PROCEDURE FOR PERFORMING THE REGULATORY GAP ANALYSIS

Revision 1

August 2010

REVISION HISTORY

RECORD OF CHANGES

[Note: Alphabetic Designations Indicate Pre-Issue Drafts]

Revision No.	Revision Made By	Description	Date
0	Bob Falk, Sam Hobbs ENERCON Services, Inc.	Initial Issue	8/4/2010
1	Bob Falk, Sam Hobbs ENERCON Services, Inc.	Revision to add Tables A1-19 and A1-20	8/16/2010

EXECUTIVE SUMMARY

Most nuclear power plants in the United States are light-water reactors (LWRs). As a result, substantial portions of the regulations and the regulatory guidance that are applicable to to the licensing of domestic U.S. nuclear power plants are oriented toward LWR technology. The U.S. Department of Energy plans to use high temperature gas-cooled reactors (HTGRs) in carrying out its Next Generation Nuclear Plant (NGNP) program. As a result, some of the existing regulations and related guidance documents are either not applicable or partially applicable because they address issues that differ from the issues involved with HTGR reactors. In some cases, these regulations have underlying public health and safety bases that would support the use of these regulations as guidance, even in cases where the regulations or the related regulatory guidance are not directly applicable.

This procedure defines a process for performing a review of existing regulations (as of June 1, 2010) and regulatory guidance for applicability to the NGNP HTGR program and for identifying where there are gaps that need to be reconciled between existing LWR regulatory requirements, general design criteria, and regulatory guidance with the specific characteristics of the HTGR technology. This procedure, when implemented, will document specific regulatory gaps, identify areas where existing regulations or guidance provide relevant guidance for HTGR technology, identify areas where new or revised regulations or guidance are potentially needed, and identify areas where further design development of the specific HTGR is needed to adequately allow the regulatory gap analysis to be completed.

The scope of the prescribed review includes the portions of the *Code of Federal Regulations* and associated regulatory guidance contained in Standard Review Plans, Regulatory Guides, and other related regulatory guidance that governs the regulation of nuclear power plants and that has been identified as potentially having technology specific elements.

The results of the documented regulatory gap analysis will provide a basis for developing a combined license application content guide for the HTGR and identify needed changes to appropriate regulatory authorities to reconcile existing regulations with the NGNP HTGR technology.

This procedure was prepared and reviewed by Entergy Nuclear (Entergy) and ENERCON Services, Inc., under the direction of Entergy.

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Note that Tables A1-1 through A1-20 (listed in detail on Attachment 1) are appended to the end of the procedure due to their voluminous size.

1.0 Introduction

Idaho National Laboratory (INL) is managing and developing the proposed Department of Energy (DOE) Next Generation Nuclear Plant (NGNP), which will be a high temperature gas-cooled reactor (HTGR) with inherent and passive safety features and underlying technology that is substantially different than light water reactors (LWRs), currently the dominant technology in the domestic commercial nuclear power arena. Many (though not all) of the existing Nuclear Regulatory Commission (NRC) regulations and associated guidance are oriented toward LWR technology and design Even the regulations and guidance that are not oriented toward LWRs may not take into account technology-specific features that need to be considered for the proposed HTGR technology. A key component for moving towards development of a Combined Operating License Application (COLA) content guide for the HTGR is to identify those regulations and guidance documents that do not apply to the HTGR and those areas where new regulations or guidance may be needed.

The general approach for development of a regulatory gap analysis and reconciliation of existing regulatory requirements was proposed in the 2001 Exelon Proposed Licensing Approach for the Pebble Bed Modular Reactor (PBMR) [Ref. 6.4]. The 2009 NGNP Licensing Plan [Ref. 6.1] adopts a similar approach.

Specific relevant prior work has also been carried out as documented in the Westinghouse *NGNP-HTGR Combined License Application Writers Guide* [Ref. 6.3], which includes a chapter by chapter preliminary evaluation of NUREG-0800 [Ref. 5.12].

Also, during the period from 1989 through 1992, DOE sponsored studies conducted by the Los Alamos National Laboratory on the New Production Reactor (NPR). These studies included a number of accident analyses, risk assessments and comparisons to regulatory requirements and guidance. These documents include comparisons to General Design Criteria, highly summarized assessments of Regulatory Guides, and an HTGR version of the Standard Review Plan. Although this information is technology specific to the NPR and the regulations have evolved considerably since that time, the documents provide useful insight and breadth that may be useful for consideration. [Refs. 6.5-6.24]

There are also other documents in the available literature that have a bearing on prior evaluations of regulations and regulatory guidance that is relevant to HTGRs. In particular, the Preliminary Safety Information Document for the Standard MHTGR (including an attachment on General Design Criteria) [Refs. 6.25 and 6.26] and an evaluation prepared by the CEGA, Inc.[1] concerning "NP-MHTGR Assessment of the Applicability of NRC Regulatory Guides and Branch Technical Positions." [Ref. 6.27]

In 2007, the NRC also published NUREG-1860 [Ref. 6.2], which includes potential changes to regulatory requirements and guidance in a highly summarized form that address the differences between the LWR and HTGR technologies. (Although these potential changes were not developed in detail and have not been adopted or endorsed through any of the formal processes used by the

[1] Note that CEGA, Inc. was a joint venture of Combustion Engineering and General Atomic, and that the corporation no longer exists. Most of the documents prepared by CEGA were initially prepared electronically in Framemaker which presented problems in the electronic conversion of documents. The only CEGA document referenced in this procedure was scanned by earlier users to obtain it in electronic format.

NRC in promulgating regulations or guidance, the information represents a technology neutral approach that was developed by the responsible regulatory agency.)

Other sources of relevant information may exist which can be used in conducting an evaluation and comparison of existing regulations and regulatory guidance, but were not identified during the preparation of this procedure.

2.0 Purpose

The purpose of this procedure is to define the process necessary to perform a Regulatory Gap Analysis that will identify the applicability of existing regulations and guidance to HTGR technology and will identify where there are gaps that need to be reconciled between existing LWR regulatory requirements, general design criteria (GDC), and regulatory guidance with the specific characteristics of HTGRs that are being considered as part of the NGNP project.

The existing regulations and guidance used in developing this procedure were those in effect on June 1, 2010. Future use of the results of the evaluation prepared during implementation of this procedure must accommodate regulatory changes that have occurred since the base date of June 1, 2010.

3.0 Objectives

The objectives of this procedure are to 1) identify applicability of existing regulations, GDCs, and regulatory guidance to the HTGR design for the NGNP, 2) identify areas where the NGNP design may need new or revised regulations or guidance, and 3) identify areas where additional design information is necessary in order to complete a regulatory applicability determination and describe how selected design options may affect regulatory applicability.

4.0 Scope

This task will address regulatory requirements and associated guidance, including, but not limited to, 10 CFR 50, Appendix A, “General Design Criteria,” of 10 CFR 50, 10 CFR 51, 10 CFR 52, 10 CFR 100, applicable Regulatory Guides, and Standard Review Plans (NUREG-0800, NUREG-1555). A more detailed list is contained in Section 5.0 of this procedure and are further delineated in the attached tables, which are to be completed by the assigned organizations and individuals.

This procedure applies to the organizations and individuals assigned responsibility for completing the regulatory gap analysis.

5.0 Regulatory Requirements and Guidance

The particular regulatory requirements and guidance to be evaluated as part of this procedure are:

5.1. 10 CFR 20 – *Standards for Protection Against Radiation*

5.2. 10 CFR 50 – *Domestic Licensing of Production and Utilization Facilities* (including appendices)

5.3. 10 CFR 51 – *Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions*

5.4. 10 CFR 52 – *Licenses, Certifications, and Approvals for Nuclear Power Plants*

5.5. 10 CFR 55 – *Operators' Licenses*

5.6. 10 CFR 70 – *Domestic Licensing of Special Nuclear Material*

5.7. 10 CFR 73 – *Physical Protection of Plants And Materials*

5.8. 10 CFR 75 – *Safeguards on Nuclear Material—Implementation of US/IAEA Agreement*

5.9. 10 CFR 100 – *Reactor Site Criteria*

5.10. 10 CFR 140 – *Financial Protection Requirements and Indemnity Agreements*

5.11. 10 CFR 961 – *Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste*

5.12. NUREG-0800 – *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition* (Standard Review Plan reviews will focus on the SRP Acceptance Criteria)

5.13. NUREG-1555 – *Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan*

5.14. Division 1 Regulatory Guides – *Power Reactors* (Regulatory Guide reviews will focus on the Regulatory Positions identified in Section C of most Regulatory Guides)

5.15. Division 4 Regulatory Guides – *Environmental and Siting*

5.16. Division 5 Regulatory Guides – *Materials and Plant Protection* (excluding the guides that are related to safeguards information)

5.17. NUREG-0933, *Resolution of Generic Safety Issues (Formerly entitled "A Prioritization of Generic Safety Issues"*; to the extent that the issues have not been incorporated in the Standard Review Plan)

5.18. NUREG-0737 (to the extent that the issues have not been incorporated into the Standard Review Plan)

5.19. NRC Generic communications such as Interim Staff Guidance, Regulatory Issue Summaries, and Generic Letters (issued since the last revision to relevant portions of the Standard Review Plan)

5.20. SECY documents and associated Staff Requirements Memoranda (SRM).

The specific individual elements of regulations, acceptance criteria, regulatory positions, and guidance are delineated on the Applicability Determination Table templates contained in the Attachments to this procedure.

6.0 Other References

6.1. PLN-3202, "NGNP Licensing Plan, Idaho National Laboratory," June 26, 2009 (Especially Sections 2.2.5.3, "LWR Regulation Reconciliation Development" and 2.2.5.4, "Regulatory Gap Analysis")

6.2. NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," Nuclear Regulatory Commission, 2007.

6.3. NGNP-NHS WEC-LIC-2, "Conceptual Design Studies for the NGNP With Hydrogen Production NGNP-HTGR Combined License Application Writers Guide," Westinghouse Electric Company, LLC, May 2009

6.4. "Proposed Licensing Approach For The Pebble Bed Modular Reactor In The United States," Exelon Generation Company, August 31, 2001

6.5. LA-NPR-3, Revision 0 (LA-CP-89-548), "Exploratory Safety Studies of the NPR-MHTGR Pre-conceptual Design," Los Alamos National Laboratory, December 1989

6.6. LA-NPR-5, Revision 3 (DOE/NP MGR-GSR-00001, Revision 0), "General Safety Requirements for the New Production Modular High Temperature Gas-Cooled Reactor, Volume 1 – Principles and Requirements," Los Alamos National Laboratory, January 30, 1991

6.7. LA-NPR-5, Revision 3 (DOE/NP MGR-GSR-00001, Revision 0), "General Safety Requirements for the New Production Modular High Temperature Gas-Cooled Reactor, Volume 2 – Appendices," Los Alamos National Laboratory, January 30, 1991

6.8. MGR-GSR-0001, Appendix A, "Justification and Discussion of General Safety Requirements for the NP-MHTGR," Los Alamos National Laboratory, January 30, 1991

6.9. MGR-GSR-0001, Appendix B, "General Design Criteria from 10 CFR 50 with Changes Shown for the NP-MHTGR," Los Alamos National Laboratory, January 30, 1991

6.10. MGR-GSR-0001, Appendix C, "Linkages Between the General Safety Principles (Section 3) and the General Safety Requirements (Section 4)," Los Alamos National Laboratory, January 30, 1991

6.11. MGR-GSR-0001, Appendix D, "Linkages Between 10 CFR and the General Safety Requirements," Los Alamos National Laboratory, January 30, 1991

6.12. MGR-GSR-0001, Appendix E, "Linkages Between DOE Orders and the General Safety Requirements," Los Alamos National Laboratory, January 30, 1991

6.13. LA-NPR-7, Revision 0 (MGR-STD-001), "Format And Content Guide for the New Production Modular High Temperature Gas-Cooled Reactor Integrated Safety Analysis Report" Los Alamos National Laboratory, December 23, 1992

6.14. LA-NPR-10, "New Production Reactor conceptual Design Safety Review Plan," Los Alamos National Laboratory, February 22, 1990

6.15. LA-NPR-11, Revision 1D (DOE/NP-000X), "Safety Review Process for the New Production Reactors Program," Los Alamos National Laboratory, October 11, 1991

6.16. LA-NPR-15, "The New Production Reactor Probabilistic Risk Assessment Detailed Review Process," Los Alamos National Laboratory, June 11, 1990

6.17. LA-NPR-16, "New Production Reactors Detailed Review Plan Development Strategy," Los Alamos National Laboratory, May 17, 1990

6.18. LA-NPR-17, "Quantitative Risk Analysis Results for the NPRG HWR-NPR Conceptual Design," Los Alamos National Laboratory, July 20, 1990

6.19. LA-NPR-21, Revision 1, "Conceptual Design Safety Review Report for the New Production Modular High Temperature Gas-Cooled Reactor," Los Alamos National Laboratory, November 26, 1990

6.20. LA-NPR-22, Revision 2, "A Limited-Scope Probabilistic Safety Analysis of the NP-MHTGR Conceptual Design: Preliminary Safety Insights," Los Alamos National Laboratory, May 10, 1991

6.21. LA-NPR-25, "Neutronic Analyses of the NP-MHTGR Pre-conceptual Reactor Design," Los Alamos National Laboratory, January 4, 1991

6.22. LA-NPR-27, "NP-MHTGR Thermal-Hydraulics/Accident Analysis Work: FY-1990 Report," Los Alamos National Laboratory, December 2, 1990

6.23. LA-NPR-31, Transition Version (DOE/NP MGR-STD-0002), "NP-MHTGR Detailed Review Plan," Los Alamos National Laboratory, March 31, 1992

6.24. LA-NPR-31 (DOE/NP MGR-STD-0002), "NP-MHTGR Detailed Review Plan," Los Alamos National Laboratory, 1991 (Chapters 1-18 and Appendices A and B have varying revision numbers)

6.25. HTGR-86-024, "Preliminary Safety Information Document for the Standard MHTGR," Volume 1 of multiple volumes, Stone & Webster Engineering Corp. (and other corporate authors), 1989

6.26. HTGR-86-024, Attachment 1 to R G.3-1, "Review of Light-Water Cooled Reactor General Design Criteria Relative to the MHTGR," Stone & Webster Engineering Corp. (and other corporate authors), 1989

6.27. CEGA-002449, "NP-MHTGR Assessment of the Applicability of NRC Regulatory Guides and Branch Technical Positions," CEGA Corporation, November 30, 1992

7.0 Definitions

Applicable – A specific regulation, regulatory guidance criteria, or position is "Applicable" if it specifies requirements or guidance that should be applied to the HTGR design without modification.

Partially Applicable – A specific regulation, regulatory guidance criteria, or position is "Partially Applicable" if there are some aspects that are applicable and some aspects that are not applicable **OR** if the underlying principle or purpose of the regulation or guidance is applicable regardless of whether the specific regulatory language related to that principle can be directly applied to an HTGR.

Not Applicable – A specific regulation, regulatory guidance criteria, or position is "Not Applicable" if it specifies requirements or guidance that does not apply to the HTGR.

Requirement – A regulatory requirement is a "requirement" specifically called out in the *Code of Federal Regulations*.

Guidance – Guidance is provided in regulatory documents such as Regulatory Guides, Standard Review Plans, Interim Staff Guidance, and NRC generic communications. For the purposes of this procedure, the user will need to distinguish between 'guidance' that reflects official NRC positions vs. 'guidance' that may be useful but does not reflect official NRC approval of a position/method for satisfying the Commission's regulations. A particular area of concern is guidance that can be used for HTGRs but that the NRC has only endorsed for LWRs.

Identified Potential Regulatory Change – A potentially needed new or modified element of a regulatory requirement or guidance identified during the applicability determination review to address an element of HTGR safety and is not addressed by existing regulation. These identified changes should include guidance that reflects an official NRC position and that can be used for HTGRs, but is only endorsed by the NRC for LWRs.

8.0 Responsibilities

8.1 Responsible Manager

8.1.1 Define appropriate qualifications for analysts (and/or teams of analysts) including, as appropriate, such factors as familiarity and experience with: regulations and their intent and application, LWR and HTGR technology, licensing (including Part 52 licensing), experience level, and education level.

8.1.2 Assure that qualified personnel are used for the analysis task and document the qualifications of the individual analysts and/or analysis teams.

8.1.3 Define the training for individual analysts and members of the analysis teams. Include specific training on the procedure and the details of how the analysis is to be conducted. Provide for general familiarization with the References (Section 6) identified in the procedure with special emphasis on references identified in the Instructions (Section 9) of this procedure.

8.1.4 Assure that analysts are trained PRIOR TO conducting the analysis task and document the training.

8.1.5 Oversee conduct of analysis as follows:

1. Assure that analysis is conducted in accordance with the procedure.
2. Assure timely, accurate and consistent completion of the analysis.
3. Assure that each element of the analysis is traceable as specified in the detailed instructions of this procedure. Note that the instructions provide for traceability on a row-by-row basis for the entries to tables that will be provided as the output of executing this procedure. Traceability for other elements of the work conducted using this procedure must also be documented appropriately. (A log or other documentation in a suitable format to allow row-by-row traceability shall be maintained.)
4. Resolve any questions or issues identified during the conduct of the analysis, this includes providing advice or guidance on the proper classification of items being evaluated.
5. Assure that the logic for difficult determinations is documented as specified in the detailed instructions of the procedure.
6. Escalate identified issues as appropriate.
7. Periodically (on a random basis) use independent reviews on a selective basis to confirm accuracy, completeness, consistency and documentation of work being conducted in carrying out the procedure.

8.1.6 Carry out the specific instructions in the procedure that are assigned to the responsible manager.

8.2 Analysts

8.2.1 Carrying out the specific instructions in this procedure (except where those instructions explicitly apply to the responsible manager).

8.2.2 Identify questions regarding conduct of the analysis or proper classification of items being evaluated to the responsible manager for resolution. Work should be continued on remaining aspects of the analysis while the resolution of such questions is obtained.

9.0 Instructions

9.1 Products to be prepared

This Regulatory Gap Analysis consists of providing three specific products:

- Applicability determination table (this is the primary product from the procedure) and is intended to provide a technology neutral gap analysis that will identify LWR technology limitations in the reviewed regulations and regulatory guidance. Although the intent is to generally identify areas where the regulations and regulatory guidance do not adequately translate to HTGRs, the gap analysis should be broad enough to accommodate the evolution and changes that may occur and should not focus on a single design in a manner that would exclude its use for evolving designs and design concepts.
- Summary table of additional design information needed to complete the vendor regulatory gap analysis (this table provides additional detailed information to the Applicability Determination Table regarding areas where additional design information may be needed)
- Summary table identifying potential new or modified regulations or guidance necessary for deployment of the proposed HTGR technology design (this table provides additional detailed information to the Applicability Determination Table regarding areas where there are potential needs for new or modified regulatory requirements or guidance).

9.2 Applicability Determination Table

A template for the Applicability Determination Table is provided in Attachment 1.

For each row in the table, the assigned analyst is responsible for:

- Determining applicability.
- Providing a basis for each applicability determination. Where a regulation is considered applicable or partially applicable the determination should be made whether the regulation is directly applicable as a legal requirement or whether the underlying principle or purpose of the requirement is applicable.
- Identifying whether further design information is needed. If further design information is needed briefly identify what that information is and whether design options may affect the applicability determination.
- Identifying areas where existing regulatory requirements and guidance do not adequately address safety aspects of the HTGR design and/or operation. For each area identified, briefly describe the appropriate changes to regulations and/or guidance.

> **NOTE:** In conducting the regulatory gap evaluation, the following kinds of considerations and previously available information should be considered.
>
> - When performing the applicability evaluation as described above, consideration should be given to:

- Limiting occupational and public radiation exposures during normal operations
- Avoiding postulated accidents
- Reducing the consequences of postulated accidents
- Limiting the environmental impact during normal operations and postulated accidents
- Providing for plant security.

At the completion of a given review, the analyst should attempt to arrive at a body of regulations and/or guidance that at least achieves the same degree of protection of the public and the environment for the HTGR that is achieved for the current generation LWRs.

- When developing the information to complete entries in the Applicability Determination Table, consideration should be given to the underlying reasons for the original requirement or guidance being evaluated. In some cases, technology dependent aspects may not always be immediately apparent. Limitations stated in requirements or guidance may be implicitly dependent on LWR technology without acknowledgement of this basis.
- Special sensitivity is warranted during evaluation of applicable requirements or guidance regarding the fuel cycle, ultimate spent fuel disposal, or waste confidence (such as 10 CFR 51.23 or other parts of 10 CFR 51, 10 CFR 961, and relevant portions of NUREG-1555). Note that detailed treatment of the overall Waste Confidence Rule or Waste Confidence Decision (which affects all reactors) is not warranted during the applicability evaluation, but these areas may be especially dependent on implicit assumptions about the fuel cycle or LWR fuel. For example, the ability to transport spent fuel may be dependent on the availability of suitable transportation casks (and possibly dependent on transportation regulations beyond the scope of this review) that should be noted during the evaluation.
- Regulations and guidance that primarily relate to operational aspects needs to be taken into consideration to the extent that there may be an impact on the proposed HTGR design.
- Prior work documented in WEC-LIC-2 [Ref. 6.3] concerning applicability of NUREG-0800 criteria is useful as guidance and should be considered but should not dictate a conclusion for the specific design under consideration.
- Although not adopted or endorsed in any formal process by the NRC, the potential changes to regulations or guidance provided in NUREG-1860 [Ref. 6.2], Appendix J, Tables J-1 through J-12 should be considered where appropriate.
- Other documents that provide general insight (such as top level safety principles) or specific insight (into regulatory requirements or guidance) should be considered as appropriate. These include but are not limited to the earlier reports concerning HTGRs and the Los Alamos NPR reports such as:
  - LA-NPR-3 [Ref. 6.5] concerning exploratory safety studies,
  - LA-NPR-5 [Refs. 6.6-6.12] concerning the general safety requirements including linkages between the cited general safety principles, the General Design Criteria, Title 10, Part 50 of the CFRs, and DOE Orders,
  - LA-NPR-31 [Ref. 6.24] which is an MHTGR version of the Standard Review Plan,

- HTGR-86-024 and attachments [Ref. 6.25 and 6.26] concerning preliminary safety and, information for the MHTGR.

Each row of the Applicability Determination Table identifies a specific regulatory requirement or guidance element. Different parts of the table are separated by rows identifying a parent requirement. The columns of the table specify 1) an element ID number, 2) a title or description, 3) the applicability of the element, 4) whether the element constitutes a requirement or guidance, 5) whether additional design information is needed, 6) whether additional regulations or guidance are needed, and 7) the basis for the determination of applicability and/or relevant comments.

Some line item entries in the tables in Attachment 1 are purely administrative in nature or are not technology dependent. These items are marked "Exclude from review" in the Basis/Comment field and do not need review. If, however, an entry that has been excluded from review is believed to be appropriate to be included in the review, the analyst or analysis team identifying that need, should escalate the issue to the responsible manager. If the responsible manager concurs, then that line item entry may be included in the review. In such cases, the responsible manager should also notify INL so that INL can consider whether other vendors or teams should also include the item in the review.

Instructions for completing the column entries for each row are as follows:

9.2.1 Element ID – This information is provided as part of the template. No further information needs to be added.

9.2.2 Title or Requirement – The information in this column is populated as part of the initial population of the first two columns. No further information needs to be added but may be modified as appropriate.

9.2.3 Applicability – Based on the HTGR design being evaluated and the definitions specified in this procedure, identify whether the element being considered is Applicable, Partially Applicable, or Not Applicable. A summary statement of the basis for the applicability determination should be provided in the Basis/Comments column. The specific entry in this column of the table should be:

- "Yes" for Applicable
- "Partial" for Partially Applicable (clear delineation of the applicability of such items must be provided in the Basis/Comments column)
- "NA" for Not Applicable.

For the body of 10 CFR 50, consider the information contained in the following documents:

- NUREG-1860 [Ref. 6.2] Appendix J
- LA-NPR-5, Appendix D [Ref. 6.11]

For the General Design Criteria (10 CFR 50, Appendix A), the evaluation documented in the following specific documents should be considered:

- HTGR-86-024, Attachment 1 to R G.3-1 [Ref. 6.26]
- LA-NPR-5, Appendix B [Ref. 6.9]

For Regulatory Guides, the information contained in the following references should be consulted:

- The CEGA evaluation of Regulatory Guides and Branch Technical Positions [Ref. 6.27]
- The portions of NUREG-1860 [Ref. 6.2] related to Division 1 Regulatory Guides

For the NUREG-0800 items, the following references should be specifically consulted:

- The applicability determination provided in WEC-LIC-2 [Ref. 6.3]
- LA-NPR-31, the MHTGR Detailed Review Plan [Ref. 6.24]
- LA-NPR-7 [Ref. 6.15] which corresponds to the Regulatory Guide 1.70 Standard Format and Content Guide is organized by Safety Analysis Report chapters and may provide useful insight into corresponding sections of NUREG-0800
- To the extent that Branch Technical Positions are involved, the CEGA evaluation of Regulatory Guides and Branch Technical Positions [Ref. 6.27]

In addition, for NUREG-0800 items, the corresponding portion of Regulatory Guide 1.206 should be included in the evaluation for the NUREG-0800 item. (Because most of the applicable portion of Regulatory Guide 1.206 is organized on a section by section basis in the same manner as NUREG-0800 and covers the same general material, this approach will avoid duplication and simplify the overall evaluation. The Basis/Comments entry should identify the portion of the evaluation regarding Regulatory Guide 1.206.) Selected portions of Regulatory Guide 1.206 that need independent evaluation are listed in the Division 1 Regulatory Guide table (Table A1-12).

9.2.4 Regulation or Guidance (abbreviated: "Reg or Guidance") for those items that are identified as being either Partially Applicable or Applicable, enter R if the item being considered is a legal requirement or GA if the item being considered is guidance found acceptable by the NRC for the HTGR, or GNA if it is useful guidance but not yet approved by the NRC for the HTGR application. An applicable or partially applicable regulation MAY be considered to be guidance (GNA) if the regulation is intended to be for an LWR design but the underlying principle or purpose is applicable to an HTGR.

For the NUREG-0800 items, information provided in WEC-LIC-2 [Ref. 6.3] on whether an item is useful as guidance should also be considered in making this determination.

In addition, other specific citations listed in the tabular note to Section 9.2.3 of this procedure should also be considered as appropriate.

9.2.5 Additional Design Information Needed (abbreviated: "Additional Design Info") – An X should be entered if the design is not developed sufficiently and that additional design information is needed to make an applicability determination. When this is the case, the entry in the Basis/Comments column should provide a brief explanation and, when appropriate, the explanation should include information on how different design options affect applicability. To the extent that the explanatory information is voluminous, it may be summarized in the column entry with the additional information contained in a note to this table. In addition to the entry in the Basis/Comments section an appropriate entry should be included in the Summary Report of Additional Design Information Needed (Attachment 2).

9.2.6 Additional Regulation or Guidance Needed (abbreviated: "Additional Reg Needed") – If, as a result of evaluating the applicability of the regulatory element, it is determined that there are areas where existing regulatory requirements and guidance do not adequately address safety aspects of the HTGR design and, therefore, additional HTGR specific regulation, policy or guidance is needed, this column should be filled in with an X. In these cases, the Basis/Comments column should identify and describe the additional regulation, policy, or guidance. In addition to the entry in the Basis/Comments section, an appropriate entry should be included in the Summary Report of Recommended Regulatory Changes (Attachment 3).

The need for additional regulation and/or guidance may be for technical and/or licensing reasons. Technical reasons could include HTGR design features which are different than LWR design features, thus requiring different guidance. Licensing reasons could include the fact that a Regulatory Guide position has only been approved for a LWR, even though it is likely to be the same for an HTGR. For example NUREG 0800 (Standard Review Plan: LWR Edition) has not been approved for an HTGR even though it is possible that an HTGR SRP would have the same content. It is anticipated that guidance categorized GNA (see step 9.2.4) would thus also be identified as "needing additional guidance" for the licensing reason that NRC endorsement for the HTGR is needed.

It is also possible that during the evaluation, a need could be identified for an entirely new regulation or regulatory guidance. If any such situations are identified, the analyst or analysis team should add a new line to the table with a unique identifier and the notation NEW in the "Additional Reg Needed" column and with a discussion in the Basis/Comments column regarding the need for the new regulation or guidance. Any such cases should be escalated to the responsible manager for concurrence. If the responsible manager concurs, INL should also be notified to provide the ability to alert other vendors or analysis teams to the new item.

Note: Information contained in NUREG-1860 [Ref. 6.2], Appendix J, Tables J-1 through J-12 should be considered in identifying areas for potential changes in regulations or regulatory guidance to accommodate the proposed NGNP HTGR designs. These tables were an early preliminary effort by selected NRC staff personnel to identify potential changes to current regulations and regulatory guidance for a technology neutral approach as well as selected HTGR technology specific areas. A number of specific items are identified in those tables related to 10 CFR 20, 50 (including the GDCs), 73, and 100 as well as Division 1 Regulatory Guides.

In addition, other specific citations listed in the tabular note to Section 9.2.3 of this procedure should also be considered as appropriate.

9.2.7 Basis/Comments – A summary statement of the basis for the applicability determination should be provided. For "Partially Applicable" classifications, particular attention is needed to clearly delineate the applicability of the item. Also, information on the need for additional design information, effects of design options, and/or need for additional regulations or guidance should be provided as appropriate. Both technical and licensing basis/comments should be included as appropriate (see step 9.2.6). In cases where the column entry providing this information would prove cumbersome to the tabular format, the information may be provided in a NOTE. In that case, the Basis/Comment entry should end: "See Notes." Also, detailed information may be placed in Table 2 with a reference to that table item.

Note: In completing the basis entries for each row of the Applicability Determination Table, reference should be made to duplicate or similar information provided in other rows of the table. Complete duplication of information should generally be avoided though with some exceptions. Where the information has been provided elsewhere, the entry for the item should be completed, but the basis information should reference the primary row of the table that contains the information. (As an example, there may be references in the Standard Review Plan elements being reviewed to specific Regulatory Guides. In such cases, the basis recorded for the Standard Review Plan element should provide a statement referring to the evaluation done for the specific Regulatory Guide.) In cases where the needed basis information is generic and is completely specified by the basis information in the referenced item of the table, the reference alone is a sufficient basis. If the basis information needed is more specific or if the referenced information does not completely specify the appropriate basis information, a cross reference should be provided along with the additional basis information. (Example: the Standard Review Plan contains system specific regulatory guidance, and it is appropriate to record the basis for determining the applicability of system specific guidance in the row of the Applicability Determination Table for that Standard Review Plan element, even if this results in some duplication of information.)

9.2.8 A notes section should be provided, if needed at the end of the Applicability Determination Table. The notes should be identified with the element ID number that refers to the specific note. A note should be added to this section of the table when the Basis/Comments entry would be too voluminous. When a note is added, the information in the corresponding row of the Basis/Comments column should provide a brief summary and a notation to "See Notes." Notes should be used sparingly.

9.2.9 The responsible manager should assure that difficult determinations regarding applicability, whether items are partially applicable and the need for new or additional regulations should assure that the logic and criteria used in those determinations is documented. That documentation may be delegated to analysts (or analysis teams) or independent reviewers or may be prepared by the responsible manager. One approach that may be useful in such documentation is the use of a decision matrix on a case by case basis.

9.2.10 The responsible manager (or designee) should review completed entries in the applicability determination table on a line by line basis to assure accuracy, completeness and consistency in the approach used for the evaluation. The responsible manager may choose to have additional independent reviews done on a selective basis to assist in assuring accuracy and consistency.

9.2.11 Documentation of personnel preparing the report, personnel conducting independent reviews (when independent reviews are used), the review by the responsible manager (or designee) and the dates of evaluation and reviews should be documented in a manner that allows row-by-row traceability of the results of the analysis contained in the completed Applicability Determination Table.

9.3 Summary report of additional design information needed

In areas where the regulatory gap analysis cannot be fully completed because of the lack of design information, the Applicability Determination Table identifies the specific additional design information needed to complete the regulatory gap analysis and describes how relevant design options might affect the applicability of the specific requirement.

9.3.1 Using the table layout specified in Attachment 2, the analyst should compile a summary of areas where additional design information is needed. This information should come from the Applicability Determination Table (including any information contained in notes to the Applicability Determination Table). Each separate area where additional design information is needed should be in a separate entry in the table. The description entry should be a narrative format. To the extent applicable, the impact of design options should be included in the description. The table column "Regulatory Gap Element Identifier" should be filled in with the appropriate corresponding "ID" entry from the Applicability Determination Table.

> Some entries in this report may have multiple element ID numbers from the Applicability Determination Table. In these cases, there should be only a single entry for the area where additional design information is needed, but all of the element ID numbers should be listed.

9.3.2 Documentation of personnel preparing the report and the dates of the work should be maintained.

9.4 Summary table of potential regulatory changes

The Applicability Determination Table also identifies areas where existing regulatory requirements and guidance do not adequately address safety aspects of the HTGR design and/or operation. This table compiles those identified potential changes needed to regulations or guidance in this separate summary table. The analyst responsible for preparing the Summary Table of Potential Regulatory Changes should complete the following.

9.4.1 Using the table layout specified in Attachment 3, compile a summary of potentially needed regulatory changes (including any completely new regulations or guidance). This information should come from the Applicability Determination Table (including any information contained in notes to the Applicability Determination Table). Each separate potentially needed regulatory change should be in a separate entry in the table. The description entry should be a narrative format. The table column "Regulatory Gap Element Identifier" should be filled in with the appropriate corresponding "ID" entry from the Applicability Determination Table.

> Some entries in this report may have multiple element ID numbers from the Applicability Determination Table. In these cases, there should be only a single entry for the potentially needed regulatory change, but all element ID numbers should be listed.

9.4.2 Documentation of personnel preparing this table and the dates of the work should be maintained.

10.0 List of Attachments

Attachment 1 – Template for Applicability Determination Table (Note that due to voluminous size of the tables, these tables are appended to the end of the procedure.)

- Table A1-1: Part 20--Standards for Protection Against Radiation
- Table A1-2: Part 50--Domestic Licensing of Production and Utilization Facilities
- Table A1-3, Part 50, Appendix A --General Design Criteria
- Table A1-4, Part 51--Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions
- Table A1-5, Part 52--Licenses, Certifications, and Approvals for Nuclear Power Plants
- Table A1-6, Part 55--Operators' Licenses
- Table A1-7, Part 70—Domestic Licensing of Special Nuclear Material
- Table A1-8, Part 73—Physical Protection of Plants And Materials
- Table A1-9, Part 100--Reactor Site Criteria
- Table A1-10, Part 140--Financial Protection Requirements and Indemnity Agreements
- Table A1-11, Part 961--Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste

- Table A1-12, Regulatory Guides (Division 1)
- Table A1-13, Regulatory Guides (Division 4)
- Table A1-14, Regulatory Guides (Division 5)
- Table A1-15, NUREG-0800, Standard review Plan
- Table A1-16, NUREG-1555, Environmental Report Standard Review Plan
- Table A1-17, Interim Staff Guidance
- Table A1-18, Generic Letters and SECY documents
- Table A1-19, Three Mile Island Requirements (NUREG-0737)
- Table A1-20, Unresolved and Generic Safety Issues (NUREG-0933)

Attachment 2 – Template for Summary Report of Additional Design Information Needed

Attachment 3 – Template for Summary Table of Potential Regulatory Changes

Attachment 1 – Template for Applicability Determination Table

Due to the voluminous nature of the detailed tables, Tables A1-1 through A1-20 are appended to the end of the procedure.

Attachment 2 –Summary Table of Addition Design Information Needed	
Regulatory Gap Element Identifier(s)	**Identification/Description of Additional Design Information Needed**

Attachment 3 –Summary Table of Potential Regulatory Changes

Regulatory Gap Element Identifier(s)	Identification/Description of Potential Regulatory Change

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Subpart A--General Provisions					Heading
20.1001	Purpose.					Exclude; Admin
20.1002	Scope.					Exclude; Admin
20.1003	Definitions.					Exclude; Admin
20.1004	Units of radiation dose.					Exclude; Admin
20.1005	Units of radioactivity.					Exclude; Admin
20.1006	Interpretations.					Exclude; Admin
20.1007	Communications.					Exclude; Admin
20.1008	Implementation.	NA				Exclude
20.1009	Information collection requirements: OMB approval.					Exclude; Admin
	Subpart B--Radiation Protection Programs					Heading
20.1101	Radiation protection programs.					Exclude; Admin
	Subpart C--Occupational Dose Limits					Heading
20.1201	Occupational dose limits for adults.					Exclude; Admin
20.1202	Compliance with requirements for summation of external and internal doses.					Exclude; Admin
20.1203	Determination of external dose from airborne radioactive material.					Exclude; Admin
20.1204	Determination of internal exposure.					Exclude; Admin
20.1205	[Reserved]	NA				Exclude
20.1206	Planned special exposures.					Exclude; Admin
20.1207	Occupational dose limits for minors.					Exclude; Admin
20.1208	Dose equivalent to an embryo/fetus.					Exclude; Admin
	Subpart D--Radiation Dose Limits for					Heading

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Individual Members of the Public					
20.1301	Dose limits for individual members of the public.					Exclude; Admin
20.1302	Compliance with dose limits for individual members of the public.					Exclude; Admin
	Subpart E--Radiological Criteria for License Termination					Heading
20.1401	General provisions and scope.					Exclude; Admin
20.1402	Radiological criteria for unrestricted use.					Exclude; Admin
20.1403	Criteria for license termination under restricted conditions.					Exclude; Admin
20.1404	Alternate criteria for license termination.					Exclude; Admin
20.1405	Public notification and public participation.					Exclude; Admin
20.1406	Minimization of contamination.					Exclude; Admin
	Subpart F--Surveys and Monitoring					Heading
20.1501	General.					Exclude; Admin
20.1502	Conditions requiring individual monitoring of external and internal occupational dose.					Exclude; Admin
	Subpart G--Control of Exposure From External Sources in Restricted Areas					Heading
20.1601	Control of access to high radiation areas.					Exclude; Admin
20.1602	Control of access to very high radiation areas.					Exclude; Admin
	Subpart H--Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas					Heading

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
20.1701	Use of process or other engineering controls.					Exclude; Admin
20.1702	Use of other controls.					Exclude; Admin
20.1703	Use of individual respiratory protection equipment.					Exclude; Admin
20.1704	Further restrictions on the use of respiratory protection equipment.					Exclude; Admin
20.1705	Application for use of higher assigned protection factors.					Exclude; Admin
	Subpart I--Storage and Control of Licensed Material					Heading
20.1801	Security of stored material.					Exclude; Admin
20.1802	Control of material not in storage.					Exclude; Admin
	Subpart J--Precautionary Procedures					Heading
20.1901	Caution signs.					Exclude; Admin
20.1902	Posting requirements.					Exclude; Admin
20.1903	Exceptions to posting requirements.					Exclude; Admin
20.1904	Labeling containers.					Exclude; Admin
20.1905	Exemptions to labeling requirements.					Exclude; Admin
20.1906	Procedures for receiving and opening packages.					Exclude; Admin
	Subpart K--Waste Disposal					Heading
20.2001	General requirements.					Exclude; Admin
20.2002	Method for obtaining approval of proposed disposal procedures.					Exclude; Admin
20.2003	Disposal by release into sanitary sewerage.					Exclude; Admin

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
20.2004	Treatment or disposal by incineration.					Exclude; Admin
20.2005	Disposal of specific wastes.					Exclude; Admin
20.2006	Transfer for disposal and manifests.					Exclude; Admin
20.2007	Compliance with environmental and health protection regulations.					Exclude; Admin
20.2008	Disposal of certain byproduct material.					Exclude; Admin
	Subpart L--Records					Heading
20.2101	General provisions.					Exclude; Admin
20.2102	Records of radiation protection programs.					Exclude; Admin
20.2103	Records of surveys.					Exclude; Admin
20.2104	Determination of prior occupational dose.					Exclude; Admin
20.2105	Records of planned special exposures.					Exclude; Admin
20.2106	Records of individual monitoring results.					Exclude; Admin
20.2107	Records of dose to individual members of the public.					Exclude; Admin
20.2108	Records of waste disposal.					Exclude; Admin
20.2109	[Reserved]	NA				Exclude
20.2110	Form of records.					Exclude; Admin
	Subpart M--Reports					Heading
20.2201	Reports of theft or loss of licensed material.					Exclude; Admin
20.2202	Notification of incidents.					Exclude; Admin
20.2203	Reports of exposures, radiation levels, and concentrations of radioactive material exceeding the constraints or limits.					Exclude; Admin
20.2204	Reports of planned special exposures.					Exclude; Admin

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
20.2205	Reports to individuals of exceeding dose limits.					Exclude; Admin
20.2206	Reports of individual monitoring.					Exclude; Admin
20.2207	Reports of transactions involving nationally tracked sources.					Exclude; Admin
	Subpart N--Exemptions and Additional Requirements					Heading
20.2301	Applications for exemptions.					Exclude; Admin
20.2302	Additional requirements.					Exclude; Admin
	Subpart O--Enforcement					Heading
20.2401	Violations.					Exclude; Admin
20.2402	Criminal penalties.					Exclude; Admin
20-App.A	Assigned Protection Factors for Respirators					Exclude; Admin
20-App.B	Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage [RELEVANT REVIEW MATERIAL IS EXCERPTED RELATED TO TABLE 2 Introduction . . . Table 2 provides concentration limits for airborne and liquid effluents released to the					Effluent concentration limits should be considered in the review.

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	general environment. . . . Table 2 The columns in Table 2 of this appendix captioned "Effluents," "Air," and "Water," are applicable to the assessment and control of dose to the public, particularly in the implementation of the provisions of § 20.1302. The concentration values given in Columns 1 and 2 of Table 2 are equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 millirem or 0.5 millisieverts). Consideration of non-stochastic limits has not been included in deriving the air and water effluent concentration limits because non-stochastic effects are presumed not to occur at the dose levels established for individual members of the public. For radionuclides, where the non-stochastic limit was governing in deriving the occupational DAC, the stochastic ALI was used in deriving the					

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	corresponding airborne effluent limit in Table 2. For this reason, the DAC and airborne effluent limits are not always proportional as was the case in appendix B to §§ 20.1-20.601. The air concentration values listed in Table 2, Column 1, were derived by one of two methods. For those radionuclides for which the stochastic limit is governing, the occupational stochastic inhalation ALI was divided by 2.4 x 109ml, relating the inhalation ALI to the DAC, as explained above, and then divided by a factor of 300. The factor of 300 includes the following components: a factor of 50 to relate the 5-rem annual occupational dose limit to the 0.1-rem limit for members of the public, a factor of 3 to adjust for the difference in exposure time and the inhalation rate for a worker and that for members of the public; and a factor of 2 to adjust the occupational values (derived for adults) so that they are applicable to other age groups. For those radionuclides for which submersion (external dose) is limiting, the occupational					

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	DAC in Table 1, Column 3, was divided by 219. The factor of 219 is composed of a factor of 50, as described above, and a factor of 4.38 relating occupational exposure for 2,000 hours per year to full-time exposure (8,760 hours per year). Note that an additional factor of 2 for age considerations is not warranted in the submersion case. The water concentrations were derived by taking the most restrictive occupational stochastic oral ingestion ALI and dividing by 7.3 x 107. The factor of 7.3 x 107 (ml) includes the following components: the factors of 50 and 2 described above and a factor of 7.3 x 105 (ml) which is the annual water intake of "Reference Man." Note 2 of this appendix provides groupings of radionuclides which are applicable to unknown mixtures of radionuclides. These groupings (including occupational inhalation ALIs and DACs, air and water effluent concentrations and sewerage) require demonstrating that the most limiting radionuclides in successive classes are absent.					

Table A1-1: PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	The limit for the unknown mixture is defined when the presence of one of the listed radionuclides cannot be definitely excluded either from knowledge of the radionuclide composition of the source or from actual measurements. [TABLE 2 SHOULD BE CONSULTED IN THE ACTUAL REGULATIONS]					
20-App.C	Quantities of Licensed Material Requiring Labeling					Exclude; Admin
20-App.D	United States Nuclear Regulatory Commission Regional Offices					Exclude; Admin
20-App.E	Nationally Tracked Source Thresholds					Exclude; Admin
20-App.F	[Reserved]	NA				Exclude
20-App.G	Requirements for Transfers of Low-Level Radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests					Exclude; Admin

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	General Provisions					
	Sec.					
50.1	Basis, purpose, and procedures applicable.					Exclude Administrative
50.2	Definitions.					
50.3	Interpretations.					Exclude Administrative
50.4	Written communications.					Exclude Administrative
50.5	Deliberate misconduct.					Exclude Administrative
50.7	Employee protection.					Exclude Administrative
50.8	Information collection requirements: OMB approval.					Exclude Administrative
50.9	Completeness and accuracy of information.					Exclude Administrative
	Requirement of License, Exceptions					
50.10	License required; limited work authorization					Exclude Administrative
50.11	Nothing in this part shall be deemed to require a license for: (a) The manufacture, production, or acquisition by the Department of Defense of any utilization facility authorized pursuant to section 91 of the Act, or the use of such facility by the Department of Defense or by a person under contract with and for the account of the Department of Defense; (b) Except to the extent that Administration facilities of the types subject to licensing pursuant to section 202 of the Energy Reorganization Act of 1974 are involved; (1)(i) The processing, fabrication or refining of special nuclear material or the separation of special nuclear material, or the separation of special nuclear material from other substances by a prime contractor of the Department under a prime contract for: (A) The performance of work for the Department at a United States government-owned or					Relevant to US government owned or contracted facilities

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	controlled site; (B) Research in, or development, manufacture, storage, testing or transportation of, atomic weapons or components thereof; or (C) The use or operation of a production or utilization facility in a United States owned vehicle or vessel; or (ii) By a prime contractor or subcontractor of the Commission or the Department under a prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontractor is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety; (2)(i) The construction or operation of a production or utilization facility for the Department at a United States government-owned or controlled site, including the transportation of the production or utilization facility to or from such site and the performance of contract services during temporary interruptions of such transportation; or the construction or operation of a production or utilization facility for the Department in the performance of research in, or development, manufacture, storage, testing, or transportation of, atomic weapons or components thereof; or the use or operation of a production or utilization facility for the Department in a United States government-owned vehicle or vessel: Provided, That such activities are conducted by a prime contractor of the Department under a prime contract with the Department. (ii) The construction or operation of a production or utilization facility by a prime contractor or subcontractor of the Commission or the Department under his prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontractor is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety. (c) The transportation or possession of any production or utilization facility by a common or contract carrier or warehousemen in the regular course of carriage for another or storage incident thereto.					
50.12	Specific exemptions.					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
50.12(a)	(a) The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part, which are-- (1) Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. (2) The Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever-- (i) Application of the regulation in the particular circumstances conflicts with other rules or requirements of the Commission; or (ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule; or (iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated; or (iv) The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption; or (v) The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation; or (vi) There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption. If such condition is relied on exclusively for satisfying paragraph (a)(2) of this section, the exemption may not be granted until the Executive Director for Operations has consulted with the Commission.					
50.12(b)	(b) Any person may request an exemption permitting the conduct of activities prior to the issuance of a construction permit prohibited by § 50.10. The Commission may grant such an exemption upon considering and balancing the following factors: (1) Whether conduct of the proposed activities will give rise to a significant adverse impact on the environment and the nature and extent of such impact, if any;					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	(2) Whether redress of any adverse environment impact from conduct of the proposed activities can reasonably be effected should such redress be necessary; (3) Whether conduct of the proposed activities would foreclose subsequent adoption of alternatives; and (4) The effect of delay in conducting such activities on the public interest, including the power needs to be used by the proposed facility, the availability of alternative sources, if any, to meet those needs on a timely basis and delay costs to the applicant and to consumers. Issuance of such an exemption shall not be deemed to constitute a commitment to issue a construction permit. During the period of any exemption granted pursuant to this paragraph (b), any activities conducted shall be carried out in such a manner as will minimize or reduce their environmental impact.					
50.13	Attacks and destructive acts by enemies of the United States; and defense activities.					Exclude Administrative
	Classification and Description of Licenses					
50.20	Two classes of licenses.					Exclude Administrative
50.21	Class 104 licenses; for medical therapy and research and development facilities.					Exclude, Not applicable to the NGNP licensing.
50.22	Class 103 licenses; for commercial and industrial facilities.					Exclude Administrative
50.23	Construction permits.					Exclude Administrative
	Applications for Licenses, Certifications, and Regulatory Approvals; Form; Contents; Ineligibility of Certain Applicants					
50.30	Filing of applications for licenses; oath or affirmation.					Exclude Administrative
50.31	Combining applications.					Exclude Administrative
50.32	Elimination of repetition.					Exclude

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						Administrative
50.33	Contents of applications; general information.					Exclude Administrative
50.34	Contents of applications; technical information.					A review of only Sections 50.34(f) and (h) are necessary.
50.34 (f)	(f) Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors. Refer to 10 CFR 50.34 for the specific criteria.					
50.34(h)	(h) *Conformance with the Standard Review Plan (SRP).* (1)(i) Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982 shall include an evaluation of the facility against the Standard Review Plan (SRP) in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later. (ii) Applications for light-water cooled nuclear power plant construction permits docketed after May 17, 1982, shall include an evaluation of the facility against the SRP in effect on May 17, 1982, or the SRP revision in effect six months before the docket date of the application, whichever is later. (2) The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria. (3) The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
50.34a	Design Objective Requirements for Equipment to Control the Release of Radioactive Material					
50.34a(a)	(a) An application for a construction permit shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an application filed on or after January 2, 1971, the application shall also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable. The term "as low as is reasonably achievable" as used in this part means as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the use of atomic energy in the public interest. The guides set out in appendix I to this part provide numerical guidance on design objectives for light-water-cooled nuclear power reactors to meet the requirements that radioactive material in effluents released to unrestricted areas be kept as low as is reasonably achievable. These numerical guides for design objectives and limiting conditions for operation are not to be construed as radiation protection standards.					
50.34a(b)	(b) Each application for a construction permit shall include: (1) A description of the preliminary design of equipment to be installed under paragraph (a) of this section; (2) An estimate of: (i) The quantity of each of the principal radionuclides expected to be released annually to unrestricted areas in liquid effluents produced during normal reactor operations; and (ii) The quantity of each of the principal radionuclides of the gases, halides, and particulates expected to be released annually to unrestricted areas in gaseous effluents produced during normal reactor operations. (3) A general description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.					
50.34a(c)	(c) Each application for an operating license shall include: (1) A description of the equipment and procedures for the control of gaseous and liquid effluents					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section; and (2) A revised estimate of the information required in paragraph (b)(2) of this section if the expected releases and exposures differ significantly from the estimates submitted in the application for a construction permit.					
50.34a(d)	(d) Each application for a combined license under part 52 of this chapter shall include: (1) A description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section; and (2) The information required in paragraph (b)(2) of this section.					
50.34a(e)	(e) Each application for a design approval, a design certification, or a manufacturing license under part 52 of this chapter shall include: (1) A description of the equipment for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section; and (2) The information required in paragraph (b)(2) of this section.					
50.35	Issuance of construction permits.					Exclude Administrative
50.36	Technical Specifications.					
50.36(a)	(a)(1) Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications. (2) Each applicant for a design certification or manufacturing license under part 52 of this chapter shall include in its application proposed generic technical specifications in accordance with the requirements of this section for the portion of the plant that is within the scope of the design certification or manufacturing license application.					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
50.36(b)	(b) Each license authorizing operation of a production or utilization facility of a type described in § 50.21 or § 50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.					
50.36(c)	(c) Technical specifications will include items in the following categories: (1) *Safety limits, limiting safety system settings, and limiting control settings*. (i)(A) Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor, except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report. (B) Safety limits for fuel reprocessing plants are those bounds within which the process variables must be maintained for adequate control of the operation and that must not be exceeded in order to protect the integrity of the physical system that is designed to guard against the uncontrolled release or radioactivity. If any safety limit for a fuel reprocessing plant is exceeded, corrective action must be taken as stated in the technical specification or the affected part of the process, or the entire process if required, must be shut down, unless this action would further reduce the margin of safety. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. If a portion of the process or the entire process has been shutdown, operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant. (ii)(A) Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report. (B) Limiting control settings for fuel reprocessing plants are settings for automatic alarm or protective devices related to those variables having significant safety functions. Where a limiting control setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that protective action, either automatic or manual, will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic alarm or protective devices do not function as required, the licensee shall take appropriate action to maintain the variables within the limiting control-setting values and to repair promptly the automatic devices or to shut down the affected part of the process and, if required, to shut down the entire process for repair of automatic devices. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant. (2) *Limiting conditions for operation*. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specifications until the condition can be met. In the case of a nuclear reactor not licensed under § 50.21(b) or § 50.22 of this part or fuel reprocessing plant, the licensee shall notify the Commission, review the matter, and record the results of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the nuclear reactor or the fuel reprocessing plant. In the case of nuclear power reactors licensed under § 50.21(b) or § 50.22, the licensee shall notify the Commission if required by § 50.72 and shall submit a Licensee Event Report to the Commission as required by § 50.73. In this case, licensees shall retain records associated with preparation of a Licensee Event Report for a period of three years following issuance of the report. For events which do not require a Licensee Event Report, the licensee shall retain each record as required by the technical specifications. (ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria: (A) *Criterion 1*. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. (B) *Criterion 2*. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. (C) *Criterion 3*. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. (D) *Criterion 4*. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. (iii) A licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995, to satisfy the criteria in paragraph (c)(2)(ii) of this section. (3) *Surveillance requirements*. Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	(4) *Design features*. Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section. (5) *Administrative controls*. Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4. (6) *Decommissioning*. This paragraph applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1) and to non-power reactor facilities which are not authorized to operate. Technical specifications involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis. (7) *Initial notification*. Reports made to the Commission by licensees in response to the requirements of this section must be made as follows: (i) Licensees that have an installed Emergency Notification System shall make the initial notification to the NRC Operations Center in accordance with §50.72 of this part. (ii) All other licensees shall make the initial notification by telephone to the Administrator of the appropriate NRC Regional Office listed in appendix D, part 20, of this chapter. (8) *Written Reports*. Licensees for nuclear power reactors licensed under § 50.21(b) and § 50.22 of this part shall submit written reports to the Commission in accordance with § 50.73 of this part for events described in paragraphs (c)(1) and (c)(2) of this section. For all licensees, the Commission may require Special Reports as appropriate.					
50.36(d)	(d)(1) This section shall not be deemed to modify the technical specifications included in any license issued prior to January 16, 1969. A license in which technical specifications have not been designated shall be deemed to include the entire safety analysis report as technical specifications.					

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	(2) An applicant for a license authorizing operation of a production or utilization facility to whom a construction permit has been issued prior to January 16, 1969, may submit technical specifications in accordance with this section, or in accordance with the requirements of this part in effect prior to January 16, 1969. (3) At the initiative of the Commission or the licensee, any license may be amended to include technical specifications of the scope and content which would be required if a new license were being issued.					
50.36(e)	(e) The provisions of this section apply to each nuclear reactor licensee whose authority to operate the reactor has been removed by license amendment, order, or regulation.					
50.36a	Technical Specifications on effluents from nuclear power reactors.					Exclude, This is not technology specific.
50.36b	Environmental conditions.					Exclude Administrative
50.37	Agreement limiting access to Classified Information.					Exclude Administrative
50.38	Ineligibility of certain applicants.					Exclude Administrative
50.39	Public inspection of applications.					Exclude Administrative
	Standards for Licenses, Certifications, and Regulatory Approvals					
50.40	Common standards.					Exclude Administrative
50.41	Additional standards for class 104 licenses.	No				Not applicable to the NGNP technology.
50.42	Additional standard for class 103 licenses.					Exclude Administrative
50.43	Additional standards and provisions affecting class 103 licenses and certifications for commercial power.					Not applicable to the NGNP technology.
50.43(a)	(a) The NRC will provide notice to the regulatory agencies, the State, utilities, and the public as warranted, through various means.					
50.43(b)	(b) If there are conflicting applications for a limited opportunity for such license, the Commission					

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	will give preferred consideration in the following order: First, to applications submitted by public or cooperative bodies for facilities to be located in high cost power areas in the United States; second, to applications submitted by others for facilities to be located in such areas; third, to applications submitted by public or cooperative bodies for facilities to be located in other than high cost power areas; and, fourth, to all other applicants.					
50.43(c)	(c) The licensee who transmits electric energy in interstate commerce, or sells it at wholesale in interstate commerce, shall be subject to the regulatory provisions of the Federal Power Act.					
50.43(d)	(d) Nothing shall preclude any government agency, now or hereafter authorized by law to engage in the production, marketing, or distribution of electric energy, if otherwise qualified, from obtaining a construction permit or operating license under this part, or a combined license under part 52 of this chapter for a utilization facility for the primary purpose of producing electric energy for disposition for ultimate public consumption.					
50.43(e)	(e) Applications for a design certification, combined license, manufacturing license, or operating license that propose nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, will be approved only if: (1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof; (ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and (iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; or (2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the					

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	testing period.					
50.44	Combustible gas control for nuclear power reactors. (a) *Definitions*					
50.44(b)	(b) *Requirements for currently-licensed reactors.* Each boiling or pressurized water nuclear power reactor with an operating license on October 16, 2003, except for those facilities for which the certifications required under § 50.82(a)(1) have been submitted, must comply with the following requirements, as applicable: (following 5 requirements) (1) *Mixed atmosphere.* All containments must have a capability for ensuring a mixed atmosphere. (2) *Combustible gas control.* (i) All boiling water reactors with Mark I or Mark II type containments must have an inerted atmosphere (ii) All boiling water reactors with Mark III type containments and all pressurized water reactors with ice condenser containments must have the capability for controlling combustible gas generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity. (3) *Equipment Survivability.* All boiling water reactors with Mark III containments and all pressurized water reactors with ice condenser containments that do not rely upon an inerted atmosphere inside containment to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). (4) *Monitoring.* (i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the					

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	containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning. (ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning. (5) *Analyses.* Each holder of an operating license for a boiling water reactor with a Mark III type of containment or for a pressurized water reactor with an ice condenser type of containment, shall perform an analysis that: (i) Provides an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75 percent of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume) and include consideration of hydrogen control measures as appropriate; (ii) Includes the period of recovery from the degraded condition; (iii) Uses accident scenarios that are accepted by the NRC staff. These scenarios must be accompanied by sufficient supporting justification to show that they describe the behavior of the reactor system during and following an accident resulting in a degraded core. (iv) Supports the design of the hydrogen control system selected to meet the requirements of this section; and, (v) Demonstrates, for those reactors that do not rely upon an inerted atmosphere to comply with paragraph (b)(2)(ii) of this section, that: (A) Containment structural integrity is maintained. Containment structural integrity must be demonstrated by use of an analytical technique that is accepted by the NRC staff in accordance with § 50.90. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. This method could include the use of actual material properties with suitable margins to account for uncertainties in					

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	modeling, in material properties, in construction tolerances, and so on; and (B) Systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including local detonations, unless such detonations can be shown unlikely to occur.					
50.44(c)	(c) *Requirements for future water-cooled reactor applicants and licensees.*[2] The requirements in this paragraph apply to all water-cooled reactor construction permits or operating licenses under this part, and to all water-cooled reactor design approvals, design certifications, combined licenses or manufacturing licenses under part 52 of this chapter, any of which are issued after October 16, 2003. (following 5 requirements) *Mixed atmosphere.* All containments must have a capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents. *Combustible gas control.* All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features. *Equipment Survivability.* Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region. *Monitoring.* (i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the					

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	containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning. (ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning. *Structural analysis.* An applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.						
	(1) (d) *Requirements for future non water-cooled reactor applicants and licensees and certain water-cooled reactor applicants and licensees.* The requirements in this paragraph apply to all construction permits and operating licenses under this part, and to all design approvals, design certifications, combined licenses, or manufacturing licenses under part 52 of this chapter, for non water-cooled reactors and water-cooled reactors that do not fall within the description in paragraph (c), footnote 1 of this section, any of which are issued after October 16, 2003. Applications subject to this paragraph must include: (following 2 requirements) Information addressing whether accidents involving combustible gases are technically relevant for their design, and If accidents involving combustible gases are found to be technically relevant, information (including a design-specific probabilistic risk assessment) demonstrating that the safety impacts of combustible gases during design-basis and significant beyond design-basis accidents have been addressed to ensure adequate protection of public health and safety and common defense and security.						
50.45	Standards for construction permits, operating licenses, and combined licenses.						Exclude, This is not

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						technology specific.
50.46	Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.					
50.46(a)	(a)(1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted. (ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models. (a)(2) The Director of Nuclear Reactor Regulation may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraphs (a)(1) (i) and (ii) of this section. (a)(3)(i) Each applicant for or holder of an operating license or construction permit issued under this part, applicant for a standard design certification under part 52 of this chapter (including an					

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	applicant after the Commission has adopted a final design certification regulation), or an applicant for or holder of a standard design approval, a combined license or a manufacturing license issued under part 52 of this chapter, shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F. (ii) For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or holder of a construction permit, operating license, combined license, or manufacturing license shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in § 50.4 or § 52.3 of this chapter, as applicable. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46 requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in §§ 50.55(e), 50.72, and 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with § 50.46 requirements. (iii) For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or holder of a standard design approval or the applicant for a standard design certification (including an applicant after the Commission has adopted a final design certification rule) shall report the nature of the change					

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	or error and its estimated effect on the limiting ECCS analysis to the Commission and to any applicant or licensee referencing the design approval or design certification at least annually as specified in § 52.3 of this chapter. If the change or error is significant, the applicant or holder of the design approval or the applicant for the design certification shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46 requirements. The affected applicant or holder shall propose immediate steps to demonstrate compliance or bring plant design into compliance with § 50.46 requirements.					
50.46(b)	*(b)(1) Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200º F. *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening. *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.					

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	Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling. *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.					
50.46(c)	(c) As used in this section: (1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. (2) An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.					
50.46(d)	(d) The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of appendix A.					
50.46a	Acceptance criteria for reactor coolant system venting systems.	No				Exclude, This is not applicable to the NGNP design.
50.47	Emergency plans.					
50.47(a)	(a)(1)(i) Except as provided in paragraph (d) of this section, no initial operating license for a					

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	nuclear power reactor will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. No finding under this section is necessary for issuance of a renewed nuclear power reactor operating license. (ii) No initial combined license under part 52 of this chapter will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. No finding under this section is necessary for issuance of a renewed combined license. (iii) If an application for an early site permit under subpart A of part 52 of this chapter includes complete and integrated emergency plans under 10 CFR 52.17(b)(2)(ii), no early site permit will be issued unless a finding is made by the NRC that the emergency plans provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. (iv) If an application for an early site permit proposes major features of the emergency plans under 10 CFR 52.17(b)(2)(i), no early site permit will be issued unless a finding is made by the NRC that the major features are acceptable in accordance with the applicable standards of 10 CFR 50.47 and 10 CFR part 50, appendix E, within the scope of emergency preparedness matters addressed in the major features. (2) The NRC will base its finding on a of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local emergency plans are adequate and whether there is reasonable assurance that they can be implemented, and on the NRC assessment as to whether the applicant's onsite emergency plans are adequate and whether there is reasonable assurance that they can be implemented. A FEMA finding will primarily be based on a review of the plans. Any other information already available to FEMA may be considered in assessing whether there is reasonable assurance that the plans can be implemented. In any NRC licensing proceeding, a FEMA finding will constitute a rebuttable presumption on questions of adequacy and implementation capability.					
50.47(b)	(b) The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards: (following 16					

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	requirements) (1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis. (2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified. (3) Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified. (4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures. (5) Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow-up messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established. (6) Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public. (7) Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of					

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	information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established. (8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained. (9) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use. (10) A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed. (11) Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides. (12) Arrangements are made for medical services for contaminated injured individuals. (13) General plans for recovery and reentry are developed. (14) Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected. (15) Radiological emergency response training is provided to those who may be called on to assist in an emergency. (16) Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.					

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50.47(c)	(c)(1) Failure to meet the applicable standards set forth in paragraph (b) of this section may result in the Commission declining to issue an operating license; however, the applicant will have an opportunity to demonstrate to the satisfaction of the Commission that deficiencies in the plans are not significant for the plant in question, that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons to permit plant operations. Where an applicant for an operating license asserts that its inability to demonstrate compliance with the requirements of paragraph (b) of this section results wholly or substantially from the decision of state and/or local governments not to participate further in emergency planning, an operating license may be issued if the applicant demonstrates to the Commission's satisfaction that: (i) The applicant's inability to comply with the requirements of paragraph (b) of this section is wholly or substantially the result of the non-participation of state and/or local governments. (ii) The applicant has made a sustained, good faith effort to secure and retain the participation of the pertinent state and/or local governmental authorities, including the furnishing of copies of its emergency plan. (iii) The applicant's emergency plan provides reasonable assurance that public health and safety is not endangered by operation of the facility concerned. To make that finding, the applicant must demonstrate that, as outlined below, adequate protective measures can and will be taken in the event of an emergency. A utility plan will be evaluated against the same planning standards applicable to a state or local plan, as listed in paragraph (b) of this section, with due allowance made both for-- (A) Those elements for which state and/or local non-participation makes compliance infeasible and (B) The utility's measures designed to compensate for any deficiencies resulting from state and/or local non-participation. In making its determination on the adequacy of a utility plan, the NRC will recognize the reality that in an actual emergency, state and local government officials will exercise their best efforts to protect the health and safety of the public. The NRC will determine the adequacy of that expected response, in combination with the utility's compensating measures, on a case-by-case basis,					

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	subject to the following guidance. In addressing the circumstance where applicant's inability to comply with the requirements of paragraph (b) of this section is wholly or substantially the result of non-participation of state and/or local governments, it may be presumed that in the event of an actual radiological emergency state and local officials would generally follow the utility plan. However, this presumption may be rebutted by, for example, a good faith and timely proffer of an adequate and feasible state and/or local radiological emergency plan that would in fact be relied upon in a radiological emergency. (2) Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.					
50.47(d)	(d) Notwithstanding the requirements of paragraphs (a) and (b) of this section, and except as specified by this paragraph, no NRC or FEMA review, findings, or determinations concerning the state of offsite emergency preparedness or the adequacy of and capability to implement State and local or utility offsite emergency plans are required prior to issuance of an operating license authorizing only fuel loading or low power testing and training (up to 5 percent of the rated power). Insofar as emergency planning and preparedness requirements are concerned, a license authorizing fuel loading and/or low power testing and training may be issued after a finding is made by the NRC that the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The NRC will base this finding on its assessment of the applicant's onsite emergency plans against the pertinent standards in paragraph (b) of this section and appendix E. Review of applicant's emergency plans will include the following standards with offsite aspects: (following 7 requirements) (1) Arrangements for requesting and effectively using offsite assistance on site have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned onsite response have been identified.					

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	(2) Procedures have been established for licensee communications with State and local response organizations, including initial notification of the declaration of emergency and periodic provision of plant and response status reports. (3) Provisions exist for prompt communications among principal response organizations to offsite emergency personnel who would be responding onsite. (4) Adequate emergency facilities and equipment to support the emergency response onsite are provided and maintained. (5) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use onsite. (6) Arrangements are made for medical services for contaminated and injured onsite individuals. (7) Radiological emergency response training has been made available to those offsite who may be called to assist in an emergency onsite.					
50.47(e)	(e) Notwithstanding the requirements of paragraph (b) of this section and the provisions of § 52.103 of this chapter, a holder of a combined license under part 52 of this chapter may not load fuel or operate except as provided in accordance with appendix E to part 50 and § 50.54(gg).					
50.48	Fire protection.					
50.48(a)	(a)(1) Each holder of an operating license issued under this part or a combined license issued under part 52 of this chapter must have a fire protection plan that satisfies Criterion 3 of appendix A to this part. This fire protection plan must: (i) Describe the overall fire protection program for the facility; (ii) Identify the various positions within the licensee's organization that are responsible for the program; (iii) State the authorities that are delegated to each of these positions to implement those responsibilities; and					

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	(iv) Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. (2) The plan must also describe specific features necessary to implement the program described in paragraph (a)(1) of this section such as— (i) Administrative controls and personnel requirements for fire prevention and manual fire suppression activities; (ii) Automatic and manually operated fire detection and suppression systems; and (iii) The means to limit fire damage to structures, systems, or components important to safety so that the capability to shut down the plant safely is ensured. (3) The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the reactor license. The licensee shall retain each superseded revision of the procedures for 3 years from the date it was superseded. (4) Each applicant for a design approval, design certification, or manufacturing license under part 52 of this chapter must have a description and analysis of the fire protection design features for the standard plant necessary to demonstrate compliance with Criterion 3 of appendix A to this part.					
50.48(b)	(b) Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate before January 1, 1979. (following 2 requirements) (1) Except for the requirements of Sections III.G, III.J, and III.O, the provisions of Appendix R to this part do not apply to nuclear power plants licensed to operate before January 1, 1979, to the extent that-- (i) Fire protection features proposed or implemented by the licensee have been accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position (BTP) APCSB 9.5-1 reflected in NRC fire protection safety evaluation reports issued before the effective date of					

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	February 19, 1981; or (ii) Fire protection features were accepted by the NRC staff in comprehensive fire protection safety evaluation reports issued before Appendix A to Branch Technical Position (BTP) APCSB 9.5-1 was published in August 1976. (2) With respect to all other fire protection features covered by Appendix R, all nuclear power plants licensed to operate before January 1, 1979, must satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O.					
50.48(c)	(c) *National Fire Protection Association Standard NFPA 805.* (following 4 requirements) (1) *Approval of incorporation by reference.* National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NFPA 805), which is referenced in this section, was approved for incorporation by reference by the Director of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. Copies of NFPA 805 may be purchased from the NFPA Customer Service Department, 1 Batterymarch Park, P.O. Box 9101, Quincy, MA 02269-9101 and in PDF format through the NFPA Online Catalog (*http://www.nfpa.org*) or by calling 1-800-344-3555 or (617) 770-3000. Copies are also available for inspection at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852-2738, and at the NRC Public Document Room, Building One White Flint North, Room O1-F15, 11555 Rockville Pike, Rockville, Maryland 20852-2738. Copies are also available at the National Archives and Records Administrative (NARA). For information on the availability of this material at NARA, call (202) 741-6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html.* (2) *Exceptions, modifications, and supplementation of NFPA 805.* As used in this section, references to NFPA 805 are to the 2001 Edition, with the following exceptions, modifications, and supplementation: (i) *Life Safety Goal, Objectives, and Criteria.* The Life Safety Goal, Objectives, and Criteria of Chapter 1 are not endorsed. (ii) *Plant Damage/Business Interruption Goal, Objectives, and Criteria.* The Plant Damage/Business Interruption Goal, Objectives, and Criteria of Chapter 1 are not endorsed.					

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	(iii) *Use of feed-and-bleed.* In demonstrating compliance with the performance criteria of Sections 1.5.1(b) and (c), a high-pressure charging/injection pump coupled with the pressurizer power-operated relief valves (PORVs) as the sole fire-protected safe shutdown path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability (i.e., feed-and-bleed) for pressurized-water reactors (PWRs) is not permitted. (iv) *Uncertainty analysis.* An uncertainty analysis performed in accordance with Section 2.7.3.5 is not required to support deterministic approach calculations. (v) *Existing cables.* In lieu of installing cables meeting (vi) *Water supply and distribution.* The italicized exception to Section 3.6.4 is not endorsed. Licensees who wish to use the exception to Section 3.6.4 must submit a request for a license amendment in accordance with paragraph (c)(2)(vii) of this section. (vii) *Performance-based methods.* Notwithstanding the prohibition in Section 3.1 against the use of performance-based methods, the fire protection program elements and minimum design requirements of Chapter 3 may be subject to the performance-based methods permitted elsewhere in the standard. Licensees who wish to use performance-based methods for these fire protection program elements and minimum design requirements shall submit a request in the form of an application for license amendment under § 50.90. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the performance-based approach; (A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (B) Maintains safety margins; and (C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability). (3) *Compliance with NFPA 805.*					

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	(i) A licensee may maintain a fire protection program that complies with NFPA 805 as an alternative to complying with paragraph (b) of this section for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979. The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under § 50.90. The application must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the licensee has identified orders, license conditions, and the technical specifications that must be revised or superseded, and that any necessary revisions are adequate. Any approval by the Director or the designee must be in the form of a license amendment approving the use of NFPA 805 together with any necessary revisions to the technical specifications. (ii) The licensee shall complete its implementation of the methodology in Chapter 2 of NFPA 805 (including all required evaluations and analyses) and, upon completion, modify the fire protection plan required by paragraph (a) of this section to reflect the licensee's decision to comply with NFPA 805, before changing its fire protection program or nuclear power plant as permitted by NFPA 805. (4) *Risk-informed or performance-based alternatives to compliance with NFPA 805.* A licensee may submit a request to use risk-informed or performance-based alternatives to compliance with NFPA 805. The request must be in the form of an application for license amendment under § 50.90 of this chapter. The Director of the Office of Nuclear Reactor Regulation, or designee of the Director, may approve the application if the Director or designee determines that the proposed alternatives: (i) Satisfy the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (ii) Maintain safety margins; and (iii) Maintain fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).					
50.48(d)	(d) [Reserved].	NA				Exclude

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50.48(e)	(e) [Reserved].	NA				Exclude
50.48(f)	(f) Licensees that have submitted the certifications required under § 50.82(a)(1) shall maintain a fire protection program to address the potential for fires that could cause the release or spread of radioactive materials (i.e., that could result in a radiological hazard). A fire protection program that complies with NFPA 805 shall be deemed to be acceptable for complying with the requirements of this paragraph. (following 3 requirements) (1) The objectives of the fire protection program are to-- (i) Reasonably prevent these fires from occurring; (ii) Rapidly detect, control, and extinguish those fires that do occur and that could result in a radiological hazard; and (iii) Ensure that the risk of fire-induced radiological hazards to the public, environment and plant personnel is minimized. (2) The licensee shall assess the fire protection program on a regular basis. The licensee shall revise the plan as appropriate throughout the various stages of facility decommissioning. (3) The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.					
50.49	Environmental qualification of electric equipment important to safety for nuclear power plants.					
50.49(a)	(a) Each holder of or an applicant for an operating license issued under this part, or a combined license or manufacturing license issued under part 52 of this chapter, other than a nuclear power plant for which the certifications required under § 50.82(a)(1) or § 52.110(a)(1) of this chapter have been submitted, shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section. For a manufacturing license, only electric equipment defined in paragraph (b) which is within the scope of the manufactured reactor must be included in the program.					
50.49(b)	(b) Electric equipment important to safety covered by this section is:					

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	(following 3 requirements) (1) Safety-related electric equipment.(1) (i) This equipment is that relied upon to remain functional during and following design basis events to ensure-- (A) The integrity of the reactor coolant pressure boundary; (B) The capability to shut down the reactor and maintain it in a safe shutdown condition; or (C) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable. (ii) Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (b)(1)(i) (A) through (C) of this section. NOTE: (1) Safety-related electric equipment is referred to as "Class 1E" equipment in IEEE 323-1974. Copies of this standard may be obtained from the Institute of Electrical and Electronics. (2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (b)(1)(i) (A) through (C) of paragraph (b)(1) of this section by the safety-related equipment. (3) Certain post-accident monitoring equipment.					

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	NOTE: Specific guidance concerning the types of variables to be monitored is provided in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Copies of the Regulatory Guide may be purchased through the U.S. Government Printing Office by calling 202-275-2060 or by writing to the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082.					
50.49(c)	(c) Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time is significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.					
50.49((d)	(d) The applicant or licensee shall prepare a list of electric equipment important to safety covered by this section. In addition, the applicant or licensee shall include the information in paragraphs (d)(1), (2), and (3) of this section for this electric equipment important to safety in a qualification file. The applicant or licensee shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment is important to safely meet the requirements of paragraph (j) of this section. (1) The performance specifications under conditions existing during and following design basis accidents. (2) The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured. (3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.					
50.49(e)	(e) The electric equipment qualification program must include and be based on the following: (following 8 requirements)					

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	(1) *Temperature and pressure*. The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional. (2) *Humidity*. Humidity during design basis accidents must be considered. (3) *Chemical effects*. The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed. (4) *Radiation*. The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects. (5) *Aging*. Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life. (6) *Submergence* (if subject to being submerged). (7) *Synergistic effects*. Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance. (8) *Margins*. Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment					

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	unless these conservatisms can be quantified and shown to contain appropriate margins.					
50.49(f)	(f) Each item of electric equipment important to safety must be qualified by one of the following methods: (following 4 requirements) (1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.(2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable. (3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable. (4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.					
50.49(g)	(g) Each holder of an operating license issued prior to February 22, 1983, shall, by May 20, 1983, identify the electric equipment important to safety within the scope of this section already qualified and submit a schedule for either the qualification to the provisions of this section or for the replacement of the remaining electric equipment important to safety within the scope of this section. This schedule must establish a goal of final environmental qualification of the electric equipment within the scope of this section by the end of the second refueling outage after March 31, 1982 or by March 31, 1985, whichever is earlier. The Director of the Office of Nuclear Reactor Regulation may grant requests for extensions of this deadline to a date no later than November 30, 1985, for specific pieces of equipment if these requests are filed on a timely basis and demonstrate good cause for the extension, such as procurement lead time, test complications, and installation problems. In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification. The schedule in this paragraph supersedes the June 30, 1982, deadline, or any other previously imposed date, for environmental qualification of electric equipment contained in certain nuclear power operating licenses.					
50.49(h)	(h) Each license shall notify the Commission as specified in § 50.4 of any significant equipment qualification problem that may require extension of the completion date provided in accordance with paragraph (g) of this section within 60 days of its discovery.					
50.49(i)	(i) Applicants for operating licenses granted after February 22, 1983, but prior to November 30,					

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	1985, shall perform an analysis to ensure that the plant can be safely operated pending completion of equipment qualification required by this section. This analysis must be submitted, as specified in § 50.4, for consideration prior to the granting of an operating license and must include, where appropriate, consideration of: (following 5 requirements) (1) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified. (2) The validity of partial test data in support of the original qualification. (3) Limited use of Administrative controls over equipment that has not been demonstrated to be fully qualified. (4) Completion of the safety function prior to exposure to the accident environment resulting from a design basis event and ensuring that the subsequent failure of the equipment does not degrade any safety function or mislead the operator. (5) No significant degradation of any safety function or misleading information to the operator as a result of failure of equipment under the accident environment resulting from a design basis event.					
50.49(j)	(j) A record of the qualification, including documentation in paragraph (d) of this section, must be maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment important to safety covered by this section: (following 2 requirements) (1) Is qualified for its application; and (2) Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.					
50.49(k)	(k) Applicants for and holders of operating licenses are not required to requalify electric equipment important to safety in accordance with the provisions of this section if the Commission has previously required qualification of that equipment in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979 (DOR Guidelines), or NUREG-0588 (For Comment version), "Interim Staff					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Position on Environmental Qualification of Safety-Related Electrical Equipment."					
50.49(l)	(l) Replacement equipment must be qualified in accordance with the provisions of this section unless there are sound reasons to the contrary.					
	Issuance, Limitations, and Conditions of Licenses and Construction Permits					
50.50	Issuance of licenses and construction permits.					Exclude Administrative
50.51	Continuation of license.					Exclude Administrative
50.52	Combining licenses.					Exclude Administrative
50.53	Jurisdictional limitations.					Exclude Administrative
50.54	Conditions of licenses.					Exclude Administrative
50.55	Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses.					Exclude; Administrative
50.55a	Codes and standards. Each construction permit for a utilization facility is subject to the following conditions in addition to those specified in § 50.55. Each combined license for a utilization facility is subject to the following conditions in addition to those specified in § 50.55, except that each combined license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section, but only after the Commission makes the finding under § 52.103(g) of this chapter. Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section in addition to those specified in § 50.55. Each manufacturing license, standard design approval, and standard design certification application under part 52 of this chapter is subject to the conditions in paragraphs (a), (b)(1), (b)(4), (c), (d), (e), (f)(3), and (g)(3) of this section. ***[Refer to the full text of 10 CFR 50.55a for detailed criteria.]***					
50.56	Conversion of construction permit to license; or amendment of license.					Exclude Administrative
50.57	Issuance of operating license.					Exclude Administrative

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
50.58	Hearings and report of the Advisory Committee on Reactor Safeguards.					Exclude Administrative
50.59	Changes, tests and experiments.					Exclude Administrative
50.60	Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.					Although this section appears to be applicable to LWRs only, it should be evaluated for guidance or partial applicability.
50.60(a)	(a) Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.					
50.60(b)	(b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12.					
50.61	Fracture toughness requirements for protection against pressurized thermal shock events.					
	Refer to the 10 CFR 50.61a document for detailed criteria.					
50.61a	Alternate fracture toughness requirements for protection against pressurized thermal shock events. ***[Refer to the full text of 10 CFR 50.61a for the detailed criteria.]***					
50.62	Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.					Although this section appears to be applicable to LWRs only, it should be evaluated for guidance or partial applicability.
50.62(a)	(a) *Applicability*. The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted.					
50.62(b)	(b) *Definition*. For purposes of this section, *Anticipated Transient Without Scram* (ATWS) means an anticipated operational occurrence as defined in appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	appendix A of this part.					
50.62(c)	(c) *Requirements*. (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system. (2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods). (3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device. (4) Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature. (5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.					

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	(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Commission as specified in § 50.4.					
50.62(d)	(d) *Implementation*. For each light-water-cooled nuclear power plant operating license issued before September 27, 2007, by 180 days after the issuance of the QA guidance for non-safety related components, each licensee shall develop and submit to the Commission, as specified in § 50.4, a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee. For each light-water-cooled nuclear power plant operating license application submitted after September 27, 2007, the applicant shall submit information in its final safety analysis report demonstrating how it will comply with paragraphs (c)(1) through (c)(5) of this section.					
50.63	Loss of all alternating current power.					
50.63(a)	(a) *Requirements*. (1) Each light-water-cooled nuclear power plant licensed to operate under this part, each light-water-cooled nuclear power plant licensed under subpart C of 10 CFR part 52 after the Commission makes the finding under § 52.103(g) of this chapter, and each design for a light-water-cooled nuclear power plant approved under a standard design approval, standard design certification, and manufacturing license under part 52 of this chapter must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors: (i) The redundancy of the onsite emergency ac power sources; (ii) The reliability of the onsite emergency ac power sources; (iii) The expected frequency of loss of offsite power; and (iv) The probable time needed to restore offsite power. (a)(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and					

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	capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Licensees are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC .					
50.63(b)	(b) *Limitation of scope*. Paragraph (c) of this section does not apply to those plants licensed to operate prior to *July 21, 1988*, if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.					
50.63(c)	(c) *Implementation*. (1) *Information Submittal*. For each light-water-cooled nuclear power plant licensed to operate on or before July 21, 1988, the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by April 17, 1989. For each light-water-cooled nuclear power plant licensed to operate after July 21, 1988, but before September 27, 2007, the licensee shall submit the information defined in this section to the Director of the Office of Nuclear Reactor Regulation, by 270 days after the date of license issuance. For each light-water-cooled nuclear power plant operating license application submitted after September 27, 2007, the applicant shall submit the information defined below in its final safety analysis report. (i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section; (ii) A description of the procedures that will be implemented for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom; and (iii) A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications. (2) *Alternate ac source*: The alternate ac power source(s), as defined in § 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time					

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	required for startup and alignment of the alternate ac power source(s) and this equipment shall be demonstrated by test. Alternate ac source(s) serving a multiple unit site where onsite emergency ac sources are not shared between units must have, as a minimum, the capacity and capability for coping with a station blackout in any of the units. At sites where onsite emergency ac sources are shared between units, the alternate ac source(s) must have the capacity and capability as required to ensure that all units can be brought to and maintained in safe shutdown (non-DBA) as defined in § 50.2. If the alternate ac source(s) meets the above requirements and can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout, then no coping analysis is required. *(3) Regulatory Assessment*: After consideration of the information submitted in accordance with paragraph (c)(1) of this section, the Director, Office of Nuclear Reactor Regulation, will notify the licensee of the Director's conclusions regarding the adequacy of the proposed specified station blackout duration, the proposed equipment modifications and procedures, and the proposed schedule for implementing the procedures and modifications for compliance with paragraph (a) this section. (c)(4) *Implementation Schedule*: For each light-water-cooled nuclear power plant licensed to operate on or before June 21, 1988, the licensee shall, within 30 days of the notification provided in accordance with paragraph (c)(3) of this section, submit to the Director of the Office of Nuclear Reactor Regulation a schedule commitment for implementing any equipment and associated procedure modifications necessary to meet the requirements of paragraph (a) of this section. This submittal must include an explanation of the schedule and a justification if the schedule does not provide for completion of the modifications within two years of the notification provided in accordance with paragraph (c)(3) of this section. A final schedule for implementing modifications necessary to comply with the requirements of paragraph (a) of this section will be established by the NRC staff in consultation and coordination with the affected licensee.					
50.64	Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors.	No				Exclude, Not applicable to the HTGR.
50.65	Requirements for monitoring the effectiveness of maintenance at nuclear power plants.					
50.65(a)	(a)(1) Each holder of an operating license for a nuclear power plant under this part and each holder of a combined license under part 52 of this chapter after the Commission makes the finding under § 52.103(g) of this chapter, shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide					

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	reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industry wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in § 50.82(a)(1) or 52.110(a)(1) of this chapter, as applicable, this section shall only apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions. (2) Monitoring as specified in paragraph (a)(1) of this section is not required where it has been demonstrated that the performance or condition of a structure, system, or component is being effectively controlled through the performance of appropriate preventive maintenance, such that the structure, system, or component remains capable of performing its intended function. (3) Performance and condition monitoring activities and associated goals and preventive maintenance activities shall be evaluated at least every refueling cycle provided the interval between evaluations does not exceed 24 months. The evaluations shall take into account, where practical, industry-wide operating experience. Adjustments shall be made where necessary to ensure that the objective of preventing failures of structures, systems, and components through maintenance is appropriately balanced against the objective of minimizing unavailability of structures, systems, and components due to monitoring or preventive maintenance. (4) Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety.					
50.65(b)	(b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety-related structures, systems, and components, as follows: (b)(1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant					

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	pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in Sec. 50.34(a)(1), Sec. 50.67(b)(2), or Sec. 100.11 of this chapter, as applicable. (b)(2) Nonsafety related structures, systems, or components: (i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or (ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or (iii) Whose failure could cause a reactor scram or actuation of a safety-related system.					
50.65(c)	(c) The requirements of this section shall be implemented by each licensee no later than July 10, 1996.					
50.66	Requirements for thermal annealing of the reactor pressure vessel.					
50.66(a)	(1) For those light water nuclear power reactors where neutron radiation has reduced the fracture toughness of the reactor vessel materials, a thermal annealing may be applied to the reactor vessel to recover the fracture toughness of the material. The use of a thermal annealing treatment is subject to the requirements in this section. A report describing the licensee's plan for conducting the thermal annealing must be submitted in accordance with § 50.4 at least three years prior to the date at which the limiting fracture toughness criteria in § 50.61 or appendix G to part 50 would be exceeded. Within three years of the submittal of the Thermal Annealing Report and at least thirty days prior to the start of the thermal annealing, the NRC will review the Thermal Annealing Report and make available the results of its evaluation at the NRC Web site, *http://www.nrc.gov*. The licensee may begin the thermal anneal after: Submitting the Thermal Annealing Report required by paragraph (b) of this section; The NRC makes available the results of its evaluation of the Thermal Annealing Report at the NRC Web site, *http://www.nrc.gov;* and					

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	The requirements of paragraph (f)(1) of this section have been satisfied.					
50.66(b)	*(b) Thermal Annealing Report*. The Thermal Annealing Report must include: a Thermal Annealing Operating Plan; a Requalification Inspection and Test Program; a Fracture Toughness Recovery and Reembrittlement Trend Assurance Program; and Identification of Unreviewed Safety Questions and Technical Specification Changes. (1) Thermal Annealing Operating Plan. The thermal annealing operating plan must include: (i) A detailed description of the pressure vessel and all structures and components that are expected to experience significant thermal or stress effects during the thermal annealing operation; (ii) An evaluation of the effects of mechanical and thermal stresses and temperatures on the vessel, containment, biological shield, attached piping and appurtenances, and adjacent equipment and components to demonstrate that operability of the reactor will not be detrimentally affected. This evaluation must include: (A) Detailed thermal and structural analyses to establish the time and temperature profile of the annealing operation. These analyses must include heatup and cooldown rates, and must demonstrate that localized temperatures, thermal stress gradients, and subsequent residual stresses will not result in unacceptable dimensional changes or distortions in the vessel, attached piping and appurtenances, and that the thermal annealing cycle will not result in unacceptable degradation of the fatigue life of these components. (B) The effects of localized high temperatures on degradation of the concrete adjacent to the vessel and changes in thermal and mechanical properties, if any, of the reactor vessel insulation, and on detrimental effects, if any, on containment and the biological shield. If the design temperature limitations for the adjacent concrete structure are to be exceeded during the thermal annealing operation, an acceptable maximum temperature for the concrete must be established for the annealing operation using appropriate test data. (iii) The methods, including heat source, instrumentation and procedures proposed for performing					

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	the thermal annealing. This shall include any special precautions necessary to minimize occupational exposure, in accordance with the As Low As Reasonably Achievable (ALARA) principle and the provisions of § 20.1206. (iv) The proposed thermal annealing operating parameters, including bounding conditions for temperatures and times, and heatup and cooldown schedules. (A) The thermal annealing time and temperature parameters selected must be based on projecting sufficient recovery of fracture toughness, using the procedures of paragraph (e) of this section, to satisfy the requirements of § 50.60 and § 50.61 for the proposed period of operation addressed in the application. (B) The time and temperature parameters evaluated as part of the thermal annealing operating plan, and supported by the evaluation results of paragraph (b)(1)(ii) of this section, represent the bounding times and temperatures for the thermal annealing operation. If these bounding conditions for times and temperatures are violated during the thermal annealing operation, then the annealing operation is considered not in accordance with the Thermal Annealing Operating Plan, as required by paragraph (c)(1) of this section, and the licensee must comply with paragraph (c)(2) of this section. (2) Requalification Inspection and Test Program. The inspection and test program to requalify the annealed reactor vessel must include the detailed monitoring, inspections, and tests proposed to demonstrate that the limitations on temperatures, times and temperature profiles, and stresses evaluated for the proposed thermal annealing conditions of paragraph (b)(1)(iv) of this section have not been exceeded, and to determine the thermal annealing time and temperature to be used in quantifying the fracture toughness recovery. The requalification inspection and test program must demonstrate that the thermal annealing operation has not degraded the reactor vessel, attached piping or appurtenances, or the adjacent concrete structures to a degree that could affect the safe operation of the reactor. (3) Fracture Toughness Recovery and Reembrittlement Trend Assurance Program. The percent recovery of RT_{NDT} and Charpy upper-shelf energy due to the thermal annealing treatment must be determined based on the time and temperature of the actual vessel thermal anneal. The recovery of RT_{NDT} and Charpy upper-shelf energy provide the basis for establishing the post-anneal RT_{NDT} and Charpy upper-shelf energy for each vessel material. Changes in the RT_{NDT} and					

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	Charpy upper-shelf energy with subsequent plant operation must be determined using the post-anneal values of these parameters in conjunction with the projected reembrittlement trend determined in accordance with paragraph (b)(3)(ii) of this section. Recovery and reembrittlement evaluations shall include: (i) Recovery Evaluations. (A) The percent recovery of both RT_{NDT} and Charpy upper-shelf energy must be determined by one of the procedures described in paragraph (e) of this section, using the proposed lower bound thermal annealing time and temperature conditions described in the operating plan. (B) If the percent recovery is determined from testing surveillance specimens or from testing materials removed from the reactor vessel, then it shall be demonstrated that the proposed thermal annealing parameters used in the test program are equal to or bounded by those used in the vessel annealing operation. (C) If generic computational methods are used, appropriate justification must be submitted as a part of the application. (ii) Reembrittlement Evaluations. (A) The projected post-anneal reembrittlement of RT_{NDT} must be calculated using the procedures in § 50.61(c), or must be determined using the same basis as that used for the pre-anneal operating period. The projected change due to post-anneal reembrittlement for Charpy upper-shelf energy must be determined using the same basis as that used for the pre-anneal operating period. (B) The post-anneal reembrittlement trend of both RT_{NDT} and Charpy upper-shelf energy must be estimated, and must be monitored using a surveillance program defined in the Thermal Annealing Report and which conforms to the intent of Appendix H of this part, "Reactor Vessel Material Surveillance Program Requirements." (4) Identification of Unreviewed Safety Questions and Technical Specification Changes. Any changes to the facility as described in the updated final safety analysis report constituting unreviewed safety questions, and any changes to the technical specifications, which are necessary to either conduct the thermal annealing or operate the nuclear power reactor following the annealing, must be identified. The section shall demonstrate that the Commission's requirements continue to be complied with, and that there is reasonable assurance of adequate					

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	protection to the public health and safety following the changes.					
50.66(c)	*(c) Completion or Termination of Thermal Annealing*. (1) If the thermal annealing was completed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall so confirm in writing to the Director, Office of Nuclear Reactor Regulation. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met. (2) If the thermal annealing was completed but the annealing was not performed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall submit a summary of lack of compliance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program and a justification for subsequent operation to the Director, Office of Nuclear Reactor Regulation. Any changes to the facility as described in the updated final safety analysis report which are attributable to the noncompliances and constitute unreviewed safety questions, and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified. (i) If no unreviewed safety questions or changes to technical specifications are identified, the licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met. (ii) If any unreviewed safety questions or changes to technical specifications are identified, the licensee may not restart its reactor until approval is obtained from the Director, Office of Nuclear Reactor Regulation and the requirements of paragraph (f)(2) of this section have been met. (3) If the thermal annealing was terminated prior to completion, the licensee shall immediately notify the NRC of the premature termination of the thermal anneal. (i) If the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, and the licensee does not elect to take credit for any recovery, the licensee need not submit the Thermal Annealing Results Report required by paragraph (d) of this section but instead shall confirm in writing to the Director, Office of Nuclear Reactor Regulation that the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program. The licensee may restart its reactor					

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	after the requirements of paragraph (f)(2) of this section have been met. (ii) If the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, and the licensee elects to take full or partial credit for the partial annealing, the licensee shall confirm in writing to the Director, Office of Nuclear Reactor Regulation that the partial annealing was otherwise performed in compliance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met. (iii) If the partial annealing was not performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, the licensee shall submit a summary of lack of compliance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program and a justification for subsequent operation to the Director, Office of Nuclear Reactor Regulation. Any changes to the facility as described in the updated final safety analysis report which are attributable to the noncompliances and constitute unreviewed safety questions, and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified. (A) If no unreviewed safety questions or changes to technical specifications are identified, the licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met. (B) If any unreviewed safety questions or changes to technical specifications are identified, the licensee may not restart its reactor until approval is obtained from the Director, Office of Nuclear Reactor Regulation and the requirements of paragraph (f)(2) of this section have been met.					
50.66(d)	(d) *Thermal Annealing Results Report*. Every licensee that either completes a thermal annealing, or that terminates an annealing but elects to take full or partial credit for the annealing, shall provide the following information within three months of completing the thermal anneal, unless an extension is authorized by the Director, Office of Nuclear Reactor Regulation: (1) The time and temperature profiles of the actual thermal annealing; (2) The post-anneal RT_{NDT} and Charpy upper-shelf energy values of the reactor vessel materials for use in subsequent reactor operation;					

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	(3) The projected post-anneal reembrittlement trends for both RT_{NDT} and Charpy upper-shelf energy; and (4) The projected values of RT_{PTS} and Charpy upper-shelf energy at the end of the proposed period of operation addressed in the Thermal Annealing Report.					
50.66(e)	(e) *Procedures for Determining the Recovery of Fracture Toughness*. The procedures of this paragraph must be used to determine the percent recovery of RT_{NDT}, R_t, and percent recovery of Charpy upper-shelf energy, R_u. In all cases, R_t and R_u may not exceed 100. (1) For those reactors with surveillance programs which have developed credible surveillance data as defined in § 50.61, percent recovery due to thermal annealing (R_t and R_u) must be evaluated by testing surveillance specimens that have been withdrawn from the surveillance program and that have been annealed under the same time and temperature conditions as those given the beltline material. (2) Alternatively, the percent recovery due to thermal annealing (R_t and R_u) may be determined from the results of a verification test program employing materials removed from the beltline region of the reactor vessel[6] and that have been annealed under the same time and temperature conditions as those given the beltline material. (3) Generic computational methods may be used to determine recovery if adequate justification is provided.					
50.66(f)	(f) *Public information and participation*. (1) Upon receipt of a Thermal Annealing Report, and a minimum of 30 days before the licensee starts thermal annealing, the Commission shall: (i) Notify and solicit comments from local and State governments in the vicinity of the site where the thermal annealing will take place and any Indian Nation or other indigenous people that have treaty or statutory rights that could be affected by the thermal annealing, (ii) Publish a notice of a public meeting in the FEDERAL REGISTER and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, to solicit comments from the public, and (iii) Hold a public meeting on the licensee's Thermal Annealing Report.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(2) Within 15 days after the NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3)(i) through (iii) of this section, the NRC staff shall make available at the NRC Web site, *http://www.nrc.gov,* a summary of its inspection of the licensee's thermal annealing, and the Commission shall hold a public meeting: (i) For the licensee to explain to NRC and the public the results of the reactor pressure vessel annealing, (ii) for the NRC to discuss its inspection of the reactor vessel annealing, and (iii) for the NRC to receive public comments on the annealing. (3) Within 45 days of NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3)(i) through (iii) of this section, the NRC staff shall complete full documentation of its inspection of the licensee's annealing process and make available this documentation at the NRC Web site, *http://www.nrc.gov.*					
50.67	Accident source term.	No				Exclude, Applicable only to earlier plants.
50.68	Criticality accident requirements.					
50.68(a)	(a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part or a combined license for a nuclear power reactor issued under Part 52 of this chapter, shall comply with either 10 CFR 70.24 of this chapter or the requirements in paragraph (b) of this section.					
50.68(b)	(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24: (1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water. (2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be					

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	performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used. (3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used. (4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. (5) The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass. (6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions. (7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight. (8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b).					
50.68(c)	(c) While a spent fuel transportation package approved under Part 71 of this chapter or spent fuel storage cask approved under Part 72 of this chapter is in the spent fuel pool: (1) The requirements in § 50.68(b) do not apply to the fuel located within that package or cask; and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(2) The requirements in Part 71 or 72 of this chapter, as applicable, and the requirements of the Certificate of Compliance for that package or cask, apply to the fuel within that package or cask.					
	Inspections, Records, Reports, Notifications					
50.69	Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors.					
50.69(a)	(a)Definitions *Risk-Informed Safety Class (RISC)-1 structures, systems, and components (SSCs)* means safety-related SSCs that perform safety significant functions. *Risk-Informed Safety Class (RISC)-2 structures, systems and components (SSCs)* means nonsafety-related SSCs that perform safety significant functions. *Risk-Informed Safety Class (RISC)-3 structures, systems and components (SSCs)* means safety-related SSCs that perform low safety significant functions. *Risk-Informed Safety Class (RISC)-4 structures, systems and components (SSCs)* means nonsafety-related SSCs that perform low safety significant functions. *Safety significant function* means a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk.					
50.69(b)	(b) Applicability and scope of risk-informed treatment of SSCs and submittal/approval process. (1) A holder of a license to operate a light water reactor (LWR) nuclear power plant under this part; a holder of a renewed LWR license under part 54 of this chapter; an applicant for a construction permit or operating license under this part; or an applicant for a design approval, a combined license, or manufacturing license under part 52 of this chapter; may voluntarily comply with the requirements in this section as an alternative to compliance with the following requirements for					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	RISC-3 and RISC-4 SSCs: (i) 10 CFR part 21. (ii) The portion of 10 CFR 50.46a(b) that imposes requirements to conform to Appendix B to 10 CFR part 50. (iii) 10 CFR 50.49. (iv) 10 CFR 50.55(e). (v) The inservice testing requirements in 10 CFR 50.55a(f); the inservice inspection, and repair and replacement (with the exception of fracture toughness), requirements for ASME Class 2 and Class 3 SSCs in 10 CFR 50.55a(g); and the electrical component quality and qualification requirements in Section 4.3 and 4.4 of IEEE 279, and Sections 5.3 and 5.4 of IEEE 603-1991, as incorporated by reference in 10 CFR 50.55a(h). (vi) 10 CFR 50.65, except for paragraph (a)(4). (vii) 10 CFR 50.72. (viii) 10 CFR 50.73. (ix) Appendix B to 10 CFR part 50. (x) The Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR part 50, for penetrations and valves meeting the following criteria: (A) Containment penetrations that are either 1-inch nominal size or less, or continuously pressurized. (B) Containment isolation valves that meet one or more of the following criteria: *(1)* The valve is required to be open under accident conditions to prevent or mitigate core damage events;					

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	(2) The valve is normally closed and in a physically closed, water- filled system; *(3)* The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and is not connected to the reactor coolant pressure boundary; or *(4)* The valve is 1-inch nominal size or less. (xi) Appendix A to part 100, Sections VI(a)(1) and VI(a)(2), to the extent that these regulations require qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the Safe Shutdown Earthquake and Operating Basis Earthquake. (2) A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information: (i) A description of the process for categorization of RISC-1, RISC-2, RISC-3 and RISC-4 SSCs. (ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs. (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i). (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions). (3) The Commission will approve a licensee's implementation of this section if it determines that the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs satisfies the requirements of § 50.69(c) by issuing a license amendment approving the licensee's use of this					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
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	section. (4) An applicant choosing to implement this section shall include the information in § 50.69(b)(2) as part of application. The Commission will approve an applicant's implementation of this section if it determines that the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs satisfies the requirements of § 50.69(c).					
50.69(c)	(c) SSC Categorization Process. (1) SSCs must be categorized as RISC-1, RISC-2, RISC-3, or RISC-4 SSCs using a categorization process that determines if an SSC performs one or more safety significant functions and identifies those functions. The process must: (i) Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. (ii) Determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. (iii) Maintain defense-in-depth. (iv) Include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of § 50.69(b)(1) and (d)(2) are small. (v) Be performed for entire systems and structures, not for selected components within a system or structure. (2) The SSCs must be categorized by an Integrated Decision-Making Panel (IDP) staffed with					

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	expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.					
50.69(d)	(d) Alternative treatment requirements.--(1) RISC-1 and RISC 2 SSCs. The licensee or applicant shall ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance. (2) RISC-3 SSCs. The licensee or applicant shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. The treatment of RISC-3 SSCs must be consistent with the categorization process. Inspection and testing, and corrective action shall be provided for RISC-3 SSCs. (i) Inspection and testing. Periodic inspection and testing activities must be conducted to determine that RISC-3 SSCs will remain capable of performing their safety-related functions under design basis conditions; and (ii) Corrective action. Conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design basis conditions must be corrected in a timely manner. For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition.					
50.69(e)	(e) Feedback and process adjustment.--(1) RISC-1, RISC-2, RISC-3 and RISC-4 SSCs. The licensee shall review changes to the plant, operational practices, applicable plant and industry operational experience, and, as appropriate, update the PRA and SSC categorization and treatment processes. The licensee shall perform this review in a timely manner but no longer than once every two refueling outages. (2) RISC-1 and RISC-2 SSCs. The licensee shall monitor the performance of RISC-1 and RISC-2 SSCs. The licensee shall make adjustments as necessary to either the categorization or treatment processes so that the categorization process and results are maintained valid. (3) RISC-3 SSCs. The licensee shall consider data collected in § 50.69(d)(2)(i) for RISC-3 SSCs to determine if there are any adverse changes in performance such that the SSC unreliability					

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	values approach or exceed the values used in the evaluations conducted to satisfy § 50.69(c)(1)(iv). The licensee shall make adjustments as necessary to the categorization or treatment processes so that the categorization process and results are maintained valid.					
50.69(f)	(f) Program documentation, change control and records. (1) The licensee or applicant shall document the basis for its categorization of any SSC under paragraph (c) of this section before removing any requirements under § 50.69(b)(1) for those SSCs. (2) Following implementation of this section, licensees and applicants shall update their final safety analysis report (FSAR) to reflect which systems have been categorized, in accordance with § 50.71(e). (3) When a licensee first implements this section for a SSC, changes to the FSAR for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.59 evaluation of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the FSAR, may be made if the requirements of this section and § 50.59 continue to be met. (4) When a licensee first implements this section for a SSC, changes to the quality assurance plan for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.54(a) review of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the quality assurance plan may be made if the requirements of this section and § 50.54(a) continue to be met.					
50.69(g)	(g) *Reporting*. The licensee shall submit a licensee event report under § 50.73(b) for any event or condition that prevented, or would have prevented, a RISC-1 or RISC-2 SSC from performing a safety significant function.					
50.70	Inspections.					Exclude Administrative
50.71	Maintenance of records, making of reports.					
50.71(a)	(a) Each licensee, including each holder of a construction permit or early site permit, shall maintain all records and make all reports, in connection with the activity, as may be required by the conditions of the license or permit or by the regulations, and orders of the Commission in effectuating the purposes of the Act, including Section 105 of the Act, and the Energy Reorganization Act of 1974, as amended. Reports must be submitted in accordance with § 50.4 or 10 CFR 52.3, as applicable.					

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50.71(b)	(b) With respect to any production or utilization facility of a type described in § 50.21(b) or 50.22, or a testing facility, each licensee and each holder of a construction permit shall submit its annual financial report, including the certified financial statements, to the Commission, as specified in § 50.4, upon issuance of the report. However, licensees and holders of a construction permit who submit a Form 10-Q with the Securities and Exchange Commission or a Form 1 with the Federal Energy Regulatory Commission, need not submit the annual financial report or the certified financial statement under this paragraph.					
50.71(c)	(c) Records that are required by the regulations in this part or part 52 of this chapter, by license condition, or by technical specifications must be retained for the period specified by the appropriate regulation, license condition, or technical specification. If a retention period is not otherwise specified, these records must be retained until the Commission terminates the facility license or, in the case of an early site permit, until the permit expires.					
50.71(d)	(d)(1) Records which must be maintained under this part or part 52 of this chapter may be the original or a reproduced copy or microform if the reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of producing a clear and legible copy after storage for the period specified by Commission regulations. The record may also be stored in electronic media with the capability of producing legible, accurate, and complete records during the required retention period. Records such as letters, drawings, and specifications, must include all pertinent information such as stamps, initials, and signatures. The licensee shall maintain adequate safeguards against tampering with, and loss of records. (2) If there is a conflict between the Commission's regulations in this part, license condition, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in this part for such records shall apply unless the Commission, pursuant to § 50.12 of this part, has granted a specific exemption from the record retention requirements specified in the regulations in this part.					
50.71(e)	(e) Each person licensed to operate a nuclear power reactor under the provisions of § 50.21 or § 50.22, and each applicant for a combined license under part 52 of this chapter, shall update periodically, as provided in paragraphs (e) (3) and (4) of this section, the final safety analysis report (FSAR) originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the applicant or licensee pursuant to					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
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	Commission requirement since the submittal of the original FSAR, or as appropriate, the last update to the FSAR under this section. The submittal shall include the effects [1] of all changes made in the facility or procedures as described in the FSAR; all safety analyses and evaluations performed by the applicant or licensee either in support of approved license amendments or in support of conclusions that changes did not require a license amendment in accordance with § 50.59(c)(2) or, in the case of a license that references a certified design, in accordance with § 52.98(c) of this chapter; and all analyses of new safety issues performed by or on behalf of the applicant or licensee at Commission request. The updated information shall be appropriately located within the update to the FSAR. (1) The licensee shall submit revisions containing updated information to the Commission, as specified in § 50.4, on a replacement-page basis that is accompanied by a list which identifies the current pages of the FSAR following page replacement. (2) The submittal shall include (i) a certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement, or that no such changes were made; and (ii) an identification of changes made under the provisions of § 50.59 but not previously submitted to the Commission. (3)(i) A revision of the original FSAR containing those original pages that are still applicable plus new replacement pages shall be filed within 24 months of either July 22, 1980, or the date of issuance of the operating license, whichever is later, and shall bring the FSAR up to date as of a maximum of 6 months prior to the date of filing the revision. (ii) Not less than 15 days before § 50.71(e) becomes effective, the Director of the Office of Nuclear Reactor Regulation shall notify by letter the licensees of those nuclear power plants initially subject to the NRC's systematic evaluation program that they need not comply with the provisions of this section while the program is being conducted at their plant. The Director of the Office of Nuclear Reactor Regulation will notify by letter the licensee of each nuclear power plant being evaluated when the systematic evaluation program has been completed. Within 24 months after receipt of this notification, the licensee shall file a complete FSAR which is up to date as of a maximum of 6 months prior to the date of filing the revision.					

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	(iii) During the period from the docketing of an application for a combined license under subpart C of part 52 of this chapter until the Commission makes the finding under § 52.103(g) of this chapter, the update to the FSAR must be submitted annually. (4) Subsequent revisions must be filed annually or 6 months after each refueling outage provided the interval between successive updates does not exceed 24 months. The revisions must reflect all changes up to a maximum of 6 months prior to the date of filling. For nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1), subsequent revisions must be filed every 24 months. (5) Each replacement page shall include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both). (6) The updated FSAR shall be retained by the licensee until the Commission terminates their license.					
50.71(f)	(f) Each person licensed to manufacture a nuclear power reactor under subpart F of 10 CFR part 52 shall update the FSAR originally submitted as part of the application to reflect any modification to the design that is approved by the Commission under § 52.171 of this chapter, and any new analyses of the design performed by or on behalf of the licensee at the NRC's request. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee with respect to the modification approved under § 52.171 of this chapter or the analyses requested by the Commission under § 52.171 of this chapter. The updated information shall be appropriately located within the update to the FSAR.					
50.71(g)	(g) The provisions of this section apply to nuclear power reactor licensees that have submitted the certification of permanent cessation of operations required under §§ 50.82(a)(1)(i) or 52.110(a)(1) of this chapter. The provisions of paragraphs (a), (c), and (d) of this section also apply to non-power reactor licensees that are no longer authorized to operate.					
50.71(h)	(h)(1) No later than the scheduled date for initial loading of fuel, each holder of a combined license under subpart C of 10 CFR part 52 shall develop a level 1 and a level 2 probabilistic risk assessment (PRA). The PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist one year prior to the scheduled date for initial loading of fuel.					

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	(2) Each holder of a combined license shall maintain and upgrade the PRA required by paragraph (h)(1) of this section. The upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade. The PRA must be upgraded every four years until the permanent cessation of operations under § 52.110(a) of this chapter. (3) Each holder of a combined license shall, no later than the date on which the licensee submits an application for a renewed license, upgrade the PRA required by paragraph (h)(1) of this section to cover all modes and all initiating events.					
50.72	Immediate notification requirements for operating nuclear power reactors.					
50.72(a)	(a) Reportable events. (1) The holder of an operating license under this part or a combined license under part 52 of this chapter (after the Commission has made the finding under § 52.103(g) of this chapter) for a nuclear power plant (licensee) shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 60 days after the discovery of the event. In the case of an invalid actuation reported under § 50.73(a)(2)(iv), other than actuation of the reactor protection system (RPS) when the reactor is critical, the licensee may, at its option, provide a telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER. Unless otherwise specified in this section, the licensee shall report an event if it occurred within 3 years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event. (2) The licensee shall report: (i)(A) The completion of any nuclear plant shutdown required by the plant's Technical Specifications. (B) Any operation or condition which was prohibited by the plant's Technical Specifications except when: (1) The Technical Specification is administrative in nature;					

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	(2) The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or (3) The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event. (C) Any deviation from the plant's Technical Specifications authorized pursuant to Sec. 50.54(x) of this part. (ii) Any event or condition that resulted in: (A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or (B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety. (iii) Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant. (iv)(A) Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section, except when: (1) The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or (2) The actuation was invalid and; (i) Occurred while the system was properly removed from service; or (ii) Occurred after the safety function had been already completed. (B) The systems to which the requirements of paragraph (a)(2)(iv)(A) of this section apply are:					

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	(1) Reactor protection system (RPS) including: reactor scram or reactor trip. (2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). (3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems. (4) ECCS for boiling water reactors (BWRs) including: high-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system. (5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system. (6) PWR auxiliary or emergency feedwater system. (7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems. (8) Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs. (9) Emergency service water systems that do not normally run and that serve as ultimate heat sinks. (v) Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat;					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident. (vi) Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function. (vii) Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident. (viii)(A) Any airborne radioactive release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration limits specified in appendix B to part 20, table 2, column 1. (B) Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentrations specified in appendix B to part 20, table 2, column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases. (ix)(A) Any event or condition that as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to:					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(1) Shut down the reactor and maintain it in a safe shutdown condition; (2) Remove residual heat; (3) Control the release of radioactive material; or (4) Mitigate the consequences of an accident. (B) Events covered in paragraph (a)(2)(ix)(A) of this section may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, licensees are not required to report an event pursuant to paragraph (a)(2)(ix)(A) of this section if the event results from: (1) A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or (2) Normal and expected wear or degradation. (x) Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.					
50.72(b)	(b) Contents. The Licensee Event Report shall contain: (1) A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence. (2)(i) A clear, specific, narrative description of what occurred so that knowledgeable readers conversant with the design of commercial nuclear power plants, but not familiar with the details of a particular plant, can understand the complete event. (ii) The narrative description must include the following specific information as appropriate for the particular event:					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	(A) Plant operating conditions before the event. (B) Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event. (C) Dates and approximate times of occurrences. (D) The cause of each component or system failure or personnel error, if known. (E) The failure mode, mechanism, and effect of each failed component, if known. (F) The Energy Industry Identification System component function identifier and system name of each component or system referred to in the LER. (1) The Energy Industry Identification System is defined in: IEEE Std 803-1983 (May 16, 1983) Recommended Practice for Unique Identification in Power Plants and Related Facilities--Principles and Definitions. (2) IEEE Std 803-1983 has been approved for incorporation by reference by the Director of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51. (3) A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies may be obtained from the Institute of Electrical and Electronics Engineers, 445 Hoes Lane, P.O. Box 1331, Piscataway, NJ 08855-1331. IEEE Std 803-1983 is available for inspection at the NRC's Technical Library, which is located in the Two White Flint North Building, 11545 Rockville Pike, Rockville, Maryland 20852-2738; or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202-741-6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html*. (G) For failures of components with multiple functions, include a list of systems or secondary functions that were also affected. (H) For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service.					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(I) The method of discovery of each component or system failure or procedural error. (J) For each human performance related root cause, the licensee shall discuss the cause(s) and circumstances. (K) Automatically and manually initiated safety system responses. (L) The manufacturer and model number (or other identification) of each component that failed during the event. (3) An assessment of the safety consequences and implications of the event. This assessment must include: (i) The availability of systems or components that could have performed the same function as the components and systems that failed during the event, and (ii) For events that occurred when the reactor was shutdown, the availability of systems or components that are needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident. (4) A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future. (5) Reference to any previous similar events at the same plant that are known to the licensee. (6) The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the plant's characteristics.					
50.72(c)	(c) *Supplemental information*. The Commission may require the licensee to submit specific additional information beyond that required by paragraph (b) of this section if the Commission finds that supplemental material is necessary for complete understanding of an unusually complex or significant event. These requests for supplemental information will be made in writing and the licensee shall submit, as specified in § 50.4, the requested information as a supplement					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	to the initial LER.					
50.72(d)	(d) *Submission of reports*. Licensee Event Reports must be prepared on Form NRC 366 and submitted to the U.S. Nuclear Regulatory Commission, as specified in § 50.4.					
50.72(e)	(e) *Report legibility*. The reports and copies that licensees are required to submit to the Commission under the provisions of this section must be of sufficient quality to permit legible reproduction and micrographic processing.					
50.72(f)	(f) [Reserved]					
50.72(g)	(g) *Reportable occurrences.* The requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications.					
50.73	License event report system.					
50.73(a)	(a) Reportable events. (1) The holder of an operating license under this part or a combined license under part 52 of this chapter (after the Commission has made the finding under § 52.103(g) of this chapter) for a nuclear power plant (licensee) shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 60 days after the discovery of the event. In the case of an invalid actuation reported under § 50.73(a)(2)(iv), other than actuation of the reactor protection system (RPS) when the reactor is critical, the licensee may, at its option, provide a telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER. Unless otherwise specified in this section, the licensee shall report an event if it occurred within 3 years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event. (2) The licensee shall report: (i)(A) The completion of any nuclear plant shutdown required by the plant's Technical Specifications. (B) Any operation or condition which was prohibited by the plant's Technical Specifications except when: (1) The Technical Specification is administrative in nature; (2) The event consisted solely of a case of a late surveillance test where the oversight was					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or (3) The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event. (C) Any deviation from the plant's Technical Specifications authorized pursuant to Sec. 50.54(x) of this part. (ii) Any event or condition that resulted in: (A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or (B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety. (iii) Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant. (iv)(A) Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section, except when: (1) The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or (2) The actuation was invalid and; (i) Occurred while the system was properly removed from service; or (ii) Occurred after the safety function had been already completed. (B) The systems to which the requirements of paragraph (a)(2)(iv)(A) of this section apply are:					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(1) Reactor protection system (RPS) including: reactor scram or reactor trip. (2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). (3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems. (4) ECCS for boiling water reactors (BWRs) including: high-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system. (5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system. (6) PWR auxiliary or emergency feedwater system. (7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems. (8) Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs. (9) Emergency service water systems that do not normally run and that serve as ultimate heat sinks. (v) Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat;					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident. (vi) Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function. (vii) Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident. (viii)(A) Any airborne radioactive release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration limits specified in appendix B to part 20, table 2, column 1. (B) Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentrations specified in appendix B to part 20, table 2, column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases. (ix)(A) Any event or condition that as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(1) Shut down the reactor and maintain it in a safe shutdown condition; (2) Remove residual heat; (3) Control the release of radioactive material; or (4) Mitigate the consequences of an accident. (B) Events covered in paragraph (a)(2)(ix)(A) of this section may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, licensees are not required to report an event pursuant to paragraph (a)(2)(ix)(A) of this section if the event results from: (1) A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or (2) Normal and expected wear or degradation. (x) Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.					
50.73(b)	(b) Contents. The Licensee Event Report shall contain: (1) A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence. (2)(i) A clear, specific, narrative description of what occurred so that knowledgeable readers conversant with the design of commercial nuclear power plants, but not familiar with the details of a particular plant, can understand the complete event. (ii) The narrative description must include the following specific information as appropriate for the particular event: (A) Plant operating conditions before the event.					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(B) Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event. (C) Dates and approximate times of occurrences. (D) The cause of each component or system failure or personnel error, if known. (E) The failure mode, mechanism, and effect of each failed component, if known. (F) The Energy Industry Identification System component function identifier and system name of each component or system referred to in the LER. (1) The Energy Industry Identification System is defined in: IEEE Std 803-1983 (May 16, 1983) Recommended Practice for Unique Identification in Power Plants and Related Facilities--Principles and Definitions. (2) IEEE Std 803-1983 has been approved for incorporation by reference by the Director of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51. (3) A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies may be obtained from the Institute of Electrical and Electronics Engineers, 445 Hoes Lane, P.O. Box 1331, Piscataway, NJ 08855-1331. IEEE Std 803-1983 is available for inspection at the NRC's Technical Library, which is located in the Two White Flint North Building, 11545 Rockville Pike, Rockville, Maryland 20852-2738; or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202-741-6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html*. (G) For failures of components with multiple functions, include a list of systems or secondary functions that were also affected. (H) For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service.					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	(I) The method of discovery of each component or system failure or procedural error. (J) For each human performance related root cause, the licensee shall discuss the cause(s) and circumstances. (K) Automatically and manually initiated safety system responses. (L) The manufacturer and model number (or other identification) of each component that failed during the event. (3) An assessment of the safety consequences and implications of the event. This assessment must include: (i) The availability of systems or components that could have performed the same function as the components and systems that failed during the event, and (ii) For events that occurred when the reactor was shutdown, the availability of systems or components that are needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident. (4) A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future. (5) Reference to any previous similar events at the same plant that are known to the licensee. (6) The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the plant's characteristics.					
50.73(c)	(c) *Supplemental information*. The Commission may require the licensee to submit specific additional information beyond that required by paragraph (b) of this section if the Commission finds that supplemental material is necessary for complete understanding of an unusually complex or significant event. These requests for supplemental information will be made in writing and the licensee shall submit, as specified in § 50.4, the requested information as a supplement to the initial LER.					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
50.73(d)	(d) *Submission of reports*. Licensee Event Reports must be prepared on Form NRC 366 and submitted to the U.S. Nuclear Regulatory Commission, as specified in § 50.4.					
50.73(e)	(e) *Report legibility*. The reports and copies that licensees are required to submit to the Commission under the provisions of this section must be of sufficient quality to permit legible reproduction and micrographic processing.					
50.73(f)	(f) [Reserved]					
50.73(g)	(g) *Reportable occurrences.* The requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications.					
50.74	Notification of change in operator or senior operator status.					Exclude Administrative
50.75	Reporting and recordkeeping for decommissioning planning.					Exclude, This is not technology dependent.
50.76	Licensee's change of status; financial qualifications.					Exclude Administrative
	US/IAEA Safeguards Agreement					
50.78	Facility information and verification.					Exclude Safeguards
	Transfers of Licenses--Creditors' Rights--Surrender of Licenses					
50.80	Transfer of licenses.					Exclude Administrative
50.81	Creditor regulations.					Exclude Administrative
50.82	Termination of license.					Exclude Administrative
50.83	Release of part of a power reactor facility or site for unrestricted use.					Exclude Administrative
	Amendment of License or Construction Permit at Request of Holder					
50.90	Application for amendment of license, construction permit, or early site permit.					Exclude Administrative
50.91	Notice for public comment; State consultation.					Exclude Administrative
50.92	Issuance of amendment.					Exclude

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
						Administrative
	Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission					
50.100	Revocation, suspension, modification of licenses, permits, and approvals for cause.					Exclude Administrative
50.101	Retaking possession of special nuclear material.					Exclude Administrative
50.102	Commission order for operation after revocation.					Exclude Administrative
50.103	Suspension and operation in war or national emergency.					Exclude Administrative
	Backfitting					
50.109	Backfitting.					Exclude, This is not technology dependent.
	Enforcement					
50.110	Violations.					Exclude Administrative
50.111	Criminal penalties.					Exclude Administrative
	Additional Standards for Licenses, Certifications, and Regulatory Approvals					
50.120	Training and qualification of nuclear power plant personnel.					
50.120(a)	(a) *Applicability*. The requirements of this section apply to each applicant for and each holder of an operating license issued under this part and each holder of a combined license issued under part 52 of this chapter for a nuclear power plant of the type specified in § 50.21(b) or § 50.22.					
50.120(b)	(b) *Requirements*. (1)(i) Each nuclear power plant operating license applicant, by 18 months prior to fuel load, and each holder of an operating license shall establish, implement, and maintain a training program that meets the requirements of paragraphs (b)(2) and (b)(3) of this section.					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(ii) Each holder of a combined license shall establish, implement, and maintain the training program that meets the requirements of paragraphs (b)(2) and (b)(3) of this section, as described in the final safety analysis report no later than 18 months before the scheduled date for initial loading of fuel. (2) The training program must be derived from a systems approach to training as defined in 10 CFR 55.4, and must provide for the training and qualification of the following categories of nuclear power plant personnel: (i) Non-licensed operator. (ii) Shift supervisor. (iii) Shift technical advisor. (iv) Instrument and control technician. (v) Electrical maintenance personnel. (vi) Mechanical maintenance personnel. (vii) Radiological protection technician. (viii) Chemistry technician. (ix) Engineering support personnel. (3) The training program must incorporate the instructional requirements necessary to provide qualified personnel to operate and maintain the facility in a safe manner in all modes of operation. The training program must be developed to be in compliance with the facility license, including all technical specifications and applicable regulations. The training program must be periodically evaluated and revised as appropriate to reflect industry experience as well as changes to the facility, procedures, regulations, and quality assurance requirements. The training program must be periodically reviewed by licensee management for effectiveness. Sufficient records must be maintained by the licensee to maintain program integrity and kept available for NRC inspection to verify the adequacy of the program.					
50.150	Aircraft impact assessment.					
50.150(a)	(a) *Assessment requirements*. (1) *Assessment*. Each applicant listed in paragraph (a)(3) shall perform a design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions:					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(i) The reactor core remains cooled, or the containment remains intact; and (ii) Spent fuel cooling or spent fuel pool integrity is maintained. (2) *Aircraft impact characteristics*.[1] The assessment must be based on the beyond-design-basis impact of a large, commercial aircraft used for long distance flights in the United States, with aviation fuel loading typically used in such flights, and an impact speed and angle of impact considering the ability of both experienced and inexperienced pilots to control large, commercial aircraft at the low altitude representative of a nuclear power plant's low profile. (3) *Applicability*. The requirements of paragraphs (a)(1) and (a)(2) of this section apply to applicants for: (i) Construction permits for nuclear power reactors issued under this part after July 13, 2009; (ii) Operating licenses for nuclear power reactors issued under this part for which a construction permit was issued after July 13, 2009; (iii)(A) Standard design certifications issued under part 52 of this chapter after July 13, 2009; (B) Renewal of standard design certifications in effect on July 13, 2009 which have not been amended to comply with the requirements of this section by the time of application for renewal; (iv) Standard design approvals issued under part 52 of this chapter after July 13, 2009; (v) Combined licenses issued under part 52 of this chapter that: (A) Do not reference a standard design certification, standard design approval, or manufactured reactor; or (B) Reference a standard design certification issued before July 13, 2009 which has not been amended to address the requirements of this section; and (vi) Manufacturing licenses issued under part 52 of this chapter that:					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(A) Do not reference a standard design certification or standard design approval; or (B) Reference a standard design certification issued before July 13, 2009 which has not been amended to address the requirements of this section.					
50.150(b)	(b) *Content of application*. For applicants identified in paragraph (a)(3) of this section, the preliminary or final safety analysis report, as applicable, must include a description of: (1) The design features and functional capabilities identified in paragraph (a)(1) of this section; and (2) How the design features and functional capabilities identified in paragraph (a)(1) of this section meet the assessment requirements in paragraph (a)(1) of this section.					
50.150(c)	(c) *Control of changes*. (1) For construction permits which are subject to paragraph (a) of this section, if the permit holder changes the information required by 10 CFR 50.34(a)(13) to be included in the preliminary safety analysis report, then the permit holder shall consider the effect of the changed feature or capability on the original assessment required by 10 CFR 50.150(a) and amend the information required by 10 CFR 50.34(a)(13) to be included in the preliminary safety analysis report to describe how the modified design features and functional capabilities continue to meet the assessment requirements in paragraph (a)(1) of this section. (2) For operating licenses which are subject to paragraph (a) of this section, if the licensee changes the information required by 10 CFR 50.34(b)(12) to be included in the final safety analysis report, then the licensee shall consider the effect of the changed feature or capability on the original assessment required by 10 CFR 50.150(a) and amend the information required by 10 CFR 50.34(b)(12) to be included in the final safety analysis report to describe how the modified design features and functional capabilities continue to meet the assessment requirements in paragraph (a)(1) of this section. (3) For standard design certifications which are subject to paragraph (a) of this section, generic changes to the information required by 10 CFR 52.47(a)(28) to be included in the final safety analysis report are governed by the applicable requirements of 10 CFR 52.63. (4)(i) For combined licenses which are subject to paragraph (a) of this section, if the licensee changes the information required by 10 CFR 52.79(a)(47) to be included in the final safety					

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	analysis report, then the licensee shall consider the effect of the changed feature or capability on the original assessment required by 10 CFR 50.150(a) and amend the information required by 10 CFR 52.79(a)(47) to be included in the final safety analysis report to describe how the modified design features and functional capabilities continue to meet the assessment requirements in paragraph (a)(1) of this section. (ii) For combined licenses which are not subject to paragraph (a) of this section but reference a standard design certification which is subject to paragraph (a) of this section, proposed departures from the information required by 10 CFR 52.47(a)(28) to be included in the final safety analysis report for the referenced standard design certification are governed by the change control requirements in the applicable design certification rule. (iii) For combined licenses which are not subject to paragraph (a) of this section but reference a manufactured reactor which is subject to paragraph (a) of this section, proposed departures from the information required by 10 CFR 52.157(f)(32) to be included in the final safety analysis report for the manufacturing license are governed by the applicable requirements in 10 CFR 52.171(b)(2). (5)(i) For manufacturing licenses which are subject to paragraph (a) of this section, generic changes to the information required by 10 CFR 52.157(f)(32) to be included in the final safety analysis report are governed by the applicable requirements of 10 CFR 52.171. (ii) For manufacturing licenses which are not subject to paragraph (a) of this section but reference a standard design certification which is subject to paragraph (a) of this section, proposed departures from the information required by 10 CFR 52.47(a)(28) to be included in the final safety analysis report for the referenced standard design certification are governed by the change control requirements in the applicable design certification rule.					
Appendix A	General Design Criteria for Nuclear Power Plants					See Applicability Determination Table for General Design Criteria
Appendix B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants					Exclude Administrative
Appendix C	Guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Construction Permits and Combined Licenses					Exclude Administrative

Table A1-2: PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Appendix D	[Reserved]	No				Exclude, not issued
Appendix E	Emergency Planning and Preparedness for Production and Utilization Facilities ***[See full text of 10 CFR 50 Appendix E for detailed requirements]***					.
Appendix F	Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities	No				Exclude, Not related to power plants.
Appendix G	Fracture Toughness Requirements ***[See full text of 10 CFR 50 Appendix G for detailed requirements]***					
Appendix H	Reactor Vessel Material Surveillance Program Requirements ***[See full text of 10 CFR 50 Appendix H for detailed requirements]***					
Appendix I	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents ***[See full text of 10 CFR 50 Appendix I for detailed requirements]***					
Appendix J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors ***[See full text of 10 CFR 50 Appendix J for detailed requirements]***					
Appendix K	ECCS Evaluation Models ***[See full text of 10 CFR 50 Appendix K for detailed requirements]***					
Appendix L	[Reserved]	No				Exclude, not issued
Appendix M	[Reserved]	No				Exclude, not issued
Appendix N	Standardization of Nuclear Power Plant Designs: Permits To Construct and Licenses To Operate Nuclear Power Reactors of Identical Design at Multiple Sites					Exclude, Not required due to 10 CFR 52.
Appendix O	[Reserved]	No				Exclude, not issued
Appendix P	[Reserved]	No				Exclude, not issued
Appendix Q	Pre-application Early Review of Site Suitability Issues	No				Exclude Administrative
Appendix R	Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979	No				Exclude, Applicable only to plants prior to 1979.
Appendix S	Earthquake Engineering Criteria for Nuclear Power Plants					Refer to 10 CFR 50, Appendix S for detailed requirements.

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
GDC 1	Quality Standards and Records					
	Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.					
GDC 2	Design Bases for Protection Against Natural Phenomena					
	Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.					
GDC 3	Fire Protection					
	Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.					
GDC 4	Environmental and Dynamic Effects Design Bases					
	Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.					
GDC 5	Sharing of Structures, Systems, and Components					
	Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.					
GDC 10	Reactor Design					
	The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to Aassure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.					
GDC 11	Reactor Inherent Protection					
	The reactor core and associated coolant systems shall be designed so that					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.					
GDC 12	Suppression of Reactor Power Oscillations					
	The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.					
GDC 13	Instrumentation and Control					
	Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.					
GDC 14	Reactor Coolant Pressure Boundary					
	The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.					
GDC 15	Reactor Coolant System Design					
	The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.					
GDC 16	Containment Design					
	Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
GDC 17	Electric Power Systems					
	An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.					
	The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.					
	Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.					
	Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.					
GDC 18	Inspection and Testing of Electric Power Systems Electric power systems important to safety shall be designed to permit					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.					
GDC 19	Control Room					
	A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.					
	Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
GDC 20	Protection System Functions					
	The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.					
GDC 21	Protection System Reliability and Testability					
	The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.					
GDC 22	Protection System Independence					
	The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.					
GDC 23	Protection System Failure Modes					
	The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
GDC 24	Separation of Protection and Control Systems					
	The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.					
GDC 25	Protection System Requirements for Reactivity Control Malfunctions					
	The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.					
GDC 26	Reactivity Control System Redundancy and Capability					
	Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.					
GDC 27	Combined Reactivity Control Systems Capability					
	The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.					
GDC 28	Reactivity Limits					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.					
GDC 29	Protection Against Anticipated Operational Occurrences					
	The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.					
GDC 30	Quality of Reactor Coolant Pressure Boundary					
	Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.					
GDC 31	Fracture Prevention of Reactor Coolant Pressure Boundary					
	The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.					
GDC 32	Inspection of Reactor Coolant Pressure Boundary					
	Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.					
GDC 33	Reactor Coolant Makeup					See Note 1
	A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.					
GDC 34	Residual Heat Removal					
	A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.					
	Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.					
GDC 35	Emergency Core Cooling					See Note 1
	A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.					
	Suitable redundancy in components and features, and suitable					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.					
GDC 36	Inspection of Emergency Core Cooling System					See Note 1
	The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.					
GDC 37	Testing of Emergency Core Cooling System					See Note 1
	The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.					
GDC 38	Containment Heat Removal					See Note 1
	A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.					
	Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.					
GDC 39	Inspection of Containment Heat Removal System					See Note 1
	The containment heat removal system shall be designed to permit					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.					
GDC 40	Testing of Containment Heat Removal System					See Note 1
	The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.					
GDC 41	Containment Atmosphere Cleanup					
	Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.					
	Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.					
GDC 42	Inspection of Containment Atmosphere Cleanup Systems					
	The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.					
GDC 43	Testing of Containment Atmosphere Cleanup Systems					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.					
GDC 44	Cooling Water					
	A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.					
	Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.					
GDC 45	Inspection of Cooling Water System					
	The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.					
GDC 46	Testing of Cooling Water System					
	The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	operation of applicable portions of the protection system and the transfer between normal and emergency power sources.					
GDC 50	Containment Design Basis					
	The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.					
GDC 51	Fracture Prevention of Containment Pressure Boundary					
	The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.					
GDC 52	Capability for Containment Leakage Rate Testing					
	The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.					
GDC 53	Provisions for Containment Testing and Inspection					
	The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an					

Table A1-3: PART 50, Appendix A--GENERAL DESIGN CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.					
GDC 54	Systems Penetrating Containment					
	Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.					
GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment					See Note 1
	Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.					
	Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	isolation valves shall be designed to take the position that provides greater safety.					
	Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.					
GDC 56	Primary Containment Isolation					
	Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.					
	Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	isolation valves shall be designed to take the position that provides greater safety.					
GDC 57	Closed Systems Isolation Valves					
	Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.					
GDC 60	Control of Releases of Radioactive Materials to the Environment					
	The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.					
GDC 61	Fuel Storage and Handling and Radioactivity Control					
	The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.					
GDC 62	Prevention of Criticality in Fuel Storage and Handling					
	Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
GDC 63	Monitoring Fuel and Waste Storage					
	Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.					
GDC 64	Monitoring Radioactivity Releases					
	Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.					

Notes: 1. GDCs do not exist for the design and review of the primary and heat removal systems proposed for the HTGR in a form approaching those available for light water reactor technology. The HTGR utilizes two safety related vessel and heat removal systems; the Vessel System (VS) and the passive Reactor Cavity Cooling System (RCCS). DOE proposes two additional systems for cooling that would not have safety related functions and would not have to fully meet safety-grade quality; the Heat Transport System (HTS) and the shutdown Cooling System (SCS). The earliest HTGR precedents abroad, and for Peach Bottom and Fort St. Vrain, generally provided favorable experience in a high temperature helium environment but no formalized criteria or industry standards were developed. Current GDCs will require review to determine if they should be modified to accommodate the HTGR design or whether new GDCs are required.

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	General Provisions					Heading
51.1	Scope.					Exclude; Admin
51.2	Subparts.					Exclude; Admin
51.3	Resolution of conflict.					Exclude; Admin
51.4	Definitions.					Exclude; Admin
51.5	Interpretations.					Exclude; Admin
51.6	Specific exemptions.					Exclude; Admin
	Subpart A--National Environmental Policy Act--Regulations Implementing Section 102(2)					Heading
51.10	Purpose and scope of subpart; application of regulations of Council on Environmental Quality.					Exclude; Admin
51.11	Relationship to other subparts. [Reserved]	NA				Exclude; Not issued
51.12	Application of subpart to ongoing environmental work.					Exclude; Admin
51.13	Emergencies.					Exclude; Admin
51.14	Definitions.					Exclude; Admin
51.15	Time schedules.					Exclude; Admin
51.16	Proprietary information.					Exclude; Admin
51.17	Information collection requirements; OMB approval.					Exclude; Admin
	Preliminary Procedures					Heading
	Classification of Licensing and Regulatory Actions					Heading
51.20	Criteria for and identification of licensing and regulatory actions requiring environmental impact statements.					Exclude; Admin
51.21	Criteria for and identification of licensing and regulatory actions requiring environmental assessments.					Exclude; Admin
51.22	Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review.					Exclude; Admin
51.23	§ 51.23 Temporary storage of spent fuel after cessation of reactor operation--generic					

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	determination of no significant environmental impact. (a) The Commission has made a generic determination that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor at its spent fuel storage basin or at either onsite or offsite independent spent fuel storage installations. Further, the Commission believes there is reasonable assurance that at least one mined geologic repository will be available within the first quarter of the twenty-first century, and sufficient repository capacity will be available within 30 years beyond the licensed life for operation of any reactor to dispose of the commercial high-level waste and spent fuel originating in such reactor and generated up to that time. (b) Accordingly, as provided in §§ 51.30(b), 51.53, 51.61, 51.80(b), 51.95, and 51.97(a), and within the scope of the generic determination in paragraph (a) of this section, no discussion of any environmental impact of spent fuel storage in reactor facility storage pools or independent spent fuel storage installations (ISFSI) for the period following the term of the reactor operating license or amendment, reactor combined license or amendment, or initial ISFSI license or amendment for which application is made, is required in any environmental report, environmental impact statement, environmental assessment, or other analysis prepared in connection with the issuance or amendment of an					

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	operating license for a nuclear power reactor under parts 50 and 54 of this chapter, or issuance or amendment of a combined license for a nuclear power reactor under parts 52 and 54 of this chapter, or the issuance of an initial license for storage of spent fuel at an ISFSI, or any amendment thereto. (c) This section does not alter any requirements to consider the environmental impacts of spent fuel storage during the term of a reactor operating license or combined license, or a license for an ISFSI in a licensing proceeding.					
	Determinations to Prepare Environmental Impact Statements, Environmental Assessments or Findings of No Significant Impact, and Related Procedures					Heading
51.25	Determination to prepare environmental impact statement or environmental assessment; eligibility for categorical exclusion.					Exclude; Admin
51.26	Requirement to publish notice of intent and conduct scoping process.					Exclude; Admin
51.27	Notice of intent.					Exclude; Admin
	Scoping					Heading
51.28	Scoping--participants.					Exclude; Admin
51.29	Scoping-environmental impact statement and supplement to environmental impact statement.					Exclude; Admin
	Environmental Assessment					Heading
51.30	Environmental assessment.	NA				Exclude
51.31	Determinations based on environmental assessment.	NA				Exclude
	Finding of No Significant Impact					Heading
51.32	Finding of no significant impact.	NA				Exclude
51.33	Draft finding of no significant impact; distribution.	NA				Exclude
51.34	Preparation of finding of no significant impact.	NA				Exclude

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51.35	Requirement to publish finding of no significant impact; limitation on Commission action.	NA				Exclude
	Environmental Reports and Information--Requirements Applicable to Applicants and Petitioners for Rulemaking					Heading
	General					Heading
51.40	Consultation with NRC staff.					Exclude; Admin
51.41	Requirement to submit environmental information.					Exclude; Admin
	Environmental Reports--General Requirements					Heading
51.45	Environmental report.					
51.45(a)	(a) General. As required by §§ 51.50, 51.53, 51.54, 51.55, 51.60, 51.61, 51.62, or 51.68, as appropriate, each applicant or petitioner for rulemaking shall submit with its application or petition for rulemaking one signed original of a separate document entitled "Applicant's" or "Petitioner's Environmental Report," as appropriate. An applicant or petitioner for rulemaking may submit a supplement to an environmental report at any time.					
51.45(b)	(b) Environmental considerations. The environmental report shall contain a description of the proposed action, a statement of its purposes, a description of the environment affected, and discuss the following considerations: (1) The impact of the proposed action on the environment. Impacts shall be discussed in proportion to their significance; (2) Any adverse environmental effects which cannot be avoided should the proposal be implemented; (3) Alternatives to the proposed action. The discussion of alternatives shall be sufficiently complete to aid the Commission in developing and					

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	exploring, pursuant to section 102(2)(E) of NEPA, "appropriate alternatives to recommended courses of action in any proposal which involves unresolved conflicts concerning alternative uses of available resources." To the extent practicable, the environmental impacts of the proposal and the alternatives should be presented in comparative form; (4) The relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity; and (5) Any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented.					
51.45(c)	(c) Analysis. The environmental report must include an analysis that considers and balances the environmental effects of the proposed action, the environmental impacts of alternatives to the proposed action, and alternatives available for reducing or avoiding adverse environmental effects. An environmental report prepared at the early site permit stage under § 51.50(b), limited work authorization stage under § 51.49, construction permit stage under § 51.50(a), or combined license stage under § 51.50(c) must include a description of impacts of the preconstruction activities performed by the applicant at the proposed site (i.e., those activities listed in paragraphs (2)(i) through (2)(x) in the definition of "construction" contained in § 51.4), necessary to support the construction and operation of the facility which is the subject of the early site permit, limited work authorization, construction permit, or combined license application. The environmental report must also contain an analysis					

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	of the cumulative impacts of the activities to be authorized by the limited work authorization, construction permit, or combined license in light of the preconstruction impacts described in the environmental report. Except for an environmental report prepared at the early site permit stage, or an environmental report prepared at the license renewal stage under § 51.53(c), the analysis in the environmental report should also include consideration of the economic, technical, and other benefits and costs of the proposed action and its alternatives. Environmental reports prepared at the license renewal stage under § 51.53(c) need not discuss the economic or technical benefits and costs of either the proposed action or alternatives except if these benefits and costs are either essential for a determination regarding the inclusion of an alternative in the range of alternatives considered or relevant to mitigation. In addition, environmental reports prepared under § 51.53(c) need not discuss issues not related to the environmental effects of the proposed action and its alternatives. The analyses for environmental reports shall, to the fullest extent practicable, quantify the various factors considered. To the extent that there are important qualitative considerations or factors that cannot be quantified, those considerations or factors shall be discussed in qualitative terms. The environmental report should contain sufficient data to aid the Commission in its development of an independent analysis.					
51.45(d)	(d) Status of compliance. The environmental report shall list all Federal permits, licenses, approvals and other entitlements which must be obtained in					

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	connection with the proposed action and shall describe the status of compliance with these requirements. The environmental report shall also include a discussion of the status of compliance with applicable environmental quality standards and requirements including, but not limited to, applicable zoning and land-use regulations, and thermal and other water pollution limitations or requirements which have been imposed by Federal, State, regional, and local agencies having responsibility for environmental protection. The discussion of alternatives in the report shall include a discussion of whether the alternatives will comply with such applicable environmental quality standards and requirements.					
51.45(e)	(e) Adverse information. The information submitted pursuant to paragraphs (b) through (d) of this section should not be confined to information supporting the proposed action but should also include adverse information.					
	Environmental Reports--Production and Utilization Facilities					Heading
51.49	Environmental report—limited work authorization.					
51.49(a)	(a) *Limited work authorization submitted as part of complete construction permit or combined license application*. Each applicant for a construction permit or combined license applying for a limited work authorization under § 50.10(d) of this chapter in a complete application under 10 CFR 2.101(a)(1) through (a)(4), shall submit with its application a separate document, entitled, "Applicant's Environmental Report—Limited Work Authorization					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	Stage," which is in addition to the environmental report required by § 51.50 of this part. Each environmental report must also contain the following information: (1) A description of the activities proposed to be conducted under the limited work authorization; (2) A statement of the need for the activities; and (3) A description of the environmental impacts that may reasonably be expected to result from the activities, the mitigation measures that the applicant proposes to implement to achieve the level of environmental impacts described, and a discussion of the reasons for rejecting mitigation measures that could be employed by the applicant to further reduce environmental impacts.					
51.49(b)	(b) *Phased application for limited work authorization and construction permit or combined license*. If the construction permit or combined license application is filed in accordance with § 2.101(a)(9) of this chapter, then the environmental report for part one of the application may be limited to a discussion of the activities proposed to be conducted under the limited work authorization. If the scope of the environmental report for part one is so limited, then part two of the application must include the information required by § 51.50, as applicable.					

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51.49(c)	(c) *Limited work authorization submitted as part of an early site permit application*. Each applicant for an early site permit under subpart A of part 52 of this chapter requesting a limited work authorization shall submit with its application the environmental report required by § 51.50(b). Each environmental report must contain the following information: (1) A description of the activities proposed to be conducted under the limited work authorization; (2) A statement of the need for the activities; and (3) A description of the environmental impacts that may reasonably be expected to result from the activities, the mitigation measures that the applicant proposes to implement to achieve the level of environmental impacts described, and a discussion of the reasons for rejecting mitigation measures that could be employed by the applicant to further reduce environmental impacts.					
51.49(d)	(d) *Limited work authorization request submitted by early site permit holder*. Each holder of an early site permit requesting a limited work authorization shall submit with its application a document entitled, "Applicant's Environmental Report—Limited Work Authorization under Early Site Permit," containing the following information: (1) A description of the activities proposed to be conducted under the limited work authorization; (2) A statement of the need for the activities; (3) A description of the environmental impacts that					

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	may reasonably be expected to result from the activities, the mitigation measures that the applicant proposes to implement to achieve the level of environmental impacts described, and a discussion of the reasons for rejecting mitigation measures that could be employed by the applicant to further reduce environmental impacts; and (4) Any new and significant information for issues related to the impacts of construction of the facility that were resolved in the early site permit proceeding with respect to the environmental impacts of the activities to be conducted under the limited work authorization. (5) A description of the process used to identify new and significant information regarding NRC's conclusions in the early site permit environmental impact statement. The process must be a reasonable methodology for identifying this new and significant information.					
51.49(e)	(e) *Limited work authorization for a site where an environmental impact statement was prepared, but the facility construction was not completed*. If the limited work authorization is for activities to be conducted at a site for which the Commission has previously prepared an environmental impact statement for the construction and operation of a nuclear power plant, and a construction permit was issued but construction of the plant was never completed, then the applicant's environmental					

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	report may incorporate by reference the earlier environmental impact statement. In the event of such referencing, the environmental report must identify: (1) Any new and significant information material to issues related to the impacts of construction of the facility that were resolved in the construction permit proceeding for the matters required to be addressed in paragraph (a) of this section; and (2) A description of the process used to identify new and significant information regarding the NRC's conclusions in the construction permit environmental impact statement. The process must use a reasonable methodology for identifying this new and significant information.					
51.49(f)	(f) *Environmental Report*. An environmental report submitted in accordance with this section must separately evaluate the environmental impacts and proposed alternatives attributable to the activities proposed to be conducted under the limited work authorization. At the option of the applicant, the "Applicant's Environmental Report—Limited Work Authorization Stage," may contain the information required to be submitted in the environmental report required under 51.50, which addresses the impacts of construction and operation for the proposed facility (including the environmental impacts attributable to the limited work authorization), and discusses the overall costs and benefits balancing					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	for the proposed action.					
51.50	Environmental report—construction permit, early site permit, or combined license stage.					
51.50(a)	(a) *Construction permit stage*. Each applicant for a permit to construct a production or utilization facility covered by § 51.20 shall submit with its application a separate document, entitled "Applicant's Environmental Report—Construction Permit Stage," which shall contain the information specified in §§ 51.45, 51.51, and 51.52. Each environmental report shall identify procedures for reporting and keeping records of environmental data, and any conditions and monitoring requirements for protecting the non-aquatic environment, proposed for possible inclusion in the license as environmental conditions in accordance with § 50.36b of this chapter.					
51.50(b)	(b) *Early site permit stage*. Each applicant for an early site permit shall submit with its application a separate document, entitled "Applicant's Environmental Report—Early Site Permit Stage," which shall contain the information specified in §§ 51.45, 51.51, and 51.52, as modified in this paragraph. (1) The environmental report must include an evaluation of alternative sites to determine whether there is any obviously superior alternative to the site proposed. (2) The environmental report may address one or					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	more of the environmental effects of construction and operation of a reactor, or reactors, which have design characteristics that fall within the site characteristics and design parameters for the early site permit application, *provided however*, that the environmental report must address all environmental effects of construction and operation necessary to determine whether there is any obviously superior alternative to the site proposed. The environmental report need not include an assessment of the economic, technical, or other benefits (for example, need for power) and costs of the proposed action or an evaluation of alternative energy sources. (3) For other than light-water-cooled nuclear power reactors, the environmental report must contain the basis for evaluating the contribution of the environmental effects of fuel cycle activities for the nuclear power reactor. (4) Each environmental report must identify the procedures for reporting and keeping records of environmental data, and any conditions and monitoring requirements for protecting the non-aquatic environment, proposed for possible inclusion in the license as environmental conditions in accordance with § 50.36b of this chapter.					
51.50(c)	(c) *Combined license stage*. Each applicant for a combined license shall submit with its application a separate document, entitled "Applicant's					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	Environmental Report—Combined License Stage." Each environmental report shall contain the information specified in §§ 51.45, 51.51, and 51.52, as modified in this paragraph. For other than light-water-cooled nuclear power reactors, the environmental report shall contain the basis for evaluating the contribution of the environmental effects of fuel cycle activities for the nuclear power reactor. Each environmental report shall identify procedures for reporting and keeping records of environmental data, and any conditions and monitoring requirements for protecting the non-aquatic environment, proposed for possible inclusion in the license as environmental conditions in accordance with § 50.36b of this chapter. The combined license environmental report may reference information contained in a final environmental document previously prepared by the NRC staff. (1) *Application referencing an early site permit*. If the combined license application references an early site permit, then the "Applicant's Environmental Report—Combined License Stage" need not contain information or analyses submitted to the Commission in "Applicant's Environmental Report—Early Site Permit Stage," or resolved in the Commission's early site permit environmental impact statement, but must contain, in addition to the environmental information and analyses					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	otherwise required: (i) Information to demonstrate that the design of the facility falls within the site characteristics and design parameters specified in the early site permit; (ii) Information to resolve any significant environmental issue that was not resolved in the early site permit proceeding; (iii) Any new and significant information for issues related to the impacts of construction and operation of the facility that were resolved in the early site permit proceeding; (iv) A description of the process used to identify new and significant information regarding the NRC's conclusions in the early site permit environmental impact statement. The process must use a reasonable methodology for identifying such new and significant information; and (v) A demonstration that all environmental terms and conditions that have been included in the early site permit will be satisfied by the date of issuance of the combined license. Any terms or conditions of the early site permit that could not be met by the time of issuance of the combined license, must be set forth as terms or conditions of the combined license. [Material referencing Design Certification and manufacturing licenses deleted.]					
51.51	Uranium fuel cycle environmental data--Table S-3.					Note: Only text portions of the regulation are

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	Under § 51.50, every environmental report prepared for the construction permit stage or early site permit stage or combined license stage of a light-water-cooled nuclear power reactor, and submitted on or after September 4, 1979, shall take Table S–3, Table of Uranium Fuel Cycle Environmental Data, as the basis for evaluating the contribution of the environmental effects of uranium mining and milling, the production of uranium hexafluoride, isotopic enrichment, fuel fabrication, reprocessing of irradiated fuel, transportation of radioactive materials and management of low-level wastes and high-level wastes related to uranium fuel cycle activities to the environmental costs of licensing the nuclear power reactor. Table S–3 shall be included in the environmental report and may be supplemented by a discussion of the environmental significance of the data set forth in the table as weighed in the analysis for the proposed facility.					listed. See the actual regulation for the layout and data requirements shown in Table S-3.
51.52	Environmental effects of transportation of fuel and waste--Table S-4. Under § 51.50, every environmental report prepared for the construction permit stage or early site permit stage or combined license stage of a light-water-cooled nuclear power reactor, and submitted after February 4, 1975, shall contain a statement concerning transportation of fuel and radioactive wastes to and from the reactor. That statement shall indicate that the reactor and this transportation either meet all of the conditions in paragraph (a) of this section or all of the conditions of paragraph (b) of this section. (a)(1) The reactor has a core thermal power level not exceeding 3,800 megawatts;					Note: Only text portions of the regulation are listed. See the actual regulation for the layout and data requirements shown in Table S-4.

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(2) The reactor fuel is in the form of sintered uranium dioxide pellets having a uranium-235 enrichment not exceeding 4% by weight, and the pellets are encapsulated in zircaloy rods; (3) The average level of irradiation of the irradiated fuel from the reactor does not exceed 33,000 megawatt-days per metric ton, and no irradiated fuel assembly is shipped until at least 90 days after it is discharged from the reactor; (4) With the exception of irradiated fuel, all radioactive waste shipped from the reactor is packaged and in a solid form; (5) Unirradiated fuel is shipped to the reactor by truck; irradiated fuel is shipped from the reactor by truck, rail, or barge; and radioactive waste other than irradiated fuel is shipped from the reactor by truck or rail; and (6) The environmental impacts of transportation of fuel and waste to and from the reactor, with respect to normal conditions of transport and possible accidents in transport, are as set forth in Summary Table S-4 in paragraph (c) of this section; and the values in the table represent the contribution of the transportation to the environmental costs of licensing the reactor. (b) For reactors not meeting the conditions of paragraph (a) of this section, the statement shall contain a full description and detailed analysis of the environmental effects of transportation of fuel and wastes to and from the reactor, including values for the environmental impact under normal conditions of transport and for the environmental risk from accidents in transport. The statement shall indicate that the values determined by the analysis					

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	represent the contribution of such effects to the environmental costs of licensing the reactor.					
51.53	Postconstruction environmental reports.					
51.53(a)	(a) *General*. Any environmental report prepared under the provisions of this section may incorporate by reference any information contained in a prior environmental report or supplement thereto that relates to the production or utilization facility or site, or any information contained in a final environmental document previously prepared by the NRC staff that relates to the production or utilization facility or site. Documents that may be referenced include, but are not limited to, the final environmental impact statement; supplements to the final environmental impact statement, including supplements prepared at the license renewal stage; NRC staff-prepared final generic environmental impact statements; and environmental assessments and records of decisions prepared in connection with the construction permit, operating license, early site permit, combined license and any license amendment for that facility.					
51.53(b)	(b) *Operating license stage*. Each applicant for a license to operate a production or utilization facility covered by § 51.20 shall submit with its application a separate document entitled "Supplement to Applicant's Environmental Report--Operating License Stage," which will update "Applicant's Environmental Report--Construction Permit Stage."					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	Unless otherwise required by the Commission, the applicant for an operating license for a nuclear power reactor shall submit this report only in connection with the first licensing action authorizing full-power operation. In this report, the applicant shall discuss the same matters described in §§ 51.45, 51.51, and 51.52, but only to the extent that they differ from those discussed or reflect new information in addition to that discussed in the final environmental impact statement prepared by the Commission in connection with the construction permit. No discussion of need for power, or of alternative energy sources, or of alternative sites for the facility, or of any aspect of the storage of spent fuel for the facility within the scope of the generic determination in § 51.23(a) and in accordance with § 51.23(b) is required in this report.					
51.53(c)	(c) *Operating license renewal stage*. (1) Each applicant for renewal of a license to operate a nuclear power plant under part 54 of this chapter shall submit with its application a separate document entitled "Applicant's Environmental Report--Operating License Renewal Stage." (2) The report must contain a description of the proposed action, including the applicant's plans to modify the facility or its administrative control procedures as described in accordance with § 54.21 of this chapter. This report must describe in detail the modifications directly affecting the					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	environment or affecting plant effluents that affect the environment. In addition, the applicant shall discuss in this report the environmental impacts of alternatives and any other matters described in § 51.45. The report is not required to include discussion of need for power or the economic costs and economic benefits of the proposed action or of alternatives to the proposed action except insofar as such costs and benefits are either essential for a determination regarding the inclusion of an alternative in the range of alternatives considered or relevant to mitigation. The environmental report need not discuss other issues not related to the environmental effects of the proposed action and the alternatives. In addition, the environmental report need not discuss any aspect of the storage of spent fuel for the facility within the scope of the generic determination in § 51.23(a) and in accordance with § 51.23(b). (3) For those applicants seeking an initial renewed license and holding an operating license, construction permit, or combined license as of June 30, 1995, the environmental report shall include the information required in paragraph (c)(2) of this section subject to the following conditions and considerations: (i) The environmental report for the operating license renewal stage is not required to contain analyses of the environmental impacts of the					

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	license renewal issues identified as Category 1 issues in Appendix B to subpart A of this part. (ii) The environmental report must contain analyses of the environmental impacts of the proposed action, including the impacts of refurbishment activities, if any, associated with license renewal and the impacts of operation during the renewal term, for those issues identified as Category 2 issues in Appendix B to subpart A of this part. The required analyses are as follows: (A) If the applicant's plant utilizes cooling towers or cooling ponds and withdraws make-up water from a river whose annual flow rate is less than 3.15×10^{12} $ft^3/year$ ($9 \times 10^{10} m^3/year$), an assessment of the impact of the proposed action on the flow of the river and related impacts on instream and riparian ecological communities must be provided. The applicant shall also provide an assessment of the impacts of the withdrawal of water from the river on alluvial aquifers during low flow. (B) If the applicant's plant utilizes once-through cooling or cooling pond heat dissipation systems, the applicant shall provide a copy of current Clean Water Act 316(b) determinations and, if necessary, a 316(a) variance in accordance with 40 CFR part 125, or equivalent State permits and supporting documentation. If the applicant can not provide these documents, it shall assess the impact of the proposed action on fish and shellfish resources					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	resulting from heat shock and impingement and entrainment. (C) If the applicant's plant uses Ranney wells or pumps more than 100 gallons (total onsite) of ground water per minute, an assessment of the impact of the proposed action on ground-water use must be provided. (D) If the applicant's plant is located at an inland site and utilizes cooling ponds, an assessment of the impact of the proposed action on groundwater quality must be provided. (E) All license renewal applicants shall assess the impact of refurbishment and other license-renewal-related construction activities on important plant and animal habitats. Additionally, the applicant shall assess the impact of the proposed action on threatened or endangered species in accordance with the Endangered Species Act. (F) If the applicant's plant is located in or near a nonattainment or maintenance area, an assessment of vehicle exhaust emissions anticipated at the time of peak refurbishment workforce must be provided in accordance with the Clean Air Act as amended. (G) If the applicant's plant uses a cooling pond, lake, or canal or discharges into a river having an annual average flow rate of less than 3.15×10^{12} ft^3/year ($9 \times 10^{10} m^3$/year), an assessment of the impact of the proposed action on public health from					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	thermophilic organisms in the affected water must be provided. (H) If the applicant's transmission lines that were constructed for the specific purpose of connecting the plant to the transmission system do not meet the recommendations of the National Electric Safety Code for preventing electric shock from induced currents, an assessment of the impact of the proposed action on the potential shock hazard from the transmission lines must be provided. (I) An assessment of the impact of the proposed action on housing availability, land-use, and public schools (impacts from refurbishment activities only) within the vicinity of the plant must be provided. Additionally, the applicant shall provide an assessment of the impact of population increases attributable to the proposed project on the public water supply. (J) All applicants shall assess the impact of highway traffic generated by the proposed project on the level of service of local highways during periods of license renewal refurbishment activities and during the term of the renewed license. (K) All applicants shall assess whether any historic or archaeological properties will be affected by the proposed project. (L) If the staff has not previously considered severe accident mitigation alternatives for the applicant's plant in an environmental impact statement or					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	related supplement or in an environmental assessment, a consideration of alternatives to mitigate severe accidents must be provided. (M) Reserved. (iii) The report must contain a consideration of alternatives for reducing adverse impacts, as required by § 51.45(c), for all Category 2 license renewal issues in Appendix B to subpart A of this part. No such consideration is required for Category 1 issues in Appendix B to subpart A of this part. (iv) The environmental report must contain any new and significant information regarding the environmental impacts of license renewal of which the applicant is aware.					
51.53(d)	(d) *Postoperating license stage*. Each applicant for a license amendment authorizing decommissioning activities for a production or utilization facility either for unrestricted use or based on continuing use restrictions applicable to the site; and each applicant for a license amendment approving a license termination plan or decommissioning plan under § 50.82 of this chapter either for unrestricted use or based on continuing use restrictions applicable to the site; and each applicant for a license or license amendment to store spent fuel at a nuclear power reactor after expiration of the operating license for the nuclear power reactor shall submit with its application a separate document, entitled "Supplement to Applicant's Environmental					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Report--Post Operating License Stage," which will update "Applicant's Environmental Report--Operating License Stage," as appropriate, to reflect any new information or significant environmental change associated with the applicant's proposed decommissioning activities or with the applicant's proposed activities with respect to the planned storage of spent fuel. Unless otherwise required by the Commission, in accordance with the generic determination in § 51.23(a) and the provisions in § 51.23(b), the applicant shall only address the environmental impact of spent fuel storage for the term of the license applied for. The ''Supplement to Applicant's Environmental Report—Post Operating License Stage'' may incorporate by reference any information contained in ''Applicants Environmental Report—Construction Permit Stage.					
51.54	Environmental report--manufacturing license.	NA				Exclude
51.55	Environmental report—standard design certification.	NA				Exclude
51.58	Environmental report-number of copies; distribution.					Exclude; Admin
	Environmental Reports--Materials Licenses					Heading
51.60	Environmental report--materials licenses.	NA				Exclude
51.61	Environmental report--independent spent fuel storage installation (ISFSI) or monitored retrievable storage installation (MRS) license.	NA				Exclude
51.62	Environmental report--land disposal of radioactive waste licensed under 10 CFR part 61.	NA				Exclude
51.66	Environmental report--number of copies; distribution.	NA				Exclude
51.67	Environmental information concerning geologic	NA				Exclude

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	repositories.					
	Environmental Reports--Rulemaking					Heading
51.68	Environmental report--rulemaking.	NA				Exclude
	Environmental Impact Statements	NA				Exclude
	Draft Environmental Impact Statements--General Requirements					Heading
51.70	Draft environmental impact statement--general.					Exclude; Admin
51.71	Draft environmental impact statement--contents.					Exclude; Admin
51.72	Supplement to draft environmental impact statement.					Exclude; Admin
51.73	Request for comments on draft environmental impact statement.					Exclude; Admin
51.74	Distribution of draft environmental impact statement and supplement to draft environmental impact statement; news releases.					Exclude; Admin
	Draft Environmental Impact Statements--Production and Utilization Facilities					Heading
51.75	Draft environmental impact statement—construction permit, early site permit, or combined license.					
51.75(a)	(a) *Construction permit stage*. A draft environmental impact statement relating to issuance of a construction permit for a production or utilization facility will be prepared in accordance with the procedures and measures described in §§ 51.70, 51.71, 51.72, and 51.73. The contribution of the environmental effects of the uranium fuel cycle activities specified in § 51.51 shall be evaluated on the basis of impact values set forth in Table S–3, Table of Uranium Fuel Cycle Environmental Data, which shall be set out in the draft environmental impact statement. With the exception of radon-222					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	and technetium-99 releases, no further discussion of fuel cycle release values and other numerical data that appear explicitly in the table shall be required.[5] The impact statement shall take account of dose commitments and health effects from fuel cycle effluents set forth in Table S–3 and shall in addition take account of economic, socioeconomic, and possible cumulative impacts and other fuel cycle impacts as may reasonably appear significant.					
51.75(b)	(b) *Early site permit stage*. A draft environmental impact statement relating to issuance of an early site permit for a production or utilization facility will be prepared in accordance with the procedures and measures described in §§ 51.70, 51.71, 51.72, 51.73, and this section. The contribution of the environmental effects of the uranium fuel cycle activities specified in § 51.51 shall be evaluated on the basis of impact values set forth in Table S–3, Table of Uranium Fuel Cycle Environmental Data, which shall be set out in the draft environmental impact statement. With the exception of radon-222 and technetium-99 releases, no further discussion of fuel cycle release values and other numerical data that appear explicitly in the table shall be required.[5] The impact statement shall take account of dose commitments and health effects from fuel cycle effluents set forth in Table S–3 and shall in addition take account of economic, socioeconomic,					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	and possible cumulative impacts and other fuel cycle impacts as may reasonably appear significant. The draft environmental impact statement must include an evaluation of alternative sites to determine whether there is any obviously superior alternative to the site proposed. The draft environmental impact statement must also include an evaluation of the environmental effects of construction and operation of a reactor, or reactors, which have design characteristics that fall within the site characteristics and design parameters for the early site permit application, but only to the extent addressed in the early site permit environmental report or otherwise necessary to determine whether there is any obviously superior alternative to the site proposed. The draft environmental impact statement must not include an assessment of the economic, technical, or other benefits (for example, need for power) and costs of the proposed action or an evaluation of alternative energy sources, unless these matters are addressed in the early site permit environmental report.					
51.75(c)	(c) *Combined license stage*. A draft environmental impact statement relating to issuance of a combined license that does not reference an early site permit will be prepared in accordance with the procedures and measures described in §§ 51.70, 51.71, 51.72, and 51.73. The contribution of the environmental effects of the uranium fuel cycle					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	activities specified in § 51.51 shall be evaluated on the basis of impact values set forth in Table S–3, Table of Uranium Fuel Cycle Environmental Data, which shall be set out in the draft environmental impact statement. With the exception of radon-222 and technetium-99 releases, no further discussion of fuel cycle release values and other numerical data that appear explicitly in the table shall be required.[5] The impact statement shall take account of dose commitments and health effects from fuel cycle effluents set forth in Table S–3 and shall in addition take account of economic, socioeconomic, and possible cumulative impacts and other fuel cycle impacts as may reasonably appear significant. The impact statement will include a discussion of the storage of spent fuel for the nuclear power plant within the scope of the generic determination in§ 51.23(a) and in accordance with§ 51.23(b). (1) *Combined license application referencing an early site permit*. If the combined license application references an early site permit, then the NRC staff shall prepare a draft supplement to the early site permit environmental impact statement. The supplement must be prepared in accordance with § 51.92(e). (2) *Combined license application referencing a standard design certification*. If the combined license application references a standard design					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	certification and the site characteristics of the combined license's site fall within the site parameters specified in the design certification environmental assessment, then the draft combined license environmental impact statement shall incorporate by reference the design certification environmental assessment, and summarize the findings and conclusions of the environmental assessment with respect to severe accident mitigation design alternatives. (3) *Combined license application referencing a manufactured reactor*. If the combined license application proposes to use a manufactured reactor and the site characteristics of the combined license's site fall within the site parameters specified in the manufacturing license environmental assessment, then the draft combined license environmental impact statement shall incorporate by reference the manufacturing license environmental assessment, and summarize the findings and conclusions of the environmental assessment with respect to severe accident mitigation design alternatives. The combined license environmental impact statement report will not address the environmental impacts associated with manufacturing the reactor under the manufacturing license.					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
51.76	Draft environmental impact statement—limited work authorization.					
51.76(a)	(a) *Limited work authorization submitted as part of complete construction permit or combined license application*. If the application for a limited work authorization is submitted as part of a complete construction permit or combined license application, then the NRC may prepare a partial draft environmental impact statement. The analysis called for by § 51.71(d) must be limited to the activities proposed to be conducted under the limited work authorization. Alternatively, the NRC may prepare a complete draft environmental impact statement prepared in accordance with § 51.75(a) or (c), as applicable.					
51.76(b)	(b) *Phased application for limited work authorization under § 2.101(a)(9) of this chapter*. If the application for a limited work authorization is submitted in accordance with § 2.101(a)(9) of this chapter, then the draft environmental impact statement for part one of the application may be limited to consideration of the activities proposed to be conducted under the limited work authorization, and the proposed redress plan. However, if the environmental report contains the full set of information required to be submitted under § 51.50(a) or (c), then a draft environmental impact statement must be prepared in accordance with §					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	51.75(a) or (c), as applicable. Siting issues, including whether there is an obviously superior alternative site, or issues related to operation of the proposed nuclear power plant at the site, including need for power, may not be considered. After part two of the application is docketed, the NRC will prepare a draft supplement to the final environmental impact statement for part two of the application under § 51.72. No updating of the information contained in the final environmental impact statement prepared for part one is necessary in preparation of the supplemental environmental impact statement. The draft supplement must consider all environmental impacts associated with the prior issuance of the limited work authorization, but may not address or consider the sunk costs associated with the limited work authorization.					
51.76(c)	(c) *Limited work authorization submitted as part of an early site permit application*. If the application for a limited work authorization is submitted as part of an application for an early site permit, then the NRC will prepare an environmental impact statement in accordance with § 51.75(b). However, the analysis called for by § 51.71(d) must also address the activities proposed to be conducted under the limited work authorization.					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
51.76(d)	(d) *Limited work authorization request submitted by an early site permit holder*. If the application for a limited work authorization is submitted by a holder of an early site permit, then the NRC will prepare a draft supplement to the environmental impact statement for the early site permit. The supplement is limited to consideration of the activities proposed to be conducted under the limited work authorization, the adequacy of the proposed redress plan, and whether there is new and significant information identified with respect to issues related to the impacts of construction of the facility that were resolved in the early site permit proceeding with respect to the environmental impacts of the activities to be conducted under the limited work authorization. No other updating of the information contained in the final environmental impact statement prepared for the early site permit is required.					
51.76(e)	(e) *Limited work authorization for a site where an environmental impact statement was prepared, but the facility construction was not completed*. If the limited work authorization is for activities to be conducted at a site for which the Commission has previously prepared an environmental impact statement for the construction and operation of a nuclear power plant, and a construction permit was issued but construction of the plant was not completed, then the draft environmental impact					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	statement shall incorporate by reference the earlier environmental impact statement. The draft environmental impact statement must be limited to a consideration of whether there is significant new information with respect to the environmental impacts of construction, relevant to the activities to be conducted under the limited work authority, so that the conclusion of the referenced environmental impact statement on the impacts of construction would, when analyzed in accordance with § 51.71, lead to the conclusion that the limited work authorization should not be issued or should be issued with appropriate conditions.					
51.76(f)	(f) *Draft environmental impact statement*. A draft environmental impact statement prepared under this section must separately evaluate the environmental impacts and proposed alternatives attributable to the activities proposed to be conducted under the limited work authorization. However, if the "Applicant's Environmental Report—Limited Work Authorization Stage," also contains the information required to be submitted in the environmental report required under § 51.50, then the environmental impact statement must address the impacts of construction and operation for the proposed facility (including the environmental impacts attributable to the limited work authorization), and discuss the overall costs and benefits balancing for the underlying proposed					

Table A1-4: PART 51-- ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	action, in accordance with § 51.71, and § 51.75(a) or (c), as applicable.					
51.77	Distribution of draft environmental impact statement.					Exclude; Admin
	Draft Environmental Impact Statements--Materials Licenses					Heading
51.80	Draft environmental impact statement--materials license.	NA				Exclude
51.81	Distribution of draft environmental impact statement.	NA				Exclude
	Draft Environmental Impact Statements--Rulemaking					Heading
51.85	Draft environmental impact statement--rulemaking.	NA				Exclude
51.86	Distribution of draft environmental impact statement.	NA				Exclude
	Legislative Environmental Impact Statements--Proposals for Legislation					Heading
51.88	Proposals for legislation.	NA				Exclude
	Final Environmental Impact Statements--General Requirements					Heading
51.90	Final environmental impact statement--general.					Exclude; Admin
51.91	Final environmental impact statement--contents.					Exclude; Admin
51.92	Supplement to the final environmental impact statement.					Exclude; Admin
51.93	Distribution of final environmental impact statement and supplement to final environmental impact statement; news releases.					Exclude; Admin
51.94	Requirement to consider final environmental impact statement.					Exclude; Admin
	Final Environmental Impact Statements--Production and Utilization Facilities					Heading
51.95	Postconstruction environmental impact statements.					Exclude; Admin; relates to Commission action.

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						(Note: Section references 10 CFR 51.23 which may be technology dependent, but does not require further evaluation.)
	Final Environmental Impact Statements--Materials Licenses					Heading
51.97	Final environmental impact statement--materials license.	NA				Exclude
	Final Environmental Impact Statements--Rulemaking					Heading
51.99	[Reserved]	NA				Exclude; Not issued
	NEPA Procedure and Administrative Action					Heading
	General					Heading
51.100	Timing of Commission action.					Exclude; Admin
51.101	Limitations on actions.					Exclude; Admin
51.102	Requirement to provide a record of decision; preparation.					Exclude; Admin
51.103	Record of decision--general.					Exclude; Admin
51.104	NRC proceeding using public hearings; consideration of environmental impact statement.					Exclude; Admin
	Production and Utilization Facilities					Heading
51.105	Public hearings in proceedings for issuance of construction permits or early site permits; limited work authorizations.					Exclude; Admin
51.105	a Public hearings in proceedings for issuance of manufacturing licenses.	NA				Exclude
51.106	Public hearings in proceedings for issuance of operating licenses.					Exclude; Admin
51.107	Public hearings in proceedings for issuance of combined licenses; limited work authorizations.					Exclude; Admin
51.108	Public hearings on Commission findings that inspections, tests, analyses, and acceptance criteria of combined licenses are met					Exclude; Admin
	Materials Licenses					Heading

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
51.109	Public hearings in proceedings for issuance of materials license with respect to a geologic repository.	NA				Exclude
	Rulemaking					Heading
51.110	[Reserved]	NA				Exclude; Not issued
	Public Notice of and Access to Environmental Documents					Heading
51.116	Notice of intent.					Exclude; Admin
51.117	Draft environmental impact statement--notice of availability.					Exclude; Admin
51.118	Final environmental impact statement--notice of availability.					Exclude; Admin
51.119	Publication of finding of no significant impact; distribution.					Exclude; Admin
51.120	Availability of environmental documents for public inspection.					Exclude; Admin
51.121	Status of NEPA actions.					Exclude; Admin
51.122	List of interested organizations and groups.					Exclude; Admin
51.123	Charges for environmental documents; distribution to public; distribution to governmental agencies.					Exclude; Admin
	Commenting					Heading
51.124	Commission duty to comment.					Exclude; Admin
	Responsible Official					Heading
51.125	Responsible official.					Exclude; Admin
Appendix A	to Subpart A--Format for Presentation of Material in Environmental Impact Statements					Exclude; Admin
Appendix B	to Subpart A--Environmental Effect of Renewing the Operating License of a Nuclear Power Plant	NA				Exclude

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	General Provisions					Heading
52.0	Scope; applicability of 10 CFR Chapter I provisions.					Exclude; Admin
52.1	Definitions.					Exclude; Admin
52.2	Interpretations.					Exclude; Admin
52.3	Written communications.					Exclude; Admin
52.4	Deliberate misconduct.					Exclude; Admin
52.5	Employee protection.					Exclude; Admin
52.6	Completeness and accuracy of information.					Exclude; Admin
52.7	Specific exemptions.					Exclude; Admin
52.8	Combining licenses; elimination of repetition.					Exclude; Admin
52.9	Jurisdictional limits.					Exclude; Admin
52.10	Attacks and destructive acts.					Exclude; Admin
52.11	Information collection requirements: OMB approval.					Exclude; Admin
	Subpart A—Early Site Permits					Heading
52.12	Scope of subpart.	NA				Exclude; early site permits.
52.13	Relationship to other subparts.	NA				Exclude; early site permits.
52.15	Filing of applications.	NA				Exclude; early site permits.
52.16	Contents of applications; general information.	NA				Exclude; early site permits.
52.17	Contents of applications; technical information.	NA				Exclude; early site permits.
52.18	Standards for review of applications.	NA				Exclude; early site permits.
52.21	Administrative review of applications; hearings.	NA				Exclude; early site permits.
52.23	Referral to the Advisory Committee on Reactor Safeguards (ACRS).	NA				Exclude; early site permits.
52.24	Issuance of early site permit.	NA				Exclude; early site permits.
52.25	Extent of activities permitted.	NA				Exclude; early site permits.
52.26	Duration of permit.	NA				Exclude; early site permits.
52.27	Limited work authorization after issuance of early site permit.	NA				Exclude; early site permits.
52.28	Transfer of early site permit.	NA				Exclude; early site permits.
52.29	Application for renewal.	NA				Exclude; early site permits.
52.31	Criteria for renewal.	NA				Exclude; early site permits.
52.33	Duration of renewal.	NA				Exclude; early site permits.

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.35	Use of site for other purposes.	NA				Exclude; early site permits.
52.39	Finality of early site permit determinations.	NA				Exclude; early site permits.
	Subpart B—Standard Design Certifications					Heading; standard design certification.
52.41	Scope of subpart.	NA				Exclude; standard design certification.
52.43	Relationship to other subparts.	NA				Exclude; standard design certification.
52.45	Filing of applications.	NA				Exclude; standard design certification.
52.46	Contents of applications; general information.	NA				Exclude; standard design certification.
52.47	Contents of applications; technical information.	NA				Exclude; standard design certification.
52.48	Standards for review of applications.	NA				Exclude; standard design certification.
52.51	Administrative review of applications.	NA				Exclude; standard design certification.
52.53	Referral to the Advisory Committee on Reactor Safeguards (ACRS).	NA				Exclude; standard design certification.
52.54	Issuance of standard design certification.	NA				Exclude; standard design certification.
52.55	Duration of certification.	NA				Exclude; standard design certification.
52.57	Application for renewal.	NA				Exclude; standard design certification.
52.59	Criteria for renewal.	NA				Exclude; standard design certification.
52.61	Duration of renewal.	NA				Exclude; standard design certification.
52.63	Finality of standard design certifications.	NA				Exclude; standard design certification.
	Subpart C—Combined Licenses					Heading
52.71	Scope of subpart.					Exclude; Admin
52.73	Relationship to other subparts.					Exclude; Admin
52.75	Filing of applications.					Exclude; Admin
52.77	Contents of applications; general information.					Exclude; Admin
52.79	Contents of applications; technical information in final safety analysis report.					
52.79(a)	(a) The application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components of the facility as a whole. The final safety analysis report shall include the following information, at a level of information sufficient to enable the Commission to reach a final					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	conclusion on all safety matters that must be resolved by the Commission before issuance of a combined license:					
52.79(a)(1)	(1)(i) The boundaries of the site; (ii) The proposed general location of each facility on the site; (iii) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated; (iv) The location and description of any nearby industrial, military, or transportation facilities and routes; (v) The existing and projected future population profile of the area surrounding the site; (vi) A description and safety assessment of the site on which the facility is to be located. The assessment must contain an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in paragraphs (a)(1)(vi)(A) and (a)(1)(vi)(B) of this section. In performing this assessment, an applicant shall assume a fission product release 5 from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that: (A) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem 6 total effective dose equivalent (TEDE). (B) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE; and					
52.79(a)(2)	(2) A description and analysis of the structures, systems, and components of the facility with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. It is expected that reactors will reflect through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The descriptions shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations. Items such		Yes			

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission: (i) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials; (ii) The extent to which generally accepted engineering standards are applied to the design of the reactor; (iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials; (iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release 7 from the core into the containment assuming that the facility is operated at the ultimate power level contemplated;					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.79(a)(3)	(3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter;					
52.79(a)(4)	(4) The design of the facility including: (i) The principal design criteria for the facility. Appendix A to part 50 of this chapter, "General Design Criteria for Nuclear Power Plants," establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units; (ii) The design bases and the relation of the design bases to the principal design criteria; (iii) Information relative to materials of construction, arrangement, and dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety.					
52.79(a)(5)	(5) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter;					
52.79(a)(6)	(6) A description and analysis of the fire protection design features for the reactor necessary to comply with 10 CFR part 50, appendix A, GDC 3, and § 50.48 of this chapter;					
52.79(a)(7)	(7) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in §§ 50.60 and 50.61(b)(1) and (b)(2) of this chapter;					
52.79(a)(8)	(8) An analysis and description of the equipment and systems for combustible gas control as required by § 50.44 of this chapter;					
52.79(a)(9)	(9) The coping analyses, and any design features necessary to address station blackout, as described in § 50.63 of this chapter;					
52.79(a)(10)	(10) A description of the program, and its implementation, required by § 50.49(a) of this chapter for the environmental qualification of electric equipment important to safety and the list of electric equipment important to safety that is required by 10 CFR 50.49(d);					
52.79(a)(11)	(11) A description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter;					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.79(a)(12)	(12) A description of the primary containment leakage rate testing program, and its implementation, necessary to ensure that the containment meets the requirements of appendix J to 10 CFR part 50;					
52.79(a)(13)	(13) A description of the reactor vessel material surveillance program required by appendix H to 10 CFR part 50 and its implementation;					
52.79(a)(14)	(14) A description of the operator training program, and its implementation, necessary to meet the requirements of 10 CFR part 55;					
52.79(a)(15)	(15) A description of the program, and its implementation, for monitoring the effectiveness of maintenance necessary to meet the requirements of § 50.65 of this chapter;					
52.79(a)(16)	(16)(i) The information with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, as described in § 50.34a(d) of this chapter; (ii) A description of the process and effluent monitoring and sampling program required by appendix I to 10 CFR part 50 and its implementation.					
52.79(a)(17)	(17) The information with respect to compliance with technically relevant positions of the Three Mile Island requirements in § 50.34(f) of this chapter, with the exception of §§ 50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v);					
52.79(a)(18)	(18) If the applicant seeks to use risk informed treatment of SSCs in accordance with § 50.69 of this chapter, the information required by § 50.69(b)(2) of this chapter;					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.79(a)(19)	(19) Information necessary to demonstrate that the plant complies with the earthquake engineering criteria in 10 CFR part 50, appendix S;					
52.79(a)(20)	(20) Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG–0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design;					
52.79(a)(21)	(21) Emergency plans complying with the requirements of § 50.47 of this chapter, and 10 CFR part 50, appendix E;					
52.79(a)(22)	(22)(i) All emergency plan certifications that have been obtained from the State and local governmental agencies with emergency planning responsibilities must state that: (A) The proposed emergency plans are practicable; (B) These agencies are committed to participating in any further development of the plans, including any required field demonstrations; and (C) These agencies are committed to executing their responsibilities under the plans in the event of an emergency; (ii) If certifications cannot be obtained after sustained, good faith efforts by the applicant, then the application must contain information, including a utility plan, sufficient to show that the proposed plans provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.					
52.79(a)(23)	(23) [Reserved]					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.79(a)(24)	(24) If the application is for a nuclear power reactor design which differs significantly from light-water reactor designs that were licensed before 1997 or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, the application must describe how the design meets the requirements in § 50.43(e) of this chapter;					
52.79(a)(25)	(25) A description of the quality assurance program, applied to the design, and to be applied to the fabrication, construction, and testing, of the structures, systems, and components of the facility. Appendix B to 10 CFR part 50 sets forth the requirements for quality assurance programs for nuclear power plants. The description of the quality assurance program for a nuclear power plant must include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 have been and will be satisfied, including a discussion of how the quality assurance program will be implemented;					
52.79(a)(26)	(26) The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements for operation;					
52.79(a)(27)	(27) Managerial and administrative controls to be used to assure safe operation. Appendix B to 10 CFR part 50 sets forth the requirements for these controls for nuclear power plants. The information on the controls to be used for a nuclear power plant shall include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 will be satisfied;					
52.79(a)(28)	(28) Plans for preoperational testing and initial operations;					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.79(a)(29)	(29)(i) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components; (ii) Plans for coping with emergencies, other than the plans required by § 52.79(a)(21);					
52.79(a)(30)	(30) Proposed technical specifications prepared in accordance with the requirements of §§ 50.36 and 50.36a of this chapter;					
52.79(a)(31)	(31) For nuclear power plants to be operated on multi-unit sites, an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multi-unit sites;					
52.79(a)(32)	(32) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter;					
52.79(a)(33)	(33) A description of the training program required by § 50.120 of this chapter and its implementation;					
52.79(a)(34)	(34) A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in § 55.59 of this chapter;					
52.79(a)(35)	(35)(i) A physical security plan, describing how the applicant will meet the requirements of 10 CFR part 73 (and 10 CFR part 11, if applicable, including the identification and description of jobs as required by § 11.11(a) of this chapter, at the proposed facility).					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR parts 11 and 73, if applicable; (ii) A description of the implementation of the physical security plan;					
52.79(a)(36)	(36)(i) A safeguards contingency plan in accordance with the criteria set forth in appendix C to 10 CFR part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for this type of license shall include the information contained in the applicant's safeguards contingency plan.8 (Implementing procedures required for this plan need not be submitted for approval.) (ii) A training and qualification plan in accordance with the criteria set forth in appendix B to 10 CFR part 73. (iii) A cyber security plan in accordance with the criteria set forth in § 73.54 of this chapter; (iv) A description of the implementation of the safeguards contingency plan, training and qualification plan, and cyber security plan; and (v) Each applicant who prepares a physical security plan, a safeguards contingency plan, a training and qualification plan, or a cyber security plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.79(a)(37)	(37) The information necessary to demonstrate how operating experience insights have been incorporated into the plant design;					
52.79(a)(38)	(38) For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass;					
52.79(a)(39)	(39) A description of the radiation protection program required by § 20.1101 of this chapter and its implementation.					
52.79(a)(40)	(40) A description of the fire protection program required by § 50.48 of this chapter and its implementation.					
52.79(a)(41)	(41) For applications for light-water-cooled nuclear power plant combined licenses, an evaluation of the facility against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement;					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.79(a)(42)	(42) Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram (ATWS) events in § 50.62 of this chapter;					
52.79(a)(43)	(43) Information demonstrating how the applicant will comply with requirements for criticality accidents in § 50.68 of this chapter;					
52.79(a)(44)	(44) A description of the fitness-for-duty program required by 10 CFR part 26 and its implementation.					
52.79(a)(45)	(45) The information required by § 20.1406 of this chapter.					
52.79(a)(46)	(46) A description of the plant-specific probabilistic risk assessment (PRA) and its results.					
52.79(a)(47)	(47) For applications for combined licenses which are subject to 10 CFR 50.150(a), the information required by 10 CFR 50.150(b).					
52.79(b)	(b) If the combined license application references an early site permit, then the following requirements apply: (1) The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the early site permit, provided, however, that the final safety analysis report must either include or incorporate by reference the early site permit site safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the design of the facility falls within the site characteristics and design parameters specified in the early site permit. (2) If the final safety analysis report does not demonstrate that design of the facility falls within					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	the site characteristics and design parameters, the application shall include a request for a variance that complies with the requirements of §§ 52.39 and 52.93. (3) The final safety analysis report must demonstrate that all terms and conditions that have been included in the early site permit, other than those imposed under § 50.36b, will be satisfied by the date of issuance of the combined license. Any terms or conditions of the early site permit that could not be met by the time of issuance of the combined license, must be set forth as terms or conditions of the combined license. (4) If the early site permit approves complete and integrated emergency plans, or major features of emergency plans, then the final safety analysis report must include any new or additional information that updates and corrects the information that was provided under § 52.17(b), and discuss whether the new or additional information materially changes the bases for compliance with the applicable requirements. The application must identify changes to the emergency plans or major features of emergency plans that have been incorporated into the proposed facility emergency plans and that constitute or would constitute a decrease in effectiveness under § 50.54(q) of this chapter. (5) If complete and integrated emergency plans are approved as part of the early site permit, new certifications meeting the requirements of paragraph (a)(22) of this section are not required.					
52.79(c)	(c) If the combined license application references a standard design approval, then the following					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	requirements apply: (1) The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design approval, provided, however, that the final safety analysis report must either include or incorporate by reference the standard design approval final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the characteristics of the site fall within the site parameters specified in the design approval. In addition, the plant-specific PRA information must use the PRA information for the design approval and must be updated to account for sitespecific design information and any design changes or departures. (2) The final safety analysis report must demonstrate that all terms and conditions that have been included in the final design approval will be satisfied by the date of issuance of the combined license.					
52.79(d)	(d) If the combined license application references a standard design certification, then the following requirements apply: (1) The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design certification, provided, however, that the final safety analysis report must either include or incorporate by reference the standard design certification final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	characteristics fall within the site parameters specified in the design certification. In addition, the plantspecific PRA information must use the PRA information for the design certification and must be updated to account for site-specific design information and any design changes or departures. (2) The final safety analysis report must demonstrate that the interface requirements established for the design under § 52.47 have been met. (3) The final safety analysis report must demonstrate that all requirements and restrictions set forth in the referenced design certification rule, other than those imposed under § 50.36b, must be satisfied by the date of issuance of the combined license. Any requirements and restrictions set forth in the referenced design certification rule that could not be satisfied by the time of issuance of the combined license, must be set forth as terms or conditions of the combined license.					
52.79(e)	(e) If the combined license application references the use of one or more manufactured nuclear power reactors licensed under subpart F of this part, then the following requirements apply: (1) The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the manufacturing license, provided, however, that the final safety analysis report must either include or incorporate by reference the manufacturing license final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site characteristics fall within the site parameters					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	specified in the manufacturing license. In addition, the plant-specific PRA information must use the PRA information for the manufactured reactor and must be updated to account for site-specific design information and any design changes or departures. (2) The final safety analysis report must demonstrate that the interface requirements established for the design have been met. (3) The final safety analysis report must demonstrate that all terms and conditions that have been included in the manufacturing license, other than those imposed under § 50.36b, will be satisfied by the date of issuance of the combined license. Any terms or conditions of the manufacturing license that could not be met by the time of issuance of the combined license, must be set forth as terms or conditions of the combined license.					
52.79(f)	(f) Each applicant for a combined license under this subpart shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.					
52.80	Contents of applications; additional technical information. The application must contain:					
52.80(a)	(a) The proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in conformity with the combined license, the provisions of the Act, and the					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	Commission's rules and regulations. (1) If the application references an early site permit with ITAAC, the early site permit ITAAC must apply to those aspects of the combined license which are approved in the early site permit. (2) If the application references a standard design certification, the ITAAC contained in the certified design must apply to those portions of the facility design which are approved in the design certification. (3) If the application references an early site permit with ITAAC or a standard design certification or both, the application may include a notification that a required inspection, test, or analysis in the ITAAC has been successfully completed and that the corresponding acceptance criterion has been met. The **Federal Register** notification required by § 52.85 must indicate that the application includes this notification.					
52.80(b)	(b) An environmental report, either in accordance with 10 CFR 51.50(c) if a limited work authorization under 10 CFR 50.10 is not requested in conjunction with the combined license application, or in accordance with §§ 51.49 and 51.50(c) of this chapter if a limited work authorization is requested in conjunction with the combined license application.					
52.80(c)	(c) If the applicant wishes to request that a limited work authorization under 10 CFR 50.10 be issued before issuance of the combined license, the application must include the information otherwise required by 10 CFR 50.10, in accordance with either 10 CFR 2.101(a)(1) through (a)(4), or 10 CFR 2.101(a)(9).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.80(d)	(d) A description and plans for implementation of the guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with the loss of large areas of the plant due to explosions or fire as required by § 50.54(hh)(2) of this chapter.					
52.81	Standards for review of applications.					Exclude; Admin
52.83	Finality of referenced NRC approvals; partial initial decision on site suitability.					Exclude; Admin
52.85	Administrative review of applications; hearings.					Exclude; Admin
52.87	Referral to the Advisory Committee on Reactor Safeguards (ACRS).					Exclude; Admin
52.89	Reserved.	NA				Not issued.
52.91	Authorization to conduct limited work authorization activities.					Exclude; Admin
52.93	Exemptions and variances.					Exclude; Admin
52.97	Issuance of combined licenses.					Exclude; Admin
52.98	Finality of combined licenses; information requests.					Exclude; Admin
52.99	Inspection during construction.					
52.99(a)	(a) The licensee shall submit to the NRC, no later that 1 year after issuance of the combined license or at the start of construction as defined in 10 CFR 50.10(a), whichever is later, its schedule for completing the inspections, tests, or analyses in the ITAAC. The licensee shall submit updates to the ITAAC schedules every 6 months thereafter and, within 1 year of its scheduled date for initial loading of fuel, the licensee shall submit updates to the ITAAC schedule every 30 days until the final notification is provided to the NRC under paragraph					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	(c)(1) of this section.					
52.99(b)	(b) With respect to activities subject to an ITAAC, an applicant for a combined license may proceed at its own risk with design and procurement activities, and a licensee may proceed at its own risk with design, procurement, construction, and pre-operational activities, even though the NRC may not have found that any one of the prescribed acceptance criteria have been met.					
52.99(c)	(c)(1) The licensee shall notify the NRC that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. The notification must contain sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. (2) If the licensee has not provided, by the date 225 days before the scheduled date for initial loading of fuel, the notification required by paragraph (c)(1) of this section for all ITAAC, then the licensee shall notify the NRC that the prescribed inspections, tests, or analyses for all uncompleted ITAAC will be performed and that the prescribed acceptance criteria will be met prior to operation. The notification must be provided no later than the date 225 days before the scheduled date for initial loading of fuel, and must provide sufficient					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	information to demonstrate that the prescribed inspections, tests, or analyses will be performed and the prescribed acceptance criteria for the uncompleted ITAAC will be met, including, but not limited to, a description of the specific procedures and analytical methods to be used for performing the prescribed inspections, tests, and analyses and determining that the prescribed acceptance criteria have been met.					
52.99(d)	(d)(1) In the event that an activity is subject to an ITAAC derived from a referenced standard design certification and the licensee has not demonstrated that the ITAAC has been met, the licensee may take corrective actions to successfully complete that ITAAC or request an exemption from the standard design certification ITAAC, as applicable. A request for an exemption must also be accompanied by a request for a license amendment under § 52.98(f). (2) In the event that an activity is subject to an ITAAC not derived from a referenced standard design certification and the licensee has not demonstrated that the ITAAC has been met, the licensee may take corrective actions to successfully complete that ITAAC or request a license amendment under § 52.98(f).					
52.99(e)	(e) The NRC shall ensure that the prescribed inspections, tests, and analyses in the ITAAC are performed.					

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(1) At appropriate intervals until the last date for submission of requests for hearing under § 52.103(a), the NRC shall publish notices in the **Federal Register** of the NRC staff's determination of the successful completion of inspections, tests, and analyses. (2) The NRC shall make publicly available the licensee notifications under paragraph (c)(1), and, no later than the date of publication of the notice of intended operation required by § 52.103(a), make available all licensee notifications under paragraphs (c)(1) and (c)(2) of this section.					
52.103	Operation under a combined license.					Exclude; Admin
52.104	Duration of combined license.					Exclude; Admin
52.105	Transfer of combined license.					Exclude; Admin
52.107	Application for renewal.					Exclude; Admin
52.109	Continuation of combined license.					Exclude; Admin
52.110	Termination of license.					Exclude; Admin
	Subpart D—Reserved	NA				Not issued.
	Subpart E—Standard Design Approvals	NA				Exclude; Standard design approval not applicable
52.131	Scope of subpart.	NA				Exclude; standard design approval.
52.133	Relationship to other subparts.	NA				Exclude; standard design approval.
52.135	Filing of applications.	NA				Exclude; standard design approval.
52.136	Contents of applications; general information.	NA				Exclude; standard design approval.
52.137	Contents of applications; technical information.	NA				Exclude; standard design approval.
52.139	Standards for review of applications.	NA				Exclude; standard design approval.
52.141	Referral to the Advisory Committee on Reactor Safeguards (ACRS).	NA				Exclude; standard design approval.
52.143	Staff approval of design.	NA				Exclude; standard design approval.

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
52.145	Finality of standard design approvals; information requests.	NA				Exclude; standard design approval.
52.147	Duration of design approval.	NA				Exclude; standard design approval.
	Subpart F—Manufacturing Licenses	NA				Exclude; Heading; manufacturing license not applicable.
52.151	Scope of subpart.	NA				Exclude; manufacturing license.
52.153	Relationship to other subparts.	NA				Exclude; manufacturing license.
52.155	Filing of applications.	NA				Exclude; manufacturing license.
52.156	Contents of applications; general information.	NA				Exclude; manufacturing license.
52.157	Contents of applications; technical information in final safety analysis report.	NA				Exclude; manufacturing license.
52.158	Contents of application; additional technical information.	NA				Exclude; manufacturing license.
52.159	Standards for review of application.	NA				Exclude; manufacturing license.
52.161	Reserved.	NA				Not issued.
52.163	Administrative review of applications; hearings.	NA				Exclude; manufacturing license.
52.165	Referral to the Advisory Committee on Reactor Safeguards (ACRS).	NA				Exclude; manufacturing license.
52.167	Issuance of manufacturing license.	NA				Exclude; manufacturing license.
52.169	Reserved.	NA				Not issued.
52.171	Finality of manufacturing licenses; information requests.	NA				Exclude; manufacturing license.
52.173	Duration of manufacturing license.	NA				Exclude; manufacturing license.
52.175	Transfer of manufacturing license.	NA				Exclude; manufacturing license.
52.177	Application for renewal.	NA				Exclude; manufacturing license.
52.179	Criteria for renewal.	NA				Exclude; manufacturing license.
52.181	Duration of renewal.	NA				Exclude; manufacturing license.
	Subpart G—Reserved	NA				Not issued.
	Subpart H—Enforcement					Heading
52.301	Violations.					Exclude; Admin
52.303	Criminal penalties.					Exclude; Admin
52 App. A	Design Certification Rule for the U.S. Advanced Boiling Water Reactor	NA				Exclude

Table A1-5: PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
52 App. B	Design Certification Rule for the System 80+ Design	NA				Exclude
52 App. C	Design Certification Rule for the AP600 Design	NA				Exclude
52 App. D	Design Certification Rule for the AP1000 Design	NA				Exclude
	Appendixes E Through M to Part 52 [Reserved]	NA				Not issued.
52 App. N	Standardization of Nuclear Power Plant Designs: Combined Licenses to Construct and Operate Nuclear Power Reactors of Identical Design at Multiple Sites					Exclude; Admin

Table A1-6: PART 55--OPERATORS' LICENSES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Subpart A--General Provisions					Heading
55.1	Purpose.					Exclude; Admin
55.2	Scope.					Exclude; Admin
55.3	License requirements.					Exclude; Admin
55.4	Definitions.					Exclude; Admin; definitions includes definition of "plant-referenced simulator."
55.5	Communications.					Exclude; Admin
55.6	Interpretations.					Exclude; Admin
55.7	Additional requirements.					Exclude; Admin
55.8	Information collection requirements: OMB approval.					Exclude; Admin
55.9	Completeness and accuracy of information.					Exclude; Admin
	Subpart B--Exemptions					Heading
55.11	Specific exemptions.					Exclude; Admin
55.13	General exemptions.					Exclude; Admin
	Subpart C--Medical Requirements					Heading
55.21	Medical examination.					Exclude; Admin
55.23	Certification.					Exclude; Admin
55.25	Incapacitation because of disability or illness.					Exclude; Admin
55.27	Documentation.					Exclude; Admin
	Subpart D--Applications					Heading
55.31	How to apply.					Exclude; Admin; includes requirements for control manipulation at the facility or a plant-referenced simulator.
55.33	Disposition of an initial application.					Exclude; Admin
55.35	Re-applications.					Exclude; Admin
	Subpart E--Written Examinations and Operating Tests					Heading
55.40	Implementation.					Exclude; Admin; reference is made to NUREG-1021 which contains broad guidance as well as some specific guidance on LWRs.
55.41	Written examination: Operators.					Exclude; Admin
55.43	Written examination: Senior operators.					Exclude; Admin
55.45	Operating tests.					Exclude; Admin

Table A1-6: PART 55--OPERATORS' LICENSES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
55.46	Simulation facilities.					Defines simulator requirements. Specific requirements listed in following sections.
55.46(a)	(a) General. This section addresses the use of a simulation facility for the administration of the operating test and plant-referenced simulators to meet experience requirements for applicants for operator and senior operator licenses.					
55.46(b)	(b) Commission-approved simulation facilities and Commission approval of use of the plant in the administration of the operating test. (1) Facility licensees that propose to use a simulation facility, other than a plant-referenced simulator, or the plant in the administration of the operating test under §§ 55.45(b)(1) or 55.45(b)(3), shall request approval from the Commission. This request must include: (i) A description of the components of the simulation facility intended to be used, or the way the plant would be used for each part of the operating test, unless previously approved; and (ii) A description of the performance tests for the simulation facility as part of the request, and the results of these tests; and (iii) A description of the procedures for maintaining examination and test integrity consistent with the requirements of § 55.49. (2) The Commission will approve a simulation facility or use of the plant for administration of operating tests if it finds that the simulation facility and its proposed use, or the proposed use of the plant, are suitable for the conduct of operating tests for the facility licensee's reference plant under §					

Table A1-6: PART 55--OPERATORS' LICENSES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	55.45(a).					
55.46(c)	(c) Plant-referenced simulators. (1) A plant-referenced simulator used for the administration of the operating test or to meet experience requirements in § 55.31(a)(5) must demonstrate expected plant response to operator input and to normal, transient, and accident conditions to which the simulator has been designed to respond. The plant-referenced simulator must be designed and implemented so that it: (i) Is sufficient in scope and fidelity to allow conduct of the evolutions listed in §§ 55.45(a)(1) through (13), and 55.59(c)(3)(i)(A) through (AA), as applicable to the design of the reference plant. (ii) Allows for the completion of control manipulations for operator license applicants. (2) Facility licensees that propose to use a plant-referenced simulator to meet the control manipulation requirements in § 55.31(a)(5) must ensure that: (i) The plant-referenced simulator utilizes models relating to nuclear and thermal-hydraulic characteristics that replicate the most recent core load in the nuclear power reference plant for which a license is being sought; and (ii) Simulator fidelity has been demonstrated so that significant control manipulations are completed without procedural exceptions, simulator performance exceptions, or deviation from the approved training scenario sequence.					

Table A1-6: PART 55--OPERATORS' LICENSES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(3) A simulation facility consisting solely of a plant-referenced simulator must meet the requirements of paragraph (c)(1) of this section and the criteria in paragraphs (d)(1) and (4) of this section for the Commission to accept the plant-referenced simulator for conducting operating tests as described in § 55.45(a) of this part, requalification training as described in § 55.59(c)(3) of this part, or for performing control manipulations that affect reactivity to establish eligibility for an operator's license as described in § 55.31(a)(5).					
55.46(d)	(d) Continued assurance of simulator fidelity. Facility licensees that maintain a simulation facility shall: (1) Conduct performance testing throughout the life of the simulation facility in a manner sufficient to ensure that paragraphs (c)(2)(ii), as applicable, and (d)(3) of this section are met. The results of performance tests must be retained for four years after the completion of each performance test or until superseded by updated test results; (2) Correct modeling and hardware discrepancies and discrepancies identified from scenario validation and from performance testing; (3) Make results of any uncorrected performance test failures that may exist at the time of the operating test or requalification program inspection available for NRC review, prior to or concurrent with preparations for each operating test or requalification program inspection; and (4) Maintain the provisions for license application,					

Table A1-6: PART 55--OPERATORS' LICENSES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	examination, and test integrity consistent with § 55.49.					
55.47	Waiver of examination and test requirements.					Exclude; Admin
55.49	Integrity of examinations and tests.					Exclude; Admin
	Subpart F--Licenses					Heading
55.51	Issuance of licenses.					Exclude; Admin
55.53	Conditions of licenses.					Exclude; Admin
55.55	Expiration.					Exclude; Admin
55.57	Renewal of licenses.					Exclude; Admin
55.59	Requalification. [Only 55.59(c)(3)(i), which contains technology specific information is included for evaluation.]					Some portions of this regulation include technology specific language.
55.59 (c)(3)(i)	(3) On-the-job training. The requalification program must include on-the-job training so that -- (i) Each licensed operator of a utilization facility manipulates the plant controls and each licensed senior operator either manipulates the controls or directs the activities of individuals during plant control manipulations during the term of the licensed operator's or senior operator's license. For reactor operators and senior operators, these manipulations must consist of the following control manipulations and plant evolutions if they are applicable to the plant design. Items described in paragraphs (c)(3)(i) (A) through (L) of this section must be performed annually; all other items must be performed on a two-year cycle. However, the requalification programs must contain a commitment that each individual shall perform or participate in a combination of reactivity control manipulations based on the availability of plant equipment and systems. Those control manipulations which are not performed at the plant					

Table A1-6: PART 55--OPERATORS' LICENSES

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	may be performed on a simulator. The use of the Technical Specifications should be maximized during the simulator control manipulations. Senior operator licensees are credited with these activities if they direct control manipulations as they are performed. (A) Plant or reactor startups to include a range that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established. (B) Plant shutdown. (C) Manual control of steam generators or feedwater or both during startup and shutdown. (D) Boration or dilution during power operation. (E) Significant (¬10 percent) power changes in manual rod control or recirculation flow. (F) Reactor power change of 10 percent or greater where load change is performed with load limit control or where flux, temperature, or speed control is on manual (for HTGR). (G) Loss of coolant, including -- (1) Significant PWR steam generator leaks (2) Inside and outside primary containment (3) Large and small, including lead-rate determination (4) Saturated reactor coolant response (PWR). (H) Loss of instrument air (if simulated plant specific). (I) Loss of electrical power (or degraded power sources). (J) Loss of core coolant flow/natural circulation. (K) Loss of feedwater (normal and emergency). (L) Loss of service water, if required for safety. (M) Loss of shutdown cooling. (N) Loss of component cooling system or cooling to					

Table A1-6: PART 55--OPERATORS' LICENSES						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	an individual component. (O) Loss of normal feedwater or normal feedwater system failure. (P) Loss of condenser vacuum. (Q) Loss of protective system channel. (R) Mispositioned control rod or rods (or rod drops). (S) Inability to drive control rods. (T) Conditions requiring use of emergency boration or standby liquid control system. (U) Fuel cladding failure or high activity in reactor coolant or offgas. (V) Turbine or generator trip. (W) Malfunction of an automatic control system that affects reactivity. (X) Malfunction of reactor coolant pressure/volume control system. (Y) Reactor trip. (Z) Main steam line break (inside or outside containment). (AA) A nuclear instrumentation failure.					
	Subpart G--Modification and Revocation of Licenses					Heading
55.61	Modification and revocation of licenses.					Exclude; Admin
	Subpart H—Enforcement					Heading
55.71	Violations.					Exclude; Admin
55.73	Criminal penalties.					Exclude; Admin

NOTE: 10 CFR 70 does not require a review under the scope of the Procedure for Performing The Regulatory Gap Analysis

Table A1-7: PART 70--DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Subpart A--General Provisions					Heading
70.1	Purpose.					Exclude; Admin
70.2	Scope.					Exclude; Admin
70.3	License requirements.					Exclude; Admin
70.4	Definitions.					Exclude; Admin
70.5	Communications.					Exclude; Admin
70.6	Interpretations.					Exclude; Admin
70.7	Employee protection.					Exclude; Admin
70.8	Information collection requirements: OMB approval.					Exclude; Admin
70.9	Completeness and accuracy of information.					Exclude; Admin
70.10	Deliberate misconduct.					Exclude; Admin
	Subpart B--Exemptions					Heading
70.11	Persons using special nuclear material under certain Department of Energy and Nuclear Regulatory Commission contracts.	NA				Exclude
70.12	Carriers.	NA				Exclude
70.13	Department of Defense.	NA				Exclude
70.14	Foreign military aircraft.	NA				Exclude
70.17	Specific exemptions.					Exclude; Admin
	Subpart C--General Licenses					Heading
70.18	Types of licenses.					Exclude; Admin
70.19	General license for calibration or reference sources.					Exclude; Admin
70.20	General license to own special nuclear material.					Exclude; Admin
70.20a	General license to possess special nuclear material for transport.					Exclude; Admin

Table A1-7: PART 70--DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
70.20b	General license for carriers of transient shipments of formula quantities of strategic special nuclear material, special nuclear material of moderate strategic significance, special nuclear material of low strategic significance, and irradiated reactor fuel.					Exclude; Admin
	Subpart D--License Applications					Heading
70.21	Filing.					Exclude; Admin; note that 50.31 allows the filing of a Part 70 application to be made as part of the facility license application.
70.22	Contents of applications.					Exclude; all information required is also included under Part 50 or Part 52 license applications.
70.23	Requirements for the approval of applications.					Exclude; Admin
70.23a	Hearing required for uranium enrichment facility.	NA				Exclude
70.24	Criticality accident requirements.	NA				Exclude; the requirements of 10 CFR 50.68(b) apply instead.
70.25	Financial assurance and recordkeeping for decommissioning.	NA				Exclude
	Subpart E--Licenses					Heading
70.31	Issuance of licenses.					Exclude; Admin
70.32	Conditions of licenses.					Exclude; Admin
70.33	Renewal of licenses.					Exclude; Admin
70.34	Amendment of licenses.					Exclude; Admin
70.35	Commission action on applications to renew or amend.					Exclude; Admin
70.36	Inalienability of licenses.					Exclude; Admin
70.37	Disclaimer of warranties.					Exclude; Admin
70.38	Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas.					Exclude; Admin
70.39	Specific licenses for the manufacture or initial transfer of calibration or reference sources.	NA				Exclude
	Subpart F--Acquisition, Use, and Transfer of					Heading

Table A1-7: PART 70--DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Special Nuclear Material, Creditors' Rights					
70.40	Ineligibility of certain applicants.					Exclude; Admin
70.41	Authorized use of special nuclear material.					Exclude; Admin
70.42	Transfer of special nuclear material.					Exclude; Admin
70.44	Creditor regulations.					Exclude; Admin
	Subpart G--Special Nuclear Material Control Records, Reports, and Inspections					Heading
70.50	Reporting requirements.	NA				Exclude; 10 CFR 50.72 takes precedent.
70.51	Records requirements.					Exclude; Admin
70.52	Reports of accidental criticality.					Exclude; Admin
70.55	Inspections.					Exclude; Admin
70.56	Tests.					Exclude; Admin
70.59	Effluent monitoring reporting requirements.	NA				Exclude
	Subpart H--Additional Requirements for Certain Licensees Authorized To Possess a Critical Mass of Special Nuclear Material					Heading
70.60	Applicability.	NA				Exclude
70.61	Performance requirements.	NA				Exclude
70.62	Safety program and integrated safety analysis.	NA				Exclude
70.64	Requirements for new facilities or new processes at existing facilities.	NA				Exclude
70.65	Additional content of applications.	NA				Exclude
70.66	Additional requirements for approval of license application.	NA				Exclude
70.72	Facility changes and change process.	NA				Exclude
70.73	Renewal of licenses.	NA				Exclude
70.74	Additional reporting requirements.	NA				Exclude
70.76	Backfitting	NA				Exclude
	Subpart I--Modification and Revocation of Licenses					Heading
70.81	Modification and revocation of licenses.					Exclude; Admin
70.82	Suspension and operation in war or national emergency.					Exclude; Admin
	Subpart J--Enforcement					Heading

Table A1-7: PART 70--DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
70.91	Violations.					Exclude; Admin
70.92	Criminal penalties.					Exclude; Admin
70.App.A	Reportable Safety Events					Exclude; Admin

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	General Provisions					Heading
73.1	Purpose and scope.					Exclude; provides extensive description of scope.
73.2	Definitions.					Exclude; Admin
73.3	Interpretations.					Exclude; Admin
73.4	Communications.					Exclude; Admin
73.5	Specific exemptions.					Exclude; Admin
73.6	Exemptions for certain quantities and kinds of special nuclear material.					Exclude; Admin
73.8	Information collection requirements: OMB approval.					Exclude; Admin
73.20	General performance objective and requirements.	NA				Exclude; exempt in accordance with 10 CFR 73.6.
73.21	Protection of Safeguards Information: Performance Requirements.					Exclude; safeguards information not being reviewed.
73.22	Protection of Safeguards Information: Specific Requirements					Exclude; safeguards information not being reviewed.
73.23	Protection of Safeguards Information—Modified Handling: Specific Requirements.					Exclude; safeguards information not being reviewed.
73.24	Prohibitions.					Exclude; Admin
	Physical Protection of Special Nuclear Material in Transit					Heading
73.25	Performance capabilities for physical protection of strategic special nuclear material in transit.	NA				Exclude; exempt in accordance with 10 CFR 73.6.
73.26	Transportation physical protection systems, subsystems, components, and procedures.	NA				Exclude; exempt in accordance with 10 CFR 73.6.
73.27	Notification requirements.	NA				Exclude; exempt in accordance with 10 CFR 73.6.
73.28	Security background checks for secure transfer of nuclear materials.					Exclude; Admin
73.37	Requirements for physical protection of irradiated reactor fuel in transit.					Exclude; Admin
	Physical Protection Requirements at Fixed Sites					Heading
73.40	Physical protection: General requirements at fixed					Exclude; Admin

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	sites.					
73.45	Performance capabilities for fixed site physical protection systems.	NA				Exclude; exempt in accordance with 10 CFR 73.6.
73.46	Fixed site physical protection systems, subsystems, components, and procedures.	NA				Exclude; exempt in accordance with 10 CFR 73.6.
73.50	Requirements for physical protection of licensed activities.	NA				Exclude; wording excludes Part 50 and Part 52 licensees.
73.51	Requirements for the physical protection of stored spent nuclear fuel and high-level radioactive waste.	NA				Exclude; wording in regulation states that the section is applicable to IFSIs, Monitored Retrievable Storage facilities, and geologic repositories.
73.54	Protection of digital computer and communication systems and networks. By November 23, 2009 each licensee currently licensed to operate a nuclear power plant under part 50 of this chapter shall submit, as specified in § 50.4 and § 50.90 of this chapter, a cyber security plan that satisfies the requirements of this section for Commission review and approval. Each submittal must include a proposed implementation schedule. Implementation of the licensee's cyber security program must be consistent with the approved schedule. **[. . . deleted requirements for submittals prior to November 23, 2009.]**					
73.54(a)	(a) Each licensee subject to the requirements of this section shall provide high assurance that digital computer and communication systems and networks are adequately protected against cyber attacks, up to and including the design basis threat as described in § 73.1. (1) The licensee shall protect digital computer and communication systems and networks associated					

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	with: (i) Safety-related and important-to-safety functions; (ii) Security functions; (iii) Emergency preparedness functions, including offsite communications; and (iv) Support systems and equipment which, if compromised, would adversely impact safety, security, or emergency preparedness functions. (2) The licensee shall protect the systems and networks identified in paragraph (a)(1) of this section from cyber attacks that would: (i) Adversely impact the integrity or confidentiality of data and/or software; (ii) Deny access to systems, services, and/or data; and (iii) Adversely impact the operation of systems, networks, and associated equipment.					
73.54(b)	(b) To accomplish this, the licensee shall: (1) Analyze digital computer and communication systems and networks and identify those assets that must be protected against cyber attacks to satisfy paragraph (a) of this section, (2) Establish, implement, and maintain a cyber security program for the protection of the assets					

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	identified in paragraph (b)(1) of this section; and (3) Incorporate the cyber security program as a component of the physical protection program.					
73.54(c)	(c) The cyber security program must be designed to: (1) Implement security controls to protect the assets identified by paragraph (b)(1) of this section from cyber attacks; (2) Apply and maintain defense-in-depth protective strategies to ensure the capability to detect, respond to, and recover from cyber attacks; (3) Mitigate the adverse affects of cyber attacks; and (4) Ensure that the functions of protected assets identified by paragraph (b)(1) of this section are not adversely impacted due to cyber attacks.					
73.54(d)	(d) As part of the cyber security program, the licensee shall: (1) Ensure that appropriate facility personnel, including contractors, are aware of cyber security requirements and receive the training necessary to perform their assigned duties and responsibilities. (2) Evaluate and manage cyber risks. (3) Ensure that modifications to assets, identified by paragraph (b)(1) of this section, are evaluated before implementation to ensure that the cyber					

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	security performance objectives identified in paragraph (a)(1) of this section are maintained.					
73.54(e)	(e) The licensee shall establish, implement, and maintain a cyber security plan that implements the cyber security program requirements of this section. (1) The cyber security plan must describe how the requirements of this section will be implemented and must account for the site-specific conditions that affect implementation. (2) The cyber security plan must include measures for incident response and recovery for cyber attacks. The cyber security plan must describe how the licensee will: (i) Maintain the capability for timely detection and response to cyber attacks; (ii) Mitigate the consequences of cyber attacks; (iii) Correct exploited vulnerabilities; and (iv) Restore affected systems, networks, and/or equipment affected by cyber attacks.					
73.54(f)	(f) The licensee shall develop and maintain written policies and implementing procedures to implement the cyber security plan. Policies, implementing procedures, site-specific analysis, and other supporting technical information used by the licensee need not be submitted for Commission review and approval as part of the cyber security plan but are subject to inspection by NRC staff on a periodic basis.					

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
73.54(g)	(g) The licensee shall review the cyber security program as a component of the physical security program in accordance with the requirements of § 73.55(m), including the periodicity requirements.					
73.54(h)	(h) The licensee shall retain all records and supporting technical documentation required to satisfy the requirements of this section as a record until the Commission terminates the license for which the records were developed, and shall maintain superseded portions of these records for at least three (3) years after the record is superseded, unless otherwise specified by the Commission					
73.55	Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage.					
73.55(a)	(a) Introduction. (1) By March 31, 2010, each nuclear power reactor licensee, licensed under 10 CFR part 50, shall implement the requirements of this section through its Commission-approved Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Cyber Security Plan referred to collectively hereafter as ‘‘security plans.’’ Current applicants for an operating license under 10 CFR part 50, or combined license under 10 CFR part 52 who have submitted their applications to the Commission prior to the effective date of this rule must amend their applications to include security plans consistent with this section. (2) The security plans must identify, describe, and account for site-specific conditions that affect the licensee's capability to satisfy the requirements of this section.					

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	(3) The licensee is responsible for maintaining the onsite physical protection program in accordance with Commission regulations through the implementation of security plans and written security implementing procedures. (4) Applicants for an operating license under the provisions of part 50 of this chapter or holders of a combined license under the provisions of part 52 of this chapter, shall implement the requirements of this section before fuel is allowed onsite (protected area). (5) The Tennessee Valley Authority Watts Bar Nuclear Plant, Unit 2, holding a current construction permit under the provisions of part 50 of this chapter, shall meet the revised requirements in paragraphs (a) through (r) of this section as applicable to operating nuclear power reactor facilities. (6) Applicants for an operating license under the provisions of part 50 of this chapter, or holders of a combined license under the provisions of part 52 of this chapter that do not reference a standard design certification or reference a standard design certification issued after May 26, 2009 shall meet the requirement of § 73.55(i)(4)(iii).					
73.55(b)	(b) General performance objective and requirements. (1) The licensee shall establish and maintain a physical protection program, to include a security organization, which will have as its objective to provide high assurance that activities					

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. (2) To satisfy the general performance objective of paragraph (b)(1) of this section, the physical protection program must protect against the design basis threat of radiological sabotage as stated in § 73.1. (3) The physical protection program must be designed to prevent significant core damage and spent fuel sabotage. Specifically, the program must: (i) Ensure that the capabilities to detect, assess, interdict, and neutralize threats up to and including the design basis threat of radiological sabotage as stated in § 73.1, are maintained at all times. (ii) Provide defense-in-depth through the integration of systems, technologies, programs, equipment, supporting processes, and implementing procedures as needed to ensure the effectiveness of the physical protection program. (4) The licensee shall analyze and identify site-specific conditions, including target sets, that may affect the specific measures needed to implement the requirements of this section and shall account for these conditions in the design of the physical protection program. (5) Upon the request of an authorized					

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	representative of the Commission, the licensee shall demonstrate the ability to meet Commission requirements through the implementation of the physical protection program, including the ability of armed and unarmed personnel to perform assigned duties and responsibilities required by the security plans and licensee procedures. (6) The licensee shall establish, maintain, and implement a performance evaluation program in accordance with appendix B to this part, to demonstrate and assess the effectiveness of armed responders and armed security officers to implement the licensee's protective strategy. (7) The licensee shall establish, maintain, and implement an access authorization program in accordance with § 73.56 and shall describe the program in the Physical Security Plan. (8) The licensee shall establish, maintain, and implement a cyber security program in accordance with§ 73.54. (9) The licensee shall establish, maintain, and implement an insider mitigation program and shall describe the program in the Physical Security Plan. (i) The insider mitigation program must monitor the initial and continuing trustworthiness and reliability of individuals granted or retaining unescorted access authorization to a protected or vital area, and implement defense-in-depth methodologies to minimize the potential for an insider to adversely					

Table A1-8: PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	affect, either directly or indirectly, the licensee's capability to prevent significant core damage and spent fuel sabotage. (ii) The insider mitigation program must contain elements from: (A) The access authorization program described in § 73.56; (B) The fitness-for-duty program described in part 26 of this chapter; (C) The cyber security program described in § 73.54; and (D) The physical protection program described in this section. (10) The licensee shall use the site corrective action program to track, trend, correct and prevent recurrence of failures and deficiencies in the physical protection program. (11) Implementation of security plans and associated procedures must be coordinated with other onsite plans and procedures to preclude conflict during both normal and emergency conditions.					
73.55(c)	(c) Security plans. (1) Licensee security plans must describe: (i) How the licensee will implement requirements of this section through the establishment and					

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	maintenance of a security organization, the use of security equipment and technology, the training and qualification of security personnel, the implementation of predetermined response plans and strategies, and the protection of digital computer and communication systems and networks. (ii) Site-specific conditions that affect how the licensee implements Commission requirements. (2) Protection of Security Plans. The licensee shall protect the security plans and other security-related information against unauthorized disclosure in accordance with the requirements of § 73.21. (3) Physical Security Plan. The licensee shall establish, maintain, and implement a Physical Security Plan which describes how the performance objective and requirements set forth in this section will be implemented. (4) Training and Qualification Plan. The licensee shall establish, maintain, and implement, and follow a Training and Qualification Plan that describes how the criteria set forth in appendix B, to this part, ‘‘General Criteria for Security Personnel,’’ will be implemented. (5) Safeguards Contingency Plan. The licensee shall establish, maintain, and implement a Safeguards Contingency Plan that describes how the criteria set forth in appendix C, to this part,‘‘Licensee Safeguards Contingency Plans,’’ will					

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	be implemented. (6) Cyber Security Plan. The licensee shall establish, maintain, and implement a Cyber Security Plan that describes how the criteria set forth in § 73.54 ‘‘Protection of Digital Computer and Communication systems and Networks’’ of this part will be implemented. (7) Security implementing procedures. (i) The licensee shall have a management system to provide for the development, implementation, revision, and oversight of security procedures that implement Commission requirements and the security plans. (ii) Implementing procedures must document the structure of the security organization and detail the types of duties, responsibilities, actions, and decisions to be performed or made by each position of the security organization. (iii) The licensee shall: (A) Provide a process for the written approval of implementing procedures and revisions by the individual with overall responsibility for the security program. (B) Ensure that revisions to security implementing procedures satisfy the requirements of this section. (iv) Implementing procedures need not be					

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	submitted to the Commission for approval, but are subject to inspection by the Commission.					
73.55(d)	(d) Security organization. (1) The licensee shall establish and maintain a security organization that is designed, staffed, trained, qualified, and equipped to implement the physical protection program in accordance with the requirements of this section. (2) The security organization must include: (i) A management system that provides oversight of the onsite physical protection program. (ii) At least one member, onsite and available at all times, who has the authority to direct the activities of the security organization and who is assigned no other duties that would interfere with this individual's ability to perform these duties in accordance with the security plans and the licensee protective strategy. (3) The licensee may not permit any individual to implement any part of the physical protection program unless the individual has been trained, equipped, and qualified to perform their assigned duties and responsibilities in accordance with appendix B to this part and the Training and Qualification Plan. Non-security personnel may be assigned duties and responsibilities required to implement the physical protection program and shall: (i) Be trained through established licensee training					

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	programs to ensure each individual is trained, qualified, and periodically re-qualified to perform assigned duties. (ii) Be properly equipped to perform assigned duties. (iii) Possess the knowledge, skills, and abilities, to include physical attributes such as sight and hearing, required to perform their assigned duties and responsibilities.					
73.55(e)	(e) Physical barriers. Each licensee shall identify and analyze site-specific conditions to determine the specific use, type, function, and placement of physical barriers needed to satisfy the physical protection program design requirements of § 73.55(b). (1) The licensee shall: (i) Design, construct, install and maintain physical barriers as necessary to control access into facility areas for which access must be controlled or denied to satisfy the physical protection program design requirements of paragraph (b) of this section. (ii) Describe in the security plan, physical barriers, barrier systems, and their functions within the physical protection program. (2) The licensee shall retain, in accordance with § 73.70, all analyses and descriptions of the physical barriers and barrier systems used to satisfy the					

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	requirements of this section, and shall protect these records in accordance with the requirements of § 73.21. (3) Physical barriers must: (i) Be designed and constructed to: (A) Protect against the design basis threat of radiological sabotage; (B) Account for site-specific conditions; and (C) Perform their required function in support of the licensee physical protection program. (ii) Provide deterrence, delay, or support access control. (iii) Support effective implementation of the licensee's protective strategy. (4) Consistent with the stated function to be performed, openings in any barrier or barrier system established to meet the requirements of this section must be secured and monitored to prevent exploitation of the opening. (5) Bullet Resisting Physical Barriers. The reactor control room, the central alarm station, and the location within which the last access control function for access to the protected area is performed, must be bullet-resisting.					

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	(6) Owner controlled area. The licensee shall establish and maintain physical barriers in the owner controlled area as needed to satisfy the physical protection program design requirements of § 73.55(b). (7) Isolation zone. (i) An isolation zone must be maintained in outdoor areas adjacent to the protected area perimeter barrier. The isolation zone shall be: (A) Designed and of sufficient size to permit observation and assessment of activities on either side of the protected area barrier; (B) Monitored with intrusion detection equipment designed to satisfy the requirements of § 73.55(i) and be capable of detecting both attempted and actual penetration of the protected area perimeter barrier before completed penetration of the protected area perimeter barrier; and (C) Monitored with assessment equipment designed to satisfy the requirements of § 73.55(i) and provide real-time and play-back/recorded video images of the detected activities before and after each alarm annunciation. (ii) Obstructions that could prevent the licensee's capability to meet the observation and assessment requirements of this section must be located outside of the isolation zone.					

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	(8) Protected area. (i) The protected area perimeter must be protected by physical barriers that are designed and constructed to: (A) Limit access into the protected area to only those personnel, vehicles, and materials required to perform official duties; (B) Channel personnel, vehicles, and materials to designated access control portals; and (C) Be separated from any other barrier designated as a vital area physical barrier, unless otherwise identified in the Physical Security Plan. (ii) Penetrations through the protected area barrier must be secured and monitored in a manner that prevents or delays, and detects the exploitation of any penetration. (iii) All emergency exits in the protected area must be alarmed and secured by locking devices that allow prompt egress during an emergency and satisfy the requirements of this section for access control into the protected area. (iv) Where building walls or roofs comprise a portion of the protected area perimeter barrier, an isolation zone is not necessary provided that the detection and, assessment requirements of this section are met, appropriate barriers are installed, and the area is described in the security plans.					

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	(v) All exterior areas within the protected area, except for areas that must be excluded for safety reasons, must be periodically checked to detect and deter unauthorized personnel, vehicles, and materials. (9) Vital areas. (i) Vital equipment must be located only within vital areas, which must be located within a protected area so that access to vital equipment requires passage through at least two physical barriers, except as otherwise approved by the Commission and identified in the security plans. (ii) The licensee shall protect all vital area access portals and vital area emergency exits with intrusion detection equipment and locking devices that allow rapid egress during an emergency and satisfy the vital area entry control requirements of this section. (iii) Unoccupied vital areas must be locked and alarmed. (iv) More than one vital area may be located within a single protected area. (v) At a minimum, the following shall be considered vital areas: (A) The reactor control room; (B) The spent fuel pool;					

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	(C) The central alarm station; and (D) The secondary alarm station in accordance with § 73.55(i)(4)(iii). (vi) At a minimum, the following shall be located within a vital area: (A) The secondary power supply systems for alarm annunciation equipment; and (B) The secondary power supply systems for non-portable communications equipment. (10) Vehicle control measures. Consistent with the physical protection program design requirements of § 73.55(b), and in accordance with the site-specific analysis, the licensee shall establish and maintain vehicle control measures, as necessary, to protect against the design basis threat of radiological sabotage vehicle bomb assault. (i) Land vehicles. Licensees shall: (A) Design, construct, install, and maintain a vehicle barrier system, to include passive and active barriers, at a stand-off distance adequate to protect personnel, equipment, and systems necessary to prevent significant core damage and spent fuel sabotage against the effects of the design basis threat of radiological sabotage land vehicle bomb assault.					

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	(B) Periodically check the operation of active vehicle barriers and provide a secondary power source, or a means of mechanical or manual operation in the event of a power failure, to ensure that the active barrier can be placed in the denial position to prevent unauthorized vehicle access beyond the required standoff distance. (C) Provide periodic surveillance and observation of vehicle barriers and barrier systems adequate to detect indications of tampering and degradation or to otherwise ensure that each vehicle barrier and barrier system is able to satisfy the intended function. (D) Where a site has rail access to the protected area, install a train derailer, remove a section of track, or restrict access to railroad sidings and provide periodic surveillance of these measures. (ii) Waterborne vehicles. Licensees shall: (A) Identify areas from which a waterborne vehicle must be restricted, and where possible, in coordination with local, State, and Federal agencies having jurisdiction over waterway approaches, deploy buoys, markers, or other equipment. (B) In accordance with the site-specific analysis, provide periodic surveillance and observation of waterway approaches and adjacent areas.					
73.55(f)	(f) Target sets. (1) The licensee shall document and maintain the process used to develop and identify target sets, to include the site-specific analyses and					

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	methodologies used to determine and group the target set equipment or elements. (2) The licensee shall consider cyber attacks in the development and identification of target sets. (3) Target set equipment or elements that are not contained within a protected or vital area must be identified and documented consistent with the requirements in § 73.55(f)(1) and be accounted for in the licensee's protective trategy. (4) The licensee shall implement a process for the oversight of target set equipment and systems to ensure that changes to the configuration of the identified equipment and systems are considered in the licensee's protective strategy. Where appropriate, changes must be made to documented target sets.					
73.55(g)	(g) Access controls. (1) Consistent with the function of each barrier or barrier system, the licensee shall control personnel, vehicle, and material access, as applicable, at each access control point in accordance with the physical protection program design requirements of § 73.55(b). (i) To accomplish this, the licensee shall: (A) Locate access control portals outside of, or concurrent with, the physical barrier system through which it controls access. (B) Equip access control portals with locking devices, intrusion detection equipment, and					

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	surveillance equipment consistent with the intended function. (C) Provide supervision and control over the badging process to prevent unauthorized bypass of access control equipment located at or outside of the protected area. (D) Limit unescorted access to the protected area and vital areas, during non-emergency conditions, to only those individuals who require unescorted access to perform assigned duties and responsibilities. (E) Assign an individual the responsibility for the last access control function (controlling admission to the protected area) and isolate the individual within a bullet-resisting structure to assure the ability of the individual to respond or summon assistance. (ii) Where vehicle barriers are established, the licensee shall: (A) Physically control vehicle barrier portals to ensure only authorized vehicles are granted access through the barrier. (B) Search vehicles and materials for contraband or other items which could be used to commit radiological sabotage in accordance with paragraph (h) of this section. (C) Observe search functions to ensure a response					

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	can be initiated if needed. (2) Before granting access into the protected area, the licensee shall: (i) Confirm the identity of individuals. (ii) Verify the authorization for access of individuals, vehicles, and materials. (iii) Confirm, in accordance with industry shared lists and databases that individuals are not currently denied access to another licensed facility. (iv) Search individuals, vehicles, and materials in accordance with paragraph (h) of this section. (3) Vehicles in the protected area. (i) The licensee shall exercise control over all vehicles inside the protected area to ensure that they are used only by authorized persons and for authorized purposes. (ii) Vehicles inside the protected area must be operated by an individual authorized unescorted access to the area, or must be escorted by an individual as required by paragraph (g)(8) of this section. (iii) Vehicle use inside the protected area must be limited to plant functions or emergencies, and keys must be removed or the vehicle otherwise disabled when not in use.					

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	(iv) Vehicles transporting hazardous materials inside the protected area must be escorted by an armed member of the security organization. (4) Vital Areas. (i) Licensees shall control access into vital areas consistent with access authorization lists. (ii) In response to a site-specific credible threat or other credible information, implement a two-person (line-of-sight) rule for all personnel in vital areas so that no one individual is permitted access to a vital area. (5) Emergency conditions. (i) The licensee shall design the access control system to accommodate the potential need for rapid ingress or egress of authorized individuals during emergency conditions or situations that could lead to emergency conditions. (ii) To satisfy the design criteria of paragraph (g)(5)(i) of this section during emergency conditions, the licensee shall implement security procedures to ensure that authorized emergency personnel are provided prompt access to affected areas and equipment. (6) Access control devices. (i) The licensee shall control all keys, locks,					

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	combinations, passwords and related access control devices used to control access to protected areas, vital areas and security systems to reduce the probability of compromise. To accomplish this, the licensee shall: (A) Issue access control devices only to individuals who have unescorted access authorization and require access to perform official duties and responsibilities. (B) Maintain a record, to include name and affiliation, of all individuals to whom access control devices have been issued, and implement a process to account for access control devices at least annually. (C) Implement compensatory measures upon discovery or suspicion that any access control device may have been compromised. Compensatory measures must remain in effect until the compromise is corrected. (D) Retrieve, change, rotate, deactivate, or otherwise disable access control devices that have been or may have been compromised or when a person with access to control devices has been terminated under less than favorable conditions. (ii) The licensee shall implement a numbered photo identification badge system for all individuals authorized unescorted access to the protected area and vital areas.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	(A) Identification badges may be removed from the protected area only when measures are in place to confirm the true identity and authorization for unescorted access of the badge holder before allowing unescorted access to the protected area. (B) Except where operational safety concerns require otherwise, identification badges must be clearly displayed by all individuals while inside the protected area and vital areas. (C) The licensee shall maintain a record, to include the name and areas to which unescorted access is granted, of all individuals to whom photo identification badges have been issued. (iii) Access authorization program personnel shall be issued passwords and combinations to perform their assigned duties and may be excepted from the requirement of paragraph (g)(6)(i)(A) of this section provided they meet the background requirements of § 73.56. (7) Visitors. (i) The licensee may permit escorted access to protected and vital areas to individuals who have not been granted unescorted access in accordance with the requirements of § 73.56 and part 26 of this chapter. The licensee shall: (A) Implement procedures for processing, escorting, and controlling visitors.					

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	(B) Confirm the identity of each visitor through physical presentation of a recognized identification card issued by a local, State, or Federal government agency that includes a photo or contains physical characteristics of the individual requesting escorted access. (C) Maintain a visitor control register in which all visitors shall register their name, date, time, purpose of visit, employment affiliation, citizenship, and name of the individual to be visited before being escorted into any protected or vital area. (D) Issue a visitor badge to all visitors that clearly indicates an escort is required. (E) Escort all visitors, at all times, while inside the protected area and vital areas. (F) Deny escorted access to any individual who is currently denied access in industry shared data bases. (ii) Individuals not employed by the licensee but who require frequent or extended unescorted access to the protected area and/or vital areas to perform duties and responsibilities required by the licensee at irregular or intermittent intervals, shall satisfy the access authorization requirements of § 73.56 and part 26 of this chapter, and shall be issued a non-employee photo identification badge that is easily distinguished from other identification badges before being allowed unescorted access to the protected and vital areas. Non-employee photo					

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	identification badges must visually reflect that the individual is a non-employee and that no escort is required. (8) Escorts. The licensee shall ensure that all escorts are trained to perform escort duties in accordance with the requirements of this section and site training requirements. (i) Escorts shall be authorized unescorted access to all areas in which they will perform escort duties. (ii) Individuals assigned to visitor escort duties shall be provided a means of timely communication with security personnel to summon assistance when needed. (iii) Individuals assigned to vehicle escort duties shall be trained and qualified in accordance with appendix B of this part and provided a means of continuous communication with security personnel to ensure the ability to summon assistance when needed. (iv) When visitors are performing work, escorts shall be generally knowledgeable of the activities to be performed by the visitor and report behaviors or activities that may constitute an unreasonable risk to the health and safety of the public and common defense and security, including a potential threat to commit radiological sabotage, consistent with § 73.56(f)(1). (v) Each licensee shall describe visitor to escort					

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	ratios for the protected area and vital areas in physical security plans. Implementing procedures shall provide necessary observation and control requirements for all visitor activities.					
73.55(h)	(h) Search programs. (1) The objective of the search program is to detect, deter, and prevent the introduction of firearms, explosives, incendiary devices, or other items which could be used to commit radiological sabotage. To accomplish this the licensee shall search individuals, vehicles, and materials consistent with the physical protection program design requirements in paragraph (b) of this section, and the function to be performed at each access control point or portal before granting access. (2) Owner controlled area searches. (i) Where the licensee has established physical barriers in the owner controlled area, the licensee shall implement search procedures for access control points in the barrier. (ii) For each vehicle access control point, the licensee shall describe in implementing procedures areas of a vehicle to be searched, and the items for which the search is intended to detect and prevent access. Areas of the vehicle to be searched must include, but are not limited to, the cab, engine compartment, undercarriage, and cargo area. (iii) Vehicle searches must be performed by at least two (2) trained and equipped security personnel, one of which must be armed. The armed individual					

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	shall be positioned to observe the search process and provide immediate response. (iv) Vehicle searches must be accomplished through the use of equipment capable of detecting firearms, explosives, incendiary devices, or other items which could be used to commit radiological sabotage, or through visual and physical searches, or both, to ensure that all items are identified before granting access. (v) Vehicle access control points must be equipped with video surveillance equipment that is monitored by an individual capable of initiating a response. (3) Protected area searches. Licensees shall search all personnel, vehicles and materials requesting access to protected areas. (i) The search for firearms, explosives, incendiary devices, or other items which could be used to commit radiological sabotage shall be accomplished through the use of equipment capable of detecting these items, or through visual and physical searches, or both, to ensure that all items are clearly identified before granting access to protected areas. The licensee shall subject all persons except official Federal, state, and local law enforcement personnel on official duty to these searches upon entry to the protected area. Armed security officers who are on duty and have exited the protected area may re-enter the protected area without being searched for firearms.					

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	(ii) Whenever search equipment is out of service, is not operating satisfactorily, or cannot be used effectively to search individuals, vehicles, or materials, a visual and physical search shall be conducted. (iii) When an attempt to introduce firearms, explosives, incendiary devices, or other items which could be used to commit radiological sabotage has occurred or is suspected, the licensee shall implement actions to ensure that the suspect individuals, vehicles, and materials are denied access and shall perform a visual and physical search to determine the absence or existence of a threat. (iv) For each vehicle access portal, the licensee shall describe in implementing procedures areas of a vehicle to be searched before access is granted. Areas of the vehicle to be searched must include, but are not limited to, the cab, engine compartment, undercarriage, and cargo area. (v) Exceptions to the protected area search requirements for materials may be granted for safety or operational reasons provided the design criteria of § 73.55(b) are satisfied, the materials are clearly identified, the types of exceptions to be granted are described in the security plans, and the specific security measures to be implemented for excepted items are detailed in site procedures. (vi) To the extent practicable, excepted materials must be positively controlled, stored in a locked					

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	area, and opened at the final destination by an individual familiar with the items. (vii) Bulk material excepted from the protected area search requirements must be escorted by an armed member of the security organization to its final destination or to a receiving area where the excepted items are offloaded and verified. (viii) To the extent practicable, bulk materials excepted from search shall not be offloaded adjacent to a vital area.					
73.55(i)	(i) Detection and assessment systems. (1) The licensee shall establish and maintain intrusion detection and assessment systems that satisfy the design requirements of § 73.55(b) and provide, at all times, the capability to detect and assess unauthorized persons and facilitate the effective implementation of the licensee's protective strategy. (2) Intrusion detection equipment must annunciate and video assessment equipment shall display concurrently, in at least two continuously staffed onsite alarm stations, at least one of which must be protected in accordance with the requirements of the central alarm station within this section. (3) The licensee's intrusion detection and assessment systems must be designed to: (i) Provide visual and audible annunciation of the alarm. (ii) Provide a visual display from which assessment					

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	of the detected activity can be made. (iii) Ensure that annunciation of an alarm indicates the type and location of the alarm. (iv) Ensure that alarm devices to include transmission lines to annunciators are tamper indicating and self-checking. (v) Provide an automatic indication when the alarm system or a component of the alarm system fails, or when the system is operating on the backup power supply. (vi) Support the initiation of a timely response in accordance with the security plans, licensee protective strategy, and associated implementing procedures. (vii) Ensure intrusion detection and assessment equipment at the protected area perimeter remains operable from an uninterruptible power supply in the event of the loss of normal power. (4) Alarm stations. (i) Both alarm stations required by paragraph (i)(2) of this section must be designed and equipped to ensure that a single act, in accordance with the design basis threat of radiological sabotage defined in § 73.1(a)(1), cannot disable both alarm stations. The licensee shall ensure the survivability of at least one alarm station to maintain the ability to perform the following functions:					

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	(A) Detect and assess alarms; (B) Initiate and coordinate an adequate response to an alarm; (C) Summon offsite assistance; and (D) Provide command and control. (ii) Licensees shall: (A) Locate the central alarm station inside a protected area. The interior of the central alarm station must not be visible from the perimeter of the protected area. (B) Continuously staff each alarm station with at least one trained and qualified alarm station operator. The alarm station operator must not be assigned other duties or responsibilities which would interfere with the ability to execute the functions described in § 73.55(i)(4)(i) of this section. (C) Not permit any activities to be performed within either alarm station that would interfere with an alarm station operator's ability to execute assigned duties and responsibilities. (D) Assess and initiate response to all alarms in accordance with the security plans and implementing procedures. (E) Assess and initiate response to other events as					

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	appropriate. (F) Ensure that an alarm station operator cannot change the status of a detection point or deactivate a locking or access control device at a protected or vital area portal, without the knowledge and concurrence of the alarm station operator in the other alarm station. (G) Ensure that operators in both alarm stations are knowledgeable of final disposition of all alarms. (H) Maintain a record of all alarm annunciations, the cause of each alarm, and the disposition of each alarm. (iii) Applicants for an operating license under the provisions of part 50 of this chapter, or holders of a combined license under the provisions of part 52 of this chapter, shall construct, locate, protect, and equip both the central and secondary alarm stations to the standards for the central alarm station contained in this section. Both alarm stations shall be equal and redundant, such that all functions needed to satisfy the requirements of this section can be performed in both alarm stations. (5) Surveillance, observation, and monitoring. (i) The physical protection program must include surveillance, observation, and monitoring as needed to satisfy the design requirements of § 73.55(b), identify indications of tampering, or otherwise implement the site protective strategy.					

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	(ii) The licensee shall provide continuous surveillance, observation, and monitoring of the owner controlled area as described in the security plans to detect and deter intruders and ensure the integrity of physical barriers or other components and functions of the onsite physical protection program. Continuous surveillance, observation, and monitoring responsibilities may be performed by security personnel during continuous patrols, through use of video technology, or by a combination of both. (iii) Unattended openings that intersect a security boundary such as underground pathways must be protected by a physical barrier and monitored by intrusion detection equipment or observed by security personnel at a frequency sufficient to detect exploitation. (iv) Armed security patrols shall periodically check external areas of the protected area to include physical barriers and vital area portals. (v) Armed security patrols shall periodically inspect vital areas to include the physical barriers used at all vital area portals. (vi) The licensee shall provide random patrols of all accessible areas containing target set equipment. (vii) Security personnel shall be trained to recognize obvious indications of tampering consistent with their assigned duties and responsibilities.					

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	(viii) Upon detection of tampering, or other threats, the licensee shall initiate response in accordance with the security plans and implementing procedures. (6) Illumination. (i) The licensee shall ensure that all areas of the facility are provided with illumination necessary to satisfy the design requirements of § 73.55(b) and implement the protective strategy. (ii) The licensee shall provide a minimum illumination level of 0.2 foot-candles, measured horizontally at ground level, in the isolation zones and appropriate exterior areas within the protected area. Alternatively, the licensee may augment the facility illumination system by means of low-light technology to meet the requirements of this section or otherwise implement the protective strategy. (iii) The licensee shall describe in the security plans how the lighting requirements of this section are met and, if used, the type(s) and application of low-light technology.					
73.55(j)	(j) Communication requirements. (1) The licensee shall establish and maintain continuous communication capability with onsite and offsite resources to ensure effective command and control during both normal and emergency situations. (2) Individuals assigned to each alarm station shall be capable of calling for assistance in accordance					

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	with the security plans and the licensee's procedures. (3) All on-duty security force personnel shall be capable of maintaining continuous communication with an individual in each alarm station, and vehicle escorts shall maintain continuous communication with security personnel. All personnel escorts shall maintain timely communication with the security personnel. (4) The following continuous communication capabilities must terminate in both alarm stations required by this section: (i) Radio or microwave transmitted two-way voice communication, either directly or through an intermediary, in addition to conventional telephone service between local law enforcement authorities and the site. (ii) A system for communication with the control room. (5) Non-portable communications equipment must remain operable from independent power sources in the event of the loss of normal power. (6) The licensee shall identify site areas where communication could be interrupted or cannot be maintained, and shall establish alternative communication measures or otherwise account for these areas in implementing procedures.					
73.55(k)	(k) Response requirements. (1) The licensee shall					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	establish and maintain, at all times, properly trained, qualified and equipped personnel required to interdict and neutralize threats up to and including the design basis threat of radiological sabotage as defined in § 73.1, to prevent significant core damage and spent fuel sabotage. (2) The licensee shall ensure that all firearms, ammunition, and equipment necessary to implement the site security plans and protective strategy are in sufficient supply, are in working condition, and are readily available for use. (3) The licensee shall train each armed member of the security organization to prevent or impede attempted acts of radiological sabotage by using force sufficient to counter the force directed at that person, including the use of deadly force when the armed member of the security organization has a reasonable belief that the use of deadly force is necessary in self-defense or in the defense of others, or any other circumstances as authorized by applicable State or Federal law. (4) The licensee shall provide armed response personnel consisting of armed responders which may be augmented with armed security officers to carry out armed response duties within predetermined time lines specified by the site protective strategy. (5) Armed responders. (i) The licensee shall determine the minimum					

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	number of armed responders necessary to satisfy the design requirements of § 73.55(b) and implement the protective strategy. The licensee shall document this number in the security plans. (ii) The number of armed responders shall not be less than ten (10). (iii) Armed responders shall be available at all times inside the protected area and may not be assigned other duties or responsibilities that could interfere with their assigned response duties. (6) Armed security officers. (i) Armed security officers, designated to strengthen onsite response capabilities, shall be onsite and available at all times to carry out their assigned response duties. (ii) The minimum number of armed security officers designated to strengthen onsite response capabilities must be documented in the security plans. (7) The licensee shall have procedures to reconstitute the documented number of available armed response personnel required to implement the protective strategy. (8) Protective strategy. The licensee shall establish, maintain, and implement a written protective strategy in accordance with the requirements of this section and part 73, appendix C, Section II. Upon					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	receipt of an alarm or other indication of a threat, the licensee shall: (i) Determine the existence and level of a threat in accordance with pre-established assessment methodologies and procedures. (ii) Initiate response actions to interdict and neutralize the threat in accordance with the requirements of part 73, appendix C, section II, the safeguards contingency plan, and the licensee's response strategy. (iii) Notify law enforcement agencies (local, State, and Federal law enforcement agencies (LLEA)), in accordance with site procedures. (9) Law enforcement liaison. To the extent practicable, licensees shall document and maintain current agreements with applicable law enforcement agencies to include estimated response times and capabilities. (10) Heightened security. Licensees shall establish, maintain, and implement a threat warning system which identifies specific graduated protective measures and actions to be taken to increase licensee preparedness against a heightened security threat. (i) Licensees shall ensure that the specific protective measures and actions identified for each threat level are consistent with the security plans and other emergency plans and procedures.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(ii) Upon notification by an authorized representative of the Commission, licensees shall implement the specific threat level indicated by the Commission representative.					
73.55(l)	(l) Facilities using mixed-oxide (MOX) fuel assemblies containing up to 20 weight percent plutonium dioxide (PuO_2). (1) Commercial nuclear power reactors licensed under 10 CFR parts 50 or 52 and authorized to use special nuclear material in the form of MOX fuel assemblies containing up to 20 weight percent PuO_2 shall, in addition to meeting the requirements of this section, protect un-irradiated MOX fuel assemblies against theft or diversion as described in this paragraph. (2) Commercial nuclear power reactors authorized to use MOX fuel assemblies containing up to 20 weight percent PuO_2 are exempt from the requirements of §§ 73.20, 73.45, and 73.46 for the onsite physical protection of un-irradiated MOX fuel assemblies. (3) Administrative controls. (i) The licensee shall describe in the security plans the operational and administrative controls to be implemented for the receipt, inspection, movement, storage, and protection of un-irradiated MOX fuel assemblies. (ii) The licensee shall implement the use of tamper-indicating devices for un-irradiated MOX fuel assembly transport and shall verify their use and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	integrity before receipt. (iii) Upon receipt of un-irradiated MOX fuel assemblies, the licensee shall: (A) Inspect un-irradiated MOX fuel assemblies for damage. (B) Search un-irradiated MOX fuel assemblies for unauthorized materials. (iv) The licensee may conduct the required inspection and search functions simultaneously. (v) The licensee shall ensure the proper placement and control of un-irradiated MOX fuel assemblies as follows: (A) At least one armed security officer shall be present during the receipt and inspection of un-irradiated MOX fuel assemblies. This armed security officer shall not be an armed responder as required by paragraph (k) of this section. (B) The licensee shall store un-irradiated MOX fuel assemblies only within a spent fuel pool, located within a vital area, so that access to the un-irradiated MOX fuel assemblies requires passage through at least two physical barriers and the water barrier combined with the additional measures detailed in this section. (vi) The licensee shall implement a material control and accountability program that includes a					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	predetermined and documented storage location for each un-irradiated MOX fuel assembly. (4) Physical controls. (i) The licensee shall lock, lockout, or disable all equipment and power supplies to equipment required for the movement and handling of un-irradiated MOX fuel assemblies when movement activities are not authorized. (ii) The licensee shall implement a two-person, line-of-sight rule within the spent fuel pool area whenever control systems or equipment required for the movement or handling of un-irradiated MOX fuel assemblies must be accessed. (iii) The licensee shall conduct random patrols of areas containing un-irradiated MOX fuel assemblies to identify indications of tampering and ensure the integrity of barriers and locks. (iv) Locks, keys, and any other access control device used to secure equipment and power sources required for the movement of un-irradiated MOX fuel assemblies, or openings to areas containing un-irradiated MOX fuel assemblies, must be controlled by the security organization. (v) Removal of locks used to secure equipment and power sources required for the movement of un-irradiated MOX fuel assemblies or openings to areas containing un-irradiated MOX fuel assemblies must require approval by both the on-duty security					

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	shift supervisor and the operations shift manager. (A) At least one armed security officer shall be present to observe activities involving the movement of un-irradiated MOX fuel assemblies before the removal of the locks and providing power to equipment required for the movement or handling of un-irradiated MOX fuel assemblies. (B) At least one armed security officer shall be present at all times until power is removed from equipment and locks are secured. (C) Security officers shall be knowledgeable of authorized and unauthorized activities involving un-irradiated MOX fuel assemblies. (5) At least one armed security officer shall be present and shall maintain constant surveillance of un-irradiated MOX fuel assemblies when the assemblies are not located in the spent fuel pool or reactor. (6) The licensee shall maintain at all times the capability to detect, assess, interdict and neutralize threats to un-irradiated MOX fuel assemblies in accordance with the requirements of this section. (7) MOX fuel assemblies containing greater than 20 weight percent PuO_2. (i) Requests for the use of MOX fuel assemblies containing greater than 20 weight percent PuO_2 shall be reviewed and approved by the Commission					

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	before receipt of MOX fuel assemblies. (ii) Additional measures for the physical protection of un-irradiated MOX fuel assemblies containing greater than 20 weight percent PuO2 shall be determined by the Commission on a case-by-case basis and documented through license amendment in accordance with 10 CFR 50.90.					
73.55(m)	(m) Security program reviews. (1) As a minimum the licensee shall review each element of the physical protection program at least every 24 months. Reviews shall be conducted: (i) Within 12 months following initial implementation of the physical protection program or a change to personnel, procedures, equipment, or facilities that potentially could adversely affect security. (ii) As necessary based upon site-specific analyses, assessments, or other performance indicators. (iii) By individuals independent of those personnel responsible for program management and any individual who has direct responsibility for implementing the onsite physical protection program. (2) Reviews of the security program must include, but not be limited to, an audit of the effectiveness of the physical security program, security plans, implementing procedures, cyber security programs, safety/security interface activities, the testing, maintenance, and calibration program, and response commitments by local, State, and Federal					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	law enforcement authorities. (3) The results and recommendations of the onsite physical protection program reviews, management's findings regarding program effectiveness, and any actions taken as a result of recommendations from prior program reviews, must be documented in a report to the licensee's plant manager and to corporate management at least one level higher than that having responsibility for day-to-day plant operation. These reports must be maintained in an auditable form, available for inspection. (4) Findings from onsite physical protection program reviews must be entered into the site corrective action program.					
73.55(n)	(n) Maintenance, testing, and calibration. (1) The licensee shall: (i) Establish, maintain, and implement a maintenance, testing and calibration program to ensure that security systems and equipment, including secondary and uninterruptible power supplies, are tested for operability and performance at predetermined intervals, maintained in operable condition, and are capable of performing their intended functions. (ii) Describe the maintenance, testing and calibration program in the physical security plan. Implementing procedures must specify operational and technical details required to perform maintenance, testing, and calibration activities to					

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	include, but not limited to, purpose of activity, actions to be taken, acceptance criteria, and the intervals or frequency at which the activity will be performed. (iii) Identify in procedures the criteria for determining when problems, failures, deficiencies, and other findings are documented in the site corrective action program for resolution. (iv) Ensure that information documented in the site corrective action program is written in a manner that does not constitute safeguards information as defined in 10 CFR 73.21. (v) Implement compensatory measures that ensure the effectiveness of the onsite physical protection program when there is a failure or degraded operation of security-related component or equipment. (2) The licensee shall test each intrusion alarm for operability at the beginning and end of any period that it is used for security, or if the period of continuous use exceeds seven (7) days. The intrusion alarm must be tested at least once every seven (7) days. (3) Intrusion detection and access control equipment must be performance tested in accordance with the security plans and implementing procedures. (4) Equipment required for communications onsite					

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	must be tested for operability not less frequently than once at the beginning of each security personnel work shift. (5) Communication systems between the alarm stations and each control room, and between the alarm stations and local law enforcement agencies, to include backup communication equipment, must be tested for operability at least once each day. (6) Search equipment must be tested for operability at least once each day and tested for performance at least once during each seven (7) day period. (7) A program for testing or verifying the operability of devices or equipment located in hazardous areas must be specified in the implementing procedures and must define alternate measures to be taken to ensure the timely completion of testing or maintenance when the hazardous condition or other restrictions are no longer applicable. (8) Security equipment or systems shall be tested in accordance with the site maintenance, testing and calibration procedures before being placed back in service after each repair or inoperable state.					
73.55(o)	(o) Compensatory measures. (1) The licensee shall identify criteria and measures to compensate for degraded or inoperable equipment, systems, and components to meet the requirements of this section. (2) Compensatory measures must provide a level of protection that is equivalent to the protection that					

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	was provided by the degraded or inoperable, equipment, system, or components. (3) Compensatory measures must be implemented within specific time frames necessary to meet the requirements stated in paragraph (b) of this section and described in the security plans.					
73.55(p)	(p) Suspension of security measures. (1) The licensee may suspend implementation of affected requirements of this section under the following conditions: (i) In accordance with §§ 50.54(x) and 50.54(y) of this chapter, the licensee may suspend any security measures under this section in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent. This suspension of security measures must be approved as a minimum by a licensed senior operator before taking this action. (ii) During severe weather when the suspension of affected security measures is immediately needed to protect the personal health and safety of security force personnel and no other immediately apparent action consistent with the license conditions and technical specifications can provide adequate or equivalent protection. This suspension of security measures must be approved, as a minimum, by a licensed senior operator, with input from the security supervisor or manager, before taking this					

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	action. (2) Suspended security measures must be reinstated as soon as conditions permit. (3) The suspension of security measures must be reported and documented in accordance with the provisions of § 73.71.					
73.55(q)	(q) Records. (1) The Commission may inspect, copy, retain, and remove all reports, records, and documents required to be kept by Commission regulations, orders, or license conditions, whether the reports, records, and documents are kept by the licensee or a contractor. (2) The licensee shall maintain all records required to be kept by Commission regulations, orders, or license conditions, until the Commission terminates the license for which the records were developed, and shall maintain superseded portions of these records for at least three (3) years after the record is superseded, unless otherwise specified by the Commission. (3) If a contracted security force is used to implement the onsite physical protection program, the licensee's written agreement with the contractor must be retained by the licensee as a record for the duration of the contract. (4) Review and audit reports must be maintained and available for inspection, for a period of three (3) years.					
73.55(r)	(r) Alternative measures. (1) The Commission may					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	authorize an applicant or licensee to provide a measure for protection against radiological sabotage other than one required by this section if the applicant or licensee demonstrates that: (i) The measure meets the same performance objectives and requirements specified in paragraph (b) of this section; and (ii) The proposed alternative measure provides protection against radiological sabotage or theft of un-irradiated MOX fuel assemblies, equivalent to that which would be provided by the specific requirement for which it would substitute. (2) The licensee shall submit proposed alternative measure(s) to the Commission for review and approval in accordance with §§ 50.4 and 50.90 of this chapter before implementation. (3) In addition to fully describing the desired changes, the licensee shall submit a technical basis for each proposed alternative measure. The basis must include an analysis or assessment that demonstrates how the proposed alternative measure provides a level of protection that is at least equal to that which would otherwise be provided by the specific requirement of this section. (4) Alternative vehicle barrier systems. In the case of vehicle barrier systems required by § 73.55(e)(10), the licensee shall demonstrate that: (i) The alternative measure provides protection					

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	against the use of a vehicle as a means of transportation to gain proximity to vital areas; (ii) The alternative measure provides protection against the use of a vehicle as a vehicle bomb; and (iii) Based on comparison of the costs of the alternative measures to the costs of meeting the Commission's requirements using the essential elements of 10 CFR 50.109, the costs of fully meeting the Commission's requirements are not justified by the protection that would be provided.					
73.56	Personnel access authorization requirements for nuclear power plants.					Exclude; Admin
73.57	Requirements for criminal history records checks of individuals granted unescorted access to a nuclear power facility or access to Safeguards Information.					Exclude; Admin. Portions of this regulation related purely to safeguards information are outside the scopeof this procedure.
73.58	Safety/security interface requirements for nuclear power reactors.					Exclude; Admin
73.59	Relief from fingerprinting, identification and criminal history records checks and other elements of background checks for designated categories of individuals.					Exclude; Admin
73.60	Additional requirements for physical protection at nonpower reactors.	NA				Exclude
73.61	Relief from fingerprinting and criminal history records check for designated categories of individuals permitted unescorted access to certain radioactive materials or other property.					Exclude; Admin
	Physical Protection of Special Nuclear Material of Moderate and Low Strategic Significance					Heading
73.67	Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance.	NA				Exclude

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Records and Reports	NA				Heading
73.70	Records.	NA				Exclude; exempt in accordance with 10 CFR 73.6.
73.71	Reporting of safeguards events.					Exclude; safeguards information not being reviewed.
73.72	Requirement for advance notice of shipment of formula quantities of strategic special nuclear material, special nuclear material of moderate strategic significance, or irradiated reactor fuel.					Exclude; Admin
73.73	Requirement for advance notice and protection of export shipments of special nuclear material of low strategic significance.	NA				Exclude
73.74	Requirement for advance notice and protection of import shipments of nuclear material from countries that are not party to the Convention on the Physical Protection of Nuclear Material.	NA				Exclude
73.75	Posting.					Exclude; Admin
	Enforcement					Heading
73.80	Violations.					Exclude; Admin
73.81	Criminal penalties.					Exclude; Admin
73 App. A	U.S. Nuclear Regulatory Commission Offices and Classified Mailing Addresses					Exclude; Admin
73 App. B	General Criteria for Security Personnel					Exclude; Admin
73 App. C	Nuclear Power Plant Safeguards Contingency Plans					Exclude; safeguards information not being reviewed.
73 App. D	Physical Protection of Irradiated Reactor Fuel in Transit, Training Program Subject Schedule					Exclude; Admin
73 App. E	Levels of Physical Protection To Be Applied in International Transport of Nuclear Material	NA				Exclude
73 App. F	Nations That Are Parties to the Convention on the Physical Protection of Nuclear Material	NA				Exclude
73 App. G	Reportable Safeguards Events					Exclude; safeguards information not being reviewed.

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
73 App. H	Weapons Qualification Criteria					Exclude; Admin
73 App. I	Category 1 and 2 Radioactive Materials					Exclude; Admin

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
100	Reactor Site Criteria					
100.1	Purpose					Exclude; Administrative
100.2	Scope					Exclude; Administrative
100.3	Definitions					Exclude; Administrative
100.4	Communications					Exclude; Administrative
100.10	Factors to be considered when evaluating sites. Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:					
100.10(a)	(a) Characteristics of reactor design and proposed operation including: 1) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials; 2) The extent to which generally accepted engineering standards are applied to the design					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of the reactor; 3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials; 4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur.					
100.10(b)	(b) Population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.					
100.10(c)	(c) Physical characteristics of the site, including seismology, meteorology, geology, and hydrology. Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," describes the nature of investigations required to obtain the geologic and seismic data necessary to determine site suitability and to provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public. It describes procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describes information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting. Meteorological conditions at the site and in the					

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	surrounding area should be considered. Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow into nearby streams or rivers or might find ready access to underground water tables.					
100.10(d)	(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.					
100.11	Determination of exclusion area, low population zone, and population center distance.					
100.11(a)	(a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release1 from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following: 1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	receive a total radiation dose to the whole body in excess of 25 rem[2] or a total radiation dose in excess of 300 rem[2] to the thyroid from iodine exposure. 2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure. 3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.					
100.11(b)	(b) For sites for multiple reactor facilities consideration should be given to the following: 1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion					

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	area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries. 2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the Commission the basis for such a reduction in the source term. 3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations. Note: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains					

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	a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation. [1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products. [2] The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.					
	Subpart B--Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997					
100.20	Factors to be Considered when Evaluating Sites. The Commission will take the following factors into consideration in determining the acceptability of a site for a stationary power reactor:					
100.20(a)	(a) Population density and use characteristics of the site environs, including the exclusion area, the population distribution, and site-related characteristics must be evaluated to determine whether individual as well as societal risk of potential plant accidents is low, and that physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans are identified.					
100.20(b)	(b) The nature and proximity of man-related hazards (e.g., airports, dams, transportation routes, military and chemical facilities) must be evaluated to establish site parameters for use in determining whether a plant design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low.					
100.20(c)	(c) Physical characteristics of the site, including seismology, meteorology, geology, and hydrology.					

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	Section 100.23, "Geologic and seismic siting factors," describes the criteria and nature of investigations required to obtain the geologic and seismic data necessary to determine the suitability of the proposed site and the plant design bases. Meteorological characteristics of the site that are necessary for safety analysis or that may have an impact upon plant design (such as maximum probable wind speed and precipitation) must be identified and characterized. Factors important to hydrological radionuclide transport (such as soil, sediment, and rock characteristics, adsorption and retention coefficients, ground water velocity, and distances to the nearest surface body of water) must be obtained from on-site measurements. The maximum probable flood along with the potential for seismically induced floods discussed in § 100.23 (d)(3) must be estimated using historical data.					
100.21	Non-Seismic Siting Criteria Applications for site approval for commercial power reactors shall demonstrate that the proposed site meets the following criteria:					
100.21(a)	(a) Every site must have an exclusion area and a low population zone, as defined in § 100.3;					
100.21(b)	(b) The population center distance, as defined in § 100.3, must be at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon					

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	consideration of population distribution. Political boundaries are not controlling in the application of this guide;					
100.21(c)	(c) Site atmospheric dispersion characteristics must be evaluated and dispersion parameters established such that: 1) Radiological effluent release limits associated with normal operation from the type of facility proposed to be located at the site can be met for any individual located offsite; and 2) Radiological dose consequences of postulated accidents shall meet the criteria set forth in § 50.34(a)(1) of this chapter for the type of facility proposed to be located at the site;					
100.21(d)	(d) The physical characteristics of the site, including meteorology, geology, seismology, and hydrology must be evaluated and site parameters established such that potential threats from such physical characteristics will pose no undue risk to the type of facility proposed to be located at the site;					
100.21(e)	(e) Potential hazards associated with nearby transportation routes, industrial and military facilities must be evaluated and site parameters established such that potential hazards from such routes and facilities will pose no undue risk to the type of facility proposed to be located at the site;					
100.21(f)	(f) Site characteristics must be such that adequate security plans and measures can be developed;					
100.21(g)	(g) Physical characteristics unique to the proposed site					

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	that could pose a significant impediment to the development of emergency plans must be identified;					
100.21(h)	(h) Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable[3]. [3] Examples of these factors include, but are not limited to, such factors as the higher population density site having superior seismic characteristics, better access to skilled labor for construction, better rail and highway access, shorter transmission line requirements, or less environmental impact on undeveloped areas, wetlands or endangered species, etc. Some of these factors are included in, or impact, the other criteria included in this section.					
100.23	Geologic and Seismic Siting Criteria This section sets forth the principal geologic and seismic considerations that guide the Commission in its evaluation of the suitability of a proposed site and adequacy of the design bases established in consideration of the geologic and seismic characteristics of the proposed site, such that, there is a reasonable assurance that a nuclear power plant can be constructed and operated at the proposed site without undue risk to					

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	the health and safety of the public. Applications to engineering design are contained in appendix S to part 50 of this chapter.					
100.23(a)	(a) *Applicability*. The requirements in paragraphs (c) and (d) of this section apply to applicants for an early site permit or combined license pursuant to Part 52 of this chapter, or a construction permit or operating license for a nuclear power plant pursuant to Part 50 of this chapter on or after January 10, 1997. However, for either an operating license applicant or holder whose construction permit was issued prior to January 10, 1997, the seismic and geologic siting criteria in Appendix A to Part 100 of this chapter continues to apply.					
100.23(b)	(b) *Commencement of construction*. The investigations required in paragraph (c) of this section are not considered "construction" as defined in 10 CFR 50.10(a).					
100.23(c)	(c) *Geological, seismological, and engineering characteristics*. The geological, seismological, and engineering characteristics of a site and its environs must be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site, to provide sufficient information to support evaluations performed to arrive at estimates of the Safe Shutdown Earthquake Ground Motion, and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. The size of the region to be investigated and the type of data pertinent to the investigations must be determined based on the nature of the region surrounding the proposed site. Data on the vibratory ground motion, tectonic surface					

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	deformation, non-tectonic deformation, earthquake recurrence rates, fault geometry and slip rates, site foundation material, and seismically induced floods and water waves must be obtained by reviewing pertinent literature and carrying out field investigations. However, each applicant shall investigate all geologic and seismic factors (for example, volcanic activity) that may affect the design and operation of the proposed nuclear power plant irrespective of whether such factors are explicitly included in this section.					
100.23(d)	(d) *Geologic and seismic siting factors*. The geologic and seismic siting factors considered for design must include a determination of the Safe Shutdown Earthquake Ground Motion for the site, the potential for surface tectonic and nontectonic deformations, the design bases for seismically induced floods and water waves, and other design conditions as stated in paragraph (d)(4) of this section. 1) Determination of the Safe Shutdown Earthquake Ground Motion. The Safe Shutdown Earthquake Ground Motion for the site is characterized by both horizontal and vertical free-field ground motion response spectra at the free ground surface. The Safe Shutdown Earthquake Ground Motion for the site is determined considering the results of the investigations required by paragraph (c) of this section. Uncertainties are inherent in such estimates. These uncertainties must be					

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	addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis or suitable sensitivity analyses. Paragraph IV(a)(1) of appendix S to part 50 of this chapter defines the minimum Safe Shutdown Earthquake Ground Motion for design. 2) Determination of the potential for surface tectonic and non-tectonic deformations. Sufficient geological, seismological, and geophysical data must be provided to clearly establish whether there is a potential for surface deformation. 3) Determination of design bases for seismically induced floods and water waves. The size of seismically induced floods and water waves that could affect a site from either locally or distantly generated seismic activity must be determined. 4) Determination of siting factors for other design conditions. Siting factors for other design conditions that must be evaluated include soil and rock stability, liquefaction potential, natural and artificial slope stability, cooling water supply, and remote safety-related structure siting. Each applicant shall evaluate all siting factors and potential causes of failure, such as, the physical properties of the materials underlying the site, ground disruption, and the effects of vibratory ground motion that may affect the design and operation of the proposed nuclear power plant.					

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	Appendix A to Part 100--SEISMIC AND GEOLOGIC SITING CRITERIA FOR NUCLEAR POWER PLANTS					
100.I.	Purpose General Design Criterion 2 of Appendix A to part 50 of this chapter requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. It is the purpose of these criteria to set forth the principal seismic and geologic considerations which guide the Commission in its evaluation of the suitability of proposed sites for nuclear power plants and the suitability of the plant design bases established in consideration of the seismic and geologic characteristics of the proposed sites. These criteria are based on the limited geophysical and geological information available to date concerning faults and earthquake occurrence and effect. They will be revised as necessary when more complete information becomes available.					Exclude; Administrative
100.II.	Scope These criteria, which apply to nuclear power plants, describe the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and provide reasonable assurance that a nuclear power plant can be constructed					Exclude; Administrative

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	and operated at a proposed site without undue risk to the health and safety of the public. They describe procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describe information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting. Other geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants are identified. The investigations described in this appendix are within the scope of investigations permitted by § 50.10(c)(1) of this chapter. Each applicant for a construction permit shall investigate all seismic and geologic factors that may affect the design and operation of the proposed nuclear power plant irrespective of whether such factors are explicitly included in these criteria. Additional investigations and/or more conservative determinations than those included in these criteria may be required for sites located in areas having complex geology or in areas of high seismicity. If an applicant believes that the particular seismology and geology of a site indicate that some of these criteria, or portions thereof, need not be satisfied, the specific sections of these criteria should be identified in the license application, and supporting data to justify clearly such departures should be presented. These criteria do not address investigations of volcanic					

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	phenomena required for sites located in areas of volcanic activity. Investigations of the volcanic aspects of such sites will be determined on a case-by-case basis					
100.III.	Definitions As used in these criteria: (a) The *magnitude* of an earthquake is a measure of the size of an earthquake and is related to the energy released in the form of seismic waves. *Magnitude* means the numerical value on a Richter scale. (b) The *intensity* of an earthquake is a measure of its effects on man, on man-built structures, and on the earth's surface at a particular location. Intensity means the numerical value on the Modified Mercalli scale. (c) The *Safe Shutdown Earthquake*[1] is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to assure: (1) The integrity of the reactor coolant pressure boundary,					

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	(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part. (d) The *Operating Basis Earthquake* is that earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant; it is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional. (e) A *fault* is a tectonic structure along which differential slippage of the adjacent earth materials has occurred parallel to the fracture plane. It is distinct from other types of ground disruptions such as landslides, fissures, and craters. A fault may have gouge or breccia between its two walls and includes any associated monoclinal flexure or other similar geologic structural feature. (f) *Surface faulting* is differential ground displacement at or near the surface caused directly by fault movement and is distinct from nontectonic types of ground					

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	disruptions, such as landslides, fissures, and craters. (g) A *capable fault* is a fault which has exhibited one or more of the following characteristics: (1) Movement at or near the ground surface at least once within the past 35,000 years or movement of a recurring nature within the past 500,000 years. (2) Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault. (3) A structural relationship to a capable fault according to characteristics (1) or (2) of this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other. In some cases, the geologic evidence of past activity at or near the ground surface along a particular fault may be obscured at a particular site. This might occur, for example, at a site having a deep overburden. For these cases, evidence may exist elsewhere along the fault from which an evaluation of its characteristics in the vicinity of the site can be reasonably based. Such evidence shall be used in determining whether the fault is a capable fault within this definition. Notwithstanding the foregoing paragraphs III(g) (1), (2) and (3), structural association of a fault with geologic structural features which are geologically old (at least					

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	pre-Quaternary) such as many of those found in the Eastern region of the United States shall, in the absence of conflicting evidence, demonstrate that the fault is not a capable fault within this definition. (h) A *tectonic province* is a region of the North American continent characterized by a relative consistency of the geologic structural features contained therein. (i) A *tectonic structure* is a large scale dislocation or distortion within the earth's crust. Its extent is measured in miles. (j) A *zone requiring detailed faulting investigation* is a zone within which a nuclear power reactor may not be located unless a detailed investigation of the regional and local geologic and seismic characteristics of the site demonstrates that the need to design for surface faulting has been properly determined. (k) The *control width* of a fault is the maximum width of the zone containing mapped fault traces, including all faults which can be reasonably inferred to have experienced differential movement during Quaternary times and which join or can reasonably be inferred to join the main fault trace, measured within 10 miles along the fault's trend in both directions from the point of nearest approach to the site. (See Figure 1 of this appendix.) (l) A *response spectrum* is a plot of the maximum responses (acceleration, velocity or displacement) of a					

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	family of idealized single-degree-of-freedom damped oscillators against natural frequencies (or periods) of the oscillators to a specified vibratory motion input at their supports.					
100.IV.	Required Investigations The geologic, seismic and engineering characteristics of a site and its environs shall be investigated in sufficient scope and detail to provide reasonable assurance that they are sufficiently well understood to permit an adequate evaluation of the proposed site, and to provide sufficient information to support the determinations required by these criteria and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. The size of the region to be investigated and the type of data pertinent to the investigations shall be determined by the nature of the region surrounding the proposed site. The investigations shall be carried out by a review of the pertinent literature and field investigations and shall include the steps outlined in paragraphs (a) through (c) of this section.					
100.IV(a)	(a) *Required Investigation for Vibratory Ground Motion*. The purpose of the investigations required by this paragraph is to obtain information needed to describe the vibratory ground motion produced by the Safe Shutdown Earthquake. All of the steps in paragraphs (a)(5) through (a)(8) of this section need not be carried out if the Safe Shutdown Earthquake					

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	can be clearly established by investigations and determinations of a lesser scope. The investigations required by this paragraph provide an adequate basis for selection of an Operating Basis Earthquake. The investigations shall include the following: 1) Determination of the lithologic, stratigraphic, hydrologic, and structural geologic conditions of the site and the region surrounding the site, including its geologic history; 2) Identification and evaluation of tectonic structures underlying the site and the region surrounding the site, whether buried or expressed at the surface. The evaluation should consider the possible effects caused by man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams or reservoirs; 3) Evaluation of physical evidence concerning the behavior during prior earthquakes of the surficial geologic materials and the substrata underlying the site from the lithologic, stratigraphic, and structural geologic studies; 4) Determination of the static and dynamic engineering properties of the materials underlying the site. Included should be properties needed to determine the behavior of the underlying material during earthquakes and the characteristics of the underlying material in					

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	transmitting earthquake-induced motions to the foundations of the plant, such as seismic wave velocities, density, water content, porosity, and strength; 5) Listing of all historically reported earthquakes which have affected or which could reasonably be expected to have affected the site, including the date of occurrence and the following measured or estimated data: magnitude or highest intensity, and a plot of the epicenter or location of highest intensity. Where historically reported earthquakes could have caused a maximum ground acceleration of at least one-tenth the acceleration of gravity (0.1g) at the foundations of the proposed nuclear power plant structures, the acceleration or intensity and duration of ground shaking at these foundations shall also be estimated. Since earthquakes have been reported in terms of various parameters such as magnitude, intensity at a given location, and effect on ground, structures, and people at a specific location, some of these data may have to be estimated by use of appropriate empirical relationships. The comparative characteristics of the material underlying the epicentral location or region of highest intensity and of the material underlying the site in transmitting earthquake vibratory motion shall be considered; 6) Correlation of epicenters or locations of highest intensity of historically reported earthquakes,					

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	where possible, with tectonic structures any part of which is located within 200 miles of the site. Epicenters or locations of highest intensity which cannot be reasonably correlated with tectonic structures shall be identified with tectonic provinces any part of which is located within 200 miles of the site; 7) For faults, any part of which is within 200 miles[2] of the site and which may be of significance in establishing the Safe Shutdown Earthquake, determination of whether these faults are to be considered as capable faults.[3],[4] This determination is required in order to permit appropriate consideration of the geologic history of such faults in establishing the Safe Shutdown Earthquake. For guidance in determining which faults may be of significance in determining the Safe Shutdown Earthquake, table 1 of this appendix presents the minimum length of fault to be considered versus distance from site. Capable faults of lesser length than those indicated in table 1 (Note: Refer to 10 CFR 100 for the table) and faults which are not capable faults need not be considered in determining the Safe Shutdown Earthquake, except where unusual circumstances indicate such consideration is appropriate; 8) For capable faults, any part of which is within 200 miles2 of the site and which may be of significance in establishing the Safe Shutdown Earthquake, determination of:					

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	(i) The length of the fault; (ii) The relationship of the fault to regional tectonic structures; and (iii) The nature, amount, and geologic history of displacements along the fault, including particularly the estimated amount of the maximum Quaternary displacement related to any one earthquake along the fault.					
100.IV(b)	(b) *Required Investigation for Surface Faulting*. The purpose of the investigations required by this paragraph is to obtain information to determine whether and to what extent the nuclear power plant need be designed for surface faulting. If the design basis for surface faulting can be clearly established by investigations of a lesser scope, not all of the steps in paragraphs (b)(4) through (b)(7) of this section need be carried out. The investigations shall include the following: Determination of the lithologic, stratigraphic, hydrologic, and structural geologic conditions of the site and the area surrounding the site, including its geologic history; Evaluation of tectonic structures underlying the site, whether buried or expressed at the surface, with regard to their potential for causing surface displacement at or near the site. The evaluation shall consider the possible effects caused by man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction					

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	of minerals, or the loading effects of dams or reservoirs; Determination of geologic evidence of fault offset at or near the ground surface at or near the site; For faults greater than 1000 feet long, any part of which is within 5 miles[5] of the site, determination of whether these faults are to be considered as capable faults;[6],[7] Listing of all historically reported earthquakes which can reasonably be associated with capable faults greater than 1000 feet long, any part of which is within 5 miles5 of the site, including the date of occurrence and the following measured or estimated data: magnitude or highest intensity, and a plot of the epicenter or region of highest intensity; Correlation of epicenters or locations of highest intensity of historically reported earthquakes with capable faults greater than 1000 feet long, any part of which is located within 5 miles5 of the site; For capable faults greater than 1000 feet long, any part of which is within 5 miles5 of the site, determination of: (i) The length of the fault; (ii) The relationship of the fault to regional tectonic structures; (iii) The nature, amount, and geologic history of displacements along the fault, including particularly the estimated amount of the maximum Quaternary displacement related to any one earthquake along the fault; and (iv) The outer limits of the fault established by mapping Quaternary fault traces for 10 miles along its trend in both directions from the point of its nearest approach to the site.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
100.IV(c)	(c) *Required Investigation for Seismically Induced Floods and Water Waves.* (1) For coastal sites, the investigations shall include the determination of: (i) Information regarding distantly and locally generated waves or tsunami which have affected or could have affected the site. Available evidence regarding the runup and drawdown associated with historic tsunami in the same coastal region as the site shall also be included; (ii) Local features of coastal topography which might tend to modify tsunami runup or drawdown. Appropriate available evidence regarding historic local modifications in tsunami runup or drawndown at coastal locations having topography similar to that of the site shall also be obtained; and (iii) Appropriate geologic and seismic evidence to provide information for establishing the design basis for seismically induced floods or water waves from a local offshore earthquake, from local offshore effects of an onshore earthquake, or from coastal subsidence. This evidence shall be determined, to the extent practical, by a procedure similar to that required in paragraphs (a) and (b) of this section. The probable slip characteristics of offshore faults shall also be considered as well as the potential for offshore slides in submarine material.					

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	(2) For sites located near lakes and rivers, investigations similar to those required in paragraph (c)(1) of this section shall be carried out, as appropriate, to determine the potential for the nuclear power plant to be exposed to seismically induced floods and water waves as, for example, from the failure during an earthquake of an upstream dam or from slides of earth or debris into a nearby lake.					
100.V.	Seismic and Geologic Design Bases					
100.V(a)	(a) *Determination of Design Basis for Vibratory Ground Motion*. The design of each nuclear power plant shall take into account the potential effects of vibratory ground motion caused by earthquakes. The design basis for the maximum vibratory ground motion and the expected vibratory ground motion should be determined through evaluation of the seismology, geology, and the seismic and geologic history of the site and the surrounding region. The most severe earthquakes associated with tectonic structures or tectonic provinces in the region surrounding the site should be identified, considering those historically reported earthquakes that can be associated with these structures or provinces and other relevant factors. If faults in the region surrounding the site are capable faults, the most severe earthquakes associated with these					

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	faults should be determined by also considering their geologic history. The vibratory ground motion at the site should be then determined by assuming that the epicenters or locations of highest intensity of the earthquakes are situated at the point on the tectonic structures or tectonic provinces nearest to the site. The earthquake which could cause the maximum vibratory ground motion at the site should be designated the Safe Shutdown Earthquake. The specific procedures for determining the design basis for vibratory ground motion are given in the following paragraphs. 1) *Determination of Safe Shutdown Earthquake*. The Safe Shutdown Earthquake shall be identified through evaluation of seismic and geologic information developed pursuant to the requirements of paragraph IV(a), as follows: (i) The historic earthquakes of greatest magnitude or intensity which have been correlated with tectonic structures pursuant to the requirements of paragraph (a)(6) of section IV shall be determined. In addition, for capable faults, the information required by paragraph (a)(8) of section IV shall also be taken into account in determining the earthquakes of greatest magnitude related to the faults. The magnitude or					

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	intensity of earthquakes based on geologic evidence may be larger than that of the maximum earthquakes historically recorded. The accelerations at the site shall be determined assuming that the epicenters of the earthquakes of greatest magnitude or the locations of highest intensity related to the tectonic structures are situated at the point on the structures closest to the site; (ii) Where epicenters or locations of highest intensity of historically reported earthquakes cannot be reasonably related to tectonic structures but are identified pursuant to the requirements of paragraph (a)(6) of section IV with tectonic provinces in which the site is located, the accelerations at the site shall be determined assuming that these earthquakes occur at the site; (iii) Where epicenters or locations of the highest intensity of historically reported earthquakes cannot be reasonably related to tectonic structures but are identified pursuant to the requirements of paragraph (a)(6) of section IV with tectonic provinces in which the site is not located, the accelerations at the site shall be determined assuming that the epicenters or locations of highest					

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	intensity of these earthquakes are at the closest point to the site on the boundary of the tectonic province; (iv) The earthquake producing the maximum vibratory acceleration at the site, as determined from paragraph (a)(1)(i) through (iii) of this section shall be designated the Safe Shutdown Earthquake for vibratory ground motion, except as noted in paragraph (a)(1)(v) of this section. The characteristics of the Safe Shutdown Earthquake shall be derived from more than one earthquake determined from paragraph (a)(1)(i) through (iii) of this section, where necessary to assure that the maximum vibratory acceleration at the site throughout the frequency range of interest is included. In the case where a causative fault is near the site, the effect of proximity of an earthquake on the spectral characteristics of the Safe Shutdown Earthquake shall be taken into account. The procedures in paragraphs (a)(1)(i) through (a)(1)(iii) of this section shall be applied in a conservative manner. The determinations carried out in accordance with paragraphs (a)(1)(ii) and (a)(1)(iii) shall assure that the safe					

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	shutdown earthquake intensity is, as a minimum, equal to the maximum historic earthquake intensity experienced within the tectonic province in which the site is located. In the event that geological and seismological data warrant, the Safe Shutdown Earthquake shall be larger than that derived by use of the procedures set forth in section IV and V of the appendix. The maximum vibratory accelerations of the Safe Shutdown Earthquake at each of the various foundation locations of the nuclear power plant structures at a given site shall be determined taking into account the characteristics of the underlying soil material in transmitting the earthquake-induced motions, obtained pursuant to paragraphs (a)(1), (3), and (4) of section IV. The Safe Shutdown Earthquake shall be defined by response spectra corresponding to the maximum vibratory accelerations as outlined in paragraph (a) of section VI; and (v) Where the maximum vibratory accelerations of the Safe Shutdown Earthquake at the foundations of the nuclear power plant structures are determined to be less than one-tenth					

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	the acceleration of gravity (0.1 g) as a result of the steps required in paragraphs (a)(1)(i) through (iv) of this section, it shall be assumed that the maximum vibratory accelerations of the Safe Shutdown Earthquake at these foundations are at least 0.1 g. 2) *Determination of Operating Basis Earthquake*. The Operating Basis Earthquake shall be specified by the applicant after considering the seismology and geology of the region surrounding the site. If vibratory ground motion exceeding that of the Operating Basis Earthquake occurs, shutdown of the nuclear power plant will be required. Prior to resuming operations, the licensee will be required to demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public. The maximum vibratory ground acceleration of the Operating Basis Earthquake shall be at least one-half the maximum vibratory ground acceleration of the Safe Shutdown Earthquake.					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
100.V(b)	(b) *Determination of Need to Design for Surface Faulting*. In order to determine whether a nuclear power plant is required to be designed to withstand the effects of surface faulting, the location of the nuclear power plant with respect to capable faults shall be considered. The area over which each of these faults has caused surface faulting in the past is identified by mapping its fault traces in the vicinity of the site. The fault traces are mapped along the trend of the fault for 10 miles in both directions from the point of its nearest approach to the nuclear power plant because, for example, traces may be obscured along portions of the fault. The maximum width of the mapped fault traces, called the control width, is then determined from this map. Because surface faulting has sometimes occurred beyond the limit of mapped fault traces or where fault traces have not been previously recognized, the control width of the fault is increased by a factor which is dependent upon the largest potential earthquake related to the fault. This larger width delineates a zone, called the zone requiring detailed faulting investigation, in which the possibility of surface faulting is to be determined. The following paragraphs outline the specific procedures for determining the zone requiring detailed faulting investigation for a capable fault. (1) *Determination of Zone Requiring Detailed Faulting Investigation*. The zone requiring detailed faulting investigation for a capable fault which was investigated pursuant to the requirement of paragraph (b)(7) of section IV shall be determined through use of the following					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	table: (Note: Refer to 10 CFR 100 for the table) The largest magnitude earthquake related to the fault shall be used in table 2. This earthquake shall be determined from the information developed pursuant to the requirements of paragraph (b) of Section IV for the fault, taking into account the information required by paragraph (b)(7) of section IV. The control width used in table 2 is determined by mapping the outer limits of the fault traces from information developed pursuant to paragraph (b)(7)(iv) of section IV. The control width shall be used in table 2 unless the characteristics of the fault are obscured for a significant portion of the 10 miles on either side of the point of nearest approach to the nuclear power plant. In this event, the use in table 2 of the width of mapped fault traces more than 10 miles from the point of nearest approach to the nuclear power plant may be appropriate. The zone requiring detailed faulting investigation, as determined from table 2, shall be used for the fault except where: (i) The zone requiring detailed faulting investigation from table 2 is less than one-half mile in width. In this case the zone shall be at least one-half mile in width; or (ii) Definitive evidence concerning the regional and local characteristics of the fault justifies use of a					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	different value. For example, thrust or bedding-plane faults may require an increase in width of the zone to account for the projected dip of the fault plane; or (iii) More detailed three-dimensional information, such as that obtained from precise investigative techniques, may justify the use of a narrower zone. Possible examples of such techniques are the use of accurate records from closely spaced drill holes or from closely spaced, high-resolution offshore geophysical surveys. In delineating the zone requiring detailed faulting investigation for a fault, the center of the zone shall coincide with the center of the fault at the point of nearest approach of the fault to the nuclear power plant as illustrated in figure 1.					
100.V(c)	(c) *Determination of Design Bases for Seismically Induced Floods and Water Waves*. The size of seismically induced floods and water waves which could affect a site from either locally or distantly generated seismic activity shall be determined, taking into consideration the results of the investigation required by paragraph (c) of section IV. Local topographic characteristics which might tend to modify the possible runup and drawdown at the site shall be considered. Adverse tide conditions shall also be taken into account in determining the effect of the floods and waves on the site. The characteristics of the earthquake to be used in					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	evaluating the offshore effects of local earthquakes shall be determined by a procedure similar to that used to determine the characteristics of the Safe Shutdown Earthquake in paragraph V(a).					
100.V(d)	(d) *Determination of Other Design Conditions*— (1) *Soil Stability*. Vibratory ground motion associated with the Safe Shutdown Earthquake can cause soil instability due to ground disruption such as fissuring, differential consolidation, liquefaction, and cratering which is not directly related to surface faulting. The following geologic features which could affect the foundations of the proposed nuclear power plant structures shall be evaluated, taking into account the information concerning the physical properties of materials underlying the site developed pursuant to paragraphs (a)(1), (3), and (4) of section IV and the effects of the Safe Shutdown Earthquake: (i) Areas of actual or potential surface or subsurface subsidence, uplift, or collapse resulting from: (a) Natural features such as tectonic depressions and cavernous or karst terrains, particularly those underlain by calcareous or other soluble deposits; (b) Man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams or reservoirs; and (c) Regional deformation. (ii) Deformational zones such as shears, joints,					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	fractures, folds, or combinations of these features. (iii) Zones of alteration or irregular weathering profiles and zones of structural weakness composed of crushed or disturbed materials. (iv) Unrelieved residual stresses in bedrock. (v) Rocks or soils that might be unstable because of their mineralogy, lack of consolidation, water content, or potentially undesirable response to seismic or other events. Seismic response characteristics to be considered shall include liquefaction, thixotropy, differential consolidation, cratering, and fissuring. (2) *Slope stability*. Stability of all slopes, both natural and artificial, the failure of which could adversely affect the nuclear power plant, shall be considered. An assessment shall be made of the potential effects of erosion or deposition and of combinations of erosion or deposition with seismic activity, taking into account information concerning the physical property of the materials underlying the site developed pursuant to paragraph (a)(1), (3), and (4) of section IV and the effects of the Safe Shutdown Earthquake. (3) *Cooling water supply*. Assurance of adequate cooling water supply for emergency and long-term shutdown decay heat removal shall be considered in the design of the nuclear power plant, taking in to account information concerning the physical properties of the materials underlying the site developed pursuant to paragraphs (a)(1), (3), and					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(4) of section IV and the effects of the Safe Shutdown Earthquake and the design basis for surface faulting. Consideration of river blockage or diversion or other failures which may block the flow of cooling water, coastal uplift or subsidence, or tsunami runup and drawdown, and failure of dams and intake structures shall be included in the evaluation, where appropriate. (4) *Distant structures*. Those structures which are not located in the immediate vicinity of the site but which are safety related shall be designed to withstand the effect of the Safe Shutdown Earthquake and the design basis for surface faulting determined on a comparable basis to that of the nuclear power plant, taking into account the material underlying the structures and the different location with respect to that of the site.					
100.VI.	Application to Engineering Design					
100.VI(a)	(a) *Vibratory ground motion*— (1) *Safe Shutdown Earthquake*. The vibratory ground motion produced by the Safe Shutdown Earthquake shall be defined by response spectra corresponding to the maximum vibratory accelerations at the elevations of the foundations of the nuclear power plant structures determine pursuant to paragraph (a)(1) of section V. The response spectra shall relate the response of the foundations of the nuclear power plant structures to the vibratory ground motion, considering such foundations to be single-degree-of-					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	freedom damped oscillators and neglecting soil-structure interaction effects. In view of the limited data available on vibratory ground motions of strong earthquakes, it usually will be appropriate that the response spectra be smoothed design spectra developed from a series of response spectra related to the vibratory motions caused by more than one earthquake. The nuclear power plant shall be designed so that, if the Safe Shutdown Earthquake occurs, certain structures, systems, and components will remain functional. These structures, systems, and components are those necessary to assure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part. In addition to seismic loads, including aftershocks, applicable concurrent functional and accident-induced loads shall be taken into account in the design of these safety-related structures, systems, and components. The design of the nuclear power plant shall also take into account the possible effects of the Safe Shutdown Earthquake on the facility foundations by ground disruption, such as fissuring, differential consolidation, cratering, liquefaction, and landsliding, as required in paragraph (d) of section V.					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The engineering method used to insure that the required safety functions are maintained during and after the vibratory ground motion associated with the Safe Shutdown Earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism. The analysis or test shall take into account soil-structure interaction effects and the expected duration of vibratory motion. It is permissible to design for strain limits in excess of yield strain in some of these safety-related structures, systems, and components during the Safe Shutdown Earthquake and under the postulated concurrent conditions, provided that the necessary safety functions are maintained. (2) *Operating Basis Earthquake*. The Operating Basis Earthquake shall be defined by response spectra. All structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subjected to the effects of the vibratory motion of the Operating Basis Earthquake in combination with					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	normal operating loads. The engineering method used to insure that these structures, systems, and components are capable of withstanding the effects of the Operating Basis Earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that the structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism. The analysis or test shall take into account soil-structure interaction effects and the expected duration of vibratory motion. (3) *Required Seismic instrumentation*. Suitable instrumentation shall be provided so that the seismic response of nuclear power plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can continue to be operated safely and to permit such timely action as may be appropriate. These criteria do not address the need for instrumentation that would automatically shut down a nuclear power plant when an earthquake occurs which exceeds a predetermined intensity. The need for such instrumentation is under consideration.					

Table A1-9: PART 100--REACTOR SITE CRITERIA						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
100.VI(b)	(b) *Surface Faulting*. (1) If the nuclear power plant is to be located within the zone requiring detailed faulting investigation, a detailed investigation of the regional and local geologic and seismic characteristics of the site shall be carried out to determine the need to take into account surface faulting in the design of the nuclear power plant. Where it is determined that surface faulting need not be taken into account, sufficient data to clearly justify the determination shall be presented in the license application. (2) Where it is determined that surface faulting must be taken into account, the applicant shall, in establishing the design basis for surface faulting on a site take into account evidence concerning the regional and local geologic and seismic characteristics of the site and from any other relevant data. (3) The design basis for surface faulting shall be taken into account in the design of the nuclear power plant by providing reasonable assurance that in the event of such displacement during faulting certain structures, systems, and components will remain functional. These structures, systems, and components are those necessary to assure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite					

Table A1-9: PART 100--REACTOR SITE CRITERIA

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	exposures comparable to the guideline exposures of this part. In addition to seismic loads, including aftershocks, applicable concurrent functional and accident-induced loads shall be taken into account in the design of such safety features. The design provisions shall be based on an assumption that the design basis for surface faulting can occur in any direction and azimuth and under any part of the nuclear power plant unless evidence indicates this assumption is not appropriate, and shall take into account the estimated rate at which the surface faulting may occur.					
100.VI(c)	(c) *Seismically Induced Floods and Water Waves and Other Design Conditions*. The design basis for seismically induced floods and water waves from either locally or distantly generated seismic activity and other design conditions determined pursuant to paragraphs (c) and (d) of section V, shall be taken into account in the design of the nuclear power plant so as to prevent undue risk to the health and safety of the public. **Figure 1**--Diagrammatic Illustration of Delineation of Width of Zone Requiring Detailed Faulting Investigations For Specific Nuclear Power Plant Location. (Note: Refer to 10 CFR 100 for the figure and notes)					

Note: For determining what portions of Part 140 should be reviewed, provisions applicable to applicants or licensees specified under both Subpart B and Subpart C were included.

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Subpart A--General Provisions					Heading
140.1	Purpose.					Exclude; Admin
140.2	Scope.					Exclude; Admin
140.3	Definitions.					Exclude; Admin
140.4	Interpretations.					Exclude; Admin
140.5	Communications.					Exclude; Admin
140.6	Reports.					Exclude; Admin
140.7	Fees.					Exclude; Admin
140.8	Specific exemptions.					Exclude; Admin
140.9	Modification of indemnity agreements.					Exclude; Admin
140.9a	Information collection requirements: OMB approval.					Exclude; Admin
	Subpart B--Provisions Applicable Only to Applicants and Licensees Other Than Federal Agencies and Nonprofit Educational Institutions					Heading
140.10	Scope. This subpart applies to each person who is an applicant for or holder of a license issued under 10 CFR parts 50 or 54 to operate a nuclear reactor, or is the applicant for or holder of a combined license issued under parts 52 or 54 of this chapter, except					

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	licenses held by persons found by the Commission to be Federal agencies or nonprofit educational institutions licensed to conduct educational activities. This subpart also applies to persons licensed to possess and use plutonium in a plutonium processing and fuel fabrication plant.					
140.11	Amounts of financial protection for certain reactors.					Note: Establishes financial protection requirements as a function of thermal power. Specifically applies to reactors used for electric power generation. Is silent on the issue of process heat.
140.11(a)	(a) Each licensee is required to have and maintain financial protection: (1) In the amount of $1,000,000 for each nuclear reactor he is authorized to operate at a thermal power level not exceeding ten kilowatts; (2) In the amount of $1,500,000 for each nuclear reactor he is authorized to operate at a thermal power level in excess of ten kilowatts but not in excess of one megawatt; (3) In the amount of $2,500,000 for each nuclear reactor other than a testing reactor or a reactor licensed under section 104b of the Act which he is authorized to operate at a thermal power level exceeding one megawatt but not in excess of ten megawatts; and (4) In an amount equal to the sum of $375,000,000 and the amount available as secondary financial protection (in the form of private liability insurance available under an industry retrospective rating plan providing for deferred premium charges equal to the pro rata share of the aggregate public liability claims and costs, excluding costs payment of which is not authorized by Section 170o.(1)(D), in excess					

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	of that covered by primary financial protection) for each nuclear reactor which is licensed to operate and which is designed for the production of electrical energy and has a rated capacity of 100,000 electrical kilowatts or more: Provided, however, that under such a plan for deferred premium charges for each nuclear reactor which is licensed to operate, no more than $111,900,000 with respect to any nuclear incident (plus any surcharge assessed under Subsection 170o.(1)(E) of the Act) and no more than $17,500,000 per incident within one calendar year shall be charged. *Except that,* where a person is authorized to operate a combination of 2 or more nuclear reactors located at a single site, each of which has a rated capacity of 100,000 or more electrical kilowatts but not more than 300,000 electrical kilowatts with a combined rated capacity of not more than 1,300,000 electrical kilowatts, each such combination of reactors shall be considered to be a single nuclear reactor for the sole purpose of assessing the applicable financial protection required under this section.					
140.11(b)	(b) In any case where a person is authorized under parts 50, 52, or 54 of this chapter to operate two or more nuclear reactors at the same location, the total primary financial protection required of the licensee for all such reactors is the highest amount which would otherwise be required for any one of those reactors; provided, that such primary financial protection covers all reactors at the location.					
140.12	Amount of financial protection required for other reactors.					
140.12(a)	(a) Each licensee is required to have and maintain					

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	financial protection for each nuclear reactor for which the amount of financial protection is not determined in § 140.11, in an amount determined pursuant to the formula and other provisions of this section: *Provided*, That in no event shall the amount of financial protection required for any nuclear reactor under this section be less than $4,500,000 or more than $74,000,000.					
140.12(b)	(b)(1) The formula is: x=B times P. (2) In the formula: x=Amount of financial protection in dollars. B=Base amount of financial protection. P=Population factor. (3) The base amount of financial protection is equal to $185 times the maximum power level, expressed in thermal kilowatts, as authorized by the applicable license. (4) The population factor (P) shall be determined as follows: (i) *Step 1*. The area to be considered includes all minor civil divisions (as shown in the 1950 Census of Population, Bureau of the Census, or later data available from the Bureau) which are wholly or partly within a circle with the facility at its center and having a radius in miles equal to the square root of the maximum authorized power level in thermal megawatts. (ii) *Step 2*. Identify all minor civil divisions according to the same census which are in whole or in part					

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	within the circle determined in Step 1. Determine the population of each such minor civil division (according to the same census or later data available from the Bureau of the Census). For each minor civil division, divide its population by the square of the estimated distance to the nearest mile from the reactor to the geographic center of the minor civil division: *Provided*, That no such distance shall be deemed to be less than one mile. If the sum of the quotients thus obtained for all minor civil divisions wholly or partly within the circle is 1,000 or less, the population factor is 1. If the sum of these quotients is more than 1,000 but not more than 3,000, the population factor is 1.2. If the sum of these quotients is more than 3,000 but not more than 5,000, the population factor is 1.4. If the sum of these quotients is more than 5,000 but not more than 7,000, the population factor is 1.6. If the sum of these quotients is more than 7,000 but not more than 9,000, the population factor is 1.8. If the sum of these quotients is more than 9,000 the population factor is 2.0.					
140.12(c)	(c) In any case where a person is authorized under parts 50, 52, or 54 of this chapter to operate two or more nuclear reactors at the same location, the total financial protection required of the licensee for all such reactors is the highest amount which would otherwise be required for any one of those reactors; provided, that such financial protection covers all reactors at the location.					
140.12(d)	(d) Except in cases where the amount of financial protection calculated under this section is a multiple of $100,000, amounts determined pursuant to this section shall be adjusted to the next highest					

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	multiple of $100,000.					
140.13	Amount of financial protection required of certain holders of construction permits and combined licenses under 10 CFR part 52. Each holder of a part 50 construction permit, or a holder of a combined license under part 52 of this chapter before the date that the Commission had made the finding under 10 CFR 52.103(g), who also holds a license under part 70 of this chapter authorizing ownership, possession and storage only of special nuclear material at the site of the nuclear reactor for use as fuel in operation of the nuclear reactor after issuance of either an operating license under 10 CFR part 50 or combined license under 10 CFR part 52, shall, during the period before issuance of a license authorizing operation under 10 CFR part 50, or the period before the Commission makes the finding under § 52.103(g) of this chapter, as applicable, have and maintain financial protection in the amount of $1,000,000. Proof of financial protection shall be filed with the Commission in the manner specified in § 140.15 of this chapter before issuance of the license under part 70 of this chapter.					Requirements for financial protection during construction.
140.13a	Amount of financial protection required for plutonium processing and fuel fabrication plants.	NA				Exclude
140.13b	Amount of liability insurance required for uranium enrichment facilities.	NA				Exclude
140.14	Types of financial protection.	Info				Exclude; description of types. No regulatory content.
140.15	Proof of financial protection.					Exclude; Admin
140.16	Commission review of proof of financial protection.					Exclude; Admin, if applicable.
140.17	Special provisions applicable to licensees furnishing					Exclude; Admin

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	financial protection in whole or in part in the form of liability insurance.					
140.18	Special provisions applicable to licensees furnishing financial protection in whole or in part in the form of adequate resources.					Exclude; Admin
140.19	Failure by licensees to maintain financial protection.					Exclude; Admin
140.20	Indemnity agreements and liens.					Exclude; Admin
140.21	Licensee guarantees of payment of deferred premiums.					Exclude; Admin
140.22	Commission guarantee and reimbursement agreements.					Exclude; Admin
	Subpart C--Provisions Applicable Only to Federal Agencies					Heading
140.51	Scope. This subpart applies only to persons found by the Commission to be Federal agencies, which have applied for or are holders of licenses issued pursuant to part 50 of this chapter authorizing operation of nuclear reactors. Note: Federal agencies are not required to furnish financial protection.					
140.52	Indemnity agreements.					
140.52(a)	(a) The Commission will execute and issue agreements of indemnity with each Federal agency subject to this subpart pursuant to the regulations in this part or such other regulations as may be issued by the Commission. Such agreements, as to any licensee, shall be effective on: (1) The effective date of the license (issued pursuant to part 50 of this chapter) authorizing the licensee to operate the nuclear reactor involved; or (2) The effective date of the license (issued pursuant to part 70 of this chapter) authorizing the licensee to possess and store special nuclear					

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	material at the site of the nuclear reactor for use as fuel in operation of the nuclear reactor after issuance of an operating license for the reactor, whichever is earlier. No such agreement, however, shall be effective prior to September 26, 1957.					
140.52(b)	(b)(1) The general form of indemnity agreement to be entered into with licensees subject to this subpart is contained in § 140.94 appendix D. (2) The form of indemnity agreement to be entered into by the Commission with any particular licensee under this subpart shall contain such modifications of the form in § 140.94, as are provided for in applicable licenses, regulations or orders of the Commission. (3) Each licensee who has executed an indemnity agreement under this subpart shall enter into such agreements amending such indemnity agreement as are required by applicable licenses, regulations or orders of the Commission.					
	Subpart D--Provisions Applicable Only to Nonprofit Educational Institutions					Heading
140.71	Scope.	NA				Exclude
140.72	Indemnity agreements.	NA				Exclude
	Subpart E--Extraordinary Nuclear Occurences					Heading
140.81	Scope and purpose.					Exclude; Admin
140.82	Procedures.					Exclude; Admin - this is the administrative process used by the Commission in determining whether there has been an Extraordinary Nuclear Occurrence.
140.83	Determination of extraordinary nuclear occurrence.					Exclude; Admin
140.84	Criterion I--Substantial discharge of radioactive material or substantial radiation levels offsite.					Exclude; Admin
140.85	Criterion II--Substantial damages to persons offsite					Exclude; Admin

Table A1-10: PART 140--FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	or property offsite.					
	Subpart F--Violations					Heading
140.87	Violations.					Exclude; Admin
140.89	Criminal penalties.					Exclude; Admin
140.91	Appendix A--Form of nuclear energy liability policy for facilities.	Info				Appendix A provides an example of an acceptable nuclear energy liability policy.
140.92	Appendix B--Form of indemnity agreement with licensees furnishing insurance policies as proof of financial protection.					Exclude; Admin - example of an indemnity agreement when insurance is used for financial protection.
140.93	Appendix C--Form of indemnity agreement with licensees furnishing proof of financial protection in the form of licensee's resources.					Exclude; Admin - example of an indemnity agreement when licensee resources are used for financial protection.
140.94	Appendix D--Form of indemnity agreement with Federal agencies.					Exclude; Admin
140.95	Appendix E--Form of indemnity agreement with nonprofit educational institutions.	NA				Exclude
140.96	Appendix F--Indemnity locations.					Exclude; specifies that geographical boundaries of indemnity locations includes the entire construction area during construction.
140.107	Appendix G--Form of indemnity agreement with licensees processing plutonium for use in plutonium processing and fuel fabrication plants and furnishing insurance policies as proof of financial protection.	NA				Exclude
140.108	Appendix H--Form of indemnity agreement with licensees possessing plutonium for use in plutonium processing and fuel fabrication plants and furnishing proof of financial protection in the form of the licensee's resources.	NA				Exclude
140.109	Appendix I.	Info				Exclude; Appendix I is the Nuclear Energy Liability Insurance Association master policy.

Table A1-11: PART 961--STANDARD CONTRACT FOR DISPOSAL OF SPENT NUCLEAR FUEL AND/OR HIGH-LEVEL RADIOACTIVE WASTE

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
961.1	Purpose. This part establishes the contractual terms and conditions under which the Department of Energy (DOE) will make available nuclear waste disposal services to the owners and generators of spent nuclear fuel (SNF) and high-level radioactive waste (HLW) as provided in section 302 of the Nuclear Waste Policy Act of 1982 (Pub. L. 97-425). Under the contract set forth in Sec. 961.11 of this part, DOE will take title to, transport, and dispose of spent nuclear fuel and/or high-level radioactive waste delivered to DOE by those owners or generators of such fuel or waste who execute the contract. In addition, the contract will specify the fees owners and generators of SNF and/or HLW will pay for these services. All receipts, proceeds, and revenues realized by DOE under the contract will be deposited in the Nuclear Waste Fund, an account established by the Act in the U.S. Treasury. This fund will pay for DOE's radioactive waste disposal activities, the full costs of which will be borne by the owners and generators under contract with DOE for disposal services.					Note that fees are expected to cover disposal costs.
961.2	Applicability. This part applies to the Secretary of Energy or his designee and any person who owns or generates spent nuclear fuel or high-level radioactive waste, of domestic origin, generated in a civilian nuclear power reactor. If executed in a timely manner, the contract contained in this part will commit DOE to accept title to, transport, and dispose of such spent fuel and waste. In exchange for these services, the owners or generators of					Note that fees are expected to cover disposal costs.

Table A1-11: PART 961--STANDARD CONTRACT FOR DISPOSAL OF SPENT NUCLEAR FUEL AND/OR HIGH-LEVEL RADIOACTIVE WASTE

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	such fuel or waste shall pay fees specified in the contract which are intended to recover fully the costs of the disposal services to be furnished by DOE. The contract must be signed by June 30, 1983, or by the date on which such owner or generator commences generation of, or takes title to, such spent fuel or waste, whichever occurs later.					
961.3	Definitions.					Exclude; Admin
961.4	Deviations.					Exclude; Admin
961.5	Federal agencies.					Exclude; Admin; however, specifics of the ownership and agency agreements may affect whether an interagency agreement is needed.
961.11	Text of the contract.					See the actual regulation for a formatted copy of the contract including tables of data. The text of the contract is heavily oriented toward LWR fuel and the basic fees are based on electricity production and do not take into account the possibility of reactors that produce process heat.

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
RG-1.1 (Rev. 0, November 1970)	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	NA				Exclude, This plant has passive safety systems so there are no safety related heat removal pumps.
RG-1.2	Withdrawn (See 56 FR 36175, 7/31/1991)	NA				Exclude.
RG-1.3 (Rev. 2, June 1974)	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-coolant Accident for Boiling Water Reactors	NA				Exclude, This regulatory guide is applicable only to BWRs and, therefore, is not applicable to the HTGR reactor. New design basis accidents and a new source term methodology will be developed for the HTGR.
RG-1.4 (Rev. 2, June 1974)	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors	NA				Exclude, This regulatory guide is applicable only to PWRs and, therefore, is not applicable to the HTGR reactor. New design basis accidents and a new source term methodology will be developed for the HTGR.
RG-1.5 (Rev. 0, March 1971)	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	NA				Exclude, This regulatory guide is applicable only to BWRs and, therefore, is not applicable to the HTGR reactor. New design basis accidents and a new source term methodology will be developed for the HTGR..
RG-1.6 (Rev. 0, March 1971)	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems					
RG-1.6.D.1	The electrically powered safety loads (a-c and d-c) should be separated into redundant load groups such that loss of any one group will not prevent the					

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	minimum safety functions from being performed.					
RG-1.6.D.2	Each a-c load group should have a connection to the preferred (offsite) power source and to a standby (onsite) power source (usually a single diesel generator). The standby power source should have no automatic connection to any other redundant load group. At multiple nuclear unit sites, the standby power source for one load group may have an automatic connection to a load group of a different unit. A preferred power source bus, however, may serve redundant load groups.					
RG-1.6.D.3	Each d-c load group should be energized by a battery and battery charger. The battery-charger combination should have no automatic connection to any other redundant d-c load group.					
RG-1.6.D.4	When operating from the standby sources, redundant load groups and the redundant standby sources should be independent of each other at least to the following extent: a. The standby source of one load group should not be automatically paralleled with the standby source of another load group under accident conditions; b. No provisions should exist for automatically connecting one load group to another load group; c. No provisions should exist for automatically transferring loads between redundant power sources; d. If means exist for manually connecting redundant load groups together, at least one interlock should be provided to prevent an operator error that would parallel their standby power sources.					
RG-1.6.D.5	A single generator driven by a single prime mover is acceptable as the standby power source for each a-c load group of the size and characteristics typical of recent applications. If other arrangements such as multiple diesel generators operated in parallel or multiple prime movers driving a single generator are proposed, the applicant should demonstrate that the proposed arrangement has an equivalent reliability. Common mode failures as well as random single failures should be considered in the					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	analysis.					
RG-1.7 (Rev 3, March 2007)	Control of Combustible Gas Concentration in Containment					
RG-1.7.C.1.	Combustible Gas Control Systems The following design guidance is applicable to combustible gas control systems installed to mitigate the risk associated with combustible gas generation attributed to beyond-design-basis accidents. Structures, systems, and components (SSCs) installed to mitigate the hazard from the generation of combustible gas in containment should be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. Equipment survivability expectations under severe accident conditions should consider the circumstances of applicable initiating events (such as station blackout[1] or earthquakes) and the environment (including pressure, temperature, and radiation) in which the equipment is relied upon to function. This guidance was presented in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs" (Ref. 6). The required system performance criteria will be based on the results of design-specific reviews that include probabilistic risk assessment as required by 10 CFR 52.47(a). Because these requirements address beyond-design-basis combustible gas control, SSCs provided to meet these requirements need not be subject to the environmental qualification requirements of 10 CFR 50.49, quality assurance requirements of Appendix B to 10 CFR Part 50, and redundancy/diversity requirements of Appendix A to 10 CFR Part 50. Guidance such as that found in Appendices A and B to Regulatory Guide 1.155 (Ref. 7) is appropriate for equipment used to mitigate the consequences of severe accidents. This guidance was used to review the design of evolutionary and passive plant designs, as documented in NUREG-1462 (Ref. 8), NUREG-1503 (Ref. 9), and NUREG-					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	1512 (Ref. 10). The combustible gas control systems in all BWRs with Mark III-type containments and all PWRs with ice condenser type containments must meet the requirements in Section 50.44. The staff considers that the combustible gas control systems installed and approved by the NRC as of October 16, 2003, are acceptable without modification. [1] Section 50.44 does not require the deliberate ignition systems used by BWRs with Mark III type containments and PWRs with ice condenser type containments to be available during station blackout events. The deliberate ignition systems should be available upon restoration of power. Additional guidance concerning the availability of deliberate ignition systems during station blackout sequences is being developed as part of the staff's review of Generic Safety Issue 189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."					
RG-1.7.C.2.	Hydrogen and Oxygen Monitors 2.1 Hydrogen Monitors Section 50.44 requires that equipment be provided for monitoring hydrogen in the containment. The equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a beyond-design-basis accident for accident management, including emergency planning. Safety-related hydrogen monitoring systems installed and approved by the NRC prior to October 16, 2003, are sufficient to meet these criteria. Non-safety-related commercial-grade hydrogen monitors can also be used to meet these criteria if they comply with the following criteria: **(1) Equipment Survivability:** The hydrogen monitoring equipment need not be qualified in accordance with 10 CFR 50.49. However, these systems					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	are required to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond-design-basis accident environment. The evaluation of survivability should consider the effects of the post-accident environment for the specific type of facility and monitoring system design. The procurement for such equipment should address equipment reliability and operability in the beyond-design-basis accident environmental conditions for the specific facility and monitoring system design. Acceptable approaches for demonstrating equipment survivability are described in Chapter 19 of the ABWR FSER (Ref. 9) and the AP1000 FSER (Ref. 11). **(2) Power Source:** The instrumentation should be energized from a high-reliability power source, not necessarily standby power, and should be backed up by batteries where momentary interruption is not tolerable. **(3) Quality Assurance:** The instrumentation should be of high-quality commercial grade and should be selected to withstand the specified service environment. **(4) Display and Recording:** The instrumentation signal may be displayed on an individual instrument or it may be processed for display on demand. If direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available on redundant dedicated recorders. Otherwise, it may be continuously updated, stored in computer memory, and displayed on demand. Intermittent displays such as data loggers and scanning recorders may be used if no significant transient response information is likely to be lost by such devices. **(5) Range:** If two or more instruments are needed to cover a particular range, overlapping of instrument span should be provided. If the required range of monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments should be					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	used. **(6) Servicing, Testing, and Calibration:** Servicing, testing, and calibration programs should be specified to maintain the capability of the monitoring instrumentation. If the required interval between testing is less than the normal time interval between plant shutdowns, a capability for testing during power operation should be provided. Whenever means for removing channels from service are included in the design, the design should facilitate administrative control of the access to such removal means. The design should facilitate administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points. Periodic checking, testing, calibration, and calibration verification should be in accordance with the applicable portions of Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems" (Ref. 12), pertaining to testing of instrument channels. (Note: Response time testing not usually needed.) **(7) Human Factors:** The instrumentation should be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules. The monitoring instrumentation design should minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator. Human factors analysis should be used in determining the type and location of displays. Rev. 3 of RG 1.7, Page 6. To the extent practicable, the same instruments should be used for accident monitoring as are used for the normal operations of the plant to enable the operators to use, during accident situations, instruments with which they are					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	most familiar. **(8) Direct Measurement:** To the extent practicable, monitoring instrumentation inputs should be from sensors that directly measure the desired variables. An indirect measurement should be made only when it can be shown by analysis to provide unambiguous information. The above provisions can be met with a program based on compliance with a pre-specified, structured program of testing and calibration; alternatively, these items can be met with a less-prescriptive, performance-based approach to assurance of the hydrogen monitoring function. Such an approach is consistent with SECY-00-0191, "High-Level Guidelines for Performance-Based Activities" (Ref. 13). Specifically, assurance of the reliability, availability, and capability of the hydrogen monitoring function can be derived through tracking actual reliability performance (including calibration) against targets established by the licensee based on the significance of this function, which is determined on a plant specific basis. Thus, for hydrogen monitoring, it is acceptable to accomplish the functions of servicing, testing, and calibration within the maintenance rule program provided that applicable targets are established based on the functions of the hydrogen monitors delineated above. Section 50.44 also requires that hydrogen monitors be functional. Functional requirements can be found in Three Mile Island (TMI) Action Plan Item II.F.1, Attachment 6, in NUREG-0737 (Ref. 14), which states that hydrogen monitors are to be functioning within 30 minutes of the initiation of safety injection. This requirement was imposed by confirmatory orders following the accident at TMI Unit 2. Since that requirement was issued, the staff has determined that the 30-minute requirement can be overly burdensome. Through the "Confirmatory Order Modifying Post-TMI Requirements Pertaining to Containment Hydrogen Monitors for Arkansas Nuclear One, Units 1 and 2" (Ref. 15), the staff developed a method for licensees to adopt a risk-informed functional requirement in lieu of the 30-minute					

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	requirement. As described in the confirmatory order, an acceptable functional requirement would meet the following requirements: (1) Procedures shall be established for ensuring that indication of hydrogen concentration in the containment atmosphere is available in a sufficiently timely manner to support the role of information in the emergency plan (and related procedures) and related activities such as guidance for the severe accident management plan. (2) Hydrogen monitoring will be initiated on the basis of the following considerations: (a) The appropriate priority for establishing indication of hydrogen concentration within containment in relation to other activities in the control room. (b) The use of the indication of hydrogen concentration by decision-makers for severe accident management and emergency response. (c) Insights from experience or evaluation pertaining to possible scenarios that result in significant generation of hydrogen that would be indicative of core damage or a potential threat to the integrity of the containment building. The NRC staff has found that adoption of this functional requirement by licensees results in the hydrogen monitors being functional within 90 minutes after the initiation of safety injection. This period of time includes equipment warm-up but not equipment calibration.					

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	2.2 Oxygen Monitors Section 50.44 requires that equipment be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. The revised rule requires the equipment for monitoring oxygen to be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a beyond-design-basis accident for combustible gas control and accident management, including emergency planning. Existing oxygen monitoring systems approved by the NRC prior to October 16, 2003, are sufficient to meet these criteria. Non-safety-related oxygen monitors would also meet these criteria if they comply with the following provisions: **(1) Equipment Survivability:** The oxygen monitoring equipment need not be qualified in accordance with 10 CFR 50.49. However, these systems are required to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond-design-basis accident environment. The evaluation of survivability should consider the effects of the post-accident environment for the specific type of facility and monitoring system design. The procurement for such equipment should address equipment reliability and operability in the beyond-design-basis accident environmental conditions for the specific facility and monitoring system design. Acceptable approaches for demonstrating equipment survivability are described in Chapter 19 of the ABWR FSER (Ref. 9) and the AP1000 FSER (Ref. 11). **(2) Power Source:** The instrumentation should be energized from a high-reliability power source, not necessarily standby power, and should be backed up by batteries where momentary interruption is not tolerable. **(3) Channel Availability:** The out-of-service interval should be based on normal technical specification requirements on out of service for the system					

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	it serves where applicable or where specified by other requirements. **(4) Quality Assurance:** The recommendations of the following regulatory guides pertaining to quality assurance should be followed: • Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)" (Ref. 16) • Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (Ref. 17) • Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" (Ref. 18) • Regulatory Guide 1.176, "An Approach for Plant-Specific, Risk-Informed Decision-making: Graded Quality Assurance" (Ref. 19) **(5) Display and Recording:** The instrumentation signal may be displayed on an individual instrument or it may be processed for display on demand. If direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available on redundant dedicated recorders. Otherwise, it may be continuously updated, stored in computer memory, and displayed on demand. Intermittent displays such as data loggers and scanning recorders may be used if no significant transient response information is likely to be lost by such devices. Rev. 3 of RG 1.7, Page 8 **(6) Range:** If two or more instruments are needed to cover a particular range, overlapping of instrument span should be provided. If the required range of monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments should be used. **(7) Interfaces:** The transmission of signals for other use should be through isolation devices that are designated as part of the monitoring					

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	instrumentation and that meet the provisions of the criteria presented here. **(8) Servicing, Testing, and Calibration:** Servicing, testing, and calibration programs should be specified to maintain the capability of the monitoring instrumentation. If the required interval between testing is less than the normal time interval between plant shutdowns, a capability for testing during power operation should be provided. Whenever means for removing channels from service are included in the design, the design should facilitate administrative control of the access to such removal means. The design should facilitate administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points. Periodic checking, testing, calibration, and calibration verification should be in accordance with the applicable portions of Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," (Ref. 12) pertaining to testing of instrument channels. (Note: Response time testing not usually needed.) The location of the isolation device should be such that it would be accessible for maintenance during accident conditions. **(9) Human Factors:** The instrumentation should be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules. The monitoring instrumentation design should minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator. Human factors analysis should be used in determining the type and location of displays.					

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	To the extent practicable, the same instruments should be used for accident monitoring as are used for the normal operations of the plant to enable the operators to use, during accident situations, instruments with which they are most familiar. **(10) Direct Measurement:** To the extent practicable, monitoring instrumentation inputs should be from sensors that directly measure the desired variables. An indirect measurement should be made only when it can be shown by analysis to provide unambiguous information.					
RG-1.7.C.3.	Atmosphere Mixing Systems Section 50.44 requires that all containments have a capability for ensuring a mixed atmosphere. This capability may be provided by an active, passive, or combination system. Active systems may consist of a fan, a fan cooler, or containment spray. For passive or combination systems that use convective mixing to mix the combustible gases, the containment internal structures should have design features that promote the free circulation of the atmosphere. All containment types should have an analysis of the effectiveness of the method used for providing a mixed atmosphere. This analysis should demonstrate that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity.[2] Atmosphere mixing systems prevent local accumulation of combustible or detonable gases that could threaten containment integrity or equipment operating in a local compartment. Active systems installed to mitigate this threat should be reliable, redundant, single-failure-proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power. The NRC staff considers atmosphere mixing systems installed and approved by the NRC as of October 16, 2003, to be acceptable without modification.					

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	References 20 through 23 provide important insights into the potential for detonation of hydrogen-air mixtures. [2] The NRC staff believes that current lumped parameter analytical codes may overestimate mixing processes (in particular, natural convection). Applicants should substantiate the applicability of these codes to their analyses through sensitivity studies, validation with data, or other means.					
RG-1.7.C.4.	Hydrogen Gas Production Materials within the containment that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.					
RG-1.7.C.5.	Section 50.44 requires that containment structural integrity be demonstrated by use of an analytical technique that is accepted by the NRC staff. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The following criteria of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 24) provide an acceptable method for demonstrating that the requirements are met: (1) Steel containments meet the requirements of the ASME Boiler and Pressure Vessel Code (edition and addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Subsubarticle NE - 3220, Service Level C Limits, considering pressure and dead load alone (evaluation of instability is not required). (2) Concrete containments meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC - 3720, Factored Load Category, considering pressure and dead load alone. As a minimum, the specific code requirements set forth for each type of					

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	containment should be met for a combination of dead load and an internal pressure of 45 psig. The staff will consider modest deviations from these criteria, if the applicant shows good cause. These criteria, which no longer are contained in Section 50.44, remain acceptable to the NRC staff for meeting the current regulations. The acceptability of licensee analyses using the ASME Code criteria remains unaffected by this rulemaking.					
RG-1.8 (Rev. 3, May 2000)	Qualification and Training of Personnel for Nuclear Power Plants					Exclude, This is a COL item for the licensee to address.
RG-1.9 (Rev. 4, March 2007)	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	NA				Exclude, The HTGR design does not rely on safety related emergency diesel generators.
RG-1.10	Withdrawn (See 46 FR 37579, 07/21/1981)	NA				Exclude,
RG-1.11 (Rev. 0, March 2010)	Instrument Lines Penetrating Primary Reactor Containment					
RG-1.11.C.1.	Instrument lines penetrating the primary containment that are connected to instruments that are part of the protection or safety systems are extensions of those systems and should support those systems achieving their requirements for redundancy, independence, and testability to ensure the systems safety functions are accomplished.					
RG-1.11.C.2.	Instrument lines penetrating the primary containment that are part of the reactor coolant boundary should be sized or orificed in such a manner as to ensure that the following occurs in the event of any breach of the line outside of the primary containment during normal reactor operation: a. The leakage is reduced to the maximum extent practical consistent with other safety requirements. b. The rate and extent of coolant loss are within the capability of the normal reactor coolant makeup system. Rev. 1 of RG 1.11, Page 4					
RG-1.11.C.3.	Instrument lines penetrating the primary containment should be provided with an automatically operated isolation valve or one that an operator can					

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	manually operate from a remote location (e.g., in the control room or in another appropriate location). The valve should be located in the line outside containment as close to containment as practical. Excess-flow check valves may provide acceptable automatic operation in this application. There should be a high degree of assurance that these valves will perform as follows: a. They will not close accidentally during normal reactor operation. b. They will close or can be readily closed if the integrity of the instrument line outside containment is lost during normal reactor operation or under accident conditions. c. They will reopen or can be readily reopened under the conditions that would prevail when reopening them is appropriate. Power-operated valves should remain "as is" upon loss of power. The status (opened and closed) of all such isolation valves should be indicated in the control room. If a remotely operated valve is provided, sufficient information should be available in the control room or other appropriate location to ensure that the operator can take timely and proper actions.					
RG-1.11.C.4.	Instrument lines penetrating the primary containment that are connected to instruments that provide input signals to the protection or safety systems and are closed systems both inside and outside of containment (e.g., for containment pressure instrumentation) are acceptable without containment isolation valves if they meet the conditions specified in Section 3.6.2 of American National Standards Institute (ANSI) N271-1976, "Containment Isolation Provisions for Fluid Systems" (Ref. 3).					
RG-1.11.C.5.	Instrument lines penetrating primary containment should be designed conservatively from containment out to and including the isolation valve and should be of a quality at least equivalent to that of containment. These portions of the lines should be located and protected so as to minimize the likelihood of their being accidentally damaged. They should be protected or separated to prevent the failure of one line from contributing to the failure of any other line. Provisions should be included to permit periodic visual inspection during plant operation, particularly of those portions of the lines					

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	outside containment out to and including the isolation valve.					
RG-1.11.C.6.	Instrument lines penetrating the primary containment should not be so restricted by components in the lines, such as valves and orifices, that the response time of the connected instrumentation could be increased unacceptably.					
RG-1.11.C.7.	Instrument lines penetrating the primary containment that are not associated with protection or safety system instrumentation should meet the provisions of the following: a. positions 2, 3, 5, and 6 above, or b. Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems" (Ref. 4).					
RG-1.12 (Rev. 2, March 1997)	Nuclear Power Plants Instrumentation for Earthquakes The type, locations, operability, characteristics, installation, actuation, remote indication, and maintenance of seismic instrumentation described below are acceptable to the NRC staff for satisfying the requirements in 10 CFR Part 20 and Paragraph IV(a)(4) of Appendix S to 10 CFR Part 50 for ensuring the safety of nuclear power plants.					
RG-1.12.C.1.	Seismic Instrumentation Type and Location 1.1 Solid-state digital instrumentation that will enable the processing of data at the plant site within 4 hours of the seismic event should be used. 1.2 A triaxial time-history accelerograph should be provided at the following locations: 1. Free-field. 2. Containment foundation. 3. Two elevations (excluding the foundation) on a structure inside the containment. 4. An independent Seismic Category I structure foundation where the response is different from that of the containment structure. *5.* An elevation (excluding the foundation) on the independent Seismic					

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	Category I structure selected in 4 above. 6. If seismic isolators are used, instrumentation should be placed on both the rigid and isolated portions of the same or an adjacent structure, as appropriate, at approximately the same elevations. 1.3 The specific locations for instrumentation should be determined by the nuclear plant designer to obtain the most pertinent information consistent with maintaining occupational radiation exposures ALARA for the location, installation, and maintenance of seismic instrumentation. In general: 1.3.1 The free-field sensors should be located and installed so that they record the motion of the ground surface and so that the effects associated with.- surface features, buildings, and components on the recorded ground motion will be insignificant. 1.3.2 The in-structure instrumentation should be placed at locations that have been modeled as mass points in the building dynamic analysis so that the measured motion can be directly compared with the design spectra. The instrumentation should not be located on a secondary structural frame member that is not modeled as a mass point in the building dynamic model. **1.3.3** A design review of the location, installation, and maintenance of proposed instrumentation for maintaining exposures ALARA should be performed by the facility in the planning stage in accordance with Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." 1.3.4 Instrumentation should be placed in a location with as low a dose rate as is practical, consistent with other requirements. 1.3.5 Instruments should be selected to require minimal maintenance and in-service inspection, as well as minimal time and numbers of personnel to					

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	conduct installation and maintenance.					
RG-1.12.C.2.	Instrumentation at Multi-Unit Sites Instrumentation in addition to that installed for a single unit will not be required if essentially the same seismic response is expected at the other units based on the seismic analysis used in the seismic design of the plant. However, if there are separate control rooms, annunciation should be provided to both control rooms as specified in Regulatory Position 7.					
RG-1.12.C.3.	Seismic Instrumentation Operability The seismic instrumentation should operate during all modes of plant operation, including periods of plant shutdown. The maintenance and repair procedures should provide for keeping the maximum number of instruments in service during plant operation and shutdown.					
RG-1.12.C.4.	Instrumentation Characteristics 4.1 The design should include provisions for in-service testing. The instruments should be capable of periodic channel checks during normal plant operation. 4.2 The instruments should have the capability for in-place functional testing. 4.3 Instrumentation that has sensors located in inaccessible areas should contain provisions for data recording in an accessible location, and the instrumentation should provide an external remote alarm to indicate actuation. 4.4 The instrumentation should record, at a minimum, 3 seconds of low-amplitude motion prior to seismic trigger actuation, continue to record the motion during the period in which the earthquake motion exceeds the seismic trigger threshold, and continue to record low-amplitude motion for a minimum of 5 seconds beyond the last accidence of the seismic trigger threshold. 4.5 The instrumentation should be capable of recording 25 minutes of					

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	sensed motion. 4.6 The battery should be of sufficient capacity to power the instrumentation to sense and record (see Regulatory Position 4.5) 25 minutes of motion over a period of not less than the channel check test interval (Regulatory Position 8.2). This can be accomplished by providing enough battery capacity for a minimum of 25 minutes of system operation at any time over a 24-hour period, without recharging, in combination with a battery charger whose line power is connected to an uninterruptable power supply or a line source with an alarm that is checked at least every 24 hours. Other combinations of larger battery capacity and alarm intervals may be used. 4.7 Acceleration Sensors 4.7.1 The dynamic range should be 1000:1 zero to peak, or greater; for example, **0.001g** to 1.0g. 4.7.2 The frequency range should be 0.20 Hz to 50 Hz or an equivalent demonstrated to be adequate by computational techniques applied to the resultant accelerogram. 4.8 Recorder 4.8.1 The sample rate should be at least 200 samples per second in each of the three directions. 4.8.2 The bandwidth should be at least from 0.20 Hz to 50 Hz. 4.8.3 The dynamic range should be 1000:1 or greater, and the instrumentation should be able to record at least 1.0g zero to peak. 4.9 Seismic Trigger The actuating level should be adjustable and within the range of **0.001g** to 0.02g.					
RG-1.12.C.5.	Instrumentation Installation 5.1 The instrumentation should be designed and installed so that the					

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	mounting is rigid. 5.2 The instrumentation should be oriented so that the horizontal components are parallel to the orthogonal horizontal axes assumed in the seismic analysis. 5.3 Protection against accidental impacts should be provided.					
RG-1.12.C.6.	Instrumentation Actuation **6.1** Both vertical and horizontal input vibratory ground motion should actuate the same time-history accelerograph. One or more seismic triggers may be used to accomplish this. 6.2 Spurious triggering should be avoided. 6.3 The seismic trigger mechanisms of the time history accelerograph should be set for a threshold ground acceleration of not more than 0.02g.					
RG-1.12.C.7.	Remote Indication Triggering of the free-field or any foundation-level time-history accelerograph should be annunciated in the control room. If there is more than one control room at the site, annunciation should be provided to each control room.					
RG-1.12.C.8.	Maintenance 8.1 The purpose of the maintenance program is to ensure that the equipment will perform as required. As stated in Regulatory Position 3, the maintenance and repair procedures should provide for keeping the maximum number of instruments in service during plant operation and shutdown. 8.2 Systems are to be given channel checks every 2 weeks for the first 3 months of service after startup. Failures of devices normally occur during initial operation. After the initial 3-month period and 3 consecutive successful checks, monthly channel checks are sufficient. The monthly channel check is to include checking the batteries. The channel functional test should be performed every 6 months. Channel calibration should be performed during each refueling outage at a minimum.					
RG-1.13 (Rev.	Spent Fuel Storage Facility Design Basis					

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RG No./Rev.	**RG Title**	**Applicable**	**Reg. or Guidance**	**Add'l Design Info**	**Add'l Reg. Needed**	**Basis/Comment**
2, March 2007)	With the additions, clarifications, and exceptions presented in Section B of this guide, the guidance presented in ANSI Standard N210-1976/ANS-57.2-1983 (Ref. 1) is acceptable for use in the design of spent fuel storage facilities at light-water nuclear power plants. The following resulting regulatory positions are formally presented for clarity.					
RG-1.13.C.1.	Seismic Design With the additions, clarifications, and exceptions presented in Section B of this guide, the guidance presented in ANSI Standard N210-1976/ANS-57.2-1983 (Ref. 1) is acceptable for use in the design of spent fuel storage facilities at light-water nuclear power plants. The following resulting regulatory positions are formally presented for clarity.					
RG-1.13.C.2.	Protection Against Extreme Winds The spent fuel storage facility should be designed to (a) keep extreme winds and missiles generated by those winds from causing significant loss of watertight integrity of the fuel storage pool, and (b) keep missiles generated by extreme winds from contacting fuel within the pool. For those nuclear plants that are located in areas of the country where tornadoes cause the strongest winds, refer to Regulatory Guide 1.76,					
RG-1.13.C.3.	Protection Against Turbine Missiles The spent fuel storage facility should be designed to protect the spent fuel from low-trajectory turbine missiles, and the storage pool should retain watertight integrity if struck by such missiles. Regulatory Guide 1.115, “Protection Against Low-Trajectory Turbine Missiles” (Ref. 9), provides guidance for appropriate protection against low-trajectory turbine missiles.					
RG-1.13.C.4.	Confinement and Filtering Systems A controlled-leakage building should enclose the fuel to limit the potential release of radioactive iodine and other radioactive materials. If necessary to					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	limit offsite dose consequences from a fuel handling accident or spent fuel pool boiling, the building should include an engineered safety feature filtration system that meets the guidelines outlined in Regulatory Guide 1.52.					
RG-1.13.C.5.	Control of Heavy Loads Cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving over the pool. Furthermore, the spent fuel storage facility design should have at least one of the following provisions with respect to the handling of heavy loads, including the spent fuel cask: (a) Cranes should be designed to provide single-failure-proof handling of heavy loads, so that a single failure will not result in the crane handling system losing the capability to perform its safety function. (b) The spent fuel cask-loading area should be designed to withstand, without significant leakage of the adjacent spent fuel storage, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted.					
RG-1.13.C.6.	Drainage Prevention Drains, permanently connected mechanical or hydraulic systems, and other features that (by maloperation or failure) could reduce the coolant inventory to unsafe levels should not be installed or included in the design. No piping penetrations through the storage pool wall should be below the minimum water level required for shielding. Siphon breakers, check valves, and other devices should be used to preclude accidental draining by hydraulic systems. In addition, the spent fuel storage facility should comply with one of the following criteria: (a) If the spent fuel pool cooling system is designed to Quality Group C, Seismic Category I requirements, drains, piping, or other systems should be unable to reduce the coolant inventory to a level that would prevent the					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	cooling system from maintaining the storage pool below its design temperature limit. (b) If the spent fuel pool is designed to allow coolant boiling during accident conditions, no drains, piping, or other systems should be installed that would allow coolant levels to drain below adequate					
RG-1.13.C.7.	Instrumentation Reliable and frequently tested monitoring equipment should be provided to alarm both locally and in a continuously manned location if the water level in the fuel storage pool falls below a predetermined level, if the water temperature exceeds a predetermined level, or if high local radiation levels are experienced. The high-radiation-level instrumentation should signal automatic ventilation and/or filtration functions that are consistent with the dose consequence evaluation for fuel-handling Accidents.					
RG-1.13.C.8.	Makeup Water A Quality Group C, Seismic Category I makeup system should be provided to add coolant to the pool. Appropriate redundancy or a backup system for filling the pool from a reliable source, such as a lake, river, or onsite Seismic Category I water-storage facility, should be provided. If the spent fuel pool cooling system is designed to the requirements of Quality Group C, Seismic Category I, the backup to the makeup system need not be permanently installed or designed to Seismic Category I requirements; however, the backup system should still take water from a Seismic Category I source. The makeup system and its backup should have redundant flow paths for providing water to the storage pool. The capacity of the makeup systems should exceed the larger of (1) the pool leakage rate, assuming spent fuel pool liner perforation resulting from a dropped fuel assembly, or (2) the evaporation rate necessary to remove 0.3 percent of the rated reactor thermal power.					
RG-1.13.C.9.	Pool Cooling					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	The spent fuel storage facility should include a system for cooling the pool water in order to maintain a bulk temperature below 60 °C (140 °F) for all heat load conditions, including full-core offloads during refueling. Administrative controls may be used to ensure that this temperature limit is not exceeded. However, the minimum heat removal capacity with the forced-circulation cooling system in operation, the pool at the design temperature of the structure, and the heat sink at its maximum design temperature should exceed 0.3 percent of the reactor rated thermal power. One of the two following conditions should also be satisfied: (a) The spent fuel pool cooling system is designed to meet Quality Group C, Seismic Category I requirements. (b) The spent fuel pool cooling system is not designed to meet Quality Group C, Seismic Category I requirements. However, the pool structure and liner are designed to withstand coolant boiling; the pool makeup system and its backup are designed to Quality Group C, Seismic Category I requirements; and the building ventilation system has the capability to vent steam or moisture to the atmosphere to protect safety-related components from high temperatures and moisture levels. If necessary to limit offsite dose consequences from venting steam or moisture during accident conditions, the ventilation system should meet the guidelines of regulatory Guide 1.52.					
RG-1.13.C.10.	Gates and Weirs Gates and weirs that isolate the spent fuel storage pool from the adjacent fuel-handling areas should be designed to prevent the coolant inventory from being drained below the top of the fuel assemblies. The volume of the fuel-handling areas adjacent to the storage pool (e.g., cask-loading area, transfer canal) should be limited so that if the seal(s) of a single gate were to fail and the pool water drained into one of these areas, pool coolant inventory would not be reduced to a level less than 3 meters (10 feet) above the top of the fuel assemblies.					
RG-1.13.C.11.	Fuel Cooling					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	The spent fuel storage racks should be designed in a manner that allows for adequate coolant flow to all stored fuel assemblies. A thermal-hydraulic analysis should demonstrate that the racks provide adequate natural circulation to prevent nucleate boiling within the stored assemblies.					
RG-1.13.C.12.	Leakage Containment The spent fuel storage pool should include a system for detecting and containing pool liner leaks. Segmented leak channels, proper drainage, and sumps for collecting and containing such leakage should be used.					
RG-1.13.C.13.	Pool Cleanup The spent fuel storage facility should be capable of maintaining safe radiation levels for personnel during anticipated operating and accident conditions. To maintain low radiation levels, a filtering system should be provided to remove radioactive materials and other contaminants from the spent fuel pool coolant. This system does not need to be safety-related, but its failure should not impair safety-related systems or cause a significant decrease in the pool coolant inventory.					
RG-1.13.C.14.	High-Burnup Fuel The mechanical properties of fuel may change with longer operating cycles. For instance, high-burnup fuel may become more brittle (i.e., possess lower ductility and fracture toughness) and, therefore, be more vulnerable to failure. In order to protect high-burnup fuel from mechanical damage, this potential vulnerability should be considered in the design of spent fuel handling and storage facilities.					
RG-1.14 (Rev. 1, August 1975)	Reactor Coolant Pump Flywheel Integrity	NA				Exclude, The HTGR design does not utilize reactor coolant pumps; there is a main circulator inside the steam generator vessel.
RG-1.15	Withdrawn (See 46 FR 37579, 07/21/1981)	NA				Exclude

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
RG-1.16	Withdrawn (See 74 FR 40244, 2/11/2009)	NA				Exclude
RG-1.17	Withdrawn, (See 56 FR 30777, 07/05/1991)	NA				Exclude
RG-1.18	Withdrawn, (See 46 FR 37579, 07/21/1981)	NA				Exclude
RG-1.19	Withdrawn, (See 46 FR 37579, 07/21/1981)	NA				Exclude
RG-1.20, (Rev. 3, March 2007)	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing In general, the NRC staff recommends that applicants use the classifications identified in Regulatory Position 1 (below) to categorize reactor internals according to design, operating parameters, and operating experience with potential prototypes. Applicants should then establish an appropriate comprehensive vibration assessment program using the guidelines specified in the succeeding regulatory positions, as they relate to the given classification(s). The comprehensive vibration assessment programs outlined in this guide are summarized in Figure 1. Note: Refer to the Regulatory Guide for detailed criteria and the figure.					
RG-1.21 (Rev. 2, June 2009)	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste ***This RG is voluminous and contains tables that cannot be formatted to fit in this table. See the full text of the RG for details. The guide describes a suitable onsite program to collect the basic meteorological data needed to determine the environmental impacts of the plant, perform consequence assessments supporting routine release and design-basis accident evaluations, and support emergency preparedness programs and other applications at power reactor sites.***					
RG-1.22 (Rev. 0, February 1972)	Periodic Testing of Protection System Actuation Functions					
RG-1.22.D.1	1. The protection system should be designed to permit periodic testing to extend to and include the actuation devices and actuated equipment.					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	a. The periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident. b. The protection system and the systems whose operation it initiates should be designed to permit testing of the actuation devices during reactor operation.					
RG-1.22.D.2	2. Acceptable methods of including the actuation devices in the periodic tests of the protection system are: a. Testing simultaneously all actuation devices and actuated equipment associated with each redundant protection system output signal; b. Testing all actuation devices and actuated equipment individually or in judiciously selected groups; c. Preventing the operation of certain actuated equipment during a test of their actuation devices; d. Providing the actuated equipment with more than one actuation device and testing individually each actuation device. Method a. set forth above is the preferable method of including the actuation devices in the periodic tests of the protection system. It shall be noted that the acceptability of each of the four above methods is conditioned by the provisions of regulatory positions 3 and 4 below.					
RG-1.22.D.3	3. Where the ability of a system to respond to a bona fide accident signal is intentionally bypassed for the purpose of performing a test during reactor operation: a. Positive means should be provided to prevent expansion of the bypass condition to redundant or diverse systems, and					

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	b. Each bypass condition should be individually and automatically indicated to the reactor operator in the main control room.					
RG-1.22.D.4	4. Where actuated equipment is not tested during reactor operation, it should be shown that: a. There is no practicable system design that would permit operation of the actuated equipment without adversely affecting the safety or operability of the plant; b. The probability that the protection system will fail to initiate the operation of the actuated equipment is, and can be maintained, acceptably low without testing the actuated equipment during reactor operation, and c. The actuated equipment can be routinely tested when the reactor is shut down.					
RG-1.23 (Rev. 1, March 2007)	Meteorological Monitoring Programs for Nuclear Power Plants ***[Note: Refer to the text of RG 1.23 for detailed criteria which are voluminous and include tables that cannot readily be formatted to include in this line item.]***					
RG-1.24 (Rev. 0, March 1972)	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	NA				Exclude, This RG is superseded by RG 1.145 for new plants.
RG-1.25 (Rev. 0, March 1972)	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors	NA				Exclude, This RG is superseded by RG 1.183 for new plants.
RG-1.26 (Rev. 4, March 2007)	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive- Waste-Containing Components of Nuclear Power Plants					
RG-1.26.C.1.	Quality Group B The Quality Group B standards given in Table 1 of this guide should be applied to water- and steam-containing pressure vessels, heat exchangers					

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RG No./Rev.	**RG Title**	**Applicable**	**Reg. or Guidance**	**Add'l Design Info**	**Add'l Reg. Needed**	**Basis/Comment**
	(other than turbines and condensers), storage tanks, piping, pumps, and valves that are either (1) part of the reactor coolant pressure boundary defined in 10 CFR 50.2 but excluded from the requirements of 10 CFR 50.55a[3] pursuant to paragraph (c)(2) of that section, or (2) not part of the reactor coolant pressure boundary but part of the following: (a) systems or portions of systems[4] important to safety that are designed for (i) emergency core cooling, (ii) postaccident containment heat removal, or (iii) postaccident fission product removal (b) systems or portions of systems[5] important to safety that are designed for (i) reactor shutdown or (ii) residual heat removal (c) those portions of the steam systems of boiling-water reactors extending from the outermost containment isolation valve up to but not including the turbine stop and bypass valves,[5] and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation; alternatively, for boiling-water reactors containing a shutoff valve (in addition to the two containment isolation valves) in the main steamline and the main feedwater line, those portions of the steam and feedwater systems extending from the outermost containment isolation valves up to and including the shutoff valve or the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation (d) those portions of the steam and feedwater systems[3] of pressurized-water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation (e) systems or portions of systems[5] that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure. [3]The regulations in 10 CFR 50.55a specify the Quality 3 Group A standards					

Table A1-12: Regulatory Guides (Division 1)						
RG No./Rev.	**RG Title**	**Applicable**	**Reg. or Guidance**	**Add'l Design Info**	**Add'l Reg. Needed**	**Basis/Comment**
	for pressure-containing components of the reactor coolant pressure boundary. [4] The system boundary includes those portions of the system necessary to accomplish the specified safety function and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required. [5] The turbine stop valve and turbine bypass valve, although not included in Quality Group B, should be subjected to a quality assurance program at a level generally equivalent to Quality Group B.					
RG-1.26.C.2.	Quality Group C (a) cooling water and auxiliary feedwater systems or portions of those systems5 important to safety that are designed for (i) emergency core cooling, (ii) postaccident containment heat removal, (ii) postaccident containment atmosphere cleanup, or (iv) residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems), although Quality Group B includes portions of those systems that are required for their safety functions and that (i) do not operate during any mode of normal reactor operation and (ii) cannot be tested adequately. (b) cooling water and seal water systems or portions of those systems5 important to safety that are designed for the functioning of components and systems important to safety, such as reactor coolant pumps, diesels, and the control room (c) systems or portions of systems5 that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure[6] (d) systems, other than radioactive waste management systems3, not covered by Regulatory Positions 2(a) through 2(c) (above) that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses [using meteorology as					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	recommended in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors" (Ref. 5), and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized-Water Reactors" (Ref. 6) that exceed 0.5 rem to the whole body or its equivalent to any part of the body; only single component failures need be assumed for those systems located in Seismic Category I structures, and no credit should be taken for automatic isolation from other components in the system or for treatment of released material, unless the isolation or treatment capability is designed to the appropriate seismic and quality group standards and can withstand loss of offsite power and a single failure of an active component [6] Components in influent lines may be classified as Quality Group D if they are capable of being isolated from the reactor coolant pressure boundary by an additional valve that has high leaktight integrity.					
RG-1.26.C.3.	Quality Group D The Quality Group D standards given in Table 1 of this guide should be applied to water and steam-containing components that are not part of the reactor coolant pressure boundary or included in Quality Groups B or C, but are part of systems or portions of systems that contain or may contain radioactive material. ***Note: See the full text of the Regulatory Guide for Table 1.***					
RG-1.27 (Rev. 2, January 1976)	Ultimate Heat Sink for Nuclear Power Plants					
RG-1.27.C.1.	The ultimate heat sink should be capable of providing sufficient cooling for at least 30 days (a) to permit simultaneous safe shutdown and cooldown of all nuclear reactor units that it serves and to maintain them in a safe shutdown condition, and (b) in the event of an accident in one unit, to limit the effects of that accident safely, to permit simultaneous and safe					

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	shutdown of the remaining units, and to maintain them in a safe shutdown condition. Procedures for ensuring a continued capability after 30 days should be available. Sufficient conservatism should be provided to ensure that a 30-day cooling supply is available and that design basis temperatures of safety-related equipment are not exceeded. For heat sinks where the supply may be limited and/or the temperature of plant intake water from the sink may eventually become critical (e.g., ponds, lakes, cooling towers, or other sinks where recirculation between plant cooling water discharge and intake can occur), transient analyses[1] of supply and/or temperature should be performed. The meteorological conditions resulting in maximum evaporation and drift loss should be the worst 30-day average combination of controlling parameters (e.g., dewpoint depression, wind speed, solar radiation). The meteorological conditions resulting in minimum water cooling should be the worst combination of controlling parameters, including diurnal variations where appropriate, for the critical time period(s) unique to the specific design of the sink. The following are acceptable methods for selecting these conditions: a. Based on regional climatological[2] information, select the most severe observation for the critical time period(s) for each controlling parameter or parameter combination, with substantiation of the conservatism of these values for site use. The individual conditions may be combined without regard to historical occurrence. b. Select the most severe combination of controlling parameters, including diurnal variations where appropriate, for the total of the critical time period(s), based on examination of regional climatological[2] measurements that are demonstrated to be representative of the site. If significantly less than 30 years of representative data are available, other historical regional data should be examined to determine controlling					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	meteorological conditions for the critical time period(s). If the examination of other historical regional data indicates that the controlling meteorological conditions did not occur within the period of record for the available representative data, then these conditions should be correlated with the available representative data and appropriate adjustments should be made for site conditions. c. Less severe meteorological conditions may be assumed when it can be demonstrated that the consequences of exceeding lesser design basis conditions for short time periods are acceptable. Information on magnitude, persistence, and frequency of occurrence of controlling meteorological parameters that exceed the design basis conditions, based on acceptable data as discussed above, should be presented. The above analysis related to the 30-day cooling supply and the excess temperature should include sufficient information to substantiate the assumptions and analytical methods used. This information should include actual performance data for a similar cooling method operating under load near the specified design conditions or justification that conservative evaporation and drift loss and heat transfer values have been used. A cooling capacity of less than 30 days may be acceptable if it can be demonstrated that replenishment or use of an alternate water supply can be effected to assure the continuous capability of the sink to perform its safety functions, taking into account the availability of replenishment equipment and limitations that may be imposed on "freedom of movement" following an accident or the occurrence of severe natural phenomena. [1] For transient analysis of small shallow cooling ponds, use may be made of the analytical techniques and computer programs contained in "Generic Emergency Cooling Pond Analysis," COO-2224-1, May 1972-October 1972, prepared for the USAEC by University of Pennsylvania, School of Engineering and Applied Science, Civil Engineering, Philadelphia, Pennsylvania 19104. For sinks other than small shallow cooling ponds,					

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	similar transient analyses should be performed to demonstrate acceptable inventory and/or maximum intake water temperature. 2 Climatological in this context pertains to a recent period of record at least 30 years in length.					
RG-1.27.C.2	2. The ultimate heat sink complex, whether composed of single or multiple water sources, should be capable of withstanding, without loss of the sink safety functions specified in regulatory position 1, the following events: a. The most severe natural phenomena expected at the site, with appropriate ambient conditions, but with no two or more such phenomena occurring simultaneously, b. The site-related events (e.g., transportation accident, river diversion) that historically have occurred or that may occur during the plant lifetime, c. Reasonably probable combinations of less severe natural phenomena and/or site-related events, d. A single failure of manmade structural features. Ultimate heat sink features, which are constructed specifically for the nuclear power plant and which are not required to be designed to withstand the Safe Shutdown Earthquake or the Probable Maximum Flood, should at least be designed and constructed to withstand the effects of the Operating Basis Earthquake (as defined in 10 CFR Part 100, Appendix A) and water flow based on severe historical events in the region.					
RG-1.27.C.3	3. The ultimate heat sink should consist of at least two sources of water, including their retaining structures, each with the capability to perform the safety functions specified in regulatory position 1, unless it can be demonstrated that there is an extremely low probability of losing the capability of a single source. For close-loop cooling systems, there should be at least two aqueducts connecting the source(s) with the intake structures of the nuclear power units and at least two aqueducts to return the cooling water to the source, unless it can be demonstrated that there is					

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	extremely low probability that a single aqueduct can functionally fail entirely as a result of natural or site-related phenomena. For once-through cooling systems, there should be at least two aqueducts connecting the source(s) with the intake structures of the nuclear power units and at least two aqueducts to discharge the cooling water well away from the nuclear power plant to ensure that there is no potential for plant flooding by the discharged cooling water, unless it can be demonstrated that there is extremely low probability that a single aqueduct can functionally fail as a result of natural or site-related phenomena. All water sources and their associated aqueducts should be highly reliable and should be separated and protected such that failure of any one will not induce failure of any other.					
RG-1.27.C.4	4. The technical specifications for the plant should include provisions for actions to be taken in the event that conditions threaten partial loss of the capability of the ultimate heat sink or the plant temporarily does not satisfy regulatory positions 1 and 3 during operation.					
RG-1.28 (Rev. 3, August 1985)	Quality Assurance Program Requirements (Design and Construction)					Exclude. Administrative.
RG-1.29 (Rev. 4, March 2007)	Seismic Design Classification					
RG-1.29.C.1.	The following SSCs of a nuclear power plant, including their foundations and supports, are designated as Seismic Category I and must be designed to withstand the effects of the SSE and remain functional. The titles and functions of these Seismic Category I SSCs for LWR designs are based on existing technology from prior applications. Certain SSCs previously considered Seismic Category I may no longer have a safety-related function requiring Seismic Category I classification, and certain passive SSCs in new LWR designs may be titled differently. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 shall apply to all activities affecting the safety-related functions of these SSCs:					

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	a. the reactor coolant pressure boundary b. the reactor core and reactor vessel internals c. systems[3] or portions thereof that are required for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident containment atmosphere cleanup (e.g., hydrogen removal system) d. systems[2] or portions thereof that are required for (1) reactor shutdown, (2) residual heat removal, or (3) cooling the spent fuel storage pool e. those portions of the steam systems of boiling-water reactors extending from the outermost containment isolation valve up to but ***not*** including the turbine stop valve, and connected piping of a nominal size of 6.35 cm (2.5 inches) or larger, up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation (the turbine stop valve should be designed to withstand the SSE and maintain its integrity) f. those portions of the steam and feedwater systems of pressurized-water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping of a nominal size of 6.35 cm (2.5 inches) or larger, up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation g. cooling water, component cooling, and auxiliary feedwater systems[2] or portions thereof, including the intake structures, that are required for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, (4) residual heat removal from the reactor, or (5) spent fuel storage pool cooling h. cooling water and seal water systems[2] or portions thereof that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps i. systems[2] or portions thereof that are required to supply fuel for emergency equipment					

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	j. all electrical and mechanical devices and circuitry between the process and the input terminals. k. systems[2] or portions thereof that are required for (1) monitoring and (2) actuating systems[4] important to safety l. the spent fuel storage pool structure, including the fuel racks m. the reactivity control systems (e.g., control rods, control rod drives, and boron injection system) n. the control room, including its associated equipment and all equipment needed to maintain the control room within safe habitability limits for personnel and safe environmental limits for vital equipment o. primary and secondary reactor containment p. systems,[2] other than radioactive waste management systems,[5] not covered by items 1.a through 1.o above that contain or may contain radioactive material and of which postulated failure would result in conservatively calculated potential offsite doses [using meteorology as recommended in the latest editions of Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors" (Ref. 6), Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors" (Ref. 7), and Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" (Ref. 3)] that are more than 0.005 Sievert (0.5 rem) to the whole body or its equivalent to any part of the body or total effective dose equivalent (TEDE), as applicable q. the Class 1E electrical systems, including the auxiliary systems for the onsite electric power supplies, that provide the emergency electric power needed for functioning of plant features included in items 1.a through 1.p above					
RG-1.29.C.2,	Those portions of SSCs of which continued function is not required but of which failure could reduce the functioning of any plant feature included in items 1.a through 1.q above to an unacceptable safety level or could result in incapacitating injury to occupants of the control room should be designed					

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	and constructed so that the SSE would not cause such failure.[6]					
RG-1.29.C.3.	At the interface between Seismic Category I and non-Seismic Category I SSCs, the Seismic Category I dynamic analysis requirements should be extended to either the first anchor point in the non-seismic system or a sufficient distance into the non-Seismic Category I system so that the Seismic Category I analysis remains valid.					
RG-1.29.C.4.	The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of those portions of SSCs covered under Regulatory Positions 2 and 3 above.					
RG-1.29.C.5.	Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants" (Ref. 8), provides guidance used to establish the design requirements for portions of fire protection SSCs to meet the requirements of GDC 2, as they relate to designing those SSCs to withstand the effects of the SSE. [4] Guide 1.151, "Instrument Sensing Lines" (Ref. 4). [5] See the latest edition of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Ref. 5). [6] Wherever practical, structures and equipment of which failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.					
RG-1.30 (Rev. 0, August 1972)	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment					Exclude, This is a COL item for the licensee to address.
	The requirements for the installation, inspection, and -testing of nuclear power plant instrumentation and electric equipment which are included in **ANSI N45.2.4-1972,** "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations" (also designated as IEEE Std 336-1971)[1] are generally acceptable and provide an adequate basis for complying with the pertinent quality					

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	assurance requirements of Appendix B to 10 CFR Part 50, subject to the following: (following 3 requirements)					
	ANSI N45.2.4-1972 should be used in conjunction with ANSI N45.2-197:1, "Quality Assurance Program Requirements for Nuclear Power Plants." (It is expected that future revisions of ANSI N45.2.4-1972 will include this provision.)					
	Section 9 of ANSI N45.2.4-1972 lists additional guides and standards made applicable by ANSI N45.2,4. The specific applicability or acceptability of these listed guides and standards has been or will be covered separately, in other safety guides or in appropriate Commission regulations.					
	Although subdivision 1.1 of ANSI N45.2.4-1972 states that the requirements promulgated apply during the construction phase of a nuclear power plant, these requirements are also to be considered applicable for the installation, inspection, and testing of instrumentation and electric equipment during the operation phase of a nuclear power plant. [1]Copies may be obtained from either the Institute of Electrical and Electronics Engineers or the American Society of Mechanical Engineers, United Engineering Center, 345 E. 47th Street, New York, N.Y. 10017					
RG-1.31 (Rev. 3, April 1978)	Control of Ferrite Content in Stainless Steel Weld Metal					
1.	Verification of Delta Ferrite Content of Filler Materials Prior to production usage, the delta ferrite content of test weld deposits from each lot and each heat of weld filler metal procured for the welding of austenitic stainless steel core support structures and Class 1 and 2 components should be verified for each process to be used in production. It is not necessary to make delta ferrite determinations for SFA-5.4 type 16-8-2 weld metal or for filler metal used for weld metal cladding. Delta ferrite determinations for consumable inserts, electrodes, rod or wire filler metal					

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	used with the gas tungsten arc welding process, and deposits made with the plasma arc welding process may be predicted from their chemical composition using an applicable constitutional diagram to demonstrate compliance. Delta ferrite verification should be made for all other processes by tests using magnetic measuring devices on undiluted weld deposits. For submerged arc welding processes, the verification tests for each wire and flux combination may be made on a production weld or simulated production weld. All other delta ferrite weld filler verification tests should be made on weld pads that contain undiluted layers of weld metal.					
2.	Ferrite Measurement Appendix A to this guide contains extracts from a future edition of the American Welding Society's AWS A5.4, "Specification for Corrosion-Resisting Chromium and Chromium-Nickel Steel Covered Welding Electrodes,"* which describes a procedure for pad preparation and ferrite measurement. The NRC staff considers this procedure acceptable for use with covered electrodes.					
3.	Instrumentation The weld pad should be examined for ferrite content by a magnetic measuring instrument which has been calibrated against a Magnegage in accordance with American Welding Society Specification AWS A4.2-74, "Procedures for Calibrating Magnetic Instruments to Measure the Delta Ferrite Content of Austenitic Stainless Steel Weld Metal."'** The Magnegage should have been previously calibrated in accordance with AWS A4.2-74 using primary standards as defined therein.					
4.	Acceptability of Test Results Weld pad test results showing an average Ferrite Number from 5 to 20 indicate that the filler metal is acceptable for production welding of Class 1 and 2 austenitic stainless steel components and core support structures.					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	The upper limit of 20 may be waived for (a) welds that do not receive post weld stress relief heat treatment or welds for which such post weld stress relief treatment is conducted at temperatures less than 900'F, (b) welds that are given a solution annealing heat treatment, and (c) welds that employ consumable inserts.					
5.	Quality Assurance The applicable provisions of 10 CFR Part 50, Appendix B, should be used in verifying compliance with requirements for delta ferrite as described herein. * This specification has been recommended by the Subcommittee on Welding of Stainless Steels of the High Alloys Committee of the Welding Research Council and has been approved by the American Welding Society (AWS). It is expected to be published as AWS A5.4-78.					
RG-1.32 (Draft Rev. 3, March 2004)	Criteria for Power Systems for Nuclear Power Plants Conformance with the requirements of IEEE Std. 308-2001, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," is acceptable to the NRC staff for satisfying the NRC's regulations with respect to the design, operation, and testing of safety-related power systems for nuclear power plants, except for sharing of dc power systems at multi-unit nuclear power plants, as described in Revision 1 of Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants."					
RG-1.33 (Rev. 2, February 1978)	Quality Assurance Program Requirements (Operation)	NA				Exclude. This is a COL item for the licensee to address.
RG-1.34 (Rev.	Control of Electroslag Weld Properties					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
0, December 1972)	Electroslag welds for core support structures should comply with the fabrication requirements specified for Section III[2] Class 1 components. Electroslag weld fabrication for core support structures and Class 1 and 2 vessels and components should comply with Section **III** and Section IX[3] supplemented by the following:					
RG-1.34.C.1.	The procedure qualification[3] for low-alloy steel and stainless steel welding should require that: a. Process variables such as slag pool depth, electrode feed rate and oscillation, current, voltage, and slag conductivity be selected to produce a solidification pattern (dendritic grain pattern) with a joining angle of less than 90 degrees in the weld center; b. A macro-etch test be performed in the longitudinal weld direction of the center plane across the weld from base metal to base metal as shown in Figure A of this guide. The test should verify that the desired solidification pattern resulting from regulatory position l.a. above has been obtained and that the weld is free of unacceptable fissures or cracks; and c. Impact testing be specified for Class 2 low-alloy steel vessels in accordance with paragraph NC-2310 of Section **III**[2]					
RG-1.34.C.2.	The results of the tests required by regulatory position 1. above should be included in the certified qualification test report.					
RG-1.34.C.3.	For longitudinal production welds of low-alloy steel vessels, material containing base metal and weld metal taken from weld prolongations should be tested as follows: a. Tensile and impact tests similar to those required for the base metal by paragraph NB-32 11(d) of Section III should be made to determine the mechanical properties of the quenched and tempered weld metal; b. To verify that the specified weld solidification pattern has been obtained and that the weld center is sound, one of the following methods should be used: (1) A macro-etch test similar to requirement of regulatory position 1.b.					

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	above, or (2) Impact testing with the specimen notch located at the weld center as shown in Figure A of this guide. c. The tests specified in regulatory positions 3.a. and 3.b. above should be applied to: (1) Each of the welds for Class 1 vessels, (2) One weld per shell course for Class 2 vessels.					
RG-1.34.C.4.	For production welds for austenitic stainless steel core support structures and fluid system components, the production welding should be monitored to verify compliance with the limits for the process variables specified in the procedure qualification.					
RG-1.34.C.5.	In the event that properties obtained from tests identified in regulatory positions 3. and 4. above are not acceptable, additional procedure qualifications should be performed in accordance with regulatory position 1. above. Note: Refer to the Regulatory Guide for the figures. 2ASME B&PVC, Section III and Summer 1972 Addenda to Section III. [3]ASME B&PVC, Section IX and Code Case 1355-3.					
RG-1.35 (Rev. 3, July 1990)	In-service Inspection of Ungrouted Tendons in Pre-stressed Concrete Containments	NA				This regulatory guide is not applicable to the HTGR reactor since there are no tendons in the containment or reactor building structures.
RG-1.35.1 (Rev. 0, July 1990	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments (Rev. 0, July 1990)	NA				Exclude, This regulatory guide is not applicable to the HTGR reactor since there is no prestressing associated with the containment or reactor building structures.

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
RG-1.36 (Rev. 0, February 1973)	Nonmetallic Thermal Insulation for Austenitic Stainless Steel The levels of leachable contaminants in nonmetallic insulation materials[6] that come in contact with austenitic stainless steels of the American Iron & Steel Institute (AISI) Type 3XX series used in fluid systems important to safety should be carefully controlled so that stress-corrosion cracking is not promoted. In particular, the 1-table chlorides and fluorides should be held to the lowest practicable levels. Insulation for the above application should meet the following conditions:					
RG-1.36.C.1.	All insulating materials should be manufactured, processed, packaged, shipped, stored, and installed in a manner that will limit, to the maximum extent practical, chloride and fluoride contamination from external sources.					
RG-1.36.C.2.	Qualification Test: Each type[3] of insulating material should be qualified by the manufacturer or supplier for use by: a. An appropriate test to reasonably assure that the insulation formulation does not induce stress corrosion. Two acceptable tests are: (1) ASTM C692-71, "Standard Method for Evaluating Stress Corrosion Effect of Wicking-Type Thermal Insulations on Stainless Steel" (Dana Test). The material should be rejected if more than one of five specimens crack; and (2) RDT M12-1T,[7] "Test Requirements for Thermal Insulating Materials for Use on Austenitic Stainless Steel," Section 5, (Knolls Atomic Power Laboratory (KAPL) Test). The material should be rejected if more than one of four specimens crack. b. Chemical analysis to determine the ion concentrations of leachable chloride, fluoride, sodium, and silicate. Insulating material that is not demonstrated by the analysis to be within the acceptable region of Figure 1 of this guide should be rejected. This analysis should also be used as a comparison basis for the production test specified in C.3. below.					
RG-1.36.C.3.	Production Test: A representative sample[8] from each production lot[5] of					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	insulation material to be used adjacent to, or in contact with, austenitic stainless steels used in fluid systems important to safety should be chemically analyzed to determine leachable chloride, fluoride, sodium, and silicate ion concentrations as in C.2.a. above. The lot should be accepted only if: a. The analysis shows the material to be within the acceptable region of Figure 1; and b. Neither the sum of chloride plus fluoride ion concentrations nor the sum of sodium plus silicate ion concentrations determined by this analysis deviates by more than 50 percent from the values determined on the sample used to qualify the insulation in C.2. above.					
RG-1.36.C.4.	Requalification: When a change is made in the type, nature, or quality of the ingredients, the formulation, or the manufacturing process, the insulation material should be requalified by repeating the tests described in C.2. above. Note: Refer the Regulatory Guide for the figure. [3]Type means material of similar composition, form, and class and of consistent quality, formulation, and manufacturing process. [5]**A** lot is defined as the thermal insulation material of the same composition, form, type, grade, and class produced at one plant under the same conditions over a limited time span and designated by the producer as a production lot. [6]Thermal insulating materials include block insulation, pipe insulation, board and blanket and the cements and adhesives employed in their application. [7]Copies may be obtained from RDT Standards Office, Oak Ridge National Laboratory, Building 1000, P.O. Box X, Oak Ridge, Tennessee 37830. [8] **A** representative sample should be fully representative of the cross					

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	section of the material; that is, it should include proportionate amounts of					
RG-1.37 (Draft Rev. 1, March 2007)	Quality Assurance Requirements for cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants The NRC staff finds that the provisions and recommendations included in ASME NQA-1-1994, Part II, Subpart 2.1 are generally acceptable for onsite cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. These provisions and recommendations provide an adequate basis for complying with the pertinent QA requirements of Appendix B to 10 CFR Part 50, subject to the following regulatory positions:					
RG-1.37.C.1.	Referenced Documents Section 7 of the Introduction to ASME NQA-1-1994, Part II, which is applicable to Subpart 2.1, states that the codes, standards, and specifications referenced in this Part may be identified with the applicable date or citation at the point of reference or in Table entitled "Codes, Standards, and Specifications Referenced in Text." The specific applicability or acceptability of these listed documents has been (or will be) covered separately in other regulatory guides or in Commission regulations, as appropriate.					
RG-1.37.C.2.	**Water Quality** Section 3.4.1 of ASME NQA-1-1994, Part II, Subpart 2.1 states that "the water quality for mixing cleaning solutions, rinsing, and flushing shall be specified by the organization responsible for cleaning unless otherwise stipulated in procurement documents or approved procedures." The water quality for final flushes of fluid systems and associated components should be at least equivalent to the quality of the operating system water.					

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RG-1.37.C.3.	**Precautions** Sections 8.2.2 and 8.2.3 of ASME NQA-1-1994, Part II, Subpart 2.1 provide precautions related to the use of alkaline cleaning solutions and chelating agents, respectively, by referencing non-mandatory Appendix 2.1 to ASME NQA-1-1994, Part III, Subpart 3.2. These precautions should be followed. In addition, a suitable chloride stress-cracking inhibitor should be added to the fresh water used to flush systems containing austenitic stainless steels.					
RG-1.38 (Rev. 2, February 2010)	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants					
RG-1.38.C.1.	The requirements for the packaging, shipping, receiving, storage, and handling of items for watercooled nuclear power plants that are included in ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants During the Construction Phase,"2 are acceptable to the NRC staff and, when supplemented by the guidelines identified in Regulatory Position 2, provide an adequate basis for complying with the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50, subject to the following: a. Subdivision 1.5 of ANSI N45.2.2-1972 states that other documents required to be included as a part of this standard are either identified at the point of reference or described in Section 9 of the standard. The specific acceptability of these listed documents has been or will be covered separately in other regulatory guides or in Commission regulations where appropriate. b. Subdivision 7.3.4 of ANSI N45.2.2-1972 delineates requirements for re-rating hoisting equipment for special lifts. This subdivision requires that re-rated equipment be given a dynamic load test over the full range of the lift, using a test weight at least equal to the lift weight. In lieu of this requirement, the test weight used in temporarily re-rating hoisting equipment for special lifts in accordance with the provisions of subdivision					

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	7.3.4 should be at least equal to 110% of the lift weight. c. Subdivision A.3.6.3(l) of ANSI N45.2.2-1972 permits desiccants and desiccant bag materials containing not more than 0.25% halogens to be used with austenitic stainless steels. In lieu of this requirement, desiccants and the materials for the desiccant bags, when used with austenitic stainless steel or nickel alloy materials, should not be compounded from or treated with chemical compounds containing elements in such quantities that harmful concentrations could be leached or be released by breakdown of the compounds under expected environmental conditions (e.g., by radiation). Examples of such compounds are those containing fluorides, chlorides, sulfur, lead, zinc, copper, and mercury. d. Although ANSI N45.2.2-1972 is entitled "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants During the Construction Phase," the requirements included in the standard are considered to be applicable during the operation phase and should be used, where applicable, consistent with the recommendations of this regulatory guide. e. Notwithstanding the provisions of subdivision 1.2 of ANSI N45.2.2-1972 with respect to the applicability of this standard and the definition of carrier contained in subdivision 1.4 of ANSI N45.2.21972, nothing contained in Section 4, "Shipping," of ANSI N45.2.2-1972 should be deemed to require a common or contract carrier transporting or shipping byproduct, source, or special nuclear material in the ordinary course of its business to comply with the provisions set forth in this section of the standard. In this situation these carriers are exempt from NRC regulation under the provisions of 10 CFR §§ 30.13, 40.12, and 70.12. Therefore, the provisions of Section 4 of ANSI N45.2.2-1972 apply only to the extent that they affect the activities of an NRC licensee (e.g., requirements related to shipping contained in 10 CFR Part 71) or a private carrier subject to NRC regulations.					
RG-1.38.C.2.	The guidelines (indicated by the verb "should") of ANSI N45.2.2-1972 contained in the following section are considered to have sufficient safety					

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	importance to be treated the same as the requirements of the standard, subject to any exceptions noted: a. Section 4.2.3-The guidelines concerning special shipments. b. Section 4.3.6-The guideline that addresses written instructions on stacking. c. Subdivision A.3.5.2(l)(a)-This guideline states that the halogen and sulfur content of tapes should not be in excess of 0.10% by weight when used in contact with austenitic stainless steel and nickel alloy surfaces. In lieu of this guideline, tapes, when used with austenitic stainless steel or nickel alloy materials, should not be compounded from or treated with chemical compounds containing elements in such quantities that harmful concentrations could be leached or be released by breakdown of the compound under expected environmental conditions (e.g., by radiation). Examples of such compounds are those containing fluorides, chlorides, sulfur, lead,					
RG-1.39 (Rev. 2, September 1977)	Housekeeping Requirements for Water-Cooled Nuclear Power Plants The requirements for the control of work activities, conditions, and environments at water-cooled nuclear power plant sites that are included in ANSI Standard N45.2.3-1973, "Housekeeping During the Construction Phase of Nuclear Power Plants,"[2] provide a method acceptable to the NRC staff for complying with the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50, subject to the following:					
RG-1.39.C.1.	Subdivision 1.5 of ANSI N45.2.3-1973 states that other documents that are required to be included as. **a** part or this standard are either identified at the point of reference or identified in Paragraph 5 of the standard. The specific acceptability of these listed documents has been or will **be** covered separately in other regulatory guides or in					

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	Commission regulations, where appropriate.					
RG-1.39.C.2.	Subdivision 3.2.3 of ANSI N45.2.3-1973 includes general guidelines and requirements for fire protection and prevention. The requirements and guidelines of Subdivision 3.2.3 are not considered a part of this regulatory guide, since this subject is addressed separately in more detail in other NRC documents. Thus, a commitment to follow this regulatory guide does not imply a commitment to follow the guidelines and requirements of Subdivision 3.2.3.					
RG-1.39.C.3.	Although ANSI N45.2.3-1973 is entitled "-Housekeeping During the Construction Phase of Nuclear Power Plants," the requirements included in the standard, subject to the provision of Regulatory Position C.2, are considered to be applicable for housekeeping activities occurring during the operations phase that are comparable to those occurring during the construction phase.					
RG-1.40 (Rev. 1, March 1973)	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants The NRC staff considers conformance with IEEE Standard 334-2006 an acceptable method for use in satisfying the Commission's regulations with respect to qualification of continuous duty safety related motors. IEEE Standard 334-2006 references several industry codes and standards. If the NRC's regulations separately incorporate a referenced standard, licensees and applicants must comply with the standard as set forth in the regulations. By contrast, if the NRC staff has endorsed a referenced standard in a regulatory guide, that standard constitutes an acceptable method of meeting a regulatory requirement as described in the regulatory guide.					
RG-1.41 (Rev. 0, March 1973)	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments As part of the initial preoperational testing program, and also after major modifications or repairs to a facility, those on-site electric power systems					

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	designed in accordance with Regulatory Guides 1.6 and 1.32 (Safety Guides 6 and 32) should be tested as follows to verify the existence of independence among redundant on-site in power sources and their load groups.					
1.	The plant electric power distribution system, not necessarily including the switchyard and the startup of the and auxiliary transformers, should be isolated from the off-site transmission network. Preferably, this isolation should be effected by direct actuation of the undervoltage-sensing relays within the on-site system.					
2.	Under the conditions of C.1. above, the on-site electric power system should be functionally tested successively in the various possible combinations of power sources and load groups with all **d-c** and on-site a-c power sources for one load group at a time completely disconnected. Each test should include injection of simulated accident signals, startup of the on-site power source(s) and load group(s) under test, sequencing of loads, and the functional performance of the loads. Each test should be of sufficient duration to achieve stable operating conditions and thus permit the onset and detection of adverse conditions which could result from improper assignment of loads, e.g., the lack of forced cooling of a vital device.					
3.	During each test, the d-c and on-site a-c buses and related loads not under test should be monitored to verify absence of voltage at these buses and loads.					
RG-1.42	Withdrawn (See 41 FR 11891, 03/22/1981)	NA				Exclude
RG-1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components Controls should be exercised to limit the occurrence of underclad cracking in low-alloy steel safety-related components clad with stainless steel. Welding processes that induce underclad cracking by generating excessive heating and promoting grain coarsening in the base metal should not be used, for cladding any grade of material that has a known susceptibility to underclad cracking. Welding procedures used for cladding these grades of					

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	material should be qualified for use to demonstrate that underclad cracking is not induced. These controls need not be applied to the cladding of materials demonstrated to be resistant to underclad cracking, such as SA-533 Grade B Class **I** plate made to fine-grain practice and heat-treated to develop a fine-grained structure. Weld cladding practices used in the fabrication of low-alloy steel safety-related components should be conducted in accordance with the following guidelines:					
1.	For weld cladding of SA-508 Class 2 forgings made to coarse-grain practice and plate material of similar composition: a. "High-heat-input" welding processes that induce underclad cracking such as the submerged-arc wide-strip welding process and the submerged-arc 6-wire process should not be used. b. Weld cladding procedures should be qualified for use in accordance with regulatory position C.2. below.					
2.	The weld cladding procedure described in regulatory position C.1. should be qualified for use by a performance test to demonstrate that it does not induce excessive underclad cracking. The test should include the following: a. Base material for the test should be of the same grade as that to be used in production. A minimum of three representative heats of material should be tested. Where less than three heats of material are used in production, these heats may be tested in lieu of the three representative heats. **b.** The qualification block from which test specimens are to be taken should be of sufficient size and thickness to develop thermal restraint conditions typical of those developed in production welding. c. The qualification block from which test specimens are to be taken should be suitably post-weld heat-treated at temperatures and times at least as great as those encountered in production heat treatment prior to removal of specimens. d. A minimum of two weld clad-overlap areas per test specimen should be evaluated.					
	The following indications on any one-inch length of evaluation test					

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	specimen should be the basis for rejection of the welding procedure: **(1** any fissures greater than 1/32 inch in length or 0.010 inch in depth. (2) more than three fissures 0.005 inch to 0.010 inch in depth.					
3.	Production welding should he monitored verify compliance with the limitations on essential variables established by the procedure qualification. In the event that the production welding procedure does not conform to these limitations, an examination for cracking should be performed on the production part from which a section of cladding has been removed or the cladding procedure should be requalified in accordance with regulatory position C.2. above.					
RG-1.44 (Rev. 0, May 1973)	Control of the Use of Sensitized Stainless Steel Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the following:					
1.45 (Rev. 0, May 1973)	Reactor Coolant Pressure Boundary Leakage Detection Systems Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the following:					
1.	Material should be suitably cleaned and suitably protected against contaminants capable of causing stress corrosion cracking during fabrication, shipment, storage, construction, testing, and operation of components and systems.					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
2.	Material from which components and systems are to be fabricated should be solution heat treated[3] to produce a non-sensitized condition in the material.					
3.	Non-sensitization of the material[4] should be verified using ASTM A 262-70, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steel," Practices A or E, or another method that can be demonstrated to show non-sensitization in austenitic stainless steel. Test Specimens should be selected from material subjected to each different heat treatment practice and from each heat.					
4.	Material subjected to sensitizing temperature in the range of 800 to 1500°F, subsequent to solution heat treating in accordance with subparagraph C.2. above and testing in accordance with subparagraph C.3. above should be L Grade material; that is, it should not have a carbon content greater than 0.03 percent. Exceptions are: (a) Material exposed to reactor coolant which has a controlled concentration of less than 0.10 ppm dissolved oxygen at all temperatures above 200°F during normal operation; or (b) Material in the form of castings or weld metal with a ferrite content of at least 5 percent, or (c) Piping in the solution annealed condition whose exposure to temperatures in the range of 800 to 1500°F has been limited to welding operations, provided it is of sufficiently small diameter so that in the event of a credible postulated failure of the piping during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.					
5.	Material subjected to sensitizing temperatures in the range of 800 to 1500°F during heat treating or processing other than welding, subsequent to solution heat treating in accordance with subparagraph C.2. above, and testing in accordance with subparagraph C.3. above, should be retested in accordance with subparagraph C.3. above, to demonstrate that is not					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	susceptible to intergranular attack, except that retest is not required for: (a) Cast metal or weld metal with a ferrite content of 5 percent or more; or (b) Material with a carbon content of 0.03 percent or less that is subjected to temperatures in the range of 800 to 1500°F for less than one hour; or (c) Material exposed to special processing, provided the processing is properly controlled to develop a uniform product and provided that adequate documentation exists of service experience and/or test data to demonstrate that the processing will not result in increased susceptibility to intergranular stress corrosion. Specimens for the above retest should be taken from each heat of material and should be subjected to a thermal treatment that is representative of the anticipated thermal conditions that the production material will undergo.					
6.	Welding practices and, if necessary, material composition should be controlled to avoid excessive sensitization of base metal heat-affected zones of weldments. An intergranular corrosion test, such as specified in subparagraph C.3. above, should be performed for each welding procedure to be used for welding material having a carbon content of greater than 0.03 percent. [3] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Three Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at http://www.asme.org/Codes/Publications. [4] All information notices (INs) listed herein were published by the U.S. Nuclear Regulatory Commission and are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; and email PDR@nrc.gov.					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
RG-1.45 (Rev. 0, May 1973)	Reactor Coolant Pressure Boundary Leakage Detection Systems					
1.	**General Positions** 1.1. The source and location of reactor coolant leakage should be identifiable to the extent practical, and the plant should measure the leakage rate. 1.2. The plant should collect or otherwise isolate leakage to the primary reactor containment from identified sources so that the following criteria are fulfilled: (i) Flow rates from identified sources are monitored separately from the flow rates from unidentified sources. (ii) The plant can establish and monitor the total flow rate. 1.3. The plant should monitor critical components of the RCPB for leaks. 1.4. The plant should monitor intersystem leakage for systems connected to the RCPB. 1.5. The capabilities of the leakage monitoring systems should be known. In addition, the capabilities should ensure effective management of leakage.					
2.	**Leakage-Monitoring-Related Positions** 2.1. Plant procedures should include the collection of leakage to the primary reactor containment from unidentified sources so that the total flow rate can be detected, monitored, and quantified for flow rates greater than or equal to 0.05 gal/min (0.19 L/min). 2.2. The plant should use leakage detection systems with a response time (not including the transport delay time) of no greater than 1 hour for a leakage rate of 1 gal/min (3.8 L/min). 2.3. Plant technical specifications should identify at least two independent and diverse instruments and/or methods that have the detection and monitoring capabilities detailed above. The methods to consider for incorporation in the technical specifications include, but are not limited to, the following:					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	(i) monitoring sump level or flow, (ii) monitoring airborne particulate radioactivity, and (iii) monitoring condensate flow rate from air coolers. In addition to the monitoring systems detailed in the technical specifications, the plant should use other systems to detect and monitor for leakage, even if it does not have the capabilities specified in Regulatory Position 2.2. These supplemental instruments/methods may include, but are not limited to, the following: (a) monitoring airborne gaseous radioactivity, (b) monitoring the humidity of the containment, (c) monitoring the temperature of the containment, (d) monitoring the pressure of the containment, (e) monitoring acoustic emission, and (f) conducting video surveillance. 2.4. At least one of the leakage monitoring systems required by the plant technical specifications (as described in Regulatory Position 2.3 above) should be capable of performing its function(s) following any seismic event that does not require plant shutdown. 2.5 The leakage monitoring systems, including those with location detection capability, should have provisions to permit calibration and testing during plant operation to ensure functionality or operability, as appropriate.					
3.	3.1. The plant should periodically analyze the trend in the unidentified and identified leakage rates. When the leakage rate increases noticeably from the baseline leakage rate, the plant should evaluate the safety significance of the leak. The plant should determine the rate of increase in the leakage to verify that plant actions can be taken before the plant exceeds technical specification limits. 3.2 The plant should establish procedures for responding to leakage. These procedures should address the following considerations and should ensure that no adverse safety consequences result from the leakage:					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	(i) Plant procedures should specify operator actions in response to leakage rates less than the limits set forth in the plant technical specifications. The procedures should include actions for confirming the existence of a leak, identifying its source, increasing the frequency of monitoring, verifying the leakage rate (through a water inventory balance), responding to trends in the leakage rate, performing a walk down outside containment, planning a containment entry, adjusting alarm setpoints, limiting the amount of time that operation is permitted when the sources of the leakage are unknown, and determining the safety significance of the leakage. (ii) Plant procedures should specify the amount of time the leakage detection and monitoring instruments (other than those required by technical specifications) may be out of service to ensure that the leakage rate is effectively monitored during all phases of plant operation (i.e., hot shutdown, hot standby, startup, transients, and power operation). 3.3 The plant should provide output and alarms from leakage monitoring systems in the main control room. Procedures for converting the instrument output to a leakage rate should be readily available to the operators. (Alternatively, these procedures could be part of a computer program so that the operators have a real-time indication of the leakage rate as determined from the output of these monitors.) Periodic calibration and testing of leakage monitoring systems should take place. The alarm should provide operators an early warning signal so that they can take corrective actions, as discussed in Regulatory Position 3.2 above. 3.4. During maintenance and refueling outages, the plant should take actions to identify the source of any unidentified leakage that was detected during plant operation. In addition, corrective action should take place to eliminate the condition resulting in the leakage.					
4.	**Technical Specification Position** 4.1. Plant technical specifications should include the limiting conditions for identified, unidentified, RCPB, and intersystem leakage, and they should					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	address the availability of various types of instruments to ensure adequate coverage during all phases of plant operation (not including cold shutdown and refueling modes of operation).					
RG-1.46	Withdrawn (See 50 FR 9732, 03/11/1985)	NA				Exclude
RG-1.47 (Rev. 0, May 1973)	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems The following regulatory positions provide the supplemental guidance for implementing IEEE Std 603-1991 to satisfy the NRC regulatory requirements with respect to the bypassed and inoperable status indication for nuclear power plant safety systems:					
1.	Administrative procedures should be supplemented by an indication system that automatically indicates, for each affected safety system or subsystem, the bypass or deliberately induced inoperability of a safety function and the systems actuated or controlled by the safety function. Provisions should also be made to allow the operations staff to confirm that a bypassed safety function has been properly returned to service.					
2.	The indicating system of Position 1 above should also be activated automatically by the bypassing or the deliberately induced inoperability of any auxiliary or supporting system that effectively bypasses or renders inoperable a safety function and the systems actuated or controlled by the safety function.					
3.	Annunciating functions for system failure and automatic actions based on the self-test or self diagnostic capabilities of digital computer-based I&C safety systems should be consistent with Positions 1 and 2 above.					
4.	The bypass and inoperable status indication system should include a capability for ensuring its operable status during normal plant operation to the extent that the indicating and annunciating functions can be verified.					
5.	Bypass and inoperable status indicators should be arranged such that the operator can determine whether continued reactor operation is permissible. The control room of all affected units should receive an indication of the bypass of shared system safety functions.					
6.	Bypass and inoperable status indicators should be designed and installed in					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	a manner that precludes the possibility of adverse effects on plant safety systems. The indication system should not be used to perform functions that are essential to safety, unless it is designed in conformance with criteria established for safety systems.					
RG-1.48	Withdrawn (See 50 FR 9732, 03/11/1981)	NA				Exclude.
RG-1.49	Withdrawn (See 72 FR 36737, 07/05/2007)	NA				Exclude.
RG-1.50 (Rev. 0, May 1973)	Control of Preheat Temperature for Welding of Low- Alloy Steel Weld fabrication[2] for low-alloy steel components should comply with the fabrication requirements specified in Section III and Section IX of the ASME B&PV Code supplemented by the following:					
Where are numbers ??	The procedure qualification should require that: a. A minimum preheat and a maximum interpass temperature be specified. b. The welding procedure be qualified at the minimum preheat temperature.					
	For production welds, the preheat temperature should be maintained until a post-weld heat treatment has been performed.					
	Production welding should be monitored to verify that the limits on preheat and interpass temperatures are maintained.					
	In the event that regulatory positions C.1., C.2. and C.3. above are not met, the weld is subject to rejection. However, the soundness of the weld may be verified by an acceptable examination procedure. [2]Does not apply to weld repairs after initial fabrication.					
RG-1.51	Withdrawn (See 40 FR 30510, 07/21/1975)	NA				Exclude.
RG-1.52 (Draft Rev. 3, June 2001)	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light- Water-Cooled Nuclear Power Plants					Note 1

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	Note: Refer to the regulatory guide for detailed criteria.					
RG-1.53 (Draft Rev. 2, November 2003)	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems Conformance with the requirements of IEEE Std 379-2000, "Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems," provides methods acceptable to the NRC staff for satisfying the NRC's regulations with respect to the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power plant safety systems. Section 2 of IEEE Std 379-2000 references several industry codes and standards. If a referenced standard has been separately incorporated into the NRC's regulations, licensees and applicants must comply with the standard as set forth in the regulation. If the referenced standard has been endorsed by the NRC staff in a regulatory guide, the standard constitutes an acceptable method of meeting a regulatory requirement as described in the regulatory guide. If a referenced standard has been neither incorporated into the NRC's regulations nor endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced standard, if appropriately justified, consistent with regulatory practice.					
RG-1.54 (Draft Rev. 1, July 2000)	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants					
1.	**Guidance in ASTM Standards** ASTM D 5144-00 (Ref. 5) and the other ASTM standards discussed below provide guidance on practices and programs that are acceptable to the NRC staff for the selection, application, qualification, inspection, and maintenance of protective coatings applied in nuclear power plants. However, the ASTM Committee has revised definitions of Service Level I, II,					

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RG No./Rev.	**RG Title**	**Applicable**	**Reg. or Guidance**	**Add'l Design Info**	**Add'l Reg. Needed**	**Basis/Comment**
	and III coatings locations to include both safety-related and non-safety-related regions as set forth below. The quality assurance provisions and guidance contained in the standards in this Regulatory Position are generally acceptable and provide methods acceptable to the NRC staff for complying with the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 subject to the following two exceptions. (1) When using this regulatory guide, NRC licensees should meet the quality assurance provisions and guidance contained in the standards in this regulatory guide and must also meet the commitments and provisions contained in their Quality Assurance Program description. (2) Service Level I, II, and III coatings are defined as: Service Level I coatings are used in areas inside the reactor containment where the coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown. Service Level II coatings are used in areas where coatings failure could impair, but not prevent, normal operating performance. The functions of Service Level 2 coatings are to provide corrosion protection and decontaminability in those areas outside the reactor containment that are subject to radiation exposure and radionuclide contamination. Service Level II coatings are not safety-related. Service Level III coatings are used in areas outside the reactor containment where failure could adversely affect the safety function of a safety-related structure, system, or component.					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	ASTM D 5144-00 (Ref. 5) addresses by reference the preparation of test specimens, radiation tolerance testing, decontaminability of coatings, physical properties, chemical resistance tests, fire evaluation tests, DBA testing, surface preparation, coating application and inspection, and thermal conductivity testing. Therefore, ASTM D 5144-00 can be viewed as a top-level ASTM standard that incorporates by reference other key ASTM standards as shown in Figure 1.					
2.	**Quality Assurance** ASTM D 3843-00 (Ref. 3) provides quality assurance practices that are acceptable to the NRC staff and are applicable to safety-related protective coating work in coating Service Level I areas of nuclear facilities. Applicable portions of practices described may be used as the basis for limited quality assurance for protective coating work in coating Service Level II areas of nuclear facilities. ASTM D 5139-96 (Ref. 6) provides guidance that is acceptable to the NRC staff on the size, composition, and surface preparation for test samples of protective coatings for use in qualification testing of coatings to be used in nuclear power plants as described in ASTM D 3911-95 and D-4082-95 (Refs. 4 and 7). ASTM D 3911-95 (Ref. 4) provides guidance that is acceptable to the NRC staff on procedures for evaluating protective coating systems test specimens under simulated DBA conditions. ASTM D 911-95 also provides guidance on conditions and test apparatus for temperature-pressure testing, conditions for radiation testing, and procedures for preparing, examining, and evaluating samples. ASTM D 4082-95 (Ref. 7) provides a standard test method that is acceptable to the NRC staff for evaluating the effects of gamma radiation on the lifetime radiation tolerance of Service Level I and II coatings.					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	ASTM D 4537-96 (Ref. 8) provides guidance that is acceptable to the NRC staff on the qualification and certification of personnel who inspect protective coatings in nuclear facilities. This standard provides guidance on inspection of the education, training, experience, qualifications, and certification of Level I, II, and III coatings inspectors.					
3.	**Training and Qualification of Protective Coatings, Inspectors, and Coatings Applicators** ASTM D 5498-94 (Ref. 9) provides guidance acceptable to the NRC staff for persons responsible for developing a training program for the indoctrination and training of personnel for inspecting coating work in nuclear facilities and also recommends areas of proficiency that are embodied in the ASTM standards shown in Table 1 in ASTM D 5498-94. ASTM D 4227-95 (Ref. 10) provides guidance acceptable to the NRC staff for the qualification of coatings applicators to verify that they are proficient and able to attain the quality required for applying specified coatings to concrete surfaces, including those in a nuclear facility. ASTM D 4228-95 (Ref. 11) provides guidance acceptable to the NRC staff for the qualification of coatings applicators to verify that they are proficient and able to attain the quality required for applying specified coatings to steel surfaces, including those in a nuclear facility. ASTM D 4286-96 (Ref. 12) provides criteria and methods that are acceptable to the NRC staff to assist utility owners, architects, engineers, and contractors in determining the overall qualifications of a coatings contractor to execute coating work for the primary containment and= other safety-related facilities of nuclear power plants. The criteria and requirements for contractors address the contractor's capability to execute nuclear coating work.					
4.	**Maintenance of Coatings** ASTM D 5163-96 (Ref. 13) provides guidelines that are acceptable to the					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	NRC staff for establishing an in-service coatings monitoring program for Service Level I coating systems in operating nuclear power plants and for Service Level II and other areas outside containment (as applicable). ASTM D 4541-95 (Ref. 14) provides guidance acceptable to the NRC staff for a procedure for evaluating the pull-off strength of coatings using fixed-alignment adhesion testers. ASTM D 3359-95, Revision A (Ref. 15), provides guidance that is acceptable to the NRC staff on test methods for measuring adhesion using tape tests. 1.54-6 ASTM D 3912-95 (Ref. 16), provides guidance that is acceptable to the NRC staff on evaluation of the chemical resistance of coatings used in light-water nuclear power plants. ASTM D 5962-96 (Ref. 17) provides guidance that is acceptable to the NRC staff on maintaining unqualified coatings (paints) within Level I areas of a nuclear power facility.					
5.	**ASTM Standard Terminology** ASTM D 4538-95 (Ref. 18) defines standard terms related to protective coating and lining work for power generation facilities that are acceptable to the NRC staff and that are also applicable to protective coatings employed in nuclear power plants.					
6.	**Additional Information** Additional information on the selection, application, inspection, and maintenance of nuclear plant safety-related protective coatings is provided in EPRI Report TR-109937 (Ref. 19), which provides a detailed discussion of important considerations related to protective coatings and can be used to supplement the ASTM Standards guidelines as deemed necessary.					
RG-1.55	Withdrawn (See 46 FR 37579, 07/21/1981)	NA				Exclude

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
RG-1.56	Withdrawn (See 75 FR 7526, 2/19/2010	NA				Exclude
RG-1.57 (Draft Rev. 1, March 2007)	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components					
1.	Code Class MC vessels, electrical and mechanical penetration assemblies, and other penetration assemblies (excluding bellows-type expansion joints) that are parts or appurtenances of the vessel. For earthquake engineering criteria, 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," would be applicable for the operating-basis earthquake (OBE) and safe-shutdown earthquake (SSE). In this manner, the OBE serves the function as an inspection-level earthquake below which the effect on the health and safety of the public would be insignificant and above witch the licensee would be required to shut down the plant and inspect for damage. Code Class MC components of primary metal containment systems that are completely enclosed within Seismic Category I structures[4] should be designed to withstand the following loads and loading combinations within the specified design limits. **1.1 Loads** **Loads** D- Dead loads. L- Live loads, including all loads resulting from platform flexibility and deformation and from crane loading, if applicable. Pt- Test pressure. Tt- Test temperature. To -Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition.					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	Ro - Pipe reactions during startup, normal operating, or shutdown conditions based on the most critical transient or steady-state condition. Po - External pressure loads resulting from pressure variation either inside or outside containment. E - Loads generated by the operating-basis earthquake including sloshing effects, if applicable. E'- Loads generated by the SSE, including sloshing effects. Pa - Pressure load generated by the postulated pipe break accident (including pressure generated by postulated small-break or intermediate-break pipe ruptures), pool swell, and subsequent hydrodynamic loads.[5] Ta -Thermal loads under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads.[5] Ra - Pipe reactions under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads.[5] Ps - All pressure loads that are caused by the actuation of safety relief valve (SRV) discharge, including pool swell and subsequent hydrodynamic loads. T s - All thermal loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic thermal loads. Rs - All pipe reaction loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic reaction loads. Yr - Equivalent static load on the structure generated by the reaction on the broken pipe during the design- Yj - Jet impingement equivalent static load on the structure generated by the broken pipe during the design-basis accident. Ym - Missile impact equivalent static load on the structure generated by or during					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	the design-basis accident, such as pipe whipping. FL - Load generated by the post-LOCA flooding of the containment, if any. Pg1 - Pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction. Pg2 - Pressure resulting from uncontrolled hydrogen burning. Pg3 - Pressure resulting from post-accident inerting, assuming carbon dioxide is the inerting agent. See Regulatory Guide 1.7 (Ref. 6) for additional guidance about the pressure load Pg3 due to combustible gas concentration. **1.2 Loading Combinations and Design Limits** The specified loads and load combinations are acceptable if found to be in accordance with the following guidance. The following load combinations include all loading combinations for which the containment might be designed for or subjected to during the expected life of the plant: **1.2.1 *Testing Condition*** This includes the testing condition of the containment to verify its leak integrity. In this case, the loading combination includes: D + L + Tt + Pt **1.2.2 *Design Conditions*** These include all design loadings for which the containment vessel or portions thereof might be designed for during the expected life of the plant. Such loads include design pressure, design temperature, and the design mechanical loads generated by the design-basis accident. In this case, the loading combination includes: D + L + Pa + Ta + Ra **1.2.3 *Service Conditions***					

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	The load combinations in these cases correspond to and include Level A service limits, Level B service limits, Level C service limits, Level D service limits and the post-flooding condition. The loads may be combined by their actual time history of occurrence, taking into consideration their dynamic effect upon the structure. **1.2.3.1 Level A Service Limits** These service limits are applicable to the service loadings to which the containment is subjected, including the plant or system design-basis accident conditions for which the containment function is required, excepting only those categorized as Level B, C, or D, or Testing Loadings. The loading combinations corresponding to these limits include the following: (1) Normal operating plant condition D + L + To + Ro + Po (2) Operating plant condition in conjunction with multiple SRV actuations D + L + Ts + Rs + Ps (3) Loss of coolant Accident D + L + Ta + Ra + Pa (4) Multiple SRV actuations in combination with a small- or intermediate-break accident D + L + Ta + Ra + Pa + Ts + Rs + Ps (5) Normal operating plant conditions in combination with inadvertent full actuation of a post-accident inerting hydrogen control system [reference					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	10 CFR 50.34(f)(3)(v)(B)(1)] D + L + To + Ro + Po + Pg3 (6) Pressure Test load to ensure that the containment will safely withstand the pressure calculated to result from carbon-dioxide inerting [reference 10 CFR 50.3(f)(3)(v)(B)(2)] D + 1.10 x Pg3 **1.2.3.2 Level B Service Limits** These service limits include the loads subject to Level A service limits, plus the additional loads resulting from natural phenomena during which the plant must remain operational. The loading combinations corresponding to these limits include the following: (1) Design-basis LOCA in combination with the operating-basis earthquake (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered) D + L + To + Ro + Po + E (2) Operating plant condition in combination with the operating-basis earthquake (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered) D + L + To + Ro + Po + E (3) Operating plant condition in combination with the operating-basis earthquake and multiple SRV actuations (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered)					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	D + L + Ts + Rs + Ps + E (4) Loss-of-coolant accident in combination with a single active component failure causing one SRV discharge D + L + Ta + Pa + Ra + Ts + Rs + Ps **1.2.3.3 Level C Service Limits** These service limits include the loads subject to Level A service limits, plus the additional loads resulting from natural phenomena for which safe shutdown of the plant is required. The loading combinations corresponding to these limits include the following: (1) Loss-of-coolant accident in combination with the SSE D + L + Ta + Ra + Pa + E' (2) Operating plant condition in combination with the SSE D + L + To + Ro + Po + E' (3) Multiple SRV actuations in combination with a small- or intermediate-break accident and SSE D + L + Ta + Ra + Pa + Ts + Rs + Ps + E' (4) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by hydrogen burning [reference 10 CFR 50.34(f)(3)(v)(A)(1); 10 CFR 50.44]					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	D + Pg1 + Pg2 [NOTE: In this load combination, Pg1 + Pg2 should not be less than 310 kPa (45 psig) and evaluation of instability is not required.] (5) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by the added pressure from post-accident inerting, assuming carbon dioxide as the inerting agent [reference 10 CFR 50.34(f)(3)(v)(A)(1); 10 CFR 50.44] D + Pg1 + Pg3 [NOTE: In this load combination, Pg1 + Pg3 should not be less than 310 kPa (45 psig) and evaluation of instability is not required.] **1.2.3.4 Level D Service Limits** These service limits include other applicable service limits and loadings of dynamic nature for which the containment function is required. The load combinations corresponding to these limits include the following: (1) Loss-of-coolant accident in combination with the SSE and local dynamic loadings D + L + Ta + Ra + Pa + Yr + Yj + Ym + E' (2) Multiple SRV actuations in combination with a small- or intermediate-break accident, SSE, and local dynamic loadings D + L + Ta + Ra + Pa + Yr + Yj + Yj + Ps + Ts + Rs + E' (3) Post-LOCA flooding of the containment in combination with the operating-basis earthquake					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	D + L + FL + E **1.3 Design Limits** Total stresses for the combination of loads delineated in Regulatory Position 1.2 (above) are acceptable if found to be within the limits defined by Articles NE-3221.1, NE-3221.2, NE-3221.3 and NE-3221.4 of the Code. **1.4 Treatment of Buckling Effects** Earthquake, thermal, and pressure loads require consideration of buckling of the shell. Buckling of shells with more complex geometries and loading conditions than those covered by Article NE-3133 of the Code should be considered in accordance with the criteria described in ASME Code Case N-284-2, pending endorsement in Regulatory Guide 1.84 (Ref. 7).[6] An acceptable approach to this problem is to perform a nonlinear analysis.					
2.	**Bellows-Type Expansion Joints that are Parts or Appurtenances of ASME Code Class MC Vessels** Bellows-type expansion joints that are parts or appurtenances of Code Class MC components that are completely enclosed within Seismic Category I structures should be designed to withstand the loads and loading combinations within the design limits specified in Regulatory Position 1 (above), as applicable, supplemented by the design limits specified in Article NE-3366.2(b) of the Code. **Ultimate Capacity of Concrete Containment** A nonlinear finite element analysis should be performed to determine the ultimate capacity of the containment. Additional information guidance is provided in the SRP 3.8.2.[6] [4] Components of primary reactor containment systems are Seismic					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	Category I for seismic design purposes in accordance with Regulatory Guide 1.29, "Seismic Design Classification" (Ref. 5). Seismic Category I SSCs are designed to remain functional if the SSE occurs. [5] For load combinations 1.2.3.1(4), 1.2.3.3(3), and 1.2.3.4(2), a small or intermediate pipe break is postulated. For all other load combinations involving a loss of coolant accident (LOCA), the design-basis LOCA is postulated. 6 Code Case N-284, "Metal Containment Shell Buckling Design Methods, Class MC Section III, Division 1," is currently being revised. Revision 1 of N-284 is unacceptable to the NRC, as discussed in Regulatory Guide 1.193 (Ref. 8). Revision 2 of N-284 is correcting errata, misprints, recommendations, and errors identified by the NRC staff, and is expected to be approved when it is published.					
RG-1.58	Withdrawn (See 56 FR 36175, 07/31/1991)	NA				Exclude
RG-1.59 (Rev. 2, August 1977)	Design Basis Floods for Nuclear Power Plants					
1.	The conditions resulting from the worst site-related flood probable at a nuclear power plant (e.g., PMF, seismically induced flood, hurricane, seiche, surge, heavy local precipitation) with attendant wind generated wave activity constitute the design basis flood conditions that safety-related structures, systems, and components identified in Regulatory Guide 1.29 (see footnote 3) must be designed to withstand and retain capability for cold shutdown and maintenance thereof. a. The PMF on streams, as defined in Appendix A and based on the analytical techniques summarized in Appendices A and B of this guide, provides an acceptable level of conservatism for estimating flood levels caused by severe hydrometeorological conditions. b. Along lakeshores, coastlines, and estuaries, estimates of flood levels resulting from severe surges, seiches, and wave action caused by hydrometeorological activity should be based on criteria comparable in conservatism to those used for Probable Maximum Floods. Criteria and					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	analytical techniques providing this level of conservatism for the analysis of these events are summarized in Appendix A of this guide. Appendix C of this guide presents an acceptable method for estimating the stillwater level of the Probable Maximum Surge from hurricanes at open-coast sites on the Atlantic Ocean and Gulf of Mexico. c. Flood conditions that could c. Flood conditions that could be caused by dam failures from earthquakes should also be considered in establishing the design basis flood. Analytical techniques for evaluating the hydrologic effects of seismically induced dam failures discussed herein are presented in Appendix A of this guide. Techniques for evaluating the effects of tsunami will be presented in a future appendix. d. Where upstream dams or other features that provide flood protection are present, in addition to the analyses of the most severe floods that may be induced by either hydrometeorological or seismic mechanisms, reasonable combinations of less-severe flood conditions and seismic events should also be considered to the extent needed for a consistent level of conservatism. The effect of such combinations on the flood conditions at the plant site should be evaluated in cases where the probability of such combinations occurring at the same time and having significant consequences is at least comparable to the probability associated with the most severe hydrometeorological or seismically induced flood. For relatively large streams, examples of acceptable combinations of runoff floods and seismic events that could affect the flood conditions at the plant arc contained in Appendix A. Less-severe flood conditions, associated with the above seismic events, may be acceptable for small streams that exhibit relatively short periods of flooding. e. The effects of coincident wind-generated wave activity to the water levels associated with the worst site-related flood possible (as determined from paragraphs a, b, c, or d above) should be added to generally define the					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	upper limit of flood potential. Acceptable procedures are contained in Appendix A of this guide.					
2.	2. As an alternative to designing *hardened protection* for all safety-related structures, systems, and components as specified in Regulatory Position 1 above, it is permissible not to provide hardened protection for some of these features if: a. **S**ufficient warning time is shown to be available to shut the plant down and implement adequate emergency procedures; b. All safety-related structures, systems, and components identified in Regulatory Guide 1.29 (see footnote 3) arc designed to withstand the flood conditions resulting from a Standard Project events with attendant wind-generated wave activity that may be produced by the worst winds of record and remain functional; c. In addition to the analyses in paragraph 2.b above, reasonable combinations of less-severe flood conditions are also considered to the extent needed for a consistent level of conservatism; and d. In addition to paragraph **2.b** above, at least those structures, systems, and components necessary for cold shutdown and maintenance thereof are designed with *hardened* protective features to remain functional while withstanding the entire range of flood conditions up to and including the worst site-related flood probable (e.g., PMF, seismically induced flood, hurricane, surge, seiche, heavy local precipitation) with coincident wind-generated wave action as discussed in Regulatory Position I above.					
3.	During the economic life of a nuclear power plant, unanticipated changes to the site environs which may adversely affect the flood-producing characteristics of the environs are possible. Examples include construction of a dam upstream or downstream of the plant or, comparably, construction of a highway or railroad bridge and embankment that obstructs the flood					

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	flow of a river and construction of a harbor or deepening of an existing harbor near a coastal or lake site plant. Significantly adverse changes in the runoff or other flood-producing characteristics of the site environs, as they affect the design basis flood, should be identified and used as the basis to develop or modify emergency operating procedures, if necessary, to mitigate the effects of the increased flood.					
4.	Proper utilization of the data and procedures in Appendices B and C will result in PMF peak discharges and PMS peak stillwater levels which will in many cases be approved by the NRC staff with no further verification. The staff will continue to accept for review detailed PMF and PMS analyses that result in less conservative estimates than those obtained by use of Appendices B and **C.** In addition, previously reviewed and approved detailed PMF and PMS analyses will continue to be acceptable even though the data and procedures in Appendices B and C result in more conservative estimates.					
RG-1.60 (Rev. 1, December 1973)	Design Response Spectra for Seismic Design of Nuclear Power Plants Note: Refer to the Regulatory Guide for the figures and tables					
1.	The horizontal component ground Design Response Spectra, without soil-structure interaction effects, of the SSE, 1/2 the SSE. or the OBE on sites underlain by rock or by soil should be linearly scaled from Figure **1**[2] in proportion to the maximum horizontal pound acceleration specified for the earthquake chosen. (Figure **1** corresponds to a maximum horizontal ground acceleration of 1.0 g and accompanying displacement of 36 in.) The applicable multiplication factors and control points are given in Table **I.** For damping ratios not included in Figure **1** or Table I, a linear interpolation should be used.					
2.	The vertical component ground Design Response Spectra, without soil-structure interaction effects, of the SSE. 1/2 the SSE, or the OBE on sites underlain by rock or by soil should be linearly scaled from Figure 2[2] in proportion to the maximum horizontal ground					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	acceleration specified for the earthquake chosen. (Figure 2 is based on a maximum *horizontal ground acceleration* of **1.0** g and accompanying displacement of 36 in.) The applicable multiplication factors and control points are given in Table **11.** For damping ratios not included in Figure 2 or Table **11,** a linear interpolation should be used. [2]**This** does not apply to sites which (1) **are** relatively **close** to the epicenter of **an** expected earthquake of (2) which have physical characteristics that **could sig**nificantly affect the spectral combination **of** input motion. The Design Response Spectra for such sites should be developed on a case-by-case basis.					
RG-1.61 (Draft Rev. 1, March 2007)	Damping Values for Seismic Design of Nuclear Power Plants The following regulatory positions provide acceptable damping values to be used in the elastic dynamic seismic analysis and design of SSCs, where energy dissipation is approximated by viscous damping unless otherwise specified. Damping values higher than those provided may be used if documented test data support the higher values. Damping values associated with soil-structure interaction analysis are not within the scope of this regulatory guide. Note: Refer to the Regulatory Guide for the figures and tables					
1.	**Structural Damping** **1.1 Acceptable Structural Damping Values for Containment Structures,** **Containment Internal Structures, and Other Seismic Category I Structures** **1.1.1** ***Safe-Shutdown Earthquake (SSE)*** Table 1 provides acceptable damping values for the SSE analysis. **1.1.2** ***Operating-Basis Earthquake (OBE)*** If the design-basis OBE ground acceleration is selected to be less than or equal to one-third of the design-basis SSE ground acceleration, then a					

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	separate OBE analysis is not required. However, if the design-basis OBE ground acceleration is selected to be greater than one-third of the design-basis SSE ground acceleration, then a separate OBE analysis should be conducted. Table 2 provides acceptable damping values for the OBE analysis. **1.2 Special Consideration for In-Structure Response Spectra Generation** The SSE damping values specified in Table 1 for linear dynamic analysis of structures have been selected based on the expectation that the structural response attributed to load combinations that include SSE will be close to applicable code stress limits, as defined in Section 3.8 of NUREG-0800 [Ref. 15]. However, there may be cases where the predicted structural response to load combinations that include SSE is significantly below the applicable code stress limits. Because equivalent viscous damping ratios have been shown to be dependent on the structural response level, it is necessary to consider that the SSE damping values specified in Table 1 may be inconsistent with the predicted structural response level. For structural evaluation, this is not a concern, because the stresses resulting from the use of damping-compatible structural response will still be less than the applicable code stress limits, as defined in Section 3.8 of NUREG-0800 [Ref. 15]. However, for in-structure response spectra generation, it is necessary to use the damping-compatible structural response. Consequently, the following additional guidance is provided for analyses used to determine in-structure response spectra: (1) Use the OBE damping values specified in Table 2, which are acceptable to the staff without further review. (2) Submit a plant-specific technical basis for use of damping values higher than the OBE damping values specified in Table 2, but not greater than the SSE damping values specified in Table 1 (e.g., see NUREG/CR-6919,					

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	Section 3.2.3), subject to staff review on a case-by-case basis. In general, for certified standard plant designs where the design-basis in-structure response spectra represent the envelope of the in-structure responses obtained from multiple analyses conducted to consider a range of expected site soil conditions, it is not necessary for combined license applicants to address this issue. However, if plant-specific seismic analyses are conducted for Category I structures and/or structures not included as part of the standard plant design, then the applicant is expected to address this issue accordingly.					
2.	Table 3 presents the constant damping values specified for SSE and OBE (where required) analyses of piping systems. These values are applicable to time-history, response spectra, and equivalent static analysis procedures for structural qualification. As an alternative for response spectrum analyses using an envelope of the SSE or OBE response spectra at all support points (uniform support motion), frequency-dependent damping values shown in Figure 1 may be used, subject to the following restrictions: • Frequency-dependent damping should be used completely and consistently, if at all. (Damping values specified in Regulatory Guide 1.61 are to be used for equipment other than piping.) • Use of the specified damping values is limited only to response spectral analyses. Acceptance of the use of the specified damping values with other types of dynamic analyses (e.g., time-history analyses or independent support motion method) requires further justification. • When used for reconciliation or support optimization of existing designs, the effects of increased motion on existing clearances and online mounted equipment should be checked. • Frequency-dependent damping is not appropriate for analyzing the dynamic response of piping systems using supports designed to dissipate energy by yielding. • Frequency-dependent damping is not applicable to piping in which stress corrosion cracking has occurred, unless a case-specific evaluation is					

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	provided and reviewed and found acceptable by the NRC staff.					
3.	**Electrical Distribution System Damping** Table 4 presents the constant damping values specified for SSE and OBE (where required) analyses of cable tray and conduit systems. These values are applicable to response spectra and equivalent static analysis procedures for structural qualification. The damping values specified in Table 4 are applicable to all types of supports, including welded supports. The use of higher damping values for cable trays with flexible support systems (e.g., rod-hung trapeze systems, strut-hung trapeze systems, and strut-type cantilever and braced cantilever support systems) is permissible, subject to obtaining NRC review for acceptance on a case-by-case basis. The analysis methodology should consider the flexibility of supports in determining the system response to seismic excitation.					
4.	**Heating, Ventilation, and Air Conditioning Duct Damping** Table 5 presents the constant damping values specified for SSE and OBE (when required) analyses of HVAC duct systems. These values are applicable to response spectra and equivalent static analysis procedures for structural qualification. The analysis methodology must consider the flexibility of supports in determining system response to seismic excitation.					
5.	**Mechanical and Electrical Component Damping** Table 6 presents the damping values for mechanical and electrical components, which are applicable to passive subcomponents that can be seismically qualified by analysis. Active subcomponents do not readily lend themselves to seismic qualification by analysis, and require qualification by test, as described in Section 3.10 of NUREG-0800 [Ref. 15].					
RG-1.62 (Rev. 0, October	Manual Initiation of Protective Actions					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
1973)						
1.	Means should be provided for manual initiation of each protective action (e.g., reactor trip, containment isolation) at the system level, regardless of whether means are also provided to initiate the protective action at the component or channel level (e.g., individual control rod, individual isolation valve).					
2.	Manual initiation of a protective action at the system level should perform all actions performed by automatic initiation such as starting auxiliary or supporting systems, sending signals to appropriate valve-actuating mechanisms to assure correct valve position, and providing the required action-sequencing functions and interlocks.					
3.	The Switches for manual initiation of protective actions at the system level should be located in the control room and be easily accessible to the operator so that action can be taken in an expeditious manner.					
4.	The amount of equipment common to both manual and automatic initiation should be kept to a minimum. It is preferable to limit such common equipment to the final actuation devices and the actuated equipment. However, action-sequencing functions and interlocks (of Position 2) associated with the final actuation devices and actuated equipment may be common if individual manual initiation at the component or channel level is provided in the control room. No single failure within the manual, automatic, or common portions of the protection system should prevent initiation of protective action by manual or automatic means.					
5.	Manual initiation of protective actions should depend on the operation of **a** minimum of equipment, consistent with **1,** 2, 3, and 4 above.					
6.	Manual initiation of protective action at the system level should be so designed that once initiated, it **will** go to completion as required in Section 4.16 of IEEE Std 279-1971.					
RG-1.63 (Rev. 3, February 1987)	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants Conformance with the requirements of IEEE Std 317-1983, "IEEE Standard					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," provides a method acceptable to the NRC staff for satisfying the Commission's regulations with respect to the design, construction, testing, qualification, and installation of electric penetration assemblies in containment structures for nuclear power plants, subject to the following: The external circuit protection of electric penetration assemblies should meet the provisions of Section 5.4 of IEEE Std 741-1986, "Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations."					
RG-1.64	Withdrawn (See 56 FR 36175, 07/31/1991)	NA				Exclude.
RG-1.65 (Rev. 0, October 1973)	Materials and Inspections for Reactor Vessel Closure Studs					
1.	Bolting Materials a. In accordance with Section III of the ASME BPV Code, as incorporated by reference into 10 CFR 50.55a, "Codes and Standards," reactor vessel closure stud bolting must be fabricated from materials that have adequate toughness throughout the life cycle of the reactor. The staff's position is that applicants can meet the applicable requirements by following this guidance to ensure that reactor vessel closure stud bolting is designed and tested in an appropriate manner: i. The measured yield strength of the stud bolting material should not exceed 1,034 MPa (150 ksi). ii. Stud bolting should not be metal-plated unless it has been demonstrated that the plating will not degrade the quality of the stud in any significant way (e.g., corrosion and hydrogen embrittlement) or reduce the quality of results attainable by the various required inspection procedures. The stud bolting may have a manganese phosphate (or other acceptable) surface treatment. Lubricants for the stud bolting are permissible, provided that they are stable at operating temperatures and are compatible with the bolting and vessel					

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	materials and with the surrounding environment.					
2.	Protection against Corrosion a. As provided in Section 3.13 of NUREG-0800, lubricants with deliberately added halogens, sulfur, or lead should not be used for any reactor coolant pressure boundary components or other components in contact with reactor water. Lubricants containing molybdenum sulfide (disulfide or polysulfide) should not be used for any safety-related applications. Fasteners should not be plated with low melting point materials such as zinc, tin, cadmium, etc. b. During the venting and filling of the pressure vessel and while the head is removed, the stud bolts and stud bolt holes in the vessel flange should be adequately protected from corrosion and contamination.					
RG-1.66	Withdrawn (See 42 FR 54478, 10/06/1977)	NA				Exclude.
RG-1.67	Withdrawn (See 48 FR 19101, 04/27/1983)	NA				Exclude.
RG-1.68 (Draft Rev. 3, March 2007)	Initial Test Programs for Water-Cooled Nuclear Power Plants This revision of Regulatory Guide 1.68 describes the general scope and depth that the NRC staff considers acceptable for ITPs for light-water-cooled nuclear power plants. Appendix A to this guide provides a representative listing of the plant SSCs and the design features and performance capabilities that should be demonstrated during the ITP. Note: Refer to the Regulatory guide for detailed criteria.					
RG-1.68.1 (Rev. 1, January 1977)	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants Comprehensive preoperational and initial startup testing programs on the feedwater and condensate systems of boiling water reactors should be performed to provide assurance that these systems will accomplish the required functions under normal operational and transient conditions as stated in the safety analysis report.	NA				Exclude . This regulatory guide is applicable only to water cooled reactors and, therefore, is not applicable to the HTGR reactor.

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	Preoperational testing is conducted prior to fuel loading to determine component operability and performance and to verify proper system installation. While as much of this type of testing as practicable should be accomplished during this phase, the ability to conduct some system-level testing and component testing is limited by the unavailability of reactor power or system flow paths, or other factors. Thus, these tests should be completed as part of the initial plant startup test program. Tests that were satisfactorily completed during preoperational testing need not be repeated.					
1.	Preoperational Testing The preoperational phase of the initial test program should include at least tests and measurements to verify the following: a. Operability of pumps utilized to provide feedwater flow (condensate, condensate booster, and feedwater pumps). Tests should confirm that the pumps satisfy all performance requirements, including required head, flow rate, suction head, and overspeed characteristics, b. Operability and correct setpoints of permissive and prohibit interlocks in the starting and shutdown controls for the pump drivers. c. Proper operation of controls used for manual and automatic starting and stopping of the pump drivers. d. Operability of valves utilized for adjusting the feedwater. flow rate. Tests should verify proper response of valves for the design operating range and correct operation of protective features such as thermal overload devices and undervoltage sensing devices incorporated in the design of valve operators and associated control circuitry. e. Operability of sensors and associated instrumentation that provide inputs to the feedwater control system. Tests should verify stable and accurate outputs in response to test signals.					

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	f. Operability of the feedwater control system. Tests should verify the proper response of individual components in the control system (including programmers, summers. and signal modifiers) and the overall response of the control system, including the final control element. Tests should also verify that the overall response of the control system to simulated limiting malfunctions in the control system (such as feedwater controller failure to maximum flow demand) and to simulated plant transients (such as main steamline isolation valve closure at full flow conditions and turbine trip without bypass at full flow conditions) is in accordance with performance requirements for the control system. Such testing should be conducted in both single-element and three-element system control modes. g. Proper operation of instrumentation and alarms utilized to monitor the performance of the systems.					
2.	Startup Testing The startup phase of the initial test program should include at least tests and measurements to verify the following: a. Operability of the feedwater system at low reactor power ($\leq$15% reactor power). b. Proper response of the feedwater control system in the manual mode of control. Tests should verify that the system can be operated in the manual mode and that transfer to the automatic mode can be accomplished in accordance with design requirements ($\leq$15% reactor power). c. The stability and response characteristics of the automatic control system are in accordance with performance requirements for normal plant operation (15% to 100% reactor power). d. The stability and response characteristics of the automatic control system following plant transients are in accordance with system performance requirements. Tests should verify that the acceptance criteria for maximum and minimum water levels in the reactor vessel are not exceeded as a result of plant transients, such as turbine trip and main steam isolation valve					

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	closure, with the control system in the automatic mode of control (15% to 100% reactor power). e. The response of the feedwater system is in accordance with performance requirements following loss of a feedwater pump (100% reactor power).. f. Vibration levels for system components and piping are within predetermined limits. g. Piping movements during heatup and steadystate and transient operation are within predetermined limits. h. Adequate margins exist between system variables and setpoints of instruments monitoring these variables to prevent spurious actuation or loss of system pumps and motor-operated valves.					
3.	Test Reports See Regulatory Position 9 in Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants," for guidance regarding preoperational and initial startup test reports.					
RG-1.68.2 (Rev. 1, July 1978)	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants The regulations in GDC 19 and QA Criterion XI of Appendix B to 10 CFR Part 50 require licensees of water-cooled nuclear power plants to develop and conduct a test program to demonstrate remote shutdown capability for each unit. The test program should contain the following elements:					
1.	Objectives a. Verify that the nuclear power plant can be safely shut down from outside the control room. b. Verify that the nuclear power plant can be maintained in a hot shutdown condition from outside the control room.					

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	c. Verify that the nuclear power plant can be safely cooled from hot shutdown to cold shutdown conditions from outside the control room.					
2.	Prerequisites a. Approved operating procedures for performing a remote shutdown should be available, including approved procedures for conducting the test. b. Communications should exist between the control room observers and the remote shutdown locations. c. The authority and responsibility of the control room observers should be established and documented in the test procedure. Licensees should make provisions for the following: (1) Assume control of the plant if an emergency or unsafe condition develops during the testing that cannot be managed by the shutdown crew. (2) Perform non safety-related activities that would not be required during an actual remote shutdown. These could include the protection of non safety-related equipment from mechanical damage during the transient and the placement of equipment into shutdown status when no longer required. Licensees should have previously defined and evaluated such activities to ensure that, if they were not performed during an actual remote shutdown, safe shutdown of the plant could still be achieved. Any additional activities should be recorded and reviewed following the test to assess their impact on the validity of the total test performance. Individuals in addition to those comprising the minimum shift crew described in Regulatory Position 3 may carry out these activities. d. Licensees should have completed preoperational testing of plant instrumentation, controls, and systems to be used at remote shutdown					

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	locations. This preoperational testing should include verification that all systems to be used during shutdown operation from outside the control room are operable in the manner in which they would be used during the operation (i.e., control from remote stations, manual operation, use of available power supplies) and that communication could be established and maintained among the personnel who will be performing the shutdown operation. In addition, if applicable to the plant design, licensees should verify that it is not possible to control transferred components from the control room after control of these components has been established at the remote shutdown locations. Licensees can verify much of this in conjunction with other tests, such as preoperational tests on individual systems or components. Once successfully completed, these verification tests need not be repeated.					
3.	Licensees should initiate the test from a location outside the control room with the reactor at a moderate power level (10–25 percent), sufficiently high that plant systems are in their normal configuration with the turbine generator in operation. Licensees should perform the test with the minimum of personnel required to be at the reactor unit at any one time (i.e., the minimum number of reactor operators and senior reactor operators onsite per shift, as required by 10 CFR 50.54(m) and the minimum complement of personnel who are not licensed operators, as required by unit technical specifications). Data obtained at locations outside the control room should verify: a. that the plant has achieved hot shutdown status, and b. that the plant can be maintained under stable hot shutdown conditions for at least 30 minutes. During the demonstration, licensees should use only that equipment for which credit would be taken in performing an actual remote shutdown. Personnel in excess of the minimum requirements may be present during the demonstration, provided that during the demonstration the additional personnel perform only non safety-related activities that would not be					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	required during an actual emergency shutdown					
4.	Cold Shutdown Demonstration Procedure Licensees do not need to demonstrate cold shutdown capability immediately following the test to achieve and maintain a safe hot shutdown from outside the control room. Rather, licensees may combine this cooldown portion of the test with another startup test requiring the reactor to be cooled down, as long as the procedures and acceptance criteria for the combined test meet all the elements of each individual test. The licensee should demonstrate the plant's cold shutdown capability by partially cooling down the plant from the hot shutdown condition using controls and instrumentation located outside the control room. This cooldown demonstration may use additional personnel who could be made available to the unit before the time when the cooldown would have to be initiated. Each licensee should establish the number and level of such personnel in the remote shutdown procedure. The test should demonstrate that: a. The reactor coolant temperature and pressure can be lowered sufficiently to permit the operation of the core decay heat removal system that is to be ultimately used to place the reactor in a refueling shutdown mode. (This demonstration should be performed with adequate steam pressure available to perform this test and avoid damaging equipment (e.g., Safety Relief Valves)). b. Operation of this decay heat removal system can be initiated and controlled. c. A heat transfer path to the ultimate heat sink can be established. d. The reactor coolant temperature can be reduced approximately 28 degrees Celsius (50 degrees Fahrenheit) using this decay heat removal system, at a rate that would not exceed technical specification limits. This cooldown test should show that the potential exists to achieve cold					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	shutdown from outside the control room. During the demonstration, the licensee should use only that equipment for which credit would be taken to perform an actual remote shutdown. Personnel in excess of the minimum requirements may be present during the demonstration, provided that during the demonstration the additional personnel perform only non-safety related activities that would not be required during an actual emergency shutdown					
5.	Reporting The licensee should retain the testing procedures and results from the hot and cold shutdown demonstration as part of the plant's historical record. In addition, the historical record should include a summary of the testing in a startup report, consistent with (Ref. 3). This summary should include the following information: a. a description of the method and objectives for each test; b. a comparison of applicable test data with the related acceptance criteria, including the systems' responses to major plant transients (such as reactor scram and turbine trip); c. a description of all design- and construction-related deficiencies discovered during testing, system modifications, the corrective actions required to correct those deficiencies, and the schedule for implementing these modifications and corrective actions, unless previously reported to the NRC; d. justification for the acceptance of systems or components that are not in conformance with design predictions or performance requirements; and e. conclusions regarding system or component adequacy.					
RG-1.68.3 (Rev. 0, April 1982)	Preoperational Testing of Instrument and Control Air Systems As part of the initial preoperational testing program and also after					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	major modification or repairs to the instrument and control **air** system or portions thereof (e.g., where air-flow-rate requirements are significantly altered or where opened systems are subject to contamination), the system and loads should be tested **as** described below to verify that **all** components function properly at normal pressures and following possible pressure increases and that the systems respond as designed to a los-of-air-pressure event.					
1.	The test program for the instrument and control air system and associated equipment should include the applicable prerequisite checks, verifications, and tests provided in Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."					
2.	Compressors, aftercoolers, oil separator units, air receivers, and pressure-reducing stations should be tested to verify proper operation according to system design. The operation of compressor unloaders, automatic and manual start and stop circuits of standby compressors, high- and low-pressure alarms, pressure indicators, and temperature indicators should be checked. Relief valve settings should be verified.					
3.	Air dryer units should be tested for proper functioning, and the units should be operated through at least one regeneration cycle. Acceptable operation at maximum flow rates should be verified. The appropriate differential pressures and proper operation of pressure switches, high- and low-pressure alarms, safety and relief valves, bypass valves, and alarms and resets should be verified.					
4.	It should be verified by test that the instrument and control air system will meet system design specifications relating to flow, pressure, and temperature of the product air.					
5.	It should be established by appropriate measurements or observations that the total air demand at normal steady-state conditions, including leakage from the system, is in accordance with design.					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
6.	The ability of the system to meet the quality requirements of the system design should be verified. ANSIJISA S73-1975, "Quality Standard for Instrument Air," is an acceptable standard with respect to oil, water, and particulate matter contained in the product air. The quality should be verified by analyzing the air at the end of each feeder line using continuous flow techniques or **by** analyzing **a** discrete sample.					
7.	When redundant components **and** air supplies are provided in the facility design to meet the single-failure criterion for a **given** safety function, it should be verified by test that the single-failure criterion a met.					
8.	It should be verified by tests that the air-operated or a air-powered loads that *are* ***a*** part of (or support the operation of) portions of the facility important to safety respond i**n** accordance with design to a **loss** of air pressure Testing should be sufficiently comprehensive to determine the response of loads to complete loss of system pressure, both sudden and gradual, and to partial reductions in system pressure. **For** valves that use multiple air connections **(e.g.,** 30 psi to pilot and 100 psi to positioner or booster relay), if failure of less than **all** ***air*** supply **sources** is credible, the tests should verify that the valve responds safely to all failure modes. The tests should verify the adequacy of design requirements relating to system pressures at which supplied loads change state (e.g., fail open, fail closed, fail *as is,* fail upscale, fail downscale, or **fail** to perform other required functions). Testing should also verify that the backup supplies for the protected loads supplied by the system, **e.g.,** accumulators and backup bottled gas supplies, will maintain sufficient air pressure to permit these loads to perform their design function. As part of the above testing, los-of-air-supply tests should be conducted on all branches of the instrument and control air system simultaneously, if practicable, or on the *largest* number of branches of the system that can be adequately managed. For each test, the valves to be tested should be placed in their normal operating position, and the rest of the plant should **be** maintained in **as** close to normal conditions as **is** practicable. (It should be					

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	noted that not all valves can be placed in the required normal operating position because of operating procedure requirements or personnel or equipment safety factors.) The following tests should be performed: a. Shut off the instrument and control air system in a manner that would simulate a sudden air pipe break and verify that the affected components respond properly. b. Repeat test a., but shut the instrument and control air system off very slowly to simulate **a** gradual loss of pressure.					
9.	Tests should be conducted, as appropriate, to demonstrate that plant equipment designated by design to be supplied by the instrument and control system is not being supplied by other compressed air supplies (such as service air) that may have less restrictive air quality requirements.					
10.	Plant components requiring large quantities of instrument and control air for operation (such **as** large valve operators) should be operated simultaneously while the system is operating at normal steady-state conditions (unless it can be shown that simultaneous operation is prohibited by interlock or appropriate procedure) to verify that pressure transients in the distribution system do not exceed acceptable values.					
11.	Functional testing of instrument and control *air* systems important to safety should be performed to ensure that credible failures resulting in an increase in the supply system pressure will not cause los of operability.					
RG-1.69 (Rev. 0, December 1973)	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants ANSI/ANS-6.4-2006, ACI 349-06, and ACI 349.1R-07 are acceptable for the construction of radiation shielding structures of hot laboratories, radiochemical plants, experimental facilities, nuclear fuel fabrication plants, and the shielding structures for nuclear power plants, with a few exceptions. Section C.1 lists specific guidelines for the combined use of the above standards in the design and construction of the concrete radiation shields for nuclear power plants. Section C.2 lists the specific provisions of the above standards that the NRC has not endorsed. Section C.3 endorses					

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	ANSI/ANS-6.3.1-1987; R2007, which describes a test program to be used in evaluating biological radiation shielding in nuclear reactor facilities under normal operating conditions, including anticipated operational occurrences.					
1.	**Guidelines for Use of ANSI/ANS-6.4-2006, ACI 349-06, and ACI 349.1R-07** a. The minimum thickness of concrete radiation shields, based on radiation shielding requirements, should be determined using the following approach: (1) Use ANSI/ANS-6.4-2006, Chapters 6, 7, and 8, as an overview of the historic calculation methodology for concrete radiation shields. (2) Use the Monte Carlo technique for radiation shielding calculations (e.g., Richard H. Olsher, "A Practical Look at Monte Carlo Variance Reduction Methods in Radiation Shielding") (Ref. 7). (3) Use the latest version of the software for radiation shielding calculations (i.e., MCNP Monte Carlo Team, X-5) (Ref. 8). The concrete composition input parameters for the MCNP5 calculations should correspond to the specific concrete used for the radiation shields. Applicant's referencing shielding codes other than the Los Alamos MCNP computer code must provide a justification to the NRC as to why these other codes represent an acceptable alternative shielding code to the MCNP computer code. The latest versions of codes which have been approved by the NRC are available through the Radiation Safety Information Computational Center (RSICC), Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN 37831-6171. b. The minimum thickness of concrete radiation shields, based on structural requirements, and other structural dimensions and reinforcement requirements should be determined in accordance with the provisions of ACI 349-06 and ACI 349.1R-07 for applicable normal loads, severe and extreme environmental loads, and abnormal loads, as defined in Section 9.1 of ACI 349-06. c. The final minimum thickness of a concrete shield structure should be the					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	greater of the following two values: (1) Thickness determined based on radiation shielding requirements in accordance with Regulatory Position C.1.a. (2) Thickness determined based on structural requirements in accordance with Regulatory Position C.1.b. d. Load and strength reduction factors for the structural design of concrete shield structures and related members should be based on those prescribed in ACI 349-06, Sections 9.2 and 9.3, respectively. e. The design of the concrete for shielding structures, including materials selection, durability requirements, quality control, mixing, placement, formwork, embedded pipes, construction joints, reinforcement, analysis, and design, should conform to provisions outlined in Chapters 3 through 8 of ACI 349-06.					
2.	**Exceptions for Use of ACI 349-06, and ACI 349.1R-07** ACI 349-06, Section 1.2.2, states that input and output data should be retained as documentation when software is used for the calculation. The software itself and other related documentation should be retained as well. It is not required that the software be updated regularly. The NRC does not endorse the following sections of ACI 349-06: a. Section 3.3.1: The exception portion of the section is not endorsed. b. Section 3.3.2: "These limitations may be waived if, in the judgment of the engineer, workability and methods of consolidation are such that concrete can be placed without honeycombs or voids." c. Section 5.4.1: "If data required by 5.3 are not available, concrete proportions shall be based on other experience or information, if approved by the engineer. The required average compressive strength ***fc′r*** of concrete produced with materials similar to those proposed for use shall be at least 1200 psi greater than ***fc′***. This alternative shall not be used if ***fc′*** is					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	greater than 5000 psi." d. Section 5.6.2.3: "When total quantity of a given class of concrete is less than 50 yd^3, strength tests may be waived by the engineer if the engineer has been provided adequate evidence of satisfactory strength." Instead, follow the provisions of Regulatory Position 5 of Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants" (Ref. 9) for strength testing. e. Section 7.10.3: "It shall be permitted to waive the lateral reinforcement requirements of 7.10, 10.16, and 18.11 where tests and structural analysis show adequate strength and feasibility of construction."					
3.	The NRC endorses the standard ANSI/ANS-6.3.1 1987; R2007, "Program for Testing Radiation Shields in Light Water Reactors (LWR)" for testing radiation shields. The standard describes a test program to be used in evaluating biological radiation shielding in nuclear reactor facilities under normal operating conditions, including anticipated operational occurrences.					
RG-1.70 (Rev. 3, November 1978)	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants The purpose of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (hereinafter "Standard Format") is to indicate the information to be provided in the SAR and to establish a uniform format for presenting the information. Use of this format will help ensure the completeness of the information provided, will assist the Commission's staff and others in locating the information, and will aid in shortening the time needed for the review process. Refer to the Regulatory Guide for the detailed criteria.					Only to the extent that technical guidance is provided that is not provided elsewhere. This is essentially the predecessor to RG 1.206.
RG-1.71 (Draft Rev. 1, March 2007)	Welder Qualification for Areas of Limited Accessibility Weld fabrication and repair should comply with the fabrication standards specified in Sections III and IX of the ASME Code, supplemented by the following:					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
(1)	Performance qualification should provide for testing the welder or welding operator under simulated access, and visibility limitations when physical conditions restrict the welder's access to a production weld to less than 30 centimeters (12 inches) in any direction from the joint and which would affect electrode manipulation, or bead progression, or require an indirect means of weld pool observation (such as a mirror).					
(2)	Requalification should be necessary when (a) the use of an indirect means is required to view the weld pool (such as a mirror) during production welding and the welder or welding operator did not qualify for welding in areas of limited accessibility using that indirect means of weld pool observation, or (b) any of the essential welding variables for welders (QW-350) or welding operators (QW-360) listed in Section IX change, or (c) the qualification expires per QW-320.					
(3)	Production welding and adherence to welding qualification criteria should be monitored.					
RG-1.72 (Rev. 2, November 1978)	Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin Safety-related spray pond piping components made from fiberglass-reinforced thermosetting resin should comply with ASME Code Case N-155-l (1792-1) supplemented by the following:					
1.	The design temperature for spray pond piping should be 100°C (212°F).					
2.	2. The allowable design stress should be the value obtained from the minimum HDB (hydrostatic design basis) in Table 3611-1 of Code Case N-155-1 (Procedure A or B) or the value determined as one-sixth of the stress obtained from a short-time burst test for the pipe being qualified, whichever is lower. The short-time burst strength should be determined by bursting the pipe (ASTM D-1599-74 using free-end mounting) after it has been exposed to $\mathbf{10}^5$ pressure cycles from atmospheric to design pressure.					
3.	The value of "K" in equation 9 of paragraph 3652.2 should be limited to 1.2 unless it can be demonstrated that with the use of a large value of K the functional capability of the system will not be impaired during upset					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	and emergency conditions.					
4.	The following items should be identified: a. The physical location of the system in relation to other safety-related systems, b. The design and service loads, and c. The value of "B" to be used in equation 1 of paragraph 3641.1, together with justification for its selection.					
5.	Pressure-relief devices may be omitted for piping systems that are open-ended and for which the system pressure is limited by other means (such as nonclogging spray nozzles and self-limiting pump characteristics) to design pressure.					
6.	RTR piping should be uninsulated or uncovered and installed under conditions that make it readily accessible for inspection.					
7.	Preoperational and inservice inspections should be as follows: a. During the preoperational testing period, tests should be made to verify that the piping is free of vibration induced by weather conditions or water flow that could fatigue the piping prematurely. b. Fiberglass-reinforced piping components should be inspected in accordance with **ASME** Code, Section XI, for Code Class 3 components.[3] In addition, all pipe supports should be inspected. c. Inspection frequency for piping should be increased to once annually if an exterior weather-resistant coating is not provided.					
RG-1.73 (Rev. 0, January 1974)	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants The procedures specified by IEEE Std 382-1972. "IEEE Trial-Use Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations,"[3] dated April 10, 1973, for conducting qualification tests of electric valve operators for service inside the containment vessel of water-cooled					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	and gas-cooled nuclear power plants are generally acceptable and provide an adequate basis for complying with the qualification testing requirements of Section **III** of Appendix B to 10 CFR Part 50 to verify adequacy of design for service under design basis event conditions. subject to the following:					
1.	To the extent practicable, auxiliary equipment (e.g., limit switches) that is not integral with the valve operator mechanism but will be part of the installed valve operator assembly should be tested in accordance with the subject standard.					
2.	The test sequence described in Section 4.5.2 of the standard should be used unless the anticipated actual service operating sequence for the valve operator is expected to create a more severe operating condition than described in Section 4.5.2. In such case, the actual service sequence should be used in the test.					
3.	To assure that the valve operator is tested under an environment of sufficient severity, the magnitude of the environmental conditions (e.g., temperature, pressure, radiation, humidity) that, simulate the conditions to which the valve operator is expected to be exposed during and following a design basis accident (Section 4.4, second paragraph) should be based on conservative calculations.					
4.	The radiological source term for qualification tests in a nuclear radiation environment should be based on the same source term used in Regulatory Guide 1.7 (Safety Guide 7), "Control, of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," for BWRs and PWRs. An equivalent source term (i.e., 100% of the noble gases, 50% of the halogens, and 1% of the remaining solids developed from maximum full-power operation of the core) should be used for HTGRs. The containment size should be taken into account in each case. For exposed organic materials, calculations should take into account both beta and gamma radiation.					
5.	Qualification testing for gas-cooled reactor (HTGR) components should follow the written description in Section 4 of IEEE Std 382-					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	1972 through at least two environmental transients of the temperature profiles depicted in Figures 2 and 3 of IEEE Std 382-1972.					
6.	Part I, Section 6, "Standard References," of IEEE Std 382-1972, dated April 10, 1972, lists additional applicable IEEE Standards. The specific applicability or acceptability of these referenced standards has been or will be covered separately in other regulatory guides, where appropriate.					
RG-1.74	Withdrawn (See 54 FR 38919, 09/21/1989)	NA				Exclude
RG-1.75 (Rev. 3, February 2005)	Physical Independence of Electric Systems					
(1)	Sections 7.1.2.1, 7.1.2.4, and 7.2.2.3 of IEEE Std. 384-1992 should be supplemented as follows: *The breaker or fuse that is automatically opened by fault current may be used as an isolation device, provided that (a) the fault current under bolted and arcing fault conditions (assuming multiple faults of all non-safety-related loads and load current of all safety-related circuits) will cause the nearest circuit breaker or fuse to interrupt the fault current prior to initiation of a trip of any upstream protection device, and (b) periodic testing of circuit breakers (visual inspection of fuses and fuse holders) during every refueling must demonstrate that the overall coordination scheme under multiple faults of non-safety-related loads remains within the limits specified in the design criteria for the nuclear power plant.*					
(2)	The summary results of the analysis performed to meet the requirements of IEEE Std. 384-1992, for example, to comply with Sections 5.5.2, 5.6, 6.1, etc., should be included in the final safety analysis report for the nuclear power plant.					
(3)	Section 6.1.1.2 of IEEE Std. 384-1992 should be supplemented as follows:					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	Cable splices in raceways should generally be avoided to the extent that it is practical to do so.					
(4)	Section 5.6(3) of IEEE Std. 384-1992 should not be construed as allowing less than minimum separation of non-safety-related circuits from safety-related circuits to be justified by analyses without treatment of the affected non-safety-related circuits as associated circuits.					
(5)	Section 3 of IEEE Std. 384-1992 references several industry codes and standards. If a referenced standard has been separately incorporated into the NRC's regulations, licensees and applicants must comply with the standard as set forth in the regulation. If a referenced standard has been endorsed by the NRC staff in a regulatory guide, the standard constitutes an acceptable method of meeting a regulatory requirement as described in the regulatory guide. If a referenced standard has been neither incorporated into the NRC's regulations nor endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced standard, if appropriately justified, consistent with regulatory practice.					
RG-1.76 (Draft Rev. 1, March 2007)	Design Basis Tornado and Tornado Missiles for Nuclear Power Plants The NRC staff has established the following regulatory positions for licensees and applicants to use in selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear power plant should be designed to withstand to prevent undue risk to the health and safety of the public.					
1.	**Design-Basis Tornado Parameters** Nuclear power plants should be designed to withstand the design-basis tornado. The parameter values specified in Table 1 for the appropriate					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	regions identified in Figure 1 are generally acceptable to the NRC staff for defining the design-basis tornado for a nuclear power plant. If a design-basis tornado proposed for a given site is characterized by less-conservative parameter values than the regional values in Table 1, a comprehensive analysis should be provided to justify the selection of the less-conservative design-basis tornado. Sites located near the general boundaries of adjoining regions may involve additional considerations. The radius of maximum rotational speed of 45.7 meters (150 feet) is used for all three tornado intensity regions.					
2.	**Design-Basis Tornado-Generated Missile Spectrum** The design-basis tornado-generated missile spectrum in Table 2 is generally acceptable to the staff for the design of nuclear power plants.					
RG-1.77 (Rev. 0, May 1974)	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors Note: Refer to the Regulatory Guide for detailed criteria on Physics and Thermal Hydraulics, and Radiological Assumptions, respectively. Acceptable assumptions and evaluation models for analyzing a rod ejection accident in PWRs are presented in Appendices A (Physics and Thermal-Hydraulics) and B (Radiological Assumptions) of this guide. By use of these appendices, it should be shown that:	NA				Exclude ,RG 1.77 was superseded by RG 1.183 for new plants.
1.	Reactivity excursions will not result in a radial average fuel enthalpy greater than 280 cal/g at any axial location in any fuel rod.					
2.	Maximum reactor pressure during any portion of the assumed transient will be less than the value that will cause-stresses to-exceed the Emergency Condition-stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code.					
3.	Offsite dose consequences will be well within the guidelines of 10 CFR Part 100, "Reactor Site Criteria."					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
1.78 (Rev. 1, December 2001)	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release Note: Refer to the Regulatory guide for the referenced tables. The following guidance is provided for evaluating the habitability of a nuclear power plant control room during a postulated hazardous chemical release.					
RG-1.79 (Rev. 1, September 1975)	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors Note: Refer to the Regulatory Guide for the detailed criteria.					Note 1
RG-1.80	Withdrawn (See 47 FR 19258, 05/04/1982)	NA				Exclude
RG-1.81 (Rev. 1, January 1975)	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plant					The design of the electrical systems for a multi unit plant has yet to be determined.
RG-1.82 (Rev. 3, November 2003)	Water Sources for Long Term Recirculation Cooling Following a Loss-of-Coolant Accident Note: Refer to the Regulatory Guide for the detailed criteria.					Note 1
RG-1.83	Withdrawn (See 74 FR 58324, 11/12/2009	NA				Exclude
RG-1.84 (Rev. 34, October 2007)	Design and Fabrication Code Case Acceptability ASME Section III Division 1					
RG-1.85	Withdrawn	NA				Exclude
RG-1.86 (Rev. 0, June 1974)	Termination of Operating Licenses for Nuclear Reactors					This is a COL item for the licensee to address.
RG-1.87 (Rev. 0, June 1975)	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors					
RG-1.88	Withdrawn (See 56 FR 36175, 07/31/1991)	NA				Exclude
RG-1.89 (Draft	Environmental Qualification of Certain Electric Equipment Important to Safety					

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RG No./Rev.	**RG Title**	**Applicable**	**Reg. or Guidance**	**Add'l Design Info**	**Add'l Reg. Needed**	**Basis/Comment**
Rev. 1, June 1984)	for Nuclear Power Plants					
RG-1.90 (Rev. 1, August 1977)	In-service Inspection of Prestressed Concrete Containment Structures With Grouted Tendons	NA				Exclude, This regulatory guide is not applicable to the HTGR reactor since there is no prestressing associated with the containment or reactor building structures.
RG-1.91 (Rev. 1, February 1978)	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites					Exclude, This is a COL item for the licensee to address (there may be a need to postulate something standard for BOP structural design purposes).
RG-1.92 (Draft Rev. 2, July 2006)	Combining Modal Responses and Spatial Components in Seismic Response Analysis					
RG-1.93 (Rev. 0, December 1974)	Availability of Electric Power Sources					
RG-1.94 (Rev. 1, April 1976)	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants					
RG-1.95	Withdrawn (01/2002)	NA				Exclude, Incorporated into Rev. 1 of RG 1.78
RG-1.96 (Rev. 1, June 1976)	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	NA				Exclude, This regulatory guide is applicable only to BWRs and, therefore, is not applicable to the HTGR reactor.
RG-1.97 (Draft Rev. 4, June 2006)	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
RG-1.98 (Rev. 0, March 1976)	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	NA				Exclude, This regulatory guide is applicable only to BWRs and, therefore, is not applicable to the HTGR reactor.
RG-1.99 Rev. 2, (May 1988)	Radiation Embrittlement of Reactor Vessel Materials					
RG-1.100 (Rev. 2, June 1988)	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants					
RG-1.101 (Draft Rev. 5, June 2005)	Emergency Planning and Preparedness for Nuclear Power Reactors					
RG-1.102 (Rev. 1, September 1976)	Flood Protection for Nuclear Power Plants					
RG-1.103	Withdrawn (See 46 FR 37579, 07/21/1981)	NA				Exclude
RG-1.104	Withdrawn (See 44 FR 49321, 08/22/1979)	NA				Exclude
RG-1.105 (Rev. 3, December 1999)	Setpoints for Safety-Related Instrumentation					
RG-1.106 (Rev. 1, March 1977)	Thermal Overload Protection for Electric Motors on Motor-Operated Valves					
RG-1.107 (Rev. 1, February 1977)	Qualifications for Cement Grouting Tendons for Prestressing Tendons in Containment Structures	NA				Exclude, This regulatory guide is not applicable to the HTGR reactor since there are no tendons in the containment or reactor building structures.
RG-1.108	Withdrawn (See 58 FR 4183, 08/05/1993)	NA				Exclude
RG-1.109 (Rev. 1, October 1977)	Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50 Appendix I					
RG-1.110 (Rev. 0, March 1976)	Cost-Benefit Analysis for Radwaste Systems for Light- Water-Cooled Nuclear					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	Power Reactors					
RG-1.111 (Rev. 1, July 1977)	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light- Water-Cooled Reactors					
RG-1.112 (Draft Rev. 1, March 2007)	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light- Water-Cooled Power Reactors					
RG-1.113 (Rev. 1, April 1977)	Estimating Aquatic Dispersion of Effluents From Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I					Exclude, This is a COL item for the licensee to address.
RG-1.114 (Rev. 3, October 2008)	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit					Exclude, This is a COL item for the licensee to address.
RG-1.115 (Rev 1, July 1977)	Protection Against Low-Trajectory Turbine Missiles					
RG-1.116 (Rev. 0-R, May 1977)	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems					
RG-1.117 (Rev. 1, April 1978)	Tornado Design Classification					
RG-1.118 (Rev. 3, April 1995)	Periodic Testing of Electric Power and Protection Systems					
RG-1.119	Withdrawn (See 42 FR 33387, 06/30/1977)	NA				Exclude.
RG-1.120	Withdrawn (08/15/2001)	NA				Exclude.
RG-1.121 (Rev. 0, August 1976)	Bases for Plugging Degraded PWR Steam Generator Tubes					
RG-1.122 (Rev. 1, February 1978)	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components					
RG-1.123	Withdrawn (See 56 FR 36175, 07/31/1991)					
RG-1.124	Service Limits and Loading Combinations for Class 1 Linear-Type					

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
(Draft Rev. 2, February 2007)	Component Supports					
RG-1.125 (Rev. 2, March 2009)	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants					
RG-1.126 (Rev. 1, March 1978)	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification					
RG-1.127 (Rev. 1, March 1978)	Inspection of Water-Control Structures Associated With Nuclear Power Plants					
RG-1.128 (Draft Rev. 2, February 2007)	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants					
RG-1.129 (Draft Rev. 2, February 2007)	Maintenance, Testing, and Replacement of Large Lead-Acid Storage Batteries for Nuclear Power Plants					
RG-1.130 (Draft Rev. 2, March 2007)	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports					
RG-1.131	Withdrawn (See 74 FR 39349, 8/6/2009)	NA				Exclude
1.132 (Draft Rev. 2, October 2003)	Site Investigations for Foundations of Nuclear Power Plants					
RG-1.133 (Rev. 1, May 1981)	Loose-Part Detection Program for the Primary System of Light- Water-Cooled Reactors					
RG-1.134 (Draft Rev. 3, March 1998)	Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses					Exclude, This is a COL item for the licensee to address.
RG-1.135	Normal Water Level and Discharge at Nuclear Power Plants					

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
(Rev. 0, September 1977)						
RG-1.136 (Draft Rev. 3, March 2007)	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments					
RG-1.137 (Rev. 1, October 1979)	Fuel-Oil Systems for Standby Diesel Generators					
RG-1.138 (Rev. 2, December 2003)	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants					
RG-1.139 (Rev. 0, May 1978)	Withdrawn (See 73 FR 32750, 06/10/2008)	NA				Exclude
RG-1.140 (Rev. 2, June 2001)	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light- Water-Cooled Nuclear Power Plants					
RG-1.141 (Rev. 0, April 1978)	Containment Isolation Provisions for Fluid Systems					
RG-1.142 (Draft Rev. 2, November 2001)	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)					
RG-1.143 (Rev. 2, November 2001)	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light- Water-Cooled Nuclear Power Plants					
RG-1.144	Withdrawn (See 56 FR 36175, 07/31/1991)	NA				Exclude
RG-1.145 (Rev. 1, February 1983)	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants					

Table A1-12: Regulatory Guides (Division 1)						
RG No./Rev.	**RG Title**	**Applicable**	**Reg. or Guidance**	**Add'l Design Info**	**Add'l Reg. Needed**	**Basis/Comment**
RG-1.146	Withdrawn (See 56 FR 36175, 07/31/1991)	NA				Exclude
RG-1.147	In-service Inspection Code Case Acceptability ASME Section XI Division 1					
RG-1.148	Withdrawn (See 75 FR 2894, 1/19/2010)	NA				Exclude
RG-1.149 (Draft Rev. 3,October 2001)	Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations					Exclude, This is a COL item for the licensee to address.
RG-1.150	Withdrawn (See 73 FR 7766, 0702/11/2008)					
RG-1.151 (Rev. 0, July 1983)	Instrument Sensing Lines					
RG-1.152 (Draft Rev. 2, January 2006)	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants					
RG-1.153 (draft Rev. 1, June 1996)	Criteria for Safety Systems					
RG-1.154 (Draft Rev. 0, January 1987)	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors	NA				Exclude, Pressurized thermal shock concerns are not applicable to the HTGR.
RG-1.155 (Rev. 0, August 1988)	Station Blackout					
RG-1.156 (Rev. 0, November 1987)	Environmental Qualification of Connection Assemblies for Nuclear Power Plants					
RG-1.157 (Rev. 0, May 1989)	Best-Estimate Calculations of Emergency Core Cooling System Performance					Note 1

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
	Note: Refer to the Regulatory Guide for the detailed criteria.					
RG-1.158 (Rev. 0, February 1989)	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants					
RG-1.159 (Draft Rev. 1, October 2003)	Assuring the Availability of Funds for Decommissioning Nuclear Reactors					Exclude, This is a COL item for the licensee to address.
RG-1.160 (Draft Rev. 2, March 1997)	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants					Exclude, This is a COL item for the licensee to address.
RG-1.161 (Draft Rev. 0, June 1995)	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb	NA				Exclude, This is a water reactor cooled concern only and not applicable to a helium gas reactor.
RG-1.162 (Draft Rev. 0, February 1996)	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels					Exclude, This is a COL item for the licensee to address.
RG-1.163 (Draft Rev. 0, September 1995)	Performance Based Containment Leak-Test Program					
RG-1.164	Not Yet Issued.	NA				Exclude
RG-1.165 (Draft Rev. 0, March 1997)	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion					
RG-1.166 (Draft Rev. 0, March 1997)	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions					Exclude, This is a COL item for the licensee to address.
RG-1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event					Exclude, This is a COL item for

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
(Draft Rev. 0, March 1997)						the licensee to address.
RG-1.168 (Draft Rev. 1, February 2004)	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants					
RG-1.169 (Draft Rev. 0, September 1997)	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants					
RG-1.170 (Draft Rev. 0, September 1997)	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants					
RG-1.171 (Draft Rev. 0, September 1997)	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants					
RG-1.172 (draft Rev. 0, September 1997)	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants					
RG-1.173 (Draft Rev. 0, September 1997)	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants					
RG-1.174 (Rev. 1, November 2002)	An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis					Exclude, This is a COL item for the licensee to address.
RG-1.175 (Rev. 0, August 1998)	An Approach for Plant-Specific, Risk-Informed Decision-making for Inservice Testing					Exclude, This is a COL item for the licensee to address.

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
RG-1.176	Withdrawn (See 73 FR 7766, 02/11/2008)	NA				Exclude
RG-1.177 (Rev. 0, August 1998)	An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications					
RG-1.178 (Draft Rev. 1, September 2003)	An Approach for Plant-Specific, Risk-Informed Decision-making: Inservice Inspection of Piping					Exclude, This is a COL item for the licensee to address.
RG-1.179 (Rev. 0, January 1999)	Standard Format and Content of License Termination Plans for Nuclear Power Reactors					Exclude, This is a COL item for the licensee to address.
RG-1.180 (Draft Rev. l, October 2003)	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems					
RG-1.181 (Draft Rev. 0, September 1999)	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)					Exclude , This is a COL item for the licensee to address.
RG-1.182 (Draft Rev. 0, May 2000)	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants					Exclude, This is a COL item for the licensee to address.
RG-1.183 (Draft Rev. 0, July 2000)	Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors					
RG-1.184 (Rev. 0, July 2000)	Decommissioning of Nuclear Power Reactors					Exclude, This is a COL item for the licensee to address.
RG-1.185 (Draft Rev. 0, July 2000)	Standard Format and Content for Post-shutdown Decommissioning Activities Report					Exclude, This is a COL item for the licensee to address.
RG-1.186	Guidance and Examples of Identifying 10 CFR 50.2 Design Bases					Exclude, This is a COL item for

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
(Draft Rev. 0, December 2000)						the licensee to address.
RG-1.187 (Draft Rev. 0, November 2000)	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments					Exclude, This is a COL item for the licensee to address.
RG-1.188 (Draft Rev. 1, September 2005)	Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses					Exclude, This is a COL item for the licensee to address.
RG-1.189 (Draft Rev. 2, October 2009)	Fire Protection for Nuclear Power Plants					Exclude, This is a COL item for the licensee to address.
RG-1.190 (Draft Rev. 0, March 2001)	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence					
RG-1.191 (Draft Rev. Rev. 0, May 2003)	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown					Exclude, This is a COL item for the licensee to address.
RG-1.192 (Draft Rev. 0, June 2003)	Operations and Maintenance Code Case Acceptability, ASME OM Code					
RG-1.193 (Draft Rev. 2, October 2007)	ASME Code Cases Not Approved for Use					
RG-1.194 (Draft Rev. 0, June 2003)	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants					
RG-1.195 (Draft Rev. 0, May 2003)	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Plants					New design basis accidents and a new source term methodology will be developed for the HTGR.

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
RG-1.196 (Draft Rev. 1, January 2007)	Control Room Habitability at Light-Water Nuclear Power Plants					
RG-1.197 (Rev. 0, May 2003)	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors					
RG-1.198 (Draft Rev. 0, November 2003)	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites					
RG-1.199 (Draft Rev. 0, November 2003)	Anchoring Components and Structural Supports in Concrete					
RG-1.200 (Draft Rev. 2, March 2009)	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities					
RG-1.201 (Draft Rev. 1 May 2006)	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance					
RG-1.202 (Draft Rev. 0 February 2005)	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Plants					Exclude, This is a COL item for the licensee to address.
RG-1.203 (Draft Rev. 0, December 2005)	Transient and Accident Analysis Methods					
RG-1.204 (Draft Rev. 0, November 2005)	Guidelines for Lightening Protection of Nuclear Power Plants					
RG-1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water					Exclude, This is a COL item for

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
(Draft Rev. 0, May 2006)	Nuclear Power Plants					the licensee to address.
RG-1.206 (Rev. 0, June 2007)	Combined License Applications for Nuclear Power Plants					Most of his RG is evaluated as part of the evaluation of NUREG-0800. Only the portions of Regulatory Guide 1.206 explicitly identified in the following rows are included in this table. These items are the relevant combined license application information that do not have a direct counterpart in NUREG-0800 but that are generally applicable to HTGRs. In each case, the information is contained in tables that cannot be readily formatted to be inserted in this table or is voluminous or both, so reference must be made to the RG to obtain the full text of the regulatory position.
RG-1.206, C.IV.1	Combined License Application Acceptance Review Checklist					Substantial portions of the Acceptance Review Checklist overlap with the material contained in Regulatory Position C.1 and NUREG-0800. However, the information in this table provides a summary and overview that is useful for judging

Table A1-12: Regulatory Guides (Division 1)

RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
						completeness of an application prior to submittal and delineates general technical areas at a high level.
RG-1.206, C.IV.4	Operational Programs					
RG-1.206, C.IV.5	General and Financial Information					
RG-1.206, C.IV.6	Limited Work Authorization and Site Redress Plan					The Limited Work Authorization portion of this guide does not contain relevant content, but the Site Redress Plan should be evaluated.
RG-1.206, C.IV.7	Pre-application activities					
RG-1.206, C.IV.8	Generic Issues					The portions of this guide specific to NUREG-0933 are generally not applicable based on reviews done during the preparation of this procedure, but the portions related to how operating experience will be considered should be evaluated.
RG-1.206, C.IV.9	Regulatory Treatment of Non-safety Systems					
RG-1.207 (Draft Rev. 0, March 2007)	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors					
RG-1.208 (Draft Rev. 0, March 2007)	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion					Exclude, This is a COL item for the licensee to address.
RG-1.209, (Draft Rev. 0,	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants					

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RG No./Rev.	RG Title	Applicable	Reg. or Guidance	Add'l Design Info	Add'l Reg. Needed	Basis/Comment
March 2007)						
RG-1.210 (Rev. 0, June 2008)	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants					
RG-1.211 (Rev. 0, April 2009)	Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants					
RG-1.212 (Rev. 0, November 2008)	Sizing of Large Lead-Acid Storage Batteries					
RG-1.213 (Rev. 0, May 2009)	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants					

Notes: 1. Regulatory Guides do not exist for the design and review of the primary and heat removal systems proposed for the HTGR in a form approaching those available for light water reactor technology. The HTGR utilizes two safety related vessel and heat removal systems; the Vessel System (VS) and the passive Reactor Cavity Cooling System (RCCS). DOE proposes two additional systems for cooling that would not have safety related functions and would not have to fully meet safety-grade quality; the Heat Transport System (HTS) and the Shutdown Cooling System (SCS). The earliest precedents abroad, and Peach Bottom and Fort St. Vrain, generally provided favorable experience fort a high temperature helium environment but no formalized criteria or industry standards were developed. Current Regulatory Guides will require review to determine if they should be modified to accommodate the HTGR design or whether new Regulatory Guides are required.

Table A1-13: Regulatory Guides (Division 4)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
RG 4.1 Rev. 2	Radiological Environmental Monitoring for Nuclear Power Plants (06/01/2009)					Note that the RG has a section with graphics and a glossary that are not included in the Regulatory Positions in this table.
RG 4.1.C.1	RG 4.1.C.1. Preoperational Radiological Environmental Monitoring Program a. A REMP should be established and implemented at least 2 years before initial facility operation. The program will contain the routine surveillances necessary to adequately characterize the radiological conditions in the vicinity of the reactor site. Once initiated, the collection of samples and analysis of data should follow the sampling and analyses schedule and should continue for the first 3 years of commercial operation. For new reactor sites that are collocated with currently operating nuclear power plants (or previously operating nuclear power plants with a currently operating REMP program), the existing operational REMP associated with the operating (or previously operating) facility will normally meet the requirements for a preoperational REMP, given that the monitoring data is relevant to the time period . b. The preoperational REMP should be conducted so that the preoperational radiological conditions are understood in sufficient detail to allow future reasonable, direct comparison with data collected after power operation of the facility. The preoperational REMP should be updated when the					

Table A1-13: Regulatory Guides (Division 4)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	land use census identifies new exposure pathways or receptor locations.					
RG 4.1.C.2	RG 4.1.C.2. Operational Radiological Environmental Monitoring Program a. Although all operating facilities will have a REMP associated with the operating reactors, some licensees may have other REMPs to satisfy other needs. An operational radiological environmental monitoring program may consist of several different parts. For example, a licensee may have (1) a REMP that is associated with the 10 CFR Part 50 licensed facility, (2) a REMP that is associated with the 10 CFR Part 72 specific-licensed facility, and (3) a REMP not explicitly required by NRC regulations (e.g., environmental samples of local community interest or samples deemed important for continuity with the preoperational REMP). This regulatory guide addresses only those REMPs required by NRC regulations, but licensees may, at their discretion, apply this information to any aspect of a REMP conducted for purposes of local community interest. b. If a licensee has a REMP as part of a 10 CFR Part 50 license and another REMP as part of a 10 CFR Part 72 specific license, the licensee may choose to establish totally separate REMPs, or it may choose to collocate surveillance equipment					

Table A1-13: Regulatory Guides (Division 4)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	where practical. In all cases, the licensee shall conduct the REMPs in accordance with the applicable regulations and the licensing bases at the site. c. The REMP is sometimes conceptualized as an offsite monitoring program. However, some portions of the REMP may be conducted on site. For example, NUREG-1301/1302 states that the inner ring of thermoluminescent dosimeters (TLDs) may be located "in the general area of the site boundary." The same is true for radioiodine and particulate sampling. NUREG-1301/1302 also describes ground water monitoring if ground water is "likely to be affected" and describes the monitoring of drinking water supplies if they "could be affected." Licensees should consider this when implementing a REMP (especially if the facility obtains drinking water from wells located down gradient from the site). In some situations, licensees should consider onsite monitoring with respect to NUREG-1301/1302, Section 3/4.12.2, "Land Use Census," Control 3.12.2. Action "a" of that control specifies that new locations be reported in the Annual Radioactive Effluent Release Report when the doses are higher than those at the current location. Action "b" of that control requires revising the REMP (and reporting to the NRC) if the licensee identifies a location yielding a dose or dose commitment that is 20 percent larger than the dose at locations from which samples are currently being taken. For example, where liquid effluents are stored in onsite ponds, and evaporation from those ponds may					

Table A1-13: Regulatory Guides (Division 4)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	contribute to the inhalation pathway for residents at locations not currently identified in the REMP, licensees may need to evaluate the potential impact on the REMP. This could also apply in situations in which ground water transports seepage containing radionuclides from such onsite ponds to an offsite surface water body where commercial or recreational fishing is allowed. In addition, sites that contain onsite independent spent fuel storage installations (ISFSIs) may include onsite REMP samples.					
RG 4.1.C.3	RG 4.1.C.3. Routinely Monitored Exposure Pathways a. Figure 1 shows the three exposure pathways (i.e., inhalation, ingestion, and direct radiation) that are routinely monitored, along with some related characteristics. Each of the three exposure pathways consists of one or more routes of exposure. For example, inspection of Figure 1 for “liquid effluents” reveals three types of sample media associated with the ingestion exposure pathway. Each of these media is involved in a different route by which radioactive material may be transferred from the environment to an individual (causing an exposure). These routes of exposure are identified based on site-specific information (e.g., receptors, receptor locations, distances, directions, and water usage) identified during the land use census. b. Using the results of the land use census, each site should develop, implement, and maintain a site-specific REMP as outlined in NUREG-					

Table A1-13: Regulatory Guides (Division 4)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	1301/1302. Some exposure pathways (e.g., inhalation) are considered to exist at all sites (because air, the transfer media, exists at all sites). Therefore, the REMP should always include air samples (e.g., particulate filters and charcoal cartridges). Other exposure pathways and routes of exposure do not exist at all sites. For example, the REMP includes drinking water samples only if drinking water sources are present and are likely to be affected by effluents.					
RG 4.1.C.4	RG 4.1.C.4. New Routes of Exposure a. If the facility is modified or otherwise changed in a manner that results in the creation of a new point of release for radioactive material, the new release point could potentially impact a receptor or receptor location associated with the existing REMP. Alternatively, plant modifications may result in the discovery of a previously unidentified effluent release point that could impact the REMP. For example, installation of a new liquid effluent settling pond located some distance away from the center of the facility may create a new release point and cause a change to the nearest maximum exposed individual identified in the most recent land use census. This change to the maximum exposed individual may be associated with (1) the receptor, (2) the receptor location, (3) the distance to the receptor, or (4) the direction of the receptor. New routes of exposure, new sample locations, or new receptor locations may result from a radioactive leak or spill. If conditions at a site create a new route of exposure, or alter the parameters associated with an existing route of					

Table A1-13: Regulatory Guides (Division 4)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	exposure, the REMP may recognize such changes after the change has occurred through environmental sample results that are different than expected (or different than those identified in the preoperational REMP). With REMP sample results different than expected (or different than those identified in the preoperational REMP), licensees may refer to the actions in NUREG-1301/1302, Controls 3.12.1 and 3.12.2. This reactive approach is an appropriate and acceptable manner to conduct a REMP and, historically has been the basis of the REMP. At the same time, operating experience indicates that, if licensees take a more proactive approach to the REMP, they may realize program improvements and reduce regulatory involvement, including regulatory actions. Such a proactive approach includes recognizing how changes, modifications, or operational occurrences at the facility could affect the REMP with respect to actions described in NUREG-1301/1302, Controls 3.12.1 and 3.12.2.					
RG 4.1.C.5	RG 4.1.C.5. Sample Media a. Figure 1 lists some common sample media associated with various exposure pathways. In general, sample media should be selected for environmental monitoring as outlined in NUREG-1301/1302. The REMP need only include sample media that actually exists at a site and are utilized in sufficient quantities (consider availability and usage/consumption factors). However, if the site-specific land use census identifies a new important route of exposure that contributes more than 20% to the calculated individual dose as determined by					

Table A1-13: Regulatory Guides (Division 4)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Regulatory Guide 1.109, then sample media associated with the route of exposure should be added to the REMP b. TLDs or other equivalent devices should be used to monitor direct radiation exposure as outlined in NUREG-1301/1302. These results may be used, as outlined in the offsite dose calculation manual (ODCM), in demonstrating compliance with the 40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations” (Ref. 9), dose limits for members of the public in the unrestricted area. c. The sample media associated with the inhalation exposure pathway should be monitored as outlined in NUREG-1301/1302. d. The sample media associated with the ingestion and ground plane exposure pathways for liquid effluents should be monitored as outlined in NUREG-1301/1302. Because Table 3.12-1 in NUREG-1301/1302 specifies ground water and drinking water separately, the REMP should include separate sampling and analysis for ground water and drinking water if they are likely to be affected by liquid releases. For example, if liquid effluents are discharged to a local pond, the licensee should evaluate whether ground water is likely to be affected and take the appropriate action. e. The sample media associated with the ingestion pathway for gaseous releases should be					

Table A1-13: Regulatory Guides (Division 4)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	monitored as outlined in NUREG-1301/1302. This includes sampling and analyzing sample media (i.e., milk, or if milk is not available, broad leaf vegetation) and other sample media if identified in accordance with Section 5.a above. For example, sampling of domesticated meat may be needed if the land use census shows that a significant amount of meat is raised locally, and an evaluation shows that meat consumption contributes a 20% dose increment to the total individual dose. Similarly, sampling meat from game animals may be necessary if hunting accounts for a significant amount of meat obtained for consumption (see usage factors in Regulatory Guide 1.109). If goat milk is produced locally (e.g., within 5 miles or 8 km) for human consumption, then sampling and analysis may be required if sufficient quantities are available for sampling purposes. However, if sufficient quantities are not available for sampling, then an alternate sample media should be sampled such as broad leaf vegetation. f. Sample media other than those identified in Figure 1 that have special local interest or are otherwise locally important should be evaluated for inclusion in the REMP. In these instances, from a regulatory perspective, the licensee need demonstrate only that the sample media and receptor location associated with the route(s) of exposure are appropriately evaluated (e.g., by using the criteria of NUREG-1301/1302, Control 3.12.2). g. Control stations should be established and					

Table A1-13: Regulatory Guides (Division 4)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	clearly distinguished from indicator stations for use in correlating control and indicator station results as specified in NUREG-1301/1302.					
RG 4.1.C.6	RG 4.1.C.6. Sampling and Analysis Schedule a. The environmental sampling and analysis program should be conducted at the frequencies specified in NUREG-1301/1302, unless otherwise evaluated and justified. The justification for deleting samples from the REMP should be based on the results of the local land use census (e.g., the census indicates the absence or unavailability of the sample media) or as otherwise justified. The deletion of sample media from the REMP should be rare. Reduction in sample frequency may be appropriate if it is shown that the reduction does not impact the effectiveness of the REMP. Advances in remote telemetry of some air samplers may provide sufficient justification for reducing the frequency of air samples. For example, it may be appropriate to reduce the frequency of analysis associated with an air sampler (from once per week to once per 2 weeks) if the licensee can demonstrate that the new equipment is more reliable and results in fewer “missed samples” Changes to the sampling and analyses program can also be made based on operational experience. b. In all cases where sample or analysis frequencies are reduced, the changes should not					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	reduce the overall effectiveness of the environmental monitoring program. If sample or analysis frequencies are reduced, the justification should also include an evaluation showing that the increased sampling interval does not impact the ability to detect radionuclides (e.g., because of half-life considerations). The licensee should document and report the basis for changes to the environmental monitoring program in the Annual Radiological Environmental Operating Report.					
RG 4.1.C.7	RG 4.1.C.7. Analytical Detection Capabilities a. Sample analysis should employ analytical techniques so that an appropriate analytical sensitivity (e.g., a priori LLD) is achieved, as specified in NUREG-1301/1302. Alternately, licensees may use the analytical detection sensitivities as determined by the licensee based on the “Multi- Agency Radiological Laboratory Analytical Protocols Manual” (MARLAP) (Ref. 10). Selection of values different from those in NUREG-1301/1302 should be justified and documented. b. The specified detection capabilities are normally achievable for routine environmental measurements. Deviations from the a priori analytical sensitivity levels are anticipated during actual sample analyses because of interference from other radionuclides or other factors but should be evaluated and documented. Licensees should report the analytical sensitivity capabilities of the REMP in the Annual Radiological Environmental Operating Report.					

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	c. If a licensee has reduced a sample frequency as outlined in Position 6 above, the licensee should evaluate the impact on the analytical detection capability. d. Analyses for C-14 in environmental media are not required since the plant produced component is a small fraction of the naturally occurring C-14.					
RG 4.1.C.8	RG 4.1.C.8. Deviations from the Radiological Environmental Monitoring Program a. Deviations from the sampling schedule are permitted if samples are unobtainable because of hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment, and other legitimate reasons. Legitimate reasons with respect to seasonal unavailability may include consideration of the migratory routes of species or the growing season of crops but should generally not include unavailability which is within the control of the licensee. Similarly, sample pump failures that occur at an unacceptably high rate or pump failure that continues to occur because of ineffective corrective actions would not be legitimate reasons for sample unavailability. Hurricanes, tornadoes, or floods may qualify as legitimate reasons for temporary sample unavailability. b. If samples are unobtainable because of sampling equipment malfunction, reasonable effort under the circumstances should be made to complete corrective action before the end of the next sampling period, or else compensatory					

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	sampling and analysis are required. The Annual Radiological Environmental Operating Report should document deviations from the sampling schedule. Deviations that should be reported include loss of sample media, inability of the sample to meet the analytical sensitivity (e.g., the sample volume was too low to provide adequately sensitive analysis results), or invalid analyses results. Minor deviations, such as use of alternate sampling media and short periods of missed collection time, do not need to be reported.					
RG 4.1.C.9	RG 4.1.C.9. Land Use Census a. Land use, exposure pathways, and the mechanisms of exposure may change over the operating life of the plant. The REMP should contain provisions to identify changes in land use, and based on this information, the licensee should revise the REMP as necessary to identify and evaluate the site-specific parameters identified in NUREG-1301/1302 (e.g., receptors and receptor locations). Licensees should refer to NUREG-1301/1302, Section 3/4.12.2, “Land Use Census,” for additional information, including the distance over which the land use census is conducted. b. In accordance with NUREG-1301/1302, a land use census should be conducted, typically annually during the growing season, to identify (1) changes in land use, (2) receptor locations, and (3) new exposure pathways (or route of exposure). Monitoring of vegetation at the site boundary can be performed in lieu of the garden census as identified in NUREG-1301/1302. The frequency of					

Table A1-13: Regulatory Guides (Division 4)						
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	the land use census may be reduced provided that (1) the frequency is outlined in the procedures (e.g., the ODCM and related procedures), (2) the licensee can demonstrate that there is no reduction in the effectiveness of the REMP, and (3) persons knowledgeable in land use census monitor usage characteristics based on knowledge gained during routine sample collection.					
RG 4.1.C.10	RG 4.1.C.10. Reporting Levels a. The results of the REMP must be evaluated using the reporting levels in NUREG-1301/1302. When applying reporting levels, licensees may use the average of the measured radionuclide concentrations during the quarterly period. The values selected for the reporting levels approximate the concentrations equivalent to the design objectives of Appendix I to 10 CFR Part 50 for the given pathway or media. If the reporting levels are exceeded, the licensee must submit a special report to the NRC. b. If a principal radionuclide (as determined in accordance with Regulatory Guide 1.21) is detected in an environmental sample and if NUREG-1301/1302 does not provide a corresponding reporting level, licensees should calculate a reporting level for that radionuclide. The basis for that calculation should be to approximate compliance with the numerical guides of Appendix I to 10 CFR Part 50. c. If it can be demonstrated that a detected radionuclide exceeding the reporting level is not					

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	the result of 10 CFR Part 50 licensed operation (e.g., from medical radioisotopes or by comparison with control station or preoperational data), a report need not be submitted. However, the licensee should describe this occurrence in the Annual Radiological Environmental Operating Report, as discussed in NUREG-1301/1302.					
RG 4.1.C.11	RG 4.1.C.11. Annual Radiological Environmental Operating Report a. An Annual Radiological Environmental Operating Report must be prepared and must include measurement summaries and trends regarding radiation and radioactive materials in the local environment. The report should include a summary description of the REMP, a map of indicator locations keyed to a table giving distances and directions from the reactor or site centerline, any changes identified in the land use census, measurements (i.e., indicator, control, and quality control), and trends in the measurements of levels of radiation and radioactive materials in the environment and other such information as NUREG-1301/1302 may specify. NUREG-1301/1302 provides more guidance on preparing the Annual Radiological Environmental Operating Report. b. This annual report should summarize the environmental data in the format specified in NUREG-1301/1302. Data should be evaluated to identify the levels of plant-related environmental radioactivity above background levels (i.e., plant-related contributions that are distinguishable from					

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	background). For data distinguishable from background levels, a comparison should be made of current environmental monitoring results with preoperational data as appropriate and previous operational measurements for the purpose of trending environmental radioactivity resulting from licensed plant operation. c. In addition, in cases where plant-related activity is detected in the environment (e.g., tritium discharged to lakes or ponds or subsurface ground water), a basic correlation should be made between predicted and measured environmental concentrations. The purpose is to determine the adequacy of the effluent measurements and dispersion modeling. In cases where plant-related activity in the environment is increasing, the impact of prior year effluent releases should be factored into the correlations to determine if the rate of increase is commensurate with plant effluents. d. For direct radiation, the direct measurement data (e.g., TLD data) should be evaluated to determine if there is a dose contribution from plant operation. For plants with onsite sources of radiation (e.g., ISFSI or low-level waste storage) that cause measurable changes in REMP TLDs, trend graphs may be appropriate to demonstrate the change in radiation levels from the preoperational (and previous operational) REMP results to current time periods. e. The Annual Radiological Environmental Operating Report for the previous calendar year					

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	should be submitted electronically or as a hard copy to the director of the NRC regional office (with a copy to the Director, Office of Nuclear Reactor Regulation) as a separate document by May 15 each year (unless otherwise specified in technical specifications or the ODCM). Note that the period of the first report should begin with the date of initial criticality and end on December 31.					
RG 4.1.C.12	RG 4.1.C.12. Environmental Program Review a. A periodic environmental program review should be conducted to reexamine the adequacy and effectiveness of the REMP to achieve its objectives. The review can be performed during preparation of the Annual Radiological Environmental Operating Report. b. The program review should evaluate the need to expand (or reduce) the environmental monitoring program given the results of the environmental data and trends in environmental radioactivity. Note that any reductions must be thoroughly evaluated and justified, given that environmental data indicating the absence of plant-related radioactivity are important. c. The review should confirm exposure pathways and sampling media. d. The review should ensure that the principal radionuclides being discharged are the same nuclides being analyzed in the environmental program.					

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	e. The review should identify whether the land use census is able to identify potential changes in exposure pathways (e.g., new drinking water locations or irrigation systems in use). f. The review should examine 10 CFR 50.75(g) files to identify leaks, spills, or other events that could affect radioactivity levels in the unrestricted area. g. The review should identify any REMP changes or special studies that may be needed as a followup to evaluations made when comparing effluent and environmental program results. h. The review should evaluate whether the sampling and measurement techniques meet the objectives of the REMP					
RG 4.2 Rev. 2	Preparation of Environmental Reports for Nuclear Power Stations (07/01/1976)					REFER TO THE TEXT OF THE REGULATORY GUIDE. REGULATORY POSITIONS IN THE GUIDE ARE VOLUMINOUS. IN ADDITION, THE RG HAS ATTACHED TABLES AND APPENDICES INCLUDING GRAPHICS THAT CANNOT BE INCORPORATED IN THIS TABLE.

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 4.2S1	Supplement 1 to Regulatory Guide 4.2, Preparation of Supplemental Environmental Reports for Applications To Renew Nuclear Power Plant Operating Licenses (09/01/2000)	NA				Exclude; OL Renewal only
RG 4.3	Withdrawn (12/01/1976)	NA				Exclude; Withdrawn
RG 4.4	Reporting Procedure for Mathematical Models Selected To Predict Heated Effluent Dispersion in Natural Water Bodies (05/01/1974)					REFER TO THE TEXT OF THE REGULATORY GUIDE. THE REGULATORY POSITIONS ARE IN THE FORM OF DETAILED REFERENCES TO A MODEL ASSESSMENT TABLE AND AN APPENDIX TO THE GUIDE.
RG 4.5	Measurements of Radionuclides in the Environment--Sampling and Analysis of Plutonium in Soil (Withdrawn 10/01/2009)	NA				Exclude; Withdrawn
RG 4.6	Measurements of Radionuclides in the Environment-- Strontium-89 and Strontium-90 Analyses (Withdrawn 10/01/2009)	NA				Exclude; Withdrawn

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RG 4.7 Rev. 2	General Site Suitability Criteria for Nuclear Power Stations (04/01/1998)					IN ADDITION TO THE REGULATORY POSITIONS INCLUDED IN THIS TABLE, SEE THE TEXT OF THE GUIDE ITSELF, FOR WHICH SECTION B HAS EXTENSIVE DISCUSSION THAT IS RELEVANT. IN ADDITION, THERE ARE TWO SUBSTANTIVE APPENDICES TO THE GUIDE.
RG 4.7.C.1	RG 4.7.C.1. GEOLOGY AND SEISMOLOGY Preferred sites are those with a minimal likelihood of surface or near-surface deformation and a minimal likelihood of earthquakes on faults in the site vicinity (within a radius of 8 km (5 miles)). Because of the uncertainties and difficulties in mitigating the effects of permanent ground displacement phenomena such as surface faulting or folding, fault creep, subsidence or collapse, the NRC staff considers it prudent to select an alternative site when the potential for permanent ground displacement exists at the site. Sites located near geologic structures, for which at the time of application the data base is inadequate to determine their potential for causing surface deformation, are likely to be subject to a longer licensing process in view of the need for extensive and detailed geologic and seismic investigations of the site and surrounding region and for the rigorous analyses of the site-plant combination.					

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	Sites with competent bedrock generally have suitable foundation conditions. In regions with few or no such sites, it is prudent to select sites with competent and stable solid soils, such as dense sands and glacial tills. Other materials may also provide satisfactory foundation conditions, but a detailed geologic and geotechnical investigation would be required to determine static and dynamic engineering properties of the material underlying the site in accordance with 10 CFR 100.23.					
RG 4.7.C.2	RG 4.7.C.2. ATMOSPHERIC EXTREMES AND DISPERSION As noted in the Discussion Section of this guide, site atmospheric conditions are site suitability characteristics, principally with respect to the calculation of radiation doses resulting from the release of fission products as a consequence of a postulated accident. Accordingly, each applicant for site approval should collect meteorological information for at least one year that is representative of the site conditions, including wind speed, wind direction, precipitation, and atmospheric stability. Nonradiological atmospheric considerations such as local fogging and icing, cooling tower drift, cooling tower plume lengths, and plume interactions between cooling tower plumes, as well as plumes from nearby industrial facilities, should be considered in evaluating the suitability of potential sites. The atmospheric data necessary					

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	for the assessment of nonradiological considerations are described in Regulatory Guide 1.23, "Onsite Meteorological Programs."					
RG 4.7.C.3	RG 4.7.C.3. EXCLUSION AREA AND LOW POPULATION ZONE An applicant for a reactor license is required by 10 CFR Part 100 to designate an exclusion area and to have authority to determine all activities within that area, including removal of personnel and property. Transportation corridors such as highways, railroads, and waterways are permitted to traverse the exclusion area provided (1) these are not so close to the facility as to interfere with normal operation of the facility and (2) appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway in the case of emergency to protect the public health and safety. According to 10 CFR 50.34(a)(1)(ii)(D)(1), the exclusion area must be of such a size that an individual assumed to be located at any point on its boundary would not receive a radiation dose in excess of 25 rem total effective dose equivalent					

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	(TEDE) over any two-hour period following a postulated fission product release into the containment. An applicant is also required by 10 CFR Part 100 to designate an area immediately beyond the exclusion area as a low population zone (LPZ). The size of the LPZ must be such that the distance to the nearest boundary of a densely populated center containing more than about 25,000 residents ("population center distance") must be at least one and one-third times the distance from the reactor to the outer boundary of the LPZ. The boundary of the population center should be determined upon consideration of population distribution, not political boundaries. According to 10 CFR 50.34(a)(1)(ii)(D)(2), the LPZ must be of such a size that an individual located on its outer radius for the course of the postulated accident (assumed to be 30 days) would not receive a radiation dose in excess of 25 rem TEDE.					
RG 4.7.C.4	RG 4.7.C.4. POPULATION CONSIDERATIONS As stated in 10 CFR 100.21(h), "Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable."					

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	Locating reactors away from densely populated centers is part of the NRC's defense-in-depth philosophy and facilitates emergency planning and preparedness as well as reducing potential doses and property damage in the event of a severe accident. Numerical values in this guide are generally consistent with past NRC practice and reflect consideration of severe accidents, as well as the demographic and geographic conditions characteristic of the United States. Preferably a reactor would be located so that, at the time of initial site approval and within about 5 years thereafter, the population density, including weighted transient population, averaged over any radial distance out to 20 miles (cumulative population at a distance divided by the circular area at that distance), does not exceed 500 persons per square mile. A reactor should not be located at a site whose population density is well in excess of the above value. If the population density of the proposed site exceeds, but is not well in excess of the above preferred value, the analysis of alternative sites should pay particular attention to alternative sites having lower population density. However, consideration will be given to other factors such as safety, environmental, or economic considerations, which may result in the site with the higher population density being found acceptable. Examples of such factors include, but are not limited to, the higher population density site having superior seismic characteristics, better rail or					

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	highway access, shorter transmission line requirements, or less environmental impact upon undeveloped areas, wetlands, or endangered species. The transient population should be included for those sites where a significant number of people (other than those just passing through the area) work, reside part-time, or engage in recreational activities and are not permanent residents of the area. The transient population should be taken into account for site evaluation purposes by weighting the transient population according to the fraction of time the transients are in the area. Projected changes in population within about 5 years after initial site approval should be evaluated for the proposed site and any alternative sites considered. Population growth in the site vicinity after initial site approval is normal and expected and will be periodically factored into the emergency plan for the site, but population increases after initial site approval will not be a factor in license renewal or, by itself, used to impose other license conditions or restrictions on an operating plant.					

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RG 4.7.C.5	RG 4.7.C.5. EMERGENCY PLANNING As stated in 10 CFR 100.21(g), "Physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans must be identified." An examination and evaluation of the site and its vicinity, including the population distribution and transportation routes, should be conducted to determine whether there are any characteristics that would pose a significant impediment to taking protective actions to protect the public in the event of emergency. Special population groups, such as those in hospitals, prisons, or other facilities that could require special needs during an emergency, should be identified. Physical characteristics of the proposed site that could pose a significant impediment to taking protective measures, such as egress limitations from the area surrounding the site, should be identified. An evacuation time estimate (ETE) should be performed to estimate the time periods that would be required to evacuate various sectors of the plume exposure emergency planning zone (EPZ), including the entire EPZ. The ETE is an emergency planning tool that assesses, in an organized and systematic fashion, the feasibility of					

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	taking protective measures for the population in the surrounding area. Information on performing an ETE analysis is given in Appendix 4 to NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (November 1980).2 The value of the ETE analysis is in the methodology required to perform the analysis rather than in the calculated ETE times. While lower ETEs may reflect favorable site characteristics from an emergency planning standpoint, there is no minimum required evacuation time in the regulations that an applicant has to meet.					
RG 4.7.C.6	RG 4.7.C.6. SECURITY PLANS According to 10 CFR 100.21(f), "Site characteristics must be such that adequate security plans and measures can be developed." Also, 10 CFR Part 73 describes physical protection requirements for nuclear power plants as well as special nuclear materials. Generally, a distance of about 110 meters (360 feet) to any vital structure or vital equipment would provide sufficient space to satisfy security measures of 10 CFR 73.55 (e.g., protected area barriers, detection equipment, isolation zones, vehicle barriers). If the distance to a vital structure or vital equipment is less than about 110 meters (360 feet), special measures or analyses may be needed to show that adequate security plans can be developed.					

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RG 4.7.C.7	RG 4.7.C.7. HYDROLOGY 7.1 Flooding To evaluate sites located in river valleys, on flood plains, or along coastlines where there is a potential for flooding, the site suitability studies described in Regulatory 1.59, "Design Basis Floods for Nuclear Power Plants,"'2 should be made. 7.2 Water Availability A highly dependable system of water supply sources must be shown to be available under postulated occurrences of natural and site-related accidental phenomena or combinations of such phenomena as discussed in Regulatory Guide 1.59. To evaluate the suitability of sites, there should be reasonable assurance that permits for consumptive use of water in the quantities needed for a nuclear power plant of the stated approximate capacity and type of cooling system can be obtained by the applicant from the appropriate State, local, or regional agency. 7.3 Water Quality The potential impacts of nuclear power stations on water quality are likely to be acceptable if effluent limitations, water quality criteria for receiving waters, and other requirements promulgated					

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	pursuant to the Federal Water Pollution Control Act are applicable and satisfied. The criteria in 10 CFR Parts 20 and 50 will be used by the NRC staff for determining permissible concentrations of radioactive materials discharged to surface water or to ground water. 13 7.4 Fission Product Retention and Transport To be able to assess fission product retention and transportation via ground water, the following information should be determined for the site: • Soil, sediment, and rock characteristics (e.g., volcanic ash, fractured limestone), • Absorption and retention coefficients for radioactive materials, • Ground-water velocity, and • Distance to nearest body of surface water. This information should be used in the environmental report required in 10 CFR Part 51 and compared to the hydrological information used in the PRA or other analyses for a certified plant design (if such a design is to be located at the site) or used in the site-specific PRA for a custom plant located at the site. Aquifers that are or may be used by large populations for domestic, municipal, industrial, or irrigation water supplies provide potential pathways for the transport of radioactive material to man in the event of an accident. To evaluate the suitability of proposed sites located over such					

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	aquifers, detailed studies of factors identified in Section 2.4.13 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,'"2 should be completed.					
RG 4.7.C.8	RG 4.7.C.8. INDUSTRIAL, MILITARY, AND TRANSPORTATION FACILITIES According to 10 CFR 100.21(e), "Potential hazards associated with nearby transportation routes, industrial and military facilities must be evaluated and site parameters established such that potential hazards from such routes and facilities will pose no undue risk to the type of facility proposed to be located at the site." The acceptability of a site would depend on establishing that (1) an accident at a nearby industrial, military, or transportation facility would not result in radiological consequences that exceed the dose specified in 10 CFR 50.34, or (2) the accident poses no undue risk because it is sufficiently unlikely to occur (less than about 10-7 per year), or (3) the nuclear power station can be designed so its safety will not be affected by the accident.					

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	Potentially hazardous facilities and activities within 8 km (5 mi) of a proposed site, and major airports within 16km (10 mi) of a proposed site, should be identified. If a preliminary evaluation of potential accidents at these facilities indicates that the potential hazards from shock waves and missiles approach or exceed those of the design basis tornado for the region or there are potential hazards such as flammable vapor clouds, toxic chemicals, or incendiary fragments, the suitability of the site should be determined by detailed evaluation of the degree of risk imposed by the potential hazard. The design basis tornado is described in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."2 The identification of design basis events resulting from the presence of hazardous materials or activities in the vicinity of a nuclear power station is acceptable if the design basis events include each postulated type of accident for which a realistic estimate of the probability of occurrence of doses in excess of the value specified in 10 CFR 50.34(a)(1) exceeds approximately **10-7** per year. Because of the difficulty of assigning precise numerical values to the probability of occurrence of the types of potential hazards generally considered in determining the acceptability of sites for nuclear stations, judgment must be used as to the acceptability of the overall risk presented by an event. In view of the low-probability events under consideration, the probability of occurrence of					

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	initiating events leading to potential consequences in excess of the dose specified in 10 CFR 50.34(a)(1) should be based on assumptions that are as realistic as is practicable. Because of the low-probability events under consideration, valid statistical data are often not available to permit accurate quantitative calculation of probabilities. Accordingly, a conservative calculation showing that the probability of occurrence of doses in excess of the value specified in 10 CFR 50.34(a)(1) is approximately 10-6 per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower. The effects of design basis events have been appropriately considered if analyses of the effects of those accidents on the safety-related features of a proposed nuclear station have been performed and appropriate measures (e.g., hardening, fire protection) to mitigate the consequences of such events have been taken.					
RG 4.7.C.9	RG 4.7.C.9. **ECOLOGICAL SYSTEMS AND** BIOTA The ecological systems and biota at potential sites and their environs should be sufficiently well known to allow reasonably certain predictions that there would be no unacceptable or unnecessary deleterious impacts on populations of important species or on ecological systems with which they are associated from the construction or operation of a nuclear power station at the site.					

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	When early site inspections and evaluations indicate that critical or exceptionally complex ecological systems will have to be studied in detail to determine the appropriate plant designs, proposals to use such sites should be deferred unless sites with less complex characteristics are not available. It should be determined whether any important species (as defined in the Discussion section of this guide under Ecological Systems and Biota) inhabit or use the proposed site or its environs. If so, the relative abundance and distribution of their populations should be considered. Potential adverse impacts on important species should be identified and assessed. The relative abundance of individuals of an important species inhabiting a potential site should be compared to available information in the literature concerning the total estimated local population. Any predicted impacts on the species should be evaluated relative to effects on the local population and the total population of the species. The destruction of, or sublethal effects on, a number of individuals that would not adversely affect the reproductive capacity and vitality of a population or the crop of an economically important harvestable population or recreationally important population should generally be acceptable, except in the case of certain endangered species. If there are endangered or threatened species at a site, the potential effects should be evaluated relative to the impact on the local population and the total estimated population over the entire range of the					

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	species as noted in the literature. It should be determined whether there are any important ecological systems at a site or in its environs. If so, determination should be made as to whether the ecological systems are especially vulnerable to change or if they contain important species habitats, such as breeding areas (e.g., nesting and spawning areas), nursery, feeding, resting, and wintering areas, or other areas of seasonally high concentrations of individuals of important species. Important considerations in balancing costs and benefits include the uniqueness of a habitat or ecological system within the region under consideration, the amount of the habitat or ecological system destroyed or disrupted relative to the total amount in the region, and the vulnerability of the reproductive capacity of important species populations to the effects of construction and operation of the station and ancillary facilities. If sites contain, are adjacent to, or may impact on important ecological systems or habitats that are unique, limited in extent, or necessary to the productivity of populations of important species (e.g., wetlands and estuaries), they cannot be evaluated as to suitability for a nuclear power station until adequate assessments for the reliable prediction of impacts have been completed and the facility design characteristics that would satisfactorily mitigate the potential ecological					

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	impacts have been defined. In areas where reliable and sufficient data are not available, the collection and evaluation of appropriate seasonal data may be required. Migrations of important species and migration routes that pass through the site or its environs should be identified. Generally, the most critical migratory routes relative to nuclear power station siting are those of aquatic species in water bodies associated with the cooling systems. Site conditions that should be identified and evaluated in assessing potential impacts on important aquatic migratory species include (1) narrow zones of passage, (2) migration periods that are coincident with maximum ambient temperatures, (3) the potential for major modification of currents by station structures, (4) the potential for increased turbidity during construction, and (5) the potential for entrapment, entrainment, or impingement by or in the cooling water system or for blocking of migration by facility structures or effluents. The potential for blockage of movements of important terrestrial animal populations caused by the use of the site for a nuclear power station and the availability of alternative routes that would provide for maintenance of the species' breeding population should be assessed. If justifiable relative to costs and benefits, the potential impacts of plant construction and operation on the biota and ecological systems can generally be mitigated by adequate engineering					

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	design and site planning and by proper construction and operations when there is adequate information about the vulnerability of the important species and ecological systems. A summary of environmental considerations, parameters, and regulatory positions for use in evaluating sites for nuclear power stations is provided in Appendix B to this guide.					
RG 4.7.C.10	RG 4.7.C.10. LAND USE AND AESTHETICS Land use plans adopted by Federal, State, regional, or local agencies should be examined, and any conflict between these plans and use of a potential site should be resolved by consultation with the appropriate agency. For a potential site on land devoted to specialty crop production where changes in land use might result in market dislocations, a detailed investigation should be provided to demonstrate that potential impacts have been identified. The potential aesthetic impact of nuclear power stations at sites near natural-resource-oriented public use areas is of concern, and evaluation of such sites is dependent on consideration of specific station design layout.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 4.7.C.11	RG 4.7.C.11. SOCIOECONOMICS The NRC staff considers that an evaluation of the suitability of nuclear power station sites near distinctive communities should demonstrate that the construction and operation of the nuclear station, including transmission and transportation corridors, and potential problems relating to community services, such as schools, police and fire protection, water and sewage, and health facilities, will not adversely affect the distinctive character of the community nor disproportionately affect minority or low-income populations. A preliminary investigation should be made to address environmental justice considerations and to identify and analyze problems that may arise from the proximity of a distinctive community to a proposed site.					
RG 4.7.C.12	RG 4.7.C.12. NOISE Noise levels at proposed sites must comply with applicable Federal, State, and local noise regulations.					
RG 4.8	Environmental Technical Specifications for Nuclear Power Plants (for Comment). (Withdrawn 06/01/2009)	NA				Exclude; Withdrawn

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 4.9 Rev. 1	Preparation of Environmental Reports for Commercial Uranium Enrichment Facilities (10/01/1975)	NA				Exclude; for uranium enrichment facilities.
RG 4.10	Withdrawn (11/01/1977)	NA				Exclude; Withdrawn
RG 4.11 Rev. 1	Terrestrial Environmental Studies for Nuclear Power Stations (08/01/1977)					NOTE THAT THERE IS ALSO EXTENSIVE MATERIAL IN SECTION B OF THE REGULATORY GUIDE PRECEDING THE REGULATORY POSITIONS WHICH ARE INCLUDED HERE.
RG 4.11.C.1	RG 4.11.C.1. It is important to coordinate all the programs discussed in Regulatory Guides 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants," and 4.2, "Preparation of Environmental Reports for Nuclear Power Stations." Since precise predictions and assessments of impacts on terrestrial ecological systems are not always possible, reasonable professional interpretations should be made when quantitative prediction is impossible.					
RG 4.11.C.2	RG 4.11.C.2. Adequate assessment of current land-use status should show (by a table, for example) major land-use categories and areas devoted to each category along with aerial photographs showing the same categories. When data are not available from existing records, an acceptable means of acquiring them would be through the use of aerial photographs in conjunction with ground reconnaissance. The scale of photographs should be appropriate to the degree of detail required. Federal, State, regional, and local planning authorities should be consulted					

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	to determine the existence of present or planned areas dedicated to the public interest or in which siting would be in conflict with preexisting zoning plans. Such contacts should be documented.					
RG 4.11.C.3	RG 4.11.C.3. Discussion of soils should include association names, capability classes,7 and percentage of site coverage by each association. When numerous associations of minor extent are present, it is acceptable to account for 10 to 15% of the total area in a miscellaneous category, except for areas of unique value. Detailed consideration of soils and their production potential is necessary for sites located in areas that are especially productive of agricultural or forestry products.					
RG 4.11.C.4	RG 4.11.C.4. Biological monitoring programs should be initially devised to be screening procedures to detect undesirable effects. If adverse biological effects are detected, detailed quantitative biological and ecological analyses may be required to determine causes and to devise remedies. If adverse effects are not detected, quantitative studies are not needed. The species inventory of the site should include important habitats and normal seasonal variations. Locally prominent or important vascular plants, mammals, birds, reptiles, amphibians, insects, and other plants and animals should be included. The inventory should be reasonably complete but may					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	be terminated when additional field effort no longer yields significant numbers of previously unobserved species. Insect surveys should provide information on important species such as disease vectors, pests, and pollinating insects. Interpretation of insect data should include consideration of the possibility of adverse consequences to animals, vegetation, or humans that might be caused by construction or operation of the station. Adverse consequences can usually be determined by consultation with State agricultural authorities. Normally, detailed field surveys of insect populations are not needed. Protection of terrestrial systems is usually adequate when it can be shown that (I) habitat losses or alterations of important species' are small with respect to the amount available within the regional or local context, (2) chemical emissions from the station are sufficiently small to permit reasonable assurance that no adverse effect will occur, and (3) no mechanism exists for causing unintended destruction of organisms, or its occurrence is infrequent enough to give reasonable assurance that whole populations will not be adversely affected. Environmental protection should be achieved by control of common sources of environmental effects. These include soil erosion, siltation, use of herbicides, dust and noise during construction, and others. Biological consequences can usually be prevented or reduced to acceptable levels through					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	proper management. If cooling towers are being considered, the mineral content of the cooling water supply should be determined in the baseline studies. An estimate should be made of the amounts and dispersion of salts expected to be deposited from the towers. The estimate should be based on the cooling water quality, manufacturer's specifications for drift release from the towers, concentration factors, and prevailing meteorological conditions at the site. Meteorological dispersion models are useful to obtain estimates of drift deposition. Estimated drift deposition from cooling towers may be plotted on a base map or graph centered on the towers and showing isopleths of salt deposition. The maps should have a radius sufficient to show the points at which the amounts of drift from the tower fall within the normal range of annual variation of background deposition from other sources. They should also show the vegetation types that occur in the drift field. Reconnaissance and inspection of biota in the drift field before and after cooling tower operation is a means recommended for detection of possible adverse effects of drift. The baseline inspection should be carried out by specialists in biology working systematically from checklists of possible adverse effects in the community. Seasonal aerial and ground-level photographs in color or infrared false color of permanent vegetation plots are often useful aids. Quantitative chemical analysis of					

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	plants, animals, and soils are needed if chemical deposits are expected to exceed toxic or injurious thresholds. Population monitoring of selected species could also be needed in such cases. The assessment of cooling lakes and transmission and access corridors should include detailed consideration of the effects of land diversion on local, regional, and State agricultural production, forest production, or recreational uses. The assessment should include both adverse and beneficial aspects. Where a cooling lake is proposed, the baseline studies should include a preliminary assessment of the potential for reclamation of the lake bottom for agricultural, ecological, or forestry use after decommissioning. It is not necessary, however, to prejudge future use of the lake site. It is sufficient to establish whether the option exists to reclaim the site for other productive uses or whether the creation of the lake constitutes an irretrievable change in land use. The assessment should also include a report of the number of hectares of the lake site that will remain undisturbed during construction, the number of hectares and vegetation that will be disturbed, the source of "borrow" material for dike construction, and the management of topsoil removed during construction. Use of topsoil stripped from the lake bottom for vegetative stabilization of dikes and for ultimate replacement on the lake bottom for rehabilitation should be considered.					

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	When soil disruption during construction at the site or in transmission corridors is expected to expose substrates or a proposed lake is to be built on substrates having a potential for affecting water quality, chemical analyses of the substrates should be performed. The elements to be measured depend on the nature of the substrate. If the substrate is formerly fertilized farmland, analysis for elements common to chemical fertilizers is needed. If the substrate is land of some special history, such as strip-mine land, appropriate chemical assessment of the water-soluble and exchangeable components of the substrate should be made to obtain an estimate of chemical input to water bodies. Special attention is given those elements that could reach toxic concentrations in water, accumulate to toxic levels in food webs, or affect the pH of water bodies. The chemical analyses should be performed on appropriate chemical extracts of the soil material. The characterization of soil material should also include determination of exchange capacity, organic matter, pH, and textural class. When a reservoir is proposed, the baseline studies should include reasonable predictions of the number of birds (especially waterfowl) expected to use the lake on an annual basis, their expected residence time, the expected impact on farmlands, and all other impacts either on the birds themselves or on the surrounding area due to their presence. The estimates should be the best obtainable based on known flyways, estimates of					

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	farm acreages nearby, literature, or local evidence of bird utilization of other reservoirs under similar conditions.					
RG 4.11.C.5	RG 4.11.C.5. Information needed for transmission and access corridor assessment is generally similnr to that for sites; however, certain considerations apply specifically to corridors. Detailed land-use information along corridors is needed. The description should include the distance transversed and locations of principal land-use types such as forests, permanent pastures, cultivated crops, parks, preserves, water bodies, recreation areas, and housing areas. Special features such as historic sites; monuments; archaeological sites; caves; mineralogical, paleontological, or geological areas of special interest; stream crossings; and road crossings should be identified and their locations specified. Information may be presented in the form of land-use maps that are keyed to descriptive text. It is often useful to subdivide long corridors into convenient segments containing similar land-use types for descriptive purposes. It is usually adequate to describe biotic communities in terms of principal vegetative associations such as oak-hickory forest. The animals most likely to be found along corridors may be determined from literature studies, local experts, or field reconnaissance. Emphasis should					

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	be placed on "important" species as defined in Regulatory Guide 4.2. Comprehensive field inventories of biota along transmission corridors are not usually needed. The potential occurrence of threatened or endangered plants and animals or their critical habitat adjacent to or within the proposed corridors should be .investigated. Local, State, and Federal authorities (e.g., the U.S. Fish and Wildlife Service and State wildlife agencies) should be consulted to determine protected species that reasonably 'could be expected to occur and the locations of possible occurrences along corridors. If potential areas are identified, field inspection of these areas may be necessary to verify, the presence or absence of the protected organisms. If proposed transmission corridors could add to the further endangerment of a protected species, realignment in the critical areas might be required.					
RG 4.11.C.6	RG 4.11.C.6. When adverse effects of construction or operation can be reasonably inferred from information obtained during the baseline phase, quantitative studies that can be compared with later studies during construction or operational phases should be initiated. Such studies include measurements of population densities of endangered species, chemical measurements of soils and biota within the potential drift field of a cooling tower, or annual aerial photography, for example. The preferred method of biological protection on					

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	many construction sites is direct control of potentially injurious work practice. Systematic inspection during construction at the site, along corridors, and in adjacent areas should be used to detect injurious or unauthorized activities. Examples of items that may be checked are: a. Traffic Control -Vehicles should be confined to authorized roadways and stream crossings. b. Dust Control -Dust should be controlled by such means as watering, graveling, or paving. Areas subject to wind erosion should be controlled by mulching, seeding, or the equivalent. c. Noise Control -Noise should be monitored at site boundaries. d. Smoke Control -Open slash burning of plant material should be conducted in accordance with local and State regulations. e. Chemical and Solid Waste Control -Cement, chemicals, fuels, sanitary wastes, lubricants, bitumens, flushing solutions, or other potentially hazardous materials should be salvaged or discharged safely in accordance with existing regulations. Spills should be cleaned up before they become a hazard. f. Soil Erosion and Sediment Control –Erosion should be controlled by piped drainage, diversion dikes, flumes, sediment control structures, ground covers, or other appropriate means. g. Dewatering -Dewatering should be confined to the area needed for construction; test wells or preexisting wells should be monitored for changes in the water table. If, after analysis of the inventory of species and					

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	consideration of potential effects of the nuclear power station, a conclusion is warranted that there will be no adverse impact on biota, there may be no need to carry out biological monitoring programs at the construction and operational stages. Special studies could be necessary if adverse effects on biota are detected and there is no obvious explanation or remedy for the effect. In the usual case, however, if habitat loss or alteration, chemical emissions, or direct destruction of organisms do not constitute a threat to a population of an important species, the effect need not be studied further.					
RG 4.12	Not published	NA				Exclude; Never Published
RG 4.13 Rev. 1	Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications (07/01/1977) The requirements and recommendations for performance specifications, testing procedures, calibration procedures, field procedures, and reporting procedures that are included in ANSI N545-1975 are generally acceptable to the NRC staff as the basis for using thermoluminescence dosimetry for the measurement of X and gamma radiation in the environs of NRC-licensed facilities subject to the following additional provisions and qualifications.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 4.13.C.1	RG 4.13.C.1. Section 3, "Performance Specifications," of ANSI N545-1975 should be supplemented by the following statement: "Subsection 4.2.4 shall apply also to the subsections 3.1 and 3.3."					
RG 4.13.C.2	RG 4.13.C.2. Instead of Section 3.1 of ANSI N545-1975, the following should be used: "The performance of the TLD system shall be determined under laboratory conditions and in a known radiation field with an exposure equal to that resulting from an exposure rate of 10 R/hr during the field cycle. Ninety-five percent of the measurements shall fall within 10% of the known exposure."					
RG 4.13.C.3	RG 4.13.C.3. Instead of Section 3.3 of ANSI N545-1975, the following should be used: "Ninety-five percent of the final values (after all appropriate corrections to the measurements are applied, including those for errors expected under field conditions) shall differ from the correct value by less than 30% of the correct value."					
RG 4.13.C.4	RG 4.13.C.4. Instead of Section 4.3.1 of ANSI N545-1975, the following should be used: "Uniformity shall be determined by giving TLDs from the same batch an exposure equal to that resulting from an exposure rate of 10 R/hr during the field cycle. The response obtained shall have a relative standard deviation (coefficient of variation) of less than 7.5%."					
RG 4.13.C.5	RG 4.13.C.5. Instead of Section 4.3.2 of ANSI N545-1975, the following should be used: "Reproducibility shall be determined by giving one TLD repeated exposures equal to that resulting from an exposure rate of 10 μR/hr during the field					

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	cycle. The responses shall have a relative standard deviation (coefficient of variation) of less than 3.0%.					
RG 4.14 Rev. 1	Radiological Effluent and Environmental Monitoring at Uranium Mills (04/01/1980)	NA				Exclude; Applicable to Uranium Mills Only
RG 4.15 Rev. 2	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) -- Effluent Streams and the Environment (07/01/2007)					REFER TO THE FULL TEXT OF THE REGULATORY GUIDE WHICH IS VOLUMINOUS AND WHICH ALSO CONTAINS EXTENSIVE SUPPLEMENTAL MATERIAL IN THE FORM OF A DETAILED GLOSSARY AND REFERENCE LIST. THE REGULATORY POSITIONS CONSIST OF THE 10 (OUT OF 18) 10 CFR 50 APPENDIX B CRITERIA THAT ADDRESS ISSUES RELEVANT TO RADIOLOGICAL MONITORING PROGRAMS.
RG 4.16 Rev. 1	Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants (including Errata, published 08/1986) (12/01/1985)	NA				Exclude; Applicable to different types of facilities than reactors

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
RG 4.17 Rev. 1	Standard Format and Content of Site Characterization Plans for High-Level-Waste Geologic Repositories (03/01/1987)	NA				Exclude; Applicable to High Level Waste Geologic Repositories Only
RG 4.18	Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste (06/01/1983)	NA				Exclude; Applicable to ERs for Disposal of Radwaste
RG 4.19	Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste (08/01/1988)	NA				Exclude; Applicable to site selection for radwaste disposal facilities only
RG 4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors (12/01/1996)	NA				Exclude; Not for power reactors
RG 4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning (06/01/2008)					NOTE THAT THE REGULATORY POSITIONS INCLUDED IN THIS TABLE CONSTITUTE ONLY A SMALL PART OF THE MATERIAL IN THE REGULATORY GUIDE ITSELF WHICH ALSO INCLUDES EXTENSIVE INTRODUCTORY DISCUSSION AND TWO APPENDICES.

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 4.21.C.1	RG 4.21.C.1.Minimizing Facility Contamination 1.1. General Statement In general, an applicant should minimize radioactive contamination of the facility by using structure, system, and component (SSC) designs and operational procedures that limit leakage and/or control the spread of contamination. The design and operational procedures should provide for the early detection of leaks thus allowing prompt assessment to support a timely and appropriate response. Applicants should note that 10 CFR 20.1406 requires that contamination be minimized "...to the extent practicable..." This implies that other competing concerns such as the implication to safety systems and the overall cost should be considered. Thus the minimization of facility contamination must be considered in the context of overall facility safety. 1.2. Minimization of Leaks and Spills and Provision of Containment Through design, worker practices, preventive maintenance, and effective operating procedures, applicants covered by 10 CFR 20.1406 should strive to minimize leaks and spills, provide containment in areas where such events might occur, and provide for detection that supports timely assessment and appropriate response. This approach should be applied in a risk-informed and performance-based manner considering the nature of the hazard. Radiologically significant leaks and					

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	spills1 need to be addressed, and containment should be considered where practical and cost effective. Areas where licensed materials are used and stored should be designed to facilitate maintenance and operations (including cleanup). Radiological work should be restricted to a localized section of the facility in order to minimize the potential area requiring decontamination. 1.3. Prompt Detection of Leakage The facility should be designed such that any SSC that has the potential for leakage is provided with adequate leak detection capability to the extent practical. In addition to design considerations to control and, if possible, prevent radioactive system leakage, it is important during operations to be able to promptly detect leakage as close as possible to the leakage source to allow timely intervention and to prevent the potential for widespread contamination. Thus, monitoring and routine surveillance programs are an important part of minimizing potential contamination. This approach should include the placement of instruments to detect leakage at readily accessible locations and the use of operational practices that will enable early detection of contamination. Because leakage detection is only the first step in minimizing contamination, the applicant also should be prepared to provide a timely assessment and response based on the location and characteristics of the leak or spill. 1.4. Avoidance of the Release of Contamination					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	from Undetected Leaks Past experience has shown that leaks of radioactive material from SSCs containing radiation can go undetected over long periods of time if these SSCs are not readily accessible for surveillance or when the amount of leaking material is below the sensitivity of the survey instrument. Under these conditions, contamination from undetected leaks can accumulate as subsurface residual radioactivity that may require remediation prior to license termination. This contamination generally occurs as minor leaks over an extended period of time. SSCs that are buried, embedded in concrete, or in contact with soil (such as spent fuel pools, underground tanks, and buried pipes) are particularly susceptible to undetected leakage. Facilities undergoing decommissioning commonly discover previously undetected contamination in the subsurface environment. These releases were generally minor leaks that occurred over an extended period of time. Many of the leaks occurred in areas where it was difficult or impossible to conduct regular inspections. This likely contributed to the failure to identify the leaks at the time of occurrence. Monitoring of some SSCs was not sufficiently sensitive to identify small leaks and leakage rates. Such situations and conditions should be avoided during facility design. Leak detection systems should be included within the facility design that are capable, to the extent practical, of detecting minor leaks that otherwise, over time, could potentially cause significant environmental					

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	contamination. 1.5. Measures for Reducing the Need To Decontaminate Equipment and Structures Leakage from components containing radioactive liquids can be reduced by: (1) the inclusion of design specifications such as the proper selection of materials (e.g., corrosion-resistant piping, double-walled pipes, and tanks with annulus monitoring); (2) improved protection of buried components (e.g., galvanic corrosion protection, coatings); (3) the use of industry consensus codes and standards for repair and/or replacement of SSCs; and (4) the application of rigorous quality control and quality assurance program requirements in procurement specifications and during installation of SSCs. 10 CFR 20.1406 applicants can decrease the probability of a release, the amount released, and the spread of a contaminant by: (1) temporary or supplemental ventilation systems, (2) treating the exhaust from vents and overflows, and (3) using techniques to control releases (i.e., capping or elevating uncontrolled drains, hard piping of drains to drain sumps, use of barriers or dikes, use of controlled sumps, and protection of SSCs from inclement weather). 1.6.Periodic Review of Operational Practices Operational practices are another important consideration in meeting the requirements of 10 CFR 20.1406. These practices should be					

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	subjected to periodic review to ensure that (1) facility personnel follow the operating procedures, (2) operating procedures are revised to reflect the installation of new or modified equipment or facility processes, and (3) personnel qualification and training are kept current with the latest versions of operational programs and procedures. Operational programs and procedures should be subjected to review and evaluation following events that resulted in leaks and spills of radioactive materials. As part of the analysis, the evaluation should determine (1) whether procedures, equipment, and/or operator errors contributed to the event and releases, and (2) identify immediate and long-term corrective actions. The results of such lessons learned should then be assessed for their broader applicability to similar or related facility operations, and then incorporated as needed into revised programs and procedures.					
RG 4.21.C.2	RG 4.21.C.2. Minimizing Contamination of the Environment 2.1. Development of a Conceptual Site Model Development In general, system design features and operational procedures that prevent and/or control releases within the facility also contribute to minimizing contamination of the environment. For systems that directly interface with the environment, the first indication of a leak may be detection in an environmental monitoring system. To control and mitigate such events, it is prudent to have a comprehensive understanding of the interface with					

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	environmental systems and the features that will control the movement of contamination in the environment. A conceptual site model based on site characterization and facility design and construction can be a significant tool in (1) understanding the site, (2) planning and implementing a contaminant monitoring program, and (3) planning and implementing mitigative actions. Therefore, the site should be characterized before construction to assess the impact that the facility will have on the site hydrogeology following construction. In addition to the conceptual site model, attention should be given to identifying the potential release mechanisms, release scenarios, and possible location of contaminant releases. 2.2. Provision for Early Detection of Leakage and Contaminant Migration Systems or structures that are buried or in contact with soil are particularly susceptible to undetected leakage. Undetected leakage commonly occurs in areas where it is not possible to conduct regular inspections; therefore, these leaks are often not identified in a timely manner. To minimize contamination of the environment, systems should be designed to facilitate early detection of leakage and contaminant migration. 2.3. Final Site Configuration Applicants covered by 10 CFR 20.1406 should consider the site configuration following					

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	construction to aid in preventing the offsite migration of radionuclides via an unmonitored pathway. 10 CFR 20.1406 applicants should identify, as early as practical in the licensing process, the potential pathways of radioactive contaminants through the surface and subsurface. Applicants covered by 10 CFR 20.1406(a) should develop an onsite monitoring program, as an integral part of the radiological environmental monitoring program. The program should provide early detection and quantification of leaks and spills and maintain a current baseline of radiological and hydrogeological parameters. Plans for responding to the detection of leaks and spills should reflect the final facility design and site configuration.					
RG 4.21.C.3	RG 4.21.C.3. Facilitation of Decommissioning 3.1. General Statement In general, the means for facilitating decommissioning begins at the design stage and should be incorporated into the procedures and operations. The objective is to ensure that throughout the life of the facility, the design and operating procedures minimize the amount of residual radioactivity that will require remediation at the time of decommissioning. 3.2. Facilitation of Decommissioning with Proper Records The provisions of 10 CFR 50.75(g) contain requirements for maintaining records "...of					

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	information important to the safe and effective decommissioning of the facility." These records are required to contain details on contaminating events and residual levels of contamination in the environment during the life of the facility. In addition, 10 CFR 30.35(g), 10 CFR 40.36(f), 10 CFR 70.25(g), and 10 CFR 72.30(d) have records retention requirements related to decommissioning. Records on events involving leaks or spills should be maintained and readily accessible to facilitate cleanup and eventual decommissioning of the facility.					
RG 4.21.C.4	RG 4.21.C.4. Minimizing the Generation of Waste Minimizing the generation of radioactive waste is both a design and operational consideration. A life-cycle approach should be taken in identifying all components used in the facility and all waste that will result from system operations and processing. Life-cycle waste management planning should also be carried out for any new waste stream to define the strategy for its conditioning, storage, or disposal. System designs should enable operators to perform decontamination efficiently while minimizing collective dose and the production of radioactive waste. 10 CFR 20.1406 applicants should evaluate design and operational options to implement measures that minimize waste generation and radioactivity levels and that fit each phase of the expected life cycle of the facility. For each phase, the implementation of such measures should consider the merits of various technological					

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	options and lessons learned from the use of earlier or similar technology; assess public health and safety, and the protection of the environment; and confirm compliance with applicable Federal, State and local regulations governing the management of radioactive waste, and wastes characterized by the presence of hazardous chemicals and radioactivity. While the measures identified in this guide focus on minimizing the generation of radioactive waste, NRC recognizes that constraints and competing factors may govern the selection of specific measures for waste minimization. In many instances, an applicant or licensee has no control over such constraints and may be forced to balance competing factors of operational flexibility and costs, while satisfying all applicable regulatory requirements at the same time. For example, access to or the availability of offsite low-level waste disposal capacity may be beyond the control of an applicant or licensee. The methods chosen to manage radioactive waste should be carefully considered for the purpose of meeting regulatory requirements for transportation and the waste acceptance criteria of specific disposal or treatment outlets. For some waste streams, a processing method that may be used to reduce the overall volume of waste might result in an increase of the specific activity of the waste, thereby increasing the difficulty in finding appropriate disposal outlets for higher activity wastes, such as Class B and C wastes under the					

Table A1-13: Regulatory Guides (Division 4)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	requirements of 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." In other instances, the amount or volume of waste is not the issue. Rather the waste's radiological and chemical properties, such as for mixed waste, which may restrict options in finding treatment and disposal outlets unless one of the hazardous properties is delisted. NRC and U.S. Environmental Protection Agency regulations control the storage of mixed wastes. Some States impose additional regulations addressing the characterization, treatment, transportation, and disposal of mixed wastes. When disposal or treatment outlets are not available, a 10 CFR 20.1406 applicant or licensee may be required to develop additional onsite storage capacity. The availability of waste disposal facilities depends on whether States or regional low-level waste compacts have provided facilities for long-term storage and disposal. For onsite storage, 10 CFR 20.1406 applicants and licensees should integrate the associated operations into existing waste management programs. They should also address decontamination and decommissioning of the storage facility and conduct periodic reassessments of waste already being stored, given that changes in future disposal requirements might possibly make stored wastes unacceptable for disposition.					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 5.1	Withdrawn (Jan-98)	NA				Withdrawn
RG 5.2	Withdrawn (Oct-79)	NA				Withdrawn
RG 5.3	Statistical Terminology and Notation for Special Nuclear Materials Control and Accountability (Feb-73)	NA				Not technically relevant
RG 5.4	Standard Analytical Methods for the Measurement of Uranium Tetrafluoride (UF4) and Uranium Hexafluoride (UF6) (Feb-73)	NA				Not technically relevant
RG 5.5	Standard Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Uranium Dioxide Powders and Pellets (Feb-73)	NA				Not technically relevant
RG 5.6	Withdrawn (Jun-85)	NA				Withdrawn
RG 5.7 Rev. 1	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas (May-80)	NA				RG implements portions of Part 73 that are NA.
RG 5.8 Rev. 1	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Drying and Fluidized Bed Operations (May-74)	NA				Not technically relevant
RG 5.9 Rev. 2	Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material (Dec-83)	NA				Not technically relevant
RG 5.10	Selection and Use of Pressure-Sensitive Seals on Containers for Onsite Storage of Special Nuclear Material (Jul-73)	NA				Not technically relevant
RG 5.11 Rev. 1	Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste (Apr-84)	NA				Not technically relevant
RG 5.12	General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials (Nov-73)					
RG 5.12.C.1.	Combination locks installed in solid doors such as those in vaults or vault-type rooms in protected areas should he three- or four-position dial-type changeable-combination locks meeting the					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Underwriters' Laboratories Standard UL-768. "Combination Locks," for Group I locks.					
RG 5.12.C.2.	Combination padlocks should be used when practicable on doors or gates to material access areas, in protected and vital area perimeters, and for access to vital equipment in preference to key padlocks. Combination padlocks should be used on dosed vehicles or containers holding SNM that are required to be locked. Combination padlocks should be three-position dial type changeable-combination padlocks meeting Federal Specification FF-P-I I OF, "Padlock, Changeable (Combination (Resistant to Opening by Manipulation and Surreptitious Attack)."					
RG 5.12.C.3.	Key locks used in lieu of combination padlocks on doors or gates to material access areas, in protected and vital area perimeters, and for access to vital equipment should provide a high degree of resistance to opening by force and tamper techniques and should meet Underwriters' Laboratories UL-437, "Key Locks."'					
RG 5.12.C.4.	Key padlocks used in lieu of combination padlocks on doors or, gates to material access areas, in protected and vital area perimeters, and for access to vital equipment should be of rugged and sturdy construction and designed for outdoor use if necessary, and should meet Interim Federal					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Specification FF-P-001480 (GSA FSS), "Padlock, Key Operated (Resistant to Opening by Force, Pick, and Bypass Techniques)."					
RG 5.12.C.5.	Electric locks should be used inside the protected area as a means of access control only if a magnetic card key systym is coupled with a pushbutton system and integrated into the alarm system. This lock combination should have features that resist tampering with the combination-changing mechanism and that alarm after a set number of errors in punching the combinations is made.					
RG 5.12.C.6.	Pushbutton mechanical locks are not recommended for use at this time because of the lack of comprehensive standards and specifications against which the locks can be evaluated.					
RG 5.12.C.7.	Mechanical locks used as panic locks on emergency exit doors within protected area perimeters should be operable only from the inside.					
RG 5.12.C.8.	Combinations, keys and locks should be controlled, protected and changed in accordance with the following requirements: a. Combinations of locks or padlocks on					

Table A1-14: Regulatory Guides (Division 5)						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	repositories containing SNM or used to secure gates or doors to material access areas, in protected and vital area perimeters, and for access to vital equipment should be known only to those authorized access to the material or to the area. They should be changed when repositories or areas are first placed in use, whenever a person knowing the combination no longer requires it as a result of reassignment of duties or termination, whenever the combination may have been compromised, or at least twice every year. A record of the combinations of locks should be kept in a location that is secured by a combination lock. b. Keys and cards to locks or padlocks on containers holding SNM or used to secure gaes or doors to material access areas and in protected and vital area perimeters should be issued only to persons authorized access to the material or to the area. Keys or cards in use should be checked in at the end of each shift or workday, and a log should be maintained showing keys and cards, users, in and out times, and other pertinent information. Keys and cards should be recovered from reassigned or terminating personnel. Locks should be immediately changed or cores replaced and an inventory conducted whenever a core, key, or card is lost or missing; the lock, care, key, or card has been compromised; or unrecorded keys or cards					

Table A1-14: Regulatory Guides (Division 5)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	are found. In a mastered system, a complete remastering of the system should be conducted whenever a core, card, master or control key, or a lock is lost or compromised. c. A record of all locks, cores, keys, and cards should be maintained and kept in a location secured by a combination lock. A physical inventory of locks, cores. keys, and cards should be conducted semiannually when the locks are used for protection of facilities and bimonthly when the locks are used for the protection of SNM. Unused locks, cores, keys, and cards should be stored in a location secured by a combination lock. A specific individual at each site should be named and placed In charge of all locks, cores, keys, and cards.					
RG 5.13	Conduct of Nuclear Material Physical Inventories (Nov-73)	NA				Deals specifically with process inventory controls such as those at enrichment, fuel fabrication, or processing facilities.
RG 5.14	Withdrawn (Jan-98)	NA				Withdrawn
RG 5.15 Rev. 1	Tamper-Indicating Seals for the Protection and Control of Special Nuclear Material (Mar-97)	NA				RG deals with acceptable ways of complying with portions of regulations that are not applicable.
RG 5.16	Withdrawn (Jun-85)	NA				Withdrawn
RG 5.17	Truck Identification Markings (Jan-74)	NA				RG deals with transportation issues and acceptable ways of complying with portions of regulations that are not applicable.
RG 5.18	Limit of Error Concepts and Principles of Calculation in Nuclear Materials Control (Jan-74)	NA				RG deals with statistical methods of calculating errors in material balances.

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 5.19	Withdrawn (Jun-85)	NA				Withdrawn
RG 5.20	Training, Equipping, and Qualifying of Guards and Watchmen (Jan-74)	NA				RG deals with acceptable ways of complying with portions of regulations that are not applicable.
RG 5.21 Rev. 1	Nondestructive Uranium-235 Enrichment Assay by Gamma Ray Spectrometry (Dec-83)	NA				For enrichment testing
RG 5.22	Assessment of the Assumption of Normality (Employing Individual Observed Values) (Apr-74)	NA				RG deals with formal statistical measures of normality used for statistical calculations in material control of processes in areas such as enrichment and processing.
RG 5.23 Rev. 1	In Situ Assay of Plutonium Residual Holdup (Feb-84)	NA				Not technically relevant
RG 5.24	Withdrawn (Jan-98)	NA				Withdrawn
RG 5.25	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Wet Process Operations (Jun-74)	NA				Not technically relevant
RG 5.26 Rev. 1	Selection of Material Balance Areas and Item Control Areas (Apr-75)	NA				Concerns control of material in material processing facilities which use item control areas and material balance areas in different parts of the facility.
RG 5.27	Special Nuclear Material Doorway Monitors (Jun-74)	NA				Concerns doorway monitors in facilities which must restrict diversion of SNM.
RG 5.28	Evaluation of Shipper-Receiver Differences in the Transfer of Special Nuclear Materials (Jun-74)	NA				Concerns methods of accounting for differences in inventory shipped and received for licensees dealing with material in a form where such inventories can have measurement differences.
RG 5.29	Withdrawn (Jan-98)	NA				Withdrawn
RG 5.30	Withdrawn (Jan-98)	NA				Withdrawn
RG 5.31 Rev. 1	Specially Designed Vehicle with Armed Guards for Road Shipment of Special Nuclear Material (Apr-75)	NA				Not technically relevant
RG 5.32 Rev. 1	Communication with Transport Vehicles (May-75)	NA				Concerns specific communication procedures and equipment used during transportation.

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 5.33	Statistical Evaluation of Material Unaccounted For (Jun-74)	NA				Concerns statistical methods for handling "Material Unaccounted For" at facilities using material balance control.
RG 5.34 Rev. 1	Nondestructive Assay for Plutonium in Scrap Material by Spontaneous Fission Detection (May-84)	NA				Not technically relevant
RG 5.35	Withdrawn (Aug-77)	NA				Withdrawn
RG 5.36	Recommended Practice for Dealing with Outlying Observations (Jun-74)	NA				Concerns statistical methods for dealing with outlying observations at facilities using material balance control.
RG 5.37 Rev. 1	In Situ Assay of Enriched Uranium Residual Holdup (Oct-83)	NA				Not technically relevant
RG 5.38 Rev. 1	Nondestructive Assay of High-Enrichment Uranium Fuel Plates by Gamma Ray Spectrometry (Oct-83)	NA				Not technically relevant
RG 5.39	General Methods for the Analysis of Uranyl Nitrate Solutions for Assay, Isotopic Distribution, and Impurity Determinations (Dec-74)	NA				Not technically relevant
RG 5.40	Withdrawn (Jun-85)	NA				Withdrawn
RG 5.41	(Not issued)	NA				Not issued
RG 5.42	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations (Jan-75)	NA				Not technically relevant
RG 5.43	Plant Security Force Duties (Jan-75)	NA				Concerns compliance with an area of 10 CFR 73 that is not applicable to reactors.
RG 5.44 Rev. 3	Perimeter Intrusion Alarm Systems (Oct-97) Note: Refer to the Regulatory Guide for detailed criteria in “Regulatory Positions” and the Attachment.					
RG 5.45	Withdrawn (Jan-98)	NA				Withdrawn
RG 5.46	(Not issued)	NA				Not issued
RG 5.47	Withdrawn (Jun-85)	NA				Withdrawn

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 5.48	Design Considerations--Systems for Measuring the Mass of Liquids (Feb-75)	NA				RG deals with liquid measurements for SNM that will not be used at a reactor
RG 5.49	Internal Transfers of Special Nuclear Material (for Comment) (Mar-75)	NA				Concerns internal transfers of SNM between areas in material processing facilities where accounting methods change from item control to material balance control.
RG 5.50	(Not issued)	NA				Not issued
RG 5.51	Management Review of Nuclear Material Control and Accounting Systems (for Comment) (Jun-75)	NA				Concerns management review of SNM material accounting programs. Not relevant for reactors.
RG 5.52 Rev. 3	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (Other than Nuclear Power Plants) (Dec-94)	NA				For other than nuclear power plants
RG 5.53 Rev. 1	Qualification, Calibration, and Error Estimation Methods for Nondestructive Assay (Feb-84)	NA				Not technically relevant
RG 5.54	Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants (for Comment) (Mar-78)	NA				Safeguards
RG 5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities (for Comment) (Mar-78)	NA				Safeguards for fuel cycle facilities
RG 5.56	Standard Format and Content of Safeguards Contingency Plans for Transportation (for Comment) (Mar-78)	NA				Safeguards for transportation
RG 5.57	Shipping and Receiving Control of Strategic Special Nuclear Material (Jun-76)	NA				Concerns compliance with an NA portion of 10 CFR 73.
RG 5.58 Rev. 1	Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements (Feb-80)	NA				Concerns process accounting measurements for material handling facilities.
RG 5.59 Rev. 1	Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance (Feb-83)	NA				Concerns methods of complying with portions of 10 CFR 73 that are not applicable to power reactors.

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 5.60	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit (Apr-80)	NA				Concerns strategic SNM in transit.
RG 5.61	Intent and Scope of the Physical Protection Upgrade Rule Requirements for Fixed Sites (Jun-80)	NA				Concerns 1979 changes made to physical security requirements for fuel cycle facilities and transportation involving strategic SNM.
RG 5.62 Rev. 1	Reporting of Safeguards Events (Nov-87)	NA				Safeguards
RG 5.63	Physical Protection for Transient Shipments (Jul-82)	NA				Concerns shipments of Strategic SNM originating or terminating at a foreign port that stops at a United States port.
RG 5.64	(Not issued)	NA				Not issued
RG 5.65	Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls (Sep-86) Note: Refer to the Regulatory Guide for detailed criteria in “Regulatory Positions” and graphics in the Attachment.					
RG 5.66	Access Authorization Program for Nuclear Power Plants (Jun-91)					This RG basically references NUMARC 89-01 as providing an acceptable approach to Access Authorization, though with some clarifications. NUMARC 89-01 is attached to the RG.
RG 5.66.C.1.	The program given in NUMARC 89-01, "Industry Guidelines for Nuclear Power Plant Access Authorization Programs," August 1989, is acceptable to the NRC staff for meeting the provisions of 10 CFR 73.56, subject to the following: 1.1. Section 7.2, "Review Process," of the Guidelines does not apply; the review procedure					

Table A1-14: Regulatory Guides (Division 5)						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	must be conducted as specified in 10 CFR 73.56(e); 1.2. To the extent that the rule differs from the Guidelines in Section 11.0, "Grandfathering," the rule will prevail. Specifically, 10 CFR 73.56(c)(1) requires that individuals who have had an unescorted access authorization for at least 180 days on the date the final rule is published in the Federal Register need not be further evaluated.					
RG 5.66.C.2.	Licensees who adopt this regulatory guide should make the following statement in their certification to the NRC that they have implemented 10 CFR 73.56: “All elements of Regulatory Guide 5.66 have been implemented to satisfy the requirements of 10 CFR 73.56.” Licensees who adopt positions different from this regulatory guide should identify these differences in their certification to the NRC. Further, positions different from the ones in the regulatory guide that would decrease the effectiveness of the access authorization program should be submitted to the NRC in accordance with 10 CFR 50.90. A separate regulatory analysis has not been provided for this regulatory guide. The regulatory					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	analysis that was prepared for the rule on access authorization programs for nuclear power plants, 10 CFR "73.56, provides the basis for this regulatory guide and examines the costs and benefits of the rule as implemented by this guide. A copy of the "Regulatory Analysis for the NRC Nuclear Power Plant Access Authorization Program" is available for inspection and copying for a fee at the Commission's Public Document Room, 2120 L Street NW., Washington, DC, under Regulatory Guide 5.66.					
RG 5.67	Material Control and Accounting for Uranium Enrichment Facilities Authorized To Produce Special Nuclear Material of Low Strategic Significance (Dec-93)	NA				For enrichment facilities
RG 5.68	Protection Against Malevolent Use of Vehicles at Nuclear Power Plants (Aug-94)					
RG 5.68.C.1.	MEASURES TO PROTECT AGAINST UNAUTHORIZED VEHICLE INTRUSION A vehicle barrier system (VBS) that is capable of preventing forced access of a land vehicle to gain proximity to vital areas should be established at each nuclear power reactor site. The VBS should provide a perimeter around vital areas of the facility such that no location along the perimeter would permit forced entry of a land vehicle. The VBS, regardless of the type of barriers used, should be of a design capable of stopping the					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	forward motion of the design basis land vehicle (DBV). The VBS may be incorporated as part of the protected area perimeter system but should not diminish or remove any requirements established for the protected area. 1.1 Passive Barriers The passive barrier portion of the VBS may include natural terrain features such as steep cliffs and large rocks, alone or in combination with man-made structures or barriers, provided the overall effectiveness of the barrier at any point is capable of stopping the forward motion of the DBV. Man-made or natural features that limit the direction and speed of the DBV may be used in conjunction with a barrier design. The separate Safeguards Information, which has already been sent to affected licensees, provides design guidance that is acceptable to the NRC on the performance capabilities of barriers and specifications for measures that reduce vehicle speed. 1.2 Active Barriers Access by vehicles to locations inside the VBS should be through active vehicle denial barriers that, in the denial position, are capable of stopping the forward motion of the DBV. Operational design					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	features of the active barrier or barrier system, when allowing access for authorized vehicles, should be capable of preventing being bypassed and allowing access of unauthorized vehicles. A single active barrier may be used in conjunction with other vehicle control measures to ensure denial of an unauthorized vehicle. The separate Safeguards Information that was sent to affected licensees provides design guidance that is acceptable to the NRC on the performance capabilities of barriers and specifications for measures that reduce vehicle speed. **1.3** Vehicle and Personnel Access Authorization Measures Vehicles and their operators should be authorized for entry prior to being permitted access inside the VBS. Vehicle authorization should also include confirmation that the vehicle has a legitimate purpose for entering the VBS. Authorization for the vehicle operator should include confirmation that the individual has a legitimate purpose for operating the vehicle inside the VBS. For VBS designs that are adjacent to the protected area boundary and whose active vehicle barrier access points are the same as the protected area vehicle access points, vehicle and personnel authorization measures for entering the protected area provide					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	adequate authorization controls. 1.4 VBS Description The security plan should contain an attachment that describes the VBS. The description should include site drawings that identify the VBS, the various components and combinations of components that compose the VBS, and access authorization measures for vehicle and personnel within the VBS.					
RG 5.68.C.2.	MEASURES TO PROTECT VITAL AREAS AGAINST A LAND VEHICLE BOMB The new 10 CFR 73.55(c)(8) requires a licensee to compare the vehicle control measures established in accordance with 10 CFR 73.55(c) (7) with the design goals and criteria for protection against a land vehicle bomb specified by the Commission. The design basis bomb size is specified in the separate Safeguards Information that has already been provided to affected licensees. 2.1 Blast Effect Analysis The comparison of vehicle control measures with the design goals and criteria for protection against a land vehicle bomb should consist of an analysis that establishes that the capability of vital					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	equipment to maintain the plant in a safe condition is not lost as a result of a detonation of a design basis bomb at the VBS boundary. Depending on the VBS design and site-specific considerations, this comparison could result in a determination that the design goals and criteria for protection against a land vehicle bomb are satisfied at the conclusion of any one of the following measures. 2.1.1 Screening Analysis This screening process determines whether a more detailed analysis of the effects of an explosive blast of the size of the design basis bomb is required. For each location along the VBS perimeter the standoff distance (distance between vital equipment or a structure housing vital equipment and the closest exterior point of the VBS) should be determined. Certain security-related electric power supplies and the central alarm station are required by 10 CFR Part 73 to be protected within vital areas; however, in the absence of safety-related equipment necessary for plant shutdown, these vital areas need not be considered as areas needing protection in the licensee's analysis. Licensees should determine whether the standoff distances for each location along the VBS provide a safe standoff distance. This determination					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	should be made by an analysis that takes into account the size of the explosive; both reflective and side-on blast loads on walls, roofs, and supporting members; the distance between the explosive and the affected structure; and the characteristics of the structure. Vital equipment can be assumed to remain operational if the structure containing the equipment provides such a level of protection that there is a low probability of damage to the equipment from an explosion occurring at the vehicle barrier. The separate Safeguards Information that has already been provided to affected licensees specifies approaches acceptable for determining safe standoff distances. If vital area structures and equipment are found to be located at distances equal to or greater than the safe standoff distance, the design goals and criteria for protection against a land vehicle bomb are considered fully met and no further analysis is necessary. 2.1.2 Detailed Analysis If the screening analysis described in Section 2. 1.1 of this guide cannot establish that vital equipment would be protected from damage by detonation of the design basis bomb at any					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	location along the VBS boundary, the analysis should then consider: (1) Whether any obstructions in the blast path would affect the level of protection provided to vital equipment. The analysis may incorporate the effects of natural topography that diminish the effects of the bomb blast effect. The analysis may also include an assessment of interior building designs (e.g., interior walls, supports) that may protect vital equipment even if the outer wall or structure is significantly damaged. The analysis should show whether or not the blast damage impacts the functional operability of the vital equipment. (2) Whether the plant can be shut down and maintained in a shutdown condition with equipment not damaged by the explosion. The evaluation may allow for damage control actions to mitigate the consequences of the explosion. These damage control actions should be included in applicable station operating procedures and referenced in the safeguards contingency procedures. In addition, the analysis should consider loss of off-site power, an assumption that is compatible with the basic premise that equipment not					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	designated and protected as vital is vulnerable to damage and is not available. If the detailed analysis determines that all vital equipment remains functional or that the ability to shut down the facility and maintain it in a shutdown condition can be provided even with the loss of vital equipment identified in the screening analysis, the design goals and criteria for protection against a land vehicle bomb are considered fully met and no further analysis is necessary. 2.1.3 Additional Protection Measures If the screening and detailed analyses determine that the design goals and criteria for protection against a land vehicle bomb cannot be fully met, a determination should be made concerning additional measures needed to fully achieve the design goals and criteria. Additional measures may include installing blast shields, changing planned vehicle barriers to extend standoff distances, strengthening current structures, or installing or relocating plant equipment or systems. If analysis of the effects of additional measures finds that vital equipment remains functional or that the ability to shut down and maintain the facility in a safe condition can be provided, the					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	design goals and criteria for protection against a land vehicle bomb are considered fully met and no further analysis is necessary. As provided in 10 CFR 73.55(c)(8), the licensee may propose to the NRC additional measures other than ones needed to fully meet the design goals and criteria, provided this approach provides substantial protection against a vehicle bomb and it can be demonstrated that the costs of measures to fully meet the design goals and criteria are not justified by the added protection that would be provided. If so, the actions in Regulatory Position 2.2 should be taken. 2.2 Alternative Measures To Protect Against Explosives As provided in 10 CFR 73.55(c)(8), a licensee may propose to the NRC additional measures other than the ones needed to meet the design goals and criteria, provided this approach provides substantial protection against a vehicle bomb and provided it can be demonstrated that the costs of measures to fully meet the design goals and criteria are not justified by the added protection that would be provided. This submittal should include: (1) The findings regarding the extent of the					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	protection against a vehicle bomb provided by the vehicle control measures designed to meet the requirements of 10 CFR 73.55(c)(7). These findings should be expressed in explicit terms such as the size of explosive for which the measures provide protection and the locations along the barrier system perimeter where the design goals for protection against a vehicle bomb cannot be fully met. (2) A description and analysis of additional measures needed to fully meet the design goals and criteria for protection against a vehicle bomb. The description should include an estimate of the cost of the measures. (3) A description and analysis of additional measures, alternative to those needed to fully meet the design goals and criteria, that are proposed to be taken. The analysis should address the enhanced protection provided by the additional measures. The description should include an estimate of the costs of the measures. (4) A comparison of the costs of the measures described in (2) and (3) above and an assessment supporting a finding that additional costs of fully meeting the design goals and criteria are not					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	justified by the added protection that would be provided.					
RG 5.68.C.3.	DOCUMENTATION In accordance with 10 CFR 73.55(c) (9), each licensee authorized to operate a nuclear power reactor is required to submit to the Commission a summary description of the proposed vehicle control measures and the results of the vehicle bomb comparative analysis. The summary description should include identification of active and passive components of the VBS and any natural terrain features or man-made obstructions that complete the VBS. A site drawing or diagram that outlines the VBS should be included with the description. The results of the vehicle bomb comparative analysis should identify the basis for determining that the Commission's design goals and criteria for protection against a land vehicle bomb are fully met. When applicable, the results of the comparison should include damage control actions that must be taken and additional security measures taken to protect against the design basis bomb. Licensees whose comparative analysis determines that they do not fully meet the design goals and criteria for protection against a vehicle bomb and who propose alternative measures should submit the analysis and justification for the alternatives as specified in Regulatory Position 2.2. Details of the "as built" VBS and of the land					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	vehicle bomb analysis should be maintained on site.					
RG 5.68.C.4.	CONTINGENCY PLANNING FOR SURFACE VEHICLE BOMBS Once implemented, the control measures required to meet these amendments to Part 73 supersede contingency requirements initiated in response to Generic Letter 89-07, "Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs," * of April 28, 1989. However, licensees whose vehicle control measures do not fully meet the NRC's design goals and measures may choose to maintain vehicle bomb contingency planning as one element of proposed alternative measures.					
RG 5.69	Not listed in NRC index	NA				Not issued
RG 5.70	Not listed in NRC index	NA				Not issued
RG 5.71	Cyber Security Programs for Nuclear Facilities (Jan-10) Note: Refer to the Regulatory Guide for detailed criteria.					
RG 5.72	Not listed in NRC index	NA				Not issued
RG 5.73	Fatigue Management for Nuclear Power Plant Personnel (Mar-09) Note: Refer to the Regulatory Guide for detailed criteria.					This RG deals with fatigue in the context of fitness for duty and compliance with 10 CFR 26. Although not related to Part 50 or Part 52 licensing, it is relevant to the operation of nuclear power plants.
RG 5.74	Managing the Safety/Security Interface (Jun-09)					
RG 5.74.C.1.	Requirements a. In accordance with 73.58(b) and (c), licensees must review planned and emergent changes and					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	activities to identify any potential adverse impact of these changes or activities on safety and security before implementation. Each licensee is responsible for establishing, implementing, and maintaining site procedures that not only ensure that personnel knowledgeable in each program area participate in the site work control process, but also provide a means of communicating proposed changes to the appropriate personnel within each program area for review. Management controls or processes used to assess proposed facility changes may be qualitative, quantitative, or a combination of both based on the complexity of the proposed changes or planned activities. b. Licensees shall assess and manage their safety and security program activities in a manner that ensures that there are no adverse impacts on the safety and security activities. The requirements of 10 CFR 73.58 are: (1) 10 CFR 73.58(b) states, “The licensee shall assess and manage the potential for adverse effects on safety and security, including the site emergency plan, before implementing changes to the plant configurations, facility conditions, or security.” (2) 10 CFR 73.58(c) states, “The scope of changes to be assessed and managed must include planned and emergent activities (such as,					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	but not limited to, physical modifications, procedural changes, changes to operator actions or security assignments, maintenance activities, system reconfiguration, access modification or restrictions, and changes to the security plan and its implementation)." (3) 10 CFR 73.58(d), states, "Where potential adverse interactions are identified, the licensee shall communicate them to appropriate licensee personnel and take compensatory and/or mitigative actions to maintain safety and security under applicable Commission regulations, requirements, and license conditions." c. Licensees should consider reviewing and updating existing procedures to reference the requirements of the interface between safety and security as outlined in 10 CFR 73.58. These procedures should clearly define processes to ensure that effective communications between the operations (safety) and security staffs is maintained at the facility. d. In accordance with 10 CFR 73.55(m), each licensee is responsible for ensuring that the reviews and audits of its site physical protection program include activities involving the safety/security interface.					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
RG 5.74.C.2.	Scope a. The licensee's established controls and processes for managing the interface between safety and security should ensure that security personnel are notified of planned or unplanned changes to the characteristics of the site's physical layout (including topographical changes); the configuration of facilities; structures, systems, and components; and the site's operations procedures. Controls and processes should also ensure that the security organization has the opportunity to review proposed changes and activities to identify potential adverse impacts on the functions and performance of the elements of the site physical protection program established within the owner-controlled area, protected area, and vital areas. When physical and/or administrative changes are driven by operation or emergency planning, the licensee should assess the potential impacts of these changes on the functions and performance of the elements of its site physical protection program to prevent the inadvertent degradation of site protective strategy. b. Personnel knowledgeable of the site physical protection program should review proposed changes to the following program areas for potential adverse effects on security:					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(1) operations, (2) maintenance, (3) work management (control and planning), (4) nuclear training, (5) nuclear engineering and support, (6) radiation protection, (7) emergency preparedness or planning, (8) fire protection, (9) chemistry (chemical safety), (10) environmental protection, (11) industrial health and safety, and (12) security. c. Personnel knowledgeable of the site physical protection program should review the following planned or emergent activities for potential adverse effects on security: (1) activities that could cause a loss of primary power to security systems, (2) the installation or removal of a barrier that could adversely impact safety, security, or emergency response, (3) the placement of trailers or heavy equipment that could obstruct detection or assessment functions or increase the response times of security personnel, (4) the installation of chemical or hazardous material storage tanks adjacent to a protected fighting position,					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(5) fire protection manual operator actions that do not account for paths of travel through the security fields of fire, which could delay or prevent operator response and invalidate safety assumptions and credit for operator actions, (6) construction activities that remove or degrade physical barriers, thus allowing established access controls to be bypassed, (7) the installation of barriers that increase the security response timelines that interfere with protected fighting positions and fields of fire, and that interfere with or prevent detection and assessment functions, and (8) changes to target set equipment that could impact its availability or operability. d. To facilitate the safety/security assessment process, the licensee may choose to evaluate changes using predetermined questions that are specifically designed to identify potential conflicts in an efficient, yet adequately detailed, manner. Current "change management" processes that licensees may consider for use in developing screening questions include, but are not limited to: (1) 10 CFR 50.54(a) process for screening changes to quality assurance plans, (2) 10 CFR 50.54(p) process for screening changes to the security (physical security, training qualification, contingency) plan,					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(3) 10 CFR 50.54(q) and 10 CFR 50.47(b) processes for screening changes to the emergency Plan, and (4) 10 CFR 50.59, “Changes, Tests, and Experiments,” and Regulatory Guide 1.187 “Guidance for Implementation of 10 CFR 50.59, ‘Changes, Tests, and Experiments’,” (Ref. 4) processes for evaluating changes, tests, and experiments. e. The following are examples of questions that may be used for the screening of planned and emergent activities or changes: (1) Could the proposed change or activity decrease the reliability or availability of a security system to perform the intended functions? (2) Could the proposed change or activity increase the likelihood of malfunctions of security equipment or systems? (3) Could the proposed change or activity decrease the effectiveness of NRC-approved security plans or invalidate the site protective strategy (e.g., communications, response timelines and pathways, equipment and systems (particularly target sets), or protected fighting positions and fields of fire)? (4) Could the proposed change or activity interfere with detection (i.e., interior and exterior sensors, zone of detection and field of view, alarm					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	communications, or access control systems) and assessment functions? (5) Could the proposed change or activity increase response times of emergency or armed security personnel (e.g., manmade or natural and active or passive vehicle barriers, vehicle access control and channeling barriers, access delay systems, exterior (protected area) delay barriers, interior delay barriers (passive, active, or dispensable))? (6) Could the proposed change or activity increase the numbers of, change configurations of, or create a new target set(s) from those previously evaluated? (7) Could the proposed change or activity reduce adversary task times? (8) Could the proposed change or activity result in noncompliance with the NRC's security regulations? If the answer to any of these screening questions is "yes," compensatory or mitigative actions or both may be necessary to maintain safety or security. If required, the licensee should communicate the action to its appropriate personnel.					
RG 5.74.C.3.	Management Controls and Processes a. For those plant changes that could affect security, the licensee should establish controls or					

Table A1-14: Regulatory Guides (Division 5)						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	processes to assess and manage operational changes to include emergency planning for both planned and emergent activities that could impact: (1) the effectiveness, reliability, and availability of the systems of the site physical protection program; (2) the effective implementation of the site protective strategy; and (3) the effectiveness of the site security plans, implementing procedures, or license conditions. The objective is to verify that a proposed change or activity will not inhibit compliance with security requirements or reduce the effectiveness, reliability, or availability of the licensee's site physical protection program credited for protection against the design basis threat. b. One acceptable method to meet the requirements of 10 CFR 73.58 is for licensees to evaluate existing and proposed programmatic controls (i.e., plant operations review committees; plant review boards; safety review committees; independent safety reviews; work planning and controls; configuration management; review and audit programs; corrective actions and reporting programs; engineering, design, and project management; maintenance; and other controls that exist at an operating nuclear power plant). Using existing controls to implement the interface between safety and security will help to ensure					

Table A1-14: Regulatory Guides (Division 5)						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	that assessment and management of facility changes and activities includes the physical protection program. c. The licensee should develop or consolidate crosscutting controls, processes, and procedures to assess and manage the potential for adverse safety and security interactions that may result from changes to the configuration of the site, changes in equipment status, and changes to site procedures. These management controls or processes typically ensure that licensee personnel identify, describe, review, approve, monitor, implement, and document emergent and planned operations or activities. d. For those security changes that could affect safety, the licensee should establish controls or processes to assess and manage security-related changes to both planned and emergent activities that could impact safe plant operations, including emergency planning. e. The licensee should use the existing management controls and processes described in 10 CFR 50.59 to evaluate proposed changes in the design or operation of its site physical protection program that could affect elements of plant operation including emergency					

Table A1-14: Regulatory Guides (Division 5)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	preparedness. f. The licensee should conduct reviews and audits to confirm that procedures established to control any changes to the plant configuration, including emergencies, comply with the licensee's security program. The review should encompass plant operations; plant modifications; and plant safety programs, processes, and procedures. The licensee may audit engineering and design, safety analysis, work controls, construction, maintenance, and other activities. The procedures governing these and other activities should include security reviews: (1) to identify safety activities or conditions that could affect security; (2) to identify security activities or conditions that could affect safety; and (3) to provide a means for resolving conflicting or competing safety and security interests. To prevent recurrence, corrections to specific or programmatic issues should be managed through the site's corrective action program for tracking, trending, communications, and completion.					
RG 5.74.C.4.	Training a. The licensee should provide training that addresses changes in the updated procedures and corresponding guidance documents to managers involved in the process of facilitating the interface					

Table A1-14: Regulatory Guides (Division 5)						
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	between safety and security.					
RG 5.75	Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities (Jul-09) Note: Refer to the Regulatory Guide for detailed criteria.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	CHAPTER 1, Introduction and General Description of Plant					
1.0 (Rev. 1, November 2007	Introduction and Interfaces					
1.0.1	There are no specific SRP acceptance criteria associated with these general requirements.					
1.0.2	For regulatory considerations, acceptance is based on addressing the regulatory requirements as discussed within this FSAR section or within the referenced FSAR section. The SRP acceptance criteria associated with the referenced section will be reviewed within the context of that review.					
1.0.3	For performance of new safety features, the information is sufficient to provide reasonable assurance that (1) these new safety features will perform as predicted in the applicant's FSAR, (2) the effects of system interactions are acceptable, and (3) the applicant provides sufficient data to validate analytical codes. The design qualification testing requirements may be met with either separate effects or integral system tests; prototype tests; or a combination of tests, analyses, and operating experience.					
	CHAPTER 2, Site Characteristics					
2.0 (Initial Issuance, March 2007)	Site Characteristics and Site Parameters					
2.0.1	For ESP, DC, and COL applications, the acceptance criteria associated with specific site characteristics/parameters and site-related design characteristics/parameters are contained in the related SRP Chapter 2 or other referenced SRP sections.					
2.0.2	For a COL application referencing an ESP, acceptance is based on the applicant’s demonstration that the design of the facility falls within the site characteristics and site-related design parameters specified in the ESP. If the final safety analysis report does not					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	demonstrate that the design of the facility falls within the site characteristics and design parameters, the application shall include a request for a variance from the ESP that complies with the requirements of 10 CFR 52.39 and 10 CFR 52.93.					
2.0.3	For a COL application referencing a DC, acceptance is based on the applicant's demonstration that the characteristics of the site fall within the site parameters of the certified design. If the actual site characteristics do not fall within the certified standard design site parameters, the COL applicant provides sufficient justification (e.g., by request for exemption or amendment from the DC) that the proposed facility is acceptable at the proposed site.					
2.0.4	For a COL application referencing an ESP and a DC, acceptance is based on the applicant's demonstration that the site characteristics and site-related design parameters specified in the ESP fall within the site parameters and design characteristics specified in the DC. If the actual site characteristics do not fall within the certified standard design site parameters, the COL applicant provides sufficient justification (e.g., by request for exemption or amendment from the DC, or request for a variance from the ESP) that the proposed facility is acceptable at the proposed site.					
2.0.5	For a COL application referencing neither an ESP nor a DC, acceptance is based on the applicant's identification of the complete set of site characteristics and site-related design characteristics needed to enable the staff to reach a conclusion on all safety matters related to siting.					
2.1.1 (Rev. 3, March 2007)	Site Location and Description					
2.1.1.1	Specification of Location: The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.34(a)(1), 10 CFR 52.17(a)(1), and 10 CFR52.79(a)(1) if it describes highways, railroads, and waterways that traverse the exclusion area in sufficient detail to					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	allow the reviewer to determine that the applicant has met the requirements in 10 CFR 100.3.					
2.1.1.2	Site Area Map: The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.34(a)(1), 10 CFR 52.17(a)(1), and 10 CFR 52.79(a)(1) if it describes the site location, including the exclusion area and the location of the plant within the area, in sufficient detail to enable the reviewer to evaluate the applicant's analysis of a postulated fission product release, thereby allowing the reviewer to determine (in SRP Sections 2.1.2 and 2.1.3 and Chapter 15) that the applicant has met the requirements of 10 CFR 50.34(a)(1) and 10 CFR Part 100.					
2.1.2 (Rev. 3, March 2007)	Exclusion Area Authority and Control					
2.1.2.1	Establishment of Authority The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.33, 10 CFR 50.34(a)(1), 10 CFR 52.17, 10 CFR 52.79, and 10 CFR Part 100 if it provides sufficient detail to enable the staff to evaluate the applicant's legal authority within the designated exclusion area. The definition in 10 CFR 100.3 states as follows: Exclusion area means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. To meet the requirements of 10 CFR Part 100, the applicant must demonstrate, before issuance of a CP or limited work					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	authorization, that it has the authority within the exclusion area as defined in 10 CFR 100.3, or must provide reasonable assurance that it will have such authority before either the start of construction or commencing activities allowed by 10 CFR 52.25. Absolute ownership of all lands within the exclusion area, including mineral rights, is considered to carry with it the required authority to determine all activities on this land and is acceptable. Where the required authority is contingent upon future procurement of ownership (e.g., by eminent domain proceedings) or by lease, easement, contract, or other means, the exclusion area may be acceptable if legal staff can determine that the information submitted by the applicant provides reasonable assurance that it will obtain the required authority before the start of construction. In cases where the applicant will acquire or complete the acquisition of ownership and control during a construction period, legal staff will conduct a special review. In addition, in cases of proposed public road abandonment or relocation, legal staff should determine that there is sufficient authority or that sufficient arrangements have been made to accomplish the proposed relocation or abandonment. In the event an ESP applicant does not have the required authority and control but provides reasonable assurance that it will acquire such authority and control, the ESP may include a condition requiring the applicant to notify the staff when the applicant has indeed acquired such authority and control and the basis for that conclusion. At the OL or COL stage of review, the applicant must have completed arrangements to obtain exclusion area authority and control. The NRC will not permit the licensee to load fuel until					

Table A1-15: NUREG-0800, Standard Review Plan

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	it has completed all efforts to establish exclusion area authority and control, including all transfers of title, easements, lease arrangements, public road abandonments or relocations, as applicable. Atomic Safety and Licensing Board and Atomic Safety and Licensing Appeal Board decisions (e.g., Refs. 1, 2, and 3) provide further guidance regarding the sufficiency of the applicant's proposed control over the exclusion area in instances where the licensee does not hold title.					
2.1.2.2	Exclusion or Removal of Personnel and Property: The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.33, 10 CFR 50.34(a)(1), 10 CFR 52.17, 10 CFR 52.79, and 10 CFR Part 100 if it provides sufficient detail to enable the staff to evaluate the applicant's legal authority for the exclusion or removal of personnel or property from the exclusion area. A highway, railroad, or waterway may traverse the exclusion area but should not be so close to the facility so as to interfere with normal operations. In addition, appropriate and effective arrangements should be made to control traffic on the highway, railroad, or waterway in the case of an emergency. Residence within the exclusion area should normally be prohibited. In the event that people live within the exclusion area, these residents should be subject to ready removal if necessary. Activities unrelated to the operation of the reactor may be permitted in an exclusion area provided that no significant hazards to the public health and safety will result.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Where the designated exclusion area extends into a body of water such as a lake, reservoir, or river that is routinely accessible to the public, the staff must determine whether the applicant has made appropriate arrangements with the local, State, Federal, or other public agency having authority over the particular body of water. The arrangements should provide for the exclusion and ready removal in an emergency, by either the applicant or the public agency in authority, of any persons on those portions of the body of water that lie within the designated exclusion area.					
2.1.2.3	Proposed and Permitted Activities: The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.33, 10 CFR 50.34(a)(1), 10 CFR 52.17, 10 CFR 52.79, and 10 CFR Part 100 if it provides sufficient detail to enable the staff to evaluate the applicant's legal authority over all activities within the designated exclusion area. Activities unrelated to plant operation within the exclusion area are acceptable under the following circumstances: A. Such activities, including accidents associated with such activities, represent no hazard to the plant or have been shown to be accommodated as part of the plant design basis (see SRP Section 2.2.3). B. The applicant is aware of such activities and has made appropriate arrangements to evacuate persons engaged in such activities in the event of an accident. C. There is reasonable assurance that, in the event of an					

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	accident, persons engaged in such activities can be evacuated without receiving radiation doses in excess of the guideline values given in 10 CFR 50.34(a)(1).					
	REFERENCES: 1. The Cleveland Electric Illuminating Company, et. al. (Perry Nuclear Power Plant, Units 1 and 2), "Supplemental Partial Initial Decision, Site Suitability and Environmental Matters," LBP-74-76, 8 AEC 701, October 20, 1974. 2. Southern California Edison Company, et. al. (San Onofre Nuclear Generating Station, Units 2 and 3), "Decision," ALAB-248, 8 AEC 951, December 24, 1974. 3. Southern California Edison Company, et al. (San Onofre Nuclear Generating Station, Units 2 and 3), "Decision," ALAB-268, 1 NRC 383, April 25, 1975.					
2.1.3 (Rev. 3, March 2007)	Population Distribution					
2.1.3.1	Population Data: The population data supplied by the applicant in the SAR is acceptable under the following conditions: A. The SAR contains population data from the latest census and projected population at the year of plant approval and 5 years thereafter, in the geographical format given in Section 2.1.3 of Regulatory Guide 1.70 and in accordance with DG-1145. B. The SAR describes the methodology and sources used to obtain the population data, including the projections. C. The SAR includes information on transient populations in the site vicinity.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
2.1.3.2	Exclusion Area: The exclusion area should either not contain any residents, or such residents should be subject to ready removal if necessary.					
2.1.3.3	Low-Population Zone: The specified LPZ is acceptable if it is determined that appropriate protective measures could be taken on behalf of the enclosed populace in the event of a serious accident.					
2.1.3.4	Nearest Population Center Boundary: The nearest boundary of the closest population center containing 25,000 or more residents is at least one and one-third times the distance from the reactor to the outer boundary of the LPZ. The boundary of the population center should be determined based on considerations of population distribution. Political boundaries are not controlling.					
2.1.3.5	Population Density: If the population density at the CP, ESP, or COL (not referencing ESP) stage exceeds the guidelines given in Regulatory Position C.4 of Regulatory Guide 4.7, the applicant must give special attention to the consideration of alternative sites with lower population densities. A site that exceeds the population density guidelines of Regulatory Position C.4 of Regulatory Guide 4.7 can nevertheless be selected and approved if, on balance, it offers advantages compared with available alternative sites when all of the environmental, safety, and economic aspects of the proposed and alternative sites are considered.					
2.2.1 (Rev. 3, March 2007)	Identification of Potential Hazards in Site Vicinity					
2.2.1.1	Data in the safety analysis report (SAR) adequately describe the locations and distances from the plant of nearby industrial,					

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	military, and transportation facilities and that such data are in agreement with data obtained from other sources, when available.					
2.2.1.2	Descriptions of the nature and extent of activities conducted at the site and in its vicinity, including the products and materials likely to be processed, stored, used, or transported, are adequate to permit identification of the possible hazards cited in Subsection III of this SRP section.					
2.2.1.3	Sufficient statistical data with respect to hazardous materials are provided to establish a basis for evaluating the potential hazards to the plant or plants considered at the site.					
2.2.2	Part of 2.2.1					
2.2.3 (Rev. 3, March 2007)	Evaluation of Potential Accidents					
2.2.3.1	Event Probability The identification of design-basis events resulting from the presence of hazardous materials or activities in the vicinity of the plant or plants of specified type (or, for ESP applications not referencing DC, falling within a PPE) is acceptable if all postulated types of accidents are included for which the expected rate of occurrence of potential exposures resulting radiological dose in excess of the 10 CFR 50.34(a)(1) as it relates to the requirements of 10 CFR Part 100 is estimated to exceed the NRC staff objective of an order of magnitude of 10^{-7} per year. If data are not available to make an accurate estimate of the event probability (see Technical Rationale 2 below), an expected rate of occurrence of potential exposures resulting radiological dose in excess of the 10 CFR 50.34(a)(1) as relates to the requirements of 10 CFR Part 100, by an order of magnitude of 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be					

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	lower.					
2.2.3.2	Design-Basis Events The effects of design-basis events have been adequately considered, in accordance with 10 CFR 100.20(b), if analyses of the effects of those accidents on the safety-related features of the plant or plants of specified type (or, for ESP applications, falling within a PPE) have been performed and measures have been taken (e.g., hardening, fire protection) to mitigate the consequences of such events.					
2.3.1 (Rev. 3, March 2007)	Regional Climatology					
2.3.1.1	The description of the general climate of the region should be based on standard climatic summaries compiled by NOAA (e.g., References 5, 6). Consideration of the relationships between regional synoptic-scale atmospheric processes and local (site) meteorological conditions should be based on appropriate meteorological data (e.g., References 6, 7).					
2.3.1.2	Data on severe weather phenomena should be based on standard meteorological records from nearby representative National Weather Service (NWS), military, or other stations recognized as standard installations that have long periods of data on record (e.g., References. 6, 7, 8). The applicability of these data to represent site conditions during the expected period of reactor operation should be substantiated.					
2.3.1.3	The tornado parameters should be based on Regulatory Guide 1.76 (Reference 9). Alternatively, an applicant may specify any tornado parameters that are appropriately justified, provided that a technical evaluation of site-specific data is conducted. Any deviations from Regulatory Guide 1.76 should be identified by the applicant.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
2.3.1.4	The basic (straight-line) 100-year return period 3-second gust wind speed should be based on appropriate standards, with suitable corrections for local conditions (e.g., References 10, 11).					
2.3.1.5	In accordance with Regulatory Guide 1.27 (Reference 12), the UHS meteorological data that would result in the maximum evaporation and drift loss of water and minimum water cooling should be based on long-period regional records that represent site conditions. If applicable, the potential for water freezing in the UHS water storage facility should also be analyzed. The maximum accumulated degree-days below freezing recorded in the site region during the winter (or during the worst-case freezing spell in warmer climates) may be a reasonable conservative site characteristic for evaluating the potential for water freezing in a UHS water storage facility. Suitable information should be compiled from at least 30 years of meteorological data found in databases for nearby representative locations (e.g., References 13, 14, 15). The bases and procedures used to select critical meteorological data should be provided and justified.					
2.3.1.6	Consistent with the staff's branch position on winter precipitation loads (Reference 16), the winter precipitation loads to be included in the combination of normal live loads to be considered in the design of a nuclear power plant that might be constructed on the proposed site should be based on the weight of the 100-year snowpack or snowfall, whichever is greater, recorded at ground level. Likewise, the winter precipitation loads to be included in the combination of extreme live loads to be considered in the design of a nuclear power plant that might be constructed on the proposed site should be based on the weight of the 100-year snowpack at ground level plus the weight of the 48-hour PMWP at ground level for the month corresponding to the selected snowpack. Depending on the location of the site, the 48-hour PMWP may not necessarily be in the form of frozen precipitation. A CP, OL, or COL applicant may choose and justify an alternative method for defining the extreme winter precipitation load by					

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	demonstrating that the 48-hour PMWP could neither fall nor remain on the top of the snowpack and/or building roofs. The weight of the 100-year return period snowpack should be based on data recorded at nearby representative climatic stations (e.g., Reference 17) or obtained from appropriate standards with suitable corrections for local conditions (e.g., References 10, 11). For the purposes of determining the extreme winter precipitation load, the 48-hour PMWP is defined as the theoretically greatest depth of precipitation for a 48-hour period that is physically possible over a 25.9-square-kilometer (10-square-mile) area at a particular geographical location during those months with the historically highest snowpacks. The weight of the 48-hour PMWP should be determined in accordance with reports published by NOAA's Hydrometeorological Design Studies Center (e.g., References 18–22).					
2.3.1.7	Ambient temperature and humidity statistics should be derived from data recorded at nearby representative climatic stations (e.g., Reference 23) or obtained from appropriate standards with suitable corrections for local conditions (e.g., Reference 10). Reference 23 provides a method for estimating 100-year return period extreme temperature values as a function of annual extreme temperature values.					
2.3.1.8	High air pollution potential information should be based on U.S. Environmental Protection Agency (EPA) studies (e.g., References 24, 25).					
2.3.1.9	All other meteorological and air quality conditions identified by the applicant as climate site characteristics for ESP applications or used as design and operating bases for CP, OL, or COL applications should be documented and substantiated.					
	REFERENCES: 5. U.S. Department of Commerce, "Climate Atlas of the United States," CD-ROM, National Climatic Data Center, NOAA. 6. U.S. Department of Commerce, "Local Climatological Data					

Table A1-15: NUREG-0800, Standard Review Plan						
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	— Annual Summary with Comparative Data," National Climatic Data Center, NOAA, published annually for all first-order NWS stations. 7. U.S. Department of Commerce, "State Climatological Summary," National Climatic Data Center, NOAA, published annually by State. 8. U.S. Department of Commerce, "Storm Data," National Climatic Data Center, NOAA, published monthly. 9. Regulatory Guide 1.76, Rev. 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants." 10. ASCE Standard No. 7-05, "Minimum Design Loads for Buildings and Other Structures," ASCE/SEI 7-05, American Society of Civil Engineers, 2006. 11. U.S. Department of Commerce, "Engineering Weather Data," CD-ROM, National Climatic Data Center, NOAA. 12. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants." 13. U.S. Department of Commerce, "Solar and Meteorological Surface Observational Network (SAMSON)," 3-volume CD-ROM set divided geographically into regions (Eastern, Central, and Western United States) covering 1961–1990, National Climatic Data Center, NOAA. 14. U.S. Department of Commerce, "Hourly United States Weather Observations 1990–1995," CD-ROM, National Climatic Data Center, NOAA. 15. U.S. Department of Commerce, "Integrated Surface Hourly Observations," 24-volume CD-ROM set divided by geographic region and time period covering 1995–2002, National Climatic Data Center, NOAA. 16. "Site Analysis Branch Position — Winter Precipitation Loads," NRC memorandum from H.R. Denton to R.R. Maccary, March 24, 1975, available in ADAMS under Accession #ML050630277. 17. U.S. Department of Commerce, "NCDC Cooperative Station					

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	Data," 3-volume CD-ROM set divided geographically into regions (Eastern, Central, and Western United States) with the period-of-record varying among stations but falling within the period from the 1850s through 2001, National Climatic Data Center, NOAA. 18. U.S. Department of Commerce, "Probable Maximum Precipitation Estimates: Colorado River and Great Basin Drainage," Hydrometeorological Report No. 49, NOAA, Reprinted 1984. 19. U.S. Department of Commerce, "Seasonal Variation of 10-Square-Mile Probable Maximum Precipitation Estimates, United States East of the 105th Meridian," Hydrometeorological Report No. 53, NOAA, April 1980. 20. U.S. Department of Commerce, "Probable Maximum Precipitation Estimates: United States, Between the Continental Divide and the 103rd Meridian," Hydrometeorological Report No. 55A, NOAA, June 1988. 21. U.S. Department of Commerce, "Probable Maximum Precipitation: Pacific Northwest States, Columbia River (including portions of Canada), Snake River and Pacific Coastal Drainages," Hydrometeorological Report No. 57, NOAA, October 1994. 22. U.S. Department of Commerce, "Probable Maximum Precipitation for California," Hydrometeorological Report No. 59, NOAA, February 1999. 23. American Society of Heating, Refrigeration, and Air Conditioning Engineers, "2005 ASHRAE Handbook — Fundamentals," 2005. 24. G.C. Holzworth, "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Office of Air Programs, EPA, January 1972. 25. J. X. L. Wang and J. K. Angell, "Air Stagnation Climatology for the United States (1948-1998)," NOAA Air Resources					

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	Laboratory Atlas No. 1, Air Resources Laboratory, Environmental Research Laboratories, Office of Oceanic and Atmospheric Research, Silver Spring, MD, April 1999.					
2.3.2 (Rev. 3, March 2007)	Local Meteorology					
2.3.2.1	Local summaries of meteorological data based on onsite measurements in accordance with Regulatory Guide 1.23 and National Weather Service station summaries or other standard installation summaries from appropriate nearby locations (e.g., within 80km (50 miles)) should be presented as specified in Regulatory Guide 1.70, Section 2.3.2, and RG 1.206, Section 2.3.2.1					
2.3.2.2	A complete topographical description of the site and environs out to a distance of 80 kilometers (50 mi) from the plant, as described in Regulatory Guide 1.70, Section 2.3.2.2, and RG 1.206, Section 2.3.2.2, should be provided.					
2.3.2.3	A discussion and evaluation of the influence of the plant and its facilities on the local meteorological and air quality conditions should be provided. Applicants should also identify potential changes in the normal and extreme values as presented in the safety analysis report (FSAR), resulting from plant construction and operation. The acceptability of the information is determined through comparison with standard assessments.					
2.3.2.4	The description of local site airflow should include wind roses and annual joint frequency distributions of wind speed and wind direction by atmospheric stability for all measurement levels using the criteria provided in Regulatory Guide 1.23.					
2.3.3 (Rev. 3, March 2007)	Onsite Meteorological Measurements Program					
2.3.3.1	The pre-operational monitoring program should be described for CP and ESP applications and for COL applications that do not					

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	reference an ESP. The operational monitoring program should be described for OL and COL applications and in those ESP applications that contain proposed complete and integrated emergency response plans. The monitoring program description should include meteorological measurements at the site and any offsite satellite facilities. The description should include: a. a site map (drawn to scale) that shows tower location and true north with respect to man-made structures, topographic features, and other features that may influence site meteorological measurements b. distances to nearby obstructions of flow in each downwind sector c. measurements made d. elevations of measurements e. exposure or instruments f. instrument descriptions g. instrument performance specifications h. calibration and maintenance procedures and frequencies i. data output and recording systems j. data processing, archiving, and analysis procedures Guidance on a suitable onsite meteorological monitoring program to provide the required meteorological data is presented in Regulatory Guide 1.23.					
2.3.3.2	Meteorological data should be presented in the form of joint frequency distributions of wind speed and wind direction by atmospheric stability class in the format described in Regulatory Guide 1.23. A hour-by-hour listing of the hourly-averaged parameters should be provided in the format described in Regulatory Guide 1.23. If possible, evidence of how well these					

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	data represent long-term conditions at the site should also be presented, possibly through comparison with offsite data. a. For CP applications, at least one annual cycle of onsite meteorological data should be provided with the application. For OL applications, at least two consecutive cycles (and preferably 3 or more whole years), including the most recent one-year period, should be provided at the time of application submittal. b. For COL applications that do not reference an ESP and for ESP applications, at least two consecutive annual cycles (and preferably 3 or more whole years), including the most recent 1-year period, should be provided with the application. If two years of onsite meteorological data are not available at the time the application is filed, the staff expects that the COL or ESP applicant will provide at least one annual cycle of meteorological data collected onsite with the application. These data should be used by the applicant to calculate (1) the short-term atmospheric dispersion estimates for accident releases discussed in SRP Section 2.3.4 and (2) the long-term atmospheric dispersion estimates for routine releases discussed in SRP Section 2.3.5. The applicant should continue to monitor the data and submit the complete 2-year data set when it has collected all the data. This supplemental submittal should also include a reanalysis of the Section 2.3.4 and 2.3.5 atmospheric dispersion estimates based on the complete 2-year data set.					
2.3.3.3	The applicant should identify and justify any deviations from the guidance provided in Regulatory Guide 1.23.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
2.3.4 (Rev. 3, March 2007)	Short-Term Atmospheric Dispersion Estimates for Accident Releases					
2.3.4.1	A description of the atmospheric dispersion models used to calculate χ/Q values for accidental releases of radioactive and hazardous materials to the atmosphere. The models should be documented in detail and substantiated within the limits of the model so that the staff can evaluate their appropriateness of use with regards to release characteristics, plant configuration, plume density, meteorological conditions, and site topography.					
2.3.4.2	Meteorological data used for the evaluation (as input to the dispersion models) which represent annual cycles of hourly values of wind direction, wind speed, and atmospheric stability for each mode of accidental release. Any dispersion estimates should be calculated from the most representative meteorological data available for the site. Guidance on appropriate onsite meteorological data is provided in Regulatory Guide 1.23. This information is also reviewed in SRP Section 2.3.3.					
2.3.4.3	A discussion of atmospheric diffusion parameters, such as lateral and vertical plume spread (σ_y and σ_z) as a function of distance, topography, and atmospheric conditions, should be related to measured meteorological data. The methodology for establishing these relationships should be appropriate for estimating the consequences of accidents within the range of distances which are of interest with respect to site characteristics and established regulatory criteria.					
2.3.4.4	Hourly cumulative frequency distributions of χ/Q values from the effluent release point(s) to the EAB and LPZ should be constructed to describe the probabilities of these χ/Q values being exceeded. All cumulative frequency distributions of χ/Q values should be presented for appropriate distances (e.g., the EAB distance and the outer boundary of the LPZ) and time periods as specified in Section 2.3.4.2 of Regulatory Guide 1.70, "Standard					

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	Format and Content of Safety Analysis Reports for Nuclear Power Plants" and Section 2.3.4.2 of RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)". The methods for generating these distributions should be adequately described. Guidance for calculating EAB and LPZ atmospheric dispersion factors is provided in Regulatory Guide 1.145.					
2.3.4.5	Atmospheric dispersion factors used for the assessment of consequences related to atmospheric radioactive releases to the control room for design basis, other accidents, and for onsite and offsite releases of hazardous airborne materials should be provided. Guidance for calculating control room χ/Q values for radiological releases and hazardous material releases is provided in Regulatory Guide 1.194 and Regulatory Guide 1.78, respectively.					
2.3.4.6	For control room habitability analysis, a site plan drawn to scale should be included showing true North and potential atmospheric accident release pathways, control room intake, and unfiltered inleakage pathways.					
2.3.5 (Rev. 3, March 2007)	Long-Term Atmospheric Dispersion Estimates for Routine Releases					
2.3.5.1	A detailed description of the atmospheric dispersion and deposition models used by the applicant to calculate annual average concentrations in air and amount of material deposited as a result of routine releases of radioactive materials to the atmosphere. The models should be sufficiently documented and substantiated to allow a review of their accuracy and validity, source configuration, suitability of input parameters, topography, and appropriateness for the site, plant, and release characteristics.					
2.3.5.2	A discussion of atmospheric diffusion parameters, such as vertical plume spread (σ_z) as a function of distance, topography, and atmospheric conditions. Use of these parameters should be substantiated as to their appropriateness for use in estimating the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	consequences of routine releases from the site boundary to a radius of 50 miles (80 kilometers) from the plant.					
2.3.5.3	Meteorological data summaries (onsite and regional) used as input to the dispersion and deposition models. Data used for this evaluation should represent hourly average values of wind speed, wind direction, and atmospheric stability which are appropriate for each mode of release and which are characteristic of annual average atmospheric dispersion and deposition conditions in the vicinity of the plant. Guidance on appropriate onsite meteorological data is provided in Regulatory Guide 1.23. This information is also reviewed under SRP section 2.3.3.					
2.3.5.4	Points of routine release of radioactive material to the atmosphere, including the characteristics (e.g., location, release mode) of each release point. Guidance for identifying release point characteristics is provided in Regulatory Guide 1.112. This information is also reviewed under SRP section 11.3.					
2.3.5.5	The specific location of potential receptors of interest (e.g., nearest vegetable garden, nearest resident, nearest milk animal, and nearest meat cow in each 22½ degree direction sector within a 5-mile (8-kilometer) radius of the site). Guidance for identifying the location of potential receptors of interest is provided in Regulatory Guide 1.109. This information is also reviewed under SRP section 11.3.					
2.3.5.6	The χ/Q and D/Q values to be used for assessment of the consequences of routine airborne radiological releases as described in Section 2.3.5.2 of Regulatory Guide 1.70 (Ref. 9) and Section 2.3.5.2 of RG 1.206 (Ref. 10): A. Maximum annual average χ/Q values and D/Q values at or beyond the site boundary and at specific locations of potential receptors of interest utilizing appropriate meteorological data for					

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	each routine venting location. B. Estimates of annual average χ/Q values and D/Q values for 16 radial sectors to a distance of 50 miles (80 kilometers) from the plant using appropriate meteorological data. Guidance for calculating these χ/Q and D/Q values is provided in Regulatory Guide 1.111.					
	REFERENCES: 9. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." 10. Regulatory Guide RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”					
2.4.1 (Rev. 3, March 2007)	Hydrologic Description					
2.4.1.1	Interface of the Plant with the Hydrosphere: The application should provide a description of hydrology in the vicinity of the site and site regions and of how the plant interfaces with the hydrosphere. The description and elevations of safety-related structures, facilities, and accesses thereto should be sufficiently complete to allow evaluation of the impact of flood design bases. Site topographic maps should be of good quality and of sufficient scale to allow independent analysis of pre- and post-construction drainage patterns. Flood maps that show the areas to be inundated by floods of difference magnitude and recurrence interval should be of appropriate scale and quality. All external plant structures and components should be identified on site maps. Data should be provided on surface water users,					

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	location with respect to the site, type of use, and quantity of surface water used. Tabulations of drainage areas, types of structures, appurtenances, ownership, seismic and spillway design criteria, elevation-storage relationships, and short and long-term storage allocations should be provided. The description of hydrologic characteristics should correspond to those of the United States Geological Survey (USGS), National Oceanic and Atmospheric Administration (NOAA), Natural Resources Conservation Service (NRCS), U.S. Army Corps of Engineers (USACE), or appropriate State and river basin agencies. Descriptions of all existing or proposed reservoirs and dams (both upstream and downstream) that could influence conditions at the site should be provided. These descriptions may be obtained from reports of the USGS, United States Bureau of Reclamation (USBR), USACE, and others. Generally, reservoir descriptions of a quality similar to those contained in pertinent data sheets of a standard USACE Hydrology Design Memorandum are adequate.					
2.4.1.2	Hydrological Causal Mechanisms: The application should provide a description of hydrological causal mechanisms that affect the safety of the plant. Mechanisms that can result in flooding at or in the vicinity of the site should be described. Mechanisms and climate in the vicinity of the site that affect low-water or drought conditions should be described.					
2.4.1.3	Surface and Ground Water Uses: The application should provide a description of surface and ground water uses in the vicinity of the site that affect the safety-related water supply to the plant. The description should include					

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	all current and future known and likely surface and ground water use that may affect safety-related water supply to the plant. This description should include both upstream and downstream uses of water in the vicinity of the site.					
2.4.1.4	Data: The application should provide a complete description of all spatial and temporal datasets used by the applicant in support of its conclusions regarding safety of the plant. Data and descriptions should be sufficiently detailed to allow the staff to review the applicant's conclusions regarding the safety of the plant and to determine of the design bases of safety-related SSC.					
2.4.1.5	Alternate Conceptual Models: The application should provide a description of alternate conceptual models of site hydrology. These alternate conceptual models should be sufficiently detailed to reasonably bound hydrological conditions at the site.					
2.4.1.6	Consideration of Other Site-Related Evaluation Criteria: The application should demonstrate that the potential effects of site-related proximity and of seismic and non-seismic information as they relate to hydrologic description in the vicinity of the proposed plant site and site regions are appropriately taken into account.					
2.4.2 (Rev. 4, March 2007)	Floods					
2.4.2.1	Local Flooding on the Site and Drainage Design: The application should include an estimate of local intense precipitation or local probable maximum precipitation (PMP) and a determination of the capacity of site drainage facilities (including drainage from the roofs of buildings and site ponding). Conclusions relating to the potential for any adverse effects of blockage of site drainage facilities by debris, ice, or snow should be based upon conservative assumptions of storm and vegetation conditions likely to exist during storm periods. If a potential					

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	hazard does exist (e.g., the elevation of ponding exceeds the elevation of plant access openings), the applicant should document and justify the design bases of affected facilities.					
2.4.2.2	Stream Flooding: The application should include documentation of the potential sources of flood and flood response characteristics. Depending on the hydrology in the watershed where the proposed site is located, estimates of tributary contributing area, PMF, coincident wind-induced waves, floods produced due to dam failures, and combinations of less severe river floods with coincident surges and seiches should be provided.					
2.4.2.3	Surges: The application should include the complete history of storm surges in the vicinity of the site. Depending on the location of the proposed site, estimates of PMH for coastal areas, PMH winds translated to inland locations, probable maximum windstorm, storm surges and waves resulting from these winds, and combinations of less severe storm surges with runoff floods should be provided.					
2.4.2.4	Seiches: The application should include the complete history of seiches in the vicinity of the site. Depending on the location of the proposed site and hydrologic and hydraulic characteristics of nearby water bodies, estimates of meteorologically induced seiches in inland lakes, coastal harbors, and embayments, seismically induced seiches in inland lakes, seiches induced by tsunamis, and a combination of less severe seiches coincident with runoff floods should be provided.					
2.4.2.5	Tsunami: The application should include the complete history of tsunami in the vicinity of the site. Both near and far-field tsunamigenic sources should be considered. Shallow seismic sources that are located on land, but are near the coast may generate an oceanic tsunami and should be considered. Far-field sources that may					

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	generate a tsunami that can travel long distances to affect the proposed site should be considered. Hillslope failures and slides that are generated on land and impact a water body (also called sub-aerial slides) may generate tsunami-like waves and should be considered. Submarine landslides are also known to generate tsunamis and should be considered. Seismic events can also generate tsunamis in inland water bodies, e.g., the tsunami-like wave generated in the Mississippi River due to the February, 1812, New Madrid earthquake. The possibility of such events affecting an inland site should be investigated.					
2.4.2.6	Seismically Induced Dam Failures (or Breaches): The application should include the flooding hazard at the plant site resulting from seismically induced dam failure upstream of the site location. A complete listing of all dams and other relevant water control structures upstream of the site that may pose a flooding hazard to the site should be presented. The effects of dam failure induced flooding in plant design bases should be provided. The maximum water surface elevation should be provided from failure of one or multiple dams during the SSE coincident with a 25-year flood and 2-year wind waves, from failure of one or multiple dams during the OBE coincident with a SPF and 2-year wind waves, from failure of one or multiple dams during other earthquakes of lesser intensity coincident with runoff, surge, or seiche flooding, and from breaches of water control structures that may be located above the site grade and sufficiently near safety-related SSC.					
2.4.2.7	Flooding Caused by Landslides: The application should include the flooding hazard at the plant site from flood waves induced by landslides and backwater effects due to stream blockage from landslides. A thorough review of historical landslides and potential for landslides including any historical flooding caused by them in the vicinity of the site and site regions should be presented. The effects of landslide-induced flooding in the plant design bases should be considered.					

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2.4.2.8	Effects of Ice Formation in Water Bodies: The application should include information concerning potential flooding at the plant site due to flood waves resulting from the collapse of an ice dam or backwater effects due to stream blockage due to an ice dam or an ice jam downstream of the plant site. A thorough review of historical ice formation in the vicinity of the site and site regions should be presented and its effects should be appropriately accounted for in plant design bases. A thorough review of the history of frazil and anchor ice and an estimate of the flooding effects of these phenomena on plant design bases should be provided.					
2.4.2.9	Combined Events Criteria: The application should include information concerning design basis flooding at the plant site, including consideration of appropriate combinations of individual flooding mechanisms in addition to the most severe effects from individual mechanisms themselves. The highest flood water surface elevation should be determined based on consideration of the worst combination of flooding mechanisms and is reported as a site characteristic in the staff's SER.					
2.4.2.10	Consideration of Other Site-Related Evaluation Criteria: The application should demonstrate that the potential effects of site-related proximity, seismic, and non-seismic information as they relate to hydrologic description in the vicinity of the proposed plant site and site regions are appropriately taken into account.					
2.4.3 (Rev. 4, March 2007)	Probable Maximum Flood on Streams and Rivers					
2.4.3.1	Design Bases for Flooding in Streams and Rivers. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the following characteristics are needed, and should be based on conservative assumptions of hydrometeorologic characteristics in the drainage area: (a) the area of the watershed used to estimate flooding in streams and					

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	rivers, (b) the total depth of PMP and the PMP hyetograph, (c) the maximum PMF water surface elevation in streams and rivers with coincident wind-waves, and (d) hydraulic characteristics that describe dynamic effects of PMF on SSC important to safety. If a potential hazard to SSC important to safety exists, the applicant should document and justify the design bases of affected facilities.					
2.4.3.2	Design Bases for Site Drainage. To meet the requirements of GDC 2, 10 CFR 52.17 and 10 CFR Part 100, estimates of the following characteristics are needed: the runoff from the immediate site area and the drainage from areas adjacent to the site, including the roofs of safety related structures. Flood response characteristics should be identified to estimate flooding adjacent to and on the plant site. The effects of erosion and sedimentation during the flooding should be identified and their effects on SSC important to safety should be determined. If a potential hazard to SSC important to safety exists, the applicant should document and justify the design bases of affected facilities.					
2.4.3.3	Consideration of Other Site-Related Evaluation Criteria. To meet the requirements of GDC 2, 10 CFR 52.17 and 10 CFR Part 100 information about the potential effects of site-related proximity, seismic, and non-seismic information as they relate to flooding in streams and rivers and local flooding adjacent to and on the plant site is needed.					
2.4.4 (Rev. 3, March 2007)	Potential Dam Failures					
2.4.4.1	Flood Waves from Severe Breaching of an Upstream Dam: To meet the requirements of GDC 2, 10 CFR 52.17, 10 CFR Part 100, and 10 CFR 100.23(d), estimates of the following characteristics are needed, and should be based on conservative assumptions of hydrometeorological, geological, and seismic characteristics in the drainage area: (a) modes of assumed dam					

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	breaches or failures, (b) consideration of flood control reservoirs at full pool level, and (c) conservatism of coincident flow rates and water surface elevations. Flood waves produced by postulated dam failure scenarios should be routed to the proposed plant site to conservatively estimate the most severe flood water surface elevation that may affect SSC important to safety.					
2.4.4.2	Domino-Type or Cascading Dam Failures: To meet the requirements of GDC 2, 10 CFR 52.17, 10 CFR Part 100, and 10 CFR 100.23(d), an appropriate configuration of the cascade of dam failures and its potential to produce the largest flood adjacent to the plant site is needed. Several possible cascading dam failures should be investigated, including those induced by seismic and hydrological failures. The failure modes should be conservatively chosen and the resulting flood waves should be conservatively routed to the proposed plant site to estimate the most severe flood water surface elevation that may affect SSC important to safety.					
2.4.4.3	Dynamic Effects on Structures: To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, an estimate of dynamic effects of flood waves, such as velocities and momentum fluxes, on SSC important to safety is needed. If a potential hazard to SSC important to safety exists from dynamic effects of flood waves, it should be documented and included in the design bases of affected facilities.					
2.4.4.4	Loss of Water Supply Due to Failure of a Downstream Dam: To meet the requirements of GDC 2, 10 CFR 52.17, 10 CFR Part 100, and 10 CFR 100.23(d), an assessment regarding loss of safety-related water supply to the plant caused by failure of a downstream dam is needed. If the possibility of loss of safety-related water supply exists, it should be documented and the design of safety-related water supply to the plant should be reassessed.					

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2.4.4.5	Effects of Sediment Deposition and Erosion: To meet the requirements of GDC 2, 10 CFR 52.17, 10 CFR Part 100, and 10 CFR 100.23(d), an assessment is needed regarding loss of functionality of safety-related water supply to the plant caused by blockages due to sediment deposition or erosion during the dam failure-induced flood event. If a hazard exists that may lead to loss of functionality of safety-related water supply, it should be documented and the design of the safety-related water supply to the plant should be reassesed.					
2.4.4.6	Failure of Onsite Water Control or Storage Structures: To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, an assessment is needed regarding the failure of any onsite water control or storage structures that may cause flooding of SSC important to safety. If a hazard exists that may lead to flooding of SSC important to safety, it should be documented and included in the design bases of affected facilities.					
2.4.4.7	Consideration of Other Site-Related Evaluation Criteria: The potential effects of site-related proximity, seismic, and non-seismic information as they relate to flooding due to upstream dam failures and loss of safety-related water supply due to blockages and failures of downstream dam failures adjacent to and on the plant site and site regions are needed to meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100.					
2.4.5 (Rev. 3, March 2007)	Probable Maximum Surge and Seiche Flooding					
2.4.5.1	Probable Maximum Hurricane. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the probable maximum hurricane and the probable maximum storm surge, i.e., the storm surge induced by the PMH, are needed. The PMH, as defined by NOAA NWS Report 23, should be estimated for coastal locations that may be exposed to these events. If a PMH is not considered a design basis for the proposed site, documentation of the reasons should					

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	be provided. The storm surge induced by the PMH should be estimated as recommended by Regulatory Guide 1.59, supplemented by current best practices.					
2.4.5.2	Probable Maximum Wind Storm. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the probable maximum wind storm and the storm surge induced by the PMWS are needed. The PMWS should be considered for locations along the Pacific and North Atlantic Coasts, and near large bodies of water such as the Great Lakes. The storm surge induced by the PMWS should be estimated as recommended by Regulatory Guide 1.59, supplemented by current best practices.					
2.4.5.3	Seiche and Resonance. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of seiche and resonance in water bodies induced by meteorological causes, tsunamis, and seismic causes are needed. An analysis of the interaction of seiche waves with the geometry of the water body should be carried out to determine if an amplification of wave heights due to oscillations at the natural periodicity of the water body is possible. An estimate of the minimum water surface elevation during the seiche activity should be provided to evaluate if safety-related water supply to the plant may be affected.					
2.4.5.4	Wave Runup. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, an estimate of wind-induced wave runup under PMH or PMWS winds is needed. The PMH or PMWS winds should be used to estimate wave runup as recommended by the U.S. Army Corps of Engineers (USACE) Coastal Engineering Manual.					
2.4.5.5	Effects of Sediment Erosion and Deposition. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, an assessment of loss of functionality of safety-related water supply to the plant caused by blockages due to sediment					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	deposition or erosion during the storm surge or seiching is needed. If a hazard to SSC important to safety exists from sediment erosion and deposition, it should be documented and included in the design bases of these SSC.					
2.4.5.6	Consideration of Other Site-Related Evaluation Criteria. The potential effects of site-related proximity, seismic, and non-seismic information as they relate to flooding and loss of safety-related water supply due to surge and seiche adjacent to the plant site and site regions are needed to meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100.					
2.4.6 (Rev. 3, March 2007)	Probable Maximum Tsunami Hazards					
2.4.6.1	Historical Tsunami Data. The application should provide a complete description of historical tsunami data near the proposed plant site. This description should be sufficient to establish the history of tsunamis and tsunami-like waves in the vicinity of the site.					
2.4.6.2	Probable Maximum Tsunami. The application should provide an assessment of the PMT for the proposed site. The PMT assessment should include a review of tsunamigenic sources from historical, geological, and physical data, both near and far field, relevant to the proposed plant site. If no tsunami hazard exists for the proposed site, it should be so stated with justification based on the history and location of the proposed site. The tsunamigenic sources in this review should include earthquakes, submarine and sub-aerial landslides, and volcanoes. The characteristics of tsunamigenic sources should be described including parameter values associated with the PMT. The results from numerical simulations of PMT waves towards the proposed site should be provided. This simulation should use shallow water wave approximation where appropriate,					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	and use nonlinear wave dynamics where the approximation is not valid.					
2.4.6.3	Tsunami Propagation Models. The application should provide a description of the tsunami wave propagation models used in the applicant's SAR. The parameters used in the PMT wave propagation simulations should be listed and discussed with respect to their conservativeness. A discussion of all data used to input the tsunami wave propagation models should also be included.					
2.4.6.4	Wave Runup, Inundation, and Drawdown. The application should provide the extents and durations of inundation and drawdown near the proposed site. The methods and models used to simulate inundation and drawdown caused by the PMT should be described. The parameters used in the simulation of inundation and drawdown should be discussed with respect to their conservativeness. The maximum extents and the longest durations of inundation and drawdown should be provided. These effects should be considered in establishing the design bases of the affected safety-related SSC.					
2.4.6.5	Hydrostatic and Hydrodynamic Forces. The application should provide a set of metrics that describes the hydrostatic and hydrodynamic forces caused by the PMT on the safety-related SSC. This set should include the inundation and drawdown depths, current speed, acceleration, inertial component, and momentum flux near the proposed locations of safety-related SSC. These effects should be considered in establishing the design bases of the affected safety-related SSC.					
2.4.6.6	Debris and Water-Borne Projectiles. The application should provide an assessment of the debris and water-borne projectiles that may accompany PMT currents. An assessment of the hazard posed by the debris and projectiles on safety-related SSC should be provided. These effects should be considered in establishing the design bases of the affected safety-					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	related SSC.					
2.4.6.7	Effects of Sediment Erosion and Deposition. The application should provide an assessment of the effects of sediment erosion and deposition near the proposed locations of safety-related SSC. A description of and an estimate of these effects on the design bases of safety-related SSC should be provided. These effects should be considered in establishing the design bases of the affected safety-related SSC.					
2.4.6.8	Consideration of Other Site-Related Evaluation Criteria. The application should provide an evaluation of the potential effects of site-related proximity, seismic, and non-seismic information as they affect tsunamis near the plant site and site regions. This assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.					
2.4.7 (Rev. 3, March 2007)	Ice Effects					
2.4.7.1	Historical Ice Accumulation: The application should include a complete history of ice formation at and in the vicinity of the site. A thorough listing of types of ice formations (ice jams, ice dams, floes, ridges, frazil, etc.), locations and durations of these formations, and descriptions of hydrometeorological characteristics accompanying these formations should be provided that are sufficient to establish the history of ice-formation at and in the vicinity of the site.					
2.4.7.2	High and Low Water Levels: The application should include estimates of water levels resulting from potential ice flooding or low flows. Flooding from collapse of an upstream ice dam or an ice jam should be considered. Backwater effects from a downstream ice dam or an ice jam that may result in flooding at the proposed site should also be					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	considered. The suggested criteria of Regulatory Guide 1.27 apply when the water supply comprises part of the ultimate heat sink.					
2.4.7.3	Ice Sheet Formation: The application should include estimates of the most severe ice-sheet formation in water storage reservoirs. The reduction in liquid water storage capacity of water storage reservoirs due to the presence of the ice sheet should be estimated. The suggested criteria of Regulatory Guide 1.27 apply when the water supply comprises part of the ultimate heat sink.					
2.4.7.4	Ice-induced Forces and Blockages: The application should provide estimates of the most severe ice-induced forces on safety-related SSC. The forces resulting from the most severe ice sheet interacting with safety-related SSC should be estimated. An assessment regarding formation of frazil ice at and in the vicinity of the site is needed. Blockages from frazil of safety-related intakes should be assessed. Ice blockage of rivers, streams, and estuaries, both upstream and downstream of the site, should be determined. The suggested criteria of Regulatory Guide 1.27 apply when the water supply comprises part of the ultimate heat sink.					
2.4.7.5	Consideration of Other Site-Related Evaluation Criteria: The application should demonstrate that the potential effects of site-related proximity, seismic, and non-seismic information as they relate to worst-case icing scenarios adjacent to and on the plant site and site regions are appropriately take into account.					
2.4.8 (Rev. 3, March 2007)	Cooling Water Canals and Reservoirs					
2.4.8.1	Hydraulic Design Bases for Protection of Structures: To meet the requirements of GDC 1, GDC 2, 10 CFR 52.17, and 10 CFR Part 100, a complete description of the hydraulic design bases for protection of structures is needed. These design bases should be consistent with site characteristics identified by staff					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	during review of other SAR sections with respect to the flood water surface elevations, hydrodynamic forces, wind-induced waves and runup, erosion, and sedimentation.					
2.4.8.2	Hydraulic Design Bases of Canals: To meet the requirements of GDC 1, GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, a complete description of the hydraulic design bases related to the capacity, protection against wind waves, erosion, sedimentation, and freeboard, and the ability to withstand a probable maximum flood (PMF), surges, etc., is needed. These design bases should be consistent with the site characteristics identified in other hydrology sections of the SAR and with the proposed plant requirements.					
2.4.8.3	Hydraulic Design Bases of Reservoirs: To meet the requirements of GDC 1, GDC 2, 10 CFR 52.17, and 10 CFR Part 100, a complete description of the design bases of safety-related reservoirs related to their capacity, PMF design basis, wind wave and runup protection, discharge facilities (e.g., low-level outlet, spillways, etc.), outlet protection, freeboard, and erosion and sedimentation processes is needed. These design bases should be consistent with the site characteristics identified in other hydrology sections of the SAR and with the proposed plant requirements.					
2.4.8.4	Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of GDC 1, GDC 2, 10 CFR 52.17, and 10 CFR Part 100, a complete description of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated design bases of safety-related canals and reservoirs is needed. This description should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.					
2.4.9 (Rev. 3, March 2007)	Channel Diversions					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
2.4.9.1	Historical Channel Diversions: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, a complete history of channel diversions at and in the vicinity of the site is needed. A thorough listing of types of phenomena (landslides, channel erosion, breached dikes, etc.), locations and durations of these events, and descriptions of hydrogeological characteristics accompanying these events should be included. This description should be sufficient to establish the history of channel diversion in the vicinity of the site (this review includes the site and adjacent watersheds displaying similar channel characteristics).					
2.4.9.2	Regional Topographic Evidence: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, a description of regional topographic evidence as it relates to channel diversions is needed. This description should be accompanied by data where possible and should be sufficient to make an assessment of the possibility of channel diversion near the site that may affect SSC important to safety.					
2.4.9.3	Ice Causes: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, estimates of the most severe ice-induced channel diversion are needed. These setimates should be consistent with the estimates in the applicant's SAR Section 2.4.7 (Ice Effects).					
2.4.9.4	Flooding of Site Due to Channel Diversions: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, estimates of the most severe channel diversion induced forces on SSC important to safety are needed. These estimates should be sufficient to demonstrate that SSC important to safety can withstand these forces without loss of their ability to function. A description of mitigation measures for flooding from channel diversions should be provided, and it should be demonstrated that these measures are consistent with the Commission's regulations regarding performance of SSC					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	important to safety.					
2.4.9.5	Human-Induced Causes of Channel Diversion: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, an assessment of the potential for human-induced channel diversions, in the vicinity of the site (e.g., land-use changes, diking, channelization, armoring or failure of such structures) is needed. An assessment of high and low water levels during channel diversion should be provided.					
2.4.9.6	Alternate Water Sources: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, assessments of alternate water sources and operating procedures are needed. These assessments should be consistent with SAR Section 2.4.11 (Low Water Considerations) and with SAR Section 2.4.14 (Technical Specifications and Emergency Operation Requirements).					
2.4.9.7	Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of GDC 1, GDC 2, 10 CFR 52.17, and 10 CFR Part 100, a description of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated worst-case channel diversion scenario for the proposed plant site is needed. This description should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.					
2.4.10 (Rev. 3, March 2007)	Flooding Protection Requirements					
2.4.10.1	Safety-related Facilities Exposed to Flooding. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, identification of all SSC exposed to flooding is needed. This identification should be consistent with site characteristics identified by the staff during review of other SAR sections with respect to flood water surface elevations, hydrodynamic forces,					

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	and coincident wind-induced waves and runup.					
2.4.10.2	Type of Flood Protection. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, an evaluation of the applicant's proposed flood protection measures is needed. This evaluation should assess the adequacy of protection provided to the SSC exposed to flooding and should be consistent with site characteristics identified in other SAR sections.					
2.4.10.3	Emergency Procedures. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, a listing of proposed emergency procedures is needed. The estimated warning time required to implement each of these procedures should be provided.					
2.4.10.4	Consideration of Other Site-Related Evaluation Criteria. To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, an assessment regarding the potential effects of site-related proximity, seismic, and non-seismic information on the postulated flooding protection is needed. The assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.					
2.4.11, (Rev. 3, March 2007)	Low Water Considerations					
2.4.11.1	Low Water from Drought: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, a complete history of low water conditions at and in the vicinity of the site is needed. A thorough listing of types of phenomena, locations and durations of these events, and descriptions of hydrometeorological characteristics accompanying					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	these events should be included. These listings and descriptions should be sufficient to establish the history of droughts in the vicinity of the site. The staff will evaluate the applicant's evidence as it relates to low water considerations. If the staff disagrees with the applicant's conclusions, they will request additional information. The applicant should fully document and justify its estimates or accept the staff's estimates and redesign SSC important to safety affected by low water levels. The suggested criteria of Regulatory Guide 1.27 apply when the water supply comprises part of the ultimate heat sink.					
2.4.11.2	Low Water from Other Phenomena: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, a complete history of low water conditions, caused by phenomena other than a drought, at and in the vicinity of the site is needed. A thorough listing of types of phenomena, locations and durations of these events, and descriptions of hydrometeorological and hydrogeological characteristics accompanying these events should be included. These listings and descriptions should be sufficient to establish the most severe low water event due to these phenomena reasonably possible in the vicinity of the site. These estimates of low water events caused by other phenomena should be consistent with the estimates in the respective SAR sections where review of these individual phenomena is carried out. In case of disagreement between the staff's and the applicant's conclusions, the applicant should fully document and justify its conclusions or accept the staff's conclusions and redesign any SSC important to safety that may be affected by low water events.					
2.4.11.3	Effect of Low Water on Safety-Related Water Supply: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, a thorough description of all safety-related water supply requirements and the effects of the most severe low water event reasonably possible at or in the vicinity of the site is needed. The staff will review the proposed requirements of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	plant with respect to the available water during the most severe low water event to assess the reliability of the proposed safety-related water supply. In case of disagreement between the staff's and the applicant's conclusions, the applicant should fully document and justify its conclusions or accept the staff's conclusions and redesign the safety-related water supply.					
2.4.11.4	Water Use Limits: To meet the requirements of GDC 2, GDC 44, 10 CFR 52.17, and 10 CFR Part 100, a thorough description of water use and discharge limitations (both physical and legal), already in effect or under discussion by responsible Federal, regional, State, or local authorities, that may affect water supply at the plant that have been considered and are substantiated by reference to reports of the appropriate agencies is needed. The staff will review these water uses and use limitations to determine the reliability of the proposed safety-related water supply to the plant. In case of disagreement between the staff's and the applicant's conclusions, the applicant should fully document and justify its conclusions or accept the staff's conclusions and redesign the safety-related water supply.					
2.4.11.5	Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, the applicant should provide an assessment of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated worst-case low-flow scenario for the proposed plant site. This assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.					
2.4.12	Groundwater					
2.4.12.1	Local and Regional Groundwater Characteristics and Use: To meet the requirements of 10 CFR 50.55a, GDC 2, GDC 4, GDC 5, 10 CFR 100.20(c)(3), 10 CFR 100.23(d), and 10 CFR 100.10(c) or 100.20(c), a complete description of regional and					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	local groundwater aquifers, sources, and sinks, local and regional groundwater use, present and known and likely future withdrawals, regional flow rates, travel time, gradients, and velocities, subsurface properties that affect movement of contaminants in the groundwater, groundwater levels including their seasonal and climatic fluctuations, groundwater monitoring and protection requirements, and any man-made changes with a potential to affect regional groundwater characteristics over a long period of time is needed. This description should be sufficient to define local and regional groundwater characteristics and groundwater use at and in the vicinity of the site.					
2.4.12.2	Effects on Plant Foundations and other Safety-Related Structures, Systems, and Components: To meet the requirements of 10 CFR 50.55a, GDC 2, GDC 4, GDC 5, 10 CFR 100.20(c)(3), 10 CFR 100.23(d), and 10 CFR 100.10(c) or 100.20(c), a complete description of the effects of groundwater levels and other hydrodynamic effects on the design bases of plant foundations and other SSC important to safety is needed. This description should be sufficient to determine any adverse effects of groundwater on plant foundations and SSC important to safety.					
2.4.12.3	Reliability of Groundwater Resources and Systems Used for Safety-Related Purposes: To meet the requirements of 10 CFR 50.55a, GDC 2, GDC 4, GDC 5, 10 CFR 100.20(c)(3), 10 CFR 100.23(d), and 10 CFR 100.10(c) or 100.20(c), a complete description of all SSC important to safety that depend on groundwater is needed. Sufficient data and analyses regarding the reliability of the groundwater source as well as that of these systems should be provided.					
2.4.12.4	Reliability of Dewatering Systems: To meet the requirements of 10 CFR 50.55a, GDC 2, GDC 4, GDC 5, 10 CFR 100.20(c)(3), 10 CFR 100.23(d), and 10 CFR 100.10(c) or 100.20(c), a complete description of the site					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	dewatering system, including its reliability to maintain the groundwater conditions within the groundwater design bases of SSC important to safety is needed. Sufficient information should be provided to determine if the dewatering system is a safety-related system as proposed in the plant design. This information should also be sufficient to determine if the dewatering system is required during plant operation.					
2.4.12.5	Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of 10 CFR 50.55a, GDC 2, GDC 4, GDC 5, 10 CFR 100.20(c)(3), 10 CFR 100.23(d), and 10 CFR 100.10(c) or 100.20(c), the applicant's assessment of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated worst-case scenario related to groundwater effects for the proposed plant site is needed. This assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.					
2.4.13 (Rev. 3, March 2007)	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters					
2.4.13.1	Alternate Conceptual Models: Alternate conceptual models of hydrology in the vicinity of the site are reviewed. The description of these alternate conceptual models should be sufficient to bound the hydrogeological conditions at the site that define the transport of radioactive liquid effluent in ground and surface water environments.					
2.4.13.2	Pathways: The bounding set of plausible surface and subsurface pathways from the points of release are reviewed. The description of these pathways should provide sufficient information including data to ensure that the bounding set of plausible pathways that may result in the worst-case contamination are adequately identified. Estimates of physical parameters necessary to calculate the transport of liquid effluent from the points of release to the site of existing or known and likely future users should be described.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
2.4.13.3	Characteristics that Affect Transport: Radionuclide transport characteristics of the groundwater environment with respect to existing and known and likely future users should be described. Estimates and bases for coefficients of dispersion, adsorption, groundwater velocities, travel times, gradients, permeabilities, porosities and potentiometric map or piezometric levels between the site and existing or known and likely future surface and groundwater users should be described and should be consistent with site characteristics and conform to the stipulation of 10 CFR 100.20(c)(3).					
2.4.13.4	Consideration of Other Site-Related Evaluation Criteria: The applicant's assessment of the potential effects of site-proximity hazards, seismic, and non-seismic events on the radioactive concentration from the postulated tank failure related to accidental release of radioactive liquid effluents to ground and surface waters for the proposed plant site is needed. This assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.					
2.4.13.5	Branch Technical Position BTP 11-6 provides guidance in assessing a potential release of radioactive liquids following the postulated failure of a tank and its components, located outside of containment, and impacts of the release of radioactive materials at the nearest potable water supply, located in an unrestricted area, for direct human consumption or indirectly through animals, crops, and food processing.					
2.4.14 (Rev. 3, March 2007)	Technical Specifications and Emergency Operation Requirements					
2.4.14.1	Bases for Emergency Actions: To meet the requirements of 10 CFR 50.36, GDC 2, 10 CFR 52.17, and 10 CFR Part 100, an assessment of the hydrological bases for emergency actions is needed. These bases should be consistent with site characteristics identified by the staff during review of other SAR sections with respect to flood water surface					

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	elevations, hydrodynamic forces, coincident wind-induced waves and runup, and water supply limitations caused by droughts and other natural phenomena.					
2.4.14.2	Available Response Time: To meet the requirements of 10 CFR 50.36, GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of available response times to initiate and complete emergency procedures are needed. These estimates are derived from the analysis of the controlling hydrological events and should be consistent with site characteristics identified during the staff's review of other SAR sections.					
2.4.14.3	Technical Specifications: To meet the requirements of 10 CFR 50.36, GDC 2, 10 CFR 52.17, and 10 CFR Part 100, the applicant's proposed technical specifications related to emergency procedures are reviewed. These technical specifications should be appropriate and should be consistent with the site characteristics.					
2.4.14.4	Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of 10 CFR 50.36, GDC 2, 10 CFR 52.17, and 10 CFR Part 100, the applicant's assessment of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated technical specifications and emergency operations is needed. This assessment should be sufficient to demonstrate that the applicant's analyses appropriately account for these effects.					
2.4.14.5	10 CFR 50, Appendix A, General Design Criterion (GDC) 44, for CP and OL applications, as it relates to providing an ultimate heat sink for normal operating and accident conditions.					
2.5.1 (Rev. 4, March 2007)	2.5.1 Basic Geologic and Seismic Information					
2.5.1.1	Regional Geology (SAR Section 2.5.1.1) In meeting requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23, SAR Section 2.5.1.1 will					

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	be considered acceptable if a complete and documented discussion is presented for all geologic (including tectonic and nontectonic), geotechnical, seismic, and geophysical characteristics, as well as conditions caused by human activities, deemed important for safe siting and design of the plant. This section should contain a review of regional geologic and tectonic history, tectonic features (with emphasis on the Quaternary period), structural geology, seismology, paleoseismology, physiography, geomorphology, stratigraphy, and lithology within a distance of 320 km (200 mi) from the site (i.e., the "site region") to provide a framework within which significance to safety can be evaluated in regard to geology, seismology, and conditions caused by human activities. Geologic maps and cross-sections constructed at scales adequate to illustrate pertinent regional features should be included in the application.					
2.5.1.2	Site Geology (SAR Section 2.5.1.2) In meeting requirements of GDC 2 in Appendix A of 10 CFR Part 50,10 CFR 52.17, 10 CFR 100.23, and regulatory positions presented in Regulatory Guides 1.165, 1.132, 1.138, 1.198, 1.208, and 4.7, SAR Section 2.5.1.2 will be considered acceptable if it contains a description and evaluation of geologic (including tectonic and non-tectonic) features, geotechnical characteristics, seismic conditions, and conditions caused by human activities at appropriate levels of detail within areas defined by circles drawn around the site using radii of 40 km (25 mi) for site vicinity, 8 km (5mi) for site area, and 1 km (0.6 mi) for site location. This subsection should contain the following information, and geologic maps and cross-sections constructed at scales adequate to clearly illustrate pertinent features in the site vicinity and site area and at the site location should be included in the application. a. Structural geology, including identification and characterization of faults, joints, and other tectonic					

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	deformation features; and discussion of the relationships between these features and regional tectonic structures. b. Seismicity, including identification of historical and instrumentally-recorded earthquakes; identification and characterization of any local seismic sources; and discussion of the relationships between local seismicity and regional tectonic structures and seismic sources. c. Geologic and tectonic history, particularly for the Quaternary Period, and its relationships to regional geologic and tectonic history. d. Evidence for paleoseismicity, or a lack of it. e. Stratigraphy and lithology of rock units and relationships to regional stratigraphic and lithologic characteristics. f. Physiography and geomorphology. g. Engineering significance of geologic and geotechnical characteristics of features and materials, including foundation materials, related to: (1) Dynamic behavior during prior earthquakes. (2) Zones of mineralization, alteration, irregular or deep weathering, or structural weakness in surface or subsurface materials. (3) Unrelieved residual stresses in bedrock. (4) Subsurface materials that could be weak or unstable due to mineralogy or physical properties. (5) Karst features in limestone terranes. (6) Effects of human activities. h. Potentially unstable natural or man-made slopes. i. Groundwater conditions, including perched aquifers.					
2.5.2 (Rev. 4, March 2007)	2.5.2 Vibratory Ground Motion					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
2.5.2.1	2.5.2.1 Seismicity. To meet the requirements in 10 CFR 100.23, this subsection is accepted when the complete historical record of earthquakes in the region is listed and when all available parameters are given for each earthquake in the historical record. The listing should include all earthquakes having Modified Mercalli Intensity (MMI) greater than or equal to IV or magnitude greater than or equal to 3.0 that have been reported within 320 km (200 miles) of the site. Large earthquakes outside of this area that would impact the SSE, should be reported. A regional-scale map should be presented showing all listed earthquake epicenters and should be supplemented by a larger-scale map showing earthquake epicenters of events within 80 km (50 miles) of the site. The following information concerning each earthquake should be provided whenever it is available: epicenter coordinates, depth of focus, date, origin time, highest intensity, magnitude, seismic moment, source mechanism, source dimensions, distance from the site, and any strong-motion recordings (sources from which the information was obtained should be identified). All magnitude designations such as m_b, M_L, M_s, Mw should be identified. In the Central and Eastern United States (CEUS), relatively little information is available on magnitudes for historic earthquakes which are reported but for which there are no instrumental recordings; hence, it may be appropriate to rely on intensity observations (descriptions of earthquake effects) or the dimensions of the area in which the event was felt to estimate magnitudes of historic events (e.g., Refs. 11). In addition, any reported earthquake-induced geologic failure, such as liquefaction (including paleoseismic evidence of large prehistoric earthquakes), landsliding, landspreading, and lurching, should be described completely, including the estimated level of strong motion that induced failure and the physical properties of the materials. The completeness of the earthquake history of the region is determined by comparison to published sources of					

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	information. When conflicting descriptions of individual earthquakes are found in the published references, the staff should determine which is appropriate for licensing decisions.					
2.5.2.2	2.5.2.2 Geologic and Tectonic Characteristics of Site and Region. Seismic sources identified and characterized by the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI) were used for studies in the CEUS in the past. For CEUS sites, the LLNL and EPRI seismic source data bases may need to be updated. Therefore to meet the requirements of 10 CFR 100.23, this subsection is acceptable when adequate information is provided to demonstrate: (1) that a thorough investigation has been conducted to identify seismic sources that could be significant in estimating the seismic hazards of the region if they exist; and (2) that existing sources (in the PSHA) are consistent with the results of site and regional investigations or the sources have been updated in accordance with Appendix E of Regulatory Guide 1.165 or Appendix C of Regulatory Guide 1.208. For sites where the LLNL or EPRI data bases have not been used, and it is necessary to identify and characterize seismic sources in meeting the requirements of 10 CFR 100.23, adequate information must be provided in this subsection to demonstrate that all seismic sources that are significant in determining the earthquake potential of the region have been identified, or that an adequate investigation has been carried out to provide reasonable assurance that there are no unidentified significant seismic sources. Information presented in Section 2.5.1 of the applicant's safety analysis report (SAR) and information from other sources dealing with the current tectonic regime should be developed into a coherent, well-documented discussion to be used as the basis for characterizing the earthquake-generating potential of seismic					

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	sources. Specifically, each seismic source, any part of which is within 320 km (200 miles) of the site, should be identified. In the CEUS, the seismic sources will most likely be seismogenic sources with large regions of diffuse seismicity, each characterized by its own recurrence model. The staff interprets seismogenic sources to be regions of assumed uniform seismicity (same frequency of occurrence) distinct from the seismicity of the surrounding area. The proposed seismogenic sources may be based on seismicity studies, differences in geologic history, differences in the current tectonic regime, or other tectonic considerations. The staff considers that the most important factors for the determination of seismic sources include both (1) development and characteristics of the current tectonic regime of the region that is most likely reflected in the Quaternary period (approximately the last 1.8 million years and younger geologic history) and (2) the pattern and level of historical seismicity. Those characteristics of geologic structure, tectonic history, present and past stress regimes, and seismicity that distinguish the various seismic sources and the particular areas within those sources where historical earthquakes have occurred should be described. Alternative regional tectonic models derived from available literature should be discussed. The model that best conforms to the observed data is accepted. In addition, in those areas where there are capable tectonic sources, the results of the additional investigative requirements described in SRP Section 2.5.1 must be presented. The discussion should be augmented by a regional-scale map showing the seismic sources, earthquake epicenters, locations of geologic structures, and other features that characterize the seismic sources.					
2.5.2.3	2.5.2.3 Correlation of Earthquake Activity with Seismic Sources. To meet the requirements in 10 CFR 100.23, acceptance of this subsection is based on the development of the relationship					

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	between the history of earthquake activity and seismic sources of a region. For CEUS sites, when the GMRS is determined using LLNL or EPRI PSHA data bases, and Regulatory Guide 1.165 or Regulatory Guide 1.208, this subsection is acceptable when adequate information is provided to demonstrate (1) that a thorough investigation has been conducted to assess the seismicity and identify seismic sources that could be significant in estimating the seismic hazards of the region if they exist, and (2) that existing sources (in the PSHA) are consistent with the results of site and regional investigations or the sources have been updated in accordance with the Appendix E of Regulatory Guide 1.165 or the Appendix C of Regulatory Guide 1.208. For sites where LLNL or EPRI data bases are not used and it is necessary to identify and characterize seismic sources in meeting the requirements of 10 CFR 100.23, adequate information must be provided in this subsection to demonstrate that all seismic sources that are significant in determining the earthquake potential of the region have been identified, or that an adequate investigation has been carried out to provide reasonable assurance that there are no unidentified significant seismic sources. The applicant's presentation is accepted when the earthquakes discussed in Subsection 2.5.2.1 of the SAR are shown to be associated with seismic sources. Whenever an earthquake hypocenter or concentration of earthquake hypocenters can be reasonably correlated with geologic structures, the rationale for the association should be developed considering the characteristics of the geologic structure (including geologic and geophysical data, seismicity, and the tectonic history) and the regional tectonic model. The discussion should include identification of the methods used to locate the					

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	earthquake hypocenters, an estimation of their accuracy, and a detailed account that compares and contrasts the geologic structure involved in the earthquake activity with other areas within the seismogenic source. Particular attention should be given to determining the recency and level of activity of faults with which instrumentally located earthquake hypocenters may be associated. Acceptance of the proposed seismic sources (those identified by the investigations) is based on the staff's independent review of the geologic and seismic information presented by the applicant and available in the scientific literature.					
2.5.2.4	2.5.2.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquakes. For CEUS sites relying on LLNL or EPRI methods and data bases, the staff will review the applicant's PSHA, including the underlying assumptions and how the results of the site investigations are used to update the existing sources in the PSHA, how they are used to develop additional sources, or how they are used to develop a new data base. To meet the requirements of 10 CFR 100.23, this subsection is acceptable when adequate information is provided to demonstrate that the PSHA adequately characterizes the regional and local seismic hazard with respect to ground motion and its uncertainty and the controlling earthquakes for the site. In addition to seismic sources, the staff will also review the ground motion attenuation models used in the PSHA. For the CEUS, the staff has previously reviewed and accepted ground motion models developed by EPRI (Ref. 14). Use of the EPRI ground motion models (Ref. 14) is acceptable as long as an adequate investigation has been carried out to provide reasonable assurance that there are no significant updates or new models that may impact on the results of the PSHA.					

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	For sites located in the WUS, the latest attenuation relationships (based on current and extensive strong motion data bases) should be used for the PSHA. Specifically, the staff will review (1) the rationale for the inclusion of each model, (2) consideration of both aleatory and epistemic uncertainty, (3) model weighting, (4) magnitude conversion, (5) distance measure adjustments, and (6) the model parameters for each spectral frequency. For each PSHA, the staff will also examine how logic trees for seismic source parameters and models (maximum magnitude, recurrence, source geometry) and attenuation models were used to incorporate model uncertainty. Epsilon, the number of standard deviations included in defining the distribution of ground motions for each magnitude and distance scenario, can have a significant impact on the results of the PSHA. The staff will review the ground motion models used for the PSHA to ensure that the value for epsilon is large enough such that natural aleatory variability in the ground motions is adequately addressed. A recent study (Ref. 17) found no technical basis for truncating the ground motion distribution at a specified number of standard deviations (epsilons) below that implied by the strength of the geologic materials. Even though very large ground motions have a low probability of occurrence, when the hazard is calculated for low annual frequencies of exceedance, low probability events need to be considered. For determining recurrence relationship parameters, the entire seismicity catalog developed in Subsection 2.5.2.1, should be used. For the seismic hazard evaluation, the use of Cumulative Absolute Velocity (CAV) provides an alternative approach to the use of minimum magnitude truncation for the PSHA (Ref. 16). The staff will review the controlling earthquakes and associated ground motions at the site derived from the applicant's PSHA to					

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	be sure that they adequately represent the local and regional seismic hazard as represented by both historical seismicity and paleoseismicity. The applicant's probabilistic analysis, including the derivation of controlling earthquakes, is considered acceptable if it follows the procedures in Appendix C of Regulatory Guide 1.165 or Appendix D of Regulatory Guide 1.208. For applicants that use Regulatory Guide 1.165, one pair of low and high frequency controlling earthquakes should be determined at the reference probability value. For applicants that use Regulatory Guide 1.208, three pairs of low and high frequency controlling earthquakes should be determined for the mean 10^{-4}, 10^{-5}, and 10^{-6} annual probabilities. For applicants that do not use PSHA, the staff will review the method used to derive the controlling earthquakes and, in particular, the methods used to address uncertainties on a case-by-case basis. For sites not in the CEUS, the staff will review the PSHA or other methods in detail. As in the reviews of CEUS sites, the staff will particularly review the approaches used to address uncertainties. The staff will assess the controlling earthquakes for the site derived from the applicant's method to be sure that they adequately represent the local and regional seismic hazard as represented by both historical seismicity and paleoseismicity. The determination of the controlling earthquakes and the seismic hazard information base for sites not in the CEUS is carried out using procedures similar to those used for CEUS. However, because of differences in seismicity rates and ground motion attenuation characteristics at these sites, alternative magnitude-distance parameters may have to be used. In addition, if Regulatory Guide 1.165 is used, an alternative reference probability may also have to be developed, particularly for sites in the active plate margin region and for sites at which a known tectonic structure dominates the hazard.					

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	The staff will perform an independent evaluation of the earthquake potential associated with each seismic source that could affect the site. For guidance in evaluating the earthquake potential and characterizing the uncertainty for sites that are assessed using methods other than the LLNL or EPRI methods and data bases, or for sites outside the CEUS, refer to the Senior Seismic Hazard Analysis Committee (SSHAC) Report (Ref. 10).					
2.5.2.5	2.5.2.5 Seismic Wave Transmission Characteristics of the Site. In the PSHA procedure described in Regulatory Guide 1.165, the controlling earthquakes are determined for generic rock conditions. Site amplification studies are performed in a distinct separate step as a part of the determination of the GMRS. In this section, the applicant's site amplification studies are reviewed in conjunction with the geotechnical and structural engineering reviews. Particular emphasis is placed on how the uncertainties inherent in this process are addressed. To meet the requirements of 10 CFR 100.23, this subsection is acceptable when adequate information is provided to demonstrate that the site response analysis adequately estimates both the mean and variability of the site response in accordance with Regulatory Position 4 and Appendix E of Regulatory Guide 1.208. To be acceptable, the seismic wave transmission characteristics (amplification or deamplification) of the materials overlying bedrock at the site are described as a function of the significant frequencies (Ref. 11). The following material properties should be determined for each stratum under the site: thickness, seismic compressional and shear wave velocities, bulk densities, soil index properties and classification, shear modulus and damping variations with strain level, and the water table elevation and its variations (Ref. 15). In each case, methods used to determine the properties should be described in					

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	Subsection 2.5.4 of the SAR and cross-referenced in this subsection. Where vertically propagating shear waves may produce the maximum ground motion, a one-dimensional equivalent-linear analysis or nonlinear analysis may be appropriate and is reviewed in conjunction with geotechnical and structural engineering. However, site characteristics (such as a dipping bedrock surface, topographic effects or other impedance boundaries) may require that analyses are also able to account for inclined waves. The staff will review the ground motions developed for each of the controlling earthquakes. Reference 12 and 13 contain a database of recorded time histories on rock for both CEUS and WUS. The staff will also review the simulation method (such as Monte Carlo) used to incorporate the variability in soil depth, shear wave velocities, layer thicknesses, and strain-dependent dynamic nonlinear material properties at the site. A sufficient number of simulations should be performed (at least 60) in order to define the mean and the standard deviation of the site response.					
2.5.2.6	2.5.2.6 Ground Motion Response Spectra. In this subsection, the staff reviews the applicant's procedure to determine the GMRS. If the applicant uses the reference probability approach, the GMRS are considered acceptable if they meet Regulatory Position 4 and Appendix F of Regulatory Guide 1.165. If the applicant uses the performance-based approach, the GMRS are considered acceptable if they meet Regulatory Position 5 of Regulatory Guide 1.208. The staff also reviews the method used to determine the vertical GMRS. Vertical response spectra are developed by combining the appropriate horizontal response spectra and the most recent					

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	V/H ratios for either CEUS or WUS sites. Appropriate V/H ratios should be determined from the most recent ground motion attenuation models. However, as there are currently no CEUS attenuation models that predict vertical motions, appropriate V/H ratios for CEUS sites should be developed in accordance with Regulatory Position 5 of Regulatory Guide 1.208. To meet the requirements in 10 CFR 100.23, the horizontal and vertical GMRS are determined in the free field on the ground surface. For sites with soil layers near the surface that will be completely excavated to expose competent material, the GMRS is specified on an outcrop or a hypothetical outcrop that will exist after excavation. Motions at this hypothetical outcrop should be developed as a free surface motion, not as an in-column motion. Although the definition of competent material is not mandated by regulation, a number of reactor designs have specified a shear wave velocity of 1000 fps as the definition of competent material. When the GMRS are determined as free- field outcrop motions on the uppermost in-situ competent material, only the effects of the materials below this elevation are included in the site response analysis. The time duration and number of cycles of strong ground motion are required for analysis and design of many plant components. The adequacy of the time history for structural analysis is reviewed under SRP Section 3.7.1. For evaluation of the liquefaction potential at the site, the time duration and number of cycles of strong ground motion are critical parameters and require additional consideration. If the controlling earthquakes for the site have magnitudes of less than 6, the time history selected for the evaluation of liquefaction potential must have a duration and number of strong motion cycles corresponding to at least an event of magnitude 6.					

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	REFERENCES: 10. Senior Seismic Hazard Analysis Committee, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," Lawrence Livermore National Laboratory, UCRL-ID-122160, August 1995, NUREG/CR-6372. 11. Electric Power Research Institute, 'Guidelines for Determining Design Basis Ground Motions," EPRI Report TR-102293, Vols. 1-4, May 1993. 12. R. K. McGuire, W.J. Silva, and C.J. Costantino, "A Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard and Risk-Consistent Ground motion Spectra Guidelines," NUREG/CR -6728. USNRC, Washington DC, Oct. 2001. 13. R. K. McGuire, W.J. Silva, and C.J. Costantino, "A Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Development of Hazard Risk-consistent Seismic Spectra for Two Sites," NUREG/CR -6769. USNRC, Washington DC, Oct. 2002. 14. EPRI Report 1009684, "CEUS Ground Motion Project Final Report," 2004. 15. USNRC," Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plant," Regulatory Guide 1.138. 16. EPRI Report 1012965, "Use of CAV in Determining Effects of Small Earthquakes on Seismic Hazard Analysis," 2006. 17. EPRI Report 1013105, "Truncation of the Lognormal Distribution and Value of the Standard Deviation for Ground Motion Models in the Central and Eastern United States," 2006.					
2.5.3 (Rev. 4, March 2007)	2.5.3 Surface Faulting					

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2.5.3.1	Geologic, Seismic, and Geophysical Investigations. Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23 are met and guidance in Regulatory Guides 1.165, 1.132, 1.198, 1.208, and 4.7 followed for this area of review if discussions of Quaternary tectonics, structural geology, stratigraphy, geochronologic methods used for age dating, paleoseismology, and geologic history of the site vicinity, site area, and site location are complete, compare well with studies conducted by others in the same area, and are supported by detailed investigations performed by the applicant. Site vicinity, site area, and site location-specific geologic maps and cross-sections constructed at scales adequate to clearly illustrate surficial and bedrock geology, structural geology, topography, and relationship of power plant foundations to these features should be included in the application. For coastal and inland sites near large bodies of water, similar detailed investigations are to be conducted and the application should include information regarding onshore and offshore geology and seismicity. In some cases, it may be possible to identify onshore expression of an offshore tectonic structure (i.e., a fault or fold) of potential concern. As expressed in Regulatory Guide 1.165, Appendix D, and RG 1.208, Appendix C, under this condition it is acceptable for the applicant to investigate expression of the offshore structure in the onshore environment, along with other investigations of the offshore feature when possible, to better evaluate characteristics of the tectonic structure in the site vicinity and site area and at the site location.					
2.5.3.2	Geologic Evidence, or Absence of Evidence, for Surface Tectonic Deformation. Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23 are met and guidance in Regulatory Guides 1.165, 1.132, 1.198, 1.208, and 4.7 followed					

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	for this area of review if sufficient surface and subsurface information is provided by the applicant for the site vicinity, site area, and site location to confirm presence or absence of surface tectonic deformation (i.e., faulting) and, if present, to demonstrate age of most recent fault displacement and ages of previous displacements. If surface faulting is present, the faults must be characterized in regard to fault geometry (i.e., fault orientation, length, and width), amount and sense of displacement, and recurrence rate. In addition to geologic field evidence that may indicate faulting, linear features interpreted from topographic maps, low and high altitude aerial photographs, satellite imagery, and other types of imagery should be examined and their use documented. To expedite the review process, an identification index and duplicates of remote sensing data used in the linear features study should be provided to staff for review. Data to assess presence or absence of tectonic deformation at or near the site is obtained by an applicant through conduct of surface (e.g., imagery analysis, geologic reconnaissance, and geologic mapping to define fault traces) and subsurface (e.g., using seismic instrumentation, geophysical surveys at the ground surface and in boreholes, geologic and geotechnical logging of soil materials and rock core in boreholes, and geologic mapping of trenches and test pits to define paleoseismic features and fault surfaces) investigations. Nature of geologic, seismic and paleoseismic, geophysical, and geotechnical investigations to determine whether or not undetected fault displacements or other tectonic deformation features (e.g., folds related to blind faults) are likely to exist will vary in degree of detail and extent required based on geologic complexity of the specific site. In the Central and Eastern United					

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	States (CEUS) region, defined as that part of the United States east of the Rocky Mountain Front, with the exception of the New Madrid Seismic Zone of the Central Mississippi Valley (extending from northeast Arkansas, southeast Missouri, western Tennessee, and western Kentucky to southern Illinois), the Meers Fault in southwestern Oklahoma, and the Cheraw fault in eastern Colorado, earthquake-generating faults either do not extend to the ground surface or there is insufficient overlying soil or rock for reliable age dating. In the Western United States, many capable faults are exposed at the ground surface and can be more readily characterized with respect to seismic hazard potential. In the Western region, capable tectonic sources (including faults related to subduction zones) also exist as blind faults which may be expressed at the surface or near-surface only by folding, uplift, or subsidence, and these phenomena should be taken into account by an applicant for a site located in that region.					
2.5.3.3	Correlation of Earthquakes with Capable Tectonic Sources. Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23 are met for this area of review if all reported historical earthquakes within the site vicinity are evaluated with respect to accuracy of hypocenter location and source of origin, and if all capable tectonic sources that could, based on fault orientation and length, extend into the site area or site location are evaluated with respect to potential for causing surface deformation. The application should include a plot of earthquake epicenters superimposed on a map showing local capable tectonic sources.					
2.5.3.4	Ages of Most Recent Deformation. Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23 are met for this area of review if every significant surface fault and feature associated with a blind fault, any part of which lies within the site area, is investigated in sufficient detail to demonstrate, or allow relatively accurate estimates of, age of most recent fault displacement and enable					

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	identification of geologic evidence for previous displacements (if such evidence exists). The application should also provide an evaluation of sensitivity and resolution of the exploratory geologic and geophysical techniques used that is adequate for staff to determine whether or not appropriate techniques were applied to assess age of the most recent displacement.					
2.5.3.5	Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures. Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23 are satisfied for this area of review by discussion of structural and genetic relationships between site area faulting or other tectonic deformation and the regional tectonic framework. In regions of active tectonism, it may be necessary to conduct detailed geological and geophysical investigations for assessing possible relationships of site area faults to regional faults which are known to be seismically active.					
2.5.3.6	Characterization of Capable Tectonic Sources. Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23 are met for this area of review when it has been demonstrated that investigative techniques employed by the applicant are sufficiently sensitive to identify all potential capable tectonic sources, such as faults or structures associated with blind faults, within the site area; and when fault geometry, length, sense of movement, amount of total displacement and displacement per faulting event, age of latest and any previous displacements, recurrence rate, and limits of the fault zone are provided for each capable tectonic source. Investigations must extend to at least 8 km (5 mi) beyond all plant site boundaries to encompass the site area, including for those sites adjacent to large bodies of water such as oceans, rivers, and lakes.					

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2.5.3.7	Designation of Zones of Quaternarv Deformation in the Site Region. Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23 regarding designation of zones of Quaternary deformation in the site region are met if the zone (or zones) designated by the applicant as requiring detailed faulting investigations is of sufficient length and width to include all Quaternary deformation features potentially significant to the site as described in Regulatory Guides 1.165 and 1.208.					
2.5.3.8	Potential for Surface Tectonic Deformation at the Site Location. To meet requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17, and 10 CFR 100.23 for this area of review, information must be presented by the applicant in this subsection if field investigations reveal that surface or near-surface tectonic deformation along a known capable tectonic structure (i.e., a known capable tectonic feature related to a fault or blind fault) must be taken into account at the site location. It is important to note that no commercial nuclear power plant has ever been constructed on a known capable tectonic deformation feature, and it is questionable whether it may be feasible to design for surface or near-surface tectonic displacements with any degree of confidence that safety-related plant features would remain intact and functional if displacements were to occur. Consequently, it is NRC policy to recommend that any site determined, based on results of detailed fault investigations, to lie on a surface or near-surface tectonic structure capable of displacement be prudently re-located to an alternate site by the applicant. If it becomes feasible in the future to design for surface or near-surface faulting with confidence that safety-related plant features would remain intact and functional should displacements occur, it would be necessary for an applicant to present the design basis for faulting and all supporting data in a high degree of detail.					

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2.5.4 (Rev. 3, March 2007)	2.5.4 Stability of Subsurface Materials and Foundations					
2.5.4.1	2.5.4.1 Geologic Features. In meeting the requirements of 10 CFR Parts 50 and 100, the section defining geologic features is acceptable if the discussions, maps, and profiles of the site stratigraphy, lithology, structural geology, geologic history, and engineering geology are complete and are supported by site investigations sufficiently detailed to obtain an unambiguous representation of the geology. The information must be presented in this subsection or cross-referenced to the appropriate subsection in Section 2.5.1 of the SAR. Geologic features are evaluated by conducting an independent literature search and comparing these results with the information included in the applicant's SAR. References used in reviewing this subsection include published or unpublished reports, maps, geophysical data, construction records, etc., by the USGS, other Federal agencies, State agencies, and private companies. In conjunction with the literature search, the staff and its advisors review the geological investigations conducted by the applicant. Using the references listed at the end of this section and other sources, the following questions are considered in detail: 1. Are the exploratory techniques used by the site investigator representative of the present state-of-the-art? Do the samples represent the in situ soil conditions? 2. Do the applicant's investigations provide adequate coverage of the site area and in sufficient detail to define the specific subsurface conditions with a high degree of confidence?					

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	3. Have all areas or zones of actual or potential surface or subsurface subsidence, uplift or collapse, deformation, alternation, solution cavities or structural weakness, unrelieved stresses in bedrock, or rocks or soils that might be unstable because of their physical or chemical properties been identified and adequately evaluated?					
2.5.4.2	2.5.4.2 Properties of Subsurface Materials. In meeting the requirements of 10 CFR Parts 50 and 100, the description of properties of underlying materials is considered acceptable if state-of-the-art methods are used to determine the static and dynamic engineering properties of all foundation soils and rocks in the site area. These methods are described, for example, in geotechnical journals published by the American Society of Civil Engineers (Refs. 14, 22, and 23), applicable standards published by the American Society for Testing and Materials (Ref. 15), publications of the Institution of Civil Engineers (Ref. 15), and various research reports prepared by universities (Ref. 17). The properties of foundation material must be supported by field (Refs. 19 and 20) and laboratory (Ref. 21) test records. Normally, a complete field investigation and sampling program must be performed to define the occurrence and properties of underlying materials at a given site (Ref. 18). Summary tables must be provided which catalog the important test results; test results should be plotted when appropriate. Also, a detailed discussion of laboratory sample preparation must be given when applicable. For critical laboratory tests, full details must be given, e.g., how saturation of the sample was determined and maintained during testing, transported and how the pore pressures were monitored during the experiment. The applicant should provide a detailed and quantitative					

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	discussion of the criteria used to determine that the samples were properly taken, and tested in sufficient number to define all the critical soil parameters for the site, together with their potential variability. For sites that are underlain by saturated soils and sensitive clays, it should be shown that all zones which could become unstable due to liquefaction or strain-softening phenomena have been adequately sampled and tested. The relative density of the soils at the site should be determined. The applicant must also show that the consolidation behavior of the soils as well as their static and dynamic strength have been adequately defined. The discussion should explain how the developed data is used in the safety analyses, how the test data is analyzed to generate appropriate design parameters and present a table indicating the value of the parameters used in the analyses. Properties of underlying materials are evaluated to determine whether or not the investigations performed (including laboratory and field testing) were sufficient to justify the soil and rock properties used in the foundation analyses. To determine whether sufficient investigations were performed, the staff carefully reviews the criteria developed and used by the applicant in laying out the boring, sampling and testing program and evaluates the effectiveness of the program in defining the specific foundation conditions at the site to ensure that all critical conditions have been adequately sampled and tested. If suitable criteria have not been developed and used by the applicant, the staff develops appropriate criteria, using Regulatory Guide 1.132 and the data given in the SAR, and determines if sufficient investigation and testing have been carried out. If criteria are given, the staff reviews them to determine if they are appropriate and have been implemented.					

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	If it is the staff's judgment that the applicant's investigations or testing are inappropriate or insufficient, additional investigations will be required. The final conclusion is based on professional judgment, considering the complexity of the site subsurface conditions. As part of the review, the staff must ascertain, often with the help of consultants, that state-of-the-art laboratory and field techniques and equipment are employed in determining the material properties.					
2.5.4.3	2.5.4.3 Foundation Interfaces. In meeting the requirements of 10 CFR Parts 50 and 100, the discussion of the relationship of foundations and underlying materials is acceptable if it includes (1) a plot plan or plans showing the locations of all site explorations, such as borings, trenches, seismic lines, piezometers, geologic profiles, and excavations with the locations of the safety-related facilities superimposed thereon; (2) profiles illustrating the detailed relationship of the foundations of all seismic Category I and other safety-related facilities to the subsurface materials; (3) logs of core borings and test pits; and (4) logs and maps of exploratory trenches in the application for an early site permit or COL. A supplemental report providing geologic maps and photographs of the excavations for the facilities of the nuclear power plant should be provided when available. Plot plans and profiles are reviewed by comparing the subsurface materials with the proposed locations (horizontal and vertical) of foundations and walls of all seismic Category I facilities. The profiles and plot plans are cross-checked in detail with the results of all subsurface investigations conducted at the site to ascertain that sufficient exploration has been carried out and to determine whether or not the interpretations made by the investigators are valid and the foundation design assumptions contain adequate margins of safety.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
2.5.4.4	2.5.4.4 Geophysical Surveys. In meeting the requirements of 10 CFR 100.23, the presentation of the dynamic characteristics of soil or rock is acceptable if geophysical investigations have been performed at the site and the results obtained therefrom are presented in detail. Completeness of the presentation is judged by whether or not the exploratory techniques used by the applicant yield unambiguous and useful information, whether they represent state-of-the-art exploration methods (Ref. 10), and whether the applicant's interpretations are supported by adequate field records in the SAR. Multiple measurements of dynamic properties should be incorporated to capture uncertainty in the primary parameters controlling site response behavior. See also Subsection 2.5.2.3. Staff evaluation consists of a detailed review of all geophysical explorations conducted at the site, including seismic refraction, reflection, and in-hole surveys and magnetic and gravity surveys. Consultant expertise regarding specific techniques may be drawn upon in this review. Logs of core borings, trenches, and test pits are reviewed and compared with data from the seismic surveys and other geophysical explorations. Results must be consistent or additional investigations are required, or the applicant must use the most conservative values. The staff will visit the site to examine the walls and floors of excavations at an appropriate time after licensing to confirm conditions as mapped in the open excavations with interpretations and assumptions derived during the investigation program.					
2.5.4.5	2.5.4.5 Excavation and Backfill. In meeting the requirements of 10 CFR Part 50, the presentation of the data concerning excavation, backfill, and earthwork analyses is acceptable if: 1. The sources and quantities of backfill and borrow are					

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	identified and are shown to have been adequately investigated by borings, pits, and laboratory property and strength testing (dynamic and static) and these data are included, interpreted, and summarized. 2. The extent (horizontally and vertically) of all Category I excavations, fills, and slopes are clearly shown on plot plans and profiles. 3. Compaction specifications and embankment and foundation designs are justified by field and laboratory tests and analyses to ensure stability and reliable performance. 4. The impact of compaction methods are incorporated into the structural design of the plant facilities. 5. Quality control methods are discussed and the quality assurance program described and referenced. 6. Control of groundwater during excavation to preclude degradation of foundation materials and properties is described and referenced. Excavations, backfill, and earthwork are evaluated by the staff as follows: 1. The investigations for borrow material, including boring and test pit logs, and compaction test data are reviewed and judged as to their adequacy. 2. Laboratory dynamic and static records of					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	tests performed on samples compacted to the design specifications are reviewed to ascertain that state-of-the-art criteria are met. 3. Analyses and interpretations are reviewed to ensure that static and dynamic stability requirements are met. 4. Excavation and compaction specifications and quality control procedures are reviewed to ascertain conformance to state-of-the-art conservative standards.					
2.5.4.6	2.5.4.6 Ground Water Conditions. In meeting the requirements of 10 CFR Parts 50 and 100, the analysis of groundwater conditions is acceptable if the following are included in this subsection or cross-referenced to the appropriate subsections in SRP Section 2.4 of the SAR: 1. Discussion of critical cases of groundwater conditions relative to the foundation settlement and stability of the safety-related facilities of the nuclear power plant. 2. Plans for dewatering during construction and the impact of the dewatering on temporary and permanent structures. 3. Analysis and interpretation of seepage and potential piping conditions during construction. 4. Records of field and laboratory permeability tests as well as dewatering induced settlements.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	5. History of groundwater fluctuations as determined by periodic monitoring of 16 local wells and piezometers. Flood conditions should also be considered. Groundwater conditions as they affect foundation stability are evaluated by studying the applicant's records of the historic fluctuations of groundwater at the site as obtained by monitoring local wells and springs and by analysis of piezometer and permeability data from tests conducted at the site. The applicant's dewatering plans during and following construction are also reviewed. Adequacy of these plans is evaluated by comparing with the results of the groundwater investigations and by professional judgment of groundwater and soil conditions at the site. The impact of these dewatering plans on temporary and permanent structures are evaluated.					
2.5.4.7	2.5.4.7 Response of Soil and Rock to Dynamic Loading. In meeting the requirements of 10 CFR Parts 50 and 100, descriptions of the response of soil and rock to dynamic loading are acceptable if: 1. An investigation has been conducted and discussed to determine the effects of prior earthquakes on the soils and rocks in the vicinity of the site. Evidence of liquefaction and sand cone formation should be included (Ref. 12). 2. Field seismic surveys (surface refraction and reflection and in-hole and cross-hole seismic explorations) have been accomplished and the data presented and interpreted to develop bounding P and S wave velocity profiles (Ref. 10).					

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	3. Dynamic tests have been performed in the laboratory on undisturbed samples of the foundation soil and rock sufficient to develop strain-dependent modulus- reduction and hysterietic damping properties of the soils and the results included. The section should be cross-referenced with Subsection 2.5.2.5 (Ref. 11). The soil-structure interaction analysis should be described in SRP Sections 3.7.1 and 3.7.2 and cross-referenced to this subsection. Response of soil and rock to dynamic loading and soil-structure interaction is evaluated by a detailed study of the results of the investigations and analyses performed. Specifically, the effects of past earthquakes on site soils or rocks (a requirement in SRP Section 2.5.2) are determined. The data from core borings, from geophysical investigations, and from dynamic laboratory tests such as sonic and resonant column, torsional shear and cyclic triaxial tests on undisturbed samples are evaluated. The object of the staff review is to ascertain that reasonably conservative dynamic soil and rock characteristic, together with their potential variability, are used in the design and analyses and that all the significant soil and rock strata have been considered in the analyses. In some cases, independent analyses and interpretations are carried out as outlined in SRP Section 2.5.2, or as required to verify the liquefaction analysis discussed in Subsection 2.5.4.8.					
2.5.4.8	Liquefaction Potential. In meeting the requirements of 10 CFR Parts 50 and 100, if the foundation materials at the site adjacent to and under Category I structures and facilities are saturated soils and the water table is above bedrock, then an analysis of the liquefaction potential at					

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	the site is required (Ref. 12). The need for a detailed analysis is determined by a study on a case-by-case basis of the site stratigraphy, critical soil parameters, and the location of safety-related foundations. Undisturbed samples obtained at the site and appropriate laboratory tests are required to show if the soils are likely to liquefy. Liquefaction potential assessments using both deterministic and probabilistic approaches are desirable. When the need for an in depth analysis is indicated, it may be based on cyclic triaxial test data obtained from undisturbed soil samples taken from the critical zones in the site area. The shear stresses induced in the soil by the postulated earthquake should be determined in a manner that is consistent with SRP Section 2.5.2. The criterion that should be used to determine when the soil samples tested "liquefied" should be taken as the onset of liquefaction (defined as the cycle when the pore pressure first equals the confining pressure). Test data showing the rate of pore pressure increase with number of pad cycles should be presented. If the behavior of the pore pressure is such that peak to peak axial strains greater than a few percent occur before liquefaction, then the applicant must include the effects of these strains in his assessment of the potential hazards that complete or partial liquefaction could have on the stability and settlement of any Category I structures. Nonseismic liquefaction (such as that induced by erosion, floods, wind loads on structures and wave action) should be analyzed using state-of-the-art soil mechanics principles. Liquefaction potential is reviewed by a study of the results of geotechnical investigations including boring logs, laboratory classification test data and soil profiles to determine if any of the site soils could be susceptible to liquefaction. The results of in-situ					

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	tests such as the standard penetration tests and the density and strength data obtained from undisturbed samples obtained in exploration borings are examined and, when appropriate, related to the liquefaction potential of in situ soils. If it is determined that there may be liquefaction-susceptible soils beneath the site, the applicant's site exploration methods, laboratory test program, and analyses are reviewed for adequacy and reasonableness. The analysis submitted by the applicant is reviewed in detail and compared to an independent study performed by the staff employing both deterministic and probabilistic methods as appropriate. As a minimum, the staff study consists of: 1. A review of appropriate standard penetration test results, other in-situ test data and groundwater conditions to assess liquefaction potential. 2. A careful review of conventional laboratory and cyclic triaxial test data to ensure that appropriate samples were obtained and tested from critical, liquefiable zones. 3. Confirmation that an adequate number of samples were properly tested and that the test results account for the natural variation in different samples as well as define the cyclic resistance to liquefaction of the soils. 4. An assessment of the liquefaction potential using a conservative envelope of the test data submitted. 5. A calculation of the stress induced by the earthquake that has been arrived at by an envelope of critical conditions calculated for the site based on variations in the properties of					

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	the soil strata. 6. Assurance that conservative ranges of relative density of granular soils or relative consistency of fine-grained soils are estimated. Estimates of the "safety factor" obtained from the applicant's analysis are compared to the safety margins estimated by the staff. (The applicant's plans to "eliminate" the liquefaction condition, usually by excavation and backfill, vibroflotation, or chemical grouting is evaluated as discussed in Subsections 2.5.4.5 and 2.5.4.12.) 7. An assessment of post-earthquake stability and settlements due to partial liquefaction using state-of-the-art techniques. 8. An assessment of nonseismic liquefaction based on state-of-the-art techniques.					
2.5.4.9	2.5.4.9 Earthquake Design Basis. In meeting the requirements of 10 CFR Part 50, the earthquake design basis analysis is acceptable if a brief summary of the derivation of the site-specific Ground Motion Response Spectrum (GMRS) is presented and references are included to Subsection 2.5.2.6. The staff's evaluation of the amplification characteristics of specific soils and rocks beneath the site as determined by procedures discussed in that section and in Subsections 2.5.4.2, 2.5.4.4, and 2.5.4.7 are summarized and cross-referenced herein. The review of Subsection 2.5.4.9 concentrates on determining its consistency or inconsistency with other subsections. Cross-referencing with other sections is expected.					
2.5.4.10	2.5.4.10 Static Stability. In meeting the requirements of 10 CFR Parts 50 and 100, the discussions of static analyses are acceptable if the stability of all safety-related facilities has					

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	been analyzed from a static stability standpoint including bearing capacity (Ref. 22), rebound, settlement, and differential settlements (Ref. 23) under deadloads of fills and plant facilities, and lateral loading conditions. The bearing capacity estimates must include consideration of settlements associated with the strength estimates. Field and laboratory test procedures and results must be included to document soil and rock properties used in the analyses. The applicant must show that the methods of analysis used are appropriate for the local soil conditions and the function of the facility. Static analyses of the bearing capacity and settlement of the supporting soils under the loads of fills, embankments, and foundations are evaluated by conventional, state-of-the-art methods (Ref. 18). In general, the evaluation procedure includes: 1. Determining whether or not the soil and rock properties used in the analyses represent the actual site conditions beneath the planned locations of plant facilities. The site investigation, sampling, and laboratory test programs must be adequate for this evaluation. 2. Determining whether or not the methods of analysis are appropriate for the planned earthworks, foundations, and soil conditions at the site. 3. Determining whether or not the bearing capacity, settlement, differential settlement, and tilt estimates indicate conservative and tolerable behavior of the planned plant foundations when these values are compared to design criteria and quality assurance specifications.					

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	4. Evaluation of particularly complex cases on the basis of accepted principles and techniques as supplemented by case histories and confirmatory measurement and analysis programs.					
2.5.4.11	2.5.4.11 Design Criteria. In meeting the requirements of 10 CFR Part 50, the discussion of criteria and design methods is acceptable if the criteria used for the design, the design methods employed, and the factors of safety obtained in the design analyses are described and a list of references presented. An explanation and verification of the computer analyses used and source references should be included. Site exploration, sampling, testing, and interpretation are judged with respect to completeness, care and technique, meaningful documentation, performance records for similar projects, published guidelines, and state-of-the-art practice. Design safety features, the applicant's proposed confirmatory tests and measurements, and monitoring of performance for planned safety-related foundations and earthworks are reviewed and evaluated on a case-by-case basis.					
2.5.4.12	2.5.4.12 Techniques to Improve Subsurface Conditions. In meeting the requirements of 10 CFR Part 50, the discussion of techniques to improve subsurface conditions is acceptable if plans, summaries of specifications, and methods of quality control are described for all techniques to be used to improve foundation conditions (such as grouting, vibroflotation, dental work, rock bolting, or anchors). Planned techniques to improve subsurface conditions are evaluated by reviewing the applicant's specifications and techniques for performance and quality control for such activities as grouting, excavation and backfill, vibroflotation, rock bolting,					

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	and anchoring.					
	REFERENCES: 10. Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants." 11. Regulatory Guide 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants." 12. Regulatory Guide 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites." 14. Journal of the Geotechnical and Geoenvironmental Engineering Division, Proceedings of the American Society of Civil Engineers. 15. Book of ASTM Standards, American Society for Testing and Materials. 17. Earthquake Engineering Research Center, University of California, Berkeley. 18. R.E. Hunt, “Geotechnical Engineering Investigation Handbook,” CRC Press, Taylor and Francis Group, Boca Raton FL, 2005. 19. Engineering Manual EM 1110-1-1906, "Engineering and Design Soil Sampling," U.S. Army Corps of Engineers, September 1996. 20. Engineering Manual EM 1110-2-1908, "Engineering and Design Instrumentation of Embankment Dams and Levees” U.S. Army Corps of Engineers, June 1995. 21. Engineering Manual EM 1110-2-1906, "Laboratory Soils Testing," U.S. Army Corps of Engineers, August 1986. 22. American Society of Civil Engineers, “Bearing Capacity of Soils,” Technical Engineering and Design Guide, 1994. 23. American Society of Civil Engineers,”Settlement Analysis,”					

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	Technical Engineering and Design Guide, 1994.					
2.5.5 (Rev. 3, March 2007)	2.5.5 Stability of Slopes					
2.5.5.1	2.5.5.1 Slope Characteristics. In meeting the requirements of 10 CFR Parts 50 and 100, the discussion of slope characteristics is acceptable if the subsection includes: 1. Cross sections and profiles of the slope in sufficient quantity and detail to represent the slope and foundation conditions. 2. A summary and description of static and dynamic properties of the soil and rock comprised by seismic Category I embankment dams and their foundations, natural and cut slopes, and all soil or rock slopes whose stability would directly or indirectly affect safety-related and Category I facilities. The text should include a complete discussion of procedures used to estimate, from the available field and laboratory data, conservative soil properties and profiles to be used in the analysis. 3. A summary and description of groundwater, seepage, and high and low groundwater conditions. Plot plans, cross sections, and profiles of all safety-related slopes in relation to the topography and physical properties of the underlying materials are reviewed and compared with exploratory records to ascertain that the most critical conditions have been addressed and that the characteristics of all slopes have been defined. The soil and rock test data are reviewed to					

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	ensure that there is sufficient relevant test data to verify the soil strength characteristics assumed for the slopes, dikes, and dams under analysis. The evaluation is to some extent a matter of engineering judgment; however, if the safety factors resulting from the analysis are not appropriate to the hazards posed by a slope failure and other than clearly conservative soil properties and profiles were used, the applicant is required to obtain additional data to verify his assumptions, or to show that, even if the worst possible conditions are assumed, there is an adequate margin of safety. With respect to seismic analysis, this subsection and subsection 2.5.5.2 are reviewed concurrently because different methods of analysis may involve different approximations, assumptions, and soil properties. In addition to generic state-of-the-art literature, other potential sources of information are those containing design, construction, and performance records of natural slopes, excavation slopes, and dams that may have been constructed in the general vicinity of the nuclear power plant.					
2.5.5.2	2.5.5.2 Design Criteria and Analyses. In meeting the requirements of 10 CFR Parts 50 and 100, the discussion of design criteria and analyses is acceptable if the criteria for the stability and design of all seismic Category I slopes are described and valid static and dynamic analyses have been presented to demonstrate that there is an adequate margin of safety. A number of different methods of analysis are available in the literature. To be acceptable, the static analyses should include calculations with different assumptions and methods of analysis to assess the following factors: 1. The uncertainties with regard to the shape of the slope,					

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	boundaries of the several types of soil within the slope and their properties, the forces acting on the slope, and pore pressures acting within the slope. 2. Failure surfaces corresponding to the lowest factor of safety. 3. The effect of the assumptions inherent in the method of analysis used. 4. Adverse conditions such as high water levels due to the probable maximum flood (PMF), sudden drawdown, or steady seepage at various levels. In general, safety factors related to the slope hazard are needed; however, actual values depend somewhat on the method of analysis, on the assumptions concerning the soil properties, on construction techniques, and on the range of material parameters. To be acceptable, the dynamic analyses must account for the effect of cyclic motion of the earthquake on soil strength properties as well as the potential effects of both horizontal and vertical components of shaking. Actual test data are needed for both the in situ soils as well as for any materials used in the construction of dams or embankments. As discussed above, the various parameters, such as geometry, soil strength, modeling method (location and number of elements (mesh) if a finite-element analysis is used), and hydrodynamic and pore pressure forces, should be varied to show that there is an adequate margin of safety. Where liquefaction is possible, major dam foundation slopes and embankments should be analyzed by state-of-the-art finite-element or finite difference methods of analysis. Where there are liquefiable soils, changes in					

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	pore pressure due to cyclic loading must be considered in the analysis to assess not only the potential for liquefaction but also the effect of pore pressure increase on the stress-strain characteristic of the soil and the post-earthquake stability of the slopes. The criteria, design techniques, and analyses are evaluated by the staff to ascertain that: 1. Appropriate state-of-the-art methods have been employed. 2. Conservative assumptions regarding soil and rock properties have been used in the design and analysis of slopes and embankments as discussed above in subsection 2.5.5.1. 3. Appropriately conservative margins of safety have been incorporated in the design. The criteria and design methods used by the applicant are reviewed to ascertain that state-of-the-art techniques are being employed. The design analyses are reviewed to be sure that the most conservative failure approach has been used and that all adverse conditions to which the slope might be subjected have been considered. Such conditions include ground motions, both horizontal and vertical, from the safe shutdown earthquake, settlement, cracking, flood or low-water steady-state seepage, sudden drawdown of an adjacent reservoir, or a reasonable assumption of the possible simultaneous occurrence of two natural events such as an earthquake and flood. The review is also concerned with determining whether or not the soil and rock characteristics derived from the investigations described in subsection 2.5.5.3 have been completely and conservatively					

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	incorporated into the design. When marginal factors of safety are indicated by the independent analyses performed by the staff and its consultants, additional substantiation and refinement is required or the applicant must use more conservative assumptions. No single method of analysis is entirely acceptable for all stability assessments; thus, no single method of analysis can be recommended. Relevant manuals issued by public agencies (such as the U.S. Navy Department, U.S. Army Corps of Engineers, and U.S. Bureau of Reclamation) are often used in reviews to ascertain whether the analyses performed by the applicant are reasonable (Refs. 14, 15, 16, and 17). Many of the important interaction effects cannot be included in current analyses and must be treated in some approximate fashion. Engineering judgment is an important factor in the staff's review of the analyses and in assessing the adequacy of the resulting safety factors. If the staff review indicates that questionable assumptions have been made by the applicant or some nonstandard or inappropriate method of analysis has been used, then the staff or its consultant may model the dam or slope in a manner which it feels is more consistent with the data and perform an independent analysis employing both deterministic and probabilistic methods as appropriate.					
2.5.5.3	2.5.5.3 Boring Logs. In meeting the requirements of 10 CFR Parts 50 and 100, the applicant should describe the borings and soil testing carried out for slope stability studies and dam and dike analyses. Because dams, dikes, and natural or cut slopes are often remote from the main plant area, results of additional exploration, tests, and analyses for these areas should also be presented in this subsection.					

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	A comprehensive program of site investigations including borings, sampling, geophysical surveys, test pits, trenches, and laboratory and field testing must be carried out by the applicant to define the physical characteristics of all soil and rock beneath safety-related and seismic Category I slopes, and borrow material that is to be used to construct safety-related dams, fills, and embankments (Refs. 10 and 11). The staff reviews these investigations to ascertain that the program has been adequate to define the in situ and earthwork soil and rock characteristics. The decision as to the adequacy of the investigation program is based on the methods discussed in SRP Section 2.5.4.					
2.5.5.4	2.5.5.4 Compacted Fill. In meeting the requirements of 10 CFR Part 50, the applicant should describe the excavation, backfill, and borrow material planned for any dams, dikes, and embankment slopes. Planned construction procedures and control of earthworks should be described. To be acceptable, the information must be given as discussed in subsection 2.5.4.5. Some of this information could be presented in subsection 2.5.4.5. Because dams, dikes, and other earthworks are often remote from the main seismic Category I structures, it is necessary to complete this information in this subsection. Quality control techniques and requirements during and following construction must also be discussed and referenced to quality assurance sections of the SAR. The preliminary specifications and quality control techniques to be used during construction are reviewed by the staff to ascertain that all design conditions are likely to be met (Refs. 5 and 9). During this part of the review the following are among those subjects reviewed for adequacy: 1. Proposed construction dewatering plan to ensure that it will not					

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	result in damage either to the natural or engineered foundation materials or to temporary or permanent structural foundations. 2. The excavation plan to remove all unsuitable materials from beneath the foundations and the quality control procedures which establish suitable materials. 3. The techniques and equipment to be used in compacting foundation and embankment materials. 4. The quality control and testing program to provide a high level of assurance that: a. The selected borrow material is as good and as relatively homogeneous as anticipated from the investigation program. b. The compacted foundation soil meets design specifications. 5. The techniques for improving the stability of natural slopes such as drainage, grouting, rock bolting, and applying shotcrete and/or gunite. 6. The plans for monitoring during and after construction to detect occurrences that could detrimentally affect the facility. Such monitoring includes periodic examination of slopes, survey of settlement monuments, and measurements of local wells and piezometers.					
	REFERENCES: 5. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."					

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	9. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)." 10. Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants." 11. Regulatory Guide 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants." 14. Corps of Engineers, "Slope Stability," Manual N. EM 1110-2-1902, Office of the Chief of Engineers, Dept. of the Army (2003). 15. Bureau of Reclamation, "Earth Manual," Third Edition, U.S. Dept. of Interior (1998). 16. Corps of Engineers, "Soils and Geology Procedures for Foundation Design of Buildings and Other Structures (Except Hydraulic Structures)," Tech. Report TM 5-818-1, Office of the Chief of Engineers, Dept. of the Army (1983). 17. Department of the Navy, "Foundations, and Earth Structures," NAVFAC DM-7, September 1986.					
	CHAPTER 3, Design of Structures, Components, Equipment, and Systems					
3.2.1, Rev. 2 (03/2007)	Seismic Classification					
3.2.1.1	To meet the requirements of GDC 2, 10 CFR Part 100, Appendix A, and10 CFR Part 50, Appendix S regarding seismic design classification are met by using guidance provided in RG 1.29 "Seismic Design Classification." This guide describes an acceptable method of identifying and classifying those plant features that shoul be designed to withstand the effects of the SSE. RG 1.151 provides guidance with regard to seismic design requirements and classification of safety-related instrumentation sensing lines. RG 1.143 provides guidance used to establish the seismic design					

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	requirements of radioactive waste management SSCs to meet the requirements of GDC 2 and 61 as they relate to designing these SSCs to withstand earthquakes. The guide identifies several radioactive waste SSCs requiring some level of seismic design consideration. RG 1.189 provides guidance used to establish the design requirements of fire protection to meet the requirements of GDC 2 a it relates to designing these SSCs to withstand earthquakes. This guide identifies portions of fire protection SSCs requiring some leve of seismic design consideration.					
3.2.2, Rev. 2 (03/2007)	System Quality Group Classification					
3.2.2.1	RG 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." This guide describes an acceptable method for determining quality standards for Quality Group B, C, and D water- and steam-containing components important to safety of water-cooled nuclear power plants.					
3.3.1, Rev. 3 (03/2007)	Wind Loading					
3.3.1.1	The wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.					
3.3.1.2	The acceptance criteria for the design wind speed, its recurrence interval, the speed variation with height, the applicable gust factors, and the bases for determining these site-related parameters, are stated in SRP Sections 2.3.1 and 2.3.2. The approved values of these parameters should serve as basic input to the review and evaluation of the structural design procedures.					
3.3.1.3	The procedures used to transform the wind speed into an equivalent pressure to be applied to structures and parts, or portions of structures, as delineated in American Society of Civil					

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	Engineers/Structural Engineering Institute (ASCE/SEI) 7-05, “Minimum Design Loads for Buildings and Other Structures,” are acceptable. In particular, the procedures used are acceptable if found in accordance with the following:					
	A. For a design wind speed, V, the velocity pressure, q_z, evaluated at height, z, is given by: $q_z = 0.00256\ K_z K_{dt} K_d V^2 I\ (lb/ft^2)$ where: K_z = velocity pressure exposure coefficient evaluated at height, z, as defined in ASCE/SEI 7-05, Table 6-3, but not less than 0.87 K_{dt} = topographic factor equal to 1.0 K_d = wind directionality factor equal to 1.0 V = design wind speed in miles per hour (mi/h) as stated in SRP Section 2.3.1 I = importance factor equal to 1.15 B. For each wind direction considered, the upwind exposure category should be based on ground surface roughness that is determined from natural topography, vegetation, and constructed facilities. Surface roughness C is defined as open terrain with scattered obstructions having heights generally less than 30 ft. This category includes flat open country, grasslands, and all water surfaces in hurricane prone regions. Because most nuclear power plants are located in relatively open country, K_z values in Table 6-3 should be selected from the Exposure C column. The definition of Exposure C is provided in ASCE/SEI 7-05, Section 6.5.6.3. B. Design wind loads should be determined in accordance with the following sections in ASCE/SEI 7-05, as applicable. i. Section 6.5.12 – Design Wind Loads on Enclosed and Partially Enclosed Buildings					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	ii. Section 6.5.13 - Design Wind Loads on Open Buildings with Monoslope, Pitched, or Troughed Roofs iii. Section 6.5.14 - Design Wind Loads on Solid Freestanding Walls and Signs iv. Section 6.5.15 - Design Wind Loads on Other Structures					
3.3.2, Rev. 3 (03/2007)	Tornado Loads					
3.3.2.1	The tornado wind and associated missiles generated by the tornado wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.					
3.3.2.2	The acceptance criteria for tornado parameters including maximum wind speed, translational speed, rotational speed, and atmospheric pressure change, and the bases for determining these parameters are defined in SRP Sections 2.3.1 and 2.3.2. Acceptance criteria for the spectrum of tornado-generated missiles and their characteristics, as well as the bases for determining these parameters, are defined in SRP Section 3.5.1.4. These parameters should serve as basic input to the review and evaluation for structural design.					
3.3.2.3	The acceptance criteria for procedures used to transform tornado parameters into equivalent loads on structures are as follows: A. Tornado Characteristics and Effects Tornados are characterized, in Table 1 of Regulatory Guide (RG) 1.76 for the contiguous United States into three geographical regions and by (1) maximum wind speed, (2) translational speed, (3) maximum rotational speed, (4) radius of maximum rotational speed, (5) pressure drop, and (6) rate of pressure drop for each of the three regions. Tornado effects are subdivided into three groups:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	i. Tornado wind effects caused by the direct action of air flow on structures, ii. Atmospheric pressure change effects caused by the differential pressure between the interior and exterior of a structure during the passage of a tornado, and iii. Tornado-generated missile impact effects. Tornado effects considered in design should include combinations of tornado wind effects, atmospheric pressure change effects, and tornado-generated missile impact effects. B. Tornado Wind Effects Procedures delineated in American Society of Civil Engineers/ Structural Engineering Institute (ASCE/SEI) 7-05, "Minimum Design Loads for Buildings and Other Structures" are acceptable for transforming tornado wind speed into pressure-induced forces applied to structures. In particular, the following shall apply: i. The maximum velocity pressure, q_z, should be based on the applicable maximum tornado wind speed, V, using the following equation from ASCE/SEI 7-05, Section 6.5.10: $q_z = 0.00256\ K_z K_{dt} K_d V^2 I$ (lb/ft^2) where: K_z = velocity pressure exposure coefficient equal to 0.87 K_{dt} = topographic factor equal to 1.0 K_d = wind directionality factor equal to 1.0 V = maximum tornado wind speed (mi/h) I = importance factor equal to 1.15 The maximum tornado wind speed, V, is the resultant of the maximum rotational speed and the translational speed of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	tornado. ii. Wind speed is assumed not to vary with the height above ground. iii. Design tornado wind loads should be determined in accordance with the following sections in ASCE/SEI 7-05, as applicable. (1) 6.5.12 Design Loads on Enclosed and Partially Enclosed Buildings (2) 6.5.13 Design Wind Loads on Open Buildings with Monoslope, Pitched, or Troughed Roofs (3) 6.5.14 Design Wind Loads on Solid Freestanding Walls and Solid Signs (4) 6.5.15 Design Wind Loads on Other Structures C. Atmospheric Pressure Change Effects RG 1.76 provides guidance for determining the pressure drop and the rate of pressure drop caused by the passage of a tornado. “Wind Effects on Structures: Fundamentals and Applications to Design,” (Third Edition, John Wiley and Sons, Inc., New York, 1996.) by E. Simiu and R. H. Scanlan, provides methods for determining loads on structures due to atmospheric pressure changes during the passage of a tornado. For a structure that is completely open subjected to a tornado, the internal and external pressures on the structure equalize rapidly during the passage of the tornado. Therefore, the atmospheric pressure change between the interior and the exterior of that structure approaches zero. For a structure that is enclosed (unvented structure), the internal pressure remains equal to the atmospheric pressure before the passage of a tornado. The atmospheric pressure					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	outside the structure changes during the passage of a tornado, which creates pressure differences between the interior and the exterior of that structure, and these differential pressures produce outward acting loads on the roof and walls of the enclosed structure. For a structure that is partially enclosed (vented structure), the determination of loads on the structure due to atmospheric pressure changes during the passage of a tornado is more complicated. If venting is adopted as a way to reduce the atmospheric pressure change effect on a structure, the review will be performed on a case-by-case basis. D. Tornado-Generated Missile Impact Effects Tornado-generated missile characteristics and the design-basis tornado missile spectrum are provided in RG 1.76. The acceptance criteria for transforming tornado-generated missile impact into equivalent static loads on structures are delineated in SRP Section 3.5.3, subsection II. E. Combined Tornado Effects After tornado-generated wind effects, W_w, atmospheric pressure change effects, W_p, and missile impact effects, W_m, are determined, the combination thereof should then be established in a conservative manner for structures. An acceptable method of combining these effects and establishing the total tornado load on a structure is as follows: $W_t = W_p$ Eq. 1 $W_t = W_w + 0.5\ W_p + W_m$ Eq. 2 where: W_t = total tornado load W_w = load from tornado wind effect					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	W_p = load from tornado atmospheric pressure change effect W_m = load from tornado missile impact effect					
3.3.2.4	The information provided to demonstrate that failure of any structure or component not designed for tornado loads will not affect the capability of other SSCs to perform necessary safety functions, is acceptable if found in accordance with either of the following: A. The postulated failure or collapse of structures and components not designed for tornado loads, including missiles, can be shown not to result in any structural or other damage to safety-related structures, systems, or components. B. Safety-related structures are designed to resist the effects of the postulated structural failure, collapse, or generation of missiles from structures and components not designed for tornado loads.					
3.4.1, Rev. 3 (03/2007)	Internal Flood Protection for Onsite Equipment Failures					
3.4.1.1	Guidance acceptable for meeting the seismic design and classification requirements of GDC 2 is found in Regulatory Guide (RG) 1.29, Position C.1 for safety-related SSCs and Position C.2 for nonsafety-related SSCs.					
3.4.1.2	The requirements of GDC 4 are met if SSCs important to safety are designed to accommodate the effects of discharged fluid resulting form high and moderate energy line breaks that are postulated in SRP sections 3.6.1 and 3.6.2.					
3.4.2, Rev. 3 (03/2007)	Analysis Procedures					
3.4.2.1	The highest flood and groundwater levels and the associated static and dynamic effects, if any, used in the design shall be the most severe ones that have been historically reported for the site and surrounding area, with sufficient margin for the limited					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	accuracy, quantity, and period of time in which the historical data have been accumulated.					
3.4.2.2	In most situations, the highest flood level is below the proposed plant grade and only its hydrostatic effects need be considered. Unless the hydrostatic head associated with the highest flood and groundwater levels is relieved by utilizing a drainage or a pumping system around the foundations of a structures, hydrostatic pressure has to be considered as a structural load on basement walls and the foundation slab of a structure. In consideration of any uplifting or floating of a structure, the total buoyancy force may be based on the highest flood level or the highest groundwater level excluding wave action. However, wave action should be included in the calculation for lateral and overturning movements of a structure.					
3.4.2.3	Where the flood level is above the proposed plant grade, the dynamic loads of wave action should be considered. Procedures for determining such dynamic loads are acceptable if they are in accordance with or equivalent to those delineated in the U.S. Army Coastal Engineering Research Center, “Shore Protection Manual” (Vol. I, June 2002, reprinted from 1973 edition and Vol. II, June 2002, reprinted from 1973 edition) or in EM 1110-2-1100, Coastal Engineering Manual, Part II, Chapter 1, “Water Wave Mechanics,” U.S. Army Corps of Engineers, April 30, 2002 as applicable.					
3.5.1.1, Rev. 3 (03/2007)	Internally Generated Missiles (Outside Containment)					
3.5.1.1.1	The applicant’s statistical significance of an identified missile can be evaluated by a probability analysis. Its statistical significance is determined by calculating the probability of missile occurrence. If this probability is less than 10^{-7} per year, the missile is not considered statistically significant. If the probability of occurrence is greater than 10^{-7} per year, the probability of impact on a significant target is determined. If the product of these two probabilities is less than 10^{-7} per year, the missile is not					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	considered statistically significant. If the product is greater than 10^{-7} per year, the probability of significant damage is determined. If the combined probability (product of all three) is less than 10^{-7} per year, the missile is not considered statistically significant. If the combined probability is greater than 10^{-7} per year, missile protection of SSCs important to safety, and of nonsafety-related SSCs whose failure could affect an intended safety function of the safety related SSCs, should be provided by one or more of the six methods listed below.					
3.5.1.1.2	Missile protection for SSCs important to safety is adequate if provided by one or more of the following methods: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing local shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, or (6) orienting missile sources to prevent missiles from striking equipment important in safety. RG 1.117 provides guidance on the SSCs that should be protected. Where barriers are used as a method of protection of SSCs from internal missiles, the design of the barriers is acceptable if it meets the guidance of RG 1.115 position C.3. Components within one train of a system with redundant trains need not be protected from missiles originating from the same train.					
3.5.1.2, Rev. 3 (03/2007)	Internally Generated Missiles (Inside Containment)					
3.5.1.2.1	The applicant's statistical significance of an identified missile can be evaluated by a probability analysis. The statistical significance for a potential missile is determined by calculating the probability of missile occurrence. If this probability is less than 10^{-7} per year, the missile is not considered significant. If the probability of occurrence is greater than 10^{-7} per year, the probability that it will impact a significant target is determined. If the product of these					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	two probabilities is less than 10^{-7} per year, the missile is not considered significant. If the product is greater than 10^{-7} per year, the probability of significant damage is determined. If the combined probability (product of all three) is less than 10^{-7} per year, the missile is not considered significant. If the combined probability is greater than 10^{-7} per year, missile protection of SSCs important to safety, and of nonsafety-related SSCs whose failure could affect an intended safety function of the safety related SSCs, should be provided by one or more of the six methods listed below.					
3.5.1.2.2	The missile protection for SSCs important to safety is adequate if provided by one or more of the following methods: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, or (6) orienting missile sources to prevent missiles from striking equipment important to safety.					
3.5.1.2.3	In summary, an Safety Analyses Report (SAR) statement that SSCs important to safety will be afforded protection by locating them in individual missile-proof structures, physically separating redundant systems or system components, or providing special protective shields or barriers is an acceptable method to meet this criterion.					
3.5.1.3, Rev. 3 <u>(03/2007)</u>	Turbine Missiles					
3.5.1.3.1	The probability of unacceptable damage resulting from turbine missiles, P_4, is expressed as the product of (a) the probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing, P_1; (b) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems, or components, P_2; and (c) the probability of struck structures, systems, or					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	components failing to perform their safety function, P_3. Stated in mathematical terms, $P_4 = P_1 \times P_2 \times P_3$. In accordance with the guidance provided in SRP Section 2.2.3 and RG 1.115, the probability of unacceptable damage from turbine missiles should be less than or equal to 1 in 10 million per year for an individual plant (i.e., P4 should be $\leq 10^{-7}$ per year per plant). Although the calculation of strike probability, P_2, is not difficult in principle (i.e., a straightforward ballistics analysis), in practice it requires numerous modeling approximations and simplifying assumptions to define the properties of missiles, interactions of missiles with barriers and obstacles, trajectories of missiles as they interact with and perforate (or are deflected by) barriers, and identification and location of safety-related targets. Specific approximations and assumptions tend to have a significant effect on the resulting value of P_2. Similarly, a reasonably accurate specification of the damage probability, P_3, is complicated by difficulties associated with defining the missile impact energy required to render safety-related systems unavailable to perform their safety functions and with postulating sequences of events that would follow a missile-producing turbine failure. Because of the uncertainties associated with calculating P_2 and P_3, the staff concludes that such analyses are "order of magnitude" calculations only. On the basis of simple estimates for a variety of plant layouts, the strike and damage probability product can be reasonably assumed to fall in a range that depends on the gross features of turbine generator orientation. A. For favorably oriented turbine generators, the product of P_2 and P_3 tends to be in the range of 10^{-4} to 10^{-3} per year per plant.					

Table A1-15: NUREG-0800, Standard Review Plan						
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	B. For unfavorably oriented turbine generators, the product of P_2 and P_3 tends to be in the range of 10^{-3} to 10^{-2} per year per plant. Favorably oriented turbine generators are located such that the containment and all, or almost all, safety-related SSCs outside containment are excluded from the low-trajectory hazard zone described in RG 1.115. Because of assumptions and modeling difficulties in the probabilistic calculations as described above, the staff does not encourage applicants to calculate P_2, P_3, or their product. Instead, the staff accepts a product of strike and damage probabilities of 10^{-3} per year per plant for a favorably oriented turbine and 10^{-2} per year per plant for an unfavorably oriented turbine. The suggested values represent the staff's best estimate of the product of P_2 and P_3, based on the results of calculations performed at the NRC (NUREG-1048, Supplement No. 6, and NUREG-0887, Supplement No. 3) and elsewhere.					
3.5.1.3.2	Operating experience indicates that turbine rotor crack (NUREG/CR-1884; PNO-111-81-104; and NRC Memorandum from E. Jordan to W. Russell), turbine stop and control valve failures (J.J. Burns, Jr.; License Event Report No. 82-132, Docket No. 50-361; and NRC Memorandum from E. Jordan to W. Russell), blade failures, and rotor ruptures can result in the generation of high-energy missiles (D. Kalderon and NRC Memorandum from E. Jordan to W Russell). Analyses indicate that missile generation can be modeled and the probability of missile generation can be strongly influenced by a suitable program of periodic inservice testing and inspection. In general, two modes of turbine rotor failure can result in turbine missile generation: (a) rotor material failure at approximately the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	rated operating speed and (b) failure of the overspeed protection system. Failure of turbine rotors at or below the design speed (nominally, 120% of normal operating speed) can be caused by small flaws or cracks that grow to critical size during operation. Failure of the turbine rotors at destructive overspeed (about 180% to 190% of normal operating speed) can result from failure of the overspeed protection system. The material properties of the turbine casing are of interest because secondary missiles could be generated if the casing fails or, alternatively, the casing could serve to arrest and contain missiles. The missile generation probability at the design speed should be related to rotor design parameters, material properties, and the intervals of inservice examinations of disks. The missile generation probability at the destructive overspeed should be related to the speed sensing and tripping characteristics of the turbine governor and overspeed protection system, the design and arrangement of main steam control and stop valves, the reheat steam intercept, reheat stop valves, and the inservice testing and inspection intervals for system components and valves. In addition, the turbine casing material in its operational environment should be evaluated for fracture toughness properties. SRP Section 10.2 provides additional guidance regarding inspection and testing of turbine generator components. Further information regarding turbine missile generation mechanisms and probabilities can be found in NUREG-1048, NRC Memorandum from E. Jordan to W. Russell, and Letter from C. Rossi (NRC) to J. Martin (Westinghouse Electric Corporation).					
3.5.1.3.3	The staff believes that maintaining an acceptably low missile generation probability, P_1, by means of a suitable program of periodic testing and inspection is a reliable method for ensuring that the objective of precluding generation of turbine missiles (and hence the possibility of damage to safety-related structures,					

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	systems, and components by those missiles) can be met. The NRC safety objective for turbine missiles (i.e., P_4 should be $\leq 10^{-7}$ per year per plant) is best expressed in terms of either of two sets of criteria applied to missile generation probability, P_1. All applicants are expected to commit to operating criteria (see Table 3.5.1.3-1) appropriate to the applicable turbine orientation. One set of criteria should be applied to favorably oriented turbines; the other should be applied to unfavorably oriented turbines. This approach places responsibility on the applicant for initially demonstrating, and thereafter maintaining, an NRC-specified turbine reliability. Accordingly, the applicant should commit to conduct appropriate inservice inspection and testing throughout the life of the plant. Accordingly, the applicant should demonstrate the capability to perform visual, surface, and volumetric (ultrasonic) examinations suitable for inservice inspection of turbine rotors and shafts and provide reports, as required, describing the applicant's methods for determining turbine missile generation probabilities (NUREG-1048 Supplement No 6; Letter from C. Rossi (NRC) to J. Martin (Westinghouse Electric Corporation); and NUREG-0887) for NRC review and approval.					
3.5.1.3.4	Applicants obtaining turbines from manufacturers that have prepared NRC-approved reports to describe their methods and procedures for calculating turbine missile generation probabilities are expected to meet criteria appropriate to the orientation of the turbine (see Table 3.5.1.3-1). Turbine manufacturers should provide applicants with tables of missile generation probabilities versus time (inservice visual, surface, and volumetric rotor inspection interval for design speed failure and inservice valve testing interval for destructive overspeed failure) for each turbine. These probabilities should be used to establish inspection and test schedules that meet NRC safety objectives. Refer to the RG for Table 3.5.1.3-1.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
3.5.1.3.5	Applicants are expected to commit to the following program if turbines are obtained from manufacturers that have not submitted, or received NRC approval for, reports describing their methods and procedures for calculating turbine missile generation probabilities: A. An inservice inspection program should be used to detect rotor or disk flaws that could lead to brittle failure at or below design speed in the steam turbine rotor assembly. The turbine rotor design should facilitate inservice inspection of all high-stress regions, including disk bores and keyways, without removal of the disks from the shaft. The volumetric inservice inspection interval for the steam turbine rotor assembly should be established according to the following guidelines: i. The initial inspection of a new rotor or disk should be performed before any postulated crack is calculated to grow to more than one-half the critical crack depth. If the calculated inspection interval is less than the scheduled first fuel cycle, the licensee should seek the manufacturer's guidance on delaying the inspection until the first refueling outage. If the calculated inspection interval is longer than the first fuel cycle, the licensee should seek the manufacturer's guidance for scheduling the first inspection during a later refueling outage. ii. Disks that have been inspected and found free of cracks or that have been repaired to eliminate all indications of cracks should be reinspected using the criterion described in (1) above. Crack growth should be calculated from the time of the last inspection. iii. Disks operating with known and measured cracks					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	should be reinspected before the elapse of one-half the time calculated for any crack to grow to one-half the critical depth. The guidance described in (1) above should be used to set the inspection date on the basis of the calculated inspection interval. iv. Under no circumstances should the volumetric inservice inspection interval for low-pressure (LP) disks exceed 3 years or two fuel cycles, whichever is longer. B. In accordance with the manufacturer's procedures, the turbine inservice inspection program should use visual, surface, and volumetric examinations to inspect turbine components such as couplings, coupling bolts, LP turbine shafts, blades and disks, and high-pressure (HP) rotors. Shafts and disks with crack(s) having depths at or near one-half the critical crack depth should be repaired or replaced. All cracked couplings and coupling bolts should be replaced. C. The inservice inspection and test program should be used for the governor and overspeed protection system to provide further assurance that flaws or component failures will be detected in the overspeed sensing and tripping subsystems, main steam control and stop valves, reheat steam intercept and stop valves, or extraction steam non-return valves — any of which could lead to an overspeed condition above that specified by the design overspeed. The inservice inspection program for operability of the governor and overspeed protection system should include, at a minimum, the following provisions: i. For typical turbine governor and overspeed protection systems, at intervals of approximately 3 years during refueling or maintenance shutdowns, at least one main					

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	steam control valve, one main steam stop valve, one reheat intercept valve, one reheat stop valve, and one of each type of steam extraction valve should be dismantled for examination. Visual and surface examinations of valve seats, disks, and stems should be conducted. Valve bushings should be inspected and cleaned, and bore diameters should be checked for proper clearance. If any valve is shown to have flaws or excessive corrosion or improper clearances, the valve should be repaired or replaced. All other valves of that type should also be dismantled and inspected. ii. At least once a week during normal operation, main steam control and stop valves, reheat intercept and stop valves, and steam extraction nonreturn valves should be exercised by closing each valve and observing directly the valve motion as it moves smoothly to a fully closed position. iii. At least once a month during normal operation, each component of the electro-hydraulic governor system (which modulates control and intercept valves), as well as the primary and backup overspeed trip devices (both of which trip the main steam control and stop valves and the reheat intercept and stop valves), should be tested. The online test failure of any one of these subsystems mandates repair or replacement of failed components within 72 hours. Otherwise, the turbine should be isolated from the steam supply until repairs are completed. Refer to SRP Section 10.2 for additional information regarding inspection and testing of turbine generator components. D. The design, inspection, and operating conditions should					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	provide assurance that the probability of turbine missile generation will not exceed those described in Table 3.5.1.3-1.					
3.5.1.3.6	An applicant may propose to install barriers or to take credit for existing structures or features as barriers. Such a decision could be based on the applicant's deterministic judgment that a SSCs is particularly vulnerable to destruction or unacceptable damage in the event of a turbine failure. The applicant should include specific details in the safety analysis report (SAR) supporting the need for such protection. If an applicant proposes to design or evaluate barriers to reduce or eliminate turbine missile hazards to equipment, the barriers should meet the acceptance criteria described in SRP Section 3.5.3. Additional design guidance is provided in “Fundamentals of Protective Design,” TM-5-885-1, Department of the Army, July 1965.					
3.5.1.4, Rev. 3 (03/2007)	Missiles Generated by Tornadoes and Extreme Winds					
3.5.1.4.1	Regulatory Guide (RG) 1.76 describes acceptable design-basis tornado-generated missile spectrum for the design of nuclear power plants.					
3.5.1.4.2	The method of identifying appropriate design-basis missiles generated by natural phenomena shall be consistent with the acceptance criteria defined for the evaluation of potential accidents from external sources in SRP Section 2.2.3. Other methodologies used by licensees and applicants with appropriate rationale may be acceptable on a case-by-case basis.					
3.5.1.5, Rev. 4 (03/2007)	Site Proximity Missiles (Except Aircraft)					
3.5.1.5.1	To meet the requirements of 10 CFR Part 100, the probability that site proximity missiles will impact the plant and cause radiological consequences greater than the 10 CFR Part 100 exposure guidelines must be less than an order of magnitude of 10^{-7} per year (see guidance in SRP Section 2.2.3). If the review indicates that the above criterion is not met, then the acceptance					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	criterion described in item 2 below applies.					
3.5.1.5.2	The plant will meet the relevant requirements of GDC 4 and will be considered appropriately protected against site proximity missiles' design if the SSCs important to safety are capable of withstanding the effects of the postulated missiles without loss of safe-shutdown capability and without causing a release of radioactivity in excess of the 10 CFR Part 100 dose guidelines					
3.5.1.6, Rev. 3 (03/2007)	Aircraft Hazards					
3.5.1.6.1	10 CFR 100.10, 10 CFR 100.20, 10 CFR 100.21, 10 CFR 52.17, and 10 CFR 52.79 requirements are met if the probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR Part 100 exposure guidelines is less than an order of magnitude of 10-7 per year (see SRP Section 2.2.3). The probability is considered to be less than an order of magnitude of 10^{-7} per year by inspection if the distances from the plant meet all of the criteria listed below: A. The plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2 B. The plant is at least 5 statute miles from the nearest edge of military training routes, including low-level training routes, except for those associated with usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation C. The plant is at least 2 statute miles beyond the nearest edge of a Federal airway, holding pattern, or approach pattern The projected number of operations in item A above, as well as the					

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	1000 flights per year in item B above, should represent the maximum aircraft activity expected during the permit term in CP and ESP applications or for the license duration in OL and COL applications.					
3.5.1.6.2	If the above proximity criteria are not met, or if sufficiently hazardous military activities are identified (see item B above), a detailed review of aircraft hazards must be performed. Aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 with a probability of occurrence greater than an order of magnitude of 10^{-7} per year should be considered in the design of the plant. If the results of the review do not support a finding that the risk from aircraft activities is acceptably low, then the design-basis acceptance criteria outlined in GDC 4 applies. The plant meets the relevant requirements of GDC 3 and GDC 4, and is considered appropriately protected against design-basis aircraft impacts and fires, if the SSCs important to safety are capable of withstanding the effects of the postulated aircraft impacts and fires without loss of safe-shutdown capability and without causing a release of radioactivity that could exceed the 10 CFR Part 100 dose guidelines. Regulatory Guide (RG) 1.117 provides acceptable methods for determining those SSCs that should be protected. The selection of SSCs to be protected is based upon not allowing offsite exposures to exceed an appropriate fraction of the offsite dose guidelines of 10 CFR Part 100. Basing the limits upon an appropriate "fraction" ensures protection for those events that are not as severe as the design-basis event, but have a higher probability of occurrence. Protecting those SSCs important to safety from the effects of externally generated missiles due to aircraft hazards prevents failure of those systems required for					

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	safe shutdown and prevents the release of radioactivity with the potential for causing exposures in excess of the 10 CFR Part 100 guidelines. The expected rate of exposure identified in 10 CFR 50.34(a)(1) dose guideline as it relates to the requirements identified in 10 CFR 100.20(b) should be about an order of magnitude of 10^{-6} per year. If it can be shown with rigorous analysis, using realistic assumptions and reasonable arguments that the estimated probability could be lower, then, in accordance with the SRP Section 2.2.3, it is acceptable.					
3.5.2, Rev. 3 (03/2007)	Structures, Systems, and Components To Be Protected From Externally-Generated Missiles					
3.5.2.1	Acceptance is based on the design meeting the guidelines of Regulatory Guide (RG) 1.13 as to the capability of spent fuel pool systems and structures to withstand the effects of externally-generated missiles and to prevent missiles from contacting stored fuel assemblies; RG 1.27 as to the capability of the ultimate heat sink and connecting conduits to withstand the effects of externally-generated missiles; RG 1.115 as to the protection of important safety-related SSCs from the effects of turbine missiles; and RG 1.117 as to the protection of important safety-related SSCs from the effects of tornado missiles.					
3.5.3, Rev. 3 (03/2007)	Barrier Design Procedures					
3.5.3.1	For Local Damage Prediction A. Concrete Sufficient thickness of concrete should be provided to prevent perforation, spalling, or scabbing of the barriers in the event of missile impact. Several empirical equations, such as the modified National					

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	Defense Research Council (NDRC) formula; proposed in "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," by R.P. Kennedy, Nuclear Engineering and Design 1976 Pages 183-203 are available to estimate missile penetration into concrete. These equations should be used to determine the required barrier thicknesses. Thicknesses resulting from such calculations should not be less than those listed in Table 1, which specifies the minimum thicknesses necessary to protect against tornado missiles. Table 1, Minimum Acceptable Barrier Thickness Requirements, provides minimum concrete barrier thickness requirements for preventing local damage against tornado generated missiles for tornado spectrum shown in Table 2 of Regulatory Guide (RG) 1.76. Barrier thicknesses less than those listed in Table 1 may be used, provided that sufficient justification (including test data) is presented to support them. These justification will be reviewed on a case-by-case basis. Other types of missiles are specified in SRP Sections 3.5.1.1 through 3.5.1.6. For turbine missile barriers, penetration and scabbing predictions should be based on empirical equations such as the modified NDRC formula or the results of a valid test program. B. Steel The results of tests conducted by the Stanford Research Institute (SRI) on the penetration of missiles into steel plates					

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	are summarized in “U.S. Reactor Containment Technology” (ORNL/NSIC-5, Vol.1, Chapter 6, Oak Ridge National Laboratory, 1965) by W.B. Cottrell and A.W. Savolainen. The equations presented in aforementioned document are acceptable. Other equations such as the Ballistic Research Laboratory formula described in, “Reactor Safeguards,” by C. R. Russell, published by MacMillan, New York, 1962, may be used, provided the results are either comparable to those obtained by using the aforementioned “U.S. Reactor Containment Technology” method or are validated by penetration tests. C. Composite Sections For composite or multi-element barriers, procedures for prediction of local damage are acceptable if the residual velocity of the missile perforating the first element is considered as the striking velocity for the next element. For determining this residual velocity, the equations presented in “Ballistic Perforation Dynamics,” Journal of Applied Mechanics, Transactions of the ASME, Vol. 30, Series E, No. 3, September 1963 by R. F. Recht and T. W. Ipson, are acceptable when the first barrier of a multi-element missile barrier is steel. When the first barrier is concrete, procedures used are reviewed on a case-by-case basis. Refer to RG for Table 1, Minimum Acceptable Barrier Thickness Requirements					
3.5.3.2	For Overall Damage Prediction The response of a structure or barrier to missile impact depends largely on the location of impact (e.g., midspan of a slab or near a support), on the dynamic properties of the target and missile, and on the kinetic energy of the missile. In general, the assumption of					

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	plastic collisions is acceptable, where all of the missile's initial momentum is transferred to the target and only a portion of its kinetic energy is absorbed as strain energy within the target. However, where elastic impacts are expected, the additional momentum transferred to the target by missile rebound should be considered in the analyses. After it has been demonstrated that the missile will not penetrate the barrier, an equivalent static load concentrated at the impact area should then be determined, from which the structural response, in conjunction with other design loads, can be evaluated using conventional design methods. An acceptable procedure for such an analysis, where the impact is assumed to be plastic, is presented in “Impact Effect of Fragments Striking Structural Elements,” Holmes and Narver, Inc., Revised November 1973 by R. A. Williamson and R. R. Alvy. Other procedures may be used, with adequate justification provided the results obtained are comparable to that of the above reference. Maximum allowable ductility ratios for steel and reinforced concrete barriers, in the above analysis, are given in American National Standard Institute/ American Institute of Steel Construction (ANSI/AISC) N690-1994 including supplement 2(2004), American National Standard Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities (1994) and in RG 1.142. respectively.					
3.6.1, Rev. 3 (03/2007)	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment					
3.6.1.1	High and moderate energy fluid systems are separated from essential systems and components, as described in Appendix B to BTP 3-3.					
3.6.1.2	High and moderate energy fluid systems, or portions thereof, are enclosed as described in item B.1.b of BTP 3-3.					

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3.6.1.3	For cases where neither physical separation nor protective enclosures are considered practical by the applicant, the reviewer will verify the following: A. The reasons for which the applicant judged both physical separation and system enclosure to be impractical as means of protection are consistent with item B.1.c. of BTP 3-3. B. Redundant design features or additional protections (assuming a single active failure in any required system) have been provided such that failure modes and effects analyses for all failure situations ensure the performance of safety features. These analyses are done under the criteria and assumptions of item B.3. of BTP 3-3.					
3.6.1.4	Design Features are in accordance with item B.2 of BTP 3-3.					
3.6.1.5	The effects of postulated failures on essential equipment and the ability of the plant to be safely shut down are analyzed in accordance with item B.3. of BTP 3-3.					
3.6.2, Rev 2 (03/2007)	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping					
3.6.2.1	Postulated Pipe Rupture Locations Inside Containment. Acceptable criteria to define postulated pipe rupture locations and configurations inside containment are specified in Branch Technical position (BTP) 3-4.					
3.6.2.2	Postulated Pipe Rupture Locations Outside Containment. Acceptable criteria to define postulated rupture locations and plant layout considerations for protection against postulated pipe ruptures outside containment are specified in BTP 3-4.					
3.6.2.3	Methods of Analysis. Detailed acceptance criteria covering pipe-whip dynamic analysis,					

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	including determination of the forcing functions of jet thrust and jet impingement, are included in subsection III, "Review Procedures," of this SRP section. The general bases and assumptions of the analysis are given in BTP 3-4, subsection 2.C.					
3.6.3, Rev. 1 (03/2007)	Leak-Before-Break Evaluation Procedures					
	Compliance with GDC 4 requires that components important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. Safety-related components should be protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failure or events and conditions outside the nuclear power unit. Meeting the requirements of GDC 4 provides assurance that SSCs important to safety will be protected from the dynamic effects of pipe rupture and capable of performing their intended safety function.					
	LBB analyses should demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. A deterministic evaluation of the piping system that demonstrates sufficient margins against failure, including verified design and fabrication and an adequate inservice inspection program, can be assumed to satisfy the extremely low probability criterion.					
3.7.1, Rev. 3 (03/2007)	Seismic Design Parameters					
3.7.1.1	Design Ground Motion A. Design Response Spectra. The site-specific GMRS reviewed under SRP Section 2.5.2 are determined in the free-field on the ground surface.					

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	For sites with soil layers near the surface that will be completely excavated to expose competent material, the GMRS is specified on an outcrop or a hypothetical outcrop that will exist after excavation. Motions at this hypothetical outcrop should be developed as a free surface motion, not as an in-column motion. Although the definition of competent material is not mandated by regulation, a number of reactor designs have specified a shear wave velocity of 1000 fps as the definition of competent material, which is considered acceptable. If non- competent material is present, any excavation and/or backfilling should not alter the development or location of the site-specific GMRS. However, the soft soil or backfill material needs to be considered in the SSI or other analyses. According to Appendix S to 10 CFR Part 50, the minimum peak ground acceleration (PGA) for the horizontal component of the SSE at the foundation level in the free-field should be 0.1g or higher. The response spectrum associated with this minimum PGA should be a smooth broad-band response spectra (e.g., RG 1.60, or other appropriate shaped spectra if justified) considered as an outcrop response spectra at the free-field foundation level. This response spectrum anchored at 0.1g will be referred in this SRP section as the minimum required response spectrum. i. <u>Non-standard Plant Design.</u> For a non-standard plant design (e.g., COL application referencing only an ESP, or a COL application not referencing a CD and ESP), the design response spectra is developed from the site-specific GMRS or from a broad band shaped spectra similar to RG 1.60 which also envelops the site-specific GMRS. Foundation level response spectra consistent with the design response spectra are determined for each					

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	seismic Category I structure. These foundation level spectra are compared to the minimum required spectrum to ensure they meet the 0.1 g pga requirement in accordance with Appendix S to 10 CFR Part 50. If the foundation level spectra do not bound the minimum required response spectrum, then the design response spectra can be adjusted/modified in order to bound the minimum required spectrum. If the design response spectra are not modified, then the use of the two separate sets of spectra in the analysis and design of SSCs need to be reviewed for adequacy. ii. Certified Standard Plant Design (CD). For a design certification (DC) application, the postulated seismic design response spectra need to bound the minimum required response spectrum anchored at 0.1g (as specified in Appendix S to 10 CFR Part 50). These design response spectra are referred to as the CSDRS when the design is certified by the Commission under 10 CFR Part 52. For a certified standard plant design (e.g., COL application that references a CD or a COL application that references a CD and ESP), a similar approach described above (under subsection II.1.a.i) is used to ensure that the CSDRS envelop the minimum required response spectrum. Foundation level response spectra consistent with the CSDRS are determined for each seismic Category I structure. These foundation level spectra are compared to the minimum required spectrum to ensure they meet the 0.1g pga requirement in accordance with Appendix S to 10 CFR Part 50. If the foundation level spectra do not bound the minimum required spectrum, then the					

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	CSDRS can be adjusted/modified in order to bound the minimum required spectrum If the CSDRS are not modified, then the use of the two separate sets of spectra in the analysis and design of SSCs need to be reviewed for adequacy. For evaluation of soil liquefaction and soil/rock stability of slopes that may affect plant safety, the use of the site-specific GMRS rather than the CSDRS is reviewed on a case-by-case basis. The free-field design response spectra (also referred to as the CSDRS for a CD)are usually developed for the 5-percent damping value. In the seismic analysis and design, the applicant needs to define the free-field design response spectra corresponding to all damping values to be used. For the case of RG 1.60 response spectra, Tables 1 and 2 of RG 1.60 provide amplification factors at four frequencies for calculating response spectra corresponding to different damping values. For the case of the free-field design response spectra that are different from RG 1.60 response spectra, Appendix C to this SRP section provides procedures to calculate response spectra for different damping values other than 5 percent. To be acceptable, the seismic design response spectra should be specified for three mutually orthogonal directions - two horizontal and one vertical. Current practice is to assume that the design response spectra (including maximum ground accelerations) in the two horizontal directions are the same. B. Design Time Histories. The SSE and OBE design ground motion time histories can					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	be either real time histories or artificial time histories. To be acceptable, the design ground motion time histories should consist of three mutually orthogonal directions - two horizontal and one vertical. For both horizontal and vertical input motions, either a single time history or multiple time histories can be used. When time histories are used, each of the three ground motion time histories must be shown to be statistically independent from the others. Each pair of time histories are considered to be statistically independent if the absolute value of their correlation coefficient does not exceed 0.16. Simply shifting the starting time of a given time history can not be used to establish a different time history. Also, artificial time histories which are not based on seed recorded time histories should not be used. For linear structural analyses, the total duration of the artificial ground motion time histories should be long enough such that adequate representation of the Fourier components at low frequency is included in the time history. The corresponding stationary phase strong-motion duration should be consistent with the longest duration of strong motion from the earthquakes defined in SRP Section 2.5.2 at low and high frequency and as presented in NUREG/CR-6728. The strong motion duration is defined as the time required for the Arias Intensity to rise from 5% to 75%. The uniformity of the growth of this Arias Intensity should be reviewed. The minimum acceptable strong motion duration should be six seconds. In addition to the duration, the ratios V/A and AD/V^2 (A, V, D are peak ground acceleration, ground velocity, and ground displacement, respectively) should be consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the uniform hazard response spectra. These parameters should be consistent with the values determined for the low and high frequency					

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	events described in Appendix D of RG 1.208. For nonlinear structural analysis problems, multiple sets of ground motion time histories should be used to represent the design ground motion. Each set of ground motion time histories shall be selected from real recorded ground motions appropriate for the characteristic low and high frequency events. The amplitude of these ground motions may be scaled but the phasing of Fourier components must be maintained. The adequacy of this set of ground motions, including-duration estimates, is reviewed on a case-by-case basis. Option 1: Single Set of Time Histories. To be considered acceptable, the response spectra generated from the artificial time history to be used as input ground motion in the free-field should satisfy the enveloping requirements for either Approach 1 or Approach 2 below: i. Approach 1. For Approach 1, the spectrum from the artificial ground motion time history must envelop the free-field design response spectra for all damping values used in the seismic response analysis. When spectral values (e.g., spectral accelerations) are calculated from the artificial time history, the frequency intervals at which spectral values are determined are to be sufficiently small. Table 3.7.1-1 (below) provides an acceptable set of frequencies at which the response spectra may be calculated. Table 3.7.1-1 Suggested Frequency Intervals for Calculation of Response Spectra					

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	(continued below)					

Frequency Range (hertz)	Increment (hertz)
0.2 - 3.0	0
3.0 - 3.6	0
3.6 - 5.0	0
5.0 - 8.0	0
8.0 - 15.0	0
15.0 - 18.0	1
18.0 - 22.0	2
22.0 - highest frequency of interest	3

Each calculated spectrum of the artificial time history is considered to envelop the design response spectrum when no more than five points fall below, and no more than 10 percent below, the design response spectrum.

Studies indicate that numerically generated artificial ground acceleration histories produce PSD functions having a quite different appearance from one individual function to another, even when all these time histories are generated so as to

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	closely envelop the same design response spectra. For example, the use of the available techniques of generating acceleration time histories that satisfy enveloping RG 1.60 spectra usually results in PSD functions that fluctuate significantly and randomly as a function of frequency. It is also recognized that the more closely one tries to envelop the specified design response spectra, the more significantly and randomly do the spectral density functions tend to fluctuate and these fluctuations may lead to unconservative results for the response of SSCs. Therefore, when a single artificial ground motion time history is used in the design of seismic Category I SSCs, it must in general satisfy requirements for both enveloping design response spectra as well as adequately matching a target PSD function compatible with the design response spectra. Therefore, in addition to the response spectra enveloping requirement, the use of a single time history should also be justified by demonstrating sufficient energy at the frequencies of interest through the generation of PSD function, which envelops the target PSD function throughout the frequency range of significance. When RG 1.60 response spectra are used as design response spectra, the requirements for a compatible target PSD are contained in Appendix A to this SRP section. Target PSD functions other than those given in Appendix A can be used if justified. For design response spectra other than RG 1.60 response spectra, a compatible target PSD should be generated. For generation of target PSD in such cases, the guidelines and procedures provided in Appendix B to this SRP section can be used. Procedures used to generate the target PSD will be reviewed on a case-by-case basis. The PSD requirements are included as secondary and minimum requirements to prevent potential deficiency of power over					

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	the frequency range of interest. It should be noted that the ground motion is still primarily defined by the design response spectrum. The use of PSD criteria alone can yield time histories that may not envelop the design response spectrum. ii. Approach 2. For Approach 2, the artificial ground motion time histories that are generated to match or envelop the design response spectra shall comply with Steps (a) through (d) below. The general objective is to generate a modified recorded or artificial accelerogram which achieves approximately mean based fit to the target response spectrum; that is, the average ratio of the spectral acceleration calculated from the accelerogram to the target, where the ratio is calculated frequency by frequency, is only slightly greater than "1." The aim is to achieve an accelerogram that does not have significant gaps in the Fourier amplitude spectrum, but which is not biased high with respect to the target. (a) The time history shall have a sufficiently small time increment and sufficiently long duration. Records shall have a Nyquist frequency of at least 50 Hz, (e.g., a time increment of at most 0.010 seconds) and a total duration of at least 20 seconds. If frequencies higher than 50 Hz are of interest, the time increment of the record must be suitably reduced to provide a Nyquist frequency (N_f =1/(2Δt), where Δt = time increment) above the maximum frequency of interest. The total duration of the record can be increased by zero packing to satisfy these frequency criteria. (b) Spectral acceleration at 5% damping shall be					

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	computed at a minimum of 100 points per frequency decade, uniformly spaced over the log frequency scale from 0.1 Hz to 50 Hz or the Nyquist frequency. The comparison of the response spectrum obtained from the artificial ground motion time history with the target response spectrum shall be made at each frequency computed in the frequency range of interest. (c) The computed 5% damped response spectrum of the accelerogram shall not fall more than 10% below the target response spectrum at any one frequency. To prevent response spectra in large frequency windows from falling below the target response spectrum, the response spectra within a frequency window of no larger than ±10% centered on the frequency shall be allowed to fall below the target response spectrum. This corresponds to response spectra at no more than 9 adjacent frequency points defined in (b) above from falling below the target response spectrum. (d) In lieu of the power spectrum density requirement of Approach 1, the computed 5% damped response spectrum of the artificial ground motion time history shall not exceed the target response spectrum at any frequency by more than 30% (a factor of 1.3) in the frequency range of interest. If the response spectrum for the accelerogram exceeds the target response spectrum by more than 30% at any frequency range, the power spectrum density of the accelerogram needs to be computed and shown to not have significant gaps in energy at any frequency over this frequency range.					

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	Artificial ground motion time histories defined as described above shall have characteristics consistent with characteristic values for the magnitude and distance of the appropriate controlling events defined for the uniform hazard response spectrum (UHRS). Option 2: Multiple Sets of Time Histories. As discussed in Section I. 1.13 and Section II.1.13 of this SRP section, the use of multiple real or artificial time histories for analyses and design of SSCs is acceptable. For linear structural analyses, a minimum of four times histories should be used. For nonlinear structural analyses, the number of time histories must be greater than four and the technical basis for the appropriate number of time histories are reviewed on a case-by-case basis. This review also includes the adequacy of the characteristics of the multiple time histories. The response spectra calculated for each individual time history need not envelop the design response spectra. However, the multiple time histories are acceptable if the average calculated response spectra generated from these time histories envelop the design response spectra. An acceptable method to demonstrate the adequacy of a set of multiple time histories, in terms of enveloping requirements and having sufficient power over the frequency range of interest, is to follow the procedures described for Approach 2 presented in subsection II.1.B.ii of this SRP. When implementing Approach 2, the criteria in paragraphs (a) and (b) of this approach need to be satisfied for each of the time histories. The criteria in paragraphs (c) and (d) of this approach can be satisfied by utilizing the results for the average of the suite of multiple time histories.					

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3.7.1.2	Percentage of Critical Damping Values. The specific percentage of critical damping values used in the analyses of Category I SSCs are considered to be acceptable if they are in accordance with RG 1.61. Damping values different from those listed in RG 1.61 (e.g., higher damping values) may be used in a dynamic seismic analysis if test data are provided to support them. These damping values will be reviewed and accepted by the staff on a case-by-case basis. In addition, a demonstration of the correlation between stress levels and damping values will be required and reviewed for compliance with the applicable regulatory position in RG 1.61. If other methods for correlation of damping values with stress level are used, they will need to be reviewed and accepted on a case-by-case basis. The material soil damping for foundation soils must be based upon validated values or other pertinent laboratory data, considering variation in soil properties and strains within the soil, and must include an evaluation of dissipation from pore pressure effects as well as material damping for saturated site conditions. The maximum soil damping value acceptable to the staff is 15 percent.					
3.7.1.3	Supporting Media for Seismic Category I Structures. To be acceptable, the description of supporting media for each Category I structure must include foundation embedment depth, depth of soil over bedrock, soil layering characteristics, design groundwater elevation, dimensions of the structural foundation, total structural height, and soil properties such as shear wave velocity, shear modulus, Poisson's ratios, and density as a function of depth. If the minimum shear wave velocity of the supporting foundation material is less than 1,000 fps, additional studies need to be performed which consider the average shear wave velocity, and its degree of variability addressing					

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	potential impact on soil-structure interaction, potential settlements and design of foundation elements.					
3.7.1.4	Review Considerations for DC and COL Applications A. <u>COL Application Referencing an ESP and CD</u> In addition to the criteria presented below, Figures 1 and 2 in Appendix D provide additional guidance in understanding the Part 52 process. ii. Site-specific GMRS are reviewed separately under SRP 2.5.2 for adequacy. For COL application referencing an ESP and CD, the GMRS are included in the ESP. ii. Confirm that the criterion for the minimum required response spectrum (in accordance with subsection II.1.A.ii) has been satisfied. Confirm that COL action items contained in the CD have been met. This includes seismic design parameters such as soil layering assumptions used in the certified design, range of soil parameters, shear wave velocity values, and minimum soil bearing capacity. Technical justification for all deviations from the range of values used in the standard plant design must be provided for review. iii. Confirm that the ESP conditions have been met or review the COL applicant's approach to address any deviations. iv. When the site-specific GMRS and the CSDRS, are calculated at the same elevation, confirm that the CSDRS envelop the GMRS. For this case the					

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	standard design is acceptable for that site, assuming no other issue is identified during the review process. If the CSDRS do not envelop the site-specific GMRS then proceed to step vii. v. When the site-specific GMRS and the CSDRS are determined at different elevations, calculate the site-specific GMRS transferred to the base elevations of each seismic Category I foundation. These site-specific GMRS at the foundation levels are referred to as foundation input response spectra (FIRS) and are derived as free-field outcrop spectra; that is, only the effects of materials that are below the base elevation of the seismic Category I structure are included in the site response analysis. For each seismic Category I structure foundation, if the CSDRS-consistent spectra at the foundation level envelop the site- specific FIRS at the foundation level, the standard design is acceptable for that site, assuming no other issue is identified during the review process. If not, then proceed to step vii. vi. Perform an SSI analysis using the site-specific FIRS and an advanced seismic analytical technique (e.g., method that considers the effects of incoherent ground motion). When such analytical methods are utilized, the detailed technical justification shall be reviewed on a case-by-case basis. Further discussion on consideration of the effects of incoherent ground motion is provided in subsection II.4.C (under the heading Input Ground Motion, Specific Guidelines for SSI Analysis) in SRP Section 3.7.2. The in-structure responses in terms of floor response spectra, building member forces, and deformations at key locations in the structure					

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	shall be obtained. The key locations for calculating the in- structure responses, proposed by the licensee, need to be evaluated to ensure that they are sufficient to represent the various locations throughout the building. Locations should include responses at peripheral locations to detect rocking and torsion, and should include responses to check overturning, torsional, and sliding stability of the structures. The dynamic models and analysis techniques need to be sufficiently refined to be able to capture the response of the structures throughout the frequency range of interest, including the high frequency responses, typically expected in the central and eastern United States (CEUS) regions. The SSI analysis shall also consider the site-specific soil variability (i.e., best estimate, lower bound estimate, and upper bound estimate). Compare these responses at the key locations in the structure to the standard design in-structure responses. If the CSDRS responses envelop the in-structure responses from the FIRS, the standard design is acceptable assuming no other issue is identified during the review process. If the responses are not enveloped, additional analyses are required to demonstrate the acceptability of the design or the design might need to be modified. If further analyses are utilized, then the analyses must consider the potentially higher responses at all locations, not only those at the key locations described above. B. <u>COL Application Referencing a CD</u>. Follow the same steps described above under A - COL Application Referencing an ESP and CD, except that step iv. does not apply to this case.					

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	C. COL Application Referencing an ESP. In addition to the criteria presented below, Figure 3 in Appendix D provides additional guidance in understanding the process when a certified design is not used. i. Site-specific GMRS are reviewed separately under SRP 2.5.2 for adequacy. For COL application referencing an ESP, the GMRS are included in the ESP. ii. Confirm that the ESP conditions have been met or review the COL applicant's approach to address any deviations. iii. Follow the acceptance criteria described in subsection II.1.A (excluding subsection II.1.A.ii), of this SRP Section to develop the seismic design response spectra. The seismic SSI analysis would then follow the conventional approach for SSI analyses. D. COL Application not Referencing an ESP and DC. In addition to the criteria presented below, Figure 3 in Appendix D provides additional guidance in understanding the process when a certified design is not used. i. Site-specific GMRS are reviewed separately under SRP 2.5.2 for adequacy. ii. Follow the acceptance criteria described in subsection II.1.A (excluding subsection II.1.A.ii), of this SRP Section to develop the seismic design response spectra. The seismic SSI analysis would then follow the conventional approach					

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	for SSI analyses.					
3.7.2, Rev. 3 (03/2007)	Seismic System Analysis					
3.7.2.1	Seismic Analysis Methods. The seismic analysis of all seismic Category I SSCs should use either a suitable dynamic analysis method or an equivalent static load analysis method, if justified. The SRP acceptance criteria primarily address linear elastic analysis coupled with allowable stresses near elastic limits of the structures. However, for certain special cases (e.g., evaluation of as-built structures), reliance on limited inelastic/nonlinear behavior when appropriate is acceptable to the staff. Analysis methods incorporating inelastic/nonlinear considerations and the analysis results are reviewed on a case-by-case basis. A. Dynamic Analysis Method. When calculating seismic responses of Category 1 structures, dynamic analysis (response spectrum analysis method or time history analysis method) should be performed. To be acceptable, dynamic analyses should consider the following: i. Use of appropriate methods of analysis (time history analysis method [time domain solution and frequency domain solution]; response spectrum analysis method), accounting for the effects of SSI, if applicable. In general, the response spectrum analysis method is not suitable for SSI analysis. ii. Seismic analysis should be performed for three orthogonal (two horizontal and one vertical) components of earthquake ground motion. iii. Consideration of the torsional, rocking, and translational					

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	responses of the structures and their foundations (including footings, basemats and buried walls). iv. Use of an adequate number of discrete mass degrees of freedom in dynamic modeling. The adequacy of the number of discrete mass degrees of freedom can be confirmed by (1) preliminary modal analysis, and (2) correlation between static analysis results using the dynamic model and static analysis results using a distributed mass representation. (1) It is important to ensure that, for each excitation direction (2 horizontal and vertical), all modes with frequencies less than the ZPA (or PGA) frequency of the corresponding spectrum are adequately represented in the dynamic solution. Preliminary modal analysis should be performed to establish that a sufficient number of discrete mass degrees of freedom have been included in the dynamic model to (a) predict a sufficient number of modes, and (2) produce mode shapes that are reasonably smooth. If a mode shape exhibits rapid change in modal displacement between adjacent mass degrees of freedom, additional mass degrees of freedom should be added until reasonably smooth mode shapes are obtained for all modes to be included in the dynamic analysis. (2) After completion of (1), simple 1g static analyses of the dynamic model should be performed for each of the three (3) excitation directions, and compared to the corresponding results obtained from static					

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	analyses that utilize a distributed mass representation. Lack of correlation, particularly in the vicinity of and at support locations, is indicative of an insufficient number of discrete mass degrees of freedom. v. When using either the response spectrum method or the modal superposition time history method, responses associated with high frequency modes (i.e., f >_ ZPA [or PGA] frequency) should be included in the total dynamic solution using the guidance and methods described in Regulatory Guide 1.92, Revision 2, Regulatory Positions C.1.4 and C.1.5. vi. Consideration of maximum relative displacements between adjacent supports of seismic Category I SSCs. vii. Inclusion of significant effects such as piping interactions, externally applied structural restraints, hydrodynamic (both mass and stiffness effects) loads, and nonlinear responses. B. Equivalent Static Load Method. An equivalent static load method is acceptable if: i. Justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses. Typical examples or published results for similar structures may be submitted in support of the use of the simplified method.					

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	ii. The simplified static analysis method accounts for the relative motion between all points of support. iii. To obtain an equivalent static load for an SSC that can be represented by a simple model, a factor of 1.5 is applied to the peak spectral acceleration of the applicable ground or floor response spectrum. A factor less than 1.5 may be used, if adequate justification is provided.					
3.7.2.2	Natural Frequencies and Responses. To be acceptable, the following information should be provided: A. A summary of modal masses, effective masses, natural frequencies, mode shapes, modal and total responses for the Category I structures, including the containment structure, or a summary of the total responses if the method of direct integration is used. B. The calculated time histories (two horizontal and one vertical), or other parameters of motion, or response spectra (two horizontal and one vertical) used in design, at the major plant equipment elevations and points of support. C. For the multiple time history analysis option, procedures used to account for uncertainties (by variation of parameters) and to develop design responses, including justification for the statistical relationship between input design time histories and output responses. (For example, if the average response spectra generated from the multiple design time histories are used to envelop the design response spectra, then the average responses generated from the multiple analyses are used in design.)					

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3.7.2.3	Procedures Used for Analytical Modeling. A nuclear power plant facility consists of very complex structural systems. To be acceptable, the stiffness, mass, and damping characteristics of the structural systems should be adequately incorporated into the analytical models. Specifically, the following items should be considered in analytical modeling: A. Designation of Systems Versus Subsystems. Category I structures that are considered in conjunction with the foundation and its supporting media are defined as "seismic systems." Other Category I SSCs that are not designated as "seismic systems" should be considered as "seismic subsystems." B. Decoupling Criteria for Subsystems. It can be shown, in general, that frequencies of systems and subsystems have a negligible effect on the error due to decoupling. It can be shown that the mass ratio, R_m, and the frequency ratio, R_f, govern the results where R_m and R_f are defined as: R_m = Total mass of the supported subsyste / Total mass of the supporting system R_f = Fundamental frequency of the supported subsist / Dominant frequency of the support motion The following criteria are acceptable: i. If $R_m < 0.01$, decoupling can be done for any R_f.					

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	ii. If 0.01 < R_m < 0.1, decoupling can be done if 0.8 >_ R_f >_ 1.25. iii. If R_m > 0.1, a subsystem model should be included in the primary system model. If the subsystem is rigid compared to the supporting system, and also is rigidly connected to the supporting system, it is sufficient to include only the mass of the subsystem at the support point in the primary system model. On the other hand, in case of a subsystem supported by very flexible connections, e.g., pipe supported by hangers, the subsystem need not be included in the primary model. In most cases, the equipment and components, which come under the definition of subsystems, are analyzed (or tested) as a decoupled system from the primary structure and the seismic input for the former is obtained by the analysis of the latter. One important exception to this procedure is the reactor coolant system, which is considered a subsystem but is usually analyzed using a coupled model of the reactor coolant system and primary structure. C. <u>Modeling of Structures.</u> Two types of structural models are widely used by the nuclear industry: lumped-mass stick model and finite element model. Either of these two types of modeling techniques is acceptable if the following guidelines are met: i. Lumped-Mass Stick Model For a lumped-mass model, the eccentricities between the centroid (the neutral axis for axial and bending deformation), the center of rigidity (the neutral axis for					

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	shear and torsional deformation), and the center of mass of structures should be included in the seismic model. For selecting an adequate number of discrete mass degrees of freedom in the dynamic modeling to determine the response of all seismic Category I and applicable non-seismic I structures, the acceptance criteria given in Subsection II.1.a.iv of this SRP section are acceptable. ii. Finite Element Model The type of finite element used for modeling a structural system should depend on the structural details, the purpose of the analysis, and the theoretical formulation upon which the element is based. The mathematical discretization of the structure should consider the effect of element size, shape, and aspect ratio on solution accuracy. The element mesh size should be selected on the basis that further refinement has only a negligible effect on the solution results. iii. In developing either a lumped-mass stick model or a finite element model for dynamic response, it is necessary to consider that local regions of the structure, such as individual floor slabs or walls, may have fundamental vibration modes that can be excited by the dynamic seismic loading. These local vibration modes should be adequately represented in the dynamic response model, in order to ensure that the in-structure response spectra include the additional amplification. Also, the additional seismic loading on the overall structure and on the local region is needed for detailed					

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	structural design. In general, three-dimensional models should be used for seismic analyses. However, simpler models can be used if justification can be provided that the coupling effects of those degrees of freedom that are omitted from the three-dimensional models are not significant. D. Representation of Floor Loads, Live Loads, and Major Equipment in Dynamic Model. In addition to the structural mass, mass equivalent to a floor load of 50 pounds per square foot should be included, to represent miscellaneous dead weights such as minor equipment, piping, and raceways. Also, mass equivalent to 25 percent of the floor design live load and 75 percent of the roof design snow load, as applicable, should be included. The mass of major equipment should be distributed over a representative floor area or included as concentrated lumped masses at the equipment locations. E. Special Consideration for Dynamic Modeling of Structures. It has been common practice that the dynamic model used to predict the seismic response of a structure is not as detailed as the structural model used for the detailed design analysis of all applicable load combinations. Therefore, a methodology is needed to transfer the seismic response loads determined from the dynamic model to the structural model used for the detailed design analysis of all applicable load combinations. This is reviewed for technical adequacy on a case-by-case basis.					
3.7.2.4	Soil-Structure Interaction A complete SSI analysis should properly account for all effects due to kinematic and inertial interaction for surface or					

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	embedded structures. Any analysis method based on either a direct approach or a substructure approach can be used provided the following conditions are met: A. The structure, foundation, and soil are properly modeled to ensure that the results of analyses properly capture spatial variation of ground motion, three dimensional effects of radiation damping and soil layering, as well as nonlinear effects from site response analyses. B. The design earthquake ground motions used as input to the SSI analyses should be consistent with the design response spectra as defined in SRP Section 3.7.1. It is noted that there is enough confidence in the current methods used to perform the SSI analysis to capture the basic phenomenon and provide adequate design information; however, the confidence in the ability to implement these methodologies is uncertain. Therefore, in order to ensure proper implementation, the following considerations should be addressed in performing SSI analysis: A. Perform sensitivity studies to identify important parameters (e.g., potential separation and sliding of soil from sidewalls, non-symmetry of embedment, location of boundaries) and to assist in judging the adequacy of the final results. These sensitivity studies can be performed by the use of well-founded and properly substantiated simple models to give better insight; B. Through the use of some appropriate benchmark problems, the user should demonstrate its capability to properly implement any SSI methodologies; and					

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	C. Perform enough parametric studies with the proper variation of parameters (e.g., soil properties) to address the uncertainties (as applicable to the given site) discussed in subsection I.4 of this SRP section. For sites where SSI effects are considered insignificant and fixed base analyses of structures are performed, bases and justification for not performing SSI analyses are reviewed on a case-by-case basis. If the SSI analysis is not required, the input motion at the base of the structures will be the design motion reviewed in SRP Section 3.7.1. The acceptance criteria for the constituent parts of the entire SSI system are summarized as follows: A. Modeling of Structure. The acceptance criteria given under subsection II.3 of this SRP section are applicable. B. Modeling of Supporting Soil. The effect of embedment of structure, groundwater effects, and the layering effect of soil should be accounted for. For the half-space modeling of the soil media, the lumped parameter (soil spring) method and the compliance function methods are acceptable provided that frequency variations and layering effects are incorporated. For the method of modeling soil media with finite boundaries, all boundaries should be properly simulated and the use of types of boundaries should be justified and reviewed on a case-by-case basis. Finite element and finite difference methods are acceptable methods for discretization of a continuum. The properties used in the SSI analysis should					

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	be those that are consistent with soil strains developed in free-field site response analyses. For structures founded on materials having a shear wave velocity of 8,000 feet per second or higher, under the entire surface of the foundation, a fixed base assumption is acceptable. C. Input Ground Motion. The acceptance criteria for generating the input ground motion to be used in the SSI analysis are summarized in the following: i. If the design earthquake ground motion is defined from generic response spectral shapes (e.g, Reg. Guide 1.60 or NUREG-0098), the location of the ground motion should be consistent with the properties of the soil profile. For profiles consisting of competent soil or rock, with relatively uniform variation of properties with depth, the ground motion should be located at the soil surface at the top of the finished grade. For profiles consisting of one or more soft and/or thin soil layers overlaying competent material, the ground motion should be located at an outcrop (real or hypothetical) at the top of the competent material in the vicinity of the site. ii. If the design earthquake ground motion is defined from site-specific evaluations of uniform hazard spectra, the location of the ground motion should be at the ground surface in the free-field. In developing the ground motion at the surface, the potential effects of soft soil layers need to be considered. For sites with soil layers					

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	near the surface that will be completely excavated to expose competent material, the ground motion response spectra are specified on an outcrop or a hypothetical outcrop that will exist after excavation. Motions at this hypothetical outcrop should be developed as a free surface motion, not as an in-column motion. Competent material is defined as in-situ material having a minimum shear wave velocity of 1,000 feet/second (fps). iii. When the guidance for SSI analysis presented above is not completely implemented, the spectral amplitude of the acceleration response spectra (horizontal component of motion) in the free field at the foundation depth shall be not less than 60 per cent of the corresponding design response spectra at the finished grade in the free field. When variation in soil properties are considered (as required by the "Specific Guidelines for SSI Analysis" below), the 60 percent limitation may be satisfied using an envelope of the three spectra corresponding to the three soil properties. If the accompanying rotational components of the input motion are ignored, no reduction is permitted in the horizontal component at the foundation level. Specific Guidelines for SSI Analysis The following specific guidelines are provided here to facilitate the review and draw the attention of reviewers to some important aspects of the SSI analysis. These guidelines are not necessarily requirements for the acceptance of any methodologies or an SSI analysis.					

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	• The behavior of soil, though recognized to be nonlinear, can often be approximated by linear techniques. Truly nonlinear analysis is not required unless the comparison of results from large-scale tests or actual earthquakes and analytical results indicate deficiencies that cannot be accounted for in any other manner. The nonlinear soil behavior may be accounted for by the following: - Using equivalent linear soil material properties typically determined from an iterative linear analysis of the free-field soil deposit. This accounts for the primary nonlinearity, or - Performing an iterative linear analysis of the coupled soil-structure system. This accounts for the primary and secondary nonlinearities. In the event the nonlinear analysis is chosen, the results of the nonlinear analysis should be judged on the basis of the linear or equivalent linear analysis (NUREG/CP-0054). • Superposition of horizontal and vertical response as determined from separate analyses is acceptable (assuming nonlinear effects are not important) considering the simple material models now available. • The strain-dependent soil properties (e.g., shear modulus, damping) estimated from analysis of the seismic motion in the free field shall be consistent with the geotechnical information reviewed in SRP Section 2.5.4. • For cases using standard plant designs, where the site					

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	specific spectra fall below the standard plant design spectra, the SSI evaluations are addressed in the standard plant design. • Enough SSI analyses should be performed so as to account for the effects of the potential variability in the properties of the soils and rock at the site. At least three soil/rock profiles should be considered in these analyses, namely, a best estimate (BE) profile, a lower bound (LB) and an upper bound (UB) profile in the evaluation of SSI effects. The properties of each layer of the site profile are typically defined in terms of its low-strain shear modulus and strain-dependent modulus degradation and strain-dependent hysteretic damping properties. These may be determined from dynamic laboratory testing of the site materials, information obtained from the published literature, or both. The set of properties appropriate for a given soil is reviewed for its adequacy. For a particular site, the iterated shear modulus and damping values are typically determined from the results of a number of free-field site response analyses, which are intended to account for the effects of the site-specific design ground motions as well as the site nonlinear properties. If only a single site response calculation is performed, with the low strain property of each material layer selected at its BE value, the resulting iterated property is then determined. The upper and lower bound values of soil/rock shear modulus (G) can then be defined in terms of their best estimate values as: $G_{LB} = G_{BE} / (1+COV)$ $G_{UB} = G_{BE} \times$					

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	(1+COV) where COV is the coefficient of variation considered appropriate for the site materials. The corresponding damping properties should be defined at the compatible strains associated with the shear moduli. If many site response calculations are performed (30 to 60 site response calculations) using Monte Carlo techniques to develop site properties, these calculations are typically used to determine the BE, LB and UB iterated site properties. The BE properties are determined from the mean of the resulting properties and the UB and LB values selected from the +/- one sigma values. A sufficient number of site response calculations need to be performed, to ensure that a stable value of sigma for each material of the profile is obtained. For well-investigated sites (see RGs 1.132 and 1. 138), the COV should be no less than 0.5. For sites that are not well investigated, the COV for shear modulus shall be at least 1.0. These COV requirements apply to the “single site response calculation”, as well as the “many site response calculations” described above. In no case should the lower bound shear modulus be less than that value consistent with standard foundation analysis that yields foundation settlement under static loads exceeding design allowables. The upper bound shear modulus should not be less than the best estimate shear modulus defined at low strain and as determined from the geophysical testing program. In no case should the material soil damping as expressed by the hysteretic damping ratio exceed 15 percent (NUREG/CR-1161).					

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	For the case of analyses using generic broad-banded ground motion spectra, the best estimate shear modulus and damping of each material of the site profile can be defined in terms of its low strain values. The upper and low bound shear moduli can then be defined at twice and one-half the best estimate values, with damping maintained at its low strain value. Alternate approaches can be reviewed on a case-by-case basis. • For dipping soil and rock strata, it is necessary to account for the coupling between the horizontal and vertical degrees of freedom in the stiffness and free- field seismic motion definitions. Also, there may be sites where the reactor building or a seismic Category I structure may have an embedded foundation close to an embankment or a natural slope that preclude the assumption of uniform foundation condition. For such sites, modeling and analysis techniques are reviewed on a case-by-case basis. • Finite Boundary Modeling or Direct Solution Technique The direct solution method is characterized as follows: - Each analysis of the soil and structures is performed in one step. - Finite element or finite difference discrete methods of analysis are used to spatially discretize the soil-structure system. - Definition of the motion along the boundaries of the model (bottom and sides) is either known, assumed, or computed as a precondition of the analysis.					

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	Dynamic analysis can be performed using either frequency-domain (limited to linear analysis) or time-integration methods. The mesh size should be adequate for representing the static stress distribution under the foundation and transmitting the frequency content of interest. The following limitations should be observed for deep soil sites: - The model depth, generally, should be at least twice the base dimension below the foundation level, which should be verified by parametric studies. - The fundamental frequency of the soil (or backfill) stratum should be well below the structural frequencies of interest. - All structural modes of significance should be included. • Half Space or Substructure Solution Technique The half space or substructure approach generally comprises the following steps: (1) Determine the motion of the massless foundation, including both translational and rotational components. (2) Determine the foundation stiffness in					

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	terms of frequency- dependent impedance functions. (3) Perform SSI analysis. The procedures, modeling assumptions and analytical bases adopted for performing the half space or substructure analysis, including use of frequency- independent soil spring parameters, and the spring and damping coefficients, will be reviewed on a case-by-case basis. • There are advanced analytical methods that are being considered by the nuclear industry (e.g., the effects of incoherent ground motion) to reduce the potential effects of high frequency ground motion input. These might be used when a site acceptability determination is performed as discussed in subsection II.4 of SRP Section 3.7.1. If incoherency is used to reduce the high frequency response, the potential effects of increasing other responses (e.g., overturning and torsional responses) shall be considered. When approved for use by the NRC, via issuance of interim staff guidance, it should be noted that the effects of incoherent ground motion may be considered either at the Design Certification stage, or at the site-specific application stage, but not both. If any advanced analytical methods are utilized, the technical basis and analysis results are subject to detailed review on a case-by-case basis.					
3.7.2.5	Development of In-Structure Response Spectra. RG 1.122 describes methods generally acceptable to the staff for developing the two horizontal and the vertical in- structure response spectra (e.g., floor response spectra) from the time					

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	history motions resulting from the dynamic analysis of the supporting structure. The topics addressed are A SRSS Combination of the three in-structure response spectra in a given direction (e.g., x direction), developed from the output time histories from separate analyses of the three directions (x, y, z) of input motion. SRSS combination is not applicable, if the three directions of the input motion are applied simultaneously in a single analysis. B. Frequency increments for calculation of spectral accelerations. C. Spectrum smoothing and broadening to account for uncertainty. The guidance in RG 1.122 is augmented as follows: (1) SRSS combination applies to all cases where the three directions of input motion are analyzed separately. There is no longer a distinction made between symmetric and unsymmetric structures. (2) The 3 Hz frequency increment in the last row of RG 1.122, Table 1, applies up to the highest frequency of interest. This typically will be the PGA frequency of the design ground response spectrum, which in some cases may significantly exceed 33 Hz. (3a) When a single set of three artificial time histories is used as the input motion to the supporting structure, the in-structure response spectra are smoothed and					

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	broadened in accordance with the provisions of RG 1.122, to account for uncertainty. (3b) When multiple sets of three time histories, derived from actual earthquake records, are used as the input motion to the supporting structure, the multiple sets of in-structure response spectra already account for some of the uncertainty. Therefore, the provisions of RG 1.122, to account for uncertainty, do not strictly apply. The use of multiple sets of time histories to generate in-structure response spectra is reviewed and accepted on a case-by-case basis. Particularly, the basis for procedures used to account for uncertainties (by variation of parameters) are evaluated. The same acceptance criteria apply to the in-structure response spectra as apply to the design ground response spectrum, reviewed in subsection II.I.13 of SRP Section 3.7.1. As an example, if the average of the multiple response spectra generated from the multiple design time histories is used to envelop the design ground response spectrum, then the average of the multiple in-structure response spectra generated from the multiple analyses (each of which used one of the multiple design time histories) are used in design. An evaluation of the statistical correlation between the input ground response spectrum and the output in-structure response spectra should also be provided.					

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	The methods used for direct generation of in-structure response spectra are reviewed and accepted on a case-by-case basis.					
3.7.2.6	Three Components of Earthquake Motion. RG 1.92, describes acceptable methods for combining the responses due to three components of earthquake motion, for both the response spectrum method and the time history method. Use of alternate methods are evaluated on a case-by-case basis for acceptability. When the three components of earthquake motion are applied simultaneously, using a set of three artificial time histories, the statistical independence of the time histories should be demonstrated. See subsection II.1.13 of SRP 3.7.1 for the acceptance criteria to demonstrate statistical independence.					
3.7.2.7	Combination of Modal Responses. RG 1.92, describes acceptable methods for combination of modal responses, including consideration of closely-spaced modes and high-frequency modes, when the response spectrum method of analysis is used to determine the dynamic response of damped linear systems. Use of alternate methods are evaluated on a case-by-case basis for acceptability. When the modal superposition time history method of analysis is used, modal responses are combined algebraically, at each output time step. In accordance with RG 1.92, only modes with natural frequencies less than or equal to the ZPA frequency of the input spectrum are included in the modal superposition time history analysis. The contribution of the higher frequency modes to the total response is calculated by the missing mass approach. Since this contribution is in-phase with the input time history, it is treated as one additional modal response, that is scaled by the input time history normalized to the ZPA, and combined algebraically with the modal					

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	superposition time history solution at each output time step.					
3.7.2.8	Interaction of Non-Category I Structures with Category I SSCs. All non-Category I structures should be assessed to determine whether their failure under SSE conditions could impair the integrity of seismic Category I SSCs, or result in incapacitating injury to control room occupants. Each non-Category I structure should meet at least one of the following criteria: A. The collapse of the non-Category I structure will not cause the non-Category I structure to strike a Category I SSC. B. The collapse of the non-Category I structure will not impair the integrity of seismic Category I SSCs, nor result in incapacitating injury to control room occupants. The non-Category I structure will be analyzed and designed to prevent its failure under SSE conditions, such that the margin of safety is equivalent to that of Category I structures. The disposition of each non-Category I structure should be formally documented. For criterion (b), it is necessary to provide the technical basis for the determination that collapse of the non-Category I structure is acceptable. This should include a description of any additional loads imposed on the Category I SSCs and the method used to conclude that these loads are not damaging. Also, any protective shields installed to prevent direct impact on Category I SSCs should be described					
3.7.2.9	Effects of Parameter Variations on Floor Response Spectra.					

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	Consideration should be given in the analysis to the effects on floor response spectra (e.g., peak width) of expected variations of structural properties, damping values, soil properties, and SSI. The acceptance criteria for the consideration of the effects of parameter variations are provided in subsection II.5 of this SRP section. In addition, for concrete structures, the effect of potential concrete cracking on the structural stiffness should be specifically addressed.					
3.7.2.10	Use of Equivalent Vertical Static Factors. The use of equivalent static load factors to calculate vertical response loads for the seismic design of Category I SSCs, in lieu of the use of a vertical seismic system dynamic analysis, is acceptable only if it can be demonstrated that the SSC is rigid in the vertical direction, or the acceptance criteria in subsection 3.7.2.II.1.b of this SRP section are satisfied. The criterion for rigidity is that the lowest frequency in the vertical direction is higher than the ZPA frequency of the input ground or in-structure spectrum.					
3.7.2.11	Methods Used to Account for Torsional Effects. An acceptable method to account for torsional effects in the seismic analysis of Category I structures is to perform a dynamic analysis that incorporates the torsional degrees of freedom. An acceptable alternative, if properly justified, is the use of static factors to account for torsional accelerations in the seismic design of Category I structures. To account for accidental torsion, an additional eccentricity of ± 5 percent of the maximum building dimension shall be assumed for both horizontal directions. The magnitude and location of the two eccentricities is determined separately for each floor elevation.					
3.7.2.12	Comparison of Responses. If both the time history analysis method and the response spectrum analysis method are used to analyze an SSC, the peak responses obtained from these two methods should be					

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	compared, to demonstrate approximate equivalency between the two methods.					
3.7.2.13	Analysis Procedure for Damping. Either the composite modal damping approach or the modal synthesis technique can be used to account for element-associated damping. Use of composite modal damping for computing the response of systems with nonclassical modes may lead to unconservative results (Miller, et al., 1985). Therefore, the composite modal damping approach is acceptable provided the composite modal damping is limited to 20 percent. One of the other methods mentioned below is generally applicable if the composite modal damping exceeds 20 percent. A. Time domain analysis using complex modes/frequencies, B. Frequency domain analysis, or C. Direct integration of uncoupled equation of motion. For the composite modal damping approach, two techniques of determining an equivalent modal damping matrix or composite damping matrix are commonly used. They are based on the use of the mass or stiffness as a weighting function in generating the composite modal damping. The formulations lead to: $\overline{\beta}_j = \{\phi\}^T [\overline{M}] \{\phi\}$		(1) (2)			

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	$$\beta_j = \frac{\{\phi\}^T [\bar{K}] \{\phi\}}{K^*}$$ where $K^* = \{\phi\}^T [K] \{\phi\}$, [K] = assembled stiffness matrix, β_j = equivalent modal damping ratio of the j^{th} mode, $[\bar{K}]$, $[\bar{M}]$ = the modified stiffness or mass matrix constructed from element matrices formed by the product of the damping ratio for the element and its stiffness or mass matrix, and $\{\phi\}$ = j^{th} normalized modal vector. For models that take SSI into account by the lumped soil spring approach, the method defined by equation (2) is acceptable. For fixed base models, either equation (1) or (2) may be used. Other techniques based on modal synthesis have been developed and are particularly useful when more detailed data on the damping characteristics of structural subsystems are available. The modal synthesis analysis procedure consists of (1) extraction of sufficient modes from the structure model, (2) extraction of sufficient modes from the finite element soil model, and (3) performance of a coupled analysis using the modal synthesis technique, which uses the data obtained in steps (1) and (2) with					

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	appropriate damping ratios for structure and soil subsystems. This method is based upon satisfaction of displacement compatibility and force equilibrium at the system interfaces and uses subsystem eigenvectors as internal generalized coordinates. This method results in a nonproportional damping matrix for the composite structure, and equations of motion have to be solved by direct integration or by uncoupling them by use of complex eigenvectors. Other techniques for estimating the equivalent modal damping of a SSI model are reviewed on a case-by-case basis.					
3.7.2.14	Determination of Seismic Overturning Moments and Sliding Forces for Seismic Category I Structures. To be acceptable, the determination of the design overturning moment and sliding force should incorporate the following items: A. Three components of input motion. B. Conservative consideration of the simultaneous action of vertical and horizontal seismic forces. Additional information on load combinations is provided in SRP Section 3.8.5.					
3.7.3, Rev. 3 (03/2007)	Seismic Subsystem Analysis					
3.7.3.1	Seismic Analysis Methods. The acceptance criteria provided in SRP Section 3.7.2, subsection II.1, are applicable.					
3.7.3.2	Determination of Number of Earthquake Cycles. During the plant life at least one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBEs), if applicable, should be assumed. The number of cycles per earthquake should be obtained from the time history used for the system analysis, or a					

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	minimum of 10 maximum stress cycles per earthquake may be assumed. When the OBE is defined as less than one-third the SSE (and therefore the OBE does not need to be considered in design), there may be certain structural elements which still need to be evaluated for fatigue due to the OBE induced stress cycles. In these instances, the guidance for determining the number of earthquake cycles for use in fatigue calculations should be the same as the guidance provided in SRM for SECY-93-087 dated July 21, 1993 for piping systems. The number of earthquake cycles to consider are two SSE events with 10 maximum stress cycles per event. This is considered to be equivalent to the cyclic load basis of one SSE and five OBEs. Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1987, Appendix D.					
3.7.3.3	Procedures Used for Analytical Modeling. The acceptance criteria provided in SRP Section 3.7.2, subsection II.3, are applicable.					
3.7.3.4	Basis for Selection of Frequencies. To avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than 1/2 or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads.					
3.7.3.5	Analysis Procedure for Damping. The acceptance criteria provided in SRP Section 3.7.2, subsection II.13, are applicable.					
3.7.3.6	Three Components of Earthquake Motion. The acceptance criteria provided in SRP Section 3.7.2, subsection II.6, are applicable.					

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3.7.3.7	Combination of Modal Responses. The acceptance criteria provided in SRP Section 3.7.2, subsection II.7, are applicable.					
3.7.3.8	Interaction of Other Systems With Seismic Category I Systems. To be acceptable, each non-seismic Category I system should be designed to be isolated from any seismic Category I system by either a constraint or barrier, or should be remotely located with regard to the seismic Category I system. If it is not feasible or practical to isolate the seismic Category I system, adjacent non-seismic Category I systems should be analyzed according to the same seismic criteria as applicable to the seismic Category I system. For non-seismic Category I systems attached to seismic Category I systems, the dynamic effects of the non-seismic Category I systems should be simulated in the modeling of the seismic Category I system. The attached non-seismic Category I systems, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the seismic Category I system. The acceptance criteria provided in SRP Section 3.7.2, subsection II.8, are applicable to all seismic Category I SSCs at the system and subsystem level.					
3.7.3.9	Multiply-Supported Equipment and Components With Distinct Inputs. Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different. A conservative and acceptable approach for analyzing equipment items supported at two or more locations is to define a uniform response spectrum (URS) that envelopes all of the individual response spectra at the various support locations. The URS is applied at all locations to calculate the maximum inertial responses of the equipment. This is referred to as the uniform					

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	support motion (USM) method. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the building structural response calculations. The support displacements can then be imposed on the supported equipment in the most unfavorable combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method. The URS method described above can result in considerable overestimation of seismic responses. In the case of multiply-supported equipment in a single structure and/or spanning between structures, an alternate method that can be used is the independent support motion (ISM) approach. Guidance and criteria for the use of the ISM method is given in NUREG-1061, Section 2, Volume 4.. If the ISM method is utilized, all of the criteria presented in NUREG-1061 related to the ISM method must be followed. In lieu of the response spectrum approach, time histories of support motions may be used as input excitations to the subsystems. The time history approach is considered to provide more realistic results as compared to the USM or ISM methods.					
3.7.3.10	Use of Equivalent Vertical Static Factors. The acceptance criteria provided in SRP Section 3.7.2, subsection II.10, are applicable.					
3.7.3.11	Torsional Effects of Eccentric Masses. For seismic Category I subsystems, when the torsional effect of an eccentric mass is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for judging the significance will be reviewed on a case-by-case basis.					
3.7.3.12	Seismic Category I Buried Piping, Conduits, and Tunnels. For seismic Category I buried piping, conduits, tunnels, and any other					

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	subsystems, the following items should be considered in the analysis: A. Two types of groundshaking-induced loadings must be considered for design. i. Relative deformations imposed by seismic waves traveling through the surrounding soil or by differential deformations between the soil and anchor points. ii. Lateral earth pressures and ground-water effects acting on structures. B. The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Procedures using the principles of the theory of structures on elastic foundations are acceptable. C. When applicable, the effects due to local soil settlements, soil arching, etc., should also be considered in the analysis. D. Actual methods used for determining the design parameters associated with seismically induced transient relative deformations are reviewed and accepted on a case-by-case basis. Additional information, for guidance purposes only, can be found in NUREG/CR-1161, page 26, in American Society of Civil Engineers (ASCE) Standard 4-98, Section 3.5.2 and in ASCE Report - Seismic Response of Buried Pipes and Structural					

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	Components.					
3.7.3.13	Methods for Seismic Analysis of Seismic Category I Concrete Dams. For the analysis of all seismic Category I concrete dams, an appropriate approach that takes into consideration the dynamic nature of forces (due to both horizontal and vertical earthquake loadings), the behavior of the dam material under earthquake loadings, soil-structure interaction (SSI) effects, and nonlinear stress-strain relations for the soil, should be used. Analysis of earthen dams is reviewed under SRP Section 2.5.5, “Stability of Slopes.”					
3.7.3.14	Methods for Seismic Analysis of Above-Ground Tanks. Most above-ground fluid-containing vertical tanks do not warrant sophisticated, finite element, fluid-structure interaction analyses for seismic loading. However, the commonly used alternative of analyzing such tanks by the "Housner-method" described in TID-7024 may be inadequate in some cases. The major problem is that direct application of this method is consistent with the assumption that the combined fluid-tank system in the horizontal impulsive mode is sufficiently rigid to justify the assumption of a rigid tank. For flat-bottomed tanks mounted directly on their bases, or tanks with very stiff skirt supports, the assumption leads to the usage of a spectral acceleration equal to the zero-period base acceleration. Recent studies (Veletsos (1974 and 1984), Veletsos and Yang (1977), Veletsos and Tang (1989), Haroun and Housner (1981), have shown that for typical tank designs, the frequency for this fundamental horizontal impulsive mode of the tank shell and contained fluid is such that the spectral acceleration may be significantly greater than the zero-period acceleration. Thus, the assumption of a rigid tank could lead to inadequate design loadings. The SSI effects may also be very important for tank responses, and they may need to be considered for both horizontal and vertical motions.					

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	The acceptance criteria below are based upon the information contained in TID-7024 and NUREG/CR-1161. Additional guidance is provided in ASCE Standard 4-98, Section 3.5.4. These references also contain acceptable calculational techniques for the implementation of these criteria. The use of other approaches meeting the intent of these criteria can also be considered if adequate justification is provided. A. A minimum acceptable analysis must incorporate at least two horizontal modes of combined fluid-tank vibration and at least one vertical mode of fluid vibration. The horizontal response analysis must include at least one impulsive mode in which the response of the tank shell and roof are coupled together with the portion of the fluid contents that moves in unison with the shell. In addition, the fundamental sloshing (convective) mode of the fluid must be included in the horizontal analysis. B. The fundamental natural horizontal impulsive mode of vibration of the fluid-tank system must be estimated giving due consideration to the flexibility of the supporting medium and to any uplifting tendencies for the tank. It is unacceptable to assume a rigid tank unless the assumption can be justified. The horizontal impulsive-mode spectral acceleration, $S_{a1,}$ is then determined using this frequency and the appropriate damping for the fluid-tank system. Alternatively, the maximum spectral acceleration corresponding to the relevant damping may be used. C. Damping values used to determine the spectral acceleration in the impulsive mode shall be based upon the system damping associated with the tank shell					

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	material as well as with the SSI, as specified in NUREG/CR-1161 and Veletsos and Tang (1989). D. In determining the spectral acceleration in the horizontal convective mode, S_{a2}, the fluid damping ratio shall be 0.5 percent of critical damping unless a higher value can be substantiated by experimental results. E. The maximum overturning moment, M_o, at the base of the tank should be obtained by the modal and spatial combination methods discussed in subsection II of SRP Section 3.7.2. The uplift tension resulting from M_o must be resisted either by tying the tank to the foundation with anchor bolts, etc., or by mobilizing enough fluid weight on a thickened base skirt plate. The latter method of resisting M_o must be shown to be conservative. F. The seismically induced hydrodynamic pressures on the tank shell at any level can be determined by the modal and spatial combination methods in SRP Section 3.7.2. The maximum hoop forces in the tank wall must be evaluated with due regard for the contribution of the vertical component of ground shaking. The effects of soil-structure interaction should be considered in this evaluation unless justified otherwise. The hydrodynamic pressure at any level must be added to the hydrostatic pressure at that level to determine the hoop tension in the tank shell. G. Either the tank top head must be located at elevation higher than the slosh height above the top of the fluid or else must be designed for pressures resulting from fluid sloshing against this head.					

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	H. At the point of attachment, the tank shell must be designed to withstand the seismic forces imposed by the attached piping. An appropriate analysis must be performed to verify this design. I. The tank foundation (see also SRP Section 3.8.5) must be designed to accommodate the seismic forces imposed on it. These forces include the hydrodynamic fluid pressures imposed on the base of the tank as well as the tank shell longitudinal compressive and tensile forces resulting from Mo. J. In addition to the above, a consideration must be given to prevent buckling of tank walls and roof, failure of connecting piping, and sliding of the tank.					
3.7.4, Rev. 2 (03/2007)	Seismic Instrumentation					
3.7.4.1	Comparison with RG 1.12. The seismic instrumentation program is considered to be acceptable if it is in accordance with guidance provided in RG 1.12. The bases for elements of the proposed seismic instrumentation program that differ from RG 1.12 must be provided. This guide recommends installation of solid-state digital time-history accelerographs at appropriate locations in order to provide time history data on the seismic response of the free-field, containment structure, and other Seismic Category I structures. The COL, DC, and construction permit (CP) applicants should provide solid-state digital instrumentation that will enable the processing of data at the plant site within 4 hours of the seismic event. A triaxial time-history accelerograph should be provided at					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	each of the locations specified in RG 1.12. Triggering of the free-field or any foundation-level accelerograph should be annunciated in the control room. In addition, applicants should provide a rationale for the placement of instrumentation which is consistent with maintaining occupational radiation exposures ALARA for the location. With regard to operability and installation, applicants should demonstrate that the seismic instrumentation will be operable during all modes of plant operation, including periods of plant shutdown. In addition, the applicant's maintenance and repair procedures should provide for keeping the maximum number of instruments in service during plant operation and shutdown. Instruments should be designed and installed so that the mounting is rigid and oriented so that the horizontal components are parallel to the orthogonal axes assumed for the seismic analysis. Also, protections against accidental impacts should be provided. With regard to capabilities and characteristics, the seismic instrumentation should include each of the specifications identified in RG 1.12. This includes provisions for in- service testing, a remote alarm to indicate actuation, recording capabilities, sufficient dynamic range and sampling rate, and a low and adjustable actuating level or trigger. In addition, both vertical and horizontal input vibratory ground motion should actuate the same time-history accelerograph.					
3.7.4.2	Comparison with RG 1.166. The seismic instrumentation program is considered to be acceptable if it contains pre-earthquake planning and post-earthquake actions in accordance with RG 1.166. The bases for elements of the proposed seismic instrumentation program that differ from RG 1.166 must be provided. This guide provides guidance for a timely evaluation after an earthquake					

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	of the recorded seismic instrumentation data and for determining whether plant shutdown is required. The COL, DC, and CP applicants should provide a description of both pre-earthquake planning and post-earthquake actions in order to make a rapid determination of the degree of severity of the seismic event. The data from the seismic instrumentation, coupled with information obtained from a plant walkdown, should be used to make the initial determination of whether the plant must be shut down. With regard to the necessary baseline data, information related to seismic instrumentation, including instrument calibration, should be kept at the plant. The applicant's program should also describe the necessary actions, such as selecting equipment and structures for inspections and the content of the baseline inspections, that are to be taken immediately after an earthquake, as described in RG 1.166. With regard to the evaluation of ground motion records, the applicant's program should describe data identification (i.e., record collection log), data collection, and record evaluation procedures. Shutdown of the nuclear power plant is required if the vibratory ground motion experienced exceeds that of the OBE. A criterion for determining exceedance of the OBE is provided in the Electric Power Research Institute (EPRI) document EPRI NP-5930, "A Criterion for Determining Exceedance of the Operating Basis Earthquake." This criterion is based on a threshold response spectrum ordinate check and a cumulative absolute velocity (CAV) check. The ground motion evaluation should consist of a check on the response spectrum and CAV and a check on the operability of the instrumentation as described in RG 1.166. This evaluation should take place within 4 hours of the earthquake.					

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3.8.1, Rev. 2 (03/2007)	Concrete Containment					
3.8.1.1	Description of the Containment. The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the criteria set forth in Section 3.8.1.1 of RG 1.206. If the concrete containment has new or unique features that are not specifically covered in RG 1.206, the reviewer determines whether the information necessary to accomplish a meaningful review of the structural aspects of these new or unique features is presented. RG 1.206 provides the basis for evaluating the description of Seismic Category I structures to be included in a DC or a COL application. RG 1.70 provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).					
3.8.1.2	Applicable Codes, Standards, and Specifications. The design, materials, fabrication, erection, inspection, testing, and inservice surveillance of concrete containments are covered by codes, standards, specifications, and guides that are applicable either in their entirety or in part. The following codes and guides are acceptable: RG — Title 1.7 — Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident 1.35 — Inservice Inspection of Ungrouted					

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ID	Title/Requirement		Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
		Tendons in Prestressed Concrete Containments					
	1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments					
	1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons					
	1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants					
	1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures					
	1.115	Protection Against Low Trajectory Turbine Missiles					
	1.136	Materials, Construction, and Testing of Concrete Containments					

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3.8.1.3	Loads and Loading Combinations. The specified loads and load combinations are acceptable if found to be in accordance with Article CC-3000 of the ASME Code with the exceptions listed below applied to the requirements specified in Table CC-3230-1. RG 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments," provides additional guidance for design requirements, including load and load combinations, which should be considered in the design of concrete containments. A. The maximum values of P_a, T_a, R_a, R_{rr}, R_{rj}, and R_{rm} should be applied simultaneously, where appropriate, unless a time-history analysis is performed to justify doing otherwise. B. Hydrodynamic loads resulting from LOCA and/or SRV actuation should be combined as indicated in the appendix to this SRP section. Fluid structure interaction associated with these hydrodynamic loads and those from earthquakes should be considered. C. As noted in Appendix S to 10 CFR Part 50, the OBE is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input. If the OBE is set at one-third or less of the SSE ground motion, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the containment remains functional and is within applicable stress, strain, and deformation limits. SRP Section 3.7 provides further guidance on the use of OBE.					

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	When the OBE is defined as less than one-third of the SSE (and therefore the OBE does not need to be considered in design), certain structural elements of the containment (e.g., penetrations or bellows) still need to be evaluated for fatigue resulting from the OBE-induced stress cycles. In these instances, the guidance for determining the number of earthquake cycles for use in fatigue calculations should be the same as the guidance provided in the staff requirements memorandum (SRM) for SECY-93-087 for piping systems. The number of earthquake cycles to consider is two SSE events with 10 maximum stress cycles per event. Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987. D. Where post-LOCA flooding is a design consideration for the plant, the load combination in the ASME Code containing LOCA flooding along with OBE should be considered. Where post-LOCA flooding is combined with the OBE set at one-third or less of the SSE for the plant, this load combination may be eliminated provided the load combination is shown to be less severe than one of the other load combinations. E. For those plants to which 10 CFR 50.34(f)(3)(v) applies, the requirements regarding loads and loading combinations include the following: Containment integrity should be maintained by meeting the requirements of Subarticle CC-3720 of the ASME Code (considering pressure and					

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	dead load alone) during an accident that releases hydrogen generated from 100-percent metal-water reaction of the fuel cladding and accompanied by either hydrogen burning or added pressure from postaccident inerting (assuming carbon dioxide is the inerting agent). At a minimum, the ASME Code requirements will be met for a combination of dead load and an internal pressure of 310 Kilo Pascals (KPa) or 45 pounds per square in gauge (psig) The containment structure should be designed against the loadings produced by the inadvertent full actuation of a postaccident inerting hydrogen control system (assuming carbon dioxide), excluding seismic or design-basis accident loadings. Under these conditions, the loadings should not produce strains in the containment liner in excess of the limits established in Subarticle CC-3720 of the ASME Code. The requirements of Subarticle CC-3720 of the ASME Code should be met when the containment structure is exposed to the following loading conditions: i. For the factored load category: $D + P_{g1} + [P_{g2} \text{ or } P_{g3}]$ ii. For the service load category, the strains in the containment liner should not exceed the limits set forth in Subarticle CC-3720 when exposed to pressure P_{g3}. iii. As a minimum design condition for either condition i					

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	or ii above, the following load combination must be satisfied: D + 310 kPa (45 psig) where D = Dead load P_{g1} = Pressure resulting from an accident that releases hydrogen generated from 100-percent metal-water reaction of the fuel cladding P_{g2} = Pressure resulting from uncontrolled hydrogen burning P_{g3} = Pressure resulting from postaccident inerting, assuming carbon dioxide is the inerting agent F. 10 CFR 50.44 requires that an analysis be performed that demonstrates that the containment structural integrity is maintained under loads resulting from combustible gases generated from metal-water reaction of the fuel cladding. An analytical technique accepted by the NRC staff should demonstrate the containment structural integrity. This analysis should include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. RG 1.7 presents further guidance on the analytical technique, loads, loading combination, and acceptance criteria. G. Other site-related or plant-related loads applicable to					

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	containment such as floods, explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures need to be considered. The staff reviews the inclusion of these loads in the factored load combinations on a case-by-case basis. H. The review considers those loads encountered during construction of the containment, which include dead loads, live loads, prestress loads, temperature, wind, earth pressure, snow, rain, and ice, and construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. Structural Engineering Institute (SEI)/American Society of Civil Engineers (ASCE) Standard 37 gives additional guidance on construction loads for use in the load combination for construction given in Table CC-3230-1 of the ASME Code. When SEI/ASCE Standard 37 and the ASME Code/SRP provide conflicting criteria, then the ASME Code/SRP should govern.					
3.8.1.4	Design and Analysis Procedures. The procedures for design and analysis used for the concrete containment, including the steel liner, are acceptable if found in accordance with those stipulated in Article CC-3300 of the ASME Code and RG 1.136 (see Subsection II.3 of this SRP section). In particular, for the areas of review outlined in Subsection I.4 above, the following procedures are, in general, acceptable: A. <u>Assumptions on Boundary Conditions.</u> The boundary conditions depend on the methods of analysis to be used and the portions of the containment shell to be separately analyzed. If the analysis is to involve the use of the finite element technique and is to include the foundation media,					

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	the boundary would be the demarcation lines separating the foundation mass taken into consideration in the analysis from the surrounding media. The boundaries of the foundation mass considered should be selected to provide comparable or conservative results to those corresponding to a further extension of the boundaries. This is reviewed on a case-by-case basis. If the analysis considers only the containment shell and its foundation mat, then the bottom of the foundation slab is the boundary of the analytical model. The foundation media should be represented by appropriate soil springs. If separate analyses of the containment shell and the base mat are to be used, it is considered acceptable if strain compatibility of the bottom portion of the shell with the base mat is maintained. B. Axisymmetric and Nonaxisymmetric Loads. Even with the large penetrations and buttresses that may be used in the shell, the overall behavior of the shell has been shown to be axisymmetric under pressure. Therefore, it is acceptable to make such an assumption with respect to the containment geometry. However, for loads such as those induced by wind, tornadoes, earthquakes, and pipe rupture, the analysis should consider the nonaxisymmetric effect of these loads. C. Transient and Localized Loads. During normal operation, a linear temperature gradient across the containment wall thickness may develop. After a LOCA, however, the sudden increase in temperature in the steel liner and the adjacent concrete may produce a nonlinear transient temperature gradient across the containment wall thickness. The analysis should consider the effects of such					

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	transient loads. In a PWR ice condenser containment, nonaxisymmetric and transient pressure loads resulting from compartmentalization inside the containment will develop after a LOCA. For a BWR pressure-suppression containment, the analysis should consider nonaxisymmetric and transient pressure loads resulting from earthquakes, LOCA, and/or SRV actuation (including fluid-structure interaction). For the effects of such localized and transient loads, the overall behavior of the containment structure should first be determined. A portion of the containment shell, within which the localized or transient load is located, should then be analyzed, using the results obtained from the analysis of the overall vessel behavior as boundary conditions. D. Creep, Shrinkage, and Cracking of Concrete. Creep and shrinkage values for concrete should be established by tests performed on the concrete to be used in the containment structure or from data obtained on completed containments constructed of the same kind of concrete. In establishing these values, the analysis should consider the differences in the environment between the test samples and the actual concrete in the structure. For some containments, cracking of concrete is expected to occur based on the structural integrity test performed in accordance with Article CC-6000 of the ASME Code. Also, based on load combinations that include the design pressure load with earthquake loads, additional concrete cracking would be expected to occur. Concrete cracking can					

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	cause redistribution of member forces because of the various loadings applied to the structure. Concrete cracking can also affect the stiffness of the containment and cause shifting of the natural frequency, thereby affecting the response/loads used to design the containment. Accordingly, the analysis used to calculate the dynamic response of the containment resulting from dynamic loads such as earthquake and hydrodynamic loads (if applicable) needs to consider the potential effects of concrete cracking, if significant. The approach used should include the effect of redistribution of the various loads caused by concrete cracking. With improvements in the development of computer programs for analysis of concrete structures, the evaluation of concrete cracking can be analyzed directly within the finite element model. Alternatively, additional analyses can treat the effect of concrete cracking by determining the response of the containment to variation in the stiffness characteristics of the containment shell (e.g., shear stiffness and tensile membrane stiffness reduction). As stated in CC-3320 of the ASME Code, the effects of reduction in shear stiffness and tensile membrane stiffness resulting from cracking of the concrete should be considered in methods for predicting the maximum strains and deformations of the containment. Thus, concrete cracking needs to be considered depending on the stress levels caused by the most severe seismic load combination. Provide technical justification, if cracking is not considered or is determined to be insignificant. Sections 3.1.3 and C 3.1.3 of ASCE 4-98 provide additional guidance for modeling the stiffness of concrete elements. The staff reviews the methods used for considering creep, shrinkage, and concrete cracking, or the					

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	justification for not considering these effects, on a case-by-case basis. E. Dynamic Soil Pressure. Consideration of dynamic lateral soil pressures on embedded walls of a concrete containment (if applicable) is acceptable if the lateral earth pressure loads are evaluated for two cases. These are (1) lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with ASCE 4-98 Section 3.5.3.2 and (2) lateral earth pressure equal to the passive earth pressure. If the above methods are shown to be overly conservative for the cases considered, then any alternative methods proposed will be reviewed on a case-by-case basis. F. Computer Programs. The computer programs used in the design and analysis should be described and validated by any of the following procedures or criteria: i. The computer program is recognized in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration. ii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained by a similar and independently written and recognized program in the public domain. The test problems should be demonstrated to be similar to or within the range of applicability of the problems analyzed by the public domain computer program.					

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	iii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained from classical solutions or from accepted experimental tests or to analytical results published in technical literature. The test problems should be demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program. A summary comparison should be provided for the results obtained in the validation of each computer program. G. Tangential Shear. Design and analysis procedures for tangential shear are acceptable if in accordance with those contained in Article CC-3000 of the ASME Code. The regulatory staff should note the exceptions taken to the provisions of this article, as contained in Subsection II.5 of this SRP section. H. Variation in Physical Material Properties. For the analysis of the effects of possible variations in the physical properties of materials on the analytical results, the upper and lower bounds of these properties should be used, wherever critical. The physical properties that may be critical include the soil modulus, modulus of elasticity, and Poisson's ratio of concrete. I. Thickened Penetrations. The effect of the large, thickened penetration regions on the overall behavior of the containment may be treated by the same method used for					

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	localized loads as discussed in Subsection II.4.C. J. <u>Steel Liner Plate and Anchors.</u> For the design and analysis of the liner plate and its anchorage system, the procedures furnished are found adequate and acceptable if in accordance with the provisions of Subarticle CC-3600 of the ASME Code. In general, the liner plate analysis should consider deviations in geometry resulting from fabrication and erection tolerances and variations of the assumed physical properties of the liner and anchor material. Since the liner plate is usually anchored at relatively closely spaced intervals, the analysis procedures are acceptable if based on either the classical plate or beam theory. Since the concrete shell is much stiffer than the liner plate, the strains in the liner will essentially follow those in the concrete. The strains in the concrete under the various load combinations as obtainable from the analysis of the shell are thus imposed on the liner plate, and the resulting strains and stresses in the liner and its anchors should be lower than the allowable limits defined in Tables CC-3720-1 and CC-3730-1 of the ASME Code. K. <u>Ultimate Capacity of Concrete Containment.</u> Regulatory criteria require a determination of the internal pressure capacity for containment structures, as a measure of the safety margin above the design-basis accident pressure. i. <u>Reinforced Concrete Containments</u> One acceptable methodology for cylindrical reinforced concrete containments is to estimate the capacity based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1 percent. The specific location of interest is the steel					

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	reinforcement in the hoop direction, closest to the inside surface of the concrete. The inside radius of the concrete wall should be used in calculating the strain in the hoop reinforcing steel. To conduct the necessary analysis, both nonlinear material behavior and nonlinear geometric behavior must be considered for the reinforcing steel. The stress-strain curve for the reinforcing steel should be based on the code-specified minimum yield strength and a stress-strain relationship above yield that is representative of the specific grade of reinforcing steel. The stress-strain curve must be developed for the design-basis accident temperature. The use of an alternate failure criteria for the analyses of noncylindrical containments and cylindrical containments are reviewed on a case by case basis. Guidance on computer modeling of reinforced concrete containments for internal pressure capacity calculations appears in NUREG/CR-6906. NOTE: In applying the analysis methodology to existing containment structures, it is permissible to use as-built material properties for the reinforcing steel and concrete. Sufficient data must be available to establish with reasonable confidence a lower bound, a median, and an upper bound value for the important material parameters. These values must be adjusted for the design-basis accident temperature. For deterministic assessments, the lower-bound values should be used. For probabilistic risk assessment, calculations of					

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	failure probability versus pressure should consider the statistical distribution of the material properties. ii. <u>Prestressed Concrete Containments</u> One acceptable methodology for cylindrical prestressed concrete containments is to estimate the capacity based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 0.8 percent. This strain limit is applicable to all materials which contribute to resisting the internal pressure (i.e., tendons, rebars, and liner (if considered)). When calculating the pressure capacity contribution from the tendons, the above-specified strain limit is applicable to the full range of strain (from 0.0 psi at 0.0-percent strain up to the tendon contribution to pressure capacity at 0.8-percent strain). The other items described previously for reinforced concrete containment, after the first paragraph identifying global strain limits, are also applicable to the approach used for prestressed concrete containments. The criteria presented for consideration of nonlinear material behavior of the reinforcing steel also apply to the tendons. iii. <u>Containment Penetrations</u> The methodologies described above apply to the containment structure. A complete evaluation of the internal pressure capacity must also address major containment penetrations, such as the removable drywell head and ventlines for BWR designs,					

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	equipment hatches, personnel airlocks, and major piping penetrations. The analysis should also address other potential containment leak paths through mechanical and electrical penetrations. iv. Special Considerations for Steel Elliptical and Torispherical Heads: Under internal pressure, a potential failure mode of steel ellipsoidal and torispherical heads is buckling, resulting from a hoop compression zone in the knuckle region. The analysis needs to evaluate this potential mode of failure to determine if it is the limiting condition for the pressure capacity of the containment. The analysis should consider nonlinear material and geometric behavior and address the effect of initial geometric imperfections either explicitly (direct modeling) or implicitly (through the use of appropriate imperfection sensitivity knockdown factors). If appropriately demonstrated, residual postbuckling strength can be considered in determining the pressure capacity. The details of the analysis and the results should be submitted in report form with the following identifiable information: (1) The original design pressure, P_a, as defined in the ASME Code (2) Calculated static pressure capacity (3) Equivalent static pressure					

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	response calculated from dynamic pressure (4) The associated failure mode (5) The stress-strain relation of the liner steel and reinforcing and/or prestressing steel and the behavior of the liner under the postulated loading conditions in relation to that of the reinforcing and/or prestressing steel (6) The criteria governing the original design and the criteria used to establish failure (7) Analysis details and general results (8) Appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure L. Structural Audit. Appendix B to SRP Section 3.8.4 describes the conduct of a structural audit. M. Design Report. The design report is considered acceptable when it satisfies the guidelines of Appendix C to SRP Section 3.8.4.					
3.8.1.5	Structural Acceptance Criteria					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A. For the structural portions of the containment, the specified allowable limits for stresses and strains are acceptable if they are in accordance with Subsection CC-3400 of the ASME Code and RG 1.136 (see Subsection II.3 of this SRP section), with the following exceptions: <u>CC-3421.5</u> For existing (older vintage) plants where a portion of the tangential shear stress, v_c, was permitted to be carried by the concrete, v_c is limited to 276 kPa or 40 pounds per square inch (psi) and 414 kPa (60 psi) for the load combinations of Table CC-3230-1, representing abnormal/severe environmental and abnormal/extreme environmental conditions, respectively. The criteria for design of steel reinforcement to resist the excess shear load above v_c should meet the provisions of the code of record for the containment design. For other plants, the concrete should carry no tangential shear stress as indicated in Subsection CC-3421.5 of the ASME Code. The tangential shear strength provided by orthogonal reinforcement should be limited to the following: $0.833\sqrt{f^{1}c}\ (MPa); (psi)]$ where the value of *f'*, is in units of MPa and psi in the first and second expression, respectively, in accordance with the ASME Code.					

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	For prestressed concrete containments, the principal tensile stress should not exceed the following: $\frac{1}{3}\sqrt{f^1c}\ (MPa)\,[4\sqrt{f^1c}\ 9\,psi)]$ where the value of f^1c, is in units of MPa and psi in the first and second expression, respectively, in accordance with the ASME Code. B. For the liner plate and its anchorage system, the specified limits for stresses and strains are acceptable if in accordance with Tables CC-3720-1 and CC-3730-1 of the ASME Code, respectively.					
3.8.1.6	<u>Materials, Quality Control, and Special Construction Techniques</u> A. The specified materials of construction are acceptable if found to be in accordance with Article CC-2000 of the ASME Code with additional guidance provided by RG 1.107 and 1.136. B. Quality control programs are acceptable if found to be in accordance with applicable portions of Articles CC-4000 and CC-5000 of the ASME Code with additional guidance provided by RG 1.136 for quality assurance requirements. C. Special construction techniques, if any, are reviewed on a case-by-case basis.					
3.8.1.7	<u>Testing and Inservice Surveillance Requirements</u>					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A. Procedures for the postconstruction, preoperational structural proof test proposed for the containment are acceptable if found in accordance with those delineated in Article CC-6000 of the ASME Code. B. For reinforced and prestressed concrete containments, 10 CFR 50.55a imposes the examination requirements of Section XI, Subsections IWL and IWE, of the ASME Code. These subsections provide preservice examination, inservice inspection, and repair/replacement requirements, and acceptance criteria. The scope of Subsection IWL includes the concrete and unbonded posttensioning systems. Subsection IWE covers examination requirements for steel liners of concrete containments and their integral attachments; metallic shell portions of containment (e.g., steel head); containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The regulations in 10 CFR 50.55a(b)(2) specify the acceptable edition of the ASME Code and additional requirements beyond those contained in these subsections of the ASME Code. 10 CFR 55a (b)(2)(viii)(E) requires that licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. C. For concrete containments, it is important to accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of containments is essential for plant safety. The staff reviews on a case-by-case basis any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in					

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	inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containments. For plants with nonaggressive ground water/soil (i.e., pH > 5.5, chlorides < 500 ppm, sulfates <1500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and foundations is to (1) examine the exposed portions of below-grade concrete for signs of degradation, when excavated for any reason; and (2) conduct periodic site monitoring of ground water chemistry, to confirm that the ground water remains nonaggressive. For plants with aggressive ground water/soil (i.e., exceeding any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible, below-grade concrete for signs of degradation. D. For prestressed concrete containments, inservice surveillance requirements for the tendons, as presented in the technical specifications of the operating license, are acceptable if in accordance with Section XI, Subsection IWL of the ASME Code; 10 CFR 50.55a; and RG 1.35 and 1.35.1 for ungrouted tendons and 1.90 for grouted tendons, respectively. E. SRP Section 6.2.6 presents the preoperational and inservice integrated leak-rate testing criteria. F. For new and unique containment designs (e.g., incorporating integrally connected passive systems with					

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	pools), the preoperational tests and inspections of containment discussed above need to consider items included in these unique features.					
3.8.2, Rev. 2 (03/2007)	Steel Containment					
3.8.2.1	Description of the Containment. The descriptive information in the safety analysis report (SAR) is acceptable if it meets the criteria set forth in Section 3.8.2.1 of RG 1.206. If the steel containment has new or unique features that RG 1.206 does not specifically cover, adequate information necessary to accomplish a meaningful review of the structural aspects of these new or unique features need to be presented such that an evaluation can be made that it is equivalent in function and complies with the applicable requirements. RG 1.70 provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).					
3.8.2.2	Applicable Codes, Standards, and Specifications. Codes, standards, and specifications, acceptable either in their entirety or in part, cover the design, materials, fabrication, erection, inspection, testing, and inservice surveillance of steel containments. The following codes and guides are acceptable: Code/Guide — Title ASME Code — Section III, Division 1, Subsection NE, "Class MC Components" ASME Code — Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Water Cooled Plants” RG 1.7 — Control of Combustible Gas Concentrations in Containment RG 1.57 — Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components					
3.8.2.3	Loads and Loading Combinations. Currently, ASME Code, Section III, Division 1, Subsection NE, and RG 1.57 do not explicitly state the loads and load combinations that should be considered in the design of steel containments. The staff has issued as a proposed revision to RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.” This draft guide or subsequent revision to RG 1.57 provides additional guidance for design requirements, including load and load combinations, which should be considered in the design of steel containments. The specified loads and load combinations are acceptable if found to be in accordance with the following: A. Loads D — Dead loads L — Live loads, including all loads resulting from platform flexibility and deformation and from crane loading, if applicable P_t — Test pressure T_t — Test temperature					

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	T_o — Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition R_o — Pipe reactions during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition P_o — External pressure loads resulting from pressure variation either inside or outside containment E — Loads generated by the OBE, including sloshing effects, if applicable E' — Loads generated by the SSE, including sloshing effects, if applicable P_a — Pressure load generated by the postulated pipe break accident (including pressure generated by postulated small-break or intermediate-break pipe ruptures), pool swell, and subsequent hydrodynamic loads Note: For loading combinations B, Service Conditions(iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated. T_a — Thermal loads under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads					

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	Note: For loading combinations B, Service Conditions(iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated. R_a — Pipe reactions under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads Note: For loading combinations B, Service Conditions(iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated. P_s — All pressure loads that are caused by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic loads, if applicable T_s — All thermal loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic thermal loads, if applicable R_s — All pipe reaction loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic reaction loads, if applicable Y_r — Equivalent static load on the structure generated by the reaction on the broken pipe					

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	during the design-basis accident Y_j — Jet impingement equivalent static load on the structure generated by the broken pipe during the design-basis accident Y_m — Missile impact equivalent static load on the structure generated by or during the design-basis accident, such as pipe whipping F_L — Load generated by the post-LOCA flooding of the containment, if applicable P_{g1} — Pressure load generated from 100-percent fuel clad metal-water reaction P_{g2} — Pressure loads generated by hydrogen burning, if applicable P_{g3} — Pressure load from postaccident inerting, assuming carbon dioxide is the inerting agent, if applicable B. Loading Combinations The loading combinations for which the containment might be designed or subjected to during the expected life of the plant include the following: i. Testing Condition This includes the testing condition of the containment to verify its leak integrity. The loading combination in this case includes—					

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	$D + L + T_t + Pt$ ii. Design Conditions These include all design loadings for which the containment vessel or portions thereof might be designed during the expected life of the plant. Such loads include design pressure, design temperature, and the design mechanical loads generated by the design-basis LOCA. The loading combination in this case includes— $D + L + P_a + T_a + R_a$ iii. Service Conditions The load combinations in these cases correspond to and include Level A service limits, Level B service limits, Level C service limits, Level D service limits, and the postflooding condition. The loads may be combined by their actual time history of occurrence taking into consideration their dynamic effect upon the structure. (1) Level A Service Limits These service limits are applicable to the service loadings to which the containment is subjected, including the plant or system design-basis accident conditions for which the containment function is required, except only those categorized as Level B, Level C, Level D, or testing loadings. The loading combinations corresponding to these limits include the					

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	following: (a) Normal operating plant condition $D + L + T_o + R_o + P_o$ (b) Operating plant condition in conjunction with the actuation of multiple SRVs $D + L + T_s + R_s + P_s$ (c) Design-basis LOCA $D + L + T_a + R_a + P_a$ (d) Multiple SRV actuations in combination with small- or intermediate-break accident $D + L + T_a + R_a + P_a + T_s + R_s + P_s$ (e) Normal operating plant conditions in combination with inadvertent full actuation of a postaccident inerting hydrogen control system (10 CFR 50.34(f)(3)(v)(B)(1)) $D + L + T_o + R_o + P_o + Pg3$ (f) Pressure test load to ensure that the containment will safely withstand the pressure calculated to result from					

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	carbon-dioxide inerting (10 CFR 50.34(f)(3)(v)(B)(2)) D + 1.10 x Pg3 (2) Level B Service Limits These service limits include the loads subject to Level A service limits plus the additional loads resulting from natural phenomena during which the plant must remain operational. The loading combinations corresponding to these limits include the following: (a) Design-basis LOCA in combination with OBE (if E s one- third E', only its contribution to cyclic loading needs to be considered) $D + L + T_a + R_a + P_a + E$ (b) Operating plant condition in combination with OBE (if E s one-third E', only its contribution to cyclic loading needs to be considered) $D + L + T_o + R_o + P_o + E$ (c) Operating plant condition in combination with OBE and multiple SRV actuations (if E s one-third E', only its contribution to cyclic loading needs to be considered)					

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	$D + L + T_s + R_s + P_s + E$ (d) Design-basis LOCA in combination with a single active component failure causing one SRV discharge $D + L + T_a + P_a + R_a + T_s + R_s + Ps$ (3) Level C Service Limits These service limits include the loads subject to Level A service limits plus the additional loads resulting from natural phenomena for which safe shutdown of the plant is required. The loading combinations corresponding to these limits include the following: (a) Design-basis LOCA in combination with SSE $D + L + T_a + R_a + P_a + E'$ (b) Operating plant condition in combination with SSE $D + L + T_o + R_o + P_o + E'$ (c) Multiple SRV actuations in combination with small- or intermediate-break accident and SSE $D + L + T_a + R_a + P_a + T_s + R_s + P_s + E'$					

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	(d) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction accompanied by hydrogen burning (10 CFR 50.34(f)(3)(v)(A)(1), 10 CFR 50.44) D + Pg1 + Pg2 Note: In this load combination, Pg1 + Pg2 should not be less than 310 kilo Pascals (kPa) or 45 pounds per square in gauge (psig). (e) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction accompanied by the added pressure from postaccident inerting, assuming carbon dioxide as the inerting agent (10 CFR 50.34(f)(3)(v)(A)(1)) D + Pg1 + P_{g3} Note: In this load combination, P_{g1} + P_{g3} should not be less than 310 kPa or (45 psig). (4) Level D Service Limits These service limits include other applicable service limits and loadings of a local dynamic nature for which the containment function is required. The load combinations corresponding to					

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	these limits include the following: (a) Design-basis LOCA in combination with SSE and local dynamic loadings $D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$ (b) Multiple SRV actuations in combination with small- or intermediate-break accident, SSE, and local dynamic loadings $D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + P_s + T_s + R_s + E'$ (5) Postflooding Condition This includes the post-LOCA flooding of the containment in combination with OBE-basis earthquake $D + L + F_L + E$ C. Construction Loads Temporary construction loads and the effects of environmental loads during the construction stage need to be considered. ASME Code, Section III, Subsection NE, does not address this. The sections of Structural Engineering Institute/American Society of Civil Engineers (SEI/ASCE) Standard 37-02 pertaining to steel structures may be used for guidance.					

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	D. External Environmental Loads A concrete shield building typically protects steel containments from the environment. If environmental loads external to the steel containment (e.g., wind, tornado, external flooding) either directly or indirectly impose loads on the steel containment, the design of the steel containment also needs to consider these loads. Load combinations and acceptance criteria that are consistent with those specified in SRP Section 3.8.1 for concrete containments should be used. As noted in 10 CFR 50, Appendix S, the OBE is only associated with plant shutdown and inspection, unless specifically selected by the applicant as a design input. If the OBE is set at one-third or less of the SSE ground motion, explicit analysis is not required. The only exceptions are the postflooding condition and cyclic loading considerations. The staff requirements memorandum for SECY-93-087 provides guidance on the treatment of cyclic loading for the OBE. If the OBE is set at a value greater than one-third of the SSE, explicit analysis must be performed to demonstrate that the applicable load combinations meet the Service Level B stress, strain, deformation, and fatigue limits.					
3.8.2.4	Design and Analysis Procedures. Article NE-3000 of ASME Code, Section III, Division 1, Subsection NE, covers design and analysis procedures for steel containments. The procedures given in the ASME Code, with additional guidance provided in the applicable provisions of RG 1.57, constitute an acceptable basis for design and analysis. Moreover, for the specific areas of review described in Subsection					

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	I.4 of this SRP section, the following criteria are acceptable: A. Treatment of Nonaxisymmetric and Localized Loads For most containments, the nonaxisymmetric loads that apply are the horizontal seismic and associated sloshing loads, pool swell, and its related hydrodynamic loads caused either by LOCA or by SRV actuation. Other possible nonaxisymmetric and localized loads are those induced by pipe rupture, such as reactions, jet impingement forces, and missiles. For the PWR ice-condenser containment, the design-basis accident may result in a nonaxisymmetric pressure load caused by compartmentation of the containment interior. For such localized loads, the analyses should include a determination of the local effects of the loads. These effects should then be superimposed on the overall effects. For the overall effects of nonaxisymmetric loads on shells of revolution, an acceptable general procedure is to expand the load by a Fourier series. Any other applicable methods proposed for a large thin shell, will be reviewed on a case-by-case basis. B. Treatment of Buckling Effects Earthquake loads and localized pressure loads (such as those encountered in PWR ice-condenser containments) require consideration of shell buckling. An acceptable approach to the problem is to perform a nonlinear dynamic analysis. If a static analysis is performed, an appropriate dynamic load factor should be used to obtain the effective static load. Subarticle NE-3133 of ASME Code, Section III, Division 1, Subsection NE, is acceptable to address buckling of shell					

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	geometries and loadings covered therein. Buckling of shells with more complex geometries or loading conditions than those covered by Subarticle NE-3133 may be considered in accordance with the criteria described in ASME Code Case N-284, Revision 1, with additional guidance provided in RG 1.193. Each application of ASME Code Case N-284, Revision 1, is subject to review on a case by case basis. Buckling of shells under internal pressure (e.g., torispherical heads) may also be considered in accordance with the criteria described in ASME Code Case N-284, Revision 1, with guidance provided in RG 1.193. Each application of ASME Code Case N-284, Revision 1, is subject to review on a case by case basis. The staff will review the use of alternate methodologies to address the buckling of steel containments on a case-by-case basis. RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" and RG 1.193, " Code cases not approved for Use," provide additional guidance for code case acceptability which should be considered in the design of steel containments. Any Code cases not currently approved by NRC requires review on a case by case basis. C. Computer Programs The computer programs used in the design and analysis should be described and validated by procedures or criteria described in Subsection II.4.e of SRP Section 3.8.1.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	D. Ultimate Capacity of Steel Containment For new reactors, regulatory criteria require a determination of the internal pressure capacity for containment structures, as a measure of the safety margin above the design-basis accident pressure. One methodology acceptable to the staff for cylindrical steel containments is to estimate the capacity based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1.5 percent. To conduct the necessary analysis, both nonlinear material behavior and nonlinear geometric behavior must be considered. The stress-strain curve for the steel containment material should be based on the code-specified minimum yield strength and a stress-strain relationship above yield that is representative of that specific grade of steel. The stress-strain curve must be developed for the design-basis accident temperature. Analyses of noncylindrical containments and analyses of cylindrical containments that use alternate failure criteria will be subject to detailed staff review, on a case-by-case basis. The NRC has published guidance on computer modeling of steel containments for internal pressure capacity calculations in NUREG/CR-6906. Note: In applying the analysis methodology to existing					

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	containment structures, it is permissible to use as-built material properties for the steel containment material. Sufficient material certification data must be available to establish with reasonable confidence a lower bound, a median, and an upper bound value for the important material parameters. These values must be adjusted for the design-basis accident temperature. For deterministic assessments, the lower bound values should be used. For probabilistic risk assessment, calculations of failure probability vs. pressure should consider the statistical distribution of the material properties. <u>Containment Penetrations:</u> The methodology described above applies to the containment structure. A complete evaluation of the internal pressure capacity must also address major containment penetrations, such as the removable drywell head and ventlines for BWR designs, equipment hatches, personnel airlocks, and major piping penetrations. Other potential containment leak paths through mechanical and electrical penetrations should also be addressed. <u>Special Considerations for Steel Ellipsoidal and Torispherical Heads:</u> Under internal pressure, a potential failure mode of steel ellipsoidal and torispherical heads is buckling, resulting from a hoop compression zone in the knuckle region. This potential mode of failure needs to be evaluated, to determine if it is the limiting condition for the pressure capacity of the containment. The analysis should consider nonlinear material and geometric behavior and address the effect of initial geometric imperfections either explicitly (direct modeling) or implicitly (through the use of appropriate imperfection sensitivity knockdown factors). If appropriately demonstrated, residual postbuckling strength					

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	can be considered in determining the pressure capacity. The details of the analysis and the results should be submitted in a report form with the following identifiable information: i. Original design pressure, P, as defined in ASME Code, Section III, Division 1, Subsection NE, Subarticle NE-3112.1 ii. Calculated static pressure capacity iii. Equivalent static pressure response calculated from dynamic pressure iv. Associated failure mode v. Criteria governing the original design and the criteria used to establish failure vi. Analysis details and general results vii. Appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure E. Structural Audit Structural audits are conducted as described in SRP Section 3.8.4, Appendix B. F. Design Report					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The design report is considered acceptable when it satisfies the guidelines provided in SRP Section 3.8.4, Appendix C.					
3.8.2.5	Structural Acceptance Criteria. Stresses at various locations of the shell of the containment for various design loads are determined by analysis. Total stresses for the combination of loads delineated in Subsection II.3 of this SRP section are acceptable if found to be within the limits defined by ASME Code, Section III, Division 1, Subsection NE, Subarticles NE-3221.1, NE-3221.2, NE-3221.3, and NE-3221.4 for Service Levels A, B, C, and D, respectively. For the postflooding load combination (Subsection II.3.b(iii)(5)), Service Level C limits apply to primary stress, and Service Level B limits apply to primary plus secondary stress. Evaluation of primary plus secondary plus peak stress is not required. If external environmental loads need to be considered in the steel containment design, the staff will review the adequacy of the approach and acceptance criteria on a case-by-case basis.					
3.8.2.6	Materials, Quality Control, and Special Construction Techniques A. The materials of construction are acceptable if in accordance with Article NE-2000 of ASME Code, Section III, Division 1, Subsection NE. The organization responsible to review material properties will review corrosion protection. B. Quality control programs are acceptable if in accordance with Articles NE-2000, NE-4000, and NE-5000 of ASME Code, Section III, Division 1, Subsection NE.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	C. The acceptability of special construction techniques, if any, are evaluated on a case-by-case basis. D. The staff will review the consideration of temporary construction loads and the effects of environmental loads during the construction stage on a case-by-case basis.					
3.8.2.7	Testing and Inservice Surveillance Requirements A. Procedures for the preoperational structural proof test are acceptable if the procedures are in accordance with Article NE-6000 of ASME Code, Section III, Division 1, Subsection NE. B. For steel containments, 10 CFR 50.55a requires examination be conducted as outlined in ASME Code Section XI, Subsection IWE. Subsection IWE provides preservice examination, inservice inspection, and repair/replacement requirements and corresponding acceptance criteria. The scope of Subsection IWE includes the steel containment shell; integral attachments; containment hatches and airlocks; seals, gaskets, and moisture barriers; and pressure-retaining bolting. 10 CFR 50.55a(b)(2) specifies the acceptable edition of the ASME Code and additional requirements beyond those contained in Subsection IWE. 10 CFR 55a (b)(2)(viii)(E) requires that licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. C. The staff will review any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of the steel containment on a case-by-case basis.					
3.8.3, Rev. 2 (03/2007)	Concrete and Steel Internal Structures of Steel or Concrete Containments					
3.8.3.1	<u>Description of the Internal Structures.</u> The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the criteria set forth in Section 3.8.3.1 of RG 1.70 or 1.206. During the application acceptance review, the reviewer identifies deficient areas of descriptive information and initiates a request for additional information. New or unique design features that are not specifically covered in RG 1.70 or RG 1.206 may require a more detailed review. The reviewer determines whether additional information is required to accomplish a meaningful review of the structural aspects of such new or unique features. RG 1.206 provides the basis for evaluating the description of structures to be included in a DC or a COL application. RG 1.70 provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).					
3.8.3.2	<u>Applicable Codes, Standards, and Specifications.</u> The design, materials, fabrication, erection, inspection, testing, and inservice surveillance, if any, of containment internal structures are covered by codes, standards, and guides that are applicable either in their entirety or in part. The following codes and guides are acceptable: <u>Code, Standard, or Specification</u> <u>Title</u>					

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ID	Title/Requirement		Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	AC I 349	Code Requirements for Nuclear Safety-Related Concrete Structures (supplemented with additional guidance by RG 1.142 and 1.199)					
	ASME Code	Section III, Division 2, Subsection CC, "Code for Concrete Reactor Vessels and Containments"					
	ASME Code	Section III, Division 1, Subsection NE, "Class MC Components"					
	ANSI/AISC N690-1994 including Supplement 2 (2004) Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities						
	Regulatory Guides						
	1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment					
	1.69	Concrete Radiation Shields for Nuclear Power Plants					
	1.136	Materials, Construction, and Testing of Concrete					

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ID	Title/Requirement		Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
		Containments					
	1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)					
	1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in LWR Plants					
	1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants					
	1.199	Anchoring Components and Structural Supports					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	in Concrete					
3.8.3.3	Loads and Load Combinations. The loads and load combinations for containment internal structures described in Subsection I.1 of this SRP are acceptable if they are consistent with the guidance given below. The loads and load combinations for the divider-barrier and ice-condenser elements of the ice-condenser PWR containment and the drywell of the BWR containment are presented following the general criteria given for concrete and steel structures. A. Concrete Structures All loads and load combinations are to be in accordance with ACI 349 and RG 1.142. Supplemental criteria on the use of loads and load combinations are presented below. Dead loads include hydrostatic loads, and, for equipment supports, they include static and dynamic head and fluid flow effects. Live loads include any movable equipment loads and other loads that vary with intensity and occurrence. For equipment supports, they also include loads caused by vibration and any support movement effects. Alternate load cases in which the magnitudes and locations of the live loads are arranged so that worst-case conditions are included in the design should be investigated, as appropriate. As per 10 CFR 50, Appendix S, the OBE is only associated with plant shutdown and inspection unless the applicant specifically selects it as a design input. If the OBE is set at one-third or less of the SSE ground motion, an explicit response or design analysis is not required. If the OBE is set at a value greater than one- third of the SSE, an analysis and design must be performed to demonstrate that the containment internal structures					

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	remain functional and are within applicable stress, strain, and deformation limits. SRP Section 3.7.1 and 3.7.2 provides further guidance on the use of OBE. For structures or structural components subjected to hydrodynamic loads resulting from LOCA and/or SRV actuation, such loads should be considered as indicated in the appendix to SRP Section 3.8.1. Fluid structure interaction associated with these hydrodynamic loads and those from earthquakes should be taken into account. The design of concrete structures must consider the loads and load combinations that may occur during their construction. These loads consist of dead loads, live loads, temperature, wind, snow, rain, and ice. Applicable construction loads include material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. Structural Engineering Institute (SEI)/ASCE Standard 37 provides additional guidance on construction loads. This standard may be used for supplemental guidance. When the standard and the Code/SRP provide conflicting criteria, the criteria provided in Code/SRP governs. B. Steel Structures All loads and load combinations are to be in accordance with ANSI/AISC N690-1994 including Supplement 2 (2004). This specification uses the allowable stress design (ASD) method. Use of the load and resistance factor design (LRFD) version of the specification (N690L) is reviewed on a case-by-case basis. The supplemental criteria on the use of loads and load combinations presented above for concrete structures also apply to steel					

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	structures. C. Divider Barrier and Ice-Condenser of the PWR Ice-Condenser Containment Specific load and load combination criteria applicable to the divider barrier and ice-condenser elements are given below. Supplemental criteria presented in Subsection II.3.A of this SRP section are also applicable. i. Divider Barrier Because the structural integrity of the divider barrier and, to a certain extent, its leaktight integrity are important to the proper functioning of the ice-condenser containment system, it is treated, for design purposes, in a manner similar to the containment itself. Accordingly, for concrete pressure-resisting portions of the divider barrier, the loads and load combinations of Article CC-3000 of the ASME Code, Section III, Division 2, with additional guidance provided by applicable portions of SRP Section 3.8.1 and RG 1.136. For other concrete portions of the divider barrier, the loads and load combinations as defined in Subsection II.3.A apply. Steel portions of the divider barrier that resist the design differential pressure and are not backed by concrete, such as penetrations, hatches, locks, and guard pipes, should be designed in accordance with the appropriate sections of Subsection NE of the ASME Code, Section III, Division 1, with additional criteria provided by applicable portions of SRP Section 3.8.2 and RG 1.57 apply.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	For other steel portions of the divider barrier, the loads and load combinations as defined in Subsection II.3.B apply. ii. Ice-Condenser Elements The structural integrity of the ice baskets, ice-bed framing, and their supports is important to the functional integrity of the ice-condenser containment system. Loads and load combinations for the ice-condenser elements are acceptable if found to be in accordance with ANSI/AISC N690-1994 including Supplement 2 (2004). For the ice-condenser, the load P_a is the LOCA pressure load induced by drag and change in the momentum of flowing air and steam. D. BWR Containment Drywell This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner. Because the structural integrity of the drywell and, to a certain extent, its leaktight integrity are critically important to the proper functioning of the pressure-suppression system, the drywell is treated, for design and testing purposes only, in a manner similar to the containment itself. Accordingly, for the concrete pressure-resisting portions of the drywell, the loads and loading combinations of Article CC-3000 of ASME Code, Section III, Division 2, will apply, with additional criteria provided by applicable portions of SRP Section 3.8.1 and RG 1.136. For steel components of the drywell that resist pressure and are not backed by concrete, the appropriate sections of					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Subsection NE of ASME Code, Section III, Division 1, should be used with additional guidance provided by applicable portions of SRP Section 3.8.2 and RG 1.57. Specifically, the loads and load combinations of Subsection II.3 of SRP Section 3.8.2 apply. Additional criteria presented in Subsection II.3.A of this SRP section are also applicable to the BWR containment drywell. For the lower vent portion of the drywell, the following conditions apply: i. If the main reinforcement of the drywell is carried down between the vent holes, and the reinforced concrete section is relied upon for structural purposes, the criteria that apply to concrete portions of the drywell as described above will apply. ii. If the main reinforcement of the drywell is terminated above the vent holes, and two steel plates lining both faces of the drywell are used for structural purposes, the criteria that apply to steel portions of the drywell as described above will apply. iii. If other structural systems are used in the vent region, the loads and load combinations are reviewed and judged on a case-by-case basis.					
3.8.3.4	Design and Analysis Procedures. The design and analysis procedures used for the containment internal structures are acceptable if found to be in accordance with the following: A. PWR Dry Containment Internal Structures i. Primary Shield Wall and Reactor Cavity					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The design and analysis procedures used for the shield wall are acceptable if found to be in accordance with ACI 349 with additional guidance provided by RG 1.142. This code is based on the strength design method. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found to be in accordance with ACI 349, Appendix B, with additional guidance provided by RG 1.199. Analyses for LOCA loads applicable to the primary shield wall, such as the cavity differential pressure combined with pipe rupture reaction forces, are acceptable if these loads are treated as dynamic time-dependent loads. This requires that either a detailed time-history analysis be performed or a static analysis using the peak of the forcing function amplified by an appropriate chosen dynamic factor be employed. Elastic behavior of the wall should be maintained under the differential pressure. However, for the concentrated accident loads, such as Y_r, Y_j, or Y_m, elasto-plastic behavior may be assumed if the deflections are limited to maintain functional requirements. Simplified methods for determining effective dynamic load factors for elastic behavior are acceptable if found to be in accordance with recognized dynamic analysis methods. ii. Secondary Shield Walls Design and analysis procedures used for the secondary shield walls are acceptable if found to be in accordance with conventional beam/slab design and analysis procedures described in ACI 349, with additional guidance provided RG 1.142. The design and analysis of anchors (steel embedments) used for					

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	component and structural supports on concrete structures are acceptable if found to be in accordance with ACI 349, Appendix B, with additional guidance provided by RG 1.199. Similar to the primary shield wall, the secondary shield walls are also subject to dynamic LOCA loads and the methods described in Subsection II.4.A.i are, therefore, applicable and acceptable. iii. Other Interior Structures Most of the other interior structures that are reviewed are combinations of reinforced concrete slabs, walls, beams, and columns, and steel beams and columns, which are classified as Category I structures subject to the loads and load combinations described in Subsection II.3 of this SRP section. Analytical techniques for these structures are acceptable if found to be in accordance with those described in ACI 349, and with additional guidance provided by RG 1.142 and 1.199 for concrete and anchors (steel embedments), respectively, and with ANSI/AISC N690-1994 including Supplement 2 (2004) for steel. B. PWR Ice-Condenser Containment Internal Structures i. Divider Barrier The most important loads that usually govern the design of the divider barrier are those induced by a LOCA, including the differential pressure across the barrier and any concentrated jet impingement loads. Because the structural integrity of the divider barrier and, to a certain extent, its leaktight integrity are important to the proper					

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	functioning of the ice-condenser containment system, it is treated, for design purposes, in a manner similar to the containment itself. Accordingly, for concrete pressure-resisting portions of the divider barrier, the design and analysis procedures of Article CC-3000 of the ASME Code, Section III, Division 2, apply with additional guidance provided by applicable portions of SRP Section 3.8.1 and RG 1.136. For the other concrete portions of the divider barrier, the design and analysis procedures are acceptable if found to be in accordance with ACI 349, with additional guidance provided by RG 1.142 and 1.199. These methods are based on linear elastic design methods unless the structure is subjected to concentrated accident loads, as discussed in Subsection II.4.A.i, in which elasto-plastic behavior may be assumed. For steel portions of the divider barrier that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found to be in accordance with the applicable provisions of Subsection NE of the ASME Code, Section III, Division 1 apply, with additional guidance provided by applicable portions of SRP Section 3.8.2 and RG 1.57. ii. Ice-Condenser Elements The design and analysis procedures for the ice-condenser and its various components are acceptable if found to be in accordance with either the elastic/linear design method of Part 1 of ANSI/AISC N690-1994 including Supplement 2 (2004), or the					

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	plastic design method of Part 2 of the same specifications. For components using experimental testing to verify the design, the testing procedures are acceptable if found to be in accordance with recognized prototype or model testing procedures that consider the effect of scaling and similitude. C. BWR Containment Internal Structures This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner. i. Drywell The design and analysis procedures used for concrete portions of the drywell are acceptable if found to be in accordance with Subsection II.4 of SRP Section 3.8.1. For steel portions of the drywell that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found to be in accordance with the applicable provisions of SRP Section 3.8.2, Subsection II.4. ii. Weir Wall One of the major loads to which the weir wall may be subjected is a jet impingement load induced by a pipe rupture in a nearby recirculation loop. The deflection of the wall under such a load must be limited so as not to impair the pressure-suppression performance. The procedures used to analyze the wall for such a dynamic time-dependent load are acceptable if a detailed time-history dynamic analysis is performed or if an equivalent static analysis is					

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	performed using the peak of the jet load amplified by an appropriately chosen dynamic load factor. The design and analysis procedures for concrete weir walls are acceptable if found to be in accordance with conventional methods described in ACI 349, with additional guidance provided by RG 1.142 and 1.199, for concrete and anchors (steel embedments), respectively. iii. Refueling Pool and Operating Floor The refueling pool and the operating floor, which may be supported on the walls of the refueling pool on one side and on the containment shell on the other side, are constructed of a combination of reinforced concrete and structural steel. The design and analysis procedures are acceptable if found to be in accordance with conventional methods described in ACI 349, with additional guidance provided by RG 1.142 and 1.199, for concrete and anchors (steel embedments), respectively, and in ANSI/AISC N690-1994 including Supplement 2 (2004) for structural steel. iv. Supports for Reactor The support system for the reactor vessel, described in Subsection I of this SRP section, should be designed to resist various combinations of loadings as indicated in Subsection II.3 of this SRP section. Among the major loads that should be considered are normal operating loads, seismic loads, and LOCA loads. The design and analysis procedures used for the reactor supports (beyond the jurisdictional boundary					

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	of the ASME-designed supports) are acceptable if found to be in accordance with the same criteria for concrete and steel that apply to the refueling pool and operating floor. v. Reactor Pedestal The reactor pedestal, which supports the reactor and must withstand the loads transmitted through the reactor supports, should be subjected to most of the loads described in Subsection II.3 of this SRP section and should be designed for all applicable load combinations. The design and analysis procedures used for the reactor pedestal are acceptable if found to be in accordance with the same criteria for concrete applicable to the refueling pool and operating floor. vi. Reactor Shield Wall This cylindrical wall, which surrounds the reactor and provides biological shielding, should be subjected to most of the loads described in Subsection II.3 of this SRP section. In many cases, the wall is used to anchor most of the pipe restraints placed around the reactor coolant system piping. A pipe rupture in the vicinity of the reactor nozzles may pressurize the space within the wall. The wall may be lined on both faces with steel plates which may constitute the major structural elements relied upon to resist the design loads. Like the reactor pedestal, the biological shield wall is also subjected to dynamic LOCA loads and the same methods are, therefore, applicable and acceptable. The design and analysis procedures used for the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	reactor shield wall are acceptable if found to be in accordance with the same criteria for concrete that apply to the refueling pool and operating floor. If the shield wall is constructed from steel plates filled with unreinforced concrete, then the design and analysis procedures are reviewed on a case-by-case basis. vii. Miscellaneous Platforms Platforms inside the drywell are usually constructed of structural steel and their main structural function is to provide foundations for the pipe restraints inside the drywell. Platforms outside the drywell are usually combinations of steel and concrete. The design and analysis procedures used for miscellaneous platforms are acceptable if found to be in accordance with the same criteria for concrete and steel that apply to the refueling pool and operating floor. Of particular interest are the dynamic loads induced on these floors by pool swell during a LOCA. D. For all containment internal structures, the design and analysis methods described in Subsections II.4 of SRP Sections 3.8.1 and 3.8.2, which are applicable to the containment internal concrete and steel structures, respectively, also need to be considered. These items include assumptions on boundary conditions, axisymmetric and nonaxisymmetric loads, transient and localized loads, shrinkage and cracking of concrete, computer programs, and evaluation of liner plates and anchors. E. Design of structures that use modular construction methods are reviewed on a case-by-case basis. NUREG/CR-6486 provides guidance related to the use of modular construction methods. Appendix B to					

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	NUREG/CR-6486 includes proposed modular construction review criteria. F. A structural design audit is conducted as described in Appendix B to SRP Section 3.8.4. G. The applicant's design report is considered acceptable if it satisfies the guidelines of Appendix C to SRP Section 3.8.4.					
3.8.3.5	Structural Acceptance Criteria. The structural acceptance criteria for containment internal structures described in Subsection I.1 of this SRP section are acceptable if found to be in accordance with the guidance given below. The acceptance criteria for the divider-barrier and ice-condenser elements of the ice-condenser PWR containment and the drywell of the BWR containment are presented following the criteria given for concrete and steel structures. The structural acceptance criteria for structures that use modular construction methods are reviewed on a case-by-case basis. See Section II.4.E of this SRP section for criteria relating to modular construction. A. Concrete Structures ACI 349 and RG 1.142 define the structural acceptance criteria for concrete structures. The structural acceptance criteria for anchors (steel embedments) used for support of systems and components to concrete structures are acceptable if found to be in accordance with Appendix B to ACI 349, with additional guidance provided by RG 1.199. B. Steel Structures ANSI/AISC N690-1994 including Supplement 2 (2004) defines the structural acceptance criteria for steel structures. This specification uses the ASD method. Use of the LRFD version of the specification (N690L) is reviewed on a case-					

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	by-case basis. C. Divider Barrier and Ice-Condenser of PWR Ice-Condenser Containment i. Divider Barrier For concrete pressure-resisting portions of the divider barrier, the specified limits for stresses and strains are acceptable if found to be in accordance with Subsection CC-3400 of ASME Code Section III, Division 2, with additional guidance provided by applicable portions of SRP Section 3.8.1 and RG 1.136. For steel portions of the divider barrier that resist pressure but are not backed by structural concrete, the design should be similar to that of steel containments. Accordingly the stress limits are acceptable if found to be in accordance with Subsection NE of the ASME Code, Section III, Division 1, with additional guidance provided by applicable portions of SRP Section 3.8.2 and RG 1.57. For the other concrete and steel portions of the divider barrier, the specified limits for stresses and strains are acceptable if found to be in accordance with those provided in Subsections II.5.A and B for concrete and steel, respectively. ii. Ice-Condenser Elements For load combination delineated in Subsection II.3 of this SRP section, the specified limits for stresses and strain are acceptable if found to be in accordance with those given in ANSI/AISC N690-1994 including Supplement 2 (2004).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	D. BWR Containment Drywell This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner. For concrete and steel portions of the drywell, the specified limits for stresses and strain are acceptable if found to be in accordance with the acceptance criteria of item II.5.C.i as described for the divider barrier. For the lower vent portion of the drywell, the following conditions apply: i. If the main reinforcement of the drywell is carried down between the vent holes, and the reinforced concrete section is relied upon for structural purposes, the structural acceptance criteria are the same as for item II.5.C.i above for concrete. ii. If the main reinforcement of the drywell is terminated above the vent holes, and two steel plates lining both faces of the wall are used for structural purposes, the acceptance criteria are reviewed on a case-by-case basis. iii. If other structural systems are used in the vent region, the acceptance criteria are also reviewed on a case-by-case basis.					
3.8.3.6	Materials, Quality Control, and Special Construction Techniques. The specified materials of construction and quality control programs are acceptable if found to be in accordance with the public code or standard as indicated in Subsection I.6 of this SRP section.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	Special construction techniques, if any, are treated on a case-by-case basis. For modular construction, the materials, quality control, and special construction techniques are also reviewed on a case-by-case basis. See Section II.4.E of this SRP section for further information.					
3.8.3.7	<u>Testing and Inservice Surveillance Requirements.</u> BWR containment drywells, such as those used for the Mark III containment, should be subjected to a structural proof test. Such a test is acceptable if found to be in accordance with the following: A. The drywell should be subjected to an acceptance test that increases the drywell internal pressure in three or more approximately equal pressure increments ranging from atmospheric pressure to at least the design pressure. The drywell should be depressurized in the same number of increments. Measurements should be recorded at atmospheric pressure and at each pressure level of the pressurization and depressurization cycles. At each level, the pressure should be held constant for at least 1 hour before the deflections and strains are recorded. B. So that the overall deflection pattern can be determined in prototype drywells, radial deflections should be measured at a minimum of three points along each of at least three meridians equally spaced around the drywell, including locations with varying stiffness characteristics. Radial deflections should be measured at the lower vent region, about mid-height, and near the top of the cylindrical design. Measurement points may be relocated, depending on the distribution of stresses and deformations anticipated in each particular design. C. In prototype drywells only, strain measurements sufficient to					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	permit an evaluation of strain distribution should be recorded for at least two opposing meridians at the following locations on the wall: i. At the bottom of the wall ii. At mid-height of the wall These strain measurements should be made at a minimum of three positions within the wall section - one at the center and one each near the inner and outer surfaces. D. In nonprototype drywells, deflection and strain measurements need not be made if strain levels have been correlated with deflection measurements during the acceptance test of a prototype drywell when measured strains and deflections are within the predefined tolerance of their predicted responses. E. Any reliable system of displacement meters, optical devices, strain gauges, or other suitable apparatus may be used for the measurements. F. If the test pressure drops as a result of unexpected conditions to or below the next lower pressure level, the entire test sequence should be repeated. Significant deviations from the previous test should be recorded and evaluated. G. If any significant modifications or repairs are made to the drywell following, and because of, the initial test, the test should be repeated. H. A description of the proposed acceptance test and instrumentation requirements should be included in the preliminary SAR.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	I. The following information should be submitted before the performance of the test: i. The numerical values of the predicted responses of the structure which will be measured ii. The tolerances to be permitted on the predicted responses iii. The bases on which the predicted responses and the tolerances were established J. The following information should be included in the final test report: i. A description of the actual test and instrumentation ii. A comparison of the test measurements with the allowable limits (predicted response plus tolerance) for deflections and strains iii. An evaluation of the accuracy of the measurements iv. An evaluation of any deviations (i.e., test results that exceed the allowable limits), the disposition of the deviations, and the need for corrective measures v. A discussion of the calculated safety margin provided by the structure as deduced from the test results For Category I structures inside containment, structures monitoring and maintenance requirements are acceptable if found to be in accordance with 10 CFR 50.65 and RG 1.160. It is important that Category I structures inside containment					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of the containment internal structures is essential for plant safety. Any special design provisions (e.g., providing sufficient physical access, providing alternative means for identifying conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containment internal structures are reviewed on a case-by-case basis.					
3.8.4, Rev. 2 (03/2007)	Other Seismic Category I Structures					
3.8.4.1	Description of the Structures. The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the criteria set forth in Section 3.8.4, and RG 1.206. New or unique design features that are not specifically covered in RG 1.70 or RG 1.206 may require a more detailed review. The reviewer determines the additional information that may be needed to accomplish a meaningful review of the structural aspects of such new or unique features. RG 1.206 provides the basis for evaluating the description of structures to be included in a DC or a COL application. RG 1.70 provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).					
3.8.4.2	Applicable Codes, Standards, and Specifications. The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Seismic Category I structures are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. A list of such documents follows: Codes/Specifications Title					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	AC I 349 "Code Requirements for Nuclear Safety-Related Concrete Structures" (with additional criteria provided in RG 1.142) ANSI/AISC N690-1994 " Specification for the Design, Fabrication and including Supplement 2 (2004) Erection of Steel Safety-Related Structures for Nuclear Facilities" RG 1.69. "Concrete Radiation Shields for Nuclear Power Plants" 1.91 "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants" 1.115 "Protection Against Low-Trajectory Turbine Missiles" 1.127 "Inspection of Water-Control Structures Associated with Nuclear					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Power Plants” 1.142 “Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)” 1.143 “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in LWR Plants” 1.160 “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” 1.199 “Anchoring Components and Structural Supports in Concrete”					
3.8.4.3	Loads and Load Combinations. The specified loads and load combinations are acceptable if found to be in accordance with the guidance given below:					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A. Concrete Structures All loads and load combinations are to be in accordance with ACI 349 and RG 1.142. Supplemental criteria on the use of loads and load combinations are presented below. Dead loads include hydrostatic loads and, for equipment supports, include static and dynamic head and fluid flow effects. Live loads include any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure. The dynamic effects of lateral soil pressure should be accounted for in accordance with the provisions of Subsection II.4(H) of this SRP section. For equipment supports, live loads also include loads resulting from vibration and any support movement effects. Alternate load cases, in which the magnitudes and locations of the live loads are arranged so that the design includes worst-case conditions, should be investigated, as appropriate. As noted in Appendix S to 10 CFR Part 50, the OBE is associated only with plant shutdown and inspection unless specifically selected by the applicant as a design input. If the OBE is set at one-third or less of the SSE ground motion, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the Seismic Category I structures remain functional and are within applicable stress, strain, and deformation limits. SRP Section 3.7 provides further guidance on the use of OBE. For structures or structural components subjected to					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	hydrodynamic loads resulting from LOCA and/or SRV actuation, the consideration of such loads should be as indicated in the appendix to SRP Section 3.8.1. Fluid structure interaction associated with these hydrodynamic loads and those from earthquakes should be taken into account. The design of concrete structures needs to consider the loads and load combinations that may occur during their construction. These loads consist of dead loads, live loads, temperature, wind, snow, rain, ice, and construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. Structural Engineering Institute (SEI)/American Society of Civil Engineers (ASCE) Standard 37 gives additional guidance on construction loads. This standard provides supplemental guidance, and in cases where the criteria in the standard and in the Code/SRP conflict, then the Code/SRP shall govern. The analysis should consider other site-related or plant-related loads applicable to Seismic Category I structures outside the containment such as floods, explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures. The inclusion of these loads and the related load combinations are reviewed on a case-by-case basis. B. Steel Structures All loads and load combinations are to be in accordance with AISC N690-1994 including Supplement 2 (2004).					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	This specification uses the allowable stress design (ASD) method. The supplemental criteria on the use of loads and load combinations presented above for concrete structures also apply to steel structures.					
3.8.4.4	Design and Analysis Procedures. The design and analysis procedures used for Seismic Category I structures, including assumptions about boundary conditions and expected behavior under loads, are acceptable if found to be in accordance with the following: A. For concrete structures, the procedures are in accordance with ACI 349, as supplemented by RG 1.142. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found in accordance with Appendix B to ACI 349, as supplemented by RG 1.199. B. The design and analysis methods described in Subsections II.4 of SRP 3.8.1 and 3.8.2, which apply to the other Category I concrete and steel structures, respectively, also need to be considered. Items to be considered include assumptions on boundary conditions, transient and localized loads, and shrinkage and cracking of concrete. C. For steel structures, the procedures are in accordance with ANSI/AISC N690-1994, including Supplement 2 (2004). D. Computer programs are acceptable if the validation provided follows the procedures delineated in Subsection II.4.E of SRP Section 3.8.1. E. The design report is considered acceptable if it contains the information specified in Appendix C to this SRP section. F. The structural audit is conducted in accordance with the provisions of Appendix B to this SRP section. G. The design of the spent fuel pool and racks is considered acceptable when it meets the criteria of Appendix D to this SRP section.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	H. Consideration of dynamic lateral soil pressures on embedded walls is acceptable if the lateral earth pressure loads are evaluated for two cases. These are (1) lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with ASCE 4-98, Section 3.5.3.2, and (2) lateral earth pressure equal to the passive earth pressure. If these methods are shown to be overly conservative for the cases considered, then the staff reviews alternative methods on a case-by-case basis. For earth retaining walls, the guidance in ASCE 4-98 Sections 3.5.3.1 through 3.5.3.3 is acceptable. I. The design of masonry walls is considered acceptable when it meets the requirements of Appendix A of this SRP. J. The design of structures that use modular construction methods are reviewed and evaluated on a case-by-case basis. NUREG/CR-6486 provides guidance related to the use of modular construction methods. Appendix B to NUREG/CR-6486 includes proposed modular construction review criteria.					
3.8.4.5	Structural Acceptance Criteria. For each of the loading combinations delineated in Subsection II.3 of this SRP section, the structural acceptance criteria appear in ACI 349 and RG 1.142 for concrete structures, and AISC N690-1994, including Supplement 2 (2004), for steel structures. The structural acceptance criteria for structures that use modular construction methods are evaluated on a case-by-case basis. See Subsection II.4.J of this SRP for information.					
3.8.4.6	Materials, Quality Control, and Special Construction Techniques. For Seismic Category I structures outside the containment, the materials and quality control programs are acceptable if found in accordance with the codes and standards					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	indicated in Subsection I.6 of this SRP section. Special construction techniques, if any, are evaluated on a case-by-case basis. For modular construction, reviewers evaluate the materials, quality control, and special construction techniques on a case-by-case basis. See Subsection II.4.J of this SRP section for more information.					
3.8.4.7	Testing and Inservice Surveillance Requirements. For Seismic Category I structures outside containment, structures monitoring and maintenance requirements are acceptable if program is in accordance with 10 CFR 50.65 and RG 1.160. For water control structures, inservice inspection programs are acceptable if in accordance with RG 1.127. Water control structures covered by this program include concrete structures, embankment structures, spillway structures and outlet works, reservoirs, cooling water channels and canals and intake and discharge structures, and safety and performance instrumentation. For Seismic Category I structures, it is important to accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of other Category I structures is essential for plant safety. The staff reviews any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high-radiation areas) to accommodate inservice inspection of other Category I structures on a case-by-case basis. For plants with nonaggressive ground water/soil (i.e., pH > 5.5, chlorides < 500 ppm, sulfates <1500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and foundations is to (1) examine the exposed portions of below-					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	grade concrete, when excavated for any reason, for signs of degradation; and (2) conduct periodic site monitoring of ground water chemistry, to confirm that the ground water remains nonaggressive. For plants with aggressive ground water/soil (i.e., it exceeds any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible, below-grade concrete for signs of degradation.					
3.8.4.8	Masonry Walls. Appendix A to this SRP section contains the acceptance criteria for masonry walls.					
3.8.5, Rev. 2 (03/2007)	Foundations					
3.8.5.1	<u>Description of the Foundation.</u> The descriptive information in the safety analysis report (SAR) is acceptable if it meets the criteria in Section 3.8.5.1 of RG 1.206. New or unique design features that are not specifically covered in RG 1.206 or RG 1.70 may require a more detailed review. The reviewer determines the additional information that may be needed to accomplish a meaningful review of the structural aspects of such new or unique features. RG 1.206 provides the basis for evaluating the description of structures to be included in a DC or a COL application. RG 1.70 provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).					
3.8.5.2	Applicable Codes, Standards, and Specifications. The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Seismic Category I foundations are covered by codes, standards, and guides that apply either in their					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	entirety or in part. Subsection II.2 of SRP Section 3.8.4 includes a list of such documents. In addition, the documents listed in Subsection II.2 of SRP Section 3.8.1 are acceptable for the containment foundation.					
3.8.5.3	<u>Loads and Load Combinations.</u> The specified loads and load combinations used in the design of Seismic Category I foundations are acceptable if found to be in accordance with those combinations referenced in Subsection II.3 of SRP Section 3.8.1 for the containment foundation and with those combinations listed in Subsection II.3 of SRP Section 3.8.4 for all other Seismic Category I foundations. In addition to the load combinations referenced above, the combinations used to check against sliding and overturning attributable to earthquakes, winds, tornadoes and against flotation because of floods are acceptable if found to be in accordance with the following: D + H + E D + H + W D + H + E' D + H + Wt D + F' Where D, E, W, E', and W_t are as referenced in Subsection II.3 of SRP Section 3.8.4, where H is the lateral earth pressure, and F' is the buoyant force of the design-basis flood. Justification should be provided for including live loads or portions thereof in these combinations. As noted in Appendix S to 10 CFR Part 50, the operating-basis earthquake (OBE), designated as E above, is only associated with plant shutdown and inspection unless the applicant specifically selects it as a design input. If the OBE is set at one-					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	third or less of the safe-shutdown earthquake (SSE) ground motion, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the Seismic Category I foundations remain functional and fall within applicable stress, strain, and deformation limits. SRP Section 3.7 provides additional guidance on OBE use.					
3.8.5.4	Design and Analysis Procedures. The design and analysis procedures used for Seismic Category I foundations are acceptable if found to be in accordance with the following: A. The design should consider the soil-structure interaction, hydrodynamic effect, and dynamic soil pressure. B. For Seismic Category I concrete foundations other than the containment foundations, the procedures are in accordance with the ACI 349, with additional guidance provided by RG 1.142. C. For Category I steel foundations, the procedures are in accordance with ANSI/AISC N690-1994 including Supplement 2 (2004). D. For the containment foundation, if in accordance with the design and analysis procedures referenced in SRP Section 3.8.1, Subsection II.4. E. The design report is acceptable if it satisfies the guidelines provided in SRP Section 3.8.4, Appendix C.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	F. The structural audit is conducted in accordance with SRP Section 3.8.4, Appendix B. Methods for determining the overturning moment attributable to an earthquake should be in accordance with the methods described in SRP Section 3.7.2. Computer programs are acceptable if the validation provided is found to be in accordance with the procedures delineated in Subsection II.4.E of SRP Section 3.8.1. In addition to the above, the design and analysis procedures for the following details are reviewed on a case-by-case basis: A. Method for determination of the bending moments and shear forces in the foundation mat for seismic loads? B. Performance of the sliding analysis method and how the analysis adequately accounts for potential foundation mat liftoff effects, if appropriate? The method to calculate the factor of safety against sliding. If sliding resistance is the sum of shear friction along the base mat and passive pressures induced by embedment effects, how these effects are considered in an analysis based on a consistent lateral displacement criterion? C. Evaluation of the capability of a foundation to transfer shear when waterproofing is used for a range of site conditions (soil sites with shear wave velocity of 1000 feet per second to hard rock)? D. The definition of dead load for uplift evaluations (floatation and seismic overturning), including the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	treatment of the stored volume of water in any pools? E. Detail explanation of how settlement (including potential effects of static or dynamic differential settlement) was considered. Evaluation and consideration of the effects of settlement on construction procedures. Evaluation of the allowable settlement (total and differential) that can be accommodated in the foundation/structures? F. The maximum toe pressure for base mat design under worst-case static and dynamic loads and its justification. G. The stiff and soft spots evaluation in the foundation soil to maximize the bending moments used in the design of the foundation mat. H. Description of the design details of critical locations, such as the junction of sidewall and base mat and the junctions of base mat to sumps. I. Detail explanation of the load path from all superstructures to the foundation mat to the subgrade. Discussion of any unique design features that occur in the load path (e.g., any safety-related function that the tendon gallery may have as part of the foundation in a prestressed containment or the connection of any internal structures to a steel containment and its supporting foundation). J. Explanation of how loads attributable to construction are considered in the design. Some examples of items to be discussed include the excavation sequence and loads from the construction					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	sequence of the foundation mat and walls, as well as the potential for loss of subgrade contact (e.g., because of loss of cement from a mud mat) that may lead to a differential pressure distribution on the mat.					
3.8.5.5	Structural Acceptance Criteria. For the loading combinations referenced in the first paragraph of Subsection II.3 of this SRP section, the allowable limits that constitute the acceptance criteria are referenced in Subsection II.5 of SRP Section 3.8.1 for the containment foundation and in Subsection II.5 of SRP Section 3.8.4 for all other foundations. In addition, for the five other load combinations in Subsection II.3 of this SRP section, the factors of safety against overturning, sliding, and flotation are acceptable if found to be in accordance with the following: Minimum Factors of Safety For Combination Overturning Sliding Flotation a. 1.5 --------------------- 1.5 -- b. 1.5 --------------------- 1.5 -- c. 1.1 --------------------- 1.1 -- d. ------------1.1 1.1 -- e. --- -------------------- --- 1					
3.8.5.6	Materials, Quality Control, and Special Construction Techniques. For the containment foundation, Subsection II.6 of SRP Section 3.8.1 references the acceptance criteria for materials, quality control, and any special construction techniques. For all other Seismic Category I foundations, the materials and quality control programs are acceptable if found to be in accordance with the codes and standards indicated in					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Subsection I.6 of this SRP section. Special construction techniques, if any, are treated on a case-by-case basis.					
3.8.5.7	Testing and Inservice Surveillance Requirements. For Category I foundations, structure monitoring and maintenance requirements are acceptable if found to be in accordance with 10 CFR 50.65 and RG 1.160. For water control structures, inservice inspection programs are acceptable if found to be in accordance with RG 1.127. Water control structures covered by this program include concrete structures, embankment structures, spillway structures and outlet works, reservoirs, cooling water channels and canals, as well as intake and discharge structures, and safety and performance instrumentation. For Category I foundations, it is important to accommodate inservice inspection of critical areas. The staff considers monitoring and maintaining the condition of Category I foundations as essential for plant safety. Any special design provisions (e.g., providing sufficient physical access, supplying a means for identification of conditions in inaccessible areas that can lead to degradation, performing remote visual monitoring of high-radiation areas) to accommodate inservice inspection of Category I foundations are reviewed on a case-by-case basis. For plants with nonaggressive ground water/soil (i.e., pH > 5.5, chlorides < 500 parts per million (ppm), sulfates < 1500 ppm), an acceptable program for normally inaccessible below-grade concrete walls and foundations is to (1) examine the exposed portions of below-grade concrete for signs of degradation, when excavated for any reason, and (2) conduct periodic site monitoring of ground-water chemistry to confirm that the ground water remains nonaggressive.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	For plants with aggressive ground water/soil (i.e., exceeding any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible below-grade concrete for signs of degradation. Subsection II.7 of SRP Section 3.8.1 covers additional testing and surveillance requirements for the containment foundation. Design of any special foundations will be reviewed on a case-by-case basis.					
3.9.1, Rev. 3 (03/2007)	Special Topics for Mechanical Components					
3.9.1.1	To meet the requirements of GDCs 1, 2, 14, 15, and 10 CFR Part 50, Appendix S, the applicant should provide a complete list of transients to be used in the design and fatigue analysis of all Code Class 1 and core support components, supports, and reactor internals within the reactor coolant pressure boundary. The number of events for each transient and the number of load and stress cycles per event and for events in combination should be included. All transients, such as startup and shutdown operations, power level changes, emergency and recovery conditions (including, for new applications, natural convection cooldown), switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients from single operator errors, inservice hydrostatic tests, seismic events as determined from the criteria specified in Appendix S to 10 CFR Part 50, and design-basis events contained in the Code-required "Design Specifications" for the components of the reactor coolant pressure boundary, should be specified.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The section of the applicant's SAR on transients will be acceptable if the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure conditions caused by those transients. To a large extent the selection of these specific transient conditions is based upon engineering judgment and experience. Some guidance on the selection of these transients and combinations can be found in SRP Section 3.9.3. Transients and consequent loads and load combinations with appropriate specified design and service limits should provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant. The staff should consider the number of transients appropriate for the design life of the plant. Also, environmental conditions to which equipment important to safety will be exposed (e.g., chemistry of the coolant water) should be considered to minimize the degradation of materials due to corrosion.					
3.9.1.2	To meet the requirements of 10 CFR Part 50, Appendix B, and GDC 1, a list of computer programs to be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items and the analyses to determine stresses should be provided. For each program the following information should be provided to demonstrate applicability and validity: A. The author, source, dated version, and facility. B. A description and the extent and limitation of its application.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	C. The computer program solutions to a series of test problems demonstrated to be substantially similar to solutions obtained from any one of sources (i) through (iv) and source (v): (i) Hand calculations (ii) Analytical results published in relevant engineering literature (iii) Acceptable experimental tests (iv) Results from a similar program within the acceptable margins. (v) The benchmark problems prescribed in NUREG/CR-1677, "Piping Benchmark Problems." Vols. I and II. A summary comparison of the solution obtained from sources (i) through (iv) should be provided in either graphical or numerical form. For source (v), the complete computer printout of the input and the solution should be submitted for every benchmark problem. These solutions may be referenced, and need not be resubmitted, in subsequent license applications, provided the information submitted under Items A and B remains unchanged.					
3.9.1.3	To meet the requirements of GDCs 1, 14, and 15, if experimental stress analysis methods are used in lieu of analytical methods for any seismic Category I Code or non-Code items, the section of the SAR addressing the experimental stress analysis methods is					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	acceptable if the information meets the provisions of Appendix II to ASME Code, Section III, Division 1 and, as in the case of analytical methods, if the information is sufficiently detailed to show the design meeting the provisions of the Code-required "Design Specifications."					
3.9.1.4	To meet the requirements of GDCs 1, 14, and 15 when Service Level D limits are specified by the applicant for Code Class 1 and core support components and for supports, reactor internals, and other non-Code items, the methods of analysis to calculate the stresses and deformations should conform to the methods outlined in Appendix F to ASME Code, Section III, Division 1, subject to the conditions addressed in subsection III.4 of this SRP section.					
3.9.2, Rev. 3 (03/2007)	Dynamic Testing and Analysis of Systems, Structures, and Components,					
3.9.2.1	Relevant requirements of GDCs 1, 2, 4, 14, and 15 are met if vibration, thermal expansion, and dynamic effects testing are conducted during startup functional testing for specified high- and moderate-energy piping and their supports and restraints. The purposes of these tests are to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service as required by the code and to confirm that no unacceptable restraint of normal thermal motion occurs. An acceptable test program to confirm the adequacy of the designs should include the following: A. A list of systems to be monitored. B. A list of the flow modes of operation and transients like pump trips, valve closures, etc. to which the components will be subjected during the test. (For					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	additional guidance see RG 1.68). For example, the transients of the reactor coolant system heatup tests should include but not necessarily be limited to: (i) Reactor coolant pump start. (ii) Reactor coolant pump trip. (iii) Operation of pressure-relieving valves. (iv) Closure of a turbine stop valve. C. A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak), pressure, or other appropriate criteria to show that the stress and fatigue limits are within the design levels should be provided. D. A list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from cold to hot position. E. A description of the thermal motion monitoring program (i.e., verification of snubber movement, adequate clearances and gaps, including acceptance criteria and how motion will be measured). F. If vibration is noted beyond the acceptance levels set by the criteria of Item II.1.C above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If, during the test, piping system restraints are determined to be inadequate or are					

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	damaged, corrective restraints should be installed and another test should determine whether the vibrations have been reduced to an acceptable level. If no snubber piston travel is measured at those stations indicated in Item II.1.D of the acceptance criteria, the corrective action to be taken to ensure that the snubber is operable should be described.					
3.9.3.2	To meet the requirements of GDC 2, acceptance criteria for the areas of review described in subsection I.2 of this SRP section are given below. Other approaches which can be justified as equivalent to or more conservative than the stated acceptance criteria may be used to confirm the ability of all Seismic Category I systems and components and their supports to function as needed during and after an earthquake. A. <u>Seismic Analysis Methods.</u> The seismic analysis of all Category I systems, components, equipment, and their supports (including supports for conduit and cable trays and ventilation ducts) should utilize either a suitable dynamic analysis method or an equivalent static load method, if justified. i. <u>Dynamic Analysis Method.</u> A dynamic analysis (e.g., response spectrum method, time history method, etc.) should be used when the use of the equivalent static load method cannot be justified. To be acceptable such analyses should consider the following items: (1) Use of either the time history or the response spectrum method. (2) Use of an adequate number of masses or degrees of freedom in dynamic modeling to determine the response of all Category I and					

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	applicable non-Category I systems and plant equipment. The number is adequate when additional degrees of freedom do not result in more than a 10-percent increase in responses. Alternately, the number of degrees of freedom may be taken as equal to twice the number of modes with frequencies less than 33 Hz. (3) Investigation of a sufficient number of modes to ensure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10-percent increase in responses. (4) Consideration of maximum relative displacements among supports of Category I systems and components. (5) Inclusion of such significant effects as piping interactions, externally-applied structural restraints, hydrodynamic (both mass and stiffness effects) loads, and nonlinear responses. ii. <u>Equivalent Static Load Method. An equivalent static load method is acceptable if:</u> (1) There is justification that the system can be realistically represented by a simple model and the method produces conservative results in responses. Typical examples or					

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	published results for similar systems may be submitted in support of the use of the simplified method. (2) The design and simplified analysis account for the relative motion between all points of support. (3) To obtain an equivalent static load of equipment or components which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. A factor of less than 1.5 may be used with adequate justification. In addition, for equipment which can be modeled adequately as a one-degree-of-freedom system, the use of a static load equivalent to the peak of the floor response spectra is acceptable. For piping supported at only two points, the use of a static load equivalent to the peak of the floor response spectra is also acceptable. B. Determination of Number of Earthquake Cycles. The number of earthquake cycles during one seismic event, the maximum number of cycles for which applicable systems and components are designed, and the criteria and the applicant's procedures to establish these parameters are reviewed by the staff in accordance with the guidance of SRP Section 3.7.3. C. Basis for Selection of Frequencies. To avoid resonance, the fundamental frequencies of components and					

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	equipment selected preferably should be less than 1/2 or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads. D. Three Components of Earthquake Motion. Depending upon what basic methods are used in the seismic analysis (i.e., response spectra or time history method) the following two approaches are acceptable for the combination of three-dimensional earthquake effects. (i) Response Spectra Method. When the response spectra method is adopted for seismic analysis, the maximum structural responses due to each of the three components of earthquake motion should be combined by taking the square root of the sum of the squares of the maximum codirectional responses caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model. (ii) Time History Analysis Method. When the time history analysis method is employed for seismic analysis, two types of analysis are generally performed depending on the complexity of the problem. (1) to obtain maximum responses to each of the three components of the earthquake motion: in this case the method for combining the three-dimensional effects is identical to that described in Item (i) except that the maximum responses are calculated by the time history method instead of the spectrum method. (2) To obtain time history responses from each of the three components of					

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	the earthquake motion and combine them at each time step algebraically: the maximum response in this case can be obtained from the combined time solution. When this method is used, to be acceptable the earthquake motions specified in the three different directions should be statistically independent. E. Combination of Modal Responses. SRP Section 3.7.2 and RG 1.92,"Combining Modal Responses and Spatial Components in Seismic Response Analysis," present criteria and guidance for modal response combination methods acceptable to the staff. F. Analytical Procedures for Piping Systems. The seismic analysis of Category I piping may use either a dynamic analysis or an equivalent static load method. The acceptance criteria for the dynamic analysis or equivalent static load methods are described in subsection II.2.A of this SRP section. G. Multiply-Supported Equipment and Components With Distinct Inputs. Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different. A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper-bound envelope of all the individual response spectra for these locations to calculate maximum inertial responses of multiply-supported items. In addition, the relative displacements at the support points should be					

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	considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the structural response calculations or, as a conservative approximation, from the floor response spectra. For the latter option, the maximum displacement of each support (S_d) is predicted by: $$S_d = S_a\, g / \omega^2$$ where S_a is the spectral acceleration in "g's" at the high frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and w is the fundamental frequency of the primary support structure in radians per second. The support displacements can then be imposed on the supported item in the most unfavorable combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method. In the case of multiple supports located in a single structure, an alternate acceptable method using the floor response spectra determines dynamic responses due to the worst single floor response spectrum selected from a set of floor response spectra at various floors and applied identically to all the floors provided there is no significant shift in frequencies of the spectra peaks. In addition, the support displacements should be imposed on the					

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	supported item in the most unfavorable combination by static analysis procedures. Further criteria and methods for the evaluation of multiple support arrangement analysis issues are described in SRP Sections 3.7.2 and 3.7.3. These methods can result in overestimation of seismic responses. Acceptable alternate response spectrum analysis methods that provide more realistic estimation of seismic responses are discussed in subsection II.9 of SRP Section 3.7.3. In lieu of the response spectrum approach, time histories of support motions may be used as excitations to the systems. Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiply-supported systems. H. Use of Constant Vertical Static Factors. The use of constant vertical load factors as vertical response loads for the seismic design of all Category I systems, components, equipment, and their supports in lieu of a vertical seismic system dynamic analysis is acceptable only if the structure is demonstrably rigid in the vertical direction. The criterion for rigidity is that the lowest frequency in the vertical direction be more than 33 Hz. I. Torsional Effects of Eccentric Masses. For Seismic Category I systems, if the torsional effect of an eccentric mass like a valve operator in a piping system is judged to					

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	be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for significance will have to be determined case by case. J. Category I Buried Piping Systems. For Category I buried piping systems, the following items should be considered in the analysis: (i) The inertial effects due to an earthquake upon buried piping systems should be adequately considered in the analysis. Use of the procedures described in the references is acceptable. (ii) The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Use of the procedures described in the references is acceptable. (iii) When applicable, the effects of local soil settlements, soil arching, etc., also should be considered in the analysis. K. Interaction of Other Piping with Category I Piping. To be acceptable, each non-Category I piping system should be designed to be isolated from any Category I piping system by either a constraint or barrier or should be located remotely from the seismic Category I piping system. If isolation of the Category I piping system is not feasible or practical, adjacent non-Category I piping should be analyzed according to the same seismic criteria applicable to the Category I piping system. For non-Category I piping					

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	systems attached to Category I piping systems, the dynamic effects of the non-Category I piping should be simulated in the modeling of the Category I piping. The attached non-Category I piping, up to the first anchor beyond the interface, also should be designed not to cause a failure of the Category I piping during an earthquake of SSE intensity. L. Criteria Used for Damping. RG 1.61,"Damping Values for Seismic Design of Nuclear Power Plants," provides acceptable values which may be used. The methods for analysis of damping should be consistent with those described in SRP Section 3.7.2.					
3.9.2.3	To meet the requirements of GDCs 1 and 4, the following guidelines, in addition to RG 1.20 “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing”, apply to the analytical solutions to predict vibrations of reactor internals for prototype plants. Generally, this analysis is required only for prototype designs and power uprate of existing plants; However, it is not required for non-prototypes except that segments of an analysis (in particular, assessments of any potential adverse flow effects) may be necessary if there are deviations from the prototype internals design or operating conditions or if the non-prototype is based on a conditional prototype which has experienced problems from adverse flow effects. If the reactor internal structures are a non-prototype design, the applicant should refer to the results of tests and analyses for the prototype reactor and give a brief summary of the results. A more detailed summary of results of assessment of the potential of any adverse flow effects also should be given. A. The results of vibration and stress calculations should consist of the following:					

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	(i) Dynamic responses to operating transients at critical locations of the internal structures should be determined and, in particular, at the locations where vibration sensors will be mounted on the reactor internals. For each location, the maximum response, the modal contribution to the total response, (in case of cyclic or resonant behavior), and the response causing the maximum stress amplitude should be calculated. (ii) The damping factors for different modes should be properly selected and substantiated. In prior submissions, utilities have cited NRC damping guidance for very low frequency seismic analyses as justification for high damping factors for mid-to-high frequency analyses. RG 1.20 corrects this guidance and requires that damping factors used in structural dynamic modeling be based on mid- to high-frequency measurements or rigorous analyses conducted on structures typical of the reactor internal structure modeled. (iii) The dynamic properties of internal structures, including the natural frequencies and shapes of the dominant modes, should be characterized. In analyses of a component structural element basis, the presence of dynamic coupling among component structure elements should be investigated. Upper bounds on the uncertainties of all natural frequencies of the relevant resonance modes should be provided. The uncertainties and bias errors of the amplitudes of the frequency response functions (FRFs) also should be provided. The uncertainties and bias errors may be estimated from comparisons of simulations to measurements made on structures similar in construction to the reactor internal being modeled. The performance of					

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	hammer tests would be expected for replacement steam dryers. (iv) Dynamic responses of reactor internals to self-excited flow oscillations should be estimated. The applicants/licensees should analyze in detail adverse flow effects generated by various excitation mechanisms like vortex-induced vibration flow-excited acoustic resonance, fluid-elastic instability, and other flow instabilities (e.g., separated and impinging flow instabilities). These mechanisms may be assessed by theoretical, numerical, or experimental techniques, including scale model testing. The analysis should clearly identify whether each mechanism will be excited during the planned operating range of the power plant. Full dynamic analysis is requested for mechanisms expected to generate adverse flow effects, including estimation of vibration and stress amplitudes at the critical locations and, in particular, where vibration sensors will be mounted on the reactor internals. RG 1.20, Section C.2.1.3 provides more guidance on self-excited flow instabilities. (v) The dependance of the dynamic response on hydrodynamic excitation forces like coolant recirculation pump frequencies and the flow path configuration should be evaluated. Any frequency coincidence between the pump blade passing frequency and the natural frequencies of the internal structures should be identified and supplemented with error and uncertainty analysis.					

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	(vi) Acceptance criteria should be established for allowable responses and for the location of vibration sensors. Such criteria relate to the code-allowable stresses, strains, and limits of deflection established to preclude loss of function of the reactor core structures and fuel assemblies. B. The forcing functions should account for the effects of transient flow conditions and the frequency content. Any potential amplification of a forcing function caused by self-excitation or "lock-in" of a flow instability with a structural or acoustic resonance should be clearly quantified (See RG 1.20, Section C.2.1.3 for more guidance on self-excited flow instabilities). Acceptable methods for formulating forcing functions for vibration prediction include the following: i. Analytical method: based on standard hydrodynamic theory, the governing differential equations for vibratory motions should be developed and solutions obtained with appropriate boundary conditions and parameters. This method is acceptable where the geometry along the fluid flow paths is mathematically tractable. ii. Test-analysis combination method: based on data obtained from plant or scale model tests (e.g., velocity or pressure distribution data), forcing functions should be formulated to include the effects of complex flow path configurations and wide variations of pressure distributions. The suitability of any approach used to define forcing functions should be assessed with expected bias errors and uncertainties of					

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	the selected approach. In addition to direct measurements in nuclear power plants, the following approaches may be used to formulate the forcing functions. (1) Scale Model Tests (SMTs): If SMTs are used to develop forcing functions, the following areas should be considered. (a) The scale model should be dynamically similar to the prototype. The dynamic similarity should cover all fluid, structural (such as piping dimensions and elbow locations), and acoustic parameters relevant to the phenomenon considered. If some distortions in the dimension-less parameters of the scale model should be made, the applicants/licensees should show that these distortions are conservative. As an example, sound attenuation in scale models is normally substantially higher than that of the prototype due to viscous heat conduction and other losses higher in small-size models tested at low pressures, leading to the requirement that the scale					

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	model size and its test pressure be sufficiently large to ensure the re-production of such specific flow phenomena as flow-induced vibration and acoustic resonance present in the prototype. (b) The effects of structural damping and sound attenuation (in the test medium) on the loading function measured in the scale model should be considered carefully. Any non-conservative deviations in these parameters from those of the prototype reactor should be corrected when the loading function is scaled to that of a full-size reactor pressure vessel (RPV). (c) The conservative simulation of boundary conditions in the scale model. (d) Whether the size of the scale model is sufficiently large to allow investigation of small relevant details in geometry (e.g., branch line					

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	openings). (e) Validation of the SMT results by measurements in nuclear power plants. (2) CFD: If CFD simulations are used to develop unsteady forcing functions, the following areas should be considered. (a) Include acoustic/vibration coupling to simulate enhancement of flow instabilities (if any). (b) Grid size sensitivity tests. (c) The Courant number requirement should be met. (d) There should be unsteady simulations using Large Eddy Simulation (LES) or Direct Numerical Simulation (DNS) at high Reynolds number flow and including compressibility					

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	effects to model any coupling of the flow with the acoustic waves in the fluid (self-excitation or lock-in effects). (e) Real gas simulation should be used (i.e., use state equation of steam as real gas). (f) The simulation procedures should be validated on similar (i.e., complex and high Reynolds number) flow situations. (3) Acoustic Modeling of Steam System: If an acoustic model of the steam system (the steam within the MSLs and the RPV) computes fluctuating pressures within the RPV and on BWR steam dryers inferred from measurements of fluctuating pressures within the MSLs connected to the RPV, the following areas should be considered. (a) There should be at least two measurement locations on each MSL in a BWR; however, three					

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	measurement locations on the MSLs improve input data to an acoustic model, particularly if the locations are spaced logarithmically, reducing uncertainty in describing the waves coming from and going into the RPV. With two or three measurement locations, there should be no acoustic sources between the measurement locations, unless justified. (b) Strain gages (at least four gages circumferentially oriented and placed at equal distance along the circumference) may be used to relate the hoop strain in the MSL to the internal pressure. Strain gages should be calibrated according to the MSL dimensions (diameter, thickness, and static pressure). Alternatively, pressure measurements made with transducers flush-mounted against the MSL internal surface may be used. The effects of flow turbulence on any direct pressure measurements					

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	should be considered, however. (c) The speed of sound in any acoustic models should not be changed from plant to plant but rather be a function of temperature and steam quality. (d) Reflection coefficients at any boundary between steam and water should be based on rigorous modeling or on direct measurement. The uncertainty of the reflection coefficients should be clearly defined. (e) Any sound attenuation coefficients should be a function of steam quality (variable between the chimney and reactor dome) rather than constant throughout a steam volume (like the volume within the RPV). (f) (f) Once validated, the same speed of sound, attenuation coefficient, and reflection coefficient should					

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	be used in other plants; however, different flow conditions (temperature, pressure, quality factor) may require adjustments of these parameters. (4) Response-deduction method: based on a derivation of response characteristics from plant or SMT data, forcing functions should be formulated; however, as such functions may not be unique and are also expected to depend on material properties and loss factors, the computational procedures and the basis for selection of the representative forcing functions should be described together with all bias errors and uncertainties (see subsection II.3.B.(ii)(1) of this SRP section, “Scale Model Tests,” for guidelines on inferring forcing functions from plant or scale model testing data). Alternately, the applicant/licensee may use other approaches to formulate the forcing function. However, sufficient supporting justification should be provided to demonstrate that the selected approach is technically sound and realistically predicts the					

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	forcing function. In addition, an assessment of bias errors and uncertainties should be provided. C. Acceptable methods of obtaining dynamic responses for vibration and stress predictions are as follows: (i) If a numerical model is used to compute mode shapes and FRFs, the modeling approach should be documented along with the model itself. Uncertainties and bias errors for both the approach and the specific model should be provided along with their bases. Additional guidance on numerical uncertainties and bias errors can be found in RG 1.20. ii. Force-response computations are acceptable if the characteristics of the forcing functions are predetermined conservatively and the mathematical model of the reactor internals is appropriately typical of the design. (ii) If the forcing functions are not predetermined, either a special analysis of response signals measured from reactor internals of similar design may predict amplitude and modal contributions or parameter studies useful for extrapolating the results from tests of internals or components of similar designs based on composite statistics may be used. The latter approach should be used only when the expectation that flow-induced vibration or acoustic resonance will not occur for the operation conditions covering the extrapolated range of the forcing functions is shown beyond doubt.					

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	D. Vibration predictions should be verified by RPV, steam, feed water and condensate piping, and safety relief valve test results. This procedure should consider all sources of bias errors and uncertainties. If the test results differ substantially from the predicted response behavior, the vibration analysis should be modified appropriately for more agreement with test results and validation of the analytical method and input forcing functions as appropriate for predicting responses of the prototype unit as well as of other units where confirmatory tests are conducted.					
3.9.2.4	For requirements of GDCs 1 and 4, the preoperational vibration and stress test program for the internals of a prototype reactor, for existing reactors under consideration for power uprate, and for non-prototype reactors whose valid or conditional prototypes have experienced structural failures due to adverse flow effects in any plant (e.g., steam dryer cracking and valve failures) should conform to the requirements for a prototype test as specified in RG 1.20, including vibration prediction, vibration monitoring, adverse flow effects (flow-induced acoustic and structural resonances, data reduction, bias errors and uncertainty analysis, and walkdown and surface inspections. The test program to demonstrate design adequacy of the reactor internals should include, but not necessarily be limited to, the following: A. The vibration testing should be conducted with the fuel elements in the core or with dummy elements with equivalent dynamic effects and flow characteristics. Testing without fuel elements in the core may be acceptable if testing in this mode is demonstrably conservative.					

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	B. The vibration monitoring instrumentation should be described briefly, including instrument types and specifications (including useful frequency and amplitude ranges) and diagrams of locations, including those with the most severe vibratory motions or the most effect on safety functions. C. Testing to evaluate potential adverse flow effects on reactor internal components should include the steam dryer and MSL valves. The instrumentation directly mounted on the steam dryer should include pressure sensors, strain gages, and accelerometers. The MSLs also should be instrumented to collect data to determine steam pressure fluctuations to identify the presence of flow-excited acoustic resonances and to allow the analysis of those pressure fluctuations to calculate MSL valve loading and vibration and steam dryer loading and stress. Accelerometers should be mounted on the main steam valves to record the presence and the level of any flow-excited acoustic resonance or vibration. D. The planned duration of the test for the normal operation modes to ensure that all critical components are subjected to at least 10^6 cycles of vibration should be provided. For instance, if the lowest response frequency of the core internal structures is 10 Hz, a total test duration of 1.2 days or more is acceptable. E. Testing should include all of the flow modes of normal operation and upset transients. The proposed set of flow modes is acceptable if it provides a conservative basis for determining the dynamic response of the tested components and is reviewed on request. The					

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	power ascension program for startup testing should include specific hold points with sufficiently long duration to allow data recording and reduction, comparisons with predetermined limit loading, and inspections and walkdowns for steam, feedwater, and condensate systems. The test program also should include details of actions to be taken if acceptance criteria are not satisfied. Further information on test procedure is addressed in RG 1.20. F. The methods and procedures to process the test data for meaningful interpretation of the vibration behavior of various components should be provided. Vibration interpretation should include the amplitude, frequency content, stress state, and possible effects on safety functions. There should be detailed analysis of bias errors and uncertainties of instrumentation, data acquisition systems, and models to estimate loading functions from the measured data. G. Vibration predictions, test acceptance criteria and bases, and permissible deviations from the criteria should be provided before the test. H. The applicant/licensee is expected to provide a summary evaluation of plant startup and power ascension to the staff within 90 days of plant startup. If full licensed power is not achieved in that time period, the applicant/licensee is expected to provide a supplemental report within 30 days after achieving full licensed power. I. There should be walkdown inspections during and visual and nondestructive surface inspections after completion					

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	of the vibration tests. The inspection program description should include the areas subject to inspection, the methods of inspection, the design access provisions to the reactor internals, and the equipment to be used for such inspections, which preferably should follow the removal of the internals from the reactor vessel. Where removal is not feasible, the inspections should be by means of equipment appropriate for in-situ inspection. The areas inspected should include all load-bearing interfaces, core restraint devices, high-stress locations, and locations critical to safety functions. MSL valves also should be inspected if adverse flow effects (flow-induced acoustic and structural resonances) are observed during the startup test. For later reactor internals with the same design, size, configuration, and operating conditions as the prototype, the vibration test program should comply with the requirements of the appropriate non-prototype program as specified in RG 1.20.					
3.9.2.5	For requirements of GDCs 2, 4, 14, and 15 dynamic system analyses should confirm the structural design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe LOCA in combination with the SSE. Where a substantial separation between the forcing frequencies of the LOCA (or SSE) loading and the natural frequencies of the internal structures can be demonstrated, the analysis may treat the loadings statically. Evaluations performed under SRP Section 3.6.3, address review of applications that propose to eliminate consideration of design loads of the dynamic effects of pipe rupture. Evaluation in this Section should interface with the evaluation in Section 3.6.3.					

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	The most severe dynamic effects from LOCA loadings generally result from a postulated double-ended rupture of a primary coolant loop near a reactor vessel inlet or outlet nozzle with the reactor in the most critical normal operating mode. However, all other postulated break locations should be evaluated and the location producing the controlling effects should be identified. Mathematical models used for dynamic system analysis for LOCAs in combination with SSE effects should include the following: A. Modeling should include reactor internals and dynamically-related piping, pipe supports, components, and fluid-structure interaction effects when applicable. Typical diagrams and the modeling basis should be developed and described. B. Mathematical models should typify system such structural characteristics as flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping). C. Any system structural partitioning and directional decoupling in the dynamic system modeling should be justified. D. The effects of flow upon the mass and flexibility properties of the system should be addressed. Typical diagrams and the basis for postulating the LOCA-induced forcing function should be provided, including a description of the governing hydrodynamic equations and the assumptions for					

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	mathematically tractable flow path geometries, tests for determining flow coefficients, and any semi-empirical formulations and scaled model flow testing for determining pressure differentials or velocity distributions. The acceptability of the hydraulic analysis, as reviewed on request, is based on established engineering practice and generic topical reviews by the staff. The methods and procedures for dynamic system analyses should be described, including the governing equations of motion and the computational scheme for deriving results. Time domain forced-response computation is acceptable for both LOCA and SSE analyses. The response spectrum modal method may be used for SSE analysis. The stability of such elements in compression as the core barrel and the control rod guide tubes under outlet pipe rupture loadings should be investigated. Either response spectra or time histories may be used for specifying seismic input motions of the SSE at the reactor core supports. The criteria for acceptance of the analytical results are described in SRP Sections 3.9.3 and 3.9.5. For PWRs, the criteria and review methods for verifying whether the applicant has appropriately addressed asymmetric blowdown loadings on reactor internals are described in SRP Section 3.9.5.					
3.9.2.6	For requirements of GDC 1, as to the correlation of tests and analyses of reactor internals, the applicant should address the following items to ensure the adequacy and sufficiency of the test and analysis results. A. Comparison of the measured response frequencies with					

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	the analytically obtained natural frequencies of the reactor internals for validation of the mathematical models used in the analysis. Comparison of the measured and predicted damping factors as a function of natural frequencies for validation of the damping assumed in the analysis. B. Comparison of the analytically obtained mode shapes with the shape of measured motion for identification of the modal combination or verification of a specific mode. C. Comparison of the response amplitude time variation and the frequency content from test and analysis for verification of the postulated forcing function. D. Comparison of the measured amplitudes, frequencies, and time variations of loads with those predicted by test-analysis combination method for validation of the predicted forcing function. E. Comparison of the maximum responses from test and analysis for verification of stress levels. F. Comparison of the mathematical model for dynamic system analysis under operational flow transients and under combined LOCA and SSE loadings for similarities. G. Comparison of measurements and predictions of any adverse flow phenomena (e.g., flow-excited acoustic and/or structural resonances) for validation of the model(s) predicting the loading induced by the phenomena.					
3.9.2.7	For new applications, test specifications should be in accordance with ASME OM-S/G-1990, "Standards and Guides For Operation of Nuclear Power Plants," Part 3, "Requirements for					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems," and Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems."					
3.9.3, Rev. 2 (03/2007)	ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures					
3.9.3.1	Loading Combinations, System Operating Transients, and Stress Limits. The design and service loading combinations, including system operating transients, and the associated design and service stress limits considered for each component and its supports should be sufficiently defined to provide the basis for design of Code Class 1, 2, and 3 components and component supports, and core support structures for all conditions. The acceptability of the combination of design and service loadings (including system operating transients), applicable to the design of Class 1, 2, and 3 components and component supports, and core support structures, and of the designation of the appropriate design or service stress limit for each loading combination, is judged by comparison with positions stated in Appendix A, and with appropriate standards acceptable to the staff, developed by professional societies and standards organizations. The design criteria for internal parts of components such as valve discs, seats, and pump shafting should comply with applicable Code or Code Case criteria. In those instances where no Code criteria exist, the design criteria are acceptable if they ensure the structural integrity of the part such that no safety-related functions are impaired.					
3.9.3.2	Design and Installation of Pressure Relief Devices.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The applicant should use design criteria for pressure relief installations specified in Appendix O, ASME Code, Section III, Division 1, "Rules for the Design of Safety Valve Installations." In addition, the following criteria are applicable: A. Where more than one valve is installed on the same pipe run, the sequence of valve openings to be assumed in analyzing for the stress at any piping location should be that sequence which is estimated to induce the maximum instantaneous value of stress at that location. B. Stresses should be evaluated, and applicable stress limits should be satisfied for all components of the pipe run and connecting systems and the pressure relief valve station, including supports and all connecting welds between these components. C. In meeting the stress limit requirements, the contribution from the reaction force and the moments resulting from that force should include the effects of a Dynamic Load Factor (DLF) or should use the maximum instantaneous values of forces and moments for that location as determined by the dynamic hydraulic/structural system analysis. This requirement should be satisfied in demonstrating satisfaction of all design limits at all locations of the pipe run and the pressure relief valve for Class 1, 2, and 3 piping. A DLF of 2.0 may be used in lieu of a dynamic analysis to determine the DLF. The SAR should also include a description of the calculational procedures, computer programs, and other methods to be used in the analysis. The analysis should include the time history or equivalent effects of changes of momentum due to fluid flow changes of direction. The fluid states considered should include					

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	postulated water slugs where water seals are used and subcooled or saturated liquid if such fluid can be discharged under postulated transient or accident conditions. Applicants for plants utilizing suppression pools should also consider the applicable pool dynamic loads on the safety relief valve system. Stress computations and stress limits must be in accord with applicable rules of the Code.					
3.9.3.3	Component Supports. The component support designs should provide adequatemargins of safety under all combinations of loadings. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination should meet the criteria in Appendix A, Regulatory Guides (RG) 1.124 and RG 1.130 and Subsection NF of the Code. A. Component supports of active pumps and valves should be considered in context with the other features of the functionality assurance and seismic qualification program as presented in SRP Section 3.10. If the component support deformation can be expected to affect the operability requirements of the supported component, then deformation limits should also be specified. Such deformation limits should be compatible with the operability requirements of the supported components. These deformation limits should be incorporated into the functionality assurance and seismic qualification program. In establishing allowable equipment deformations, the possible movements of the support base structures must be taken into account. B. Criteria for snubber functionality assurance should contain the following elements:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(i) Structural Analysis and Systems Evaluation. Systems and components which utilize snubbers as shock and vibration arresters should be analyzed to ascertain the interaction of such devices with the systems and components to which they are attached. Snubbers may be used as shock and vibration arresters and in some instances as dual purpose snubbers, and when so used fatigue strength should be considered. Important factors in the fatigue evaluation include: (1) unsupported system component movement or amplitude, (2) force imparted to snubber and corresponding reaction on system or component due to restricting motion (damped amplitude), (3) vibration frequency or number of load cycles, and (4) verification of system or component and snubber fatigue strength. Snubbers used as shock arresters need not undergo a fatigue evaluation if it can be demonstrated that: (a) the number of load cycles which the snubber will experience during normal plant operating conditions is small (<2500) or (b) motion during normal plant operating					

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	conditions does not exceed snubber dead band. Snubbers utilized in systems or components which may experience high thermal growth rates, either during normal operating conditions or as a result of anticipated transients, should be checked to ensure that such thermal growth rates do not exceed the snubber lock-up velocity. (ii) Characterization of Mechanical Properties. An important aspect of the structural analysis is realistic characterization of snubber mechanical properties (i.e., spring rates) in the analytical model. Since the "effective" stiffness of a snubber is generally greater than that for the snubber support assembly (i.e., the snubber plus clamp, transition tube extension, back-up support structure, etc.) the snubber response characteristics may be "washed out" by the added flexibility in the support structure. The combined effective stiffness of the snubber and support assembly should therefore be considered in evaluating the structural response of the system or component. Snubber spring rate should be determined independent of clearance/lost motion, activation level, or release rate. The stiffness should be based on structural and hydraulic compliance, and the effects of temperature should be considered. The snubber end fitting clearance, mismatch of end fitting clearances, mismatch of activation and release rates, and lost motion should be minimized and should					

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	be considered when calculating snubber reaction loads and stress which are based on a linear analysis of the system or component. This is especially important in multiple snubber applications where mismatch of end fitting clearance has a greater effect on the load sharing of these snubbers than does the mismatch of activation level or release rate. Equal load sharing of multiple snubber supports should not be assumed if mismatch in end fitting clearance exists. (iii) Design Specifications. The required structural and mechanical performance of snubbers is determined from the applicant's structural analysis described in Subsections II.3.B(i) and (ii). The snubber Design Specification is the instrument provided by the purchaser to the supplier to ensure that the requirements are met. The Design Specification should contain: (1) the general functional requirements, (2) operating environment, (3) applicable codes and standards, (4) materials of construction and standards for hydraulic fluids and lubricants, (5) environmental, structural, and performance design verification tests, including the required dynamic qualification, testing and extrapolation methods supporting qualification of large bore hydraulic snubbers with rated load capacities of 50 Kips or					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	more as recommended in NUREG/CR-5416. (6) production unit functional verification tests and certification, (7) packaging, shipping, handling, and storage requirements, and (8) description of provisions for attachments and installation. In addition, the procurement program should include provisions for the snubber manufacturer to submit quality assurance and assembly quality control procedures for review and acceptance by the purchaser. (iv) Use of Additional Snubbers. Snubbers could in some instances be installed during or after plant construction. These snubbers may not have been included in the design analysis. This could occur as a result of unanticipated piping vibration, as discussed in SRP Section 3.9.2, or interference problems during construction. The effects of such snubbers should be fully evaluated and documented to demonstrate that normal plant operations and safety are not diminished.					
3.9.4, Rev. 3 (03/2007)	Control Rod Drive Systems					
3.9.4.1	The descriptive information is determined to be sufficient provided the minimum requirements for such information meet Section 3.9.4 of Regulatory Guide (RG) 1.29.					

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3.9.4.2	Construction (as defined in NCA-1110 of Section III of the ASME Code) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable: A. For pressurized portions of equipment classified as Quality Group A, B, C (RG 1.26): Section III of the ASME Code, Class 1, 2, or 3 as appropriate. B. For pressurized portions of equipment classified as Quality Group D (RG 1.26): i. Section VIII, Division 1, of the ASME Code for vessels and pump casings. ii. For piping systems (American National Standards Institute, ANSI):(1) B 16.5 Steel Pipe Flanges and Flanged Fittings B 16.9 Steel Butt Welding Fittings B16.11 Steel Socket Welding Fittings B16.25 Butt Welding Ends B 16.34 Steel Valves with Flanged and Butt B31.1 Welding Ends Power Piping MSS-SP-25 Marking for Valves, Fittings, Flanges, and Unions C. For nonpressurized equipment (Non-ASME Code): Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. A justification of any					

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	decreases in design margins should be provided.					
3.9.4.3	For the various design and service conditions defined in NB-3113 of Section III of the ASME Code, load combination sets are as given in SRP Section 3.9.3. The stress limits applicable to pressurized and nonpressurized portions of the control rod drive systems should be as given in SRP Section 3.9.3 for the response to each loading set. For BWRs, the CRDS design should adequately consider water hammer loads to assure that system safety functions can be achieved.					
3.9.4.4	The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and ability to overcome a stuck rod meet system design requirements.					
3.9.5, Rev. 3 (03/2007)	Reactor Pressure Vessel Internals					
3.9.5.1	Requirements for loads, loading combinations, and limits applicable to those portions of reactor internals constructed to					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Subsection NG of the ASME Code are presented in SRP Section 3.9.3.					
3.9.5.2	The design and construction of the core support structures should comply with the requirements of Subsection NG, "Core Support Structures," of the ASME Code and SRP Section 3.9.3.					
3.9.5.3	The design criteria, loading conditions, and analyses that provide the bases for the design of reactor internals other than the core support structures should meet the guidelines of NG-3000 and be constructed not to affect the integrity of the core support structures adversely (NG-1122). If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified					
3.9.5.4	Deformation limits for reactor internals should be established by the applicant and presented in the safety analysis report. The basis for these limits should be included. The stresses of these displacements should not exceed the specified limits. The requirements for dynamic analysis of these components are addressed in SRP Section 3.9.2.					
3.9.5.5	The reactor internals should be designed to accommodate asymmetric blowdown loads from postulated pipe ruptures. The applicant's evaluation of such loads should demonstrate that they do not exceed the limits imposed by the applicable codes and standards. Where double-ended guillotine break of reactor coolant piping is postulated, criteria for evaluating loading transients and structural components are specified in NUREG-0609.					
3.9.5.6	Potential adverse flow effects of flow-induced vibration (FIV) and acoustic resonances on reactor internals (including the steam dryer in BWRs) should be adequately addressed in accordance with relevant criteria stated in the Appendix to this SR Section.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
3.9.6, Rev. 3 (03/2007)	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints					
3.9.6.1	Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints A. For new plant applications, safety-related pump, valve, and piping designs should include provisions to allow testing of pumps and valves at the maximum flow specified in the plant accident analyses. B. Functional design and qualification of each safety-related pump and valve should be accomplished such that each pump and valve is capable of performing its intended function for a full range of system differential pressure and flow, ambient temperatures, and available voltage (as applicable) under all conditions ranging from normal operating to design-basis accident conditions. C. Acceptance criteria for the design of dynamic restraints (snubbers) are provided in SRP Section 3.9.3. D. Acceptance criteria for the design and installation of safety and relief valves are provided in SRP Section 3.9.3. E. Acceptance criteria for the seismic and dynamic qualification of mechanical and electrical equipment are provided in SRP Section 3.10. F. As required by GDC 14, safety-related valves that are part of the RCPB should be designed and tested such that these valves will not experience any abnormal leakage, or increase in leakage, from their loading, as addressed in SRP Section 3.10.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	G. For new plant applications, dynamic restraints in safety-related systems must include provisions to allow access for IST program activities.					
3.9.6.2	Inservice Testing Program for Pumps A. The scope of the applicant's test program is acceptable if it includes all of the ASME Code Class 1, 2, and 3 pumps described in 10 CFR 50.55a(f) and Subsection ISTA-1100 of the OM Code and, in addition, includes pumps not categorized as ASME Code Class 1, 2, or 3 but which the staff considers to be safety-related. Since the pump test program is based on the detection of changes in the hydraulic and mechanical condition of a pump relative to a reference test specified in Subsection ISTB-3000 of the OM Code, the establishment of a set of reference values and a consistent test method are basic criteria of the program. B. The pump test program is acceptable if it meets the requirements for establishing reference values and the periodic testing schedule described in Subsection ISTB-3000 of the OM Code. Subsections ISTB-3000, ISTB-5000, and ISTB-6000 of the OM Code establish the allowable ranges of IST quantities (e.g., flow rates and pressure differential), corrective actions, and vibration tests. The pump test schedule is required to comply with these rules. C. The frequency of ISTs and test parameters are acceptable if the provisions of Subsection ISTB-3000 of the OM Code are met. D. The methods of measurement are acceptable if the test program meets the requirements of Subsection ISTB-5000 of the OM Code with regard to instruments, pressure measurements, rotational speed, vibration measurements,					

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	and flow measurements. E. The instruments are acceptable if they meet the accuracy and range requirements of Subsection ISTB-3500 of the OM Code. F. The duration of the test is acceptable if the provisions of Subsection ISTB-5000 of the OM Code are met.					
3.9.6.3	Inservice Testing Program for Valves A. To be acceptable, the SAR valve test list must contain all safety-related ASME Code Class 1, 2, and 3 valves required by 10 CFR 50.55a(f) and the OM Code, except those nonsafety-related valves excepted by Subsection ISTC-1200 of the OM Code. It should also include valves not categorized as ASME Code Class 1, 2, or 3 but which are safety related. The SAR valve list must include a valve categorization that complies with the provisions of Subsection ISTC-1300 of the OM Code. The SAR should list each specific valve to be tested under the rules of Subsection ISTA-1100 of the OM Code by type, valve identification number, code class, and valve category. B. The valve test procedures, acceptance criteria, and corrective actions are acceptable if the provisions of Subsection ISTC of the OM Code, as incorporated by reference in 10 CFR 50.55a, are met with regard to preservice and periodic inservice valve testing. C. Refer to the RG for additional acceptance criteria for specific valve or actuator types, and leak testing.					
3.9.6.4	Inservice Testing Program for Dynamic Restraints A. The IST program for dynamic restraints is acceptable if it					

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	meets the requirements of the ASME Code, Section XI, or the ASME OM Code as incorporated by reference in 10 CFR 50.55a. The IST program for dynamic restraints must comply with these provisions. B. In 10 CFR 50.55a(b)(3)(v), the regulations state that Subsection ISTD of the ASME OM Code, 1995 edition through the latest edition and addenda and incorporated by reference in 10 CFR 50.55a(b)(3), may be applied in place of the requirements for snubbers in the ASME Code, Section XI, IWF-5200(a) and (b) and IWF-5300(a) and (b), by making appropriate changes to technical specifications or licensee-controlled documents. The regulations also state that preservice and inservice examinations must be performed using the VT-3 visual examination method prescribed in IWA-2213. C. The FSAR should identify and tabulate all safety-related components that use snubbers in their support systems. The tabulation should include the following information: i. Identification of the systems and components in those systems that use snubbers ii. The number of snubbers used in each system and on components in that system iii. The type(s) of snubber (hydraulic or mechanical) and the corresponding supplier iv. Specification whether the snubber was constructed in accordance with the ASME Code, Section III, Subsection NF					

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	v. Statement whether the snubber is used as a shock, vibration, or dual purpose snubber vi. For snubbers identified as either dual purpose or vibration arrestor type, an indication of whether both snubber and component were evaluated for fatigue strength, the evaluation is performed under SRP Section 3.9.3 Appendix A. D. The applicant should provide assurance that all snubbers are properly installed before preoperational piping vibration and plant startup tests. The applicant may use visual observation of piping systems and measurement of thermal movements during plant startup tests to verify that snubbers are operable (not locked up). The piping preoperational vibration and plant startup test programs should discuss the provisions for such examinations and measurements as described in SRP Section 3.9.2. E. The applicant should discuss accessibility provisions for maintenance, inservice inspection and testing, and possible repair or replacement of snubbers consistent with the provisions of the applicable NRC standard technical specifications.					
3.9.6.5	Relief Requests and Proposed Alternatives A. The applicant should identify the component identified for which it requests relief: i. Name and number as given in SAR ii. Component functions iii. ASME Code, Section III, Code Class					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	iv. Valve category as defined in Subsection ISTC-1300 of the OM Code v. Pump group as defined in Subsection ISTB-2000 of the OM Code B. The applicant should identify the ASME OM Code requirement(s) from which it is requesting relief. C. The applicant should specify the basis under which it is requesting relief and then explain why complying with the OM Code is impractical. D. For alternatives to the OM Code requirements, the applicant should provide sufficient details to demonstrate that (1) the proposed alternative will provide an acceptable level of quality and safety, or (2) compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. E. The applicant should specify a schedule for the implementation of the relief request or alternative. F. The approval of relief requests or alternatives involves the following: i. Approval of relief for impractical code requirements Pursuant to 10 CFR 50.55a(f)(6)(i) for pumps and valves, and 10 CFR 50.55a(g)(6)(i) for dynamic restraints, the Commission may grant relief from impractical code requirements because of design limitations upon application by the applicant. The NRC					

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	will consider the burden on the applicant as a factor in its review and evaluation. ii. Approval of alternatives to the OM Code requirements Pursuant to 10 CFR 50.55a(a)(3), the staff may authorize alternatives to IST program requirements of the OM Code if the applicant has adequately demonstrated either of the following: (1) Proposed alternatives to the Code requirements or portions thereof will provide an acceptable level of quality and safety. (2) Compliance with the Code requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.					
3.9.6.6	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the preservice testing program, inservice testing program, inservice inspection program, and motor-operated valve testing program are reviewed in accordance with 10 CFR 50.55a(f), 10 CFR 50.55a(g) and 10 CFR 50.55a(b)(3)(ii). The implementation milestones for the specific programs are specified below and included as license conditions for preservice testing and motoroperated valve testing programs: A. Preservice testing program Per ASME OM Code, Subsection ISTA-2000, defines the preservice test period as the period of time following the completion of construction activities related to the					

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	component and before first electrical generation by nuclear heat. B. IST program Per ASME OM Code, Subsection ISTA-2000, prior to first electrical generation by nuclear heat C. Inservice inspection program related to dynamic restraints Per ASME Code, Section XI, IWA-2430(b), before placement of the plant into commercial service D. MOV program Per ASME OM, Subsection ISTA-2000, prior to first electrical generation by nuclear heat					
3.9.7 (08/1998)	Risk-Informed Inservice Testing of Pumps and Valves					
	Refer to the SRP for the detailed criteria					
3.9.8 (09/2003)	Risk-Informed Inservice Inspection of Piping					
II.1	Element 1: Define the Proposed Change to ISI Program The licensee's RI-ISI submittal should have defined the proposed changes to the ISI program in general terms. The licensee should have confirmed that the plant is designed and operated in accordance with the currently approved requirements and that the PRA used in support of their RI-ISI program submittal reflects the actual plant. The licensee should identify those aspects of the plant's licensing bases that may be affected by the proposed change, including, but not limited to, rules and regulations, FSAR, technical specifications, and licensing conditions. In addition, the licensee should identify any changes to commitments. The licensee's programs and procedures that guide future changes to					

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	the ISI program without prior NRC approval should provide for engineering analyses, internal reviews, and a degree of traceability consistent with the magnitude of the changes the licensee intends to make. The particular piping systems, segments, and welds that are affected by the change in the ISI program should be identified. Specific revisions to inspection scope, schedules, locations, and techniques should also be identified. In addition, plant systems and functions that rely on the affected piping should be identified. Industry and plant-specific experience with inspection program results should be obtained and characterization relative to the effectiveness of past inspections of the piping and the flaws that have been observed should be described					
II.2	Element 2: Engineering Analysis After the proposed changes to the licensee's ISI program have been defined, the licensee should conduct an engineering analysis of the proposed changes using a combination of traditional engineering analysis with supporting insights from a PRA. Regulatory Guides 1.174 and 1.178 provide guidance for the performance of this evaluation.					
II.2.1	Traditional Analysis The traditional engineering analyses conducted should assess whether the impact of the proposed ISI changes (individually and cumulatively) is consistent with the principles that defense in depth and adequate safety margins are maintained. The primary regulations governing ISI of piping are 10 CFR 50.55a and Appendix A to 10 CFR Part 50. The intent of these regulations is to maintain the structural integrity of piping in a nuclear power plant. The regulations reference other codes and requirements that define the elements of a defensein-depth philosophy to ensure the structural integrity of piping. For each of					

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	the regulations and licensing bases relevant to the ISI of piping, the licensee should ensure that the proposed changes to the ISI program do not deviate from the regulations and licensing bases. ASME B&PVC Section XI is referenced in 10 CFR 50.55a for the detailed requirements regarding piping ISI. The objective of the ISI requirements of the ASME Code has been to identify conditions, such as flaw indications, that are precursors to leaks and ruptures in pressure boundaries that may impact plant safety. The licensee should verify that the proposed changes to the ISI program meet or exceed the intent of ASME B&PVC Section XI to identify conditions that are precursors to leaks and ruptures and to provide plans for additional and more frequent inspections in response to detection of flaws and degradation mechanisms. The plans for additional inspections following detection of a flaw should be targeted toward locations with the same degradation mechanism that may have contributed to the unacceptable flaw development. The nuclear industry has implemented augmented inspection programs to address generic industrywide piping degradation problems such as IGSCC and FAC. The licensee should identify whether the proposed changes in the ISI program affect previous licensee commitments for augmented inspection programs for piping degradation problems such as IGSCC and FAC.					
II.2.2	Probabilistic Risk Assessment The quality of the PRA should be compatible with the safety implications of the ISI change being requested and the degree that the justification of the change request depends on the PRA analysis. Guidance relating the acceptable scope, level of detail, and quality of the PRA analysis based on the anticipated change in risk can be found in Regulatory Guide 1.174, in Section 2.2.3, “Quality of PRA Analysis,” and SRP Chapter 19.0, in Section					

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	III.2.2.4, "Quality of a PRA for Use in Risk-Informed Regulation." The PRA performed should realistically reflect the actual design, construction, and operational practices and reflect the impact of previous changes made to the approved requirements. All calculations using the PRA model should be performed correctly and in a manner that is consistent with accepted practices. Limitations and approximations in the PRA and the PRA techniques that can influence the interpretation of the results required to support the ISI application should be clearly described and appropriately addressed. Parameter uncertainty, model uncertainty, and completeness uncertainty should be addressed in accordance with the guidelines of Regulatory Guide 1.174. The programs and procedures regarding the long-term maintenance, update, and use of the PRA should be sufficient to ensure that any anticipated changes in the ISI program that do not require NRC notification or approval will always be based on an appropriately generated set of risk insights.					
II.2.2.1	Scope of Piping Systems The piping systems included in the RI-ISI program for the purpose of evaluating the impact of the proposed changes in the ISI program on total plant risk and for the purpose of screening to classify the safety significance of piping systems should be such that any proposed increases in CDF and risk are small and are consistent with the intent of the Commission's Safety Goal Policy Statement					
II.2.2.2	Piping Segments An acceptable method for modeling a run of a pipe in a PRA or to define its ISI requirements is to divide the pipe run into segments. Portions of piping within the piping systems that have the same consequences of failure should be systematically identified.					

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	Consequences of failure include an initiating event, loss of a particular train, loss of a system, or a combination thereof. The location of the piping in the plant, and whether inside or outside the containment, should be taken into account in defining piping segments. Piping sections subjected to the same degradation mechanism should be systematically identified. Most of the degradation mechanisms present in nuclear power plant piping are dependent on a combination of design characteristics, fabrication processes and practices, operating conditions, and service experience. The degradation mechanisms to be considered include, but may not be limited to, vibration fatigue, thermal fatigue, corrosion cracking, primary water stress corrosion cracking (PWSCC), IGSCC, microbiologically induced corrosion (MIC), erosion, cavitation, and FAC. Piping segments should be defined taking into account the potential degradation mechanism and the consequence of failure at any point in the segment. Segments with the same consequences but a different degradation mechanism may be combined for consequence characterization, but the development of the inspection program should explicitly address the different degradation mechanisms within such segments. In addition, consideration should be given to identifying distinct segment boundaries at locations of branching points such as flow splits or flow joining points, locations of size changes, isolation valve, motor-operated valves (MOVs) and air-operated valves (AOVs). Distinct segment boundaries should be defined if the break potential is expected to be significantly different for various portions of piping.					
II.2.2.3	Evaluating Pipe Failures with PRA The licensee's methodology should systematically use risk					

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	insights from the PRA and PRA results to characterize the impact of each segment's failure on the plant's risk. The characterization should allow for the determination of the relative safety significance of the different pipe segments and should support the final determination regarding the impact of implementing the program on plant risk. Generally, three or four primary system LOCA sizes and two steam line rupture locations that represent the spectrum of demands on the mitigating systems are modeled in PRAs. An internal events flooding analysis is also included in most PRAs performed in response to Generic Letter 88-20. Much of this analysis will be used as a basis for determining the consequence of pipe failures. The review should focus on the robustness of the above models and methods in the baseline PRA, as well as appropriate use of this information to investigate the impact of the change in risk that is due to RI-ISI implementation. One acceptable approach is to investigate the change in risk due to an ISI program change is based on developing the pipe elements' failure potentials into probabilities, and integrating these probabilities into the existing quantitative PRA framework. The contribution to risk from each piping element may be ranked and the safety significance of the element determined. An alternative acceptable approach is based on categorizing each segment's failure potential and the consequences of each segment's failures. These two elements of risk, failure potential and consequences, are then systematically combined to determine the safety significance of each segment.					
II.2.2.4	Piping Failure Potential The determination of the degradation mechanisms present at each weld within all pipe runs included in the scope of the					

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	submittal is central to the success of the ISI application. The process used to identify the degradation mechanism at each weld should be well defined, systematic and applied to all welds within the scope. The documentation and engineering evaluations upon which the process is based should be capable of supporting the identification of all applicable degradation mechanisms. The determination of failure potential of piping segments, either as a quantitative estimate or a categorization into groups, should be based on appropriate design, operational, and inspection parameters in conjunction with the identified degradation mechanisms. The evaluation should include a determination of whether the potential failure of each segment is best characterized as a demand failure while responding to a plant transient or an operational failure which causes a plant transient. When data analysis is utilized to develop a quantitative estimate, the data should be appropriate and complete. When elicitation of expert opinion is used in conjunction with, or in lieu of probabilistic fracture mechanics or data analysis, a systematic procedure should be developed for conducting such elicitation and a suitable team of experts should be selected and trained. When categorization based on the degradation mechanism is used, the justification for the relationship between the degradation mechanism and the assigned category should be appropriate and complete. The assessment of piping failure potential should take into account uncertainties. These uncertainties include, but are not limited to, design versus fabrication differences, variation in material properties and strength, the effect of various degradation and aging mechanisms, variation in steady-state and transient loads, availability and accuracy of plant operating history, availability of inspection and maintenance program data, and					

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	capabilities of analytical methods and models to predict realistic results. The methodology, process, and rationale used to determine the failure potential of piping segments should be reviewed and approved by the plant expert panel as part of its deliberations during the final classification of the safety significance of each segment. This process should be justified, documented, and included in the submittal. When computer codes are used to develop quantitative estimates, the techniques should be verified and validated against established industry codes.					
II.2.2.5	Consequences of Failure The impact on risk that is due to piping pressure boundary failure should consider both direct and indirect effects. Consideration of direct effects should include failures that cause initiating events or that disable single or multiple components, trains, or systems, or a combination of these effects. Indirect effects of pressure boundary failures that affect other systems, components, or piping segments, also referred to as spatial effects such as pipe whip, jet impingement, flooding, or consequential initiation of fire protection systems, should also be considered. The direct and indirect effects of pipe failures should be characterized to incorporate appropriate failure mechanisms and dependencies into the PRA model. The possibility of different leak sizes ranging from minor leaks to full rupture should be considered. In general, the leak size resulting in the most severe consequence should be selected to characterize the consequence for each segment. An acceptable method of incorporating pipe failures is to classify pipe failures as leaks, disabling leaks, and breaks. Each of these failure modes may be characterized with a different failure					

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	probability or potential and a corresponding potential for degrading system performance through direct or indirect effects or both. The time available for operator actions also depends on the break size, and this timing dependence should be recognized and incorporated into the analysis as appropriate.					
II.2.2.6	Risk Impact of ISI Changes The guidelines discussed in Regulatory Guide 1.174, in Section 2.2.5, “Comparison of PRA Results with the Acceptance Guidelines,” are applicable to ISI change requests. General guidance for reviewing the risk impact from changes to the current licensing basis can be found in SRP Chapter 19 in Section III.2.2.5, “Evaluation of Risk Impact.” The methods used to determine the piping failure potential, the piping failure consequence, and the impact of the change in the number of inspections should together provide confidence that any increase in CDF or risk is small and acceptable in accordance with Regulatory Guide 1.174 guidelines and consistent with the intent of the Commission's Safety Goal Policy Statement. Increase in risk caused by changes in the ISI program could arise from a decrease in the number of welds inspected, reduced efficiency from simplified weld inspections, or both. Decreases in risk could arise from inspecting welds not currently being inspected in the program, improved weld inspections, or both. The greater the potential risk increase that is due to the proposed change in the ISI program (e.g., the larger the reduction in the number of welds to be inspected and of replacements of detailed inspections with simplified inspections), the more rigorous and detailed the risk analyses needed. The licensee should demonstrate that principle four in Regulatory Guide 1.174 and Regulatory Guide 1.178 is met. Principle four states that proposed increases in CDF and risk should be small					

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	and consistent with the intent of the Commission's Safety Goal Policy Statement. A direct evaluation of the fulfillment of principle four may be based on: • Risk importance measures or bounding estimates capable of characterizing plant specific pipe element failure potential and consequences categories, • A systematic process to combine failure potential and consequence to determine pipe element safety significance, • Pipe segmentation and element inspection selection process that provides for changes in the ISI program based on the safety significance of the pipe element, and • A discussion and evaluation of the aggregate risk impact of the set of changes requested in the ISI program, including an evaluation of uncertainty indicating that the uncertainties do not invalidate the conclusions. Alternatively, principle four may be shown to be met by calculating the expected change in CDF and LERF. The expected change can be calculated using the baseline PRA and before change versus after change piping failure potential expressed as failure probabilities. An evaluation of the uncertainty in the results should be performed, which indicates that the uncertainties do not invalidate the conclusions.					
II.2.3	Integrated Decisionmaking The integrated decisionmaking must address all five key safety principles presented in Section I, "Areas of Review," in this SRP and should address each of the expectations discussed in Section 2, "An Acceptable Approach to Risk-Informed Decisionmaking," of Regulatory Guide 1.174. The integrated decisionmaking should also ensure that the proposed ISI program is consistent with the intent of each of the elements related to defense in depth and safety margins discussed in 2.2.1.1, "Defense in Depth," and					

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	2.2.1.2, "Safety Margins," of Regulatory Guide 1.174. The results of the different elements of the engineering analysis discussed in Sections I.2.1 and I.2.2 must be considered in an integrated decisionmaking process. For ISI application, traditional requirements are outlined in 10 CFR 50.55a and the General Design Criteria in Appendix A to 10 CFR Part 50. To be acceptable, the traditional engineering analysis should address all the relevant regulations and the licensing bases of the plant. The acceptability of the impact of the proposed change in the ISI program is based on the adequacy of the traditional engineering analysis, acceptable change in plant risk relative to the criteria, and the adequacy of the proposed implementation and performance monitoring plan. The intent of the ASME B&PVC to maintain integrity of reactor coolant system boundary by ISI should be preserved under the RI-ISI program. An acceptable approach for the risk ranking of piping segments and elements is the use of risk reduction worth (RRW), risk achievement worth (RAW), conditional core damage probability (CCDP), conditional large early release probability (CLERP), or other importance measures. RRW is a measure of the maximum possible reduction in total CDF or LERF due to pressure boundary failures in plant piping systems that can result from making a component perfectly reliable. RAW, CLERP, and CCDP characterize the increase in risk associated with the pressure boundary failure. The risk ranking methodology must be able to systematically identify all safety- significant pipe segments within the scope of the RI-ISI program. Guidelines for using risk importance measures to categorize SSCs with respect to safety significance can be found in Appendix A, "Use Of Risk-Importance Measures To Categorize Structures, Systems, and Components with Respect to Safety Significance," to Regulatory Guide 1.174.					

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	The classification of piping segments should be evaluated to determine whether any piping segment is inappropriately classified. Consideration should be given to the limitations resulting from the PRA structure, PRA scope, and risk-importance measures. Operational insights from previous inspection results, industry data on pipe failures, and Maintenance Rule impacts should also be taken into account. Piping that is subject to ISI under ASME B&PVC Section XI requirements but has no segments exceeding the piping segment screening criteria should be further reviewed. Each ASME Class coded system should have some segments inspected for defense-in-depth considerations. The criteria for determining how many structural elements should be selected for inspection should be based on the safety significance of the segment and the failure potential within that segment. The potential for pipe failure directly drives the need for selecting elements for inspection and the location within a segment to be inspected. The sampling program for the selection of the number of elements to be inspected should be fully justified. Guidelines for an acceptable methodology for selection of structural elements for inspection within pipe segments are provided in the Regulatory Guide 1.178. The intent of the ASME B&PVC to maintain integrity of the reactor coolant system boundary by ISI should be preserved under the RI-ISI program. Appropriate consideration should be given to implementation and performance monitoring strategies so that piping performance can be assessed under the proposed ISI program change to confirm the assumptions and analyses that were conducted to justify the ISI program change.					
II.3	Element 3: Implementation and Monitoring Programs					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Careful consideration should be given to implementation and performance-monitoring strategies. The primary goal of this element is to assess piping performance under the proposed RI-ISI program by establishing performance-monitoring strategies to confirm the assumptions and analyses that were conducted to justify the changes in the ISI program. As discussed in Regulatory Guide 1.178, performance monitoring encompasses feedback and modification of the RI-ISI program resulting from changes in plant design features, plant procedures, equipment performance, examination results, and individual plant and industry failure information. Inspection scope and examination methods for the RI-ISI program should provide an acceptable level of quality and safety as stipulated in 10 CFR 50.55a(a)(3)(i). Inspection strategies should ensure that failure mechanisms of concern have been addressed and that there is a sufficiently high probability of detecting damage before structural integrity is impacted. Safety significance of piping segments should be taken into account in defining the inspection scope for the RI-ISI program. Degradation mechanisms, postulated failure modes, and configuration of piping structural elements should be incorporated in the definition of the inspection scope and inspection locations. For piping segments that are included in the existing plant FAC or IGSCC (Category B-G) inspection programs, the inspection locations should be the same as in the existing programs. For segments not in these programs, inspection locations should be mainly based on specific degradation mechanism and industry as well as plant-specific cracking experience. Determination of inspection locations for segments with no known degradation mechanism but high failure consequence should be based on sensitized weld locations, stress concentration, geometric discontinuities, and terminal ends. Plantspecific pipe cracking					

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	experience should be considered in selecting inspection locations. To be acceptable, alternative examination methods should be specified to ensure structural integrity in cases where examination methods cannot be applied due to limitations, such as inaccessibility or radiation exposure hazards. System pressure tests and visual examination of ASME piping structural elements should continue to be performed regardless of whether the segments contain locations that have been classified as safety significant or low safety significant. Safety-significant non-Code piping should be treated as ASME Code Class piping for purposes of examination and pressure testing. The qualifications of nondestructive examination (NDE) personnel, processes, and equipment should be demonstrated to be in compliance with ASME B&PVC Section XI. The acceptance criteria for flaw evaluation should meet the requirements of ASME B&PVC Section XI. For inspections outside the scope of Section XI, the acceptance criteria should meet existing regulatory guidance applicable to those programs. The risk-informed inspection program should specify appropriate inspection intervals consistent with the relevant degradation rate if the data on the degradation mechanism suggest that an inspection interval shorter than that stated in the ASME Section XI is required. In such cases, inspection intervals should be sufficiently short so that degradation too small to be detected during one inspection does not grow to an unacceptable size before the next inspection is performed. Updates to the RI-ISI program should be performed at least on a 10-year interval basis to coincide with the ISI requirements in ASME B&PVC Section XI. Significant changes to the PRA model, plant design feature changes, plant procedure changes, and equipment performance changes should be included for review in					

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	the RI-ISI program update if needed to support the update. Leakage, flaws, or indications identified during scheduled RI-ISI program NDE examinations and system pressure tests should be evaluated as part of the RI-ISI program update. Periodic updates of RI-ISI programs should include individual plant as well as industry failure information. Appropriate modifications of the ISI plan should be developed if new or unexpected degradation mechanisms occur. The adequacy of the reliability of the implemented NDE methods should be monitored. The adequacy of NDE performance levels and inspection intervals along with the appropriateness of the selected ISI locations should be considered valid only if the ISI program is successful in detecting degradation before it leads to leakage or rupture of piping.					
3.10, Rev. 3 (03/2007)	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment					
3.10.1	The qualification of electrical equipment and its supports should meet the requirements and recommendations of American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) Std 344-1987 as endorsed by RG 1.100. (Subsequent revision to RG 1.100 will provide guidance with exceptions for use of Appendix QR-A of ASME QME-1-2007 for seismic qualification of active mechanical equipment and other qualifications of mechanical components, and IEEE Std 344-2004 for seismic qualification for Class 1 E equipment.) These documents are generally applicable to all types of equipment and should be used to the extent practicable for the qualification of mechanical equipment. Specifically, conformance to the following criteria should be demonstrated. A. Qualification for Equipment Functionality					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	i. Tests and analyses are required to confirm the functionality of all mechanical and electrical equipment during and after an earthquake of magnitude up to and including the OBE and SSE and for all static and dynamic loads from normal, anticipated operational occurrence, and accident conditions. Before SSE qualification, the applicant should demonstrate that the equipment can withstand the equivalent effect of five OBE excitations without loss of structural integrity. Analyses alone, without testing, are acceptable as a basis for qualification only if the necessary function of the equipment is ensured by its structural integrity alone. When complete testing is impractical, a combination of tests and analyses is acceptable. Equipment that has been previously qualified by means of tests and analyses equivalent to those described herein is acceptable provided that the applicant submits proper documentation of such tests and analyses. ii. Equipment should be tested in the operational condition. Functionality should be verified during and/or after the testing, as applicable to the equipment being tested. Loadings simulating those of plant normal operation, such as thermal and flow-induced loading, if any, should be concurrently superimposed upon the seismic and other pertinent dynamic loading to the extent practicable. iii. Response spectrum or time history methods should specify the characteristics of the required seismic and dynamic input motions. These characteristics, derived from the seismic and dynamic analyses of the					

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	structures or systems, should be representative of the input motions at the equipment mounting locations, except as noted in subsection II.2 (under SRP Acceptance Criteria) of this SRP Section. iv. For seismic and dynamic loads, the actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (i.e., the test response spectrum (TRS) should closely resemble and envelop the required response spectrum (RRS) over the critical frequency range). v. Since seismic and dynamic load excitation generally has a broad frequency content, multifrequency vibration input motion should be used. However, single-frequency input motion, such as sine beats, is acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), or the anticipated response of the equipment is adequately represented by one mode, or in the case of structural integrity assurance, the input has enough intensity and duration to produce sufficiently high levels of stress for such assurance. Components that have been previously tested to IEEE Std 344-1971 should be reevaluated to justify the appropriateness of the input motion used and requalified if necessary. vi. For the seismic and dynamic portion of the loads, the test input motion should be applied to one vertical axis					

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	and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously, unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally. Components that have been previously tested to IEEE Std 344-1971 should be requalified using biaxial test input motions unless the applicant provides justification for using a single-axis test input motion. vii. Dynamic coupling between the equipment and related systems, if any, such as connected piping and other mechanical components, should be considered. viii. The fixture design should simulate the actual service mounting and should not cause any extraneous dynamic coupling to the test item. ix. For pumps and valves, the loads imposed by the attached piping should be considered. To ensure functionality under combined loadings, the stresses resulting from the applied test loads should envelop the specified service stress limit for the intended function of the component. Stresses in valve					

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	bodies and pump casings should be limited to the particular material's elastic limit when the pump or valve is subject to the combination of normal operating loads, SSE, and other applicable dynamic loads. x. If the dynamic testing of a pump or valve assembly proves to be impracticable, static testing of the assembly is acceptable provided that the end loadings are conservatively applied and are equal to or greater than postulated event loads, all dynamic amplification effects are accounted for, the component is in the operating mode during and after the application of loads, and an adequate analysis is made to show the validity of the static application of loads. xi. The in situ application of vibratory devices to simulate the seismic and dynamic vibratory motions on a complex active device is acceptable to confirm the functionality of the device when the applicant shows that a meaningful test can be made in this way. xii. The test program may be based on selective testing of a representative number of components according to type, load level, size, and the like on a prototype basis. xiii. Selection of damping values for equipment to be qualified should be made in accordance with RG 1.61 and ANSI/IEEE Std 344-1987. Higher damping values may be used if justified by documented test data with proper identification of the source and mechanism.					

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	xiv. When complete testing is not practicable, the features listed below should be incorporated into a test and analysis functionality assurance program for pumps and valves. Similar programs can be developed for other types of equipment. (1) Simple and passive elements, such as valve and pump bodies and their related piping and supports, may be analyzed to confirm structural integrity under postulated event loadings. However, complex active devices such as pump motors, valve operators and gate or disk assemblies, and other electrical, mechanical, pneumatic, or hydraulic appurtenances which are vital to the pump or valve operation should be tested for functionality. (2) The following analyses are acceptable provided they are correlated to classical problems, elementary laboratory tests, or in situ tests: a. An analysis is performed to determine the vibratory input to the valve or pump. b. An analysis is performed to determine the system's natural frequencies and the movement of the pump or valve during the dynamic events. c. An analysis is performed to determine the pressure differential and the impact energy on					

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	a valve disc during a loss-of-coolant accident (LOCA) and to verify the design adequacy of the disc. d. An analysis is performed to determine the forcing functions of the axial and radial loads imposed on a pump rotor because of a LOCA, such that combined LOCA and vibratory effects on the shaft and rotor assembly can be evaluated. e. An analysis is performed to determine the speed of the pump shaft as a result of postulated events and to compare it with the design critical speed. f. An analysis is performed to verify the design adequacy of the wall thickness of valve and pump pressure retaining bodies. g. An analysis is performed to determine the natural frequencies of a pump shaft and rotor assembly to ascertain whether they are within the frequency range of the vibratory excitations. If the minimum natural frequency of the assembly is beyond the excitation frequencies, a static deflection analysis of the shaft is acceptable to account for dynamic effects. If the assembly's natural frequencies are close to the excitation frequencies, an acceptable dynamic analysis must be performed to determine the structural response of the assembly to the excitation frequencies.					

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	h. When analyses are used for qualification, the combination of multimodal and multidirectional responses should be made in accordance with RG 1.92. B. Design Adequacy of Supports i. Analyses or tests should be performed for all supports of mechanical and electrical equipment to ensure their structural capability. ii. The analytical results should include the required input motions to the mounted equipment as obtained and characterized in the manner stated in subsection II.1.A.iii above, and the combined stresses of the support structures should be in accordance with the criteria specified in SRP Section 3.9.3. iii. Supports should be tested with equipment installed or with a dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. If the equipment is installed in a nonoperational mode for the support test, the response in the test at the equipment mounting location should be monitored and characterized in the manner stated in subsection II.1.A.iii above. In such a case, equipment should be tested separately for functionality, and the actual input motion to the equipment in this test should be more conservative in amplitude and frequency content than the monitored response from the support test.					

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	iv. The criteria of subsections II.1.A.iii thru II.1.A.xiii above apply when tests are conducted on the equipment supports. C. Verification of Seismic and Dynamic Qualification. The seismic and dynamic qualification testing performed in accordance with ANSI/IEEE Std 344-1987, as endorsed by RG 1.100, Revision 2, as part of an overall qualification program should be performed in the sequence indicated in Section 6 of IEEE Std 323-1974 (endorsed with exceptions by RG 1.89).					
3.10.2	Instrumentation described in RG 1.97, including associated mountings, should be tested under appropriate seismic and dynamic loadings as described in the regulatory guide, thereby ensuring that the instruments will continue to monitor plant variables and systems after a seismic event and/or accident.					
3.10.3	If the applicant proposes qualification by an experience-based approach, the details of the experience database, including applicable implementation methods and procedures to ensure structural integrity and functionality of the in-scope mechanical and electrical equipment, must meet the functionality of equipment for the defined load condition as presented in paragraphs 1 and 2 above. Supporting documentation for equipment identified in the database should confirm that such equipment remained functional during and after an SSE and the equivalent effect of five postulated occurrences of OBE in combination with other relevant static and dynamic loads.					
3.10.4	GDC 1 and 10 CFR Part 50, Appendix B, Criteria XVII establish requirements for records concerning the qualification of equipment. To satisfy these requirements, complete and auditable records must be available, and the applicant must maintain them, for the life of the plant, at a central location. These files should describe the qualification method used for all					

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	equipment in sufficient detail to document the degree of compliance with the criteria of this SRP section. These records should be updated and kept current as equipment is replaced, further tested, or otherwise further qualified. The equipment qualification file should contain a list of all systems, equipment, and the equipment support structures, as defined in the second paragraph of subsection I of this SRP Section. The equipment list should identify which equipment is supplied by the nuclear steam supply system (NSSS) and which equipment is supplied by the balance of plant (BOP). The equipment qualification file should also include qualification summary data sheets for each piece of equipment (i.e., each mechanical and electrical component of each system) which summarize the component's qualification. These data sheets should include the following information: A. Identification of equipment, including vendor, model number, and location within each building. Valves that are part of the reactor coolant pressure boundary (RCPB) should be so identified. B. Physical description, including dimensions, weight, and field mounting condition, and identification of whether the equipment is pipe-, floor-, or wall-supported. C. A description of the equipment's function within the system. D. Identification of all design (functional) specifications and qualification reports and their locations. Functional specifications for active valve assemblies should conform to RG 1.148.					

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	E. Description of the required loads and their intensities for which the equipment must be qualified. F. If qualification by test, identification of the test methods and procedures, important test parameters, and a summary of the test results. G. If qualification by analysis, identification of the analysis methods and assumptions and comparisons between the calculated and allowable stresses and deflections for critical elements. H. If qualification by an experience-based approach, identification of the type of experience and the source of experience database. I. The natural frequency (or frequencies) of the equipment. J. Identification of whether the equipment may be affected by vibration fatigue cycle effects and a description of the methods and criteria used to qualify the equipment for such loading conditions. K. Indication of whether the equipment has met the qualification requirements. L. Availability for inspection (i.e., statement of whether the equipment is already installed). M. A compilation of the required response spectra (or time history) and corresponding damping for each seismic and dynamic load specified for the equipment together					

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	with all other loads considered in the qualification and the method of combining all loads.					
3.10.5	GDC 14 requires, in part, that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage. 10 CFR 50, Appendix A, GDC 30 further requires, in part, that components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. As discussed under acceptance criteria in SRP Section 3.9.6, to satisfy these requirements, the qualification program for valves that are part of the RCPB should include testing or testing and analyses demonstrating that these valves will not experience any leakage, or increase in leakage, as a result of any loading or combination of loadings for which the valves must be qualified.					
3.10.6	The implementation of the qualification program described above should be documented in the following ways: A. The preliminary safety analyses report (PSAR) or DC application should contain the following: i. A detailed description of NSSS and architect/engineer (A/E) practice followed in qualification, including criteria, methods, and procedures used in conducting testing and analysis, which demonstrate the extent of compliance with the criteria set forth in subsections II.1 thru II.5 above. ii. If equipment qualification by using earthquake experience data and/or test experience data is proposed, a detailed description of the experience database, including applicable implementation methods and procedures to					

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	ensure structural integrity and functionality of the in-scope mechanical and electrical equipment subject to the defined load condition as presented in paragraphs 1 and 2 above. Supporting documentation for equipment identified in the database should confirm that such equipment remained functional during and after an SSE and the equivalent effect of five postulated occurrences of OBE in combination with other relevant static and dynamic loads. Note: For electrical equipment earthquake and/or experience data should not be used without adequate justification. iii. Information regarding administrative control of component qualification, especially a description of the equipment qualification file, the handling of documentation, internal acceptance review procedures, identification of the scope of NSSS and A/E suppliers, and the procedures for interchange of information between NSSS, A/E, equipment vendors, and testing laboratories. B. In addition to the information contained in the PSAR, as revised, the final safety analyses report (FSAR) should contain the following: i. A list of all systems required to perform the functions defined in the second paragraph of subsection I of this SRP section. ii. A description of the results of any in-plant tests, such as in situ impedance tests, and any plans for operational tests which will be used to confirm the qualification of any item of equipment.					

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	C. The seismic qualification report (SQR) should contain the following: i. The list of systems required to perform the functions defined in the second paragraph of subsection I of this SRP section. ii. The list of equipment, and its supports, associated with each system and any other equipment required in accordance with the second paragraph of subsection I of this SRP section. iii. The summary data sheets for each piece of equipment (i.e., each component) listed. iv. A detailed description of the experience database similar to item II.6.A.ii above for in-scope equipment not covered in DC. D. COL applications should include the information described in subsections II.6.A, II.6.B, and II.6.C, as well as the following: i. A description of the environmental parameters applicable to the specific plant and its equipment qualification program. ii. Documentation to demonstrate that properly defined and enveloped seismic and dynamic input response spectra have been applied to the specific plant and its equipment qualification program.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
3.11, Rev. 3 (03/2007)	Environmental Qualification of Mechanical and Electrical Equipment					
3.11.1	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment," Revision 1, July 1981 provides staff positions applicable to existing plants for assessing the compliance of an environmental qualification program with 10 CFR 50.49. For future plants, Regulatory Guide 1.89 provides the principal guidance for implementing the requirements and criteria of 10 CFR 50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment. However, certain NUREG-0588 Category I guidance may be used if relevant guidance is not provided in Regulatory Guide 1.89. NUREG-0588 includes two sets of qualification criteria, Category I and Category II. Category I refers to IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1 E Equipment for Nuclear Power Generating Stations." Category I applies to plants whose CP SERs were dated after July 1, 1974. Category II refers to IEEE Std 323-1971, and is not applicable to any future plants.					
3.11.2	IEEE Std 323 contains the principles and criteria that are generic to the environmental qualification process. The following clarification related to the criteria in IEEE Std 323 should be considered. IEEE Std 323 requires that the service environment, including the installed configuration of the equipment, be considered as part of the qualification process. In meeting this requirement, the potential for flooding of electrical equipment that are installed above the flood level, but are subject to water and moisture intrusion, should be considered as part of environmental qualification. Operating experience (e.g., Information Notice 89-63) shows that electrical enclosures that are located above the flood level and are subject to water and moisture intrusion could result in submergence of electrical components inside the enclosures, if the enclosures do not have					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	drainage holes. The reviewer should confirm that equipment in such locations, whose design is such that water accumulation is possible, should have measures to preclude such accumulation (e.g., enclosure drain holes) or the affected equipment should be qualified for the anticipated submergence.					
3.11.3	Regulatory Guide 1.40, “Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants,” endorses IEEE Std 334, "IEEE Trial Use Guide for Type Tests of Continuous-Duty Class 1 Motors Installed Inside the Containment of Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental design and qualification of Class 1E motors, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental design and qualification of Continuous-Duty Class 1 E Motors.					
3.11.4	Regulatory Guide 1.63, “Electrical Penetration Assemblies in Containment Structures for Nuclear Power Plants,” endorses IEEE Std 317, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations." These documents contain general guidance that is acceptable to the staff for the environmental design and qualification of electrical penetration assemblies, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental design and qualification of electrical penetration assemblies.					
3.11.5	Regulatory Guide 1.73, “Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants,” endorses IEEE Std 382, "IEEE Trial Use Guide for Type Test of Class 1 E Electric Valve Operators for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental design and qualification of Class 1 E electric valve operators, and should be					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental design and qualification of Class 1 E electric valve operators.					
3.11.6	Regulatory Guide 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety in Nuclear Power Plants,” provides guidance for implementing the requirements and criteria of 10 CFR 50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment. Regulatory Guide 1.89 endorses the provisions of IEEE Std 323 as being acceptable to the staff, and provides supplementary guidance for satisfying the Commission's regulations regarding the environmental qualification of electrical equipment located in a harsh environment.					
3.11.7	Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," provides guidance acceptable to the staff for the environmental qualification of the post-accident monitoring equipment described in Subsection I, Item 1(f), of this SRP section, as well as instruments and controls for the equipment described in Subsection I, Items 1(a) to 1(e), of this SRP section. These criteria, as supplemented by those of Regulatory Guide 1.89, should be used to evaluate the environmental qualification of the I&C equipment.					
3.11.8	Draft Regulatory Guide 1.131, “Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants,” endorses IEEE Std 383, "Standard for Type Test of Class 1 E Electric Cables and Field Splices for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental qualification of Class 1 E electric cables and field splices, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental qualification of Class 1 E electric cables and field splices. Pending issuance of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	"Final" version, the Draft version of RG 1.131 may be used as guidance.					
3.11.9	Regulatory Guide 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants," endorses IEEE Std 572, "IEEE Standard for Qualification of Class 1 E Connection Assemblies for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental qualification of Class 1E connection assemblies, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental qualification of Class 1 E connection assemblies.					
3.11.10	Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants," endorses IEEE Std 535, "IEEE Standard for Qualification of Class 1 E Lead Storage Batteries for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental qualification of Class 1 E lead storage batteries, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental qualification of lead storage batteries.					
3.11.11	Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio- Frequency Interference in Safety-Related Instrumentation and Control Systems," provides guidance acceptable to the staff for determining electromagnetic compatibility for I&C equipment during service. These criteria, as supplemented by those of Regulatory Guide 1.89, should be used to evaluate the environmental design and qualification of safety-related I&C equipment. New digital systems and new advanced analog systems may require susceptibility testing for electromagnetic interference/radio-frequency interference (EMI/RFI) and power surges, if the environments are significant to the equipment being qualified. The functional descriptions of I&C equipment are provided in SRP Chapter 7.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
3.11.12	Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provides guidance acceptable to the staff for determining the radiation dose and dose rate for equipment during postulated accident conditions. These criteria, as supplemented by those of Regulatory Guide 1.89, should be used to evaluate the accident source term used in the environmental design and qualification of equipment important to safety. 10 CFR 50.67, "Accident Source Term," provides the requirements for licensees to revise the accident source term used in design basis radiological analyses for plants licensed prior to January 10, 1997. Radiation dose and dose rate used to determine the radiation environment for qualification of electrical and mechanical equipment must be based on an NRC staff-approved source term and methodology, as discussed in NUREG-0588 and as supplemented by Section II.B.2 of NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License,” or as discussed in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." The radiation environment must be based on the integrated effects of the normally expected radiation environment over the equipment's installed life, plus the effects associated with the most severe design basis event during or following which the equipment is required to remain functional. The effects of beta radiation must also be considered in the qualification process. The effects of radiation exposure due to recirculatory fluid must be considered for equipment located outside the containment.					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The staff's definition of what constitutes a mild radiation environment for electronic components, such as semiconductors or electronic components containing organic material, differs from that for other equipment. The staff's position, as stated in NUREG-1503, "Final SER ABWR, Chapter 3, Design of Structures, Components, Equipment, and Systems," and NUREG-1793, "Final SER AP1000, Chapter 3, Design of Structures, Components, Equipment, and Systems," is that a mild radiation environment for electronic equipment is a total integrated dose less than 10 Gy (1 E3 rad), and a mild radiation environment for other equipment is less than 100 Gy (1 E4 rad). Environmental qualification for electrical equipment located in a "Radiation harsh" environment (i.e., locations where radiation is the only harsh environmental condition) can be accomplished in accordance with 10 CFR 50.49(f)(4) using analysis of test data (from identical materials) combined with radiation test information (i.e., partial test data), and appropriate consideration of margin and aging effects for nonmetallic components/materials when sufficient documentation is available to preclude the need for a type test.					
3.11.13	The effects of chemical exposure must be addressed in the environmental qualification process. The concentration of chemicals used for qualification must be equivalent to, or more severe than that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling system initiation, or recirculation phase). If the chemical composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical environment that results from a single failure in the spray system must be assumed. If only demineralized water spray is used, then the effect of the demineralized water spray must be included in the equipment					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	qualification.					
3.11.14	Mechanical components must be designed to be compatible with postulated environmental conditions, including those associated with loss-of-coolant accidents (LOCAs). A process must be established to determine the suitability of materials, parts, and equipment needed for safety-related functions, and to verify that the design of such materials, parts, and equipment is adequate. Also, equipment records must be maintained, and these records must include the results of tests and material analyses used as part of the environmental design and qualification process for each component. For mechanical equipment, the staff concentrates its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). The reviewer confirms that the applicant has (1) identified safety-related mechanical equipment located in harsh environment areas, including its required operating time; (2) identified nonmetallic subcomponents of such equipment; (3) identified the environmental conditions and process parameters for which this equipment must be qualified; (4) identified nonmetallic material capabilities; and (5) evaluated environmental effects.					
3.11.15	For electrical and mechanical equipment located in a mild environment, acceptable environmental design can be demonstrated by the "design/purchase" specifications for the equipment. The specifications must contain a description of the functional requirements for a specific environmental zone during normal environmental conditions and anticipated operational occurrences. A well-supported maintenance/surveillance program, in conjunction with a good preventive maintenance program, is sufficient to ensure that equipment that meets the					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	design/purchase specifications is qualified for the designed life. Compliance with 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and associated guidance in Regulatory Guide 1.160 are sufficient to provide reasonable assurance that environmental considerations established during design are reviewed every refueling outage and maintained on a continuing basis to ensure that the qualified design life has not been reduced by thermal, radiation, and/or cyclic degradation resulting from unanticipated operational occurrences or service conditions. Modification to the replacement program and/or replacement of equipment should be based on the review of maintenance/surveillance data.					
3.11.16	For COL reviews, the description of the operational program and proposed implementation milestone(s) for the environmental qualification program are reviewed in accordance with 10 CFR 50.49. The implementation milestone for the environmental qualification program is to have all qualification requirements met prior to the loading of fuel. Implementation is required by a license condition.					
3.12 (03/2007)	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports					
3.12.1	A. Piping Analysis Methods i. Experimental Stress Analysis Methods If experimental stress analysis methods are used in lieu of analytical methods for Seismic Category I ASME Code and non-Code piping system designs, the applicant should provide sufficient information to show the validity of the design. It is recommended, prior to use of the experimental stress analysis methods, that details of the method as well as the scope and extent of					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	its application, be submitted for approval. The experimental stress analysis methods provided in Appendix II to ASME Code, Section III, Division 1 are applicable. ii. <u>Modal Response Spectrum Method</u> The SRP acceptance criteria provided in SRP Section 3.9.2, Subsection II.2 are applicable. iii. <u>Response Spectra Method- Independent Support Motion Method</u> This method may be used in lieu of the response spectra method when there is more than one supporting structure. The acceptance criteria provided in NUREG-1061, Volume 4 are applicable. iv. <u>Time History Method</u> The SRP acceptance criteria provided in SRP Section 3.7.2, Subsection II.6 are applicable. v. <u>Inelastic Analysis Method</u> If inelastic analysis methods are used for the piping design, the applicant will provide sufficient information to show the validity of the analysis. It is recommended, prior to use of the inelastic analysis method that details of the method, as well as the scope and extent of its application and acceptance criteria, be submitted for approval. The inelastic analysis methods provided in SRP Section 3.9.1,					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Subsection II.4 are applicable. vi. Small Bore Piping Method The SRP acceptance criteria provided in SRP Section 3.9.2, Subsection II.2(A) are applicable. vii. Nonseismic/Seismic Interaction (II/I) The acceptance criteria provided in Section 3.9.2, Subsection II.2.(K) are applicable. viii. Category I Buried Piping, Conduits, and Tunnels The acceptance criteria provided in SRP Section 3.7.3, Subsection II.12 are applicable. B. Piping Modeling Techniques i. Computer Codes The acceptance criteria provided in SRP Section 3.9.1, Subsection II.2 are applicable. ii. Dynamic Piping Model The acceptance criteria provided in SRP Section 3.9.2, Subsection II.2 are applicable. iii. Piping Benchmark Program The computer programs are benchmarked with the appropriate NRC benchmarks.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	iv. Decoupling Criteria The acceptance criteria provided in SRP Section 3.7.2, Subsection II.3(b) are applicable. C. Piping Stress Analysis Criteria i. Seismic Input The acceptance criteria provided in SRP Section 3.7.2 Subsection II.5 are applicable. ii. Design Transients The acceptance criteria provided in SRP Section 3.9.1, Subsection II.1 are applicable. iii. Loadings and Load Combinations The acceptance criteria provided in SRP Section 3.9.3, Subsection II.1 are applicable. iv. Damping Values The acceptance criteria provided in SRP Section 3.9.2, Subsection II.2(L) are applicable. v. Combination of Modal Responses The acceptance criteria provided in SRP Section 3.9.2, Subsection II.2(E) are applicable. vi. High-Frequency Modes					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The acceptance criteria provided in SRP Section 3.9.3, Subsection II.2 are applicable. vii. <u>Fatigue Evaluation for ASME Code Class 1 Piping</u> The acceptance criteria in Section III of the ASME Code are applicable. viii. <u>Fatigue Evaluation of ASME Code Class 2 and 3 Piping</u> The acceptance criteria for provided in Section III of the ASME Code are applicable. ix. <u>Thermal Oscillations in Piping Connected to the RCS</u> The operating experience insights contained in NRC Bulletin (BL) 88-08 and supplements are applicable for the identification and evaluation of piping systems susceptible to thermal stratification, cycling, and striping. x. Thermal Stratification The operating experience insights contained in NRC BL 79-13 and BL 88-11 are applicable for the identification and evaluation of long runs of horizontal piping susceptible to thermal stratification. xi. <u>Safety Relief Valve Design, Installation, and Testing</u> The acceptance criteria provided in SRP					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Section 3.9.3, Subsection II.2 are applicable. xii. Functional Capability The acceptance criteria provided in NUREG-1367, "Functional Capability of Piping Systems," may be used to ensure piping functionality under level D loading conditions. Alternative criteria will be reviewed on a case by case basis. xiii. Combination of Inertial and SAM Effects The acceptance criteria provided in SRP Section 3.9.2, Subsection II.2(G) are applicable for enveloped support motion analysis. The acceptance criteria provided in NUREG-1061, Volume 4 are applicable for independent support motion analysis. xiv. OBE as a Design Load Appendix S to 10 CFR Part 50, "Earthquake Engineering Criteria for Nuclear Power Plants," allows the use of operating basis earthquake ground motion. The criteria is provided in paragraph IV.(a)(2). The detail criteria for use of such an option was provided in NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, Section 3.1.1.2." xv. Welded Attachments Support members, connections, or attachments					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	welded to piping should be designed such that their failure under unanticipated loads does not cause failure at the pipe pressure boundary. The applicant may use Code Cases for the design of the welded attachments. Acceptable Code Cases are listed in RG 1.84. xvi. <u>Modal Damping for Composite Structures</u> The acceptance criteria provided in SRP Section 3.7.2, Subsection II.13 are applicable. xvii. <u>Temperature for Thermal Analyses</u> The applicant should perform thermal expansion analyses for piping systems that operate at temperatures above or below the stress-free reference temperature. The stress-free reference temperature for a piping system is typically defined as a temperature of 70°F. The applicant should provide justification if thermal expansion analyses are not performed. The justification will be reviewed on a case by case basis. xviii. <u>Intersystem LOCA</u> The acceptance criteria for the design of the piping system should be such that over pressurization of low-pressure piping systems due to RCPB isolation failure will not result in rupture of the low-pressure piping outside containment. The criteria provided in Staff Requirements Memoranda (SRM) dated June 26, 1990 in response to Commission Papers (SECY)-90-					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	016 dated January 12, 1990 are applicable. xix. <u>Effects of Environment on Fatigue Design</u> The guidance provided in Regulatory Guide 1.207[1] is applicable. D. <u>Piping Support Design</u> The piping system supports must provide adequate margins of safety to maintain the functionality of the piping components under all combinations of loadings. i. <u>Applicable Codes</u> The design of ASME Code, Section III, Class 1, 2, and 3, piping supports should comply with the design criteria requirements of ASME Code, Section III, Subsection NF. ii. <u>Jurisdictional Boundaries</u> The jurisdictional boundaries between pipe supports and interface attachment points should comply with ASME Code, Section III, Subsection NF. iii. <u>Loads and Load Combinations</u> The criteria provided in SRP Section 3.9.3, Subsection II.1 are applicable. iv. <u>Pipe Support Baseplate and Anchor Bolt Design</u> The design of the pipe support baseplates and anchor					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	bolts should comply with guidance provided in NRC BL 79-02, Revision 2. v. <u>Use of Energy Absorbers and Limit Stops</u> The evaluation typically consists of iterative response spectra analyses of the piping and support system. The analyses will be reviewed on a case by case basis. vi. <u>Use of Snubbers</u> The acceptance criteria provided in SRP Section 3.9.3, Subsection II.3 are applicable. vii. <u>Pipe Support Stiffness</u> The acceptance criteria provided in SRP Section 3.9.3, Subsection II.3 are applicable. viii. <u>Seismic Self-Weight Excitation</u> The acceptance criteria provided in SRP Section 3.9.3, are applicable for loads caused by the seismic excitation of the pipe support. ix. <u>Design of Supplementary Steel</u> The design of structural steel for use as pipe supports should comply with the ASME Code, Section III, Subsection NF. x. <u>Consideration of Friction Forces</u>					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The design of sliding type supports, such as guides or box supports, should include evaluation of the friction loads induced by the pipe on the support. The applicant should provide the friction coefficients used in the evaluation. The proposed friction coefficient will be reviewed on a case by case basis. xi. Pipe Support Gaps and Clearances Small gaps are generally provided for frame type supports. The gap allows for radial thermal expansion of the pipe and for pipe rotation. This gap must account for the diametrical expansion of the pipe due to temperature and pressure. The acceptance criteria for the minimum gap (total of opposing sides) between the pipe and the support and will be reviewed on a case by case basis. xii. Instrumentation Line Support Criteria The acceptance criteria provided in ASME Code, Section III, Subsection NF are applicable. xiii. Pipe Deflection Limits The allowable deflections of the piping at support locations resulting from design loadings should be controlled to ensure that the pipe deflections do not cause the failure of the supports. This criteria will be reviewed on a case by case basis. This criteria applies to following type of pipe supports: limit stops, snubbers, rods, hangers, and sway struts.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	NOTE: 1. Regulatory Guide 1.207 was issued for comment July 2006. Regulatory Guide 1.207 is being developed concurrent with the development of this SRP section.					
3.13 (03/2007)	Threaded Fasteners - ASME Code Class 1, 2, and 3					
	<u>Design Aspects</u> A. Materials Selection The selection of materials used for the design of threaded fasteners is acceptable if the ASME Code, Section III criteria shown in Table 3.13-1 of this SRP section are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems. B. Mechanical Testing, Special Process and Controls The criteria for mechanical property testing of threaded fastener materials are provided in the particular ASME Code Section II, Part A, specification under which the material was procured. The material heat treatment and tensile test coupon preparation criteria for threaded fasteners that are fabricated from ferritic materials (i.e., carbon steel, low-alloy steel, quenched and tempered steel) are acceptable if the ASME Code, Section III criteria shown in Table 3.13-1 are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems. The applicant should apply criteria of ASME Code Section III Subparagraphs NB-2200, NC-2200, ND-2200 rather than the criteria of the material specification applicable to the mechanical testing if there is a conflict between the two sets of criteria.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Lubricants and sealants in mechanical connections secured by threaded fasteners should be specified to ensure they are compatible with the threaded fasteners. Any mechanical joint using threaded fasteners should be designed to preclude galvanic corrosion. C. Fracture Toughness Requirements for Ferritic Materials The fracture toughness of ferritic bolts, studs, and nuts (i.e., made from either low-alloy steel or carbon steel materials) is acceptable if the ASME Code, Section III criteria shown in Table 3.13-1 are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems. Ferritic bolts, studs, and nuts (i.e., bolts, studs, and nuts made from either low-alloy steel or carbon steel materials) used in RCPB applications must also meet the fracture toughness requirements of 10 CFR Part 50, Appendix G. D. Fabrication Inspection The examination criteria for threaded fasteners are acceptable if the ASME Code, Section III criteria shown in Table 3.13-1 are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems. E. Quality Records The applicant should provide assurance that the CMTRs will be retained in accordance with the requirements of 10 CFR 50.70. The CMTR should identify the material specification for which the material was procured along with the associated material properties tests (including fracture toughness tests) and inspections that apply to the particular material specification.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Refer to the RG for Tables 3.13-1 and 3.13-2					
	<u>Preservice and Inservice Inspection Requirements</u> The preservice and inservice inspection provisions for mechanical joints are acceptable if the ASME Code, Section XI criteria shown in Table 3.13-2 are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems. For system pressure testing, the requirements of 10 CFR 50.55a(b)(2)(xxvii) for visual examination of certain insulated bolting or studs during system pressure testing should also be identified.					
Branch Technical Position 3-1, Rev. 2 (03/2007)	Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants					
BTP 3-1.1	The main steam line components of BWR plants should conform to the criteria listed in items 1 through 5 of the attached Table A-1. BWRs that do not include a main steam isolation valve leakage control system or main steam line shutoff valves and that credit fission product hold-up and retention in main steam piping and/or the condenser to address main steam isolation valve leakage in analyses of accident radiological consequences, should also conform to the criteria specified in item 6 of Table A-1. Figure A-1 illustrates acceptable quality group and seismic classifications for BWR main steam piping and components.					
Branch Technical Position 3-2, Rev. 2	Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
(03/2007)						
BPT 3-2.1	The main steam and feedwater system components of BWR/6 plants should be classified in accordance with Branch Technical Position (BTP) 3-1, or alternately, in accordance with the attached Table B-1. The classifications indicated are consistent with the guidelines currently specified in RG 1.26 and RG 1.29. As an additional criterion, a suitable interface restraint should be provided at the point of departure from the Class I structure where the interface exists between the safety and nonsafety-related portions of the MSL and MFL. A sketch is attached (Figure B-1) to clarify the specified alternate classification system.					
Branch Technical Position 3-3, Rev. 3 (03/2007)	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment					
BTP 3-3.1	Plant Arrangement Protection of essential systems and components1 against postulated piping failures in high or moderate energy fluid systems that operate during normal plant conditions and that are located outside of containment, should be provided by items a, b, or c below in order of their preference. a. Plant arrangements should separate fluid system piping from essential systems and components. Separation should be achieved by plant physical layouts that provide sufficient distances between essential systems and components and fluid system piping such that the environmental effects of any postulated piping failure therein cannot impair the integrity or operability of essential systems and components. The following considerations should also be made:					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(1) Even though portions of the main steam and feedwater lines meet the break exclusion requirements of item 2.A(ii) of BTP 3-4, they should be separated from essential equipment. Designers are cautioned to avoid concentrating essential equipment in the break exclusion zone. Essential equipment must be protected from the environmental effects of an assumed nonmechanistic longitudinal break of the main steam and feedwater lines. Each assumed nonmechanistic longitudinal break should have a cross sectional area of at least one square foot and should be postulated to occur at a location that has the greatest effect on essential equipment. (2) The main steam and feedwater lines should not be routed around or in the vicinity of the control room. b. Fluid system piping or portions thereof not satisfying the provisions of item B.1.a should be enclosed within structures or compartments designed to protect nearby essential systems and components. Alternatively, essential systems and components may be enclosed within structures or compartments designed to withstand the effects of postulated piping failures in nearby fluid systems. c. Plant arrangements or system features that do not satisfy the provisions of either item B.1.a or item B.1.b should be limited to those for which the above provisions are impractical because of the stage of design or construction of the plant; because the plant design is based upon that of an earlier plant accepted by the staff as a base plant under the Commission's standardization and replication policy; or for other substantive reasons such as particular design features of the fluid systems.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Such cases may arise, for example, (1) at interconnections between fluid systems and essential systems and components, or (2) in fluid systems having dual functions (i.e., required to operate during normal plant conditions as well as to shut ovided by designing or testing essential systems and components to withstand the environmental effects associated with postulated piping failures.					
BTP 3-3.2	Design Features a. Essential systems and components should be designed to meet the seismic design requirements of Regulatory Guide (RG) 1.29. b. Protective structures or compartments, and other protective measures should be designed in accordance with the following: Protective structures or compartments needed to implement Section B.1 should BTP 3-3-4 Revision 3 - March 2007 be designed to seismic Category I requirements. The protective structures should be designed to withstand the effects of a postulated piping failure (i.e., pipe whip, jet impingement, pressurization of compartments, water spray, and flooding, as appropriate) in combination with loadings associated with the design basis earthquake within the respective design load limits for structures. c. Fluid system piping in containment penetration areas should be designed to meet the break exclusion provisions contained in item 2.A(ii) of BTP 3-4. d. Piping classification as recommended by RG 1.26 should be maintained without change until beyond the outboard restraint. If the restraint is located at the isolation valve, a classification change at the valve interface is acceptable.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
BTP 3-3.3	Analyses and Effects of Postulated Piping Failures a. To show that the plant arrangement and design features provide the necessary protection of essential systems and components, piping failures should be postulated in accordance with BTP 3-4 and postulated to include fullcircumferential ruptures of non-seismic moderate energy piping (since BTP 3-4 only applies during normal conditions, not seismic events). Each longitudinal or circumferential break or leakage crack should be considered separately as a single postulated initial event occurring during normal plant conditions. An analysis should be made of the effects of each such event, taking into account the provisions BTP 3-4 and of the system and component operability considerations of item B.3.b. below. The effects of each postulated piping failure should be shown to result in offsite consequences within the guidelines of 10 CFR Part 100 and to meet the provisions of items B.3.c. and d. below. b. In analyzing the effects of postulated piping failures, the following assumptions should be made with regard to the operability of systems and components: (1) Offsite power should be assumed to be unavailable if a trip of the turbine-generator system or reactor protection system is a direct consequence of the postulated piping failure. Also, offsite power should be assumed unavailable following seismic events. (2) A single active component failure should be assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted in item B.3.b.(3) below. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	consequences of the piping failure, such as unit trip and loss of offsite power. (3) Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system (e.g., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the postulated piping failure), single active failures of components in the other train or trains of that system or other systems necessary to mitigate the consequences of the piping failure and shut down the reactor, need not be assumed provided the systems are designed to seismic Category I standards, are powered from both offsite and onsite sources, and are constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems. Examples of systems that may, in some plant designs, qualify as dual-purpose essential systems are service water systems, component cooling systems, and residual heat removal systems. (4) All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account should be taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions should be judged on the basis of ample time and adequate access to equipment being available for the proposed actions. For breaks in non-seismic piping systems, only seismically-qualified systems should be assumed to be available to mitigate the consequences of the failure since a seismic event may have caused the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	pipe break. c. The environmental effects of a postulated piping failure should not preclude habitability of the control room or access to surrounding areas important to the safe control of reactor operations needed to cope with the consequences of the piping failure. d. The functional capability of essential systems and components should be maintained after a failure of piping not designed to seismic Category I standards, assuming a concurrent single active failure. e. The considerations related to the leak-before-break approach should conform with the provisions of SRP Section 3.6.3.					
BTP 3-3.4	Implementation a. Designs of plants for which CP applications are tendered after July 1, 1975 should conform to the provisions of this position. b. Designs of plants for which CP applications are tendered after July 1, 1973 and before July 1, 1975 should conform to the provisions of either (a) the letter of July 12, 1973 from J. F. O'Leary, Appendix C to this position, or (b) this position, at the option of the applicants. c. Designs of plants for which CP applications were tendered before July 1, 1973 and operating licenses are issued after July 1, 1975 should follow the guidance provided in the December 1972 letter from A. Giambusso, Appendix B to this position and provide analyses of moderate energy lines made in conformance with Section B.3 of this position, as part of the operating license application for these plants to demonstrate that acceptable protection against the effects of piping failures outside containment has been provided. Alternately, this position may be used in its entirety as an acceptable basis for					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	this finding. For plants in this category for which CP are not issued as of February 1, 1975, acommitment by the applicant to either (a) follow the guidance of Appendix B and submit Section B.3 analyses of moderate energy lines with the plant final safety analysis report (FSAR), or (b) conform the plant design to the provisions of this position, should provide an acceptable basis for issuance of the construction permit with regard to effects of piping failures outside containment. d. Designs of plants for which OL are issued before July 1, 1975 are considered acceptable with regard to effects of piping failures outside containment on the basis of the analyses made and measures taken by applicants and licensees in response to the December 1972 letter from A. Giambusso, and the staff review and acceptance of these analyses and measures. For plants in this category for which the staff review and acceptance of protection against the effects of piping failures outside containment is not substantially complete as of February 1, 1975, a commitment by the applicant to carry out analyses according to Section B.3 of this position, to submit them for staff review, and to carry out any system modifications found necessary before extended operation of the plant at power levels above one-half the license power level, should provide an acceptable basis for issuance of the operating license.					
Branch Technical Position 3-4, Rev. 2 (03/2007)	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
BTP 3-4.A	A. High-Energy Fluid Systems Piping i. Fluid Systems Separated From Essential Systems and Components. For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of Branch Technical Position (BTP) 3-3, a review of the piping layout and plant arrangement drawings should clearly show that the effects of postulated piping breaks at any location are isolated or are physically remote from essential systems and components.[1] At the designer's option, break locations as determined from 2A(iii) of this position may be assumed for this purpose. ii. Fluid System Piping in Containment Penetration Areas. Breaks and cracks need not be postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves, provided they meet the design criteria of the ASME Code, Section III, Subarticle NE-1120, and the following additional design criteria: 1) The following design stress and fatigue limits should not be exceeded: For ASME Code, Section III, Class 1 Piping (a) The maximum stress range between any two load sets (includingthe zero load set) should not exceed 2.4 S_m and should be calculated[2] by Eq. (10) in ASME Code, Section III, NB-3653. If the calculated maximum stress range of Eq. (10) exceeds 2.4 S_m, the stress ranges calculated by both Eq. (12) and Eq. (13)					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	in Paragraph ASME Code, Section III, NB-3653 should meet the limit of 2.4 S_m. (b) The cumulative usage factor should be less than 0.1. (c) The maximum stress, as calculated by Eq. (9) in ASME Code, Section III, NB-3652 under the loadings resulting from a postulated piping failure beyond these portions of piping, should not exceed 2.25 S_m and 1.8 S_y, except that following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed and operability of the valves with such stresses is ensured in accordance with the criteria specified in SRP Section 3.9.3. Primary loads include those which are deflection-limited by whip restraints. For ASME Code, Section III, Class 2 Piping (d) The maximum stress ranges as calculated by the sum of Eqs.(9) and (10) in Paragraph NC-3653, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified in the system's design specification (i.e., sustained loads, occasional loads, and thermal expansion), including an OBE event (if applicable), should not exceed 0.8(1.8 S_h + S_A). The S_h and S_A are allowable					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III. (e) The maximum stress, as calculated by ASME Code, Section III, NC-3653, paragraph Eq. (9) under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping, should not exceed 2.25 S_h and 1.8 S_y. Primary loads include those which are deflection-limited by whip restraints. The exceptions permitted in (c) above may also be applied, provided that when the piping between the outboard isolation valve and the restraint is constructed in accordance with the Power Piping Code ANSI B31.1, the piping should either be of seamless construction with full radiography of all circumferential welds or all longitudinal and circumferential welds should be fully radiographed. 2) Welded attachments, for pipe supports or other purposes, to these portions of piping should be avoided, except where detailed stress analyses, or tests, are performed to demonstrate compliance with the limits of 2.A(ii)(1). 3) The number of circumferential and longitudinal piping welds and branch connections should be minimized. Where guard pipes are used, the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	enclosed portion of fluid system piping should be seamless construction and without circumferential welds unless specific access provisions are made to permit inservice volumetric examination of the longitudinal and circumferential welds. 4) The length of these portions of piping should be reduced to the minimum length practical. 5) The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe-whip restraints) should not need welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings may be used), except where such welds are 100% volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of 2.A(ii)(1). 6) Guard pipes provided for those portions of piping in the containment penetration areas should be constructed in accordance with the criteria of the ASME Code, Section III, Subsection NE, Class MC, where the guard pipe is part of the containment boundary. In addition, the entire guard pipe assembly should be designed to meet the following criteria and tests: (a) The design pressure and temperature should not be less than the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions. (b) The Level C stress limits in, ASME					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	Code, Section III, NE-3220 should not be exceeded under the loading associated with containment design pressure and temperature in combination with the safe shutdown earthquake. (c) Guard pipe assemblies should be subjected to a single pressure test at a pressure not less than its design pressure. (d) Guard pipe assemblies should not prevent the access necessary to conduct the inservice examination specified in 2.A(ii)(7). Inspection ports, if used, should not be located in that portion of the guard pipe through the annulus of dual barrier containment structures. 7) A 100% volumetric inservice examination of all pipe welds should be conducted during each inspection interval as defined in ASME Code, Section XI, IWA-2400. iii. Postulation of Pipe Breaks in Areas Other Than Containment Penetration (1) With the exceptions of those portions of piping identified in 2.A(ii), breaks in Class 1 piping (ASME Code, Section III) should be postulated at the following locations in each piping and branch run: (a) At terminal ends.[3] (b) At intermediate locations where the maximum stress range[2] as calculated by					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Eq. (10)and either Eq. (12) or Eq. (13) xceeds 2.4 S_m. (c) At intermediate locations where the cumulative usage factor exceeds 0.1. As a result of piping reanalysis, the highest stress locations may be shifted; however, the initially determined intermediate break locations need not be changed unless one of the following conditions exists: i. The dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe-whip restraints and jet shields. ii. A change is necessary in pipe parameters such as major differences in pipe size, wall thickness, and routing. (2) With the exceptions of those portions of piping identified in 2A(ii), breaks in Class 2 and 3 piping (ASME Code, Section III) should be postulated at the following locations in those portions of each piping and branch run: (a) At terminal ends. (b) At intermediate locations selected by one of the following criteria: i. At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	valve. Or, where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure. ii. At each location where stresses are calculated[2] by the sum of Eqs. (9) and (10) in NC/ND-3653 of ASME Code, Section III, to exceed 0.8 times the sum of the stress limits given in NC/ND-3653. As a result of piping reanalysis, due to differences between the design configuration and the as-built configuration, the highest stress locations may be shifted however, the initially determined intermediate break locations may be used unless redesign of the piping resulting in a change in pipe parameters (diameter, wall thickness, routing) is necessary, or the dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe-whip restraints and jet shields. (3) Breaks in seismically analyzed non-ASME Class piping are postulated according to the same criteria as for ASME Class 2 and 3 piping above.[4] (4) Applicable to (1), (2), and (3) above: If a structure separates a high-energy line from					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure, irrespective of the fact that the above criteria might not need such a break location to be postulated. (5) Safety-related equipment should be environmentally qualified in accordance with SRP Section 3.11. Appropriate pipe ruptures and leakage cracks (whichever controls) should be included in the design bases for environmental qualification of electrical and mechanical equipment both inside and outside the containment. iv. The designer should identify each piping run it considered in order to postulate the break locations pursuant to 2.A(iii) above. In complex systems such as those containing arrangements of headers and parallel piping running between headers, the designer should identify and include all such piping within a designated run in order to postulate the number of breaks pursuant to these criteria. v. With the exceptions of those portions of piping identified in 2.A(ii), leakage cracks should be postulated as follows: (1) For ASME Code, Section III, Class 1 piping, at axial locations where the calculated stress range[2] by Eq. (10) in NB-3653 exceeds 1.2 S(m). (2) For ASME Code, Section III, Class 2 and 3 or nonsafety-class (not ASME Class 1, 2, or 3) piping, at axial locations where the calculated stress2 by the sum of Eqs. (9) and (10) in					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	NC/ND-3653 exceeds 0.4 times the sum of the stress limits given in NC/ND-3653. (3) Nonsafety-class piping that has not been evaluated to obtain stress information should have leakage cracks postulated at axial locations that produce the most severe environmental effects. B. Moderate-Energy Fluid System Piping i. Fluid Systems Separated from Essential Systems and Components. For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of BTP 3-3, a review of the piping layout and plant arrangement drawings should clearly show that the effects of through-wall leakage cracks at any location in piping designed to seismic and nonseismic standards are isolated or physically remote from essential systems and components. ii. Fluid System Piping in Containment Penetration Areas. Leakage cracks need not be postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves, provided 1) they meet the criteria of the ASME Code, Section III, NE-1120, and 2) the stresses calculated[2] by the sum of Eqs. (9) and (10) in ASME Code, Section III, NC-3653 do not exceed 0.4 times the sum of the stress limits given in NC-3653. iii. Fluid Systems in Areas Other Than Containment Penetration. (1) Leakage cracks should be postulated in piping located adjacent to structures, systems, or components important to safety, except:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(a) Where exempted by 2.B(ii) or 2.B(iv), (b) For ASME Code, Section III, Class 1 piping, where the stress range calculated2 by Eq. (10) in NB-3653 is less than 1.2 S(m), and (c) For ASME Code, Section III, Class 2 or 3 and nonsafety-class piping, where the stresses calculated[2] by the sum of Eqs. (9) and (10) in NC/HD-3653 are less than 0.4 times the sum of the stress limits given in NC/ND-3653. (2) Leakage cracks, unless the piping system is exempted by (1) above, should be postulated at axial and circumferential locations that result in the most severe environmental consequences. (3) Leakage cracks should be postulated in fluid system piping designed to nonseismic standards as necessary to satisfy B.3.d of BTP 3-3. iv. Moderate-Energy Fluid Systems in Proximity to High-Energy Fluid Systems. Leakage cracks need not be postulated in moderate-energy fluid system piping located in an area in which a break in high-energy fluid system piping is postulated, provided such leakage cracks would not result in more limiting environmental conditions than the high-energy piping break. Where a postulated leakage crack in the moderate-energy fluid system piping results in more limiting environmental conditions than the break in proximate high-energy fluid system piping, the provisions of 2.B(iii) should be applied.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	v. Fluid Systems Qualifying as High-Energy or Moderate-Energy Systems. Through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems [5] that qualify as high-energy fluid systems for only a short operational period but qualify as moderate-energy fluid systems for the major operational period. C. Type of Breaks and Leakage Cracks in Fluid System Piping i. Circumferential Pipe Breaks The following circumferential breaks should be postulated individually in high-energy fluid system piping at the locations specified in 2.A of this position: (1) Circumferential breaks should be postulated in fluid system piping and branch runs exceeding a nominal pipe size of 1 inch, except where the maximum stress range2 exceeds the limits specified in 2.A(iii)(1) and 2A(iii)(2), but the circumferential stress range is at least 1.5 times the axial stress range. Instrument lines, as well as 1 inch and less nominal pipe or tubing size, should meet the provisions of Regulatory Guide 1.11. (2) Where break locations are selected without the benefit of stress calculations, breaks should be postulated at the piping welds to each fitting, valve, or welded attachment. (3) Circumferential breaks should be assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections					

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	unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading). (4) The dynamic force of the jet discharge at the break location should be based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge. (5) Pipe whipping should be assumed to occur in the plane defined by the piping geometry and configuration and to initiate pipe movement in the direction of the jet reaction. ii. Longitudinal Pipe Breaks The following longitudinal breaks should be postulated in high-energy fluid system piping at the locations of the circumferential breaks specified in 2C(i): (1) Longitudinal breaks in fluid system piping and branch runs should be postulated in nominal pipe sizes 4-inch and larger, except where the maximum stress range[2] exceeds the limits specified in 2.A(iii)(1) and 2.A(iii)(2), but the axial stress range is at least 1.5 times the circumferential stress range. (2) Longitudinal breaks need not be postulated at					

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	terminal ends. (3) Longitudinal breaks should be assumed to result in an axial split without pipe severance. Splits should be oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reactions cause out-of-plant bending of the piping configuration. Alternatively, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis). (4) The dynamic force of the fluid jet discharge should be based on a circular or elliptical (2D x ½D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge. (5) Piping movement should be assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis. iii. Leakage Cracks Leakage cracks should be postulated at those axial locations specified in 2.A(v) for high-energy fluid system piping and in those piping systems not exempted in					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	2.B(iii)(1) for moderate-energy fluid system piping. (1) Leakage cracks need not be postulated in 1-inch and smaller piping. (2) For high-energy fluid system piping, the leakage cracks should be postulated to be in those circumferential locations that result in the most severe environmental consequences. For moderate-energy fluid system piping, see 2.B(iii)(2). (3) Fluid flow from a leakage crack should be based on a circular opening of area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width. (4) The flow from the leakage crack should be assumed to result in an compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects should be determined on the basis of a conservatively estimated time period necessary to effect corrective actions.environment that wets all unprotected components within the					
	NOTES: 1. Systems and components necessary to shut down the reactor and mitigate the consequences of a postulated pipe rupture without offsite power. 2. For those loads and conditions for which Level A and Level B stress limits have been specified in the design specification (including the operating basis earthquake). 3. This is defined as the extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion					

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	and thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior. In piping runs that are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve), a terminal end of such a runs is the piping connection to this closed valve. 4. Note that, in addition, breaks in nonseismic (i.e., non-Category I) piping should be taken into account as described in Section II.2.k, "Interaction of Other Piping with Category I Piping," of SRP Section 3.9.2. 5. The operational period is considered "short" if the fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is about 2% of the time that the system operates as a moderate-energy fluid system (e.g., systems such as the reactor decay heat removal system qualify as moderate-energy fluid systems; however, systems such as auxiliary feedwater systems operated during PWR reactor startup, hot standby, or shutdown qualify as high-energy fluid systems).					
	CHAPTER 4, Reactor					
4.0	Reactor					
4.2 (Rev. 3, March 2007)	Fuel System Design					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
4.2.1	Design Bases					
	The fuel system design bases must reflect the four objectives described in Subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following paragraphs:					
	A. Fuel System Damage This subsection applies to normal operation, and Section 4.2 of the safety analysis report should contain the information to be reviewed. To meet the requirements of GDC 10, as it relates to SAFDLs for normal operation, including AOOs, fuel system damage criteria should be included for all known damage mechanisms. Fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. Note: Refer to the RG for the complete damage criteria. B. Fuel Rod Failure This subsection applies to normal operation, AOOs, and postulated accidents. Items 1.B.i through 1.B.iii below address failure mechanisms that are more limiting during normal operation; Section 4.2 of the safety analysis report should contain the information to be reviewed. To meet the requirements of (1) GDC 10 as it relates to SAFDLs					

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	for normal operation, including AOOs and (2) 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be provided for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although the staff recognizes that it is impossible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, the review must ensure that fuel does not fail as a result of specific causes during normal operation and AOOs. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis. Fuel rod failures can be caused by overheating, PCI, hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. When applicable, the fuel rod failure criteria should consider high burnup effects based on irradiated material properties data. Note: Refer to the RG for complete fuel failure criteria. C. Fuel Coolability This subsection applies to postulated accidents, and Chapter 15 of the safety analysis report will contain most of the information to be reviewed. Item 1.C.v below addresses the combined effects of two accidents, and Section 4.2 of the safety analysis report should include that information. To meet the requirements of GDC 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be provided for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. This subsection also addresses					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	control rod insertability criteria. Note: Refer to the RG for complete coolability criteria.					
4.2.2	Description and Design Drawings					
	The reviewer determines that the fuel system description and design drawings provide an accurate representation and supply the information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description: • Type and metallurgical state of the cladding • Cladding outside diameter • Cladding inside diameter • Cladding inside roughness • Pellet outside diameter • Pellet roughness • Pellet density • Pellet resintering data • Pellet length • Pellet dish dimensions • Pellet grain size and open porosity • Burnable poison content • Insulator pellet parameters • Fuel column length • Overall rod length • Rod internal void volume • Fill gas type and pressure • Sorbed gas composition and content • Spring and plug dimensions • Fissile enrichment • Equivalent hydraulic diameter					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	• Coolant pressure • Design-specific burnup limit • Control blade/rod descriptions, dimensions, and lifetime limits • Fit of control blade/rod interference with surrounding structure (e.g., channel box or guide tube) The following design drawings and dimensions are also necessary for an acceptable fuel system description: • Fuel assembly cross section • Fuel assembly outline • Fuel rod schematic • Spacer grid cross section • Guide tube and nozzle joint • Guide tube with respect to control rod dimensions • Control blade/rod assembly cross section • Control rod assembly outline • Control rod schematic • Burnable poison rod assembly cross section • Burnable poison rod assembly outline • Burnable poison rod schematic • Orifice and source assembly outline					
4.2.3	Design Evaluation:					
	The reviewer will evaluate the methods for demonstrating that the design bases are met. Methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in the safety analysis report by reference.					
	A. Operating Experience Operating experience with fuel systems of the same or similar design should be described, including the maximum burnup experience. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	testing and design analyses that were performed before gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner. B. Prototype Testing When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed, when practical, to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested before irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer: • Spacer grid structural tests • Control rod structural and performance tests • Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping) • Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, fuel rod fretting (should account for spacer spring relaxation), and assembly wear and life) In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The maximum burnup or fluence experience associated with such tests should also be reviewed and considered in relation to the specified maximum burnup or fluence limit for the new design. The following phenomena have been tested in this manner in new designs and will serve as a guide to the reviewer: • Fuel and burnable poison rod growth • Fuel rod bowing • Fuel rod, spacer grid, and channel box oxidation and hydride levels					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	• Fuel rod fretting • Fuel assembly growth • Fuel assembly bowing • Channel box wear and distortion • Fuel rod ridging (PCI) • Crud formation • Fuel rod integrity • Holddown spring relaxation • Spacer grid spring relaxation • Guide tube wear characteristics In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished before operation of the design's full core. The inability to perform in-reactor testing may result from an incompatibility of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see Subsection II.4 below). C. Analytical Predictions Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Note: Refer to the RG for the complete criteria.					
4.2.4	Testing, Inspection, and Surveillance Plans					
	Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel (See items A, B, and C below).					
	A. Testing and Inspection of New Fuel Testing and inspection plans for new fuel should verify cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Quality control reports should document the details of the manufacturer's testing and inspection programs and should be referenced and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	summarized in the safety analysis report. The program for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described. When the overall testing and inspection programs are essentially the same as those for previously approved plants, a statement to that effect should be made. In that case, the safety analysis report need not include program details, but an appropriate reference should be cited and a summary (tabular) should be presented. B. Online Fuel System Monitoring The applicant's online fuel rod failure detection methods should be reviewed. Both the sensitivity of the instruments and the applicant's commitment to use the instruments should be evaluated. NUREG-0401 and NUREG/CR-1380 evaluate several common detection methods and should be used in this review. Surveillance is also needed to assure that B4C control rods are not losing reactivity. Boron compounds are susceptible to leaching in the event of a cladding defect. Periodic reactivity worth tests such as those described in NUREG-0308 are acceptable. C. Postirradiation Surveillance A postirradiation fuel surveillance program should be described for each plant to acceptable program will depend on the history of the fuel design being considered (i.e., whether the proposed fuel design is the same as current operating fuel or incorporates new design features). For a fuel design similar to that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing, dimension changes, or crud deposition. The program should also					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	commit to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should address the disposition of failed fuel. In addition to the plant-specific surveillance program, a continuing fuel surveillance effort should exist for a given type, make, or class of fuel that can be suitably referenced by all plants using similar fuel. In the absence of such a generic program, the reviewer should expect more detail in the plant-specific program. For a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes should be described. This program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with the prototype testing discussed in Subsection II.3.B. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.					
4.2.A	Evaluation of Fuel Assembly Structural Response to Externally Applied Forces					
	Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. This appendix describes the review that should be performed of the fuel assembly structural response to seismic and LOCA loads. NUREG-0609, NUREG/CR-1018, NUREG/CR-1019, and NUREG/CR-1020 provide background material for this appendix.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Note: Refer to the RG for the complete criteria.					
4.2.B	Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents					
	This appendix provides the interim acceptance criteria and guidance for the reactivity-initiated accident (RIA). RIAs consist of postulated accidents which involve a sudden and rapid insertion of positive reactivity. These accident scenarios include a control rod ejection (CRE) for pressurized water reactors (PWRs) and a control rod drop accident (CRDA) for boiling water reactors (BWRs). The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion which promptly increases local core power. Fuel temperatures rapidly increase, prompting fuel pellet thermal expansion. The reactivity excursion is initially mitigated by Doppler feedback and delayed neutron effects followed by reactor trip. Standard Review Plan (SRP) Section 15.4.8 and 15.4.9 provide further detail on the CRE and CRDA respectively. The technical and regulatory basis of this interim criteria is documented in a memorandum dated January 19, 2007 (ADAMS ML070220400). Note: Refer to the RG for the complete criteria.	10				
4.3 (Rev. 3, March 2007)	Nuclear Design					
4.3.1	Power Densities and Power Distribution					
	There are no direct or explicit criteria for the power densities and power distributions allowed during (and at the limits of) normal operation, either steady-state or load following. These limits are determined from an integrated consideration of fuel limits (SAR Section 4.2), thermal limits (SAR Section 4.4), scram limits (SAR Chapter 7), and transient and accident analyses (SAR Chapter					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	15). The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 1204EC (2200EF) peak cladding temperature allowed for loss-of-coolant accidents (LOCAs), are not exceeded during design-basis accidents. Consideration must also be made to the effect of coolant temperatures and enthalpy on the fuel and cladding temperatures. The limiting power distributions are then determined such that the limits on power densities and peaking factors can be maintained in operation. These limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic scrams), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed.					
	The acceptance criteria in the area of power distribution are that the information presented should satisfactorily demonstrate that: A. A reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor, taking into account the analytical methods and data for the design calculations; uncertainty analyses and experimental comparisons presented for the design calculations; the sufficiency of design cases calculated covering times in cycle, rod positions, load-follow transients, etc.; and special problems such as power spikes due to densification, possible asymmetries, and misaligned rods. A reasonable probability exists that in normal operation the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation; the processing of that information, including calculations involved in the processing; the requirements for periodic check measurements; the accuracy of design calculations used in developing correlations when primary variables are not directly					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	measured; the uncertainty analyses for the information and processing system; and the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., tilt alarms for control rod misalignment).					
	Criteria for acceptable values and uses of uncertainties in operation, instrumentation numerical requirements, limit settings for alarms or scram frequency and extent of power distribution measurements, and use of ex-core and in-core instruments and related correlations and limits for offsets and tilts, all vary with reactor type. They can be found in staff safety evaluation reports and in appropriate sections of the technical specifications and accompanying bases for reactors similar to the reactor under review. The organization responsible for the review/assessment of nuclear design has enunciated Branch Technical Position 4-1 for Westinghouse reactors that employ constant axial offset control.					
	Acceptance criteria for power spike models can be found in a NUREG report on fuel densification, and are discussed in Regulatory Guide (RG) 1.126.					
	Generally, special or newly emphasized problems related to core power distributions will not be a direct part of normal reviews but will be handled in special generic reviews. Fuel densification effects and the related power spiking and the use of uncertainties in design limits are examples of these areas.					
4.3.2	Reactivity Coefficients					
	The only directly applicable GDC in the area of reactivity coefficients is GDC 11, which states "...the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity," and is considered to be satisfied in light water reactors (LWRs) by the existence of the Doppler and negative power coefficients. There are no criteria that explicitly establish acceptable ranges of coefficient values or preclude the acceptability of a positive moderator temperature					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	coefficient (MTC) such as may exist in PWRs at beginning of core life.					
	The acceptability of the coefficients in a particular case is determined in the reviews of the analyses in which they are used, e.g., control requirement analyses, stability analyses, and transient and accident analyses. The use of spatial effects such as weighting approximations as appropriate for individual transients are included in the analysis reviews. The judgement to be made under this SRP section is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment includes the use to be made of a coefficient, i.e., the analyses in which it is important; the state of the art for calculation of the coefficient; the uncertainty associated with such calculations, experimental checks of the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review.					
4.3.3	Control Rod Patterns and Reactivity Worths					
	Acceptance criteria relative to control rod patterns and reactivity worths include: A. The predicted control rod worths and reactivity insertion rates must be reasonable bounds to values that may occur in the reactor. These values are used in the transient and accident analyses and judgment as to the adequacy of the uncertainty allowances are made in the review of the transient and accident analyses. B. Equipment, operating limits, and procedures necessary to restrict potential rod worths or reactivity insertion rates should be shown to be capable of performing these functions. It is a position of the organization responsible for the review/assessment of nuclear design to require, where feasible, an alarm when any limit or restriction is violated or is about to be violated.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
4.3.4	Analytical Methods and Data					
	There are no specific criteria that must be met by the analytical methods or data that are used by an applicant or reactor vendor. In general, the analytical methods and database should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate representations of fuel designs in the reactor and encompass a sufficient range of variables and operating conditions.					
4.4 (Rev. 2, March 2007)	Thermal and Hydraulic Design					
4.4.1	Fuel Design Limits					
	One criterion provides assurance that there be at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience a DNB or transition condition during normal operation or AOOs. Uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculational methods used in the assessment of thermal margin should be treated with at least a 95-percent probability at the 95-percent confidence level. The assessment of thermal margin should also consider the uncertainties in instrumentation. The origin of each uncertainty parameter, such as fabrication uncertainty, computational uncertainty, or measurement uncertainty e.g., reactor power, coolant temperature, flow), should be identified. Each uncertainty parameter should be identified as statistical or deterministic and should clearly describe the methodologies used to combine uncertainties. Core design and operating changes for extended power uprates					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(EPUs) should be performed in a manner that ensures adequate safety margin. At a minimum, there should be a 95-percent probability at the 95-percent confidence level that a hot fuel rod in the reactor core will not experience a DNB or a transition condition during normal operation or AOOs. Specifically, this safety criterion should be satisfied while accounting for changes in radial and bundle power distribution, including any changes in critical heat flux ratio (CHFR) and CPR. The reviewer should confirm the adequacy of the flow-based average power range monitor flux trip and safety limit minimum critical power ratio at the uprated conditions (Review Standard RS-001). The reviewer should also ensure that the correlations used in the EPU analysis do not exceed their validation range under uprated normal operation and AOO conditions. The following are two examples of acceptable approaches to meeting this criterion: A. For departure from nucleate boiling ratio (DNBR), CHFR or CPR correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs. B. The limiting (minimum) value of DNBR, CHFR, or CPR correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a DNB or boiling transition during normal operation or AOOs. Correlations of critical heat flux are continually being revised as a result of additional experimental data, changes in fuel assembly design, and improved calculational techniques involving coolant mixing and the effect of axial power distributions.					
4.4.2	DNBR and CPR Limits					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Problems affecting DNBR or CPR limits, such as fuel densification or rod bowing, are accounted for by an appropriate design penalty which is determined experimentally or analytically. Subchannel hydraulic analysis codes, such as those described in "TEMPThermal Enthalpy Mixing Program," BAW-10021, Babcock and Wilcox Company, April 1970 and "THINC-IC-An Improved Program for Thermal-Hydraulic Analysis Of Rod Bundle Cores," WCAP-7956, Westinghouse Electric Corporation, June 1973, should be used to calculate local fluid conditions within fuel assemblies for use in PWR DNB correlations. The acceptability of such codes must be demonstrated by measurements made in large lattice experiments or power reactor cores. The review should include the effects of radial pressure gradients in the core flow distribution. The reviewer should also confirm that calculations of BWR fluid conditions for use in CHF correlations have been made in accordance with the models specified in "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, Appendix C, General Electric Company, April 1971 and "General Electric Company Analytical Model for Loss of Coolant Accident Analysis in Accordance with 10 CFR Part 50, Appendix K, "NEDO-20566, General Electric Company, November 1975.					
4.4.3	Core Oscillations and Thermal Hydraulic Instabilities					
	The design should address core oscillations and thermal-hydraulic instabilities as described in SRP Section 15.9.					
4.4.4	Methods for calculating single-phase and two-phase fluid flow in the reactor vessel					
	Methods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components should include classical fluid mechanics relationships and appropriate empirical					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	correlations. For components of unusual geometry, such as those listed below, these relationships should be confirmed empirically using representative databases from approved reports: A. Reactor vessel (“Reactor Vessel Model Flow Tests,” BAW-10037 (nonproprietary version of BAW-10012), Rev. 2, Babcock and Wilcox Company, September 1968). B. Jet pump (“Design and Performance of General Electric Boilng Water Reactor Jet Pumps,” APED-5460, General Electric Company, September 1968). C. Core flow distribution (BAW-10037 and “Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello,” NEDO-10299, General Electric Company, January 1971, DRAFT Rev. 2, April 1996). D. Void fraction distribution for BWRs.					
4.4.5	Proposed Technical Specifications					
	The proposed technical specifications should ensure that the plant can be safely operated at steady-state conditions under all expected combinations of system parameters. The safety limits and limiting safety settings must be established for each parameter, or combinations of parameters, to satisfy specific acceptance criterion 1, above.					
4.4.6	Preoperational and Initial Startup Test Programs					
	Preoperational and initial startup test programs should follow the recommendations of Regulatory Guide 1.68, as it relates to measurements and the confirmation of thermalhydraulic design aspects.					
4.4.7	Loose Parts Monitoring System					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The design description and proposed procedures for use of the loose parts monitoring system should be consistent with the requirements of Regulatory Guide 1.133.					
4.4.8	Effects of Crud					
	The thermal-hydraulic design should account for the effects of crud in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should assure the capability to detect a 3-percent drop in the reactor coolant flow. The flow should be monitored every 24 hours.					
4.4.9	Instrumentation					
	Instrumentation provided for an unambiguous indication of ICC, such as primary coolant saturation meters in PWRs, reactor vessel measurement systems, and core exit thermocouples, should meet the design requirements of TMI Action Plan Item II.F.2 of NUREG-0737. Applicants subject to 10 CFR 50.34(f) should meet the requirements of 10 CFR 50.34(f)(2)(xviii). Procedures for detection and recovery from conditions of ICC must be consistent with technical guidelines, including applicable EPGs developed pursuant to the TMI action plan, that incorporate response predictions based on appropriate analyses.					
4.4.10	Thermal-Hydraulic Stability During an ATWS					
	Thermal-hydraulic stability performance of the core during an ATWS event should not exceed acceptable fuel design limits. SRP Sections 15.8 and 15.9 describe an acceptable method for performing such an analysis for BWR and PWR cores.					
4.5.1 (Rev. 3, March 2007)	Control Rod Drive Structural Materials					
4.5.1.1	Materials Specifications					
	The properties of the materials selected for the CRDM should be equivalent to those of Section III, Appendix I, Division 1 of the ASME Code or Section II, Parts A, B, C, and D of the ASME					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Code. Cold-worked austenitic stainless steels should have a 0.2 percent offset yield strength no greater than 620 MPa (90,000 psi), to reduce the probability of stress corrosion cracking in these systems. Regulatory Guide (RG) 1.85 describes the acceptable code cases that may be used with these specifications.					
4.5.1.2	Austenitic Stainless Steel Components					
	Acceptance criteria include criteria described in SRP Section 5.2.3, Subsections II.4.D and E, and the criteria described below. RG 1.44 describes accepted methods for preventing intergranular corrosion of stainless steel components. Furnace-sensitized material should not be allowed, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling, storage, testing, and fabrication and for determining the degree of sensitization during welding. The controls for abrasive work on austenitic stainless steel surfaces should be adequate for preventing contamination that promotes stress corrosion cracking. The final surfaces should meet the acceptance standards specified in ASME NQA-1-1994 Edition, "Quality Assurance Requirements for Nuclear Facilities." Tools that contain materials that could contribute to stress-corrosion cracking or that, from previous usage, may be contaminated with such materials should not be used on austenitic stainless steel surfaces.					
4.5.1.3	Other Materials					
	All materials for use in this system should be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardening stainless steels should be specified for assurance that these materials will not deteriorate from stress corrosion cracking in service. Acceptable heat					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	treatment temperatures include aging at 565E - 595EC (1050E - 1100EF) for Type 17-4 PH and 565EC (1050EF) for Type 410 stainless steel.					
4.5.1.4	Cleaning and Cleanliness Control					
	Onsite cleaning and cleanliness control should be in accordance with ASME NQA-1-1994 edition. The oxygen content of the water in vented tanks need not be controlled. Vented tanks with deionized or demineralized water are normal sources of water for final cleaning or flushing of finished surfaces. Halogenated hydrocarbon cleaning agents should not be used.					
4.5.2 (Rev. 3, March 2007)	Reactor Internal and Core Support Structure Materials					
4.5.2.1	Materials Specifications					
	For core support structures and reactor internals, the permitted material specifications are those given in the ASME Code, Section III, Division 1, Sub-subarticle NG-2120. The properties of these materials are specified in Tables 2A, 2B and 4 of Section II of the Code. Additional permitted materials and their applications are identified in ASME Code Cases approved for use as described in Regulatory Guide 1.84, “Design, Fabrication, and Material Code Case Acceptability, ASME, Section III.”					
4.5.2.2	Controls on Welding					
	Methods and controls for core support structures and reactor internals welds shall be in accordance with ASME Code, Section III, Division 1, Article NG-4000. The examination requirements and acceptance criteria for these welds are specified in Article NG-5000.					
4.5.2.3	Nondestructive Examination					
	Nondestructive examinations shall be in accordance with the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	requirements of ASME Code, Section III, Division 1, Subarticle NG-2500. The nondestructive examination acceptance criteria shall be in accordance with the requirements of ASME Code, Section III, Division 1, Subarticle NG-5300.					
4.5.2.4	Austenitic Stainless Steels					
	The acceptance criteria for this area of review are given in SRP Section 5.2.3, subsections II.2 and II.4.a, b, d, and e. Regulatory Guide 1.44 provides acceptance criteria for preventing intergranular corrosion of stainless steel components. In conformance with this guide, furnace sensitized material should not be allowed. Methods described in this guide should be followed for cleaning and protecting austenitic stainless steel from contamination during handling, storage, testing, and fabrication, and for determining the degree of sensitization that occurs during welding.					
4.5.2.5	Other Materials					
	All materials used for reactor internals and core support structures must be selected for compatibility with the reactor coolant, as specified in Subsubarticles NG-2160 and NG-3120 of Section III, Division 1 of the ASME Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be specified to provide assurance that these materials will not deteriorate in service. Acceptable heat treatment temperatures are 565EC - 595EC (1050EF - 1100EF) for aging of Type 17-4 PH and 565EC (1050EF) for tempering of Type 410 stainless steel. Other materials shall have similar appropriate heat treat and fabrication controls in accordance with strength and compatibility requirements.					
4.6 (Rev. 2, March 2007)	Functional Design of Control Rod Drive System					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
4.6.1	Reactor Shutdown Capabilities					
	To meet the requirements of GDC 4, the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions and after postulated accidents.					
4.6.2	CRDS Failure					
	To meet the requirements of GDC 23, the CRDS should fail in an acceptable condition, even under adverse conditions, that prevents damage to the fuel cladding and excessive reactivity changes during failure.					
4.6.3	Reactivity Control Systems					
	To meet the requirements of GDC 25, the design of the reactivity control systems should assure that a single malfunction of the CRDS will not result in exceeding acceptable fuel design limits.					
4.6.4	Operational Control and Relibility					
	To meet the requirements of GDC 26, the CRDS should be capable of providing sufficient operational control and reliability during reactivity changes during normal operation and anticipated operational occurrences.					
4.6.5	Control of Reactivity Changes					
	To meet the requirements of GDC 27, the combined capability of CRDS and emergency core cooling system should reliably control the reactivity changes to assure the capability to cool the core under accident conditions.					
4.6.6	Reactivity Accidents					
	To meet the requirements of GDC 28, the CRDS should be designed to assure that reactivity accidents do not result in damage to the reactor coolant pressure boundary, or result in sufficient damage to the core or support structures so as to significantly impair coolability.					
4.6.7	Anticipated Operational Occurrences					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The CRDS should be designed to ensure an extremely high probability of functioning during anticipated operational occurrences to in conformance is GDC, 29.ticipated Operational Occurrences					
4.6.8	Alternate Rod Injection System					
	To meet the requirements of 10 CFR 50.62(c)(3), BWR plants should have an alternate rod injection system that is diverse and independent from the reactor trip system and should have redundant scram air header exhaust valves.					
	CHAPTER 5, Reactor Coolant System and Connected Systems					
5.2.1.1, Rev. 3 (03/2007)	Compliance With the Codes and Standards Rule, 10 CFR 50.55a					
5.2.1.1.1	Technical Rationale					
	The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:					
	1. Compliance with GDC 1 requires that components be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed. RG 1.26 provides quality group classifications for water-, steam-, and radioactive waste-containing components (pressure vessels, piping, pumps, valves, and storage tanks) commensurate with the importance of the safety functions they perform. For compliance with these quality group classifications, RCPB and other components containing radioactive materials must meet the requirements					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of ASME Code, Section III. These components will perform acceptably, commensurate with their intended safety functions, when designed in accordance with ASME Code requirements. The staff considers the requirements outlined in GDC 1 to be adequate for assurance that these components will perform acceptably, commensurate with the importance of their safety functions, in containing radioactive materials.					
5.2.1.1.2	2. 10 CFR 50.55a requires that components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions performed. 10 CFR 50.55a specifies that RCPB components and Quality Group B and C components (as defined in RG 1.26) must meet ASME Code, Section III requirements. These components will perform acceptably, commensurate with their intended safety functions, when designed in accordance with ASME Code requirements. The staff considers these requirements adequate for assurance that these components will perform acceptably, commensurate with the importance of their safety functions, in containing radioactive materials.					
5.2.1.2, Rev. 3 (03/2007)	Applicable Code Cases					
	Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1." This guide lists those Section III, Division 1, ASME Code Cases oriented to design, fabrication, materials, and testing, which are acceptable to the staff for implementation in the licensing of nuclear power plants.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." This guide lists those Section XI ASME Code Cases which are acceptable to the staff for use in the inservice inspection of components and their supports, as described in the first paragraph of subsection I, of this SRP. Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code." This guide lists ASME OM Code Cases oriented to operation and maintenance for nuclear power plant components which are acceptable to the staff for implementation in the licensing of nuclear power plants.					
	If the proposed Code Cases provide an acceptable level of quality and safety; or If compliance with the specified requirements of 10 CFR 50.55a would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.					
5.2.2, Rev. 3 (03/2007)	Overpressure Protection					
5.2.2.1	Material Specifications					
	The requirements of GDC 1, GDC 30, and 10 CFR 50.55a regarding quality standards are met for material specifications by compliance with the applicable provisions of the ASME Code and by acceptable application of material code cases, as described in Regulatory Guide 1.84. The specifications for permitted materials are identified in Appendix I to Section III of the ASME Code or described in detail in Parts A, B, and C of Section II of the ASME Code. Regulatory Guide 1.84 describes acceptable material code cases and guidelines for application in light-water-cooled nuclear power plants that may be used in conjunction with the above					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	specifications.					
5.2.2.2	Design Requirements for BWRs Operating at Power					
	A. For overpressure protection during power operation of the BWR reactor, the designs of the pilot-operated relief valves with auxiliary actuation devices, isolation condensers, or other pressure dissipation systems should have sufficient capacity to preclude actuation of safety valves during normal operational transients when assuming the following conditions at the plant: i. The reactor is operating at the licensed core thermal power level. ii. All system and core parameters have values within normal operating range that produce the highest anticipated pressure. iii. All components, instrumentation, and controls function normally. B. The design of safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe AOO with reactor scram, as specified by ASME Code Article NB-7000. Sufficient available margin should account for uncertainties in the design and operation of the plant, assuming the following: i. The reactor is operating at a power level that will produce the most severe overpressurization transient. ii. All system and core parameters have values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure. iii. The second safety-grade signal from the reactor					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	protection system initiates the reactor scram. iv. The discharge flow is based on the rated capacities specified in ASME III for each type of valve. v. The design of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe infrequent event, as specified by ASME Code Article NB-7000. C. A single malfunction or failure of an active component should not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power. Full credit is allowed for spring-loaded safety valves designed in accordance with the requirements of ASME Code Article NB-7511.1.					
5.2.2.3	Design Requirements for PWRs Operating at Power					
	A. For overpressure protection during power operation of the PWR reactor, the design of the PORVs or the pressurizer should have sufficient capacity to preclude actuation of safety valves during normal operational transients, when assuming the following conditions at the plant: i. The reactor is operating at the licensed core thermal power level. ii. All system and core parameters have values within normal operating range that produce the highest anticipated pressure.					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	iii. All components, instrumentation, and controls function normally. B. The designs of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe AOO with reactor scram, as specified by ASME Code Article NB-7000. Also, sufficient available margin should account for uncertainties in the design and operation of the plant assuming: i. The reactor is operating at a power level that will produce the most severe overpressurization transient. ii. All system and core parameters have values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure. iii. The second safety-grade signal from the reactor protection system initiates the reactor scram. iv. The discharge flow is based on the rated capacities specified in ASME Code Article NB-7000 for each type of valve. In addition, the designs of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe infrequent event, as specified by ASME Code Article NB-7000. C. A single malfunction or failure of an active component should not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions,					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	including loss of offsite power. Full credit is allowed for spring-loaded safety valves designed in accordance with the requirements of ASME Code Article NB-7511.1.					
5.2.2.4	Design Requirements for PWRs Operating at Low Temperature (Startup, Shutdown)					
	The design of the low-temperature overpressure protection (LTOP) system or the cold overpressure mitigation system (COMS) should be in accordance with the requirements of Branch Technical Position (BTP) 5-2. The LTOP system or COMS should be operable during startup and shutdown conditions below the enable temperature defined in paragraph II.2 of BTP 5-2.					
5.2.2.5	Testing and Inspections					
	The performance of tests and inspections should occur before operation and during startup to functionally demonstrate that the overpressure protection system, as installed, meets all design requirements.					
5.2.2.6	Technical Specifications					
	The technical specifications should specify appropriate limiting conditions of operation and inservice surveillance to ensure continued system reliability, including, for PWRs, specific limiting conditions of operation and testing of the LTOP system as specified in NUREG-1430 through NUREG-1434, Generic Letters No. 82-16, 83-02, and 90-06.					
5.2.2.7	TMI Action Plan Requirements					
	Section II.D.1 of the TMI Action Plan requires an applicant submit a plant specific report regarding relief valve (RV) and safety valve (SV) testing. Section II.D.3 of the TMI Action Plan requires that RVs and SVs be provided with direct valve position indication. Generic Letters No. 82-16 and 83-02 requires sections II.D.1 and					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	II.D.3 be covered by technical specifications while NUREG –0737 section II.K.3.3 specifies reporting for section II.D.1 and II.D.3.					
5.2.3, Rev. 3 (03/2007)	Reactor Coolant Pressure Boundary Materials					
5.2.3.1	Material Specifications					
	The requirements of GDC 1, GDC 30, and § 50.55a regarding quality standards are met for material specifications by compliance with the applicable provisions of the ASME Code and by acceptable application of materials Code Cases as described in Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III." The specifications for permitted materials are those identified in the ASME Code, Section III, Appendix I, or described in detail in the ASME Code, Section II, "Materials, Parts A, B, and C. Regulatory Guide 1.84 describes acceptable materials Code Cases and guidelines for their application in light-water-cooled nuclear power plants that may be used in conjunction with the above specifications. Staff positions related to BWR piping materials and materials processing are described in Attachment A to Generic Letter 88-01. The technical bases for the positions provided in Generic Letter 88-01 and similar recommendations related to minimizing stress corrosion cracking in susceptible piping of BWRs are detailed in NUREG-0313.					
5.2.3.2	Compatibility of Materials with the Reactor Coolant					
	The requirements of GDC 4 relative to compatibility of components with environmental conditions are met by compliance with the applicable provisions of the ASME Code and by compliance with the positions of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Ferritic low alloy steels and carbon steels, which are used in many principal pressure-retaining components, are clad with a layer of austenitic stainless steel. If cladding is not used, conservative corrosion allowances must be indicated for all exposed surfaces of carbon and low alloy steels, as indicated in the ASME Code, Section III, NB-3121, "Corrosion." Regulatory Guide 1.44 contains staff positions related to unstabilized austenitic stainless steel of the AISI Type 3XX series used for components of the RCPB. Positions related to BWR piping materials, including verification of nonsensitization of the material by an approved test, are described in Attachment A to Generic Letter 88-01. The technical bases for the positions provided in Generic Letter 88-01 and similar recommendations related to minimizing stress corrosion cracking in susceptible piping of BWRs are detailed in NUREG-0313, Revision 2.					
5.2.3.3	Fabrication and Processing of Ferritic Materials					
	A. The acceptance criteria for fracture toughness are the requirements of Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50. These criteria satisfy the requirements of GDC 14 and GDC 31 regarding prevention of fracture of the RCPB. Appendix G requires that the pressure-retaining components of the RCPB that are made of ferritic materials shall meet the requirements for fracture toughness anticipated operational occurrences. With respect to absorbed energy in J (ft-lbs) and lateral expansion as shown by Charpy V-notch (C_v) impact tests, all materials shall meet the acceptance standards of Article NB-2300 of the Code, Section III, and the requirements of Sections IV of Appendix G, 10 CFR Part					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	50, as follows: (1) Materials for piping (i.e., pipes, tubes, and fittings), pumps, and valves, excluding bolting materials, shall meet the requirements of the Code, Section III, Paragraph NB-2331 or NB-2332 (as applicable based upon thickness), and Appendix G, Paragraph G-3100 to the Code, Section III. The required Cv values for piping, pumps, and valves are specified in Table NB-2332(a)-1 of the Code, Section III. (2) Materials for bolting for which impact tests are required shall meet the requirements of the Code, Section III, Paragraph NB-2333. (3) Calibration of instruments and equipment shall meet the requirements of the Code, Section III, Paragraph NB-2360. The special acceptance requirements and staff positions for fracture toughness of reactor vessels are covered by SRP Section 5.3.1. B. The acceptance criteria for control of ferritic steel welding are based upon the following regulatory guides and ASME Code provisions to satisfy the quality standards requirements of GDC 1, GDC 30, and § 50.55a: (1) The amount of specified preheat must be in accordance with the requirements of the Code, Section III, Appendix D, Paragraph D-1210. These requirements are supplemented by positions described in Regulatory Guide 1.50, “Control of Preheat Temperature for Welding of Low Alloy Steel.” The supplemental acceptance criteria for control of preheat temperature are as follows:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(a) According to the welding procedure qualification minimum preheat and maximum interpass temperatures should be specified and the welding procedure should be qualified at the minimum preheat temperature. For production welds, the preheat temperature should be maintained until a post-weld heat treatment has been performed. (b) Production welding should be monitored to verify that the limits on preheat and interpass temperatures are maintained. In the event that the above criteria are not met, the weld is subject to rejection. (2) The acceptance criteria for electroslag welds are presented in Regulatory Guide 1.34, "Control of Electroslag Weld Properties." These criteria specify acceptable solidification patterns and impact test limits (for qualification of welds in Class 1 and Class 2 components) and the criteria for verifying conformance during production welding. (3) Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," provides the following criteria for requalification of welders: the performance qualification should require testing of the welder when conditions of accessibility to a production weld are less than 30 to 35 cm (12-14 inches) in any direction from the joint; and requalification is required for different restricted accessibility conditions or when any of the essential variables listed in the Code, Section IX, "Welding and Brazing Qualifications" are changed.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Qualification of the welder or welding operators for limited accessibility may be waived provided that 100% radiographic and/or ultrasonic examination of the completed welded joint is performed. Examination procedures and acceptance standards should meet the requirements of the ASME Section III of the Code. Records of the examination reports and radiographs should be retained and made part of the Quality Assurance Documentation for the completed weld. (4) Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," provides criteria to limit the occurrence of underclad cracking in low-alloy steel safety-related components clad with stainless steel. According to these criteria, material known to have susceptibility to underclad cracking should not be weld clad by high-heat-input welding processes and should be qualified for use to demonstrate that underclad cracking is not induced. C. For nondestructive examination of ferritic steel tubular products, the requirements of GDC 1, GDC 30, and § 50.55a regarding quality standards are met by compliance with the applicable provisions of the ASME Code. The acceptance criteria are given in Section III of the Code, Paragraphs NB-2550 through NB-2570.					
5.2.3.4	Fabrication and Processing of Austenitic Stainless Steel					
	A. The requirements of GDC 4 relative to compatibility of components with environmental conditions are met with measures to avoid sensitization in austenitic stainless steels. The acceptance criteria for testing, alloy					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	compositions, and heat treatment, to avoid sensitization in austenitic stainless steels, are covered in Regulatory Guide 1.44 and additional criteria for BWRs are specified in Attachment A to Generic Letter 88-01 based upon the technical information provided in NUREG-0313, Revision 2. Similar recommendations related to minimizing stress corrosion cracking in susceptible piping of BWRs are described in NUREG-0313, Revision 2. Regulatory Guide 1.44 also identifies acceptable methods for verification of non-sensitization of austenitic stainless steel materials and qualification of welding processes employed in production including testing using ASTM A-262 Practice A or E or another method which can be demonstrated to show non-sensitization. Alternative tests that have been previously accepted, based upon the adequacy of justifications presented and circumstances of proposed use, include the use of ASTM A-708. B. The requirements of GDC 4 relative to compatibility of components with environmental conditions are met with additional controls to avoid stress corrosion cracking in austenitic stainless steels. These controls consist of acceptance criteria on prevention of contamination, cleaning, and upper limit on yield strength. Additional controls for avoiding stress corrosion cracking are applied to BWRs as described below. Controls to avoid stress corrosion cracking in austenitic stainless steels are also covered in Regulatory Guide 1.44. This guide provides acceptance criteria on the cleaning and protection of the material against contaminants capable of causing stress corrosion cracking. Acid pickling is to be avoided on fabricated stainless steels. Necessary pickling is					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	to be done only with appropriate controls. Pickling should not be performed upon sensitized stainless steels. The quality of water used for final cleaning or flushing of finished surfaces during installation should be in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." Vented tanks with deionized or demineralized water are an acceptable source of water for final cleaning or flushing of finished surfaces. The oxygen content of the water need not be controlled. The controls for abrasive work on austenitic stainless steel surfaces should, as a minimum, be equivalent to the controls described in Regulatory Guide 1.37 position C.5 to prevent contamination which promotes stress corrosion cracking. Tools which contain materials that could contribute to intergranular or stress corrosion cracking or which, because of previous usage, may have become contaminated with such materials, should not be used on austenitic stainless steel surfaces. Laboratory stress corrosion tests and service experience provide the basis for the criterion that cold-worked austenitic stainless steels used in the reactor coolant pressure boundary should have an upper limit on the yield strength of 620 MPa (90,000 psi). Additional controls, beyond those described above, are warranted to avoid intergranular stress corrosion cracking (IGSCC) in and near welds in BWR austenitic stainless steel piping. The affected piping and the additional controls are described in Attachment A to Generic Letter 88-01 or					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	NUREG-0313. These controls include material and weldment specifications for IGSCC resistant materials, processing techniques, categorization of the IGSCC resistance of installations based upon material properties, treatment history, and post-weld treatments. The technical bases for these controls are described in NUREG-0313. C. The acceptance criteria for compatibility of austenitic stainless steel with thermal insulation are based on Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," to satisfy GDC 14 and 31 relative to prevention of failure of the RCPB. The compatibility of austenitic stainless steel materials with thermal insulation is dependent upon the type of insulation. The thermal insulation is acceptable if either reflective metal insulation is employed or a nonmetallic insulation which meets the criteria of Regulatory Guide 1.36 is used. The acceptance criteria for nonmetallic insulation for stainless steel are based on the levels of leachable contaminants in the material and are presented in position C.2.b and Figure 1 of the guide. D. The acceptance criteria for control of welding of austenitic stainless steels are based on NUREG-0313 as described below and on Regulatory Guides 1.31, 1.34, and 1.71, to satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a. The acceptance criteria for delta ferrite in austenitic stainless steel welds are given in Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." These acceptance criteria cover (1) verification of delta ferrite content of filler metals, (2) ferrite measurement, (3) instrumentation, (4) acceptability of test results, and (5)					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	documentation of weld pad verification tests. For the BWR austenitic stainless steel RCPB piping specified in Generic Letter 88-01, the weld metal ferrite content should be controlled as described in the positions of Attachment A to Generic Letter 88-01 or the recommendations of NUREG-0313, Revision 2. The acceptance criteria for electroslag welds in austenitic stainless steel are given in Regulatory Guide 1.34, “Control of Electroslag Weld Properties.” These criteria specify acceptable solidification patterns for qualification of austenitic stainless steel welds and the basis for verifying conformance during production welding. Regulatory Guide 1.71 provides the following criteria for requalification of welders: (1) The performance qualification should require testing of the welder when conditions of accessibility to a production weld are less than 30 to 35 cm (12-14 inches) in any direction from the joint. (2) Requalification should be required for different restricted accessibility conditions or when other essential variables listed in the Code, Section IX, are changed. An alternate acceptance criterion is as stated in Subsection II.3.B of this SRP section. E. For nondestructive examination of austenitic stainless steel tubular products, the quality standards requirements of GDC 1, GDC 30, and § 50.55a are met by compliance with the applicable provisions of the ASME Code. The acceptance criteria are given in Section III of the Code, Paragraphs NB-2550 through NB-2570.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	G. Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Inservice Inspection and Inservice Testing Programs are reviewed in accordance with 10 CFR 50.55a(g) and 10 CFR 50, Appendix A. The implementation milestones in the Inservice Inspection and Inservice Testing Programs are identified under SRP Section 5.2.4.					
5.2.4, Rev. 2 (03/2007)	Reactor Coolant Pressure Boundary Inservice Inspection and Testing					
5.2.4.1	System Boundary Subject to Inspection.					
	The applicant's or licensee's definition of the RCPB is acceptable if it is in agreement with the following criteria: for pressurized water reactor (PWR) and boiling water reactor (BWR) nuclear power systems, the inspection requirements of 10 CFR 50.55a, as detailed in Section XI of the ASME Code, must be met for all Class 1 pressure-containing components (and their supports). The system boundary, as defined in 10 CFR 50.2, includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant system, up to and including:					
	A. The outermost containment isolation valve in system piping that penetrates the primary reactor containment. B. The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment. C. The reactor coolant system safety and relief valves.					
5.2.4.2	Accessibility					
	The design and arrangement of system components are					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	acceptable if adequate clearance is provided in accordance with Subarticle IWA-1500, "Accessibility," 5.2.4.3of the ASME Code, Section XI.					
5.2.4.3	Examination Categories and Methods					
	The examination categories and methods specified in the SAR are acceptable if they are in agreement with the criteria in Article IWB-2000, "Examination and Inspection," of Section XI of the ASME Code. Every area subject to examination should fall within one or more of the examination categories in Article IWB-2000 and should be examined at least to the extent specified. The methods of examination for the components and parts of the pressure retaining boundaries are also listed in the requirements of Article IWB-2000 of Section XI of the ASME Code. The applicant's or licensee's examination techniques and procedures used for preservice examination or inservice inspection of the system are acceptable if they are in agreement with the following criteria:					
	A. The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with Article IWA-2000, "Examination and Inspection," and Article IWB-2000, "Examination and Inspection," of Section XI of the ASME Code. B. The acceptance standards for the examination results required by 3.A above are given in Section XI, Article IWB-3000, "Acceptance Standards." C. The methods, procedures, and requirements for qualification of personnel performing ultrasonic examination are in accordance with the requirements of Appendix VII to Division 1 of Section XI					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of the ASME Code. D. Performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws is in accordance with the requirements of Appendix VIII of Section XI of the ASME Code. E. The methods, procedures, and requirements for ultrasonic examination of reactor-vessel-to-flange welds, closure-head-to-flange welds, and integral attachment welds incorporate the regulatory positions provided in Regulatory Guide 1.150, unless qualified by performance demonstration in accordance with the requirements of Appendix VIII of Section XI of the ASME Code.					
5.2.4.4	Inspection Intervals					
	The required examinations and pressure tests must be completed during each ten-year interval of service, hereinafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of Article IWA-2000, “Examination and Inspection,” concerning inspection intervals of Section XI of the ASME Code.					
5.2.4.5	Evaluation of Examination Results					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The standards for evaluation of examination results are acceptable if they are in accordance with the requirements of Section XI, Article IWB-3000, "Acceptance Standards." B. The proposed program regarding repair or replacement of components containing defects is acceptable if the program is in accordance with the requirements of Section XI, Article IWA-4000, "Repair/Replacement Activities." The criteria that establish the need for repair or replacement are described in Section XI, Article IWB-3000, "Acceptance Standards." C. The standards for evaluation of examination results should be in accordance with the requirements of Sections XI, Article IWB-3000, "Acceptance Standards," if Regulatory Guide 1.150 is used.					
5.2.4.6	System Pressure Tests					
	The pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program is in accordance with the requirements of Section XI, Article IWB-5000, "System Pressure Tests," and the technical specification requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. In some cases, these limitations may be more severe than those in Article IWB-5000.					
5.2.4.7	Code Exemptions					
	Exemptions from Code examinations should be permitted if the criteria in Subsubarticle IWB-1220, "Components Exempt from Examination," are met. The applicant's or licensee's program should list the exemptions taken in accordance with the ASME Code.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
5.2.4.8	Code Cases					
	ASME code cases referenced by the COL application are reviewed for acceptability and compliance with Regulatory Guide 1.147. Code cases not specifically referenced in Regulatory Guide 1.147 will be reviewed and accepted on a case-by-case basis.					
5.2.4.9	Augmented ISI to Protect Against Postulated Piping Failures					
	The reviewer verifies that the high-energy system piping between containment isolation valves should receive an augmented ISI as follows: A. Protective measures, pipe whip restraints, structures, supports and guard pipes should not prevent access required to conduct the inservice examinations specified in the ASME Code, Section XI, Division 1. B. For those portions of high-energy fluid system piping between containment isolation valves, the inservice examination completed during each inspection interval should provide 100% volumetric examination of circumferential and longitudinal pipe welds. C. For those portions of high-energy fluid system piping enclosed in guard pipes, inspection ports should be provided in the guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in the portion of the guard pipe passing through the annulus of dual-barrier containment structures. D. The areas subject to examination should be defined in accordance with the Examination Category for Class 1 piping welds specified in Article IWB-2000.					
5.2.4.10	Other Inspection Programs					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A. For BWR plants, the reviewer ascertains that the ISI program addresses the staff positions concerning augmented inspections for intergranular stress corrosion cracking (IGSCC) provided in Generic Letter 88-01, Supplement 1 to Generic Letter 88-01, and NUREG-0313, Revision 2. B. For BWR plants, the reviewer ascertains that the ISI program adequately addresses the augmented inspections of feedwater and control rod drive nozzles as discussed in NUREG-0619. The staff may approve alternatives to the inspection guidelines in NUREG-0619. C. For PWR plants, the reviewer verifies that the applicant or licensee has established a program to detect and correct potential RCPB corrosion caused by boric acid leaks, as described in Generic Letter 88-05. D. For Westinghouse PWR plants, the reviewer verifies that the applicant or licensee has established an inspection program to periodically confirm the integrity of incore neutron-monitoring system thimble tubes, as described in NRC Bulletin 88-09.					
5.2.4.11	Operational Programs					
	For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Preservice Inspection, Inservice Inspection and Inservice Testing Programs are reviewed in accordance with 10 CFR 50.55a(g) and 10 CFR Part 50, Appendix A.					
5.2.5, Rev. 2 (03/2007)	Reactor Coolant Pressure Boundary Leakage Detection					
5.2.5.1	For GDC 2, acceptance is based on the guidelines of RG 1.29, Positions C.1 and C.2.					
5.2.5.2	For GDC 30, acceptance is based on meeting the guidelines of RG 1.45.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
5.3.1, Rev. 2 (03/2007)	Reactor Vessel Materials					
5.3.1.1	Materials					
	The requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for materials, as detailed below: A. Acceptable materials for the reactor vessel and its appurtenances and attachments are those identified in the Code, Section III, Appendix I. The materials must also meet the requirements of 10 CFR Part 50, Appendix G. B. The acceptability of materials not specified in the Code are considered on an individual basis. Their suitability is evaluated on the basis of data submitted in accordance with the requirements of Code Section III, Appendix IV-1000 and 10 CFR Part 50, Appendix G. These data must include information on mechanical properties, weldability, and physical changes of the material.					
5.3.1.2	Special Processes Used for Manufacture and Fabrication of Components					
	The requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for fabrication of components. The reactor vessel and its appurtenances are fabricated and installed in accordance with Code Section III, Paragraph NB-4100. The manufacturer or installer of such components is required to certify, by application of the appropriate Code Symbol and completion of an appropriate data report in accordance with Code Section III, Article NCA-8000, that the materials used comply with the requirements of NB-2000, and that the fabrication or installation comply with the requirements of NB-4000.					
5.3.1.3	Special Methods for Nondestructive Examination					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the ASME Code, Section III, for fabrication nondestructive testing. The acceptance criteria for examination of the reactor vessel and its appurtenances by nondestructive examination are those specified in Code Section III, NB-5000.					
5.3.1.4	Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels					
	The acceptance criteria for special controls and processes in welding austenitic or ferritic steel components are based upon the following regulatory guides, ASME Code provisions, and other regulatory documents necessary to satisfy the relevant requirements of GDC 1, 4, 14, and 30; Appendix B; and 10 CFR 50.55a. A. Only those welding processes capable of producing welds in accordance with the welding procedure qualification requirements of Code Sections III and IX may be used. Any process used shall be such that the records required by NB-4300 of Section III can be made, with the exception of stud welding, which is acceptable only for minor nonpressure attachments. B. ASME Code Sections III and IX criteria for welding ferritic steel are supplemented by the regulatory positions in Regulatory Guides (RGs) 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” and 1.34, “Control of Electroslag Weld Properties.” C. The regulatory positions of RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,” provide the acceptance criteria to avoid underclad cracking of stainless steel clad ferritic components. D. ASME Code Sections III and IX criteria for welding austenitic					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	stainless steels are supplemented by the regulatory positions in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and RG 1.34. For the BWR austenitic stainless steel reactor vessel attachments and appurtenances specified in Generic Letter (GL) 88-01, the weld metal ferrite content should be controlled as described in the positions of Attachment A to GL 88-01 or the recommendations of NUREG-0313, Revision 2. E. The regulatory positions of RGs 1.44, "Control of the Use of Sensitized Stainless Steel," and 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," provide the acceptance criteria to avoid sensitization and contamination of stainless steel. RG 1.44 states that non-sensitization should be verified using ASTM A-262 Practices A or E, or another method that can be demonstrated to show nonsensitization of austenitic stainless steel. Alternative tests to those in ASTM A-262 that have been previously accepted include ASTM A 708. For BWRs, the control of sensitized steel per RG 1.44 should be modified as necessary to conform with the positions in Attachment A to GL 88-01 or the recommendations of NUREG-0313. The controls for abrasive work on austenitic stainless steel surfaces should, as a minimum, be equivalent to the controls described in RG 1.37 position C.5 to prevent contamination which promotes stress corrosion cracking. Tools which contain materials that could contribute to intergranular or stress-corrosion cracking or which, because of previous usage, may have become contaminated with such materials, should not be used on austenitic stainless steel surfaces. F. Additional controls, beyond those described above, are					

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	considered necessary to avoid intergranular stress corrosion cracking (IGSCC) in and near welds in BWR austenitic stainless steel reactor vessel attachments and appurtenances. The additional controls are described in Attachment A to GL 88-01 and in NUREG-0313, Revision 2. These controls include material and weldment specifications for IGSCC resistant materials, processing techniques, categorization of the IGSCC resistance of installations based upon material properties, treatment history, and post-weld treatments. The technical bases for these controls are described in NUREG-0313, Revision 2. The referenced regulatory guides are described in detail in the acceptance criteria of SRP Section 5.2.3.					
5.3.1.5	Fracture Toughness					
	The acceptance criteria for this area of review are the requirements of Appendix G of 10 CFR Part 50. These criteria satisfy the requirements of GDC 31 and 10 CFR 50.60 regarding materials testing and acceptance standards for fracture toughness. Appendix G requires that the reactor vessel and appurtenances thereto which are made of ferritic materials shall meet the following minimum requirements for fracture toughness during system hydrostatic tests, conditions of normal operation, and anticipated operational occurrences: A. The ferritic materials shall be tested in accordance with the ASME Code paragraph NB-2300 including: i. T_{NDT} shall be determined for each material by means of a drop weight test. ii. The materials shall meet the acceptance standards of paragraph NB-2330 of the Code, which states that at a					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	temperature not greater than (T_{NDT} + 33°C)[(T_{NDT} + 60°F)] each Charpy C_v specimen tested shall exhibit at least 0.89 mm (35 mils) lateral expansion and not less than 68 J (50 ft-lbs) of absorbed energy. When these requirements are met, T_{NDT} is defined as the reference temperature, RT_{NDT}. iii. In the event that the above requirements are not met, additional C_v notch impact tests are performed (in groups of three specimens) to determine the temperature T_{cv} at which they are met. In this case the reference temperature RT_{NDT} = T_{cv} - 33°C (RT_{NDT} = T_{cv}- 60°F). Thus the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{cv} - 33°C) [(T_{cv} - 60°F)] iv. When a C_v impact test has not been performed at (T_{NDT} + 33°C) [(T_{NDT} + 60°F)], or when the C_v impact test at (T_{NDT} + 33°C) [(T_{NDT} + 60°F)] does not exhibit a minimum of 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion, a temperature representing a minimum of 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v impact tests performed. B. In addition to the above criteria, the requirements of paragraphs IV.A.1, IV.A.2, and IV.B of Appendix G of 10 CFR Part 50 and 10 CFR 50.61(b)(2) (for PWRs) shall be met. i. SRP Section 5.3.2 discusses the requirements of paragraphs IV.A.2 and of Appendix G in detail. ii. The acceptance criteria discussed in paragraph IV.A.1 of Appendix G states that reactor vessel belt-line materials shall have a minimum upper shelf energy of 102 J (75 ft-lbs) as determined from Charpy V-notch impact tests on unirradiated specimens in accordance with paragraph NB-2331(a) of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Code, Section III. Reactor vessel belt-line materials must also maintain an upper shelf energy no less than 68 J (50 ft-lb) throughout the life of the vessel. These two requirements do not apply, however, if it is demonstrated to the Commission by appropriate data and analyses based on other types of tests that lower values of upper shelf fracture energy are adequate. C. The neutron radiation embrittlement effects on reactor vessel materials shall be determined in accordance with 10 CFR Part 50, Appendix G, Section III, and RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials."					
5.3.1.6	Material Surveillance					
	The material surveillance acceptance criteria are the requirements of Section III of Appendix H of 10 CFR Part 50. Complying with the acceptance criteria satisfies the requirements of GDC 32 regarding an appropriate material surveillance program for the reactor vessel. Section III of Appendix H requirements are: A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E > 1 MeV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2. B. Reactor vessels constructed of ferritic materials which do not meet the conditions in paragraph a. shall have their belt-line regions monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) standard ASTM E-185, except as modified by Appendix H to 10 CFR Part 50.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	C. The surveillance program shall meet the following requirements: i. Surveillance specimens shall be taken from locations alongside the fracture toughness test specimens required by Section III of Appendix G of 10 CFR Part 50. The specimen types shall comply with the requirements of Section III.B of Appendix H, except that drop-weight specimens are not required. ii. Surveillance capsules containing the surveillance specimens shall be located near the inside vessel wall in the belt-line region, so that the neutron flux received by the specimens approximates that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface. If the capsule holders are attached to the vessel wall or cladding, inspection shall be done according to the requirements for permanent structural attachments as given in ASME Code Sections III and XI. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in paragraph III.B.1 of Appendix H. iii. The required number of capsules, which will vary from three to five depending upon the adjusted reference temperature at the end of the service lifetime of the reactor vessel, and their withdrawal schedules, shall be in accordance with the requirements of paragraph III.B.2 of Appendix H. iv. For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis in accordance with the requirements of paragraph III.C of Appendix H. The material surveillance program criteria of ASTM E-185 cited in					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	10 CFR Part 50, Appendix H, is predicated on an assumed 40-year reactor vessel design life. For those applicants proposing a facility with greater than a 40-year design life, the criteria of ASTM E-185 must be supplemented to provide for monitoring of the reactor vessel materials for the entire reactor vessel design life. Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Reactor Vessel Material Surveillance Program are reviewed in accordance with 10 CFR 50.60 and 10 CFR 50, Appendix H. The Reactor Vessel Material Surveillance Program and associated implementation milestone(s) are included within the license condition on operational program implementation.					
5.3.1.7	Reactor Vessel Fasteners					
	The acceptance criteria for the reactor vessel bolting material are given by paragraph IV.A of Appendix G to 10 CFR Part 50 and by the recommendations of RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs.” These acceptance criteria satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a, and meet the requirements of GDC 31 regarding prevention of fracture of the RCPB. Regulatory Positions C.1 and C.2 of RG 1.65 recommend the following: A. Materials for reactor vessel studs (and other fasteners) that are considered suitable are SA-540 Grades B-23 and B-24, SA-193 Grade B-7, SA-194 Grade 7, and SA-320 Grade L-43, as presented in Section II of the ASME Code. B. The fastener material should not have an ultimate tensile strength over 1170 MPa (170 ksi), and the fracture toughness tests and acceptance levels of NB-2333 of Section III of the Code must be met as required by paragraph IV.A of Appendix G to 10					

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	CFR Part 50. C. Surface treatments, plating, or thread lubricants used should be shown to be compatible with the materials, and stable at operating temperatures. D. Nondestructive examination should be performed according to Section III of the Code, subsubarticle NB-2580 including additional recommendations given in Regulatory Position C.2 of RG 1.65.					
5.3.2, Rev. 2 (03/2007)	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock					
5.3.2.1	Pressure-Temperature Limits					
	A. Applicable Regulations, Codes, and Basis Documents. The regulations in 10 CFR 50.60 and associated Appendix G to 10 CFR Part 50 describe the conditions that require P-T limits and provide the general basis for these limits. Appendix G specifically requires that P-T limits must be at least as conservative as limits obtained by following Appendix G to Section XI of the ASME Code during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins when the reactor core is critical. Since the regulations may not have included specific fracture toughness testing requirements for the ferritic materials in the pressure-retaining components at the time some of the reactor facilities were designed and constructed, Branch Technical Position (BTP 5-3) describes procedures for making estimates and assumptions concerning the fracture toughness properties of materials in the older plants. Although Appendix G to Section III of the ASME Code is usually referenced with regard to facility design and construction, the					

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	reviewer should instead apply the provisions of Appendix G to Section XI of the ASME Code when using this SRP. The following provide the rationale for using Appendix G to Section XI of the ASME Code instead of Appendix G to Section III of the ASME Code: i. Appendix G to 10 CFR Part 50 specifically references Appendix G to Section XI to the ASME Code, and Appendix G to Section III to the ASME Code contains similar provisions. ii. The differences between Appendix G to Section XI of the ASME Code and Appendix G to Section III of the ASME Code have resulted from a series of ASME code cases, including N-588, N-640, and N-641. Appendix G to Section III of the ASME Code has not been updated since those code cases were developed. However, the staff expects that Appendix G of Section III of the ASME Code will be updated to be consistent with Appendix G to Section XI of the ASME Code. B. Pressure-Temperature Requirements. Appendix G to 10 CFR Part 50 requires that the pressure-temperature (P-T) limits defined in that Appendix be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code, as stated below: i. Pressure-Temperature Limits for Preservice Hydrostatic Tests During preservice hydrostatic tests (if fuel is not in the vessel), a material's lower bound static crack initiation fracture toughness, K_{Ic}, must be greater than the K_I caused by pressure stresses acting on a defined, conservative hypothetical flaw, as shown in					

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	the following expression: $K_{applied} = K_I(\text{pressure}) < K_{Ic}$ ii. Pressure-Temperature Limits for Inservice Leak and Hydrostatic Tests During performance of inservice leak and hydrostatic tests, a material's K_{Ic} must be greater than 1.5 times the K_I caused by pressure, as shown in the following expression: $K_{applied} = 1.5\, K_I(\text{pressure}) < K_{Ic}$ iii. Pressure-Temperature Limits for Heatup and Cooldown Operations At all times during heatup and cooldown operations, a material's K_{Ic} must be greater than the sum of 2 times the K_I caused by pressure and the K_I caused by thermal gradients, as shown in the following expression: $K_{applied} = 2K_I(\text{pressure}) + K_I(\text{thermal}) < K_{Ic}$ iv. Pressure-Temperature Limits for Core Critical Operation At all times that the reactor core is critical (except for low-power physics tests), the temperature must be higher than that required for inservice hydrostatic testing. In addition, the P-T relationship must provide at least a 22 °C (40 °F) margin over that required for heatup and cooldown operations.					
5.3.2.2	Upper-Shelf Energy					
	A. Applicable Regulations, Codes, and Basis Documents. Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have a Charpy USE value in the transverse direction for base material and along the weld for weld material according to the ASME Code of no less than 102 J (75 ft-lb) initially and must maintain a Charpy USE value throughout the life of the vessel of no less than 68 J (50 ft-lb), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.					

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	B. Upper-Shelf Energy Requirements. Appendix G to 10 CFR Part 50 contains the following USE requirements: i. Initially, the USE value in the transverse direction for base material and along the weld must not be less than 102 J (75 ft-lb). ii. Charpy USE throughout the life of the vessel must be maintained at no less than 68 J (50 ft-lb), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.					
5.3.2.3	Pressurized Thermal Shock					
	A. Applicable Regulations, Codes, and Basis Documents. Projected values of RT_{PTS} must be determined for PWR reactor vessel beltline materials in accordance with 10 CFR 50.61. For RT_{PTS} values projected to exceed the screening criteria, safety analyses must be provided that include proposed flux reduction programs or other corrective actions to prevent potential PTS-related failure of the reactor vessel if continued plant operation beyond the screening criterion is allowed. B. Pressurized Thermal Shock Requirements. In accordance with 10 CFR 50.61, values of RT_{PTS} projected using the methods of 10 CFR 50.61 for the time of the initial application submittal and for the projected expiration date of the operating license must not exceed the screening criteria of 132 °C (270 °F) for plates, forgings, and axial weld materials, and 149 °C (300 °F) for circumferential weld materials, throughout the facility's licensed operating permit. This assessment must be updated whenever projected values of RT_{PTS} change significantly, or upon request for a change in the expiration date for operation of the facility. For RT_{PTS} values projected to exceed the screening criteria, safety					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	analyses must be provided that include proposed flux reduction programs or other corrective actions to prevent potential PTS-related failure of the reactor vessel if continued plant operation beyond the screening criterion is allowed.					
5.3.3, Rev. 2 (03/2007)	Reactor Vessel Integrity					
5.3.3.1	Design					
	With regard to compatibility of design with material properties and fabrication methods, the quality standards requirements of GDC 1, GDC 30, and § 50.55a are met by compliance with the provisions of the ASME boiler and pressure vessel code. The basic acceptance criteria for the design of the vessel are the requirements of Section III of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"). The design of the reactor vessel must be compatible with the properties of the materials used, and must permit construction by the use of standard and well proven fabrication methods. The design details should not include new or novel concepts unless they are substantiated by a comprehensive justification showing that no aspects of the design will compromise the overall integrity of the vessel in any manner. The design details must be adequate to permit all required inspections and to provide required access to all areas requiring inservice inspection in conformance with Section XI of the Code, as detailed in SRP Section 5.2.4. This satisfies the requirements of GDC 32 and § 50.55a regarding inservice inspection. If the procedures of Section IV.A of Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 do not indicate the existence of an equivalent safety margin, then Section IV.B allows the reactor vessel beltline to be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of 10 CFR 50.66, "Requirements for					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	thermal annealing of the reactor pressure vessel." Annealing of the reactor vessel provides assurance that fracture toughness properties can be restored to satisfy the fracture toughness requirements of GDC 31.					
5.3.3.2	Materials of Construction					
	The basic acceptance criteria for the materials used in the construction of the reactor vessel, and the regulations that they satisfy, are detailed in SRP Sections 5.2.3 and 5.3.1. These criteria are the requirements of Appendix G,10 CFR Part 50, as augmented by Sections III and IX of the Code. The materials must be compatible with the design requirements in the GDC. Acceptability is based on standard practice and engineering judgement, with consideration being given to such factors as material form, size-related variations in properties, and nonisotropic characteristics. Although many materials are acceptable for reactor vessels according to Section III of the Code, the special considerations relating to fracture toughness and radiation effects effectively limit the basic materials that are currently acceptable for most parts of reactor vessels to SA 533 Gr B C1 1, SA 508 C1 2, and SA 508 C1 3. Acceptability criteria for other grades will have to be developed before they can be used. Material compositions and expected neutron fluence must be compatible with the requirements for the material surveillance program. The reviewer uses published data to ensure that the predicted shift in toughness properties (RTNDT and upper shelf energy) is conservative, based on actual material composition and predicted fluence. The predicted shift in toughness properties should be at least as conservative as that obtained by use of the most recent revision of Regulatory Guide (RG) 1.99. Acceptability of the material surveillance program, as specified in					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR Part 50, depends on these relationships.					
5.3.3.3	Fabrication Methods					
	Acceptance criteria for the basic fabrication processes and their qualification and control requirements, and the regulations satisfied by these criteria, are detailed in SRP Section 5.3.1. These criteria are given in Sections III and IX of the Code. Although a particular fabrication process (such as multiple wire-high heat input welding) may be generally acceptable, it may not be suitable for reactor vessel fabrication for some materials without further justification or qualification. The reviewer uses "state-of-the-art" criteria and past practice to evaluate the acceptability of materials process combinations. Because fabrication methods, materials, and the effectiveness of nondestructive evaluation methods are interrelated, the reviewer should rely on state-of-the-art knowledge and past practice to determine whether the proposed combinations are compatible and acceptable.					
5.3.3.4	Inspection Requirements					
	The basic requirements for performing nondestructive inspections, the quality assurance criteria for the reactor vessel, and the regulations that all of these criteria satisfy, are detailed in SRP Section 5.3.1. These requirements and criteria are contained in Section III of the Code. Additional criteria are contained in Section V of the Code. Acceptance criteria for compatibility with materials and fabrication areas are discussed in previous sections. Very important relationships are those among in-process and					

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	final shop inspections, and the inservice inspection requirements of Section XI of the Code. The reviewer should determine whether the methods of inspection, the sensitivity levels, and flaw evaluation criteria are compatible with Section XI, and whether the results of the preservice baseline inspection can be correlated with the results of later inservice inspections.					
5.3.3.5	Shipment and Installation					
	The basic acceptance criteria for procedures and care to maintain proper cleanliness and freedom from contamination during all stages of shipping, storage, and installation of the reactor vessel, and the regulations that these criteria satisfy, are given in SRP Section 5.2.3. The purpose of this area of review is to verify that the as-built characteristics of the reactor vessel are not degraded by improper handling. Acceptability in these areas is assured for current designs and materials by compliance with the basic acceptance criteria. If nonstandard materials or designs are used, the reviewer should determine whether criteria will be adequate, based on current technology. If the basic criteria are not followed, either intentionally or through error, the reviewer should evaluate, on a case basis, whether the integrity of the reactor vessel is compromised, using current technology, past practice, and experience as applicable.					
5.3.3.6	Operating Conditions					
	Acceptance criteria for operating limits for the reactor vessel, and the regulations that they satisfy, are detailed in SRP Section 5.3.2. These acceptance criteria are given in Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50 and for PWRs, 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”					

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	The criterion for acceptable behavior is that the vessel remains leaktight enough to support adequate core cooling. The generally accepted principles and procedures of linear elastic fracture mechanics provide the basis for acceptance of analyses that support conformance with this criterion.					
5.3.3.7	Inservice Surveillance					
	The acceptance criteria for adequacy of the reactor vessel materials surveillance program, and the regulations satisfied by the criteria, are detailed in SRP Section 5.3.1. The criteria are based on the requirements of Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50. The SAR also provides information regarding the inservice inspections to be performed on the reactor vessel. The acceptance criteria for accessibility and inspection plan details, and the regulations that they satisfy, are detailed in SRP Section 5.2.4. These criteria are those of Section XI of the Code.					
5.3.3.8	Operational Programs					
	For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Inservice Inspection and Reactor Vessel Material Surveillance Programs are reviewed under SRP Section 5.2.4 and 5.3.1 respectfully, in accordance with 10 CFR 50.55a(g), 10 CFR 50.60 and 10 CFR 50, Appendix H. The Reactor Vessel Material Surveillance Program and associated implementation milestone(s) are included within the license condition on operational program implementation.					
5.4, Rev. 2 (03/2007)	Reactor Coolant System Component and Subsystem Design					
	Refer to the BTP for the detailed criteria.					
5.4.1.1, Rev. 2 (03/2007)	Pump Flywheel Integrity (PWR)					
5.4.1.1.1	Materials Selection and Fabrication					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	The applicant's materials selection and fabrication are acceptable if they comply with the following criteria, which are derived from Subsections C.1.a and C.1.c of RG 1.14. The flywheel material is acceptable if it is produced by a process (such as vacuum melting or degassing) that minimizes flaws in the material and improves its fracture toughness properties. If the flywheel is flame cut from a plate or forging, at least 1.3 cm (1/2 inch) of material should be left on the outer and bore radii for machining to final dimensions.					
5.4.1.1.2	Fracture Toughness					
	The pump flywheel fracture toughness properties are acceptable if they comply with the following criteria, which are derived from Subsection C.1.b and supplemented by Subsection B of RG 1.14 and the ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Appendix G, Protection Against Nonductile Failure. The material should be examined and tested to establish its fracture toughness property. The minimum KIC of the material at the normal operating temperature of the flywheel should be 165 MPa /m (150 ksi /in). Use of the direct test method to obtain KIC is encouraged. Direct Test. The plane-strain fracture toughness, KIC, should be obtained in accordance with ASTM E 399-05 if linear elastic fracture mechanics is used in the fracture mechanics analysis. The J-resistance curve should be obtained in accordance with ASTM E 1820-05a if elastic-plastic fracture mechanics is used. Either test should be conducted at or below the operating temperature of the pump flywheel. Indirect Tests for Certain Steel. For flywheel materials made of ASME SA-533-B Class 1, ASME SA-508 Class 2, ASME SA-508 Class 3, and ASME SA-516 Grade 65 steel, the fracture					

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	toughness values can be found in the ASME Code, Section XI, Appendix A as a function of the difference between operating temperature (T) and the RT_{NDT} of the flywheel material, i.e., T-RT_{NDT} . The RT_{NDT} of the flywheel material should be determined in accordance with NB-2320 and NB-2330 of the ASME Code, Section III based on the nil-ductility transition temperature (T_{NDT}) determined by dropweight tests (DWT) and the impact energy determined by Charpy V-notch (C_v) tests. NB-2320 specifies ASTM E-208-95a as the Standard for DWT tests and ASTM A-370 as the Standard for C_v tests. If this indirect approach is applied to flywheel materials other than ASME SA-533-B Class 1, ASME SA-508 Class 2, ASME SA-508 Class 3, or ASME SA-516 Grade 65 steel, justification should be given to establish equivalence of fracture toughness between the proposed flywheel material and those mentioned here.					
5.4.1.1.3	Preservice Inspection					
	The applicant's preservice inspection program, including finish machining and ultrasonic and surface inspections, is acceptable if it complies with the following criteria, which are derived from Subsection C.4.a of RG 1.14. A. Each finished flywheel should be subjected to a 100% volumetric examination by ultrasonic methods using procedures and acceptance criteria specified in ASME Code, Section III, NB-2530 for plates, and NB-2540 for forgings. B. If the flywheel is flame cut from a plate or forging, at least 1.3 cm (1/2 inch) of material should be left on the outer and bore radii for machining to final dimensions. C. Finish machined bores, keyways, splines, and drilled holes should be subjected to magnetic particle or liquid penetrant examination.					

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	D. The inspection results should be appropriately documented to establish initial flywheel conditions, accessibility, and practicality of the program to be used as baseline information for future inservice inspections.					
5.4.1.1.4	Flywheel Design					
	The applicant's flywheel design is acceptable if it complies with the following criteria, which are derived from Subsection C.2 of RG 1.14. The flywheel should be designed to withstand normal conditions, anticipated transients, the design basis loss of coolant accident, and the safe shutdown earthquake without loss of structural integrity. The design of the pump flywheel should also meet the following criteria: A. The combined stresses at the normal operating speed due to centrifugal forces and the interference fit of the wheel on the shaft, should not exceed 1/3 of the minimum specified yield strength or 1/3 of the measured yield strength in the weak direction of the material if appropriate tensile tests have been performed on the actual material of the flywheel. B. The design overspeed of a flywheel should be at least ten percent above the highest anticipated overspeed. The anticipated overspeed should include consideration of the maximum rotational speed of the flywheel if a break occurs in the reactor coolant piping in either the suction or discharge side of the pump. An acceptable basis for the assumed design overspeed, addressing pipe breaks consistent with the design basis for reactor coolant piping, should be submitted to the staff for review. C. The combined stresses at the design overspeed, due to					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	centrifugal forces and the interference fit, should not exceed 2/3 of the minimum specified yield strength, or 2/3 of the measured yield strength in the weak direction if appropriate tensile tests have been performed on the actual material of the flywheel. D. The shaft and the bearings supporting the flywheel should be able to withstand any combination of loads from normal operation, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake. E. A fracture mechanics analysis should be conducted for the life time of the flywheel, including extended operation, to predict the critical speed for fracture of the flywheel. The ratio of K_{IC} to the maximum tangential stress at speeds from normal to design overspeed should be at least 2 √in (consistent with SRP 10.2.3, "Turbine Disk Integrity"), or alternatively, the ratio of K_{IC} to the applied K should be 3.16 for normal and upset conditions and 1.41 for emergency and faulted conditions (consistent with the ASME Code approach). This fracture mechanics analysis should consider crack growth due to identified degradation mechanisms for the largest flaw which could be missed by inspection (use the NRC accepted value of 0.25 inch for Westinghouse Owners Group [WOG] and ABB Combustion Engineering Owners Group [CEOG] flywheels if a smaller value can not be justified). The analysis should be submitted as a topical report to the NRC staff for evaluation.					
5.4.1.1.5	Overspeed Test					
	The applicant's commitment to perform an overspeed test is acceptable if each flywheel assembly is tested at the design overspeed of the flywheel. This criterion is taken from Subsection C.3 of RG 1.14.					
5.4.1.1.6	Inservice Inspection (ISI)					
	The applicant's ISI program is acceptable if it complies with the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	following criteria, which are derived from Subsection C.4.b of RG 1.14, operating experience, and staff's evaluation of WOG's and CEOG's fracture mechanics analyses on reactor coolant pump flywheels of operating plants. A. A volumetric examination by ultrasonic methods of the areas of higher stress concentration at the bore and keyway extending to half of the flywheel radius, or a surface examination by liquid penetrant or magnetic particle methods of all exposed surfaces, at approximately 10 operating year intervals, during the refueling or maintenance shutdown coinciding with the inservice inspection schedule as required by the ASME Code, Section XI. Removal of the flywheel is not required. B. Examination procedures and acceptance criteria should be in conformance with the requirements specified in Subsection II.3.A of this SRP section.					
5.4.1.1.7	Operational Programs					
	For COL reviews, the description of the operational program and proposed implementation milestones for the Pre-Service Inspection, Inservice Inspection, and Inservice Testing Programs are reviewed in accordance with 10 CFR Part 50, Appendix A, 10 CFR 50.55a(a)(1) and 10 CFR 50.55a(f). The implementation milestone are completion prior to initial plant start-up, prior to commercial service and after generator on-line on nuclear heat.					
5.4.2.1, Rev. 3 (03/2007)	Steam Generator Materials					
5.4.2.1.1	Selection, Processing, Testing, and Inspection of Materials					
	The materials selected for the steam generator form portions of the primary and secondary system pressure boundary. In addition, certain materials used for nonpressure- retaining components (including tube supports) can have a direct impact on the integrity of the pressure boundary (e.g., denting of the steam					

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	generator tubes from corrosion of a tube support or mechanical damage to the tubes from the generation of loose parts). As a result, the materials selected for the steam generator must be fabricated and tested to quality standards, as required by GDC 1. In addition, the materials selected for the RCPB must be fabricated and tested to the highest quality standards, as required by GDC 30. The materials selected for use in fabricating the steam generator are acceptable from a fabrication/manufacturing standpoint if they comply with 10 CFR 50.55a. In general, this regulation - specifically 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e) - requires that the components satisfy the requirements of Section III of the ASME Code. Provisions in 10 CFR 50.55a(b) permit ASME Code cases, as discussed in Regulatory Guide 1.84, to be used to select, fabricate, and test materials for the steam generator. Section III of the ASME Code establishes - through articles such as NCA-1000, NB-2000 (for Class 1 components), and NC-2000 (for Class 2 components) - requirements for selecting, processing, testing, inspecting (during fabrication/manufacturing), and certifying materials. In general, Section III of the ASME Code references Parts C and D of Section II of the ASME Code for permitted material specifications (e.g., in Articles NB-2120 and NC-2120). Examples of materials that are currently used for Class 1 components in the steam generator include the following: Tubing: ASME SB-163, N06690, Thermally Treated Pressure Plates: ASME SA-533, Grade B, Class 1 Pressure Forgings ASME SA-508, Grade 3,					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(including nozzles — Class 2 (formerly referred to as Class 3a) Nozzle Safe Ends: ASME SA-182, F316LN Channel Heads: ASME SA-508, Grade 3, Class 2 (formerly referred to as Class 3a) Cladding, Buttering, and welds: ASME SFA 5.4 (308L, 309L), 5.9 (308L, 309L), 5.11 (ENiCrFe-7), and 5.14 (ERNiCrFe-7) Pressure Boundary Welds: Low Alloy Steel, SFA 5.5, 5.23, 5.28 Manway Studs: ASME SA-193, Grade B7 Manway Nuts: ASME SA-194 Examples of materials that are currently used for Class 2 components in the steam generator include the following: Pressure Plates: ASME SA-533, Grade B, Class 1 Bolting: ASME SA-193, Grade B7 Tube Support Structures (including antivibration bars/fan-bars): ASME SA-240, Type 405 and Type 410S In summary, for the purposes of satisfying GDC 1 and GDC 30,					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	the materials used in fabricating the steam generator are acceptable if they are selected, fabricated, tested, and inspected (during fabrication/manufacturing) in accordance with the ASME Code.					
5.4.2.1.2	Steam Generator Design					
	The design of the steam generator should limit the potential for degradation so that the integrity of the steam generator, including the tubes, is maintained during the operating interval between inspections. Degradation of the steam generator tubes and other secondary side components that could affect tube integrity should be manageable through the steam generator program (reviewed under SRP Section 5.4.2.2). Degradation of other steam generator pressure boundary materials should be manageable through the inservice inspection program (the RCPB inservice inspection program is reviewed under SRP Section 5.2.4). The steam generator design is acceptable from a degradation standpoint if it accomplishes the following: A. Limits the crevice between the tube and the tube supports. This can be accomplished by using openings of various shapes (e.g., trifoil or quatrefoil) in tube support plates or by using lattice grid (eggcrate) tube supports. The design of the tube supports should promote high-velocity flow along the tubes. Limiting the crevices will limit the buildup of corrosion product and sludge, which can lead to corrosion of the tubes and the supports. B. Uses appropriate corrosion-resistant materials or employs cladding for materials susceptible to corrosion. To limit the potential for denting the tubes, the tube support structures should use a corrosion-resistant material. Tube denting is a phenomenon associated with corrosion of the tube support structures, creating a hard corrosion product that fills the crevice between the tube and the tube support. Denting of tubes can result in the restriction					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of primary coolant flow and stress-corrosion cracking of the tubes. To limit the steam generator tube's susceptibility to corrosion, the tubes should be heat-treated (e.g., thermally treated), as needed, to optimize their microstructure from a corrosion resistance standpoint. To reduce residual stresses in the U-bend region of short-radius tubes (and therefore the material's susceptibility to corrosion), the U-bend region of short-radius tubes should be stress-relieved after bending. The materials that support the tubes and other materials on the secondary side should be sufficiently resistant to degradation to ensure that the tubes will remain adequately supported and to reduce the potential for the generation of loose parts, which can result in a loss of tube integrity. The corrosion-resistant cladding on the tubesheet and on other primary side components should be weld-deposited, fabricated, and inspected according to the requirements in Part QW of Section IX of the ASME Code. C. Limits the crevice and residual stresses in the tubesheet region. The extent of the tube-to-tubesheet crevice should be limited. This can be accomplished by expanding the tube throughout the tubesheet region, if practical (given other design considerations such as the desired preload in the tube for once-through steam generators). The choice of the method for expansion should consider limiting the stresses in the tube. Limiting the crevices will restrict the buildup of corrosion product and sludge that can lead to corrosion. Limiting the stresses will diminish the potential for stress-corrosion cracking. D. Includes an appropriate allowance for deterioration (including corrosion) of the steam generator materials. This is accomplished through compliance with Section III of the ASME Code (Articles NB-2160 and NB-3121 for Class 1 components and Articles NC-2160 and NC-3121 for Class 2 components).					

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	E. Uses bolting material that will perform adequately under the expected service conditions and that is not subject to stress-corrosion cracking. This can be accomplished by following the regulatory positions in Regulatory Guide 1.65. Although Regulatory Guide 1.65 provides guidance for the design of reactor vessel closure studs, it is also appropriate for the selection of suitable steam generator bolting material. The integrity of bolting and threaded fasteners is also reviewed under SRP Section 3.13. The above criteria, in conjunction with the acceptance criteria for interfacing reviews and appropriately performed inservice inspections, as discussed above, provide assurance that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture will be extremely low, (2) the design conditions of the RCPB are not exceeded during operation, and (3) sufficient margin is available to prevent rapidly propagating failure, consistent with the requirements of GDC 14, 15, and 31.					
5.4.2.1.3	Fabrication and Processing of Ferritic Materials					
	A. Fracture Toughness The steam generator is part of the primary and secondary system pressure boundary. As a result, the materials selected should be sufficient to avoid rapidly propagating failure and to ensure that the design conditions will not be exceeded during operation, consistent with the requirements of GDC 14, 15, and 31. The pressure-retaining ferritic materials selected for use in steam generators are acceptable from a fracture toughness standpoint if they (1) comply with Appendix G to 10 CFR Part 50 and with 10 CFR Part 50, 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e) and (2) follow the provisions of Appendix G to Section III of the ASME Code. In general, the regulations cited above require the use of Section III of the ASME Code. Articles NB-2300 and NC-2300 of Section					

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	III of the ASME Code address fracture toughness requirements for Class 1 and 2, respectively. Appendix G to Section III of the ASME Code includes additional fracture toughness criteria. B. Welding The joining of the materials used to fabricate a steam generator is critical to ensuring that it can properly function. Consistent with the requirements of GDC 1 and GDC 30 (for RCPB materials), the welding qualification, weld fabrication processes, and inspection during fabrication and assembly of the steam generator are performed by using quality standards (supplemented and modified, as necessary) commensurate with the importance of the functions to be performed. Ferritic steel welding of steam generator components is acceptable if it complies with 10 CFR Part 50, 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 0.55a(e) and meets the following: i. Controls the amount of specified preheat in accordance with the requirements of paragraph D-1210 of Appendix D to Section III of the ASME Code, as supplemented by Regulatory Guide 1.50. ii. Follows Regulatory Guide 1.34. iii. Follows Regulatory Guide 1.71. With respect to the qualification of the welder or welding operators when limited accessibility is an issue, these qualifications may be waived provided that 100-percent radiographic and/or ultrasonic examination of the completed welded joint is performed. In these cases, the examination procedures and acceptance standards should meet the requirements of Section III of the ASME Code. Records of the examination reports and radiographs should be retained as part of the quality assurance documentation for the completed weld.					

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	iv. Follows Regulatory Guide 1.43.					
5.4.2.1.4	Fabrication and Processing of Austenitic Stainless Steel (if austenitic stainless steel is used for pressure boundary applications)					
	A. Limiting Susceptibility to Cracking Various factors can make austenitic stainless steel susceptible to stress-corrosion cracking. These factors include the yield strength of the material, exposure of the material to contaminants during cleaning and operation, and presence or absence of material sensitization. Consistent with GDC 14, 15, and 31, limiting the potential for stress-corrosion cracking provides assurance that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture is extremely low, (2) the RCPB design conditions are not exceeded during operation, and (3) sufficient margin is available to prevent rapidly propagating failure. The fabrication and processing of austenitic stainless steel steam generator components is acceptable if it complies with 10 CFR Part 50, 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e) and meets the following: i. Limits the yield strength to 620 megapascal (MPa) (90,000 pounds per square inch (psi)). Laboratory stress-corrosion cracking tests and service experience provide the basis for the criterion that the cold-worked austenitic stainless steels used in the RCPB should have an upper limit on yield strength. ii. Follows Regulatory Guide 1.37. With respect to the source of water for final cleaning or flushing of finished surfaces, vented tanks with deionized or demineralized water are an acceptable source. The oxygen content of this water need not be controlled; however, the concentrations of other chemical species (e.g., chloride, fluoride) should be limited to the values listed in					

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	Regulatory Guide 1.44. iii. Controls abrasive work on austenitic stainless steel surfaces in accordance with Position C.5 of Regulatory Guide 1.37, at a minimum. iv. Follows Regulatory Guide 1.44. In addition to the methods discussed in Regulatory Guide 1.44 for verifying that austenitic stainless steel is not sensitized, alternative tests that have been previously accepted, based upon the adequacy of justifications presented and circumstances of proposed use, include the use of ASTM A-708. v. Follows Regulatory Guide 1.36. The thermal insulation is acceptable if either reflective metal insulation is employed or a nonmetallic insulation that meets the criteria of Regulatory Guide 1.36 is used. B. Welding The joining of the materials used to fabricate a steam generator is critical to ensuring that it can properly function. Consistent with the requirements of GDC 1 and GDC 30 (for RCPB materials), the welding qualification, weld fabrication processes, and inspection during fabrication and assembly of the steam generator are performed using quality standards (supplemented and modified, as necessary) commensurate with the importance of the functions to be performed. Austenitic stainless steel welding of steam generator components is acceptable if it complies with 10 CFR Part 50, 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e) and meets the following: i. Regulatory Guide 1.31					

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	ii. Regulatory Guide 1.34 iii. Regulatory Guide 1.71 iv. NUREG-0313, which may be appropriate for any austenitic stainless steel steam generator materials					
5.4.2.1.5	Compatibility of Materials with the Primary (Reactor) and Secondary Coolant and Cleanliness Control					
	The materials used in the steam generator (including the tubes) can degrade. The degree of susceptibility to degradation and the rate of degradation depend, in part, on the materials, water chemistry, and operating environment (e.g., temperature). To ensure that the materials are compatible with the environment, consistent with the requirements of GDC 4, the primary and secondary coolant water chemistry should be controlled. In addition, material damage or deterioration can occur during construction and operation as a result of improper cleaning or cleanliness control. This damage/deterioration can result from chemical impurities or from particulate matter. As a result, it is important to establish measures to control the cleaning of material and equipment, consistent with the requirements of Criterion XIII of Appendix B to 10 CFR 50. The overall purpose of determining the compatibility of the material with the environment is to ensure that the inservice inspection program is sufficient to manage any degradation. The intention of this approach is ultimately to ensure that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture is extremely low, (2) the RCPB design conditions are not exceeded during operation, and (3) sufficient margin is available to prevent rapidly propagating failure, consistent with the requirements of GDC 14, 15, and 31.					

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	The primary water chemistry program is reviewed under SRP Sections 5.2.3 and 9.3.4. In addition, Regulatory Guide 1.44 discusses appropriate chemistry limits for the reactor coolant. The secondary water chemistry program is acceptable if (1) the coolant chemistry is maintained and monitored as described in the Branch Technical Position, BTP 5-1, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generator," (2) the secondary water chemistry requirements in the latest revisions of the Standard Technical Specifications, NUREG-1430, NUREG-1431, and NUREG-1432 are incorporated into the facility's Technical Specifications (the secondary water chemistry program in the Standard Technical Specifications meets the requirements of 10 CFR 50.36), and (3) the chemical additives that limit the steam generator's susceptibility to corrosion are such that any degradation to which the steam generator remains susceptible can be managed through the inservice inspection program. The operating environment (temperature, pressure, and flow) includes important variables that must be considered in evaluating the effectiveness of the chemical additives in limiting the steam generator's susceptibility to corrosion. The onsite cleaning and cleanliness controls of the steam generator are acceptable if they meet the regulatory provisions of Regulatory Guide 1.37, consistent with the requirements of Criterion XIII of Appendix B to 10 CFR Part 50.					
5.4.2.1.6	Provisions for Accessing the Secondary Side of the Steam Generator					
	Corrosion products (including deposits and sludge) and other contaminants can accumulate in the secondary side of the steam generator. For example, corrosion products and contaminants have been observed along the length of the steam generator tubes, in the crevice between the tube and the tube supports, and at the top of the tubesheet. Depending on the nature of these corrosion products and contaminants, degradation of the tubes (or					

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	other components) can occur. Because this degradation could lead to degradation of the pressure boundary, the design of the steam generator should provide access for the removal of these corrosion products and contaminants. These provisions will supplement the removal of corrosion products and contaminants by blowdown, which is reviewed under SRP Section 10.4.8. In addition to corrosion products and other contaminants, foreign objects (including loose parts) can be introduced into the steam generator. These objects can also lead to degradation of the pressure boundary; therefore, the design of the steam generator should provide access for removing these objects. The steam generator design is considered acceptable from a secondary-side access standpoint if it provides adequate access to the internals so that tools may be inserted to inspect and remove (1) corrosion products and contaminants (such as those found on the tubesheet and at the tube-to-tube support crevice) that may lead to corrosion and (2) foreign objects (including loose parts) that may affect tube integrity. These provisions, in conjunction with appropriately performed inservice inspections, as discussed above, provide assurance that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture is extremely low and (2) the RCPB design conditions are not exceeded during operation, consistent with the requirements of GDC 14 and 15.					
5.4.2.2, Rev. 2 (03/2007)	Steam Generator Program					
5.4.2.2.1	Steam generator tubes are susceptible to degradation. This degradation can occur anywhere along the length of the tube. As a result, each tube is required to be accessible for inspection along its entire length and removable from service if unacceptable flaws are observed. The entire length of each tube must be					

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	inspectable using currently available nondestructive examination methods and techniques capable of finding the forms of degradation that may occur during the service life of the steam generators. The design of the steam generators should permit tubes with unacceptable flaws to be removed from service to ensure that tube integrity will be maintained. Tubes with unacceptable flaws should also be capable of being stabilized if it is determined that a plugged tube potentially may sever (as a result of continued degradation) and subsequently affect the integrity of an active tube. Access to both the primary and secondary sides of the steam generator tubes is required because conditions may exist on either side of the steam generator tubes that could affect their structural and leakage integrity. This should include, for example access to the secondary face of the tubesheet, open tube lanes, feedwater inlet area (e.g., J-tubes or preheater inlet), and other locations that may impair tube integrity. Degradation of secondary side internals can result in the generation of loose parts, inadequate tube support, and mechanical damage to the tubes. In addition, the introduction of foreign objects (including loose parts) into the steam generator during fabrication, maintenance, or operation of the steam generators could impact tube integrity. Sludge buildup and deposits on the tubes can increase the susceptibility of the tubing to corrosion and make it more difficult to inspect the tubing (e.g., because of noise in eddy current data or obstructions in a visual inspection). As a result, the design of the steam generators is considered acceptable for the purposes of implementing the steam generator program if it (1) ensures that all steam generator tubes are accessible for periodic inspection, testing, and repair (including plugging and stabilizing), (2) permits an inspection of the full length of every tube, using currently available nondestructive					

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	examination methods and techniques, (3) allows access to the tubes from the primary side, (4) permits access to the secondary side of the steam generator for assessing the condition of SSCs that may affect tube integrity and for taking appropriate corrective action if adverse/anomalous conditions are identified, (4) permits inspection for, and removal of, foreign objects (including loose parts), and (5) allows the removal of each tube from service.					
5.4.2.2.2	A steam generator program is needed to ensure the effective monitoring and management of tube degradation and degradation precursors (so as to ensure steam generator tube integrity). This permits prompt preventive and corrective actions to ensure that the structural and leakage integrity of the steam generator tubes is maintained. The steam generator program should include elements such as an assessment of degradation, inspection requirements for the tubes and any repairs to the tubes (including plugging), integrity assessment procedures, tube plugging and repairs, primary-to-secondary leak monitoring, foreign material exclusion (including management of loose parts), maintenance of steam generator secondary side integrity, contractor oversight, self assessment, and reporting. For light water reactors (LWRs), Nuclear Energy Institute (NEI) 97-06 discusses many of the elements of a steam generator program. The water chemistry portion of the steam generator program is reviewed under BTP 5-3, “Monitoring of Secondary Side Water Chemistry in PWR Steam Generators.”					
5.4.2.2.3	The latest revisions of NUREG-1430, NUREG-1431, and NUREG-1432 provide for the establishment and implementation of a steam generator program to ensure that tube integrity is maintained for the operating interval between tube inspections, consistent with the requirements of GDC 32. The Technical Specifications provide the objectives of the steam generator program, maximum limits on the quantity of primary-to-secondary leakage permitted during operation, maximum time interval between inspections, objectives of the techniques used to inspect					

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	the tubes, tube repair criteria, and tube repair methods. "Notice of Availability of Model Application Concerning Technical Specification Improvement To Modify Requirements Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process," and " Notice of Opportunity To Comment on Model Safety Evaluation on Technical Specification Improvement To Modify Requirements Regarding the Addition of LCO 3.4.[17] on Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process" include the staff's evaluation of these Technical Specifications. The intention of implementing this program is to ensure tube integrity consistent with the original design criteria for the tubes. Certain aspects of the steam generator portion of the Standard Technical Specifications specify a plant-specific evaluation. For example, the tube repair criteria and tube repair methods are evaluated on a plant-specific basis.					
5.4.2.2.4	With respect to the steam generator tube repair criteria, Regulatory Guide (RG) 1.121 describes a methodology acceptable to the NRC staff for determining the repair criteria specified in the Technical Specifications. Specifically, RG 1.121 describes a methodology for determining the minimum acceptable steam generator tube wall thickness. This methodology accounts for flaw growth and the uncertainty in measuring the size of the flaw (i.e., nondestructive examination uncertainty). The general principles of RG 1.121 can also be used to evaluate the acceptability of alternate tube repair criteria, that is, to assess tube repair criteria based on inspection parameters (e.g., flaw length) other than the depth of the flaw (i.e., other than a minimum wall-thickness repair criterion). Tubes with flaws that exceed the repair criteria, as determined by the steam generator program, are removed from service consistent with the objective of the steam generator program to maintain tube integrity.					
5.4.2.2.5	With respect to tube repair methods, the review of these methods					

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	ensures that the repair is accessible for inspection and that techniques are available to find the forms of degradation to which the repair may be susceptible. The acceptability of any materials used in the repair is evaluated under SRP Section 5.4.2.1. The review of the acceptability of the mechanical design of the repair is consistent with the design requirements of the ASME Code and the steam generator performance criteria in the Standard Technical Specifications. The repair criteria for the repair method are reviewed under the guidance in RG 1.121.					
5.4.2.2.6	The latest revisions of NUREG-1430, NUREG-1431, and NUREG-1432 address ISI; however, preservice inspections are essential in assessing the nature and significance of indications detected during ISI. As a result, it is important to inspect all tubes before placing the steam generators in service, using techniques that should be used during subsequent inspections (i.e., ISI). Although preservice inspections should use techniques that are expected to be employed during ISI, this expectation should not be construed to inhibit the use of new technology or to imply that the techniques used during the preservice inspection will always remain acceptable (i.e., different techniques may be appropriate based on operating experience).					
5.4.2.2.7	10 CFR 50.55a(b)(2)(iii) specifically addresses the inspection of steam generator tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern. This requirement is intended to resolve any conflict between the requirements in the ASME Code and the Technical Specifications. If a conflict (i.e., difference) does not exist pertaining to a specific requirement, both the requirements of the ASME Code and the Technical Specifications must be met. In general, the requirements in the ASME Code and the Technical Specifications are complementary.					
5.4.2.2.8	For applicants referencing a certified design, the Standard Technical Specifications associated with the referenced design					

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	will specify the guidelines for periodic inspection and testing of the steam generator tube portion of the RCPB.					
5.4.2.2.9	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Steam Generator Program are reviewed in accordance with 10 CFR.55a(g) as it relates to periodic inspection and testing of the steam generator tubes as detailed in Section XI of the ASME Code. The implementation milestone is the establishment and completion of an acceptable steam generator program per Article IWA-2430(b) of Section XI of the ASME Code before placing the plant into commercial service. The steam generator program is acceptable if it: A. Complies with 10 CFR 50.55a as it relates to periodic inspection and testing of the steam generator tubes as detailed in Section XI of the ASME Code. B. Complies with 10 CFR 50.65 as it relates to monitoring SSCs and establishing goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. C. Incorporates the steam generator program requirements in the latest revisions of the Standard Technical Specifications, NUREG-1430, NUREG-1431, and NUREG-1432, into the facility's Technical Specifications (the steam generator program in the Standard Technical Specifications meets the requirements of 10 CFR 50.36). D. Verifies that all potential conflicts between the Technical Specifications and the ASME Code are identified. E. Verifies that the steam generator program includes the elements discussed above.					

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	F. Ensures that all tubes are inspected before being placed in service, using techniques that are expected to be used during subsequent inspections.					
5.4.6, Rev. 4 (03/2007)	Reactor Core Isolation Cooling System (BWR)					
5.4.6.1	The general objective of the review is to determine that the RCIC system-in conjunction with the HPCS (or HPCI) system, the safety and relief valves (SRVs), and the suppression pool cooling mode of the residual heat removal (RHR) system-meets the requirements of GDC 34 by providing the capability for decay heat removal to enable complete shutdown of the reactor under conditions requiring its use. The system must maintain the reactor water inventory above the top of the active fuel until the reactor is sufficiently depressurized to permit operation of the low-pressure cooling systems. The RCIC system-in conjunction with the HPCS (or HPCI) system, SRVs, and suppression pool cooling mode of the RHR system-must be capable of removing fission product decay heat and other residual heat from the reactor core following shutdown thus precluding fuel damage or reactor coolant pressure boundary overpressurization. Because the RCIC system, in conjunction with the HPCS (or HPCI) system, provides makeup inventory in some modes of RHR, these systems should jointly meet the guidelines of Branch Technical Position (BTP) 5-4, “Design Requirements of the Residual Heat Removal System.”					
5.4.6.2	The RCIC system also supplies reactor coolant makeup for small leaks. Accordingly, the system must meet the relevant requirements of GDC 33.					
5.4.6.3	Historically, credit has been taken for the RCIC system capability to mitigate the consequences of certain abnormal events; however, because the cooling function is redundant to the HPCI, HPCS, or HPCF system, the RCIC system itself is not required to meet the single failure criterion, but it must do so in conjunction with the HPCS, HPCI, or HPCF system. In addition, the RCIC					

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	system must perform its function without the availability of any alternating current (ac) power, per the requirements of GDC 34, and, in conjunction with the HPCS, HPCI, or HPCF system, must be designed to ensure an extremely high probability of accomplishing its safety function, as required by GDC 29.					
5.4.6.4	As a system that must respond to certain abnormal events, the design of the RCIC system must conform to seismic Category I standards (as discussed in SRP Section 3.2.1), and must not be shared among nuclear power units, except as permitted by GDC 5.					
5.4.6.5	The RCIC system and the HPCS, HPCI, or HPCF system must be protected against natural phenomena, external or internal missiles, pipe whip, and jet impingement forces so that such events cannot cause both systems to fail simultaneously. SRP Sections 3.3.1 through 3.6.2 discuss relevant acceptance criteria.					
5.4.6.6	The RCIC system must meet the requirements of GDC 54 regarding leak detection and isolation provisions for lines passing through the primary containment. SRP Sections 6.2.4 and 6.2.6 describe other containment isolation criteria for the RCIC system.					
5.4.6.7	The RCIC system should meet the following task action plan item recommendations of NUREG-0737 and NUREG-0718: A. Section II.K.1.22 with regard to actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems that are used when the main feedwater system is not operable. The regulations at 10 CFR 50.34(f)(2)(xxi) establish an equivalent requirement for those applicants subject to the requirements of 10 CFR 50.34(f). B. Section II.K.3.13 with regard to separation of the initiation levels of the HPCI and RCIC systems so that the RCIC system initiates at a higher water level than the HPCI system and so that the RCIC system initiation logic will restart the RCIC system on a low water level. The regulations at 10 CFR 50.34(f)(1)(v) establish					

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	an equivalent requirement for those applicants subject to the requirements of 10 CFR 50.34(f). C. Section II.K.3.15 with regard to preventing spurious isolation of the RCIC system from the line break detection logic. D. Section II.K.3.22 with regard to automatic switchover of the RCIC system suction from the CST to the suppression pool when the CST level is low. E. Section II.K.3.24 with regard to space cooling to ensure reliable long-term operation of the RCIC system following a complete loss of offsite power to the plant for at least 2 hours. The regulations at 10 CFR 50.34(f)(1)(ix) establish an equivalent requirement for those applicants subject to the requirements of 10 CFR 50.34(f). F. Section III.D.1.1 with regard to leakage detection and control in the design of systems outside containment that include (or might include) radioactive source term materials following an accident. The regulations at 10 CFR 50.34(f)(2)(xxvi) establish an equivalent requirement for those applicants subject to the requirements of 10 CFR 50.34(f).					
5.4.6.8	If the RCIC system is used to control or mitigate the consequences of an accident, either by itself or as a backup to another system, it must meet the requirements of an engineered safety feature (ESF). The RCIC system should have adequate martgin with the containment at atmospheric pressure.					
5.4.6.9	To satisfy the requirements of GDC 4, design features and operating procedures that are designed to prevent damaging water hammer attributable to mechanisms such as voided discharge lines, steam bubble collapse, and water entrainment in steamlines shall be provided.					
5.4.6.10	If the RCIC system supports the demonstration of adequate plant SBO coping capability as required by 10 CFR 50.63, acceptance					

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	may be based on the positions in Regulatory Guide 1.155 regarding RCIC system design.					
5.4.7, Rev. 4 (03/2007)	Residual Heat Removal (RHR) System					
5.4.7.1	The system or systems must satisfy the functional, isolation, pressure relief, pump protection, and test requirements specified in Branch Technical Position BTP 5-4.					
5.4.7.2	To meet the requirements of GDC 4, design features and operating procedures should be provided to prevent damaging water hammer caused by such mechanisms as voided lines.					
5.4.7.3	Interfaces between the RHR system and the RCIC and component or service water systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the RHR system must conform to GDC 5.					
5.4.7.4	When the RHR system is used to control or mitigate the consequences of an accident, it must meet the design requirements of an engineered safety feature system. This includes meeting the guidelines of Regulatory Guide 1.82 regarding water sources for long term recirculation cooling following a loss-of-coolant accident.					
5.4.8, Rev. 3 (03/2007)	Reactor Water Cleanup System (BWR)					
5.4.8.1	The system should be capable of maintaining acceptable reactor water purity in normal operation and during anticipated operational occurrences, e.g., reactor startup, refueling, and condensate demineralizer breakthrough to ensure reactor coolant pressure boundary integrity in accordance with the requirements of GDC 14. The following should be included in the system design: A. The system should be designed to maintain reactor water					

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	purity within the guidelines provided in the latest version in the Electric Power Research Institute (EPRI) report series, "BWR Water Chemistry Guidelines," and the technical specifications for water chemistry of reactor coolant systems for boiling water reactors. The system should provide demineralization of reactor water through mixed bed resins (beads or powdered) at approximately one percent of the main steam flow rate. B. The nonregenerative heat exchangers should be designed to reduce the temperature of cleanup flow to the demineralizer operating temperature when the regenerative heat exchanger cooling capacity is reduced as a result of partially bypassing a portion of the return flow to the main condenser or radwaste system. C. The RWCS should have the capability to permit processing excess reactor water during startups, shutdowns, and hot standby conditions. Interconnections between the reactor water cleanup and liquid waste and condensate storage systems to enable sharing of the processing burden are acceptable. D. The RWCS should be designed to permit processing reactor water during periods of single active component failures or equipment downtime.					
5.4.8.2	The reactor water cleanup system should include the following: A. Means for automatically isolating the RWCS from the reactor coolant system in the event the liquid poison system is actuated for reactor shutdown. B. Means for automatically isolating the RWCS in the event the nonregenerative heat exchanger effluent temperature exceeds the prescribed resin operating temperature for the cleanup demineralizer resins. C. Means for automatically maintaining flow through filter / demineralizer beds to prevent bed loss in the event of low					

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	process flow or loss of process flow through the system. The recirculation loop and holding pump subsystem provided for precoating can serve this purpose if it is activated on loss of flow or low flow conditions. D. Means of transferring resins. Sight glass provisions (bull's eyes) are acceptable for monitoring resin transfers. Systems should be designed to prevent "resin traps" in sluice lines. Consideration should be given in the design of transfer lines to avoid resin traps; e.g., resin transfer lines may be designed to avoid resins collecting in valves, low points, or stagnant areas. E. Means for draining and venting RWCS components through a closed system (i.e., not to the immediate atmosphere) in accordance with the requirements of General Design Criteria 60 and 61. The system design should include vent lines that run to a ventilation duct exhausting from the plant for normal system operation when the probability of releases of radiological materials are minimal. F. Means of resin strainers in return lines to the reactor system or condensate system that are capable of removing resin particles contained in demineralizer effluents. G. Means to prevent inadvertent opening of the filter / demineralizer backwash valves during normal operation.					
5.4.8.3	To meet the requirements of GDC 1 and 2, Regulation Position C.2.c of Regulatory Guide 1.26 and C.1, C.2, C.3, and C.4 of Regulatory Guide 1.29 are applicable so that the portion of the RWCS extending from the reactor vessel and recirculation loops to the outermost drywell isolation valves should be designed to seismic Category I and Quality Group A. The remainder of the system outside the primary containment should be designed to Quality Group C and need not be seismic Category I. The precoating unit for demineralizers need not be designed to Quality Group C and need not be seismic Category I.					

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5.4.8.4	The RWCS should include equipment for monitoring: A. System effluent to ensure the conductivity is below the threshold for immediate reactor shutdown. Instrumentation should have sufficient range to measure conductivities up to the value requiring immediate shutdown of the reactor. B. Temperature upstream of the demineralizer, to ensure the ion exchange resin temperature limits are not exceeded. C. Differential pressure, to ensure the design limits on filter/ demineralizer septums and resin strainers are not exceeded.					
5.4.11, Rev. 3 (03/2007)	Pressurizer Relief Tank					
5.4.11.1	Acceptance as it relates to the protection of essential systems from the effects of earthquakes is based on meeting the guidelines in Position C.2 of Regulatory Guide 1.29 regarding the location of the tank in relation to other plant systems (the design of the tank system should be such that the plant safety-related systems would continue to perform their safety functions in the event of a tank failure) and in Position C.3 regarding the extension of seismic Category I boundaries.					
5.4.11.2	The staff uses the following specific criteria to determine whether the requirements of GDC 4 are met: A. The rupture disks have a relief capacity that at least equals the combined capacity of the pressurizer relief and safety valves, with sufficient allowance for rupture disk tolerance. B. The pressurizer relief tank volume and the quantity of water initially stored in the tank should be such that no steam or water will be released to containment under any normal operating conditions or AOOs. It should be assumed that the initial temperature of water inside the tank will be no lower than 49 EC (120 EF). Systems performing similar functions should also be shown to have					

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	no release to containment during normal operations and AOOs. C. The design of the pressurizer relief tank and rupture disk should accommodate full vacuum so that the tank will not collapse if the contents are cooled after a discharge of steam without the addition of nitrogen. D. Alarms for high temperature, high pressure, and high and low liquid levels for the pressurizer relief tank have been provided. Systems performing similar functions should also have appropriate instrumentation to inform the operator about the condition of the systems. E. The location of the tank should be such that the rupture discs do not pose a missile threat to safety-related equipment.					
5.4.12, Rev. 1 (03/2007)	Reactor Coolant System High Point Vents					
5.4.12.1	The reactor coolant vent design must ensure that use of these vents during and following an accident does not aggravate the challenge to containment or the course of the accident.					
5.4.12.2	Vent capability should be provided on high points of the RCS (including the pressurizer on PWRs and the hot legs on Babcock and Wilcox designs) to vent gases which may inhibit core cooling. For reactors with U-tube steam generators, procedures should be developed to remove sufficient gas from the U-tubes to ensure continued core cooling, since it is impractical to individually vent the thousands of U-tubes. In general, vent paths are not required for local high points at locations where gas accumulation would not be expected to jeopardize core cooling such as a reactor coolant pump valve body.					
5.4.12.3	A single failure of a vent valve, power supply, or control system should not prevent isolation of the vent path. On boiling water reactors, block valves are not required in lines with safety valves used for venting.					

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5.4.12.4	The design should incorporate sufficient redundancy to minimize the probability of inadvertent actuation. Other methods to reduce the chances of inadvertent actuation, such as removing power or administrative controls, may be considered.					
5.4.12.5	Since the RCS vent will be part of the RCPB, all requirements for the RCPB must be met.					
5.4.12.6	The size of the vent should be smaller than the size corresponding to the definition of a LOCA (Appendix A to 10 CFR Part 50, 10 CFR 52.47(a)(1)(ii), and 10 CFR 52.79(b)) to avoid unnecessary challenges to the ECCS, unless the applicant provides justification for a larger size.					
5.4.12.7	Vent paths to the containment should discharge into areas that provide good mixing with containment air and are able to withstand steam, water, noncondensibles, and mixtures of the above.					
5.4.12.8	The vent system should be operable from the control room and provide positive valve position indication. Power should be supplied from emergency buses.					
5.4.12.9	It is important that the control room displays and controls for the RCS vents do not increase the potential for operator error. A human-factor analysis should be performed that considers the following: A. The use of this information by an operator during both normal and abnormal plant conditions B. Integration into emergency procedures C. Integration into operator training D. Other alarms during an emergency and need for prioritization of alarms					
5.4.12.10	The design should have provisions for testing the operability of the reactor coolant vent system. Testing should be performed in accordance with Subsection IWV of Section XI of the ASME Code for Category B valves.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
5.4.12.11	The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) should be seismically and environmentally qualified in accordance with IEEE 344, as supplemented by Regulatory Guide 1.100 and Regulatory Guide 1.92. Environmental qualifications must be in accordance with 10 CFR 50.49.					
5.4.12.12	The reactor coolant vent system should be designed to withstand the dynamic loads that will be encountered during operation from high RCS pressure to the approximate atmospheric pressure at the vent system exhaust.					
5.4.12.135.4.12.13	Procedures to effectively operate the vent system must consider when venting is needed and when it is not needed. A variety of initial conditions for which venting may be required should be considered. Operator actions and the necessary instrumentation should be identified.					
5.4.12.14	The reactor coolant vent system should meet the quality assurance acceptance criteria provided in SRP Chapter 17.					
5.4.13 (03/2007)	Isolation Condenser System (BWR)					
5.4.13.1	GDC 2 provides the design bases for plant structures, systems and components (SSCs) for protection from natural phenomena, i.e., earthquakes, tornados, hurricanes, floods, tsunamis, and seiche without loss of capability or loss of safety function. In the application, consideration should be given to the historical data of the phenomena, potential combination of normal and accident conditions, and the importance of the safety functions performed by the SSCs. With respect to GDC 2, the application should demonstrate that SSCs are designed to withstand the effects of the above listed phenomena without loss of integrity or capability to perform their safety function. The application should demonstrate that all quality assurance requirements of 10 CFR 50, Appendix B have been applied to the activities affecting safety related functions of these SSCs.					

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5.4.13.2	With respect to GDC 4, the application should demonstrate compatibility of components with environmental conditions that are acceptable by compliance with the applicable provisions of the ASME Code and by compliance with the positions of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Regulatory Guide 1.44 contains staff positions related to unstabilized austenitic stainless steel of the AISI Type 3XX series used for components of the RCPB. Positions related to BWR piping materials, including verification of nonsensitization of the material by an approved test, are described in Attachment A to Generic Letter 88-01. The technical bases for the positions provided in Generic Letter 88-01 and similar recommendations related to minimizing stress corrosion cracking in susceptible piping of BWRs are detailed in NUREG-0313, Revision 2. Upon resolution of GSI-191, the review should include consideration of the resolution of this issue.					
5.4.13.3	Pursuant to GDC 5, SSCs that are important to safety should not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. With respect to GDC 5, the application should demonstrate that the ICS design's ability to accomplish these safety-related functions is not compromised for each unit regardless of equipment failures or other events that may occur in another unit.					
5.4.13.4	With respect to GDC 17, the application should demonstrate conformance with the guidelines in RG 1.93 with respect to providing onsite and offsite electric power systems to permit functioning of SSCs important to safety to ensure their safety function assuming either power system is not functioning. The application should demonstrate sufficient capacity and capability					

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	to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents and possible loss of power generated by the nuclear power plant.					
5.4.13.5	With respect to GDC 33, the application should demonstrate that the system can supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary and in the event that either onsite or offsite ac power is unavailable.					
5.4.13.6	With respect to GDC 34, the application should demonstrate that the system can transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design pressure of the reactor coolant pressure boundary are not exceeded. The application should demonstrate that the ICS provides the capability for decay heat removal, and that component redundancy, leak detection and isolation capabilities are provided. ICS operation should be assured with a single active failure including electric power.					
5.4.13.7	With respect to GDC 35, the application should demonstrate that the system design is capable of providing abundant emergency core cooling following any loss of coolant such that: (1) fuel and clad damage that could interface with continued effective core cooling is prevented and, (2) clad metal-water reaction is limited to negligible amounts. The application should demonstrate that the ICS provides water inventory and DHR capability following reactor shutdown, and that component redundancy, leak detection and isolation capabilities are provided. ICS operation should be assured with a single active failure including electric power.					
5.4.13.8	With respect to GDC 36, the application should demonstrate that the ECCS is designed in a manner permitting appropriate					

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	inspection of important components, to assure the readiness, integrity, and capability of the system.					
5.4.13.9	With respect to GDC 37, the application should demonstrate that the ICS is designed to permit appropriate periodic pressure and functional testing to assure: (1) structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.					
5.4.13.10	With respect to GDC 54, the application should demonstrate that piping systems penetrating primary containment are provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems. An acceptable design of such piping systems must have the capability to test periodically the operability of the isolation valves and associated apparatus and to determine whether valve leakage is within acceptable limits.					
5.4.13.11	With respect to 10 CFR 50.46, the application should demonstrate that an acceptable evaluation model is used so that the cooling performance is sufficient to assure that the most severe postulated LOCA is calculated. The application should demonstrate that: 1) the maximum fuel element cladding temperature shall not exceed 2200EF, 2) the maximum total cladding oxidation shall nowhere exceed 0.17 times the total cladding thickness before oxidation, 3) the total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding					

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	surrounding the plenum volume, were to react, 4) changes in core geometry shall be such that the core remains amenable to cooling, and 5) after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.					
5.4.13.12	For 10 CFR 50.63, the application should demonstrate compliance with RG 1.155, which identifies acceptable methods to support the plant’s ability to withstand for a specified duration and recover from an SBO.					
Branch Technical Position 5-1, Rev. 3 (03/2007)	Monitoring of Secondary Side Water Chemistry in PWR Steam Generators					
BPT 5-1.1	The applicant’s final safety analysis report (FSAR) should describe the implementation of a secondary water chemistry monitoring and control program in accordance with the supplier’s recommended procedure to inhibit steam generator corrosion and tube degradation. The applicant should address how its program meets industry guidelines (e.g., EPRI’s secondary water chemistry guidelines and Nuclear Energy Institute (NEI) 97-06). In addition, this program should cover all operational modes. Each of the modes should be defined with regard to percent rated thermal power and approximate temperature range.					
BPT 5-1.2	The secondary water chemistry monitoring and control program should identify a sampling schedule for critical parameters during each mode of operation, as well as the acceptance control criteria for these parameters. At a minimum, the program should control pH, cation conductivity, sodium, and dissolved oxygen. However, other parameters merit consideration, such as specific conductivity, chloride, fluoride, suspended solids, silica, total iron,					

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	copper, sulfate, lead, ammonia, and residual hydrazine. Additives to each steam generator should be controlled separately.					
BPT 5-1.3	The reviewer will evaluate the secondary water chemistry control and monitoring program of each individual plant. Significant deviations from the industry guidelines should be noted and justified technically. Records should be made of the monitored item values and should be made available for audit and inspection when deemed necessary.					
BPT 5-1.4	Routine changes to the secondary water chemistry control and monitoring program should be reported as part of the biannual FSAR update, as required by 10 CFR 50.71. Changes shall be evaluated in accordance with the requirements of 10 CFR 50.					
Branch Technical Position 5-2, Rev. 3 (03/2007)	Overpressure Protection of Pressurized-Water Reactors While Operating at Low Temperatures					
BTP 5-2.1	A system should be designed and installed that will prevent exceeding the applicable technical specifications and Appendix G limits for the RCS while operating at low temperatures. The system should be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to satisfy the technical specification limits, particularly while the RCS is in a water-solid condition.					
BTP 5-2.2	The low-temperature overpressure protection system should be operable during startup and shutdown conditions below the enable temperature, defined as the water temperature corresponding to a metal temperature of at least RT(NDT) + 50°C (90°F) at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations.					
BTP 5-2.3	The system should be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must demonstrate that the system will					

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	provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event (e.g., operator error, component malfunction) should not be considered as the single active failure. The analyses should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event. All potential overpressurization events should be considered when establishing the worst-case event. Some events may be prevented by using protective interlocks or by locking out power. These events should be identified individually. If the analysis excludes the events, the controls to prevent these events should be in the plant technical specifications.					
BTP 5-2.4	The design of the system should use Institute of Electrical and Electronics Engineers (IEEE) Standard 603 as guidance. The system may be manually enabled; however, an alarm should be provided to alert the operator to enable the system at the correct plant condition during cooldown. Positive indication should be provided to indicate when the system is enabled. An alarm should activate when the protective action is initiated. The reviewer responsible for instrumentation and controls will assist in reviews of the design criteria and the design for the low-temperature overpressure protection system controls and instrumentation, as described in Subsection I of SRP Section 5.2.2.					
BTP 5-2.5	To ensure operational readiness, the overpressure protection system should be testable. Technical specification surveillance requirements should include the following: A. A test performed to ensure operability of the system (exclusive of relief valves) before each shutdown. B. A test for valve operability, as a minimum, to be conducted as specified in the ASME Code Section XI.					

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BTP 5-2.6	The system must meet the requirements of Regulatory Guide 1.26 and Section III of the ASME Code.					
BTP 5-2.7	The design of the overpressure protection system should function during an operating-basis earthquake. It should not compromise the design criteria of any other safety-grade system with which it would interface, such that the requirements of Regulatory Guide 1.29 are met.					
BTP 5-2.8	The overpressure protection system should not depend on the availability of offsite power to perform its function. The system should be operable from battery-backed power sources, not necessarily Class 1E buses.					
BTP 5-2.9	Overpressure protection systems that take credit for active component(s) to mitigate the consequences of an overpressurization event should include additional analyses considering inadvertent system initiation/actuation or should provide justification that existing analyses bound such an event.					
BTP 5-2.10	If pressure relief is from a low-pressure system not normally connected to the primary system, interlocks that would isolate the low-pressure system from the primary coolant system should not defeat the overpressure protection function (see Branch Technical Position 7-1).					
Branch Technical Position 5-3, Rev. 2 (03/2007)	Fracture Toughness Requirements					
BTP 5-3.1	Preservice Fracture Toughness Test Requirements. The fracture toughness of all ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary shall be evaluated in accordance with the requirements of Appendix G, 10 CFR Part 50, as augmented by the criteria of Section III of the ASME Code. The fracture toughness test requirements for plants with construction permits prior to August 15, 1973 may not comply with the new Codes and Regulations in					

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	all respects. The fracture toughness of the materials for these plants should be assessed by using the available test data to estimate the fracture toughness in the same terms as the new requirements. This should be done because the operating limitations imposed on old plants should provide the same safety margins as are required for new plants. 1.1 Determination of RT_{NDT} for Vessel Materials Temperature limitations are determined in relation to a characteristic temperature of the material, RT_{NDT}, that is established from the results of fracture toughness tests. Both drop weight nil-ductility transition temperature (NDTT) tests and Charpy V-notch tests should be run to determine the RT_{NDT}. The NDTT temperature, as determined by drop weight tests (ASTM E-208-1969) is the RT_{NDT} if, at 33 °C (60 °F) above the NDTT, at least 68 J (50 ft-lbs) of energy and 0.89 mm (35 mils) lateral expansion (LE) are obtained in Charpy V-notch tests on specimens oriented in the weak direction (transverse to the direction of maximum working). In most cases, the fracture toughness testing performed on vessel material for older plants did not include all tests necessary to determine the RT_{NDT} in this manner. Acceptable estimation methods for the most common cases, based on correlations of data from a large number of heats of vessel material, are provided below for guidance in determining RT_{NDT} when measured values are not available. (1) If dropweight tests were not performed, but full Charpy V-notch curves were obtained, the NDTT for SA-533 Grade B, Class 1 plate and weld material may be assumed to be the temperature at which 41 J (30 ft-lbs) was obtained in Charpy V-notch tests, or -18 °C (0 °F), whichever was higher.					

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	(2) If dropweight tests were not performed on SA-508, Class II forgings, the NDTT may be estimated as the lowest of the following temperatures: (a) 33 °C (60 °F). (b) The temperatures of the Charpy V-notch upper shelf. (c) The temperature at which 136 J (100 ft-lbs) was obtained on Charpy V-notch tests if the upper-shelf energy values were above 136 J (100 ft-lbs). (3) If transversely-oriented Charpy V-notch specimens were not tested, the temperature at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) LE would have been obtained on transverse specimens may be estimated by one of the following criteria: (a) Test results from longitudinally-oriented specimens reduced to 65% of their value to provide conservative estimates of values expected from transversely oriented specimens. (b) Temperatures at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) LE were obtained on longitudinally-oriented specimens increased 11 °C (20 °F) to provide a conservative estimate of the temperature that would have been necessary to obtain the same values on transversely-oriented specimens. (4) If limited Charpy V-notch tests were performed at a single temperature to confirm that at least 41 J (30 ft-lbs) was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 61 J (45 ft-lbs) was obtained if the specimens were longitudinally oriented. If the minimum value obtained was less than 61 J (45 ft-lbs), the RT_{NDT} may be					

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	estimated as 11 °C (20 °F) above the test temperature. 1.2 Estimation of Charpy V-Notch Upper Shelf Energies For the beltline region of reactor vessels, the upper shelf toughness must account for the effects of neutron radiation. Reactor vessel beltline materials must have Charpy upper shelf energy, in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 102 J (75 ft-lbs) initially and must maintain Charpy upper shelf energy throughout the life of the vessel of no less than 68 J (50 ft-lbs). If Charpy upper shelf energy values were not obtained, conservative estimates should be made using results of tests on specimens from the first surveillance capsule removed. If tests were only made on longitudinal specimens, the values should be reduced to 65% of the longitudinal values to estimate the transverse properties. The predicted end-of-life Charpy upper shelf energy and adjusted reference temperature for the reactor vessel materials must meet the requirements of 10 CFR 50, Appendix G, paragraph IV.B. Reactor vessel materials that do not meet the specified end-of-life acceptance criteria are reviewed in accordance with paragraphs V.C and V.D of 10 CFR 50, Appendix G. NUREG-0744 provides an acceptable methodology for performance of fracture analysis for demonstrating adequate margins of safety for continued operation in accordance with 10 CFR Part 50, Appendix G, paragraph V.C.3. 1.3 Reporting Requirements					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	Fracture toughness information identified by the Code and by Appendix G, 10 CFR Part 50, should be reported in the FSAR to provide a basis for evaluating the adequacy of the operating limitations given in the Technical Specifications. In the case of older plants, the data may be estimated, using the procedures listed above, or other methods that can be shown to be conservative.					
BTP 5-3.2	Operating Limitations for Fracture Toughness 2.1 Pressure-Temperature Operating Limitations As required by Appendix G, 10 CFR Part 50, the following operating limitations shall be determined and included in the Technical Specifications. The basis for determination shall be reported, and is the responsibility of the applicant, but in no case shall the limitations provide less safety margin than those determined in accordance with Appendix G, 10 CFR Part 50, and Appendix G to Section III of the Code. (1) Minimum temperatures for performing any hydrostatic test involving pressurization of the reactor vessel after installation in the system. (2) Minimum temperatures for all leak and hydrostatic tests performed after the plant is in service. (3) Maximum pressure-minimum temperature curves for operation, including startup, upset, and cooldown conditions. (4) Maximum pressure-minimum temperature curves for core operation. 2.2 Recommended Bases for Operating Limitations					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	2.2.1 Leak and Hydrostatic Tests (1) It is recommended that no tests at pressures higher than design pressure be conducted with fuel in the vessel. (2) For system and component hydrostatic tests performed prior to loading fuel in the reactor vessel, it is recommended that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60 °F. (3) For system and component hydrostatic tests performed subsequent to loading fuel in the reactor vessel, the minimum test temperature should be determined as discussed in Section III of SRP 5.3.2. 2.2.2 Heatup and Cooldown Limit Curves Heatup and cooldown pressure-temperature limit curves may be determined using single pr/t stress calculations, using the method given in Appendix G of the Code. The effect of thermal gradients may be conservatively approximated by the procedures in Appendix G of the Code or from the report, “Tabulation of thermally-Induced Stress Intensity Factors (KIT) and Crack Tip Temperatures for Generating P-T Curves per ASME Section XI-Appendix G,” ORNL/NRC/LTR- 03/03. Calculations need only be performed for the beltline region, if the RT_{NDT} of the beltline is demonstrated to be adequately higher than the RT_{NDT} for all higher stressed regions. Alternatively, more rigorous analytical procedures may be used, provided that the intent of the Code is met, and adequate technical justification for all assumptions and bases is provided.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	2.2.3 Core Operation Limits To provide added margins during actual core operation, Appendix G, 10 CFR Part 50 requires a minimum temperature during core operation, and a 22 °C (40 °F) margin in temperature over the pressure-temperature limits as determined for heatup and cooldown in 2.2.2 above. The minimum temperature, regardless of pressure, is the temperature calculated for the inservice hydrostatic test according to 2.2.1 above. 2.2.4 Upset Conditions The pressure-temperature limits described in 2.2.2 and 2.2.3 above are applicable to upset conditions. Normal operating procedures should permit variations from intended operation, including all upset conditions, without exceeding the limit curves. 2.2.5 Emergency and Faulted Conditions It is recognized that the severity of a transient resulting from an emergency or faulted condition is not directly related to operating conditions, and resulting temperature-stress relationships in the reactor coolant boundary components are primarily system dependent, and therefore not under direct control of the operator. For these reasons, operating limits for emergency and faulted conditions are not a requirement of the Technical Specifications. The SAR should present descriptions of the continued integrity of all vital components of the RCPB during postulated faulted conditions. It is recommended that such descriptions be made in as realistic a manner as possible, avoiding grossly over conservative assumptions and procedures.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	2.3 Reporting Requirements The Technical Specifications should include the operating and test limits discussed above, and the basis for their determination. The Technical Specifications should also include information on the intended operating procedures, and justify that adequate margins between the expected conditions and the limit conditions will be provided to protect against unexpected or upset conditions.					
BTP 5-3.3	Inservice Surveillance of Fracture Toughness The reactor vessel may be exposed to significant neutron radiation during the service life. This will affect both the tensile and toughness properties. A material surveillance program in conformance with Appendix H, 10 CFR Part 50, must be carried out. 3.1 Surveillance Program Requirements The minimum requirements for the surveillance program are covered by Appendix H, 10 CFR Part 50. The selection of material to be included in the surveillance program should be in accordance with ASTM E-185-82, unless the intent of the program is better realized by using more rigorous criteria. For example, the approach of estimating the actual RTNDT and upper shelf toughness of each plate, forging, or weld in the beltline as a function of service life, and choosing as the surveillance materials those that are expected to be most limiting, may be preferable in some cases. This would include consideration of the initial RTNDT, the upper shelf toughness, the expected radiation sensitivity of the material (based on copper and nickel content, for example) and the neutron fluence expected at its location in the vessel. 3.2 SAR Criteria					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	With respect to the adequacy of the surveillance program, information requested for beltline materials includes the following: (1) Tensile properties. (2) Dropweight test and Charpy V test results used to determine RTNDT. (3) Charpy V test results to determine the upper shelf toughness. (4) Composition, specifically the copper and nickel content. (5) Estimated maximum fluence for each beltline material. (6) List of materials included in the surveillance program, with basis used for their selection. 3.3 Surveillance Test Procedures Surveillance capsules must be removed and tested at intervals in accordance with Appendix H, 10 CFR Part 50. The proposed removal and test schedule should be included in the Technical Specifications. 3.4 Reporting Criteria All information used to evaluate results of the tests on surveillance materials, evaluation methods, and results of the evaluation should be submitted with the evaluation report. This should include: (1) Original properties and compositions of the materials.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(2) Fluence calculations, including original predictions, for both surveillance specimens and vessel wall. (3) Test results on surveillance specimens. (4) Basis for evaluation of changes in RTNDT and upper shelf toughness. (5) Updated prediction of vessel properties. 3.5 Technical Specification Changes Changes in the operating and test limits recommended as a result of evaluating the properties of the surveillance material, together with the basis for these changes, shall be submitted to the Office of Nuclear Reactor Regulation for approval.					
BTP 5-3.4	Pressurized Thermal Shock (PWR only) 4.1 Pressurized Thermal Shock Requirements As required by 10 CFR 50.61, the following is a summary of requirements for the PWR reactor vessels: (1) RT_{PTS} values must be projected using end-of-life fluence for each weld, plate or forging in the reactor vessel beltline region. The projected EOL RT_{PTS} values must be approved by the NRC. (2) PTS screening criteria is 132 °C (270 °F) for plates, forgings, and axial weld materials, and 149 EC (300 EF) for circumferential weld materials. (3) If reactor vessel is projected to exceed the PTS screening criteria, 10 CFR 50.61(b)(3) requires the applicant to implement a flux reduction program that is reasonably practicable to avoid					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	exceeding the PTS screening criteria. (4) If the flux reduction program does not prevent the reactor vessel from exceeding the PTS screening criterion at the end of life, the applicant choose between the two options in 10 CFR 50.61 to meet PTS requirements: (a) submit a safety analysis pursuant to 10 CFR 50.61(b)(4) to determine what, if, any, modifications to equipment, systems, and plant operation to prevent failure of the reactor vessel from a postulated PTS event, (b) perform a thermal-annealing treatment of the reactor vessel pursuant 10 CFR 50.61(b)(7) to recover fracture toughness. 10 CFR 50.61 requires details of the approach selected to be submitted for NRC approval at least 3 years before the reactor vessel is projected to exceed the PTS screening criteria.					
Branch Technical Position 5-4, Rev. 4 (03/2007)	Design Requirements of the Residual Heat Removal System					
BTP 5-4.1	Functional Requirements The system(s) that can be used to take the reactor from normal operating conditions to coldshutdown[1] shall satisfy the following functional requirements: A. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems. These systems shall satisfy GDC 1 through 5. B. The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to ensure that for onsite electrical power system operation (assuming offsite power is not available) and offsite electrical power system operation (assuming onsite power					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	is not available) the system function can be accomplished assuming a single failure. C. The system(s) shall be capable of being operated from the control room (including instrumentation for monitoring and control functions) with either only onsite or offsite power available. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified. D. The system(s) shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure.					
	NOTE: 1. Processes involved in cooldown are heat removal, depressurization, flow circulation, and reactivity control. The cold shutdown condition, as described in the Standard Technical Specifications, refers to a subcritical reactor with a reactor coolant temperature no greater than 93.3°C (200°F) for a pressurized-water reactor (PWR) and 100°C (212°F) for a boiling-water reactor.					
BTP 5-4.2	RHR System Isolation Requirements The RHR system shall satisfy the following isolation requirements: A. The following shall be provided in the suction side of the RHR system to isolate it from the RCS: i. Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	ii. The valves shall have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Failure of a power supply shall not cause any valve to change position. iii. The valves should have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of the RHR system, to the extent that such interlocks will not degrade high system reliability during shutdown operations (see Generic Letter 88-17). B. One of the following shall be provided on the discharge side of the RHR system to isolate it from the RCS: i. The valves, position indicators, and interlocks described in items 1(a) through 1(c) above. ii. One or more check valves in series with a normally closed power-operated valve. The power-operated valve position shall be indicated in the control room. If the RHR system discharge line is used for an ECCS function, the power-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure. iii. Three check valves in series. iv. Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leak tightness and the testing is performed at least annually.					
BTP 5-4.3	Pressure Relief Requirements The RHR system shall satisfy the following pressure relief					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	requirements: A. To protect the RHR system against accidental over pressurization when it is in operation (not isolated from the RCS), pressure relief in the RHR system shall be provided with relieving capacity in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The most limiting pressure transient during the plant operating condition when the RHR system is not isolated from the RCS shall be considered when selecting the pressurerelieving capacity of the RHR system. For example, during shutdown cooling in a PWR with no steam bubble in the pressurizer, inadvertent operation of an additional charging pump or inadvertent opening of an ECCS accumulator valve should be considered in the selection of the design bases. B. Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not do the following: i. Result in flooding of any safety-related equipment ii. Reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated loss-of-coolant accident iii. Result in a nonisolatable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of the containment C. If interlocks are provided to automatically close the isolation valves when the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
BTP 5-4.4	Pump Protection Requirements The design and operating procedures of any RHR system shall have provisions to prevent damage to the RHR system from overheating, cavitation, or loss of adequate pump suction fluid.					
BTP 5-4.5	Test Requirements The isolation valve operability and interlock circuits must be designed so as to permit online testing when operating in the RHR mode. Testability shall meet the requirements of Institute of Electrical and Electronics Engineers Std 338-1987 and Regulatory Guide 1.22. The preoperational and initial startup test program shall be in conformance with Regulatory Guide 1.68. The programs for PWRs shall include tests with supporting analysis to (1) confirm that adequate mixing of borated water added before or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (2) confirm that cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with the performance of previously tested plants of similar design may be substituted for these tests.					
BTP 5-4.6	Operational Procedures The operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with Regulatory Guide 1.33. For PWRs, the operational procedures shall include specific procedures and information required for cooldown under natural circulation conditions. These natural					

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	circulation cooldown procedures and analyses should consider the potential for a voiding event in the reactor vessel head and incorporate appropriate controls to address such an occurrence (Generic Letter 92-02).					
BTP 5-4.7	Auxiliary Feedwater Supply The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least 4 hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure.					
BTP 5-4.8	Implementation For the purposes of implementing the requirements for plant heat removal capability to comply with this position, plants are divided into the following three classes: Class 1 Full compliance with this position for all plant applications that are docketed on or after January 1, 1978. See Table 1 for possible solutions for full compliance. Class 2 Partial implementation of this position for all plants (custom or standard) for which construction permit or PDA applications are docketed before January 1, 1978, and for which issuance of an operating license is expected on or after January 1, 1979. See Table 1 for recommended implementation for Class 2 plants. Class 3 The extent to which the implementation guidance in Table 1 will be backfitted for all operating reactors and all other plants					

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	(custom or standard) for which issuance of the operating license occurred before January 1, 1979, will be based on the combined I&E and DOR review of related plant features for operating reactors.					
	CHAPTER 6, Engineered Safety Features					
6.1.1, Rev. 2 (03/2007)	Engineered Safety Features Materials					
6.1.1.1	Materials and Fabrication. To meet the requirements of GDC 1 and 10 CFR 50.55a to assure that SSCs important to safety shall are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed, codes and standards should be identified and records maintained. The materials specified for use in these systems must be as given in Parts A, B and C of Section II of the ASME Code and Appendix I to Section III, Division 1 of the Code. Regulatory Guide (RG) 1.84 describes acceptable Code Cases that may be used in conjunction with the above specifications. Fracture toughness of the materials should be as stated in SRP Section 10.3.6, “Steam and Feedwater System Materials,” subsection II.1. A. Austenitic Stainless Steels. To meet the requirements of GDC 4 relative to compatibility of components with environmental conditions; GDC 14 with respect to fabrication and testing of the RCBP such that there is an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture; and the quality assurance requirements of Appendix B of 10 CFR Part 50, the following guidelines should be used:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	i. RG 1.44 describes acceptable criteria for preventing intergranular corrosion of stainless steel components of the ESF. Furnace-sensitized material should not be allowed in the ESF, and methods described in this guide should be followed for testing the materials prior to fabrication, and for ensuring that no deleterious sensitization occurs during welding. ii. RG 1.31 describes acceptable criteria for assuring the integrity of welds in austenitic stainless steel ESF components. The control of delta ferrite content of weld filler metal is specified in this guide, which sets forth an acceptable basis for delta ferrite content of weld filler metal. iii. The controls for abrasive work on austenitic stainless steel surfaces should, at a minimum, be equivalent to the controls described in RG 1.37, position C.5 to prevent contamination, which promotes stress corrosion cracking. Tools that contain materials that could contribute to intergranular or stress-corrosion cracking or which, because of previous usage, may have become contaminated with such materials, should not be used on austenitic stainless steel surfaces. iv. Criteria to assure adequate resistance to intergranular stress corrosion cracking (IGSCC) for susceptible boiling water reactors (BWR) austenitic stainless steel ESF piping are described in NUREG-0313 and in Attachment A to Generic Letter (GL) 88-01. The technical bases for the positions provided in GL 88-01 are detailed in NUREG-0313. These criteria are applied to piping specified in GL 88-01. GL 88-01 and NUREG-0313 criteria used for the evaluation of initial material selection and fabrication include welding controls (e.g., delta ferrite content limits) and material specifications (e.g., carbon content specifications) that are more stringent than specified in RGs 1.31 and 1.44 and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	should supplant the regulatory guides to assure adequate resistance of susceptible piping to IGSCC. B. Ferritic Steel Welding. To meet the requirements of GDC 1 related to general quality assurance and codes and standards; Appendix B to 10 CFR Part 50, related to control of special processes; and 10 CFR 50.55a, the following acceptance criteria for ferritic steel welding should be used: i. The amount of minimum specified preheat must be in accordance with the recommendations of the Code, Section III, Appendix D, Article D-1000, and RG 1.50, unless an alternate procedure is justified. ii. Moisture control on low hydrogen welding materials shall conform to the requirements of the Code, Section III, Articles NB, NC, ND-2000 and 4000, and AWS D1.1, unless alternate procedures are justified. iii. For areas of limited accessibility, the criteria of Regulatory Guide 1.71 apply a discussed in SRP Section 10.3.6.					
6.1.1.2	Composition and Compatibility of ESF Fluids. In meeting the requirements of GDC 4 and 41 that SSCs important to safety are designed to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including loss-of-coolant accidents, and to assure that the concentration of hydrogen in the containment atmosphere following postulated accidents is controlled to maintain containment integrity, hydrogen generation resulting from the corrosion of metals by containment sprays during a design-basis accident should be controlled as described in RG 1.7, position C.6.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A. Pressurized Water Reactors (PWRs). To meet the requirements of GDC 4, 14, and 41, the composition of containment spray and core cooling water should be controlled to ensure a minimum pH of 7.0, as addressed in Branch Technical Position (BTP) 6-1, “pH for Emergency Coolant Water for PWRs.” Experience has shown that maintaining the pH of borated solutions at this level will help to inhibit initiation of stress corrosion cracking of austenitic stainless steel components. Hydrogen generation from the corrosion of materials within containment, such as aluminum and zinc, depends upon the corrosion rate, which in turn depends upon such factors as the coolant chemistry, the coolant pH, the metal and coolant temperature, and the surface area exposed to attack by the coolant. The assumed corrosion rates of materials in containment should be consistent with standard corrosion rate data. B. Boiling Water Reactors (BWRs). To meet the requirements of GDC 4, 14, and 41, the water used in the ESF systems should be controlled to provide assurance against stress corrosion cracking of unstabilized austenitic stainless steel components. Water used for emergency core cooling systems and spray systems should be controlled to ensure the following limits: Conductivity ≤0.5 mS/m (≤5 μmhos/cm) @ 25 °C Chloride (Cl^-) < 0.20 ppm pH = 5.3 to 8.6 @ 25 °C Hydrogen generation in BWR containments is assumed to follow the same characteristics as in pressurized water reactors (PWRs) in that the rates of hydrogen generation will rise with increasing					

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	zinc corrosion as the temperature rises, and will change with any change in pH.					
6.1.1.3	Component and Systems Cleaning. To meet the requirements of Appendix B to 10 CFR Part 50, Criteria IX and XIII, measures should be established to control the cleaning of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. Components and systems should be cleaned in conformance with the positions of RG 1.37.					
6.1.1.4	Thermal Insulation. To meet the requirements of GDC 1, 14, and 31, the RCPB should be designed, fabricated, erected, and tested in conformance with the following guidelines, such that there is an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture: A. The composition of nonmetallic thermal insulation on ESF components should be controlled as described in RG 1.36. B. The use of nonmetallic insulation on nonaustenitic stainless steel components should be controlled as described in RG 1.36. Moisture dripping from wet insulation can affect austenitic stainless steel components at lower elevations. C. Concentrations of leachable contaminants and added inhibitors should be controlled as specified in position C.2.b and Figure 1 of RG 1.36 to reduce the probability of stress corrosion cracking of austenitic stainless steel components.					
6.1.2, Rev. 3 (03/2007)	Protective Coating Systems (Paints) - Organic Materials					

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6.1.2.1	Appendix B to 10 CFR Part 50 requires a quality assurance program which comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. It is important to prevent the deterioration of protective coatings by one, all, or a combination of the following conditions: ionizing radiation; contamination by radioactive nuclides and subsequent decontamination processes; chemical and water sprays; high-temperature; high-pressure steam; and abrasion or wear. The protective coatings must be resistant to causing generation of combustible gases like hydrogen and methane and gaseous formation of radioactive organic iodides. If the protective coatings deteriorate by flaking, peeling, etc., they may form solid debris which can reach the containment recirculation sump and have a negative impact on the performance of post-accident cooling safety systems. Regulatory Guide 1.54, Rev. 1, describes an acceptable method of complying with the quality assurance requirements in regard to protective coatings applied to ferritic steels, aluminum, stainless steel, zinc-coated (galvanized) steel, concrete, or masonry surfaces of nuclear facilities. Compliance with Appendix B to 10 CFR Part 50 is important to ensure the overall quality and safety performance of protective coatings under normal and accident conditions.					
6.2.1, Rev. 3 (03/2007)	Containment Functional Design					
	Specific SRP acceptance criteria are provided in other SRP Sections.					
6.2.1.1.A, Rev. 3 (03/2007)	PWR Dry Containments, Including Subatmospheric Containments					
6.2.1.1.A.1	To satisfy the requirements of GDC 16 and 50 regarding sufficient design margin, for plants at the construction permit (CP) stage of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a loss-of-coolant accident, or a steam or					

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	feedwater line break. For plants at the operating license (OL) stage of review, the peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break, should be less than the containment design pressure. In general, the peak calculated containment pressure should be approximately the same as at the construction permit or design certification stage of review. However, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in the margin.					
6.2.1.1.A.2	To satisfy the requirements of GDC 38 to rapidly reduce the containment pressure, the containment pressure should be reduced to less than 50% of the peak calculated pressure for the design basis loss-of-coolant accident within 24 hours after the postulated accident. If analysis shows that the calculated containment pressure may not be reduced to 50% of the peak calculated pressure within 24 hours, the organization responsible for SRP Section 15.0.3 should be notified.					
6.2.1.1.A.3	To satisfy the requirement of GDC 38 to rapidly reduce the containment pressure, the containment pressure for subatmospheric containments should be reduced to below atmospheric pressure within one hour after the postulated accident, and the subatmospheric condition maintained for at least 30 days.					
6.2.1.1.A.4	To satisfy the requirements of GDC 38 and 50 with respect to the containment heat removal capability and design margin, the loss-of-coolant accident analysis should be based on the assumption of loss of offsite power and the most severe single failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure.					
6.2.1.1.A.5	To satisfy the requirements of GDC 38 and 50 with respect to the containment heat removal capability and design margin, the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	containment response analysis for postulated secondary system pipe ruptures should be based on the most severe single active failure in the containment heat removal systems (e.g., a fan, pump, or valve failure) or the secondary system isolation provisions (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis. Acceptable methods for the calculation of the containment environmental response to main steam line break accidents are found in NUREG-0588.					
6.2.1.1.A.6	To satisfy the requirements of GDC 38 and 50 with respect to the functional capability of the containment heat removal systems and containment structure under loss-of-coolant accident conditions, provisions should be made to protect the containment structure against possible damage from external pressure conditions that may result, for example, from inadvertent operation of containment heat removal systems. The provisions made should include conservative structural design to assure that the containment structure is capable of withstanding the maximum expected external pressure; or interlocks in the plant protection system and administrative controls to preclude inadvertent operation of the systems. If the containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event.					
6.2.1.1.A.7	In accordance with the requirements of GDC 13 and 64, and 10 CFR 50.34(f)(2)(xvii) (for those applicants subject to 10 CFR 50.34(f)), instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water level and temperature following an accident. The					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. See Item II.F.1 of NUREG-0737 and NUREG-0718, and Branch Technical Position 7-10, "Guidance on Application of Regulatory Guide 1.97."					
6.2.1.1.A.8	In accordance with 10 CFR 50.46 Appendix K, I.D.2, the minimum calculated containment pressure should not be less than that used in the analysis of the emergency core cooling system capability (See SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies").					
6.2.1.1.A.9	In accordance with GDC 4, containment internal structures and system components (e.g., reactor vessel, pressurizer, steam generators) and supports should be designed to withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment subcompartments (See SRP Section 6.2.1.2, "Subcompartment Analysis").					
6.2.1.1.A.10	In meeting the requirements of 10 CFR 50.34(f)(3)(v)(A)(1), applicants subject to this section should evaluate an accident that releases hydrogen generated from a 100% fuel clad metal-water reaction. The evaluation should demonstrate that the appropriate article for service level C limits (considering pressure and dead load only), for either concrete or steel containments, from ASME Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from hydrogen burning in containment should be analyzed.					
6.2.1.1.B, Rev. 3 DRAFT (04/1996)	Ice Condenser Containments					
	a.[23] In meeting the requirements of General Design Criteria 16, 38, and 50 regarding the functional capability of the containment and associated heat removal system to preserve containment					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	integrity under postulated high-energy line break accident conditions, the containment pressure and temperature response should be calculated using the LOTIC-1 (or an equivalent) computer code (Reference. 22).[24] For plants under review for construction permits, the containment design pressure should provide at least a 20% margin above the highest calculated accident pressure. For plants under review for operating licenses, the highest calculated accident pressure should not exceed the design pressure of the containment. The containment pressure and temperature response to postulated secondary system pipe ruptures should be based on the most severe single active failure of the isolation provisions in the secondary system (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature, depending on the purpose of the analysis. Acceptable methods for the calculation of the containment environmental response to main steam line break accidents are found in NUREG-0588 (Reference. 29).[25] b. In meeting the requirements of General Design Criterion 50 regarding the integrity of containment internal structures, the containment subcompartment or control volume differential (internal) pressures should be calculated using the Transient Mass Distribution (TMD) computer code as described in the proprietary report WCAP-8077[26] (Reference. 19)[27], without the augmented critical flow correlation. The TMD calculation[28] should incorporate the heat transfer correlation developed from the 1974 full-scale ice condenser tests and should include the compressibility factor "Y" in the incompressible flow equation.					

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	For plants being reviewed for construction permits, the design differential pressures for all ice condenser control volumes or subcompartments, and system components (e.g., reactor vessel, pressurizer, steam generators) and supports, should provide at least a 40% margin above the highest calculated differential pressures. For plants being reviewed for operating licenses, the highest calculated differential pressures for all ice condenser control volumes or subcompartments should not exceed the corresponding design differential pressures. The operating deck, steam generator and pressurizer enclosures, and ice condenser lower inlet doors should be designed to withstand the maximum calculated reverse differential pressures between the upper and lower compartments using the LOTIC-2 computer code (Reference. 23).[29] To account for uncertainties in the analysis of reverse differential pressures, an adequate margin should be provided above the maximum calculated reverse differential pressure. c. In meeting the requirements of General Design Criteria 16 and 38 regarding the functional capability of the containment heat removal system to reduce rapidly, and without exceeding containment design conditions, the containment pressure and temperature under postulated accident conditions, the maximum allowable area for steam bypass of the ice condenser should be greater than the identifiable bypass area for the plant (e.g., the drainage provisions to allow containment spray water to return from the upper compartment to the sumps in the lower compartment). The bypass area capability of the plant should be based on analyses of the spectrum of postulated reactor coolant system pipe breaks, and should be about 3.3 square meters (35 square feet) [30] or greater.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	d. In meeting the requirements of General Design Criteria 39 and 40 regarding the inspection and testing of containment heat removal systems, the design of the ice condenser system and return fan system should incorporate provisions for periodic inservice inspection and testing of essential system components; e.g., the ice baskets and doors, the ice condenser temperature monitoring system, the available mass of ice, and return air fan performance and controls. e. In meeting the requirements of General Design Criterion 16 regarding the containment design conditions important to safety, inadvertent operation of engineered safety features (e.g., the return air fan system or the containment spray system) should not cause the external design pressure of the primary containment to be exceeded. This may be accomplished through conservative containment design, use of vacuum relief devices, or electrical interlocks that preclude inadvertent operation of the spray and fan systems. f. In meeting the requirements of General Design Criteria 13 and 64, and 10 CFR 50.34(f)(2)(xvii) (for those applicants subject to 10 CFR 50.34(f)),[31] instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water level and temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. Item II.F.1 of NUREG-0737 and NUREG-0718 (References 24 and 25)[32], and Regulatory Guide 1.97, "Instrumentation For Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," should be followed. g. The minimum calculated containment pressure as determined					

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	by the LOTIC-2 Code [33] should not be less than that used in the analysis of the emergency core cooling system capability (see SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies"). h. In meeting the requirements of 10 CFR 50, §50.34(f)(3)(v)(A)(1), applicants subject to this article should evaluate an accident that releases hydrogen generated from a 100% fuel clad metal-water reaction. The evaluation should demonstrate that the appropriate article for service level C limits (considering pressure and dead load only), for either concrete or steel containments, from ASME Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from either hydrogen burning in containment or initiation of the post-accident inerting system, if installed, should be analyzed. Unless specifically known, the post-accident inerting gas should be assumed to be carbon dioxide.[34] i. In meeting the requirements of 10 CFR 50, §50.34(f)(3)(v)(B)(1), applicants subject to this article should evaluate the containment design's capability to withstand full actuation of the post-accident inerting system, if installed. The peak pressure caused by inadvertent actuation of the post-accident inerting system should be less than the containment design pressure.[35]					
	NOTES: 23. Specific acceptance criteria were changed from a number format to a letter format. Numbers are already used above in the general acceptance criteria. Using numbers for both could lead to confusion when referencing specific criterion. This change is consistent with other SRP sections. 24. Format change to make the citation of references consistent					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	with the SRP-UDP format requirements. Additionally, this reference cannot be verified to be the most current reference that is still being used by the NRC. Also, corrected the reference number to be consistent with changes to the SRP 6.2.1 Reference section. 25. Format change to make the citation of references consistent with the SRP-UDP format requirements. Also, corrected the reference number to be consistent with changes to the SRP 6.2.1 Reference section. 26. Added the phrase "as described in the proprietary report WCAP-8077" for clarity and to be consistent with a later citation of the same reference. 27. Format change to make the citation of references consistent with the SRP-UDP format requirements. Additionally, this reference cannot be verified to be the most current reference that is still being used by the NRC. Also, corrected the reference number to be consistent with changes to the SRP 6.2.1 Reference section. 28. Changed "TMD" to "The TMD calculation" since this sentence is referring to a calculation to be performed with the TMD code, not the code itself. 29. Format change to make the citation of references consistent with the SRP-UDP format requirements. Additionally, this reference cannot be verified to be the most current reference that is still being used by the NRC. Also, corrected the reference number to be consistent with changes to the SRP 6.2.1 Reference section. 30. The existing criteria of 35 square feet for the approximate size					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of the ice condenser steam bypass area was converted to 3.3 square meters using the guidance of Federal Standard 376B. See enclosed conversion documentation 31. Added citation of 10 CFR 50.34(f)(2)(xvii) related to the existing citation of II.F.1 of NUREG 0737/NUREG 0718. 32. Format change to make the citation of references consistent with the SRP-UDP format requirements. 33. Format change to make the citation of references consistent with the SRP-UDP format requirements. 34. Added a specific criterion for 10CFR50.34(f)(3)(v)(A)(1) regarding designing containment to meet hydrogen burning or post-accident inerting system actuation during an accident.					
	REFERENCES: 19. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. 22. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." 23. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following An Accident." 24. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance." 25. NRC Safety Evaluation Report, Babcock and Wilcox Company, Reference Safety Analysis Report, B-SAR-205, May 1978. 29. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.					
6.2.1.1.C, Rev.	Pressure-Suppression Type BWR Containments					

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7 (03/2007)						
6.2.1.1.C.1	In meeting the requirements of GDC 16 and 50 regarding the design margin for BWR pressure-suppression plants at the operating license stage of review, the peak calculated values of pressure and temperature for the drywell and wetwell should not exceed the respective design values. Also, the peak deck differential pressure for Mark II plants should not exceed the design value. Acceptable methods for the calculation of BWR pressure-suppression containment environmental response to loss-of-coolant accidents are found in NUREG-0588. For Mark III plants, the calculated results for drywell pressure and temperature, containment pressure and temperature, and differential pressure between the drywell and containment should be based on the General Electric Mark III analytical model that was used in the ABWR and Grand Gulf analyses. The use of this model at the construction permit stage is acceptable if an appropriate margin (see below) between the calculated and design differential pressures is used. The Mark III analytical model has been verified by the large-scale Mark III test results. If an analytical model other than the General Electric Mark III analytical model identified above is used, the model should be demonstrated to be physically appropriate and conservative to the extent that the General Electric model has been found acceptable. In addition, it will be necessary to demonstrate its performance with suitable test data in a manner similar to that described above. For ABWR plants, the calculated results for containment short-term and long-term response to postulated line breaks are based on the General Electric Mark III (ABWR) analytical model that was used in the ABWR standard plant analysis evaluated by the NRC in the ABWR FSER.					

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	For Mark III plants at the construction permit stage, the containment design pressure should provide at least a 15% margin above the peak calculated containment pressure, and the design differential pressure between drywell and containment should provide at least a 30% margin above the peak calculated differential pressure. For BWR pressure-suppression plants at the operating license stage, the peak calculated containment pressure and differential pressure should be less than the design values. In general, it is expected that the peak calculated pressures will be about the same as at the construction permit stage. However, it is possible that the margins may be affected by revised or improved analytical models, test results, or minor changes in the as-built design of the plant.					
6.2.1.1.C.2	In meeting the requirement of GDC 4, regarding the dynamic effects associated with normal and accident conditions, calculation of dynamic loads should be based on appropriate analytical models and supported by applicable test data. Consideration should be given to loads on suppression pool retaining structures and structures which may be located directly above the pool, as a result of pool motion during a loss-of-coolant accident or following actuation of one or more reactor coolant system safety/relief valves. The acceptability of pool dynamic loads for plants with Mark I containments is based on conformance with NRC acceptance criteria found in NUREG-0661. The acceptability of loss-of-coolant accident related pool dynamic loads for plants with Mark II containments is based on conformance with the generic loads previously reviewed and found acceptable by the NRC and NRC acceptance criteria. The loss-of-coolant accident related pool dynamic loads and criteria					

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	are as discussed in NUREG-0808, and Appendix B to this SRP section. Pool dynamic loads and criteria associated with the actuation of one or more reactor coolant system safety/relief valves are specified in Appendix A of NUREG-0802. The acceptability of pool dynamic loads for plants with Mark III containments is based on conformance with the NRC acceptance criteria identified in Appendix C of NUREG-0978. For Mark III plants at the construction permit stage, conformance with the NRC acceptance criteria can be demonstrated if a previously analyzed Mark III plant has sufficient similarity in plant characteristics to make the analyses performed for that plant design applicable to the Mark III plant design under consideration. The acceptability of pool dynamic loads associated with the actuation of one or more reactor coolant system safety/relief valves in Mark III containment are specified in Appendix B of NUREG-0802. The acceptability of pool dynamic loads for plants with ABWR containments is based on the GE analytical model provided in Appendix 3B of the ABWR SSAR which, in part, conforms with NUREGS 0802, 0808, and 0978. This model was used in the standard plant analysis and evaluated by the NRC in the ABWR FSER.					
6.2.1.1.C.3	In meeting the requirements of GDC 16 and 50 regarding the containment design margin for Mark III and ABWR plants, high energy lines passing through the containment should be provided with guard pipes or enclosed in other types of protective structures to assure that the suppression pool is not bypassed. If guard pipes are used, they should be designed in accordance with acceptance criteria set forth in SRP Section 3.6.2. The allowable leakage areas for steam bypass of the suppression pool should be determined for a spectrum of postulated reactor coolant					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	system pipe breaks. The maximum allowable bypass area of the plant should be based on conservative analyses which consider available energy removal mechanisms and the containment design pressure.					
6.2.1.1.C.4	In meeting the requirement of GDC 53 regarding periodic testing at containment design pressure for Mark I, II, and III containments, the maximum allowable leakage area for steam bypass of the suppression pool should be greater than the technical specification limit for leakage measured in periodic drywell-wetwell leakage tests. Specific acceptance criteria for the three types of containments are discussed in Appendix A. Plants with ABWR containments should follow the specific acceptance criteria for Mark II containments.					
6.2.1.1.C.5	In meeting the requirement of GDC 50 with respect to the design leakage rate for Mark III containments, justification should be provided for any reduction in the containment leak rate claimed for times less than 30 days after a postulated pipe break accident. This also includes meeting the regulatory position C.1.e of Regulatory Guide 1.3. For plants with ABWR containments, the design leakage rate for primary containment should be assumed for the duration of the loss-of-coolant accident consistent with Regulatory Guide 1.3.					
6.2.1.1.C.6	In meeting the requirement of GDC 16, provisions should be made in one of the following ways to protect the drywell and wetwell (or containment) of Mark I, II, III, and ABWR plants, and the operating deck of Mark II plants, against loss of integrity from negative pressure transients or post accident atmosphere cooldown: A. Structures should be designed to withstand the maximum calculated external pressure. B. Vacuum relief devices should be provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code,					

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	Section III, Subsection NE, to assure that the external design pressures of the structures are not exceeded. The vacuum relief valve guidelines are set forth in Appendix A to this SRP section.					
6.2.1.1.C.7	In meeting the requirements of GDC 50, with respect to design margin for item 6. above, the external design pressures of the structures, including the design upward deck differential pressure for Mark II plants, should provide an adequate margin above the maximum calculated external pressures to account for uncertainties in the analyses.					
6.2.1.1.C.8	In meeting the requirements of GDC4, the acceptability of the reactor coolant system safety/relief valve in-plant confirmatory test program shall be based on conformance with the guidelines specified in Section 6, 7, and 8 of NUREG-0763. If the applicant/licensee elects not to perform the SRV in-plant tests, the acceptability of this exception shall be determined in conformance with the guidelines specified in Section 4 of NUREG-0763.					
6.2.1.1.C.9	NUREG-0783 specifies that, for BWR pressure-suppression plants, the local suppression pool temperature should not exceed 93 C (200 F) or the acceptance criteria specified in Section 5.1 of NUREG 0783. This criterion may be eliminated provided that the SRV discharges are delivered to the suppression pool through a "T" or "X" quencher device previously approved by the staff and described in NUREG-0802 and NUREG-0978. NEDO-30832 concluded that unstable condensation oscillation loads due to suppression pool temperatures approaching the saturation temperature are bounded by air clearing hydrodynamic loads when the "T" or "X" quencher is used. The NRC review and approval of this conclusion is documented in a August 29, 1994 safety evaluation.					

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	This NRC safety evaluation also stated that there was no basis for permitting the deletion of local pool temperature requirements when a plant has an Emergency Safety Feature (ESF) pump inlet located at or above the quencher elevation due to concern that steam discharged from the quencher may be ingested at the pump inlet and cause pump cavitation or a water hammer. An analysis based on the plant specific geometry of the quenchers and pump intakes may be used to demonstrate that a steam plume discharged from the quencher will not be ingested by the pump intakes.					
6.2.1.1.C.10	In meeting the requirements of General Design Criteria 13 and 64, and 10 CFR 50.34(f)(2)(xvii) (for those applicants subject to 10 CFR 50.34(f)), instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the suppression pool water level and temperature following an accident. Regulatory guidance is contained in Branch Technical Position 7-10, "Guidance on Application of Regulatory Guide 1.97."					
6.2.1.1.C.11	In meeting the requirements of 10 CFR 50.34(f)(3)(v)(A)(1), applicants subject to this section should evaluate an accident that releases hydrogen generated from a 100% fuel clad metal-water reaction. The evaluation should demonstrate that the appropriate articles for service level C limits (considering pressure and dead load only), for either concrete or steel containments, from ASME Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from hydrogen burning in containment should be analyzed.					
6.2.1.2, Rev. 3 (03/2007)	Subcompartment Analysis					
6.2.1.2.1	Nodalization Schemes. Subcompartment nodalization schemes should be chosen so that					

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	there is no substantial pressure gradient within a node. A sensitivity study which includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes should be used to verify the nodalization scheme. The guidelines of Section 3.2 of NUREG-0609 (Ref. 1) should be followed and a nodalization sensitivity study should be performed, which should include the consideration of spatial pressure variations (e.g., pressure variations circumferentially, axially, and radially within the subcompartment). These variations are use to calculate the transient forces and moments acting on components.					
6.2.1.2.2	Initial Thermodynamic Conditions. The initial atmospheric conditions within a subcompartment should maximize the resultant differential pressure. An acceptable model would assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. If the assumed initial atmospheric conditions differ from this model, the selected values should be justified by the applicant. Another acceptable model that may be used for a restricted class of subcompartments involves simplifying the air model outlined above. In this case, the initial atmosphere within the subcompartment is modeled as a homogeneous water-steam mixture with an average density equivalent to the dry air model. This approach should be limited to subcompartments that have choked flow within the vents because the adequacy of this simplified model for subcompartments having primarily subsonic flow through the vents has not been established.					
6.2.1.2.3	Vent Flow Path and Distribution of Mass and Energy Released. Assumptions with regard to the distribution of mass and energy release should be biased towards maximizing the subcompartment pressure. The vent flow behavior through all					

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	flowpaths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100-percent water entrainment. In addition, the selected vent critical flow correlation should be conservative with respect to available experimental data. Currently acceptable vent critical flow correlations are the "frictionless Moody" (Ref. 2), with a multiplier of 0.6 for water-steam mixtures, and the thermal homogeneous equilibrium model for air-steam-water mixtures. If vent flowpaths are used that are not immediately available at the time of pipe rupture, the following criteria apply: A. The vent area and resistance as a function of time after the break should be based on a dynamic analysis of the subcompartment pressure response to pipe ruptures. B. The validity of the analysis should be supported by experimental data, or a testing program should be proposed at the construction permit or DC stage that will support this analysis. C. To meet the requirements of GDC 4, the safety analysis should consider the effects of missiles that may be generated during the transient.					
6.2.1.2.4	Design Pressure. For the review of a construction permit (CP) preliminary safety analysis report (PSAR) or a factor of 1.4 should be applied to the peak differential pressure which is calculated in a manner acceptable to the reviewer for the subcompartment structure, and the enclosed components for use in the design of the structure and the component supports. For the review of the operating license (OL) final safety analysis report (FSAR), DC or COL FSAR, the peak calculated differential pressure should not exceed the design pressure. It is expected that the peak calculated differential pressure will not be substantially different					

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	from that of the construction permit. However, improvements in the analytical models or changes in the as-built subcompartment may affect the available margin.					
	REFERENCES: 1. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981. 2. F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Jour. Of Heat Transfer, Trans. Am. Soc. of Mechanical Engineers, Vol. 87, No. 1, February 1965.					
6.2.1.3, Rev. 3 (03/2007)	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)					
6.2.1.3.1	General Design Criterion 50 and Appendix K to 10 CFR Part 50 A. Sources of Energy. The sources of stored and generated energy that should be considered in analyses of LOCAs include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the steam generator tubing and secondary water. Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A. However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA. An example of this would be accomplished by maximizing the sensible heat stored in the reactor coolant system (RCS) and steam generator metal and increasing the RCS and steam generator secondary mass to account for uncertainties and thermal expansion.					

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	The requirements of paragraph I.B in Appendix K to 10 CFR Part 50, concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core. B. Break Size and Location i. The staff's review of the applicant's choice of break locations and types is discussed in SRP Section 3.6.2. ii. Of several breaks postulated on the basis of a., above, the break selected as the reference case for subcompartment analysis should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area. iii. Containment design basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified. C. Calculations. In general, calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment subcompartment response). The criteria given below for each phase of the accident indicate the conservatism that should exist. i. Subcompartment Analysis The analytical approach used to compute the mass and energy					

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	release profile will be accepted if both the computer program and volume nodding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. The computer programs that are currently acceptable include SATAN-V CRAFT-2, CE FLASH-4, and RELAP4, when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation. ii. Initial Blowdown Phase Containment Design Basis The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data. Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection. Calculations of heat transfer from the secondary coolant to the steam generator tubes for PWRs should be based on natural convection heat transfer for tube surfaces immersed in water and					

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	condensing heat transfer for the tube surfaces exposed to steam. iii. PWR Core Reflood Phase (Cold Leg Breaks Only) Following initial blowdown, which includes the period from the accident initiation (when the reactor is in a steady-state full power operation condition) to the time that the reactor coolant system broken loop pressure equalizes to the containment pressure, the water remaining in the reactor vessel should be assumed to be saturated. Justification should be provided for the refill period, which is the time from the end of the blowdown to the time when the emergency core cooling system (ECCS) refills the vessel lower plenum. An acceptable approach is to assume a water level at the bottom of the active core at the end of blowdown so there is no refill time. Calculations of the core flooding rate should be based on the ECCS operating condition during the core reflood phase, which begins when the water starts to flood the core and continues until the core is completely quenched, or the post-reflood phase, which is the period after the core has been quenched and energy is released to the RCS primary system by the RCS metal, core decay heat, and the steam generators, that maximizes the containment pressure. Calculations of liquid entrainment, i.e., the carryout rate fraction, which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the PWR full length emergency cooling heat transfer experiments. Liquid entrainment should be assumed to continue until the water level in the core is 61 cm (2 feet) from the top of the core. An acceptable approach is to assume a carryout rate fraction (CRF) of 0.05 to the 46 cm (18-inch) core level, a linearly increasing CRF to 0.80 at the 61 cm (24-inch) level, and a constant CRF of 0.80 until the water level is					

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	61 cm (2 feet) from the top of the core. Above this level, a CRF of 0.05 may be used. The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the CRF of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water. Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant iv. PWR Post-Reflood Phase All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase. Steam quenching should be justified by comparison with applicable experimental data. The results of post-reflood analytical models should be compared to applicable experimental data. v. PWR Decay Heat Phase The dissipation of core decay heat should be considered during this phase of the accident. The fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in SRP Section 9.2.5. Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water.					

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	The following methods and computer models are acceptable for calculating the mass and energy releases for containment design basis calculations: Babcock and Wilcox / Framatome ANP: CRAFT, CRAFT-2, RELAP5/MOD2-B&W, Revision 1 and RELAP5/MOD2-B&W, Revision 4. Combustion Engineering: CEFLASH-4A and CESSAR System 80. General Electric: M3CPT, NEDO-20533, and SHEX. Westinghouse: WCAP-8312, SATAN-V, WCAP-10325, SATAN-VI, and WREFLOOD. Theses codes and methods have been referenced in licensee submittals and on a case by case basis have been found to be acceptable for these purposes. Other methods will be acceptable if they are found to be conservative for these calculations.					
6.2.1.3.2	10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.					
6.2.1.3.3	10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the					

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	facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations. 10 CFR 52.47(a)(1)(vi) provides the requirement for ITAAC for design certification reviews.					
6.2.1.4, Rev. 2 (03/2007)	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures					
6.2.1.4.1	Sources of Energy. The sources of energy that should be considered in the analyses of steam and feedwater line break accidents include the stored energy in the affected steam generator's metal, including the vessel tubing, feedwater line, and steamline; stored energy in the water contained within the affected steam generator; stored energy in the feedwater transferred to the affected steam generator before closure of the isolation valves in the feedwater line; stored energy in the steam from the unaffected steam generator(s) before the closure of the isolation valves in the steam generator crossover lines; and energy transferred from the primary coolant to the water in the affected steam generator during blowdown. The steamline break accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102 percent of full power. The applicant need only analyze the 102-percent power condition if it can demonstrate that the feedwater flows and fluid inventory are greatest at full power.					
6.2.1.4.2	Mass and Energy Release Rate. In general, calculations of the mass and energy release rates during a steam or feedwater line break accident should be performed in a conservative manner from a containment response standpoint (i.e., the post accident containment pressure and temperature are maximized). The following criteria indicate the degree of conservatism that is desired: A. Mass release rates should be calculated using the Moody					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	model (Ref. 1) for saturated conditions or a model that is demonstrated to be equally conservative. B. Calculations of heat transfer to the water in the affected steam generator should be based on nucleate boiling heat transfer. C. Calculations of mass release should consider the water in the affected steam generator and feedwater line, feedwater transferred to the affected steam generator before the closure of the isolation valves in the feedwater lines, steam in the affected steam generator, and steam coming from the unaffected steam generator(s) as the secondary system is being depressurized before the closure of the isolation valves in the steam generator crossover lines. D. If liquid entrainment is assumed in the steamline breaks, experimental data should support the predictions of the liquid entrainment model. The effect on the entrained liquid of steam separators located upstream from the break should be taken into account. A spectrum of steamline breaks should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur. This will allow selection of the maximum release case. If no liquid entrainment is assumed, a spectrum of the steamline breaks should be analyzed beginning with the double-ended break and decreasing in area until it has been demonstrated that the maximum release rate has been considered. E. Feedwater flow to the affected steam generator should be calculated considering the diversion of flow from the other steam generators, feedwater flashing, and increased feedwater pump flow caused by the reduction in steam generator pressure. An acceptable method for computing feedwater flow is to assume all					

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	feedwater travels to the affected steam generator at the pump runout rate before isolation. After isolation, the unisolated feedwater mass should be added to the affected steam generator. Operator action to terminate auxiliary feedwater flow will be reviewed under SRPSection 10.4.9. Any general-purpose thermal-hydraulics computer codes that the responsible reviewing organization for the subject application finds acceptable may be used to compute mass and energy releases from steam and feedwater line break accidents.					
6.2.1.4.3	Single-Failure Analyses. Steam and feedwater line break analyses should assume a single active failure in the steam or feedwater line isolation provisions or feedwater pumps to maximize the containment peak pressure and temperature. For the assumed failure of a safety-grade steam or feedwater line isolation valve, operation of nonsafety-grade equipment may be relied upon as a backup to the safety-grade equipment. In this event, the reviewer will confer with the responsible organizations for SRP Sections 3.2.1, 3.2.2, 3.6.2, and 10.4.9 to ensure a consistent staff position regarding the acceptability of the design criteria for the nonsafety-grade equipment.					
	REFERENCES: 1. F. J. Moody, “Maximum Flow Rate of a Single Component, Two-Phase Mixture,” Revision Jour. of Heat Transfer, Trans. Am. Soc. of Mechanical Engineers, Vol. 87, No. 1, February 1965.					
6.2.1.5, Rev. 3 (03/2007)	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies					
6.2.1.5.1	To meet the requirements of 10 CFR 50.46(a)(1)(i), the model to determine minimum containment pressure for ECCS studies should comply with Regulatory Guide (RG) 1.157, Position C.3.12.1, which describes acceptable containment pressure models for ECCS performance analysis.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
6.2.1.5.2	To meet the requirements of 10 CFR Part 50.46(a)(1)(ii), the following specific criteria indicate the conservatism that analyses of the containment response to LOCAs should have for determining the minimum containment pressure for ECCS performance capability studies: A. Calculations of the mass and energy released during postulated LOCAs should be based on the requirements of 10 CFR Part 50, Appendix K. B. Branch Technical Position 6-2, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," delineates the calculation approach that should be followed for a conservative prediction of the minimum containment pressure.					
6.2.2, Rev. 5 (03/2007)	Containment Heat Removal Systems					
6.2.2.1	In meeting GDC 16 requirements for functional capability of the secondary containment, the analysis of pressure and temperature response of the secondary containment to a LOCA in the primary containment should follow these guidelines: A. Heat transfer from the primary to the secondary containment should be considered. i. Heat transfer from the primary containment atmosphere to the primary containment structure should be calculated by conservative heat transfer coefficients like those in Branch Technical Position (BTP) 6-2. ii. Conductive heat transfer through the primary containment structure and convective heat transfer to the secondary containment atmosphere should be considered. iii. Radiant heat transfer to the secondary containment should be					

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	considered. B. Adiabatic boundary conditions should be assumed for the surface of the secondary containment structure exposed to the outside environment. C. The compressive effect of primary containment expansion on the secondary containment atmosphere should be considered. D. Secondary containment in-leakage should be considered. E. No credit should be taken for secondary containment out-leakage. F. For secondary containment response analyses loss of offsite power and the most severe single active failure in the emergency power system (e.g., a diesel generator failure), in the primary containment heat removal systems, in the core cooling systems, or in the secondary containment depressurization and filtration system should be assumed. Any delay due to system design in secondary containment depressurization and filtration system actuation should be considered. G. Heat loads generated within the secondary containment (e.g., equipment heat loads) should be considered. H. Fan performance characteristics should be considered in evaluating secondary containment depressurization.					
6.2.2.2	To meet the GDC 4 requirement to protect SSCs important to safety against dynamic effects, high-energy lines passing through the secondary containment should have guard pipes. Design criteria for guard pipes are in SRP Section 3.6.2. If there are no guard pipes, analyses should demonstrate that both primary containment and secondary containment structures are capable					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of withstanding the effects of a high-energy pipe rupture inside the secondary containment without loss of integrity.					
6.2.2.3	In meeting GDC 16 requirements for the functional capability of the secondary containment, the following criteria apply: A. The secondary containment depressurization and filtration systems should meet the guidelines of Regulatory Guide (RG) 1.52 and be capable of maintaining a uniform negative pressure throughout the secondary containment as well as other areas served by the systems. B. The negative pressure differential to be maintained in the secondary containment and other contiguous plant areas should be no less than 0.063 kPa (0.25 inches water gauge) compared to adjacent regions under all wind conditions up to the wind speed at which diffusion becomes sufficient to assure site boundary exposures less than those calculated for the design basis accident even if exfiltration occurs. If the leakage rate exceeds 100 percent of the volume per day, there should be a special exfiltration analysis. C. All openings like personnel doors and equipment hatches should be under administrative control with readout position indicators and alarms in the main control room. The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated and confirmatory preoperational tests conducted. D. Some plants may have only portions of the primary containment enclosed rather than a secondary containment structure or shield building completely enclosing the primary containment. These enclosures are areas into which the primary containment most likely would leak, and they may be equipped with air filtration systems. Quantitative credit cannot be given for					

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	the holdup effect of these enclosed areas or for the air filtration systems to mitigate the radiological consequences of a postulated accident unless the magnitude of unprocessed leakage can be adequately demonstrated. Quantitative credit for leakage collection in a partial-dual containment will be reviewed case by case. E. The external design pressure of the secondary containment structure should provide an adequate margin above the maximum expected external pressure.					
6.2.2.4	In meeting GDC 43 and 10 CFR Part 50, Appendix J, requirements for secondary containment system testing the following criteria apply: A. The fraction of primary containment leakage bypassing the secondary containment and escaping directly to the environment should be specified. BTP 6-3 provides guidance for detecting leakage paths to the environment which may bypass the secondary containment. The periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly bypass the secondary containment and other contiguous areas served by ventilation and filtration systems should be described. Individual tests should be according to procedures from technical specifications or their bases. B. There should be provisions in the design of the secondary containment system for inspections and monitoring of the functional capability. Preoperational and periodic test programs determine the depressurization time, the secondary containment in-leakage rate, the uniformity of negative pressure throughout the secondary containment and other contiguous areas, and the potential for ex-filtration.					
6.2.3, Rev. 3 (03/2007)	Secondary Containment Functional Design					

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6.2.3.1	In meeting GDC 16 requirements for functional capability of the secondary containment, the analysis of pressure and temperature response of the secondary containment to a LOCA in the primary containment should follow these guidelines: A. Heat transfer from the primary to the secondary containment should be considered. i. Heat transfer from the primary containment atmosphere to the primary containment structure should be calculated by conservative heat transfer coefficients like those in Branch Technical Position (BTP) 6-2. ii. Conductive heat transfer through the primary containment structure and convective heat transfer to the secondary containment atmosphere should be considered. iii. Radiant heat transfer to the secondary containment should be considered. B. Adiabatic boundary conditions should be assumed for the surface of the secondary containment structure exposed to the outside environment. C. The compressive effect of primary containment expansion on the secondary containment atmosphere should be considered. D. Secondary containment in-leakage should be considered. E. No credit should be taken for secondary containment out-leakage. F. For secondary containment response analyses loss of offsite power and the most severe single active failure in the emergency					

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	power system (e.g., a diesel generator failure), in the primary containment heat removal systems, in the core cooling systems, or in the secondary containment depressurization and filtration system should be assumed. Any delay due to system design in secondary containment depressurization and filtration system actuation should be considered. G. Heat loads generated within the secondary containment (e.g., equipment heat loads) should be considered. H. Fan performance characteristics should be considered in evaluating secondary containment depressurization.					
6.2.3.2	To meet the GDC 4 requirement to protect SSCs important to safety against dynamic effects, high-energy lines passing through the secondary containment should have guard pipes. Design criteria for guard pipes are in SRP Section 3.6.2. If there are no guard pipes, analyses should demonstrate that both primary containment and secondary containment structures are capable of withstanding the effects of a high-energy pipe rupture inside the secondary containment without loss of integrity.					
6.2.3.3	In meeting GDC 16 requirements for the functional capability of the secondary containment, the following criteria apply: A. The secondary containment depressurization and filtration systems should meet the guidelines of Regulatory Guide (RG) 1.52 and be capable of maintaining a uniform negative pressure throughout the secondary containment as well as other areas served by the systems. B. The negative pressure differential to be maintained in the secondary containment and other contiguous plant areas should be no less than 0.063 kPa (0.25 inches water gauge) compared to adjacent regions under all wind conditions up to the wind speed at which diffusion becomes sufficient to assure site boundary					

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	exposures less than those calculated for the design basis accident even if exfiltration occurs. If the leakage rate exceeds 100 percent of the volume per day, there should be a special exfiltration analysis. C. All openings like personnel doors and equipment hatches should be under administrative control with readout position indicators and alarms in the main control room. The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated and confirmatory preoperational tests conducted. D. Some plants may have only portions of the primary containment enclosed rather than a secondary containment structure or shield building completely enclosing the primary containment. These enclosures are areas into which the primary containment most likely would leak, and they may be equipped with air filtration systems. Quantitative credit cannot be given for the holdup effect of these enclosed areas or for the air filtration systems to mitigate the radiological consequences of a postulated accident unless the magnitude of unprocessed leakage can be adequately demonstrated. Quantitative credit for leakage collection in a partial-dual containment will be reviewed case by case. E. The external design pressure of the secondary containment structure should provide an adequate margin above the maximum expected external pressure.					
6.2.3.4	In meeting GDC 43 and 10 CFR Part 50, Appendix J, requirements for secondary containment system testing the following criteria apply: A. The fraction of primary containment leakage bypassing the secondary containment and escaping directly to the environment					

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	should be specified. BTP 6-3 provides guidance for detecting leakage paths to the environment which may bypass the secondary containment. The periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly bypass the secondary containment and other contiguous areas served by ventilation and filtration systems should be described. Individual tests should be according to procedures from technical specifications or their bases. B. There should be provisions in the design of the secondary containment system for inspections and monitoring of the functional capability. Preoperational and periodic test programs determine the depressurization time, the secondary containment in-leakage rate, the uniformity of negative pressure throughout the secondary containment and other contiguous areas, and the potential for ex-filtration.					
6.2.4, Rev. 3 (03/2007)	Containment Isolation System					
6.2.4.1	Regulatory Guide (RG) 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines closed both inside and outside containment are designed to withstand pressure and temperature conditions following a loss-of-coolant accident (LOCA) and dynamic effects are acceptable without isolation valves.					
6.2.4.2	Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems may include remote-manual valves, but should detect possible leakage from these lines outside containment.					
6.2.4.3	Containment isolation provisions for lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) may include remote-manual valves, but there should be provisions for detecting leakage from such lines outside containment.					

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6.2.4.4	Containment isolation provisions for lines in the systems of items 2 and 3 normally consist of one isolation valve inside and one outside containment. If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment. For this type of isolation valve arrangement, the valve nearer the containment and the piping between the containment and the valve should be enclosed in a leak-tight or controlled-leakage housing. If, in lieu of housing, the piping and valve are designed to preclude a breach of piping integrity, the design should comply with SRP Section 3.6.2 requirements. Design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet seals.					
6.2.4.5	Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve is acceptable if system reliability can be shown to be greater, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment should be protected from missiles, designed to seismic Category I and Group B quality standards, and have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak-tested unless system integrity can be shown to be maintained during normal plant operations. For this type of isolation valve arrangement the valve is located outside containment, and the piping between the containment and the valve should be enclosed in leak-tight or controlled-leakage housing. If, in lieu of housing, piping and valve are designed conservatively to preclude a breach of piping integrity, the design should comply with SRP Section 3.6.2 requirements. Design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet					

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	seals.					
6.2.4.6	Sealed-closed barriers may be used in place of automatic isolation valves. Sealed-closed barriers include blind flanges and sealed-closed isolation valves which may be closed manual valves, closed remote-manual valves, or closed automatic valves which remain closed after a LOCA. Sealed-closed isolation valves should be under administrative control so they cannot be opened inadvertently. Administrative control includes mechanical devices to seal or lock the valve closed or to prevent power supply to the valve operator.					
6.2.4.7	Relief valves may be used as isolation valves provided the relief setpoint is greater than 1.5 times the containment design pressure.					
6.2.4.8	10 CFR 50.34(f)(2)(xiv) requires that systems penetrating the containment be classified as either essential or nonessential. Reference 26 presents guidance on the classification of systems as essential and nonessential. Essential systems, like those described in items 2 and 3, may include remote-manual containment isolation valves, but there should be provisions for detecting leakage from the lines outside containment. 10 CFR 50.34(f)(2)(xiv) also requires that nonessential systems be isolated automatically by the containment isolation signal.					
6.2.4.9	Isolation valves outside containment should be located as close to it as practical, as required by GDCs 55, 56, and 57.					
6.2.4.10	To meet the requirements of GDCs 55 and 56, upon loss of actuating power, automatic isolation valves should take the position of greatest safety. The position of an isolation valve for normal and shutdown plant operating and post-accident conditions depends on the fluid system function. If a fluid system has no post-accident function, the isolation valves in the lines should be closed automatically. For engineered safety feature or engineered safety feature-related systems, isolation valves in the lines may remain open or be opened. In a power failure to the					

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	valve operator isolation valves should be in the "safe" position, normally the post-accident valve position. For lines equipped with motor-operated valves, a loss of actuating power leaves the affected valve in the "as-is" position, which may be the open position; however, redundant isolation barriers ensure that the isolation function for the line is satisfied. All power-operated isolation valves should have position indications in the main control room.					
6.2.4.11	To improve the reliability of the isolation function, addressed in GDC 54, 10 CFR 50.34(f)(2)(xiv) requires reduction of the containment setpoint pressure that initiates containment isolation for nonessential penetrations to the minimum value compatible with normal operating conditions.					
6.2.4.12	There should be diversity in the parameters sensed for the initiation of containment isolation to satisfy the GDC 54 requirement for reliable isolation capability.					
6.2.4.13	To improve the reliability of the isolation function, addressed in GDC 56, system lines which provide open paths from the containment to the environs (e.g., purge and vent lines addressed in 10 CFR 50.34(f)(2)(xiv)) should be equipped with radiation monitors capable of isolating these lines upon a high-radiation signal, which should not be considered a diverse containment isolation parameter.					
6.2.4.14	In meeting GDC 54 requirements, the performance capability of the isolation function should reflect the safety importance of isolating system lines. Consequently, containment isolation valve closure times should be selected for rapid isolation of the containment following postulated accidents. Valve closure time for a power-operated valve to be in the fully-closed position after the actuator power has reached the operator assembly does not include the time to reach actuation signal setpoints or instrument delay times, which, with system design capabilities, should be considered for establishing valve closure times. For lines providing open paths from the containment to the environs (e.g.,					

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	the containment purge and vent lines), isolation valve closure times of five seconds or less may be necessary. The closure times of these valves should be established to minimize the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and to prevent degradation of emergency core cooling system effectiveness by reduced containment back-pressure. Analyses of the radiological consequences and the effect on the containment back-pressure of the release of containment atmosphere should justify the selected valve closure time. Branch Technical Position (BTP) 6-4 presents additional guidance on the design and use of containment purge systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby, and hot shutdown). Containment purge valves that do not satisfy the operability criteria of Branch Technical Position 6-4 must be sealed closed as defined in subsection II.6 of this SRP section during operational conditions 1, 2, 3, and 4. Furthermore, closure of these valves must be verified at least every 31 days. These requirements should be incorporated into the technical specifications for plant operation.					
6.2.4.15	The use of a closed system inside containment as one of the isolation barriers is acceptable if the closed system design satisfies the following requirements: A. The system does not connect with either the reactor coolant system or the containment atmosphere. B. The system is protected against missiles and pipe whip. C. The system is designated seismic Category I. D. The system is classified Quality Group B. E. The system is designed to withstand temperatures equal to at					

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	least that of the containment design. F. The system is designed to withstand the external pressure from the containment structure acceptance test. G. The system is designed to withstand the LOCA transient and environment. As to the structural design of containment internal structures and piping systems, the protection against loss of function from missiles, pipe whip, and earthquakes is acceptable if 1) isolation barriers are located behind missile barriers; 2) pipe whip was considered in the design of pipe restraints and the location of piping penetrating the containment; and 3) the isolation barriers, including the piping between isolation valves, are designated seismic Category I, i.e., designed to withstand the effects of the safe-shutdown earthquake, as recommended by Regulatory Guide 1.29.					
6.2.4.16	To meet the requirements of GDCs 1, 2, 4, and 54, appropriate reliability and performance considerations should be included in the design of isolation barriers to reflect the safety importance of their integrity (i.e., containment capability) under accident conditions. The design criteria for components performing a containment isolation function, including the isolation barriers and the piping between them or the piping between the containment and the outermost isolation barrier, are acceptable if: A. Group B quality standards, as defined in RG 1.26, apply to the components, unless the service function dictates that Group A quality standards apply. B. The components are designated seismic Category I in accordance with RG 1.29.					
6.2.4.17	GDC 54 requires reliable isolation capability; therefore, for					

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	remote-manual isolation valves, the design of the containment isolation system is acceptable if there are provisions to allow the operator in the main control room to know when to isolate fluid systems equipped with remote-manual isolation valves. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.					
6.2.4.18	GDC 54 specifies requirements for the containment isolation system; therefore, to satisfy GDC 54, the design of the containment isolation system should provide for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. The isolation valve testing program should be consistent with that proposed for other engineered safety features. SRP Section 6.2.6 presents acceptance criteria for the leakage rate testing program for containment isolation barriers.					
6.2.4.19	GDC 54 requires reliable isolation capability. To satisfy this requirement, the design of the containment isolation system should reduce the possibility of unintended isolation valve reopening following isolation. 10 CFR 50.34(f)(2)(xiv) requires control systems for automatic containment isolation valves be designed for resetting the isolation signal without automatically reopening the valves. Reopening of containment isolation valves should require deliberate operator action and combined reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be valve by valve or line by line, provided that electrical independence and other single-failure criteria remain satisfied. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method for meeting this design requirement.					
6.2.4.20	In meeting 10 CFR 50.34(f)(2)(xv) purging requirements, the regulatory guidance of BTP 6-4, "Containment Purging During Normal Plant Operations," should be used to establish compliance with this regulation.					

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6.2.4.21	RG 1.155, "Station Blackout," Regulatory Position C.3.2.7, provides guidance for meeting the requirements of the SBO rule, 10 CFR 50.63(a)(2), for containment isolation valves and valve position indication.					
6.2.4.22	10 CFR Part 50, Appendix K, provides guidance for the determination of the extent of fuel failure (source term) in the radiological calculations.					
6.2.5, Rev. 3 (03/2007)	Combustible Gas Control in Containment					
6.2.5.1	In meeting the requirements of 10 CFR Part 50, § 50.44, and GDC 41 to provide systems to control the concentration of hydrogen in the containment atmosphere, materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.					
6.2.5.2	In meeting the requirements of 10 CFR Part 50, § 50.44, and GDC 41 to provide systems to control the concentration of hydrogen or oxygen in the containment atmosphere, the applicant should demonstrate by analysis, for non-inerted containments, that the design can safely accommodate hydrogen generated by an equivalent of a 100 percent fuel clad-coolant reaction, while limiting containment hydrogen concentration, with the hydrogen uniformly distributed, to less than 10 percent (by volume), and while maintaining containment structural integrity.					
6.2.5.3	In meeting the requirements of 10 CFR Part 50, § 50.44(c)(3), regarding equipment survivability, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment structural integrity should perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel clad-coolant reaction including the environmental conditions created by activation of the combustible gas control system.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
6.2.5.4	In meeting the requirements of 10 CFR Part 50, § 50.44, to provide the capability for ensuring a mixed atmosphere in the containment during design bases and significant beyond-design-bases accidents, and of GDC 41 to provide systems as necessary to ensure that containment integrity is maintained, this capability may be provided by an active, passive, or combination system. Active systems may consist of a fan, a fan cooler, or containment spray. For passive or combination systems that use convective mixing to mix the combustible gases, the containment internal structures should have design features which promote the free circulation of the atmosphere. For all containment types, an analysis of the effectiveness of the method used for providing a mixed atmosphere should be provided. This analysis is acceptable if it shows that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity. Atmosphere mixing systems prevent local accumulation of combustible or detonable gases which could threaten containment integrity or equipment operating in a local compartment. Active systems installed to mitigate this threat should be reliable, redundant, single-failure proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power.					
6.2.5.5	In meeting the requirements of 10 CFR Part 50, § 50.44, and GDC 41 regarding the functional capability of the combustible gas control systems to ensure that containment integrity is maintained, the design should meet the provisions of RG 1.7, Revision 3, Section C.1.					
6.2.5.6	To satisfy the design requirements of GDC 41: A. Performance tests should be performed on system components, such as hydrogen igniters and combustible gas monitors. The tests should support the analyses of the functional					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	capability of the equipment. B. Combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal and accident conditions. The instrumentation should be capable of determining that a system is performing its intended function, or that a system train or component is malfunctioning and should be isolated. The instrumentation should have readout and alarm capability in the control room. The containment hydrogen and oxygen monitors should meet the provisions of RG 1.7, Revision 3, Section C.2.					
6.2.5.7	To satisfy the inspection and test requirements of GDC 41, 42, and 43, combustible gas control systems should be designed with provisions for periodic inservice inspection, operability testing, and leak rate testing of the systems or components.					
6.2.5.8	In meeting the requirements of 10 CFR Part 50, § 50.44(c)(5), regarding containment structural integrity, an analysis must demonstrate containment structural integrity, using an analytical technique that is accepted by the NRC staff and including sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by combustible gas burning. Systems necessary to ensure containment integrity must also demonstrate the capability to perform their functions under these conditions. One acceptable analytical technique is a demonstration that specific criteria of the ASME Boiler and Pressure Vessel Code, described in RG 1.7, Revision 3, Section C.5, are met.					
6.2.5.9	In meeting the requirements of 10 CFR Part 50, § 50.44(c), and GDC 41 for the design and functional capability of the combustible gas control systems, preliminary system designs and statements of intent in the SAR are acceptable at the CP stage of review if the guidelines of RG 1.7, Revision 3, are endorsed.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
6.2.6, Rev. 3 (03/2007)	Containment Leakage Testing					
6.2.6.1	Containment boundaries that do not constitute potential containment atmospheric leakage pathways during and following a design-basis loss-of-coolant accident (DB LOCA);					
6.2.6.2	Containment boundaries sealed with a qualified seal system;					
6.2.6.3	Test connections, vents, and drains between containment isolation valves which: A. are one inch or less in size, and B. administratively secured closed, and C. consist of a double barrier (e.g., two valves in series, one valve with a nipple and cap, one valve and a blind flange). This guidance may be applied to either Option A or Option B of Appendix J. Examples of Case No. 1 are lines that terminate below the minimum post-accident water level of the suppression pool in a BWR or the recirculation sump in a PWR. For Case No. 2, a qualified seal system is defined in ANSI/ANS-56.8-1994 as a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 Pa, for at least 30 days following the DB LOCA. The staff's position is that the analysis of the sealing capability includes the assumption of the most limiting single failure of any active component. Also, unless there is a virtually unlimited supply of sealing liquid (such as from a suppression pool or recirculation sump), limits for liquid leakage rate should be assigned to these valves based on analysis and included in the plant technical specifications. Periodic leakage					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	rate testing, using the sealing liquid as the test medium, is then needed to ensure that the technical specification limits are maintained. For Case No. 3, to ensure that containment integrity is restored following testing, the test, vent, and drain connections that are used to facilitate local leakage rate testing and the performance of the CILRT should be under administrative control and should be subject to periodic surveillance, to ensure their integrity and to verify the effectiveness of administrative controls. The testing requirements for BWR drywell steam bypass are discussed in SRP Section 6.2.1.1.C. Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Containment Leak Rate Testing Program are reviewed in accordance with 10 CFR Part 50, Appendix J. The implementation milestones are as follows: A. Appendix J, Option A, Section III: Type A, B and C test: prior to any reactor operating period. B. Appendix J, Option B, Section III.A: Type A test: after the containment has been completed and is ready for operation. Type B & C test: prior to initial criticality					
6.2.7, Rev. 1 (03/2007)	Fracture Prevention of Containment Pressure Boundary					
6.2.7.1	To meet the requirements of GDC 1, 16 and 51, ferritic containment pressure boundary materials should meet the fracture toughness criteria and requirements for testing identified					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	in Article NE-2300 of Section III, Division 1 or Article CC-2520 of Section III, Division 2 of the ASME Code or, for materials that were not fracture toughness tested as discussed below, the fracture toughness criteria for Class 2 components identified in the Summer 1977 Addenda to Section III, Division 1, Subsection NC of the ASME Code.					
6.2.7.2	Mandatory fracture toughness testing of ASME Code Section III Class 2 materials was first identified in the Summer 1977 Addenda Code Class 2 rules. As a result, cases exist where Class 2 ferritic materials of the reactor containment pressure boundary were not fracture toughness tested, because the ASME Code Edition and Addenda in effect at the time the components were ordered, did not require that they be tested. The staff's assessment of the fracture toughness of materials that were not fracture toughness tested is based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, “Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports,” and ASME Code Section III, Summer 1977 Addenda, Subsection NC. The metallurgical characterization of these materials, with respect to their fracture toughness, is developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addend of the ASME Code Section III, provides the technical basis for the staff's evaluation of the compliance with Code Class 2 requirements of the materials which were not fracture toughness tested.					
6.3, Rev. 3 (03/2007)	Emergency Core Cooling System					
6.3.1	In regard to the ECCS acceptance criteria of 10 CFR 50.46, the five major performance criteria deal with:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A. Peak cladding temperature. B. Maximum calculated cladding oxidation. C. Maximum hydrogen generation. D. Coolable core geometry E. Long-term cooling. Guidance, procedures and methods that are acceptable for meeting the requirements for a realistic or best-estimate evaluation model for ECCS performance can be found in Regulatory Guide 1.157. This method must identify and account for uncertainties in the analysis method and inputs such that there is a high level of probability that the acceptance criteria is not exceeded (addresses Generic Issue C-4). Alternatively, Appendix K to 10 CFR Part 50 contains guidance for conservative ECCS evaluation models. These areas are reviewed as a part of the effort associated with the LOCA analysis (SRP Section 15.6.5). However, the impact of various postulated single failures on the operability of the ECCS, ECCS response times, break locations (including ECCS break locations), and break sizes impacting ECCS capabilities are evaluated under this SRP section.					
6.3.2	The ECCS must meet the requirements of GDC 35. The system must have alternate sources of electric power, as required by GDC 17, and must be able to withstand a single failure. The ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident. A passive failure in a fluid system is a breach in the fluid pressure					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	boundary or mechanical failure that adversely affects a flowpath. SECY-94-084 states the approved position that passive advanced light-water reactor designs need not assume passive component failures in addition to the initiating failure in the application of single-failure criterion to assure safety of the plant. In addition, the staff considers, on a long-term basis, passive component failures in fluid as potential accident initiators, in addition to initiating events. Check valves in the passive safety systems (except those for which proper function can be demonstrated and documented) are considered components subject to single-failure consideration.					
6.3.3	The ECCS must be designed to permit periodic inservice inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, piping, pumps, and valves in accordance with the requirements of GDC 36. The ECCS must be designed to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation, as required by GDC 37.					
6.3.4	The combined reactivity control system capability associated with ECCS must meet the requirements of GDC 27 and should conform to the recommendation of Regulatory Guide 1.47. The primary mode of actuation for the ECCS must be automatic, and actuation must be initiated by signals of suitable diversity and redundance. Provisions should also be made for manual actuation, monitoring, and control of the ECCS from the reactor control room.					
6.3.5	The design of the ECCS should conform to the recommendations of Regulatory Guide 1.1.					
6.3.6	Design features and operating procedures, designed to prevent damaging water hammer due to such mechanisms as voided discharge lines and water entrainment in steam lines shall be provided, in order to meet the requirements of GDC 4.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
6.3.7	The design of those portions of the system which are not safety related, whose failures could have an adverse effect on the ECCS system, must be in accordance with GDC 2, and acceptance is based on meeting Position C2 of Regulatory Guide 1.29. Also see SECY-94-084 for policy and technical issues associated with the regulatory treatment of non-safety systems in passive plant designs.					
6.3.8	Interfaces between the ECCS and component or service water systems must be such that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems, e.g., residual heat removal (RHR) and containment heat removal systems, the ECCS must conform to GDC 5.					
6.3.9	The requirements of Task Action Plan Item II.K.3(15) of NUREG-0737 and NUREG-0718, which involves isolation of HPCI and RCIC for BWR plants, should also be satisfied.					
6.3.10	The requirements and guidance regarding ECCS outage times and reports on ECCS unavailability, contained in Task Action Plan Item II.K.3.17, and Generic issue B-61, must also be satisfied.					
6.4, Rev. 3 (03/2007)	Control Room Habitability System					
6.4.1	Control Room Emergency Zone The control room emergency zone should include the following: A. Instrumentation and controls necessary for a safe shutdown of the plant, i.e., the control room, including the critical document reference file; B. Computer room, if it is used as an integral part of the emergency response plan; C. Shift supervisor's office; and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	D. Operator washroom and the kitchen. E. The control room emergency zone should conform to the guidelines of Regulatory Guide 1.196, May 2003, "Control Room Habitability at Light Water Nuclear Power Reactors," and Regulatory Guide (RG) 1.197, May 2003, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."					
6.4.2	Ventilation System Criteria. The ventilation system should include the following design features: A. Isolation dampers used to isolate the control zone from adjacent zones or the outside should be low leakage dampers or valves. The degree of leaktightness should be documented in the SAR. B. Single failure of an active component should not result in loss of the system's functional performance. All the components of the control room emergency filter train should be considered active components. See Appendix A to this SRP for criteria regarding valve or damper repair.					
6.4.3	Pressurization Systems. Ventilation systems that will pressurize the control room during a radiation emergency should meet the following criteria: A. Systems having pressurization rates of greater than or equal to 0.5 volume changes per hour should be subject to periodic verification (every 18 months) that the makeup is ± 10% of design value. During plant construction or after any modification to the control room that might significantly affect its capability to maintain a positive pressure, measurements should be taken to verify that the control room emergency zone is pressurized to at least to the value used in the accident analysis relative to all surrounding air spaces while					

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	applying makeup air at the design rate. B. Systems having pressurization rates of less than 0.5 and equal to or greater than 0.25 volume changes per hour should have identical testing requirements as indicated in acceptance criteria 1 above. In addition, at the construction permit (CP), combined license, or standard design certification stage, an analysis should be provided (based on the planned leaktight design features) that ensures the feasibility of maintaining the tested differential pressure with the design makeup airflow rate. C. 90 Systems having pressurization rates of less than 0.25 volume changes per hour should meet all the criteria for acceptance criteria 2 above, except that periodic verification of control room pressurization (every 18 months) should be specified.					
6.4.4	Emergency Standby Atmosphere Filtration System. Iodine removal for this system should be in accordance with the guidelines of Regulatory Guide 1.52. For new applications, the system should also conform with ASME Code AG-1, "Code on Nuclear Air and Gas Treatment" including the AG-1a-92 Addenda (Reference 14). Protection of control room personnel from releases of chlorine or other toxic gases is addressed in Regulatory Guide 1.78 as discussed in the criteria below.					
6.4.5	Relative Location of Source and Control Room. The control room inlets should be located with consideration of the potential release points of radioactive material and toxic gases. Specific criteria as to radiation and toxic gas sources are as follows: A. Radiation sources. As a general rule the control room					

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	ventilation inlets should be separated from the major potential release points by at least 31 meters (100 feet) laterally and by 16 meters (50 feet) vertically. However, the actual minimum distances should be based on the dose analyses (Ref. 9). B. Toxic gases. The minimum distance between the toxic gas source and the control room is dependent upon the amount and type of the gas in question, the container size, and the available control room protection provisions. The acceptance criteria for the control room habitability system are provided in the regulatory positions of Regulatory Guide 1.78 with respect to postulated hazardous chemical releases in general.					
6.4.6	Radiation Hazards A. For current operating reactors that do <u>not</u> implement an alternative source term under 10 CFR 50.67, 10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC 19) “Control room,” requires that “Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.” In accordance with GDC 19, these doses to an individual in the control room should not be exceeded for any postulated design basis accident. The whole body gamma dose consists of contributions from airborne radioactivity inside and outside the control room, as well as direct shine from all radiation sources. i. For current operating reactors the dose guidelines for evaluating the emergency zone radiation protection provisions are as follows:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	whole body gamma: 50 mSv (5 rem) thyroid: 300 mSv (30 rem) beta skin dose: 300 mSv (30 rem)[1] ii. For current operating reactors conforming to and implementing the guidance of RG 1.195 in conjunction with RG 1.196, the dose guidelines for evaluating the emergency zone radiation protection provisions are relaxed as follows: whole body gamma: 50 mSv (5 rem) thyroid: 500 mSv (50 rem)[2] beta skin dose: 500 mSv (50 rem)[1, 2] B. Applicants for and holders of construction permits and operating licenses under 10 CFR Part 50 who apply on or after January 10, 1997, applicants for design certifications under 10 CFR Part 52 who apply on or after January 10, 1997, applicants for and holders of combined licenses under 10 CFR Part 52 who do not reference a standard design certification, or holders of operating licenses using an alternative source term under 10 CFR 50.67, shall meet the requirements of GDC 19, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident.					

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6.4.7	Toxic Gas Hazards. Three exposure categories are defined: protective action exposure (2 minutes or less), short-term exposure (between 2 minutes and 1 hour), and long-term exposure (1 hour or greater). Because the physiological effects can vary widely from one toxic gas to another, the following general restrictions should be used as guidance: there should be no chronic effects from exposure; acute effects, if any, should be reversible within a short period of time (several minutes) without benefit of any measures other than the use of self-contained breathing apparatus. The allowable limits should be established on the basis that the operators should be capable of carrying out their duties with a minimum of interference caused by the gas and subsequent protective measures. The limits for the three categories normally are set as follows: A. Protective action limit (2 minutes or less): Use a limit that will ensure that the operators will quickly recover after breathing apparatus is in place. In determining this limit, it should be assumed that the concentration increases linearly with time from zero to two minutes and that the limit is attained at two minutes. B. Short-term limit (2 minutes to 1 hour): Use a limit that will ensure that the operators will not suffer incapacitating effects after a 1-hour exposure. C. Long-term limit (1 hour or greater): Use a limit assigned for occupational exposure (40-hour week). The protective action limit is used to determine the acceptability of					

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	emergency zone protection provisions during the time personnel are in the process of fitting themselves with self-contained breathing apparatus. The other limits are used to determine whether the concentrations with breathing apparatus in place are applicable. They are also used in those cases where the toxic levels are such that emergency zone isolation without use of protective gear is sufficient. Self-contained breathing apparatus for the control room personnel (at least 5 individuals) should be on hand. A 6-hour onsite bottled air supply should be available with unlimited offsite replenishment capability from nearby location(s). As an example of appropriate limits, the following are the three levels for chlorine gas: protective action: 15 ppm by volume short-term: 4 ppm by volume long-term: 1 ppm by volume Regulatory Guide 1.78 provides a partial list of protective action levels for other toxic gases.					
	NOTES: 1. The whole-body gamma, thyroid, and beta skin doses are consistent with the recommendations of International Committee on Radiation Protection (ICRP) 26, which were used in the May 21, 1991, revision of 10 CFR Part 20. 10 CFR 20.1201 limits organ dose to 50 rem annually. 2. Credit for the beta radiation shielding afforded by special protective clothing and eye protection is acceptable if the applicant commits to its use during severe radiation releases. However, even though protective clothing is used, the calculated unprotected skin dose should not exceed 750 mSv (75 rem). The skin and thyroid dose levels are to be used only for judging the acceptability of the design provisions for protecting control room operators under postulated design					

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	basis accident conditions. They are not to be interpreted as acceptable emergency doses. The dose levels quoted here are derived for use in the controlled plant environment and should not be confused with the conservative dose computation assumptions used in evaluating exposures to the general public for the purposes of comparison with the values of 10 CFR Part 100.					
	REFERENCES: 9. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974. 14. ASME Code AG-1, "Code for Nuclear Air and Gas Treatment," 1991 (including the AG-1a-92 Addenda thereto).					
6.5.1, Rev. 3 (03/2007)	ESF Atmosphere Cleanup Systems					
	Refer to the BTP for the detailed criteria.					
	Relevant aspects of the requirements are met by the regulatory positions of Regulatory Guide (RG) 1.52 to the design, testing, and maintenance of ESF atmosphere cleanup system air filtration and adsorption units.					
6.5.2, Rev. 4 (03/2007)	Containment Spray as a Fission Product Cleanup System					
6.5.2.1	Design Requirements for Fission Product Removal. The containment spray system should be designed in accordance with the requirements of ANSI/ANS 56.5, except that the requirements for any spray additive or other pH control system in this reference need not be followed. A. System Operation. The containment spray system should be designed to be initiated automatically by an appropriate accident signal and transferred automatically from the injection mode to the recirculation mode to					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	ensure continuous operation until the design objectives of the system have been achieved. In all cases, the operating period should not be less than 2 hours. Additives to the spray solution may be initiated manually or automatically or stored in the containment sump to be dissolved during the spray injection period. B. Coverage of Containment Building Volume. To ensure full spray coverage of the containment building volume, the following should be observed: i. The spray nozzles should be located as high in the containment building as practicable to maximize the spray drop fall distance. ii. The layout of the spray nozzles and distribution headers should be such that the cross-sectional area of the containment building covered by the spray is as large as practicable and the spray produced is a nearly homogeneous distribution in the containment building space. Unsprayed regions in the upper containment building and, in particular, an unsprayed annulus adjacent to the containment building liner should be avoided wherever possible. iii. In designing the layout of the spray nozzle positions and orientations, the effects of the postaccident atmosphere should be considered, including the effects of postaccident conditions that result in the maximum possible density of the containment atmosphere. C. Promotion of Containment Building Atmosphere Mixing. Because the effectiveness of the containment spray system depends on a well-mixed containment atmosphere, consideration should be given to all design features enhancing postaccident mixing.					

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	D. Spray Nozzles. The nozzles used in the containment spray system should be designed to minimize the possibility of clogging while producing drop sizes effective for iodine absorption. The nozzles should not have internal moving parts such as swirl vanes and turbulence promoters. They should not have orifices or internal restrictions which narrow the flow passage to less than 0.64 cm (0.25 inch) in diameter. E. Spray Solution. The partition of iodine between liquid and gas phases and retention of iodine in the liquid is enhanced by the alkalinity of the solution. The spray system should be designed so that the spray solution is within material compatibility constraints. Iodine-scrubbing credit is given for spray solutions whose chemistry, including any additives, has been demonstrated to be effective for iodine absorption and retention under postaccident conditions. F. Containment Sump Solution Mixing. The containment sump should be designed to permit mixing of emergency core cooling system (ECCS) and spray solutions. Drains to the engineered safety features sump should be provided for all regions of the containment which would collect a significant quantity of the spray solution. Alternatively, allowance should be made for “dead” volumes in the determination of the pH of the sump solution and the quantities of additives injected. G. Containment Sump and Recirculation Spray Solutions. The pH of the aqueous solution collected in the containment sump after completion of injection of containment spray and ECCS water and all additives for reactivity control, fission product removal, or other purposes should be maintained at a level sufficiently high to provide assurance that significant long-term iodine reevolution does not occur. The expected long-term					

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	partition coefficient is used to calculate the long-term iodine retention. Long-term iodine retention may be assumed only when the equilibrium sump solution pH, after mixing and dilution with the primary coolant and ECCS injection, is above 7. This pH value should be achieved by the onset of the spray recirculation mode. H. Storage of Additives. The design should provide facilities for the long-term storage of any spray additives. These facilities should be designed so that the additives required to achieve the design objectives of the system are stored in a state of continuous readiness whenever the reactor is critical for the design life of the plant. The storage facilities should be designed to prevent freezing, precipitation, chemical reaction, and decomposition of the additives. For sodium hydroxide storage tanks, heat tracing of tanks and piping is required whenever exposure to temperatures below 4.5 EC (40 EF) is predicted. An inert cover gas should be provided for solutions that may deteriorate when exposed to air. I. Single Failure. The system should be able to function effectively and meet all the criteria in Subsection II with a single failure of an active component in the spray system, in any of its subsystems, or in any of its support systems.					
6.5.2.2	Testing. Tests should be performed to demonstrate that the containment spray system, as installed, meets all design requirements for an effective fission-product-scrubbing function. Such tests should include preoperational verification of: A. freedom of the containment spray piping and nozzles from obstructions, B. the capability of the system to deliver the required spray flow, and					

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	C. the capability of the system to deliver spray additives (if any are needed) and to achieve the sump solution pH specified in the SAR. For a system such as a gravity feed system, whose performance is sensitive to the as-built piping layout, the testing should be performed at full flow.					
6.5.2.3	Technical Specifications. The technical specifications should specify appropriate limiting conditions for operation, tests, and inspections to provide assurance that the system is capable of performing its design function whenever the reactor is critical. These specifications should include: A. The operability requirements for the system, including all active and passive devices, as a limiting condition for operation (with acceptable outage times). The following items should be specifically included: containment spray pumps, additive pumps (if any), additive mixing devices (if any), valves and nozzles, additive quantity and concentration in additive storage tanks, and nitrogen (or other inert gas) pressure in additive storage tanks. B. Requirements for periodic inspection and sampling of the contents of additive storage tanks to confirm that the additive quantity and concentrations are within the limits established by the system design. C. Requirements for periodic testing and exercising of the active components of the system and verification that essential piping and passive devices are free of obstructions. Acceptable methods for computing fission product removal rates by the spray system are given in Subsection III.4.c, “Fission Product Cleanup Models.”					

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	Although credit is granted for containment spray removal of fission products in the calculations of accident doses, the acceptance criteria of containment leakage in SRP Section 6.2.1.1.A and the acceptance criteria of the engineered safety feature atmosphere cleanup systems in SRP Section 6.5.1 should still be met.					
6.5.3, Rev. 3 (03/2007)	Fission Product Control Systems and Structures					
6.5.3.1	Primary Containment. Primary containment design leakage rates for which credit is given should not be less than 0.1% per day due to difficulties in measuring lower leakage rates. Containment isolation methods and times must be such that the calculated radiological doses resulting from the escape of radioactive material prior to and following isolation after a LOCA do not exceed the applicable dose requirements of 10 CFR Part 100 and GDC 19. The primary reactor containment and associated systems should be designed so that periodic inspections and functional testing can be performed.					
6.5.3.2	Secondary Containment. To be classified as a secondary containment for the purpose of fission product control, a structure or structures should completely surround the primary containment, and at least should be held at a pressure of 0.6 cm (0.25 in) (water), below adjacent regions, under all wind conditions up to the wind speed at which diffusion becomes great enough to ensure site boundary exposures less than those calculated for the design basis accidents even if exfiltration occurs. Acceptance of other fission product control structures for collection and control of postaccident releases will be determined following consultation with the organization responsible for the review of reactor accident consequence assessment, (specifically					

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	design basis containment and ventilation performance) and the organization responsible for structural design of containment and ventilation systems, on a case-by-case basis. The leakage and filtration rates of such structures are acceptable provided that the offsite doses calculated by the organization responsible for radiation protection under SRP Section 15.6.5 will meet the dose guidelines of 10 CFR Part 100 and provided that the preoperational testing and appropriate technical specifications are acceptable. Other criteria include specifications for intake and return headers on recirculation systems. These should be placed as far away from each other as practical. The return header should provide a wide distribution over the secondary containment. The purpose of this placement is to ensure some degree of mixing of the return flow in the secondary containment volume before it is again drawn into the system intake. With judicious placement, up to 50% mixing may be assumed. A claim for greater than 50% mixing must be supported by the applicant to the satisfaction of the staff. Spacing between intake and return headers is reviewed on a case-by-case basis. Adjustments in the mixing fraction to less than 50% may be indicated by some designs. Past practice has been to allow mixing in 50% of the volume between — and within 3 or 6 meters (10 or 20 feet) of — the inlet and outlet headers if both have distributed openings or if one has distributed openings and the other is at the top of the containment. Partial dual containments should meet the same basic criteria as secondary containments in order to be given credit for fission product holdup and removal. The fraction of leakage source considered to be controlled by such partial fission products control structures is determined after consultation with the SCSB					

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	reviewer on a case-by-case basis.					
6.5.3.3	NUREG 0800 does not contain an item 3					
6.5.3.4	Other Fission Product Control Systems. Fission product retention credit may be taken by the applicant for other systems- e.g., containment spray systems as evaluated in SRP Section 6.5.2, pressure suppression pools as evaluated in SRP Section 6.5.5, and filtration and adsorption units as described in Regulatory Guide 1.52. Justification for fission product retention systems should include analytical bases addressing the important physical and chemical variables of the fission product removal and retention processes.					
6.5.4,Rev. 4 DRAFT (06/1996)	Ice Condenser as a Fission Product Cleanup System					
6.5.4.1	The ice condenser system is acceptable for elemental iodine removal if the ice contains a quantity of the proposed chemical additive sufficient to ensure that the pH of the post-accident recirculating solution is above 7 (Reference. 7)[15].					
6.5.4.2	The technical specifications are acceptable if they specify appropriate limiting conditions for operations, tests, and inspections to ensure that the system is capable of its design function whenever the reactor is critical. These specifications should include: the operability requirements for the system, and periodic sampling and testing requirements of the ice to confirm that the concentration of the chemical additive in the ice melt is within the limits established by the system design. While granting credit for ice condenser scrubbing of fission products in the calculations of accident doses, the acceptance criteria of containment leakage in SRP Section 6.2.1.1.B and the acceptance criteria of the engineered safety feature atmosphere cleanup systems in SRP Section 6.5.1 should still be met.					
	Notes:					

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	15. Format change to make the citation of references consistent with the SRP-UDP format requirements. Revised the reference number to reflect the reordering of the references in the Reference subsection.					
	REFERENCES: 7. C. C. Lin, "Chemical Effects of Gamma Radiation on Iodine in Aqueous Solutions," Journal of Inorganic and Nuclear Chemistry, 42, pages 1101-1107 (1980).					
6.5.5, Rev. 1 (03/2007)	Pressure Suppression Pool as a Fission Product Cleanup System					
6.5.5.1	The drywell and its penetrations must be designed to ensure that, even with a single active failure, all releases that include fission products from the reactor core must pass into the suppression pool, except for small bypass leakage.					
6.5.5.2	The bypass leakage assumed for purposes of evaluating fission product retention must be no less than that accepted in the review under SRP Section 6.2.1.1.C, and must be demonstrated in periodic tests by the license technical specifications also reviewed under that section.					
6.5.5.3	For plants that have already received a construction permit, the iodine retention calculated using this section must not be used to justify removal of the standby gas treatment or other filtered exhaust system from status as engineered safety features, and any change in plant design, proposed testing, surveillance or maintenance must be supported by considerations of lowered operator dose and other projected benefits. For such plants, the charcoal filters must be maintained at least to the minimum level of Table 1 in Regulatory Guide 1.52, Rev. 3. Acceptable methods for computing fission product retention by the suppression pool are given in this document in subsection III, "REVIEW PROCEDURES."					
6.6, Rev. 2 (03/2007)	Inservice Inspection and Testing of Class 2 and 3 Components					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
6.6.1	Components Subject to Inspection. The applicant's definition of ASME Code Class 2 and 3 components and systems subject to an ISI program is acceptable if it is in agreement with the NRC quality group classification system or the definitions in Article NCA-2000 of Section III of the ASME Code. The classification of components by the applicant is subject to review under SRP Section 3.2.2 for compliance with safety criteria pertaining to component classification. Where a specific item will be subject to inspection requirements different in any way from the ASME Code Section XI requirements corresponding to the item's Code Class, the exceptions for the item, including the inservice inspection requirements to be applied, should be clearly identified and described. Exceptions involving less stringent inspection requirements for Code Class 2 or 3 items other than those required by Section XI must be adequately justified. (Refer to SRP Section 3.2.2 or Article NCA-2000 of Section III of the ASME Code.)					
6.6.2	Accessibility. The design and arrangement of Class 2 and 3 systems should include allowances for adequate clearances to conduct the examinations specified in Articles IWC-2000 and IWD-2000 at the frequency specified. The design and arrangement of system components are acceptable if adequate clearance is provided in accordance with Subarticle IWA-1500. Special design considerations are given to those systems that are intended to be examined during normal reactor operation.					
6.6.3	Examination Categories and Methods. The examination categories and requirements specified in the SAR are acceptable if they are in agreement with the rules of Articles IWA-2000, IWC-2000, and IWD-2000. Every area subject to examination should fall within one or more of the examination categories and must be examined at least to the extent specified.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The applicant's examination techniques and procedures used for preservice inspection and inservice inspection are acceptable if they are in agreement with the following criteria: A. The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with Article IWA-2000. B. Alternative examination methods, combination of methods, or newly developed techniques to those given in A. above are acceptable provided that the results are equivalent or superior. The acceptance standards for these alternate methods are given in Articles IWC-3000 and IWD-3000. C. The methods, procedures, and requirements regarding qualification of personnel performing ultrasonic examination reflect the guidance provided in Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," to Division 1 of Section XI of the ASME Code. D. Performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws are in accordance with the requirements of Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Division 1 of Section XI of the ASME Code.					
6.6.4	Inspection Intervals. The ISI program schedule provided in the SAR is acceptable if the required examinations and pressure tests are specified for completion during each tenyear interval, hereinafter designated as the “inspection interval,” and as required by ASME Section XI, Articles IWA-2000, IWC-2000, and IWD-2000.					
6.6.5	Evaluation of Examination Results. The methods for evaluation of examination results are reviewed for compliance with Articles IWC-3000 and IWD-3000 in the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Code. If the applicable edition of the Code states that these articles are in the course of preparation, the rules of Article IWB-3000 shall apply. The repair procedures are acceptable if they are in compliance with ASME Section XI, Article IWA-4000.					
6.6.6	System Pressure Tests. The program provided in the SAR for Class 2 and 3 system pressure testing is acceptable if it meets the criteria of ASME Section XI, Articles IWC-5000 and IWD-5000.					
6.6.7	Augmented ISI to Protect Against Postulated Piping Failures. The augmented ISI program for high-energy fluid system piping between containment isolation valves is acceptable if it specifies the following requirements: A. Protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the Division 1 of Section XI of the ASME Code. B. For those portions of high energy fluid system piping between containment isolation valves, the extent of inservice examination completed during each inspection interval should provide 100% volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping. C. For those portions of high-energy fluid system piping enclosed in guard pipes, inspection ports should be provided in the guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures. D. The areas subject to examination should be defined in accordance with Article IWC-2000, Examination Category C-F for Class 2 piping welds.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
6.6.8	Code Exemptions. The exemptions from Code examination requirements identified by the applicant are acceptable if they have been permitted by Subsubarticles IWC-1220 or IWD-1220 of Section XI of the ASME Code.					
6.6.9	Relief Requests. Request for relief from the ASME Code Section XI examination requirements that are found to be impractical due to the limitations of design, geometry, or materials of construction of components are evaluated in accordance with 10 CFR 50.55a.					
6.6.10	Code Cases. The exemptions from Code examination requirements identified by the applicant or licensee are acceptable if they have been permitted by appropriate ASME code cases.					
6.6.11	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Preservice Inspection and Inservice Inspection and testing programs for Class 2 and 3 components are reviewed in accordance with the requirements of 10 CFR 50.55a, “Codes and Standards.” The implementation milestone for the inservice inspection program is when the plant enters into commercial operation.					
6.7, Rev. 3 DRAFT (06/1996)	Main Steam Isolation Valve Leakage Control System (BWR)	NA				Exclude; Not applicable to the HTGR design.
BTP 6-1, Initial Issuance (03/2007)	pH For Emergency Coolant Water for Pressurized Water Reactors					
BTP 6-1.1	Minimum pH should be 7.0.					
BTP 6-1.2	For the spray water recirculated from the containment sump, the higher the pH in the 7.0 to 9.5 range, the greater the assurance that no stress corrosion cracking will occur. See SRP Section 6.5.2 for additional water chemistry requirements related to fission					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	product removal.					
BTP 6-1.3	If a pH greater than 7.5 is used, consideration should be given to the hydrogen generation problem from corrosion of aluminum in the containment.					
BTP 6-2, Rev. 3 (03/2007)	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation					
BTP 6-2.1	Input Information for Model A. Initial Containment Internal Conditions. The minimum containment gas temperature, minimum containment pressure, and maximum humidity encountered under limiting normal operating conditions should be used. Ice condenser plants should use the maximum containment gas temperature. B. Initial Outside Containment Ambient Conditions. A reasonably low ambient temperature external to the containment should be used. C. Containment Volume. The maximum net free containment volume should be used. This maximum free volume should be determined from the gross containment volume minus the volumes of such internal structures as walls and floors, structural steel, major equipment, and piping. The individual volume calculations should reflect the uncertainty in the component volumes. D. Purge Supply and Exhaust Systems. If purge system operation is proposed during the reactor operating modes of startup, power operation, hot standby, and hot shutdown, the system lines should be assumed to be initially open.					
BTP 6-2.2	Active Heat Sinks					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	A. Spray and Fan Cooling Systems. The operation of all engineered safety feature containment heat removal systems operating at maximum heat removal capacity (i.e., with all containment spray trains operating at maximum flow conditions and all emergency fan cooler units operating) should be assumed. In addition, the minimum temperature of the stored water for the spray cooling system and the cooling water supplied to the fan coolers, based on technical specification limits, should be assumed. Deviations from the foregoing are accepted if the worst conditions for a single active failure, stored water temperature, and cooling water temperature can be shown to have been selected from the standpoint of the overall ECCS model. B. Containment Steam Mixing With Spilled ECCS Water. The spillage of subcooled ECCS water into the containment provides an additional heat sink as the subcooled ECCS water mixes with the steam in the containment. The effect of the steam-water mixing should be considered in the containment pressure calculations. C. Containment Steam Mixing With Water from Ice Melt. The water from ice melting in an ice condenser containment provides an additional heat sink as the subcooled water mixes with the steam while draining from the ice condenser into the lower containment volume. The effect of the steam-water mixing should be considered in the containment pressure calculations.					
BTP 6-2.3	Passive Heat Sinks A. Identification. The passive heat sinks that should be included in the containment evaluation model should be established by identifying structures					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	and components within the containment that could influence the pressure response. Structures and components that should be included are listed in Table 1. Data on passive heat sinks have been compiled from previous reviews and used as a basis for the simplified model outlined below. This model is acceptable for minimum containment pressure analyses for construction permit applications until a complete identification of available heat sinks can be made (i.e., at the operating license review). Where no detailed listing of heat sinks within the containment is provided, the following procedure may model the passive heat sinks within the containment: (i) Use the surface area and thickness of the primary containment steel shell or steel liner, anchors, and concrete, as appropriate. (ii) Estimate the exposed surface area of other steel heat sinks in accordance with Figure 1 and assume an average thickness of 9.53 mm (3/8 inch). (iii) Model the internal concrete structures as a slab with a thickness of 30.5 cm (one foot) and exposed surface of 15,000 m2 (160,000 ft2). Acceptable heat sink thermo-physical properties are shown in Table 2. Applicants should provide a detailed list of passive heat sinks with appropriate dimensions and properties. B. Heat Transfer Coefficients. The following conservative condensing heat transfer coefficients for heat transfer to the exposed passive heat sinks during the blowdown and post-blowdown phases of the loss-of-coolant accident should be used: (i) During the blowdown phase, assume a linear increase in the condensing transfer coefficient from $hi_{initial}$ = 8 Btu/hr-ft^2-°F, at t = 0, to a peak value four times greater than the maximum					

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	calculated condensing heat transfer coefficient at the end of blowdown, using the Tagami correlation, $h_{max}=7.25(Q/Vt_p)^{0.62}$ where h_{max} = maximum heat transfer coefficient, Btu/hr-ft^2-°F Q = primary coolant energy, Btu V = net free containment volume, ft^3 t_P = time interval to end of blowdown, sec. (ii) During the long-term post-blowdown phase of the accident characterized by low turbulence in the containment atmosphere, assume condensing heat transfer coefficients 1.2 times greater than those predicted by the Uchida data and given in Table 3. (iii) During the transition phase of the accident between the end of blowdown and the long-term post-blowdown phase, a reasonably conservative exponential transition in the condensing heat transfer coefficient should be assumed (See Figure 2). The calculated condensing heat transfer coefficients based on this method should be applied to all exposed passive heat sinks, both metal and concrete, and for both painted and unpainted surfaces. Heat transfer between adjoining materials in passive heat sinks should be based on the assumption of no resistance to heat flow at the material interfaces. An example is the containment liner to concrete interface. (iv) Variations from these guidelines may be acceptable if the overall ECCS performance evaluation model produces an acceptable peak calculated fuel cladding temperature.					
	Refer to the RG for Tables 1, 2, and 3 and Figures 1 and 2.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
BTP 6-3, Rev. 3 (03/2007)	Determination of Bypass Leakage Paths in Dual Containment Plants					
BTP 6-3.1	A secondary containment structure should enclose the primary containment structure completely with the exception of such parts imbedded in the soil as the base mat of the containment structure. For partial dual containment concepts, leak rates less than the design leak rate of the primary containment should not be used in the calculation of the radiological consequences of a LOCA unless the magnitude of unprocessed leakage can be adequately demonstrated. Quantitative credit for leakage collection in a partial-dual containment will be reviewed case by case.					
BTP 6-3.2	Direct leakage from the primary containment to the environment, equivalent to the design leak rate of the primary containment, should be assumed to occur following a postulated LOCA whenever the secondary containment volume is at a "positive" pressure (i.e., greater than -0.063 kPa (-0.25 inches water gauge)). Positive pressure periods should be determined by a pressure response analysis of the secondary containment volume including thermal loads from the primary containment and infiltration leakage.					
BTP 6-3.3	The secondary containment depressurization and filtration systems should be designed in accordance with Regulatory Guide (RG) 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants." Preoperational and periodic inservice inspection and test programs should be proposed for these systems and should include means for determining the secondary containment infiltration rate and the capability of the systems to draw down the secondary containment to the prescribed negative pressure in a prescribed time.					
BTP 6-3.4	For secondary containments with design leakage rates greater than 100 volume percent per day, there should be an exfiltration analysis.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
BTP 6-3.5	The following leakage barriers in paths which do not terminate within the secondary containment should be considered potential bypass leakage paths around the leakage collection and filtration systems of the secondary containment: A. Isolation valves in piping which penetrates both the primary and secondary containment barriers. B. Seals and gaskets on penetrations which pass through both the primary and secondary containment barriers. C. Welded joints on penetrations (e.g., guard pipes) which pass through both the primary and secondary containment barriers.					
BTP 6-3.6	The total leakage rate for all potential bypass leakage paths, as identified in item 5, should be determined realistically, considering equipment design limitations and test sensitivities. This value should be used in calculating the offsite radiological consequences of postulated LOCAs and in setting technical specification limits with margin for bypass leakage.					
BTP 6-3.7	There should be provisions for preoperational and periodic leakage rate testing similar to the Type B or C tests of 10 CFR Part 50, Appendix J, for each bypass leakage path listed in item 5. An acceptable alternative for local leakage rate testing for welded joints would be a soap bubble test of the welds concurrently with the integrated (Type A) leakage test of the primary containment required by Appendix J. Any detectable leakage determined would require repair of the joint.					
BTP 6-3.8	If air or water sealing systems or leakage control systems are proposed to process or eliminate leakage through valves, these systems should be designed, to the extent practical, according to the guidelines for leakage control systems in RG 1.96.					
BTP 6-3.9	If a closed system is proposed as a leakage boundary to preclude bypass leakage, then the system should:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A. Either (a) not directly communicate with the containment atmosphere or (b) not directly communicate with the environment following a LOCA. B. Be designed in accordance with Quality Group B standards, as defined by RG 1.26. (Systems designed to Quality Group C or D standards that qualify as closed systems to preclude bypass leakage will be considered case by case.) C. Meet seismic Category I design requirements. D. Be designed to at least the primary containment pressure and temperature design conditions. E. Be designed for protection against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features. F. Be tested for leakage unless it can be shown that during normal plant operations the system integrity is maintained.					
BTP 6-4, Rev. 3 (03/2007)	Containment Purging During Normal Plant Operations					
BTP 6-4.1	The on-line purge system should be designed in accordance with the following criteria: A. GDC 54 requires that the reliability and performance capabilities of containment isolation valves reflect the safety importance of isolating the systems penetrating the containment boundary; therefore, the performance and reliability of the purge system isolation valves should be consistent with the operability assurance program of SRP Section 3.10. The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum and the supply line and exhaust line flows as a function of time up to and during valve closure.					

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	B. The number of supply and exhaust lines should be limited to one supply line and one exhaust line to improve the reliability of the isolation function as required by GDC 54 and to facilitate compliance with the requirements of 10 CFR Part 50, Appendix K, for the containment pressure used in the evaluation of ECCS effectiveness and 10 CFR Part 100 for offsite radiological consequences. C. The size of the lines should not exceed about eight inches in diameter without detailed justification for larger line sizes to improve the reliability and performance capability of the isolation and containment functions as required by GDC 54 and to facilitate compliance with the requirements of 10 CFR Part 50, Appendix K, for the containment pressure used in evaluating ECCS effectiveness and 10 CFR Part 100 for the offsite radiological consequences. D. As required by GDC 54, the containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features (i.e., quality, redundancy, testability and other appropriate criteria) to reflect the importance to safety of isolating these lines. GDC 56 establishes explicit requirements for isolation barriers in purge system lines. E. To improve the reliability of the isolation function addressed in GDC 54, instrumentation and control systems isolating the purge system lines should be independent and actuated by diverse parameters (e.g., containment pressure, safety injection actuation, and containment radiation level). Furthermore, if energy is required to close the valves, at least two sources of energy must be provided, either of which can effect the isolation function. F. Purge system isolation valve closure times, including					

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	instrumentation delays, should not exceed five seconds to facilitate compliance with 10 CFR Part 100 for offsite radiological consequences. G. Isolation valve closure must not be prevented by debris which could become entrained in the escaping air and steam.					
BTP 6-4.2	The purge system should not be relied on for temperature and humidity control within the containment.					
BTP 6-4.3	The need for purging of the containment should be minimized by containment atmosphere cleanup systems within the containment.					
BTP 6-4.4	The availability of the isolation function and the leakage rate of the isolation valves during reactor operation should be tested.					
BTP 6-4.5	The following analyses should justify the containment purge system design: A. An analysis of the radiological consequences of a LOCA should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valve closures should be identified. The source term in the radiological calculations should be based on a calculation under the terms of 10 CFR Part 50, Appendix K, to the extent of fuel failure and the concomitant release of fission products and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR Part 100 guideline values. B. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment (e.g., fans, filters, and ductwork) located beyond the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	purge system isolation valves against loss of function in the environment created by the escaping air and steam. C. An analysis of the reduction in the containment pressure caused by the partial loss of containment atmosphere during the accident for ECCS back pressure determination. D. The maximum allowable leak rate of the purge isolation valves should be specified case by case with appropriate consideration for valve size, maximum allowable leakage rate for the containment (as defined in 10 CFR Part 50, Appendix J), and, where appropriate, the maximum allowable bypass leakage fraction for dual containments.					
BTP 6-5, Rev. 3 (03/2007)	Currently the Responsibility of Reactor Systems Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps					
BTP 6-5.1	Branch Position A. The single active failure criterion defined in (a) and (b) above will be applied in evaluating the design of the piping systems that connect the safety injection pumps to the RWST (BWST) and the containment sumps. B. The piping systems, including valves, shall be designed to satisfy the requirements listed below without the need to disconnect the power to any valve. C. The valves and piping between the RWST (or BWST) and the safety injection pumps must be arranged so that no single failure will prevent the minimum flow to the core required to satisfy 10 CFR 50.46. D. The valves and piping between the RWST (or BWST) and safety injection pumps must be arranged so that no single active					

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	failure will result in damage to pumps such that the minimum flow requirements for long-term core and containment cooling after a LOCA are not satisfied. E. The valves and piping that connect the RWST (or BWST) and the containment sump(s) to the safety injection pumps must be arranged so as not to preclude automatic switchover from the injection mode of ECCS operation to recirculation cooling from the sump. These piping systems must be arranged so that the differential pressure between the sump and the RWST (or BWST), even if there is a single active failure, will not result in a loss of core cooling or a path that permits release of radioactive material from the containment to the environment.					
BTP 6-5.2	Implementation A. Applicants for a construction permit for which an SER was published prior to April 16, 1975 will not be required to comply with the provisions of this item. B. For plants with an operating license issued prior to July 1981 and operating license applications docketed prior to July 1981 the position will not be completely applied. Specifically, locking out power to valves will be permitted. For most plants it is expected that this will be sufficient to meet the single failure criteria. However, in other plants changes to the piping and valving arrangements may be required to satisfy the single failure criteria. C. Applications docketed on or after July 1981 will be reviewed according to the provisions of this item.					
	CHAPTER 7, Instrumentation and Controls					
7.0, Rev. 5 (03/2007)	Instrumentation and Controls - Overview of Review Process					

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	Refer to the BTP for the detailed criteria.					
7.0-A, Rev. 5 (03/2007)	Review Process for Digital Instrumentation and Control Systems					
	1. Qualification of Digital Instrumentation and Control Systems and Components Digital I&C systems require additional design and qualification approaches than are typically employed for analog systems. The performance of analog systems can typically be predicted by the use of engineering models. These models can also be used to predict the regions over which an analog system exhibits continuous performance. The ability to analyze design using models based on physics principles and to use these models to establish a reasonable expectation of continuous performance over substantial ranges of input conditions are important factors in the qualification of analog systems design. These factors enable extensive use of type testing, acceptance testing, and inspection of design outputs in qualifying the design of analog systems and components. If the design process assures continuous behavior over a fixed range of inputs, and testing at a finite sample of input conditions in each of the continuous ranges demonstrates acceptable performance, performance at intermediate input values between the sampled test points can be inferred to be acceptable with a high degree of confidence. Digital I&C systems are fundamentally different from analog I&C systems in that minor errors in design and implementation can cause them to exhibit unexpected behavior. Consequently, the performance of digital systems over the entire range of input conditions cannot generally be inferred from testing at a sample of input conditions. Inspections, type testing, and acceptance testing of digital systems and components do not alone accomplish design					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	qualification at high confidence levels. To address this issue, the staff's approach to the review of design qualification for digital systems focuses to a large extent on confirming that the applicant/licensee employed a high-quality development process that incorporated disciplined specification and implementation of design requirements. Inspection and testing are used to verify correct implementation and to validate desired functionality of the final product, but confidence that isolated, discontinuous point failures will not occur derives from the discipline of the development process.					
	2. Defense Against Common-Cause Failure In digital I&C safety systems, code, data transmission, data, and hardware may be common to several functions to a greater degree than is typical in analog systems. Although this commonality is the basis for many of the advantages of digital systems, it also raises a key concern: a design using shared data or code has the potential to propagate a common-cause failure via software errors, thus defeating the redundancy achieved by the hardware architectural structure. Greater commonality or sharing of hardware among functions within a channel increases the consequences of the failure of a single hardware module and reduces the amount of diversity available within a single safety channel. Because of this concern, the staff's review of digital I&C protection systems emphasizes quality and diversity and defense-in-depth as protection against propagation of common-cause failures within and between functions. Additional guidance on assessment of diversity and defense-in-depth is provided in SRP BTP 7-19.					
	3. System Aspects of Digital Instrumentation and Control Certain functional requirements that apply to I&C safety					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	systems involve system aspects that pose assurance challenges when applied to digital systems. These aspects include real-time performance, independence, and on-line testing. The review process for these topics must recognize the special characteristics of digital systems.					
7.1, Rev. 5 (03/2007)	Instrumentation and Controls - Introduction					
7.1.1	SRP Table 7-1, Section 3 (Staff Requirements Memoranda), Section 4 (Regulatory Guides), and Section 5 (Branch Technical Positions), list the SRP acceptance criteria applicable to I&C systems important to safety. Sources of the acceptance criteria are as follows: • Commission Papers (SECY) are issue papers submitted by the staff to the NRC commissioners to inform them about policy matters. Staff Requirements Memoranda (SRM) provide the NRC's decisions and directions on the issues discussed in the SECY. • Regulatory guides describe acceptable methods for meeting regulatory requirements and provide guidance to applicant/licensees. Industry codes and standards set forth industry consensus requirements and recommended practices applicable to I&C systems for nuclear power plants. These standards are endorsed by regulatory guides, with or without modification, and provide acceptable methods for meeting the requirements of the NRC's regulations. • Branch technical positions (BTP) document the resolution of significant technical issues or questions of interpretation that have arisen in past reviews. BTPs outline acceptable approaches to a particular issue. The approaches taken in BTPs, like the recommendations of regulatory guides, are not mandatory.					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	SECY and associated SRM, regulatory guides and their endorsed industry codes and standards, and BTPs are the guidelines used as SRP acceptance criteria for the evaluation of conformance to the requirements of the NRC's regulations.					
7.1.2	Use of IEEE Std. 603-1991 and IEEE Std. 279-1971 for Non-Safety Systems. IEEE Std. 603-1991 is an NRC requirement for safety systems and IEEE Std. 279-1971 is an NRC requirement only for protection systems. However, these standards require that protection and safety systems be appropriately isolated from non-safety systems. Consequently, the requirements of IEEE Std. 603-1991 and IEEE Std. 279-1971 apply to the interface between safety and non-safety systems. The quality and reliability of systems important to safety that are not classified as safety systems should still be sufficient to minimize challenges to safety systems and to fulfill their overall role in plant non-safety strategy. Although IEEE Std. 603-1991 and IEEE Std. 279-1971 are not requirements for non-safety I&C systems, these standards describe concepts that are useful in any situation in which functional reliability is a goal. Consequently, although these standards are not SRP acceptance criteria for non-safety I&C systems, they are a source of design concepts that may be useful for the reviewer to consider. The scope of IEEE Std. 603-1991 is broader than that of IEEE Std. 279-1971, and the guidance of IEEE Std. 603-1991 is consequently readily adaptable for use in the review of non-safety I&C systems.					
7.1.3	Location of Detailed Acceptance Criteria and Review Methods • SRP Appendix 7.1-A provides guidance on the applicability and review methods to be used in evaluating conformance to the regulatory requirements and SRP acceptance criteria for I&C systems important to safety.					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	In three cases the discussion of review methods are extensive and is located in separate appendices that are referenced by SRP Appendix 7.1-A. These appendices are: • SRP Appendix 7.1-B provides guidance for evaluating conformance to the requirements of IEEE Std. 279-1971. • SRP Appendix 7.1-C provides guidance for evaluating conformance to IEEE Std. 603-1991. • SRP Appendix 7.1-D provides guidance for evaluating conformance to SRP acceptance criteria contained in IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."					
7.1-T, Second Rev. 5 (03/2007)	Table 7-1 Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety					
	SRP Table 7-1 identifies the regulatory requirements (denoted by "R"), and SRP acceptance criteria (denoted by "A") and their applicability to the various sections of Chapter 7 of the safety analysis report (SAR). Refer to the RG for Table 7-1. Regulatory Requirements (R), and SRP Acceptance Criteria (A) for Instrumentation and Control Systems Important to Safety					
Appendix 7.1-A, Second Rev. 5 (03/2007)	Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety					
	Refer to the SRP for the detailed criteria					
Appendix 7.1-	Guidance for Evaluation of Conformance to IEEE Std. 279					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
B, Rev. 5 (03/2007)						
7.1-B.1	**General Functional Requirements** (IEEE Std. 279-1971 Clause 4.1) Clause 4.1 of IEEE Std. 279-1971 requires in part that the protection system shall, with precision and reliability, automatically initiate protective action for the range of conditions and performance enumerated in Clauses 3(7) through 3(9) of IEEE Std. 279-1971. The applicant/licensee's analysis should confirm that the protection system has been qualified to demonstrate that the performance requirements are met. The evaluation should confirm that the general functional requirements have been appropriately allocated to the various system components. Automatic initiation is required for all protective functions; a manual initiation capability is also a requirement (see Clause 4.17 of IEEE Std. 279-1971 and Regulatory Guide 1.62, "Manual Initiation of Protection Actions"). The evaluation of the precision of the protection system is addressed to the extent that setpoints, margins, errors, and response times are factored into the analysis. The topic of reliability is addressed in the following paragraphs. Staff acceptance of system reliability is based on the deterministic criteria described in IEEE Std. 279-1971 rather than on quantitative reliability goals. The NRC staff does not endorse the concept of quantitative reliability goals as a sole means of meeting the requirements for reliability of protection systems. Quantitative reliability determination, using a combination of analysis, testing, and operating experience can provide an added level of confidence in the reliable performance of the I&C system. The applicant/licensee should justify that the degree of					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	redundancy, diversity, testability, and quality provided in the protection system design is adequate to achieve functional reliability commensurate with the safety functions to be performed					
7.1-B.2	**Single-Failure Criterion** (IEEE Std. 279-1971 Clause 4.2) Clause 4.2 of IEEE Std. 279-1971 requires in part that any single failure within the protection system shall not prevent proper protective action at the system level when required. The applicant/licensee's analysis should confirm that the requirements of the single-failure criterion are satisfied. Guidance in the application of the single-failure criterion is provided in Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Safety Systems," which endorses IEEE Std. 379-2000, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems." Where it is determined that the spatial dependence of a parameter requires several sensor channels to ensure plant protection, the redundancy requirements are determined for the individual case. In certain designs, for example, adequate monitoring of core power requires a minimum number of sensors arranged in a given configuration to provide adequate protection. This aspect of redundancy is dealt with in coordination with the organization responsible for the review of reactor systems to establish redundancy requirements. Components and systems not qualified for seismic events or accident environments and non-safety-grade components and systems are assumed to fail to function if failure adversely affects protection system performance. Conversely, these components and systems are assumed to function if functioning adversely affects protection system performance. All failures in the protection system that can be predicted as a result of an event for					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	which the protection system is designed to provide a protective function are assumed to occur if the failure adversely affects the protection system performance. In general, the lack of equipment qualification may serve as a basis for the assumption of certain failures. After assuming the failures of non-safety-grade, non-qualified equipment and those failures caused by a specific event, a random single failure is arbitrarily assumed. With these failures assumed, the protection system must be capable of performing the protective functions required to mitigate the consequences of the specific event.					
7.1-B.3	**Quality of Components and Modules** (IEEE Std. 279-1971 Clause 4.3) The applicant/licensee should confirm that quality assurance provisions of Appendix B to 10 CFR Part 50 are applicable to the protection system. The evaluation of the adequacy of the quality assurance program is addressed in the review of Chapter 17 of the SAR.					
7.1-B.4	**Equipment Qualification** (IEEE Std. 279-1971 Clause 4.4) The applicant/licensee should confirm that the protection system equipment is designed to meet the functional performance requirements over the range of environmental conditions for the area in which it is located, as identified by Clauses 3(7) and 3(8) of IEEE Std. 279-1971, and discussed in subsection 3 above. The organization responsible for the review of I&C reviews mild environment qualification and electromagnetic interference (EMI) qualification of protection system I&C equipment, and consults with other organizations to confirm qualification for harsh environments and seismic loads. The review of harsh environment qualification is coordinated with the organization responsible for the review of environmental qualification. The review of seismic qualification is coordinated with the organization					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	responsible for the review of seismic qualification. Mild environment qualification should conform with the applicable guidance of IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." Additionally, the applicant/licensee should confirm that a single failure within the environmental control system, for any area in which protection system equipment is located, will not result in conditions that could result in damage to the protection system equipment, nor prevent the balance of the protection system not within the area from accomplishing its safety function. In this regard, the loss of an environmental control system is treated as a single failure that should not prevent the protection system from accomplishing its safety functions. Because the loss of environmental control systems does not usually result in prompt changes in environmental conditions, the design bases may rely upon monitoring environmental conditions and taking appropriate action to ensure that extremes in environmental conditions are maintained within non-damage limits until the environmental control systems are returned to normal operation. If such bases are used, the applicant/licensee should confirm that there is independence between environmental control systems and sensing systems that would indicate the failure or malfunctioning of environmental control systems. Review of mild environment qualification should also include confirmation that the environmental protection of instrument sensing lines conforms with the guidance of Regulatory Guide 1.151, "Instrument Sensing Lines." EMI qualification in accordance with the guidance of Regulatory Guide 1.180, Revision 1, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Related Instrumentation and Control Systems," is an acceptable means of meeting the qualification requirements for EMI and electrostatic discharge. Lightning protection should be addressed as part of the review of electromagnetic compatibility. Lightning protection features should conform to the guidance of Regulatory Guide 1.204, "Guidelines for Lightning Protection of Nuclear Power Plants." The organizations responsible for the review of equipment qualification to harsh environments and seismic events will perform the evaluation of conformance to the requirements of GDC 2 and 4 and 10 CFR 50.49 to ensure the requirements for equipment qualification to harsh environments and seismic events are met. Guidance for the review of this equipment qualification is given in SRP Sections 3.10 and 3.11.					
7.1-B.5	**Channel Integrity** (IEEE Std. 279-1971 Clause 4.5) Information provided in Clauses 3(7) and 3(8) of IEEE Std. 279-1971 is reviewed to confirm that the design includes the qualification of equipment for the conditions identified in the design bases. Failures may not be credited to protect the integrity of other equipment. The review should confirm that tests have been conducted on protection system equipment components and the system racks and panels as a whole to demonstrate the functional performance requirements of the protection system over the range of transient and steady-state conditions of both the energy supply and the environment. Where tests have not been conducted, the applicant should confirm that the protection system components are conservatively designed to operate over the range of service conditions. Auxiliary features necessary to support protection system performance should meet all of the requirements of IEEE Std.					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	279-1971. Other auxiliary features that are part of the protection system, but not isolated from the protection system, should be designed to meet the criteria of IEEE Std. 279-1971 as necessary to assure that these components and systems do not degrade the protection systems below an acceptable level. SRP BTP 7-9 provides specific guidance for the review of anticipatory trips that are auxiliary features of a reactor protection system. The sharing of structures, systems, and components between units in multi-unit stations is permissible provided that the ability to simultaneously perform required safety functions in all units is not impaired. The review of shared displays and controls should be coordinated with the organization responsible for the review of human factors to confirm that shared user interfaces are sufficient to support the operator needs for each of the shared units. The organizations responsible for the review of electrical systems and balance of plant systems review power source requirements. Reviewers in the organization responsible for the review of I&Cs should coordinate with these organizations to confirm that I&C protection system power sources are adequate. The review of channel integrity should confirm that the design provides for protection systems to fail in a safe state, or into a state that has been demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy, or adverse environments are experienced. This aspect is typically evaluated through evaluation of the applicant/licensee's failure modes and effects analysis. The analysis should justify the acceptability of each failure effect. RTS functions should typically fail in the tripped state. ESFAS functions should fail to a predefined safe state. For many ESFAS functions this predefined safe state will be that the actuated component remains as-is.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
7.1-B.6	**Channel Independence** (IEEE Std. 279-1971 Clause 4.6) Two aspects of independence should be addressed: • Physical independence. • Electrical independence. Guidance for evaluation of physical and electrical channel independence is provided in Regulatory Guide 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," which endorses IEEE Std. 384-1992, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." The applicant/licensee should confirm that the protection system design precludes the use of components that are common to redundant channels, such as common switches for actuation, reset, mode, or test; common sensing lines; or any other features that could compromise the independence of redundant channels. Physical independence is attained by physical separation and physical barriers. Electrical independence shall include the utilization of separate power sources. The organization responsible for the review of electrical systems reviews power source requirements. Reviewers in the organization responsible for the review of I&Cs should coordinate with the electrical systems reviewers to confirm that I&C protection system power sources are adequate. Transmission of signals between independent channels should be through isolation devices. SRP BTP 7-11 provides guidance for the application and qualification of isolation devices.					
7.1-B.7	**Control and Protection System Interaction** (IEEE Std. 279-1971 Clause 4.7) Control and protection system interaction involves more than examining the electrical isolation and interconnection. The					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	functional performance of control systems must be such that a control system cannot prevent proper action of a protection system. Clause 4.7 of IEEE Std. 279-1971, with regard to isolation devices and multiple failures resulting from a credible single event, is explained by example in the document (See Clause 4.2 of IEEE Std. 279-1971). The applicant/licensee's analysis should confirm that the requirements for control and protection system interaction are satisfied.					
7.1-B.8	**Derivation of System Inputs** (IEEE Std. 279-1971 Clause 4.8) A protection system that requires loss of flow protection would, for example, normally derive its signal from flow sensors. A design might use an indirect parameter such as a pressure signal or pump speed. However, the applicant/licensee should verify that any indirect parameter is a valid representation of the desired direct parameter for all events. Even a directly measured variable should be reviewed and its response to postulated events compared with the credit taken for the parameter in the events for which it provides protection.					
7.1-B.9	**Capability for Sensor Checks** (IEEE Std. 279-1971 Clause 4.9) The most common method used to verify the availability of the input sensors is by cross checking between redundant channels that have available readout. When only two channels of readout are provided, the applicant/licensee should state the basis used to ensure that an operator will not take incorrect action when the two channel readouts differ. The applicant/licensee should state the method to be used for checking the operational availability of non-indicating sensors.					
7.1-B.10	**Capability for Test and Calibration** (IEEE Std. 279-1971 Clause 4.10) Guidance on periodic testing of the protection system is provided					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	in Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions," and in Regulatory Guide 1.118, Revision 3, "Periodic Testing of Electric Power and Protection Systems," which endorses IEEE Std. 338-1987, "Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems." The extent of test and calibration capability provided bears heavily on whether the design meets the single-failure criterion. Any failure that is not detectable must be considered concurrently with any random postulated, detectable, single failure. Periodic testing should duplicate, as closely as practical, the overall performance required of the protection system. The test should confirm operability of both the automatic and manual circuitry. The capability should be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be such that the tests do, in fact, overlap from one test segment to another. Test procedures that require disconnecting wires, installing jumpers, or other similar modifications of the installed equipment are not acceptable test procedures for use during power operation. The review of test and calibration provisions should be coordinated with the organization responsible for the review of technical specification format and content to confirm that the system design supports the types of testing required by the technical specifications. The system design should also support the compensatory actions required by technical specifications when limiting conditions for operation are not met. Typically, the design should allow for tripping or bypass of individual functions in each protection system channel.					
7.1-B.11	**Channel Bypass and Removal from Operation** (IEEE Std. 279-1971 Clause 4.11) The review of bypass and removal from operations should be					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	coordinated with the organization that is responsible for the format of technical specifications to confirm that the provisions for this bypass are consistent with the required actions of the proposed plant technical specifications.					
7.1-B.12	**Operating Bypass** (IEEE Std. 279-1971 Clause 4.12) The requirement for automatic removal of operational bypasses means that the reactor operator shall have no role in such removal. The operator may take action to prevent the unnecessary initiation of a protective action.					
7.1-B.13	**Indication of Bypass** (IEEE Std. 279-1971 Clause 4.13) Guidance on bypasses and inoperable status indication is provided in Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System."					
7.1-B.14	**Access to Means for Bypassing** (IEEE Std. 279-1971 Clause 4.14) Administrative control is acceptable to ensure that access to the means for bypassing is limited to qualified plant personnel and that permission of the control room operator is obtained to gain access.					
7.1-B.15	**Multiple Setpoints** (IEEE Std. 279-1971 Clause 4.15) The staff interpretation of "positive means" is that automatic action is provided to ensure that the more restrictive setpoint is used when required. SRP BTP 7-3 provides additional guidance on multiple setpoints used to allow operation with reactor coolant pumps out of service.					
7.1-B.16	**Completion of a Protective Action Once it is Initiated** (IEEE Std. 279-1971 Clause 4.16) The staff review of this item should include review of functional and logic diagrams to ensure that "seal-in" features are provided					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	to enable system-level protective actions to go to completion. The seal-in feature may incorporate a time delay as appropriate for the safety function. Additionally, the seal-in feature need not function until it is confirmed that a valid protective command has been received, provided the system meets response time requirements.					
7.1-B.17	**Manual Initiation** (IEEE Std. 279-1971 Clause 4.17) Features for manual initiation of protective action should conform with Regulatory Guide 1.62, "Manual Initiation of Protection Action." The review of manual controls should be coordinated with the organization responsible for the review of human factors to confirm that the functions controlled and the characteristics of the controls (e.g., location, range, type, and resolution) allow plant operators to take appropriate manual actions. The review of manual controls should include confirmation that the controls will be functional (e.g., power will be available and command equipment is appropriately qualified) during plant conditions under which manual actions may be necessary.					
7.1-B.18	**Access to Setpoint Adjustments, Calibrations, and Test Points** (IEEE Std. 279-1971 Clause 4.18) The review of access control should confirm that design features provide the means to control physical access to protection system equipment, including access to test points and means for changing setpoints. Typically such access control includes provisions such as alarms and locks on protection system panel doors, or control of access to rooms in which protection system equipment is located.					
7.1-B.19	**Identification of Protective Actions and Information Read-Out** (IEEE Std. 279-1971 Clauses 4.19 and 4.20)					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The review of information displays should be coordinated with the organization that is responsible for the review of reactor systems to confirm that the information displayed and characteristics of the displays (e.g., location, range, type, and resolution) support operator awareness of system and plant status and will allow plant operators to make appropriate decisions. The review of information displays for manually controlled actions should include confirmation that displays will be functional (e.g., power will be available and sensors are appropriately qualified) during plant conditions under which manual actions may be necessary. Protection system bypass and inoperable status indication should conform with the guidance of Regulatory Guide 1.47.					
7.1-B.20	**Information Read-Out** (IEEE Std. 279-1971 Clause 4.20) See subsection 4.19 above.					
7.1-B.21	**System Repair** (IEEE Std. 279-1971 Clause 4.21) Protection systems may include self-diagnostic capabilities to aid in troubleshooting.					
7.1-B.22	**Identification** (IEEE Std. 279-1971 Clause 4.22) Guidance on identification is provided in Regulatory Guide 1.75, which endorses IEEE Std. 384-1992. The preferred identification method is color coding of components, cables, and cabinets.					
Appendix 7.1-C, Rev 5 (03/2007)	Guidance for Evaluation of Conformance to IEEE Std. 603					
	Refer to the SRP for the detailed criteria					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Appendix 7.1-D Second Issuance (03/2007)	Guidance for Evaluation of Conformance to IEEE Std. 7-4.3.2					
	Refer to the SRP for the detailed criteria					
7.2, Rev. 5 (03/2007)	Reactor Trip System					
7.2.1	SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h).					
7.2.2	SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h)(2).					
7.2.3	IEEE Std 7-4.3.2-2003, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by Regulatory Guide 1.152, Revision 2, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants,” provides guidance on applying the safety system criteria to computer-based safety systems. SRP Appendix 7.1-D provides SRP acceptance criteria for safety and protection systems using digital computer-based technology.					
7.2.4	Item II.Q, “Defense Against Common-Mode Failures in Digital Instrument and Control Systems,” of the Staff Requirements Memorandum on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs,” provides guidance on Diversity and Defense-in-Depth. SRP BTP 7-19 provides additional guidance.					
7.3, Rev. 5 (03/2007)	Engineered Safety Features Systems					
7.3.1	SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h).					
7.3.2	SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
7.3.3	IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," provides guidance on applying the safety system criteria to computer-based safety systems. SRP Appendix 7.1-D provides SRP acceptance criteria for safety and protection systems using digital computer-based technology.					
7.3.4	Item II.Q, "Defense Against Common-Mode Failures in Digital Instrument and Control Systems," of the Staff Requirements Memorandum on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," provides guidance on Diversity and Defense-in-Depth. SRP BTP 7-19 provides additional guidance.					
7.4, Rev. 5 (03/2007)	Safe Shutdown Systems					
7.4.1	SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h).					
7.4.2	SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).					
7.4.4	SRP Appendix 7.1-D provides SRP acceptance criteria for the digital I&C compliance with IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."					
7.5, Rev. 5 (03/2007)	Information Systems Important to Safety					
7.5.1	SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h).					
7.5.2	SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
7.5.3	SRP Appendix 7.1-D provides SRP acceptance criteria for the application of the requirements of IEEE Std. 603-1991 to digital I&C. Appendix 7.1-D discusses the application of the guidance in IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2.					
7.5.4	Item II.Q, "Defense Against Common-Mode Failures in Digital Instrument and Control Systems," of the Staff Requirements Memorandum on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," provides guidance on Diversity and Defense-in-Depth. SRP BTP 7-19 provides additional guidance.					
7.5.5	Regulatory Guide 1.97, Revision 2, 3, and 4, describe methods acceptable to the NRC staff for providing instrumentation to monitor variables for accident conditions. For plants with operating licenses issued before June 2006, Regulatory Guide 1.97, Revision 2 and 3, are still effective. Licensees of these plants may, however, convert to the criteria of Revision 4 or use the criteria of Revision 4 when performing modifications that do not involve a conversion. The guidance contained in Regulatory Position 1 of Regulatory Guide 1.97, Revision 4, should be followed in these cases. Plants that obtained an operating license after June 2006 should reference the guidance of Regulatory Guide 1.97, Revision 4. SRP BTP 7-10 provides guidance on the application of Regulatory Guide 1.97.					
7.6, Rev. 5 (03/2007)	Interlock Systems Important to Safety					
7.6.1	SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h).					
7.6.2	SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).					
7.6.3	SRP Appendix 7.1-D provides SRP acceptance criteria for digital I&C compliance with IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."					
7.7, Rev. 5 (03/2007)	Control Systems					
7.7.1	SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h). Although compliance with IEEE Std. 603-1991 is required by 10 CFR 50.55a(h) only for safety systems, the criteria of IEEE Std. 603-1991 may be used as review guidance for any I&C system. Therefore, for control systems, the reviewer may use the concepts in IEEE Std. 603-1991 as a starting point.					
7.7.2	SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).					
7.7.3	SRP Appendix 7.1-D provides SRP acceptance criteria for digital I&C compliance with IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."					
7.8, Rev. 5 (03/2007)	Diverse Instrumentation and Control Systems					
7.8.1	For plants with a digital RTS or ESFAS, the NRC position on D3 should be especially noted. This position is contained in Item II.Q, "Defense Against Common-Mode Failures in Digital Instrument and Control Systems," of the Staff Requirements Memorandum on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." SRM requirements applicable to diverse I&C functions are as follows: "If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	same common-mode failure [as the safety system], shall be required to perform either the same function [as the safety system function that is vulnerable to common mode failure] or a different function [that provides adequate protection]. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary functions under the associated event conditions." "A set of displays and controls located in the main control room shall be provided for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system[s] …"					
7.8.2	SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h).					
7.8.3	SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).					
7.8.4	SRP Appendix 7.1-D provides SRP acceptance criteria for digital I&C compliance with IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2.					
7.9, Rev. 5 (03/2007)	Data Communication Systems					
7.9.1	SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h).					
7.9.2	SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).					
7.9.3	SRP Appendix 7.1-D provides SRP acceptance criteria for digital I&C compliance with IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152,					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Revision 2.					
	Appendix 7-A, Rev.5, Branch Technical Positions (BTP) (02/20/2007), has been separated into individual sections.					
Branch Technical Position 7-1, Rev. 5 (03/2007)	Guidance on Isolation of Low-Pressure Systems From the High-Pressure Reactor Coolant System					
BTP 7-1.1	At least two valves in series should be provided to isolate any subsystem whenever the primary system pressure is above the pressure rating of the subsystem.					
BTP 7-1.2	For system interfaces where both valves are motor-operated, the valves should have independent and diverse interlocks to prevent both from opening unless the primary system pressure is below the subsystem design pressure. Also, the valve operators should receive a signal to close automatically whenever the primary system pressure exceeds the subsystem design pressure.					
BTP 7-1.3	For those system interfaces where one check valve and one motor-operated valve are provided, the motor-operated valve should be interlocked to prevent the valve from opening whenever the primary pressure is above the subsystem design pressure, and to close automatically whenever the primary system pressure exceeds the subsystem design pressure.					
BTP 7-1.4	Suitable valve position indication should be provided in the control room for the interface valves.					
BTP 7-1.5	For those interfaces where the subsystem is required for emergency core cooling system operation, the above recommendations need not be implemented. System interfaces of this type should be evaluated on an individual basis, as discussed in GL 87-12 and GL 88-17.					
BTP 7-1.6	The system should satisfy the requirements of the General Design Criteria and Section 50.55a(h) of 10 CFR Part 50. 10					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	CFR 50.55a(h), "Protection and Safety Systems," requires compliance with IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Station," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plantspecific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971 and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations". SRP Appendix 7.1-B provides procedures for reviewing systems against IEEE Std 279-1971. SRP Appendix 7.1-C provides procedures for reviewing systems against IEEE Std 603-1991.					
Branch Technical Position 7-2, Rev. 5 (03/2007)	Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines					
BPT 7-2.1	Automatic opening of the valves when either primary coolant system pressure exceeds a preselected value (to be specified in the technical specifications), or a safety injection signal is present. Both primary coolant system pressure and safety injection signals should be provided to the valve operator.					
BPT 7-2.2	Visual indication in the control room of the open or closed status of the valve.					
BPT 7-2.3	Bypassed and inoperable status indication in accordance to Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System."					
BPT 7-2.4	Utilization of a safety injection signal to remove automatically (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	provisions of the technical specifications).					
Branch Technical Position 7-3, Rev. 5 (03/2007)	Guidance on Protection System Trip Point Changes for Operation With Reactor Coolant Pumps Out of Service					
BTP 7-3.1	If more restrictive safety trip points are required for operation with a reactor coolant pump out of service, and if operation with a reactor coolant pump out of service is of sufficient likelihood to be a planned mode of operation, the change to the more restrictive trip points should be accomplished automatically.					
BTP 7-3.2	Plants with designs not in accordance with the above should have included in the plant technical specifications a requirement that the reactor be shut down prior to changing the set points manually					
Branch Technical Position 7-4, Second Rev. 5 (03/2007)	Guidance on Design Criteria for Auxiliary Feedwater Systems					
BTP 7-4.1	The auxiliary feedwater system should be capable of satisfying the system functional requirements after a postulated break in the auxiliary feedwater piping inside containment together with a single electrical failure. The basis for the position is that an auxiliary feedwater piping break would result in tripping the unit and, in turn, might cause loss of offsite power. Standard staff assumptions for analyzing postulated accidents include the assumption of loss of offsite power if the affected unit generator is tripped by the accident. Such a circumstance would leave the plant without adequate means for removal of afterheat even though the reactor coolant pressure boundary was intact - an unacceptable result. Plant heat removal systems must, in any postulated piping break, be capable of removing afterheat to the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	ultimate heat sink assuming a single electrical (active) failure anywhere in the auxiliary feedwater system or in the onsite power system.					
Branch Technical Position 7-5, Rev. 5 (03/2007)	Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors					
BTP 7-5.1	GDC 20 requires that the protection system shall be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences. GDC 25 requires that these limits shall not be exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection) of control rods. Within the context of GDC 20 the staff considers operator error to be an anticipated operational occurrence, in addition to the consideration of single malfunction requirements of GDC 25, for which conformance to these requirements is to be evaluated. The applicant should perform analyses of the reactivity control systems1 and analyze the consequences of operator error to assess the impact of these events on fuel design limits. If the results of these analyses show that specified acceptable fuel design limits may be exceeded for these events, the protection system must be designed to detect and terminate these events prior to exceeding these limits. With regard to the evaluation of malfunctions within the reactivity control systems, consideration should be given to failures that cause actions as well as prevent actions, such that all possible effects are examined. Further, failures that could lead to single or multiple rod position changes or out-of-sequence rod patterns should be analyzed, as well as failures that could lead to reactivity changes by boron control systems.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Branch Technical Position 7-6, Rev. 5 (03/2007)	Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode					
BTP 7-6.1	A design that provides manual initiation at the system level of the transfer to the recirculation mode, while not ideal, is sufficient and satisfies the intent of IEEE Std 279-1971 or IEEE Std 603-1991, provided that adequate instrumentation and information display are available to the operator so that he or she can make the correct decision at the correct time. Furthermore, it should be shown that, in case of operator error, sufficient time and information are available so that the operator can correct the error, and that the consequences of such an error are acceptable.					
BTP 7-6.2	Automatic transfer to the recirculation mode is preferable to manual transfer, for the reasons cited above, and should be provided for standard plant designs submitted for review on a generic basis under the Commission's standardization policy.					
Branch Technical Position 7-8, Rev. 5 (03/2007)	Guidance for Application of Regulatory Guide 1.22					
PTB 7-8.1	The protection system should satisfy the requirements of the General Design Criteria and Section 50.55a(h) of 10 CFR Part 50. 10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991, and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971 and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971. SRP Appendix					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	7.1-B provides guidance for reviewing systems against IEEE Std 279-1971. SRP Appendix 7.1-C provides guidance for reviewing systems against IEEE Std 603-1991.					
Branch Technical Position 7-9, Rev. 5 (03/2007)	Guidance on Requirements for Reactor Protection System Anticipatory Trips					
BTP 7-9.1	All reactor trips incorporated in the reactor protection system should be designed to meet the requirements of IEEE Std 279-1971, or IEEE Std 603-1991. This position applies to the entire trip function, from the sensor to the final actuated device. For sensors located in non-seismic areas, the installation (including circuit routing) and design should be such that the effects of credible faults (i.e., grounding, shorting, application of high voltage, or electromagnetic interference) or failures in these areas could not be propagated back to the reactor protection system and degrade the reactor protection system performance or reliability. The sensors should be qualified to operate in a seismic event, i.e., not fail to initiate a trip for conditions which would cause a trip.					
Branch Technical Position 7-10, Rev. 5 (03/2007)	Guidance on Application of Regulatory Guide 1.97					
	Refer to the BTP for the details of Table 1					
B. Item 3.1	Environmental Qualification 10 CFR 50.49(b)(3), “Certain post-accident monitoring equipment,” has been interpreted by the staff as follows: For plants using Revisions 2 or 3 of Regulatory Guide 1.97,					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	accident monitoring equipment that falls within the scope of Category 1 or 2 equipment should be environmentally qualified as required by 10 CFR 50.49, or the applicant/licensee should provide an acceptable alternative for complying with 10 CFR 50.49(b)(3). For plants using Revision 4 of Regulatory Guide 1.97, accident monitoring equipment identified as Type A, B, or C in accordance with that guide should be environmentally qualified as required by 10 CFR 50.49. Type D variables should be environmentally qualified for the particular accident's postulated environment at the installed location in accordance with the plant's licensing basis. Licensees converting to Revision 4 or performing modifications based on Revision 4 may reference previously accepted alternatives as their basis for deviations from the environmental qualification criteria in Revision 4.					
B. Item 3.2	Seismic Qualification For plants using Revisions 2, 3, or 4 of Regulatory Guide 1.97, if a reactor licensing basis does not include a commitment to Regulatory Guide 1.100, “Seismic Qualification of Electric Equipment for Nuclear Power Plants,” and credit is taken for original equipment in meeting the guidelines identified in Regulatory Guide 1.97, installation of the original equipment in conformance with the licensing basis for seismic qualification is acceptable, provided the other guidelines identified in Regulatory Guide 1.97 and this BTP are satisfied. However, for all reactors, new instrumentation that is installed to satisfy Regulatory Guide 1.97 or to replace original equipment for which credit was taken in satisfying Regulatory Guide 1.97 should satisfy the seismic qualification guidelines identified in Regulatory Guide 1.97.					
B. Item 3.3	Redundancy For Category 1 variables under Revisions 2 and 3 of Regulatory					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Guide 1.97, no single failure should prevent the operators from being presented information necessary to determine the safety status of the plant and to maintain the plant in a safe condition following an accident. Channels provided to monitor a Revision 2 or 3 Category 1 variable do not need to meet this criterion during channel maintenance, test, or calibration, provided the duration of such testing satisfies the applicable requirements of the licensing basis. For example, the time interval required for a test, calibration, or maintenance operation could be shown to be so short that it would have an insignificant effect on overall availability of the accident monitoring instrumentation system. This BTP does not supplement Regulatory Guide 1.97, Revision 4 and IEEE Std. 497-2002 guidance on this topic.					
B. Item 3.4	Independence of Redundant Instrumentation For plants using Revisions 2, 3, or 4 of Regulatory Guide 1.97, if a reactor licensing basis does not include a commitment to Regulatory Guide 1.75, "Criteria for Independence of Electric Systems," and credit is taken for original equipment in meeting the guidelines identified in Regulatory Guide 1.97, installation of the original equipment in conformance with the licensing basis requirements for separation and independence is acceptable, provided the other guidelines identified in Regulatory Guide 1.97 and this BTP are satisfied. However, for all reactors, new instrumentation that is installed to satisfy Regulatory Guide 1.97 or to replace original equipment for which credit was taken in satisfying Regulatory Guide 1.97 should satisfy the separation and isolation guidelines in Regulatory Guide 1.97.					
B. Item 3.5	Display and Recording Revisions 2 and 3 of Regulatory Guide 1.97 state in part that if direct or immediate trend or transient information is essential for operator information or action, the recording should be					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	continuously available on dedicated recorders. Otherwise, the information may be continuously updated, stored in computer memory, and displayed on demand. For the latter non-essential applications, the use of Category 2 computers or dedicated Category 2 recorders is acceptable for recording Category 1 information, provided the Category 1 instrumentation is isolated from the Category 2 instrumentation using qualified isolation devices. Guidance on the application and qualification of isolation devices is provided in SRP BTP 7-11. This BTP does not supplement Regulatory Guide 1.97, Revision 4 and IEEE Std. 497-2002 guidance on this topic.					
B. Item 3.6	Range Deviations from the range values identified in Revisions 2 and 3 of Regulatory Guide 1.97 may be acceptable if supported by analyses demonstrating that the indication would remain on scale with appropriate margins for any design basis event or accident for which the instrumentation might be required for operator information. An appropriate margin should include allowance for analytical uncertainties and instrumentation uncertainties. However, Regulatory Guide 1.97 identifies that, for a limited number of functionally significant variables (e.g., containment pressure, primary system pressure), instrument ranges should extend beyond values that the selected variables can attain under limiting conditions. Guidance on uncertainties is provided in SRP BTP 7-12. The ranges and footnotes for radiation and meteorological instrumentation that are provided in Revision 3 of Regulatory Guide 1.97 should be applicable for plants using Revision 4 of Regulatory Guide 1.97. Applicants/licensees using Revision 4 should document differences from the Revision 3 ranges and footnotes for radiation and meteorological instrumentation.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
B. Item 3.7	Minimizing Measurements To the extent practicable, the same instruments should be used for accident monitoring as are used for normal operations of the plant. In cases in which a single display may indicate the reading of more than one instrument, the intent of this recommendation is met if the same variable and same display are used for accident monitoring even though the sensor(s) providing the signal are different.					
B. Item 3.8	Alternate Instrumentation The use of alternate instrumentation to monitor variables different from those identified in Regulatory Guide 1.97, Revisions 2 and 3, is acceptable, provided that all three of the following criteria are met: a. The alternate instrumentation fulfills the purpose of the variables identified in Regulatory Guide 1.97. b. The alternate instrumentation conforms to the design and qualification criteria for the variables identified in Regulatory Guide 1.97. c. No credit is taken by the applicant/licensee in post-accident procedures, emergency operating procedures, or functional recovery guidelines for indication of the variables identified in Regulatory Guide 1.97 for which the alternative instrumentation is proposed. Revision 4 of Regulatory Guide 1.97 does not identify specific variables to be displayed; therefore, this topic does not apply to Revision 4.					
B. Item 3.9	Guidance for Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) Variables Table 1, BWR Variables, and Table 2, PWR Variables, of					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Regulatory Guide 1.97, Revision 2, and Table 2, BWR Variables, and Table 3, PWR Variables, of Regulatory Guide 1.97, Revision 3, identify guidelines for the range, design/qualification category, and purpose for specific BWR and PWR variables. For selected BWR and PWR variables identified in the tables, acceptable deviations or clarifications are identified in Tables 1 and 2, respectively, of this BTP. Tables 1 and 2 of this BTP list Regulatory Guide 1.97 variables and types of deviations from Regulatory Guide 1.97 guidelines (e.g., deviations with respect to category, redundancy, range, direct measurement), and provide a summary of the acceptance guidelines or clarification associated with the deviations. Revision 4 of Regulatory Guide 1.97 does not identify specific variables; therefore Tables 1 and 2 of this BTP do not apply to Revision 4.					
B. Item 3.10	Conversion to Revision 4 Applicants/licensees of reactors that are committed to either Revision 2 or 3 of Regulatory Guide 1.97 may voluntarily convert the plant's entire accident monitoring program to the criteria of Revision 4 of Regulatory Guide 1.97. Conversion means revising the commitment for the plant's entire accident monitoring program from the current licensing basis (either Revision 2 or 3) to the guidance in Revision 4. The conversion to Revision 4 could include physical changes (e.g., replacing an instrument), licensing changes (e.g., technical specification changes), changes in variable types and associated design and qualification criteria, changes in the function or purpose of the variable, and changes in the range being monitored. This conversion should be supported by a complete analysis of the plant's accident monitoring program against all of the criteria in Revision 4. The applicant/licensee should document the results of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	analysis in a table format that includes for each variable the following: variable name, current type, current function or purpose, current range, proposed type, proposed function or purpose, proposed range, and any criteria that would be changed. For variables for which there is a proposed change in type or purpose, the applicant/licensee should document the rationale for the change.					
B. Item 3.11	Modifications Using Revision 4 Applicants/licensees of reactors that are committed to either Revision 2 or 3 of Regulatory Guide 1.97 may voluntarily use the criteria of Revision 4 of Regulatory Guide 1.97 to perform modifications that do not involve a conversion. The applicant/licensee should first perform an analysis to determine the complete list of variables and their associated types in accordance with the selection criteria of Revision 4. Without such an analysis, there would not be a means to correlate Revision 4 criteria being applied to the modification of variables that have been previously licensed to the criteria of Revision 2 or 3. The applicant/licensee should document the proposed modifications in a table format that includes for each variable the following: variable name, current type, current function or purpose, current range, proposed type, proposed function or purpose, proposed range, and any criteria that would be changed. For variables for which there is a proposed change in type or purpose, the applicant/licensee should document the rationale for the change. Licensees may make modifications within the plant's current licensing basis without referencing Revision 4 of Regulatory Guide 1.97.					
Branch Technical	Guidance on Application and Qualification of Isolation Devices					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Position 7-11, Rev. 5 (03/2007)						
B Item 3.1	Description of Device Application Isolation devices should be classified as part of the safety system and powered in accordance with criteria of IEEE Std 603-1991 or IEEE Std 279-1971 and the guidelines of Regulatory Guide 1.75. If non-safety power sources interface to the isolation device, the applicant/licensee should verify that the non-safety power is not required for the device to perform its isolation function. MCF requirements should be established by analysis of proximate circuits that are credible sources of the fault, either through inadvertent application from human error or through a fault or failure postulated to occur that involves proximate circuits, cabling, or terminations (e.g., a “hot short” from an adjacent conductor). The determination of specific MCF characteristics is plant-specific. Surge waveforms and characteristics should be defined for the worst-case conditions expected at the installation. The acceptable leakage current into the safety system should be identified for specified MCFs.					
B Item 3.2	Description of Device Design The design of isolation devices should conform to IEEE Std 603-1991, or IEEE Std 279-1971, and Regulatory Guide 1.75 guidelines for (1) independence of redundant safety divisions, and (2) independence between protection (safety) and control (non-safety) systems. The isolation device should include design features for which					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	credit is taken (e.g., surge protectors or barriers) and identification of the application limits of the device. The device should be designed for postulated electrical faults or failures, including open circuits, short circuits, ground, and application of an MCF. The specified MCF should equal or exceed the application requirements. Regulatory Guide 1.75 suggests that the MCF include the levels and duration of the fault current on the non-safety side of the device. ANSI Std C84.1-1989, Table 1, “Standard Nominal System Voltages and Voltage Ranges,” provides an acceptable basis for identifying nominal voltages and guidelines for steady-state tolerances. The device design should accommodate the fault voltage and current waveforms and characteristics defined for the application. Appropriate industry standards should be used as a basis for establishing the fault-transient exposure level (e.g., IEEE Std C62.41.1-2002, IEEE Std C62.41.2-2002). The physical arrangement of components in the isolation device should be configured to prevent, in the event of failure, the effects of shattered parts or material (e.g., solder spatter), fire, and smoke from breaching the isolation barrier.					
B Item 3.3	Description of Test Method A description of the specific testing performed for each type of isolation device should be provided. This should include elementary or schematic diagrams as necessary to describe the test configuration and how the MCF and surges will be applied to the devices during the test. The basis for the set of postulated electrical faults and failures should be included in the test program.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A specific definition of pass/fail acceptance criteria for each type of device should be provided. This should include justification that the pass/fail acceptance criterion is sufficient to demonstrate that the tested device meets the requirements of IEEE Std 603-1991, Clause 5.6, or IEEE Std 279-1971, Clause 4.7.2. Regulatory Guide 1.75 recommends that: • The maximum credible voltage or current transient applied to the device output should not degrade below an acceptable level the operation of the circuit connected to the device input. • Shorts, grounds, or open circuits occurring in the output will not degrade below an acceptable level the circuit connected to the device input. • Transient voltages that may appear in the output circuit (e.g., surges) should also be considered. • The qualification should consider the levels and duration of the fault current on the non-safety side of the device. For safety/non-safety isolation, during and following the application of the MCF or surge test, there should be no degradation or distortion of the isolation device input that would have a detrimental effect on the performance of the safety system. For isolation of redundant safety circuits, there should be no degradation or distortion of the redundant channel that would have a detrimental effect on the performance of the safety system. Applicable industry standards should be used as the basis for performing the qualification testing (e.g., IEEE Std C62.45-2002). Devices might be used either for isolation of safety circuits from					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	non-safety circuits or for isolation of redundant safety divisions. For qualification testing, the detailed device configuration depends on the objective of the isolation and the specific type and configuration of the isolation device (e.g., relay, isolation amplifier, optical-electronic device). The MCF represents the application of the maximum credible alternating current (AC) and direct current (DC) voltages and currents that are applied to the device in common and differential modes (as defined by IEEE Std 100-2000, "The Authoritative Dictionary of IEEE Standards Terms 7th Edition") that exist based on the installation of the device. Common mode refers to faults between both signal terminals and a common reference plane (ground) and causes the potential of both sides of the transmission path to be changed simultaneously and by the same amount relative to the common reference plane (ground). Differential mode refers to faults between the signal terminals that cause the potential of one side of the signal transmission path to be changed relative to the other side. The mode of application should satisfy the following guidelines for test configurations. For isolation of safety circuits from non-safety circuits: • MCFs and surges should be applied between the signal output terminals of the (non-safety) circuits (differential mode) and between any output terminal and ground (common mode). • Surges should be applied to power terminals. The guidance of IEEE Std C62.45-2002 is acceptable for surge testing at the power input. • The signal input terminals should be monitored to assure that no unacceptable interactions (degradations or distortions) between the safety and non-safety circuits would occur.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	For isolation between redundant safety circuits: • MCFs should be applied between the signal input terminals (differential mode) and between any input terminal and ground (common mode); the output should be monitored to assure that no unacceptable interactions (degradations or distortions) between redundant safety circuits will occur. • Surges should be applied to power terminals. The guidance of IEEE Std C62.45-2002 is acceptable for surge testing at the power input. • MCFs should also be applied to the output terminals in the differential mode and between any output terminal and ground (common mode); the input should be monitored to assure that no unacceptable interactions (degradations or distortions) between redundant safety circuits will occur. MCFs should be applied to the isolation device for a sufficient duration to allow any measurable effects to occur on the isolation device and to allow monitored values or effects to reach steady-state.					
B Item 3.4	Description of Test Results Test data and results should verify that the design basis faults, including short circuits, open circuits, grounds, MCF, and surge were applied to the device in all of the applicable connection modes (i.e., applicable input, output, power, and ground connection modes). Test data and results should verify that the test acceptance criteria are met.					
Branch	Guidance on Establishing and Maintaining Instrument					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
Technical Position 7-12, Rev. 5 (03/2007)	Setpoints					
B Item 3.1	Setpoint Documentation The following information on the licensee/applicant's setpoint program should be provided for review: • Facility setpoint list identifying safety setpoints and non-safety setpoints for functions providing protective functions important to safety or that are relevant to compliance with technical specification limiting conditions for operation. • Identification of safety setpoints that are not safety-limit-related LSSS and the basis for this determination. • Identification of setpoints that trigger procedural actions that are important to safety. • Description of the setpoint methodology and procedures used in determining setpoints, including information sources, scope, assumptions, interface reviews, and statistical methods. • Terminology used to describe limits, allowances, and tolerances, and environmental or other effects used to support setpoint calculations. • Technical specifications and basis for LSSSs. • Basis for acceptable as-found band and acceptable as-left band and determination of the instrument operability based on acceptable as-found band and acceptable as-left band. • Basis for calibration intervals. • Basis for assumptions regarding instrument uncertainties and a discussion of the method used to determine uncertainty values.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	• Description of the provisions for control of measuring and test equipment used for calibration of the instrument. • Description of the program and methodology used to monitor and manage instrument uncertainties, including drift. A documented basis for safety system setpoint should be available for Staff review. Documentation should conform with the guidance of Regulatory Guide 1.105, Revision 3. The description of the instrument channel in accordance with ISA-S67.04-1994, Part I, should include: • Description of the functional and performance criteria for the initiation and execution of the safety functions at the setpoints. • Instrument specifications, including range, accuracy, repeatability, hysteresis, dynamic response, environmental qualification, calibration reference, and calibration intervals for each instrument type. • Instrument loop diagrams showing all hardware elements of the instrument loop(s). • Instrument and tubing layout drawings and installation details showing locations and elevations of instruments and tubing relative to a reference datum, as well as the points where the instrument interfaces with the monitored process. • For digital instrumentation, the configuration database for the instrumentation functions, and identification of digital elements (hardware and software) where error could be introduced into the measurement-for example, errors that could result from analog-to-digital or digital-to-analog conversion or from numerical					

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	methods used in the software (e.g., curve fitting). The description of assumptions in accordance with ISA-S67.04-1994, Part I, should include the environmental allowances (temperature, pressure, humidity, radiation, vibration, seismic, and electrical) for the instruments.					
B Item 3.2	Analysis Supporting Establishment of Setpoints and Instrumentation Tolerances The applicant/licensee should document the bases and the calculations of measurement uncertainties. The methods by which setpoints are calculated should conform to the guidance of Regulatory Guide 1.105, Revision 3.					
B Item 3.3	Statistical Guidelines for Instrument Uncertainty In the review of uncertainties in determining a trip setpoint and its allowable values, the NRC staff typically uses 95/95 tolerance limits as an acceptable criterion, i.e., a 95 percent probability that the constructed limits contain 95 percent of the population of interest for the surveillance interval selected.					
B Item 3.4	Guidelines for Graded Approach ISA-S67.04-1994, Part I, Section 4 states that the safety significance of various types of setpoints important to safety may differ, and thus a less rigorous setpoint determination method for certain functional units and limiting conditions of operation may be applied. The use of a graded approach allows a less rigorous setpoint determination method based on the safety significance of the instrument function. However, the grading technique chosen by the applicant/licensee should be consistent with the standard and should consider and bound all known applicable uncertainties regardless of setpoint application. Additionally, the application of the standard using a graded approach is also					

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	appropriate for non-safety system instrumentation maintaining design limits in the technical specifications					
B Item 3.5	Basis for Instrument Calibration Intervals The applicant/licensee should evaluate the effects of extended calibration intervals on instrument uncertainties, equipment qualification, and vendor maintenance provisions to assure that an extended surveillance interval does not result in exceeding the assumptions stated in the safety analysis. Generic Letter 91-04, Enclosure 2, "Guidance for Addressing the Effect of Increased Surveillance Intervals on Instrument Drift and Safety Analysis Assumptions," provides acceptable guidance for justifying extended calibration intervals through the use of data analysis, monitoring, and assessment. This approach has been used for plants to accommodate a 24-month fuel cycle change. For changes to surveillance test intervals for reasons other than a 24-month fuel cycle, the submittals have followed the risk informed approach and followed the guidance of Regulatory Guides 1.174, 1.177, and 1.200.					
Branch Technical Position 7-13, Rev. 5 (03/2007)	Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors					
B Item 3.1	Supporting Analysis Analyses, and information on the instrument maintenance and calibration program should be provided to support the adequacy of the cross-calibration program. The analysis should, as a minimum, address the following topics. • Justification that the cross-calibration program is consistent with the characteristics of the RTD sensors, including RTD specifications, range, accuracy,					

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	repeatability, dynamic response, installed configuration, environmental qualification, calibration reference, calibration history, and calibration intervals. • The specific methods or analyses used for signal conditioning or processing (for example, averaging, biasing, failure detection, data quality determination, and error compensation). • The planned process for cross-calibration and response time determination. • Justification that the performance requirements and failure criteria assumed in the plant accident/event analyses are satisfied by the cross-calibration process and testing results. • The technical basis for the acceptance criteria and values of cross-calibration points monitored in-situ throughout the RTD range, to ensure that the data are adequate for detecting degradation or systematic drift.					
B Item 3.2	Traceability of the Installed Reference RTD to Laboratory Calibration Data Laboratory calibration involves measuring the RTD's resistance at several known temperatures. The data are then used to provide a calibration curve for the device. In addition, the RTD response time can be determined under laboratory conditions using controlled temperature baths and a methodology to calculate the RTD response time over the measuring temperature range. The installation of a calibrated RTD should include a test procedure to demonstrate the response time applicability of the laboratory test results. Loop current step response (LCSR) testing is an acceptable way to verify that the conditions of the installed RTD are adequately correlated to the laboratory test data. Response time testing of the installed RTDs using LCSR should					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	use an analytical technique such as the LCSR transformation identified in NUREG-0809, "Review of Resistance Temperature Detector Time Response Characteristics," to correlate the in-situ results with the results of a laboratory-type temperature test.					
B Item 3.3	Acceptable Methods for In-Situ Testing Verification of RTD calibrations should be accomplished by installing a newly calibrated reference RTD sensor and then cross-correlating with the measurements of the other RTDs subject to the same temperature and flow environment. A critical element in this approach is providing assurance that all sensor elements are subject to sufficiently similar temperature and flow environments. Other methods, such as using a diverse parameter to provide a crosscorrelation reference, can be used if adequate justification is provided. Before installing a reference or new RTD, the sensor should either be calibrated in a laboratory or, if the manufacturer's calibration data are to be used, the applicant/licensee should perform an analysis or test to verify the RTD has retained its calibration. The application temperatures should be within the manufacturer's highest calibration range. All data should be taken at isothermal plant conditions and all loops (hot legs and cold legs) should be at similar temperatures. If this condition can not be assured then the applicant/licensee should provide for removal of one or more of the RTDs at each representative location and for replacement with a newly calibrated RTD. The applicant/licensee should provide an analysis which states the limits of acceptable calibration, response times, and in-situ testing of the RTDs. Test procedures, with acceptance criteria, should state the limits of the calibration, particularly the					

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	dependency of the data on uniform coolant temperature and flow. Correction factors or bias values should be established to compensate for non-isothermal conditions. Because plant temperatures cannot be perfectly controlled, fluctuations and drift in the primary coolant temperature might occur during in-situ testing. The test data should be corrected for the fluctuations and drift in the coolant temperature. If during the testing incomplete mixing of the reactor coolant should occur, the test data should be corrected for the temperature differences. Reactor coolant temperatures should be stable and uniform. In the event this is not the case the data should be corrected to account for these effects. Equipment used in the test should be accurate to within the necessary tolerance and have stable performance. See BTP 7-12 for guidance on determining plant instrumentation tolerances.					
B Item 3.4	Response Time Testing Even though response time testing is independent from the cross-calibration test, it should be performed for the existing and the newly installed reference sensors to account for installation effects and to identify degradation. The resulting test data and analysis should support correlation of each of the existing sensors in the common flow path to its laboratory response time test data, and also to the laboratory response time test data for the reference sensor. Correlation between LCSR test results for the existing sensors and LCSR test results for the reference sensor may be used to establish the correlation with the reference RTD laboratory test data. As-Found/As-Left Surveillance Data The applicant/licensee should maintain a database of the "as-left"					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	and "as-found" calibration and response time tests for each sensor. To monitor systematic drift or degradation, at each refueling cycle, or as required by the plant's technical specifications, a newly calibrated RTD or a new RTD with recent calibration data should be installed at representative location(s) determined by analysis. The cross-correlation to the reference RTD(s) should be monitored using "as found" and "as left" data records. Test data and analysis should identify and account for differences in isothermal conditions and demonstrate that the drift is random and is within an acceptable band as determined by setpoint analyses, and that systematic drift is not exhibited. If historical data reveals potential drift problems which would exceed the allowable values of temperature drift in testing for any sensor then the applicant/licensee should verify the calibration of the deviating sensor(s) and identify appropriate corrective action. Analysis to project RTD drift should be available for all RTDs within the protection system.					
B Item 3.5	Control/Protection Interaction and Common-Cause Failure During In-Situ Testing If the applicant/licensee uses test equipment common to redundant channels, qualified isolation should be provided to preclude single-failure effects on redundant channels or unacceptable protection/control interactions.					
Branch Technical Position 7-14, Rev. 5 (03/2007)	Guidance on Software Reviews for Digital Computer-Based Instrumentation and Controls Systems					
	Refer to the BTP for the detailed criteria.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Branch Technical Position 7-16 Withdrawn	Guidance on Level of Detail Required for Design Certification Applications Under 10 CFR Part 52 (see ML070450253)	NA				Exclude
Branch Technical Position 7-17, Rev. 5 (03/2007)	Guidance on Self-Test and Surveillance Test Provisions					
B Item 3.1	Failure Detection Failures detected by hardware, software, and surveillance testing should be consistent with the failure detectability assumptions of the single-failure analysis and the failure modes and effects analysis.					
B Item 3.2	Self-Test Features Digital computer-based I&C systems should include self-test features to confirm computer system operation on system initialization. Digital computer-based I&C systems should generally include continuous self-testing. Some small, stand-alone, embedded digital computers may not need self-testing. Typical self-tests include monitoring memory and memory reference integrity, using watch-dog timers or processors, monitoring communication channels, monitoring central processing unit status, and checking data integrity. Other self-testing features that are candidates for incorporation into digital computer-based I&C systems include plausibility checks for intermediate results, evaluation using different methods, ranges of variables, array bound checking, well-defined outputs for detected failures, reporting of errors for which error recovery techniques are used, use of counters and reasonableness traps, and correctness verification of transferred					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	parameters. SRP BTP 7-14 discusses a number of functional characteristics for software design, such as robustness and timing, which could give rise to self-testing features. Self-tests may also include automatic calibration tests such as the use of fundamental physical principles in Johnson noise thermometry to calibrate resistance temperature detectors (RTDs). The design of automatic self-test features should maintain channel independence, maintain system integrity, and meet the single-failure criterion during testing. The scope and extent of interfaces between software that performs protection functions and software for other functions such as self-test should be designed to minimize the complexity of the software logic and data structures. The safety classification of the hardware and software used to perform automatic self-testing should be equivalent to that of the tested system unless physical, electrical, and communications independence are maintained such that no failure of the test function can inhibit the performance of the safety function. The positive aspects of self-test features should not be compromised by the additional complexity that may be added to the safety system by the self-test features. The improved ability to detect failures provided by the self-test features should outweigh the increased probability of failure associated with the self-test feature. Self-test functions should be verified during periodic functional tests.					
B Item 3.3	Surveillance Testing Systems should be able to conduct periodic surveillance testing consistent with the technical specifications and plant procedures. As delineated in Regulatory Guide 1.118, periodic testing consists					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of functional tests and checks, calibration verification, and time response measurements. As required by IEEE Std. 279-1971, Clause 4.13, or IEEE Std. 603-1991, Clause 5.8.3, and as stated in Regulatory Guide 1.47, if the protective action of some part of a protection or safety system is bypassed or deliberately rendered inoperative for testing, that fact should be continuously indicated in the control room. Provisions should also be made to allow operations staff to confirm that the system has been properly returned to service. Regulatory Guide 1.118 states in part that test procedures for periodic tests should not require makeshift test setups. For digital computer-based systems, makeshift test setups, including temporary modification of code or data that must be appropriately removed to restore the system to service, should be avoided. If automatic test features are credited with performing surveillance test functions, provisions should be made to confirm the execution of the automatic tests during plant operation. The capability to periodically test and calibrate the automatic test equipment should also be provided. The balance of surveillance and test functions that are not performed by the automatic test feature should be performed manually to meet the intent of Regulatory Guide 1.118. In addition, the automatic test feature function should conform to the same requirements and considerations (e.g., test interval) as the manual function. The safety classification and quality of the hardware and software used to perform periodic testing should be equivalent to that of the tested system. The design should maintain channel independence, maintain system integrity, and meet the single-failure criterion during testing. Commercial digital computer-based equipment used to perform periodic testing should be					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	appropriately qualified for its function.					
B Item 3.4	Actions on Failure Detection The design should have either the automatic or manual capability to take compensatory action on detection of any failed or inoperable component. The design capability and plant technical specifications, operating procedures, and maintenance procedures should be consistent with each other. Plant procedures should specify manual compensatory actions and mechanisms for recovery from automatic compensatory actions. Mechanisms for operator notification of detected failures should comply with the system status indication provisions of IEEE Std. 603-1991 and should be consistent with, and support, plant technical specifications, operating procedures, and maintenance procedures.					
Branch Technical Position 7-18, Rev. 5 (03/2007)	Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems					
	Purchased PLC hardware; embedded and operating systems software, programming tools, and peripheral components should be qualified to a level commensurate with the system they are designed to support. EPRI TR-106439 and EPRI TR-107330 describe an acceptable process for qualifying commercial systems. NUREG/CR-6421 provides additional information on the characteristics of an acceptable process for qualifying existing software, and discusses the use of engineering judgment and compensating factors for purchased PLC software. See the discussion of the commercial dedication of predeveloped software					

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	(PDS) in SRP Appendix 7.0-A. PLC hardware, embedded and operating system software, and peripheral components built specifically for nuclear power plant applications should meet the appropriate quality criteria. The embedded and operating system software should meet the acceptance criteria contained in SRP BTP 7-14, appropriately graded for the application in which the PLC will be used. The application software (ladder logic or other) should meet the acceptance criteria contained in SRP BTP 7-14 commensurate with the system it is designed to support. Application software should conform with the recommended practices of NUREG/CR-6463. Tools for developing application software or loading it into the PLC should be qualified to a level commensurate with the system they are designed to support. PLC-based functions should conform with the guidance regarding real-time performance and testing outlined in SRP BTP 7-21 and SRP BTP 7-17. Administrative or hardware lockout controls that prevent unauthorized modification of the PLC software should be in place. This is particularly important because many PLCs are designed so that their software is easy to modify. All software changes should be under configuration management control. In particular, administrative procedures for maintaining control of the software implemented in the PLC should be detailed in the configuration management plan.					
Branch Technical Position 7-19,	Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Rev. 5 (03/2007)						
B Item 3.1	For each anticipated operational occurrence in the design basis occurring in conjunction with each single postulated common-cause failure, the plant response calculated using best-estimate (realistic assumptions) analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary. The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action.					
B Item 3.2	For each postulated accident in the design basis occurring in conjunction with eachsingle postulated common-cause failure, the plant response calculated using best-estimate (realistic assumptions) analyses should not result in radiation release exceeding the 10 CFR 100 guideline values, violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment (i.e., exceeding coolant system or containment design limits). The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action.					
B Item 3.3	When a failure of a common element or signal source shared by the control system and RTS is postulated and the common-cause failure results in a plant response that requires reactor trip and also impairs the trip function, then diverse means that are not subject to or failed by the postulated failure should be provided to perform the RTS function. The diverse means should assure that the plant response calculated using best-estimate (realistic assumptions) analyses does not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or					

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	violation of the integrity of the primary coolant pressure boundary.					
B Item 3.4	When a failure of a common element or signal source shared by the control system and ESFAS is postulated and the common-cause failure results in a plant response that requires engineered safety features (ESF) and also impairs the ESF function, then diverse means that are not subject to or failed by the postulated failure should be provided to perform the ESF function. The diverse means should assure that the plant response calculated using best-estimate (realistic assumptions) analyses does not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.					
	No failure of monitoring or display systems should influence the functioning of the RTS or ESFAS. If plant monitoring system failure induces operators to attempt to operate the plant outside safety limits or in violation of the limiting conditions of operation, the analysis should demonstrate that such operator-induced transients will be compensated for by protection system function.					
B Item 3.5	The adequacy of the diversity provided with respect to the above criteria must be justified. Interconnections between the RTS and ESFAS (for interlocks providing for reactor trip if certain ESFs are initiated, ESF initiation when a reactor trip occurs, or operating bypass functions) are permitted if it can be demonstrated that the functions required by the ATWS rule (10 CFR 50.62) are not impaired. NUREG/CR-6303, Section 3.2, describes six types of diversity and describes how instances of different types of diversity might be combined into an overall case for the sufficiency of the diversity provided. Typically, several types of diversity should exist, some of which should exhibit one or more of the stronger attributes listed in NUREG/CR-6303. Functional diversity and					

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	signal diversity are considered to be particularly effective. The following cautions should be noted where applicable: • The justification for equipment diversity, or for the diversity of related system software such as a real-time operating system, must extend to the equipment's components to assure that actual diversity exists. For example, different manufacturers might use the same processor or license the same operating system, thereby incorporating common failure causes. Claims for diversity based just on difference in manufacturer name are insufficient without consideration of the above. • With respect to software diversity, experience indicates that independence of failure causes may not be achieved in cases where multiple versions of software are developed using the same software requirements. Other considerations, such as functional and signal diversity, that lead to different software requirements form a stronger basis for diversity. Displays and manual controls provided for compliance with Point 4 of the NRC position on D3 should be sufficient both for monitoring the plant state and to enable control room operators to actuate the systems that will place the plant in a hot shutdown condition. In addition, the displays and controls should be sufficient for the operator to monitor and control the following critical safety functions: reactivity level, core heat removal, reactor coolant inventory, containment isolation, and containment integrity. This additional manual capability is necessary in new reactors because all of the protection and control systems are digital-computer-based and thus vulnerable to common-cause failure. These displays and controls provide plant operators with information and control capabilities that are not subject to					

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	common-cause failures due to software errors in the plant's automatic digital I&C safety system because they are independent and diverse from that system. The point at which the manual controls are connected to safety equipment should be downstream of the plant's digital I&C safety system outputs. These connections should not compromise the integrity of interconnecting cables and interfaces between local electrical or electronic cabinets and the plant's electromechanical equipment. To achieve system-level actuation at the lowest possible level in the safety system architecture, the controls may be connected either to discrete hardwired components or to simple (e.g., component function can be completely demonstrated by test), dedicated, and diverse, software-based digital equipment that performs the coordinated actuation logic. The displays may include digital components that are dedicated exclusively to the display function. Functional characteristics (e.g., range, accuracy, time response) should be sufficient to provide operators with the information needed to place and maintain the plant in a hot shutdown condition. Human factors engineering principles and criteria should be applied to the selection and design of the displays and controls. Human-performance requirements should be described and related to the plant safety criteria. Recognized human-factors standards and design techniques should be employed to support the described human-performance requirements.					
Branch Technical Position 7-21, Rev. 5 (03/2007)	Guidance on Digital Computer Real-Time Performance					

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B Item 3.1	Limiting Response Times Limiting response times should be shown to be consistent with safety requirements (e.g., suppress power oscillations, prevent fuel design limits from being exceeded, prevent a non-coolable core geometry). Setpoint analyses and limiting response times should also be shown to be consistent. The reviewer should verify that limiting response times are acceptable to the organizations responsible for reactor systems, electrical systems, and plant systems before accepting the limiting response times as a basis for timing requirements.					
B Item 3.2	Digital Computer Timing Requirements Digital computer timing should be shown to be consistent with the limiting response times and characteristics of the computer hardware, software, and data communications systems. Computer system timing requirements that should be addressed in a software requirements specifications are described in SRP BTP 7-14.					
B Item 3.3	Architecture The level of detail in the architectural description should be sufficient that the Staff can determine the number of message delays and computational delays interposed between the sensor and the actuator. An allocation of time delays to elements of the system and software architecture should be available. In initial design phases (e.g., at the point of design certification application), an estimated allocation of time delays to elements of the proposed architecture should be available. Subsequent detailed design and implementation should develop refined timing allocations down to unit levels in the software architecture. A design should be feasible with currently known methods and representative equipment. Design timing feasibility may be					

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	demonstrated by allocating a timing budget to components of the system architecture so that the entire system meets its timing requirements. See NUREG/CR-6083, Sections 2.2, 2.3.1, and 2.3.2, and NUREG/CR-6082. The timing budget should include internal and external communication delays, with adequate margins. Any non-deterministic delays should be noted and a basis provided that such delays are not part of any safety functions, nor can the delays impede any protective action. Software architectural timing requirements should be addressed in a software architectural description as described in SRP BTP 7-14. Databases, disk drives, printers, or other equipment or architectural elements subject to halting or failure should not be able to impede protective system action.					
B Item 3.4	Design Commitments Design basis documents should describe system timing goals. Timing requirements should be satisfied by design commitments. A design should consider data rates, data bandwidths, and data precision requirements for normal and off-normal operation, including the impact of environmental extremes. There should be sufficient excess capacity margins to accommodate likely future increases in demands or software or hardware changes to equipment. Design basis documents should identify design practices that the applicant/licensee will use to avoid timing problems. Risky design practices such as non-deterministic data communications, non-deterministic computation, use of interrupts, multitasking, dynamic scheduling, and event-driven design should be avoided.					

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	When such practices are allowed, the applicant/licensee should describe methods for control of the associated risk. NUREG/CR-6082 and NUREG/CR-6083 describe risky design practices in more detail.					
B Item 3.5	Performance Verification The means proposed, or used, for verifying a system's timing should be consistent with the design. Testing and/or analytic justification should show that the system meets limiting response times for a reasonable, randomly selected subset of system loads, conditions, and design basis events. The subset should include some limiting load conditions and be chosen by persons independent of the persons who designed the system. Both analytical and test techniques of timing analysis have drawbacks. It is difficult to demonstrate completeness of timing tests. Completeness is easier to demonstrate for analyses, but analyses predict extreme times that are not actually possible. Therefore, analysis and testing are often combined in a complementary manner to confirm that a system can meet the limiting response times. Measurement methods should be appropriate to the resolution and detail required. Timing measurements should meet projections or the anomalies should be satisfactorily explained (NUREG/CR-6083, Sections 2.1, 2.3.3, and 2.3.4).					
B Item 3.6	Use of Cyclic Real-Time Executive In systems that include a cyclic real-time executive (operating system), a typical cycle includes application modules, diagnostic					

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	modules, and other support modules. A watch-dog timer is normally set at the beginning of each cycle and reset at the end. If the cycle is not completed before the watch-dog timer period is complete, an error is generated. A basis should be provided that describes the cycle and demonstrates that the watch-dog timer is correctly implemented, the time required for the application modules does not exceed the allotted time given in the architecture timing budget, and diagnostic and other support modules will not cause the allotted time to be exceeded. Examples of solutions acceptable to the Staff may be found in the Safety Evaluation Reports for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, "Issuance of Amendments on the Core Protection Calculator System Upgrade," dated October 24, 2003, and the Siemens Power Corporation, Topical Report EMF-2110(NP), "Teleperm XS: A Digital Reactor Protection System," dated May 5, 2000.					
B Item 3.7	Use of Part-Scale Prototypes In systems that have not been implemented and tested on a full scale, expected system delays on scale-up should be calculated and shown to be less than limiting system response times (NUREG/CR-6083, Sections 2.1.3 and 2.1.4). A basis should be provided that describes the effects of adding sensors, divisions, communication links, controllers, computer nodes, or actuation devices required to scale the test system to full scale. Test data should confirm scaling as well as performance projections. Exceptions are considered anomalies or abnormal events.					

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	Prototypes designed to demonstrate scaling should include all significant architectural elements plus enough additional elements to show the scaling effects to be measured.					
	CHAPTER 8, Electric Power					
8.1, Rev. 3 (03/2007)	Electric Power - Introduction					
8.1.1	Specific SRP acceptance criteria are contained in SRP Sections 8.2, 8.3.1, 8.3.2, and 8.4.					
8.2, Rev. 4 (03/2007)	Offsite Power System					
8.2.1	GDC 2 is satisfied as it relates to structures, systems, and components of the offsite power system being capable of withstanding the effects of natural phenomena such as high and low atmospheric temperatures, high wind, rain, lightning discharges, ice and snow conditions, and weather events causing regional effects as established in Chapter 3 of the SAR, and reviewed by the organizations with primary responsibility for the reviews of plant systems, civil engineering and geosciences, and mechanical engineering.					
8.2.2	GDC 4 is satisfied as it relates to structures, systems, and components of the offsite power system being protected against dynamic effects, including the effects of missile that may result from equipment failures during normal operation, maintenance, testing, and postulated accidents, as established in Chapter 3 of the SAR and reviewed by the organizations with primary responsibility for the reviews of plant systems, materials, and chemical engineering.					
8.2.3	GDC 5 is satisfied as it relates to: sharing of structures, systems, and components of the preferred power systems; guidelines of Regulatory Guide 1.32 as related to its endorsement of Section 7					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of IEEE Std 308, relating to sharing of structures, systems, and components of the Class 1E power system at multi-unit stations; and guidance related to the sharing of structures, systems, and components of the offsite power system (preferred power supply) at multi-unit stations, previously addressed in the 1980 and earlier versions of IEEE Std 308, but now covered in the industry standard for preferred power supply (Reference 52).					
8.2.4	GDC 17 is satisfied as it relates to the preferred power system's (i) capacity and capability to permit functioning of structures, systems, and components important to safety; (ii) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies; (iii) physical independence; (iv) availability and the guidelines of Regulatory Guide 1.32 (see also IEEE Std 308) as related to the availability and number of immediate access circuits from the transmission network; and (v) capability to meet the guidelines of Appendix A to SRP Section 8.2 as related to acceptability of generator circuit breakers and generator load break switches. For evolutionary light water reactor design applications, as documented in SECY 94-084 for designs such as the CE-ABB System 80+ and the GE ABWR, the design should provide at least one offsite circuit to each redundant safety division that is supplied directly from an offsite power source with no intervening non-safety buses, thereby permitting the offsite source to supply power for safety buses in the event the non-safety bus(es) fails. The design should also include an alternate power source to non-safety loads, unless it can be demonstrated that existing design margins will ensure that transients for loss of non-safety power events are no more severe than those associated with the turbine-trip-only event specified in current plant designs					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(References 33 and 35). These issues are reviewed in detail in SRP Section 8.3.1 For passive reactor design applications, the passive safety-related systems only require electric power for valves and related instrumentation, which can be supplied from the onsite Class 1E batteries and associated dc and ac distribution systems. The acceptability of this design for the AP 1000 is documented in SECY-05-0227 and FSER NUREG-1793. If no offsite power is available, it is expected that the non-safety-related diesel generators would be available for important plant functions, but this non-safetyrelated ac power is not relied on to maintain core cooling or containment integrity. Therefore, this passive reactor design supports an exemption to the requirement of GDC 17 for two physically independent offsite circuits, by providing safety-related passive safety systems for core cooling and containment integrity (see also References 33, 34, 35). However, one offsite power source with sufficient capacity and capability from the transmission network must be provided to power the safety-related systems and all other auxiliary systems under normal, abnormal, and accident conditions. The offsite power source should be designed to minimize to the extent practical the likelihood of its failure under normal, abnormal, and accident conditions.					
8.2.5	GDC 18 is satisfied as it relates to the inspection and testing of the offsite electric power system.					
8.2.6	GDCs 33, 34, 35, 38, 41, and 44 are satisfied as they relate to the operation of the offsite electric power system, encompassed in GDC 17, to ensure that the safety functions of the systems described in GDC's 33, 34, 35, 38, 41, and 44 are accomplished, assuming a single failure where applicable.					
8.2.7	10 CFR 50.63 is satisfied as it relates to an AAC power source (as defined in 10 CFR 50.2) provided for safe shutdown in the event of a station blackout (non-DBA), and the guidelines of Regulatory Guide 1.155 are followed as they relate to the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	adequacy of the AAC source and the independence of the AAC power source from the offsite power system and onsite power system and sources.					
8.2.8	Except for passive reactor designs described in subsection II (2) above, new applications must provide an adequate AAC source of diverse design (with respect to ac onsite emergency sources) that is consistent with the guidance in Regulatory Guide 1.155 and capable of powering at least one complete set of normal safe shutdown loads. These issues are reviewed in detail in SRP Section 8.4. 8. 10 CFR 50.65, Section 50.65(a)(4), as it relates to the requirements to assess and manage the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. Acceptance is based on meeting the following specific guidelines: A. Regulatory Guide 1.160, as related to the effectiveness of maintenance activities for onsite emergency ac power sources including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase LOOP frequency, or reduce the capability to cope with a LOOP or SBO). B. Regulatory Guide 1.182, as related to implementing the provisions of 10 CFR 50.65 (a)(4) by endorsing Section 11 to NUMARC 93-01, “Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, February 22, 2000.					
	REFERENCES: 33. SECY 94-084 “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs,” dated March 28, 1994. Approved in the SRM of June 30, 1994. 34. SECY-95-132 “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS)					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	in Passive Plant Designs." Approved in the SRM of June 28, 1995. 35. NRC Memorandum; From: D. Crutchfield; To: File; Subject: Consolidation of SECY-94- 084 and SECY-95-132, July 24, 1995. SECY-94-084 was approved in the SRM of June 30, 1994. SECY-95-132 was approved in the SRM of June 28, 1995. 52. IEEE Standard 765-1983, "IEEE Standard for Preferred Power Supply (PPS) for Nuclear Power Generating Stations." (2002 is latest revision)					
8.3.1, Rev. 3 (03/2007)	AC Power Systems (Onsite)					
8.3.1.1	GDC 2 is satisfied as it relates to SSCs of the onsite ac power system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapter 3 of the SAR, and reviewed by the organizations with primary responsibility for the reviews of plant systems, civil engineering and geosciences, and mechanical engineering.					
8.3.1.2	GDC 4 is satisfied as it relates to SSCs of the ac power system being capable of withstanding the effects of missiles and environmental conditions associated with normal operation and postulated accidents, as established in Chapter 3 of the SAR and reviewed by the organizations with primary responsibility for the reviews of plant systems, materials, and chemical engineering.					
8.3.1.3	GDC 5 is satisfied as it relates to the sharing of SSCs of the ac power system and the following guidelines: A. Regulatory Guide 1.32, as it relates to the sharing of SSCs of the Class 1E power system at multi-unit stations. B. Regulatory Guide 1.81, as it relates to the sharing of SSCs of the ac power system, positions C.2 and C.3.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
8.3.1.4	GDC 17 is satisfied as it relates to the onsite ac power system's: (a) capacity and capability to permit functioning of SSCs important to safety; (b) independence, redundancy, and testability to perform its safety function assuming a single failure; and (c) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network. Acceptance is based on meeting the following specific guidelines: A. Regulatory Guide 1.6, as it relates to the independence of the onsite ac power system, positions D.1, D.2, D.4, and D.5. B. Regulatory Guide 1.9 (see also IEEE Std 387). C. Regulatory Guide 1.32 (see also IEEE Std 308), as it relates to design criteria for onsite ac power systems. D. Regulatory Guide 1.53 (see also IEEE Stds 279 and 603), as it relates to the application of the single-failure criterion to safety systems. E. Regulatory Guide 1.75 (see also IEEE Std 384), as it relates to the onsite ac power system. F. Regulatory Guide 1.153 G. Regulatory Guide 1.155, as it relates to the use of onsite emergency ac power sources for station blackout. H. Regulatory Guide 1.204 (see also IEEE Stds 665, 666, 1050, and C62.23), as it relates to the lightning and surge protection for the onsite ac power system.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	I. NUREG/CR-0660 is incorporated as it relates to the following recommendations: i. The diesel generator sets should be capable of operation at less than full load for extended periods of time without degradation of performance or reliability. With offsite power available, no-load operation of the diesel generators will occur following a safety injection signal. Extended no-load operation of this equipment should be minimized. Operating procedures should be provided that limit extended no-load operation of the diesel generators. The procedures should include loading the diesel engine to a minimum of 25% of full load for 1 hour after 8 hours of continuous no-load operation or to a load as recommended by the engine manufacturer. ii. A complete formal training program should be provided for all personnel who will be responsible for the maintenance and availability of the diesel generators. The depth and quality of training shall be at least equivalent to that provided by major diesel engine manufacturers' training programs. iii. A preventive maintenance program should be provided which encompasses investigative testing of components which have a history of repeated malfunctioning and a plan for the replacement of those components that require constant attention and repair with other products of proven reliability. iv. Repair and maintenance procedures should provide for a final equipment check prior to an actual start-run-					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	load test to ensure that all electrical circuits are functional (i.e., fuses in place, no loose wires, test leads removed, etc.) and all valves are in the proper position. The test procedure(s) should explicitly state that upon satisfactory test completion the diesel generator unit should be returned to a ready automatic standby service under the control of the control room operator. v. Except for sensors and other equipment that need to be directly mounted on the engine or associated piping, the controls and monitoring instruments should be installed on a free-standing, floor-mounted panel located on a vibration-free floor area. [NOTE: If the floor is not vibration free, the panel should be equipped with vibration mounts.] J. Acceptance criteria for the interface between the onsite ac power system and the offsite power system to satisfy the requirements of GDC 17 in evolutionary light water reactor design applications are documented in SECY-91-078, which states that the design should include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon the failure of any non-safety bus. The evolutionary light water reactor design should also include an alternate power source to non-safety loads, unless it can be demonstrated that existing design margins will ensure that transients for loss of non-safety power events are no more severe than those associated with the turbine-trip-only event specified in current plant designs.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Passive light water reactor design applications provide passive safety systems that do not need Class 1E ac electric power, other than that provided by the Class 1E dc batteries and their inverters, to accomplish the plant's safety-related functions for 72 hours. However, in accordance with SECY-94-084, SECY-95- 132, and Regulatory Guide 1.206 Section C.IV.10, ac power system features will be evaluated using the process for regulatory treatment of non-safety systems (RTNSS) for electrical distribution issues on passive designs. The AP1000 passive plant design certification, for example, includes an exemption to the requirement of GDC 17 for two physically independent offsite circuits, by providing safety-related passive safety systems for core cooling and containment integrity. However, even for this design, one offsite power source with sufficient capacity and capability from the transmission network should be provided to power the safety-related systems and all other auxiliary systems under normal, abnormal, and accident conditions. The offsite power source should be designed to minimize to the extent practical the likelihood of its failure under normal, abnormal, and accident conditions. Detailed reviews of the offsite ac power system and its interface with the onsite power system for ALWR design applications are covered in Section 8.2, "Offsite Power System."					
8.3.1.5	GDC 18 is satisfied as it relates to the testability of the onsite ac power system, and the following guidelines: A. Regulatory Guide 1.32 (see also IEEE Std 308), as it relates to capability for testing of the onsite ac power					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	system. B. Regulatory Guide 1.47, with respect to indicating the bypass or inoperable status of portions of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the system it actuates to perform their safety-related functions. C. Regulatory Guide 1.118 (see also IEEE Std 338), as it relates to the capability for testing the onsite ac power system. D. Regulatory Guide 1.153 (see also IEEE Std 603), as it relates to the onsite ac power system.					
8.3.1.6	The design requirements for an onsite ac power supply for systems covered by GDCs 33, 34, 35, 38, 41, and 44 are encompassed in GDC 17.					
8.3.1.7	GDC 50 is satisfied as it relates to the design of containment electrical penetrations containing circuits of the ac power system, and the guidelines of Regulatory Guide 1.63 are followed (see also IEEE Stds 242, 317, and 741), as related to the capability of electric penetration assemblies in containment structures to withstand a LOCA without loss of mechanical integrity and the external circuit protection for such penetrations, as well as to ensure that electrical penetrations will withstand the full range of fault current (minimum to maximum) available at the penetration.					
8.3.1.8	10 CFR 50.63, as it relates to use of the redundancy and reliability of diesel generator units as a factor in limiting the potential for station blackout events. Acceptance is based on meeting the following specific guidelines: A. Regulatory Guide 1.9, as it relates to the adequacy of the diesel generator surveillance criteria provided to attain and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	maintain the target reliability levels of diesel generator units. B. Regulatory Guide 1.155, as it relates to use of the reliability of emergency onsite ac power sources as a factor in determining the coping duration for station blackout and the establishment of a reliability program for attaining and maintaining source target reliability levels. Determination of station blackout coping time is reviewed in detail in SRP Section 8.4. Except for passive reactor designs described in the acceptance criteria of SRP 8.3.1 subsection II.4.J above, new applications should provide an adequate AAC source of diverse design (with respect to onsite ac emergency sources) that is consistent with the guidance in Regulatory Guide 1.155 and capable of powering at least one complete set of normal safe shutdown loads. These issues are reviewed in detail under SRP Section 8.4.					
8.3.1.9	10 CFR 50.65, Section 50.65(a)(4), as it relates to the requirements to assess and manage the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. Acceptance is based on meeting the following specific guidelines: A. Regulatory Guide 1.160, as it relates to the effectiveness of maintenance activities for onsite emergency ac power sources including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase loss of offsite power (LOOP) frequency, or reduce the capability to cope with a LOOP or station blackout (SBO)). B. Regulatory Guide 1.182, as it relates to implementing the provisions of 10 CFR 50.65 (a)(4) by endorsing Section 11					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	to NUMARC 93-01, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, February 22, 2000.					
8.3.1.10	10 CFR 50.55a(h) as it relates to protection systems for plants with construction permits issued after January 1, 1971, but before May 13, 1999, which must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std 279-1971. Nuclear power plants with applications filed on or after May 13, 1999 for preliminary and final design approvals (10 CFR Part 52, Appendix O), design certification, construction permits, operating licenses, and combined licenses that do not reference a final design approval or design certification, must meet the requirements for safety systems in IEEE Std 603-1991 and the correction sheet dated January 30, 1995. Branch technical positions and industry standards that are acceptable to the staff for implementing the requirements of GDCs 2, 4, 5, 17, 18, and 50 are identified in SRP Section 8.1, and Table 8.1. In addition, 10 CFR 50.34(f)(2)(v), (xiii), and (xx), related to Task Action Plan items I.D.3, II.E.3.1 and II.G.1 of NUREG-0718 and NUREG-0737, provide additional guidance for the reviewer.					
8.3.2, Rev. 3 (03/2007)	DC Power Systems (Onsite)					
8.3.2.1	Regulatory Guide 1.6 positions D.1, D.3, and D.4, as they relate to the independence between redundant onsite dc power sources and between their distribution systems.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
8.3.2.2	Regulatory Guide 1.32, as it relates to the design, operation, and testing of the safety-related portions of the onsite dc power system. Except for sharing of safety-related dc power systems in multi-unit nuclear power plants, RG 1.32 endorses IEEE Std. 308-2001.					
8.3.2.3	Regulatory Guide 1.75, as it relates to the physical independence of the circuits and electrical equipment that comprise or are associated with the onsite dc power system.					
8.3.2.4	Regulatory Guide 1.81, as it relates to the sharing of structures, systems, and components of the dc power system. Regulatory Position C.1 states that multi-unit sites should not share dc systems.					
8.3.2.5	Regulatory Guide 1.128, as it relates to the installation of vented lead-acid storage batteries in the onsite dc power system.					
8.3.2.6	Regulatory Guide 1.129, as it relates to maintenance, testing, and replacement of vented lead-acid storage batteries in the onsite dc power system.					
8.3.2.7	Regulatory Guide 1.118, as it relates to the capability to periodically test the onsite dc power system.					
8.3.2.8	Regulatory Guide 1.153, as it relates to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety systems of nuclear plants, including the application of the single failure criterion in the onsite dc power system. As endorsed by Regulatory Guide 1.153, IEEE Std. 603 provides a method acceptable to the staff to evaluate all aspects of the electrical portions of the safety-related systems, including basic criteria for addressing single failures. However, as stated in 10 CFR 55a(h), all plants are not required to comply with IEEE Std. 603. Only applications filed on or after May 13, 1999, for preliminary and final design approvals (10 CFR Part 52, Appendix O), design certification, and construction permits; operating licenses and combined licenses that do not reference a final design approval or design certification must meet the requirements for safety systems in IEEE Std. 603-1991 and the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	correction sheet dated January 30, 1995. Operating nuclear power plants are encouraged, but not required to, comply with IEEE Std. 603 for future system-level modifications.					
8.3.2.9	Regulatory Guide 1.53, as it relates to the application of the single-failure criterion.					
8.3.2.10	Regulatory Guide 1.63, as it relates to the capability of electric penetration assemblies in containment structures to withstand a loss of coolant accident without loss of mechanical integrity and the external circuit protection for such penetrations.					
8.3.2.11	Regulatory Guide 1.155, as it relates to the capability and the capacity of the onsite dc power system for an SBO, including batteries associated with the operation of the alternate ac (AAC) power source(s) (if used).					
8.3.2.12	The guidelines of Regulatory Guide 1.160, as they relate to the effectiveness of maintenance activities for dc power systems. Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17.					
8.3.2.13	The guidelines of Regulatory Guide 1.182, as they relate to conformance to the requirements of 10 CFR 50.65(a)(4) for assessing and managing risk when performing maintenance.					
8.4 (03/2007)	Station Blackout					
8.4.1	The guidelines of RG 1.155, as they relate to compliance to 10 CFR 50.63. NUMARC-8700, Revision 0, also provides guidance acceptable to the staff for meeting these requirements. Table 1 of RG 1.155 provides a cross-reference to NUMARC-8700, Revision 0, and notes when the RG takes precedence.					
8.4.2	The guidelines of RG 1.155, as they relate to compliance to 10 CFR 50.63. NUMARC-8700, Revision 0, also provides guidance acceptable to the staff for meeting these requirements. Table 1 of RG 1.155 provides a cross-reference to NUMARC-8700, Revision 0, and notes when the RG takes precedence.					
8.4.3	The guidelines of RGs 1.9 (Ref. 6) and 1.155, as they relate to					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	the reliability program implemented to ensure that the target reliability goals for onsite EAC power sources (typically diesel generator units) are adequately maintained.					
8.4.4	The guidelines of RG 1.160 (Ref. 8), as they relate to the effectiveness of maintenance activities for onsite EAC power sources, including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase LOOP frequency, or reduce the capability to cope with a LOOP or SBO). Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17.					
8.4.5	The guidelines of RG 1.182 (Ref. 9), as they relate to conformance to the requirements of 10 CFR 50.65(a)(4) for assessing and managing risk when performing maintenance.					
	REFERENCE: 6. Regulatory Guide 1.9, “Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants.” 8. Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” 9. Regulatory Guide 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.					
8-A, Rev. 1 (03/2007)	General Agenda, Station Site Visits					
	Refer to the BTP for the detailed criteria.					
Branch Technical Position 8-1, Rev. 3 (03/2007)	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines					
BTP 8-1.1	Automatic opening of the valves when either primary coolant system pressure exceeds a preselected value (to be specified in the technical specifications) or a safety injection signal is present.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Both primary coolant system pressure and safety injection signals should be provided to the valve operator.					
BTP 8-1.2	Visual indication in the control room of the open or closed status of the valve					
BTP 8-1.3	An audible and visual alarm, independent of item 2, above, that is actuated by a sensor on the valve when the valve is not in the fully open position.					
BTP 8-1.4	Use of a safety injection signal to remove automatically (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with provisions of the technical specifications). Conformance with the relevant criteria for operating bypasses described in IEEE Std. 603, as endorsed in RG 1.153, constitutes an acceptable alternative approach. It should be noted that BTP 8-4 may also be applied to these isolation valves and should be used, when applicable, in conjunction with this Branch Technical Position. It should also be noted that IEEE Std. 1290 provides information on motor-operated valve protection, control, and testing.					
Branch Technical Position 8-2, Rev. 3 (03/2007)	Use of Diesel-Generator Sets for Peaking					
BTP 8-2.1	The staff's position regarding the use of onsite emergency power diesel-generator sets for purposes other than that of supplying standby power when needed is that such use should be prohibited. In particular, emergency power diesel-generator sets should not be used for peaking service.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Branch Technical Position 8-3, Rev. 3 (03/2007)	Stability of Offsite Power Systems					
BPT 8-3.1	The staff has concluded, from a review of appropriate reliability data, that power systems with supporting grid interties meet the grid availability criterion with some margin. This conclusion is applicable to the review of most plants located on the U.S. mainland.					
BPT 8-3.2	A strong indication exists that an isolated system large enough to justify inclusion of a nuclear unit will also meet this criterion. However, as a conservative approach, the staff will examine the generating capacity of a system, including interties if available, available to withstand outage of the largest unit. If the available capacity is judged marginal in its ability to provide adequate stability of the grid, additional measures should be taken. These may include provisions for additional capability and margin for the onsite power system beyond the normal requirements or other measures that may be appropriate in a particular case. The additional measures to be taken should be determined on an individual case basis.					
Branch Technical Position 8-4, Rev. 3 (03/2007)	Application of the Single Failure Criterion to Manually Controlled Electrically Operator Valves					
BPT 8-4.1	Failures of components in electrical systems, including valves and other fluid system components, in both the “fail to function” sense and the “undesirable function” sense, should be considered in designing against a single failure, even though the valve or other fluid system component may not be called upon to function in a given safety operational sequence.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
BPT 8-4.2	When it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component, and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component. The plant technical specifications should include a list of all electrically operated valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.					
BPT 8-4.3	Electrically operated valves that are classified as “active” valves (i.e., are required to open or close in various safety system operational sequences, but are manually controlled) should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless (1) electrical power can be restored to the valves from the main control room, (2) valve operation is not necessary for at least 10 minutes following occurrence of the event requiring such operation, and (3) it is demonstrated that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant technical specifications should include a list of the required positions of manually controlled, electrically operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.					
BPT 8-4.4	When the single failure criterion is satisfied by removal of electrical power from valves described in items 2 and 3, above, these valves should have redundant position indication in the main control room, and the position indication system should, itself, meet the single failure criterion.					
BPT 8-4.5	The phrase “electrically operated valves” includes both valves					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	operated directly by an electrical device (e.g., a motor-operated valve or a solenoid-operated valve) and those valves operated indirectly by an electrical device (e.g., an air-operated valve with an air supply controlled by an electrical solenoid valve).					
Branch Technical Position 8-5, Rev. 3 (03/2007)	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems					
BPT 8-5.1	The bypass indicators should be arranged to enable the operator to determine the status of each safety system and whether continued reactor operation is permissible.					
BPT 8-5.2	When a protective function of a shared system can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit.					
BPT 8-5.3	The means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated cases of erroneous indications cannot be eliminated by another practical design.					
BPT 8-5.4	Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to safety. Administrative procedures should not require immediate operator action based solely on the bypass indications.					
BPT 8-5.5	The indication system should be designed and installed in a manner that precludes the possibility of adverse effects on plant safety systems. Failure or bypass of a protective function should not be a credible consequence of failures occurring in the indication equipment, and the bypass indication should not reduce the required independence between redundant safety systems.					
BPT 8-5.6	The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Branch Technical Position 8-6, Rev. 3 (03/2007)	Adequacy of Station Electric Distribution System Voltages					
BTP 8-6.1	In addition to the undervoltage scheme provided to detect LOOP at the Class 1E buses, a second level of undervoltage protection with time delay should be provided to protect the Class 1E equipment. This second level of undervoltage protection should satisfy the following criteria: a. The selection of undervoltage and time delay setpoints should be determined from an analysis of the voltage requirements of the Class 1E loads at all onsite system distribution levels. b. Two separate time delays should be selected for the second level of undervoltage protection based on the following conditions: i. The first time delay should be long enough to establish the existence of a sustained degraded voltage condition (i.e., something longer than a motor-starting transient). Following this delay, an alarm in the control room should alert the operator to the degraded condition. The subsequent occurrence of a safety injection actuation signal (SIAS) should immediately separate the Class 1E distribution system from the offsite power system. In addition, the degraded voltage relay logic should appropriately function during the occurrence of an SIAS followed by a degraded voltage condition. ii. The second time delay should be limited to prevent					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	damage to the permanently connected Class 1E loads. Following this delay, if the operator has failed to restore adequate voltages, the Class 1E distribution system should be automatically separated from the offsite power system. The bases and justification for such an action must be provided in support of the actual delay chosen. c. The voltage sensors should be designed to satisfy the following applicable requirements derived from IEEE Std. 279 and/or IEEE Std. 603, as endorsed by RG 1.153: i. Class 1E equipment should be used and should be physically located at and electrically connected to the Class 1E switchgear. ii. An independent scheme should be provided for each division of the Class 1E power system. iii. The undervoltage protection should include coincidence logic on a per bus basis to preclude spurious trips of the offsite power source. iv. The voltage sensors should automatically initiate the disconnection of offsite power sources whenever the voltage setpoint and time delay limits (cited in item 1.b.2 above) have been exceeded. v. Capability for test and calibration during power operation should be provided. vi. Annunciation must be provided in the control room for any bypasses incorporated in the design.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	d. The technical specifications should include limiting conditions for operations, surveillance requirements, trip setpoints, and maximum and minimum allowable values for the first level of undervoltage protection (LOOP) relays and the second-level (degraded voltage) protection sensors and associated time delay devices.					
BTP 8-6.2	The Class 1E bus load shedding scheme should automatically prevent shedding during sequencing of the emergency loads to the bus. The load shedding feature should, however, be reinstated upon completion of the load sequencing action. The technical specifications must include a test requirement to demonstrate the operability of the automatic load shedding features at least once every refueling outage/cycle. An adequate basis must be provided if the load shedding feature is retained during the above load sequencing of the emergency loads to the bus.					
BTP 8-6.3	The voltage levels at the safety-related buses should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power sources by appropriate adjustment of the voltage tap settings of the intervening transformers. The tap settings selected should be based on an analysis of the voltage at the terminals of the Class 1E loads. The analyses performed to determine minimum operating voltages should typically consider maximum unit steady-state and transient loads for events, such as a unit trip, loss-of-coolant accident, startup or shutdown, with the offsite power supply (grid) at minimum anticipated voltage and only the offsite source being considered available. Maximum voltages should be analyzed with the offsite power supply (grid) at maximum expected voltage concurrent with minimum unit loads (e.g., cold shutdown, refueling). A separate set of the above analyses should be performed for each available connection to the offsite power supply.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
BTP 8-6.4	The analytical techniques and assumptions used in the voltage analyses cited in item 3 above must be verified by actual measurement. The verification and test should be performed before initial full-power reactor operation on all sources of offsite power by taking the following actions: a. Loading the station distribution buses, including all Class 1E buses down to the 120/208-volt level, to at least 30 percent b. Recording the existing grid and Class 1E bus voltages and bus loading down to the 120/208-volt level at steady-state conditions and during the start of both a large Class 1E and non-Class 1E motor (not concurrently) Note: To minimize the number of instrumented locations (recorders) during the motor-starting transient tests, the bus voltages and loading need only be recorded on that string of buses that previously showed the lowest analyzed voltages from item 3 above. c. Using the analytical techniques and assumptions of the previous voltage analyses cited in item 3 above, and the measured existing grid voltage and bus loading conditions recorded during conduct of the test, calculate a new set of voltages for all the Class 1E buses down to the 120/208-volt level d. Compare the analytically derived voltage values against the test results With good correlation between the analytical results and the test results, the test verification requirement will be met. That is, the validity of the mathematical model used to perform the analyses					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of item 3 will have been established, thereby establishing the validity of the results. In general the test results should not be more than 3 percent lower than the analytical results; however, the difference between the two, when subtracted from the voltage levels determined in the original analyses, should never be less than the Class 1E equipment-rated voltages.					
Branch Technical Position 8-7, Rev. 3 (03/2007)	Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status					
BTP 8-7.1	Diesel-generator unit bypass or deliberately induced inoperability status should be automatically indicated in the control room when the bypass or deliberately induced inoperable condition can be expected to occur more frequently than once per year and can render the unit unavailable to adequately respond to an automatic or operator-initiated emergency demand. Manually induced indication may be desirable and is permitted for diesel-generator unit bypass or deliberately induced inoperability status for those conditions expected to occur less frequently than once per year					
BTP 8-7.2	All status indication should be sufficiently precise to prevent misinterpretation. Furthermore, disabling or bypass indicators should be separate from nondisabling indicators and should be physically arranged to enable the operator to clearly determine the status of each diesel-generator unit. An acceptable design includes a separate alarm for each disabling condition or a single shared alarm with reflash capability. The alarms should be displayed in the control room and at the diesel-generator unit for all disabling conditions, with wording that indicates that the diesel-generator unit is incapable of adequately responding to an emergency demand.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
BTP 8-7.3	When a shared diesel-generator unit can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit.					
BTP 8-7.4	The indication system should be designed and installed to preclude the possibility of adverse effects on the diesel-generator units. Failures in the indication equipment should not result in diesel-generator unit failure or bypass of the diesel-generator unit, and the bypass indication should not reduce the required independence between redundant diesel-generator units.					
BTP 8-7.5	The indication system should be capable of ensuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified.					
BTP 8-7.6	RG 1.9, positions C.1.6 through C.1.8, contains further guidance to be addressed regarding status and anomalous conditions indication and alarms for diesel-generators.					
	CHAPTER 9, Auxiliary Systems					
9.1.1, Rev. 3 (03/2007)	Criticality Safety of Fresh and Spent Fuel Storage and Handling					
9.1.1.1	The criteria for GDC 62 are specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.1, ANSI/ANS 57.2, and ANSI/ANS 57.3, as they relate to the prevention of criticality accidents in fuel storage and handling.					
9.1.2, Rev. 4 (03/2007)	New and Spent Fuel Storage					
9.1.2.1	Acceptance for meeting the relevant aspect of GDC 2 is based on compliance with positions C.1 and C.2 of Regulatory Guide (RG) 1.13 and applicable portions of RG 1.29, and RG 1.117. For the spent fuel storage facility, additional guidance acceptable for meeting this criterion is found in American Nuclear Society (ANS)					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	57.2, paragraphs 5.1.1, 5.1.3, 5.1.12.9, and 5.3.2. For the new fuel storage facility, additional guidance acceptable for meeting this criterion is found in ANS 57.3, paragraphs 6.2.1.3(2), 6.2.3.1, 6.3.1.1, 6.3.3.4, and 6.3.4.2.					
9.1.2.2	Acceptance for meeting the relevant aspect of GDC 4 is based on positions C.2 and C.3 of RG 1.13, and RG 1.115 and 1.117.					
9.1.2.3	GDC 5 is met by sharing the SSCs important to safety between the units in a manner that does not degrade the performance of their safety functions.					
9.1.2.4	Acceptance for meeting the relevant aspect of GDC 61 for the spent fuel storage facility is based on compliance with positions C.4, C.6, C.10, C.11 and C.12 of RG 1.13 and the appropriate paragraphs of ANS 57.2. Acceptance for meeting this criterion for the new fuel storage facility is based on compliance with the appropriate paragraphs of ANS 57.3. Acceptance is also based on meeting the fuel storage capacity requirements noted in subsection III.1 of this SRP section. The following design considerations are evaluated: A. Provisions for periodic inspections of components important to safety. B. Suitable shielding for radiation protection, including adequate water levels. C. Appropriate containment and confinement systems. D. Residual heat removal capability by effective coolant flow through the storage racks for spent fuel assemblies. E. Prevention of reduction in fuel storage coolant inventory under accident conditions.					
9.1.2.5	Acceptance for meeting the relevant aspect of GDC 63 for spent fuel storage is based on compliance with position C.7 of RG 1.13					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	and paragraph 5.4 of ANS 57.2. Acceptance for meeting this criterion for the dry storage of new fuel is based on radiation monitoring pursuant to 10 CFR 70.24 or acceptable prevention of an increase in effective multiplication factor (K_{eff}) beyond safe limits as described in 10 CFR 50.68.					
9.1.2.6	In meeting the requirements of 10 CFR 20.1101(b), positions C.2.f(2) and C.2.f(6) of RG 8.8 are the bases for acceptance with respect to provisions for decontamination. For spent fuel storage, paragraph 5.1.5 of ANS 57.2 and appropriate positions of RG 1.13 are the bases for acceptance. For new fuel storage, paragraphs 6.3.3.7 and 6.3.4 of ANS 57.3 are the bases for acceptance.					
9.1.2.7	10 CFR 50.68 allows the applicant to follow the guidelines of 10 CFR 70.24 for criticality monitors or the guidelines described therein for significant margins of subcriticality.					
9.1.3, Rev. 2 (03/2007)	Spent Fuel Pool Cooling and Cleanup System					
9.1.3.1	Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations are included in the Requirements subsection, above. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.					
9.1.4, Rev. 3 (03/2007)	Light Load Handling System (Related to Refueling)					
9.1.4.1	Acceptance for meeting the relevant aspects of GDC 2 is based on RG 1.29, Positions C.1 and C.2.					
9.1.4.2	Acceptance for meeting the relevant aspects of GDC 5 is embodied within the other acceptance criteria					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.1.4.3	Acceptance for meeting the relevant aspects of GDC 61 is based in part on the guidelines of American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.1-1992.					
9.1.4.4	Acceptance for meeting the relevant aspects of GDC 62 is based in part on ANSI/ANS 57.1-1992.					
9.1.5, Rev. 1 (03/2007)	Overhead Heavy Load Handling Systems					
9.1.5.1	Acceptance for meeting the relevant aspects of GDC 1 is based in part on NUREG-0554 for overhead handling systems and ANSI N14.6 or ASME B30.9 for lifting devices.					
9.1.5.2	Acceptance for meeting the relevant aspects of GDC 2 is based in part on position C.2 of RG 1.29 and Section 2.5 of NUREG-0554.					
9.1.5.3	Acceptance for meeting the relevant aspects of GDC 4 is based in part on position C.5 of RG 1.13.					
9.1.5.4	Acceptance for meeting the relevant aspects of GDC 5 is embodied within the other acceptance criteria.					
9.2.1, Rev. 5 (03/2007)	Station Service Water System					
9.2.1.1	Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the SWS and the SWS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the SWS and Position C.2 for nonsafety-related portions of the SWS are appropriately addressed.					
9.2.1.2	Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the SWS, are met: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1. In addition, the information will be considered acceptable if the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	design provisions presented in GL 96-06 and to GL 96-06, Supplement 1 are appropriately addressed.					
9.2.1.3	Sharing of Structures, Systems, and Components. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the SWS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s). In addition, the information will be considered acceptable if the provisions GL 89-13 and GL 91-13 are appropriately addressed.					
9.2.1.4	Cooling Water System. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if a system to transfer heat from SSCs important to safety to an ultimate heat sink is provided. In addition, the SWS can transfer the combined heat load of these SSCs under normal operating and accident conditions, assuming loss of offsite power and a single failure, and that system portions can be isolated so the safety function of the system is not compromised.					
9.2.1.5	Cooling Water System Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the design of the SWS permits inservice inspection of safety-related components and equipment and operational functional testing of the system and its components.					
9.2.1.6	Cooling Water System Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if the SWS is designed for testing to detect degradation in performance or in the system pressure boundary so that the SWS will function reliably to provide decay heat removal and essential cooling for safety-					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	related equipment.					
9.2.2, Rev. 4 (03/2007)	Reactor Auxiliary Cooling Water Systems					
9.2.2.1	Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the reactor auxiliary CWS and the reactor auxiliary CWS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the reactor auxiliary CWS and Position C.2 for nonsafety-related portions of the reactor auxiliary CWS are appropriately addressed.					
9.2.2.2	Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the reactor auxiliary CWS, are met: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1. In addition, the information will be considered acceptable if the design provisions presented in GL 96-06 and GL 96-06, Supplement 1 are appropriately addressed.					
9.2.2.3	Sharing of Structures, Systems, and Components. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the reactor auxiliary CWS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s).					
9.2.2.4	Cooling Water System. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if the reactor auxiliary					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	CWS and its components will continue to perform their required safety functions, assuming a single, active failure or a moderate-energy line crack as defined in Branch Technical Position ASB 3-1 and to seismic Category I, Quality Group C, and American Society of Mechanical Engineers (ASME) Section III Class 3 requirements concurrent with the loss of offsite power. In addition, the information will be considered acceptable based on appropriate application of IEEE Std 603, as endorsed by RG 1.153, and appropriate application of RG 1.155, Position C.3.3.4.					
9.2.2.5	Cooling Water System Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the periodic inspection of important reactor auxiliary CWS components ensures system integrity and capability to perform design safety functions.					
9.2.2.6	Cooling Water System Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if periodic system pressure and function testing of the reactor auxiliary CWS will ensure the leak-tight integrity and operability of its components, as well as the operability of the system as a whole, at conditions as close to the design basis as practical.					
9.2.3 - Withdrawn	Demineralized Water Makeup System (see ML063320108)	NA				Exclude
9.2.4, Rev. 3 (03/2007)	Potable and Sanitary Water Systems					
9.2.4.1	Control of Releases of Radioactive Materials to the PWSW. Information that addresses the requirements of GDC 60 in regards to controlling radioactive effluent releases is considered acceptable if the following are met: A. There are no interconnections between the PSWS and systems having the potential for containing radioactive material.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	B. The potable water system is protected by an air gap, where necessary. C. An evaluation of potential radiological contamination, including accidental, and safety implications of sharing (for multi-unit facilities) indicates that the system will not result in contamination beyond acceptable limits.					
9.2.5, Rev. 3 (03/2007)	Ultimate Heat Sink					
9.2.5.1	Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the UHS and the UHS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.27, Positions C.2 and C.3 are appropriately addressed.					
9.2.5.2	Sharing of Structures, Systems, and Components. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the UHS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the other unaffected unit(s).					
9.2.5.3	Cooling Water System. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if the guidance of RG 1.27, Positions C.2 and C.3; RG 1.72, Positions C.1, C.4, C.5, C.6, and C.7.; and American National Standards Institute/American Nuclear Society (ANSI/ANS) 5.1 are applied appropriately.					
9.2.5.4	4. Cooling Water System Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the design of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	UHS permits inservice inspection of safety-related components and equipment.					
9.2.5.5	Cooling Water System Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if the UHS is designed for testing of safety-related systems or components for structural integrity and leak-tightness, operability, performance of active components, and the capability of the system to function as intended under accident conditions.					
9.2.6, Rev. 3 (03/2007)	Condensate Storage Facilities					
9.2.6.1	Protection Against Natural Phenomena. Acceptance for meeting the relevant aspects of GDC 2 is based in part on meeting the guidance of Position C.1 of Regulatory Guide 1.29 if any portion of the system is deemed to be safety related and the guidance of Position C.2 for nonsafety-related portions. Also, acceptance is based in part on (1) meeting the guidance of Regulatory Guide 1.117 with respect to identifying portions of the system that should be protected from tornadoes and (2) meeting the guidance of Regulatory Guide 1.102 with respect to identifying portions of the system that should be protected from flooding.					
9.2.6.2	Sharing of Structures, Systems, and Components. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the CSF in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s).					
9.2.6.3	Condensate Storage Facility. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if a system to transfer heat from SSCs important to safety to an ultimate heat sink is provided. In addition, the CSF can transfer the combined					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	heat load of these SSCs under normal operating and accident conditions, assuming loss of offsite power and a single failure, and that system portions can be isolated so the safety function of the system is not compromised.					
9.2.6.4	Condensate Storage Facility Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the design of the CSF permits inservice inspection of safety-related components and equipment and operational functional testing of the system and its components.					
9.2.6.5	Condensate Storage Facility Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if the CSF is designed for testing to detect degradation in performance or in the system pressure boundary so that the CSF will function reliably to provide decay heat removal and essential cooling for safety-related equipment.					
9.2.6.6	Control of Radioactive Releases to the Environment. Acceptance for meeting the relevant aspects of GDC 60 is based on meeting the guidance of Regulatory Guide 1.143.					
9.2.6.7	Loss of All Alternating Current Power. Acceptance for meeting the relevant aspects of 10 CFR 50.63 is based on meeting the guidance of Regulatory Guide 1.155.					
9.3.1, Rev. 2 (03/2007)	Compressed Air System					
9.3.1.1	Acceptance for meeting the relevant aspect of GDC 1 is based on compliance with the criteria specified in American National Standards Institute/Instrument Society of America (ANSI/ISA) S7.3-R1981 related to minimum instrument air quality standards.					
9.3.1.2	Acceptance for meeting the relevant requirements of GDC 2 as it relates to seismic classification is based on compliance to guidance provided in RG 1.29, Positions C.1 and C.2.					
9.3.1.3	Acceptance for meeting the relevant requirements of GDC 5 as it relates to the sharing of safety-related SSCs is based on the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	criteria set forth here for CAS SSCs shared among multiple units.					
9.3.1.4	Acceptance for meeting the relevant requirements of 10 CFR 50.63 as it relates to the CAS design and the ability of a plant to withstand for a specified duration and recover from a station blackout is based on RG 1.155.					
9.3.2, Rev. 3 (03/2007)	Process and Post-Accident Sampling Systems					
9.3.2.1	The applicant's design is such that the PSS has the capability to sample all normal process systems and principal components, including provisions for obtaining samples from at least the points indicated below. The guidelines of Regulatory Guide (RG) 1.21, Position C.2, the Electric Power Research Institute (EPRI) BWR Water Chemistry Guidelines, and the Electric Power Research Institute (EPRI) PWR Water Chemistry Guidelines are used to meet the requirements of the relevant GDC.					

For a Pressurized Water Reactor (PWR)	GDC
Reactor coolant (e.g., letdown system)	13, 14, 26
Refueling (borated) water storage tank	13, 26
ECCS core flooding tank	13
Boric acid mix tank	13, 26
Boron injection tank	13
Chemical additive tank	13, 14, 41
Spent fuel pool	63, 60
Secondary coolant (e.g., condensate hotwell)	13, 14
Pressurizer tank	64, 60
Steam generator blowdown (if applicable)	14, 64, 60
Secondary coolant condensate treatment waste	64, 60
Sumps inside containment	64, 60
Containment atmosphere	64, 60
Gaseous radwaste storage tanks	63, 64, 60
For a Boiling Water Reactor (BWR)	GDC
Main condenser evacuation system offgas, and	64, 60

<table>
<tr><th colspan="7">Table A1-15: NUREG-0800, Standard Review Plan</th></tr>
<tr><th>ID</th><th>Title/Requirement</th><th>Applicable</th><th>Reg or Guidance</th><th>Additional Design Info</th><th>Additional Reg Needed</th><th>Basis/Comment</th></tr>
<tr><td></td><td>
<table>
<tr><td>charcoal delay or decay beds</td><td></td></tr>
<tr><td>Reactor coolant (inlet and outlet of reactor water cleanup system)</td><td>13, 14, 64, 60</td></tr>
<tr><td>Standby liquid control system tank</td><td>13, 26</td></tr>
<tr><td>Sumps inside containment</td><td>64, 60</td></tr>
<tr><td>Spent fuel pool</td><td>63, 60</td></tr>
<tr><td>Drywell atmosphere (Mark I & II)</td><td>64, 60</td></tr>
<tr><td>Inlet and outlet of gaseous radwaste storage tank</td><td>63, 64, 60</td></tr>
<tr><td>Inlet and outlet of condensate polishing system</td><td>13, 14</td></tr>
</table>
SRP Section 11.5 gives other sample points that may be included in the PSS but do not require remote sampling.</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>9.3.2.2</td><td>The plant Technical Specifications include the required analysis and frequencies.</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>9.3.2.3</td><td>The following guidelines should be used to determine the acceptability of the PSS functional design:

A. Provisions should be made to ensure representative samples from liquid process streams and tanks. For tanks, provisions should be made to sample the bulk volume of the tank and to avoid sampling from low points or from potential sediment traps. For process stream samples, sample points should be located in turbulent flow zones. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet these criteria.

B. Provisions should be made to ensure representative samples from gaseous process streams and tanks in accordance with American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.1-1999. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet this criterion.</td><td></td><td></td><td></td><td></td><td></td></tr>
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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	C. Provisions should be made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing). The guidelines of Regulatory Position C.7 in RG 1.21 are followed to meet this criterion. D. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures at ALARA levels. The guidelines of Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8 are followed to meet this criterion. E. Isolation valves should fail in the closed position, in accordance with the requirements of GDC 60 to control the release of radioactive materials to the environment. F. Passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures to ALARA levels and the requirements of GDC 60 to control the release of radioactive materials to the environment. The guidelines of Regulatory Position 2.i.(6) in RG 8.8 should be followed to meet this criterion. Redundant environmentally qualified, remotely operated isolation valves may replace passive flow restrictions in the sample lines to limit potential leakage. The automatic containment isolation valves should close on containment isolation signals or safety injection signals.					
9.3.2.4	To meet the requirements of GDCs 1 and 2, the applicant's seismic design and quality group classification of sampling lines, components, and instruments for the PSS should conform to the classification of the system to which each sampling line and component is connected (e.g., a sampling line connected to a					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Quality Group A and seismic Category I system should be designed to Quality Group A and seismic Category I classification), in accordance with Regulatory Positions C.1, C.2, and C.3 in RG 1.26; Regulatory Positions C.1, C.2, C.3, and C.4 in RG 1.29, and the guidelines of RG 1.97. Components and piping downstream of the second isolation valve may be designed to Quality Group D and nonseismic Category I requirements, in accordance with Regulatory Position C.3 in RG 1.26.					
9.3.3, Rev. 3 (03/2007)	Equipment and Floor Drainage System					
9.3.3.1	Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of safety-related system portions of the EFDS to withstand the effects of natural phenomena. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section. If no portion is safety-related, the EFDS need not meet GDC 2.					
9.3.3.2	Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding the capability to withstand the effects of and to be compatible with the environmental conditions (flooding) of normal operation, maintenance, testing, and postulated accidents (pipe break, tank ruptures) will be considered acceptable if the EFDS is designed to prevent flooding that could affect SSCs important to safety (i.e., necessary for safe shutdown, accident prevention, or accident mitigation) adversely.					
9.3.3.3	Control of Releases of Radioactive Material to the Environment. Information that addresses the requirements of GDC 60 regarding the suitable control of the release of radioactive materials in liquid effluent, including anticipated operational occurrences will be considered acceptable if the EFDS is designed to prevent the inadvertent transfer of contaminated fluids to a noncontaminated drainage system for disposal.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.3.4, Rev. 3 (03/2007)	Chemical and Volume Control System (PWR) (Including Boron Recovery System)					
9.3.4.1	The CVCS safety-related functional performance should be maintained in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. For compliance with GDC 29, 33 and 35, the CVCS should provide sufficient pumping capacity to supply borated water to the RCS, maintain RCS water inventory within the allowable pressurizer level range for all normal modes of operation, and function as part of the ECCS, if so designed, to supply reactor coolant makeup in the event of small pipe breaks assuming a single active failure coincident with the loss of offsite power. Also, Regulatory Guide 1.155 describes a means acceptable to the NRC staff for meeting the requirements of 10 CFR 50.63, “Loss of all alternating current power.” If the CVCS is necessary to support a plant SBO coping capability as required by 10 CFR 50.63, the positions in Regulatory Guide 1.155 regarding CVCS design provide an acceptable method for showing compliance.					
9.3.4.2	SECY-77-439 describes the concept of single failure criteria and the application of the single failure criterion that involves a systematic search for potential single failure points and their effects on prescribed missions. Application of the single failure assumption in system design and analysis provides redundancy and defense-in-depth to ensure functional performance of the CVCS. Also, the requirements of GDC 5 prohibiting the sharing among nuclear units the SSCs important to safety would be met by the use of a separate CVCS for each unit.					
9.3.4.3	10 CFR 50.55(a) requires that components of the RCPB be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III of the ASME					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Boiler and Pressure Vessel Code or equivalent quality standards. Regulatory Guide 1.26 describes a quality classification system that may be used to determine quality standards acceptable to the NRC staff for satisfying GDC 1 for other safety related components containing water, steam, or radioactive materials in light-water-cooled nuclear power plants. RG 1.29 describes a method acceptable to the NRC staff for identifying and classifying those features of LWRs that should be designed to withstand the effects of the safe shutdown earthquake (SSE). The requirements of GDC 1 regarding the quality standard are met by acceptable application of quality group classifications and application of quality standards as described in RG 1.26. The requirement of GDC 2 regarding the protection against natural phenomena are met by meeting the guidance of RG 1.29, Position C.1, for safety-related portions of the system and Position C.2 for nonsafety-related portion.					
9.3.4.4	The CVCS design and arrangement should be that all components and piping that can contain boric acid will either be heat traced or will be located within heated rooms to prevent precipitation of boric acid. As additional specific criteria used to review the CVCS and BRS design, the CVCS should include provisions for monitoring: (a) temperature upstream of the demineralizer to assure that resin temperature limits are not exceeded, and (b) filter demineralizer differential pressure to assure that pressure differential limits are not exceeded. In addition, the CVCS should have provision for automatically diverting or isolating the CVCS flow to the demineralizer in the event the demineralizer influent temperature exceeds the resin temperature limit.					
9.3.4.5	10 CFR 50.34(f)(2)(xxvi), as applicable, specifies the provisions regarding detection of reactor coolant leakage outside containment. These requirements will be met, in part, by					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	providing leakage control and detection systems in the CVCS and implementation of appropriate leakage control program.					
9.3.4.6	Implementation of Action 1 specified in Bulletin 80-05 provides an acceptable means for the system to prevent the CVCS holdup tanks, which can contain radioactive release, from the formation of such vacuum conditions that could cause wall inward buckling and failure. The requirements of GDC 60 and 61 can be met, in part, by providing in the CVCS appropriately designed venting and draining closed systems to confine the radioactivity associated with the effluents.					
9.3.4.7	10 CFR 52.47(a)(1)(vi) specifies that the application of a design certification should contain proposed ITAAC necessary and sufficient to assure the plant is built and will operate in accordance with the design certification. 10 CFR 52.97(b)(1) specifies that the COL identifies the ITAAC necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. SRP 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(a)(1)(vi) and 10 CFR 52.97(b)(1) will be met, in part, by identifying inspections, tests, analyses, and acceptance criteria of the top-level design features of the CVCS in the design certification application and the combined license, respectively.					
9.3.5, Rev. 3 (03/2007)	Standby Liquid Control System (BWR)					
9.3.5.1	Acceptance for meeting the relevant aspects of GDC 2 is based on meeting the guidance of Regulatory Guide 1.29, Position C-1.					
9.3.5.2	Acceptance for meeting the relevant aspects of GDC 4 is embodied within SRP Section 3.9.2.					
9.3.5.3	Acceptance for meeting the relevant aspects of GDC 5 is based on not sharing the SSCs important to safety between the units (except as identified).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.3.5.4	Acceptance for meeting the relevant aspects of GDC 26 is based on the provision of two independent reactivity control systems of different design principles (control rod drive system and SLCS system).					
9.3.5.5	Acceptance for meeting the relevant aspect of GDC 27 is based on the system having suitable redundancy in components and features to assure system safety function assuming a single failure. For some newer designs such as the ESBWR, GDC 27 is met by the provision of SLCS as part of the ECCS.					
9.4.1, Rev. 3 (03/2007)	Control Room Area Ventilation System					
9.4.1.1	Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the CRAVS and the CRAVS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the CRAVS and Position C.2 for nonsafety-related portions of the CRAVS are appropriately addressed.					
9.4.1.2	Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the CRAVS, are met: SRP Sections 3.5.1.1, 3.5.2, and 3.6.1.					
9.4.1.3	Sharing of Structures, Systems, and Components. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the CRAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.4.1.4	Control Room. Information that addresses the requirements of GDC 19 regarding the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain the plant in a safe condition under accident conditions, including loss-of-coolant accidents will be considered acceptable if adequate protection against radiation and hazardous chemical releases are provided to permit access to and occupancy of the control room under accident conditions. RG 1.78 provide guidance acceptable to the staff for meeting these control room occupancy protection requirements.					
9.4.1.5	Control of Releases of Radioactive Material to the Environment. Information that addresses the requirements of GDC 60 regarding the suitable control of the release of gaseous radioactive effluents to the environment will be considered acceptable if the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants are appropriately addressed. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.					
9.4.1.6	Loss of All Alternating Current Power. Information that addresses the requirements of 10 CFR 50.63 regarding the necessary support systems providing sufficient capacity and capability for coping with a station blackout event will be considered acceptable if the guidance of RG1.155, including position C.3.2.4 is applied appropriately.					
9.4.2, Rev. 3 (03/2007)	Spent Fuel Pool Area Ventilation System					
9.4.2.1	For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions and Position C.2 for					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	nonsafety-related portions.					
9.4.2.2	For GDC 5, acceptance is based on the determination that the use of the SFPAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).					
9.4.2.3	For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.					
9.4.2.4	For GDC 61, acceptance is based on the guidance of RG 1.13 as to the design of the ventilation system for the spent fuel storage facility, Position C.4.					
9.4.3, Rev. 3 (03/2007)	Auxiliary and Radwaste Area Ventilation System					
9.4.3.1	For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions, and Position C.2 for nonsafety-related portions.					
9.4.3.2	For GDC 5, acceptance is based on the determination that the use of the ARAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).					
9.4.3.4	For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.					
9.4.4, Rev. 3 (03/2007)	Turbine Area Ventilation System					
9.4.4.1	For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions.					
9.4.4.2	For GDC 5, acceptance is based on the determination that the use of the TAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s)					
9.4.4.3	For GDC 60, acceptance is based on guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 Revision 2, the applicable regulatory position is C.2. For RG 1.52 Revision 3, the applicable regulatory position is C.3. For RG 1.140 Revision 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 Revision 2, the applicable regulatory positions are C.2 and C.3.					
9.4.5, Rev. 3 (03/2007)	Engineered Safety Feature Ventilation System					
9.4.5.1	For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.					
9.4.5.2	For GDC 4, acceptance is based on meeting the acceptance criteria in the following SRP sections, as they apply to the ESFVS: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.4.5.3	For GDC 5, acceptance is based on the determination that the use of the ESFVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).					
9.4.5.4	For GDC 17, acceptance is based on the guidance of item 2 under Subsection A and item 1 under Subsection C of the NUREG-CR/0660 section "Recommendations" for protection of essential electrical components from failure due to the accumulation of dust and particulate materials.					
9.4.5.5	For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.					
9.4.5.6	For 10 CFR 50.63, acceptance is based on the applicable guidance of RG 1.155, including Position C.3.2.4.					
9.5.1, Rev. 5 (03/2007)	Fire Protection Program					
9.5.1.1.1	RG 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment In Risk- Informed Decisions on Plant-Specific Changes to the Licensing Basis,” as it applies to the use of PRA in support of changes to the fire protection licensing basis for nuclear power plants. Appropriate techniques for performing a Fire PRA are presented in NUREG/CR-6850 (EPRI TR-1011989), “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities.”					
9.5.1.1.2	RG 1.188, Revision 1, “Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses,”					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	as it applies to FPP considerations for license renewal such as equipment aging issues. This RG endorses the guidance in Nuclear Energy Institute (NEI) document, NEI 95-10, Revision 0, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule."					
9.5.1.1.3	RG 1.189, Revision 1, "Fire Protection for Nuclear Power Plants," which provides comprehensive staff positions and guidelines on fire protection for nuclear power plants.					
9.5.1.1.4	RG 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown," which establishes the fire protection objectives and staff positions for implementing fire protection for those nuclear power plants that have submitted the necessary certifications for license termination under 10 CFR Part 50.82(a).					
9.5.1.1.5	RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it applies to the FPP of any new reactor COL application submitted in accordance with 10 CFR Part 52.					
9.5.1.1.6	Enhanced fire protection criteria for new reactor designs as documented in SECY 90-016, SECY 93-087, and SECY 94-084. SECY 90-016 established enhanced fire protection criteria for evolutionary light-water reactors (LWRs). SECY 93-087 recommended that the enhanced criteria be extended to include passive reactor designs. SECY 90 016 and SECY 93-087 were approved by the Commission in staff requirements memoranda. SECY 94-084, in part, establishes criteria defining safe shutdown conditions for passive LWR designs.					
9.5.1.1.7	For COL reviews, the description of the operational program and proposed implementation milestone(s) for the FPP are reviewed in accordance with 10 CFR 50.48. The operational program for fire protection should be fully implemented prior to fuel receipt at the plant site.					
9.5.1.2, Rev. 0 (12/2009)	Risk-Informed (RI), Performance-Based (PB) Fire Protection Program (FPP)					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
9.5.1.2.1	NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions, Interim Enforcement Policy," May 1, 2000, which provides the Commission's policy on enforcement discretion for non-compliant conditions, either existing or identified during transition to a RI/PB FPP in accordance with 10 CFR 50.48(c).					
9.5.1.2.2	RG 1.205, Revision 1, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," which provides NRC guidance on an acceptable approach to meeting 10 CFR 50.48(c), including endorsement (with exceptions) of NEI 04-02, Revision 2, "Guidance for Implementing a Risk-Informed Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," and portions of NEI 00-01, Revision 1, "Guidance for Post-Fire Safe Shutdown Circuit Analysis." ISG that will be considered in future revisions of RG 1.205 is documented in approved NFPA 805 FAQs.					
9.5.1.2.3	RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," which provides NRC guidance on an acceptable method to assess the nature and impact of licensing basis changes using risk information within the context of applicability under 10 CFR 50.48(c) and RG 1.205.					
9.5.1.2.4	RG 1.189, Revision 2, "Fire Protection for Nuclear Power Plants," which provides general guidance on acceptable FPPs.					
9.5.1.2.5	Section 19.1 of the SRP, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," which provides review guidance on determining the technical adequacy of probabilistic risk assessment (PRA) models for RI initiatives.					
9.5.1.2.6	Section 19.2 of the SRP, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," which provides guidance on reviewing risk information used to support plant-specific changes to the licensing basis.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.5.1.2.7	RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities," March 2009, which provides guidance with respect to acceptable methods and PRA quality.					
9.5.1.2.8	NUREG/CR-6850, EPRI 1011989, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," Volumes 1 and 2, issued September 2005, which provides a method for developing a fire PRA in support of adopting a RI/PB FPP, within the context of the additional clarifications provided by the staff via the NFPA 805 FAQ process.					
9.5.1.2.9	NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," issued October 2007, which provides qualitative methods to demonstrate that Operator Manual Actions (OMAs) are feasible and reliable.					
9.5.1.2.10	NUREG-1824, EPRI 1011999, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," Volumes 1–7, issued May 2007, which provides guidance on verification and validation (V&V) of fire models.					
9.5.1.2.11	NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection", issued October 28, 2004, which reflects NRC's organizational responsibilities and authorities with respect to the management of facility-specific backfits.					
9.5.2, Rev. 3 (03/2007)	Communications Systems					
9.5.2.1	Information regarding the requirements of Appendix E to 10 CFR Part 50, Part IV.E(9), will be found acceptable if adequate provisions are made and described for emergency facilities and equipment, including: at least one onsite and one offsite communications system; each system shall have a backup power source.					
9.5.2.2	For those applicants subject to either 10 CFR 50.34(f) or the TMI Action Plan, information regarding the requirements of 10 CFR 50.34(f)(2)(xxv) and TMI Action Plan Item III A.1.2 will be found					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	acceptable if provisions are made for an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility.					
9.5.2.3	Information regarding the requirements of 10 CFR 50.47(a)(8) will be found acceptable if adequate emergency facilities and equipment to support the response are provided and maintained					
9.5.2.4	Information regarding the requirements of 10 CFR 50.55a will be found acceptable if SSCs are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.					
9.5.2.5	Information regarding the requirements of GDC 1 will be found acceptable if SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.					
9.5.2.6	Information regarding the requirements of GDC 2 will be found acceptable if SSCs important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) appropriate consideration of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.					
9.5.2.7	Information regarding the requirements of GDC 3 will be found acceptable if SSCs important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on SSCs important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.					
9.5.2.8	Information regarding the requirements of GDC 4 will be found acceptable if SSCs important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.					
9.5.2.9	Information regarding the requirements of GDC 19 will be found acceptable if equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls (I&C) to maintain the unit in a safe					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.					
9.5.2.10	Information regarding the requirements of 10 CFR 73.45(e)(2)(iii) will be found acceptable if communications subsystems and procedures are provided for notification of an attempted unauthorized or unconfirmed removal of strategic special nuclear material so that response can be such as to prevent the removal and satisfy the general performance objective and requirements of § 73.20(a).					
9.5.2.11	Information regarding the requirements of 10 CFR 73.45(g)(4)(i) will be found acceptable if communications networks are provided to transmit rapid and accurate security information among onsite forces for routine security operation, assessment of a contingency, and response to a contingency.					
9.5.2.12	Information regarding the requirements of 10 CFR 73.46(f) will be found acceptable if each guard, watchman, or armed response individual on duty shall be capable of maintaining continuous communication with an individual in each continuously manned alarm station required by 10CFR 73.46(e)(5), who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from law enforcement authorities; each alarm station required by 10 CFR 73.46(e)(5) shall have both conventional telephone service and radio or microwave transmitted two-way voice communication, either directly or through an intermediary, for the capability of communication with the law enforcement authorities; and non-portable communications equipment controlled by the licensee and required by 10CFR 73.46(f) shall remain operable from independent power sources in the event of the loss of normal power.					
9.5.2.13	Information regarding the requirements of 10 CFR 73.55(e) will be found acceptable if all alarms required by 10 CFR 73.55 annunciate in a continuously manned central alarm station					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	located within the protected area and in at least one other continuously manned station not necessarily onsite, so that a single act cannot remove the capability of calling for assistance or otherwise responding to an alarm. The onsite central alarm station must be considered a vital area and its walls, doors, ceiling, floor, and any windows in the walls and in the doors must be bullet-resisting. The onsite central alarm station must be located within a building in such a manner that the interior of the central alarm station is not visible from the perimeter of the protected area. This station must not contain any operational activities that would interfere with the execution of the alarm response function. Onsite secondary power supply systems for alarm annunciator equipment and non-portable communications equipment as required 10 CFR 73.55(f) of this section must be located within vital areas. All alarm devices including transmission lines to annunciators shall be tamper indicating and self-checking, e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when the system is on standby power. The annunciation of an alarm at the alarm stations shall indicate the type of alarm (e.g., intrusion alarms, emergency exit alarm, etc.) and location. All emergency exits in each protected area and each vital area shall be alarmed.					
9.5.2.14	Information regarding the requirements of 10 CFR 73.55(f) will be found acceptable if each guard, watchman or armed response individual on duty is capable of maintaining continuous communication with an individual in each continuously manned alarm station required by 10 CFR 73.55(e)(1), who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from local law enforcement authorities. The alarm stations required by 10 CFR 73.55(e)(1) shall have conventional telephone service for communication with the law enforcement authorities as described in 10 CFR 73.55(f)(1). To provide the capability of continuous communication, radio or microwave transmitted two-way voice					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	communication, either directly or through an intermediary, shall be established, in addition to conventional telephone service, between local law enforcement authorities and the facility and shall terminate in each continuously manned alarm station required by 10 CFR 73.55(e)(1). Non-portable communications equipment controlled by the licensee and required by 10 CFR 73.55 shall remain operable from independent power sources in the event of the loss of normal power.					
9.5.3, Rev. 3 (032007)	Lighting Systems					
9.5.3.1	Acceptance criteria of the design of the normal and emergency lighting systems, as described in the applicant's safety analysis report (SAR), is based in part on the degree of similarity of the systems design with those for previously reviewed plants with satisfactory operating experience.					
9.5.3.2	The normal lighting system(s) is acceptable if the integrated design of the system(s) will provide adequate station lighting in all areas, from power sources described in Section 8.2 of the SRP that are required for control and maintenance of equipment and plant access routes during normal plant operations.					
9.5.3.3	The emergency lighting system(s) is acceptable if the integrated design of the system(s) will provide adequate emergency station lighting in all areas, required for fire fighting, control and maintenance of equipment used for implementing safe shutdown of the plant during all plant operating conditions, and the access routes to and from these areas.					
9.5.3.4	The lighting systems designs will be acceptable if they conform to the lighting levels recommended in NUREG-0700, which is based on the Illuminating Engineering Society of North America (IESNA) Lighting Handbook (Reference 2) as related to systems design and illumination levels recommended for industrial facilities.					
	REFERENCE: 2. NUREG-1793, "Final Safety Evaluation Report Related to					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Certification of the AP1000 Standard Design.”					
9.5.4, Rev. 3 (03/2007)	Emergency Diesel Engine Fuel Oil Storage and Transfer System					
9.5.4.1	GDC 2 requirements for which SSCs must be protected from, or be capable of withstanding, the effects of such natural phenomena as earthquakes, tornadoes, hurricanes, and floods apply to safety-related EDEFSS SSCs. The identification of SSC required to withstand earthquakes without the loss of capability to perform safety functions is listed in RG 1.29. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.					
9.5.4.2	GDC 4 requirements for which SSCs must be protected from, or be capable of withstanding the effects of externally-and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks apply to safety-related EDEFSS SSCs. Comprehensive compliance with GDC 4 is reviewed under other SRP sections as specified in subsection I of this SRP section.					
9.5.4.3	GDC 5 requirements for sharing of SSCs important to safety among nuclear power units are met if each unit has its own diesel generator(s) and each diesel generator has an independent fuel oil system.					
9.5.4.4	GDC 17 as to the capability of the fuel oil system to meet independence and redundancy criteria and the guidance and positions of the following: A. RG 1.137 as to the diesel engine fuel oil system design, fuel oil quality, and tests which are specified in regulatory positions C1 and C2. The regulatory position C1 addresses the design criteria for the fuel oil system such as materials, physical arrangement, and applicable codes and regulations. The physical arrangements 9.5.4-6 Revision 3 - March 2007 of the fuel oil system should provide for inservice inspection and testing in accordance with ASME					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Boiler and Pressure Vessel Code Section XI ,"Rules for Inservice Inspections." Criteria for oil quality are addressed in the position C2. The fuel oil stored in the fuel supply tank or used for filling or refilling the supply tank should meet the Federal Fuel Oil, ASTM, or diesel-generator manufacturer requirements. The quality of fuel oil is determined by performing suitable tests and when it does not meet the prescribed standards it is replaced. Also, prior to adding new fuel oil to the supply tank the test for specific gravity, water sediment and viscosity testing should be performed and the fuel oil not meeting the test requirements should not be added to the tank. B. NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability" C. Each diesel engine with its own EDEFSS. D. ANSI/ANS-59.51 regarding the onsite fuel oil storage for each diesel generator being sufficient to operate the diesel generator following any design basis event and a continuous loss of off-site power either for seven days, or for the time required to replenish the fuel from sources outside the plant site following any design event without interruption of the operation of the diesel generator, whichever is longer.					
9.5.5, Rev. 3 (03/2007)	Emergency Diesel Engine Cooling Water System					
9.5.5.1	GDC 2 requirements for which SSCs must be protected from, or be capable of withstanding, the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods apply to safety-related EDECWS SSCs. The identification of SSC required to withstand earthquakes without the loss of capability to perform safety functions is listed in RG 1.29. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.5.5.2	GDC 4 requirements for which SSCs must be protected from, or be capable of withstanding, the effects of externally-and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks apply to safety-related EDECWS SSCs. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.					
9.5.5.3	GDC 5 requirements for sharing of SSCs important to safety among nuclear power units are met if each unit has its own diesel generator(s) and each diesel generator has an independent and reliable cooling water system.					
9.5.5.4	GDC 17 requirements for the capability of the cooling water system to meet independence and redundancy criteria are met when: B. Each diesel generator has a separate and independent EDECWS. C. NRC recommendations specified in NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," are implemented.					
9.5.5.5	GDC 44 requirements are met when the EDECWS has: A. The capability to transfer heat from systems and components to a heat sink under transient or accident conditions. B. Redundancy of components for performance of safety functions under accident conditions, assuming a single active component failure, or each diesel generator has a separate and independent EDECWS. C. The capability to isolate system or piping components if required to maintain the system safety function.					
9.5.5.6	GDC 45 as to design provisions for periodic inspection of safety-related system components and equipment.					
9.5.5.7	GDC 46 as to design provisions for appropriate functional testing of safety-related systems or components for structural integrity					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	and leak-tightness, operability, performance of active components, and the capability of the system to function as intended under accident conditions.					
9.5.6, Rev. 3 (03/2007)	Emergency Diesel Engine Starting System					
9.5.6.1	GDC 2 requirements for SSCs to withstand or be protected from the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods apply to safety-related EDESS SSCs. The identification of SSC required to withstand earthquakes without loss of capability to perform safety functions is listed in RG 1.29. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.					
9.5.6.2	GDC 4 requirements for SSCs to be protected against the effects of externally-and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks apply to safety-related EDESS SSCs. Comprehensive compliance with GDC 4 is reviewed under other SRP sections as specified in subsection I of this SRP section.					
9.5.6.3	GDC 5 requirements for sharing of SSCs important to safety among nuclear power units are met if each unit has its own diesel generator(s) and each diesel generator an independent starting system.					
9.5.6.4	GDC 17 as to the capability of the diesel engine air starting system to meet independence and redundancy criteria. Specific criteria and guidance necessary to meet GDC 17 requirements are as follow: A. NUREG/CR-0660 "Enhancement of Onsite Emergency Diesel Generator Reliability." B. Each diesel engine should have a dedicated air starting system consisting of an air compressor, an air dryer, one or more air receiver(s), piping, injection lines and valves, and devices to crank the engine as recommended by the engine manufacturer.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	C. As a minimum, the air starting system should be capable of cranking a cold diesel engine five times without recharging the receiver(s). The air starting system capacity should be determined as follows: (i) each cranking cycle duration should be approximately three seconds, (ii) consist of two to three engine revolutions, or (iii) air start requirements per engine start provided by the engine manufacturer, whichever air start requirement is larger. D. Alarms should alert operating personnel if the air receiver pressure falls below the minimum allowable value. E. Provisions for the periodic or automatic blowdown of accumulated moisture and foreign material in the air receiver(s) and other system critical points. F. Starting air should be dried to a dew point of not more than 10EC (50EF) when installed in a normally-controlled 21EC (70EF) environment; otherwise, the starting air dew point should be controlled to at least 5.5EC (10EF) less than the lowest expected ambient temperature.					
9.5.7, Rev. 3 (03/2007)	Emergency Diesel Engine Lubrication System					
9.5.7.1	GDC 2 requirements for SSCs to withstand or be protected from the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods apply to safety-related EDELS SSCs. The identification of SSCs required to withstand earthquakes without the loss of capabilities to perform safety functions is listed in RG 1.29. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.					
9.5.7.2	GDC 4 requirements for SSCs to be protected against the effects of externally- and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks apply to safety-related EDELS SSCs. Comprehensive compliance with GDC 4 is reviewed under other SRP sections as specified in subsection I of this SRP section.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.5.7.3	GDC 5 requirements for sharing of SSCs important to safety among nuclear power units are met if each unit has its own diesel generator(s), each with an independent lubrication system.					
9.5.7.4	GDC 17 requirements of independence and redundancy criteria are applicable to the EDELS. Acceptance is based on the following specific criteria: A. NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." B. System operating pressure, temperature differentials, flow rate, and heat removal rate external to the engine in accordance with engine manufacturer recommendations. C. Sufficient system protective measures to maintain required oil quality during engine operation. D. Protective measures (e.g., relief ports) to prevent unacceptable crankcase explosions and to mitigate consequences of such events. E. A keep-warm oil lubricating system to maintain engine lubricating oil passages in a warmed and filled state when the diesel engine is in the standby mode. F. System design to circulate lubricating oil to the diesel engine during standby to enhance starting capability in conditions under which the engine-driven oil pump can pressurize the system quickly following engine starts. G. Each diesel engine lubricating oil system completely independent of other diesel engines so a single failure will not cause a loss of the required minimum diesel generator capacity as specified in ANSI/ANS-59.52. H. Onsite lubricating oil storage capacity for each diesel engine sufficient for seven days operation after any design basis event and a continuous loss of off-site power as specified in ANSI/ANS-59.52.					
9.5.8, Rev. 3 (03/2007)	Emergency Diesel Engine Combustion Air Intake and Exhaust System					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
9.5.8.1	GDC 2 requirements for SSCs to withstand or be protected from the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods, apply to safety-related EDECAIES SSCs. The identification of SSCs required to withstand earthquakes without the loss of capabilities to perform safety function is listed in RG 1.29. Compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.					
9.5.8.2	GDC 4 requirements of SSCs to be protected against the effects of externally- and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks, apply to safety-related EDECAIES SSCs. Compliance with GDC 4 is reviewed under other SRP sections as specified in subsection I of this SRP section.					
9.5.8.3	GDC 5 requirements for sharing of SSC important to safety are met when each diesel generator has its own independent and reliable combustion air intake and exhaust system.					
9.5.8.4	GDC 17 as related to the capabilities of the diesel engine combustion and air intake exhaust system to meet independence and redundancy criteria. Acceptance is based on meeting the following specific criteria: A. NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." i. Engine combustion air should be through piping directly from outside the building with the air intake sufficiently (20 feet) above ground level and filtered to preclude any degradation of continuous engine function. ii. The piping for room ventilation air should be separate from that for engine combustion air. iii. Engine exhaust gas should not circulate back into the diesel generator room, fuel storage room, or any part of the power plant.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	B. Each emergency diesel engine should have an independent and reliable combustion air intake and exhaust system sized and physically arranged for no degradation of engine function when the diesel generator set must operate continuously at the maximum rated power output. C. The combustion air intake system must have a means of reducing airborne particulate material over the entire time period requiring emergency power, assuming the maximum airborne particulate concentration at the combustion air intake. D. Suitable design precautions must preclude degradation of the diesel engine power output due to exhaust gases and other diluents that could reduce oxygen content below acceptable levels.					
	CHAPTER 10, Steam and Power Conversion Systems					
10.2, Rev. 3 (03/2007)	Turbine Generator					
10.2.1	Specific criteria necessary to meet the requirements of GDC 4 are as follows: A. A turbine control and overspeed protection system should control turbine action under all normal or abnormal operating conditions and should ensure that a fullload turbine trip will not cause the turbine to overspeed beyond acceptable limits. Under these conditions, the control and protection system should permit an orderly reactor shutdown by use of either the turbine bypass system and main steam relief system or other engineered safety systems. The overspeed protection system should meet the single failure criterion and should be testable when the turbine is in operation.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	B. The turbine main steam stop and control valves and the reheat steam stop and intercept valves should protect the turbine from exceeding set speeds and should protect the reactor system from abnormal surges. The reheat stop and intercept valves should be capable of closure concurrent with the main steam stop valves, or of sequential closure within an appropriate time limit, to ensure that turbine overspeed is controlled within acceptable limits. The valve arrangements and valve closure times should be structured so that a failure of any single valve to close will not result in excessive turbine overspeed in the event of a TGS trip signal. C. The TGS should have the capability to permit periodic testing of components important to safety while the unit is operating at rated load.					
10.2.2	An inservice inspection program for main steam and reheat valves should be established and should include the following provisions: A. At intervals of approximately 3-1/3 years, during refueling or maintenance shutdowns coinciding with the inservice inspection schedule required by Section XI of the American Society of Mechanical Engineers (ASME) Code for reactor components, at least one main steam stop valve, one main steam control valve, one reheat stop valve, and one reheat intercept valve should be dismantled, and visual and surface examinations should be conducted of valve seats, disks, and stems. If this process detects unacceptable flaws or excessive corrosion in a valve, all other valves of that type should be dismantled and inspected. Valve bushings should be inspected and cleaned, and bore diameters should be checked for proper clearance. B. Main steam stop and control valves should be exercised at a frequency recommended by the turbine vendor or valve					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	manufacturer.					
10.2.3	The arrangement of connection joints between the low-pressure turbine exhaust and the main condenser should prevent adverse effects on any safety-related equipment in the turbine room in the event of a rupture (it is preferable not to locate safety-related equipment in the turbine room).					
10.2.3, Rev. 2 (03/2007)	Turbine Rotor Integrity					
10.2.3.1	Materials Selection. The turbine forged or welded rotor should be made from a material and by a process that tends to minimize flaw occurrence and maximize fracture toughness properties, such as a NiCrMoV alloy processed by vacuum melting or vacuum degassing. The material should be examined and tested to meet the following criteria: A. Chemical analysis should be performed for each forging. Elements that have a deleterious effect on toughness, such as sulfur and phosphorus, should be controlled to low levels. B. The 50% fracture appearance transition temperature (FATT) as obtained from Charpy tests performed in accordance with specification ASTM A-370 should be no higher than -18°C (O°F) for low-pressure turbine rotors. The nil-ductility transition (NDT) temperature obtained in accordance with specification ASTM E-208 may be used in lieu of FATT. NDT temperatures should be no higher than -35°C (-30°F). C. The Charpy V-notch (C_v) energy at the minimum operating temperature of each low-pressure rotor in the tangential direction should be at least 8.3 kg-m (60 ft-lbs). A minimum of three C_v specimens should be tested in accordance with specification ASTM A-370.					
10.2.3.2	Fracture Toughness. The low-pressure turbine disk forged or welded rotor fracture toughness properties are acceptable if the following criteria are met.					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	The ratio of the fracture toughness (K_{Ic}) of the rotor material to the maximum tangential stress at speeds from normal to design overspeed should be at least 10 √mm (2 √in), at minimum operating temperature. Bore stress calculations should include components due to centrifugal loads, interference fit, and thermal gradients. Sufficient warmup time should be specified in the turbine operating instructions to ensure that toughness will be adequate to prevent brittle fracture during startup. Fracture toughness properties can be obtained by any of the following methods: A. Testing of the actual material of the turbine rotor to establish the K_{Ic} value at normal operating temperature. B. Testing of the actual material of the turbine rotor with an instrumented Charpy machine and a fatigue precracked specimen to establish the K_{Ic} (dynamic) value at normal operating temperature. If this method is used, K_{Ic} (dynamic) shall be used in lieu of K_{Ic} (static) in meeting the toughness criteria above. C. Estimating of K_{Ic} values at various temperatures from conventional Charpy and tensile data on the rotor material using methods are presented in J. A. Begley and W. A. Logsdon, Scientific Paper 71-1E7-AMSLRF-P1. This method of obtaining K_{Ic} should be used only on materials which exhibit a well-defined Charpy energy and fracture appearance transition curve and are strain-rate insensitive. The staff should review the test data and the calculated toughness curve submitted by the applicant. D. Estimating "lower bound" values of K_{Ic} at various temperatures using the equivalent energy concept developed by F. J. Witt and T. R. Mager, ORNL-TM- 3894. The staff should review the load-displacement data from the compact tension specimens and the calculated					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	toughness data submitted by the applicant.					
10.2.3.3	Pre-service Inspection. The applicant's pre-service inspection program is acceptable if it meets the following criteria: A. Forged or welded rotors should be rough machined prior to heat treatment. B. Each finished forged or welded rotor should be subjected to 100% volumetric (ultrasonic), surface, and visual examinations using procedures and acceptance criteria equivalent to those specified for Class 1 components in the ASME Boiler and Pressure Vessel Code, Sections III and V. Before welding and/or brazing, all surfaces prepared for welding and/or brazing should be surface examined. After welding and/or brazing, all surfaces exposed to steam should be surface examined, giving particular attention to stress risers and welds. Welds should be ultrasonically examined in the radial and radial-tangential sound beam directions. C. Finish machined bores, keyways, and drilled holes should be subjected to magnetic particle or liquid penetrant examination. No flaw indications in keyway or hole regions are allowed. D. Each turbine rotor assembly should be spin tested at 5% above the maximum speed anticipated during a turbine trip following loss of full load.					
10.2.3.4	Turbine Rotor Design. The turbine assembly should be designed to withstand normal conditions, anticipated transients, and accidents resulting in a turbine trip without loss of structural integrity. The design of the turbine assembly should meet the following criteria: A. The design overspeed of the turbine should be 5% above the highest anticipated speed resulting from a loss of load. The staff should review the basis for the assumed design					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	overspeed. B. The combined stresses of low-pressure turbine rotor at design overspeed due to centrifugal forces, interference fit, and thermal gradients should not exceed 0.75 of the minimum specified yield strength of the material, or 0.75 of the measured yield strength in the weak direction of the materials if appropriate tensile tests have been performed on the actual rotor material. C. The turbine shaft bearings should be able to withstand any combination of the normal operating loads, anticipated transients, and accidents resulting in a turbine trip. D. The natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20% overspeed should be controlled in the design and operation stages so as to cause no distress to the unit during operation. E. The turbine rotor design should facilitate inservice inspection of all high stress regions, including bores and keyways, without the need for removing the disks from the shaft.					
10.2.3.5	Inservice Inspection. The applicant's inservice inspection program is acceptable if it meets the following criteria: The inservice inspection program for the steam turbine assembly should provide assurance that rotor flaws that might lead to brittle failure of a rotor at speeds up to design speed will be detected. The inservice inspection and maintenance program for the turbine assembly should comply with the manufacturers recommendations. Inservice inspection and maintenance activities may be performed during plant shutdown coinciding with the inservice inspection schedule as required by ASME Boiler and Pressure Vessel Code, Section XI, and should include complete inspection of all					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	significant turbine components, such as couplings, coupling bolts, turbine shafts, low-pressure turbine blades, low-pressure rotors, and high-pressure rotors. This inspection should consist of visual, surface, and volumetric examinations, as required by the code.					
10.3, Rev. 4 (03/2007)	Main Steam Supply System					
10.3.1	Acceptance of GDC 2 is based on meeting the guidance of Regulatory Guide 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions.					
10.3.2	Acceptance of GDC 4 is based on the guidance of Regulatory Guide 1.115, Position C.1, as it relates to the protection of SSCs important to safety from the effects of turbine missiles. In addition, the system design should adequately consider water (steam) hammer and relief valve discharge loads to assure that system safety functions can be performed and should assure that operating and maintenance procedures include adequate precautions to prevent water (steam) hammer and relief valve discharge loads. The system design should also include protection against water entrainment.					
10.3.3	Compliance with GDC 5 requires that structures, systems, and components important to safety shall not be shared by nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their intended safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. Meeting the requirements of GDC 5 provides assurance that the main steam system and its associated components will continue performing their required safety functions even if they are shared by multiple nuclear power units.					
10.3.4	Acceptance of GDC 34 is based on the following: A. The positions in Branch Technical Position 5-4, as they relate to the design requirements for residual heat removal					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(RHR) B. Issue Number 1 of NUREG-0138, as it relates to credit being taken for all valves downstream of the main steam isolation valves (MSIVs) to limit blowdown of a second steam generator if a steamline were to break upstream of the MSIV					
10.3.5	Acceptance of 10 CFR 50.63 is based on meeting Regulatory Guide 1.155 as it relates to the MSSS design.					
10.3.6	Regulatory Guide 1.29, Positions C.1.a, C.1.e, C.1.f, C.2 and C.3, as it relates to the seismic design classification of system components.					
10.3.7	Regulatory Guide 1.117, Appendix Position 2 and 4, as it relates to the protection of SSCs important to safety from the effects of tornado missiles.					
10.3.8	SECY 93-087, as it applies to BWR plants that do not incorporate an MSIVLCS and for which main steamline fission product holdup and retention are credited in the analysis of design-basis accident radiological consequences as follows: A. Seismic Category I is the classification for the main steamlines extending from the outermost containment isolation valve to the seismic interface restraint and connected piping up to the first normally closed valve. B. The nonseismic Category I classification can apply to the main steamlines from the seismic interface restraint up to, but not including, the turbine stop valve (including connected piping to the first normally closed valve) if the following criteria are met: i. A dynamic seismic analysis method analyzed the lines to demonstrate their structural integrity under SSE loading conditions. ii. All pertinent quality assurance requirements of					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Appendix B to 10 CFR Part 50 are applied. iii. For lines used as an MSIV leakage path to the condenser, reliable power sources must be available for control and isolation valves so that a control operator can establish the flowpath, assuming a single active failure. C. Main steamlines and other main steam system components are assigned a quality group classification in accordance with the criteria of Branch Technical Position 3-1.					
10.3.6, Rev. 3 (03/2007)	Steam and Feedwater System Materials					
10.3.6.1	Materials Selection and Fabrication of Class 2 and 3 Components A. The materials specified for use in Class 2 and 3 components should conform to Appendix I to Section III of the Code and to Parts A, B, and C of Section II of the Code. B. Regulatory Guide 1.84, describes acceptable Code Cases that may be used in conjunction with the above specifications. Appendix IV to Section III of the Code provides requirements for approval of new materials. C. Regulatory Guide 1.71 provides the following guidelines for assuring the integrity of welds in locations of restricted direct physical and visual accessibility. i. The performance qualification should require testing of the welder under simulated conditions when conditions of accessibility to production welds are less than 30 to 35 cm (12 to 14 inches) in any direction from the joint. ii. Requalification should be required for significantly different restricted accessibility conditions or when any essential welding variables listed in Code Section IX are changed.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	D. Regulatory Guide 1.50 provides methods to control preheat temperatures for welding low alloy steel. For carbon steel and low alloy steel materials, Section III, Appendix D, Article D-1000 of the ASME Code specifies preheat temperatures. E. Regulatory Guide 1.37 and ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," describe acceptable procedures for cleaning and handling Class 2 and 3 components of the steam and feedwater systems. F. Acceptance criteria for nondestructive examination of tubular products are provided in the relevant paragraphs of Subsections NC and ND of Section III of the ASME Code.					
10.3.6.2	Fracture Toughness of Class 2 and 3 Components The fracture toughness properties of the ferritic materials of these components should meet the following requirements of the editions and addenda of Section III of the Code, as specified in 10 CFR 50.55a: A. NC-2300, "Fracture Toughness Requirements for Material" (Class 2) B. ND-2300, "Fracture Toughness Requirements for Material" (Class 3)					
10.4.1, Rev. 3 (03/2007)	Main Condensers					
10.4.1.1	The requirements of GDC 60 are met when the MC design includes provisions to prevent excessive releases of radioactivity to the environment which may result from a failure of a structure, system or component in the MC. Acceptance is based on meeting the following: A. SECY 93-087 gives guidance for new BWR plants that do					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	not incorporate an MSIVLCS and for which MC holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequence. It states that seismic analyses are to be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after an SSE. B. If there is a potential for explosive mixtures to exist, the MC is designed to withstand the effects of an explosion and instrumentation is provided to detect and annunciate the buildup of potentially explosive mixtures, dual instrumentation is provided to detect, annunciate, and effect control measures to prevent the buildup of potentially explosive mixtures, as outlined in SRP Section 11.3, subsection II, “Acceptance Criteria,” SRP Acceptance Criteria, Item 6.					
10.4.2, Rev. 3 (03/2007)	Main Condenser Evacuation System					
40.4.2.1	The requirements of General Design Criteria 60 (GDC 60) are met when the MCES design includes provisions to prevent excessive releases of radioactivity to the environment which may result from a failure of a structure, system or component in the MC. Acceptance is based on meeting the following: A. If there is a potential for explosive mixtures to exist, the MCES is designed to withstand the effects of an explosion and instrumentation is provided to detect and annunciate the buildup of potentially explosive mixtures, dual instrumentation is provided to detect, annunciate, and effect control measures to prevent the buildup of potentially explosive mixtures, as outlined in SRP Section 11.3, subsection II, "Acceptance Criteria," SRP Acceptance Criteria, Item 6. B. Such a potential does not exist on systems designed to					

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	maintain the steam content above 58% by volume in hydrogen-air mixtures or nitrogen content above 92% by volume in hydrogen-oxygen mixtures in all MCES components. The design pressure and normal operational absolute pressure should be provided for MCES components containing potentially explosive mixtures.					
10.4.3, Rev. 3 (03/2007)	Turbine Gland Sealing System					
	There is no specific acceptance criteria associated with this SRP section.					
10.4.4, Rev. 3 (03/2007)	Turbine Bypass System					
10.4.4.1	Piping Failures. The requirements of GDC 4 related to the ability of structures, systems and components important to safety to meet environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions is met by demonstrating that failure of the TBS due to a pipe break or malfunction of the TBS will not adversely affect essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation).					
10.4.4.2	Residual Heat Removal. The requirements of GDC 34 related to providing a reliable system that removes residual heat during normal plant shutdown is met by demonstrating the ability to use the turbine bypass system for shutting down the plant during normal operations. The operation of the TBS eliminates the need to rely solely on safety systems, which are required to meet the redundancy and power source requirements of this criterion.					
10.4.4.3	MSIV Alternate Leakage Path (ALP). For BWR plants that do not incorporate an MSIVLCS and for which TBS holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequences, guidance from SECY 93-087 is applicable. Specifically, the turbine bypass lines from the first valve up to the condenser inlet do not need to be classified as seismic category I if the following criteria are met:					

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	A. They have been analyzed using a dynamic seismic analysis method to demonstrate their structural integrity under SSE loading conditions. B. All pertinent QA requirements of Appendix B to 10 CFR Part 50 are applied. C. For lines utilized as an MSIV leakage path to the condenser, reliable power sources must be available for control and isolation valves so that a control operator can establish the flow path assuming a single active failure. In addition, the TBS lines and other components utilized as an MSIV leakage path to the condenser are assigned a quality group classification in accordance with the criteria of Branch Technical Position 3-1.					
10.4.5, Rev. 3 (03/2007)	Circulating Water System					
10.4.5.1	The requirements of GDC 4 are met when the circulating water system design includes provisions to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS. Acceptance is based on meeting the following: A. Means should be provided to prevent or detect and control flooding of safety-related areas so that the intended safety function of a system or component will not be precluded due to leakage from the CWS. B. Malfunction or a failure of a component or piping of the CWS, including an expansion joint, should not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components.					
10.4.6, Rev. 3 (03/2007)	Condensate Cleanup System					
10.4.6.1	For direct cycle (boiling-water reactor (BWR)) plants, SRP Section 5.4.8 provides the criteria for acceptable water purity.					

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	SRP Section 5.4.8 refers to the guidelines provided in the latest version in the Electric Power Research Institute (EPRI) report series, "BWR Water Chemistry Guidelines," and the technical specifications for the water chemistry of BWR reactor coolant systems.					
10.4.6.2	For indirect cycle (pressurized-water reactor (PWR)) plants, SRP Section 5.4.2.1 provides the criteria for acceptable secondary water chemistry. SRP Section 5.4.2.1 refers to the guidelines provided in the latest version in the EPRI report series, "PWR Secondary Water Chemistry Guidelines."					
10.4.7, Rev. 4 (03/2007)	Condensate and Feedwater System					
10.4.7.1	Seismic Events. The requirements of GDC 2 are met by demonstrating that structures, systems, and components important to safety will be designed to withstand the effects of natural phenomena such as earthquakes. Acceptance is based on meeting the guidance of Regulatory Guide 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions.					
10.4.7.2	Fluid Instabilitites. The requirements of GDC 4 as related to protecting structures, systems and components against the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation as well as during upset or accident conditions are met by: A. Meeting the guidance contained in the Branch Technical Position 10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators," for reducing the potential for water hammers in steam generators; and B. Meeting the guidance related to feedwater-control-induced water hammer. Guidance for water hammer prevention and mitigation is found in NUREG-0927, Revision 1.					
10.4.7.3	Sharing of Structures, Systems, and Components. The requirements of GDC 5 are met by demonstrating the capability of important to safety components in the CFS which are shared					

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	by multiple units to perform their required safety functions.					
10.4.7.4	Heat Removal Capability. The requirements of GDC 44, as related to the capability to transfer heat from structures, systems and components important to safety to an ultimate heat sink are met by demonstrating that the CFS is capable of providing heat removal under both normal operating and accident conditions. Sufficient redundancy of components is demonstrated so that under accident conditions the safety function can be performed assuming a single active component failure (which may be coincident with the loss of offsite power for certain events.) The system demonstrates capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.					
10.4.7.5	Inspection. The requirements of GDC 45 are met by demonstrating that the design contains provisions to permit periodic inservice inspection of system components and equipment.					
10.4.7.6	Testing. The requirements of GDC 46 are met by demonstrating that the design contains provisions to permit appropriate functional testing of the system and components to ensure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.					
10.4.7.7	Flow Accelerated Corrosion. Piping system designs, including material standards and inspection programs, shall incorporate adequate considerations to avoid erosion and corrosion. Guidance for acceptable inspection programs is found in Generic Letter 89-08 and in EPRI NP-3944, "Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines."					
10.4.7.8	Feedwater Nozzle Design. For BWRs, feedwater nozzle design, inspection, and testing procedures, and CFS operating procedures are adequate to minimize nozzle cracking at low					

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	feedwater flow. The review criteria for this issue are stated in NUREG-0619 and in associated Generic Letters 80-95 and 81-11.					
10.4.8, Rev. 3 (03/2007)	Steam Generator Blowdown System					
10.4.8.1	The requirements of GDC 1 and GDC 2 are met when the design of the SGBS includes the following: A. The design is seismic Category I and Quality Group B, from its connection to the steam generator inside primary containment up to and including the first isolation valve outside containment. B. The design is in accordance with the provisions of Regulatory Guide 1.143, Position C.1.1 downstream of the outer containment isolation valves.					
10.4.8.2	The requirements of GDC 13 are met when the SGBS design includes provisions to monitor system parameters and maintain them within a range that allows the system to perform its impurity removal function and thereby assist in maintaining the integrity of the reactor coolant pressure boundary.					
10.4.8.3	The requirements of GDC 14 are met when the SGBS design includes provisions to control secondary water chemistry to maintain the integrity of the primary coolant boundary. Acceptance is based on meeting the following: A. The SGBS is sized to accommodate the design blowdown flow needed to maintain secondary coolant chemistry for normal operation, including anticipated operational occurrences. B. Equipment capacities are based on design blowdown flow rates and are such that temperature limits for heat-sensitive processes are not exceeded.					
10.4.9, Rev. 3 (03/2007)	Auxiliary Feedwater System (PWR)					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
10.4.9.1	Acceptance for meeting the relevant aspects of GDC 2 is based in part on meeting the guidance of Position C.1 of Regulatory Guide 1.29 if any portion of the system is deemed to be safety related and the guidance of Position C.2 for nonsafety-related portions. Also, acceptance is based in part on (1) meeting the guidance of Regulatory Guide 1.117 with respect to identifying portions of the system that should be protected from tornadoes and (2) meeting the guidance of Regulatory Guide 1.102 with respect to identifying portions of the system that should be protected from flooding.					
10.4.9.2	Acceptance for meeting the relevant aspects of GDC 4 is based on identification of essential portions of the system as protected from dynamic effects including internal and external missiles. In part, this information should be consistent with the guidance of Regulatory Guide 1.117 with respect to identifying portions of the system that should be protected from tornado missiles and the guidance of BTP 3-3 with respect to identifying portions of the system that should be protected from the dynamic effects of pipe breaks.					
10.4.9.3	Acceptance of GDC 5 is based on provision of information that addresses the capability of shared portions of the AFW system to perform required safety functions during an accident in one unit such that the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s) is not significantly affected.					
10.4.9.4	Acceptance of GDC 19 is based on meeting BTP 5-4 with regards to cold shutdown from the control room using only safety grade equipment.					
10.4.9.5	Acceptance of GDC 34 and 44 is based on the system having sufficient flow capacity so that the system can remove residual heat over the entire range of reactor operation and cool the plant to the decay heat removal system cut-in temperature and the					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	system design conforming to the guidance of BTP 10-1 as it relates to AFW pump drive and power supply diversity. In addition, the recommendations of NUREG-0611 and NUREG-0635 shall also be met. TMI Action Plan item II.E.1.1 of NUREG 0737 and 10 CFR 50.34(f)(1)(ii) for applicants subject to 10 CFR 50.34(f) require an AFWS reliability analysis. An acceptable AFWS should have an unreliability in the range of 10^{-4} to 10^{-5} per demand exclusive of station blackout scenarios. Compensating factors (e.g., other methods of accomplishing AFWS safety functions of the AFWS or other reliable methods for cooling the reactor core during abnormal conditions) may be considered to justify a larger AFWS unavailability.					
10.4.9.6	Acceptance of GDC 45 is based on provision of information describing how the design of the AFW system permits inservice inspection of safety-related components and equipment.					
10.4.9.7	Acceptance of GDC 46 is based on provision of information describing how the design of the AFW system, including instrumentation, permits periodic operational functional testing of safety-related components and equipment.					
10.4.9.8	Acceptance of 10 CFR 50.62 is based on design provisions for automatic initiation of the AFW system in an ATWS.					
10.4.9.9	Acceptance of 10 CFR 50.63 is based on conformance with the guidance of RG 1.155as related to the AFWS design.					
Branch Technical Position, 10-1, Rev. 3 (03/2007)	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants					
BTP 10-1.1	The AFWS should have at least two full-capacity, independent systems with diverse power sources.					
BTP 10-1.2	Other AFWS powered components also should have separate and multiple sources of motive energy (e.g., two separate					

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	auxiliary feedwater trains, each capable of removing the reactor system after-heat load, one separate train powered from either of two alternating current sources and the other powered wholly by steam and direct current electric power).					
BTP 10-1.3	The piping arrangements, both intake and discharge, for each train should be designed for the pumps to supply feedwater to any combination of steam generators. This arrangement should be designed for pipe failure, active component failure, power supply failure, or control system failure that could prevent system function. One acceptable arrangement is crossover piping with valves operable by remote manual control from the control room applying the power diversity principle to the valve operators and actuation systems.					
BTP 10-1.4	The AFWS design should have suitable redundancy to offset the consequences of any single-active component failure; however, each train need not have redundant active components.					
BTP 10-1.5	For a high-energy line break, the system should be arranged to assure the capability to supply necessary emergency feedwater to the steam generators despite the postulated rupture of any high-energy section of the system, assuming a concurrent, single, active failure.					
Branch Technical Position, 10-2, Rev. 4 (03/2007)	Design Guidelines for Avoiding Water Hammers in Steam Generators					
B. Item 1.1	Top-Feed Steam Generator Designs To eliminate or reduce possible water hammer in the feedwater system: 1. Prevent or delay water draining from the feed ring following a drop in steam generator water level by means such as top discharge J-Tubes and limiting feed ring seal assembly leakage.					

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	2. Minimize the volume of feedwater piping external to the steam generator which could pocket steam using the shortest possible (less than 2.1 m (7 ft)) horizontal run of inlet piping to the steam generator feed ring. 3. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater and possible draining of the feed ring. Provide the procedures for these tests for approval before conducting the tests and submit the results from such tests. 4. Implement pipe refill flow limits where practical.					
B Item 1.2	Preheat Steam Generator Designs 1. Minimize the horizontal lengths of feedwater piping between the steam generator and the vertical run of piping by providing downward turning elbows immediately upstream of the main and auxiliary feedwater nozzles. 2. Provide a check valve upstream of the auxiliary feedwater connection to the top feedwater line. 3. Maintain the top feedwater line full at all times. 4. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater. Also perform a water hammer test at the power level at which feedwater flow is transferred from the auxiliary feedwater nozzle to the main feedwater nozzle. The test shall be					

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	performed by pumping feedwater through the auxiliary feedwater (top) nozzle at the lowest feedwater temperature that the plant standard operating procedure (SOP) allows and then switching the feedwater at that temperature from the auxiliary feedwater nozzle to the main feedwater (bottom) nozzle by following the SOP. Submit the results of such tests.					
B Item 1.3	Once Through Steam Generator (OTSG) Designs 1. Provide auxiliary feedwater to the steam generator through an externally mounted supply top discharge header. 2. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater. Provide the procedures for these tests for approval before conducting the tests, and submit the results of such tests.					
	CHAPTER 11, Radioactive Waste Management					
11.1, Rev. 3 (03/2007)	Source Terms					
11.1.1	All normal and potential sources of radioactive effluent delineated above in Subsection I will be considered.					
11.1.2	For each source of liquid and gaseous waste considered above in Subsection I.1, the volumes and concentrations of radioactive material given for normal operation and anticipated operational occurrences should be consistent with those given in NUREG-0016 or NUREG-0017.					
11.1.3	Decontamination factors for inplant control measures used to reduce gaseous effluent releases to the environment, such as iodine removal systems and high-efficiency particulate air (HEPA) filters for building ventilation exhaust systems and containment internal cleanup systems should be consistent with					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	those given in Regulatory Guide 1.140. The building mixing efficiency for containment internal cleanup should be consistent with NUREG-0017.					
11.1.4	Decontamination factors for inplant control measures used to reduce liquid effluent releases to the environment, such as filters, demineralizers and evaporators, should be consistent with those given in NUREG-0016 or NUREG-0017.					
11.1.5	Radwaste augments used in the calculation of effluent releases to the environment are consistent with the findings of a cost-benefit analysis, which may be performed using the guidance of Regulatory Guide 1.110. The provisions that require a cost-benefit analysis are stated in Section II.D of Appendix I to 10 CFR Part 50.					
11.1.6	Effluent concentration limits at the boundary of the unrestricted area do not exceed the values specified in Table 2 of Appendix B to 10 CFR Part 20.					
11.1.7	The source terms result in meeting the design objectives for doses in unrestricted areas as set forth in Appendix I to 10 CFR Part 50.					
11.1.8	For evaluating the source terms, the applicant should provide the relevant information in the SAR as required by 10 CFR 50.34, and 10 CFR 50.34a. This technical information should include all the basic data listed in Appendix A (BWRs) and Appendix B (PWRs) to Regulatory Guide 1.112 in order to calculate the releases of radioactive material in liquid and gaseous effluents (the source terms). An acceptable method for satisfying the criteria given in items 1 through 5 consists of using the Gaseous and Liquid Effluent (GALE) Computer Code and the source term parameters given in NUREG-0016 or NUREG-0017 for BWRs and PWRs, respectively. Complete listings of the GALE Computer Codes for BWRs and PWRs are given in NUREG-0016 and NUREG-0017, respectively.					
11.1.9	If the applicant's calculational technique or any source term parameter differs from that given in ANSI/ANS 18.1-1999,					

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	NUREG-0016, or NUREG-0017, they should be described in detail and the bases for the methods and/or parameters used should be provided.					
11.2, Rev. 3 (03/2007)	Liquid Waste Management System					
11.2.1	The LWMS should have the capability to meet the dose design objectives and include provisions to treat liquid radioactive wastes such that the following is true: A. The calculated annual total quantity of all radioactive materials released from each reactor at the site to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 0.03 millisievert (mSv) (3 millirem (mrem)) to the total body or 0.1 mSv (10 mrem) to any organ. Regulatory Guides 1.109, 1.112, and 1.113 provide acceptable methods for performing this analysis. B. In addition to 1.A, the LWMS should include all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return for a favorable cost-benefit ratio, can effect reductions in doses to the population reasonably expected to be within 80 kilometers (km) (50 miles (mi)) of the reactor. Regulatory Guide 1.110 provides an acceptable method for performing this analysis. C. The concentrations of radioactive materials in liquid effluents released to unrestricted areas should not exceed the concentration limits in Table 2, Column 2, of Appendix B, to 10 CFR Part 20.					
11.2.2	The LWMS should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process liquid wastes during periods when major processing equipment may be down for maintenance (single					

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	failures) and during periods of excessive waste generation. Systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences, are acceptable. To meet these processing demands, interconnections between subsystems, redundant equipment, mobile equipment, and reserve storage capacity will be considered.					
11.2.3	The seismic design of structures housing LWMS components, the quality group classification of liquid radwaste treatment equipment, and provisions to prevent and collect spills from indoor and outdoor storage tanks should conform to the guidelines of Regulatory Guide 1.143 for liquids and liquid wastes produced during normal operation and anticipated operational occurrences. For the purpose of this SRP, the dose limit cited in Section 5 of Regulatory Guide 1.43, addressing unmitigated releases of radioactive materials, is revised to be consistent with that of 10 CFR Part 20.1301. The annual dose limit of Part 20.1301 is 100 mrem for members of the public located in unrestricted areas.					
11.2.4	System designs should contain provisions to control leakage and facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143 and industry standards cited in this regulatory guide for liquids and liquid wastes produced during normal operation and anticipated operational occurrences.					
11.2.5	System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste, in accordance with the guidelines of Regulatory Guide 1.143, for liquids and liquid wastes produced during normal operation and anticipated operational occurrences, and the requirements of 10 CFR 20.1406, or the DC application, update in the SAR, or the					

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	COL application, to the extent not addressed in a referenced certified design.					
11.2.6	For an ESP application, the dose estimates to a hypothetical maximally exposed member of the public from liquid effluents using radiological exposure models are developed based on Regulatory Guides 1.109, 1.111, and 1.113, and appropriate computer codes, such as the LADTAP II computer code (NUREG/CR-4013) for liquid effluents. Refer to the RG for the relevant regulatory guides and Branch Technical Positions.					
11.3, Rev. 3 (03/2007)	Gaseous Waste Management System					
11.3.1	The GWMS should have the capability to meet the dose design objectives and should include provisions to treat gaseous radioactive wastes such that the following is true: A. The calculated annual total quantity of all radioactive materials released from each reactor to the atmosphere will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 0.05 mSv (5 mrem) to the total body or 0.15 mSv (15 mrem) to the skin. Regulatory Guides 1.109, 1.111, and 1.112 provide acceptable methods for performing this analysis. B. The calculated annual total quantity of radioactive materials released from each reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 0.01 cGy (10 millirads) for gamma radiation or 0.02 cGy (20 millirads) for beta radiation. Regulatory Guides 1.109, 1.111, and 1.112 provide acceptable methods for performing this analysis.					

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	C. The calculated annual total quantity of radioiodines, carbon-14, tritium, and all radioactive materials in particulate form released from each reactor at the site in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such releases for any individual in an unrestricted area from all pathways of exposure in excess of 0.15 mSv (15 mrem) to any organ. Regulatory Guides 1.109, 1.111, and 1.112 provide acceptable methods for performing this analysis. D. In addition to 1.A, 1.B, and 1.C, above, the GWMS should include all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, for a favorable cost-benefit ratio, can effect reductions in dose to the population reasonably expected to be within 80 km (50 mi) of the reactor. Regulatory Guide 1.110 provides an acceptable method for performing this analysis. E. The concentrations of radioactive materials in gaseous effluents released to an unrestricted area should not exceed the limits specified in Table 2, Column 1, of Appendix B to 10 CFR Part 20. F. The regulatory position contained in Regulatory Guide 1.140 is met, as it relates to the design testing and maintenance of normal ventilation exhaust system air filtration and adsorption units at nuclear power plants. G. The regulatory position contained in Regulatory Guide 1.143 is met, as it relates to the seismic design and quality group classification of components used in the structures housing the GRS and the provisions used to control leakages of gaseous wastes produced during normal operation and anticipated operational occurrences.					

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	H. The regulatory position contained in Regulatory Guide 1.143 is met, as it relates to the definition of the boundary of the GWMS, beginning at the interface from plant systems to the point of controlled discharges to the environment as defined in the ODCM, or at the point of storage in holdup tanks or decay beds for gaseous wastes produced during normal operation and anticipated operational occurrences.					
11.3.2	The GWMS should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process gaseous wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. Systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences, are acceptable. To meet these processing demands, the reviewer will consider shared systems, redundant equipment, mobile equipment, and reserve storage capacity.					
11.3.3	The seismic design and quality group classification of components used in the GWMS and structures housing the system should conform to Regulatory Guide 1.143. The design should include precautions to stop continuous leakage paths (i.e., to provide liquid seals downstream of rupture discs) and to prevent permanent loss of the liquid seals in the event of an explosion due to gaseous wastes produced during normal operation and anticipated operational occurrences.					
11.3.4	System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste in accordance with Regulatory Guide 1.143, for gaseous wastes produced during normal operation and anticipated operational occurrences, and the requirements of 10 CFR 20.1406 or the DC					

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	application, update in the SAR, or the COL application to the extent not addressed in a referenced certified design.					
11.3.5	System designs should use the guidelines in Regulatory Guide 1.140 for the design testing and maintenance of HEPA filters and charcoal adsorbers installed in normal ventilation exhaust systems. If decontamination factors for radioiodines that differ from those specified in Regulatory Guide 1.140 are used for design purposes, they should be supported by test data under operating or simulated operating conditions (temperature, pressure, humidity, expected iodine concentrations, and flow rate). The test data should also support the effects of aging and poisoning by airborne contaminants.					
11.3.6	If the potential for explosive mixtures of hydrogen and oxygen exists, the GRS portion of the GWMS should either be designed to withstand the effects of a hydrogen explosion or be provided with dual gas analyzers with automatic control functions to preclude the formation or buildup of explosive mixtures. The GRS is normally the only portion of the system that is vulnerable to potential hydrogen explosion. A. For a system designed to withstand the effects of a hydrogen explosion, the design pressure of the system should be approximately 20 times the operating absolute pressure (including the intermediate stage condenser for BWR offgas systems). B. Small allowances should be made to conform to standard design pressures for off-the-shelf components (e.g., if the system operating pressure is nominally 103 kPa (15 psia) but could approach 138 kPa (20 psia) by design, piping could be designed to 2413 kPa (350 psia), since the next higher standard pressure rating is 4137 kPa (600 psia)). C. The process gas stream should be analyzed for potentially explosive mixtures and annunciated both locally and in the control					

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	room. D. For systems not designed to withstand a hydrogen explosion, dual gas analyzers (with dual being defined as two independent gas analyzers continuously operating and providing two independent measurements verifying that hydrogen and/or oxygen are not present in potentially explosive concentrations) with automatic control functions are required to preclude the formation or buildup of both locally and in the control room. Analyzer "high alarm" setpoints should be set at approximately 2 percent and "high-high alarm" setpoints should be set at a maximum of 4 percent hydrogen or oxygen. Control features to reduce the potential for explosion should be automatically initiated at the "high-high alarm" setting. The automatic control features should be as follows: i. For systems designed to preclude explosions by maintaining either hydrogen or oxygen below 4 percent, the source of hydrogen or oxygen (as appropriate) should be automatically isolated from the system (valves should fail in closed position). ii. For systems using recombiners, if the downstream hydrogen and/or oxygen concentration exceeds 4 percent (as appropriate), acceptable control features include automatic switching to an alternate recombiner train. iii. Injection of diluents to reduce concentrations below the limits specified herein. Systems designed to operate below 4 percent hydrogen and below 4 percent oxygen may be analyzed for either hydrogen or oxygen; systems designed to operate below 4 percent hydrogen only (no oxygen restrictions) should be analyzed for hydrogen; and systems designed to operate above 4 percent hydrogen					

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	should be analyzed for oxygen. For BWR systems with steam dilution upstream of the recombiners, analysis for hydrogen (oxygen is not an acceptable alternative) should be downstream of the recombiners and upstream of the delay portions of the system (analysis upstream of the recombiners is not required if the system is designed to assure the availability of dilution steam during operation). For PWR systems using recombiners, analysis for hydrogen and/or oxygen should be downstream of the recombiners. In addition, unless the system design features preclude explosive gas mixtures of hydrogen and oxygen upstream of the recombiners, analysis for hydrogen and/or oxygen (as appropriate) should be upstream of the recombiners as well. The number of gas analyzers and control features at each location should be in accordance with this SRP section. One gas analyzer upstream and one gas analyzer downstream of the recombiners should not be construed as dual gas analyzers. For systems involving pressurized storage tanks (excluding surge tanks), at least one gas analyzer is required between the compressor and the storage tanks. Dual gas analyzers set to sequentially measure concentrations both upstream and downstream of a recombiner are acceptable for a PWR. When two or more potentially explosive process streams are combined before entering a component, each stream or the combination thereof, is required to have dual gas analyzers. If gas analyzers are to be used to sequentially measure several points in a system not designed to withstand a hydrogen explosion, at least one gas analyzer which is continuously on stream is required. The continuous gas analyzer should be located at a point common to streams and measured sequentially (i.e., the analyzer should be sampling the combined stream).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Gas analyzers should have daily sensor checks, monthly functional checks, and quarterly calibrations. Gas analyzers installed in systems designed to withstand a hydrogen explosion should be capable of withstanding a hydrogen explosion; gas analyzers installed in the systems not designed to withstand a hydrogen explosion need not be capable of withstanding a hydrogen explosion (similar requirements apply to radiation monitors which are internal to lines containing potentially explosive mixtures). All gas analyzer instrumentation systems shall be nonsparking.					
11.3.7	Branch Technical Position (BTP) 11-5, as it relates to potential releases of radioactive materials (noble gases) as a result of postulated leakage or failure of a waste gas storage tank or offgas charcoal delay bed.					
11.3.8	For an ESP application, the dose estimates to a hypothetical maximally exposed member of the public from gaseous effluents using radiological exposure models are developed based on Regulatory Guides 1.109 and 1.111, and appropriate computer codes, such as the GASPAR II computer code (NUREG/CR-4653) for gaseous effluents.					
11.4, Rev. 3 (03/2007)	Solid Waste Management System					
11.4.1	The SWMS design parameters are based on expected radionuclide distributions and concentrations consistent with reactor operating experience for similar designs, as evaluated under SRP Section 11.1					
11.4.2	Processing equipment is sized to handle the design SWMS inputs, that is, the types of liquid, wet, and solid wastes; radionuclide distributions and concentrations; radionuclide removal efficiencies and decontamination factors; waste volume reduction and increase factors; waste volumes; and waste generation rates.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
11.4.3	All liquid and wet wastes will be stabilized in accordance with a PCP before offsite shipment, or provisions will be made to verify the absence of free liquid in each container and procedures to reprocess containers in which free liquid is detected in accordance with the requirements of Branch Technical Position (BTP) 11-3.					
11.4.4	Other forms of wet wastes will be stabilized or dewatered (subject to the licensed disposal facility's waste acceptance criteria) in accordance with a PCP, or provisions will be made to verify the absence of free liquid in each container and procedures to reprocess containers in which excess water is detected in accordance with the requirements of BTP 11-3.					
11.4.5	SWMS design objectives, design criteria, treatment methods, expected effluent releases, process and effluent radiation monitoring and control instrumentation, and methods for establishing process and effluent instrumentation control set points, as they relate to the PCP and ODCM under this SRP Section and SRP Section 11.5.					
11.4.6	Waste containers, shipping casks, and methods of packaging wastes meet all applicable Federal regulations (e.g., 10 CFR Part 71, addressing the packaging and transportation of radioactive materials; 10 CFR 20.2006 and Appendix G to 10 CFR Part 20, addressing the transfer and manifesting of radioactive waste shipments; and 49 CFR Parts 171–180, addressing U.S. Department of Transportation (DOT) regulations for the shipment of radioactive materials); and 10 CFR Part 61 or corresponding State regulations addressing applicable waste acceptance criteria of the disposal facility or waste processors.					
11.4.7	Onsite waste storage facilities provide sufficient storage capacity to allow time for shorter lived radionuclides to decay before shipping in accordance with the requirements of BTP 11-3. The SAR should give the bases for determining the duration of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	storage.					
11.4.8	SWMS components and piping systems, as well as structures housing SWMS components, are designed in accordance with the provisions of Regulatory Guide 1.143, as it relates to the seismic design and quality group classification of components, and BTP 11-3 for wastes produced during normal operation and anticipated operational occurrences.					
11.4.9	The SWMS contains provisions to reduce leakage and facilitate operations and maintenance in accordance with the provisions of Regulatory Guide 1.143 and BTP 11-3, as they relate to wastes produced during normal operation and anticipated operational occurrences.					
11.4.10	For long-term onsite storage (e.g., for several years, but within the operational life of the plant), the storage facility should be designed to the guidelines of Appendix 11.4-A to this SRP section, including updated guidance from SECY 93-323 and SECY 94-198.					
11.4.11	Liquid, wet, and dry solid wastes will be processed and disposed of in accordance with 10 CFR 61.55 and 10 CFR 61.56 requirements for waste classification and characteristics and with the waste acceptance criteria of the chosen licensed radioactive waste disposal site. The PCP should present the process and methods used to meet these 10 CFR Part 61 requirements.					
11.4.12	Mixed wastes (characterized by the presence of hazardous chemicals and radioactive materials) will be processed and disposed in accordance with 10 CFR 20.2007, as it relates to compliance with other applicable Federal, State, and local regulations governing any other toxic or hazardous properties of radioactive wastes.					
11.4.13	All effluent releases (gaseous and liquid) associated with the operation (normal and anticipated operational occurrences) of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	SWMS will comply with 10 CFR Part 20 and Regulatory Guide 1.143, as they relate to the definition of the boundary of the SWMS beginning at the interface from plant systems, including multiunit stations, to the points of controlled liquid and gaseous effluent discharges to the environment or designated onsite storage locations, as defined in the PCP and ODCM.					
11.4.14	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the PCP aspect of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with 10 CFR 20.1301 and 20.13.2, 10 CFR 50.34a, 10 CFR 50.36a, and 10 CFR 50, Appendix I, section II and IV. Its implementation is required by a license condition.					
11.5, Rev. 4 (03/2007)	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems					
11.5.1	Provisions should be made for the installation of instrumentation and monitoring equipment and/or sampling and analyses of all normal and potential effluent pathways for release of radioactive materials to the environment, including nonradioactive systems that could become radioactive through interfaces with radioactive systems. For GDC 64 and the requirements specified in 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii), the system designs should meet the provisions of Regulatory Guide 1.21 (Position C and Appendix A), Regulatory Guide 1.97 (Position C and Table 1 or 2, as applicable), Regulatory Guide 4.15 (Position C), and Appendix A to Regulatory Guide 1.33. SRP Branch Technical Position (BTP) 7-10 (see SRP Section 7.5) provides additional guidance on the application of Regulatory Guide 1.97. A. The gaseous and liquid process streams or effluent release points should be monitored and sampled according to Tables 1 and 2 of this SRP. B. For both boiling water-reactors (BWRs) and pressurized-water					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	reactors (PWRs), liquid waste and gaseous waste (contained in tanks) should be sampled on a batch basis before their release, in accordance with Regulatory Guide 1.21. Open structures, such as PWR turbine buildings or atmospheric vents for liquid waste tanks containing treated or processed liquid waste and located outside of buildings, do not require continuous gaseous effluent monitors. For liquid and gaseous effluents that cannot be easily monitored or sampled on a batch basis, one of the following representative sampling methods should be provided: i. Use of a continuous proportioning sampling system, with at least two sample collection tanks. The system should be designed to collect a sample at a fixed ratio established between the sample collection flow rate and the effluent stream discharge flow rate. ii. Use of a periodic automatic grab sampling system, with at least two sample collection tanks. The system should be designed to collect a sample at a fixed volume established at a rate that is proportional to the effluent stream discharge flow rate. iii. For radioactive materials, other than noble gases in gaseous effluents, a continuous sampling system should be used with replaceable particulate filters and radioiodine adsorbers. The system should be designed to automatically take representative samples at a known flow rate established in accordance with American National Standards Institute/Health Physics Society (ANSI/HPS) N13.1-1999. iv. For intermittently operating effluent release points, the system should be designed to automatically take samples whenever flow is in the effluent stream using a known ratio between the discharge and sampling stream flow rates.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	v. Periodic sampling and analysis frequencies and types of radiological analyses should be specified for all samples described above in the SREC, ODCM, and/or PCP.					
11.5.2	Provisions should be made for the installation of instrumentation and monitoring equipment and/or periodic or continuous sampling and analysis of radioactive waste process systems. For GDC 60 and 63, as they relate to radioactive waste systems, detection of excessive radiation levels, and initiation of appropriate safety actions, the design of systems should meet the guidelines of Appendix 11.5-A, Regulatory Guide 1.21 (Position C, as applicable), Regulatory Guide 1.97 (Position C and Table 1 or 2, as applicable), Regulatory Guide 4.15 (Position C), and Appendix A to Regulatory Guide 1.33. SRP BTP 7-10 (see SRP Section 7.5) provides additional guidance on the application of Regulatory Guide 1.97. A. Provisions should be made to ensure representative sampling from radioactive process streams and tank contents. Recirculation pumps for liquid waste tanks (collection or sample test tanks) should be capable of recirculating at a rate of not less than two tank volumes in 8 hours. For gaseous and liquid process stream samples, provisions should be made for purging sampling lines and for reducing the plate-out of radioactive materials in sample lines. Provisions for gaseous sampling from ducts and stacks should be consistent with ANSI/HPS N13.1-1999. B. When practicable, provisions should be made to collect samples from process waste streams at central sample stations to reduce leakage, spillage, and radiation exposures to operating personnel in accordance with SRP Section 9.3.2 and 10 CFR 20.1406. C. Provisions should be made to purge and drain sample					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	streams back to the system of origin or to an appropriate waste treatment system.					
11.5.3	Provisions should be made for administrative and procedural controls for the installation of necessary auxiliary or ancillary equipment, for the inclusion of special features in instrumentation and radiological monitoring sampling systems, and for the analysis of process and effluent streams. For GDC 63 and 64 (including the requirements specified in 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii)), as they relate to radioactive waste process systems and effluent discharge paths, the design of systems and the implementation of administrative and procedural controls should meet the guidelines of Appendix 11.5-A, Regulatory Guide 1.21 (Position C), Regulatory Guide 4.15 (Position C), and Appendix A to Regulatory Guide 1.33. Instrumentation, sampling, and monitoring provisions should conform to the following: A. Sampling frequencies, required analyses, instrument alarm/trip setpoints, calibration and sensitivities, and provisions for preparing composite samples for low-level radioactivity analyses should conform to Regulatory Guides 1.21, 1.33, and 4.15. The plant's SREC, ODCM, and/or PCP should indicate sampling frequencies and required analyses. B. Provisions should be made for the necessary instrumentation and facilities to perform gross beta-gamma and gross alpha measurements, isotopic or radionuclide-specific analyses, and other routine analyses in conformance with Regulatory Guides 1.21, 1.33, and 4.15. C. Provisions should be made to perform routine instrument calibration, maintenance, and inspections in conformance with Regulatory Guides 4.15 and 1.33. Instrumentation calibration					

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	procedures should consider whether instrumentation response is expected to change given that radionuclide distributions may vary with the operating status of the plant (i.e., normal operation, anticipated operational occurrences, and post-accident conditions). The plant's SREC, ODCM, and/or PCP should indicate the frequency of such actions. Provisions should also be made to replace or decontaminate instrumentation or sampling equipment without opening the process system or losing the capability of isolating the effluent stream. D. Isolation valves, dampers, or diversion valves with automatic control features should fail in the closed or safe position. The plant's SREC, ODCM, and/or PCP should establish setpoints for actuation of automatic control features initiating actuation of isolation valves, dampers, or diversion valves. The bases for establishing instrumentation alarm or system activation setpoints should be provided, taking into consideration the following: i. For liquid effluents, in-plant effluent dilution factors and dilution factors beyond the point of discharge to the site boundary and nearest offsite dose receptors ii. For gaseous and particulate effluents from plant stacks and building vents, atmospheric dispersion (χ/Q) and deposition (D/Q) factors to the site boundary and offsite dose receptors E. Non-ESF instrumentation provisions for automatic termination or diversion of releases should conform to the design guidance contained in Appendix 11.5-A. SRP Sections 7.6 and 13.3 address the review the ESF instrumentation provisions for automatic termination or diversion of releases. F. The process used to develop, review, verify, validate, and audit digital computer software used in radiation monitoring and					

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	sampling equipment, including software used to terminate or divert process and effluent streams. This aspect addresses software developed by the applicant, purchased through a vendor, or included with the instrumentation.					
11.5.4	Provisions should be made for monitoring instrumentation, sampling, and sample analyses for all identified gaseous effluent release paths in the event of a postulated accident. For GDC 64, as it relates to potential gaseous effluent release paths, the design of systems should meet the provisions of NUREG-0718 and NUREG-0737 (item II.F.1 and Attachments 1 and 2), 10 CFR 50.34(f)(2)(vxii) and 10 CFR 50.34(f)(2)(vxxii), Appendix 11.5-A, and Regulatory Guide 1.97 (Position C). SRP BTP 7-10 (see SRP Section 7.5) provides additional guidance on the application of Regulatory Guide 1.97. In addition, the design of the gaseous waste collection and processing system should meet the guidelines referenced in SRP Sections 9.3.2, 11.3, and 13.3, as well as the following conditions: A. Administrative controls and procedures in conformance with Acceptance Criterion 3 of this SRP section are to be in effect to minimize inadvertent or accidental releases of radioactive gaseous and particulate effluents. B. Gaseous and particulate radiological effluent monitors are to be provided for the automatic termination of releases in the event that effluent release setpoints are exceeded, as provided in Acceptance Criterion 1 of this SRP section and as established in the plant's SREC, ODCM, and/or PCP.					
11.5.5	Provisions should be made for monitoring instrumentation, sampling, and sample analysis for all identified liquid effluent release paths in the event of a postulated accident. These provisions should be in accordance with GDC 64 and the requirements of 10 CFR 50.34(f)(2)(vxii) and 10 CFR 50.34(f)(2)(vxxii), as they relate to postulated accidents and					

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	identified liquid effluent release paths. In addition, the design of the liquid waste collection and processing system should meet the guidelines referenced in SRP Sections 9.3.2, 11.2, and 13.3, as well as the following conditions: A. Administrative controls and procedures in conformance with Acceptance Criterion 3 of this SRP section are to be in effect to minimize inadvertent or accidental releases of radioactive liquids. B. Liquid effluent radiological monitors are to be provided for the automatic termination of releases in the event that effluent release setpoints are exceeded, as provided in Acceptance Criterion 1 of this SRP section and as established in the plant's SREC, ODCM, and/or PCP.					
11.5.6	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the RETS/SREC, ODCM and REMP aspects of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with 10 CFR 20.1301 and 20.13.2, 10 CFR 50.34a, 10 CFR 50.36a, and 10 CFR Part 50, Appendix I, Section II and IV. Its implementation is required by a license condition.					
Branch Technical Position 11-3, Rev. 3 (03/2007)	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants					
BTP 11-3.1	Processing Requirements A. Dry Wastes i. Compaction devices for compressible dry wastes (rags, paper, and clothing) should include a ventilated shroud around the waste container to control the					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	release of airborne radioactivity generated during the compaction process. ii. Activated charcoal, HEPA filters, and other dry wastes that do not normally require stabilization processing should be treated as radioactively contaminated solids and packaged for disposal in accordance with applicable Federal, State, and local regulations addressing the presence of other toxic and hazardous materials. B. Wet Wastes i. Liquid wet wastes, such as evaporator and reverse osmosis concentrates, should be rendered immobile by combining with a suitable binding agent (e.g., cement, asphalt) to form a homogeneous solid matrix (absent of free water) before offsite shipment. Adsorbents such as vermiculite are not acceptable substitutes for binding agents. ii. Spent resins and filter sludge, if acceptable to the receiving burial site, may be shipped dewatered. These dewatered wastes are subject to (1) Subsections II.2.A.ii and II.2.B below, (2) to the receiving burial site's maximum free-liquid criteria (upon receipt at the burial site), and (3) applicable DOT regulations under 49 CFR Parts 171–180. Furthermore, the activity level of the dewatered wastes, subject to receiving burial site requirements, may dictate the type of container to be used. Stabilization or encapsulation of spent resins and filter sludge in a suitable binder is also an acceptable alternative. iii. Spent cartridge filter elements may be packaged in a					

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	shielded container with a suitably acceptable absorber, although solidifying the elements in a suitable binder is desirable.					
BTP 11-3.2	Assurance of Complete Stabilization or Dewatering Operators should assure the complete stabilization or dewatering of wet wastes by implementing a PCP or by methods to detect free liquids within container contents before shipment. A. Process Control Program i. Stabilization, encapsulation, or solidification (binding) agents and potential waste constituents should be tested and a set of process parameters (e.g., pH, ratio of waste to agent) should be established with boundary conditions that reasonably assure that stabilization will be complete, with essentially zero free liquid and appropriate waste form characteristics. ii. Dewatering procedures, equipment, and potential waste constituents should be tested and a set of process parameters (e.g., settling time, drain time, drying time) should be established with boundary conditions that reasonably assure that dewatering will be complete, with essentially zero free liquid. iii. The solid waste processing system (or liquid waste processing system, as appropriate) should include appropriate instrumentation and wet waste sampling capability necessary to successfully implement and/or verify the PCP described in Subsections II.2.A.i and/or II.2.A.ii, above. iv. The plant operator should provide assurance that the process is run within the parameters established under Subsections II.2.A.i and/or II.2.A.ii, above. The licensee should maintain appropriate records for					

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	individual batches, showing conformance with the established parameters. B. Free Liquid Detection Using suitable methods, the operator should check each container filled with stabilized or dewatered wet wastes to verify the absence of free liquids using the PCP. An alternate method may be used if an offnormal condition exists during processing, but the alternate method must be documented and its effectiveness must be demonstrated using real or simulated waste material. Visual inspection of the upper surface of the waste in the container is not alone sufficient to ensure that free water is not present in the container. Provisions to be used to verify the absence of free liquids should consider actual stabilization procedures which may create a thin layer of encapsulation or solidifying agent on top without affecting the lower portion of the container, possibly leaving pools of freestanding liquids within the waste matrix.					
BTP 11-3.3	Waste Storage A. Tanks accumulating spent resins from reactor water purification systems should be capable of accommodating at least 60 days of waste generation at normal generation rates. Tanks accumulating spent resins from other sources and tanks accumulating filter sludge should be able to accommodate at least 30 days of waste generation at normal generation rates. B. Storage areas for processed wet wastes (i.e., stabilized or dewatered wastes) should be capable of accommodating at least 30 days of waste generation at normal generation rates. These storage areas should be located indoors. C. Storage areas for dry wastes and packaged contaminated equipment should be capable of accommodating at least one full offsite waste shipment.					

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BTP 11-3.4	Portable Solid Waste Systems The following supplementary guidance should be incorporated into the design and use of portable (mobile) volume reduction, stabilization, and/or dewatering systems: A. Tanks containing wet wastes are limited to inplant installation and should not be part of the portable system. B. The use of flexible lines (hoses and connections) should be limited to necessary interfaces with plant systems. Pressure testing of all temporary and flexible lines, as connected to plant system piping embedded in concrete, and effluent discharge lines or piping buried in soils should be done. Such piping shall have a pressure rating equal to or greater than the system design pressure. Before its use, all flexible piping should be hydrostatically tested to at least 1.5 times the interfacing system design pressure and maintained for at least 30 minutes without leakage or structural deformation to ensure the integrity of the flexible piping and associated fittings. C. Corrosion-resistant properties should be used for all system piping and valves associated with transfer lines to storage tanks and discharge piping, including features designed for the early detection of leaks and spills. D. Portable systems should be located, as a minimum, on concrete pads with curbs and drainage provisions to process drains and drip pans or containment boxes to contain radioactive leaks. Provisions should be available for interfacing system drains with the plant's liquid radwaste system. Other safety features may include backflow preventers, siphon breakers, self-sealing quick-disconnects, and operational interlocks to prevent spills. Portable systems should have integral ventilation systems with self-contained filters or interface with the plant's ventilation exhaust system.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	E. Mobile liquid waste processing systems with interconnections to permanently installed plant liquid waste management subsystems should include provisions that (1) avoid the contamination of nonradioactive systems, (2) prevent uncontrolled and unmonitored releases of radioactive materials in the environment, and (3) avoid interconnections with potable and sanitary water systems. F. Designs should minimize the potential for spills and leaks to the extent practicable, consistent with maintaining radiation doses ALARA during operations and for the purpose of facilitating decommissioning. G. Regulatory Guide 1.143 seismic criteria for structures housing portable solid waste systems are not applicable.					
BTP 11-3.5	Additional Design Features The following additional features should be incorporated into the design of the solid waste system. A. Process concentrate piping and tanks should have heat tracing if the concentrates are likely to solidify at ambient temperatures (indoor or outdoor). B. Components and piping that contain radioactive slurries should have flushing connections and piping runs that minimize the number of bends and traps that may retain radioactivity and lead to increased ambient external radiation exposure rates. C. Stabilization or encapsulation agents should be stored in low radiation areas, generally less than 0.025 mSv/hour (2.5 mrem/hour), with provisions for sampling. D. Tanks or equipment that use compressed gases for transport or drying of resins or filter sludge should be vented directly to the plant ventilation exhaust system, which includes HEPA filters, as a minimum, and charcoal filters for radioiodines. The vent design should prevent					

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	liquids and solids from entering the plant ventilation system.					
Branch Technical Position 11-5, Rev. 3 (03/2007)	Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure					
BTP11-5.1	Waste Gas System Leak or Failure Analysis A. Criteria The SAR (Section 11.3) should provide an analysis of the radiological consequences of a single failure of an active component in the waste gas system. The analysis should provide reasonable assurance that, in the event of a postulated failure or leak of the waste gas system, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 25 mSv (2.5 rem) for systems designed to withstand explosions and earthquakes, or 1 mSv (0.1 rem) for systems not designed to withstand explosions and earthquakes. The bases for the analysis should include the assumption that the waste gas system fails to meet its design intent as required by 10 CFR 50.34a(c) and GDC 60 of Appendix A to 10 CFR Part 50. B. Source Term The safety analysis on the radiological consequences of a single failure of an active component in the waste gas system should use a system design-basis source term for light-water-cooled nuclear power plants. The NRC staff method of calculation for this analysis is based on conservative assumptions to maximize the design capacity source term (sustained power operation). These assumptions are given below: i. For a PWR: 1 percent of the operating fission product inventory					

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	in the core being released to the primary coolant ii. For a BWR: A fission product release rate consistent with the noble gas release to the reactor coolant of 100 µCi/s per MWt (after 30-minute decay) The analysis should assume principal parameters and conditions typical of the equipment designed to remove radioactive gases from the coolant and to process and treat these gases during normal operation, including anticipated operational occurrences, by the waste gas system. The NRC staff believes that no major alteration would occur in the use or performance of gas separation, reduction, and decay equipment before and immediately following this unique unplanned release affected by the waste gas system maximum design capacity source term. The source terms and releases may be developed using the BWR-GALE Code (NUREG-0016) or PWR-GALE Code (NUREG-0017) with appropriately justified adjustments made in modeling a specific type of event. C. Release The NRC staff considers that the release to the environment resulting from the postulated event will occur via a pathway not normally used for planned releases, and the release will require a reasonable time to detect and take remedial action to terminate the release. The NRC staff considers that the release of a compressed gas storage tank of a batch-type waste gas system or the inadvertent bypass of the main decay portion of a continuous-type waste gas system (such as charcoal delay beds in a BWR-augmented offgas system) will provide a conservative assumption for the release, while the input to the waste gas system is at the system design-basis source term. Only the radioactive noble gases (xenon and krypton) are to be considered since the assumed transit time is long enough to					

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	permit major radioactive decay of oxygen and nitrogen isotopes. Particulates and radioiodines are assumed to be removed by pretreatment, gas separation, and intermediate radwaste treatment equipment. The release should be assumed to occur within the building structure housing the waste gas system storage tank or the main decay position of the system. It should further be assumed that the effluent resulting from the postulated event will be released to the environs without continuous effluent radiation monitoring to automatically isolate and/or terminate the effluent release. In addition, ground-level release without credit for a building wake factor should be assumed, and a conservative (5 percent) short-term diffusion estimate (X/Q), as determined by a method outlined in the acceptance criteria in SRP Section 2.3.4, should be assumed. No deposition is assumed to occur during downwind transport.					
BTP11-5.2	Staff Method for Analysis A. Pressurized Storage Tanks: The safety analysis for the radiological consequences of a single failure of an active component in a waste gas system with compressed gas storage (holdup or decay) tanks or cover gas tanks assumes that the tank being filled has a major leak to the environs. The following general procedural steps should be used for this analysis: i. The radioactive noble gas inventory in the tank, at 100-percent capacity, should be determined based on the maximum expected radioactive source term and the system design capacity using the parameters and principal components considered for pretreatment and collection of waste gas to the waste gas system tanks during normal operation, including anticipated operational occurrences. The assumptions and parameters used in the analysis should be described and justified to include among others: a description of the event leading to the release, release path from the affected system and building to the environment,					

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	type of release, duration of the release, basis for the noble gas source term, assumed receptor location, atmospheric dispersion parameters, and any modifying factor specific to the event. ii. The radiological impact should be determined using the noble gas radionuclide inventory determined step 1 above, total-body dose factor listed as DFB_i in Table B-1 of Regulatory Guide 1.109, in mrem-m^3/pCi-yr, any modifying factor specific to the event, and the relative concentration (X/Q, in s/m^3) at the nearest exclusion area boundary given in Figure 1 of Regulatory Guide 1.24 for ground-level releases. iii. The dose, summed over all radionuclides, shall not exceed 25 mSv (2.5 rem) for systems designed to withstand explosions and earthquakes, or 1 mSv (0.1 rem) for systems not designed to withstand explosions and earthquakes. Using the same parameters, a corresponding TS can be defined to set a curie limit on a tank, based on the maximum of 25 mSv (2.5 rem) or 1 mSv (0.1 rem) at the nearest exclusion area boundary and same noble gas mixture to assure that the BTP criteria are met at the exclusion area boundary. B. Charcoal Delay Units: The safety analysis for the radiological consequences of a single failure of an active component in a waste gas system with charcoal delay or decay beds assumes that the charcoal unit is bypassed with a 1-hour release to the environs. The staff considers that either a line bypass valve malfunction, control error, or a charcoal bed bypass will require a remedial action by isolation and that starting an alternate charcoal unit, if available, or reducing reactor power could take up to 2 hours. The following general procedural steps should be used for this analysis: i. The radioactive noble gas inventory should be determined					

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	based on the maximum expected radioactive source term and the system design capacity using the parameters and principal components considered for pretreatment and collection of waste gas to the waste gas charcoal delay or decay beds during normal operation, including anticipated operational occurrences. The assumptions and parameters used in the analysis should be described and justified to include among others: a description of the event leading to the release, release pathway from the affected system and building to the environment, type of release, duration of the release, basis for the noble gas source term after 30-minute decay, assumed receptor location, atmospheric dispersion parameters, and any modifying factor specific to the event. ii. The radiological impact should be determined using the noble gas radionuclide inventory determined step 1 above, total-body dose factor listed as DFB_i in Table B-1 of Regulatory Guide 1.109, in mrem-m^3/pCi-yr, any modifying factor specific to the event, and the relative concentration (X/Q, in s/m^3) at the nearest exclusion area boundary given in Figure 1 of Regulatory Guide 1.24 for ground-level releases. iii. The dose, summed over all radionuclides, shall not exceed 1 mSv (0.1 rem). Using the same parameters, a corresponding TS can be defined to set a maximum release rate to the waste gas system of 100 µCi/s per MWt (after 30-minute decay) or use the value of Q_i (in µCi/s) as determined above. Using the lowest of these two values will assure that the BTP criteria are met for an exposure duration of 2 hours at the exclusion area boundary.					
Branch Technical Position 11-6 (03/2007)	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
BTP11-6.1	Site Geology and Hydrology and Conceptual Transport Models The staff will review the site's geologic and hydrologic features in assessing the potential consequences of a release radioactive materials associated with the failure of a tank and its components on current and likely future users of ground or surface water. The review of information on surface and ground water hydrology, parameters governing the movement of liquids and mobility of radioactivity through soils, and potential dilution in water is performed under SRP Section 2.4.13. Briefly, these sections of the SRP address information describing streams and lakes, regional and local ground water aquifers, sources, and sinks, local and regional ground water users, known and likely future withdrawal rates, regional flow rates, travel time, gradients, and velocities, subsurface properties that affect movement of contaminants in ground water, ground water levels including their seasonal and climatic fluctuations, ground water monitoring and protection requirements, man-made changes that may affect regional ground water characteristics over time, and local practices in using water resources.					
BTP11-6.2	Radioactive Source Term The proposed radionuclide concentrations assumed for the postulated failure of a tank and its components will be reviewed by the staff using the information presented by the applicant. The analysis assumes that a tank and its components fail to meet the design bases as required by 10 CFR Part 50.34a, and General Design Criteria 60 and 61. The staff will evaluate the basis and assumptions used in developing the source terms, radionuclide distributions and concentrations to ensure that the highest potential radioactive material inventory is selected among the					

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	expected types of liquid and wet waste streams processed by the LWMS. The radionuclide inventory for the tank and its components assumed to fail is based on 80% of the volume capacity of that tank and its component. The radionuclides selected for the radioactive source term and total inventory should include those that have the highest potential exposure consequences to users of water resources, including long-lived fission and activation products and environmentally mobile radionuclides. The radionuclide concentrations and total inventory of radioactive materials is based on the expected failed fuel fraction, i.e., 0.12% of the fuel producing power in a pressurized water reactor (PWR) as per NUREG-0017, or consistent with an offgas release rate of 0.555 MBq/sec per MWt (15 µCi/sec per MWt) after a 30-minute delay for a boiling water reactor (BWR) as per NUREG-0016. The radionuclide inventory in failed components is calculated based on the methods given in Chapter 4 and Appendices A and B of NUREG-0133, or by using equivalently documented techniques. The staff will confirm that the initial inventory of radioactive materials corresponds to the highest expected concentrations and inventory of radioactivity in systems and components used to process, treat, or store liquid and wet wastes products associated with normal operation and anticipated operational occurrences, The reviewer will determine whether the tank and its components, for which a failure is assumed, will result in the highest concentrations of radioactive materials at the nearest potable water supply located in an unrestricted area.					
BTP11-6.3	Mitigating Design Features The staff will determine whether the analysis has considered the					

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	use of design features, e.g., steel liners or walls in areas housing components, dikes for outdoor tanks, and overflow provisions incorporated to mitigate the effect of a postulated tank failure. The types of failed components are typically waste collector tanks or sample tank, among others. However, the components selected for the analysis should realistically reflect the specific design features of the plant, as described in Sections 11.2 and 11.4 of the application. The staff will coordinate this part of the evaluation with the organization responsible for the review of systems and components that are part of the balance of plant. The purpose of this review is to ensure that the analysis considered the proper selection of the failed equipment, and appropriate release mechanisms from the selected equipment and buildings housing such systems. Credit for liquid retention by unlined building foundations will not be given regardless of the building seismic category because of the potential for cracks. Credit is not allowed for retention by coatings or leakage barriers outside the building foundation.					
BTP11-6.4	Specifications on Tank Waste Radioactivity Concentration Levels The reviewer will evaluate the proposed technical specification limiting the radioactivity content (becquerel, curie) of liquid-containing tanks to ensure that the technical specification is consistent with the safety evaluation. Chapter 16 of the SRP identifies the requirements for this technical specification. The radioactivity content (becquerel, curie) is based on that quantity which would not exceed the concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply, located in an unrestricted area, in the event of an uncontrolled release of the tank's contents.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	CHAPTER 12, Radiation Protection					
12.1, Rev. 3 (03/2007)	Assuring that Occupational Radiation Exposures Are As Low as is Reasonably Achievable					
12.1.1	Policy Considerations. Acceptability will be based on evidence that a policy for ensuring that ORE will be ALARA has been formulated in accordance with the training requirements in 10 CFR 19.12 and the ALARA provisions of 10 CFR 20.1101(b), and that the policy has been described, displayed, and will be implemented in accordance with the provisions of Regulatory Guides 8.8 (Regulatory Position C.1) and 8.10 (Regulatory Position C.1) and NUREG-1736, as it relates to maintaining doses ALARA. A specific individual(s) will be designated and assigned responsibility and authority for implementing ALARA policy. Alternative proposed policies will be evaluated on the basis of a comparison with the above regulatory guides and NUREG-1736.					
12.1.2	Design Considerations. Acceptability will be based on evidence that the design methods, approach, and interactions are in accordance with the ALARA provisions of 10 CFR 20.1101(b) and Regulatory Guide 8.8 (Regulatory Position C.2) and will include incorporation of measures for reducing the need for time spent in radiation areas; maintenance; measures to improve the accessibility to components requiring periodic maintenance or inservice inspection; measures to reduce the production, distribution, and retention of activated corrosion products throughout the primary system; measures for assuring that ORE during decommissioning will be ALARA; reviews of the design by competent radiation protection personnel; instructions to designers and engineers regarding ALARA design; experience from operating plants and past designs; and continuing facility design reviews. Alternative proposed design policies will be evaluated on the basis of a comparison with the design guidance in Regulatory Guide 8.8 (Regulatory Position C.2).					
12.1.3	Operational Considerations. Acceptability will be based on					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	evidence that the applicant has a program to develop plans and procedures in accordance with Regulatory Guides 1.33, 1.8, 8.8, and 8.10 that can incorporate the experiences obtained from facility operation into facility and equipment design and operations planning and that will implement specific exposure control techniques.					
12.1.4	Radiation Protection Considerations. Acceptability will be based on evidence that overall facility operations, as well as the radiation protection program, integrate the procedures necessary to ensure that radiation doses are ALARA, including work scheduling, work planning, design modifications, and radiological considerations.					
12.2, Rev. 3 (03/2007)	Radiation Sources					
12.2.1	Regulatory Guide 1.3[2], as it relates to assumptions used in evaluating gaseous concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident, for boiling-water reactors (BWRs).					
12.2.2	Regulatory Guide 1.4[2], as it relates to assumptions used in evaluating gaseous concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident, for pressurized-water reactors (PWRs).					
12.2.3	Regulatory Guide 1.183[3], as it relates to the assumptions used in evaluating the concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident.					
12.2.4	Regulatory Guide 1.7, as it relates to methods for determining gaseous concentrations of radionuclides in containment following an accident.					
12.2.5	Regulatory Guide 1.112, as it relates to complying with the Commission's regulations under 10 CFR 20.1301 concerning the calculation of realistic radiation levels and radioactive materials source terms for the evaluation of waste treatment systems.					
12.2.6	NUREG-0737, Task Action Plan Item II.B.2, as it relates to the identification of specific postaccident sources of radiation in the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	facility.					
12.2.7	American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 18.1, as it relates to the establishment of typical long-term concentrations of principal radionuclides in fluid streams of light-water-cooled nuclear power plants.					
	Compliance with the following specific acceptance criteria is necessary to meet the relevant requirements of the regulations identified above. Descriptions should be provided for all radiation sources that require (1) shielding, (2) special ventilation systems, (3) special storage locations and conditions, (4) traffic or access control, (5) special plans or procedures, or (6) monitoring equipment. The source descriptions should include all pertinent information required for (1) input to shielding codes used in the design process, (2) establishment of related facility design features, (3) development of plans and procedures, and (4) assessment of occupational exposure. For contained sources, the description should include plan scale drawings of each floor of the plant that show all sources identified so that they can easily be related to tables containing the pertinent and necessary quantitative source parameters. Their position should be located accurately, indicating the approximate size and shape. Neutron and gamma streaming into containment from the annulus between the reactor pressure vessel and the biological shield should be analyzed to determine the radiation fields that could occur in areas that may require occupancy. Relevant experience from operating reactors may be used. Airborne sources that are created by leakage, opening formerly closed containers, storage of leaking fuel elements, and other mechanisms should be identified by location and magnitude so that they can be used for designing appropriate ventilation systems and in specifying appropriate monitoring systems.					

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	Airborne radioactivity concentrations in frequently occupied areas should be a small fraction of the concentrations related to 10 CFR 20.1203, 10 CFR 20.1204, and Appendix B to 10 CFR Part 20. The assumptions made in arriving at quantitative values for these various sources should be specified, either in this section or by reference to SAR Chapter 11. Shielding and ventilation design fission product source terms will be acceptable if developed using these bases: • An offgas rate of 370 MBq/s (100,000 µCi/s) after a 30-minute delay for BWRs. • 0.25-percent fuel cladding defects for PWRs. • Postaccident shielding (for vital area access, including work in the area) source terms from NUREG-0737, Item II.B.2, or Regulatory Guide 1.183. Coolant and corrosion activation products source terms should be based on applicable reactor operating experience. The buildup of activated corrosion products in various components and systems should be addressed. Any allowances made in design source terms for the buildup of activated corrosion products should be explained. Neutron and prompt gamma source terms should be based on reactor core physics calculations and applicable reactor operating experience. The tables of source parameters, which can be placed in SAR Chapter 12 or referenced to SAR Chapter 11, will be acceptable if the accompanying text either in this section or other referenced sections makes it clear how the values are used in a shield design calculation or in a ventilation system design. In addition, the quantities will be acceptable if the specific values given in the tables are consistent with ANSI/ANS Standard 18.1 and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Regulatory Guide 1.112 for coolant and corrosion activation products source terms. For PWRs designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations in the regions specified in item I.2 above should be based on a primary coolant concentration of 1.3x10 Bq/gm (3.5 µCi/gm).					
	NOTES: 2. Regulatory Guides 1.3 and 1.4 provide guidance related to Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." This guidance is applicable to a holder of an operating license issued prior to January 10, 1997 or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997. These license holders may voluntarily revise the accident source term. 3. Regulatory Guide1.183 is applicable to applicants or license holders issued after January 10, 1997.					
12.3-12.4, Rev. 3 (03/2007)	Radiation Protection Design Features					
	The following regulatory guides, NUREGs, and industry standards provide information, recommendations, and guidance and in general describe a basis acceptable to the staff for implementing the requirements of the regulations identified above: 1. Regulatory Guide 1.3[2], as it relates to assumptions used in evaluating gaseous concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident for boiling-water reactors (BWRs). 2. Regulatory Guide 1.4[2], as it relates to assumptions used in evaluating gaseous concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident for pressurized-water reactors (PWRs).					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	3. Regulatory Guide 1.7, as it relates to methods for determining gaseous radionuclides in containment following an accident. 4. Regulatory Guide 1.52, as it relates to radiation protection considerations for engineered safety feature (ESF) atmosphere cleanup systems operable under postulated design-basis accident (DBA) conditions, to be designated as “primary systems.” 5. Regulatory Guide 1.69, as it relates to the requirements and recommended practices acceptable for construction of facilities that apply to occupational radiation protection shielding structures for nuclear power plants. 6. Regulatory Guide 1.97, Revision 3, as it relates to a method acceptable to the staff for complying with the Commission’s regulations to provide instrumentation for radiation monitoring following an accident in a light-water-cooled nuclear power plant. 7. Regulatory Guide 1.183[3], as it relates to the assumptions and methods for evaluating doses to individuals accessing the facility during and following an accident in accordance with NUREG-0737, item II.B.2. 8. Regulatory Guide 8.2, as it relates to general information on radiation monitoring programs for administrative personnel. 9. Regulatory Guide 8.8, as it relates to actions taken during facility design, engineering, construction, operation, and decommissioning to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, concerning the radiation protection information to be supplied in SAR Section 12. 10. Regulatory Guide 8.10, as it relates to the commitment by management and vigilance by the radiation protection manager and staff to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003 .					

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	11. Regulatory Guide 8.19, as it relates to a method acceptable to the staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process so that such exposures will be ALARA. 12. Regulatory Guide 8.25, as it relates to a method acceptable to the staff for continuous monitoring for airborne radioactive materials in plant spaces. 13. Regulatory Guide 8.38, as it relates to the physical controls for personnel access to high and very high radiation areas. 14. NUREG-1430, as it relates to radiation protection considerations in the applicability, format, and implementation of the Babcock and Wilcox Technical Specification package. 15. NUREG-1433, as it relates to radiation protection considerations in the applicability, format, and implementation of the General Electric Technical Specification package. 16. NUREG-1434, as it relates to radiation protection considerations in the applicability, format, and implementation of the General Electric Technical Specification package. 17. NUREG-1432, as it relates to radiation protection considerations in the applicability, format, and implementation of the Combustion Engineering Technical Specification package. 18. NUREG-1431, as it relates to radiation protection considerations in the applicability, format, and implementation of the Westinghouse Technical Specification package. 19. ANSI/ANS-HPSSC-6.8.1-1981, as it relates to criteria for the establishment of locations for fixed continuous area gamma radiation monitors and for design features and ranges of measurement. 20. ANSI N13.1-1999, as it relates to the principles that apply in					

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	obtaining valid samples of airborne radioactive materials, and acceptable methods and materials for gas and particle sampling. 21. ANSI/ANS-6.4-1997 (R2004), as it relates to requirements and recommended practices for the construction of concrete radiation shielding structures. 22. Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, dated October 10, 2006, and NUREG/CR-3587, as they relate to the design issues that need to be addressed to meet the requirements of 10 CFR 20.1406. NOTES: 2. Regulatory Guides 1.3 and 1.4 provide guidance related to Technical Information Document (TID) 14844, “Calculation of Distance Factors for Power and Test Reactor Sites.” This guidance is applicable to a holder of an operating license issued prior to January 10, 1997 or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997. These license holders may voluntarily revise the accident source term. 3. Regulatory Guide1.183 is applicable to applicants or license holders issued after January 10, 1997.					
12.3-12.4.1	Facility Design Features The acceptability of the facility design features will be based on evidence that the applicant has fulfilled the dose limiting requirements of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, and 10 CFR 20.1207, as well as the radiation protection aspects of GDC 19 and 61, and 10 CFR 50.34. This includes radioactive material handling and processing, inservice inspection, calibration, decommissioning, and recovery from accidents) have been considered in plant design and that radiation protection features incorporated into the design will keep					

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	potential radiation exposure from these activities ALARA in accordance with 10 CFR 20.1101(b), the definition of ALARA in 10 CFR 20.1003, and Regulatory Guides 8.8 and 8.10. Such features may include (1) the ease of accessibility to work, inspection, and sampling areas, (2) the ability to reduce source intensity, (3) design measures to reduce the production, distribution, and retention of activated corrosion products, (4) the ability to reduce time required in radiation fields, and (5) a provision for portable shielding and remote handling tools. Access control will be judged for acceptability in accordance with the requirements of 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903 or access control alternatives in Standard Technical Specifications (NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433, and NUREG-1434). Facility design, to the extent practicable, should minimize the potential for creating a very high radiation area during normal operations, including abnormal operational occurrences (such as dropping a fuel bundle during fuel handling operations). High and very high radiation areas should be remote from normally occupied rooms and corridors such that personnel access to these areas can be controlled in accordance with 10 CFR 20.1601 and 10 CFR 20.1602 and the guidance in Regulatory Guide 8.38. All accessible portions of the spent fuel transfer tube or canal that are capable of having radiation levels greater than 1 gray (Gy) per hour (100 rads per hour) should be shielded during fuel transfer. This shielding should be such that the resultant contact radiation levels are no greater than 1 Gy per hour (100 rads per hour). All accessible portions of the spent fuel transfer tube are clearly marked with a sign stating that potentially lethal radiation fields are possible during fuel transfer. If removable shielding is used for the fuel transfer tubes, it must also be explicitly marked as above. If other than permanent shielding is used, local audible and visible alarming radiation monitors must					

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	be installed to alert personnel if temporary fuel transfer tube shielding is removed during fuel transfer operations. Similar precautions should also apply to any other plant radiation source having radiation levels greater than 1 Gy per hour (100 rads per hour). The areas inside the plant structures, as well as in the general plant yard, should be subdivided into radiation zones, with maximum design dose rate zones and the criteria used in selecting maximum dose rates identified. Maximum zone dose rates should be defined for each zone, depending on anticipated occupancy and access control. The areas that must be occupied on a predictable basis (based on the number of people and stay or transit times) during normal operations and anticipated operational occurrences (including refueling; purging; fuel handling and storage; radioactive material handling; processing, use, storage, and disposal; normal maintenance; routine operational surveillance; inservice inspection; and calibration) should be zoned such that this occupancy results in an annual dose to each of the involved individuals that is as far below the limits of 10 CFR Part 20 as is reasonably achievable, and a total personsievert (person-rem) dose that is ALARA. Based on current operating experience and on predictions being made for new plant designs, it is expected that the plant shielding can be designed, the plant can be zoned, and sufficient radiation protection design features can be incorporated, such that individuals in shielded areas would receive a small fraction of the 10 CFR Part 20 limits. All vital areas, in which radiation may unduly limit personnel occupancy during operations following an accident resulting in a degraded core, should be identified. Personnel access to these areas under accident conditions should be demonstrated in accordance with 10 CFR 50.34(f)(2)(vii), using the methods listed					

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	in Section II.B.2 of NUREG-0737. The analysis should consider access to, stay time in, and egress from these vital areas.					
12.3-12.4.2	Shielding The staff will evaluate the shielding design in terms of the assumptions used to calculate shield thickness, the calculational methods used, and the parameters chosen. A number of acceptable shielding calculational codes are available that are effective for determining the necessary shield thickness for gamma ray and combination neutrongamma sources. The code description file of the Radiation Safety Information Computational Center (formerly the Radiation Shielding Information Center) at Oak Ridge National Laboratory includes most of the codes used by shield designers, which means that the codes have been tested and authenticated for operation but not for reliability and accuracy. Radiation shielding codes vary in complexity and accuracy from the relatively simple point-kernel methods, to the more complex discrete ordinates methods, to the still more rigorous Monte Carlo methods. The staff may use these codes, as necessary, to calculate dose rates for given shield designs and source strengths as a confirmation of the applicant's method. The applicant's shielding design is acceptable if the methods are comparable to commonly accepted shielding calculations and if assumptions regarding source terms, cross sections, shield and source geometries, and transport methods are realistic. Labyrinth shielded access ways and penetrations should be used to minimize radiation steaming and scatter around shields. Composition of the shielding material should be selected to minimize, to the extent practicable, the potential for the shield itself to become a radiation source (either from activation of the shield material or production of secondary radiation resulting from interactions with the primary radiation). Effective shield design is essential to meeting the criteria that ORE will be ALARA.					

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	In addition, Regulatory Guide 1.69 and ANSI/ANS-6.4-1997 provide guidance on the fabrication and installation of concrete shields for occupational radiation protection at nuclear power plants. Acceptability of the shield construction will be based on an indication that the guidance of these documents have been implemented in facility construction, or that acceptable alternatives have been proposed. Regulatory Guide 8.8 provides additional acceptance criteria regarding shielding and isolation in radiation protection design.					
12.3-12.4.3	Ventilation The ventilation system will be acceptable for radiation protection purposes if the criteria and bases for ventilation rates within the areas covered in SAR Section 12.2.2 will ensure that air will flow from areas of low potential airborne radioactivity to areas of higher airborne radioactivity and then to filters or vents, that the concentrations of radioactive material in areas normally occupied can be maintained in accordance with the requirements 10 CFR 20.1701, and that the dose limits of 10 CFR 20.1201 are met consistent with the requirements of 10 CFR 20.1202, 10 CFR 20.1203, and 10 CFR 20.1204. The system has adequate capability to reduce concentrations of airborne radioactivity to 1.0 derived air concentration (DAC), as specified in Appendix B to 10 CFR Part 20, in areas not normally occupied where maintenance or inservice inspection must be performed. The system is designed so that filters containing radioactivity can be easily maintained and will not create an additional radiation hazard to personnel maintaining them, or those in adjacent occupied areas. Acceptability of the ventilation system, relative to radioactive gases and particulates, will also be based on evidence that the applicant has applied the guidance of Regulatory Guide 8.8 or proposed acceptable alternatives.					

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	Regulatory Guide 1.52, particularly Sections C.3.10 and 4.10, provides guidance that can be used in this review, although the guide relates to mitigating accidents involving airborne radioactivity. Good practices in that regard apply to normal operation as well, since the release of radioactivity in normal operational occurrences is usually different only in quantity from some of the accident cases.					
12.3-12.4.4	Area Radiation and Airborne Radioactivity Monitoring Systems A. The area radiation monitoring systems will be acceptable if they meet the provisions of 10 CFR 20.1501, 10 CFR 50.34(f)(2)(xvii); the guidance in NUREG-0737, Regulatory Guide 8.25, and Regulatory Guide 1.97, Revision 3; and the following criteria: i. The detectors are located in areas that normally may be occupied without restricted access and that may have a potential for radiation fields in excess of the radiation zone designations discussed in the third paragraph under item 1, above, in accordance with ANSI/ANS-HPSSC-6.8.1. ii. The detectors provide on-scale readings of dose rate that include the design maximum dose rate of the radiation zone in which they are located as well as the maximum dose rate for anticipated operational occurrences and accidents. iii. The detectors are calibrated during fuel outages and after the performance of any maintenance work on the detector. iv. Each monitor has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms. v. Readout and annunciation are provided in the control room.					

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	vi. The in-containment high-range radiation monitors meet the criteria of 10 CFR 50.34(f)(2)(xvii). vii. Emergency power is initiated after a loss of offsite power. B. The airborne radioactivity monitoring system will be acceptable if it is consistent with the guidance on continuous air sampling in Regulatory Guide 8.25 and meets the following criteria: i. Engineering controls provide the principal protection against the intake of radioactive materials. ii. Air should be sampled at normally occupied locations where airborne radioactivity may exist, such as solid waste handling areas, spent fuel pools, reactor operating floors, and BWR turbine buildings. The monitoring system should be capable of detecting 10 DAC-hours of particulate and iodine radioactivity from any compartment that has a possibility of containing airborne radioactivity and that normally may be occupied by personnel, taking into account dilution in the ventilation system. Continuous monitoring of air being exhausted from locations within the facility during normal operation is an acceptable method. Noble gas monitors should be calibrated such that, when monitoring for ^{133}Xe, the instrument response will determine concentrations accurately. iii. Representative air concentrations are measured at the detectors, which are located as close to the sampler intakes as possible. iv. Ventilation monitors are upstream of high-efficiency particulate air filters. v. The detectors are calibrated routinely and after any maintenance work is performed on the detector.					

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	vi. Each location has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms. vii. Readout and annunciation are provided in the control room. viii. Emergency power is initiated after a loss of offsite power. C The in-plant accident radiation monitoring systems will be acceptable if they meet the following criteria: i. Personnel have the capability to assess the radiation hazard in areas that may be accessed during the course of an accident, in accordance with the criteria of 10 CFR 50.34(f)(2)(xvii); NUREG-0737, item II.F.1; and Regulatory Guide 1.97, Revision 3. ii. Portable instruments to be used in the event of an accident should be placed so as to be readily available to personnel responding to an emergency. iii. Emergency power should be provided for installed accident monitoring systems. iv. The accident monitoring systems should have usable ranges that include the maximum calculated accident levels and should be designed to operate properly in the environment caused by the accident. v. Two high-range radiation monitors are provided in containment in accordance with the requirements of 10 CFR 50.34(f)(2)(xvii) and item II.F.1 of NUREG-0737. D. Appendix A to Regulatory Guide 1.21 provides useful guidance about effluent monitoring that applies to the acceptability of in-plant airborne radioactivity monitoring. Regulatory Guide 8.2 includes guidance on surveys to evaluate radiation hazards. The detailed guidance in ANSI					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	N13.1-1999 covers the sampling of airborne radioactive materials in ventilation ducts and stacks of nuclear facilities and may be used for acceptance criteria on the actual sampling process and certain techniques involved. Regulatory Guide 8.8 provides further guidance on monitoring systems. E Instrumentation for monitoring areas where reactor fuel is stored or handled will be acceptable if it meets the criteria of 10 CFR 50.68.					
12.3-12.4.5	Dose Assessment The dose assessment will be acceptable if it documents in appropriate detail the assumptions made, calculations used, results for each radiation zone (including numbers and types of workers involved in each), expected and design dose rates, and projected person-Sievert (person-rem) doses, in accordance with Regulatory Guide 8.19.					
12.5, Rev. 3 (03/2007)	Operational Radiation Protection Program					
	The following regulatory guides, NUREGs, and industry standards provide information, recommendations, and guidance and in general describe a basis acceptable to the staff to implement the requirements of 10 CFR Part 19, 10 CFR Part 20, and 10 CFR Part 50: 1. Regulatory Guide 1.8, as it relates to compliance with the Commission's regulations regarding qualification of nuclear power plant personnel 2. Regulatory Guide 1.33, as it relates to compliance with the Commission's quality assurance regulatory requirements during nuclear power plant operations. 3. Revision 3 of Regulatory Guide 1.97, as it relates to compliance with the Commission's regulations to provide					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	instrumentation to monitor plant variables and systems during and following an accident. 4. Regulatory Guide 8.2, as it relates to general information on radiation monitoring programs for administrative personnel. 5. Regulatory Guide 8.4, as it relates to standards for direct-reading and indirect-reading pocket dosimeters used for personnel dose or dose rate measurements. 6. Regulatory Guide 8.6, as it relates to testing the operating characteristics of Geiger-Mueller counters before making calibrations and measurements. 7. Regulatory Guide 8.7, as it relates to the specification of records necessary to describe the ORE of individuals and to the conditions under which the exposure may occur. 8. Regulatory Guide 8.8, as it relates to meeting the requirements of 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003 by providing radiation protection information pertaining to actions taken during the design, construction, operation, and decommissioning to ensure that ORE remains ALARA. 9. Regulatory Guide 8.9, as it relates to appropriate concepts, models, equations, and assumptions to be used in determining the extent of an individual's intake of radioactive materials and resulting committed organ dose. 10. Regulatory Guide 8.10, as it relates to meeting the requirements of 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003 concerning commitment by the applicant's management and vigilance by the radiation protection manager and the radiation protection staff to maintain ORE ALARA. 11. Regulatory Guide 8.13, as it relates to the description of the instruction to be provided concerning biological risks to embryos or fetuses resulting from prenatal ORE. 12. Regulatory Guide 8.15, as it relates to elements of acceptable respiratory protection programs.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	13. Regulatory Guide 8.20, as it relates to the development and implementation of a bioassay program for any licensee handling or processing of iodine-125 or iodine-131. 14. Regulatory Guide 8.25, as it relates to monitoring the levels of airborne radioactivity within the facility. 15. Regulatory Guide 8.26, as it relates to bases used by the NRC staff in evaluating the need for license provisions on bioassay programs for workers subject to internal radiation exposure from the inhalation or ingestion of licensed materials. 16. Regulatory Guide 8.27, as it relates to a radiation protection training and retraining program consistent with the ALARA objective and acceptable to the NRC staff for meeting the training requirement of 10 CFR Part 19. 17. Regulatory Guide 8.28, as it relates to the appropriate use of audible alarm dosimeters and the conditions under which they should not be relied on to perform their intended function. 18. Regulatory Guide 8.29, as it relates to providing appropriate instruction on the risks associated with ORE to individuals who might be exposed that are acceptable to the NRC staff for meeting the training requirement of 10 CFR Part 19. 19. Regulatory Guide 8.32, as it relates to monitoring individuals for exposure to tritium. 20. Regulatory Guide 8.34, as it relates to criteria acceptable to the NRC staff that licensees may use to determine when monitoring is required, as well as methods acceptable to the NRC staff for calculating occupational doses when intake is known. 21. Regulatory Guide 8.35, as it relates to guidance on the conditions and prerequisites for permitting planned special exposures, as allowed by 10 CFR Part 20, and the associated specific monitoring and reporting requirements. 22. Regulatory Guide 8.36, as it relates to determination of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	total radiation dose to the embryo/fetus as the sum of the deep-dose equivalent to, and dose to the embryo/fetus from, intakes of the declared pregnant worker. 23. Regulatory Guide 8.38, as it relates to guidance on acceptable methods to control access to high- and very-high-radiation areas in nuclear power plants that follows the requirements specified in 10 CFR Part 20. 24. NUREG-0041, as it relates to the provision of technical information to licensees on the appropriate application of respiratory protective devices for protection against airborne radioactive materials, including selection and maintenance of equipment and personnel training. 25. NUREG-0731, as it relates to appropriate staffing levels and technical expertise considered essential within a utility to support nuclear power plant operation properly. 26. NUREG-1736, as it relates to the requirements for a radiation protection program (including program review and audit) and compliance with 10 CFR Part 20. 27. American National Standards Institute (ANSI)/American Nuclear Society (ANS) 3.1-1978, reaffirmed 1999, as it relates to criteria for selection, qualifications, responsibilities, and training of personnel in operating and support organizations, as appropriate for the safe and efficient operation of nuclear power plants. 28. ANSI N13.6-1999, as it relates to guidance to the employer for the systematic generation and retention of records relating to ORE. 29. ANSI/Health Physics Society (HPS) N13.11-2001, as it relates to the performance criteria for personal radiation dosimeters that require processing. 30. ANSI/HPS N13.14-1994, as it relates to personnel monitoring. 31. ANSI/HPS N13.30-1996, as it relates to detection and dosimetry of internally deposited radionuclides.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	32. ANSI/HPS N13.42-1997, as it relates to monitoring radiation dose from internally deposited radionuclides. 33. ANSI Institute for Electrical and Electronics Engineers (IEEE) 309-1991, as it relates to guidance on specification of test conditions - such as associated electronic circuitry, environment, and counting rate - to ensure that operating characteristics can be appropriately evaluated 34. ANSI N42.20-2003, as it relates to the accuracy and overall performance of personnel radiation monitors 35. ANSI N42.17A-1989, as it relates to the accuracy and overall performance of portable survey instruments 36. ANSI N323A-1997, as it relates to the calibration and maintenance of portable radiation survey instruments 37. Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, dated October 10, 2006, and NUREG/CR-3587, as they relate to operating programs that facilitate decommissioning.					
12.5.1	Organization Acceptance will be based on a determination that the organization described, and the duties, qualifications, and training of the individuals responsible for ensuring that ORE will be ALARA; (1) are in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1103; Regulatory Guides 1.8, 8.2, 8.8, and 8.10; and 10 CFR 19.12; and (2) are such that doses resulting from licensed activities fall within the limits of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1301, 10 CFR 20.1302, 10 CFR 50.120, NUREG-0731, and NUREG-1736. Alternatives will be evaluated on the basis of a comparison with the referenced regulatory guides.					
12.5.2	Equipment, Instrumentation, and Facilities Acceptance will be based on a determination of the following:					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	A. Sufficient sampling and analysis capabilities for reactor coolant and containment samples are available during normal and accident conditions, consistent with 10 CFR 50.34(f)(2)(viii). B. The radiochemistry laboratory is equipped to perform the routine analyses required for personnel protection, surveys, and related radiation protection functions, in accordance with 10 CFR 20.1501. C. The counting room (low background) has the necessary instrumentation to perform routine counting on all plant radioactivity samples (e.g., water, air, swipes) in conformance with 10 CFR 20.1501 and with GDC 64 in Appendix A to 10 CFR Part 50. Counting room equipment normally includes the following: i. Radionuclide spectrometry equipment (such as a multichannel gamma pulse height analyzer). ii. Low-background alpha-beta proportional counter and gamma and alpha-beta scintillation counters. iii. End-window Geiger-Mueller type counter. D. Instruments for measuring radiation or radioactivity in accordance with 10 CFR 20.1501 normally include the following: i. Portable low- and high-range ion chamber rate meters (see Revision 3 of ii. Regulatory Guide 1.97 for ranges). iii. Portable Geiger-Mueller counters. iv. Portable alpha scintillation or proportional counter rate meters. v. Portable neutron dose equivalent rate meters. vi. Fixed and portable air samplers for use with particulate filters and iodine collection devices (such as charcoal cartridges or equivalent filters) and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	airborne radioactivity monitors. vii. High-range instruments, in accordance with Revision 3 of Regulatory Guide 1.97. viii. Fixed area monitors with local and remote readouts and alarm functions. ix. Small item contamination (i.e., box) counters. E. Personnel monitoring instruments in accordance with 10 CFR 20.1501 and 10 CFR 20.1502 include the following: i. Personnel contamination monitors (e.g., friskers, hand-and-foot monitors, standup portal monitors). ii. Self-reading low and intermediate pocket dosimeters, including audible alarm dosimeters (for early evaluation of individual doses). Performance and other requirements conform to Regulatory Guides 8.4 and 8.28 or to appropriate proposed alternatives. iii. Remote and local reading alarm dosimeters (coupled with direct or electronic surveillance equipment) for monitoring workers in highdose/ high-dose-rate environments. iv. Personal dosimeters (e.g., film badges, thermoluminescent dosimeters (TLD), ocularly stimulated dosimeters) of sufficient range and sensitivity that are processed and evaluated by a processor accredited by the National Voluntary Laboratory Accreditation Program (NVLAP), as appropriate, in conformance with 10 CFR 20.1501(c). v. Provisions for bioassays (in vivo and in vitro as appropriate) and facilities capable of detecting intakes of expected radionuclides (e.g., mixed fission and activation products, tritium, and alpha-emitting nuclides) to meet the requirements of 10 CFR 20.1204 and Regulatory Guides 8.9, 8.20, 8.26, and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	8.32 or to satisfy appropriate proposed alternatives. F. Utility-issued personnel protection equipment include the following: i. Anticontamination clothing. ii. Plastic suits for contamination control in wet work environments. iii. Head covers, shoe covers, gloves, face shields, and safety-related items (including provisions for personnel cooling in high-temperature work environments). iv. Pressure demand (e.g., full-facepiece) air line respirators. v. Pressure demand self-contained breathing apparatus. vi. Air purifying respirators (e.g., full-face negative pressure, powered air purifying). vii. Respiratory protection equipment and facilities that meet the requirements of 10 CFR 20.1703. viii. Work efficiency equipment (e.g., ice vests, air-supplied suits, or other heat stress coping equipment). G. At a minimum, the following radiation protection support facilities or areas will be provided: i. Portable instrument calibration and storage area. The latter should be easily accessible. ii. Personnel decontamination area with necessary monitoring equipment. This facility should be located and designed to expedite rapid cleanup of male and female personnel and should not be used as a multiple-purpose area. iii. Facility and equipment to clean, sanitize, repair, and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	decontaminate personnel protective equipment, monitoring instruments, respirators, and associated equipment. iv. A change room for donning protective clothing (i.e., anticontamination suits) and storage of personal items. v. Control points for entrance into, or exit from, controlled access areas of the plant, condition signs, labels, and signals, in accordance with 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, 10 CFR 20.1903, 10 CFR 20.1904, and 10 CFR 20.1905. vi. Storage and control capability for licensed materials in unrestricted areas, in accordance with 10 CFR 20.1801 and 10 CFR 20.1906. vii. One or more radiation protection stations, which may be used as locations for storage and issuance of portable radiation survey equipment, respiratory protective equipment, personnel monitoring equipment, and contamination control supplies. The equipment should be readily accessible, and the stations should be equipped to facilitate communication throughout the plant. viii. Training facilities for conducting general employee training, health physics technician hands-on practical factors exercises, and prework ALARA mockup training. ix. Radiation work control stations (and/or remote surveillance facilities) for overseeing work in high-radiation and very-high-radiation areas. H. Special shields and equipment include the following: i. Lead blankets ii. Remote tools and handling equipment					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	iii. Portable ventilation equipment Acceptance will also be based on implementation of the guidance of Regulatory Guide 8.8 or the provision of acceptable alternatives.					
	CHAPTER 13, Conduct of Operations					
13.1.1, Rev. 5 (03/2007)	Management and Technical Support Organization					
13.1.1.1	General Requirements In reviewing and evaluating this SAR section, the following points should be considered. A. The corporate level management and technical support structures should be free of ambiguous assignments of primary responsibility, as demonstrated by organizational charts and descriptions of functions and responsibilities. B. The corporate officer responsible for nuclear activities should be identified and have no ancillary responsibilities that might detract attention from nuclear safety matters. C. Design and construction responsibilities should be reasonably well defined in both numbers and experience of persons required to implement the project. D. Similarly, management and organizational responsibilities should be clearly defined to address human factors engineering (HFE) considerations in human-system interface issues. This subject is covered in more detail in NUREG-0711 and in SRP Chapter 18.					
13.1.1.2	Specific Requirements Specific criteria are described below for meeting 10 CFR 50.40(b) with respect to the CP, OL, COL reviews and 10 CFR 50.80 with					

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	respect to license transfer reviews. A. CPs and COLs i. The applicant has identified and functionally described the organizational groups responsible for implementing the project. ii. The applicant has described how it will carry out its responsibilities to consider safety first in designing and construct the project and during the transition to operation and to control major contractors iii. The organizational units involved in the design and construction of the project communicate fully and frankly among each other and with management, and management clearly and unambiguously controls the project. iv. Manpower with suitable experience is available to implement the project. v. The applicant has clearly described the role and function of the AE and the NSSS vendor during both design and construction and has demonstrated appropriate control over the project-related activities of the AE and NSSS vendor. vi. The applicant has designated the organizations responsible for the test program, and early plans give reasonable assurance that the designated organizations can collectively provide staff with the skills and experience necessary to develop and conduct the test program. vii. The applicant plans to utilize the plant operating and technical staff in developing and conducting the test program and in reviewing test results. viii. For COL applicants subject to 10 CFR 50.34(f)[1], the applicant has identified plans for the organization and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	staffing to oversee design and construction of the nuclear facility, in accordance with the guidelines of Item II.J.3.1 of NUREG-0718, as related to the requirements of 10 CFR 50.34(f)(3)(vii). As referenced in SRP Section 18.0, the review criteria for the human factors engineering (HFE) design team are provided in NUREG-0711, Chapter 2, "Element 1 - HFE Program Management." B. For OL or COL Holders The review and evaluation of management and technical organizational structure for OL and COL applicants is based on the guidelines of Three Mile Island (TMI) Action Plan Item I.B.1.2, originally described in NUREG-0694. Specific criteria are as follows: i. The applicant has identified and described the organizational groups responsible for implementing the initial test program and providing technical support for the operation of the facility. ii. The applicant has described how it will carry out its responsibilities to conduct the initial test program, provide sufficient technical support, and safely operate of the facility. iii. The1 organizational structure provides for integrated management of activities that support the operation and maintenance of the facility. iv. Clear management control and effective lines of authority and communications exist among the organizational units involved in managing, operating, and providing technical support for the facility. v. Manpower with suitable experience is available to conduct the initial test program and provide technical					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	support for the operation of the facility. The need to add experienced personnel to the corporate structure during the initial years of operation will be determined on case-by-case basis. vi. Qualifications of members of the technical support organization should meet or exceed those endorsed by Regulatory Guide 1.8. vii. The technical staff will be utilized in the initial test program to the maximum extent practicable. Before testing begins, participants in the test program should receive plant-specific training on the administrative controls for the test program. The level of staffing should be adequate in the reviewer's judgment. C. Reviews of OL Transfers Following are criteria for reviewing the management and technical-support organizational structures of license transfer applicants are as follows: i. The applicant has identified and described the organizations responsible for the technical support for the operation of the facility. ii. The applicant has described how it will obtain the necessary technical support. iii. The organizational structure provides for integrated management of activities that support the operation and maintenance of the facility. iv. There is clear management control of the organizational units involved in operating and providing technical support for the facility, and there are clear lines of authority between management and these groups and effective communications among them and with management.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	v. Manpower with sufficient experience is available to provide the technical support for the operation of the facility. vi. The qualifications of members of the technical support organization meet or exceed those endorsed by Regulatory Guide 1.8. NOTE: 1. For Part 50 applicants not listed in 10 CFR 50.34(f), the provisions of 50.34(f) will be made a requirement during the licensing process.					
13.1.2-13.1.3, Rev. 6 (03/2007)	Operating Organization					
	Refer to the BTP for the details of Table 1					
II.1	General Requirements					
	Plant staff organizational structures are not rigidly fixed. However, experience has shown that certain components are common to and necessary for all plants. Among these are operational, onsite technical support, and maintenance groups under the direction and supervision of a plant manager.					
	The operating organization should be free of ambiguous assignments of primary responsibility. Operating responsibilities should be reasonably well defined in both numbers and experience of persons required to implement the project.					
	The total on-shift manpower available should include enough full operating-shift crews that excessive overtime is not routinely scheduled. 4.					
	Any requests for exemptions from the requirements of 10 CFR 50.54(m) concerning the number of licensed personnel should be justified and reviewed using the NRC's "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)"					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(NUREG-1791).					
	Specific Requirements. Specific criteria to meet the relevant requirements of 10 CFR 50.40(b), 10 CFR 50.80, and 10 CFR 50.54(j), (k), (l), and (m) are as follows: A. ANSI N18.7/ANS-3.2, Section 3.4, "Operating Organization," as endorsed by Regulatory Guide 1.33, should be met. In addition, the following criteria should be satisfied: i. The reporting responsibility and authority of the functional areas of radiation protection, quality assurance, and training should ensure independence from operating pressures. In utilities with large commitments to nuclear power plants, overall management and technical direction in these areas may be concentrated at the home office. ii. There should be clear lines of authority to the plant manager. iii. Responsibility for all activities important to the safe operation of the facility should be clearly defined. iv. Distinct functional areas should be separately supervised and/or managed. v. There should be sufficient managerial depth to provide qualified backup if the incumbent is absent. B. Responsibilities and authorities of operating organization personnel should conform to the requirements of ANSI N18.7/ANS-3.2, Section 5.2, "Rules of Practice"; ANSI Section 4.4, "Onsite Review," as endorsed by Regulatory Guide 1.33; Branch Technical Position SPLB 9.5-1; and Regulatory Guide 1.8 for the operating organization. In addition, the organization should reflect the staff position in TMI Action Plan item I.C.3 of NUREG-0694 by clearly					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	defining the command duties of the shift supervisor position and making top management responsibility for the safe operation of the plant. C. Assignments of onsite shift operating crews shall be made in accordance with 10 CFR 50.54(j), (k), (l), and (m). In addition, the staffing should follow the staff positions of TMI Action Plan items I.A.1.1 and I.A.1.3 of NUREG-0737, as follows: i. A shift supervisor with a senior reactor operator's license, who is also a member of the station supervisory staff, shall be on site at all times when at least one unit is loaded with fuel. ii. In addition to the licensed personnel specified in 10 CFR 50.54(m), as a minimum, an auxiliary operator (nonlicensed) shall be assigned to each reactor and an additional auxiliary operator shall be assigned for each control room for an operating reactor. These operators shall be properly qualified to support the unit to which they are assigned. (The shift composition described above is shown in tabular form in Table 1.) iii. To meet TMI Action Plan item I.A.1.1 of NUREG-0737, engineering expertise shall be onsite at all times a licensed pressurized water reactor (PWR) is being operated in Modes 1–4 or a licensed boiling water reactor (BWR) is being operated in Modes 1–3. This engineering expertise should be consistent with one of the options in the Commission's Policy Statement on Engineering Expertise on Shift. iv. A health physics technician shall be on site at all times when there is fuel in a reactor.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	v. A rad/chem technician shall be on site at all times when a PWR is being operated in Modes 1 through 4 or a BWR in Modes 1 through 3. vi. Assignment, stationing, and relief of operators and senior operators within the control room shall be as described in Regulatory Guide 1.114. D. Any deviation from the Specific Criterion B.3.a-f and/or the staffing- related requirements of 10 CFR Part 50 can be justified and reviewed using the guidance set forth in NUREG-1791. E. The total complement of licensed and unlicensed personnel for onsite shift operating crews should be sufficient to avoid the routine heavy use of overtime. (SRP Section 13.5.1 contains guidance on work hour limitations.) To meet this policy, staffing plans should provide for no less than the number required for five shift rotations. F. The plant operating and technical staff should be used as much as possible in the initial test program for the facility. G. Assignments of personnel to the fire brigade should follow the guideline of SRP Section 9.5.1, including the following: i. The responsibilities of the fire brigade members under normal conditions should not conflict with their responsibilities during a fire emergency. ii. The minimum number of fire brigade members available on site for each shift operation crew should be consistent with the activities required to combat the most significant fire. The minimum size of the fire brigade shift should be five persons unless a site evaluation has been completed and some other number justified.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	H. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," sets forth the staff position on plant personnel qualifications and training. In addition, although the qualification levels of the standards are endorsed as acceptable minimums for each position, it is expected that the collective qualifications of the plant staff will be greater than the sum of the minimum individual requirements described in the standard, particularly in the area of nuclear power plant experience and in supervisory and managerial positions involved in operating the facility. If the collective qualifications do not exceed the sum of the minimums for individual positions, additional technical support for the plant staff may be required. This will be determined on a case-by-case basis.					
13.2.1, Rev. 3 (03/2007)	Reactor Operator Requalification Program; Reactor Operator Training					
13.2.1.1	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the Reactor Operator Requalification Program are reviewed in accordance with 10 CFR 50.34(b), 10 CFR 50.54(i), and 10 CFR 55.59. The implementation milestone is within 3 months after issuance of license or the date that the Commission makes the finding under 10 CFR 52.103(g) per 10 CFR 50.54(i-1). The description of the operational program for the Reactor Operator Training Program is reviewed in accordance with 10 CFR 55.13, 10 CFR 55.31, 10 CFR 55.41, 10 CFR 55.43, and 10 CFR 55.45. Its implementation is required by a license condition.					
13.2.2, Rev. 3 (03/2007)	Non-Licensed Plant Staff Training					
13.2.2.1	The nonlicensed plant personnel should be trained in accordance with an appropriate ANSI standard as endorsed by Regulatory Guide 1.8. 2.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
13.2.2.2	Training programs shall be developed, established, implemented, and maintained using a systems approach to training as required by 10 CFR 50.120 and 10 CFR 52.78 and as defined in 10 CFR 55.4. Training program development will be evaluated by the staff using the guidance contained in NUREG-0711 and training program content and effectiveness will be evaluated using NUREG-1220.					
13.2.2.3	Simulation facilities used for training nonlicensed plant personnel should meet the guidelines of Regulatory Guide 1.149.					
13.2.2.4	Personnel to be granted access to protected areas or to emergency operations facilities shall be trained to ensure understanding of information related to the fitness-for-duty program, including the associated policies and procedures, the hazards and effects of drugs and alcohol, available employee assistance programs, responsibilities under the policy, and the consequences that may result from lack of adherence to the policy, as required in 10 CFR 26.21. Managers, supervisors, and persons assigned to escort duties must be trained to ensure they understand the roles and responsibilities of personnel involved in the fitness-for-duty program, techniques for recognizing drugs and indications of drug possession or use, techniques for behavioral observation, and procedures for initiating corrective actions under the program, as required in 10 CFR 26.22.					
13.2.2.5	Training programs related to radiological emergencies shall meet the requirements of 10 CFR Part 50, Appendix E, Section II.F or IV.F, as applicable. The detailed evaluation criteria and methods for the verification of overall compliance with these requirements are contained in SRP Section 13.3.					
13.2.2.6	Formal segments of the initial training program should be substantially completed when the preoperational test program begins.					
13.2.2.7	The number of people for whom training is planned prior to fuel load should be sufficient to ensure that applicable technical specification conditions with respect to the number of plant					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	personnel can be met from the time of initial fuel loading of the first unit, with due allowance for contingencies and the need to avoid planned overtime for supervisory personnel during the startup phase.					
13.2.2.8	Refresher training for nonlicensed personnel should be periodic and not less frequent than every 2 years and should include, at a minimum, refresher instruction on administrative, radiation protection, emergency, and security procedures.					
13.2.2.9	The detailed guidance and criteria for review of radiological protection training and retraining programs, including the evaluation of their adequacy in informing and instructing personnel pursuant to the requirements of 10 CFR 19.12, is described in SRP Section 12.5.					
13.2.2.10	Fire Protection Training A. Fire Brigade Training. The fire brigade training program shall in general follow the guidelines of Branch Technical Position (BTP) SPLB 9.5-1 to ensure that the capability to fight potential fires is established and maintained. The program shall consist of an initial classroom instruction program followed by periodic classroom instruction, firefighting practice, and fire drills as follows: i. Instruction (1) The initial classroom instruction shall include: (a) Indoctrination in the plant firefighting plan with specific identification of each individual's responsibilities. (b) Identification of the type and location of fire hazards and associated types of fires that could occur in the plant. (c) The toxic and corrosive characteristics of					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	expected products of combustion. (d) Identification of the location of firefighting equipment for each fire area and familiarization with the layout of the plant, including access and egress routes to and from each area. (e) The proper use of available firefighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays, hydrogen fires, fires involving flammable and combustible liquids or hazardous process chemicals, fires resulting from construction or modifications (welding), and record file fires. (f) The proper use of communication, lighting, ventilation, and emergency breathing equipment. (g) The proper method for fighting fires inside buildings and confined spaces. (h) The direction and coordination of the firefighting activities (fire brigade leaders only). (i) Detailed review of firefighting strategies and procedures. (j) Review of the latest plant modifications and corresponding changes in firefighting plans. Note: Items ix and x may be deleted from the training of no more than two of the nonoperations personnel who may be assigned to the fire brigade.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(2) The instruction shall be provided by qualified individuals who are knowledgeable, experienced, and suitably trained in fighting the types of fires that could occur in the plant and in using the types of equipment available in the nuclear power plant. (3) Instruction shall be provided to all fire brigade members and fire brigade leaders. (4) Regular planned meetings shall be held at least every 3 months for all brigade members to review changes in the fire protection program and other subjects as necessary. (5) Periodic refresher training sessions shall be held to repeat the classroom instruction program for all brigade members over a 2-year period. These sessions may be concurrent with the regular planned meetings. ii. Practice. Practice sessions shall be held for each shift fire brigade on the proper method of fighting the various types of fires that could occur in a nuclear power plant. These sessions shall provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under the strenuous conditions encountered in firefighting. These practice sessions shall be provided at least once per year for each fire brigade member. iii. Drills (1) Fire brigade drills shall be performed in the plant so that the fire brigade can practice as a team.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(2) Drills shall be performed at regular intervals not to exceed 3 months for each shift fire brigade. Each fire brigade member should participate in each drill, but must participate in at least two drills per year. A sufficient number of these drills, but not less than one for each shift fire brigade per year, shall be unannounced to determine the firefighting readiness of the plant fire brigade, brigade leader, and fire protection systems and equipment. Persons planning and authorizing an unannounced drill shall ensure that the responding shift fire brigade members are not aware that a drill is being planned until it is begun. Unannounced drills shall not be scheduled more frequently than 4 weeks apart. At least one drill per year shall be performed on a “backshift” for each shift fire brigade. (3) The drills shall be preplanned to establish the training objectives of the drill and shall be critiqued to determine how well the training objectives have been met. Unannounced drills shall be planned and critiqued by members of the management staff responsible for plant safety and fire protection. Performance deficiencies of a fire brigade or of individual fire brigade members shall be remedied by scheduling additional training for the brigade or members. Unsatisfactory drill					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	performance shall be followed by a repeat drill within 30 days. (4) At 3-year intervals, a randomly selected unannounced drill shall be critiqued by qualified individuals independent of the licensee's staff. A written report from such individuals shall be available for NRC review. (5) Drills shall, as a minimum, include the following: (a) Assessment of fire alarm effectiveness, time required to notify and assemble the fire brigade, selection, placement, and use of equipment, and firefighting strategies. (b) Assessment of each brigade member's knowledge of his or her role in the firefighting strategy for the area assumed to contain the fire. Assessment of the brigade member's compliance with established plant firefighting procedures and use of firefighting equipment, including self-contained emergency breathing apparatus, communication equipment, and ventilation equipment, to the extent practicable. (c) The simulated use of firefighting equipment required to cope with the situation and type of fire selected for the drill. The area and type of fire chosen for the drill should differ from those used in the previous drill so that brigade members are trained in fighting fires in various plant areas. The situation selected should					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	simulate the size and configuration of a fire that could reasonably occur in the area selected, allowing for fire development due to the time required to respond, obtain equipment, and organize for the fire, assuming the loss of automatic suppression capability. (d) Assessment of the thoroughness, accuracy, and effectiveness of the brigade leader's direction of the firefighting effort. iv. Records. Individual records of training provided to each fire brigade member, including drill critiques, shall be maintained for at least 3 years to ensure that each member receives training in all parts of the training program. These records of training shall be available for NRC review. Retraining or broadened training for firefighting within buildings shall be scheduled for all brigade members whose performance records show deficiencies. v. Fire Protection Staff. Training for the fire protection staff members shall include courses in: (1) Design and maintenance of fire detection, suppression, and extinguishing systems. (2) Fire prevention techniques and procedures. (3) Training and manual firefighting techniques and procedures for plant personnel and the fire brigade. vi. Other Station Employees					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(1) Instruction (a) Instruction shall be provided for all employees once a year. It shall be repeated on an annual basis. The instruction shall be given, as appropriate, on (i) the fire protection plan (ii) the evacuation routes, and (iii) the procedure for reporting a fire. (b) Instruction for security personnel shall address (i) entry procedures for outside fire departments, (ii) crowd control for people exiting the station, and (iii) procedures for reporting potential fire hazards observed in touring the facility. (c) Instruction for all shift personnel should complement the instruction given to members of the fire brigade. (d) Instruction shall be provided to temporary employees so that they are familiar with (i) evacuation signals, (ii) evacuation routes, and (iii) the procedure for reporting fires. (2) Drills. All employees should participate in an annual evacuation drill.					
13.2.2.11	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the Non-licensed Plant Staff Training Program are reviewed in accordance with 10 CFR 50.120 and 10 CFR 52.78. The implementation milestone is 18 months prior to scheduled fuel load per 10 CFR 50.120(b).					
13.3, Rev. 3	Emergency Planning					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
(03/2007)						
13.3.1	All of the standards of 10 CFR 50.47(b), as supported by the guidance in the corresponding planning standards and evaluation criteria of NUREG-0654/FEMA-REP-1, Rev. 1, (including the March 2002 addenda) must be met before an OL is issued pursuant to 10 CFR 50.57 or a COL is issued pursuant to 10 CFR 52.97. In addition, for the first reactor at a site, Appendix E to 10 CFR Part 50 requires that a full participation exercise be conducted within 2 years before NRC issuance of an operating license for full power (i.e., one authorizing operation above 5 percent of rated power). Because this exercise would be included in the ITAAC required for a COL, it's acceptance criteria would have to be satisfied before fuel loading pursuant to a COL (see Table 14.3.10-1).					
13.3.2	The onsite and, except as provided in 10 CFR 50.47(d), offsite emergency response plans for nuclear power reactors must meet the standards established in 10 CFR 50.47(b) and applicable requirements of Appendix E to 10 CFR Part 50. Compliance with these regulations is determined by using the guidance in Regulatory Guide (RG) 1.101, Rev. 2, which endorses NUREG-0654/FEMA-REP-1, Rev. 1, and through it NUREG-0396, and NUREG-0696. NUREG-0654/FEMA-REP-1, Rev. 1, establishes an acceptable basis for NRC licensees and State, tribal and local governments to develop radiological emergency plans and procedures, and improve their overall state of emergency preparedness. NUREG-0696 discusses the facilities and systems to be provided by nuclear power plant licensees to aid the licensee's response to emergency situations. Additional guidance is provided in NUREG-0718,[6] NUREG-0737, Supplement 1 to NUREG-0737, NUREG-0814, and Supplement 3 to NUREG-0654/ FEMA-REP-1, Rev. 1.					
13.3.3	10 CFR 50.47(b)(4) requires a standard emergency classification and action level scheme. Section IV.C, "Activation of Emergency					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Organization," of Appendix E identifies the four emergency classes. Section IV.B, "Assessment Actions," of Appendix E to 10 CFR Part 50 also requires emergency action levels. The emergency plan should include the emergency classification level scheme described in Appendix 1 and Supplement 3 to NUREG-0654. The staff anticipates that any new application will use an emergency action level scheme similar to that described in Revision 4 of NEI 99-01, "Methodology for Development of Emergency Action Levels," dated January 2003, which was endorsed in Revision 4 Regulatory Guide (RG) 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," dated October 2003. However, Revision 4 of NEI 99-01, "Methodology for Development of Emergency Action Levels," dated January 2003, is not considered to be entirely applicable to advanced light water reactor designs. Even though the majority of Revision 4 of NEI 99-01 may be applicable to any reactor design and should be used, the unique characteristics of the new reactor should be addressed in the development of emergency action levels specific to the new plant and the site. The format of the emergency action level scheme should follow the convention established in Regulatory Information Summary 2003-18, "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels," Revision 4, dated January 2003, and its supplements. Section IV.B. "Assessment Actions," of Appendix E to 10 CFR Part 50 also requires that the initial emergency actions be discussed and agreed on by the State and local governmental authorities. The applicant should provide some form of confirmation of the agreement, such as a letter signed by State and local governmental authorities, in the emergency plan, if the applicant provides emergency action levels different from those for the existing reactor(s) on the site.					
13.3.4	Appendix 2, "Meteorological Criteria for Emergency Preparedness at Operating Nuclear Power Plants," to NUREG-0654/FEMA-REP-1, Rev. 1, provides guidance related to the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	planning standards codified in 10 CFR 50.47(b)(8) and (9) and the requirements of Section IV.E.2 of Appendix E to 10 CFR Part 50. Proposed revision 1 to Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants," is referenced in Appendix 2 to NUREG-0654 as a source of acceptance criteria for meteorological measurements. Since Appendix 2 was issued, additional guidance related to meteorological systems has been developed. NUREG-0696, "Functional Criteria for Emergency Response Facilities," refers to the guidance in proposed Revision 1 to Regulatory Guide 1.23, Revision 2 to Regulatory Guide 1.97, and Appendix 2 to NUREG-0654/FEMA-REP-1, Rev. 1. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements," (Generic Letter 82-33) clarifies the guidance in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-water-cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," and contains guidance related to the need to provide reliable indication of meteorological variables in the control room, Technical Support Center, and Emergency Operations Facility in the vicinity (up to about 10 miles) of the plant site. Revision 3 of Regulatory Guide 1.97 was issued in May 1983 and Revision 4 was issued in June 2006. Revision 1 to Regulatory Guide 1.23 was issued in March 2007.					
13.3.5	Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements," (Generic Letter 82-33) clarifies the guidance in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-water-cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," and contains guidance related to upgrading emergency response facilities and meeting the requirements of 10 CFR 50.47(b)(6), (8), (9) and Section IV.E of 10 CFR Part 50.					
13.3.6	Appendix 3, "Means for Providing Prompt Alerting and Notification of Response Organizations and the Population," to NUREG-0654/FEMA-REP-1, Rev. 1, provides guidance related					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	to 10 CFR 50.47(b)(5) and (6).					
13.3.7	Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents," to NUREG-0654/FEMA-REP-1, Rev.1, provides guidance for the development of protective action recommendations for the public for severe reactor accidents. The guidance updates and simplifies the decision-making process for protective actions for severe reactor accidents given in Appendix 1 to NUREG-0654/FEMA-REP-1, Rev.1.					
13.3.8	RG 1.101, Rev. 2, states that the criteria and recommendations in NUREG-654/FEMAREP-1, Rev. 1, are considered by the NRC staff to be acceptable methods for complying with the standards in 10 CFR 50.47. Except in those cases in which the applicant or licensee proposes acceptable alternative methods for complying with specific portions of the regulations, the methods described in NUREG-0654/FEMA-REP-1, Rev. 1, will be used as a basis for evaluating the adequacy of the emergency plans. If an applicant chooses to propose an alternative practice or method for complying with the regulations, the application should provide an appropriate justification.					
13.3.9	In addition to NUREG-0654/FEMA-REP-1, Rev. 1, FEMA will evaluate State, tribal, and local government planning and preparedness on the basis of applicable policies and guidance,[7] including approved alternative approaches and methods. FEMA will base its findings and determinations, relating to the adequacy of offsite radiological emergency planning and preparedness, on these evaluations.					
13.3.10	10 CFR 50.33(g), 10 CFR 50.47(c)(2), and Section I of Appendix E to 10 CFR Part 50 require that the size of the EPZ for a nuclear power plant shall be determined in relation to local emergency response needs and capabilities, as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. 10 CFR 52.77 requires that the COL application must contain all of the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	information required by 10 CFR 50.33. 10 CFR 50.33(g) requires that an applicant for an operating license submit radiological emergency response plans of State and local government entities that are wholly or partially within the 10-mile plume exposure EPZ, as well as the plans of State governments wholly or partially within the 50-mile ingestion pathway EPZ. An applicant should also submit plans for tribal governmental entities affected by the 10-mile EPZ. NUREG-0396 provides additional guidance relating to the definition of the EPZs.					
13.3.11	Section IV of Appendix E to 10 CFR Part 50, through 10 CFR 52.79(a)(21) and 10 CFR 50.34, requires that an application for an OL or COL provide an analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ; i.e., an ETE. The NRC regulations do not specify a limit for such estimated evacuation times. An ETE can identify physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans. An ETE provides an analysis of the time required to evacuate and for taking other protective actions for various sectors and distances within the plume exposure EPZ. This information can be used by decision makers in responding to an actual emergency to aid in deciding what protective actions to implement. Appendix 4 to NUREG-0654/FEMAREP-1, Rev. 1, and Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, provide guidance relating to performing an ETE analysis. NUREG/CR-6863 provides additional information on ETEs.					
13.3.12	Section VI of Appendix E to 10 CFR Part 50 requires an emergency response data system (ERDS). The ERDS is a direct near real-time electronic data link between a licensee's onsite computer system and the NRC Operations Center, and provides for the automated transmission of a limited data set of selected parameters from a licensee's installed onsite computer system in the event of an emergency. NUREG-1394 provides the minimum standards and acceptable methods that may be used to					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	implement and comply with the ERDS requirements.					
13.3.13	Insofar as emergency planning and preparedness requirements are concerned, 10 CFR 50.47(d) provides that a license authorizing fuel loading and/or low-power testing and training (up to 5 percent of the rated power) may be issued after a finding is made by the NRC that the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The assessment of the applicant's onsite emergency plan will be based on the pertinent standards in 10 CFR 50.47(b) and the requirements of Appendix E to 10 CFR Part 50. However, the acceptability of an applicant's emergency plans will be reviewed against the standards with offsite aspects presented in 10 CFR 50.47(d)(1)-(7).					
13.3.14	Where an applicant for an OL or COL asserts that its inability to demonstrate compliance with the offsite emergency planning requirements of 10 CFR 50.47(b) is wholly or substantially the result of the non-participation of State and/or local governments, an operating license may be issued if the applicant demonstrates to the Commission's satisfaction those elements listed in 10 CFR 50.47(c)(1)(i)-(iii). (See 10 CFR 50.47(c)(1) and 10 CFR 52.79(a)(22)(ii).) Supplement 1 to NUREG-0654/FEMA-REP-1, Rev. 1, provides guidance for the development, review, and evaluation of utility offsite radiological emergency response planning and preparedness, for those situations in which State and/or local governments decline to participate in emergency planning.					
13.3.15	The minimum acceptance criteria for all ESP applications, located in 10 CFR 52.17(b)(1), require that ESP applications identify physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans. If such physical characteristics are identified, the applicant must also identify measures that would, when					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	implemented, mitigate or eliminate the significant impediment. Applications providing only the information required by 10 CFR 52.17(b)(1) must also include a description of contacts and arrangements (preferably letters of agreement) made with local, State, and Federal governmental agencies with emergency planning responsibilities, in accordance with 10 CFR 52.17(b)(4). The applicant may choose to submit additional emergency planning information in the ESP application to address the two options in 10 CFR 52.17(b)(2). The two options allow an ESP applicant to propose either major features of the emergency plans, or to provide complete and integrated emergency plans. While neither option is required, each would provide for a more definitive finding concerning emergency plans and preparedness at the ESP stage than would be the case for submittal of only the minimum required information. Complete and integrated emergency plans in an ESP application will be reviewed in accordance with the applicable requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50. Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, provides guidance relating to emergency planning information in an ESP application.					
13.3.16	For an ESP application, a preliminary analysis of evacuation times is one example of how some significant impediments to the development of emergency plans may be identified. Other factors, such as the availability of adequate shelter facilities, in consideration of local building practices and land use (e.g., outdoor recreation facilities, including camps, beaches, hunting or fishing areas), and the presence of large institutional or other special needs populations (e.g., schools, hospitals, nursing homes, prisons) should also be addressed when identifying significant impediments to the development of emergency plans. Any ETE analysis or other identification of physical impediments should include the latest population census numbers and reflect the most recent local conditions.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Appendix 4 to NUREG-0654/FEMA-REP-1, Rev. 1, and Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, provide guidance relating to performing an ETE analysis. NUREG/CR-6863 provides additional information on ETEs.					
13.3.17	For applications that require site approval for a stationary power reactor subject to 10 CFR Part 50 or 10 CFR Part 52 (e.g., CP, OL, ESP and COL), 10 CFR 100.1 and 10 CFR 100.21(g) require the identification of physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans. This siting requirement is similar to that in 10 CFR 52.17(b)(1) for an ESP application, and the means for identifying significant impediments (e.g., an analysis of evacuation times or ETE) could apply to non-ESP applications. Further, if such physical characteristics are identified, the application must also identify measures that would, when implemented, mitigate or eliminate the significant impediment. Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards (see 10 CFR 100.10(d), which applies to applications submitted before January 10, 1997). The application should provide a projection of the population within the 10-mile EPZ throughout the requested duration of the application; including a discussion of the sources of information and methodology that supports the population projection. The application should specifically address whether the projected population creates a significant impediment to the development of emergency plans over the requested duration of the ESP or COL application, including how it would affect the ETE. If a significant impediment is created, then the applicant should identify measures that would, when implemented, mitigate or eliminate the significant impediment. Additional site-related guidance is provided in RG 4.7, and in ESP-related guidance documents (e.g., Supplement 2 to NUREG-654/FEMA-REP-1,					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Rev. 1).[8]					
13.3.18	Copies of letters of agreement or other certifications, reflecting contacts and arrangements made with local, State, and Federal agencies with supporting emergency responsibilities, should be included in a CP, OL, ESP or COL application, as required by 10 CFR 52.17(b)(4), 10 CFR 52.79(a)(22), or Section II.B of Appendix E to 10 CFR Part 50.[9] The agreement information should be up-to-date when the application is submitted, and should reflect use of the proposed site for possible construction of a new reactor (or reactors). In addition, a discussion of the details associated with any ambiguous or incomplete language in the letters of agreement should be provided in the application. For an existing reactor site, the letters of agreement or other certifications[10] should clearly address the presence of an additional reactor (or reactors) at the site, and any impact that would have on governmental agency or private organization emergency planning responsibilities, including acknowledgment by the agencies or organization of the proposed expanded responsibilities. If the applicant is unable to make arrangements with local, tribal, State, and Federal governmental agencies with emergency planning responsibilities, for whatever reason, the applicant should discuss its efforts to make such arrangements and describe any compensatory measures the applicant has taken or plans to take because of the lack of such arrangements. Supplement 1 to NUREG-654/FEMA-REP-1, Rev. 1, provides guidance for the development, review, and evaluation of utility offsite radiological emergency response planning and preparedness (i.e., a utility plan), for those situations in which State and/or local governments decline to participate in emergency planning. (See also 10 CFR 50.47(c)(1).)					
13.3.19	Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, will be used as the primary guidance for the review of emergency preparedness information and plans submitted with an ESP					

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	application pursuant to Subpart A of 10 CFR Part 52. For a pre-existing nuclear facility, all major features of the emergency plan (i.e., all 14 planning standards) identified in Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, should be addressed in the ESP application. The detailed, specific evaluation criteria for each of the major features in Supplement 2 should be addressed for both a pre-existing nuclear facility, as well as for applicable major features associated with a site without a pre-existing nuclear facility. If emergency planning information is not provided on all 14 major features (including the detailed, specific evaluation criteria) in Section V of Supplement 2, the ESP application will not be rejected. The review and evaluation will, however, only be based on, and specifically limited to, the submitted information that relates to the guidance in Supplement 2 of NUREG-0654/FEMA-REP-1, Rev. 1.					
13.3.20	The planning standards and evaluation criteria for preparing and evaluating an ESP application containing complete and integrated emergency plans are provided in NUREG-0654/FEMA-REP-1, Rev. 1. Under this ESP option, the applicant should make a good-faith effort to obtain from the government agencies certifications that (1) the proposed emergency plans are practicable; (2) these agencies are committed to participating in any further development of the plans, including any required field demonstrations; and (3) these agencies are committed to executing their responsibilities under the plans in the event of an emergency. The application must contain any certifications that have been obtained. If these certifications cannot be obtained, the application must contain information, including a utility plan pursuant to 10 CFR 50.47(c)(1), sufficient to show that the proposed plans nonetheless provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site. The utility-prepared emergency plans and preparedness will be reviewed and evaluated using the guidance in Supplement 1 to NUREG-					

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	0654/FEMA-REP-1, Rev. 1.					
13.3.21	10 CFR 52.17(b)(3) allows an applicant for an ESP, that proposes major features of the emergency plans or complete and integrated emergency plans, to include proposed ITAAC which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the license, the provisions of the Atomic Energy Act, and the NRC's regulations.					
13.3.22	10 CFR 52.47(b)(1) allows an applicant for a design certification to include proposed ITAAC, including those applicable to emergency planning, which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.					
13.3.23	10 CFR 52.80(a) requires that an application for a combined license includes proposed emergency planning ITAAC which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.					
13.3.24	Table 14.3.10-1[11] provides an acceptable set of generic emergency planning ITAAC that an applicant may use to develop application-specific ITAAC, tailored to the specific reactor design and emergency planning program requirements. A smaller set of ITAAC is acceptable if the application contains information that fully addresses emergency preparedness requirements associated with any of the generic ITAAC in Table 14.3.10-1 that are not used. Table 14.3.10-1 is not all-inclusive, or exclusive of other ITAAC an applicant may propose. Additional plant-specific					

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	emergency planning ITAAC (i.e., beyond those listed in Table 14.3.10-1) may be proposed, and they will be examined to determine their acceptability on a case-by-case basis. In general, ITAAC are inappropriate for procedure-level details associated with the emergency plans, in that procedure adequacy and implementation can be evaluated under the exercise ITAAC, and should be limited to those aspects of emergency planning and preparedness that can not reasonably be addressed prior to construction of the plant. Each EP-ITAAC must have an objective acceptance criteria stated.					
13.3.25	For those licensees subject to 10 CFR 50.34(f),[12] 10 CFR 50.34(f)(2)(xxv) requires that an applicant provide a TSC, OSC, and, for a CP application only, a near-site emergency operations facility (EOF) (TMI Item III.A.1.2[13]). NUREG-0696, Appendix B to NUREG-0718, NUREG-0737, and Supplement 1 to NUREG-0737 provide guidance relating to the design and implementation of emergency response facilities (e.g., TSC, OSC, EOF). In addition, 10 CFR 50.47(b)(8) and Subsection IV.E.8 of Appendix E to 10 CFR Part 50 requires that the design should include adequate emergency facilities and equipment to support emergency response. NUREG-0696, NUREG-0737, and Supplement 1 to NUREG-0737 provide guidance relating to occupancy and radiological habitability of vital areas (including the TSC), which aid in the mitigation of or recovery from an accident.					
13.3.26	For those licensees subject to 10 CFR 50.34(f), 10 CFR 50.34(f)(2)(iv) requires that an applicant seeking an operating license shall provide an SPDS in both the TSC and EOF (TMI Item I.D.2). The SPDS includes the minimum set of plant parameters needed to assess the safety status of the plant in a timely manner, and is capable of indicating when process limits are being approached or exceeded. Supplement 1 to NUREG-0737, NUREG-0696, and NUREG-0814 provide guidance regarding the SPDS. (The SPDS is reviewed under SRP					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Sections 7.5 and 18.2.)					
13.3.27	For those licensees subject to 10 CFR 50.34(f), 10 CFR 50.34(f)(2)(viii) requires that an applicant provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials, while ensuring that no individual receives radiation exposure in excess of 0.05 Sv (5 rem) to the whole body or 0.5 Sv (50 rem) to the extremities (TMI Item II.B.3). In addition, 10 CFR 50.47(b)(9) requires adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition. To address this regulation, the NRC has concluded that source term information should be obtained and analyzed, to continuously assess and refine dose assessments and confirm or modify initial protective action recommendations. Finally, 10 CFR 50.47(b)(11) requires the establishment of the means for controlling radiological exposure to emergency workers. Post-accident sampling systems are discussed in the October 31, 2000, Model Safety Evaluation, as it relates to the development of contingency plans for sampling and analysis of highly radioactive samples from the reactor coolant system, containment sump, and containment atmosphere.					
13.3.28	For those licensees subject to 10 CFR 50.34(f), 10 CFR 50.34(f)(2)(xvii) requires instrumentation to measure, record and readout of various containment parameters, including noble gas effluents at all potential, accident release points. In addition, an applicant must provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples (TMI Item II.F.1). RG 1.97 provides guidance relating to instrumentation to assess plant and environmental conditions during and following an accident.					
13.3.29	10 CFR 50.72(a)(3) and (c)(3) require the notification of the NRC					

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	Operations Center following the declaration of an emergency in accordance with the licensee's approved emergency plans, and the establishment of an open and continuous communications channel when requested by the NRC. 10 CFR 50.72(a)(4) establishes requirements for the activation of the ERDS following the licensee's declaration of an alert, site area emergency, or general emergency. NUREG-1022 provides the minimum standards and acceptance methods that may be used to comply with these NRC reporting requirements. 10 CFR 73.71(a) requires the notification of the NRC Operations Center, after the discovery of an imminent or actual safeguards threat against the facility or other safeguards events. Regulatory Guide 5.62 provides the minimum standards and acceptance methods that may be used to comply with these NRC reporting requirements.					
13.3.30	The emergency planning and preparedness standards and requirements in 10 CFR Part 50, 10 CFR Part 52, and 10 CFR Part 100 are supplemented by various generic communications and Commission Orders.[14] Those generic communications that relate to emergency planning and are currently in effect are identified in Subsection VI (below). They provide additional guidance and criteria for meeting the relevant emergency planning standards and requirements. Any subsequently issued generic communications or Commission Orders that pertain to emergency planning and preparedness and are relevant to the application should also be addressed by the applicant.					
13.3.31	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Emergency Planning program are reviewed in accordance with 10 CFR 50.47, Part 50 Appendix E. The implementation milestones are as follows: full participation exercise conducted within 2 years of scheduled date for initial loading of fuel per 10 CFR 50, Appendix E.IV.F.2a(ii); onsite exercise conducted within 1 year before the schedule date for initial loading of fuel per 10 CFR Part 50, Appendix E.IV.F.2a(ii); and applicant's detailed					

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	implementing procedures for its emergency plan submitted no less than within 180 days prior to scheduled date for initial loading of fuel per 10 CFR Part 50, Appendix E.V.					
	NOTES: 6. The applicability of NUREG-0718, Rev. 2, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," January 1982, is addressed in 10 CFR 50.34(f). 7. In addition to the current REP-series guidance documents and associated memoranda, offsite plans and procedures are reviewed against the requirements and policies incorporated in the REP Program Planning Guidance Document: "Radiological Emergency Preparedness: Planning Guidance" (see 68 FR 9669, February 28, 2003). 8. The identification of significant impediments, while common to all site approval decisions (per 10 CFR 100.21(g)), is more fully addressed for an ESP application under 10 CFR 52.17, which also requires that the applicant identify measures to mitigate or eliminate any identified significant impediment (see 10 CFR 52.18). The adequate compensating engineering safeguards language, which is taken from 10 CFR 100.10(d) and applies to applications prior to January 10, 1997, is intended to address this societal risk siting factor for emergency planning, and is included in order to determine the acceptability of the site if significant impediments are identified. 9. Agreements or other arrangements with tribal agencies and private organizations should be included in the application. 10. Another acceptable method of addressing this issue would be through the use of separate correspondence. Such correspondence might be appropriate, for example, in a case for which an existing letter of agreement is written in a way that is broad enough to cover an expanded site use, and does not need to be revised. The correspondence would					

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	identify this fact. 11. See SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005; and SRM SECY-05-0197, February 22, 2006. The generic EP ITAAC in SECY-05-0197 formed the basis for Table 14.3.10-1. 12. NUREG-0933, "A Prioritization of Generic Safety Issues," August 2004, presents priority rankings for generic safety issues, and is periodically updated. 10 CFR 50.34(f) identifies the pending applications that are subject to additional Three Mile Island (TMI)-related requirements. 13. Alphanumeric designations correspond to the related action plan items in NUREG-0718 and NUREG-0660, relating to the TMI accident in 1979 (see 10 CFR 50.34(f)(a)(1), footnote 10). 14. See also 10 CFR 52.79(a)(37), which requires that a COL application contain information which demonstrates how operating experience insights.					
13.4, Rev. 3 (03/2007)	Operational Programs					
	Refer to the RG for table FSAR 13.4-x, Operational Programs Required by NRC Regulation and Program Implementation					
13.5.1.1 (03/2007)	Administrative Procedures - General					
	Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations are identified within the Requirements section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance					

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	criteria provide acceptable methods of compliance with the NRC regulations. Refer to the BTP for the detailed criteria.					
13.5.1.2 DRAFT	Administrative Procedures - Initial Test Program (Content subsumed into SRP Section 14.2)					
13.5.1.2.1	Test Procedures The applicant's administrative and organizational system that will be used to develop, review, and approve individual test procedures should provide for appropriate levels of review prior to final approval. The individuals performing these functions should meet the qualification requirements described in Section 4.4.6 of ANS 3.118 draft revision dated 12-6-79. The applicant should utilize system designers to provide the test objectives and acceptance criteria used in developing detailed test procedures. The participating system designers should include those of the nuclear steam supply system vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as appropriate.					
13.5.1.2.2	Conduct of Test Program a. The test program should be conducted by appropriately qualified personnel using detailed procedures approved by designated management positions within the applicant's organization. b. The controls used by the applicant to assure ensure that test prerequisites are met should include requirements for inspections, checks, etc.; require identification of test personnel completing data forms or checksheets; and require identification of dates of completion.					

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	c. The controls provided for plant modification and repairs, identified as a result of plant testing, are found to be acceptable if (1) the controls are sufficient to assure ensure the required repairs modifications will be made, (2) the controls will assure ensure retesting is conducted following such modifications or repairs, and (3) the controls will assure ensure a review of any proposed facility modifications by the original design organization or other designated design organizations. The applicant's requirements for documentation associated with such controls should permit audits to be made to assure ensure proper implementation of controls. d. The controls pertaining to adherence to test procedures and to methods for changing test procedures are found to be acceptable based on the reviewer's judgment. Modifications to startup test procedures should be made in accordance with technical specifications for post-fuel loading tests.					
13.5.1.2.3	Review, Evaluation, and Approval of Test Results a. The controls that will govern the review, evaluation, and approval of test results should provide for a technical evaluation of test results of by19 qualified personnel and approval of test results in or by20 personnel in designated management positions in the applicant's organization. b. Design organizations should be notified and should participate in the resolution of problems involving design that result in or contribute to a failure to meet test acceptance criteria. c. The applicant should establish the requirement that test data for each major test phase will be reviewed and approved prior to beginning the next phase of testing. d. The applicant should establish the requirement that test data at each major power test plateau or power/flow test					

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	condition will be reviewed and approved before proceeding to the next test level during the power ascension test phase.					
13.5.2.1, Rev. 1 (03/2007)	Operating and Emergency Operating Procedures					
13.5.2.1.1	Operating Procedure Schedule. A generally acceptable target date for completion of operating procedures is about 6 months before fuel loading to allow adequate time for plant staff familiarization and to allow NRC staff adequate time to develop operator license examinations. The PGP for EOPs must be submitted not later than 3 months before the date formal operator training on EOPs is to begin.					
13.5.2.1.2	Control Room and Plant Procedures. The following regulations and staff guidelines applicable to operating procedures are to be used in the control room and locally in the plant: A. 10 CFR 50.34(a)(6) and (10) and 10 CFR 50.34(b)(6)(iv) and (v). B. 10 CFR Part 50, Appendix B, Criteria V and VI, establish criteria for development, approval, and control of procedures for all activities affecting quality. C. The review criteria for procedures in NUREG-0711, Chapter 9, "Element 8 - Procedure Development." D. NUREG-0737, “Clarification of TMI Action Plan,” item I.C.1, “Guidance for the Evaluation and Development of Procedures for Transients and Accidents.” (emergency operating procedures only) E. Supplement 1 to NUREG-0737, TMI Action Plan items I.C.1 and I.C.9, “Requirements for Emergency Response Capability,” Item 7, Subsections 7.1 and 7.2, “Upgrade of Emergency Operating Procedures.” (emergency operating					

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	procedures only) F. The guidelines in the Regulatory Position section of Regulatory Guide 1.33. G. The guidelines of ANSI/ANS 3.2-1982, Section 5.3. H. Appendix A to SRP, Section 13.5.2.1, "Guidelines for the Evaluation of Procedures Generation Packages." (emergency operating procedures only) I. Supplement 1 to NUREG-1358, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures," 1992.					
13.5.2.2 DRAFT	Maintenance and Other Operating Procedures (Content subsumed into SRP Section 17.5)					
	Refer to the BTP for the detailed criteria.					
13.6 (03/2007)	Physical Security					Exclude ; Administrative
13.6.1 (03/2007)	Physical Security - Combined License					
13.6.1.1	The physical security plan (PSP) is the physical protection program that provides high assurance against the design basis threat outlined in 10 CFR Part 73.1 (a) to ensure activities involving special nuclear material are not inimical to common defense and security and do not constitute an unreasonable risk to the public health and safety (Appendix 1). A. 10 CFR 73.21 - 10 CFR 73.21 establishes the requirements for the protection of Safeguards Information. The physical security plan, safeguards contingency plan, and any elements of the guard training and qualification plan that disclose information related to the physical					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	security system or response procedures are considered Safeguards Information. The unauthorized disclosure of this information could compromise the ability of the security organization to provide an appropriate level of protection against, and response to, threats, theft and radiological sabotage. Compliance with 10 CFR 73.21 provides assurance that Safeguards Information is protected against unauthorized disclosure. B. Section (b) of 10 CFR 73.55 - Physical security organization. The licensee shall establish a security organization, including guards, to protect the facility against radiological sabotage. Security personnel, including guards, shall comply with the requirements of 10 CFR Part 73, Appendix B, "General Criteria for Security Personnel." These general criteria establish requirements for the selection, training, equipping, testing, and qualification of individuals who will be responsible for the protecting of special nuclear materials, nuclear facilities, and nuclear shipments. C. Section (c) of 10 CFR 73.55 - Physical Barriers. The licensee shall locate vital equipment only within a vital area, which, in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers as defined in 10 CFR 73.2. Isolation zones adjacent to the protected area perimeter shall also be provided. Isolation zone and protected area lighting shall meet the requirements of 10 CFR 73.55(c) including the interpretations in 10 CFR 8.5(b) and (c). The reactor control room perimeter boundaries shall be bullet resisting. Vehicle control measures shall be established in accordance with the requirements of 10 CFR 73.55(c) (7). D. Section (d) of 10 CFR 73.55 - Access Requirements. The licensee shall control all points of personnel and vehicle					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	access into a protected area. Identification and search of all individuals, packages, and vehicles, unless otherwise provided in 10 CFR 73. shall be made and authorization shall be checked at such points. Access authorization systems shall be designed to accommodate the rapid ingress and egress of authorized individuals and vehicles during emergency conditions or situations that could lead to emergency conditions. The access authorization systems shall ensure vital area access is controlled during nonemergency conditions through individual access authorizations which are periodically reviewed; through maintenance of positive control over vital area access for authorized individuals; and by locking and alarming unoccupied vital areas. Locking devices, including keys and combinations, related to access control to protected and vital areas shall be controlled. Records, in accordance with 10 CFR 73.70(d), shall document the vital area entry and exit of individuals. 10 CFR 73.2 provides additional guidance relative to the use and control of locks, keys, and combinations. E. Section (e) of 10 CFR 73.55 - Detection Aids. All alarms required pursuant to this part shall annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station, not necessarily onsite, such that a single act cannot remove the capabilities of calling for assistance or otherwise responding to an alarm. The central alarm station shall be considered a vital area, shall be bullet resisting, the interior shall not be visible from the protected area perimeter, and associated onsite secondary power supplies for alarm annunciators and non-portable communication equipment must be located within vital areas. All emergency exits from protected and vital areas shall be alarmed. Alarm devices and transmission lines					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	must be tamper indicating and self checking. Regulatory Guide 5.44 (Testing option 1 or 2) provides additional guidance relevant to perimeter intrusion and vital area alarm systems, respectively. In addition, section B.4.e of Attachment 2 to the ICM Order provides guidance for surveillance equipment and security patrols. F. Section (f) of 10 CFR 73.55 - Communication Requirements. Each guard, watchman or armed response individual, or any other individual performing an active security function on duty, shall be capable of maintaining continuous communications with an individual in each continuously manned alarm stations. Conventional telephone and radio or microwave transmitted two-way voice communications shall be established with local law enforcement authorities. G. Section (g) of 10 CFR 73.55 - Testing and Maintenance. Each licensee shall test and maintain intrusion alarms, emergency alarms, communications equipment, access control equipment, physical barriers, and other security-related devices or equipment. Intrusion alarms should be tested in accordance with guidance in Regulatory Guide 5.44. Additional guidance for Protected and Vital Area physical barriers can be found in paragraph C.3.a. of Attachment 2 to the Training Order. In addition to security system testing and maintenance requirements, licensees shall independently audit the continued effectiveness of the overall security program per the requirements of 10 CFR 73.55(g)(4) and 10 CFR 50.54(p)(3), and the access authorization and fitness for duty programs in accordance with 10 CFR 73.56(g) and 10 CFR 26.80 respectively. Periodic reviews of the interface between the security program and plant and personnel safety shall be addressed in accordance with the					

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	requirements of 10 CFR 73.55(d)(7)(ii)(B). H. Section (h) of 10 CFR 73.55 - Response Requirements. The licensee shall establish, maintain, and follow an NRC-approved safeguards contingency plan. The licensee shall maintain liaison with local law enforcement authorities. Each licensee shall maintain an adequate number of guards for response and assessment of possible security threats. Each licensee shall require that the security organization take steps to evaluate and neutralize the threat when detected with sufficient force to counter the force of the threat. The licensee shall provide a means to observe the isolation zones and physical barrier at the perimeter of the protected area. I. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations; J. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.					
13.6.1.2	The Training and Qualification Program (T&QP) (Appendix B of					

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	an applicant's security plan) should provide the general criteria for security personnel, which establishes requirements for the selection, training, equipping, testing, and qualification of individuals who are responsible for the protection of special nuclear materials, nuclearfacilities, nuclear shipments, and personnel that perform security duties (Appendix 2).					
13.6.1.3	The Safeguards Contingency Program (SCP) (Appendix C of an applicant's security plan) should provide specific, defined objectives in the event of threats, thefts, or radiological sabotage relating to special nuclear material or nuclear facilities. The plan contains 1) a predetermined set of decisions and actions to satisfy specified objectives 2) an identification of the data, criteria, procedures, and mechanisms necessary to efficiently implement decisions, and 3) a description of the individual, group, or organizational entity responsible for each decision and action (Appendix 3).					
13.6.1.4	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the weapons training, weapons qualification, and requalification program are reviewed in accordance with 10 CFR 73.55(b)(4)(I) and 10 CFR Part 73, Appendix B, paragraphs II.A, B and C. The description of the operational program and proposed implementation milestone(s) for the physical security program are reviewed in accordance with 10 CFR 73.55. The description of the operational program and proposed implementation milestone(s) for the access authorization program are reviewed in accordance with 10 CFR 73.56(g). The description of the operational program and proposed implementation milestone(s) for the vehicle control program are reviewed in accordance with 10 CFR 73.55(c)(7). The description of the operational program and proposed implementation milestone(s) for the fitness-for-duty program are reviewed in accordance with 10 CFR 26.80. The implementation milestone for all operational programs in this section is prior to the arrival of					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	nuclear fuel onsite per a license condition.					
13.6.2 (03/2007)	Physical Security - Design Certification					
13.6.2.1	Section (c) of 10 CFR 73.55 - Physical Barriers. The licensee shall locate vital equipment only within a vital area, which in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers as defined in 10 CFR 73.2. The physical barriers at the perimeter shall be separated from any other barrier designated as physical barrier for a vital area within the protected area. Isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the protected area permit observation. Intrusion detection system detects penetration or attempted penetration of the protected area (PA) barrier. All exterior areas within the protected area are illuminated. The external walls, doors, ceiling and floors in the main control room are bullet resistant. Vehicle control measures which include vehicle barrier systems protect against the use of land vehicle.					
13.6.2.2	Section (d) of 10 CFR 73.55 - Access Requirements. The licensee shall control all points of personnel and vehicle access into a protected area, to include detection equipment capable of detecting firearms, explosives and incendiary devices. Unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in both the central and secondary alarm stations upon intrusion into a vital area. The individual responsible for the last access control function (controlling admission to the protected area) must be isolated within a bullet-resisting structure.					
13.6.2.3	Section (e) of 10 CFR 73.55 - Detection Aids. All alarms required pursuant to this part shall annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station, not necessarily onsite, such that a single act cannot remove the capabilities of					

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	calling for assistance or otherwise responding to an alarm. The central alarm station shall be considered a vital area, shall be bullet-resisting, the interior will not be visible from the protected area perimeter, and associated onsite secondary power supplies for alarm annunciators and non-portable communication equipment must be located within vital areas. Alarm devices and transmission lines must be tamper indicating and self checking. Alarm annunciation shall indicate type of alarm and location. All emergency exits from protected and vital areas shall be alarmed.					
13.6.2.4	Section (f) of 10 CFR 73.55 - Communication Requirements. Each security officer, watchman or armed response individual shall be capable of maintaining continuous communications with an individual in each continuously manned alarm stations. Conventional telephone and radio or microwave transmitted two-way voice communications shall be established with local law enforcement authorities.					
13.6.2.5	Section (g) of 10 CFR 73.55 - Testing and Maintenance. Each applicant shall develop test and maintenance provisions for intrusion alarms, emergency alarms, communication equipment, access control equipment, physical barriers, and other security-related devices or equipment.					
13.6.3 (03/2007)	Physical Security - Early Site Permit					
13.6.3.1	Section (c) of 10 CFR 73.55 - Physical Barriers. The licensee shall locate vital equipment only within a vital area, which in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers as defined in 10 CFR 73.2. The physical barriers at the perimeter shall be separated from any other barrier designated as a physical barrier for a vital area within the protected area. Isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the protected area permit observation. An intrusion detection system detects penetration or attempted penetration of the protected area (PA) barrier. All exterior areas					

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	within the protected area are illuminated. The external walls, doors, ceiling and floors in the main control room are bullet resistant. Vehicle control measures which include vehicle barrier systems protect against the use of a land vehicle.					
13.6.3.2	Section (d) of 10 CFR 73.55 - Access Requirements. The licensee shall control all points of personnel and vehicle access into a protected area, to include detection equipment capable of detecting firearms, explosives and incendiary devices. Unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in both the central and secondary alarm stations upon intrusion into a vital area. The individual responsible for the last access control function (controlling admission to the protected area) must be isolated within a bullet-resisting structure.					
13.6.3.3	Section (e) of 10 CFR 73.55 - Detection Aids. All alarms required pursuant to this part shall annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station, not necessarily onsite, such that a single act cannot remove the capabilities of calling for assistance or otherwise responding to an alarm. The central alarm station shall be considered a vital area, shall be bullet-resisting, the interior will not be visible from the protected area perimeter, and associated onsite secondary power supplies for alarm annunciators and non-portable communication equipment must be located within vital areas. Alarm devices and transmission lines must be tamper indicating and self checking. Alarm annunciation shall indicate type of alarm and location. All emergency exits from protected and vital areas shall be alarmed.					
13.6.3.4	Section (f) of 10 CFR 73.55 - Communication Requirements. Each security officer, watchman or armed response individual shall be capable of maintaining continuous communications with an individual in each continuously manned alarm stations. Conventional telephone and radio or microwave transmitted two-way voice communications shall be established with local law					

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	enforcement authorities.					
13.6.3.5	Section (g) of 10 CFR 73.55 - Testing and Maintenance. Each applicant shall develop test and maintenance provisions for intrusion alarms, emergency alarms, communication equipment, access control equipment, physical barriers, and other security-related devices or equipment.					
	CHAPTER 14, Initial Test Program and ITAAC-Design Certification					
14.2, Rev. 3 <u>(03/2007)</u>	Initial Plant Test Program - Design Certification and New License Applicants					
14.2.1	Summary of Test Program and Objectives This SRP section lists the general criteria of RG 1.68 that a DC, COL, or OL applicant or holder should address in its safety analysis report (SAR). DC/COL/OL Applicants A. The ITP should describe its objectives, including a description of the objectives for each of the major phases of the test program. B. The ITP should describe the criteria for selection of plant features to be tested by the applicant. C. Objectives and testing selection criteria should be consistent with the general guidelines and applicable regulatory positions in RG 1.68. Applicants should appropriately justify exceptions.					
14.2.2	Test Program's Conformance with Regulatory Guides DC/COL/OL Applicants A. The applicant should commit to the revision of RG 1.68 and the RGs listed in RG 1.68, that are referenced in this SRP					

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	and are in effect six months prior to submittal. The applicant may propose exceptions or alternatives to the specific criteria in any of these RGs, and the staff may find them acceptable if the applicant provides adequate justification. The reviewer responsible for the RG evaluates any exceptions or alternatives. The safety evaluation report (SER) should also list such exceptions or alternatives.					
14.2.3	Initial Test Program Administrative Procedures DC Applicant The applicant should provide a summary description of the following areas: A. The applicant should provide general guidance to control ITP activities, including administrative controls that will be used to develop, review, and approve individual test procedures, coordination with organizations involved in the test program, participation of plant operating and technical staff, and review, evaluation, and approval of test results. B. The applicant should include general guidance for the review of relevant operating and testing experiences at other facilities. This guidance should recognize reportable occurrences of repeatedly experienced safety concerns and other operating experiences that could potentially impact the performance of the test program. C. The applicant should include general guidance about how, and to what extent, the test program will use and/or test plant operating, emergency, and surveillance procedures. D. The applicant should provide test abstracts of SSCs and unique design features that will be tested to verify that system and component performance is in accordance with the design. These test abstracts should include the objectives, tests, and acceptance criteria that will be					

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	included in the test procedures. COL/OL Applicants The applicant should provide a detailed description of the following areas: A. Management Organizations i. The applicant should provide organizational descriptions for principal management positions responsible for the planning, execution, and documentation of preoperational and startup testing activities. ii. The applicant should provide (1) the organizational descriptions for any augmenting organizations or other personnel who will manage or execute any phase of the test program, and (2) the responsibilities, interfaces, and authorities of the principal participants. B. Conduct of the Initial Test Program i. The applicant should conduct the ITP using detailed procedures approved by designated managers in the applicant's organization. ii. Administrative controls should be established to ensure that the designated construction-related inspections and tests are completed before preoperational testing begins. The applicant should also include in the ITP adequate controls for the evaluation and approval of reoperational test results before initial startup tests begin. iii. Administrative controls should address adherence to approved test procedures during the conduct of the					

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	test program and the methods for effecting changes to approved test procedures. iv. The controls that the applicant uses to ensure that the test prerequisites are met should include requirements for (1) inspections, checks, and similar controls, (2) identification of test personnel completing data forms or checksheets, and (3) identification of dates of completion. Each major phase of the test program as well as individual tests should satisfy these requirements. v. The staff will find that the controls provided for plant modification and repairs, identified as a result of plant testing, are acceptable if the controls (1) are sufficient to ensure that the required repairs or modifications will be made, (2) will ensure retesting is conducted following such modifications or repairs, and (3) will ensure a review of any proposed facility modifications by the original design organization or other designated design organizations. The applicant's requirements for documentation associated with such controls should permit audits to be conducted to ensure its proper implementation. C. Test Program Schedule and Sequence i. The applicant should develop a schedule for conducting each major phase of the ITP. ii. The schedule should establish that the safety of the plant will not depend on the performance of untested SSCs. iii. Overlapping test program schedules (for multiunit sites) should not result in significant divisions of responsibilities or dilutions of the staff implementing the test program.					

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	iv. The sequential schedule for individual startup tests should establish that test requirements will be completed in accordance with plant technical specification requirements for SSC operability before changing plant modes. D. Staff Responsibilities, Authorities, and Qualifications i. The applicant should describe the education, training, and experience requirements established for each management and operating staff member—including the NSSS vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as appropriate—who will conduct the preoperational and startup tests and will develop testing, operating, and emergency procedures. ii. The applicant should develop a training program for each functional group of employees in the organization relative to the schedule for preoperational testing and initial startup testing to ensure that the necessary plant staff are ready to begin the test program. E. Development, Review, and Approval of Test Procedures i. The applicant is responsible for the preparation of preoperational and startup test procedures. This includes the methodology used for the generation, review, and approval of test procedures. ii. The applicant should use the NSSS vendor, architect-engineer, and other major contractors, as appropriate, to provide the test objectives and acceptance criteria used in developing detailed test procedures. iii. The applicant's administrative system for use in					

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	reviewing and approving individual test procedures should provide for appropriate levels of review before approval. iv. Controls should be in place to ensure that test procedures include appropriate prerequisites, test objectives, safety precautions, testing of initial conditions, methods to direct and control test performance, and acceptance criteria for evaluating the test. v. The applicant should include provisions to ensure that retesting that is required for modifications or maintenance remains in compliance with ITAAC commitments. vi. The format for the test procedures should be similar to that in RG 1.68, or the reviewer should consider whether the justification provided by the applicant for exception is acceptable. The format should include checklists and signature blocks to control the sequencing of testing. vii. Approved test procedures should be in a form suitable for review by regulatory inspectors at least 60 days before their intended use. Licensees should provide timely notification to NRC of changes in approved test procedures that have been made available for NRC review. F. Review, Evaluation, and Approval of Test Results i. The applicant should develop the procedures that will govern the review, evaluation, and approval of test results for each phase of the test program. Specific procedures should be implemented to ensure notification of responsible organizations, such as design organizations, when test acceptance criteria					

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	are not met and specific controls have been established to resolve such problems. ii. Before proceeding with testing, the applicant should provide controls relating to (1) the methods and schedules for approval of test data for each major phase, and (2) the methods used for initial review of individual parts of multiple tests (e.g., hot functional testing). iii. The controls that will govern the review, evaluation, and approval of test results should provide a technical evaluation of test results by qualified personnel and approval of such results by personnel in designated management positions in the applicant's organization. iv. The applicant should include provisions to allow design organizations to participate in the resolution of design-related problems that result in, or contribute to, a failure to meet test acceptance criteria. v. Provisions should be in place to retain test reports, including test procedures and results, as part of the plant historical records. Startup test reports should be prepared in accordance with RG 1.16, or the reviewer should consider whether the justification provided by the applicant for exception is acceptable. G. Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program i. The applicant should provide a summary of the principal conclusions or findings from the review of operating and testing experiences at other reactor					

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	facilities and their effect on the test program. This review should recognize categories of reportable, repeatedly experienced occurrences and other operating experiences that could potentially impact the performance of the test program. H. Trial Use of Plant Operating and Emergency Procedures i. The applicant should incorporate, to the extent practicable, the plant operating, emergency, and surveillance procedures into the test program or otherwise verify these procedures through use during the test program. ii. The applicant should provide additional operator training and participation based on the performance and evaluation of the test results of certain initial tests. An acceptable program will satisfy the criteria described in Three Mile Island (TMI) Action Plan Item I.G.1 of NUREG-0660 and NUREG-0737.					
14.2.4	Initial Startup Tests DC Applicant The applicant should provide a summary description of the following areas: A. Initial Fuel Loading/Initial Criticality/Low-Power/Power Ascension Testing i. The applicant should include in the ITP a description of the general provisions and precautions for fuel loading, initial fuel loading, initial criticality, low-power testing, and power ascension phases. Precautions, prerequisites, and measures should be consistent with					

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	the guidelines and regulatory positions in RG 1.68. This includes guidance for (1) the completion of all ITAAC associated with preoperational tests before fuel load, (2) measures to review and evaluate the results of the completed preoperational tests, (3) appropriate remedial actions to take if acceptance criteria are not satisfied, (4) applicable technical specification requirements, and (5) actions to take if unanticipated errors or malfunctions occur. COL/OL Applicants The applicant should provide a detailed description of the following areas: A. Initial Fuel Loading and Initial Criticality i. The applicant should provide measures to ensure that preoperational tests are evaluated and approved before fuel loading begins. ii. The procedures that will guide initial fuel loading and initial criticality should include precautions, prerequisites, and measures consistent with the guidelines and regulatory positions in RG 1.68. The staff will review exceptions to regulatory positions and their associated justification on a case-by-case basis. iii. Technical specifications should be instituted to ensure the operability of systems required for fuel loading. iv. The applicant should describe the minimum conditions for initial core loading, which may include, but are not limited to: 1) The reactor containment structure should be complete, and containment integrity should be					

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	demonstrated according to technical specifications. 2) Fuel handling tools and equipment should be available, and operators should be familiar with the use and operation of equipment. 3) The reactor vessel and associated components should be ready to receive fuel. 4) Nuclear instrumentation should be tested and verified to be operable. v. The applicant should include provisions to verify that core flux levels are within predicted or acceptable values. vi. The applicant should provide measures to stop core loading operations if an unexpected or unanalyzed condition occurs. vii. At the completion of fuel loading, the applicant should perform sufficient tests, as necessary, to ensure that the facility is in a final state of readiness to achieve initial criticality and to perform low-power tests. B. Low-Power/Power Ascension Testing i. The applicant should include procedures that will control low-power and power ascension testing. These procedures should include precautions, prerequisites, and measures consistent with the guidelines and regulatory positions in RG 1.68. The staff will review exceptions to regulatory positions and their associated justifications for acceptability on a case-by-case basis.					
14.2.5	Individual Test Descriptions/Abstracts DC/COL/OL Applicants A. The applicant should provide abstracts of planned tests to					

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	demonstrate and verify the performance capabilities of SSCs and design features that serve the following functions: i. Used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintenance of the reactor in a safe condition for an extended shutdown period ii. Used for safe shutdown and cooldown of the reactor under transient conditions (infrequently or moderately frequent events) and postulated accident conditions and for maintenance of the reactor in a safe condition for an extended shutdown period following such condition iii. Used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications iv. Classified as engineered safety features or used to support or ensure the operations of engineered safety features within design limits v. Assumed to function, or for which credit is taken, in the accident analysis for the facility, as described in the DCD or SAR (as applicable) vi. Used to process, store, control, measure, or limit the release of radioactive materials vii. Used in a special low-power testing program to be conducted at power levels no greater than 5 percent for the purpose of providing meaningful technical information beyond that obtained in the normal startup test program, as required for the resolution of TMI Action Item I.G.1 vii. Identified as risk significant in the design-specific probabilistic risk assessment B. The abstracts should include test objectives, prerequisites,					

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	test methods, significant parameters and plant performance characteristics to be monitored, and acceptance criteria in sufficient detail to establish the functional adequacy of the SSCs and design features tested. C. For new, unique, or first-of-a-kind design features used in the facility, the functional testing requirements and acceptance criteria necessary to verify their performance should be submitted for review and approval. D. If the testing method will not subject the SSC to representative design operating conditions, the test abstract should contain sufficient information to justify the proposed test method.					
14.2.6	Initial Test Program Acceptance Criteria DC Applicants A. The applicant should provide in Tier 1 a general description of the preoperational and power ascension test programs and the major program documents that define how the ITP will be conducted and controlled (i.e., a site-specific startup administrative manual, test specifications, and test procedures). Tier 2, Chapter 14.2, should contain a complete description of the ITP. B. The applicant should describe the key elements of the ITP in Tier 1 to ensure that the COL applicant cannot unilaterally initiate subsequent changes in the conduct of the ITP. C. The applicant should include provisions to ensure that test procedures and test specifications are made available to the NRC. COL/OL Applicants					

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	A. Applicants referencing a certified design should provide a clearly and sufficiently described ITP in terms of scope and level of detail in accordance with the rule certifying the design and the design control document. B. An applicant which does not reference a certified design should provide a clearly and sufficiently described ITP in terms of scope and level of detail in accordance with RG 1.68. C. Refer to SRP Section 14.3.10 for additional guidance.					
14.2.1 (08/2006)	Generic Guidelines for Extended Power Uprate Testing Programs					Exclude; Not applicable to scope of review.
14.3 (03/2007)	Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.1	Acceptance on the scope of the ITAAC is based on the complete facility or for a DC application, limited to the SSCs covered by the DC.					
14.3.2	Acceptance criteria on the sufficiency of the ITAAC for the areas of review are specified in SRP Section 14.3 subsections.					
14.3.1 (03/2007)	[Reserved]	NA				Exclude
14.3.2 (03/2007)	Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.2.1	The reviewer should primarily utilize the NRC rules and regulations to review the top level commitments in Tier 1. Other sources of review guidelines include RGs, SRP guidelines, and PRA insights from the standard design safety and severe accident analyses and operating experience. If applicable, the staff also must adhere to policy decisions by the Commission. Examples of these are contained in the SRM related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance in the SRM related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor					

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	Designs." The SRM related to SECY-93-087 is dated July 21, 1993.					
14.3.2.2	Design descriptions, figures (including key dimensions) and ITAAC should be developed and grouped by systems and building structures. For building structures, the structural capability is typically verified by performing an analysis to reconcile the as-built data with the structural design bases for each safety-related building. System-specific performance tests are typically conducted to demonstrate that the system can perform its intended function. For major components, the verification of design, fabrication, testing, and performance requirements should be partially addressed in conjunction with the specific system ITAAC. The review checklists for fluid systems, electrical systems, and building structures in Appendix C of SRP Section 14.3 should be used as aids for establishing consistency and completeness for the Tier 1 information.					
14.3.2.3	Review of the Standard Design Structural Integrity. The scope of structural design covers the major structural systems in the standard design plant, including the RPV, ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, reactor building, control building, turbine building, service building, and radwaste building). For PWRs, this includes the reactor vessel (RV), ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, nuclear island structures, turbine building, component cooling water (CCW) heat exchanger structures, diesel fuel storage structures (DFSSs), and radwaste building). The RPV, piping systems, and primary containment (For PWRS, RV, piping systems, and primary containment) are included because they provide the defense-in-depth principle for nuclear plants. The major building structures house those systems and components that are important to safety. In establishing the top level requirements for structural design, the					

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	staff used the General Design Criteria (GDC) of 10 CFR Part 50, Appendix A, as its basis. The primary general design criteria pertaining to the major structural system design are GDC 1, "Quality Standards and Records," GDC 2, "Design Bases for the Protection Against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Basis," GDC 14, "Reactor Coolant Pressure Boundary," GDC 16, "Containment Design," and GDC 50, "Containment Design Basis." GDC 1 requires, in part, the need for structures, systems and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. GDC 2 requires, in part, the need to design structures, systems, and components important to safety to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods without loss of capability to perform their safety functions, including the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena. GDC 4 requires, in part, the need to protect structures, systems, and components important to safety from dynamic effects including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. GDC 14 requires, in part, the need for the reactor coolant pressure boundary to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 16 requires, in part, the need for the reactor containment to					

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	provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. GDC 50 requires, in part, the need for the reactor containment structure including access openings and penetrations to be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. Using the above GDC as its basis, the following top-level attributes should be verified by ITAAC: A. pressure boundary integrity (GDC 14, 16 and 50) B. normal loads (GDC 2) C. seismic loads (GDC 2) D. suppression pool hydrodynamic loads (GDC 4) E. flood, wind, and tornado (GDC 2) F. rain and snow (GDC 2) G. pipe rupture (GDC 4) H. codes and standards (GDC 1) I. 10 CFR 50, Appendix J (GDC 16) In addition, to ensure that the final as-built plant conforms to the certified design, applicants should provide ITAAC to reconcile the as-built plant with the structural design basis. A summary of the top-level structural design requirements for the major structural systems that are verified by the structures and systems in Tier 1 and the piping design information in Tier 1.					
14.3.2.4	Pressure Boundary Integrity. To ensure that the applicable requirements of GDC 14, 16, and 50 have been adequately addressed, ITAAC should be established to verify the pressure boundary integrity of the RPV, piping, and primary containment					

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	(For PWRs, RV, piping, and primary containment) for the standard design. GDC 16, GDC 50, and 10 CFR 50, Appendix J apply to the primary containment and GDC 14 applies to the RPV (RV for PWRs) and the reactor coolant pressure boundary piping systems. The pressure integrity for these major structural systems are needed to ensure the defense in-depth principle. For the RPV and piping, hydrostatic tests performed in conjunction with the ASME Boiler and Pressure Vessel Code, Section III, should be required by ITAAC. See the standard ITAAC for hydrostatic tests in Appendix D to SRP Section 14.3. For the primary containment, a structural integrity test and containment integrated leakage rate test should be required by ITAAC to be performed on the pressure boundary components of the primary containment in accordance with the ASME Boiler and Pressure Vessel Code, Section III, and 10 CFR 50, Appendix J. Because the requirements of GDC 14, 16, and 50 do not apply to the reactor, control, turbine, service, and radwaste buildings (nuclear island structures, turbine building, CCW heat exchanger structures, DFSSs, and radwaste building for PWRs), ITAAC are not required to verify the pressure integrity for these other buildings.					
14.3.2.5	Normal Loads. To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC should be established to verify that the normal and accident loads have been appropriately combined with the effects of natural phenomena. For piping systems, ITAAC should require an analysis to reconcile the as-built piping design with the design-basis loads (which include the appropriate combination of normal and accident loads). See SRP Section 14.3.3 for additional information. For the RPV, the fabrication may be performed primarily in the vendor's shop where adherence to design drawings is tightly controlled.					

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	Therefore, ITAAC for the as-built reconciliation of normal loads with accident loads for the RPV are inappropriate. Instead, ITAAC should verify that the ASME Code-required reports exist to document that the RPV has been designed, fabricated, inspected, and tested to Code requirements to ensure adequate safety margin. Similarly, for safety-related buildings, ITAAC should require an analysis for reconciling the as-built plant with the structural design basis loads (which include the combination of normal and accident loads with the effects of natural phenomena). The analysis results should be documented in a structural analysis report, the scope and contents of which must be described in Tier 2. The staff may determine that the design of certain structures does not require verification by ITAAC, based on their safety significance. In particular, these ITAAC should apply only to safety-related structures and are not applicable to the service and turbine buildings (radwaste and turbine building for PWRs). However, ITAAC for other design aspects of structures may be appropriate.					
14.3.2.6	Seismic Loads. To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC are established to verify that the safety-related systems and structures have been designed to seismic loadings. Component qualification for seismic loads should be addressed by ITAAC for verifying the basic configuration of systems. See the standard ITAAC for basic configuration in Appendix D to SRP Section 14.3 for additional information, and the discussion in SRP Section 14.3.3. As discussed above for normal loads on piping systems and the RPV, ITAAC should require an analysis to reconcile the as-built piping design with the design basis loads (which include seismic loads). See also the discussion in SRP Section 14.3.3. For the RPV, ITAAC for the as-built reconciliation of seismic loads for the					

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	RPV are deemed to be inappropriate as previously discussed. Instead, ITAAC verify that the ASME Code-required reports exist for the RPV ensuring that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements. For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design-basis loads (which include seismic loads). The analysis results are to be documented in a structural analysis report, as discussed above. These ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings (radwaste and turbine building for PWRs). However, because the leakage path for fission products includes components within the turbine building, the turbine building should be able to withstand the effects of a safe-shutdown earthquake, if not, ITAAC should be established to verify that, under seismic loads, the collapse of the turbine building will not impair the safety-related functions of any safety-related SSCs located adjacent to or within the turbine building. For non-seismic Category I SSCs, the need for ITAAC to verify that their failure will not impair the ability of near-by safety-related SSCs to perform their safety-related functions should be assessed based on the specific design. If the design detail and as-built and as-procured information for many non-safety-related systems (e.g., field-run piping and balance-of-plant systems) is not provided by the applicant for design certification and the spatial relationship between such systems and seismic Category I SSCs cannot be established until after the as-built design information is available, the non-seismic to seismic (II/I) interaction cannot be evaluated until the plant has been constructed. Accordingly, the design criteria for ensuring acceptable II/I interactions and a commitment for the COL applicant to describe the process for completion of the design of					

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	balance-of-plant and non-safety related systems to minimize II/I interactions and proposed procedures for an inspection of the as-built plant for II/I interactions should be specified as a COL action item in Tier 2.					
14.3.2.7	Suppression Pool Hydrodynamic Loads (BWRs only). To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC should be established to verify that the safety-related systems and structures have been designed to withstand suppression pool hydrodynamic loadings, which include safety relief valve discharge and loss-of-coolant accident (LOCA) loadings. Component qualification for suppression pool hydrodynamic loads may be addressed by ITAAC established for verifying the basic configuration of systems. As discussed above for seismic loads on piping systems and the RPV, ITAAC should require an analysis to reconcile the as-built piping design with the design- basis loads (which include suppression pool hydrodynamic loads). For the RPV, ITAAC should verify that the ASME Code-required reports exist to ensure that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements. For the reactor building and primary containment including the internal structures, ITAAC should require an analysis for reconciling the building as-built configuration with the structural design basis loads (which include suppression pool hydrodynamic loads). The as-built analysis results should be documented in a structural analysis report as discussed above. This report may be able to be satisfied using the ASME Code-required reports for the reconciliation analysis for the primary containment. The effects of suppression pool hydrodynamic loads do not extend beyond the reactor building, and, thus, ITAAC are not required to verify these loadings for the building structures outside the reactor building.					

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	ITAAC also should require the verification of the horizontal vent system, water volume, and the safety-relief valve discharge line quencher arrangement to ensure adequacy of the suppression pool hydrodynamic loads used for design.					
14.3.2.8	Flood, Wind, Tornado, Rain, and Snow. To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC should be established to verify that the safety-related systems and structures have been designed to withstand the effects of natural phenomena other than those associated with seismic loadings. The effects include those associated with flood, wind, tornado, rain, and snow. These loadings do not apply to the RPV, the ASME Code Class 1, 2, and 3 piping systems and components, nor the primary containment (except for the exposed portions of the concrete containments) because they are all housed within the safety-related buildings. For safety-related buildings, ITAAC should require an analysis for reconciling the as-built plant with the structural design basis loads (which include the flood, wind, tornado, rain, and snow loads). Based on their safety significance, these ITAAC need apply only to safety-related structures and need not be applicable to the service and turbine buildings (radwaste and turbine building for PWRs). For flooding, site parameters are specified that require the maximum flood level and ground water level be below the finished plant grade level. ITAACs also require inspections to verify that divisional flood barriers and water-tight doors exist, and penetrations (except for water-tight doors) in the divisional walls are sealed up to the internal and external flood levels. In addition, for safety-related buildings, flood barriers are established up to the finished plant grade level to protect against water seepage, and flood doors and flood barrier penetrations are provided with flood protection features.					

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	ITAAC should also require inspections to verify that water-tight doors exist, penetrations (except for water-tight doors) in the divisional walls are at least 2.5 m above the floor, and safety-related electrical, instrumentation, and control equipment are located at least 20 cm above the floor surface. In addition, for safety-related buildings, ITAAC should require that external walls below flood level are equal to or greater than 0.6 m to protect against water seepage, and penetrations in the external walls below flood level are provided with flood protection features.					
14.3.2.9	Pipe Break. To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC should be established to verify that the safety-related SSCs have been designed to the dynamic effects of pipe breaks. Component qualification for the dynamic effects of pipe breaks should be addressed by ITAAC established for verifying the basic configuration of systems. For the RPV, ITAAC that verify the basic configuration of the RPV system require an inspection of the critical locations that establish the bounding loads in the LOCA analyses for the RPV to ensure that the as-built areas not exceed the postulated break areas assumed in the LOCA analyses. In addition, ITAAC should be established to verify by inspections of as-built, high-energy pipe break mitigation features and of the pipe break analysis report that safety-related SSCs be protected against the dynamic and environmental effects associated with postulated high-energy pipe breaks. ITAAC to verify pipe break loads are not required for the turbine, service, and radwaste buildings (turbine and radwaste buildings for PWRs) either because they are not safety-related structures or there are no high-energy lines located within the structure.					
14.3.2.10	Codes and Standards. To ensure that the applicable requirements					

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	of GDC 1 have been adequately addressed, ITAAC should be established to verify that appropriate codes and standards are used in the design and construction of safety-related systems and components. In general, the staff considers those codes and standards endorsed by the regulations under 10 CFR 50.55a in determining which codes and standards were appropriate for Tier 1 verification. The ASME Boiler and Pressure Vessel Code, Section III for Code Class 1, 2, and 3 systems and components is established as the code for the design and construction of standard design piping systems and the RPV. For safety-related building designs, the staff should base its safety findings on audits of standard design calculations which relied on specific codes and standards. These codes and standards are contained in the appropriate sections of DCD Tier 2 Chapter 3. Inspections will be conducted as a part of ITAAC to verify that ASME Code-required documents exist that demonstrate that the RPV, piping systems and containment pressure boundaries have been designed and constructed to their appropriate Code requirements. For other ASME Code components and equipment, the verification of Code compliance will be performed in conjunction with the quality assurance programs and by the authorized inspection agency as required by the ASME Boiler and Pressure Vessel Code. This DCD Tier 2 material should be considered for designation as Tier 2* information. Tier 2* information is information that, if considered for a change by an applicant or licensee that references the certified standard design, would require NRC approval prior to implementation of the change. Tier 2* material is discussed further in SRP Section 14.3.					
14.3.2.11	As-built Reconciliation. As discussed in various sections above, to ensure that the final as-built plant structures are built in accordance with the certified design as required by 10 CFR Part					

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	52, structural analyses should be performed which reconcile the as-built configuration of the plant structures with the structural design bases of the certified design. The structural analyses should be documented in structural analysis reports. Structural analysis reports should be verified in conjunction with ITAAC for the primary containment and the reactor, control, radwaste, and turbine buildings (nuclear island structures, radwaste building, CCW heat exchangers, DFSSs, and turbine building for PWRs). The detailed supporting information on what is required for an acceptable analysis report should be contained in DCD Tier 2 Chapter 3. Similarly for piping systems, an as-built analysis should be performed using the as-designed and as-built information. ITAAC should verify the existence of acceptable final as-built piping stress reports that conclude the as-built piping systems are adequately designed. See SRP Section 14.3.3 for additional information. For the RPV, the key dimensions of the RPV system should be verified in conjunction with the basic configuration check of the system. The key dimensions of the RPV system and the acceptable variations of the key dimensions should be provided in the certified design description. Alternatively, acceptable variations and the bases for them should be provided in Tier 2. For component qualification, tests, analyses, or a combination of tests and analyses should be performed for seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) to demonstrate that the as-built equipment and associated anchorages are qualified to withstand design basis dynamic loads without loss of safety function. These test and analyses should be performed as a part of ITAAC to verify the basic configuration of the system in which the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	equipment is located. See Section 14.3.3 for additional information.					
14.3.3 (03/2007)	Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.3.1	Generic Piping Design. DC applicants may provide less than the complete design information for piping design before DC because the design may depend upon as-built and as-procured information. Instead, applicants may provide the processes and design acceptance criteria (DAC) by which design details in this area would be developed and evaluated. Implementation of the processes is the responsibility of the COL applicant or licensee. The DAC are discussed further in to SRP Section 14.3, Appendix A. The reviewer should use the SRP guidelines to evaluate the piping design information in Tiers 1 and 2 and audit the piping design criteria in detail, including sample calculations. The staff should evaluate the adequacy of the structural integrity and functional capability of safety-related piping systems. The review is not limited to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Classes 1, 2, and 3 piping and supports, but includes buried piping, instrumentation lines, the interaction of non-seismic Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation should include the analysis methods, design procedures, acceptance criteria, and related ITAAC (and DAC if applicable) that are to be used for the completion and verification of the standard design piping design. The staff's evaluation should include both DCD Tier 1 and Tier 2 information on the applicable codes and standards, analysis methods to be used for completing the piping design, modeling techniques, pipe stress analyses criteria, pipe support design criteria, high-energy line break					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	criteria, and leak-before-break (LBB) approach applicable to the standard design. Design descriptions and the associated DAC should be specified in Tier 1. The scope of the standard design to which the piping design information applies should be stated in Tier 1. This may be done on a generic basis using a single ITAAC applicable to multiple systems of the design, or applied to individual system ITAAC. If done using a generic piping design ITAAC, the Tier 1 should address its application to piping systems classified as both nuclear safety-related and non-nuclear safety systems. The nuclear safety-related piping systems must remain functional during and following a safe-shutdown earthquake (SSE), and should be designated in Tier 1 as seismic Category I and further classified as ASME Code Class 1, 2, or 3 in the individual systems of the standard design. Tier 1 should ensure that the piping systems will be designed to perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. The material in Tier 1 should also address the consequential effects of pipe ruptures such as jet impingement, potential missile generation, and pressure and temperature effects. The scope of the piping to be verified by the generic Piping ITAAC includes all ASME Code Class 1, 2, or 3 piping systems and high-energy piping systems. Tier 1 includes ASME Code Class piping systems because the ASME Boiler and Pressure Vessel Code, Section III is referenced in 10 CFR 50.55a. Nuclear power plant components classified as Quality Groups A, B, and C are required by 10 CFR 50.55a to meet the requirements for ASME Code Class 1, 2, or 3, respectively. In each system description, the functional drawing identifies the boundaries of the ASME Code classification for the piping systems. The piping pressure					

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	boundary and structural integrity are required to be maintained because they are directly involved in preventing or mitigating an accident or event under the defense-in-depth principle. An acceptable approach to Tier 1 information for piping design is to specify distinct ITAAC that ensure the design process for piping systems occurs as described in the design description. For example, the first ITAAC specified in Tier 1 should require that an ASME Code certified stress report exists to ensure that the ASME Code Class 1, 2, or 3 piping systems and components are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design basis loads. The specific contents and requirements of the certified stress report are contained in the ASME Code. The particular certified stress report to be used to satisfy the ITAAC should be specified in Tier 2. An acceptable version of an ASME Code certified stress report is the design document required by ASME Code, Section III, Subarticle NCA-3550. A certified piping stress report provides assurance that requirements of the ASME Code, Section III for design, fabrication, installation, examination, and testing have been met and that the design complies with the design specifications. A second ITAAC should require that a pipe break analysis report exists that documents that SSCs that are required to be functional during and following an SSE have adequate high-energy pipe break mitigation features. The design description should discuss the criteria used to postulate pipe breaks, the analytical methods used to perform pipe breaks, and the method to confirm the adequacy of the results of the pipe break analyses. The design description should be verified in a Pipe Break Analysis Report that provides assurance that the high-energy line break analyses have been completed. For postulated pipe breaks, the report confirms whether (A) piping stresses in the containment penetration area					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	are within allowable stress limits, (B) pipe whip restraints and jet shield designs can mitigate pipe break loads, (C) loads on safety-related SSCs are within design load limits, and (4) SSCs are protected or qualified to withstand the environmental effects of postulated failures. The Pipe Break Analysis Report shall conclude that, for each postulated piping failure, the reactor can be shut down safely and maintained in a safe, cold shutdown condition without offsite power. Detailed information that supports this ITAAC should be contained in DCD Tier 2 Chapter 3. If the design uses Leak-Before-Break (LBB) methods, a third ITAAC should require that a LBB evaluation report exists which documents that LBB acceptance criteria are complied with for the as-built piping and piping materials. Bounding limits should be specified in Tier 2 using preliminary piping analysis results to establish a window of acceptable piping stress values for selected piping materials. The ITAAC verifies that these values are complied with using actual material properties and final piping configurations, and reconciles the as-built piping data with the LBB assumptions. Detailed information that supports this ITAAC should be contained in DCD Tier 2 Chapter 3. A fourth ITAAC should require that an as-built piping stress report exists that documents the results of an as-built reconciliation analysis confirming that the final piping system has been built in accordance with the ASME Code certified stress report. The report provides an overall verification by inspection that the as-constructed piping system, including supports, are consistent with the certified design commitments. Specific attributes to be inspected should be described in the DCD Tier 2. Although similar to the first ITAAC, this verification also provides assurance that the as-built documentation used for construction has been reconciled with the documentation used for design analysis and with the certified stress report discussed above. The inspection					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	will also involve a review of the as-built, high-energy pipe break mitigation features (e.g., pipe whip restraints and jet impingement shields) to ensure that the installed features are consistent with the pipe break analysis report. The methodology and specific attributes to be inspected are described in the DCD Tier 2. Alternatively, if an NRC-approved LBB report exists, then the dynamic effects from those postulated high-energy pipe breaks could be excluded. The documentation for this as-built reconciliation review may become part of the certified stress report. Selected material in DCD Tier 2 Chapter 3 provides design information and defines design processes that are acceptable for use in meeting the piping DAC in Tier 1. However, Tier 2 information may be changed by a COL applicant 1 or licensee referencing the certified design in accordance with a "50.59-like" process specified in the rule certifying the design. The staff's evaluation of the standard design for piping systems is based on the design processes and acceptance criteria material in the DAC and Tier 2. Consequently, the staff should consider designating selected aspects of these piping design processes as Tier 2* information. Tier 2* information is Tier 2 information that, if considered for a change by a COL applicant or licensee, requires NRC approval prior to implementation of the change. Consideration should also be given to allowing the designation of Tier 2* to expire at the first full power when the detailed design is complete and performance characteristics of the facility are known. Although applicants for design certification should propose designating similar Tier 2* information to that in the DCDs for the evolutionary designs, the NRC bears the final responsibility for designating which material is Tier 2*. The basis for the use of Tier 2* should be discussed in the staff's safety evaluation report. The Tier 2* information is discussed further in Appendix A to SRP Section 14.3.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Regulations, Codes and Standards. The use of codes and standards in the certified design material (CDM) should be minimized with exceptions granted case by case. Instead, the applicable requirements from the regulations, codes, or standards should be stated in the CDM, rather than reference them. This ensures that the requirement is clear, and allows flexibility if the reference changes. References to various parts of ASME Sections III and XI may verify issues like pressure boundaries or pre-service inspection requirements. Also, references to 10 CFR Part 20 may be required for radiation protection. The specific code edition, volume, version, date, etc., should be specified in the site safety analysis report rather than Tier 1. This provides for specific requirements that are acceptable, yet allows the code to be updated via the change process in the rule certifying the design. It is important to note that, due to the provisions of 10 CFR 52.63 and the rule certifying the design, changes to the codes and standards in 10 CFR 50.55a would not necessarily be requirements for the certified design.					
14.3.3.2	Verifications of Components and Systems. In addition to the generic approach to piping design in Tier 1, the verification of piping and component classification, fabrication, dynamic and seismic qualification, and selected testing and performance requirements is also addressed by specific ITAAC in the individual Tier 1 systems. A. Piping and Component Safety Classification. 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, requires that safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions performed. Nuclear power plant components classified as Quality Groups A, B, and C are required by 10 CFR 50.55a to meet the requirements for ASME Code Class 1, 2, or 3,					

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	respectively; therefore, SSC safety classifications should be in each system's design description, and the functional drawings should identify the ASME Code classification boundaries applicable to the safety class. The ASME Code classes in ASME Code, Section III, allow a choice of rules that provide assurance of structural integrity and quality commensurate with the relative importance assigned to the individual items of the nuclear power plant. The ASME Boiler and Pressure Vessel Code class requirements may be verified by either a generic piping design ITAAC or by each system ITAAC. The use of other codes and standards (e.g., American Institute of Steel Construction manual for building structural steel) is within the Tier 2 scope, and the DCD Tier 2 describes the applicable codes and standards for these other safety-related SSCs not designed to the ASME Boiler and Pressure Vessel Code, Section III. B. Fabrication (Welding). 10 CFR Part 50, Appendix A, GDC 14, requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage. In addition, GDC 30 requires that component parts of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. The ASME Code class welds are included in Tier 1 because the ASME Boiler and Pressure Vessel Code, Section III is referenced in 10 CFR 50.55a, which requires nuclear power plant components classified as Quality Groups A, B, and C to meet ASME Code Class 1, 2, or 3 requirements, respectively. In each system description, the functional drawing shows the boundaries of the ASME Code classification. The integrity of the pressure boundary					

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	is required to be maintained because it is directly involved in preventing or mitigating an accident or event under the defense-in-depth principle. ASME Code Class 1, 2, or 3 structural welds (e.g., pipe support welds) are not within Tier 1 scope because they indirectly prevent or mitigate accidents or events (e.g., pipe supports protect the piping but the piping itself is needed for accident mitigation). Thus, ASME Code Class 1, 2, or 3 structural welds are in the Tier 2 scope. The integrity of the pressure boundary in the plant will be ensured, in part, through a verification of the welding quality. This verification is performed as a part of the basic configuration ITAAC of each specific system. The basic configuration ITAAC, one of the standard ITAAC listed in SRP Section 14.3, Appendix D, is required for most systems in Tier 1. The provisions of the basic configuration check that must be specified in Tier 1 include non-destructive examination of the as-built pressure boundary welds for the ASME Code Class 1, 2, or 3 SSCs in the design description. The acceptance criteria for the welds are the ASME Code, Section III weld examination requirements. The specific weld examination requirements for a particular ASME Code Class 1, 2, or 3 component and weld type are tabulated in Tier 2. The specific weld examination requirements are considered Tier 2 because they could change depending on future revisions to the ASME Code, Section III requirements. Other welding activities (non-ASME Code) include: i. Pressure-boundary welds other than					

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	ASME Code, Section III welds, ii. Structural and building steel welds, iii. Electrical cable tray and conduit support welds, iv. Heating, ventilation, and air-conditioning support welds, and v. Refueling cavity and spent fuel pool liner welds. These other types of welding are included in the Tier 2 scope. Tier 2 describes the applicable codes and standards for the other types of welding and the weld acceptance criteria. Similar to the ASME Code Classes 1, 2, and 3 structural welds, these other welds are needed for protection of safety-related SSCs but do not directly (or are redundant) prevent accidents or events. Accordingly, these other types of welding were deemed inappropriate for Tier 1 scope. C. Hydrostatic Test. The integrity of the pressure boundary is required to be maintained because it is directly involved in preventing or mitigating an accident or event under the defense-in-depth principle. The pressure boundary integrity is also ensured, in part, through a hydrostatic test verifying the leak-tightness of the ASME Code piping systems. A hydrostatic test is generally specified by the ASME Code, Section III, for ASME Code Class 1, 2, and 3 SSCs to verify whether pressure integrity is maintained in the process of fabricating the overall piping system, including any welding and bolting requirements. However, the ASME piping stress report in the generic piping ITAAC does not include the results of hydrostatic tests; therefore, the standard hydrostatic test ITAAC in SRP Section 14.3, Appendix D, should be specified in each system ITAAC					

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	with ASME Code Class 1, 2, or 3 SSCs. The hydrostatic test ITAAC also may be specified in other appropriate Tier 1 systems. D. Equipment Seismic and Dynamic Qualification. The basic configuration ITAAC listed in SRP Section 14.3, Appendix D, include verifications of the dynamic qualification (e.g., seismic, loss-of-coolant accident, and safety relief valve discharge loads) of seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) in the design descriptions and figures. This inspection verifies the capability of mechanical and electrical equipment in as-built condition, including anchorages, to perform safety functions during and following a SSE. Detailed supporting information for dynamic qualification requirements, including seismic qualification records, is in DCD Tier 2, Chapter 3. The Tier 2 information describing dynamic qualification of equipment should be considered for designation as Tier 2*. Tier 2* information is addressed further in SRP Section 14.3, Appendix A. E. MOVs and Other Valves. The verification of the design qualification of valves is performed in conjunction with the basic configuration check for mechanical equipment as discussed above. For MOVs in particular, a special inspection is part of the basic configuration check to verify the records of vendor tests that demonstrate MOV ability to function under design conditions. The list of MOVs in Tier 1 should include, but not be limited to, those with active safety-related functions. These may be listed in Tier 2 in the inservice testing plan or other locations. The DCD Tier 2, Chapter 3 material should have detailed supporting information for the CDM for the methods of the COL					

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	applicant or licensee for the design, qualification, and testing of MOVs to demonstrate their design-basis capability. This material should be considered for designation as Tier 2* information. Tier 2* information is addressed further in SRP Section 14.3, Appendix A. In-situ testing of installed MOVs, POVs, and check valves, to verify whether they can perform intended functions under various fluid flow, differential pressure, electrical, and temperature conditions, should be conducted as appropriate in the applicable system ITAAC. Standard ITAAC are provided in Appendix D to SRP Section 14.3 for verification of the performance of these valves. These may be performed as part of the pre-operational test program. Tier 2 information should be provided that defines that these tests will be conducted under maximum achievable pre-operational conditions and describes the analyses that will be performed to show how the test results demonstrate that the valves will function under design basis conditions (See Tier 2 Section 3.9.6). For significant operating problems with other types of valves, or with pumps in general, the proper operation of these components may be implicitly tested, if applicable, as part of other functional tests in the system ITAAC. They also may be tested in the pre-operational or power ascension test program.					
14.3.4 (03/2007)	Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.4.1	Appendix A of SRP 14.3 describes and provides guidance relative to the content of the DCD for a design certification application and defines Tier 1 and Tier 2 design-related information that is to be ultimately incorporated by reference into the design certification rules. The basis for identifying Tier 1 information as derived from Tier 2 information, which is essentially the same information as is required for a design certification application, is that the top-level					

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	design features and performance standards (Tier 1) are those that are most important to safety, including safety-related and defense-in-depth features and functions, and non-safety-related systems that potentially impact safety. Tier 1 should be reviewed to verify that plant safety analyses, such as for core cooling, transients, overpressure protection, steam generator tube rupture, and anticipated transients without scram (ATWS), are adequately addressed. Applicants should provide tables in DCD Tier 2 Section 14.3 to show how the important input parameters used in the transient and accident analyses for the design are verified by the ITAAC. For intersystem LOCAs, the design pressure of the piping of the systems that interface with the reactor coolant pressure boundary should be specified in the design descriptions or figures. The specific fuel, control rod, and core designs presented in Tier 2 constitute an approved design that may be used for the COL first-cycle core loading without further NRC staff review. If any other core design is requested for the first cycle, the COL applicant or licensee will be required to submit for staff review those specific fuel, control rod, and core design analyses as described in DCD Tier 2 Chapters 4, 6, and 15. Much of the detailed supporting information in Tier 2 for the nuclear fuel, fuel channel, and control rods, if considered for a change by a COL applicant or licensee that references the certified standard design, would require prior NRC approval. Therefore, for the evolutionary and passive designs, the staff concluded that this information should be designated as Tier 2* information (see Appendix A of SRP Section 14.3 for a definition). However, staff will allow some of the Tier 2* designations to expire after the first full-power operation of the facility when the detailed design has been completed and the core performance characteristics are known from the startup and power-ascension test programs. The NRC					

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	bears the final responsibility for designating which material in Tier 2 is Tier 2*. The following issues are identified to ensure comprehensive and consistent treatment of Tier 1 based on the safety significance of the system being reviewed: a. System purpose and functions b. Location/functional arrangement of system c. Key design features of the system d. System operation in various modes e. Seismic and ASME code classifications f. Materials—weld quality and pressure-boundary integrity g. Controls, alarms, and displays h. Logic i. Interlocks j. Class 1E electrical power sources and divisions k. Equipment to be qualified for harsh environments l. Valve qualification and operation m. Interface requirements with other systems n. Numeric performance values (flow rates, capacities, etc.) o. Accuracy and quality of figures p. Active systems that provide defense-in-depth functions designated as non-safety systems Appendix C to SRP 14.3 provides “checklists” for the fluid systems as an aid for establishing consistency and comprehensiveness in the review of the system.					
14.3.4.2	The source of information used to determine safety significance of SSCs for the design of reactor and core cooling systems include applicable rules and regulations, general design criteria,					

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	unresolved safety issues, and generic safety issues, NRC generic correspondence, PRA, insights from the standard design's safety and severe accident analyses, and operating experience. Inputs from the PRA review, including shutdown safety evaluations, and severe accident analyses ensure important insights and design features from these analyses are incorporated into Tier 1. For both PRA and severe accident analyses, although large uncertainties and unknowns may be associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses should be selected for treatment in Tier 1.					
14.3.4.3	The passive-designed reactors use safety systems that employ passive means (natural forces), such as gravity, natural circulation, condensation and evaporation, and stored energy, for accident mitigation. These designs also include active systems that provide defense-in-depth capabilities for reactor-coolant makeup and decay heat removal. These active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. SECY-95-132, "Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)" provides certain guidance and positions for ensuring consistent and complete treatment of those systems that might be classified as non-safety-related by the designer or applicant but are important to safety or otherwise provide defense-in-depth functions.					
14.3.4.4	Applicable regulatory guidance from the Commission for selected policy and technical issues related to particular design should be followed. For the severe accident analyses, the basis for the staff's review for the evolutionary and passive standard designs was the Commission guidance related to SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and					

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	Their Relationship to Current Regulatory Requirements.” SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs” generically presents guidance and NRC positions on evolutionary and passive LWR design certification issues. For guidance, positions, and issues related to specific designs, guidance is available in such documents as SECY-97-044, “Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design” or SECY-92-137, “Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) Requirements for the General Electric (GE) Advanced Boiling Water Reactor (ABWR).” Regarding DAC, SECY-02-059, “Use of Design Acceptance Criteria for the AP1000 Standard Plant Design,” presents staff conclusions on acceptable use of DAC for I&C, control room, and piping design areas, contingent upon Westinghouse’s and the staff’s agreeing on adequate DAC during the design certification review. In SECY-92- 053, “Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Process,” the staff noted that DAC is defined as “a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification.” In some instances, an applicant may employ DAC to provide the staff with information to support its safety determination process. In SECY-92-053, the staff noted that “the concept of DAC would enable the staff to make a final safety determination, subject only to satisfactory design implementation and verification by the COL licensee through appropriate use of ITAAC.” The staff defined DAC as “a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support					

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	a design certification. The DAC are to be objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. That is, the acceptance criteria for DAC become the acceptance criteria for ITAAC, which are part of the design certification." The use of DAC by applicants use for I&C is considered acceptable given the rapidly changing technology for digital I&C systems. For many of the design features, it might be impractical to test their functionality because of the absence of simulated severe accident conditions. An example might be the ability of the reactor cavity to absorb the heat and radiation effects of a molten core. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown and confirmatory test reports or analysis, may be considered sufficient Tier 1 treatment. Another example in which passive designs would be difficult to verify prior to fuel loading as related to normal operations involves natural circulation. Passive designs, compared to previous designs, can include elongated-reactor-core designs to create the pressure differential for establishing natural circulation. Evidence of prior testing and analysis providing conclusive results may have to suffice for suitable acceptance criteria for ITAAC purposes.					
14.3.4.5	Appendix D of SRP 14.3 lists acceptable "Standard ITAAC Entries" in the standard three-column format for ITAAC entries for configuration of systems, hydrostatic tests, net positive suction head for pumps, divisional power supply, etc., that should be contained in the overall set of ITAAC entries, as appropriate. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," contains guidance for developing ITAAC assuming that a COL applicant does not reference a certified design and/or an early site permit. Guidance in Section III for COLs referencing a certified design notes that the ITAAC					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	contained in the certified design must apply to those portions of the facility design that have been approved. Appendix C.II.2-A provides "general ITAAC development guidance" on fluid, I&C, and electrical systems.					
14.3.5 (03/2007)	Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.5.1	The methodology for selecting SSCs that will be subject to ITAAC as well as the criteria for establishing the necessary and sufficient ITAAC should be appropriate for and consistently applied to I&C systems.					
14.3.5.2	Tier 1 Design Descriptions (for DC and for COL referencing DC) and ITAAC Design Descriptions or ITAAC references to the FSAR (for COL not referencing DC) should describe the top-level I&C design features and performance characteristics that are significant to safety. For safety systems, this should include a description of system purpose, safety functions, equipment quality (e.g., meet the functional requirements of IEEE Std. 603-1991 and the digital system life cycle design process), equipment qualification, automatic decision-making and trip logic functions, manual initiation functions, and design features (e.g., system architecture) provided to achieve high functional reliability. The functions and characteristics of other I&C systems important to safety should also be discussed to the extent that the functions and characteristics are necessary to support remote shutdown, support required operator actions or assessment of plant conditions and safety system performance, maintain safety systems in a state that assures their availability during an accident, minimize or mitigate control system failures that would interfere with or cause unnecessary challenges to safety systems, or provide diverse back-up to protection systems.					

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	SRP Section 14.3, Appendix A, Subsection B.1, provides additional guidance on the content of Tier 1 Design Descriptions, ITAAC Design Descriptions, or ITAAC references to the FSAR.					
14.3.5.3	ITAAC should identify the significant features of the I&C systems on which the Staff is relying to assure compliance with each NRC requirement identified in SRP Appendix 7.1-A. Tests, analyses, and acceptance criteria associated with each design commitment should, when taken together, be sufficient to provide reasonable assurance that the final as-built I&C system fulfills NRC requirements. SRP Appendix 7.1-C provides an expanded discussion of SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h). SRP Appendix 7.1-D further discusses SRP acceptance criteria for safety and protection systems using digital computer-based technology. SRP Section 14.3, Appendix A, Subsection B.2, provides additional guidance on the expected scope, content, and format of ITAAC.					
14.3.5.4	For DC or for COL applications referencing a DC, Tier 1 Design Descriptions and ITAAC design commitments should be based on and consistent with the Tier 2 material. For a COL application not referencing a DC, the ITAAC Design Descriptions (if provided) and ITAAC design commitments should be based on and consistent with the FSAR portion of the application.					
14.3.5.5	The applicant may provide design acceptance criteria (DAC) in lieu of detailed system design information. In this case, the DAC should be sufficiently detailed to provide an adequate basis for the Staff to make a final safety determination regarding the design, subject only to satisfactory design implementation and verification of the DAC by the COL applicant or licensee.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Implementation of the DAC should be verified as part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design.					
14.3.6 (03/2007)	Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria					
	The standard design Class 1E electrical systems may include: (1) the Class 1E electrical power distribution system, (2) the emergency diesel generators (EDGs), (3) the Class 1E direct current power supply, and (4) the Class 1E vital ac and Class 1E instrument and control power supplies. Using the above regulations, IEEE standards, operating experience, and PRA as its bases, the applicant should establish top-level design commitments for the Class 1E electrical systems of the standard design to be included in the design descriptions and verified by ITAAC. The top-level design commitments for the Class 1E electrical systems include design aspects related to: 1. Equipment qualification for seismic and harsh environment To ensure that the seismic design requirements of GDC 2 and the EQ requirements of 10 CFR 50.49 have been adequately addressed, a "basis configuration" standard ITAAC may be established for applicable systems to verify these design aspects of electrical equipment important to safety. The Design Description should identify that Class 1E equipment is seismic Category 1 and equipment located in a harsh environment is qualified. The basic configuration standard ITAAC may be used to verify these areas. EQ of safe-shutdown equipment may be verified as part of the basic configuration ITAAC for safety-related systems. EQ treatment in the ITAAC would then be discussed in the					

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	General Provisions section of Tier 1. Verification may include type tests or a combination of type tests and analyses of Class IE electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed. Qualification of systems and components for seismic and harsh environments should be verified by ITAAC. Electrical equipment located in a "mild" environment should be discussed in the applicable sections of the COL application only. An exception is made for state-of-the-art digital instrumentation and control (I&C) equipment and digital control and protection systems located in an "other than harsh" environment. Operational experience has shown these state-of-the-art equipment and systems to be sensitive to temperature. ITAAC should be included to verify the qualification of equipment whose performance may be impacted by sensitivity to particular environmental conditions not considered by regulations to be harsh. 2. Redundancy and independence To ensure that the Class 1E electric systems meet the single failure requirements of GDC 17 (and other GDC), ITAAC may be established to verify the redundancy and independence of the Class 1E portion of the electrical design. For the electrical systems, ITAAC should verify the Class 1E divisional assignments and independence of electric power by both inspections and tests. The independence may be established by both electrical isolation and physical separation. Identification of the Class 1E divisional equipment should be included to aid in					

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	demonstrating the separation. (The detailed requirements are specified in Tier 2. For example, separation distances and identification are outlined in Tier 2). These attributes should be verified all the way to the electrically powered loads by a combination of the electrical system ITAAC and the ITAAC of the individual fluid, I&C, and heating, ventilation and air conditioning (HVAC) systems which also cover the electrical independence and divisional power supply requirements. ITAAC should be included to verify adequate separation, required inter-ties (if any), required identification (e.g., color coding), proper routing/termination (i.e., location), separation of non-Class 1E loads from 1E buses. Post-fire safe shutdown separation of electrical circuits should be addressed in the fire protection system ITAAC. 3. Capacity and Capability To ensure that the electrical systems have the capacity and capability to supply the safety-related electrical loads, ITAAC should be established to verify the adequate sizing of the electrical system equipment and its ability to respond (e.g., automatically in the times needed to support the accident analyses) to postulated events. This includes the Class 1E portion and the non-Class 1E portion to the extent that it is involved in supporting the Class 1E system. ITAAC should be included to analyze the as-built electrical system and installed equipment (diesel generators, transformers, switchgear, batteries, etc.) to verify its ability to power the loads. In addition, the ITAAC should also include tests to demonstrate the operation of the equipment. Testing should be included in ITAAC to verify EDG capacity					

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	and capability based on the Technical Specifications. In some cases regulatory guidance specifies the need for margin in capacity to allow for future load growth. If it is only for future load growth, ITAAC does not need to check for the additional margin. ITAAC should be developed to verify the initiation of the Class 1E equipment necessary to mitigate postulated events for which the equipment is credited (e.g., loss of coolant accident (LOCA), loss of offsite power (LOOP), and degraded voltage conditions). ITAAC should be included to analyze the as-built electrical power system for its response to a LOCA, LOOP, combinations of LOCA and LOOP (including LOCA with delayed LOOP and LOOP with delayed LOCA), and degraded voltage, including tests to demonstrate the actuation of the electrical equipment in response to postulated events. Analyses to demonstrate the acceptability of a voltage drop should be included in ITAAC to verify adequacy for supporting the accomplishment of a direct safety function. The applicable section of the COL application should include a discussion of how the voltage analyses will be performed, i.e., reference to industry standards. Testing should be included in ITAAC to verify the EDG voltage and frequency response is acceptable and is the same as that specified in the Technical Specifications. 4. Electrical protection features To ensure that the electrical power system is protected against potential electrical faults, ITAAC should be					

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	established to verify the adequacy of the electrical circuit protection included in the design. Operating experience and NRC Electrical Distribution System Functional Inspections (EDSFIs) have indicated some problems with the short circuit rating of some electrical equipment and breaker and protective device coordination. Inclusion in ITAAC should be based on the potential for preventing safety functions and the operating experience. ITAAC should be included to analyze the as-built electrical system equipment for its ability to withstand and clear electrical faults. ITAAC should also be included to analyze the protection feature coordination to verify its ability to limit the loss of equipment due to postulated faults. Equipment short circuit capability and breaker coordination should be verified by specifying ITAAC for analyses. The description of the analyses should be included in the applicable section of the application. Similarly, diesel generator protective trips (and bypasses if applicable) should be considered. 5. Displays/controls/alarms To help ensure that the electrical power system is available when required, ITAAC should be included to verify the existence of monitoring and controls for the electrical equipment. The minimum set of displays, alarms, and controls is based on the emergency procedure guidelines. In some cases, additional displays, alarms, and controls may be specified based on special considerations in the design and/or operating experience. ITAAC should be included to inspect for the ability to retrieve the information (displays and alarms), and to control the electrical power system in the main control room and/or at					

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	locations provided for remote shutdown. Detection of undervoltage conditions along with the starting and loading of EDG should be included in ITAAC. This is a direct safety function in response to design basis event of loss of power. Problems with relay settings should be considered in this requirement.					
	Other Electrical Equipment Important to Safety In addition to the Class 1E systems addressed above, other aspects of the electrical design that are deemed to be important to safety and the top-level design commitments are included in Tier 1. 1. Offsite Power To ensure that the requirements of GDC 17 for the adequacy and independence of the preferred offsite power sources within the standard design scope were met, ITAAC should verify the capacity and capability of the offsite sources to feed the Class 1E divisions, and the independence of those sources. ITAAC should be included to inspect the direct connection of the offsite sources to the Class 1E divisions and to inspect for the independence/separation of the offsite sources. ITAAC should be developed to inspect for appropriate lightning protection and grounding features. In addition, the Design Description includes "interface" requirements for the portions of the offsite power outside of the standard design scope; however, no ITAAC are included for the interfaces. The interfaces define the requirements that the offsite portion of the design (that is out-of-scope) must meet to support and not degrade the in-scope design					

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	(See also Appendix A to SRP Section 14.3). 2. Containment Electrical Penetrations To ensure the containment electrical penetrations (both those containing Class 1E circuits and those containing Non Class 1E circuits) do not fail due to electrical faults and potentially breach the containment, ITAAC should verify that all electrical containment penetrations are protected against postulated currents greater than their continuous current rating. 3. Alternate AC Power Source (if applicable) To ensure the availability of the alternate AC (AAC) power source for station blackout events, ITAAC should be developed to verify, through inspection and testing, the AAC power source (combustion gas turbines, diesel generators, or hydro units) and its auxiliaries along with its independence from other AC sources. 4. Lighting ITAAC should be included to verify the continuity of power sources for plant lighting systems to ensure that portions of the plant lighting remain available during accident scenarios and power failures. The basis for inclusion may be more related to defense in-depth, support function, operating experience, or PRA rather than "accomplishing a direct safety function." 5. Electrical Power For Non-Safety Plant Systems To ensure that electrical power is provided to support the					

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	non-safety plant systems, Design Descriptions cover portions of the non-Class 1E electrical systems. ITAAC should be included to verify the functional arrangement of electrical power systems provided to support non-safety plant systems to the extent that those systems perform a significant safety function.					
14.3.7 (03/2007)	Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.7.1	The reviewer should utilize the SRP in its review of Tier 1 to determine the safety significance of SSCs. Other sources include applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from the standard design's safety and severe accident analyses, and operating experience. Tier 1 should be reviewed for consistency with the initial test program described in DCD Tier 2 Chapter 14.2. The reviewer should also use the review checklists provided in Appendix C to SRP Section 14.3 as an aid for establishing consistency and comprehensiveness in his review of the systems. If applicable, the reviewer should utilize regulatory guidance from the Commission for selected policy and technical issues related to particular design. Examples of these are contained in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The SRM related to this is dated July 21, 1993.					
14.3.7.2	Tier 1 should be reviewed for treatment of design information proportional to the safety significance of the SSC for that system. Many items may be judged to be important to safety, and thus should be included in Tier 1. The following issues are identified to ensure comprehensive and consistent treatment in Tier 1 based on the safety significance of the system being reviewed: (1) System purpose and functions (2) Location of system					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	(3) Key design features of the system (4) Seismic and ASME code classifications (5) System operation in various modes (6) Controls, alarms, and displays (7) Logic (8) Interlocks (9) Class 1E electrical power sources and divisions (10) Equipment to be qualified for harsh environments (11) Interface requirements (12) Numeric performance values (13) Accuracy and quality of figures					
14.3.7.3	Standard ITAAC entries should be utilized to verify selected issues, where appropriate. The reviewer should ensure consistent application and treatment of the standard ITAAC entries for basic configuration ITAAC, net positive suction head, and physical separation for appropriate systems in Tier 1. In particular, the general provision for environmental qualification aspects of SSCs invoked by the basic configuration ITAAC should be reviewed to ensure appropriate treatment in Tier 1.					
14.3.7.4	Environmental qualification (EQ) of safe-shutdown equipment may be verified as part of the basic configuration ITAAC for safety-related systems. EQ treatment in the ITAAC would then be discussed in the General Provisions section of Tier 1. Verification may include type tests or a combination of type tests and analyses of Class 1E electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed.					
14.3.7.5	The design features in Tier 1 should be selected to ensure that the integrity of the analyses are preserved in an as-built facility. For example, 3-hour fire boundaries and divisional separation may be shown in the building figures. Also, flooding features such as structure elevations should be specified in the site parameters,					

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	flood doors may be shown on the building figures, and elevations are shown on the buildings to verify that the approximate physical location of components and relative elevations of buildings minimize the effects of flooding. As-built reconciliation reports for fires and floods to ensure consistency with Tier 2 analyses should be required by the appropriate system ITAAC (e.g., fire protection system) and selected building ITAAC, respectively.					
14.3.7.6	Other specific issues that should be addressed include heat removal capabilities for design-basis accidents and tornado and missile protection. Heat removal capabilities may be verified through heat removal requirements for core cooling system heat exchangers and interface requirements for site-specific systems. Tornado and missile protection may be provided by inlet and outlet dampers in ventilation systems, and through the structural design of buildings.					
14.3.7.7	The areas of review for radioactive waste systems include design objectives, design criteria, identification of all expected releases of radioactive effluents, methods of treatment, methods used in calculating effluent source terms and releases of radioactive materials in the environment, and operational programs in controlling and monitoring effluent releases and for assessing associated doses to members of the public. The radioactive waste systems include the liquid waste management system (LWMS), gaseous waste management system (GWMS), and the solid waste management system (SWMS). These systems deal with the management of radioactive wastes, as liquid, wet, and dry solids, produced during normal operation and anticipated operational occurrences. In addition, the review includes an evaluation of the process and effluent radiological monitoring instrumentation and sampling systems (PERMISS) which are used to monitor liquid and gaseous process streams and effluents and solid wastes generated by these systems. The PERMISS includes subsystems used to collect process and effluent samples during normal operation, anticipated operational occurrences, and					

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	under post-accident conditions. The lead branch responsible in implementing the review should coordinate the review of these systems and operational programs and receive input on the design and compliance with acceptance criteria listed in SRP Sections 11.2 to 11.5 from other branches, including, balance of plant, structural, instrumentation and controls, HVAC, quality assurance, technical specifications, and emergency planning.					
14.3.7.8	The reviewer should receive inputs on the treatment of issues identified above from other branches such as the structural, electrical and I&C branches. In addition, the secondary review branches specified in SRP Section 14.3 should provide inputs on selected issues. These issues include key insights and assumptions from PRA and severe accident analyses, as well as inputs for issues such as treatment of alarms, displays and controls, and functionality of MOVs. Cross-references from Tier 2 to Tier 1 for key insights and assumptions from PRA and severe accidents should be provided by applicants in Tier 2 together with these analyses.					
14.3.7.9	Tier 1 should address and verify at least the minimum inventory of alarms, controls, and indications as derived from the Emergency Procedure Guidelines, the requirements of RG 1.97, and probabilistic risk assessment insights. These may be specified in the MCR and the Remote Shutdown System (RSS) ITAAC, or addressed in the appropriate ITAAC, and verified to exist. Other controls, indications and alarms should be identified in the system ITAAC based on their safety significance. Locations for these should be shown on system figures if important to system design and function. The ability of these controls, indications, and alarms to function should be checked during operation of the system for the functional tests required by the system ITAAC. Because the intent of the ITAAC is to verify the final as-built condition of the plant, the operation of the system during the completion of the functional tests required in the system ITAAC should be conducted from the MCR. Therefore, the verification that the					

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	system can be operated from the MCR need not be a separate ITAAC. Also, because the operation of the equipment from the control room demonstrates the control function, continuity checks between the RSS and the equipment demonstrates that the control signal will be received by the component and provides adequate assurance that the equipment can be operated by the RSS. The results of the pre-operational test program may be utilized to demonstrate the ability to operate plant equipment by the RSS.					
14.3.8 (03/2007)	Radiation Protection Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.8.1	The reviewer should primarily use the applicable rules and regulations, general design criteria, regulatory guides, unresolved safety issues, and generic safety issues in the review of Tier 1 to determine the safety significance of SSCs with respect to the radiation protection for occupational workers and the general public they provide. Other sources include the SRP and applicable U.S. Nuclear Regulatory Commission (NRC) generic correspondence. The reviewer should use the guidance in Appendix C to SRP Section 14.3 as an aid for ensuring the comprehensiveness and consistency of this review.					
14.3.8.2	Radiation Protection: The reviewer should ensure that Tier 1 identifies and describes, commensurate with their safety significance, those SSCs that provide radiation shielding, confinement or containment of radioactivity, ventilation of airborne contamination, or radiation (or radioactivity concentration) monitoring for normal operations and during accidents. Tier 1 identifies and describes the measures that need to be employed during first-of-a-kind engineering to ensure that final design details (i.e., materials and component selection, equipment placement, and pipe routing) are consistent with the radiation protection commitments (including the commitment that radiation exposures will be as low as is reasonably achievable (ALARA)) in the certified design. Tier 1 contains ITAAC that					

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	ensure that the identified SSCs will function in a manner consistent with the certified design.					
14.3.8.3	Design Processes and Design Acceptance Criteria: A DC applicant may not provide sufficient detail in selected aspects of the design, including sufficient information to stipulate the source terms needed to verify the design of the shielding, ventilation, and airborne radioactivity monitoring systems. The applicant may choose to provide design processes and DAC for this material, as discussed in Appendix A to SRP Section 14.3. The applicant should document in DCD Tier 2, Section 14.3, its rationale for determining which areas of the design should use design processes and acceptance criteria. Essentially, the applicant should extract the most important design processes and acceptance criteria from DCD Chapter 12 of Tier 2 and identify them in Tier 1. This may be done either in a separate section of Tier 1 or in the applicable systems of Tier 1. A COL applicant or licensee must meet these criteria in the design of the plant, and the staff can audit the facility's design documentation to ensure that the criteria are met. The following discussion is specific to the review of design processes and acceptance criteria in this area. DC applicants may not provide the complete design information in this design area before the design is certified because the radiation shielding design and the calculated concentrations of airborne radioactive material depend on as-built and as-procured information about plant systems and components. Therefore, applicants may be unable to describe the standard design's radiation source terms (i.e., the quantity and concentration of radioactive materials contained in, or leaking from, plant systems) in sufficient detail to allow the staff to verify the adequacy of the shielding design, ventilation system designs, or the design and placement of the airborne radioactivity monitors. Instead, applicants may provide the processes and acceptance criteria by which the details of the design in this area are to be developed,					

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	designed, and evaluated. The design description should state the scope of the material in Tier 1. The application could, for example, encompass the radiological shielding and ventilation design of the reactor building, turbine building, control building, service building, and radwaste building. The COL applicant or licensee is responsible for the implementation of the process and the design. The DAC may be taken from the acceptance criteria in the applicable sections of Chapter 12 of the SRP. The analysis methods and source term assumptions specified in the DAC should be consistent with the approved methods and assumptions listed in the SRP. The SRP is the basis for the staff's safety review of the standard design. Therefore, demonstrating that the final design meets these DAC with the methods and assumptions specified in Tier 1 ensures that the as-built design will meet the applicable acceptance criteria of the SRP and the associated regulations and staff technical positions. The DAC in Tier 1 should address the verification of the plant radiation shielding design and the plant airborne concentrations of radioactive materials (e.g., the ventilation system and airborne monitoring system designs). The DAC should require the COL applicant to calculate radiation levels and airborne radioactivity levels within the plant rooms and areas to verify the adequacy of these design features during plant construction (concurrently with the verification of the ITAAC). The plant rooms and areas to which the DAC apply may be given in figures in Tier 1. The appropriate section of DCD Tier 2, Chapter 12, should include detailed supporting information for the DAC. The criteria in Tier 1 should ensure that the radiation shielding design (as provided by the plant structures or by permanent or temporary shielding included in the design) is adequate so that the maximum radiation levels in plant areas are commensurate					

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	with the areas' access requirements. This will allow radiation exposures to plant personnel to be maintained ALARA during normal plant operations and maintenance. Tier 1 should ensure that adequate shielding is provided for those plant areas that may require occupancy to permit an operator to aid in the mitigation of or the recovery from an accident. Tier 1 should ensure that the contribution of gamma shine to the radiation dose (particularly from the turbine building) to a member of the public (off site) will be a small fraction of the U.S. Environmental Protection Agency's dose limits in found at 40 CFR Part 190. The criteria in Tier 1 should ensure that the plant provides adequate containment and ventilation flow rates to control the concentrations of airborne radioactivity to levels commensurate with the access requirements of areas in the plant. Tier 1 should ensure that once the concentrations of airborne radioactivity are determined, the required airborne monitors are placed in the appropriate locations in the plant.					
14.3.9 (03/2007)	Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.9.1	SRP Chapter 18 provides guidance for the NRC staff to use in determining whether an applicant has proposed an acceptable HFE design. The applicant's HFE program will be evaluated in accordance with the review criteria of SRP Chapter 18 and NUREG-0711, "Human Factors Engineering Program Review Model." As indicated in Chapter 18, the HFE program technical information for the DC or COL review may be based on a design and implementation process plan. Therefore, the DC or COL ITAAC may be based on a design and implementation process plan. For example, acceptance criteria for the task analysis program element may be stated as "a report exists and concludes that function-based task analyses were conducted in conformance with the task analysis implementation plan and					

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	include the following functions . . .”					
14.3.9.2	If an implementation plan, rather than a completed HFE element, was accepted as part of the design certification process, then ITAAC should address the completion of the HFE program element.					
14.3.9.3	If an implementation plan was not reviewed and approved as part of the design certification, then the ITAAC should address both the development of the plan as well as item 2 above.					
14.3.9.4	The reviewer will verify that HFE-related ITAAC information is provided based on accepted HFE principles and program elements as discussed in SRP Chapter 18 and incorporated into the plant’s design.					
14.3.9.5	HFE-related ITAAC should primarily address verification of products (e.g., the control room, the human-system interfaces, etc.) or results reports from implementing the HFE program element implementation plan.					
14.3.9.6	Minimum Inventory of Displays, Alarms and Controls: Tier 1 includes a minimum inventory of displays, controls, and alarms that are necessary to carry out the vendor’s emergency procedure guidelines (i.e., Owners’ Groups Generic Technical Guidelines) and critical actions identified from the applicant’s PRA and task analysis of operator actions. The reviewers evaluation of the minimum inventory will encompass a multi-disciplinary effort consisting of human factors, I&C, PRA, and plant, reactor, and electrical system engineering. The minimum inventory list has been implemented through the rule-making process for four certified designs (10 CFR Part 52 Appendixes A, B, C, and D). The criteria used to determine acceptability of the inventory includes assuring that: (1) the scope of these					

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	items in the Generic Technical Guidelines and PRA effort are adequately considered, (2) the task analysis is detailed and comprehensive, (3) RG 1.97, Revision 3, Category 1 variables or RG 1.97, Revision 4, Type A, B, and C variables for accident monitoring are included, and (4) important system displays and controls described in Tier 1 system design descriptions necessary for transient mitigation are included.					
14.3.10 (03/2007)	Initial Test Program and D-RAP - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.10.1	The reviewer should ensure that for a design certification where an applicant has chosen to address emergency response facilities that the information provided adequately discusses facilities for emergency response. These include a habitable technical support center (TSC) with space, data retrieval capabilities and dedicated communications equipment, and an operational support center (OSC) with adequate communications, consistent with the applicable criteria in Supplement 1 to NUREG-0737 and NUREG-0696.					
14.3.10.2	A generic set of acceptable emergency planning EP-ITAAC was developed through coordination efforts between the NRC and the Nuclear Energy Institute (NEI) and resulted in the development of generic EP-ITAAC that are provided in Table 14.3.10-1(Table C.II.2-B1[1] of RG 1.206). These EP-ITAAC were established on a generic basis; they are not associated with any particular site or design. As such, several of the generic EP-ITAAC require the COL applicant to provide more specific acceptance criteria that reflect the plant-specific design and site-specific emergency response plans and facilities. This generic set is applicable to ESP applications that include ITAAC information. The reviewer should consider this set of EP-ITAAC in the review of application-specific EP-ITAAC that is tailored to the specific reactor design and emergency planning program requirements for					

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	the proposed plant and site. A smaller set of EP-ITAAC is acceptable if the application contains information that fully addresses emergency preparedness requirements associated with any of the generic ITAAC contained in Table 14.3.10-1 which is not all-inclusive, or exclusive of other ITAAC an applicant may propose. Additional plant-specific EP-ITAAC (i.e., beyond those listed in Table 14.3.10-1) may be proposed, and they will be examined to determine their acceptability on an applicant-specific basis. Table 14.3.10-1 also includes ITAAC associated with emergency response facilities that are within the scope of the design certification. COL applications referencing a certified design must include these design certification ITAAC on emergency response facilities. EP-ITAAC are proposed by the COL applicant and, except for EP-ITAAC from the referenced design certification or ESP, are subject to NRC review and a hearing with respect to whether they satisfy the “necessary and sufficient” requirement of 10 CFR 52.80(a). The complete set of EP-ITAAC will be incorporated into the COL as a license condition to be satisfied prior to fuel load. A COL holder may request a change in one or more of the EP-ITAAC, except those provided in the referenced certified design, via the license amendment process applicable to 10 CFR Part 52.					
	NOTE: 1. See SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,”					
14.3.11 (03/2007)	Containment Systems and Severe Accidents - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.11.1	The reviewer should primarily utilize the SRP sections related to					

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	containment systems in its review of Tier 1 to determine the safety significance of SSCs. Other sources include applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from the standard design's safety and severe accident analyses, and operating experience. The reviewer should also use the review checklists provided in Appendix C to SRP Section 14.3 as an aid for establishing consistency and comprehensiveness in the review of the systems.					
14.3.11.2	Tier 1 should be reviewed to verify that key parameters and insights from containment safety analyses, such as loss of coolant accident, main steamline break, main feedline break, subcompartment analyses, and suppression pool bypass are adequately addressed. Applicants should provide cross references in DCD Tier 2 Section 14.3 to show how the important input parameters used in the transient and accident analyses for the design are verified by the ITAAC. The reviewer should ensure that appropriate treatment of severe accident design features and containment design features are included in Tier 1. The supporting information regarding the detailed design and analyses should remain in Tier 2. For many of the design features, it may be impractical to test their functionality because of the absence of simulated severe accident conditions. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown, may be considered sufficient Tier 1 treatment. Applicants should provide cross references in the appropriate sections of Tier 2 to show how the important parameters from PRA, including shutdown risk, and severe accident analyses are verified by the ITAAC. For both PRA and severe accident analyses, although large uncertainties and unknowns may be associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses should be selected for treatment in Tier 1.					

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14.3.11.3	If applicable, the reviewer should utilize regulatory guidance from the Commission for selected policy and technical issues related to the particular design. Examples of these are contained in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." The SRM related to this is dated July 21, 1993.					
14.3.11.4	Containment isolation may be addressed by a combination of the system ITAACs or in a single system ITAAC. The containment isolation valves (CIVs) should be specified in Tier 1, and are most clearly shown on the system figures. The verification of the design qualification of the motor operated CIVs may be verified by the basic configuration check in each system ITAAC. In addition, in-situ tests should be required for containment isolation motor operated valves (MOV) and check valves in each system ITAAC. The ITAAC should verify that the CIVs close on receipt of an isolation signal. Actual closure of the containment isolation valves may be checked using the manual isolation switches in the main control room (MCR). Other ITAAC may verify that a containment isolation signal is generated for each of the process variables that will cause a containment isolation; the intent is to preclude multiple cycling of the containment isolation valves during the testing.					
14.3.11.5	Tier 1 should address and verify at least the minimum inventory of alarms, displays, and controls in Design Control Document (DCD) Tier 2 Chapter 18. These are derived from Generic Technical Guidelines (e.g., Emergency Procedure Guidelines, Emergency Response Guidelines), the guidance of RG 1.97, and severe accident and PRA insights. They may be specified in the MCR and the Remote Shutdown System (RSS) ITAAC, or addressed in the appropriate ITAAC, and are verified to exist. Other controls, displays, and alarms should be identified in the system ITAAC based on their safety significance. Locations for these should be shown on system figures if important to system					

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	design and function.					
14.3.12 (03/2007)	Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria					
14.3.12.1	Appendix A to this SRP section provides an acceptable set of generic PS-ITAAC that an applicant may use to develop application-specific PS-ITAAC, tailored to specific physical security hardware requirements.					
14.3.12.2	Appendix A is **not** all-inclusive or exclusive of other PS-ITAAC that an applicant may propose.					
14.3.12.3	Additional plant-specific PS-ITAAC (i.e., other than those listed in Appendix A) may be proposed and will be examined to determine acceptability on a case-by-case basis.					
	CHAPTER 15, Accident Analysis					
15.0, Rev. 3 (03/2007)	Introduction - Transient and Accident Analyses					
	Subsection I.2 of this SRP section discusses general acceptance criteria, and SRP Chapter 15 subsections discuss specific acceptance criteria for transients or accidents.					
15.0.1 (07/2000)	Radiological Consequence Analyses Using Alternate Source Terms					
	Refer to the BTP for the detailed criteria.					
15.0.2 (01/2006)	Review of Transient and Accident Analysis Methods					
15.0.2.1	Documentation The submittal must identify the specific accident scenarios and plant configurations for which the codes will be used. The evaluation model documentation must be scrutable, complete, unambiguous, accurate, and reasonably self-contained. Consistent nomenclature must be used throughout the entire model documentation. Any referenced material must be readily					

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	available from a technical library. Copies of any referenced documents that are not readily obtainable from a technical library or the NRC Public Document Room, including proprietary reports, must be included with the documentation or provided upon request. The code documentation must be sufficiently detailed that a qualified engineer can understand the documentation without recourse to the originator as required of any design calculation that meets the design control requirements of Appendix B to 10 CFR Part 50, and the documentation requirement in Appendix K to 10 CFR Part 50. It is desirable that the documentation include the responses to requests for additional information, sorted according to the review issue so that it is easy to follow the entire review history for a single issue. The reviewer can help obtain this goal by issuing RAI's organized by review issue. The documentation must include the following components: A. An overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation. B. A complete description of the accident scenario including plant initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and systems and/or component interactions that influence the outcome of the accident. C. A complete description of the code assessment comprising a description of each assessment test, why it was chosen, success criteria, diagrams of the test facility that show the location of instrumentation that is used in the assessment, a code model nodalization diagram, and all code options used in the calculation.					

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	D. A determination of the code uncertainty for a sample plant accident calculation. (Appendix K models do not require a determination of the code uncertainty.) E. A theory manual that is a self-contained document and that describes (a) field equations, (b) closure relationships, (c) numerical solution techniques, (d) simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods, (e) pedigree or origin of closure relationships used in the code, and (f) limits of applicability for all models in the code. F. A user manual that provides (a) detailed instructions about how the computer code is used, (b) a description of how to choose model input parameters and appropriate code options, (c) guidance about code limitations and options that should be avoided for particular accidents, components, or reactor types, and (d) if multiple computer codes are used, documented procedures for ensuring complete and accurate transfer of information between different elements of the evaluation model. G. A quality assurance plan that describes the procedures and controls under which the code was developed and assessed, and the corrective action procedures that are followed when an error is discovered. It is not important that the documentation be provided in exactly the format stated above but the information in the review package must be clearly organized in a reasonable manner.					
15.0.2.2	Evaluation Model					

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	Models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view. The degree of imprecision that is allowed in the models will ultimately be determined by the amount of uncertainty that can be tolerated in the calculation. Models that cause non-physical predictions to the extent that misinterpretation of the calculated results or trends in the results may occur, are not acceptable. For Appendix K LOCA analyses, emergency core cooling system (ECCS) evaluation models must meet the specific requirements contained in Appendix K to 10 CFR Part 50.					
15.0.2.3	Accident Scenario Identification Process The purpose of the accident scenario identification process is to identify and rank the reactor component and physical phenomena modeling requirements based on (a) their importance to the modeling of the scenario and (b) their impact on the figures of merit for the calculation. The accident scenario identification process must be a structured process. It must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario. The models that are present in the code and their degree of fidelity in predicting physical phenomena must be consistent with the results of this process. For example, if the accident scenario identification process determines that a certain physical phenomenon is important to the scenario under consideration, the code must have a relatively accurate model for that phenomenon and a detailed assessment of that model must be provided. Phenomena that have lower ranking may be represented by models with larger inherent uncertainty. The					

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	formality and complexity of this process should be commensurate with the complexity and importance of the event under consideration.					
15.0.2.4	Code Assessment Assessments of all code models intended to be used in the evaluation model must be provided. All assessments must be performed with the frozen version of the evaluation model that has been submitted for review. Assessments performed with other versions of the evaluation model should be justified on a case by case basis because even “small” changes to the evaluation model can have unintended consequences on calculation results that were thought to not be impacted by the changes. Separate effects testing must be performed to demonstrate the adequacy of the physical models to predict physical phenomena that were determined to be important by the accident scenario identification process. Separate effects testing must also be used to determine the uncertainty bounds of individual physical models. Integral effects testing must be performed to demonstrate that the interactions between different physical phenomena and reactor coolant system components and subsystems are identified and predicted correctly. Assessments against both separate effects tests and integral effects tests must be performed with the code. All models need to be assessed over the entire range of conditions encountered in the transient or accident scenario. Assessments must also compare code predictions to analytical solutions, where possible, to show the accuracy of the numerical methods used to solve the mathematical models. Code options used in the assessment calculations must be the same as those used in plant accident					

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	calculations. A scaling analysis must be performed that identifies important non-dimensional parameters related to geometry and key phenomena. Scaling distortions and their impact on the code assessment must be identified and evaluated in the assessment. Calculations of actual plant transients or accidents can be considered, but only as confirmatory supporting assessments for the evaluation model. This is because the data available from plant instrumentation is usually not detailed enough to support code assessment of specific models. Plant data can be used for code assessment if it can be demonstrated that the available instrumentation provides measurements of adequate resolution to assess the code. The assessment cases must compare code predictions to all important measured variables in order to show that good predictions of one test variable do not result from compensating errors. Assessments must include a description of all assessment cases, specific models that are being assessed in each case, and acceptance criteria used. Acceptance criteria must be supported by quantitative analysis whenever possible. ECCS evaluation models must include a specific assessment to meet the criteria in Appendix K to 10 CFR Part 50. Small-break ECCS evaluation models must also meet the assessment requirements of TMI Action Item II.K.3.30, where applicable.					
15.0.2.5	Uncertainty Analysis The uncertainty analysis must address all important sources of code uncertainty, including the mathematical models in the code and user modeling such as nodalization. The major sources of uncertainty must be addressed consistent with the results of the accident sequence identification process. When the code is used in a licensing calculation, the combined code and application					

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	uncertainty must be less than the design margin for the safety parameter of interest. The analysis must include a sample uncertainty evaluation for a typical plant application. In some cases, bounding values are used for input parameters as described in SRP sections or Regulatory Guides and are used for plant operating conditions such as accident initial conditions, set points, and boundary conditions.					
15.0.2.6	Quality Assurance Plan The code must be maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50.					
15.0.3 (03/2007)	Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors					
15.0.3.1	Offsite Radiological Consequences of Postulated Design Basis Accidents. The acceptance criteria are based on the requirements of 10 CFR 50.34(a)(1) as related to mitigating the radiological consequences of an accident in accordance with 10 CFR 52.17(a)(1) [early site permits], 10 CFR 52.47(a)(1) [standard design certifications] and 10 CFR 52.79(b) [combined licenses]. The plant design features intended to mitigate the radiological consequences of accidents, site atmospheric dispersion characteristics and the distances to the exclusion area boundary (EAB) and to the low population zone (LPZ) outer boundary are acceptable if the total calculated radiological consequences for the postulated fission product release fall within the following exposure acceptance criteria specified in 10 CFR 50.34(a)(1)(ii)(D): A. An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the					

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	postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE), and B. An individual located at any point on the outer boundary of the LPZ, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25 rem TEDE. For CP, OL, DC and COL reviews, the application is acceptable with regard to the radiological consequences of analyzed DBAs if the calculated TEDEs at the EAB and the LPZ outer boundary do not exceed the dose acceptance criteria listed in Table 1 below. For ESP applications that neither reference the standard reactor designs certified by NRC nor use the PPE approach, the staff may establish dose acceptance criteria lower than those stated above for certain DBAs based on the probability of occurrence. Examples of such criteria are illustrated in Table 1. For COL applications using an ESP with a PPE approach, these acceptance criteria may be applied at that time. Such applicants bear the burden of ensuring sufficient margin is provided in the design parameters (for example, PPE values) in the ESP application to compensate for uncertainty in those parameters. The margin should be large enough such that the actual design submitted at the COL stage, coupled with the site characteristics as described in the ESP, will comply with NRC regulations.					
15.0.3.2	Control Room Radiological Habitability. The acceptance criterion is based on the requirements of GDC 19 that mandate a control room design providing adequate radiation protection to permit access and occupancy of the control room under accident conditions for the duration of the accident, without personnel					

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	receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. These requirements are incorporated by reference in 10 CFR 52.47(a)(1) [standard design certifications] and 10 CFR 52.79(b) [combined licenses]. The radiation protection design of the control room is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified in GDC 19 of 5 rem TEDE for the duration of the accident					
15.0.3.3	Technical Support Center Radiological Habitability. This acceptance criterion is based on the requirement of Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 to provide an onsite TSC from which effective direction can be given and effective control can be exercised during an emergency. The radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified for the control room of 5 rem TEDE for the duration of the accident.					
	Refer to RG for Table 1 Accident Dose Criteria					
15.1.1 - 15.1.4, Rev. 2 (03/2007)	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve					
	1. To identify which of the moderate-frequency initiating events that result in increased heat removal are the most limiting. 2. To verify that, for the most limiting initiating events, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.					
	1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.					

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	2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). 3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. 4. To meet the requirements of General Design Criteria 10, 13, 15, 20, and 26 the positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section. 5. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53.					
	1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. 2. Conservative scram characteristics are assumed, i.e., for a PWR - maximum time delay with the most reactive rod held out of the core, and for a BWR - a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b)					

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	the uncertainty has otherwise been accounted for (see SAR or DCD) Section 4.4. 3. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, doppler coefficient, axial power profile, and radial power distribution. 4. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105. Compliance with Regulatory Guide 1.105 is determined by the Instrumentation and Control Systems.					
15.1.5, Rev. 3 (03/2007)	Steam System Piping Failures Inside and Outside of Containment (PWR)					
	Specific criteria necessary to meet the relevant requirements of the above regulations are as follows: 1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures. 2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact					

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	with no loss of core cooling capability. 3. The radiological criteria used in the evaluation of steam system pipe break accidents (PWRs only) appear in SRP section 15.0.3. 4. The integrity of the reactor coolant pumps should be maintained such that loss of ac power and containment isolation will not result in pump seal damage. 5. The auxiliary feedwater system or other means of decay heat removal must be safety related and, when required, automatically initiated. In the case of AP1000 the PRHR provides the safety related means of decay heat removal. 6. Tripping of the reactor coolant pumps should be consistent with the resolution to Task Action Plan item II.K.3.5.					
	There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below: 1. The reactor power level and number of operating loops assumed at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular NSSS design, and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced in the SAR. 2. Assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the					

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	consequences of the accident. A loss of offsite power may occur simultaneously with the pipe break or during the accident, or offsite power may not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The reviewer should note that the assumption that offsite power is not lost may maximize heat removal from the core and reactor system and thereby maximize containment pressure and reactivity feedback within the core. The analyses should take account of the effect that loss of offsite power has on reactor coolant pump and main feedwater pump trips and on the initiation of auxiliary feedwater flow, and the effects on the sequence of events for these accidents. For new applications, loss of offsite power should be considered in addition to any limiting single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Analysis Report for the ABB-CE System 80+ design certification.) 3. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of postulated steam line breaks on other systems should be considered in a manner consistent with the intent of Branch Technical Position (BTP) 3-3 and BTP 3-4. 4. The worst single active component failure should be assumed to occur. For new applications, loss of offsite power should not be considered as a single failure, (see assumption b above). The assumed single failure may cause more than one steam generator to blow down, failure of main feedwater to isolate, or may be in any of the systems required to control the transient. 5. The maximum-worth rod should be assumed to be held in					

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	the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be used. Local power peaking at the location of the stuck out control rod should be considered. Local power peaking will affect the DNBR analysis in the initial period as the safety rods are entering the core and during any subsequent return to power resulting from reactivity addition to the core from the cooldown. 6. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. 7. The initial core flow assumed for the analysis of the steam line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results; however, for the analysis of steam line break accidents, this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the assumed value should be justified. 8. Failure of a steam line at a plant with multiple coolant loops will cause asymmetric temperatures within the reactor core. Asymmetric core temperatures will affect the local power distribution and the DNBR analysis. Assumptions for mixing in the downcomer and the reactor vessel lower plenum will affect the predicted core temperature distributions, reactivity feedback and local power. Assumptions for mixing should					

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	be chosen so as to be conservative for predicting maximum local core power and DNBR. 9. For postulated pipe failure in nonseismically qualified portions of the main steam line (outside containment and downstream of the main steam isolation valves, (MSIVs) due to a seismically initiated event, only safety related equipment should be assumed operative to mitigate the consequences of the break. 10. For postulated instantaneous pipe failures in seismically qualified portions of the main steam line (inside containment and upstream of the MSIVs), only safety related equipment should be assumed operative. If, in addition, a single malfunction or failure of an active component is postulated, credit may be taken for the use of a backup nonsafety-related component to mitigate the consequences of the break. 11. During the initial 10 minutes of the transient, should credit for operator action be required (e.g., reactor coolant pump trip), an assessment for the limiting consequence must be performed in order to account for operator delay and/or error.					
15.1.5.A, Rev. 2 (07/1981)	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR					
	The acceptance criteria are based on the relevant requirements of 10 CFR Part 100 as related to the radiological consequences of a postulated accident. The plant site and the dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated MSLB outside containment of a PWR facility if the calculated whole-body and thyroid doses at the exclusion area and the low population zone					

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	outer boundaries do not exceed the following exposure guidelines: 1. for an MSLB with an assumed preaccident iodine spike and for an MSLB with the highest worth control rod stuck out of the core, the calculated doses should not exceed the guideline values of 10 CFR Part 100, Section 11 (Ref. 1), and 2. for an MSLB with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e., 10 percent or 2.5 rem and 30 rem respectively, for the whole-body and thyroid doses. The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Regulatory Guide 1.4 (Ref. 8) except for the atmospheric dispersion factors which are reviewed under SRP Section 2.3.4. Plant technical specifications are required for the iodine activity in the primary and secondary coolant system and for the leak rate from the primary to the secondary coolant system in the steam generator(s). These specifications are acceptable if the calculated potential radiological consequences from the MSLB accident are within the exposure guidelines for the above two cases.					
	REFERENCES: 1. 10 CFR Part 100, Section 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance." 8. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant					

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	Accident for Pressurized Water Reactors."					
15.2.1-15.2.5, Rev. 2 (03/2007)	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)					
	The basic objectives of the review of the initiating events listed in subsection I of this SRP section: A. To identify which moderate-frequency event that results in an unplanned decrease in secondary system heat removal is the most limiting, in particular as to primary pressure, secondary pressure, and long-term decay heat removal. B. To verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure. C. To verify whether the plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in RG 1.105. D. To verify whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the RG 1.206 regulatory guidelines.					
	With the ANS standards as guidance, specific criteria meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency. A. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.					

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	B. Fuel cladding integrity must be maintained by the minimum departure from nucleate boiling ratio (DNBR) remaining above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remaining above the minimum CPR safety limit for BWRs based on acceptable correlations (see SAR (or DCD) Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design. C. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently. D. The requirements in RG 1.105, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs addressed in this SRP section. E. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR Part 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17. F. Performance of nonsafety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY 77-439, SECY 94-084, and RG 1.206					
	The applicant should analyze these events using an acceptable analytical model. Any other analytical method proposed by the applicant is evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by the appropriate organization for reactor systems.					

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	The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable: A. The reactor is initially at 102 percent of the rated (licensed) core thermal power (to account for a 2 percent power measurement uncertainty unless a lower number can be justified through measurement uncertainty methodology and evaluation or unless the uncertainty otherwise is accounted for (see SAR (or DCD) Section 4.4)), and primary loop flow is at the nominal design flow less the flow measurement uncertainty. B. Conservative scram characteristics are assumed (*i.e.*, for a PWR maximum time delay with the most reactive rod held out of the core, for a BWR a 0.8 design conservatism multiplier on the predicted reactivity insertion rate) unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is otherwise accounted for (see SAR (or DCD) Section 4.4). C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with Regulatory Guide 1.105.					
15.2.6, Rev. 2 (03/2007)	Loss of Non-Emergency AC Power to the Station Auxiliaries					
15.2.6.1	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
15.2.6.2	Fuel cladding integrity should be maintained by keeping the minimum departure from nucleate boiling ratio (DNBR) above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) above the minimum critical power ratio safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).					
15.2.6.3	An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.					
15.2.6.4	For the requirements of GDCs 10 and 15, the positions of Regulatory Guide (RG) 1.105, "Instrument Setpoints for Safety-Related Systems," have impact on the plant response to the type of transient addressed in this SRP section.					
15.2.6.5	The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53. The applicant's analysis of the loss of ac power transient should be based on an acceptable and NRC-approved model. If the applicant proposes analytical methods notapproved, these are evaluated by the staff for acceptability and approval. For new generic methods, the reviewer requests an appropriate evaluation. The parameter values in the analytical model should be suitably conservative. The following values are acceptable: A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level. The number of loops (RCS loop requirements as applicable for BWR design) operating at the initiation of the event should correspond to the operating condition which maximizes the					

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	consequences of the event. B. Conservative scram characteristics are assumed (i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate). C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile, and radial power distribution. D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105. Compliance with RG 1.105 is determined.					
15.2.7, Rev. 2 (03/2007)	Loss of Normal Feedwater Flow					
15.2.7.1	The basic objective in the review of the loss of normal feedwater transient is to confirm that the following criteria are met: A. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met. B. There is sufficient capacity for long term decay heat removal for the plant to reach a stabilized condition. C. The plant protection systems setpoints assumed in the transient analyses are selected with adequate allowance for measurement uncertainties as delineated in Regulatory Guide 1.105. D. The event evaluation takes into consideration single failures, operator errors, and performance of non-safety related systems					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	that are consistent with regulatory guidelines set forth in RG 1.206.					
15.2.7.2	Using the ANS standards as guidance, specific criteria have been developed to meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency and they are as follows: A. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values. B. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SAR (or DCD) Section 4.4), as well as by satisfaction of any other SAFDL that may be applicable to the particular reactor design. C. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. D. To meet the requirements of GDCs 10 and 15, the positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section. E. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53 and GDC 17. F. The guidance provided in SECY 77-439, SECY 94-084 and RG 1.206 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.					
15.2.7.3	The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. If the applicant proposes to use analytical methods which have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods the reviewer requests an evaluation by the appropriate organization for reactor systems. The value of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model. A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. B. Conservative scram characteristics are assumed, i.e., for a PWR – maximum time delay with the most reactive rod held out of the core and for a BWR – a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) the uncertainty has otherwise been accounted for (see SAR (or DCD) Section 4.4). C. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile and radial power distribution. D. Mitigating systems should be assumed to be actuated in the					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105.					
15.2.8, Rev. 2 (03/2007)	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)					
15.2.8.1	Requirements for maintenance of adequate decay heat removal by the AFWS are in 10 CFR 50.34(f)(1)(ii), (TMI issue II E 1.1) and 10 CFR 50.34(f)(2)(xii), (TMI issue II E 1.2). Requirements for reactor coolant pump (RCP) operation are in 10 CFR 50.34(f)(1)(iii), (TMI issue 2 K 2). The reviewer should see Chapter 20 of the NRC FSAR for AP1000 to see how these post TMI requirements are met by the PRHR, the non-safety related start-up feedwater system (SUFWS) and the canned-motor RCPs of AP1000.					
15.2.8.2	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures (American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III) for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks.					
15.2.8.3	The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods not meeting these criteria unless, from an acceptable fuel damage model (see SRP Section 4.2) including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability.					
15.2.8.4	Calculated doses at the site boundary from any activity release must be a small fraction of the 10 CFR Part 100 guidelines.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
15.2.8.5	The integrity of the RCPs should be maintained so loss of alternating current power and containment isolation do not result in seal damage.					
15.2.8.6	The AFWS must be safety grade and automatically initiated when required.					
15.2.8.7	Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures: A. The power level assumed and number of loops operating at the initiation of the transient should correspond to the operating condition which maximizes accident consequences. These assumed initial conditions vary with the particular nuclear steam supply system and sensitivity studies are required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report as references if applicable. B. The assumptions as to whether offsite power is lost and the time of loss should be conservative. Offsite power may be lost simultaneously with the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should determine the most conservative assumption appropriate to the plant design reviewed. The study should take account of the effects that loss of offsite power (LOOP) has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater and the consequent modification of the sequence of events. C. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of the postulated feedwater line breaks on other systems should be considered consistently					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	with the intent of Branch Technical Positions (BTP) 3-3 and BTP 3-4. D. The worst single active component failure should be assumed to occur in the systems required to control the transient. For new applications, LOOP should not be considered a single failure; feedwater pipe breaks should be analyzed with and without LOOP, as in assumption B, in combination with a single, active failure. (This position is based upon interpretation of GDC 17 as documented in the FSER for the ABB-CE System 80+ DC.) E. The maximum rod worth should be assumed to be held in the fully withdrawn position per GDC 25. An appropriate rod reactivity worth versus rod position curve should be assumed.					
15.3.1-15.3.2, Rev. 2 (03/2007)	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions					
15.3.1.1	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.					
15.3.1.2	Fuel-cladding integrity must be maintained by the minimum DNBR remaining above the 95 percent probability/95 percent confidence DNBR limit for PWRs and the critical power ratio (CPR) remaining above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).					
15.3.1.3	An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.					
15.3.1.4	The requirements stated in RG 1.105, "Instrument Spans and Setpoints," are evaluated for their impact on the plant response to AOOs addressed in this SRP section.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
15.3.1.5	Onsite and offsite electric power systems must be maintained so safety-related SSCs function during normal operation and AOOs.					
15.3.1.6	The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR 50, Appendix A, must be assumed in the analysis and should follow the guidance of RG 1.53.					
15.3.1.7	The performance of nonsafety-related systems during transients and accidents and of single failures of active and passive systems (especially the performance of check valves in passive systems), must be evaluated and verified by the guidance of SECY 77-439, SECY 94-084 and RG 1.206.					
15.3.1.8	The applicant's analysis of the most limiting AOOs should use an acceptable model. Unapproved analytical methods proposed by the applicant are evaluated by the staff for acceptability.					
15.3.1.9	Parameter values in the analytical model should be suitably conservative. The following values are acceptable: A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of 2 percent to account for power measurement uncertainty unless (i) a lower number can be justified through the measurement uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise (see SRP 4.4). The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. B. Conservative scram characteristics are assumed (e.g., maximum time delay with the most reactive rod held out of the core for a PWR, a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate for a BWR), unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise (see SRP Section 4.4).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. D. Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.					
15.3.3-15.3.4, Rev. 3 (03/2007)	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break					
	1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures. 2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR or CPR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of core cooling capability. 3. Any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. 4. The integrity of the reactor coolant pumps should be maintained such that loss of ac power and containment isolation will not result in pump seal damage. 5. The auxiliary feedwater system must be safety grade and, when required, automatically initiated. 6. A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers. 7. Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component. For new applications, loss of offsite power should not be considered a single failure; reactor coolant pump rotor seizures and shaft breaks should be analyzed with a loss of off-site power (see item 9, below) in combination with a single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Evaluation Report for the ABB-CE System 80+ design certification.) 8. The ability to achieve and maintain long-term core cooling should be verified. 9. This event should be analyzed assuming turbine trip and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	coincident loss of offsite power and coastdown of undamaged pumps.					
	1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating, plus an allowance to account for power measurement uncertainties. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. 2. The local flow conditions used in the core thermal-hydraulics model should be calculated based upon an inlet flow distribution corresponding to N–1 reactor coolant pumps (initial minus faulted pump) and a conservative time-dependent flow coastdown. Note that the inlet flow distribution will change as more pumps begin to coastdown following turbine trip and coincident loss of offsite power. 3. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core, and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate. 4. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.					
15.4.1, Rev. 3 (03/2007)	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition					
15.4.1.1	The requirements of GDC 10, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when: A. The thermal margin limits (DNBR for PWRs and MCPR for BWRs) as specified in SRP Section 4.4 are met.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	B. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2 do not exceed the melting point. C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 does not exceed 1%.					
15.4.2, Rev. 3 (03/2007)	Uncontrolled Control Rod Assembly Withdrawal at Power					
15.4.2.1	The requirements of General Design Criteria 10, 17, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when: A. The thermal margin limits departure from nucleate boiling ratio for PWRs and maximum critical power ratio for BWRs as specified in SRP Section 4.4, subsection II.1, are met. B. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point. C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1%.					
15.4.3, Rev. 3 (03/2007)	Control Rod Misoperation (System Malfunction or Operator Error)					
15.4.3.1	The thermal margin limits (departure from nucleate boiling ratio for PWRs) as specified in SRP Section 4.4, subsection II.1, are met.					
15.4.3.2	Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.					
15.4.3.3	Uniform cladding strain as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1%.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
15.4.4-15.4.5, Rev. 2 (03/2007)	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate					
15.4.4.1	Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainty, unless (a) a lower number can be justified through the measurement uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see SRP Section 4.4). An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two or three loops initially operating) or the effects referenced to a limiting case.					
15.4.4.2	Conservative scram characteristics are assumed, e.g., maximum time delay with the most reactive rod held out of the core for a PWR and a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate for a BWR, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see SRP Section 4.4).					
15.4.4.3	The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile and radial power distribution.					
15.4.4.4	Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with Regulatory Guide 1.105 as determined by the organization responsible for instrumentation and controls. The reviewer shall verify that the protection system (1) automatically initiates the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded for this event, and (2) senses the plant conditions and initiates the operation of SSCs important to safety.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	For BWR plants where flow control is part of the reactivity control system, GDCs 26 and 28 must be satisfied for this event; otherwise, GDCs 26 and 28 are not applicable. Where applicable, GDCs 26 and 28 are satisfied if compliance with GDCs 10 and 15 is demonstrated.					
15.4.6, Rev. 2 (03/2007)	Inadvertent Decrease in Boron Concentration in the Reactor Coolant (PWR)					
15.4.6.1	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.					
15.4.6.2	Fuel cladding integrity must be maintained so the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations with SRP Section 4.4.					
15.4.6.3	An incident of moderate frequency should not generate a more serious than moderate plant condition without other faults occurring independently.					
15.4.6.4	If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost: A. During refueling: 30 minutes. B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.					
15.4.6.5	The applicant's analysis of moderator dilution events should use an acceptable analytical model. Staff must evaluate any proposed unreviewed analytical methods. The reviewer initiates an evaluation of new generic methods. The following plant initial conditions should be considered in the analysis: refueling, startup, power operation (automatic control and manual modes), hot standby, hot shutdown and cold shutdown. Parameters and assumptions in the analytical model should be suitably conservative. The following values and assumptions are					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	acceptable: A. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent to account for power-measurement uncertainty. The analysis may use a smaller power-measurement uncertainty if justified adequately. B. The boron dilution is assumed to occur at the maximum possible rate. C. Core burnup and corresponding boron concentration must yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. The core burnup must be justified by either analysis or evaluation. D. All fuel assemblies are installed in the core. E. A conservatively low value is assumed for the reactor coolant volume. F. For analyses during refueling, all control rods are withdrawn from the core. An alternate assumption requires adequate justification and delineation of necessary controls so the alternate assumption remains valid. G. For analyses during power operation, the minimum shutdown margin allowed by the technical specifications (usually 1 percent) is assumed prior to boron dilution.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	H. A conservatively high reactivity addition rate is assumed for each analyzed event to take into account the effect of increasing boron worth with dilution. I. Conservative scram characteristics are assumed (*i.e.*, maximum time delay with the most reactive rod out of the core).					
15.4.7, Rev. 2 (03/2007)	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position					
15.4.7.1	To meet the requirements of GDC 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel-loading errors after fueling operations.					
15.4.7.2	In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 criteria. A small fraction is interpreted to be less than 10% of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of any fuel-loading error should include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.					
15.4.8, Rev. 3 (03/2007)	Spectrum of Rod Ejection Accidents (PWR)					
15.4.8.1	General Design Criterion (GDC) 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
15.4.8.2	Acceptance criteria are based on meeting GDC 28 requirements as to the effects of postulated reactivity accidents that result in neither damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficient damage to impair significantly core cooling capacity. Regulatory positions and specific guidelines necessary to meet the relevant requirements of GDC 28 are in Regulatory Guide 1.77 and SRP Section 4.2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that result in stresses that exceed the "Service Limit C" as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.					
15.4.8.3	10 CFR 100.11 and 10 CFR 50.67 establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation under SRP Section 15.0.3. SRP Section 4.2 describes fuel rod failure mechanisms. Guidance for calculating radiological consequences is in Regulatory Guides 1.183 and 1.195.					
15.4.8.A, Rev. 2 (07/1981)	Radiological Consequences of a Control Rod Ejection Accident (PWR)					
	Refer to the BTP for detailed criteria.					
15.4.9, Rev. 3 (03/2007)	Spectrum of Rod Drop Accidents (BWR)					
	Refer to the BTP for the detailed criteria.					
15.4.9.A, Rev. 2 (07/1981)	Radiological Consequences of Control Rod Drop Accident (BWR)					
15.4.9.A.1	Reactivity excursions should not result in radially averaged fuel rod enthalpy greater-than 280 cal/gm at any axial location .in any					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	fuel rod.					
15.4.9.A.2	The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code. (Ref. 3).					
15.4.9.A.3	The number of fuel rods predicted to reach assumed fuel failure thresholds and associated parameters such as the amount of fuel reaching melting conditions will be an input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.					
	REFERENCE: 3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.					
15.5.1-15.5.2, Rev. 2 (03/2007)	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory					
	The specific acceptance criteria derived from GDC 10, 13, 15, and 26, and from the aforementioned ANS standards, are: 1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values in accordance with the ASME Boiler and Pressure Vessel Code. 2. Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	acceptable correlations (see SRP Section 4.4). 3. An AOO should not generate a more serious plant condition without other faults occurring independently.					
	The applicant's analysis of events leading to an increase of reactor coolant inventory should be performed using an acceptable analytical model. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer performs an evaluation of the new method as part of its review under this SRP section. The values of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model: 1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. 2. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate. 3. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	radial power distribution.					
15.6.1, Rev. 2 (03/2007)	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve					
15.6.1.1	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.					
15.6.1.2	Fuel cladding integrity is maintained if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) above the minimum critical power ratio safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).					
15.6.1.3	An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant. To meet the requirements of GDCs 10, 13, 15, and 26, the positions of Regulatory Guide (RG) 1.105, "Instrument Setpoints for Safety-Related Systems," are useful as to their impact on the plant response to the type of transient addressed in this SRP section. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, should be assumed in the analysis and should satisfy the positions of RG 1.53. The applicant's analysis of this transient should use an acceptable analytical model. If the applicant proposes to use analytical methods not previously reviewed and approved by the staff, the staff evaluates them for acceptability. For new generic methods, the reviewer initiates an evaluation of the new analytical model.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable. A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to operate plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level. The number of loops operating at the initiation of the event should correspond to the operating condition that maximizes the consequences of the event. B. Conservative scram characteristics are assumed (*i.e.*, for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate). C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105.					
15.6.2, Rev. 2 (07/1981)	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment					
	The plant site and the dose mitigating engineered safety feature (ESF) systems are acceptable with respect to the radiological consequences of a postulated failure outside the containment of a small line carrying reactor coolant if the calculated whole-body and thyroid doses at the exclusion area and the low population					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	zone outer boundaries do not exceed a small fraction of the exposure guideline values of 10 CFR Part 100, §100.11 (Ref. 3) as stated in position C.1.b of Regulatory Guide 1.11 (Ref. 2). A "small fraction" of 10 CFR Part 100 means 10 percent of these exposure guideline values, that is, 2.5 rem and 30 rem for the whole-body and thyroid doses, respectively. A plant-specific technical specification is required for the iodine activity in the primary coolant system. The specification is acceptable with respect to the postulated failure if the calculated doses resulting from the failure are within the above exposure guidelines.					
	REFERENCES: 2. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Containment." 3. 10 CFR Part 100, §100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."					
15.6.3, Rev. 2 (07/1981)	Radiological Consequences of Steam Generator Tube Failure (PWR)					
	The acceptance criteria are based on the relevant requirements of 10 CFR Part 100 as it relates to mitigating the radiological consequences of an accident. The plant site and the dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated steam generator tube failure accident at a PWR facility if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed the following exposure guidelines: (1) for the postulated accident with an assumed preaccident iodine spike in the reactor coolant and for the postulated accident with the highest worth control rod stuck out of the core the calculated doses should not exceed the guideline values of 10 CFR Part 100, Section 11 (Ref. 1), and					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	(2) for the postulated accident with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e., 10 percent or 2.5 rem and 30 rem, respectively, for the whole-body and thyroid doses. The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Regulatory Guide 1.4 (Ref. 2) except for the atmospheric dispersion factors which are reviewed under SRP Section 2.3.4. Plant technical specifications are required for iodine activity in the primary and secondary coolant systems. These specifications are acceptable if the calculated potential radiological consequences from the steam generator tube failure accident are within the exposure guidelines for the above two cases.					
	REFERENCES: 1. 10 CFR Part 100, Section 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance." 2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."					
15.6.4, Rev. 2 (07/1981)	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)					
	The acceptance criteria are based on the requirements of 10 CFR Part 100 as related to the radiological consequences of an accident. The plant site and the dose mitigating engineered safety features (ESF) are acceptable with respect to the radiological consequences of a postulated MSLB outside containment of a BWR facility if the calculated whole body and thyroid doses at the exclusion area and the low population zone boundaries do not exceed the-following exposure guidelines:					

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ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
	1. For a MSLB with an assumed preaccident iodine spike corresponding to the maximum iodine concentration stated in the NSSS vendor standard technical specifications, the calculated doses should not exceed the guideline values of 10 CFR Part 100, paragraph 11 (Ref. 1). 2. For a MSLB with an assumed iodine concentration corresponding to the equilibrium value for continued full power operation stated in the NSSS vendor standard technical specifications, the doses should not exceed a small fraction of the above guideline values, i.e., 10 percent or 2.5 rem and 30 rem respectively, for the whole body and thyroid doses. 3. The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Regulatory Guide 1.5 (Ref. 2) except for the atmospheric dispersion factors which are reviewed under SRP Section 2.3.4. 4. A plant specific technical specification is required for both cases of iodine activity in the primary coolant. This specification is acceptable if the calculated potential radiological consequences from the MSLB accident are within the exposure guidelines for the above two cases.					
	REFERENCES: 1. 10 CFR Part 100, Paragraph 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance." 2. Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors."					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
15.6.5, Rev. 3 (03/2007)	Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary					
15.6.5.1	An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of 10 CFR 50.46. Regulatory Guide 1.157 and Section I of Appendix K to 10 CFR Part 50 provide guidance on acceptable evaluation models. For the full spectrum of reactor coolant pipe breaks, and taking into consideration requirements for reactor coolant pump operation during a small break loss-of-coolant accident, the results of the evaluation must show that the specific requirements of the acceptance criteria for ECCS are satisfied as given below. This also includes analyses of a spectrum of large break and small break LOCAs to assure boric acid precipitation is precluded for all break sizes and locations. The analyses should be performed in accordance with 10 CFR 50.46, including methods referred to in 10 CFR 50.46(a)(1) or (2). The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). Additionally the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying 10 CFR 50.46(c)(2), and the analysis results should meet the performance criteria in 10 CFR 50.46(b). A. The calculated maximum fuel element cladding temperature does not exceed 1200 ^{o}C (2200 ^{o}F).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	B. The calculated total local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident. C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. D. Calculated changes in core geometry are such that the core remains amenable to cooling. E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.					
	The radiological consequences of the most severe LOCA are within the guidelines of and 10 CFR 100 or 10 CFR 50.67. For applications under 10 CFR Part 52, reviewers should use SRP Section 15.0.3, "Radiological Consequences of Design Basis Accidents - for ESP, DC and COL Applications."					
	The TMI Action Plan requirements for II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31, and II.K.3.40 have been met.					
15.6.5.A, Rev. 2 (07/1981)	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution					
	The acceptance criteria are based on the requirements of 10 CFR					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Part 100 as related to mitigating the radiological consequences of an accident. Specific acceptance criteria for the total calculated doses and for the containment leakage contribution are as follows: 1. The distances to the exclusion area boundary and to the low population zone outer boundary are acceptable if the total calculated radiological consequences (i.e., thyroid and whole body doses) for the hypothetical LOCA fall within the appropriate exposure guideline values specified In 10 CFR Part 100, §100.11 (Ref. 1). The total dose is the combined dose from all release paths from the containment to the atmosphere. At the construction permit (CP) review stage, the staff applies exposure guideline values of 150 rem to the thyroid and 20 rem to the whole body in accordance with Regulatory Guides 1.3 and 1.4. This is to allow for uncertainties ip meteorology and other site-related data and to allow for system design changes that might influence the final design of engineered safety features or the dose reduction factors of these features. These lower values are applied at the CP stage to provide reasonable assurance that the 10 CFR Part 100 guideline values can be met at the operating license (OL) review stage. 2. The model for and the calculation of the post-LOCA leakage contribution -to the total whole body and thyroid doses of a hypothetical LOCA are acceptable if they incorporate the appropriate conservative design basis assumptions outlined in the regulatory positions of Regulatory Guide 1.3 (Ref. 2) for a BWk facility and of Regulatory Guide 1.4 (Ref. 3) for a PWR facility with the exception of the guidelines for the atmospheric dispersion fusion factors (X/Q values). The acceptability of the X/Q values is determined under SRP Section 2.3.4.					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	REFERENCES: 1. 10 CFR Part 100, §100.11, "DeterminatIon of Exclusion Area, Low Population Zone, and Population Center Distance." 2. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." 3. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."					
15.6.5.B, Rev. 2 (07/1981)	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Leakage From Engineered Safety Feature Components Outside Containment					
	The acceptance criteria are based on the requirements of 10 CFR Part 100 (Ref. 2) as related to mitigating the radiological consequences of an accident. Specific criteria necessary to meet this requirement are as follows: (1) ESF systems that circulate water outside the containment are assumed to leak during their intended operation (e.g., valve stem leakage) and as a result of a failure of a passive component; Both types of leakage are included in the review. ESF atmosphere filtration systems should be provided in those areas where such leakage is postulated to occur in order to mitigate the radiological consequences from the fission product release. (2) The radiological consequences from the postulated leakage should be calculated using conservative assumptions. 50% of the core iodine inventory, based upon the maximum reactor power level, should be assumed to be mixed in the sump water being circulated through the containment external piping systems, in accordance with the values listed in Table 1 of Regulatory Guide 1.7 (Ref. 1). The atmospheric dispersion factors (X/Q values) as determined under SRP Section 2.3.4 should be used in the					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	analysis. (3) The radiological consequences from ESF component leakage, as calculated by the staff, should be combined, under SRP Section 15.6.5 Appendix A, with the consequences from other fission product release paths to determine the total calculated radiological consequences from the hypothetical LOCA. The acceptability of the site, with respect to the total radiological consequences, is determined by the adequacy of the exclusion area and low population zone outer boundary distances in conjunction with the operation of dose-mitigating ESF systems. For operating license applications, the total doses should be within the exposure guidelines of 10 CFR Part, 100, § 100.11 (Ref. 2) and for a construction permit application, the total doses should be within the guideline value of Regulatory Guides 1.3 (Ref. 3) and 1.4 (Ref. 4), as appropriate. This acceptability is determined under SRP Section 15.6.5, Appendix A.					
	REFERENCES: 1. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." 2. 10 CFR Part 100, § 100.11, "Determination of Exclusion-Area Low Population Zone and Population Center Distance." 3. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." 4. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."					
15.6.5.D, Rev. 2 (07/1981)	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR)					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	The radiological consequences associated with the operation of the MSIVLCS following a postulated LOCA are combined, under SRP Section 15.6.5, Appendix A, with the consequences from other LOCA fission product release paths to determine the total calculated radiological consequences from the hypothetical LOCA. The acceptability of the site, with' respect to the total radiological consequences, is determined by the adequacy of the exclusion area and low population zone boundary distances in conjunction with the operation of dose mitigating ESF systems. For operating license applications, the total doses should be within the exposure guidelines of 10 CFR Part 100, paragraph 11 (Ref. 2), and for a construction permit application, the total doses should be within the guideline values of Regulatory Guide 1.3 (Ref. 3). The acceptability is determined under SRP Section 15.6.5, Appendix A.					
	REFERENCES 2. 10 CFR Part 100, Paragraph 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance." 3. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." Revision 2.					
15.7.3, Rev. 2 (07/1981)	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (content of this section has been relocated to BTP 11-6)					
	Refer to the BTP for the detailed criteria.					
15.7.4, Rev. 2 (07/1981)	Radiological Consequences of Fuel Handling Accidents					
II.1	The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated fuel handling accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10 CFR Part 100,					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	paragraph 11. "Well within" means 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for the whole-body doses.					
II.2	The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building are acceptable if they meet the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," (Ref. 2) with respect to appropriate containment, confinement and filtering systems.					
II.3	The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative assumptions in Regulatory Guide 1.25 (Ref. 3) with the exception of the guidelines for the atmospheric dispersion factors (X/Q values). The acceptability of the X/Q values is determined under SRP Section 2.3.4.					
II.4	An ESF grade atmosphere clean-up system is required for the spent fuel storage area to reduce the potential radiological consequences.					
II.5	The containment design is acceptable with respect to a postulated fuel handling accident if it possesses the capability for prompt radiation detection by use of redundant radiation monitors and automatic isolation if fuel handling operations inside containment occur when the containment is open to the environment (i.e., with a containment purge exhaust system). An acceptable alternative approach is containment venting through an ESF atmosphere cleanup system or containment isolation during fuel handling operations.					
	REFERENCES: 2. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control." 3. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
15.7.5, Rev. 2 (07/1981)	Spent Fuel Cask Drop Accidents					
II.1	The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10 CFR Part 100, paragraph 11. "Well within" means 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for the whole-body doses.					
II.2	The radioactivity control features of the fuel storage and spent fuel cask handling system in the fuel building are acceptable if they meet the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," (Ref. 2) with respect to appropriate containment, confinement 'and filtering systems.					
II.3	The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative assumptions in Regulatory Guide 1.25 (Ref. 3) with respect to gap inventory as stated in positions C.1.d,e, and f of the guide. The acceptability of the atmospheric dispersion factors, X/Q values, is determined under SRP Section 2.3.4.					
II.4	An ESF grade atmospheric cleanup system is required for the fuel handling building to reduce the potential radiological consequences of the fuel cask drop accident.					
II.5	The plant design with regard to spent fuel cask drop accidents is acceptable without calculation of radiological consequences if potential cask drop distances are less than 30 feet and appropriate impact limiting devices are employed during cask movements, as determined by ASB.					
	REFERENCES: 2. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control." | 3. Regulatory Guide 1.25, "Assumptions Used for Evaluating					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."					
15.8, Rev. 2 (03/2007)	Anticipated Transients Without Scram					
15.8.1	Acceptance criteria for Boiling Water Reactors (BWRs): A. Equipment shall be provided to initiate an automatic reactor coolant re-circulation pumps trip under conditions indicative of an ATWS. B. An alternate rod injection system (ARI) is provided independent and diverse from the reactor trip system sensor output to the final actuation device. The system shall have independent scram air header exhaust valve. C. A standby by liquid control system (SLCS) shall be provided that is capable of initiating reactivity control equivalent to injection of 326 liters per minute (or 86 gallons per minute) of 13 weight percent sodium pentaborate decahydrate solution of Boron-10 into a 638 centimeters (251 inches) inside diameter reactor pressure vessel operating at a power density consistent with the original licensed thermal power (OLTP). D. The SLCS initiation is automatic for the plants specified in 10 CFR 50.62(c)(4). E. For BWRs, reactor coolant system pressures should not exceed ASME Service Level C limits (approximately 10.3 MPa (1500 psi).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	F. Each plant Emergency Operating Procedures (EOPs) or Instructions (EOIs) must implement the ATWS/Stability Mitigation Actions as described in References 8 and 10. The two main mitigation actions are: i. Following a failure to scram, the reactor vessel water level must be lowered to a level below the feedwater spargers that will allow vessel steam to preheat the cold feedwater. ii. if unstable power oscillations are detected following a failure to scram, boron injection through the SLCS must be initiated manually.					
15.8.2	For Pressurized Water Reactors (PWRs): A. Provide measures to automatically initiate the auxiliary (or emergency) feedwater system and a turbine trip under conditions indicative of an ATWS. This equipment shall be independent and diverse from the reactor trip system from sensor output to the final actuation device. B. Combustion Engineering or Babcock and Wilcox reactors applicants shall have provision for a scram system that is independent and diverse from the reactor trip system, from sensor output to the points of interruption of power to the control rods. C. These system and equipment shall be demonstrated to provide reasonable assurance that unacceptable plant conditions do not occur in the event of an anticipated transients					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	D. The reactor coolant system (RCS) pressure shall not exceed ASME Service Level C limits (approximately 22 MPa or 3200 psig) containment safety parameters (e.g., temperature or pressure) should not exceed design limits					
15.8.3	For Evolutionary Plants A. For evolutionary plants where the ATWS rule does not explicitly require a diverse scram system, the applicant may provide either of the following: i. A diverse scram system satisfying the design and quality assurance criteria specified in SRP Section 7.2 ii. Demonstrate that the consequences of an ATWS event are within acceptable values. B. For evolutionary plants, some of the equipment required to satisfy the rule may not be apply. For example, passive BWRs do not have recirculation pumps; therefore, these designs cannot provide equipment to trip them as required by the rule. For these designs provision of an equivalent action such as reducing the vessel water level may be acceptable. C. Applicants must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small because either: (1) the criteria are met, or (2) a diverse scram system is installed that reduces significantly the probability of a failure to scram. The analysis leading to the ATWS rule in NUREG-0460 used the following ATWS success					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	criteria, which have their bases in the Commission regulations and GDC listed above. Applicant's design shall maintain: i. Coolable geometry for the reactor core. If fuel and clad damage were to occur following a failure to scram, GDC 35 requires that this condition should not interfere with continued effective core cooling. 10 CFR 50.46 defines three specific core-coolability criteria: (1) Peak clad temperature shall not to exceed 1221°C (2200°F), (2) Maximum cladding oxidation shall not to exceed 17% the total cladding thickness before oxidation, and (3) Maximum hydrogen generation shall not to exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen. ii. Maintain reactor coolant pressure boundary integrity. Appendix A to WASH-1270 states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the 'emergency conditions' as defined in the ASME Nuclear Power Plant Components Code, Section III." The acceptance criteria for reactor coolant pressure, based upon the ASME Service Level C limits, are approximately 10.3 MPa (1500 psig) for BWRs and approximately 22MPa (3200 psig) for PWRs. ii. Maintain containment Integrity. Following a failure to scram, the containment pressure and					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	temperature must be maintained at acceptably low levels based on GDC 16 and 38. The containment pressure and temperature limits are design dependent; but to satisfy GDC 50, those limits must ensure that containment design leakage rates are not exceeded when subjected to the calculated pressure and temperature conditions resulting from any ATWS event.					
	REFERENCES: 8. A.C. Thadani, "Acceptance for Referencing of Topical Reports NEDO-32047 and NEDO-32164, Revision 0, BWR Owners' Group Evaluation of ATWS Rule Issues and Mitigative Actions," U.S. Nuclear Regulatory Commission, February 5, 1994. 10. NEDO-32164, Revision 0, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," General Electric Company, December 1992.					
15.9 (03/2007)	Boiling Water Reactor Stability					
15.9.1	To meet requirements of GDC 12 the reactor core and its systems should be designed with sufficient margin to be free of undamped oscillations and other thermal-hydraulic instabilities for all conditions of steady-state operation (including single-loop operation and extended-cycle operation with reduced feedwater temperature where these operating conditions are proposed) and for anticipated operational occurrences (AOOs).					
15.9.2	If potential oscillations cannot be eliminated, design proposals should detect and suppress them reliably and readily.					
15.9.3	Methodologies for resolving BWR density-wave stability issues are presented in the BWR Owners' Group topical report NEDO-31960 along with Supplement 1 and were approved by the NRC in Reference 2. These reports provide LTSs to BWR stability issues as well as methodologies developed to support the design					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	of systems needed for plants to comply with GDCs 10 and 12.					
15.9.4	A reactor is considered stable if it satisfies one of the following criteria: A. The calculated decay ratio (DR) for all three common stability modes (core-wide, regional, and channel) satisfies the relationship DR < (1 - σ) where σ is the uncertainty of the calculation. Staff review and approve both the calculation methodology and its uncertainty. The value of σ is typically 0.2 but is methodology-dependent. This value includes the code uncertainty and some degree of variability of the input parameters. B. Use of an approved correlation to estimate the stability of the regional stability mode based on calculated core-wide and channel DRs is permitted. One example is the FABLE/BYPSS Stability Criteria reviewed and approved by staff and documented in NEDO-31960.					
15.9.5	An acceptable LTS methodology to satisfy GDC 12 reduces the operating domain by defining an exclusion region where the reactor is not allowed to operate. The exclusion region, defined by the area in the operating map where stability criteria are not met, should be enforced automatically with an approved stability LTS. In addition to the exclusion region, the LTS defines a larger buffer region enforced with administrative controls. The buffer region minimizes challenges to the reactor protection system.					
15.9.6	An alternative acceptable LTS methodology to satisfy GDC 12 will readily detect and suppress unstable power oscillations by scramming the reactor before SAFDLs are violated. An approved D&S stability LTS should be implemented. SAFDL requirements					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	are specified in SRP Section 4.2, "Fuel System Design," and SRP Section 4.4, "Thermal and Hydraulic Design."					
15.9.7	Detect and Suppress stability LTSs rely on calculations of the reduction in critical power ratio margin for oscillations of a given amplitude. The response to these D&S LTS hardware oscillations should be modeled by a series of likely oscillation-amplitude contours and randomly failed local power range monitor (LPRM) instruments. Delta CPR over Initial MCPR Versus Oscillation Magnitude (DIVOM) is a staff-reviewed and approved methodology documented in NEDO-32465A.					
15.9.8	All stability LTS implementations should have backup options in case the licensing solution is declared inoperable. Technical specifications should require that the primary licensing solution be restored in a relatively short period (no longer than 120 days). Backup options in effect for short periods may rely on administrative controls and manual operator actions only if operator actions required to prevent SAFDL exceedences can be accomplished within the two minutes allowed for operator action in the demonstration calculations. Backup solution exclusion regions should be confirmed for specific cycles and specified in the core operating limits report (COLR).					
15.9.9	A number of stability LTSs has been reviewed and approved by the staff. As reactor and fuel designs evolve, the industry may propose new stability LTSs. The following criteria judge the acceptability of new stability LTSs and facilitates meeting the requirements of GDC 20: A. The LTS should protect against SAFDL violations automatically. B. The LTS should demonstrate by analysis that either (i) the probability of instabilities in the allowed operating region is sufficiently small or (ii) unstable power oscillations can be detected and suppressed					

Table A1-15: NUREG-0800, Standard Review Plan						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	readily without SAFDL violations. The LTS may use a combination of both demonstrations for different instability modes. C. If the licensing basis option is declared inoperable the LTS should provide a backup option that may implement manual or administrative actions only if operator actions required to prevent SAFDLs can be accomplished within the two minutes allowed for operator action in the demonstration calculations. D. The LTS option should include generic technical specifications that address: i. The methodology for setpoint and region calculation and documentation of the setpoint on a cycle-specific basis (e.g., COLR). ii. Operability and surveillance requirements for the licensing basis option. iii. A time limit (120 days maximum) for operation under the backup option.					
15.9.10	To meet requirements of GDC 13, stability-related instrumentation functionality should be demonstrated by analysis. Hardware implementation should follow SRP Section 7.2.					
15.9.11	In addition to the density-wave instability modes, the applicant should ensure that the plant is free from other instability modes that could violate SAFDLs (e.g., startup or control system instabilities) or that oscillations can be detected and suppressed readily. Note: Some instability modes may be acceptable with no potential for SAFDL violation, (e.g., bi-stable flow or small-flow oscillations during low-pressure startup).					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	REFERENCES: 2. A. C. Thadani, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, BWR Owners Group Long-Term Stability Solutions Licensing Methodology," U.S. Nuclear Regulatory Commission, July 12, 1993.					
	CHAPTER 16, Technical Specifications					
16.0, Rev. 2 (03/2007)	Technical Specifications					
II.1	The proposed plant–specific TS satisfy 10 CFR 50.34, 10 CFR 50.36, and 10 CFR 50.36a and are therefore acceptable if consistent with the regulatory guidance of the following STS documents and present plant–specific values for parameters at the indicated level of detail: • NUREG-1430, STS, Babcock and Wilcox Plants • NUREG-1431, STS, Westinghouse Plants • NUREG-1432, STS, Combustion Engineering Plants • NUREG-1433, STS, General Electric Plants, BWR/4 • NUREG-1434, STS, General Electric Plants, BWR/6					
II.2	In TS change requests for facilities with TS based on previous STS, licensees should comply with comparable provisions in these STS NUREGs to the extent possible or justify deviations from the STS. Acceptable justifications for deviation would include retention of existing TS requirements, non-adoption of STS requirements not represented in existing TS (e.g., an LCO in STS but not in existing TS), editorial preference, facility design, and a technically justified alternative presentation equivalent to the STS intent. In some cases, comparison to the previous STS may help evaluate the proposed changes by clarifying the TS intent. The previous STS NUREGs are as follows: • NUREG-0103, STS, Babcock and Wilcox Plants					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	• NUREG-0452, STS, Westinghouse Plants • NUREG-0212, STS, Combustion Engineering Plants • NUREG-0123, STS, General Electric Plants For applicants referencing a certified design, the DCD GTS of the referenced design provide the guidelines for the evaluation of proposed plant-specific TS.					
16.1, Rev. 1 (03/2007)	Risk-Informed Decision Making: Technical Specifications					
16.1.1	Traditional Engineering Guidelines 1. Defense in Depth. The licensee's engineering evaluation should state whether the impact of the proposed TS change is consistent with the defense-in-depth philosophy. The intent is to maintain the philosophy of defense in depth, not to prevent changes in achieving defense in depth. The defense-in-depth philosophy traditionally has been applied in reactor design and operation for multiple means of performing safety functions and preventing the release of radioactive material. It continues to be effective in accounting for uncertainties in equipment and human performance. When a comprehensive risk analysis can be done, it can help determine the appropriate extent of defense in depth (e.g., balance among core damage prevention, containment failure, and consequence mitigation) to protect public health and safety. When a comprehensive risk analysis is not done, traditional defense-in-depth considerations should account for uncertainties. The evaluation should consider intent of the general design criteria (GDCs), national standards, such engineering principles as the single-failure criterion, the impact of the proposed TS change on barriers (both preventive and mitigative) to core damage, containment failure or bypass, and the balance among defense-in-depth					

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	attributes. The licensee should select the engineering analysis techniques, whether quantitative or qualitative, traditional or probabilistic, appropriate to the proposed TS change. The licensee should assess whether the proposed TS or TS change meets the defense-in-depth principle. Defense in depth consists of numerous elements that can be assessment guidelines. Other equivalent acceptance guidelines also may be used. Consistency with the defense-in-depth philosophy is maintained if: (i) A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved (i.e., the proposed TS change does not change the balance among these principles of prevention and mitigation to the extent required by 10 CFR 50.36 (Reference 9)). TS change requests should consider whether anticipated operational changes made by a TS change could introduce or could increase the likelihood of new accidents or transients (as required by 10 CFR 50.92) (Reference 14). (ii) Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided (e.g., use of high reliability estimates based primarily on optimistic program assumptions). (iii) System redundancy, independence, and diversity are maintained commensurate with the expected frequency and					

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	consequences of challenges to the system (e.g., there are no risk outliers). The licensee should consider: (1) Whether appropriate restrictions are in place to preclude simultaneous equipment outages that would erode the principles of redundancy and diversity. (2) Whether compensatory actions when entering the modified AOT for pre-planned maintenance are identified. (3) Whether the TS change specifies that voluntary removal of equipment from service should not be scheduled when adverse weather conditions or other situations that likely may subject the plant to abnormal conditions are predicted. (4) Whether the TS change impact on the safety function should be considered (e.g., impact of an AOT change for the low-pressure safety injection system on the overall availability and reliability of the low-pressure injection function). (iv) Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed (e.g., TS change requests should consider whether the anticipated operational changes from an					

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	AOT or STI change could introduce any new common cause failure modes not previously considered). (v) Independence of physical barriers is not degraded. TS change requests should address the independence of barriers as not degraded by the change (e.g., containment system TS change). (vi) Defenses against human errors are maintained. TS change requests should consider whether the anticipated operation changes from an AOT or STI change could change the expected operator response or introduce any new human errors not previously considered (e.g., change from maintenance during shutdown to maintenance at power when different personnel and different activities may be involved). (vii) The intent of the GDCs in 10 CFR Part 50, Appendix A (Reference 15), is maintained. B. Safety Margins. The engineering evaluation should assess whether the impact of the proposed TS change is consistent with the principle of maintaining sufficient safety margins (Principle 3). An acceptable set of guidelines for that assessment are summarized here. Other equivalent guidelines are acceptable. Sufficient safety margins are maintained when: (i) Codes and standards (e.g., American Society of Mechanical Engineers, Institute of Electrical and Electronic Engineers) or alternatives approved by the NRC are met					

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	(e.g., the proposed TS AOT or STI change is not in conflict with approved codes and standards for the subject system). (ii) Safety analysis acceptance criteria in the final safety analysis report (FSAR) are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties (i.e., the proposed TS AOT or STI change does not adversely affect any assumptions or inputs to the safety analysis or justification ensures continued sufficient safety margin). For TS AOT changes, the effect on FSAR acceptance criteria should be assessed, assuming the plant is in the AOT (i.e., the subject equipment is inoperable) and there are no additional failures. The assessment should identify all situations in which entry into the proposed AOT could result in failure to meet an intended safety function. C. Need for and Adequacy of Change. The licensee has demonstrated that the change is needed for adequate reliability and availability of significant safety systems. D. Justification. The licensee has provided the justification for the change based on the guidance of subsection III.A of this SRP section.					
16.1.2	Probabilistic Guidelines. The guidelines stated in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," Sections 2.2.4 and 2.2.5 (Reference 16), apply to TS change requests. Those sections present risk-acceptance guidelines as functions of the licensee's risk analysis of predicted changes in total core damage frequency (CDF) and large early					

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	release frequency (LERF) for the TS change requested. In addition, those sections address cases when the scope of the licensee's PRA does not include a Level 2 (containment performance) analysis, and when, according to the guidelines of RGs 1.174 and 1.177, such an analysis is needed. TS submissions for AOT changes should be evaluated against the risk acceptance guidelines in this section in addition to those in RG 1.174. Application of all risk acceptance guidelines to TS modification proposals will be consistent with the fundamental principle that TS changes result in small increases in the risk to the health and safety of the public (Principle 4, as described in the "Discussion" section of RG 1.177) (Reference 8). General guidance for evaluating the risk impact from TS and other changes is in SRP Section 19.1. TS change evaluations may involve some small increase in risk as quantified by PRA models. The usual argument is that such a small increase is offset by the many beneficial effects of the change not modeled by the PRA. The numerical guidelines ensure that the risk increase is small and provide a quantitative basis for the risk increase according to modeled or quantified aspects of the TS change. The numerical guidelines for an acceptable TS change are taken into account along with other traditional considerations, operating experience, lessons learned from previous changes, and practical considerations for test and maintenance practices. The final acceptability of the proposed change should be based on all of these considerations and not solely on PRA-informed results compared to numerical acceptance guidelines. The numerical guidelines ensure that any increased risk is within acceptable limits; traditional considerations ensure that the change meets rules and regulations in effect; practical					

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	considerations judge the acceptability of the change; and lessons learned from past experience ensure that mistakes are not repeated. Using the risk measures addressed in RG 1.177, the change in risk should be calculated for the TS changes and compared against the numeric guidelines referenced in this section. In calculating the risk impact of the change, additional changes from the change can be credited (e.g., for an STI change, if the test strategy also is changed, the effect should be incorporated in the risk evaluation). However, this SRP and RGs 1.177 and 1.174, apply only to permanent (as opposed to temporary or "one-time") changes to TS requirements. TS AOT changes are permanent but, because AOTs are entered infrequently and are temporary by their very nature, the following TS acceptance guidelines for AOT changes evaluate the risk of the revised AOT additionally to the evaluation by the RG 1.174 acceptance guidelines. A. The licensee has demonstrated that the TS AOT modification has only a small quantitative impact on plant risk. An incremental conditional core damage probability (ICCDP)[2] of less than 5.0E-7[3] is small for a single TS AOT modification. An incremental conditional large early release probability (ICLERP)[4] of 5.0E-8 or less is also small. Also, the ICCDP contribution should be distributed in time so any increase in conditional risk is small and within the normal operating background (risk fluctuations) of the plant (Tier 1). B. The licensee has demonstrated appropriate restrictions on dominant risk-significant configurations of the modifications (Tier 2). C. The licensee has a risk-informed plant configuration					

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	control program with procedures to utilize, maintain, and control it (Tier 3). In the context of integrated decision-making, application of the acceptance guidelines should not be overly prescriptive. They are intended to indicate, in numerical terms, what is acceptable. The numerical values are approximate and indicate changes generally acceptable. The intent in comparing PRA results to the acceptance guidelines is to demonstrate with reasonable assurance that Principle 4, addressed in the "Discussion" section of RG 1.177 (Reference 8), is met. The decision must be based on a full understanding of the contributors to the PRA results and the impacts of the uncertainties, both those explicitly considered in the results and those not. A nonquantitative assessment of risk (either alone or accompanied by quantitative assessment) may suffice to justify TS changes. The licensee is expected to use judgment on the acceptability (to support regulatory decision-making) of the risk argument, including the appropriate blend of quantitative and qualitative assessments.					
	NOTES: 2. ICCDP = [(conditional CDF with the subject equipment out of service)-(baseline CDF with nominal expected equipment unavailabilities)] x (duration of single AOT under consideration). 3. The ICCDP acceptance guideline of 5.0E-7 is based upon the hypothetical situation of subject equipment at a representative plant out of service for five hours, causing the CDF of the plant with an assumed baseline CDF of 1.0E-4 per reactor year to increase conditionally to 1.0E-3 per reactor year during the five-hour period. This basis assumes that the majority of repairs can be made in five					

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	hours or less and that the NRC has accepted this level of risk for operating plants. 4. ICLERP = [(conditional LERF with the subject equipment out of service)-(baseline LERF with nominal expected equipment unavailabilities)] x (duration of single AOT under consideration).					
	REFERENCES: 8. USNRC, "An Approach for Plant-Specific Risk-Informed Decisionmaking: Technical Specifications," Regulatory Guide 1.177, August 1998. 9. NRC, Statement of Considerations, "Technical Specifications for Facility Licensees; Safety Analyses Reports," Federal Register, 33 FR 18612, December 17, 1968. 14. USNRC, 10 CFR 50.92, "Issuance of Amendment," Federal Register, 51 FR 7767, March 6, 1986. 15. USNRC, Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Federal Register, 52 FR 41294, October 27, 1987. 16. USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," RG 1.174, July 1998.					
	CHAPTER 17, Quality Assurance					
17.1, Rev. 2 (07/1981)	Quality Assurance During the Design and Construction Phases					
	Refer to the BTP for the detailed criteria.					
17.2, Rev. 2 (07/1981)	Quality Assurance During the Operations Phase					
	Refer to the BTP for the detailed criteria.					

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17.3 (07/1981)	Quality Assurance Program Description					
	Refer to the BTP for the detailed criteria.					
17.4 (03/2007)	Reliability Assurance Program (RAP)					
17.4.1	A. DESIGN CERTIFICATION The application describes the following RAP information: 1. The scope and purpose. The scope and purpose of the RAP are described in Subsections I and II of this SRP section. 2. The application of the quality elements associated with organization, design control, procedures and instructions, records, corrective action, and audit plans as follows: a. Organization 1) The organizations responsible for formulating and implementing the RAP and the coordination of RAP program activities, including those performed within the design, PRA, RAP, and risk and reliability organizations as well as work completed by the architect-engineers and other supporting organizations that develop deterministic and other methods used to identify SSCs that are significant contributors to plant safety. 2) How the reliability and design organizations manage interface issues. For example, how does the risk and reliability organization keep the design staff cognizant of SSCs that are					

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	significant contributors to plant safety, program needs, and status, and how does the feedback process ensure that significant design assumptions related to equipment reliability are realistic and achievable. 3) The risk and reliability organization participation in the design change control process for the purpose of providing RAP related inputs in the design process. 4) The risk and reliability organization involvement in design reviews. b. Design Control 1) The configuration control process for maintaining the list of SSCs within the scope of RAP similar to the control of a quality list. 2) How the design control and change processes provide a feedback mechanism for notifying the appropriate organization of changes in the design of SSCs within the scope of the RAP that could affect the probabilistic/PRA, deterministic, or other methods used to identify SSCs that are significant contributors to plant safety. 3) The interface between the risk and reliability and the design organizations for determining that the performance of SSCs within the scope of the RAP relate to the reliability assumptions in the probabilistic/PRA, deterministic, or other					

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	methods used to identify SSCs that are significant contributors to plants safety. 4) Engineering design controls applied for determining the SSCs within the scope of the RAP. 5) The process for proposing an alternative design to reliability performance. For example, is the revised design reviewed to provide confidence that the current reliability assumptions are still valid. c. The controls for procedures and instructions used to implement the RAP. d. The controls for records of activities involving SSCs within the scope of RAP. e. The corrective action process applied to SSCs within the scope of RAP. f. The audit plans for conducting QA audits of RAP activities. 3. The expert panel qualifications in the areas of personnel knowledgeable in the design, operation and maintenance of a plant, and experience necessary to perform the SSC selections if an expert panel is utilized. 4. Deterministic or other methods of analysis used to identify SSCs included in the RAP and the SSCs affected. 5. A non-system-based ITAAC for the RAP that provides					

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	reasonable assurance that the design of SSCs within the scope of the RAP is consistent with their assumed design reliability. The ITAAC acceptance criteria should ensure that the estimated reliability of each as-built SSC is at least equal to the assumed design reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs. 6. A COL action item that a COL applicant referencing a certified design will identify the site-specific SSCs within the scope of the RAP.					
	B. COL APPLICANT The COL applicant should include the following RAP information: 1. The same information provided in the previous Sections A.2.a, A.2.b and A.3 and A.4 for the site-specific phase of the RAP if not previously addressed in Section A. 2. How procurement, fabrication, construction, and test specifications for the SSCs within the scope of the RAP ensure that significant assumptions, such as equipment reliability, are realistic and achievable. 3. How QA requirements are implemented during the procurement, fabrication, construction and preoperation testing of SSCs within the scope of the RAP. 4. A description of the integration of reliability assurance activities into existing programs (e.g., maintenance rule, surveillance testing, in-service inspection, inservice testing, and QA). The description should address the					

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	following: a. How the reliability performance goals for SSCs within the scope of the RAP are established. For example, implementation of the maintenance rule following the guidance contained in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," is one acceptable method for establishing performance goals provided that SSCs are categorized as high safety significant. b. The feedback mechanism for periodically evaluating reliability assumptions on the basis of actual equipment, train, or system performance. This description should include any key assumptions and determinations of risk significance that are derived from probabilistic/PRA, deterministic, or other methods that consider operations, maintenance, and monitoring activities for identifying component reliability and failure data. The description should also include how industry operational experience will be used to verify that reliability assumptions remain valid. (The reliability performance monitoring does not need to statistically verify the numerical values. However, it provides a feedback mechanism for periodically evaluating equipment reliability on the basis of actual equipment, train, or system performance and other operational history.) 5. The process for providing corrective actions for design and operational errors that degrade nonsafety-related SSCs within the scope of the RAP.					

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	6. Controls for the procedures and instructions used to implement reliability assurance activities. 7. The controls for records of activities involving SSCs within the scope of RAP. 8. The corrective action process applied to SSCs within the scope of RAP. 9. The audit plans for conducting QA audits of reliability assurance activities.					
17.5 (03/2007)	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants					
	Refer to the BTP for the detailed criteria.					
17.6 (03/2007)	Maintenance Rule					
17.6.1	NUMARC 93-01 as endorsed by RG 1.160 represents an acceptable approach for implementing a Maintenance Rule program in accordance with 10 CFR 50.65. For 50.65(a)(4), the guidance contained in the February 22, 2000, revision to Section 11 of NUMARC 93-01, as endorsed by RG 1.182, is effective until this guidance has been incorporated into a revision of NUMARC 93-01 later than Revision 3 and endorsed by a revision of RG 1.160 later than Revision 2, which will supersede RG 1.182. The applicant's program should be consistent with the industry guidance as endorsed. Deviations should be explained and justified.					
17.6.2	Operational Programs For COL reviews, the description of the operational program and					

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	proposed implementation milestones for the Maintenance Rule program are reviewed in accordance with 10 CFR 50.65. The implementation milestones are plant specific except that 50.65 will require that the program be fully implemented by the time fuel load is authorized.					
	CHAPTER 18, Human Factors Engineering					
18.0, Rev. 2 (03/2007)	Human Factors Engineering					
18.0.1	A. Review of the HFE Aspects of a New Plant A.1 HFE Program Management The objective of this review is to confirm that the applicant has adequately considered the role of HFE and the means by which HFE activities will be accomplished. The review should verify that: • The applicant has identified plans to oversee design and construction of the nuclear facility in accordance with the requirements of 10 CFR 50.34(f)(3)(vii), as described in SRP Section 13.1.1, "Management and Technical Support Organization." • The applicant has an HFE design team with the responsibility, authority, placement within the organization, and composition to ensure that the design commitment to HFE is achieved. There is, however, no assumption that HFE is the responsibility of a single organization or that there is an organizational unit called the HFE design team. • The team is guided by an HFE program plan to ensure the properdevelopment, execution, oversight, and documentation of the HFE program. • The overall HFE program appropriately considers and address the deterministic aspects of the design, as discussed in RG 1.174. The HFE program plan should describe the technical program					

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	in sufficient detail to ensure that all aspects of the HSIs, procedures, and training are developed, designed, and evaluated on the basis of a structured top-down systems analysis using accepted HFE principles. The applicant's HFE program management should be evaluated in accordance with the review criteria of NUREG-0711, "Human Factors Engineering Program Review Model." A.2 Operating Experience Review The objective of this review is to verify that the applicant has identified and analyzed HFE-related problems and issues in previous designs so that these problems and issues may be avoided in the development of the new design. This review should also verify that the applicant has retained positive features of previous designs. The operating experience review (OER) should be evaluated in accordance with the review criteria of NUREG-0711 and should satisfy the requirements of 10 CFR 50.34(f)(3)(i) and 52.49(a)(21). A.3 Functional Requirements Analysis and Function Allocation Functional requirements analysis is the identification and analysis of those functions that must be performed to satisfy plant safety objectives; that is, to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Function allocation analysis is the analysis of requirements for plant control and the assignment of control functions to (1) personnel (e.g., manual control), (2) system elements (e.g., automatic control and passive, self-controlling phenomena), and (3) combinations of personnel and system elements (e.g., shared control, automatic systems with manual backup).					

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	The objective of this review is to verify that (1) the plant's functions that must be performed to satisfy plant safety objectives have been defined, and (2) that the allocation of those functions to human and system resources has resulted in a role for personnel that takes advantage of human strengths and avoids human limitations. Functional requirements analysis and function analysis should be evaluated in accordance with the review criteria of NUREG-0711. A.4 Task Analysis Task analysis is the analysis of human performance that results from the allocation of functions to personnel and the identification of HSI characteristics needed to support personnel task accomplishment. The objective of this review is to ensure that the applicant's task analysis identifies the specific tasks that are needed for function accomplishment and their information, control, and task-support requirements. The task analysis should be evaluated in accordance with the review criteria of NUREG-0711. A.5 Staffing and Qualifications The objective of this review is to verify that the applicant has analyzed the requirements for the number and qualifications of personnel in a systematic manner that includes a thorough understanding of task requirements and applicable regulatory requirements. The applicant's staffing and qualifications analyses should be evaluated in accordance with the review criteria of NUREG-0711 and should satisfy the requirements of 10 CFR 50.54 (i) through (m). If an exemption from these requirements is being sought, the analysis and justifications should be presented [see also NUREG/CR-6838, "Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear					

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	Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)" and NUREG-1791, "Guidelines for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operating Staff Requirements Specified in 10 CFR 50.54(m) — Final Report"]. The full staffing program is considered to be an operational program as discussed in SECY-05-197 and in RG-1.206 "Combined License Applications for Nuclear Power Plants (LWR Edition)", Section C.IV.4, "Operational Programs." A.6 Human Reliability Analysis Human reliability analysis (HRA) is an evaluation of the potential for and mechanisms of human error that may affect plant safety. The objectives of this review are to ensure that (1) the applicant has addressed human-error mechanisms in the design of the HFE aspects of the plant to minimize the likelihood of personnel error, and detect errors and recover from them; and (2) the HRA activity effectively integrates the HFE program and PRA. A design-specific PRA/HRA is required by 10 CFR 50.34(f)(1)(i), 52.47(b)(1) and 52.79, and is addressed in SRP Chapter 19 and RG 1.206 Section C.II.1. RG 1.206 Section C.II.1 specifies the purpose and objectives of the PRA, as well as the required scope and level of detail. In order to accomplish the above objectives, the HRA/PRA and the modeling of HAs must be of sufficient quality (see SRP Chapter 19 and RG 1.206 Section C.II.1). Review of the HRA should be coordinated with SRP Section 6.3.III.19 and RG 1.206 Section C.I.6.3.2.8 as they relate to manual actions for ECCS. The integration of the applicant's HRA with the HFE program should be evaluated in accordance with the review criteria of NUREG-0711.					

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	A.7 Human-System Interface Design The HSI design process represents the translation of function and task requirements into HSI characteristics and functions. The objective of this review is to evaluate the process by which HSI design requirements are developed and HSI designs are identified and refined. The review should verify that the applicant has appropriately translated functional and task requirements to the detailed design of alarms, displays, controls, and other aspects of the HIS through the systematic application of HFE principles and criteria. The applicant's HSI design process should be evaluated in accordance with the review criteria of NUREG-0711, and the final design evaluated in accordance with the review criteria of NUREG-0700, "Human-System Interface Design Review Guidelines." The HSI design should address those subsections of 10 CFR 50.34(f)(2) that are applicable to the plant's design from the following list: 10 CFR 50.34(f)(2)(i), (iii), (iv), (v), (xi), (xii), (xiii), (xv), (xvii), (xviii), (xix), (xxi), (xxiv), (xxv), & (xxvii). In addition to the HFE considerations discussed above, the following specific HIS design guidance should also be addressed: 1. Safety parameter display system requirements, as described in 10 CFR 50.34(f)(2)(iv), NUREG-0835, NUREG-1342, and Supplement 1 of NUREG-0737. 2. Periodic testing of protection systems actuation functions, as described in Regulatory Guide 1.22. 3. Bypassed and inoperable status indication for NPP safety systems, as described in Regulatory Guide 1.47. 4. Manual initiation of protective actions, as described in					

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	Regulatory Guide 1.62. 5. Instrumentation for light-water-cooled nuclear power plants to access plant and environmental conditions during and following an accident, as described in Regulatory Guide 1.97. 6. Instrumentation setpoints, as described in Regulatory Guide 1.105. 7. Functional criteria for emergency response facilities, as described in NUREG-0696. 8. A minimum inventory of controls, displays and alarms. The HSI design should describe the process, after the plant is in operation, by which (1) HSIs are modified and updated, (2) temporary HSI changes are made (such as set point modification) and (3) operator defined HSIs are created (such as temporary displays defined by operators for monitoring a specific situation). The HSI design review should be coordinated with the instrumentation and controls review in SRP Chapter 7. A.8 Procedure Development The objective of this review is to confirm that the applicant's procedure development program incorporates HFE principles and criteria, along with all other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to utilize, validated, and in conformance with 10 CFR 50.34(f)(2)(ii). Because procedures are considered an essential component of the HFE design, they should be derived from the same design process and analyses as the other components of the HSI (e.g., displays, controls, operator aids) and subject to the					

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	same evaluation processes. The applicant's procedure development program should be evaluated in accordance with the review criteria of NUREG-0711. The review should be coordinated with the review of procedures described in SRP Section 13.5. The full procedures program is considered to be an operational program as discussed in SECY-05-197 and in RG-1.206 Section C.IV.4. A.9 Training Program Development The objective of this review is to ensure that the applicant has a systematic approach for the development of personnel training. The training development should include the following five activities: • A systematic analysis of tasks and jobs to be performed • Development of learning objectives derived from an analysis of desired performance following training • Design and implementation of training based on the learning objectives • Evaluation of trainee mastery of the objectives during training • Evaluation and revision of the training based on the performance of trained personnel in the job setting The training program should be developed in accordance with 10 CFR 50.120, 10 CFR 52.79, and 10 CFR Part 55 to ensure that personnel's qualifications are commensurate with the performance requirements of their jobs. The applicant's training program should be evaluated in accordance with the review criteria of NUREG-0711 and should address applicable guidance provided in SRP Section 13.2, "Training." The full training program is considered to be an operational program as discussed in SECY-05-197 and in RG-1.206 Section C.IV.4.					

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	A.10 Verification and Validation Verification and validation (V&V) evaluations seek to comprehensively determine that the design conforms to HFE design principles and that it enables plant personnel to successfully perform their tasks to achieve plant safety and other operational goals. The overall scope for V&V should include the main control room, the remote shutdown panel, and local control stations (including the central alarm system (CAS) and secondary alarm system (SAS) associated with the risk important HAs. The applicant's V&V activities include operational condition sampling, design verification, integrated system validation, and human engineering discrepancy (HED) resolution. The objectives of the staff review of each of these activities are identified in the following subsections. A.10.1 Operational Conditions Sampling The applicant's sampling methodology identifies the range of operational conditions that guide V&V activities. The objectives of the review are to ensure that the applicant has identified a sample of operational conditions that (1) includes conditions that are representative of the range of events that could be encountered during operation of the plant, (2) reflects the characteristics that are expected to contribute to system performance variation, and (3) considers the safety significance of HSI components. The use of risk importance to help select failure events, transients, and accidents for use in V&V is appropriate. The applicant's operational conditions sampling should be evaluated in accordance with the review criteria of NUREG-0711. A.10.2 Design Verification					

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	The applicant's verification should demonstrate that the design meets task and human requirements. Verification activities require a characterization of the HSI. The staff's review of design verification has the following objectives: • Inventory and Characterization Review - The objective of this review is to evaluate whether the applicant's HSI inventory and characterization accurately describes all HSI displays, controls, and related equipment that are within the defined scope of the HSI design review. • HSI Task Support Verification Review - The objective of this review is to evaluate whether the applicant verifies that the HSI provides all alarms, information, and control capabilities required for personnel tasks. • HFE Design Verification Review - The objective of this review is to evaluate whether the applicant verifies that the characteristics of the HIS and the environment in which it is used conform to HFE guidelines. The applicant's design verification should be evaluated in accordance with the review criteria of NUREG-0711. A.10.3 Integrated System Validation The objective of integrated system validation is to confirm that the integrated system design (i.e., hardware, software, and personnel elements) acceptably supports safe operation of the plant. Validation is based on performance-based tests. The applicant's integrated system validation should be evaluated in accordance with the review criteria of NUREG-0711. A.10.4 Human Engineering Discrepancy (HED) Resolution HED resolution is the process of evaluating and resolving issues that are identified in V&V evaluations. The objectives of the staff's review are to verify that the applicant's HED evaluation acceptably					

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	prioritizes HEDs in terms of their need for improvement and that design solutions and a realistic schedule for implementation is developed to address those HEDs selected for correction. The applicant's HED resolution should be evaluated in accordance with the review criteria of NUREG-0711. A.11 Design Implementation The objective of this review is to verify that the applicant's as-built design will conform to the verified and validated design that resulted from the HFE design process. The applicant's design implementation process should be evaluated in accordance with the review criteria of NUREG-0711. This review should also ensure the acceptability of the applicant's plans for determining the operability of the MCR, RSP, LCSs, Technical Support Center and Emergency Operations Facility. A.12 Human Performance Monitoring The objective of this review is to assure that the applicant has prepared a human performance monitoring strategy for ensuring that no significant safety degradation occurs because of any changes that are made in the plant and to verify that the conclusions that have been drawn from the evaluation remain valid over the life of the plant. The applicant's performance monitoring strategy should be evaluated in accordance with the review criteria of NUREG-0711.					
18.0.2	Review of the HFE Aspects of Control Room Modifications License amendments involving major changes to the HSIs, such as control room modernization, should be reviewed using the guidance contained in Section II.A of this SRP chapter. However, since the extent of such modifications can vary, the staff's review should be tailored using the additional guidance from NUREG-0711 and presented in this section. B.1 HFE Program Management					

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	The goals of the HFE program should address the need to consider the effects that the modification may have on the performance of personnel. The review should address the applications plan with respect to the following: • Planning the installation to minimize disruptions to work of plant personnel • Coordinating training and procedure modifications with implementing the modification to verify that both accurately reflect the characteristics of the modification • Conducting training to maximize personnel's knowledge of and skill with the new design before its implementation B.2 Operating Experience Review (OER) The operating experience of the plant being modified and plants with similar modifications should be reviewed as part of the OER. The OER should provide information on past performance of predecessor designs or earlier designs on which the new plant is based. B.3 Functional Requirements Analysis and Function Allocation Functional requirements analysis and function analysis should consider the following: • Functional requirements analyses for modifications that are likely to change existing safety functions, introduce new functions for systems supporting safety functions, or involve unclear functional requirements that may be important to safety. • Function allocation analyses for modifications that are likely to change the allocation between personnel and plant systems of functions important to safety. • A change in an operator's role due to a modification should be examined within the context of its effects on the operator's					

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	overall responsibilities. B.4 Task Analysis The following considerations should be addressed in the review of plant modifications that are likely to affect human actions (HAs) previously identified as risk-important, cause existing HAs to become risk-important, or create new actions that are risk-important: • The tasks analyses should be revised and updated to reflect requirements of the modification; the scope should include tasks involving the modification and its interactions with the rest of the plant, including those resulting from functions addressed in the analyses of functional requirements and function allocation. For maintenance, tests, inspections, and surveillances, attention should be given to risk-important actions that are new or supported by new technologies (e.g., new capabilities for online maintenance). • The task analysis should identify the design characteristics of the existing HSIs that support the performance of experienced personnel (e.g., support high levels of performance during demanding situations). B.5 Human-System Interface Design The following considerations should be addressed in the review of design modifications: • The extent to which HSI modifications are consistent with users' existing strategies and the licensee's SAR and Chapter 18 commitments. • The extent to which HSI modifications support crew coordination					

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	• The degree to which the HSI reflects changes resulting from integration among plant systems The final design modifications should be reviewed in accordance with the review criteria of NUREG-0700, as applicable. B.6 Procedure Development The review should evaluate whether procedures are modified and whether their content, format, and integration accurately reflect changes in the plant, human actions, and HSIs. B.7 Training Program Development The review should evaluate whether any changes in training content or frequency are warranted following plant modernization programs. B.8 Verification and Validation 1. Operational Conditions Sampling. V&V of the modification should reflect expected operational conditions and should address the potential effect of negative transfer of learning when the new and old components are different and impose different demands on personnel. The applicant's sampling should also consider any effects on performance of having both old and new versions of the same HSI components in place. 2. HSI Task Support Verification. HSI task support verification should focus on the HSIs that are relevant to the modification. For modifications to plant systems that do not include modifications of the HSIs, task support verification should identify any new demands for monitoring and control, and determine whether they are adequately addressed by the existing HSI design. HSIs for temporary configurations and situations where both old and new HSIs are left in place should be evaluated for their potential to					

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	negatively impact performance. 3. HFE Design Verification. HFE design verification should focus on the HSIs that are relevant to the modification. HSIs for temporary configurations and situations where both old and new HSIs are left in place should be evaluated for their potential to negatively impact performance. 4. Integrated System Validation. The applicant should perform an integrated system validation for all modifications that may (1) change personnel tasks; (2) change task demands, such as by changing task dynamics, complexity, or workload; or (3) interact with or affect HSIs and procedures in ways that may degrade performance. Integrated system validation may not be needed when a modification results in minor changes to personnel tasks such that they may reasonably be expected to have little or no overall effect on workload and the likelihood of error. The staff should verify that the applicant validates that the functions and tasks allocated to plant personnel can be accomplished effectively when the integrated design is implemented. The applicant's test objectives and scenarios should be developed to address aspects of performance that are affected by the modification design, including personnel functions and tasks affected by the modification. B.9 Design Implementation The objective of this review is to verify that the applicant's implementation of plant changes considers the effect on personnel performance and provides the necessary support for safety of operations. The applicant's design implementation should be evaluated in accordance with the review criteria of NUREG-0711. The following aspects of the design process should be addressed.					

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	1. General Criteria. The staff's review should address whether the applicant has provided assurance that: • The reactor fuel is safely monitored during the shutdown time period while the physical modifications are being implemented in the control room. • Operations and maintenance crews are fully trained and qualified to operate and maintain the plant prior to starting up with the new systems and HSIs in place. • Modifications in plant procedures and training reflect changes in plant systems, crew roles and responsibilities, HSIs, and that procedures required for the testing and operation of new systems and HSIs are in place prior to the modification being placed into service. • The applicant has a plan to monitor the system performance to identify and address any problems that arise. 2. Modernization Programs Consisting of Many Small Modifications. The staff's review should address whether the applicant can verify that each modification follows an HFE program for the maintenance of standardization and consistency, and that modifications fulfill a clear operational need and do not interfere with existing systems. 3. Modernization Programs Consisting of Large Modifications During Multiple Outages. The staff's review should address whether the applicant can verify that: • Task analysis is performed for each interim configuration to verify that the task demands that are unique to interim configurations are known. • HRA addresses any unique tasks that may affect risk or any changes to existing tasks due to the interim configuration.					

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	• The HSIs needed to perform important tasks are consistent and standardized. • Procedures are developed for temporary configurations of systems and HSIs that are used by personnel when the plant is not shut down. • Training is developed for temporary configurations of systems, HSIs, and procedures that are used by personnel when the plant is not shut down. • Temporary operational configurations are evaluated using V&V. 4. Modernization Programs Where Both Old and New Equipment Are Left in Place. The staff's review should address whether the applicant can verify that the potential for negative effects on personnel performance has been evaluated. 5. Modernization Programs Where New Nonfunctional HSIs Are In Place In Parallel With Old Functional HSIs. The staff's review should address whether the applicant can verify that the potential for negative effects on personnel performance due to control room or HSI clutter arising from having both old and new HSIs available in parallel is evaluated and that the nonfunctional state of the HSIs is clearly indicated.					
18.0.3	Review of the HFE Aspects of Modifications Affecting Risk-Important Human Actions The staff's review of license amendments and actions involving plant changes that affect important human actions (HAs) use a graded, risk-informed approach in conformance with Regulatory Guide (RG) 1.174. The staff's review uses a two-phase approach. The first phase is a screening analysis to determine the risk associated with the plant modification and its associated HAs using both quantitative and qualitative information (see Section					

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	C.1 below). This approach can be accomplished for submittals by licensees that are either risk-informed or non-risk-informed. Plan modifications and Has are categorized into regions of high, medium, and lower risk. This categorization is used to determine the level of HFE review needed. The second phase of the review is performed by the human factors analyst and consists of the HFE review. Changes that involve more risk-significant HAs receive a detailed review (see Section C.2.1 below), while those of moderate risk significance receive a less detailed review (see Section C.2.2 below). HAs in the lowest risk region receive minimal HFE review (see Section C.2.3 below). C.1 Phase I - Risk Screening C.1.1 Screening Process for Risk-Informed Change Requests If the submittal is appropriately risk-informed, applicants should evaluate the risk associated with the proposed modification and the HAs associated with it. The applicant's risk screening should be evaluated in accordance with the review criteria of "Guidance for the Review of Changes to Human Actions" (NUREG-1764), as summarized in the four paragraphs below. Determine the Risk of the Entire Modification. The first review step is to perform a risk-informed screening of the entire modification, including both equipment and HAs, in accordance with the review criteria of NUREG-1764, for both permanent and temporary changes. As part of this evaluation, the staff should determine whether the PRA information submitted as part of the risk-informed (R-I) submittal is suitable. The review criteria defined in RG 1.174 and SRP Chapter 19 should be used. If the staff determines that the information is not suitable, a generic					

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	method screening process should be used (see item C.1.2 below). RG 1.174 notes that licensee applications that lie in Region I of the acceptance guidelines for core damage frequency (or for large early release frequency) are generally not permitted. Proposed changes that are calculated to be in the Region I of three risk regions are identified as most risk significant. If the entire modification is in Region I, the staff determines whether the modification is rejected. If it is rejected, then no additional HFE review is needed. If it is not rejected, the staff determines whether the modification contains only HAs or if it includes both equipment and HAs. If the modification contains only HAs (no equipment modifications) and was determined to be in Region I, then the HA should be reviewed using the Level I criteria in Section C.2.1 below. If the modification contains equipment and HAs, then the risk importance of the HA should be evaluated (see item 2 below). Determine the Risk of the HAs. The second review step is to perform a risk-informed screening of the HA portion of the modification in accordance with the review criteria of NUREG-1764. This is done by evaluating both the risk achievement worth (RAW) and the Fussell-Vesely (FV) risk importance measures. HAs will be preliminarily sorted into the three Levels. Perform Qualitative Screen of the HAs. The third risk-screening step is to identify whether there are qualitative factors that should be taken into account when determining the risk importance of the HA. This step may be used to adjust the review level either up or down. This evaluation should be in accordance with the review criteria of NUREG-1764. Integrated Assessment of Human Actions Safety Significance. This step provides guidance on how to integrate the results from Steps 1 through 3 of the screening process for risk-informed licensing basis change requests.					

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	C.1.2 Screening Process for Non-risk-informed Change Requests If the submittal is appropriately non-risk-informed, the NRC will perform the risk screening as follows: Review of Non-Risk-Informed Submittals. In keeping with RG 1.174, a licensee submittal to the NRC may or may not be risk-informed) at the licensee's option. If it is not risk informed, then the staff may choose to use an Estimated Risk Method or a Generic Method to determine risk in accordance with the review criteria of NUREG-1764. These methods will result in a proposed Level (I, II, or III) for the review. Qualitative screening is then applied to the proposed level to see if it needs to be adjusted. Alternatively, the staff may choose to perform a deterministic review without using the risk screening methodology. Also, using guidance provided in SRP Chapter 19 and NRC Regulatory Issue Summary 2001-02, "Guidance on Risk-Informed Decision Making in License Amendment Reviews", the staff may determine that "special circumstances" exist that could result in the staff requesting the license to submit risk information. Integrated Assessment of Human Actions Safety Significance. The integrated assessments of HA safety significance for non risk-informed applications is similar to that for risk-informed applications, but simpler because there are fewer inputs to integrate. C.1.3 Determine the Level of HFE Review. Based on the quantitative and qualitative information available, the staff should classify the HA into one of three HFE review levels in accordance with the review criteria of NUREG-1764. • Level I HAs, high risk, are reviewed using the criteria in Section C.2.1 below.					

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	• Level II HAs, moderate risk, are reviewed using the criteria in Section C.2.2 below. • Level III HAs, minimal risk, are reviewed using the criteria in Section C.2.3 below. C.2 Phase II - HFE Review C.2.1 Level I HFE Review HAs in the high-risk category should be reviewed using the Level I review criteria provided below. 1. General Deterministic Review Criteria. The applicant should provide adequate assurance that deterministic aspects of design, such as whether the change meets current regulations, does not compromise defense-in-depth, and maintains sufficient safety margins, as discussed in RG 1.174, have been appropriately addressed. The staff should evaluate the deterministic aspects of the design in accordance with the review criteria of NUREG-1764. 2. Operating Experience Review. The applicant should identify and analyze HFE-related problems and issues encountered previously in designs and human tasks that are similar to the planned modification so that issues that could potentially hinder human performance can be addressed. The OER should address the operating histories of plant systems, HAs, procedures, and HSI technologies related to the proposed changes to HAs. The staff's evaluation should be conducted in accordance with the review criteria of NUREG-1764. 3. Functional Requirements Analysis And Functional Allocation. The applicant should define any changes in the plant's safety functions (functional requirements analysis), and provide evidence that the allocation of functions between humans and					

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	automatic systems provides an acceptable role for plant personnel; i.e., the allocations take advantage of human strengths and avoid functions that would be negatively affected by human limitations (functional allocation). The staff's review should address all plant functions affected by the change in HAs, including changes to the functions and to their allocation between personnel and automatic systems in accordance with the review criteria of NUREG-1764. 4. Task Analysis. The applicant should identify the behavioral requirements of the tasks personnel are required to perform. The task analysis should form the basis for specifying the requirements for the HSI, procedures, and training. The task analyses should address HAs in their entirety, including all pertinent plant conditions, situational factors, and performance-shaping factors. While the primary focus is licensed operator tasks, tasks performed by other personnel (e.g., emergency actions, maintenance, testing, inspection, and surveillance) that occur at the same time as the HAs and directly influence the actions are included in the task analysis. The staff should review the applicant's task analysis in accordance with the review criteria of NUREG-1764. 5. Staffing and Qualifications. The applicant should analyze the proposed change in HAs to determine the number and qualifications of personnel based on task requirements and applicable regulatory requirements. The analysis should addresses personnel requirements for all conditions in which the HA may be performed. The staffing and qualification review should be conducted in accordance with the review criteria of NUREG-1764. 6. Probabilistic Risk and Human Reliability Analysis. For risk-informed submittals, the applicant should (1) update the PRA model to reflect system, component, and HA changes that are					

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	necessary based on the proposed modification or HAs; (2) perform an analysis of the potential effects of the proposed changes upon plant safety and reliability, in a manner consistent with current, accepted PRA/HRA principles and practices, and (3) use the risk insights derived from the results in the selection of HAs and the development of procedures, HSI component lists, and training in order to limit risk and the likelihood of personnel error and to provide for error detection and recovery capability. The staff's HRA review should be conducted in accordance with the review criteria of NUREG-1764. 7. Human-System Interface Design. The applicant should translate function and task requirements into the detailed HSI design through the systematic application of HFE principles and criteria. The applicant's HIS design should be evaluated in accordance with the review criteria of NUREG-1764. The staff's review should address the design of temporary and permanent modifications to the HSI, including new HIS components and the modification of existing ones, for the proposed changes in the HAs. Where changes in HAs result in modifications to large portions of the HSI or in the use of HSI technologies that do not have proven operating histories, the review may also examine the HIS design process using the review criteria of NUREG-0711, Rev. 1. The review addresses aspects of the HSI and the work environment that affect the ability of the personnel to perform the HAs. The final design should be reviewed in accordance with the review criteria of NUREG-0700, as applicable. 8. Procedure Design. The applicant should modify applicable plant procedures and, where needed, provide guidance for the successful completion of the HAs. The procedures should adequately reflect changes in plant equipment and HAs. In the procedure development process, the applicant should apply HFE principles and criteria along with all other design requirements to develop procedure modifications that are technically accurate,					

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	comprehensive, explicit, easy to use, and validated. The applicant's procedure design should be evaluated in accordance with the review criteria of NUREG-1764. 9. Training Program Design. The applicant should develop and conduct adequate training for the HAs, including any changes in qualifications, as described in NRC Information Notice 97-78, “Crediting of Operation Actions In Place of Automatic Actions and Modification of Operator Actions, Including Response Times.” The training program should include all licensed and non-licensed personnel who perform the changed HAs. The applicant's training program should be evaluated in accordance with the review criteria of NUREG-1764. 10. Human Factors Verification and Validation. The applicant should conduct V&V evaluations to (1) provide assurance that the HFE/HSI design provides all necessary alarms, displays, and controls to support plant personnel tasks (HSI task support verification); (2) provide assurance that the HFE/HSI design conforms to HFE principles, guidelines, and standards (HFE design verification); (3) provide adequate assurance that the HFE/HSI design can be effectively operated by personnel within all performance requirements applicable to the HA (integrated system validation); and (4) provide adequate assurance that the final product as built conforms to the verified and validated design that resulted from the HFE design process (final plant HFE/HSI design verification). The applicant's V&V should be evaluated in accordance with the review criteria of NUREG-1764. 11. Human Performance Monitoring Strategy. The applicant should have a human performance monitoring strategy to verify that no adverse safety degradation occurs because of the changes that are made, to provide adequate assurance that the conclusions that have been drawn from the evaluation remain					

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	valid over time, and to provide adequate assurance that personnel have maintained the skills necessary to accomplish the assumed actions. The applicant's human performance monitoring strategy should be evaluated in accordance with the review criteria of NUREG-1764. C.2.2 Level II HFE Review HAs in the medium-risk category should be reviewed using the Level II review criteria provided below. 1. General Deterministic Review Criteria. The applicant should provide adequate assurance that deterministic aspects of design, as discussed in RG 1.174, have been appropriately addressed. The staff should evaluate the deterministic aspects of the design, including that the change meets current regulations and does not compromise defense-in-depth, in accordance with the review criteria of NUREG-1764. 2. Analysis. The applicant should analyze the changes to the HA in terms of OER, functional and task analysis, and staffing and qualifications, and should identify HFE inputs for any modifications to the HSI, procedures, and training that may be necessary. The applicant's HFE analyses should be evaluated in accordance with the review criteria of NUREG-1764. 3. Design of HSIs, Procedures, and Training. The applicant should support the HA by appropriate modifications to the HSI, procedures, and training. The applicant's HSIs, procedures, and training design should be evaluated in accordance with the review criteria of NUREG-1764. Design modifications to the HSI should be reviewed in accordance with the review criteria of NUREG-0700.					

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	4. Human Action Verification. The applicant should verify that the HA can be successfully accomplished with the modified HSI, procedures, and training. The applicant's verification should be evaluated in accordance with the review criteria of NUREG-1764. C.2.3 Level III HFE Review For an HA classified in third level, the staff review should verify that the action is, in fact, in Level III. Verification is accomplished by reviewing the licensee's analysis methods that show the placement of the action in that level. Typically no detailed HFE review is necessary. However, the staff may specify specific areas for review based on the results of the risk-screening process.					
	CHAPTER 19, Severe Accidents					
19.0 Rev. 2	Probabilistic Risk Assessment and Severe Accident Evaluation					
19.0.1	NRC Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 FR 32138, August 8, 1985.					
19.0.2	NRC Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants," 51FR 28044, August 4, 1986.					
19.0.3	NRC Policy Statement, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987.					
19.0.4	NRC Policy Statement, "Regulation of Advanced Nuclear Power Plants," 59 FR 35461, July 12, 1994.					
19.0.5	NRC Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," 60 FR 42622, August 16, 1995.					
19.0.6	SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory					

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	Requirements," ADAMS Accession No. ML003707849, January 12, 1990, and the related staff requirements memorandum (SRM), ADAMS Accession No. ML003707885, June 26, 1990.					
19.0.7	SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," ADAMS Accession No. ML003708021, April 2, 1993, and the related SRM, ADAMS Accession No. ML003708056, July 21, 1993.					
19.0.8	SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708224, June 12, 1996, and the related SRM, ADAMS Accession No. ML003708192, January 15, 1997.					
19.0.9	SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708316, February 18, 1997, and the related SRM, ADAMS Accession No. ML003708232, June 30, 1997.					
	The first five NRC policy statements provide guidance regarding the appropriate course of action to address severe accidents and the use of PRA. The Commission SRMs relating to SECY-90-016, SECY-93-087, SECY-96-128, and SECY-97-044 provide Commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur. For the first aspect of the review, the staff's acceptance criteria consists of a determination that the applicant has adequately demonstrated that the design properly balances preventive and mitigative features and represents a reduction in risk when compared to existing operating plants. For the second aspect of the review, the staff should ensure that					

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	the applicant has used the PRA results and insights, including those from uncertainty analyses, importance analyses, and sensitivity studies, in an integrated fashion to identify and establish specifications and performance objectives (e.g., ITAAC, technical specifications, RAP, RTNSS, and COL action items) for the design, construction, testing, inspection, and operation of the plant. The specific programs establish the staff's acceptance criteria. For example, Section C.I.17.4 of Regulatory Guide 1.206 presents the RAP submittal guidance and SRP Section 17.4 gives the associated staff review guidance, including acceptance criteria. For designs that have evolved from current plant technology through the incorporation of several features intended to make the plant safer, more available, and easier to operate, the results of the PRA should indicate that the design represents a reduction in risk compared to existing operating plants. The staff review should include a broad (qualitative and quantitative) comparison of risks, by initiating event category, between the proposed design and existing operating plant designs (from which the proposed design evolved) to identify the major design features that contribute to the reduced risk of the proposed design compared to existing plant designs (e.g., passive systems, less reliance on offsite and onsite power for accident mitigation, and divisional separation). The staff review should also consider the impact of data uncertainties on the risk estimates. The uncertainty analysis should identify major contributors to the uncertainty associated with the estimated risks. In addition, the staff review should address the applicant's risk importance studies that are performed at the system, train, and component level to provide insights about (1) the systems that contribute the most in achieving the low risk level assessed in the PRA, (2) events (e.g., component					

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	failures or human errors) that contribute the most to decreases in the built-in plant safety level, and (3) events that contribute the most to the assessed risk. The staff should also review the applicant's sensitivity studies performed to gain insights about the impact of uncertainties (and the potential lack of detailed models) on the estimated risk. The objectives of the sensitivity studies should include (1) determining the sensitivity of the estimated risk to potential biases in numerical values, such as initiating event frequencies, failure probabilities, and equipment unavailabilities, (2) determining the impact of the potential lack of modeling details on the estimated risk, and (3) determining the sensitivity of the estimated risk to previously raised issues (e.g., motor-operated valve reliability). For designs using passive safety systems and active defense-in-depth systems, the staff should review the sensitivity studies performed to investigate the impact of uncertainties on the PRA results under the assumption of plant operation without credit for the nonsafety-related defense-in-depth systems. These studies provide additional insights about the risk importance of the defense-in-depth systems that are taken into account in selecting nonsafety-related systems for regulatory treatment according to the RTNSS process. To have confidence that the applicant's PRA and severe accident evaluation results and insights are adequate, the PRA staff must also determine that the scope, level of detail, and technical adequacy of the design-specific and plant-specific PRA are appropriate for the DC and COL, respectively, and any identified uses and risk-informed applications, as follows: 1. The applicant's analyses should be comprehensive in scope, and address all applicable internal and external events and all plant operating modes. Since some aspects					

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	of the applicant's approach may involve non-PRA techniques to address specific events (e.g., PRA-based seismic margins), the PRA staff review should ensure that the scope of the applicant's analyses is appropriate for their identified uses and applications, which may involve a scope, level of detail, and/or technical adequacy for the affected areas that is greater than that needed for a COL application. 2. The level of detail of the applicant's PRA should be commensurate with the identified uses and applications of the PRA (e.g., sufficient to gain risk-informed insights and use such insights, in conjunction with assumptions made in the PRA, to identify and support requirements important to the design and plant operation). The PRA should reasonably reflect the actual plant design, construction, operational practices, and relevant operational experience of the applicant and the industry. The burden is on the applicant to justify that the PRA approach, methods, and data, as well as the requisite level of detail necessary for the NRC staff's review and assessment, are appropriate. Regulatory Guides 1.174 and 1.200 provide additional guidance on the level of detail that should be included in the PRA. If detailed design information (e.g., regarding cable and pipe routing) is not available or if it can be shown that detailed modeling does not provide significant additional information, it is acceptable to make bounding-type assumptions consistent with the guidelines in Regulatory Guide 1.200. However, the risk models should still be able to identify vulnerabilities as well as design and operational requirements such as ITAAC and COL action items. In addition, the bounding assumptions should not mask any risk-significant information about the design and its operation. 3. Consistent with the guidance in Section 2.5 of Regulatory					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Guide 1.174 regarding QA, the staff expects that the applicant will have subjected its PRA to quality control. The following methods are acceptable to the NRC staff to ensure that the pertinent QA requirements of Appendix B to 10 CFR Part 50 are met and that the PRA is sufficient: A. Use of personnel qualified for the analysis B. Use of procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses C. Documentation and maintenance of records, including archival documentation as well as submittal documentation D. Use of procedures that ensure that appropriate attention and corrective actions are taken if assumptions, analyses, or information used previously are changed or determined to be in error Toward this end, the applicant's PRA submittal should be consistent with prevailing PRA standards, guidance, and good practices as needed to support its uses and applications and as endorsed by the NRC (e.g., Regulatory Guide 1.200 and SRP Section 19.1).[4] In addressing the technical adequacy of the PRA, the applicant should include (1) a discussion of prior NRC staff review of the PRA (e.g., during the DC process), findings (i.e., facts and observations) from that review, disposition of those findings, and the relevance of that review to the technical adequacy of the current plant-specific PRA, (2) a discussion of the scope, level of detail, and technical					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	adequacy needed to support the specific uses and risk-informed applications, (3) a discussion regarding the method used for determination of technical adequacy for pertinent PRA scope areas for which the NRC has not endorsed PRA standards (i.e., identify the guidance and good practices documents relied upon to determine the technical adequacy of the PRA), (4) a discussion on the use of and criteria for independent peer reviews, and (5) a discussion on the process for dispositioning independent peer review findings and maintaining or upgrading the PRA, as appropriate, to ensure that it reasonably reflects the as-designed, as-built, and as-operated plant, including the corrective action and feedback mechanisms involving the periodic evaluation of the PRA, consistent with its uses and risk-informed applications, on the basis of actual plant-specific equipment, train, and system performance and relevant industry operational experience. As noted in Element 1.1 of Table A-1 in Appendix A to Regulatory Guide 1.200, special emphasis should be placed on PRA modeling of novel and passive features in the design, as well as addressing issues related to those features, such as digital instrumentation and control, explosive (squib) valves, and the issue of T-H uncertainties.5 The staff should confirm that the assumptions made in the applicant's PRA during design development/certification, in which a specific site may not have been identified or all aspects of the design (e.g., balance of plant) may not have been fully developed, are identified in the DC application and either remain valid or are adequately addressed within the COL application. In addition, a DC and COL applicant may request NRC approval					

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	to implement one or more risk-informed applications. The applicant's submission and staff review of these risk-informed applications follow specific regulatory guidance, approved topical reports, and SRP sections. For example, if an applicant requests to implement a risk-informed inservice inspection program concurrent with its COL application, the application should address the guidance provided in Regulatory Guide 1.178, following a specific methodology contained in identified approved topical reports and approved industry code cases, and SRP Section 3.9.8 will guide the staff review of this program. Chapter 19 of the applicant's FSAR should identify this risk-informed application, with a cross-reference to Section 3.9.8 of the applicant's FSAR, which should describe the applicant's risk-informed inservice inspection program.					
	NOTE: 4. The applicant's adherence to the recommendations provided in Regulatory Guides 1.174 and 1.200 pertaining to quality and technical adequacy will result in a more efficient and consistent NRC staff review process. Alternatively, the applicant may identify, and justify the acceptability of, alternative measures for addressing PRA quality and technical adequacy, and the staff should specifically review the acceptability of these alternative measures in the context of the specific uses and applications of the PRA.					
19.1 Rev. 2	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed					
	In order for the NRC staff to conclude that a PRA is of sufficient technical adequacy to support an application, the staff needs to be assured that (1) the parts of the PRA needed to support the application have been appropriately identified and (2) those parts have been performed in a manner consistent with current good					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	PRA practice. The former needs to be addressed as part of the assessment of the application. The latter can be met by determining that the necessary parts of the PRA have been performed in accordance with the staff position on consensus PRA standards or industry programs as documented in the appendices to Regulatory Guide 1.200. Where there are differences in approach to performing a specific part, the staff must determine that the approach used by the applicant is either equivalent to, or better than, that supported by the staff position.					
19.2	Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance					
	To evaluate licensee-initiated LB changes that are consistent with currently approved staff positions (e.g., regulatory guides, standard review plans, or branch technical positions), the staff normally uses traditional engineering analyses. Licensees generally would not be expected to submit risk information in support of such proposed changes. However, circumstances may arise in which new information reveals an unforeseen hazard or a substantially greater potential for a known hazard to occur, even when all regulatory requirements are met. In such situations, the NRC has the statutory authority to require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. The use of risk information in the review of such license amendment requests is addressed in Appendix D of this SRP section. To evaluate licensee-initiated LB changes that go beyond current staff positions, the reviewers may use traditional engineering analyses as well as the risk-informed approach set forth in this SRP section. In such instances, licensees may be requested to submit supplemental risk information or traditional engineering information if such information is not already included as part of the original submittals. If risk information on the proposed LB					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	changes is not provided, the reviewers will determine if the application can be approved on the basis of the information provided using traditional methods and will either approve or reject the application based upon this information. For those licensee-initiated LB changes that a licensee chooses (or is requested by the staff) to support with risk information, this SRP section describes the scope and content of the staff's review by considering engineering issues and applying risk insights. Licensees submitting risk information to support changes to their LB (whether on their own initiative or at the request of the staff) should address each of the principles of risk-informed regulation discussed in Regulatory Guide 1.174. The reviewers should then determine if the licensees' selected approaches and methods (whether quantitative or qualitative, and traditional or probabilistic), data, and criteria for considering risk are appropriate for the decision to be made. For each risk-informed application, reviewers should ensure that the proposed changes meet the following principles. (Subsections of this SRP section dealing with review guidance for each principle are identified in brackets.) 1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption, i.e., a "specific exemption" under 10 CFR 50.12. [Subsection III.2.1]. 2. The proposed change is consistent with the defense-in-depth philosophy. [Subsection III.2.1] 3. The proposed change maintains sufficient safety margins. [Subsection III.2.1] 4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety					

Table A1-15: NUREG-0800, Standard Review Plan

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Goal Policy Statement (60 FR 42622). [Subsections III.2.2 and III.2.3] 5. The impact of the proposed change should be monitored using performance measurement strategies. [Subsection III.3] In demonstrating adherence to the above principles, reviewers should ensure that licensees address the following issues as part of their submittals:					
II.1	All safety impacts of the proposed change are evaluated in an integrated manner as part of an overall risk management approach in which the licensee is using risk analysis to improve operational and engineering decisions broadly by identifying and taking advantage of opportunities to reduce risk, and not just to eliminate requirements the licensee sees as desirable. For those cases when risk increases are proposed, the benefits should be described and should be commensurate with the proposed risk increases. The approach used to identify changes in requirements was used to identify areas where requirements should be increased as well as where they could be reduced. [Subsection III.2.3]					
II.2	The scope, level of detail, and quality of the engineering analyses (including traditional and probabilistic analyses) conducted to justify the proposed LB change are appropriate for the nature and scope of the change and are based on the as-built, as-operated, and maintained plant, including reflecting operating experience at the plant. [Subsection III.2.2]					
II.3	The plant-specific PRA supporting the licensee's proposals has been subjected to quality controls such as an independent peer review or certification. [Subsection III.2.2]					
II.4	Appropriate consideration of uncertainty is given in analyses and interpretation of findings, including using a program of monitoring, feedback, and corrective action to address significant					

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	uncertainties. [Subsections III.2.2 and III.3]					
II.5	The use of core damage frequency (CDF) and large early release frequency (LERF) as bases for probabilistic risk assessment guidelines is an acceptable approach to addressing Principle 4. Use of the Commission's Safety Goal quantitative health objectives (QHOs) in lieu of LERF is acceptable in principle and licensees may propose their use. However, in practice, implementing such an approach would require an extension to a Level 3 PRA, in which case the methods and assumptions used in the Level 3 analysis, and associated uncertainties, would require additional attention. [Subsection III.2.2]					
II.6	Increases in estimated CDF and LERF resulting from proposed LB changes will be limited to small increments. The cumulative effect of such changes should be tracked and considered in the decision process. [Subsection III.2.2]					
II.7	The acceptability of the proposed changes should be evaluated by the licensee in an integrated fashion that ensures that all principles are met. [Subsection III.2.3]					
II.8	Data, methods, and assessment criteria used to support regulatory decisionmaking must be well documented and available for public review. [Subsection III.4]					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
1.0 (Draft Rev. 1, July, 2007)	Introduction to the Environmental Impact Statement					Exclude, Administrative done by NRC
1.1 (Draft Rev. 0, March, 2000)	The Proposed Project					Exclude, Administrative
1.1 (Draft Rev. 0, March, 2000)	The Proposed Project					
1.2 (Draft Rev. 0, March 2000)	Status of Reviews, Approvals, and Consultations					Exclude, Administrative
2.1 (Draft Rev. 0, March 2000)	Station Location					Exclude, not reactor design related
2.2 (Draft Rev. 0, March 2000)	Land					Exclude, not reactor design related
2.2.1 (Draft Rev. 0, March 2000)	The Site and Vicinity					Exclude, not reactor design related
2.2.2 (Draft Rev. 0, March 2000)	Transmission Corridors and Offsite Areas					
	Acceptance criteria for the review of land use in transmission line corridors, access corridors, and other offsite areas that will be modified for the sole purpose of supporting construction or operation of the proposed project are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.71(d) with respect to analysis requirements to be included in environmental impact statements (EISs) prepared by NRC.					
	The ER must comply with the relevant requirements of 10 CFR					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	51, Appendix A(7), with respect to discussion in EISs prepared by NRC of possible conflicts between alternatives and the objectives of applicable land-use plans.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the relevant requirements of Chapter 2.1 of NRC Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), which sets out the land-use information requirements for inclusion in an applicant's ER.					
	The ER must comply with the relevant requirements of Rev. 2 to NRC Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998), which sets forth the land-use and aesthetic considerations related to site suitability.					
	The reviewer's analysis of land-use characteristics should be closely linked with the impact assessment review described in ESRP Chapters 4.0 and 5.0 (e.g., 4.1.2 and 5.1.2) to establish the land-use characteristics most likely to be affected. With this in mind, the reviewer should take the following steps: (1) Identify the present land use within the transmission corridors, access corridors, and offsite areas according to categories defined in USGS (1997). • Base the level of detail in selecting land-use categories on the needs of subsequent assessments. • Identify total area by land-use categories. • Compare the land use of the corridors that would be changed in ESRP Chapters 4.0 and 5.0 with land use within the region as described in ESRP 2.2.3.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(2) Identify the following characteristics of the transmission corridors, access corridors, and offsite areas: • waterways, highways, roads, railroads, airports, and airplane flight paths • natural gas, electrical transmission lines, communication lines, and other utilities • golf courses and picnic, swimming, fishing, boating, and other recreational areas • residential areas and industrial or commercial facilities • Federal, State, regional, local, and Native American tribal land-use plans • special land-use classifications.					
2.2.3 (Draft Rev. 0, March 2000)	The Region					
	Acceptance criteria for the review of land use in the region are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to analysis requirements to be included in environmental impact statements (EISs) prepared by NRC.					
	The ER must comply with the requirements of 10 CFR 51, Appendix A(7), with respect to discussion in EISs prepared by NRC of possible conflicts between alternatives and the objectives of applicable land-use plans.					
	The reviewer's analysis of land-use characteristics should be closely linked with the impact assessment review described in ESRPs Chapters 4.0 and 5.0 to establish the land-use characteristics most likely to be affected by the proposed project. With this in mind, the reviewer should take the following steps: (1) Identify present land use within the region according to the					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	categories defined by the USGS (1997): • Determine the level of detail used in selecting land-use categories in consultation with the reviewers for construction and operational impacts on land use and socioeconomics and with the reviewer for radiological impacts. • Provide land-use categories for the entire region. • Include all land-use categories used by the reviewer of ESRP 2.2.1. (2) Identify the following characteristics of the region: • major waterways, highways, roads, railroads, airports, and other transportation routes within the region. Of particular interest are those routes that would be used during construction or operation of the proposed project and routes that could be affected by construction or operational activities. • electric-transmission corridors and other utility rights-of-way (e.g., natural gas line corridors) within the region • principal agricultural products, crop areas, and average annual yields • special land-use classifications within the region (e.g., Native American or military reservations, wild and scenic rivers, State and national parks, national forests, designated coastal-zone areas, wildlife refuges, and wilderness areas) • Federal, State, regional, local, and Native American tribal land-use plans.					
2.3 (Draft Rev. 0, March 2000)	Water					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	The introductory paragraph prepared under this ESRP should be consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
2.3.1 (Draft Rev. 0, March 2000)	Hydrology					
	Acceptance criteria for the review of the hydrology at the proposed plant site are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction Activities.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to the National Pollutant Discharge Elimination System (NPDES) permit conditions for discharges, including stormwater discharges.					
	The ER must comply with the requirements of 40 CFR 124 with respect to the NPDES process.					
	The ER must comply with the requirements of 40 CFR 227 with respect to criteria for evaluating environmental impacts.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole source aquifer.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the requirements of State and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	The ER must comply with the requirements of Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the United States Supreme Court granted the States additional authority to limit hydrological alterations beyond the State's role in regulating water rights.					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs including hydrology, water-use, and water-quality issues.					
	The reviewer's analysis of hydrology will be closely linked with the environmental reviews described by ESRP Chapters 3.0, 4.0, 5.0, and 6.0 to establish the hydrological characteristics that are most likely to be affected and the adequacy of the related monitoring programs. The reviewer should take the following steps: (1) Identify the monthly and annual ranges and averages, and the historical extremes of the physical and hydrological characteristics of the hydrosphere potentially affecting or affected by plant construction and operation. (2) Adjust the historical data to present or known future conditions (e.g., reservoirs built and operated during the period of record, scheduled construction of dams). (3) Develop data or take measurements using acceptable hydrological techniques if observations are incomplete or unavailable. (4) Determine if the site or any plant-related structure or alteration of the natural topography is on a floodplain or wetland. (5) Use river-basin commissions, State agencies, and Federal agencies, such as the Corps of Engineers and the U.S. Geological Survey (USGS), as possible sources for site-specific data, including the following: • comprehensive framework studies of water and related lands by river basin planning organizations and regional					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	interagency committees • reports and data from Federal agencies, including the USGS, Bureau of Reclamation, Natural Resources Conservation Service, Forest Service, Agricultural Research Service, Weather Service, Coast and Geodetic Survey, National Oceanic and Atmospheric Administration (NOAA), Coast Guard, National Marine Fisheries Service, U.S. Fish and Wildlife Service, and the Federal Highway Administration • reports and data by regional power administrations such as the Bonneville Power Administration and Tennessee Valley Authority • STOrage and RETrieval System for Water and Biological Data (STORET) water-quality data for specified geographic area, time period, and water-quality constituents from the EPA • State 303(d) list • well logs from water well drillers • reports and data from State agencies, including ecology, conservation, public health, fish and game, forestry, agriculture, water resources, State lands, State engineer, and highway departments and special natural resources commissions (names and functions vary from State to State), and from Native American tribes • standard handbooks (Maidment 1992; Linsley, Kohler, and Paulhus 1982; Mays 1996). The depth and extent of the input to the environmental impact statement (EIS) will be governed by the hydrological resources that could affect or be affected by plant construction or operation and by the nature and magnitude of the expected impacts. With this in mind, the reviewer should take the following steps:					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(1) Ensure that • data are sufficient to provide quantitative information on the hydrological resources potentially affecting or affected by plant construction and operation • Federal, State, regional, local, and affected Native American tribal agencies appropriate to the objectives of this environmental review have been consulted • sufficient data are provided for the assessment of anticipated impacts during the period of plant operation. (2) Where necessary, evaluate the collection of additional data and the substantiation of methodology used to estimate hydrological parameters. (3) Assess the hydrological descriptions with respect to relevancy, completeness, reliability, and accuracy of input to the impact assessments of other sections. (4) Verify that the measurements and data development programs use accepted hydrological practice (which includes those identified in the references listed in this ESRP).					
2.3.2 (Draft Rev. 0, March 2000)	Water Use					
	Acceptance criteria for the review of water use are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction Activities.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to National Pollutant Discharge Elimination System (NPDES) permit conditions for discharges including storm water discharges.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole-source aquifer.					
	The ER must comply with the requirements of Federal, State, regional, local, and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental-quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) should establish its own impact determination.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the United States Supreme Court granted the States additional authority to limit hydrological alterations beyond the State's role in regulating water rights.					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water-use, and water-quality issues.					
	The reviewer's analysis of surface-water and groundwater use should consider the aspects of water use that are concerned with consumptive use, nonconsumptive use, and effluent pathways. The depth of analysis will be related to the importance of water use and proximity of the use to the plant. With this in mind, the reviewer should take the following steps: (1) Identify consumptive water uses that could affect the water supply of the plant or that may be adversely affected by the plant, including the following important characteristics: • water source • locations of diversions and returns • amount and time variation of use • water rights. (2) Identify recreational, navigational, and other nonconsumptive water uses, including those that could be affected by transmission line and offsite area construction and operation. The important characteristics to be quantified are					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• location • activity • amount and time variation of use. (3) Identify the water uses that provide potential pathways for both radiological and nonradiological effluents, including the following important characteristics: • water sources • location of diversions for consumptive uses • location of receptors for nonconsumptive uses • amount and time variation of use for each. (4) In addition to information obtained from the applicant's ER and from responses to subsequent questions to the applicant, use additional sources of data, such as • local water supply companies or agencies • river basin commissions • State agencies (e.g., water resources, fish and wildlife) • various Federal agencies, such as the Corps of Engineers and the U.S. Geological Survey, and Native American tribal agencies when needed to complete the analysis. Local water users may be questioned during the site visit. (5) Using the above information, compile and tabulate water uses by the categories and characteristics described in this ESRP section, but limit the analysis to consideration of present and known future water uses. (6) Ensure that water-use data and information are adequate to					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	serve as a basis for assessing the impacts of proposed project construction and operation on consumptive and nonconsumptive water uses. (a) In evaluating the adequacy of this material, the reviewer should ensure that data are • sufficient to provide quantitative information on water-use characteristics to be impacted by construction and operation • are adequate to predict water-use impacts to the plant during construction and operation. (b) Consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies in making this evaluation.					
2.3.3 (Draft Rev. 0, March 2000)	Water Quality					
	Acceptance criteria for the review of water quality in water bodies affected by the proposed project are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction Activities.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122-133					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	with respect to the National Pollutant Discharge Elimination System (NPDES) permit conditions for discharges including storm-water discharges.					
	The ER must comply with the requirements of 40 CFR 147 with respect to restrictions on waste disposal options.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole source aquifer.					
	The ER must comply with the requirements of 40 CFR 165 with respect to the disposal and storage of pesticides and pesticide containers.					
	The ER must comply with the requirements of 40 CFR 227 with respect to criteria for evaluating environmental impacts.					
	The ER must comply with the requirements of 40 CFR 403 with respect to waste effluents.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of 40 CFR 700-716 with respect to practices and procedures for managing toxic chemicals.					
	The ER must comply with the requirements of State and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action,					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC should consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (to the degree possible in conjunction with the permitting authority and other agencies having relevant expertise) should establish its own impact determination.					
	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S. Supreme Court granted the States additional authority to limit hydrological alterations beyond the State's role in regulating water rights.					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water-use, and water-quality issues.					
	The reviewer's analysis of water quality should be closely linked with the reviews described in the Review Interfaces section of this ESRP to ensure that the physical, chemical, and biological water-quality parameters that could affect or be affected by plant construction or operation have been described. With this in mind, the reviewer should take the following steps:					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(1) Identify the location and spatial distribution of the physical, chemical, and biological characteristics, the monthly and annual ranges, and the historical extremes of those water-quality characteristics that could potentially affect or be affected by plant construction or operation. • Adjust the data for present day conditions. • If historical observations are incomplete or unavailable for the locations of concern, obtain these data through consultation with the applicant or with appropriate resource agencies. (2) Determine the presence of environmental stresses related to existing water quality. • Determine stresses on the bases of the quality criteria requirements of other water users, as indicated by the approved water-use classification (such as 303(d) lists) or water-resource planning documents for the water body in question. • As part of the determination, consult the historical literature addressing water-quality issues for the water body in question. (3) When applicable, discuss the water-quality conditions, water rights, and agreements as they affect water-quality and water-resource plans for the site and vicinity with Federal, State, regional, local, and affected Native American tribal water resource and pollution control and monitoring agencies. (4) Obtain the information primarily from the applicant's • ER					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• responses to questions to the applicant • consultation with Federal, State, regional, local, and affected Native American tribal agencies. Use sources of data, such as river basin planning organizations, and State and Federal agencies, such as the EPA, the U.S. Army Corps of Engineers, and the U.S. Geological Survey, if additional information or verification is deemed necessary. (5) Ensure that • data are sufficient to provide quantitative information on the physical, chemical, and biological water-quality characteristics potentially affecting or affected by plant construction or operation • the water-quality descriptions are sufficient, with respect to relevancy, completeness, reliability, and accuracy for input to the impact assessments of other sections • Federal, State, regional, local, and affected Native American tribal agencies appropriate to the objectives of this environmental review have been consulted. (6) When evaluating the adequacy of this material, • consult the applicable standards and guides for this environmental review and use the site visit and/or consultations to permitting agencies to evaluate the completeness of the water-quality descriptions • evaluate, when necessary, the collection of additional data, the verification of data, and the substantiation of the methodology used to estimate water-quality parameters.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(7) Include the appropriate depth and extent of the input to the environmental impact statement (EIS) as governed by the water-quality characteristics that could affect or be affected by plant construction or operation and by the nature and magnitude of the expected impacts. The following information should be included as input to the EIS: • descriptions of site and vicinity surface-water and groundwater quality that could affect or be affected by plant construction and operation. The description may consist of statistical summaries of the water-quality characteristics, including mean, mean low and high, and historical low and high values (as available) for the site and vicinity. The data included should be commensurate with the anticipated impacts. Figures may be used to show long-term and seasonal trends, such as variations in dissolved oxygen and nutrient concentrations and pH variations. • a description of the water-quality-related environmental stresses in the site and vicinity.					
2.4 (Draft Rev. 0, March 2000)	Ecology		Exclude			Exclude, Administrative
2.4.1 (Draft Rev. 0, March 2000)	Terrestrial Ecology					
	Acceptance criteria for the review of terrestrial ecology on and in the vicinity of the site and transmission corridors are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to descriptions of the environment affected by the issuance of a construction permit.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the requirements of 10 CFR 52, Subpart A, with respect to descriptions of the environment affected by the issuance of an early site permit.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the preparation of supplemental environmental impact statements (EISs) in support of the issuance of an operating license.					
	The ER must comply with the requirements of Bald and Golden Eagle Protection Act with respect to the prohibition of taking, possessing, selling, transporting, importing, or exporting the bald or golden eagle, dead or alive, without a permit.					
	The ER must comply with the requirements of Endangered Species Act of 1973 with respect to identifying threatened and endangered species, critical habitats, formal or informal consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act of 1958 with respect to consideration of fish and wildlife resources in the planning of development projects that affect water resources.					
	The ER must comply with the requirements of Migratory Bird Treaty Act with respect to declaring that it is unlawful to take, import, export, possess, buy, sell, purchase, or barter any migratory bird. Feathers or other parts of nests and eggs, and products made from migratory birds are also covered by the Act. "Take" is defined as pursuing, hunting, shooting, poisoning, wounding, killing, capturing, trapping, or collecting.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998), contains guidance concerning the ecological systems and biota at potential sites and their environs should be sufficiently well-known to allow reasonably certain predictions that there would be no unacceptable or unnecessary deleterious impacts on populations of important species or on ecological systems with which they are associated from the construction or operation of a nuclear power station at the site. The reviewer should ensure that the applicant's description of the site and transmission corridors identifies important species or ecological systems that could potentially be impacted by station and transmission corridor construction or operation.					
	The ER must comply with the requirements of Regulatory Guide 4.11, Rev. 1, Terrestrial Environmental Studies for Nuclear Power Stations (NRC 1977), contains technical information for the design and execution of terrestrial environmental studies, the results of which may be appropriate for inclusion in the applicant's ER. The reviewer should ensure that the appropriate results are included in the ER.					
	The reviewer should ensure that the ecological information is adequate to serve as a basis for assessment of the impacts of design and siting of the plant, and plant construction and operation. In evaluating the adequacy of the description of terrestrial resources of the site and offsite areas, the reviewer should consult the applicable acceptance criteria of this ESRP. Within these criteria, the reviewer will find a framework of those descriptive features of terrestrial resources judged adequate for most situations of nuclear power station siting. The reviewer should also become familiar with the provisions of the legislation listed in this ESRP.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	With these guidelines in mind, the reviewer should take the following steps: (1) Identify the species and habitats that will be considered "important" ecological resources of the site, vicinity, transmission corridors, and offsite areas for evaluation of potential impacts on them, using Table 2.4.1-1 as a reference. (2) Consult with local offices of the appropriate Federal, State, regional, local, and affected Native American tribal agencies to determine the possible presence of such species. (3) Identify the threatened and endangered species that, based on known distributions, could be present within these areas, but that have not been recorded by documented observations. (4) In the case of commercially or recreationally valuable species, list the types of wildlife and plants that could be adversely impacted by the proposed action, and in addition to the applicant's ER, consult with State or local agencies or organizations that maintain records of harvest levels of these species. (5) Review the available site-specific data for adequacy, accuracy, and completeness.					
2.4.2 (Draft Rev. 0, March 2000)	Aquatic Ecology					
	Acceptance criteria for the review of aquatic ecology on and in the vicinity of the site and transmission corridors are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to descriptions of the environment affected by the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	issuance of a construction Permit.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the preparation of supplemental environmental impact statements (EISs) in support of the issuance of an operating license.					
	The ER must comply with the requirements of 10 CFR 52, Subpart A, with respect to descriptions of the environment affected by the issuance of an early site permit.					
	The ER must comply with the requirements of Coastal Zone Management Act of 1972 with respect to natural resources, and land or water use of the coastal zone.					
	The ER must comply with the requirements of Endangered Species Act of 1973 with respect to identifying threatened and endangered species, critical habitats, and initiating formal or informal consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service.					
	The ER must comply with the requirements of Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, Amendments of 1972 with respect to restoration and maintenance of the chemical, physical, and biological integrity of water resources.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act of 1958 with respect to consideration of fish and wildlife resources in the planning of development projects that affect water resources.					
	The ER must comply with the requirements of Marine Mammal Protection Act of 1972 with respect to the protection of marine mammals.					
	The ER must comply with the requirements of Marine Protection, Research, and Sanctuaries Act of 1972 with respect					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	to dumping of dredged material into the ocean.					
	The ER must comply with the requirements of Rivers and Harbors Appropriations Act of 1899 with respect to the deposition of debris in navigable waters, or tributaries to such waters.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), details the means by which the applicant collects baseline data used to compare subsequent data to evaluate plant construction and operation impacts. The reviewer should ensure that the applicant's measurement of conditions before site preparation includes all environmental parameters necessary to evaluate impacts during station operation, as well as during site preparation and construction.					
	The ER must comply with the requirements of Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (1998), contains guidance concerning the ecological systems and biota at potential sites and requires that their environs be sufficiently well-known to allow reasonably certain predictions that there would be no unacceptable or unnecessary deleterious impacts on populations of important species or on ecological systems with which they are associated from the construction or operation of a nuclear power station at the site. The reviewer should ensure that the applicant's description of the site and transmission corridors identify important species or ecological systems that could potentially be impacted by station and transmission corridor construction or operation.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The reviewer should ensure that the regional and site-specific aquatic ecological information is adequate to serve as a basis for assessment of the effects of design and siting of the plant, construction, and operation. In assessing the adequacy of the description of aquatic resources of the site and offsite areas, the reviewer should consult the applicable acceptance criteria of this ESRP section. Within these criteria, the reviewer may find a framework of those descriptive features of aquatic resources judged adequate for most situations of nuclear power station siting. The reviewer should also become familiar with the provisions of the legislation listed in the "Acceptance Criteria" section. With these guidelines in mind, the reviewer should take the following steps: (1) Identify the species and habitats that will be considered "important" ecological resources of the site, vicinity, transmission corridors, and offsite areas for evaluation of potential impacts on them, using Table 2.4.2-1 as a reference. (2) Consult with local offices of the appropriate Federal agencies and the appropriate State agencies to verify the possible occurrence of such species. (3) Identify the threatened or endangered species that, based on known distributions, could be present within these areas, but that have not been recorded by documented observations. (4) In the case of commercially or recreationally valuable species, list the types of wildlife and plants that could be adversely impacted by the proposed action, and in addition to the applicant's ER, consult with State or local agencies or					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	organizations that maintain records of harvest levels of these species. (5) Review the available site-specific data for adequacy, accuracy, and completeness.					
2.5 (Draft Rev. 0, March 2000)	Socioeconomics					Exclude, Administrative
2.5.1 (Draft Rev. 0, March 2000)	Demography					
	Acceptance criteria for the review of socioeconomic demographics are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50.34(a)(1) with respect to site acceptance, which is based on the consideration of factors relating to the proposed reactor design and the characteristics peculiar to the site. One of the factors involves population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.					
	The ER must comply with the requirements of 10 CFR 51.45(c) with respect to analysis of socioeconomic data.					
	The ER must comply with the requirements of 10 CFR 51.45(d) and 51.71(d) with respect to the analyses required in the development of the ER and environmental impact statement (EIS). In accordance with 10 CFR 51.45(d), the applicant is required to submit in the ER information needed for evaluating these factors. Similar information is required to be present in the EIS pursuant to 10 CFR 51.71.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the requirements of 10 CFR 52.18 with respect to reviewing applications for early site permits.					
	The ER must comply with the requirements of 10 CFR 52.81 with respect to reviewing applications for combined licenses.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), addresses population distribution within the vicinity of the plant.					
	To analyze the population distribution within an 80-km (50-mi) radius of the proposed site, the reviewer should take the following steps: (1) Prepare population distribution charts that provide population data for both permanent and transient populations as they presently exist and as predicted at the time of plant startup and for 10-year increments reaching 40 years from the latest decennial census; present the data as shown in Table 2.5.1-1. (2) Determine that the data are based on the appropriate geographical coordinates. (3) Review the following: • data used to update the basic decennial census data • the methods used to establish population data within 80 km (50 mi) of the site • the applicant's methods for population projections.					
2.5.2 (Draft Rev. 0, March 2000)	Community Characteristics					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Acceptance criteria for the review of community characteristics are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.45(c) with respect to analysis of socioeconomic data.					
	The ER must comply with the requirements of 10 CFR 51.45(d) and 51.71(d) with respect to the analyses required in the development of the ER and environmental impact statement (EIS). In accordance with 10 CFR 51.45(d), the applicant is required to submit in the ER information needed for evaluating these factors. Similar information is required to be present in the EIS pursuant to 10 CFR 51.71.					
	The ER must comply with the requirements of 10 CFR 51.45 with respect to reviewing applications for early site permits.					
	The ER must comply with the requirements of 10 CFR 52.81 with respect to reviewing applications for combined licenses.					
	The reviewer's analysis of community characteristics should be closely linked with the impact assessment review described by the ESRP Chapters 4.0 and 5.0 to establish the site-specific community characteristics that are most likely to be affected (see Generic Environmental Impact Statement of Nuclear Plants [NRC 1996]). When analyzing the community characteristics, the reviewer should take the following steps: (1) Describe community characteristics for those communities within the region (see the footnote in Areas of Review in ESRP 2.5.2 for definition of "relevant region") that are expected to be impacted. (2) Conduct an initial screening of the community structure and characteristics within an approximate 80-km (50-mi) radius of the site to make a preliminary determination of the potentially					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	affected subregions and communities. • Address the following factors in the screening process to identify population influx: - settlement patterns - labor force - transportation - housing availability - public services - economics. • Discuss the results of the initial screening with the reviewers of ESRP Chapters 4.0 and 5.0 to establish any other predicted construction or operating impacts that might affect results of the screening process. (3) Describe potentially impacted areas of the region and their associated communities in the following terms (the extent and detail of the descriptions should be in proportion to the magnitude of the impacts anticipated and only those terms necessary for subsequent impact evaluation should be used): • political structure • social structure • demography • housing • economic base • social services and public facilities • highways and transportation • water and sewer facilities • education • public safety • health					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• recreation • taxation • land-use planning and zoning.					
2.5.3 (Draft Rev. 0, March 2000)	Historic Properties					
	Acceptance criteria for the review of the historic properties that could be impacted by proposed project construction or operation are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 36 CFR 800 defines the process by which a Federal agency meets the requirements under Sections 106 and 110 of the National Historic Preservation Act (NHPA) to ensure that agency-assisted or -licensed undertakings consider the effects of the undertaking on historic properties included in or eligible for the National Register. Under this regulation, the Federal agency is required to identify and evaluate all historic properties in the project area and take measures to mitigate adverse effects as being significant.					
	The ER must comply with the requirements of 36 CFR 63 contains guidance by which historic properties are evaluated and determined eligible for listing on the National Register.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are provided as follows:					
	The ER must comply with the requirements of Nuclear Reactor Regulation (NRR) Office Letter No. 906, Revision 1 (NRC 1996) contains guidance for complying with the requirements contained in the National Historic Preservation Act. NRR Office Letter No. 906 is revised periodically. Obtain a copy of the latest revision for current guidance.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The reviewer's analysis of historic properties should be closely linked with the impact assessment review described by ESRPs 4.1.3 and 5.1.3 to establish the historical and archaeological characteristics that are most likely to be affected. The reviewer should take the following steps: (1) Contact the appropriate State Historic Preservation Officer (SHPO) to determine if there are any additional comments or information concerning the proposed station site. • Make initial contact by phone and invite the SHPO to participate in the site visit. • If the SHPO has comments or information that add to or amplify that which was provided by the applicant, request that the SHPO forward, by letter to the staff, these additional comments. (2) Contact the Archeology and Ethnography Program (AEP) of the National Park Service (NPS), U.S. Department of Interior. This office is a particularly useful source of expertise in the area of historic and cultural preservation and is staffed with professionals who can assist in the environmental review and in analyzing the results of the applicant's surveys and investigations. (3) In consultation with the SHPO, apply the National Register criteria outlined by the U.S. Department of the Interior (NPS 1990; 1991) to historic properties that are on the station site or that will be directly affected by plant construction. If a property appears to meet the criteria, or if it is questionable whether the criteria are met, the staff should request, in writing, an opinion from the U.S. Department of the Interior with respect to the property's eligibility for inclusion in the National Register. The					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	request for determination of eligibility should be sent directly to the Keeper of the National Register of Historic Places, National Park Service, U.S. Department of the Interior, Washington, D.C. 20013-7127. (4) Have the NPS-AEP staff assist in • defining the requirements of additional surveys and investigations that the staff decides should be completed by the applicant • reviewing the results of these surveys. (5) Consult the National Register to verify the list of National Register properties provided by the applicant. Note: A proposed station can have a visual or noise impact on cultural and historic resources that are located some distance from the proposed station site. Therefore, National Register properties within 16 km (10 mi) of the proposed station site or within 2 km (1.2 mi) of transmission line routes, access corridors, and offsite areas should be identified. (6) Meet with the SHPO and, where appropriate, the State Archaeologist and State Historian, to discuss the information provided to the applicant by the SHPO. The SHPO can alert the staff to relevant State and local laws, orders, ordinances, or regulations aimed at the preservation of cultural resources within the applicant's State. Be sure to discuss the following: • the data necessary for Items 1 through 4 above • a list of additional organizations or individuals that might be able to assist in identifying and locating archaeological and historic resources. Of particular importance are university and Native American tribal					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	archaeological and historical staffs. (7) Contact the SHPO of each affected State for sites located on or near State boundaries, or where transmission line routes, access corridors, or offsite areas pass through more than one State. (8) Compare the information provided by the applicant with that obtained from the SHPO and the National Register and resolve any differences in identification and location of cultural and historic resources.					
2.5.4 (Draft Rev. 1, July, 2007)	Environmental Justice					
	The acceptance criteria for the review of environmental justice information are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to complying with environmental quality standards and requirements that have been imposed by Federal, State, regional, local, and affected Native American tribal agencies.					
	The ER must comply with the requirements of 10 CFR 52.18 with respect to reviewing applications for early site permits.					
	The ER must comply with the requirements of 10 CFR 52.81 with respect to reviewing applications for combined licenses.					
	The ER must comply with the requirements of 10 CFR 100.10 with respect to requirements that the site acceptance be based on the consideration of factors relating to the proposed reactor design and the characteristics peculiar to the site.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	NRC specific policy on treatment of environmental justice matters can be found in "Policy Statement on the Treatment of Environmental Justice Matters in NRC Regulatory and Licensing Actions." Federal Register, 69 FR 52040, August 24, 2004.					
	The ER must comply with the requirements of Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998a), notes that environmental justice is one of the considerations on which site acceptance is based and provides specific information for making the determinations required.					
	The Council on Environmental Quality provides guidance for addressing environmental justice, "Environmental Justice: Guidance Under the National Environmental Policy Act," CEQ Guidance, December 10, 1997 (CEQ 1997). This guidance is not binding on the NRC staff, but should be followed as appropriate.					
	Guidelines for specific information requirements for environmental justice determinations are described in Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Issues, Appendix D to Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-203, NRC Office of Nuclear Rector Regulation, Washington, D.C. (NRC 2004). NRR Office Office Instruction LIC-203 is revised periodically. Obtain the latest revision for current guidance on this subject.					
	The ER must comply with the requirements of Commission Order CLI-02-20. In the Matter of Private Fuel Storage L.L.C. (Independent Spent Fuel Storage Installation). Docket No. 72-22-ISFSI. October 01, 2002 (NRC 2002).					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	The ER must comply with the requirements of Commission Order CLI-98-3. In the Matter of Louisiana Energy Services (Claiborne Enrichment Center). Docket No. 70-3070-ML. April 3, 1998 (NRC 1998b).					
	The reviewer's analysis of minority and low-income populations should be closely linked with the impact-assessment review of environmental issues described by the ESRPs 2.2.1 through 2.5.3, 2.6 through 2.8.6, 4.1.1 through 4.6, 5.1.1 through 5.6, 7.1, and 7.3 to establish the environmental pathways by which minority and low-income households are most likely to be disproportionately affected, if any. For example, the reviewer should take the following steps: • contact the lead staff responsible for reviews of these ESRPs • contact local university departments of economics and sociology. These are particularly useful sources of expertise in the area of environmental justice, particularly those that are state repositories for Bureau of Census data. These offices are staffed with professionals who can assist the reviewer in analyzing the results of the applicant's surveys and investigations and can assist in the environmental review. • contact the cognizant personnel in each affected state, for sites located on or near state boundaries, or where transmission line routes, access corridors, or offsite areas pass through more than one state.					
2.6 (Draft Rev. 0, March 2000)	Geology					
	The potential for geological environmental impacts (e.g., subsidence from cooling pond loading) is small, and the staff's					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	experience has been that actual occurrence of such impacts is infrequent. Further, any such potential would be established and evaluated during the staff's safety evaluation and described in the staff's SER or SSER. On this basis, no environmental review of geology is required, but the reviewer's analysis should consist of the following two steps: (1) Consult with the staff's safety evaluation reviewers to determine if there is any potential for geological environmental impact. (2) When any such impacts can be predicted, notify the reviewers for ESRPs 4.1 and 5.1 to develop, in consultation with the safety reviewers, an analysis and evaluation of the potential impacts.					
2.7 (Draft Rev. 0, March 2000)	Meteorology and Air Quality					
	Acceptance criteria for the review of site meteorology and air quality are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50, Appendix I, with respect to calculation of air doses from gaseous emissions.					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to the reliability of the meteorological and climatological information.					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to compliance with environmental quality standards and requirements that have been imposed by Federal, State, regional, local, and affected Native American tribal agencies.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the requirements of 10 CFR 52.18 with respect to reviewing applications for early site permits.					
	The ER must comply with the requirements of 10 CFR 52.81 with respect to reviewing applications for combined licenses					
	The ER must comply with the requirements of 10 CFR 100.10(c) and 10 CFR 100.20(c) with respect to meteorological conditions at the site and in the surrounding area.					
	The ER must comply with the requirements of 40 CFR 50 with respect to definition of criteria pollutants and National Ambient Air Quality Standards.					
	The ER must comply with the requirements of 40 CFR 51, Subpart W, with respect to requirements related to determination that the proposed Federal action conforms to applicable implementation plans.					
	The ER must comply with the requirements of 40 CFR 51, Appendix W, with respect to air-quality models.					
	The ER must comply with the requirements of 40 CFR 81, Subpart C, with respect to attainment status designations approved by the EPA.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The description of the general climate of the region, including severe weather, should be based on published climatological summaries from nearby representative sites with long periods of record (see references in this ESRP).					
	At least one annual cycle from the onsite meteorological program should be used to relate local meteorological conditions to local and regional climatology. Regulatory Guide 1.23, Onsite Meteorological Programs (NRC 1972), provides					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	guidance related to onsite meteorology programs. ESRP 6.4 sets forth the staff review plan for evaluation of the onsite meteorological program.					
	Atmospheric dispersion models and assumptions described in Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (NRC 1977), should be used for estimating relative atmospheric concentrations and relative deposition used in calculating individual and population doses from routine releases of radioactive effluents to the atmosphere.					
	Atmospheric dispersion models and assumptions described in Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (NRC 1983), should be used for estimating relative atmospheric concentrations and relative deposition used in calculating individual doses from accidental releases of radioactive effluents to the atmosphere.					
	Atmospheric dispersion models and assumptions promulgated by the EPA should be used for air quality assessments.					
	The reviewer's analysis of meteorology should be closely linked with the impact assessment review described by ESRPs 5.3.3.1 and 5.4 to establish the meteorological characteristics that are most likely to be affected. To evaluate the applicant's climatological descriptions and meteorological data, the reviewer should compare them with the climatological data available from the National Climatic Data Center (NCDC) and information in climatological references. These references include					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• standard climatological references, such as Weather and Climate (Koeppe and Delong 1958) and Applied Climatology (Griffiths 1963) that describe the relationship between climate and geography • other climatological texts, such as Boundary Layer Climates (Oke 1978) and The Climate Near the Ground (Geiger, Aron, and Todhunter 1995), that describe local climate variability and climate modifications related to man's activities • climate descriptions for specific regions in the United States that have been prepared by the U.S. Department of Commerce (1968), National Oceanic and Atmospheric Administration and that are found in publications such as Climatic Atlas of the United States, Climates of the States, and Local Climatological Data Annual Summaries with Comparative Data. These publications contain information on meteorological extremes as well as typical conditions. • up-to-date climatological data and summaries that are available electronically from the NCDC through the Geographical Environmental & Siting Information System (GEn&SIS) • severe-weather data related to extreme winds, hurricanes, and tornadoes that have been summarized by Cry (1965), Alaka (1968), Simpson and Lawrence (1971), Changery (1982a, b), Ramsdell and Andrews (1986), and Ramsdell et al. (1987) • more recent severe weather statistics that are available through GEn&SIS and are updated monthly in Storm Data published by the NCDC. To evaluate the applicant's atmospheric transport and dispersion modeling, the reviewer should compare it with the standard					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	dispersion modeling techniques, such as • atmospheric dispersion modeling techniques that are described in detail in texts including Meteorology and Atomic Energy—1968 (Slade 1968), Handbook on Atmospheric Diffusion (Hanna, Briggs, and Hosker Jr. 1982), Atmospheric Science and Power Production (Randerson 1984), and Workbook of Atmospheric Dispersion Estimates: An Introduction to Dispersion Modeling (Turner 1994) • climatological data specifically related to air quality and atmospheric dispersion that are found in the summaries available from NCDC and in journal articles by Hosler (1961 and 1964) and Holzworth (1972).					
	<u>Regional Climatological and Local Meteorological Characteristics</u> When analyzing regional and local meteorological characteristics, the reviewer should take the following steps: (1) Assess the general climatic description of the region for completeness and accuracy. • Evaluate climatic parameters such as air masses, general airflow, pressure patterns, frontal systems, and temperature and humidity conditions reported by the applicant by comparing them with standard references. • Verify the applicant's description of the role of synoptic scale and mesoscale atmospheric processes on local (site) meteorological conditions by comparing it with the descriptions provided in standard references and the reviewer's knowledge of the area. (2) Examine the regional meteorological averages and extremes,					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	including severe weather phenomena and air quality conditions, to establish that the data represent site conditions by comparing • concurrent offsite and onsite data (e.g., monthly averages of wind speed, wind direction frequency, and precipitation, and monthly averages and diurnal variations of temperature and humidity) • offsite data for the concurrent period of onsite data with long-term (about 30 years) offsite data • the locations of the stations with respect to major topographic features and airflow patterns (e.g., valley flow, land-sea (lake) breeze circulations, principal storm tracks). (3) Evaluate the local (site) meteorological parameters and topographic descriptions of the site area to establish that the data represent conditions at the site and its immediate vicinity by examining the location of the onsite meteorological tower (and other local sources of meteorological data) with respect to local topographic characteristics that could impact local airflow patterns (e.g., local circulation conditions such as "drainage flow") and meteorological parameters such as temperature and humidity. (4) Determine if the regional and local meteorological data are appropriate as bases for the applicant's evaluation of potential changes in normal and extreme values, severe weather phenomena, and air quality conditions resulting from station construction and operation. (This information may be cross-referenced from Chapter 5.0 of the applicant's ER.) (5) Analyze the proposed terrain modifications (e.g., removal of trees, leveling of ground, installation of lakes and ponds)					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	resulting from station construction and predict the potential effects of these modifications on local meteorological characteristics with respect to the adequacy of available data considering these modifications. (6) Determine the adequacy of data on regional climatological and local meteorological conditions and phenomena as bases for assessing the effects on design and siting of the station and heat dissipation system and as bases for assessing the impact on the atmospheric environment resulting from station construction and operation. (7) Review regional and local meteorological data for appropriateness as input to predictive models for assessing cooling system impacts on the atmospheric environment by considering the types and frequencies of available meteorological measurements, the elevations at which measurements are made, the selected cooling system design, and the height of effluent release to the atmosphere.					
	<u>Meteorological Input to Individual Dose Assessment</u> When analyzing meteorological input to individual dose assessment, the reviewer should take the following steps: (1) Obtain the following information from the ESRP reviewers listed below: • ESRP 3.5—a description of release point characteristics (i.e., elevation above grade, inside vent or stack diameter, physical shape, flow rate, effluent temperature, exit velocity, release frequency, and duration and type of effluent) for each point of routine release of radioactive effluent to the atmosphere					

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	• ESRP 5.4.1—the locations of the nearest receptors (cow, goat, vegetable garden, residence, and site boundary) in each 22½q sector. (2) Compare the atmospheric transport and diffusion models used by the applicant for calculations of χ/Q and D/Q to transport and diffusion modeling concepts (as described in Regulatory Guide 1.111) applicable to local topographic and meteorological characteristics and to the type and mode of release appropriate to the plant. (3) Examine atmospheric transport and diffusion parameters for applicability to local topographic and meteorological characteristics by considering the experimental bases for these parameters with respect to the local conditions. (4) Compare the meteorological data provided by the applicant for use in the atmospheric transport and diffusion modes for compatibility with the models used and verify the completeness and adequacy of the description of local atmospheric transport and diffusion characteristics (as discussed in Regulatory Guides 1.23 and 1.111). • Evaluate the meteorological data for appropriateness of heights of measurement of wind speed, wind direction, and atmospheric stability. - Winds measured at the 10-m level and temperature difference measurements (as an indicator of atmospheric stability) between the 10-m level and height of the building or vent are acceptable for consideration of ground-level releases. - For releases considered elevated, (1) winds					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	reasonably representative of conditions at the height of release, and (2) temperature difference measurements reasonably representative of the atmospheric layer, into which the effluent will be released, are acceptable. • Examine mixing height data for considerations of restrictions to the vertical spread of the effluent. • Examine precipitation data for considerations of the effects of washout on estimates of atmospheric transport, diffusion, and deposition. (5) Evaluate estimates of relative concentration (including consideration of radioactive decay during transport and depletion of radioiodines and particulates) and relative deposition (including the effects of wet deposition) used by the applicant for assessing the individual doses resulting from routine releases of radioactive effluent to the atmosphere to verify that these estimates are complete and appropriate to local conditions. Depending on the level of confidence in the applicant's model and considering the extent, applicability, and representative nature of the available meteorological data, the reviewer may make an independent analysis of relative concentration and relative deposition values at each receptor using the transport and dispersion models described in Regulatory Guide 1.111.					
	<u>Meteorological Input to Population-Dose Assessment</u> When evaluating meteorological input to population dose assessment, the reviewer should take the following steps: (1) Verify that the release point characteristics are the same as those used for input to the individual dose assessments.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(2) Compare the atmospheric transport and diffusion models used by the applicant for calculations of relative concentration and relative deposition with transport and diffusion modeling concepts (as described in Regulatory Guide 1.111) applicable to regional (i.e., out to a distance of 80 km from the site) modeling. • Give special consideration to topographic and meteorological characteristics (narrow, deep valleys, land sea [lake] breeze regimes, restricted mixing heights, fumigation conditions, and low-level subsidence inversions of temperature) to ensure that they are applicable to the type and mode of releases from the plant. • Examine the atmospheric transport and diffusion parameters for applicability to regional topographic and meteorological characteristics by considering the experimental bases for these parameters with respect to regional conditions. (3) Compare the meteorological data provided by the applicant for use in the atmospheric transport and diffusion models for compatibility with the models used and verify the completeness and adequacy of the description of regional atmospheric transport and diffusion characteristics as discussed in Regulatory Guides 1.23 and 1.111. • Evaluate meteorological data for appropriateness of heights of measurements of wind speed, wind direction, and atmospheric stability. - Winds measured at the 10-m level and temperature difference measurements to indicate atmospheric stability between the 10-m level and height of the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	building or vent are acceptable for consideration of ground-level releases. - For releases considered elevated, winds reasonably representative of conditions at the height of release and reasonable estimates of the temperature of the atmospheric layer into which the effluent will be released are acceptable. • Examine mixing height data for considerations of restrictions to the vertical spread of the effluent. • Examine precipitation data for considerations of the effects of washout on estimates of atmospheric transport and diffusion. (4) Evaluate estimates of relative concentration (including consideration of radioactive decay during transport and depletion of radioiodines and particulates) and relative deposition used by the applicant for an assessment of the population doses resulting from routine releases of radioactive effluent to the atmosphere to verify that these estimates are complete and appropriate to regional conditions. These estimates should encompass all individuals living within 80 km of the facility. Depending on the level of confidence in the applicant's model and considering the extent, applicability, and representativeness of the available meteorological data, the reviewer may independently analyze relative concentration and relative deposition values for 16 directions in segments of 0.8-1.6 km (0.5-1 mi), 1.6-3.2 km (1-2 mi), 3.2-4.8 km (2-3 mi), 4.8-6.4 km (3-4 mi), 6.4-8.0 km (4-5 mi), 8.0-16 km (5-10 mi), 16-32 km (10-20 mi), 32-48 km (20-30 mi), 48-64 km (30-40 mi), and 64-80 km (40-50 mi) using the transport and diffusion models described in Regulatory Guide 1.111.					
	<u>Meteorological Input to Plant-Accident Assessments</u>					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	When analyzing meteorological input to plant accident assessments, the reviewer should take the following steps: (1) Compare the atmospheric transport and diffusion models used by the applicant for calculations of χ/Q and D/Q for accident consequence assessments to state-of-the-art transport and diffusion modeling concepts (as described in Regulatory Guide 1.145) applicable to local topographic and meteorological characteristics and to the type and mode of release appropriate to the plant. For environmental assessment purposes, nominal meteorological conditions are determined rather than the adverse conditions determined for safety assessments. (2) Examine atmospheric transport and diffusion parameters for applicability to local topographic and meteorological characteristics by considering the experimental bases for these parameters with respect to the local conditions. The release point characteristics should be the same as those used for input to the individual dose assessments.					
	<u>Regional and Local Air Quality Characteristics</u> When analyzing regional and local air quality characteristics, the reviewer should take the following steps: (1) Assess the description of the existing regional air quality for completeness and accuracy. (2) Identify the air pollutants for which there are non-attainment or maintenance areas in the region. (3) Determine the emissions expected from plant construction or operation activities, as appropriate. Work force vehicular					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	emissions should be estimated. (4) Evaluate the impact of emissions from plant construction and operation on existing air quality. If the site is within or near a non-attainment or maintenance area, a conformity analysis may be required (see 40 CFR 51, Subpart W). (5) Determine whether appropriate permits have been obtained.					
	<u>Early Site Permit Reviews</u> When conducting a meteorological review of an early site permit (ESP) application, the reviewer should take the following steps: (1) Refer to 10 CFR 52, which specifies the requirements and procedures applicable to the Commission's issuance of early site permits for approval of a site or sites for one or more nuclear power facilities separate from the filing of an application for a construction permit (CP) or combined license (COL). (2) Note that application for an early site permit must include the • Number • type and thermal power levels of the facilities for which the site may be used • boundaries of the site • proposed general location of each facility • maximum radiological and thermal effluents that each facility will produce • types of cooling systems that may be associated with each facility • meteorological characteristics of the proposed site. The scope and level of detail needed for meteorological review					

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	of an ESP application are the same as for review of a CP application under 10 CFR 51, except that the focus of the review is on the effects of construction and operation of a reactor, or reactors, which have characteristics that fall within the postulated site parameters.					
2.8 (Draft Rev. 0, March 2000)	Related Federal Project Activities					
	Acceptance criteria for the review of information on related Federal-project activities and the possible need for one or more cooperating agencies in preparation of the EIS are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 40 CFR 1508.25 and 10 CFR 51.14(b) with respect to the scope of an EIS and consideration of the cumulative impacts of connected, cumulative, and similar actions.					
	The ER must comply with the requirements of 40 CFR 1501.6, 10 CFR 51.10(b)(2), and 10 CFR 51.14 with respect to the possible need for cooperating agencies in the preparation of the EIS.					
	The ER must comply with the requirements of 10 CFR 51.29(a)(7) with respect to the possible need to identify cooperating agencies.					
	Data provided by the applicant will generally be adequate if future actions of other Federal agencies that are connected with, cumulative with, or similar to the NRC action are identified and described in sufficient detail to enable an assessment to be made.					
	When analyzing the related Federal-project activities, the reviewer should take the following steps:					

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	(1) Identify the planned activities of other Federal agencies that are directly related to the proposed project (i.e., that either would not be undertaken or would be of lesser scope if the project had not been proposed or is not approved). As noted in Section I (Areas of Review), above, activities of other Federal agencies related only to the granting of licenses, permits, or approvals will not be considered in this review. • When relevant Federal activities are identified, contact the EPA Office of Federal Activities for assistance and regional and local representatives of Federal agencies to obtain relevant information. • When no such Federal activities can be identified, terminate the review and state in ESRP 2.8 that the review identified no related Federal activities. (2) Determine the specific relationships of each identified activity with the proposed project by categorizing them as • activities that are requisites to project construction (e.g., sale or transfer of Federal land) • activities that justify some of the need for power (e.g., a planned Federal project that will depend on power to be supplied by the proposed project) • a planned Federal project that will not or cannot be accomplished unless the plant is constructed. (3) Determine the significance of any related Federal activity on the project by conducting a preliminary analysis of each identified Federal activity to determine in general terms the nature and extent of the environmental impacts that would be cumulative with those of the proposed project.					

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	• When the reviewer determines that these impacts are minor, no further consideration of the activity is required. • As a general rule, if the Federal agency responsible for the Federal activity has determined that preparation of an EIS is required, the staff may conclude that the impacts are of sufficient scope to merit further analysis of the activity to determine those impacts that would be cumulative with those of the proposed project. (4) Consider whether the Federal agency should be a cooperating agency on the NRC EIS. (5) If the environmental impacts of the related Federal activity could be significant, conduct a further analysis of each such activity to the extent necessary to identify those probable environmental impacts (and potential benefits) that could be expected as a result of construction and operation of the proposed project. • Limit the impacts and benefits to be considered to those having a direct relationship with the proposed project and those that will add to or subtract from an impact or benefit (e.g., land use, transmission corridor clearing, and/or aquatic impacts) predicted for the proposed project. • Consider only those activities associated with the primary functions of the related activity (e.g., construction and operation of a Federal facility) and, except for unusual circumstances, do not address secondary effects (such as induced industrial/community growth).					

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	• Provide this information to the appropriate ESRP Chapter 4.0 and 5.0 reviewers for their consideration in determining the cumulative impacts of the proposed project and the related Federal activity. (6) Ensure that • relevant Federal activities have been identified • their interrelationships with the proposed project have been described • all activities having potentially significant environmental impacts have been described in sufficient detail to permit a subsequent environmental impact analysis to determine the cumulative effects of these impacts with those of the proposed project. In particular, take the following steps: - Based on an overview of the proposed project activities, consultations with local and regional representatives of Federal agencies, and any input supplied by cooperating agencies, determine if relevant Federal activities have been identified and whether their interrelationships with the proposed project have been described. - Based on your experience and on consultation with the appropriate ESRP Chapter 4.0 and 5.0 reviewers, determine which of the identified Federal activities will have environmental impacts that would be cumulative with impacts of the proposed project and that are of sufficient magnitude to be considered in subsequent ESRP Chapter 4.0 and 5.0 assessments of cumulative impacts. - Ensure that the Federal activities selected for consideration have been described in sufficient detail to					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

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	permit an environmental impact assessment to be made. - make a preliminary determination as to whether any other Federal agency (or in some cases a State, regional, local, or affected Native American tribal agencies) should be contacted about their interest in becoming a cooperating agency on the NRC EIS.					
3.0 (Draft Rev. 0, March 2000)	Plant Description		Exclude			Administrative
3.1 (Draft Rev. 0, March 2000)	External Appearance and Plant Layout					
	Acceptance criteria for the review of the external appearance and plant layout are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.45 with respect to requirements of a description of the affected environment.					
	Regulatory positions and specific criteria to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the location and orientation of the principal station structures.					
	The reviewer should ensure that planning, layout, and external appearance information is adequate to serve as a basis for (1) assessing land-use impacts, (2) determining potential visual and aesthetic impacts to the surrounding environment, and (3) determining the extent to which aesthetics were considered in integrating the proposed project with the surrounding					

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	environment. When analyzing the external appearance and plant layout, the reviewer should take the following steps: (1) Review plant and station layout and external appearance data to the extent needed to prepare a description of the plant and station. This includes visiting the site to ensure that the major features of the site and station have been recorded and that the descriptive material to be used in the environmental impact statement (EIS) is correct. (2) Determine the potential visibility of plant structures in relationship to locations of local facilities that might be affected in the site vicinity (e.g., large business establishments with a high degree of visitor use, recreation areas, other public-use facilities, residential areas, or any National Register properties). • Let the extent of this analysis be governed by the potential for visual (aesthetic) impact. • Consider seasonal effects (e.g., presence or absence of foliage) in determining potential visibility. (3) Determine the relationship of the plant design and layout to the surrounding environment, including any aesthetic amenities of the site and vicinity.					
3.2 (Draft Rev. 0, March 2000)	Reactor Power Conversion System					
	Acceptance criteria for evaluating the description of the reactor and plant system are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 52.17 with					

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	respect to the number of units, type, and thermal-power level associated with the proposed facility.					
	The ER must comply with the requirements of 10 CFR 51.52 with respect to the environmental effects that arise from the transportation of fuel and waste from the facility. Note: Evaluation of transportation issues per Table S-4 should make use of the design power levels and projected actual burn up rate rather than those identified in Table S-4.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the inclusion of information concerning the reactor-power conversion system.					
	Generic determinations have been made that the environmental effects of transportation of spent fuel are bounded by those in Table S-4 for enrichment up to 5% uranium-235 by weight and fuel irradiation to 62,000 megawatt days per ton, provided that the fuel is shipped more than 5 years after discharge from the reactor (NRC 1996, NRC 1999a, 64 FR 48496).					
	These review procedures are used for applications for early site permits, construction permits, and combined licenses. Because the material to be reviewed is informational in nature, no specific analysis of the data is required. Ensure that adequate information is available to meet the purpose and scope of this ESRP. When reviewing the reactor-power conversion system, the reviewer should take the following steps: (1) Compare the proposed design parameters with those of similar operating plants and identify any features of the proposed					

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	system that represent a departure from previously reviewed plants. (2) Identify the reactor-power conversion and engineered safety feature systems and the basic design-performance data. As a rule, if the data listed under "Data and Information Needs" above are provided, this objective will be met. (3) Compare reactor design and performance data with the criteria of subparagraphs (1), (2), and (3) of paragraph (a) of 10 CFR 51.52 and notify the reviewer for ESRP 3.8 of any departures from these criteria.					
3.3 (Draft Rev. 0, March 2000)	Plant Water Use					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
3.3.1 (Draft Rev. 0, March 2000)	Water Consumption					
	Acceptance criteria for the review of proposed plant water use are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction Activities.					

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	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to NPDES permit conditions for discharges including storm water Discharges.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole-source aquifer.					
	The ER must comply with the requirements of Federal, State, regional, local, and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					

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	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the States were granted additional authority to limit hydrological alterations beyond the State's role in regulating water rights.					
	The ER must comply with the requirements of Regulatory Guide 4.2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water use, and water-quality issues.					
	ESRP 3.3.1 is intended to give a brief description of the water use in plant systems and the principal subsystems. The reviewer's analysis should be closely linked with the reviews listed in the Review Interfaces section of this ESRP to establish the plant water-use characteristics of concern to those reviews. Details of the principal subsystems are described in ESRPs 3.4.2, 3.5, and 3.6. Therefore, the reviewer of ESRP 3.3.1 should concentrate on the description of principal flow paths from the sources of water through each subsystem to the receiving water bodies without detailed flow patterns within each subsystem. With this in mind, the reviewer should take the following steps: • Analyze the flow diagrams of plant water systems by performing simple mass balance computations to ascertain whether the reported flow rates (water source withdrawals, different plant water system needs, and discharge flows) are consistent for each plant operating mode. • Consider periods of maximum water consumption, minimum water availability, and average operation by month.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• Determine if there are other station facilities with water uses not associated with operation of the proposed plant and include these uses in the analysis.					
3.3.2 (Draft Rev. 0, March 2000)	Water Treatment					
	Acceptance criteria for the review of water treatment processes are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 40 CFR 122 with respect to National Pollutant Discharge Elimination System (NPDES) permit conditions for discharges, including storm water discharges.					
	The ER must comply with the requirements of 40 CFR 165 with respect to chemicals and biocides used for treating water.					
	The ER must comply with the requirements of 40 CFR 403 with respect to effluent limitations.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of State and Native American tribal water laws and water rights.					
	The ER must comply with the requirements of WASH 1355, Nuclear Power Facility Performance Criteria for Making Environmental Impact Assessments (NRC 1974).					
	The ER must comply with the requirements of Safe Drinking Water Act.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the States were granted additional authority to limit hydrological alterations beyond the State's role in regulating water rights.					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water-use, and water-quality issues.					
	The reviewer's analysis of water treatment should be closely linked with the impact assessment review of ESRPs 4.2 and 5.2 to establish which water-treatment systems and processes have a potential for environmental impact. With this in mind, the reviewer should take the following steps when analyzing the proposed water treatment systems, to the extent needed to prepare a description of the purpose and nature of each system: Note: The principal types of treatment systems that should be described include those necessary to condition (1) the intake water for noncooling-system use within the plant and (2) water used in the plant cooling system and treatment systems required					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	for providing potable water. Chemicals used in these systems should be described. (1) Include a brief description of treatment system operating procedures, including plant operational and seasonal variations (AWWA 1990). (2) Further define each treatment system in terms of the purpose of the proposed processes and the chemicals required. (3) Identify the proposed use of chemicals. Only the systems that result in a waste discharge need to be analyzed in detail, and the reviewer should emphasize the systems that have a potential for requiring an NPDES permit. (4) Verify that • All water streams identified in ESRP 3.3.1 have been considered. • All chemicals (identification and quantities) to be used have been considered or described. • The status of NPDES permits and consultations with NPDES administrative agencies have been discussed. • The proposed systems have been described in sufficient detail to permit assessment of environmental impacts resulting from their operation. (5) Ensure that the water treatment information is adequate to serve as a basis for assessing the impacts of station construction and operation on water use.					
3.4 (Draft Rev. 0, March 2000)	Cooling System					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
3.4.1 (Draft Rev. 0, March 2000)	Description and Operational Modes					
	Acceptance criteria for the review of the cooling system for potential environmental impacts are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 52.17 (a)(1)(v) with respect to early site permits related to the type of cooling systems, intakes, and outflows that may be associated with the facility.					
	The ER must comply with the requirements of 10 CFR 50.34 with respect to a description and analysis of the structure, systems, and components of the facility.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), addresses the inclusion of information about the reactor and power conversion system.					
	For the review of the cooling system description and operational modes, the reviewer should take the following steps: (1) Ensure that sufficient information on plant operational modes is available to define cooling system performance for each identified mode of operation.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(2) Verify that plant water consumption and flow-rate data are consistent with the water-use analysis prepared by the reviewer for ESRP 3.3.1. (3) Analyze the overall cooling-system design for the following: • compatibility with the water-use descriptions of ESRP 3.3.1 • consistency with good engineering design (4) Identify and describe nonemergency modes of operation, including the following (as applicable): • design normal, with estimated monthly maximum, average, and minimum values of the operating parameters • heat treatment (thermal bio-control) • de-icing • reduced intake flow (pump outage) (5) Consider the following operating parameters for each mode of operation: • intake flow rates • discharge flow rates • circulating water (condenser) flow rates • other major plant system flow rates • temperature rise across the condenser • temperature rise across heat exchangers in the service water systems • heat dissipation system discharge temperatures					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• chemical concentration factors for major cooling system components • frequency and duration of operation for each mode.					
3.4.2 (Draft Rev. 0, March 2000)	Component Descriptions					
	Acceptance criteria for the review of the cooling system components are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50.34 with respect to the need for a description of the components of the facility.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Station (NRC 1976), with respect to providing a description of the applicant's planned cooling system components.					
	The reviewer's analysis of the intake, discharge, and heat dissipation system component descriptions should be closely linked with the assessment of construction and operational impacts directed by ESRP Chapters 4.0 and 5.0. The intent of this analysis is to identify and describe the design and performance characteristics of the proposed cooling components that can be expected to cause environmental impacts as a result of construction or operation. The characteristics generally considered are listed under "Data and Information Needs" in this ESRP. Each cooling system component should be analyzed, and the reviewer should prepare descriptions of the design and performance characteristics that are generally					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	expected to result in environmental impacts (e.g., intake configuration, flow velocity through traveling screens, cooling tower drift). The review should be based on the cooling system components described in the applicant's ER and should consider component performance for the operational modes described by the reviewer for ESRP 3.4.1. With this in mind, the reviewer should take the following steps: (1) For all systems, evaluate intake and discharge temperatures and the temperature rise across the condenser. (2) For cooling towers, determine average discharge temperatures for each month of the year using cooling tower performance curves. The average discharge temperature will be calculated by using the average wet-bulb temperature for the month. (3) For spray systems, analyze the applicant's estimates of average monthly discharge temperatures. The depth and extent of this analysis should depend on the seriousness of the predicted impacts of the heated effluent on the receiving body of water and the level of confidence in the applicant's model. (4) In the cases where auxiliary systems are employed to further cool the blowdown discharged from the main cooling system, determine the final discharge temperature. (5) Consult with the appropriate ESRP Chapters 4.0 and 5.0 reviewers to determine additional cooling system component design or performance characteristics to be analyzed and described. (6) Compare the cooling system descriptions with those of similar					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	operating plants and identify design or operating features of the proposed cooling system that represent a major departure from previously reviewed systems. (7) Determine if the cooling system component descriptions are consistent, accurate, and given in sufficient detail to serve the needs of the reviews of intake, discharge, and heat dissipation system impacts. (8) Ensure that • Descriptions of the intake, heat dissipation, and discharge systems are sufficiently complete to serve the purposes of the evaluations described by the appropriate ESRP Chapters 4.0 and 5.0, including any special descriptive information needed to evaluate compliance with applicable regulations (e.g., noise, Federal Water Pollution Control Act [FWPCA], commonly Clean Water Act). • The predicted operational characteristics (e.g., flow rates and velocities) are consistent with system design. • The proposed systems are consistent with good engineering practice. • Unusual system designs are identified. (9) Verify all significant performance characteristics and, if necessary, conduct independent analyses to ensure that performance characteristics are accurately described. The following are examples of such analyses: • intake system flow rates, flow velocities, and velocity distributions • cooling tower performance (e.g., approach to wet-bulb					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	temperature, drift rate and droplet size, noise-level contours) • cooling pond performance (e.g., capacity, mean temperature) • spray system performance • discharge system performance (e.g., flow velocity).					
3.5 (Draft Rev. 0, March 2000)	Radioactive Waste Management System.					
	Acceptance criteria for the review of radioactive waste management systems are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 20 with respect to requirements for waste disposal and doses to the public from those wastes.					
	The ER must comply with the requirements of 10 CFR 50 Appendix I, with respect to the guidelines for effluent releases based on maximum individual dose and population dose.					
	The ER must comply with the requirements of 10 CFR 50.34a with respect to effluent releases.					
	The ER must comply with the requirements of 10 CFR 20.1301(d), with respect to standards set to limit the release of radioactive materials from power reactors.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are provided in the following:					
	The ER must comply with the requirements of Regulatory Guide 1.112, Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (NRC 1977), with respect to determining the releases of radioactive effluents from power reactors.					
	The ER must comply with the requirements of NUREG-0016, Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling-water reactors (NRC 1976a),					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	with respect to determining the releases of radioactive effluents from a boiling-water reactor (BWR).					
	The ER must comply with the requirements of NUREG-0017, Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized-water reactors (NRC 1976b), with respect to determining the releases of radioactive effluents from a pressurized-water reactor (PWR).					
	The ER must comply with the requirements of Regulatory Guide 4.2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976c), with respect to determining the benefit-cost of waste-management systems.					
	The detailed analysis and evaluation of the radioactive waste management and effluent control systems and the capability of these systems to meet the requirements of 10 CFR 20, Subparts D and K; 10 CFR 50, Appendix I; and 10 CFR 20.1301(d) should be presented in the staff's safety evaluation report (SER). The SER should be prepared before the environmental review, and the schedules for SER analysis, evaluation, and conclusions should be compatible with the environmental review schedules. No additional analysis should be needed, and the reviewer should proceed to the Evaluation Findings section of this ESRP. When the environmental review precedes the SER, the following analysis should be performed to the level of detail necessary to support the staff input to the environmental impact statement (EIS). The reviewer should make full use of data available from any safety review activity relative to design and performance of the radioactive waste management and effluent control systems. The reviewer should analyze the proposed radioactive waste management and effluent control systems, process and instrumentation diagrams, and system process flow diagrams to determine sources of waste, points of collection of waste, flow					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	paths through the systems (including all bypasses), the treatment provided, and the points of release of effluents to the environment. Using this information, the reviewer should calculate the quantity of radioactive materials released annually in effluents (source term) during normal operation, including anticipated operational occurrences. The reviewer should use the parameters and calculational techniques described in NUREG-0016 or NUREG-0017, as appropriate, to make these calculations. If the applicant has provided a source term that is consistent with these parameters and calculational techniques, the reviewer should accept it and should not perform a separate calculation. The results of this analysis should be provided to the reviewer for ESRP 5.4 for calculation of the maximum individual and population doses expected to result from these effluent quantities. The reviewer should evaluate the tabulated parameters and components considered in the benefit-cost balance, along with the dollar/person sievert reduction. The reviewer should use the following evaluation procedure: (1) Initially, evaluate the proposed radioactive waste management and effluent control systems to ensure that they are adequately described and provide reasonable assurance of performing the function as specified. (2) Ensure that the source terms provided to the reviewer for ESRP 5.4 correctly identify the radioactive materials (and their release points) released annually in effluents during normal operation. (3) Compare the maximally exposed individual doses calculated by the reviewer for ESRP 5.4 with the design objectives described in 10 CFR 50, Appendix I, to determine if these					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	objectives have been met. If it is determined that the proposed radioactive waste management and effluent control systems will not meet the requirements of 10 CFR 50, Appendix I, the reviewer should consult with the applicant to obtain the applicant's commitment to include additional treatment equipment and effluent control measures pursuant to Section 2.d of Appendix I that will provide reasonable assurance of conformance with the applicable regulations. This consultation should be through the EPM and reflect appropriate NRC management procedures. (4) If additional equipment or control measures are needed, repeat the analysis and evaluation procedures of this ESRP, and when necessary, request additional dose calculations from the reviewer for ESRP 5.4, until the reviewer concludes that the doses calculated from the source terms are consistent with the design objectives in 10 CFR 50, Appendix I. At this point, you may conclude that the proposed radioactive waste management and effluent control systems have the capability to control and maintain releases of radioactive materials in effluents to meet the design objectives of Appendix I to 10 CFR 50 and the requirements of 10 CFR 50.34a.					
3.6 (Draft Rev. 0, March 2000)	Nonradioactive Waste Systems					
	The introductory paragraph prepared under this ESRP should be consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
3.6.1 (Draft Rev. 0, March	Effluents Containing Chemicals or Biocides					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
2000)						
	Acceptance criteria for review of waste effluents are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 40 CFR 122 with respect to the National Pollutant Discharge Elimination System (NPDES) permit conditions for discharges, including storm water discharges.					
	The ER must comply with the requirements of 40 CFR 147 with respect to restrictions or waste disposal.					
	The ER must comply with the requirements of 40 CFR 165 with respect to liquid effluents.					
	The ER must comply with the requirements of 40 CFR 403 with respect to effluent standards					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of Federal, State, local, regional, and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly known as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water use, and water-quality issues.					
	The regulatory position necessary to meet this objective requires documentation of consultations with NPDES authority.					
	The reviewer's analysis of nonradioactive effluent systems containing chemicals or biocides should be closely linked with the impact assessment review for ESRPs 5.3.2 and 5.5 to establish the waste stream characteristics that are most likely to result in environmental impacts. With this in mind, the reviewer should take the following steps: (1) Establish that the information necessary for subsequent impact analyses is available. (2) Review each system effluent stream to determine that treatment processes, points of chemical additions or alterations, flow characteristics, maximum and average concentrations of added and ambient water constituents, and point of discharge are identified. (3) Review the applicant's calculations of concentrations in the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	effluent streams. (4) Consider any separate discharge system such as for sludge disposal. (5) Consider the concentrations and flow of the treated low-volume wastes (e.g., demineralizer wastes or boiler blowdowns) before dilution by high-volume streams. (6) Consider any waste system not described in ESRPs 3.3.2, 3.6.2, or 3.6.3 in this section (e.g., waste treatment/disposal ponds and clarifiers). (7) Consider site-related problems concerning water quality or special plant operating conditions (e.g., low oxygen levels, high concentration of nutrients, toxic materials, and high concentration factors within the plant), paying particular attention to the treatment of biocide residues. (8) Ensure that the effluent information is adequate to serve as a basis for assessing the impacts of plant operation resulting from the expected performance of the systems. (9) In evaluating the adequacy of this material, consult the applicable standards and guides for this environmental review (see Acceptance Criteria in this ESRP). Ensure that Federal, State, regional, local, and affected Native American tribal agencies appropriate to the objectives of this environmental review have been consulted and that the provisions of any applicable Memoranda of Understanding with the NRC have been considered. Also ensure that compliance with applicable Federal, State, regional, local, and affected Native American tribal standards have been determined.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Note: For wastes discharged to surface waters, issuance of a NPDES permit provides determination of compliance. (10) Evaluate the descriptions of the treatment systems and their effluent streams to determine that • all identified waste streams have been considered • all discharged chemicals and biocides have been considered unusual procedures or site-specific problems that could result in unusual environmental impacts are identified.					
3.6.2 (Draft Rev. 0, March 2000)	Sanitary System Effluents					
	Acceptance criteria for review of sanitary system effluents are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to NPDES permit conditions for discharges, including storm-water Discharges.					
	The ER must comply with the requirements of 40 CFR 133 with respect to sanitary effluents.					
	The ER must comply with the requirements of 40 CFR 403 with respect to sanitary wastes.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of Federal, State, local, regional, and Native American tribal water laws and water					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the States were granted additional authority to limit hydrological alterations beyond the State's role in regulating water rights.					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water-use, and water-quality issues.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	The regulatory position necessary to meet this objective requires documentation of consultations with NPDES authority.					
	When reviewing sanitary effluent systems, the reviewer should take the following steps: (1) Describe the sanitary treatment/disposal system effluent characteristics and quantities, system capacity, unit loading factors, impact of storm water runoff, and predicted quality. (2) Determine the characteristics, including point of discharge or place of ultimate disposal of any separate discharge system such as sludge disposal. (3) Compare the pollutant release levels with applicable regulations and water-quality standards. (4) Ensure that the sanitary system effluent information is adequate to serve as a basis for assessing the impacts of plant construction and operation resulting from the expected performance of the system. (a) In evaluating the adequacy of this material, consult the applicable standards and guides for this environmental review. (b) Ensure that the requirements of Federal, State, regional, local, and affected Native American tribal agencies appropriate to the objectives of this environmental review have been considered and that the system as proposed is capable of meeting these requirements. (5) Ensure that the proposed systems are adequate and the proposed system operating procedures are consistent with good engineering practice and with the degree of waste					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	treatment needed (AWWA 1990).					
3.6.3 (Draft Rev. 0, March 2000)	Other Effluents					
	Acceptance criteria for the review of the effluents of the proposed plant sites are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 40 CFR 122 with respect to NPDES permit conditions for discharges, including storm water Discharges.					
	The ER must comply with the requirements of 40 CFR 147 with respect to effluent-disposal limitation.					
	The ER must comply with the requirements of 40 CFR 227 with respect to criteria for evaluating environmental impacts.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of Relevant Federal, State, local, regional, and Native American tribal regulations.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including miscellaneous gaseous and liquid and solid effluents.					
	The description of these miscellaneous sources of nonradioactive wastes should be closely linked with the impact assessment review for ESRPs 5.5.1 and 5.5.2 to establish the					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	nonradioactive waste characteristics that are most likely to result in environmental impacts. With this in mind, the reviewer should take the following steps: (1) Establish that the information necessary for subsequent impact analysis and comparison with regulatory standards is available and consider the manner of proposed waste treatment and control. (2) Describe the procedures for effluent handling and disposal. (3) Compare the proposed effluent systems with standard designs to determine the adequacy of the system (e.g., equipment to remove oil from storm drainage). (4) Consider the handling of dangerous materials. (5) Compare atmospheric emissions with applicable Federal, State, regional, local, and affected Native American tribal standards. (6) Identify any unusual site-related conditions (e.g., air quality standards) that would affect treatment or release of miscellaneous nonradioactive wastes. (7) Ensure that the descriptions of miscellaneous effluents and treatment systems are adequate to serve as a basis for assessing the impacts of these discharges during plant construction and operation. (8) Ensure that the requirements of Federal, State, regional, local, and affected Native American tribal agencies appropriate to the objectives of this environmental review have been					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	considered. (9) Evaluate the descriptions of miscellaneous wastes and waste systems to determine that • comparison of amounts and concentrations of waste discharges have been made with appropriate standards and criteria • all waste streams and discharged wastes not considered in ESRPs 3.6.1 and 3.6.2 have been considered • proposed procedures are consistent with good engineering practice and are consistent with the degree of waste treatment needed • unusual procedures or site-specific problems that could result in unusual environmental impacts are identified.					
3.7 (Draft Rev. 1, July, 2007)	Power Transmission System					
	Acceptance criteria for the review of power transmission line siting are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 18 CFR Part 35 with respect to the interconnection procedures (when applicable).					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), in which the level of detailed description for the construction and maintenance of these structures and their rights-of-way are identified.					
	The ER must comply with the requirements of Institute of					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Electrical and Electronic Engineers, Inc. (current NESC) with respect to electric shock hazards.					
	The ER must comply with the requirements of Applicable Federal, State, regional, local, and affected Native American tribal standards, guidelines, and requirements.					
	The reviewer's analysis of the proposed power transmission system should be closely linked with the impact assessment review described within ESRPs 4.1, 4.3.1, 4.3.2, 4.4.3, 5.1.2, 5.1.3, 5.6, and 5.8.3 to establish the general power transmission system characteristics that are most likely to affect these reviews. Because this plan is primarily for description, the information can usually be obtained from the ER or from responses to questions asked of the applicant. When an applicant has identified a specific corridor or corridors as the proposed transmission line route or routes, only those corridors need to be considered in this review. (Alternative corridors should be considered by the reviewer for ESRP 9.4.3 on Alternative Transmission Systems.) If no specific corridors are identified, the reviewer should consider in this review all potential corridors identified by the applicant.					
3.8 (Draft Rev. 1, July, 2007)	Transportation of Radioactive Materials					
	Acceptance criteria for the description of the transportation of radioactive materials are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.52 with respect to the design and operational parameters related to the transportation of fuel and waste to and from the reactor.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	There are no regulatory positions specific to this ESRP. Note, however, that the NRC has generically considered the environmental impacts of spent nuclear fuel with U-235 enrichment levels up to 5% and irradiation levels up to 62,000 megawatt-days per metric ton and found that the environmental impacts of spent nuclear fuel transport are bounded by the impacts listed in Table S-4 provided that more than 5 years has elapsed between removal of the fuel from the reactor and shipment of the fuel offsite (NRC 1996; NRC 1999). However, these analyses cannot serve as the initial licensing basis for new reactors.					
	The reviewer's analysis of the data and information is required to support the reviewer's evaluation for conformance with 10 CFR 51.52(a) (see Evaluation Findings in this ESRP). The analysis should consist of assembling the data listed in the procedures below and verifying their accuracy. The reviewer may consult with the reviewers for ESRPs 3.2 and 3.5 to verify the data. The reviewer should take the following steps: (1) Prepare the following information about packaging and shipping parameters: • onsite storage of irradiated fuel – information about the minimum time between removal from the reactor and shipment offsite • radioactive wastes other than fuel – information about the form of packaged waste to be shipped offsite (The reviewer should consider the proposed solid waste treatment and packaging procedures in evaluating this criterion.)					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• transport modes for new fuel shipment to the plant • transport mode for irradiated fuel shipments offsite • transport mode for other radioactive-waste shipments offsite • average heat load for irradiated fuel casks in transit • maximum gross vehicle weight for truck and rail shipments of unirradiated fuel, spent fuel, and other radioactive waste. (2) Review the transportation analyses in ESRPs 5.7.2 and 7.4 to determine if they are solely a comparison to the reactor and transportation conditions in 10 CFR 51.52(a) or a full description and detailed analysis of the environmental effects of transportation of fuel and waste to and from the reactor. If the former, the review of this section is complete. If the latter, additional transportation parameters should be provided in this section to support a full and detailed analysis, including: 1. general description of packaging systems for unirradiated fuel, spent fuel, and waste(e.g., approximate dimensions, weight) 2. packaging system capacity 3. shipment mode and capacity 4. radiation dose rates for loaded packages 5. locations of fuel fabrication facilities and potential destinations for shipments of spent fuel and radioactive waste that will be used to determine shipping route information.					
4.0 (Draft Rev. 0, March 2010)	Environmental Impacts of Construction					Exclude, Administrative
4.1 (Draft Rev. 0, March	Land-Use Impacts					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
2000)						
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
4.1.1 (Draft Rev. 1, July 2007)	The Site and Vicinity					
	Acceptance criteria for the review of land-use impacts at the site of the nuclear power station and in its vicinity are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to analysis requirements to be included in draft environmental impact statements (EISs) prepared by NRC.					
	The ER must comply with the requirements of 10 CFR 51, Appendix A(7), with respect to discussion in EISs prepared by NRC of possible conflicts between alternatives and the objectives of applicable land-use plans.					
	The ER must comply with the requirements of guidance and requirements for particular land types shown in Table 4.1.1-1.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998), with respect to land-use considerations rendering a proposed site unsuitable for a nuclear power station.					
	Because some portions of land-use impacts are covered in					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	ESRP 4.1.3, "Historic/Archaeological Sites"; ESRP 4.3.1, "Terrestrial Ecosystems"; and ESRP 4.4, "Socioeconomic Impacts"; this ESRP is limited to those direct physical changes and restrictions on land use at the site and vicinity due to plant construction. For each of these, the impact analysis should include consideration of the potential changes in land use as a result of the siting decision and the direct physical impacts on the site and vicinity as a result of construction activities. The reviewer should direct the analysis toward conclusions with respect to the following: • long-term restrictions of land use that would result from the licensing action and long-term physical changes in land use of the site and vicinity • short-term physical changes in land use of the site and vicinity and the applicant's plans for mitigation of adverse impacts • construction impacts on the geologic environment. The reviewer should take the following steps: (1) Evaluate Long-Term Restrictions of Land Use that would Result from the Licensing Action and Long-Term Physical Changes in Land Use of the Site and Vicinity: (a) Identify changes in land use that would occur as a consequence of the licensing action. Consider land-use changes in the context of the amount and quality of land affected after proposed measures, if any, have been implemented.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Review restrictions on the use of farm land, recreational areas, housing areas, and other similar areas. Consider any restrictions or modifications of lands classified as floodplain, wetlands, or coastal zone. (b) If appropriate, analyze the degree of change and its acceptability by comparing changes in land use with existing standards, guides, regulations, or legislation; or to Federal, State, regional, local, and affected Native American tribal land-use plans and zoning ordinances, consulting with these sources, and ensuring consistency with them where required or desirable. • Refer to the Federal sources listed in Table 4.1.1-1 (and comparable State sources applicable to the applicant's proposed site) for particular types of land. • If there are no relevant standards, guides, regulations, legislation, or land-use plans, analyze the severity of the impact without these aids. (c) Analyze the restriction on the use of land such as farmland or forests in the context of the amount and quality of the land in the vicinity of the plant. • Removal of less than 2% of such land, or up to 500 hectares (1235 ac), generally has minor effects, particularly if the land is not unique or otherwise distinguished. • When larger land areas are to be committed for a proposed nuclear station (e.g., greater than500 hectares (1235 ac)) or if the reviewer for ESRP 2.2.1 indicates that the proposed land areas are unique or otherwise distinguished, further analysis is needed to					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	determine the quality of the land. There are three indices of land quality that may be used for guidance. The first is the definitions of prime and unique farmland in the Farmland Protection Policy Act of 1981. The second is the land relative value rating prepared by the NRCS. The third and oldest index is the land capability classification system first published by the U.S. Department of Agriculture (Klingebiel and Montgomery 1961). The indices are further defined as follows: - Prime and Unique Farmland. The terms "prime farmland" and "unique farmland" are defined in the Farmland Protection Policy Act of 1981. Prime farmland is defined to be land that has the best combination of physical and chemical characteristics for producing food, feed, fiber, forage, oilseed, and other agricultural crops with minimum inputs of fuel, fertilizer, pesticides, and labor, and without intolerable soil erosion, as determined by the Secretary of Agriculture. Prime farmland includes land that possesses the above characteristics but is being used urrently to produce livestock and timber. It does not include land already in or committed to urban development or water storage. Unique farmland is defined in the Act to be land other than prime farmland that is used for production of specific high-value food and fiber crops, as determined by					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	the Secretary of Agriculture. It has the special combination of soil quality, location, growing season, and moisture supply needed to economically produce sustained high quality or high yields of specific crops when treated and managed according to acceptable farming methods. Examples of such crops include citrus, tree nuts, olives, cranberries, fruits, and vegetables. - Relative Value Rating. The NRCS will compute a relative value rating for a tract of land upon request from a Federal agency. Procedures are described at 7 CFR 658.4 and 658.5. The rating is based on a variety of data, including soil potential, productivity ratings, and land capability classifications (see below). The reviewer of ESRP 4.1.1 can request that NRCS prepare a relative value rating for a proposed site involving farmland. - Land Capability Classification. This classification places land in one of eight categories based on soil characteristics (Klingebiel and Montgomery 1961). The eight classifications are listed in Table 4.1.1-2. Land in capability Classes I and II is usually the most productive and, therefore, should be subject to the most detailed analysis when it is to be committed. Commitment of land in Classes III through VIII is less important. (d) If the land at the proposed site (1) meets the statutory definition of prime or unique, (2) has a relative value rating placing it within the top half in terms of agricultural production in the local government jurisdiction, or (3) has a land capability					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	classification of I or II, the reviewer should assess the productivity of the land to provide input to the benefit-cost balance in ESRP 10.4. The reviewer should consider a State's published documents on agricultural statistics, including crop and animal production statistics and land areas by county, and consult with State and local agricultural, soil conservation, and cooperative extension agencies to complete this assessment. (2) Analyzing the Short-Term Physical Changes in Land Use of the Site and Vicinity and the Applicant's Plans for Mitigation of Adverse Impacts: (a) Consider mitigation measures for adverse impacts. Matters that can be assessed include earth leveling, revegetation, landscaping, cleanup and disposal of debris, erosion control structures, land management practices, stabilization of spoil piles, and stabilization of dikes on cooling lakes. (3) Analyzing the Construction Impacts on the Geologic Environment: (a) Consult with the staff safety evaluation reviewers for geology (ESRP 2.6) for an evaluation of the impact of station construction on the geologic environment and for appropriate licensing/permit conditions. (b) Determine whether construction of the plant would prevent the exploitation at the proposed site or in the vicinity of mineral resources (e.g., sand and gravel, coal, oil, natural gas, or ores) of commercial value. (c) Determine if any such mineral extraction is currently in process or is planned, and the extent to which plant construction					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	would affect such operations. (d) Consult with the staff's safety evaluation reviewers for geology for assistance in this review and for an analysis of any other impacts of plant construction on the geologic environment.					
4.1.2 (Draft Rev. 1, July 2007)	Transmission Corridors and Offsite Areas					
	Acceptance criteria for the review of land-use impacts of transmission corridors and offsite areas are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to analysis requirements to be included in draft environmental impact statements (EISs) prepared by NRC.					
	The ER must comply with the requirements of 10 CFR 51, Appendix A(7), with respect to discussion in EISs prepared by NRC of possible conflicts between alternatives and the objectives of applicable land-use plans.					
	The ER must comply with the guidance and requirements for particular land types shown in Table 4.1.2-1.					
	Regulatory positions and specific criteria to meet the regulations identified above are					
	There are no conflicts between the proposed transmission corridors and offsite areas and the objectives of Federal, State, regional, and local (and in the case of proposed location on a reservation, Native American tribe) land-use plans and the Federal sources shown in Table 4.1.2-1 (plus comparable State sources).					
	If there are or are likely to be conflicts, the extent of the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	conflicts, the possibilities of resolving the conflicts, and the seriousness of the impact of the proposal on land-use plans and policies and the effectiveness of land-use control mechanisms for the area can be adequately evaluated and discussed in the EIS or other environmental document.					
	Limited portions of land-use impacts are covered in ESRPs 4.1.3, 4.3.1, and 4.4; therefore, this ESRP is limited to direct physical changes and restriction on land use in the corridors and offsite areas due to construction. For each of these, the impact analysis should include consideration of the direct physical land-use impacts that occur in the corridors and offsite areas due to construction activities. The reviewer should direct the analysis toward conclusions with respect to the following: • long-term physical changes in land use of the corridors and offsite areas • short-term changes in land use of the corridors and offsite areas and the applicant's plans for mitigation of adverse impacts • construction impacts on the geologic environment. The reviewer should take the following steps: (1) Evaluating Long-Term Physical Changes in Land Use of the Corridors and Offsite Areas: (a) Consider land-use changes in the context of the amount and quality of land affected after mitigating measures, if any, have been implemented.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Review restrictions imposed by the presence of transmission lines on use of farm land, recreational areas, housing areas, and other similar areas. (b) If appropriate, analyze the degree of change and its acceptability by comparing changes in land use with existing standards, guides, regulations, or legislation or to Federal, State, regional, local, and affected Native American tribal land-use plans and zoning ordinances, consulting with these sources and ensuring consistency with them where required or desirable. • Refer to the Federal sources listed in Table 4.1.2-1 (and comparable State sources applicable to the proposed transmission line corridors and offsite areas) for particular types of land. • If there are no relevant standards, guides, regulations, legislation, or land-use plans, analyze the severity of the impact without them. (c) Analyze the restrictions on use of land such as farm land or forests in the context of the amount and quality of the land generally available in the region as compared with that changed due to the corridors and offsite areas, recognizing that the use of some of the land of the corridors may not be changed from its current use. Modification of use for the amount of land usually used for transmission corridors and offsite areas generally has minor effects, if the land is not unique or otherwise distinguished. (d) If the land to be changed due to the corridors and offsite areas (1) meets the statutory definition of prime or unique, or (2) has a relative value rating placing it within the top half in terms of agricultural production in the local government jurisdiction, or (3) has a land capability classification of I or II, (see "Land Capability					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Classifications" under "Review Procedures" in ESRP 4.1.1), assess the productivity of the land to determine the need for mitigation or avoidance of any predicted impact. (2) Analyzing the Short Term Changes in Land Use of the Corridors and Offsite Areas and the Applicant's Plans for Mitigation of Adverse Impacts: (a) Consider mitigation measures for adverse impacts. Matters to be reviewed include revegetation, landscaping, cleanup and disposal of debris, erosion control, land-management practices, and use of chemicals. (3) Analyzing the Construction Impacts on the Geologic Environment: (a) Consult with the safety evaluation reviewers for geology for an analysis of the potential impacts of corridor and offsite area construction on the geologic environment.					
4.1.3 (Draft Rev. 0, March 2000)	Historic Properties					
	Acceptance criteria for the review of historic properties that could be impacted by proposed construction are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 36 CFR 800 with respect to the process by which a Federal agency meets its requirements under Sections 106 and 110 of the National Historic Preservation Act (NHPA) to ensure that agency assisted or licensed undertakings consider the effects of the undertaking on historic properties that are evaluated and determined eligible for listing on the National Register.					
	The ER must comply with the requirements of 43 CFR 10 with					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	respect to guidelines and procedures for Federal agencies to follow in the event of inadvertent discoveries of human remains, funerary objects, sacred objects, or objects of cultural patrimony during construction projects on Federal or Native American tribal lands.					
	Regulatory positions and specific criteria to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Nuclear Reactor Regulation (NRR) Office Letter No. 906, Revision 1, which includes guidance for complying with the requirements contained in the NHPA with respect to protection of historic properties during the construction phase and for handling inadvertent discoveries during construction (NRC 1996). NRR Office Letter No. 906 is revised periodically. Obtain a copy of the latest revision for current guidance.					
	The information is acceptable if it permits an evaluation of potential impacts and mitigation measures to historic properties.					
	The reviewer's analysis of construction impacts on historic and cultural resources should be linked to the environmental review directed by ESRP 2.5.3 to ensure that the environmental factors most likely to be impacted by proposed construction activities are described in that section. An additional source of expertise in the area of historic and cultural preservation is the Archaeology and Ethnography Program (AEP) of the National Park Service, Department of Interior. With this in mind, the reviewer should take the following steps: (1) With the assistance of the AEP and in consultation with the SHPO, consider the historic properties that are listed in or are eligible for inclusion in the National Register and that may be affected by construction of the proposed project. (2) Use the output of appropriate environmental reviews					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	describing proposed construction activity to identify the construction activities that could result in potential impacts. (3) When assessing the potential impacts on these resources, refer to 36 CFR 800, which describes in detail how to assess the impact of a proposed action on properties that are listed in or are eligible for inclusion in the National Register. (4) Recognize that there are generally two types of impacts on a resource: direct impacts (e.g., destruction during excavation) and indirect impacts (e.g., visual impact, denial of access); and consult with the reviewer for ESRPs 3.1 and 3.7 for assistance in analyzing indirect impacts. (5) Although historic properties that are neither listed in nor eligible for inclusion in the National Register are not protected by the provisions of the NHPA, as amended, or 36 CFR 800, consider the potential impacts on these resources and measures and controls to avoid adverse impacts. (6) For properties that are not eligible for inclusion in the National Register, get assistance from the SHPO, the Office of Archaeology and Historic Preservation, or other qualified individuals, as needed. (7) Consider alternatives to reduce the impact on the cultural and historic resources and make a determination of the cost of each alternative versus the benefit derived. (8) Include the cost of the recovery required by the Historical and Archaeological Preservation Act of 1974 in the consideration of alternatives.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(9) When the evaluation does not justify preservation of the resource, request that the applicant recover archaeological, historic, architectural, and cultural data related to the resource. • This recovery may include recording by photographs and measured drawings, archaeological excavations to uncover data and material, removal of structures or salvage of architectural features, and other steps that will ensure full knowledge of the lost resource. • Salvaged artifacts and materials should be deposited where they are of public and educational benefit. (10) Assess the operational impacts on historic properties concurrently with this review.					
4.2 (Draft Rev. 0, March 2000)	Water-Related Impacts					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
4.2.1 (Draft Rev. 0, March 2000)	Hydrologic Alterations					
	Acceptance criteria for the review of the hydrological alterations at the proposed plant sites are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction activities.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to NPDES permit conditions for discharges, including storm water discharges.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole source aquifer.					
	The ER must comply with the requirements of 40 CFR 227 with respect to criteria for evaluating environmental impacts.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of Federal, State, local, regional, and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts of striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S. Supreme Court granted the States additional authority to limit hydrological alterations beyond the States' role in regulating water rights.					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of including hydrology, water-use, and water-quality issues.					
	The reviewer should ensure that the construction activities that result in hydrologic alterations have been identified and seek confirmation that the alterations that result in environmental impacts have been described in sufficient detail to allow for the subsequent analysis and assessment of these impacts. The reviewer should take the following steps: (1) Identify alterations in water quantity in the various construction affected hydrologic systems under the existing and known future water rights and allocations.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(2) Describe the physical effects of identified alterations in the quantity of water available on other consumptive water users. (3) Describe the physical effects of altered hydrologic geometry, flow and circulation patterns, and mixing processes on nonconsumptive water users and to terrestrial and aquatic ecology. (a) Cooperate with the reviewers for ESRPs 4.1.1, 4.2.2, 4.3.1, and 4.3.2 in (1) determining the extent and magnitude of the resulting impacts and (2) evaluating means to mitigate or avoid these impacts. (b) When project construction or construction activity within a floodplain or wetland has been proposed, evaluate the extent of compliance with applicable floodplain or wetland protection standards and give particular attention to the consideration of alternatives to avoid adverse effects. (c) Assist the reviewer for ESRP 4.2.2 in evaluating the impacts of any construction or construction-related activity located in the floodplain or wetland. (d) Assist the appropriate ESRP 9.4 reviewers in the identification and analysis of alternatives that would avoid construction or construction activity in the floodplain or wetlands. (4) Describe the physical effects of altered erosional, depositional, and sediment characteristics on other water users, on nearby property, and to aquatic ecology. The reviewer should identify the alterations by associating the previously identified activities with changes in (1) water quantity					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	and availability, (2) hydrological geometries (especially within the floodplain or wetland), flow and circulation patterns, and mixing processes, and (3) erosion, deposition, and sediment transport. The reviewer should take the following steps: (1) Analyze the water quantity and availability by analyzing the construction activities that can alter the quantities of water physically available in nearby hydrologic systems and determine the alterations. (a) Consider all water used during construction: • the sources of the water • points of discharge • all water diversions that change the quantities of water in various parts of water systems (e.g., construction dewatering). (b) For the hydrologic systems where alterations in water quantities due to construction have been identified, determine the physical effects (e.g., altered well yields, water levels relative to intake pipes) likely to have impacts on other water users. (2) Analyze the hydrologic geometry, flow and circulation patterns, and mixing processes by evaluating the construction activities that can alter hydrologic geometries, flow and circulation patterns, and mixing processes, and determining the alterations. (a) Consider all construction activities within water bodies and diversions of water during construction. (b) Give particular attention to construction and related activities					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	located in the floodplains or wetlands. (c) Identify any Federal, State, regional, local, or Native American tribal floodplain or wetland protection standards and analyze proposed project construction and construction-related activities with respect to these standards. (3) Analyze the erosion, deposition, and sediment transport by evaluating the construction activities that can alter erosional, depositional, and sediment transport characteristics and determine the alterations. (a) Consider all construction activities within water bodies in relation to the natural processes occurring before construction. (b) For those areas where alterations in the natural erosional, depositional, and sediment transport processes have been identified, determine the physical effects (e.g., beach erosion, channel shoaling) likely to have impacts on other water users. (4) Be familiar with the provisions of standards, guides, and agreements pertinent to the hydrological aspects of plant construction. (a) Determine compliance and the adequacy of commitments to comply with applicable regulations and guides. (b) Consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies to make this determination.					
4.2.2 (Draft Rev. 0, March 2000)	Water-Use Impacts					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Acceptance criteria for the review of the water-use impact at the proposed plant sites are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction activities.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to the National Pollutant Discharge Elimination System (NPDES) permit conditions for discharges, including storm water discharges.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole-source aquifer.					
	The ER must comply with the requirements of Federal, State, regional, local, and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing					

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	the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts of striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination. Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S. Supreme Court granted the States additional authority to limit hydrological alterations beyond the States' role in regulating water rights. Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water-use, and water-quality issues.					
	The reviewer should take the following steps: (1) Evaluate water quantity and availability by identifying water users potentially impacted by alterations in water quantity and availability: (a) Describe any impacts of reduced water quantity and availability. (b) Describe the possibility for inequalities between proposed construction water use and existing and known future water					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	rights and allocations and the probable nature and extent of these inequalities. (2) Evaluate the construction activities and the hydrologic alterations identified in ESRP 4.2.1 with respect to their potential impacts to water users or water-use areas: (a) Compare the effects of these alterations (e.g., increased temperature, salinity, erosion, sedimentation) with pre-construction conditions to assess the magnitude of the impact. (b) Evaluate the impacts for individual water users and for water-use areas. (c) Identify and describe proposed construction or construction activities located on a floodplain or wetland as follows: • Consult with appropriate Federal, State, regional, local, and Native American tribal agencies to determine the extent to which any such activities will conform with applicable floodplain and wetland standards. • Ensure that the analysis has considered short-term effects (e.g., floodplain alterations resulting from temporary construction structures or activities) as well as the long-term alteration caused by the completed plant. • Consult with the reviewer for ESRP 4.2.1 and the reviewers for ESRP 9.4.1 to analyze alternatives to any proposed activity located in the floodplain. The intent of this instruction is to ensure that alternatives to avoid adverse effects and incompatible development in a floodplain or wetland have been considered.					

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	(d) Identify construction and construction activities that will alter or restrict shoreline access (e.g., beach closure) and surface oriented water uses (e.g., commercial and recreational fishing, navigation) including the following: • Describe the effects of construction to water users. • If potential adverse impacts are predicted, identify alternative design, construction practices, or procedures that could mitigate or avoid the impacts. (3) Analyze water quality: (a) Identify hydrologic alterations and construction activities affecting water quality and describe their effects on water users or water-use areas. (b) Describe the time duration or time periods when the impact will be experienced, and the number of water users or extent of water-use areas affected. (When necessary, consult with Federal, State, regional, local, and affected Native American tribal agencies for assistance in evaluating the identified impacts.) (c) Review consultation with appropriate agencies regarding compliance with Federal, State, regional, local, and affected Native American tribal water-quality standards. The reviewer's analysis of construction impacts on water use should be coordinated with the hydrologic alteration descriptions provided by the environmental review for ESRP 4.2.1. This coordination should ensure that the environmental factors most likely to be impacted by hydrologic alterations are described in					

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	sufficient detail to permit assessment of the predicted impacts. The reviewer should independently identify and analyze those construction activities expected to affect the quality of receiving water bodies. The reviewer should consult with the reviewers for ESRPs 2.3.2, 4.1, 4.3, and 4.4 to establish the location and nature of those water users potentially impacted by hydrologic alterations and water-quality changes. The reviewer should take the following steps: (1) Analyze reduced water availability: (a) Initiate this analysis if the reviewer for ESRP 4.2.1 determines that construction activities will result in decreased water availability. (b) When this is predicted to occur, identify the location of those water users likely to be affected and consult with the reviewer for ESRP 4.2.1 to determine the hydrologic effects at these locations. (c) Consider these effects (e.g., lowered groundwater table, reduced well yields, lowered surface-water levels at intake structures) and determine their impacts on individual water users or water-use areas. (d) Consider seasonal requirements for water and temporal variations in water availability. (e) Consider the potential for impacts when the reviewer for ESRP 4.2.1 predicts an incompatibility between water availability as affected by project construction activity and existing and known future water rights and allocations. For these cases,					

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	analyze the potential for future inequalities in water availability to determine their probable nature and extent. (2) Analyze the construction activity and hydrologic alterations identified by the reviewer for ESRP 4.2.1 and compare them with present and predicted future water uses that could be affected: (a) Analyze in further detail any alterations that can be shown to represent a potential for water-use impacts. (b) Consider both short-term impacts (e.g., from temporary channel diversions) that will occur only during the construction period, and long-term impacts (e.g., channel restriction by a breakwater) that will occur for the period of plant operation. (c) Identify individual water users or water-use areas and predict impacts to these users or areas. (d) Identify the proposed construction activities that will restrict non-consumptive water use or water access and identify the water users so affected, categorizing the impacts as either short- or long-term. (e) Give special consideration to hydrologic alterations that affect floodplains. When such alterations are predicted, consult with the reviewer for ESRP 4.1.1 or 4.1.2 to complete the analysis of any resulting impacts. (3) Analyze water quality by considering the construction activities and hydrologic alterations expected to result in altered water quality and the water users or water-use areas that could be impacted by the water-quality alterations:					

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	(a) Consult with the reviewer for ESRP 4.2.1 to identify the affected receiving water bodies and the hydrologic alterations (e.g., erosion, sedimentation) that could affect water quality. (b) Consult with the reviewer for ESRP 2.3.3 to determine the baseline water quality of the receiving water bodies and with the reviewer for ESRP 2.3.2 to identify potentially affected water users. (c) Identify the water bodies receiving construction effluents, the flow rates and chemical composition of these effluents, and the potential for and nature of any contaminants that could be released to surface or groundwater as a result of substrate exposure during construction. (d) Consider potential impacts to water users in terms of the intended usage (e.g., heavy metals as a contaminant affecting a municipal water supply, suspended solids affecting industrial use). (e) Consult with nearby Federal, State, regional, local, and affected Native American tribal agencies in analyzing potential water-quality impacts. (f) Finally, consult with the reviewer for ESRP 4.3.2 to coordinate the analysis of impacts to water quality and to avoid any duplication of effort in this analysis.					
4.3 (Draft Rev. 0, March 2000)	Ecological Impacts					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
4.3.1 (Draft Rev. 1, July 2007)	Terrestrial Ecosystems					
	The reviewer should become familiar with the provisions of standards, guides, and agreements that are pertinent to the construction of nuclear power stations. Acceptance criteria for the review of construction impacts on terrestrial ecology on and in the vicinity of the site and transmission corridors are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71 with respect to including in the EIS information on impacts to the terrestrial environment due to construction.					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to analysis of impacts to the terrestrial environment affected by the issuance of an early site permit, combined license, or construction permit.					
	The ER must comply with the requirements of Bald and Golden Eagle Protection Act with respect to the prohibition of taking, possessing, selling, transporting, importing, or exporting the bald or golden eagle, dead or alive, without a permit.					
	The ER must comply with the requirements of Clean Water Act with respect to dredging and filling, and avoiding/minimizing impacts to terrestrial resources in the vicinity of affected navigable waters, including wetlands.					
	The ER must comply with the requirements of Coastal Zone Management Act with respect to natural resources, and land or					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
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	water use of the coastal zone.					
	The ER must comply with the requirements of Endangered Species Act with respect to identifying impacts to threatened or endangered species and/or designated critical habitat by means of informal and/or formal consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act with respect to consideration of wildlife resources in the planning of development projects.					
	The ER must comply with the requirements of Migratory Bird Treaty Act with respect to declaring that it is unlawful to take, import, export, possess, buy, sell, purchase, or barter any migratory bird. Feathers or other parts of nests or eggs, and products made from migratory birds are also covered by the Act. "Take" is defined as pursuing, hunting, shooting, poisoning, wounding, killing, capturing, trapping, or collecting.					
	Regulatory positions and specific criteria necessary to meet the regulations and other statutory requirements identified above are as follows:					
	The ER must comply with the requirements of LIC-203, Revision 1, Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Impacts (NRC 2004), with respect to NRC compliance with the Endangered Species Act.					
	The "Second Memorandum of Understanding and Policy Statement Regarding Implementation of Certain NRC and EPA Responsibilities," serves as the legal basis for NRC decisionmaking concerning licensing matters covered by NEPA and Section 511 of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(CWA).					
	The "Memorandum of Understanding between the Corps of Engineers, U.S. Army, and the NRC for the Regulation of Nuclear Power Plants," 40 FR 60115, provides guidance with respect to the NRC exercising the primary responsibility in conducting environmental reviews and in preparing EISs for nuclear power stations. The Corps of Engineers should be consulted regarding (1) coastal erosion and other shoreline modifications, (2) siltation and sedimentation processes, (3) dredging activities and disposal of dredged materials, and (4) location of structures affecting navigable waters.					
	Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (1998), contains guidance that the ecological systems and biota at potential sites and their environs should be sufficiently well known to allow reasonably certain predictions of impacts that there would be no unacceptable or unnecessary deleterious impacts on populations of important species or on ecological systems from the construction of a nuclear power station.					
	Regulatory Guide 4.11, Rev. 1, Terrestrial Environmental Studies for Nuclear Power Stations (1977), contains technical information for the design and execution of terrestrial environmental studies, the results of which should be included in the applicant's ER.					
	When evaluating the data and information acquired under "Data and Information Needs," which is necessary to determine the impacts on terrestrial ecology from station construction, the reviewer should take the following steps: (1) Identify the construction activities that could affect terrestrial					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	ecology and the types of impacts that could result. This may be done by comparing the construction footprint on and in the vicinity of the site and along transmission corridors and at any other offsite areas, relative to the location and areal extent of terrestrial habitats/plant communities, including occurrences of "important" terrestrial species and habitats (definitions of "important" species and habitats are in Table 2.4.1-1). The following steps should be useful: • Prepare a map superimposing construction areas on and in the vicinity of the site and along transmission corridors and at any other offsite areas over terrestrial habitats/plant communities, including occurrences of "important" terrestrial species and habitats (from the ER). • During the site visit, inspect construction areas, emphasizing those where alteration of the terrestrial environment is expected to be greatest due to construction (e.g., sites proposed for facilities or new water bodies for plant cooling, etc.), and where construction activities and occurrences of "important" terrestrial species and habitats are closely juxtaposed or intersect. • Supplement activities conducted under the above two bullets with information obtained from consultations with Federal, State, regional, local, and affected Native American tribal agencies (at a minimum the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service and appropriate State resource agency). • Consider how construction activities would affect terrestrial habitats/plant communities and associated wildlife, including occurrences of "important" terrestrial					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	species and habitats (e.g., consider the spatial extent and functional loss of modified habitat, the effects on critical species life stages, etc.) (from the ER) (2) Determine the magnitude of the types of impacts identified in (1) above, which may include, but are not limited to, the following: • the area of each generic terrestrial habitat/plant community type, including "important" habitats, that would be permanently or temporarily disturbed, relative to the abundance of these habitats/plant community types in the region. This includes de-watering or filling of wetlands, ponds, or seepages, or altered surface drainage patterns that support "important" habitats, and changes in terrestrial habitat resulting from creating new water bodies to provide cooling water. Consider relation of activities to introduction and/or spread of invasive and/or exotic species. Consider the nature and duration of function lost for habitats/plant communities that would be temporarily disturbed (e.g., wetlands), and evaluate the efficacy of plans to restore these in light of recognized "best management practices." Consider the adequacy of plans to prevent soil erosion in light of recognized "best management practices." Based on all the above, estimate the overall magnitude of habitat impacts. • estimate the magnitude of construction impacts to general wildlife, including State-listed species, based on habitat disturbance (e.g., tree removal), effects on critical life stages (e.g., migratory bird nesting), impediments to migrations/dispersal/movements, noise, avian collisions with elevated structures (e.g., cranes),					

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	etc. • estimate the magnitude of construction impacts for Federally listed threatened or endangered species, and/or Federally designated critical habitat. This should be done based on the factors specified in the above two bullets for estimating the magnitude of impacts to habitats and species. • if Federally threatened endangered species and/or Federally designated critical habitat occur in the project area, and the proposed project could adversely affect the species or habitat, prepare a biological assessment and consult with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service, as applicable, under Section 7 of the Endangered Species Act. The results of the biological assessment should be used to estimate the magnitude of construction impacts for Federally listed threatened or endangered species, and/or Federally designated critical habitat, noted in the preceding bullet. Note that under an Early Site Permit, adverse impacts can result only with a Limited Work Authorization, and a biological assessment should be prepared if Federally protected species and/or habitats could be affected. If a Limited Work Authorization is not being sought under an Early Site Permit, construction is not authorized, no impacts to the terrestrial ecosystem would be possible, and thus a biological assessment should not be prepared. Note that because construction is inherent in an application for a COL, a biological assessment should always be prepared if					

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	Federally protected species and/or habitats could be affected.					
4.3.2 (Draft Rev. 1, July 2007)	Aquatic Ecosystems					
	Acceptance criteria for the review of construction impacts on aquatic ecology in the vicinity of the site and transmission corridors are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71 with respect to including in the EIS information on impacts to the terrestrial environment due to construction.					
	The ER must comply with the requirements of 10 CFR 51.75, with respect to analysis of impacts to the aquatic environment affected by the issuance of a construction permit, early site permit, or combined license.					
	The ER must comply with the requirements of 10 CFR 52, Subpart A and C, with respect to analysis of impacts to the aquatic environment affected by the issuance of an early site permit or combined license.					
	The ER must comply with the requirements of Coastal Zone Management Act with respect to natural resources and land or water use in the coastal zone.					
	The ER must comply with the requirements of Endangered Species Act with respect to identifying impacts on Federally threatened or endangered species and/or Federally designated critical habitats by means of informal and/or formal consultations with the U.S. Fish and Wildlife Service and/or the National Marines Fisheries Service.					
	The ER must comply with the requirements of Federal Water Pollution Control Act, as amended, commonly referred to as the Clean Water Act, with respect to (1) activities associated with the discharge of dredge or fill materials into waters of the United States and (2) restoration and maintenance of the chemical,					

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	physical, and biological integrity of water resources.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act with respect to consideration of fish and wildlife resources in planning development projects that affect water resources.					
	The ER must comply with the requirements of Magnuson-Stevens Fishery Conservation and Management Act, as amended, with respect to identifying impacts on Federally designated essential fish habitat (EFH) in the vicinity of the site and transmission corridors by means of consultation with the National Marine Fisheries Service.					
	The ER must comply with the requirements of Marine Mammal Protection Act with respect to the protection of marine mammals.					
	The ER must comply with the requirements of Marine Protection, Research, and Sanctuaries Act with respect to the dumping of dredged material into the ocean.					
	The ER must comply with the requirements of Rivers and Harbors Appropriations Act with respect to construction of any bridge, causeway, dam, or dike over or in any port, roadstead, haven, harbor, canal, navigable river, or any other navigable water of the United States.					
	Regulatory positions and specific criteria necessary to meet the regulations and other statutory requirements identified above are as follows:					
	The ER must comply with the requirements of LIC-203, Revision 1, Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Impacts (NRC 2004), with respect to NRC compliance with the Endangered Species Act.					
	Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998), contains guidance that the ecological systems and biota at potential sites and their environs should be sufficiently well known to allow reasonably certain					

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	predictions of impacts and that there would be no unacceptable or unnecessary deleterious impacts on populations of important species or on ecological systems from the construction of a nuclear power station.					
	Memorandum of Understanding between the U.S. Army Corps of Engineers and the NRC for the Regulation of Nuclear Power Plants (40 FR 37110) provides guidance with respect to the NRC exercising the primary responsibility in conducting environmental reviews and in preparing EISs for nuclear power stations. The Corps of Engineers should be consulted regarding (1) coastal erosion and other shoreline modifications, (2) siltation and sedimentation processes, (3) dredging activities and disposal of dredged materials, and (4) location of structures affecting navigable waters.					
	Second Memorandum of Understanding and Policy Statement Regarding Implementation of Certain NRC and EPA Responsibilities, serves as the legal basis for NRC decision-making concerning licensing matters covered by NEPA and Section 511 of the Federal Water Pollution Control Act, commonly referred to as the Clean Water Act.					
	When reviewing the impacts of station construction on aquatic ecology, the reviewer should take the following steps: (1) Review the general data and information necessary to determine the impacts of station construction on aquatic ecology: (a) Identify the construction activities that affect "important" aquatic species and habitats on and in the vicinity of the site, transmission corridors, and offsite areas. (b) Determine the areal extent and location of construction activities on and in the vicinity of the site, transmission corridors, and offsite areas, and occurrences of "important" aquatic species					

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	and habitats within reasonable buffers of these areas. • Obtain a map superimposing construction impact areas over aquatic resource areas, emphasizing occurrences of "important" aquatic species and habitats (from ER). • During the site visit, inspect construction areas, emphasizing areas where construction activities and occurrences of "important" aquatic species and habitats intersect. • Supplement the data and information specified in this part through consultations with Federal, State, regional, local, and affected Native American tribal agencies (e.g., the U.S. Fish and Wildlife Service and/or the National Marine Fisheries Service and State resource agencies). (2) Review construction activities and discuss the following impacts on aquatic ecology: (a) Determine how construction activities would affect "important" species and their habitats (e.g., those resulting from scouring and siltation, dredging and soil disposal, exposure to physically and chemically altered habitat, altered hydrology, and interference with shoreline processes), and estimate the magnitude and duration of such impacts. (b) Determine the impacts of construction on Federally threatened or endangered species and/or Federally designated critical habitat, evaluating these impacts relative to the local population and the total estimated population over the entire range of the species as noted in the literature. (c) Identify water bodies receiving construction effluents and the					

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	expected average and maximum flow rates, composition, and physical and chemical characteristics of these effluents (from ESRP 4.2). (d) Describe proposed construction best management practices for the amelioration of impacts (from the ER). For example, best management practices would be to avoid narrow reaches of water bodies and “important” habitats as sites for locating intake or discharge structures, and providing a zone of passage that permits normal movement of “important” species populations. (e) For important species having commercial or recreational value, estimate the magnitude and duration of the impact on the species and their habitats. (f) If “important” species or habitats occur in the project area, and the proposed project could adversely affect the species or habitat, consult: • with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service under Section 7 of the Endangered Species Act. If construction is authorized, then prepare a biological assessment. If construction is not authorized (e.g., Early Site Permit without a limited work authorization), a biological assessment should not be prepared. • with the National Marine Fisheries Service under the Magnuson-Stevens Fishery Conservation and Management Act concerning potential impacts on essential fish habitat. (g) Identify potential disturbances of benthic areas by:					

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	• placement of intake and discharge structures • channel modifications for navigation or flow control • placement and removal of cofferdams • construction of bulkheads, piers, jetties, seawalls, dikes, berms, basins, and storm sewers • direct dredging, including the area that may be affected by resulting siltation and turbidity. (h) Relate the critical life history and habitat needs of "important" aquatic species (e.g., seasonal requirements, migration routes, spawning areas, nursery grounds, and feeding and wintering areas) to the plant location and construction schedule and consider whether impacts are likely to be of short duration or otherwise reversible. (i) In analyzing such impacts, consider: • percent or magnitude of the water body cross section that might be obstructed by construction activity at any time • time and duration of such obstruction • potential changes to water quality caused by construction activities. (j) Identify potential clearing along reaches of streams, rivers, and other water bodies. • Identify water bodies where such habitat alterations would occur and indicate the extent of such changes. • Compare the area of altered habitat with the extent of remaining similar habitats in the region. (k) Identify potential dewatering effects on groundwater supply,					

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	wetlands (protected under Executive Order 11990 as amended by Executive Order 12608), and other aquatic habitats. • Determine the location and areal extent of any wetlands that would be drained or filled. • Assess the relative importance to the ecosystem of the affected wetlands by comparing them with the areal extent of similar wetlands in the region. • Evaluate the potential for reversibility of impacts via natural attenuation or wetland restoration following construction. (l) Identify disposal plans for dredged material and placement of fill material. • Identify the areal extent of any water bodies or wetlands that would receive dredge spoils during construction. • Consider the relative extent of similar water bodies and wetlands in the region, and in this context, analyze the importance of the impacted wetlands and water bodies to the ecosystem. (m) Ensure that aquatic species expected to become established in water bodies affected by the cooling system are identified. • Ensure that the applicant has described in the ER the aquatic species that are expected to become established in such water bodies. • Consider how these colonizations may affect aquatic species in adjacent water bodies (e.g., food chain effects) and wetlands in the site and vicinity. (n) In addition to the above analyses, consider any other site-					

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	specific construction impacts to aquatic ecosystems that can be predicted on the basis of construction and the local aquatic ecosystem, consulting with the reviewers for ESRPs 2.3, 2.4.2, 3.6, and 4.2 to identify such additional impacts. (o) Ensure that the initial evaluation of environmental impacts has been submitted by the applicant if the applicant wishes to accelerate the start of construction. • Ensure that an applicant wishing to accelerate the start of construction by early submittal of the ER has submitted in the ER an initial evaluation of environmental impacts based on an analysis of at least 6 months of field data related to the proposed facility. Ensure that the applicant has also submitted suitable projections of the remaining seasonal periods if information has already been provided on the critical life stages and biologically significant activities (e.g., spawning, migration) that increase the vulnerability of the potentially affected biota at the proposed site. • If the preceding step has been taken, the reviewer should ensure that the applicant makes a commitment to furnish, within 6 months of the time of filing, a final evaluation based on a full year of field data. • Applicant must show that the relevance of the information used in the monitoring program is appropriate and acceptable for the areal extent for the evaluation of impacts of construction on aquatic ecology. (p) Become familiar with the provisions of standards, guides, and agreements pertinent to the construction of nuclear power stations:					

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	• Refer to the "Acceptance Criteria" section of this ESRP for a list of the standards that are applicable to this environmental review. • As required by these provisions, consult with the reviewer of ESRP 2.3 and with the appropriate agencies (e.g., the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service and the State resource agencies) to ensure compliance with the applicable regulations. • Analyze construction activities in light of recognized best management practices.					
4.4 (Draft Rev. 0, March 2000)	Socioeconomic Impacts					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, and analytic, and written in plain language.					
4.4.1 (Draft Rev. 0, March 2000)	Physical Impacts					
	Acceptance criteria are based on meeting the relevant requirements for noise, dust, air pollution, and visual aesthetics of the following regulations:					
	The ER must comply with the requirements of Clean Air Act of 1970, as amended, with respect to air quality during construction activities.					
	The ER must comply with the requirements of 40 CFR 50-90 as related to National Primary and Secondary Air Quality					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Standards.					
	The ER must comply with the requirements of Noise Control Act of 1972, as amended, with respect to noise from construction.					
	The ER must comply with the requirements of 10 CFR 51.71 and 10 CFR 51.45 with respect to describing the significance or potential significance of physical impacts of plant-construction activities on nearby communities.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to economic and social impact of siting and construction activities.					
	The reviewer's analysis of construction impacts on the community should be linked to the environmental reviews directed by ESRPs 2.1, 2.2, 2.5.1, 2.5.2, 3.1 and 3.7 to ensure that the environmental factors most likely to be impacted by the proposed construction are adequately described. The reviewer should ensure that information presented in the applicant's ER is complete and accurate. The reviewer should recognize that physical impacts to a community from construction of a nuclear plant are not markedly different from any other large heavy construction project. With this in mind, the reviewer should take the following steps: (1) For any particular construction related activity, first consider the distribution of residents and transients who could be affected, including determination of sensitive use patterns (e.g., hospitals, residences, recreational areas) and the allowable limits of impacts.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(2) Identify the potential impacts on the community and predict their extent and magnitude, including impacts from dust, noise, shock from blasting, and polluting gases and particles. • Consider impacts in qualitative terms where the effect on the community is expected to be minor. • Where adverse impacts (i.e., impacts that should be mitigated or avoided) can be predicted, conduct a more detailed analysis and where practical, make quantitative estimates of the magnitude of the impacts. (3) Identify the applicant's commitments to mitigate the physical impacts. These include • wetting down roadways and construction sites • scheduling noisy operations during daytime hours • suppressing blast and shock effects by using mats. (4) Consider the major physical impacts of plant construction. The specific impacts should include the impact of construction on transportation and the aesthetic characteristics of the region. (5) Become familiar with the provisions of standards, guides, and agreements pertinent to the construction of nuclear power plants. (6) Refer to the "Acceptance Criteria" section of this ESRP for a list of those generally pertinent to this environmental review. (7) Consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies to verify that current, applicable regulations and guides are available. This should include, for example, consultation with the EPA and State and local agencies for current ambient air quality standards and air					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	pollutant levels and Occupational Safety and Health Administration guidelines and standards applicable to facility construction. (8) Verify that the applicant has made commitments to comply with these applicable regulations and guides. (9) Become familiar with general references on construction practices and impacts. (10) Examine proposed construction activities in light of recognized "good practice." The term "good practice" as used here refers to those activities that tend to mitigate noise levels and adverse construction impacts on the community.					
4.4.2 (Draft Rev. 0, March 2000)	Social and Economic Impacts					
	Acceptance criteria for including socioeconomic impacts during construction are based on meeting the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.45(c) with respect to the analysis of socioeconomic data.					
	The ER must comply with the requirements of 10 CFR 51.45(d) and 51.71(d) with respect to the analyses required in the development of the ER and EIS.					
	The ER must comply with the requirements of 10 CFR 52.18 with respect to reviewing applications for early site permits.					
	The ER must comply with the requirements of 10 CFR 52.81 with respect to reviewing applications for combined licenses.					
	The ER must comply with the requirements of 10 CFR 51 and 52 with respect to describing the significance or potential					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	significance of socioeconomic impacts of plant construction activities.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), addresses benefits and costs to nearby populations from construction activities.					
	The reviewer's analysis of the social and economic impacts of construction should be linked to the environmental descriptions provided by the reviewer for ESRP 2.5.2 (Community Characteristics). The reviewer should ensure that the environmental factors most likely to be impacted by plant construction are described in sufficient detail to permit assessment of the predicted impacts. Based on these descriptions, the reviewer should identify and analyze components of the regional and community social, political, and economic systems that would be potentially impacted. The reviewer should take the following steps: (1) From the full scope of potential impacts, determine the impacts that are minor and those that are likely to be adverse and thus need detailed analysis. • Where practical, develop quantitative measures of adverse impacts. • Consider all impacts identified during the analysis to the extent practical, in terms of location, duration, and magnitude. • Be aware that the duration of some impacts will be longer than the construction period and that the character of such impacts may be altered due to					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	completion of construction and dispersal of the construction labor force. • Confer with the reviewers for ESRP 4.1, Land-Use Impacts; 4.2, Water-Use Impacts; and 4.3, Ecological Impacts, to determine if any of the construction impacts identified under these sections are of sufficient social or economic consequence to be examined further under this plan. (2) Consider the socioeconomic impacts of construction on regional housing and public services such as safety, social services, tourism and recreation, public utilities, education, transportation, and offsite land use. (3) For analytical purposes, it is effective to categorize impacts into those directly resulting from plant construction and those resulting from the activities and demands of the construction labor force. Analyze the social and economic impacts directly associated with construction, as follows: • Estimate the annual value of the major categories of materials and services to be purchased within the region and compare that value with the estimated value of the materials and services that would have been produced without plant construction. • Estimate the annual construction labor force requirements (for each quarter year, if possible) over the construction period and compare them with the number of workers available from within the region. Where necessary, determine these requirements for the major construction crafts, using standard craft categories. • Identify the jurisdictions receiving significant tax					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	revenues derived from plant construction purchased services and materials. • Estimate the physical demands placed by plant construction on local public facilities and services (e.g., fire, police, sewer, and water) and compare these demands with existing facilities and services. • In consultation with appropriate reviewers, determine if any impacts identified under land-use, water-use, and ecological impacts require further analysis regarding social and economic consequences. Such impacts could include economic impacts of changes in visual quality or recreation resources. • Determine the families or households to be displaced by plant construction. Analysis should - determine any equitable compensation for relocation and include analysis of adequacy of mitigation plans - address socioeconomic effects of labor force mobility, and residential choices. (4) Analyze the socioeconomic impacts associated with the construction labor force, as follows: • From the previous estimates of construction labor requirements and the number of workers available within the region, predict the number of workers originating from within the region and the number of in-migrants. • Estimate the number of construction force in-migrants, and predict their temporal and geographic distribution. • Estimate the number of induced in-migrants, and predict their temporal and geographic distribution.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• Estimate the overall impact of in-movers and procurements of goods and services on regional income, employment, and population, and identify critical services and goods for the affected region. • Predict potential changes in regional housing patterns (e.g., introduction of mobile homes). • Estimate the additional level of public facilities and services required to support in-migrants as a function of their probable location. Types of facilities and services that should be considered include education, water and sewer, safety, health, welfare, transportation, and recreation. • Identify adverse traffic conditions caused by transportation of workers and materials to and from the site. • Identify the jurisdictions expected to receive significant tax revenues generated by the project payroll and induced economic activity. • Compare the total flow of tax revenues from the various sources associated with plant construction to the expenditures required to meet the additional demand for public facilities and services.					
4.4.3 (Draft Rev. 1, July 2007)	Environmental Justice Impacts					
	The acceptance criteria for environmental justice impacts during construction are based on the following:					
	The ER must comply with the requirements of 10 CFR 51.45(c) with respect to analysis of socioeconomic data					
	NRC specific policy on treatment of environmental justice matters can be found in "Policy Statement on the Treatment of Environmental Justice Matters in NRC Regulatory and Licensing					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Actions." Federal Register,69 FR 52040, August 24, 2004.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The Council on Environmental Quality (CEQ) guidance for addressing environmental justice (CEQ 1997) is not binding, but should be followed as appropriate.					
	The guidelines for specific information requirements for environmental justice determinations, which are described in Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Issues, Appendix D to Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-203, NRC Office of Nuclear Rector Regulation, Washington, D.C. (NRC 2004). NRR Office Office Instruction LIC-203 is revised periodically. Obtain the latest revision for current guidance. Information submitted by the applicant is adequate and meets the 10 CFR 51.45 requirements and NRR guidelines if it permits the identification of potential disproportionate and negative impacts on minority and low-income populations as required in that guidance.					
	The ER must comply with the requirements of Regulatory Guide 4.7, Rev. 2., General Site Suitability Criteria for Nuclear Power Stations (NRC 1998a), which specifies the avoidance of disproportionately high and adverse impacts on minority and low-income populations during plant siting.					
	To determine which impacts are likely to be of concern and, therefore, what environmental impact areas should be discussed, the reviewer should take the following steps: (1) Coordinate with the reviewers of ESRP 2.5.4 and ESRPs 4.1 through 4.6 to ensure that the appropriate impact areas are being discussed. (2) Examine the record of the National Environmental Policy Act					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(NEPA) public scoping process to determine whether appropriate environmental impact areas are being discussed with respect to environmental justice. ESRP 2.5.4 in particular discusses specific efforts that may have been made to interview representatives of minority communities and other regional contacts (such as social service agencies) having specific knowledge about the locations, resource dependencies, customs and practices, and pre-existing health and socioeconomic conditions of minority and low-income populations in the region. The results of this additional outreach, if any, should also be evaluated. (3) Contact the cognizant personnel of each affected State for sites located on or near State boundaries, or where transmission line routes, access corridors, or offsite areas pass through more than one State. (4) Analyze the potential impacts on minority and low-income populations. (a) Briefly describe pathways by which any environmental impact during construction may interact with cultural, economic, or human health circumstances that may result in disproportionate environmental impacts on minority and low-income populations. If there are none, so state, and provide a brief discussion of why the potential pathways do not result in impact. (b) Assess (qualitative or quantitative, as appropriate) the degree to which each minority or low-income population is disproportionately receiving adverse human health or environmental (including socioeconomic) impacts during construction as compared with impacts on the general population in the impacted area.					

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	(c) Assess the degree to which each minority and low-income population is disproportionately receiving any benefits compared with the general population. (d) Assess (qualitatively or quantitatively, as appropriate) the significance or potential significance of such environmental impacts on each minority and low-income population. Significance is etermined by considering the disproportionate exposure, multiple-hazard, and cumulative hazard onditions outlined in the Environmental Justice: Guidance Under the National Environmental Policy Act (CEQ 1997). (e) Discuss any mitigative measures for which credit is being taken to reduce environmental justice concerns. (f) When alternative sites are being evaluated, similar reviews should be conducted for each site, using reconnaissance-level data (see ESRP 9.3).					
4.5 (Draft Rev. 0, March 2000)	Radiation Exposure to Construction Workers					
	Acceptance criteria for the analysis and evaluation of radiation exposure to construction workers are based on the relevant requirements of the following regulations:					
	The ER must comply with the requirements of 10 CFR 20; 20.1301; 20.1302 with respect to public dose limits; or 10 CFR 20; 20.1001; 20.1201; 20.1203; 20.1204; and 20.1205, with respect to occupational dose limits requirements for summation of internal and external doses and the determination of the dose if construction workers need to be classified as radiation					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	workers.					
	The ER must comply with the requirements of 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents" with respect to design objectives for dose when construction workers are considered members of the public.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 8.8, Rev. 3, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable (NRC 1978b) with respect to methods of ensuring that the calculated occupational doses are as low as is reasonably achievable.					
	In the analysis of the potential radiation exposures to construction workers, the reviewer should first determine whether there is a need to consider radiological impacts to construction workers. The reviewer should consult the site and vicinity maps of ESRP 2.1 and the NRC list of operational nuclear facilities. If there are or will be no adjacent operating nuclear facilities during the proposed project construction period, the review should be terminated. The reviewer should prepare an input for the environmental impact statement (EIS) stating that there will be no expected radiation exposure to construction workers during construction of the proposed project. If the reviewer determines that there is or will be an adjacent operating nuclear facility during the construction period, the reviewer should take the following steps:					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(1) Identify the sources of radiation that will contribute to the radiation exposure of construction workers. • Base this identification on review of the adjacent nuclear facility description provided by the applicant as appropriate. • Consult with the reviewers of ESRP 3.1 and Chapter 12 of the applicant's PSAR, if available, and participate in or get information from reviewers who participate in the site visit to complete this portion of the analysis. Sources to be considered in this portion of the analysis have been identified in the "Data and Information Needs" of this ESRP. (2) Determine the source strength for each of the sources identified in Item 1, above. • Accomplish this determination by either direct reviewer calculation of these values or by analysis to validate and accept the applicant's data. • When the latter procedure is used, conduct this portion of the analysis by comparing the applicant's data with available data from similar systems. (3) From the information provided in the ER or PSAR (if available), determine the location, number, duration of stay, and possible shielding of construction workers. • If shielding is not practical, consider these workers to be occupationally exposed. Consult with the reviewer of ESRP 3.1 or Chapter 12 of the applicant's PSAR, if available, to confirm plant and station layout and					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	establish possible worker shielding factors and plant construction schedules through the site visit and consultation with the applicant. (4) Determine the radiation dose rates at the principal onsite locations where construction workers will be present and at locations where particularly high dose rates could be expected on the basis of the source strengths determined in Item 2 above. • Accomplish this either by direct reviewer calculation of these values or by analysis to validate and accept the applicant's data. Acceptable codes and methods include the following: - The SKYSHINE computer code, developed by Radiation Research Associates and available through the Radiation Shielding Information Center at Oak Ridge National Laboratory, is an acceptable code for calculating dose rate at distances from nitrogen-16 sources in BWR steam system components. - The GASPAR code, described in Regulatory Guide 1.109, Rev. 1, Calculation of Annual Doses to Man from Routing Releases of Reactor Effluents for the Purpose of Evaluating Compliance with CFR 50, Appendix I, (NRC 1976), is an appropriate code for calculating dose due to gaseous-effluent-plume immersion. • The dose rate may also be determined through comparison with measured results, such as those available in EPRI NP-243 and HASL-305. • When the applicant has used these codes or methods to predict dose rates, the reviewer's determination may					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	be limited to verification of the techniques of calculation and input. (5) Based on the doses or dose rates determined in Item 4 above, and the number, location, and duration of stay of construction workers determined in Item 3 above, determine the estimated individual and annual collective dose to construction workers at the proposed site. The reviewer's evaluation of radiation exposure to construction workers involves (1) a determination that the predicted doses are realistic and accurate and (2) an evaluation of the predictions with respect to the requirements of 10 CFR 20 for doses to individuals in restricted areas. The reviewer should take the following steps for estimating the doses to determine if predicted doses are realistic and accurate: (1) Analyze radiation sources and source strength. • Verify that all potential radiation sources associated with the adjacent nuclear facility have been identified and that their source strengths have been accurately predicted. • Determine this on the basis of a site visit to the adjacent facility and through comparison of the facility sources and source strengths with similar facilities. (2) Analyze the impacts on the work force. • Verify that the size of the projected work force is consistent with work force data from similar projects and that the locations of workers and the duration of their					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	stay at a particular location are consistent with the proposed plant layout, the schedule of construction, and the nature of the construction task. • Evaluate the realism of any radiation shielding factors proposed by the applicant to take credit for work force shielding provided by the plant structures under construction. (3) Evaluate dose rates and collective doses by verifying that • dose rates have been calculated on the basis of accepted computational models or on the basis of actual measurements. • dose rates or doses have been calculated for those site locations where principal concentrations of construction workers will be located and that appropriate work force/work duration data have been used. • the individual and collective doses to the construction work force are realistic and accurate. - When the evaluation establishes that there are significant differences in the determination of radiation exposure to construction workers and the applicant's determinations of radioactive exposure, consult with the applicant to determine the reasons for these differences. - Request that additional data be provided or that calculations be repeated until the reviewer and the applicant are in reasonable agreement about the estimated individual and collective doses. The reviewer should take the following steps to evaluate the predicted doses with respect to 10 CFR 20 requirements:					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• Determine whether public or occupational dose limits apply to construction workers. • If public dose limits apply, determine whether construction personnel will be monitored in accordance with 10 CFR 20.1302(a). • If occupational limits apply, determine whether monitoring of construction personnel under the requirements of 10 CFR 20.1205 is required. • Summarize measures necessary to meet the requirements of 10 CFR 20 and prepare input to appropriate EIS sections, identifying their merit. • When advised that such measures have been implemented, recalculate the construction-worker doses.					
4.6 (Draft Rev. 1, July 2007)	Measures and Controls to Limit Adverse Impacts During Construction					
	Acceptance criteria for the summary of measures to monitor and control adverse impacts during construction are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50.36b with respect to environmental conditions in an NRC license or permit for the protection of the nonaquatic environment. Such conditions can cover reporting, recordkeeping, and monitoring.					
	The ER must comply with the requirements of 10 CFR 51, Appendix A to Subpart A, with respect to discussion of alternatives and mitigating measures to avoid or minimize adverse impacts.					
	The ER must comply with the requirements of 10 CFR 52.24 with respect to issuing early site permits containing the conditions and limitations as the Commission deems					

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	appropriate and necessary.					
	Regulatory positions and specific criteria to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976) with respect to the inclusion of a construction-impact control program in an application.					
	The reviewer's analysis should include identification and tabulation of potentially adverse construction impacts, identification of the applicant's commitments that limit and control these impacts, and comparison of applicant commitments with the staff's list of impacts needing mitigation. The reviewer should take the following steps: (1) Identify and tabulate the impacts of construction (see reviewers for ESRPs 4.1.1 through 4.5) that are of sufficient severity to need mitigation (i.e., measures and controls to limit the impact). (2) List the applicant's commitments for mitigating the impact. (3) Based on consultation with appropriate staff reviewers, identify the applicant commitments that will satisfy the staff's concerns for mitigation. (4) When it is determined that there are no applicant commitments to control or limit an adverse impact, consult with reviewers for the appropriate ESRPs 4.1 through 4.5, the reviewers for ESRPs 9.4.1 through 9.4.3, and the EPM to identify and evaluate available mitigation measure(s). Also note those impacts for which no appropriate measures and controls to limit					

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	the impact can be identified. (5) Prepare a table similar to that shown in Table 4.6-1 to compare potentially adverse impacts of construction with the applicant's commitments and identify those adverse impacts that cannot be mitigated or for which mitigation is not practical. Following the analysis above (Steps 1-5), the reviewer should seek confirmation that (1) the tabulated impacts are adverse and that measures and controls to limit the magnitude of the impact are required, (2) the measures and controls are reasonable and specific, and (3) benefits/costs have been considered. To do this, the reviewer should take the following steps: (1) Confirm that the construction impacts, when considered on a site-specific basis, are adverse and should be mitigated. • Make this determination through consultation with the appropriate reviewers for ESRPs 4.1 through 4.5, and take into account experience gained in the review of other projects having similar impacts. • Ensure that adequate documentation is available to support the staff conclusions with respect to the nature and severity of those impacts requiring mitigation. (2) Confirm that the selected measures and controls to limit each impact have been evaluated to verify that a practical level of mitigation can be achieved by the methods and controls to be applied. • Confirm that each measure and control is reasonable (i.e., involves methods and techniques that are appropriate and achievable on a site-specific basis).					

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	• Confirm that the measures and controls are specific, unambiguous, and are structured such that their application and results can be verified through subsequent field reviews and inspections. (3) Confirm that environmental, economic, and social costs of the available measures and controls to limit adverse impacts have been balanced against the expected benefits to be achieved. • Consult with the appropriate benefit-cost reviewers in conducting this portion of the evaluation. Benefit-cost reviews cannot be used as a basis for noncompliance with NRC regulations. • Note that when mitigation techniques do not lead to an improvement in the overall benefit-cost ratio and if mitigation is not required by law, the impact may be accepted without mitigation and considered in the overall project benefit-cost balancing. (6) Document any construction-related reporting, recordkeeping, and monitoring requirements that should be included in any environmental protection plan attached to the proposed license or permit.					
4.7 (Draft Rev. 0, July 2007)	Cumulative Impacts Related to Construction Activities					
	Acceptance criteria for the summary of cumulative impacts associated with proposed construction activities are the following:					
	The ER must comply with the requirements of 10 CFR 51.10(a) with respect to NRC policy to voluntarily take account, subject to certain conditions, of the regulations of CEQ implementing NEPA. The CEQ regulations specify that an EIS discuss					

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	cumulative impacts [40 CFR 1508.25(c)(3)].					
	Regulatory positions and specific criteria to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976) with respect to the inclusion in an application of an assessment of (1) cumulative and projected long-term effects from the point of view that each generation is trustee of the environment for each succeeding generation, and (2) any cumulative buildup of radionuclides in the environment.					
	The reviewer's analysis should include identification and tabulation of potentially adverse cumulative impacts associated with construction of the proposed plant. The reviewer should take the following steps: (1) Identify past, present, and reasonably foreseeable Federal, non-Federal, and private actions that could have meaningful cumulative impacts with the proposed action. Review of the aggregate effects of past actions is needed to the extent that the review provides information regarding the proposed action (CEQ 2005). (2) Identify the geographic area to be considered in evaluating cumulative impacts. CEQ guidance is to use natural ecological or sociocultural boundaries (CEQ 1997). Possible geographic areas that could be used to determine the appropriate geographic area for a cumulative impact analysis are in Table 2-2 of CEQ (1997). (3) Identify and tabulate the cumulative impacts associated with construction of the proposed plant. Input should be obtained from the reviewers for ESRPs 4.1 through 4.5. CEQ guidance					

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	is that agencies should focus on cumulative impact information that is relevant to reasonably foreseeable significant adverse impacts, is essential to a reasoned choice among alternatives, and can be obtained without exorbitant cost (CEQ 2005). Cumulative effects may result from the accumulation of similar effects or the synergistic interaction of different effects (CEQ 1997).					
5.0 (Draft Rev. 0, March 2000)	Environmental Impacts of Station Operation					Exclude, Administrative
5.1 (Draft Rev. 0, March 2000)	Land-Use Impacts					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.1.1 (Draft Rev. 0, March 2000)	The Site and Vicinity					
	Acceptance criteria for the review of land-use impacts at the site of the nuclear-power station and in its vicinity are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to analysis requirements to be included in draft environmental impact statements (EISs) prepared by NRC.					
	The ER must comply with the requirements of 10 CFR 51, Appendix A(7), with respect to discussion in EISs prepared by					

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	NRC of possible conflicts between alternatives and the objectives of applicable land-use plans.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	There are no conflicts between the applicant's proposed facility and the objectives of Federal, State, regional, and local (and in the case of proposed location on a reservation, Native American tribal) land-use plans and the Federal sources shown in Table 4.1.1-1 (plus comparable State sources).					
	or					
	If there are or are likely to be conflicts, the extent of the conflicts, the possibilities of resolving the conflicts, and the seriousness of the impact of the applicant's proposal on land-use plans and policies and the effectiveness of land-use control mechanisms for the area can be adequately evaluated and discussed in the EIS or other environmental document.					
	Land-use impacts to the site and vicinity because of construction are covered in ESRP 4.1.1 and limited portions of land-use impacts on the vicinity are covered in ESRPs 4.1.3, 4.3.1, 4.4, 5.3.3.1, and 5.3.3.2. As a general rule, the land-use changes considered in the staff's environmental reviews of construction impacts (ESRP 4.0) are sufficient to cover most land-use impacts on the site and vicinity due to the physical presence of the plant. Such land-use changes on the site will not be altered during subsequent plant operation, and thus the above referenced analyses of these changes should suffice for plant operation. For example, where plant construction preempts the exploitation of mineral resources, the analysis of this impact as prepared by the reviewer for ESRP 4.1.1 should be used because the operational impact is only an extension in time of the construction impact. This ESRP should be limited to those direct restrictions on land					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	use in the site vicinity resulting from plant operation. When assessing the impacts of plant operation on land use in the vicinity, the reviewer should take the following steps: (1) Using the results of the related reviews, assess the probable impacts of plant operation on crops or other vegetation or on transportation systems to establish if any would be severe enough to result in a change in land-use patterns in the site vicinity. (a) Realize that the impacts on land use resulting from plant operation are primarily those related to salt drift from cooling tower or spray pond operation and are thus limited in scope. (b) Explore all possibilities of “special case” land-use impacts (e.g., operational impacts to floodplain land use and reallocation of irrigation water to plant cooling water), but specific instructions for such special cases are not provided in this ESRP. (2) Using the predictions of drift and plume from the cooling system (ESRP 5.3.3.1), establish the areas in which there is potential for fogging, icing, or drift damage (ESRP 5.3.3.2) of sufficient magnitude to result in potential land-use changes. (a) Add the additional land area potentially changed to the area already committed by plant construction (ESRP 4.1.1). (b) Conduct an analysis as outlined in ESRP 4.1.1, preferably as a part of the analysis called for in ESRP 4.1.1. (3) Plants with once-through cooling systems have no general impacts on land use because of plant operation; nevertheless,					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	conduct a limited inquiry to reveal any site-specific or unusual impacts. The evaluation of land-use impacts at the site or in the vicinity resulting from station operation should follow the procedures outlined in the "Review Procedures" of ESRP 4.1.1 for any additional land area potentially changed beyond that land area committed because of plant construction.					
5.1.2 (Draft Rev. 1, July 2007)	Transmission Corridors and Offsite Areas					
	Acceptance criteria for the review of land-use impacts at the site of transmission corridors and offsite areas are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to analysis requirements to be included in draft environmental impact statements (EISs) prepared by NRC.					
	The ER must comply with the requirements of 10 CFR 51, Appendix A(7), with respect to discussion in EISs prepared by NRC of possible conflicts between alternatives and the objectives of applicable land-use plans.					
	Regulatory positions and specific criteria necessary to meet regulations identified above are as follows:					
	There are no conflicts between the proposed transmission corridors and the objectives of Federal, State, regional, and local (and in the case of proposed location on a reservation, Native American tribal) land-use plans and the Federal sources shown in Table 4.1.1-1 (plus comparable State sources)					
	or					
	If there are or are likely to be conflicts, the extent of the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	conflicts, the possibilities of resolving the conflicts, and the seriousness of the impact of the proposal on land-use plans and policies and the effectiveness of land-use control mechanisms for the area can be adequately evaluated and discussed in the EIS or other environmental document.					
	The analysis of the land-use impacts of operation of the transmission corridors, access corridors, and offsite areas is an extension of the analysis conducted under the review of ESRP 4.1.2. The same considerations outlined in the Review Procedures of ESRP 4.1.2 should apply. Additional considerations include land-use restrictions or changes that could occur because of maintenance practices, access-corridor use, noise, or electric or magnetic fields. The reviewer should conduct the evaluation of land-use impacts in transmission corridors and other offsite areas resulting from station operation using the procedures outlined in ESRP 4.1.2.					
5.1.3 (Draft Rev. 0, March 2000)	Historic Properties					
	Acceptance criteria for the review of historic properties that could be impacted by proposed operation are based on the relevant requirements of the following regulations:					
	36 CFR 800 defines the process by which a Federal agency meets its requirements under Section 106 of the National Historical Preservation Act (NHPA) to ensure that agency assisted or agency licensed undertakings acknowledge the effects of the undertakings on historic properties that are eligible for listing in the National Register of Historic Places. Compliance will be necessary for any new construction or					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	ground disturbing modifications during the operational phase.					
	Regulatory positions and specific criteria to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Section 110 of the NHPA, which deals with agency responsibilities for ensuring that historic preservation is fully integrated into ongoing programs and missions of Federal agencies. The NRC is responsible for ensuring compliance with Section 110 of the NHPA during operation of the plant.					
	The ER must comply with the requirements of Nuclear Reactor Regulation (NRR) Office Letter No. 906, Revision 1 (NRC 1996), which includes guidance for complying with the requirements contained in the NHPA pertaining to protection and preservation of significant historic properties during operation of the plant. NRR Office Letter No. 906 is revised periodically. Obtain a copy of the latest revision for current guidance.					
	The reviewer's analysis and evaluation of operational impact on historic and archaeological resources should be based on the concurrent review of construction impacts (ESRP 4.1.3). Only the impacts of operation that differ from those resulting from construction need be assessed. In this respect, a temporal extension of an impact from the construction phase through the operational life of the project is not a different impact. Where the reviewer determines that the impacts of operation on cultural and historic resources have been adequately considered by the review directed by ESRP 4.1.3, no further review should be required. If the reviewer determines that there will be an impact of operation that would not have been considered by the reviewer for ESRP 4.1.3 (e.g., the impact of					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	the visual plume from a cooling tower), the reviewer should assess that operational impact as an extension of the review directed by ESRP 4.1.3.					
5.2 (Draft Rev. 0, March 2000)	Water-Related Impacts					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.2.1 (Draft Rev. 0, March 2000)	Hydrologic Alterations and Plant Water Supply					
	Acceptance criteria for the review of the hydrologic alterations at the proposed plant sites are based on the relevant requirements of the following regulations:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction activities.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to the National Pollutant Discharge Elimination System (NPDES) permit conditions for discharges, including storm water discharges.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole source aquifer.					
	The ER must comply with the requirements of 40 CFR 227 with respect to criteria for evaluating environmental impacts.					
	The ER must comply with the requirements of Federal, State, regional, local, and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Supreme Court granted the States additional authority to limit hydrological alterations beyond the States' role in regulating water rights.					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water-use, and water-quality issues.					
	This section of the environmental impact statement (EIS) should be planned to accomplish the following objectives: (1) public disclosure of the hydrologic alterations resulting from plant operation and the comparison of plant water needs with water availability, (2) a discussion of the effects of these alterations and water supply/need comparisons, and (3) presentation of staff conclusions regarding the adequacy of plant-water supply to meet plant-water needs. The reviewer's analysis of hydrologic alterations and water supply/water consumption comparison should be linked to the environmental descriptions provided by the environmental reviews for ESRPs 2.3 and 3.3 to ensure that the environmental factors most likely to be affected by operational hydrologic alterations and plant water consumption are described in sufficient detail to permit subsequent assessment of any potential impacts. The reviewer should coordinate the analysis of hydrologic alterations with the analysis prepared by the reviewer for ESRP 4.2.1 because the analyses for many of the hydrologic alterations resulting from plant construction will be sufficient to cover subsequent (period of plant operation) alterations due to the physical presence of the plant. Where these alterations will not be further changed by plant operation, the analysis prepared by the reviewer for Section 4.2.1 should suffice for plant operation. This environmental					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	review should be limited to consideration of hydrologic parameters directly associated with plant operation. The reviewer's identification of plant operational activities that could result in hydrologic alterations will require knowledge of the site and vicinity physiography, hydrology, and water uses. In addition, the reviewer should be familiar with Federal, State, regional, local, and Native American tribal regulations with respect to hydrology and water use. When evaluating hydrologic alterations resulting from plant operation and the adequacy of the water sources proposed to supply plant water needs, the reviewer should take the following steps: (1) Consider appropriate plant operating conditions (including periods of maximum plant water use, minimum water availability, average plant operation by month and during shutdown) and hydrologic variations affecting water use. (2) Determine if all known future water uses (including aquatic ecosystems) have been considered. (3) Estimate the effects of operational hydrologic alterations and restrictions on water availability on these users. (4) Identify and analyze any measures proposed by the applicant to minimize or limit these alterations and restrictions. (5) When analyzing water availability, coordinate this review with the reviewer for ESRP 3.3.1. (6) When analyzing hydrologic alterations, coordinate this review					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	with the reviewer for ESRP 4.2.1 to ensure that the reviewer is aware of the scope and extent of these related reviews and to avoid any duplication of effort. (7) In consultation with the reviewer for ESRP 2.3.1, establish the physical availability of the proposed water sources, including consideration of the drought of record for the region and the 7-day once-in-10-years low flow. (8) In consultation with the reviewer for ESRP 2.3.2, identify the other water uses, rights, and restrictions of the surface waters and groundwaters, including existing station water uses (e.g., an operating steam electric plant). (9) In consultation with the reviewer for ESRP 3.3.1, determine plant needs for the following plant operating conditions: maximum water consumption, minimum water availability, average operation by month, and plant shutdown. (10) Establish by comparison the adequacy of the water supply to accommodate anticipated plant operating modes. (11) Analyze all operational activities that can alter the quantities of water physically available in nearby hydrologic systems and determine the alterations. • Consider all water to be used during operation, under various plant operating (ESRP 3.3.1) and hydrologic (ESRP 2.3.1) conditions. • Consider all water diversions that change the quantities of water in various parts of water systems (e.g., permanent dewatering) and water rights or allocations obtained for the plant.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• Determine the physical effects (e.g., altered well yields, water levels relative to intake pipes) likely to affect other water users and aquatic ecosystems for those hydrologic systems in which alterations in water quantities have been identified. (12) Analyze the operational activities that can alter hydrologic geometries, flow and circulation patterns, and mixing processes and determine the alterations. Hydrologic alterations due to the intake or discharge system are covered in ESRPs 5.3.1.1 and 5.3.2.1. • Consider other hydrologic alterations (e.g., maintenance dredging, permanent dewatering) with the potential for impacts to water users. • Report any operational activity that will result in hydrologic alterations to the floodplain to the EPM and to the reviewer for ESRP 5.2.2. • Analyze and evaluate such alterations in accordance with the instructions provided the reviewer for ESRP 4.2.1. (13) Analyze the operational activities that can alter erosional, depositional, and sediment transport characteristics and determine the alterations. (Note that alterations resulting from intake or discharge system operation are addressed by the reviewers for ESRPs 5.3.1.1 and 5.3.2.1). • Consider operational activities in relation to the natural processes that would occur in the absence of plant operation. • For those areas in which alterations in the natural erosional, depositional, and sediment transport					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	characteristics have been identified, determine the physical effects (e.g., beach erosion, increased turbidity) likely to affect other water users. (14) Ensure that those operational activities resulting in hydrologic alterations have been identified, and seek confirmation that those alterations resulting in environmental impacts have been described in sufficient detail to allow for the subsequent analysis and assessment of these impacts. (15) Evaluate the adequacy of plant water supplies with respect to plant water needs, using the following evaluation procedures: • Determine if the identified alterations in water quantity in the various operationally affected hydrologic systems are compatible with existing and known future water rights and allocations. • Describe the physical effects of identified alterations in the quantity of water available to other consumptive water users. • Describe the physical effects of altered hydrologic geometry, flow, and circulation patterns in relation to non-consumptive water users. When proposed operational activities involving hydrologic alterations to the floodplain are identified, complete the evaluation of these alterations in accordance with the evaluation instructions of Section 4.2.1. • Describe the physical effects of altered erosional, depositional, and sediment characteristics in relation to other water users, to property and (for those effects not addressed by the reviewers of ESRPs 5.3.1.1 and 5.3.2.1) to aquatic biota. • Determine if the sources of water proposed to supply					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	plant-water needs will be adequate for these needs, taking into account seasonable variations in water supply and the variations in water needs as a function of operating conditions. If the sources are determined to be inadequate under some conditions, describe the conditions, including seasonal/plant operating-mode factors, the estimated time duration of the inadequacy, and the predicted effect on plant operation.					
5.2.2 (Draft Rev. 0, March 2000)	Water-Use Impacts					
	Acceptance criteria for the water-use impacts at the proposed plant sites are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to permit conditions for discharges, including stormwater discharges.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole source aquifer					
	The ER must comply with the requirements of Federal, State, regional, local, and affected Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S. Supreme Court granted the States additional authority to limit hydrological alterations beyond the States' role in regulating water rights. As a result of this ruling, the States may regulate the quantity of water as a part of the definition of water quality.					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs including hydrology, water-use, and water-quality issues.					
	The review conducted with this plan should be directed toward accomplishing the following objectives: (1) public disclosure of major direct water-use consequences of plant operation, (2) presentation of the basis for the staff analysis, and (3)					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	presentation of staff evaluations, conclusions, and conditions regarding water use. The reviewer should coordinate this input with the reviewer of ESRP 5.2.1 to avoid duplication. The reviewer's analysis of operational impacts on water use should be linked to the environmental descriptions provided by ESRPs 2.3 and 3.3 to ensure that the environmental factors most likely to be impacted by the proposed plant operation are described in sufficient detail to permit assessment of the predicted impacts. The reviewer should coordinate this analysis with the reviewer for ESRP 2.3.3 and with the reviewers for ESRPs 5.3.2.2 and 5.5 to identify and analyze those water-quality changes affecting water use. The reviewer should also coordinate this review with the analysis of construction impacts described in ESRP 4.2.2 because the analyses for many of the water-use changes considered in the staff's environmental review of construction impacts will be sufficient to cover subsequent (period of plant operation) impacts due to the physical presence of the plant. Where these changes will not be further altered by plant operation, the plant construction impact analyses (environmental standard) will suffice for plant operation. This environmental review should be limited to consideration of the impacts on water use that are direct results of plant operation. Unless the reviewers for ESRP 2.3 indicate a potential for operational water-use impacts along transmission corridors or at offsite areas, this review may be limited to potential site and vicinity water-use impacts.					
	Site Visit					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	During the site visit, the reviewer should • Observe the general pattern of water use at the site and vicinity and at those identified offsite and transmission corridor areas where operational activities could be expected to impact water use. • Identify those water users and water-use areas that should be considered. • Consult with appropriate nearby Federal, State, regional, local, and affected Native American tribal agencies for further identification of water users, water-use areas, or water-quality considerations that should be analyzed. • Consider appropriate plant operating conditions (including periods of maximum plant water use, minimum water availability, average plant operation by month and shutdown water requirements) and hydrologic variations in analyzing potential water-use impacts.					
	<u>Areas of Impact</u> The reviewer should evaluate the impacts of water use on water availability, hydrologic alterations, and water quality.					
	<u>Water Availability</u> When addressing water availability, the reviewer should take the following steps: (1) Ensure that the water users and water-use areas potentially impacted by alterations in water quantity and availability as a result of plant operation have been identified and that any					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	impacts of reduced water quantity and availability have been identified and assessed. • Make this assessment through consultation with the reviewers for ESRPs 5.1 and 5.8 and, where necessary, with the assistance of nearby Federal, State, regional, local, and affected Native American tribal agencies. • When adverse impacts have been identified, consult with the reviewer for ESRP 5.2.1 for assistance in identifying design or procedure modifications that could mitigate the impact. (2) Ensure that the possibility for conflicts between proposed plant water use and existing and known future water rights and allocations has been considered and that the probable nature and extent of these conflicts has been described. (3) Ensure that any transfer of water rights (e.g., from irrigation use to plant consumptive use) has been described and that the impacts associated with such transfers have been identified and assessed.					
	<u>Hydrologic Alterations</u> When addressing hydrologic alterations, the reviewer should take the following steps: (1) Ensure that the hydrologic alterations identified by the reviewers for ESRPs 5.2.1, 5.3.1.1, and 5.3.2.1 have been analyzed with respect to their potential impacts to water users or water-use areas. • Compare the effects of these alterations (e.g., turbidity, erosion, sedimentation) with preoperational conditions					

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	to assess the extent of the impact. • Evaluate impacts for individual water users and for water-use areas. • Consult with the reviewer for ESRP 5.5 for assistance in this evaluation and to coordinate the overall evaluation of operational impacts due to hydrologic alterations. • When necessary, consult with Federal, State, regional, local, and affected Native American tribal agencies for assistance. • Seek means to mitigate or avoid any identified adverse impacts. (2) Seek confirmation that any operational activities affecting a floodplain or wetland have been described by the reviewer for ESRP 5.2.1. • Consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies to determine the extent to which such activities will conform with applicable floodplain and wetlands standards. • Consult with the reviewer for ESRP 5.2.1 and the reviewers for ESRP 9.4 to analyze alternatives to any such activity affecting a floodplain or wetland. (3) Ensure that operational activities that will alter or restrict surface oriented water uses (e.g., commercial and recreational fishing or navigation) have been identified and that their effects on water users have been described. • Ensure that structurally related impacts on surface oriented water use (e.g., breakwaters or jetties having impacts to navigation) have been addressed by the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	reviewer for ESRP 4.2.2. • Identify and assess any operational impacts (e.g., altered current velocities associated with cooling water discharges) that would increase or modify these structurally related impacts. • Seek confirmation that identified hydrologic alterations resulting from plant operation comply with applicable Federal, State, regional, local, and affected Native American tribal standards and regulations. • Consider site- and region-specific water-use type, frequency, and magnitude because many of the impacts resulting from hydrologic alterations do not permit development of specific criteria for determining adversity. • When potential adverse impacts are predicted, identify alternative designs or operating procedures that could mitigate the impacts.					
	<u>Water Quality</u> When addressing water quality, the reviewer should take the following steps: (1) Ensure that hydrologic alterations and operational activities affecting water quality have been identified and their effects on water users or water-use areas described. (2) Consult with the reviewers for ESRPs 2.3.2 and 2.3.3 to ensure that potentially affected water users have been identified and that baseline water-quality data for the affected users and water bodies are available. (3) Evaluate impacts on the basis of altered water quality, taking into account the nature of the impact, the time duration or time					

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	periods when the impact will be experienced, the number of water users or extent of water-use areas affected, and the water-quality requirements of the affected users or areas. • Consult with the reviewer for ESRPs 5.3.2.2 and 5.5 to coordinate this evaluation and to avoid duplication of effort with other ESRP Chapter 5.0 reviewers. • When necessary, consult with Federal, State, regional, local, and affected Native American tribal agencies for assistance in evaluating the identified impacts. • When adverse impacts have been identified, seek alternative operational procedures to avoid the impact. (4) Consult with the reviewers for ESRP 3.6 to determine the flow rates and chemical composition of plant effluents. Consider potential impacts on water users or water-use areas in terms of the intended usage (e.g., chemical contaminants affecting a municipal water supply, suspended solids affecting industrial use, turbidity affecting recreational use). (5) Determine if operational activities affecting surface-water and groundwater quality will comply with Federal, State, regional, local, and affected Native American tribal agency water-quality standards for effluents and receiving water bodies. This evaluation should be made in consultation with the reviewer for ESRP 5.5 to avoid any duplication of effort in the evaluation of water-quality impacts.					
5.3 (Draft Rev. 0, March 2000)	Cooling System Impacts					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the					

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	following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.3.1 (Draft Rev. 0, March 2000)	Intake System					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.3.1.1 (Draft Rev. 1, July 2007)	Hydrodynamic Descriptions and Physical Impacts					
	Acceptance criteria for the hydrodynamic physical impacts at the proposed plant sites are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.45 with respect to ERs and the analysis of potential impacts contained therein.					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to descriptions of the environment affected by the issuance of a construction permit, early site permit, or combined license.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the preparation of supplemental environmental impact statements (EISs) in support of the issuance of an operating license.					

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	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of Clean Water Act with respect to Section 316(b) and Section 401.					
	The ER must comply with the requirements of 40 CFR 122 and 125 with respect to National Pollutant Discharge Elimination System (NPDES) permit conditions.					
	The ER must comply with the requirements of Rivers and Harbors Appropriation Act of 1899.					
	The ER must comply with the requirements of Federal, State, regional, local, and affected Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Reports for Nuclear Power Stations (NRC 1976), provides guidance on the format and content of ERs including hydrology, water-use, and water-quality issues.					
	The ER must comply with the requirements of LIC-203, Revision 1, Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Impacts (NRC 2004), with respect to NRC compliance with the Endangered Species Act.					
	The reviewer's description of intake hydrodynamics should be linked to the environmental descriptions provided by ESRPs 2.3.1 2.3.2, 2.3.3, 3.3, and 3.4 to ensure that water body characteristics affecting intake hydrodynamics are described in sufficient detail to allow prediction of the flow field induced by the operation of the intake system. The reviewer's analysis of physical impacts of intake system operation should be linked to the environmental descriptions and impact analyses of ESRPs 2.4.2, 5.3.1.2, 5.3.2.1, 5.4.1, and 5.4.2 to ensure that those environmental factors most likely to be affected are described in sufficient detail to permit assessment of the predicted changes or impacts. The extent of the description of intake hydrodynamics and analysis of physical impacts should be governed by the magnitude of potential intake system impacts to aquatic biota.					
	<u>Intake-Hydrodynamic Description</u> The reviewer should take the following steps to develop a description of the intake hydrodynamics: (1) Conduct a simple independent hydrodynamic analysis (e.g., calculate of the induced potential flow field by standard procedures and prepare an intake system hydrodynamic description.					

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	• The reviewer needs to determine the range of low water surface elevation at the intake. Guaranties of future water commitments from upstream dam operators are necessary to bound the operational conditions for the intake before velocities at the intake can be computed. • Discuss this with reviewers for ESRPs 2.4.2 and 5.3.1.2 to determine its adequacy for use in predicting intake system impacts to aquatic biota. • When determined that the induced flow fields would result in only minor impacts on aquatic biota (or that no biota would be affected), this portion of the analysis is complete. (2) When it is determined that the simple hydrodynamic analysis is insufficient (e.g., the analysis results in predictions of significant adverse impact; there are large populations of "important" aquatic biota in the vicinity of the intake), prepare a detailed independent analysis of intake hydrodynamics consisting of • a review of any applicant supplied flow field predictions or • a reviewer prepared prediction of the induced flow field based on modeling procedures. - Consult with the reviewers for ESRPs 2.4.2 and 5.3.1.2 to determine the extent of the surface-water body to be analyzed. - Consult with the reviewers for ESRP 5.3.2.1 and ESRP 5.4.2 to ensure that the area of the water body to be analyzed is sufficient to permit analysis of potential					

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	recirculation of discharged cooling water, if applicable. - Provide a quantitative description of the induced flow field taking into account the ambient currents. - Provide velocity vectors or other descriptors showing the areal extent of the region affected by the induced flow field.					
	<u>Physical Impacts of Intakes</u> The reviewer should take the following steps to analyze the physical impacts of the intake system: (1) Identify and analyze physical changes resulting from intake system operation, including • shoreline erosion • bottom scouring • induced turbidity • silt buildup • alterations of stratification patterns. Staff experience has indicated that the impacts listed above are generally minor. However, impacts to other resources (e.g., aquatic ecology) may be more significant. Impact findings in this ESRP are limited to those not covered in other ESRPs. (2) Unless adverse impacts have been identified, no further evaluation is required. The reviewer should ensure that the description of the intake flow field is adequate to serve as a basis for the impact assessment of ESRPs 5.3.1.2, 5.4.1, and 5.4.2. and for providing flow patterns necessary for the assessment of potential heated water					

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	recirculation conducted in ESRP 5.3.2.1. The reviewer should ensure that analyses involving mathematical or physical modeling of intake flow fields are appropriate for the specific situation being modeled, have been verified or shown to be conservative, and are documented and referenced. The reviewer should consider the procedures of Regulatory Guides 4.4, Reporting Procedure for Mathematical Models Selected for Predict Heated Effluent Dispersion in Natural Water Bodies (NRC 1974), and 1.125, Rev. 1, Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (NRC 1978), in making this evaluation. However, reviewers should be aware that these documents are dated and may not represent current standard engineering practice in some areas. For analyses involving less detailed procedures than mathematical or physical models, the reviewer should ensure that the procedures used by the applicant were appropriate for the specific situation and were adequately conservative. For specific physical impacts identified by the "Review Procedures" section, the reviewer should evaluate each impact with regard to water standards and guides or good operating procedures for intake systems. Unless potentially severe impacts have been identified, no further evaluation is required.					
5.3.1.2 (Draft Rev. 1, July 2007)	Aquatic Ecosystems					
	Acceptance criteria for the review of operation impacts on aquatic resources in the vicinity of the site and transmission corridors are based on the relevant requirements of the following:					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	The ER must comply with the requirements of 10 CFR 51.45 with respect to ERs and the analysis of potential impacts contained therein.					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to descriptions of the environment affected by the issuance of a construction permit, early site permit, or combined license.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the preparation of supplemental environmental impact statements (EISs) in support of the issuance of an operating license.					
	The ER must comply with the requirements of 40 CFR 122 and 125 with respect to NPDES permit conditions specified in the Federal Water Pollution Control Act, commonly referred to as the Clean Water Act.					
	The ER must comply with the requirements of Coastal Zone Management Act, as amended, with respect to natural resources and land or water use of the coastal zone.					
	The ER must comply with the requirements of Endangered Species Act, as amended, with respect to identifying Federal threatened and endangered, and/or Federally designated critical habitats, and initiating formal or informal consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service.					
	The ER must comply with the requirements of Federal Water Pollution Control Act, as amended, commonly referred to as the Clean Water Act, with respect to restoration and maintenance of the chemical, physical, and biological integrity of water resources.					
	The ER must comply with the requirements of Fish and Wildlife					

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	Coordination Act, as amended, with respect to consideration of fish and wildlife resources in the planning of development projects that affect water resources.					
	The ER must comply with the requirements of Magnuson-Stevens Fishery Conservation and Management Act, as amended, with respect to identifying impacts on essential fish habitat (EFH) in the vicinity of the site and initiating consultation with the National Marine Fisheries Service.					
	The ER must comply with the requirements of Marine Mammal Protection Act, as amended, with respect to the protection of marine mammals.					
	The ER must comply with the requirements of Marine Protection, Research, and Sanctuaries Act, as amended, with respect to the dumping of dredged material into the ocean.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance to the applicant concerning the analysis of potential impacts of operation of the cooling water intake system. The reviewer should ensure that the applicant's analysis is sufficient to evaluate impacts during station operation.					
	Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998) contains guidance concerning the ecological systems and biota at potential sites and requires that their environs be sufficiently well known to allow reasonably certain predictions of impacts and that there would be no unacceptable or unnecessary deleterious impacts on populations or habitats of important species or on ecological systems from the operation of a nuclear power station. This					

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	guide also provides regulatory positions concerning entrainment, impingement, entrapment, and effects of cooling systems on aquatic species, their habitats, and their migration routes.					
	Compliance with environmental quality standards and requirements of the Clean Water Act is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will conduct its own assessment.					
	Memorandum of Understanding Between the U.S. Army Corps of Engineers and the NRC for the Regulation of Nuclear Power Plants (40 FR 37110) provides guidance with respect to the NRC exercising the primary responsibility in conducting environmental reviews and in preparing EISs for nuclear power stations. The Corps of Engineers should be consulted regarding (1) coastal erosion and other shoreline modifications, (2) siltation and sedimentation processes, (3) dredging activities and disposal of dredged materials, and (4) location of structures affecting navigable waters.					
	Second Memorandum of Understanding and Policy Statement Regarding Implementation of Certain NRC and EPA					

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	Responsibilities, serves as the legal basis for NRC decisionmaking concerning licensing matters covered by the National Environmental Policy Act (NEPA) and Section 511 of the Federal Water Pollution Control Act , commonly referred to as the Clean Water Act.					
	The ER must comply with the requirements of LIC-203, Revision 1, Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Impacts (NRC 2004), with respect to NRC compliance with the Endangered Species Act.					
	The impacts from cooling water intake are regulated through the National Pollutant Discharge Elimination System (NPDES) permit system. The Clean Water Act requires that the location, design, construction, and capacity of the cooling water intake structure reflect the best technology available for minimizing environmental impacts. Responsibility for making this determination rests with the EPA or with its designees. However, compliance with environmental quality standards and requirements of the Clean Water Act is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider mitigation measures and alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will					

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	conduct its own assessment. In the most practical terms, the reviewer's final evaluation is determined through professional judgment based on the pertinent data and analyses. The reviewer may refer to earlier NRC environmental reviews in which evaluation of intake system operational impacts has been important. The reviewer should take the following steps depending on whether or not the new facility is being located at a site close to an existing nuclear facility. If the facility is located at a site close to an existing nuclear facility: Determine whether the applicant has a current NPDES permit with a Clean Water Act Section 316(b) determination, if appropriate, or equivalent State permits and supporting documentation. If these documents are not available, not current, or do not reflect conditions associated with the proposed facility, continue the analysis below for a site that is not located close to an existing nuclear facility. Otherwise, prepare an assessment of entrapment, entrainment or impingement of aquatic biota for the new plant based on the records of historical data of the existing facility emphasizing the "important" aquatic organisms. The statement for the EIS would: • summarizes the permitting documents that have been reviewed • compares the estimated future entrapment, entrainment and impingement losses from the new facility to the entrapment, entrainment and impingement losses from the existing facility					

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	• discusses the differences in the siting, orientation and structure between the existing and new facilities • evaluate the potential cooling water intake system impacts for entrapment, entrainment, or impingement on aquatic species. If the facility is not located at a site close to an existing nuclear facility: (1) Identify the "important" aquatic organisms and their life stages susceptible to entrapment, impingement, or entrainment, coordinating efforts with the reviewer of ESRP 2.4.2 to ensure that these susceptible "important" species are also described in that ESRP. If "important" aquatic species are present and are susceptible to entrapment, entrainment, or impingement, and effects would neither be detectable nor noticeably alter or destabilize population levels, then continue the analysis at Step (2). Otherwise, prepare a statement for the EIS describing the potential for entrapment, entrainment, or impingement of aquatic species that • summarizes the permitting information, species data, and methods for quantifying entrainment, entrapment, and impingement data that have been reviewed • states there are no populations of aquatic species present in the vicinity of the site that would be entrained, entrapped, or impinged by the cooling water intake system to the point where changes in their population levels are detectable • states that design and operation meet Clean Water Act Section 316(b) Phase I guidelines.					

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	(2) Estimate the levels of susceptibility in either qualitative or quantitative terms, or both. Methods for quantifying entrapment and impingement susceptibilities are not well developed; therefore, it may be necessary to draw on the experience of comparable, currently operating power stations to predict the magnitude of the potential impact for the proposed plant. Methods for quantifying entrainment susceptibilities are available; however, they are generally applicable to specific habitat species station characteristics. • Ensure that assumptions made in available model developments are valid for the case under review. • Consider habitat type in determining levels of susceptibility. (3) After identifying the "important" species and determining their susceptibility, estimate the survival rates for those species entrapped, impinged or entrained by relying on experience at other stations. Certain species have been shown to be especially fragile (e.g., threadfin shad, menhaden, and bay anchovy), whereas some shellfish are much hardier (e.g., blue crab and penaeid shrimp). • Consider the design and proposed operation of any proposed screen wash and fish return system. • Consider the potential value of such a system, if a return system is not proposed. • Assume 100% mortality for all entrained biota. (4) Consider the potential for altered hydrodynamic characteristics induced by inlet system operation (e.g., altered circulation patterns) to affect attraction and entrapment of aquatic					

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	biota, and consult with the reviewer for ESRP 5.3.1.1 to determine the extent and seasonal variation of any such hydrological alterations. (5) Consult with the reviewer for ESRP 5.3.2.1 to determine if there is any potential for the recirculation of heated effluent from the plant discharge system. If recirculation is predicted, analyze the potential effects of increased impacts of entrapment, entrainment, and impingement. (6) Finally, estimate the magnitude of the potential entrapment, impingement and entrainment impacts on the species populations and the aquatic ecosystem. • Use the results of Step 2 as the starting point (i.e., the potential station cropping rates for phytoplankton, zooplankton, and meroplankton, including vegetative spores, fish eggs and larvae, and juvenile stages of "important" species). • Consider these cropping rates in relation to natural mortality rates, reproductive rates, and standing stock estimates for the species populations. • Consider other existing stresses (cumulative mortality) to the fragile species (e.g., impacts of other electrical generating stations sited nearby). In general, the entrainment cropping of phytoplankton and zooplankton would not affect these communities due to the short reproductive cycles for these species. More detailed consideration should be given those species with annual reproductive cycles, such as most fish and shellfish. The reviewer may assume, for a first approximation, that					

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	entrainment cropping translates directly to a reduction in the harvestable or parent stocks. Where possible, this impact should be expressed in quantitative units such as (1) catch per unit effort, (2) harvestable stock by weight, (3) recruitment in numbers, (4) dollar values, and (5) numbers or percentages of specific size, age group, or life stage. The reviewer may use more refined analyses (e.g., population modeling or compensation factors) when results suggest that additional precision is needed.					
5.3.2 (Draft Rev. 0, March 2000)	Discharge System					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.3.2.1 (Draft Rev. 0, March 2000)	Thermal Description and Physical Impacts					
	Acceptance criteria for the review of thermal impacts at the proposed plant sites are based on the relevant requirements of the following regulations:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with					

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	respect to National Pollution Discharge Elimination System (NPDES) permit conditions for discharges.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of Federal, State, regional, local, and affected Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. If no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will determine the impact.					
	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S. Supreme Court granted the States additional authority to limit					

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	hydrological alterations beyond the State's role in regulating water rights.					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs including hydrology, water-use, and water-quality issues.					
	The reviewer's analysis of the thermal discharges should be linked to the environmental descriptions provided by ESRPs 2.3, 2.4.2, 2.7, 3.3, and 3.4 to ensure that the physical environmental factors most likely to be impacted by the proposed plant operation are described in sufficient detail to permit assessment of the predicted impacts. The reviewer should take the following steps: (1) Coordinate with the reviewer for ESRP 5.3.2.2 to ensure that those biotic environmental factors (e.g., aquatic biota) most likely to be impacted by the thermal discharge are described in sufficient detail to permit assessment of the predicted changes or impacts. If the proposed plant is to be located at a station with an existing generating plant and the proposed plant thermal discharges will be mixed with thermal discharges from the existing plant, limit the analysis (and subsequent evaluation) to the incremental impacts resulting from operation of the proposed plant. (2) Determine dilution factors at specific receiving water body locations when requested to do so by the reviewers for ESRPs 5.4 or 5.5. (3) Consider impacts that may result from operation of the following:					

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	• once through cooling systems starting at the condenser discharge • cooling towers, including helper towers, starting at the point of the cooling tower water blowdown • spray canals, including helper spray canals, starting at the point of the spray canal water blowdown • cooling lakes and multi-purpose cooling ponds, starting at the point of the condenser discharge • cooling ponds used only for heat dissipation, starting at the point of pond discharge to receiving water bodies. (4) Scale the scope of the analysis to the level of the anticipated impacts. • If the thermally affected discharge area will be relatively small and have low ecological impacts, then use simple methods of analysis and conservative assumptions. • If the available data indicate a significant potential for problems, such as development of a thermal block, recirculation of heated effluent to the cooling water intake and thermal buildup, discharge plumes attaching to shorelines, violation of thermal standards, or important impacts to biota, then perform a hydrothermal analysis sufficient to produce a sound basis for evaluating the potential environmental impacts. (5) Base analysis of the hydrothermal data on the applicant's mathematical and/or physical models and on field or tracer studies performed by the applicant. • Consult Regulatory Guides 4.4, Reporting Procedure for Mathematical Models Selected to Predict Heated					

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	Effluent Dispersion in Natural Water Bodies (NRC 1974) and 1.125, Rev. 1, Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (NRC 1978), to analyze the applicant's mathematical or physical models. • If the reviewer's evaluation of these data verifies the validity of the applicant's approach and results, this should constitute an adequate independent analysis. • If the reviewer is unable to verify the applicant's results by this method, perform an independent assessment, using the methods described below. (6) Select an appropriate modeling procedure based on the following considerations: (1) the type of outfall and discharge characteristics, (2) physical characteristics of the receiving water bodies, (3) hydrological flow regimes, (4) hydrodynamic characteristics of the receiving water, (5) water-use patterns in the vicinity of the station, (6) quantity and temperature of the effluents, (7) meteorology, and (8) thermal assimilative capacity of the receiving waters. • See EPA (1993) and Fisher et al. (1979) for discussions on the applicability of a variety of mathematical thermal discharge models. • Also consider new models or improved existing models when selecting a mathematical model. (7) Assess physical changes resulting from the discharge system operation, including shoreline erosion, bottom scouring, increased turbidity and siltation. • If no severe impacts can be predicted, no further analysis is necessary.					

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	• If potentially severe impacts are identified, consider using mathematical modeling or physical modeling to quantify them. (8) Determine compliance with applicable regulations. • Where required, consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies. • Become familiar with the provisions of the Second Memorandum of Understanding between NRC and EPA.					
5.3.2.2 (Draft Rev. 0, March 2000)	Aquatic Ecosystems					
	Acceptance criteria for the review of impacts to aquatic ecosystems from the discharge system are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.45 with respect to ERs and the analysis of potential impacts contained therein.					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to analysis of impacts to the terrestrial environment affected by the issuance of a construction permit.					
	The ER must comply with the requirements of 10 CFR 52, Subpart A, with respect to analysis of impacts to the terrestrial environment affected by the issuance of an early site permit.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the preparation of supplemental environmental impact statements (EISs) in support of the issuance of an operating license.					

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	The ER must comply with the requirements of 40 CFR 122 with respect to EPA administered programs, especially the National Pollutant Discharge Elimination System (NPDES).					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent guidelines and thermal standards.					
	The ER must comply with the requirements of Coastal Zone Management Act of 1972 with respect to natural resources, and land or water use of the coastal zone.					
	The ER must comply with the requirements of Endangered Species Act of 1973, as amended, with respect to identifying threatened or endangered species and critical habitats and formal or informal consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service.					
	The ER must comply with the requirements of Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act (CWA), Amendments of 1972, Sections 402 and 316[a]), with respect to restoration and maintenance of the chemical, physical, and biological integrity of water resources.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act of 1958 with respect to consideration of fish and wildlife resources and the planning of development projects that affect water resources.					
	The ER must comply with the requirements of Marine Mammal Protection Act of 1972 with respect to the protection of marine animals.					
	The ER must comply with the requirements of Marine Protection, Research, and Sanctuaries Act of 1972 with respect to the dumping of dredged material into the ocean.					
	The ER must comply with the requirements of Rivers and					

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	Harbors Appropriations Act of 1899 with respect to the deposition of debris in navigable waters, or tributaries to such waters.					
	Regulatory guidance and specific criteria to meet the requirements identified above are presented in the following guidance documents:					
	Compliance with environmental quality standards and requirements of the Clean Water Act is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will conduct its own assessment and use it in its determination of the overall benefit-cost balance.					
	The ER must comply with the requirements of Memorandum of Understanding between the Corps of Engineers, U. S. Army, and the USNRC for the Regulation of Nuclear Power Plants, with respect to the NRC exercising the primary responsibility in conducting environmental reviews and in preparing EISs for nuclear power stations. However, the Corps of Engineers will participate with the NRC in the preparation of EISs by helping to draft material for sections covering (1) coastal erosion and other shoreline modifications, (2) siltation and sedimentation					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	processes, (3) dredging activities and disposal of dredged materials, and (4) location of structures affecting navigable waters.					
	Regulation of impacts from cooling system discharges is accomplished via the NPDES permit system administered by the EPA and the permitting States under Sections 316(a) and 402 of the CWA. The CWA requires that discharge system operation must ensure the protection and propagation of a balanced, indigenous population of shellfish, fish, and wildlife in and on the receiving water body. Responsibility for making this determination (or for reassigning the responsibility) rests with the EPA. Discharge system impacts on aquatic biota may result from the effects of thermal, chemical, and physical alterations to the receiving water body. Major alterations are usually confined to a limited discharge area (the mixing zone), whereas lesser alterations may extend over a larger portion of the receiving-water body. Adverse effects on biota that are transported through, migrate through, or are attracted to the mixing zone may be acute or chronic, and impacts may be reflected as changes in the populations of “important” species and in the structure and function of the ecosystem. The reviewer should take the following steps to evaluate the impacts of the plant’s discharge system: (1) Determine whether the applicant has provided a current NPDES permit with a 316(a) determination (if required) or equivalent State permits and supporting documentation. If these documents are not available, are not current, or do not reflect conditions during the license-renewal term, continue the analysis at Step (2). Otherwise, prepare a statement for the SEIS					

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	describing the potential for discharge impacts to aquatic biota at the site that • summarizes the permitting documents reviewed • states that the required current NPDES permit and 316(a) determination are available and current • concludes that there are no discharge impacts to aquatic organisms that may occur as a result of plant-cooling-system discharges to receiving water bodies. (2) If "important" aquatic species are present and are susceptible to heat shock resulting from plant-cooling-system discharges to the receiving water bodies such that the effects will be detectable or may destabilize or noticeably alter population levels, then continue the analysis at Step (3). Otherwise, prepare a statement for the SEIS describing the potential for thermal impacts to aquatic biota at the site that • summarizes the permitting information, species data, and methods for quantifying thermal stresses due to heat shock to aquatic biota that have been reviewed • states that there are no populations of "important" aquatic biota present in the vicinity of the site that will be adversely affected by plant-cooling-system thermal discharges to the point where changes in their population levels are detectable • concludes that, because aquatic biota populations will remain stable even if some are affected by heat shock, the cooling-system discharge impacts on aquatic biota are SMALL within the context of the analysis in NUREG-1437 and that mitigation is not warranted.					

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	(3) Determine and assess the levels of potential biological impacts. • Consider the biological effects of thermal, chemical, and physical alterations to the receiving water body on the identified "important" aquatic species, including combined effects (e.g., thermal plus chemical effects) and the potential for gas-bubble disease. • Give particular attention to the relationship of these stresses to life history requirements (e.g., growth, reproduction, migration). • Evaluate the discharge system impacts of the plant as described below. Procedures for reviewing specific impacts of thermal, chemical, and physical alterations are listed below. Analyze the impacts for the parameter when considered alone and the impacts for the parameter when combined with other parameters. The review should be based on general habitat types such as • rivers and streams • lakes and reservoirs • estuaries • seacoast.					
	<u>Thermal Effects</u> The reviewer should consider species in the vicinity of the station and their susceptibility to thermal effects. (1) Consider the following: • maximum sustained temperatures for each season that are consistent with maintaining desirable levels of					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	productivity • maximum levels of metabolic acclimation to warm temperatures that will permit return to ambient winter temperatures if artificial sources of heat cease • temperature limitations for survival of brief exposures to temperature extremes, both upper and lower • if spawning or nursery areas are affected, restricted temperature ranges for various stages of reproduction, including (for fish) gonad growth and gamete maturation, spawning migration, release of gamete, development of the embryo metamorphosis, emergence, and other activities of early life stages, such as commencement of independent feeding by juveniles, and temperature required • thermal limits for diverse compositions of species of aquatic communities, particularly where nuisance growths of certain organisms create reduction in diversity or where important food sources or chains are altered • thermal requirements of downstream aquatic life where upstream warming of a cold-water source will adversely affect downstream temperature requirements • areal extent of the plume • percent of unaffected area • physical concentrating factors. (2) Identify the most thermally intolerant "important" species expected to be affected. (3) Quantify the magnitude of potential thermal impacts to the aquatic ecosystem. (4) Evaluation of thermal impacts, addressing the following					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	recommendations: • Growth of aquatic species should be maintained at levels necessary for sustaining actively growing and reproducing populations if the maximum weekly average temperature in the zone inhabited by the species at that time does not exceed one-third the range between the optimum temperature and the ultimate upper incipient lethal temperature of the species, and the temperatures above the weekly average do not exceed the criterion for short term exposures. • After the specific limiting temperatures and exposure times have been determined by studies tailored to local conditions, the reproductive activity of selected species should be protected in those areas in which (1) temperature regimes required for gonad growth and maturation are preserved, (2) no temperature differentials are created that block spawning migrations, although some delay or advancement of timing based upon local conditions may be tolerated, (3) temperatures are not raised to a level at which necessary spawning or incubation temperatures of winter spawning species cannot occur, (4) sharp temperature changes are not induced in spawning areas, either in mixing zones or in mixed water bodies (the thermal and geographic limits to such changes will be dependent upon local requirements of species, including spawning microhabitat, e.g., bottom gravels, littoral zone, and surface strata), (5) timing of reproductive events is not altered to the extent that synchrony is broken where reproduction or rearing of certain life stages is shown to be dependent upon cyclic					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	food sources or other factors at remote locations, and (6) normal patterns of gradual temperature changes throughout the year are maintained. • Nuisance growths of organisms may develop where there are increases in temperature or alterations of the temporal or spatial distribution of heat in either the receiving water bodies (e.g., rivers, lakes) or in onsite cooling ponds. Some nuisance conditions may be created by operation of cooling ponds that may not affect receiving water body biota, but that may affect the aesthetic quality of the site and vicinity. The reviewer should consider such factors (e.g., odors from algal or macrophyte growth and decomposition) in making this evaluation. There should be careful evaluation of all factors contributing to nuisance growths at any site before establishment of thermal limits based upon this response, and temperature limits should be set in conjunction with restrictions on certain other factors (e.g., eutrophication).					
	Chemical Effects The reviewer should consider species in the vicinity of the station and their susceptibility to chemicals released. (1) Consider the following parameters: • acute toxicity • chronic toxicity • accumulation • biomagnification • sublethal and behavioral effects. (2) Determine if applicant needs to perform bioassays for					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	important chemicals such as copper, chlorine, or related components, and scale inhibitors based on site-specific conditions. (3) Compare the concentrations of chemicals at the discharge points with concentrations of the same chemicals in ambient waters. • Consider dilution and mixing of chemical discharges. • Obtain estimates of concentrations at various distances from the release point. • Assess the effects of variable environmental and plant operation conditions on injury or mortality of suspectable organisms. • Determine the potential for bioconcentration, biomagnification, and interacting effects for certain chemicals. (4) Determine the biological losses from chemical stress based upon • plume configuration • time and concentration • worst and average conditions. (5) Determine if losses of either resident or migratory species will occur given proposed specifications for chemical releases. (6) Evaluations of chemical impacts should address the following: • the possible environmental effect of certain chemicals, like chlorine (hypochlorite), chlorination byproducts,					

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	other biocides, and scale and corrosion inhibitors • alternatives to the biocide treatment of condenser tubing.					
	Physical Effects The reviewer should consider species in the vicinity of the station and their susceptibility to physical effects. (1) Consider the following parameters: • reduction in density, species composition, and community structure of the benthos • loss or alteration of habitat • alteration of migratory pathways. (2) Consider the potential effects of the following on habitat loss and species composition • altered current patterns • current velocity • littoral drift • scouring • siltation • increased turbidity • gas supersaturation (gas-bubble disease) • low dissolved oxygen • predation • parasitism • disease among organisms exposed to sublethal stresses. (3) Note effects associated with loss or alteration of habitat and					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	the resultant potential reduction in species composition and community structure. (4) Evaluation of physical impacts should address the following: • potential loss or alteration of unique habitat • potential effects of altered migratory pathways • potential effects of other biotic changes.					
5.3.3 (Draft Rev. 0, March 2000)	Heat-Discharge System					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.3.3.1 (Draft Rev. 0, March 2000)	Heat Dissipation to the Atmosphere					
	Acceptance criteria for the review of the impacts of heat dissipation on the atmosphere are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to the review of environmental issues associated with heat dissipation to the atmosphere.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the post construction review of environmental issues associated with heat dissipation to the atmosphere.					
	The ER must comply with the requirements of 10 CFR 52.18 with respect to review of environmental issues associated with					

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	heat dissipation to the atmosphere for early site permits.					
	The ER must comply with the requirements of 10 CFR 52.89 with respect to review of environmental issues associated with heat dissipation to the atmosphere for combined licenses.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The reviewer should ensure that heat dissipation system impacts have been identified and described in sufficient detail to enable the reviewers for ESRPs 5.3.3.2 and 5.8.2 to evaluate and assess the environmental effects resulting from heat dissipation system. The reviewers for these plans should be consulted as part of this evaluation.					
	The staff used operational data to review several potential environmental impacts associated with cooling systems. The results of these reviews are presented in NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NRC 1996), and codified for use in environmental reviews associated with license renewal in 10 CFR 51.					
	The reviewer should analyze the applicant's estimates of the atmospheric effects of cooling system operation. The reviewer should consult with the reviewers for ESRPs 2.2.1, 2.5.3, and 3.1 to determine those locations for which analyses should be performed. (1) Evaluate the potential impacts on transportation caused by fogging and icing on the basis of the predicted additional hours of fogging and icing resulting from heat dissipation system. • When these additional hours represent a significant fraction of the naturally occurring hours (determined by					

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	the reviewer for ESRP 2.7), and the affected transportation routes will be used by the general public, identify and evaluate means to mitigate the impact. (2) Compare predictions of the occurrence of plume interaction with • existing pollutant sources • weather modification in terms of cloud development • shadowing • humidity increases • increased precipitation due to cooling tower plume or drift with operating experience at other sites. (3) Evaluate unusual heat dissipation system impacts (e.g., drift deposition on switch yards and other structures) not considered by the reviewers for ESRPs 5.1.1, 5.1.3, 5.3.3.2, and 5.8.1, and identify and evaluate means to avoid or mitigate any such impacts that are sufficiently adverse to warrant this action. (4) For spray canals, existing literature values for drift deposition rates may be used. Drift from a cooling pond or lake need not be considered. (5) Use the following references to find appropriate models for conducting any additional analyses needed: • See Hanna et al. (1982) and Hanna (1984) for information on the atmospheric impacts of heat dissipation. • See Carhart et al. (1982) for an evaluation of models that predict the rise and length of plumes from natural draft cooling towers. The best models of the period					

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	predict the visible plume rise within a factor of 2 and plume length within a factor of 2.5 about 50% of the time. • See Carhart and Policastro (1991) for a more recent model for natural draft and mechanical cooling towers that predicts the plume rise within a factor of 2 about 75% of the time and visible plume length within a factor of 2.5 about 70% of the time. • See Carhart et al. (1992) for the use of this model in predicting the long shadowing and resultant decrease in solar radiation caused by cooling tower plumes. • See Policastro et al. (1994), which extends the description to use of the model for estimating seasonal and annual cooling tower impacts, including drift deposition, icing, and fogging. (6) Perform independent analysis of additional hours of ground level fogging, icing, drift, humidity increase, and deposition of pollutants generated by offsite sources. • The need for this analysis will depend on the level of the potential impact, the level of confidence in the applicant's model, and the extent, applicability, and representative nature of the available meteorological data and observational experience at operating stations. • Coordinate this analysis with the reviewers for ESRPs 5.1.1, 5.1.3, 5.3.3.2, and 5.8.1 to ensure that appropriate heat dissipation system factors are considered and to avoid duplication of any environmental analyses. (7) For an independent analysis, use the following procedure:					

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	• For towers, use hourly onsite meteorological data, tower performance specifications, and an appropriate model to generate information on the spatial distribution of the elevated plume, annual plus seasonal and/or monthly estimates of ground level fogging, icing, and drift deposition as a function of distance and direction from the tower. These data should be compared with the meteorological data provided by the reviewer for ESRP 2.7 to determine the additional amount of ground level fogging and icing and to calculate the amount of drift deposition for the appropriate site-vicinity locations. • For cooling systems employing spray canals or a cooling pond, assume the following: • - The plume will exist as ground level fog, but will evaporate within 300 m or lift to become stratus for wind speeds greater than 2.2 m/sec. - The plume will exist as fog over the pond, lifting to become stratus for winds less than or equal to 2.2 m/sec.					
5.3.3.2 (Draft Rev. 0, March 2000)	Terrestrial Ecosystems					
	Acceptance criteria for the review of impacts on terrestrial ecosystems from the heat dissipation system are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.45 with respect to ERs and the analysis of potential impacts contained therein.					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to analysis of impacts on the terrestrial					

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	environment affected by the issuance of a construction permit.					
	The ER must comply with the requirements of 10 CFR 52, Subpart A, with respect to analysis of impacts on the terrestrial environment affected by the issuance of an early site permit.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the preparation of supplemental environmental impact statements (EISs) in support of the issuance of an operating license.					
	The ER must comply with the requirements of Endangered Species Act of 1973, as amended, with respect to identifying threatened or endangered species and critical habitats and formal or informal consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act of 1958 with respect to consideration of fish and wildlife resources and the planning of development projects that affect water resources.					
	Regulatory guidelines and specific criteria to meet the regulations and identified above are as follows:					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance for the preparation of ERs. With respect to the heat-dissipation system, it specifies that detailed descriptions of the expected effects of the system on the local environment with respect to fog, icing, precipitation modifications, humidity changes, cooling-tower blowdown and drift, and noise should be included in the ER. The reviewer should ensure that the appropriate data and analyses are provided in the ER.					
	Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998), contains guidance on					

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	factors that should be considered in the site-selection process. In specific regard to cooling-tower drift, this guide states "The potential loss of important terrestrial species and other resources should be considered."					
	Regulatory Guide 4.11, Rev. 1, Terrestrial Environmental Studies for Nuclear Power Stations (NRC 1977), contains technical information for the design and execution of terrestrial environmental studies, the results of which may be appropriate for inclusion in the applicant's ER. The reviewer should ensure that the appropriate results concerning potential effects of the heat-dissipation system on the terrestrial environment are included in the ER.					
	The depth and extent of the input to the EIS will be governed by the environmental characteristics of the terrestrial ecology that could be affected by operation of the stationGs heat dissipation systems and by the magnitude of the expected impacts to the terrestrial environment. The most apparent effects of heat dissipation systems on terrestrial ecosystems are those associated with cooling-tower or spray pond operation. These include the effects of vapor plumes, icing, and salt drift on the terrestrial ecosystems. The potential for bird collision with cooling towers should be addressed by the reviewer for ESRP 4.3.1. To date, at stations using once through cooling systems, no adverse impacts to terrestrial ecosystems have occurred that require mitigating actions. In circumstances where once through cooling is proposed, the analysis may terminate without further consideration unless unusual environmental circumstances make more analysis necessary. (1) Consider the impacts of drift deposition on plants.					

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	• Drift deposition has the potential for adversely affecting plants, but the tolerance levels of native plants, ornamentals, and crops are not known with precision. • General guidelines for predicting effects of drift deposition on plants suggest that many species have thresholds for visible leaf damage in the range of 10 to 20 kg/ha/mo of NaCl deposited on leaves during the growing season. • These effects can be altered by the frequency of rainfall, humidity, type of salt, and sensitivity of species. • Use maps of the site and vicinity showing drift isopleths that were produced by recognized drift-dispersion models to define areas of possible botanical injury. • Use an order-of-magnitude approach, as follows, to analyze operational impacts from salt drift: - Deposition of salt drift (NaCl) at rates of 1 to 2 kg/ha/mo is generally not damaging to plants. - Deposition rates approaching or exceeding 10 kg/ha/mo in any month during the growing season could cause leaf damage in many species. - Deposition rates of hundreds or thousands of kg/ha/yr could cause damage sufficient to suggest the need for changes of tower-basin salinities or a reevaluation of tower design, depending on the amount of land impacted and the uniqueness of the terrestrial ecosystems expected to be exposed to drift deposition. (2) Consider the detrimental effects increased fogging could have on local vegetation if the increase in humidity induces an increase in fungal or other phytopathological infections.					

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	Increased icing can cause physical damage to vegetation due to increased structural pressure on tree branches or by damaging fruit or leaf buds. • Use an order of magnitude approach as follows to analyze operational impacts from fog or ice: - Fogging or icing of vegetation on the order of a few hours per year is generally not severe. - Fogging or icing on the order of tens of hours per year may cause detectable damage to vegetation. - Fogging or icing occurring for hundreds of hours per year could be severe enough to suggest the need for design changes, depending on the amount of land impacted and the uniqueness of the terrestrial ecosystems expected to be exposed to drift deposition. • Consider soil salinization: - The risk from this source is generally considered to be low. - In arid areas (deserts), salts could accumulate in soils over long time intervals and cause damage. (3) Consider the impact to terrestrial biota when new shoreline habitats are created along ponds and reservoirs built for cooling purposes. Riparian tree/shrub communities that form around these new ponds or reservoirs may attract "important" species. If endangered or threatened species could be affected, agency					

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	level formal or informal consultation with the U.S. Fish and Wildlife Service under Section 7 of the Endangered Species Act is required.					
5.3.4 (Draft Rev. 1, July 2007)	Impacts to Members of the Public					
	Acceptance criteria for the analysis and evaluation of the nonradiological health impacts of the cooling system on humans are based on the following:					
	The ER must comply with the requirements of 10 CFR 51.45 with respect to ERs and the analysis of potential impacts contained therein.					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to descriptions of the environment affected by the issuance of a construction permit, early site permit, or combined license.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the preparation of supplemental environmental impact statements (EISs) in support of the issuance of an operating license.					
	Regulatory positions and specific criteria are as follows:					
	The Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NUREG-1437) (NRC 1996) contains an analysis of the effects of cooling system discharges on thermophilic microorganisms that have the potential to adversely affect human health. This analysis can provide guidance to the staff in determining the significance of the potential effects of these discharges and the extent of the analysis required.					
	The review procedures for impacts from etiologic agents are					

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	discussed separately from the procedures for impacts from noise.					
	Etiologic Agents (formerly Thermophilic Microorganisms) Consideration of the impact of etiologic agents on the public health is important, especially for those plants using cooling ponds, lakes, canals, or small rivers because the operation of these plants may significantly increase the presence and numbers of harmful waterborne diseases. Additional information regarding these organisms can be found in the Appendix to this ESRP. The following review procedures should be used: (1) Review of available data, site description, and cooling system description, to determine whether a potential exists of a detrimental impact from the thermal discharges on the concentration levels of deleterious etiological agents. If this potential exists, then further analysis of any available data would be appropriate, especially if public recreation occurs within the vicinity of the discharge or if the plant is located in the southern regions of the United States. The minimum review should include: • Consultation with the State Public Health Department. • Review of any records associated with waterborne disease outbreaks in the region. (2) If it appears to be likely that thermal discharges from the plant would increase the number of deleterious etiologic agents to levels that could cause a public health problem, the applicant should be requested to consider mitigative measures to minimize the potential impacts. • Mitigative measures may include:					

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	- setting up and executing a monitoring program for etiologic agents or other harmful biological agents to insure acceptable levels. - limiting public activities that allow contact with discharge waters in the vicinity of the site. - the use of respirators and protective clothing by plant workers to protect against mists from cooling towers or dusts inhaled during cleaning processes or limiting maintenance activities on the cooling system to times when the structures or components are dewatered. • The reviewer should analyze any mitigative measures and forward them to the reviewer for ESRP 5.10. (3) Irrespective of the plant cooling system design or the type of station discharge water body, if there has been an outbreak of waterborne disease during the previous 10 years in the vicinity of the site, at the minimum, mitigative measures may include: • Consultation with the State Public Health Department. • In the absence of monitoring data, consideration should be made of limiting public activities that allow contact with discharge waters in the vicinity of the site. • The use of respirators and protective clothing by plant workers to protect against mists from cooling towers or dusts inhaled during cleaning processes or limiting maintenance activities on the cooling system to times when the structures or components are dewatered.					
	<u>Noise</u>					

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	The primary responsibility of regulating noise was transferred to the State or local government level in 1982 and, as a result, the review of cooling system impacts will require familiarity with the applicable State and local requirements. When noise levels are below the levels that result in hearing loss, impacts have been judged primarily in terms of adverse public reactions to the noise. The principal sources of noise from plant operations include natural-draft and mechanical-draft cooling towers. Other occasional noise sources may include auxiliary equipment, such as pumps to supply cooling water from a remote reservoir. Generally, power-plant sites do not result in offsite noise levels greater than 10 dB(A) above background (NRC 1996). Noise level increases larger than 10 dB(A) would be expected to lead to interference with outdoor speech communication, particularly in rural areas or low-population areas where the day-night background noise level is in the range of 45 to 55 dB(A). Surveys around major sources of noise, such as major highways or airports, have found that when the day-night level increases beyond 60 to 65 dB(A), noise complaints increase significantly. Noise levels below 60 to 65 dB(A) are considered to be of small significance (NRC 1996). More recently, the impact of noise was considered in the Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, Supplement 1 Regarding the Decommissioning of Nuclear Power Reactors (NRC 2002). In that document, the criterion for assessing the level of significance was not expressed in terms of sound levels. Rather, the level of significance was based on the effect of noise on human activities and threatened or endangered species. The criterion in NUREG-0586 Supplement 1 (NRC 2002) is stated as follows:					

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	The noise impacts ... are considered detectable if sound levels are sufficiently high to disrupt normal human activities on a regular basis. The noise impacts ... are considered destabilizing if sound levels are sufficiently high that the affected area is essentially unsuitable for normal human activities, or if the behavior or breeding of a threatened or endangered species is affected. (1) The reviewer should become familiar with the applicable State noise limits for residential areas and other types of land use. (2) The reviewer should determine whether the plant has or will have cooling towers or other components of the cooling system capable of contributing to offsite noise levels. • If no cooling towers or other noise-producing components of the cooling system are anticipated, the analysis is complete. • If cooling towers or other noise-producing components of the cooling system are present, the reviewer should compare the anticipated day night average level of noise determined at the site boundary (based on the dB(A-scale)) from the cooling system with applicable State noise limits. • If no State noise limits are available and if the day-night noise level is below 60 to 65 dB(A), no further analysis is needed. • If the noise levels exceed the State noise limits or in the absence of such limits if the day-night noise level exceeds 65 dB(A), the reviewer should request the applicant to propose measures for mitigating the					

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	impact from the noise. The reviewer should analyze these mitigation measures and forward them to the reviewer for ESRP 5.10.					
5.4 (Draft Rev. 0, March 2000)	Radiological Impacts of Normal Operation					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.4.1 (Draft Rev. 0, March 2000)	Exposure Pathways					
	Acceptance criteria for analyzing the radiological impacts of normal operations with respect to exposure pathways are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50, Appendix I, with respect to guidelines for assessing radiological impacts from normal operations. Note: This criterion is not applicable to early site permit applications.					
	The ER must comply with the requirements of 10 CFR 20.1301(d) with respect to exposure pathways.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 1.109, Rev. 1, Calculation of Annual Doses to Main from Routing Releases of Reactor Effluents for the Purpose of Evaluating Compliance with CFR 50, Appendix I (NRC 1976),					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	with respect to calculating individual and population doses from routine effluents.					
	In this analysis, the reviewer should identify potential pathways for the transfer of radioactive materials from the plant or plant effluent streams to individuals. The analysis should consist of two parts: (1) identification of the pathways leading to maximum individual dose commitments and (2) identification of the pathways that will be used to calculate the overall dose estimate due to plant operation. Figures 5.4.1-1 and 5.4.1-2 represent the usual pathways associated with the transfer of radioactive materials to individuals and other biota and should be used by the reviewer to determine on a site-specific basis the pathways of interest for the proposed plant or site. The following pathways should be considered: • direct radiation from the plant (for determining compliance with 10 CFR 20.1301(d)), including onsite independent spent fuel storage installations and onsite waste facilities • for gaseous effluents: - immersion in the gaseous plume - inhalation of iodines and particulates - ingestion of iodines and particulates through the milk cow, milk goat, meat animal, and vegetation pathways - radiation from iodines and particulates deposited on the ground. • for liquid effluents:					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	- drinking water - ingestion of fish and invertebrates - shoreline activities for water containing radioactive effluents. In addition, the reviewer should examine site-specific data to look for any unusual pathways uniquely associated with the proposed plant, and when any such exist, the reviewer should include them in the analysis.					
	<u>Pathways for Maximum Individual Doses</u> To identify the pathways leading to maximum individual dose commitments, take the following steps: (1) Based on information provided by the applicant, information obtained during the site visit, consultation with appropriate ESRP Chapter 2.0 reviewers, and consultation with appropriate Federal, State, regional, local, and affected Native American tribal agencies, • Develop a list of "nearest" receptors as described in this ESRP. • For each such location, categorize the important pathways (i.e., direct radiation or gaseous or liquid effluent) by which radiation can be transferred to the receptor. (2) For gaseous pathways • Give the location data to the reviewer for ESRP 2.7, Meteorology, for determining atmospheric transport and					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	diffusion characteristics needed to determine dose commitments at these locations. • Give the reviewer for ESRP 2.7 assistance as needed to complete this interrelated portion of the environmental review. Note: For early site permit applications, consult with the ESRP 2.7 reviewer to determine a conservative effluent release point for the hypothetical plant. (3) For liquid pathways • Consult with the reviewer of ESRP 3.5 to complete the analysis of the information required in this ESRP and with the reviewer of ESRP 2.3.1, to complete the analysis of transit and dilution times. When these reviewers determine that the applicant supplied values for transit time and dilution are conservative (e.g., with respect to stream flow and velocity), the applicant's data may be used without further analysis. • For early site permit applications, consult with the reviewers of ESRPs 2.3.1 and 3.5 to determine the optimum effluent release point given the applicant's general statements about proposed cooling systems. • If the applicant's data are not conservative or if subsequent dose-estimations calculations and analyses made by the reviewers of ESRPs 5.4.2 and 5.4.3 predict that doses will exceed the 10 CFR 50, Appendix I, guidelines, request that the reviewer of ESRP 2.3.1 provide detailed hydrological dispersion factors.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• As needed, provide assistance to the reviewer of ESRP 2.3.1 to complete this interrelated portion of the environmental review.					
	<u>Pathways for Overall Dose</u> To identify pathways that will be used to calculate the overall dose estimate, take the following steps: (1) Refer to the information identified in the "Data and Information Needs" of this ESRP. • Base the review on information supplied by the applicant and supplemented by information obtained during the site visit, consultation with appropriate ESRP Chapter 2.0 reviewers, and consultation with appropriate Federal, State, regional, local, and affected Native American tribal agencies. • Using these data, develop the appropriate exposure pathways, and document all assumptions. • Obtain assistance from and coordinate with the reviewer of ESRPs 2.3.1, 2.7, and 3.5 in the same manner as described for the review of maximum individual dose pathways. (2) Ensure that the analysis of exposure pathways has resulted in the following identifications and determinations: • the locations of all important receptors • the important exposure pathways to each receptor • atmospheric transport and diffusion calculations (by the reviewer for ESRP 2.7) at each appropriate receptor location					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• effluent release points, transit times to unrestricted area boundaries, and diluted stream flows at these boundaries (to be verified by the reviewer for ESRP 3.5). Note: For early site permit applications, certain assumptions may need to be made regarding effluent release points. • transit times and dilution factors at each appropriate receptor location (to be verified in consultation with the reviewer for ESRP 2.3.1) • population distribution data for 5 years following the time of the license action being considered (to be verified by the reviewer for ESRP 2.5.1) • present annual milk, meat, and vegetable production (to be verified in consultation with the reviewers for ESRP 2.2). (3) As a final step in the evaluation process, consult with the reviewer of ESRP 5.4.2 to ensure that sufficient data have been provided to permit calculation of individual and population dose commitments.					
5.4.2 (Draft Rev. 0, March 2000)	Radiation Doses to Members of the Public					
	Acceptance criteria for the analysis and evaluation of doses resulting from radioactive gaseous and liquid effluents released during normal operations are based on relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50, Appendix I, with respect to determination of doses.					
	The ER must comply with the requirements of 10 CFR 50.34a with respect to determination of estimated dose.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the requirements of 10 CFR 20.1301(d) with respect to doses to members of the public as a result of exposures to discharges of radioactive material, radon, and direct radiation from a site.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 1.109, Rev. 1, Calculation of Annual Doses to Man from Routing Releases of Reactor Effluents for the Purpose of Evaluating Compliance with CFR 50, Appendix I (NRC 1976), with respect to determination of doses to the public					
	The ER must comply with the requirements of NUREG-0543, Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (NRC 1980), with respect to determination of compliance with 10 CFR 20.1301(d).					
	The reviewer's analysis of doses is usually an iterative process in coordination with the reviewer for ESRP 3.5 and adheres to the following general steps: (1) Calculate the dose to the maximally exposed individual and collective dose estimates. • Forward these estimates to the reviewer for ESRP 3.5 for comparison with the design objectives and to evaluate the radwaste cost estimate described in 10 CFR 50, Appendix I. If the reviewer for ESRP 3.5 determines that the doses do not meet these design objectives, additional analysis may be needed, and on this basis, source terms may be revised by the ESRP 3.5 reviewer.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• If the source terms are revised, use them to calculate another set of individual maximum doses and collective population doses and forward it to the reviewer for ESRP 3.5 for evaluation. • Repeat this procedure until the reviewer for ESRP 3.5 determines that the applicant's radioactive waste management system meets the design objectives of 10 CFR 50, Appendix I.					
	<u>Estimation of Doses from Gaseous and Liquid Radioactive Releases</u> In conducting the following analysis, the reviewer should be thoroughly familiar with the information and procedures specified in Regulatory Guide 1.109 (NRC 1976) and with the GASPAR and LADTAP computer codes used to estimate doses from gaseous and liquid radioactive releases. The reviewer should take the following steps in performing the analyses of releases: (1) Assemble the gaseous and liquid source term data provided by the reviewer for ESRP 3.5, the receptor location and exposure pathway data (including hydrological and meteorological dispersion factors) provided by the reviewer for ESRP 5.4.1, and any additional hydrological and meteorological data provided by the reviewers for ESRPs 2.3.1 and 2.7. (2) For iodines and particulates in gaseous effluents, examine the receptor locations, associated pathways, and relative deposition (D/Q) values. Select those locations expected to result in the maximum individual dose for input to the GASPAR computer code. (3) For noble gases in gaseous effluents, examine the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	normalized concentration (χ/Q) values at the site boundary for each of the 16 compass sectors that intersect land. • For sites that have water boundaries, examine the meteorological atmospheric dispersion factors for land in sectors beyond the water boundary to determine if any of these locations have higher factors than the other land site-boundary factors. • Determine the location at which the meteorological atmospheric dispersion factor will result in the maximum beta and gamma air dose and the maximum total body and skin dose to an individual. • Select data from this location for input to the GASPAR computer code. (4) For liquid pathways, examine the receptor locations, hydrological data, and associated exposure pathways to select the location expected to result in the maximum individual dose input to the LADTAP computer code. (5) For the locations identified in Items 2, 3, and 4 above, assemble and enter the appropriate data needed to run the GASPAR and LADTAP computer codes. • If input data needed by these codes are lacking and cannot be supplied, use default values (as provided in Regulatory Guide 1.109 [NRC 1976]) for these parameters. • If either code is not sufficient because some important pathways identified by the reviewer for ESRP 5.4.1 are not included in the codes, employ special calculations. These calculations may involve the review of available literature and development of a model describing the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	pathway. (6) When site-specific conditions are so that it is not obvious that the particular location will result in maximum individual dose, select two or more locations for input to the GASPAR and LADTAP codes, then identify the "maximum" location based on the code outputs. (7) Do not analyze the doses resulting from the transportation of radioactive material unless the reviewers for ESRPs 3.8 or 5.4 indicate that an analysis of these pathways is needed. • When this is the case, extend the analysis to cover these pathways, using an analysis and evaluation procedure developed in consultation with these reviewers. • An analysis of occupational radiation exposure from the transportation of radiology materials is not required. (8) Analyze direct radiation doses to individuals in the vicinity of the site. During this analysis, evaluate the applicant's estimates of doses from direct radiation. • If these estimates appear reasonable and justified, they may be used directly in the staff's analysis. • If not, ask the applicant to submit additional information so that the staff can adequately evaluate these sources. • The doses from direct radiation are combined with the doses from gaseous and liquid effluents. • The dose is to be calculated at a point in the offsite environment. Each unit's contribution to the dose at that point should be added to determine the total. For example, contributions of stored waste to the dose at					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	any point from each unit will vary depending on the distance from that unit to the point at which dose from the site is evaluated. The reviewer should take the following steps in performing the evaluation: (1) Assess the computer outputs to ensure that data were entered properly and that the outputs appear normal. (2) For noble gases in gaseous effluents from the GASPAR output, determine the maximum beta and gamma dose in air and the maximum dose to total body of an individual and dose to the skin of an individual. Identify the site boundary location for these doses. (3) For iodines and particulates released to the atmosphere, from the GASPAR output, determine the dose to any organ from all pathways. This dose should be for the age group (adult, teenager, child, or infant) receiving the highest dose. The dose should include the ground plane and inhalation pathways that are present at all receptor locations, plus those pathways that are applicable to the particular location. The plume pathway from the GASPAR code is due to noble gases and is not included in the iodine and particulate release pathways. Identify the receptor location for these doses. (4) For liquid effluents, from the LADTAP output, determine the maximum total body and organ dose to an individual. This dose should be for the age group (adult, teenager, or child) receiving the highest dose. It should be the sum of the pathways that are present in the vicinity of the site, although not necessarily at the same location. Thus, an individual fishing in the plant outfall					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	region is assumed to be obtaining drinking water from the nearest potable water intake affected by plant operation. Identify the receptor location for these doses. (5) From the GASPAR and LADTAP outputs, determine the dose to the total body from all pathways and the dose to any organ from all pathways. (6) Compare the dose data from Items 2 through 5 (above) with the dose data calculated by the applicant. For significant differences, consult with the reviewer for ESRP 3.5 and with the applicant to determine the reasons for these variations. (7) Consult with the reviewer for ESRP 3.5 to determine if the dose commitments calculated above meet the design objectives of Appendix I to 10 CFR 50. If the reviewer for ESRP 3.5 determines that the dose does not meet these design objectives, the following procedure should be used: • Ask the reviewer for ESRP 5.4.1 to re-evaluate the exposure pathway data. The objective of this re-evaluation is to determine if conservative estimates have been used, and if so, to see if more realistic pathway data can be identified that would result in decreased dose predictions. When more realistic input data can be identified, repeat the preceding review procedures of this ESRP and provide the reviewer for ESRP 3.5 with the revised dose calculations. • If, upon re-analysis, the exposure pathway data are shown to be realistic and still result in a prediction that doses will not meet 10 CFR 50 design objectives, request that the applicant commit to additional treatment equipment and effluent control measures.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	When advised that such commitments have been made, the reviewer for ESRP 3.5 should calculate revised source terms, and you should repeat the preceding instructions to provide the reviewer for ESRP 3.5 with the revised dose calculations. Note: For the early site permits, this re-analysis is not necessary. (8) Compare the doses from all pathways (including direct radiation) for all units at the site with the dose criteria referenced by 10 CFR 20.1301(d). If the doses from the site exceed the criteria in 40 CFR 190, request that the applicant commit to additional shielding or other source control measures as appropriate.					
5.4.3 (Draft Rev. 0, March 2000)	Impacts to Members of the Public					
	Acceptance criteria for determining the radiological impacts to individuals from releases during routine operations including anticipated operational occurrences of the reactor are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50, Appendix I, with respect to radiological impacts to individuals from the radiological effluent releases from reactors.					
	The ER must comply with the requirements of 10 CFR 20.1301 with respect to the guidelines for radiological effluent releases from reactors. Note: In accordance with the statement of considerations for 10 CFR 20 (5 CFR 23360), demonstration of compliance with the limits of 40 CFR 190 (as referenced in 10 CFR 20.1301(d)) is considered to be in compliance with the 0.1-rem limit (10 CFR 20.1301).					
	Regulatory guides and specific criteria necessary to meet the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 1.109, Rev. 1, Calculation of Annual Doses to Man from Routing Releases of Reactor Effluents for the Purpose of Evaluating Compliance with CFR 50, Appendix I (NRC 1976), with respect to determining doses to the public from reactor effluents					
	The ER must comply with the requirements of NUREG-0543, Methods for Demonstrating LWR Compliance with the EPA (NRC 1980), with respect to comparing doses to 10 CFR 20.1301(d) requirements as they relate to 40 CFR 190.					
	The analysis of radiological impacts to individuals should be based on the dose estimates prepared by the reviewer for ESRP 5.4.2 that have been evaluated by the reviewer for ESRP 3.5 and determined to be within the design objective guidelines of 10 CFR 50, Appendix I. The reviewer should take the following steps when analyzing radiological impacts. (1) Prepare a table that compares these doses, on a per-unit (individual reactor) basis, with the Appendix I design objectives, using the format shown in Table 5.4.3-1. (2) Determine the 80-km (50-mi) collective total body doses per reactor unit for liquid effluents, noblegas effluents, and radioiodines and particulates, and compare these doses to the natural radiation background for this population. (3) Include an estimate of the collective occupational dose using the format of Table 5.4.3-2. (4) Consult with the reviewers for ESRPs 3.5 and 5.4.2 to verify the accuracy and completeness of the summary table based on					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Table 5.4.3-1. (5) Verify the availability and accuracy of the following data that should be included as input to the environmental impact statement (EIS): • the maximum individual doses and the collective doses to the population within 80 km (50 mi) of the plant, based on individual reactor releases • the individual and collective doses due to total natural background radiation to the population within 80 km (50 mi) of the plant • the estimated occupational collective dose.					
5.4.4 (Draft Rev. 0, March 2000)	Impacts to Biota Other than Members of the Public					
	Acceptance criteria for the review of the potential for significant radiological impacts to biota other than members of the public are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 40 CFR 190 with respect to radiation dose criteria to members of the public.					
	The reviewer should take the following steps: (1) Identify the exposure pathways for the biota not considered in the review in ESRP 5.4.1. (a) Consider the exposure pathways to biota other than members of the public and determine if any of these pathways could be expected to result in estimated doses significantly greater than those evaluated by the reviewer for ESRP 5.4.3. (b) If no such pathways can be identified, end the review and					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	proceed to the "Evaluation Findings" of this ESRP. (2) If exposure pathways for biota other than members of the public are identified for which significantly greater (1 m Gy/day) (100 rad/day) doses could be predicted, then consult with the appropriate reviewers for ESRP 2.4 to determine how the biota at these locations could be affected, and calculate doses to these biota, using models and procedures described in Volume 2, Analytical Models and Calculations, of the BEIR (1972) report. • If the doses are of approximately the same order of magnitude or less than the dose criteria in 40 CFR 190, no further review is necessary. • If significantly higher doses can be predicted, determine if these doses can be expected to affect species population stability. Make this determination through the review of appropriate literature, if available, and through consultation with authorities in the field of radiological effects to biota.					
5.5 (Draft Rev. 0, March 2000)	Environmental Impacts of Waste					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.5.1 (Draft Rev. 0, March 2000)	Nonradioactive-Waste-System Impacts					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Acceptance criteria for the evaluation of nonradioactive waste impacts are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71(d) with respect to quantification of impacts and analysis of compliance with environmental quality standards and requirements.					
	The ER must comply with the requirements of 40 CFR 133 with respect to treatment of wastewater and sewage.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitation guidelines on chemical and biocide discharges.					
	Numerous public laws have a bearing on the handling and disposal of nonradioactive wastes. The most relevant of these include the following:					
	The ER must comply with the requirements of Solid Waste Disposal Act of 1965, which includes the Resource Conservation and Recovery Act of 1976, with respect to Federal, State, regional, local, and affected Native American tribal standards and regulations for disposal of solid wastes.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act of 1958.					
	The ER must comply with the requirements of Federal Water Pollution Control Act (FWPCA) Amendments of 1972 (as amended and now commonly referred to as the Clean Water Act [CWA]).					
	The ER must comply with the requirements of Marine Protection, Research, and Sanctuaries Act of 1972 (most recently amended 1994).					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	The ER must comply with the requirements of Endangered Species Act of 1973 (amended 1988).					
	The ER must comply with the requirements of Clean Air Amendments of 1970 and 1977 (most recently amended 1995).					
	The ER must comply with the requirements of Memorandum of Understanding Between NRC and the Army Corps of Engineers, August 25, 1975.					
	The ER must comply with the requirements of Applicable Memoranda of Understanding Between State Governments and NRC.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to waste discharges and monitoring programs.					
	The analysis should be closely linked with the nonradioactive waste system descriptions provided by the reviewers of ESRP 3.6 and with the environmental descriptions provided by the reviewers of ESRP Chapter 2.0 to establish the nonradioactive waste treatment system characteristics and effluents that are most likely to result in adverse environmental impacts. The reviewer should consult with the reviewers for ESRPs 5.1.1, 5.2.2, and 5.3.2.2 as an initial step in establishing the scope of this analysis. As a general rule, impacts affecting land use, water use, and aquatic biota will be covered by the reviewers for ESRPs 5.1.1, 5.2.2, and 5.3.2.2. This review should address impacts on terrestrial biota, air-quality impacts, water-use impacts not covered by the reviewer for ESRP 5.2.2 (e.g., sanitary waste					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	system effluents), and any other nonradioactive waste system impacts identified in the consultation with the reviewers for ESRPs 5.1.1, 5.2.2, and 5.3.2.2, but not addressed by these reviewers. The reviewer should follow the analysis procedures outlined in ESRPs 5.1.1, 5.2.2, and 5.3.2.2, depending on the nature of the impacts that could be expected. The reviewer should follow the general analysis procedure of ESRP 4.3.1 to analyze and evaluate impacts on terrestrial ecosystems. For land disposal of nonradioactive wastes, the reviewer should consider the potential for short- and long-term damage to terrestrial ecosystems, especially for movement of toxic chemical materials to groundwater, root uptake, and transfer to shoots and into food chains from both dry and liquid waste disposal to the ground. The reviewer should determine the nature and quantities of wastes to be disposed of by licensed waste disposal contractors, but will not assess the impacts of such disposals. The reviewer should prepare a list of all nonradioactive effluents (liquid, solid, and gaseous) and should assess the impacts of those discharges not considered by other ESRP Chapter 5.0 reviewers. The reviewer may use the assessments prepared by other reviewers or by other Federal or State agencies when these are available. With these guidelines in mind, the reviewer should complete the following steps: (1) Ensure that all potential impacts resulting from operation of nonradioactive waste systems have been addressed in this review or by other ESRP Chapter 5.0 reviewers. (2) Ensure that the extent of compliance with Federal, State, regional, local, and affected Native American tribal effluent and receiving water standards (e.g., the CWA) has been assessed.					

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	(3) Follow the evaluation procedures of ESRPs 4.3.1, 5.1.1, 5.2.2, and 5.3.2.2 to evaluate the identified potential impacts not addressed by other Chapter 5.0 reviewers. For terrestrial ecosystems, potential impacts that could require mitigation or avoidance include the following: • disposal sites that preempt habitat critical to the survival of threatened or endangered species or preempt more than a few percent of "important" species' habitat on a regional basis • disposal sites or discharge practices that permit toxic materials to contaminate ground or surface water or to be suspended and dispersed through the air. (4) Evaluate the impact to determine whether waste minimization and/or pollution prevention have been considered and how their implementation could change the effect of the impact.					
5.5.2 (Draft Rev. 0, March 2000)	Mixed Waste Impacts					
	Acceptance criteria for the analysis and evaluation of the impacts resulting from the production, storage, and disposal of mixed waste are based on the following:					
	The ER must comply with the requirements of Resource Conservation and Recovery Act of 1976 (RCRA) with respect to mixed waste, which must meet EPA's requirements for hazardous waste in 40 CFRs 261, 264, and 265 before final transfer offsite in route to burial. This includes the maintenance of records identifying each physical location or unit where					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	mixed waste is stored and identifying the method of storage (40 CFR 264.73(b) and 265.73(b)). An inspection of these storage areas for compliance with applicable RCRA standards for storage methods, including an assessment of compliance with storage-facility standards of 40 CFR 264 or 265 (interim status), should be performed regularly (see 40 CFR 264.15 and 265.15).					
	The ER must comply with the requirements of 10 CFR 20 with respect to the NRC requirements for general radiation protection and occupational dose limits, and waste disposal requirements.					
	Facility owners/operators are required by RCRA regulations to maintain sufficient information to identify their mixed wastes. The information required includes RCRA waste codes for the hazardous components, the source of the hazardous constituents, a discussion of how the waste was generated, the generation rate and volumes of mixed waste in storage, and any information used to identify mixed wastes or make determinations that the wastes are prohibited by land disposal restrictions. Each owner/operator is required (under RCRA regulations) to develop a waste minimization plan that identifies process changes that can be made to reduce or eliminate mixed wastes, methods to minimize the volume of regulated wastes through better segregation of materials, and the substitution of nonhazardous materials. The reviewer should take the following steps to assess the applicant's plans or capabilities for mixed waste disposal: (1) Ensure that the waste minimization plan includes a schedule for implementation, projections of volume reductions to be achieved, and assumptions that are critical to the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	accomplishment of projected volume reductions. (2) Review the nature and quantities of mixed wastes to be disposed of or that must be stored onsite. (3) Assess what, if any, environmental impacts (both radiological and nonradiological) would result from storage of the mixed wastes. (4) Compare impacts resulting from occupational dose related to the storage of mixed wastes with the occupational dose limit criteria given in 10 CFR 20. (5) Ensure that the applicant has anticipated a method for disposal, treatment, or storage of the mixed wastes. (6) Ensure that a mixed waste minimization plan has been formulated and that it identifies changes that can be made to reduce or eliminate mixed wastes.					
5.6 (Draft Rev. 0, March 2000)	Transmission System Impacts					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.6.1 (Draft Rev. 1, July 2007)	Terrestrial Ecosystems					
	The reviewer should become familiar with the provisions of					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	standards, guides, and agreements that are pertinent to the operation and maintenance of transmission systems. Acceptance criteria for the review of impacts on terrestrial ecology as a result of transmission system operation and maintenance are relevant requirements of the following:					
	The ER must comply with the requirements of Bald and Golden Eagle Protection Act with respect to the prohibition of taking, possessing, selling, transporting, importing, or exporting a bald or golden eagle, dead or alive, without a permit.					
	The ER must comply with the requirements of Coastal Zone Management Act of 1972 with respect to natural resources and land or water uses of the coastal zone.					
	The ER must comply with the requirements of Endangered Species Act of 1973 with respect to identifying Federally threatened and endangered species and/or designated critical habitat and initiating formal or informal consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act with respect to consideration of wildlife resources in the planning and development of projects.					
	The ER must comply with the requirements of Migratory Bird Treaty Act with respect to declaring that it is unlawful to take, import, export, possess, buy, sell, purchase, or barter any migratory bird. Feathers and other parts, such as nests or eggs, and products made from migratory birds are also covered by the Act. "Take" is defined as pursuing, hunting, shooting, poisoning, wounding, killing, capturing, trapping, or collecting.					
	The ER must comply with the requirements of Executive Order 13112 with respect to invasive species.					

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	Regulatory positions and specific criteria necessary to meet the regulations and other statutory requirements identified above are as follows:					
	The "Second Memorandum of Understanding and Policy Statement Regarding Implementation of Certain NRC and EPA Responsibilities," serves as the legal basis for NRC decision making concerning licensing matters covered by NEPA and Section 511 of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act (CWA).					
	The "Memorandum of Understanding between the Corps of Engineers, U.S. Army, and the NRC for the Regulation of Nuclear Power Plants," 40 FR 60115, provides guidance with respect to the NRC exercising the primary responsibility in conducting environmental reviews and in preparing EISs for nuclear power stations. The Corps of Engineers should be consulted regarding (1) coastal erosion and other shoreline modifications, (2) siltation and sedimentation processes, (3) dredging activities and disposal of dredged materials, and (4) location of structures affecting navigable waters.					
	Regulatory Guide 4.11, Rev. 1, Terrestrial Environmental Studies for Nuclear Power Stations (NRC 1977), contains technical information for the design and execution of terrestrial environmental studies, the results of which should be included in the applicant's ER and the EIS.					
	When evaluating the data and information acquired under "Data and Information Needs," which is necessary to determine the impacts on terrestrial ecology from transmission system operation and maintenance, the reviewer should take the following steps:					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(1) Review the following categories of impacts: general effects of rights-of-way maintenance (cutting and herbicide application), the special case of rights-of-way maintenance impacts on floodplains and wetlands, bird collisions with power lines, and the effects of EMFs on flora and fauna (plants, agricultural crops, honeybees, wildlife, and livestock) • Consider the conclusions presented in the GEIS (NRC 1996) for all the above impact categories. The conclusions in the GEIS for all the above categories of impacts are that they are generic and SMALL. These conclusions are based on the similarity and insignificance of the known effects (or lack thereof) of transmission system operation and maintenance at operating nuclear power plants at the time the GEIS was issued in 1996. The reviewer should determine whether this magnitude of impact is valid for the above impact categories at the proposed reactor(s) under review by assessing whether conditions there are substantially different from those at operating reactors, and/or whether there has been new and significant information published on the subject since issuance of the GEIS in 1996. In so doing, the review should include, but not be limited to the following: ○ Become familiar with the provisions of standards and guides pertinent to transmission corridor right-of-way maintenance. ○ Determine whether the proposed right-of-way maintenance procedures are those generally recognized as environmentally responsible. Following are examples of such procedures: ▪ maintaining ground cover in rights-of-					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	way to avoid runoff and siltation ▪ avoiding the use of herbicides and defoliants near waterways and using only licensed herbicide and/or pesticide applicators ▪ avoiding unnecessary removal of vegetation that shades streams. ○ Identify any impacts which should be mitigated or avoided and appropriate measures for doing so (e.g., in clearing vegetation from stream banks, make certain it is limited to that necessary for placement of structures). ○ Review any new and significant information on the impact categories of general effects of rights-of-way maintenance (cutting and herbicide application), the special case of rights-of-way maintenance impacts on floodplains and wetlands, bird collisions with power lines, and the effects of EMFs on flora and fauna (plants, agricultural crops, honeybees, wildlife, and livestock) in light of the conclusion presented for them in the GEIS (i.e., SMALL impact). Based on the above review, determine whether the conclusion presented in the GEIS should be applied to the proposed reactor(s) under review. If not, estimate the appropriate impact level for these subject areas. (2) Review impacts to Federally listed threatened or endangered species and/or designated critical habitat by doing the following: • Note that the conclusion presented in the GEIS (NRC					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	1996) for Federally listed threatened or endangered species and/or designated critical habitat is that the magnitude of impact is site-specific and thus may vary. Therefore, the GEIS is not useful for reviewing the effects of transmission system operation and maintenance on Federally protected species and habitats. • Utilize a map and superimpose transmission corridors over occurrences of "important" terrestrial species and habitats, including any Federally listed threatened or endangered species and/or designated critical habitat (from consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service, the appropriate State agency, and the ER). • Determine the magnitude of potential impact (i.e., SMALL, MEDIUM, LARGE) to Federally listed threatened or endangered species and/or designated critical habitat from transmission line right-of-way maintenance (e.g., soil erosion, destruction of habitat, and animal mortality due to chemical and mechanical vegetation control), bird collisions with power lines, etc. (3) Review the potential for the introduction of invasive species by the creation of new transmission corridors or the maintenance practices on such corridors.					
5.6.2 (Draft Rev. 0, March 2000)	Aquatic Ecosystems					
	Acceptance criteria for the review of impacts to aquatic ecology as a result of transmission system operation and maintenance are the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.45					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	with respect to ERs and the analysis of potential impacts contained therein.					
	The ER must comply with the requirements of 10 CFR 51.75 with respect to analysis of impacts to the aquatic environment affected by the issuance of a construction permit.					
	The ER must comply with the requirements of 10 CFR 52, Subpart A, with respect to analysis of impacts to the aquatic environment affected by the issuance of an early site permit.					
	The ER must comply with the requirements of 10 CFR 51.95 with respect to the preparation of supplemental environmental impact statements (EISs) in support of the issuance of an operating license.					
	The ER must comply with the requirements of Coastal Zone Management Act of 1972 with respect to natural resources and land or water uses of the coastal zone.					
	The ER must comply with the requirements of Endangered Species Act of 1973 with respect to identifying threatened and endangered species and critical habitats and initiating formal or informal consultation with the U.S. Fish and Wildlife Service and/or National Marine Fisheries Service.					
	The ER must comply with the requirements of The Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, with respect to restoration and maintenance of the chemical, physical, and biological integrity of water resources.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act of 1958 with respect to consideration of fish and wildlife resources in the planning and development of projects that affect water resources.					
	The ER must comply with the requirements of Rivers and					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Harbors Appropriations Act of 1899 with respect to the deposition of debris in navigable waters or tributaries to such waters.					
	Regulatory guidance and specific criteria to meet the regulations and other statutory requirements identified above are as follows:					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance for the preparation of ERs. With respect to the transmission corridors, it specifies that discussions of temporary or permanent changes in the biological processes of plants and wildlife in the vicinity of the transmission corridors, which result from construction of new access roads or changes in the use of herbicides or pesticides, be addressed in the ER. The reviewer should ensure that the appropriate data and analyses are provided in the environmental report and are included in the EIS.					
	To evaluate the impacts to aquatic ecosystems from transmission facility operating and maintenance, the reviewer should take the following steps: (1) Identify operational and maintenance activities associated with transmission facilities and consider those that could adversely affect those "important" aquatic species and habitats identified by the reviewer for ESRP 2.4.2. • The resources to be considered include marshlands, wetlands, impoundments, and water bodies. • Potential impacts on these resources include heating of water bodies from removal of shade trees, siltation and turbidity resulting from increased runoff and erosion,					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	runoff of defoliants and herbicides, recreational access by the public, and high energy electrical fields associated with underwater transmission facilities. (2) Consult with the reviewer for ESRP 5.2.1 for any needed hydrological data. When potential impacts are anticipated • inventory the "important" aquatic species or habitats vulnerable to the identified operation and maintenance practices • predict the environmental impacts on these aquatic species and habitats. (3) Compare proposed transmission system operation and maintenance with the provisions of standards and guides pertinent to the operation and maintenance of transmission facilities and corridors. (4) Determine whether the proposed operation and maintenance procedures are those generally recognized as environmentally responsible. Following are examples of such procedures: • maintaining ground cover in rights-of-way to avoid runoff and siltation • avoiding the use of herbicides and defoliants near waterways and using only licensed herbicide and/or pesticide applicators • burying underwater transmission lines • avoiding unnecessary removal of vegetation that shades streams. (5) Provide a summary of consultations with appropriate Federal, State, regional, local, and affected Native American					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	tribal agencies.					
5.6.3 (Draft Rev. 0, March 2000)	Impacts to Members of the Public					
	Acceptance criteria for the review of transmission system impacts on man are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.53(c)(3)(ii)(H) with respect to assessing shock hazard impacts of transmission systems.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), identifies the level of detailed description needed to evaluate impacts from land use, the construction and maintenance of these structures and their rights-of-way, and potential hazards to aerial navigation.					
	The ER must comply with the requirements of National Electrical Safety Code (NESC) (1997) with respect to shock hazards.					
	This procedure applies to the review of applications for construction permits, operating licenses, and combined licenses. The reviewer's analysis of the proposed power-transmission system should be closely linked with the environmental review for ESRP 3.7 in order to establish the general transmission characteristics that are most likely to result in environmental impacts. The analysis should be governed by the magnitude of potential impacts on members of the public. The reviewer should coordinate this review with the reviewer for ESRP 5.6.1 to avoid					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	duplication of effort. With the preceding guidelines in mind, the reviewer should take the following steps: (1) Become familiar with the provisions of standards and guides pertinent to the operation and maintenance of transmission lines and corridors, including applicable State standards. Compare predicted noise levels with applicable State noise limits for residential areas and for other types of land use. The authority for environmental noise control was given to the States in the 1972 Noise Control Act. (2) Identify the operational and maintenance activities associated with transmission facilities having impacts on man and determine whether the proposed operational parameters and maintenance procedures are those generally recognized as environmentally acceptable. Potential adverse impacts resulting from operation and maintenance activities include electric shock hazard and electromagnetic field effects, corona discharges (including resultant noise), and potential visual impacts (e.g., design parameters and maintenance activities affecting visual impacts at major road crossings, areas of significant ridges, and concentrated human settlement). For transmission lines energized at 765 kV or less, experience has shown that there are no known adverse impacts resulting from ozone formation. (3) Check for conformance with the National Electric Safety Code (NESC 1997), which provides design criteria that limit hazards from steady-state currents. Adherence to the NESC design criteria limits the short-circuit current to ground, produced by the largest anticipated vehicle or object, to less than 5 mA.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The chronic effects of exposure to electric and magnetic fields have been under investigation for some time. Although some of the recent studies suggest that the effects, if they exist, are below measurable levels, conclusions regarding this potential hazard are premature. If a scientific consensus is reached about these fields, the NRC may request that the applicant address this issue and the staff review the potential impacts on the public.					
5.7 (Draft Rev. 1, July 2007)	Uranium Fuel Cycle Impacts					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.7.1 (Draft Rev. 0, July 2007)	Uranium Fuel Cycle Impacts					
	Acceptance criteria for the evaluation of the impacts of the uranium fuel cycle for light-water reactor designs are based on the relevant requirements of the following:					
	The ER must comply with the requirements of Paragraph (a) of 10 CFR 51.51, “Uranium fuel cycle environmental data—Table S–3" (Federal Register Notices 49 FR 9381, March 12, 1984, and 49 FR 10922, March 23, 1984) with respect to the impacts to the environment from the hazards associated with the fuel cycle.					
5.7.2 (Draft Rev. 0, July	Transportation of Radioactive Materials					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
2007)						
	Acceptance criteria for the description of the transportation of radioactive materials are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.52 with respect to the design and operational parameters related to the transportation of fuel and waste to and from the reactor.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	There are no regulatory positions specific to this ESRP. Note, however, that the NRC has generically considered the environmental impacts of spent nuclear fuel with U-235 enrichment levels up to 5% and irradiation levels up to 62,000 megawatt-days per metric ton and found that the environmental impacts of spent nuclear fuel transport are bounded by the impacts listed in Table S-4 provided that more than 5 years has elapsed between removal of the fuel from the reactor and shipment of the fuel offsite (NRC 1996; NRC 1999). However, these analyses cannot serve as the initial licensing basis for a new reactor.					
	The reviewer's analysis of the data and information is required to support the reviewer's evaluation for conformance with 10 CFR 51.52(a) (see Evaluation Findings in this ESRP). The analysis should consist of assembling the data listed in the procedures below and verifying their accuracy. The reviewer may consult with the reviewers for ESRPs 3.2 and 3.5 to verify the data. The reviewer should take the following steps: (1) Compare the verified data (listed under Data and Information Needs above) with the following criteria:					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• reactor type – light-water cooled (LWR) • rated core thermal power level – 3800 MW maximum (see ESRP 3.2 for a definition of "rated.") • fuel assemblies – zircaloy fuel rods, sintered low enrichment uranium dioxide (maximum 4% by weight of 235U) pellets (Use of 5% enriched fuel in conjunction with irradiation levels above 33,000 megawatt-days per metric ton has been considered generically for existing reactors but cannot serve as the initial licensing basis for a new reactor.) • average irradiation level of irradiated fuel – 33,000 megawatt-days per metric ton maximum for use of Table S-4 directly. For existing reactors, irradiation between 33,000 megawatt-days per metric ton and 62,000 megawatt-days per metric ton maximum requires references to other environmental documents. These references cannot be used as the initial licensing basis for new reactors. New reactors must meet the 10 CFR 51.52(a) conditions or provide a full description and detailed analysis of the impacts of transporting fuel and waste to and from the reactor. • onsite storage of irradiated fuel – minimum of 90 days between removal from the reactor and shipment offsite (5 years, if the irradiation exceeds 33,000 megawatt-days per metric ton for existing reactors). (The reviewer should consider the proposed capacity of the facility to store irradiated fuel in evaluating this criterion.) • radioactive wastes other than fuel – packaged as solid waste prior to offsite shipment. (The reviewer should consider the proposed solid waste treatment and packaging procedures in evaluating this criterion.) • new fuel shipment to the plant – by truck					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• irradiated fuel shipments offsite – by truck, rail, or barge • other radioactive-waste shipments offsite – by truck or rail. (2) When the above criteria are met, conclude that the routine (incident-free) environmental impacts of transportation of fuel and radioactive wastes during reactor operations are represented by the values given in 10 CFR 51.52(c), Table S-4, and instruct the reviewers for ESRP 7.4 to adopt this table as representing the environmental impacts of radioactive materials transportation accidents. (3) For existing reactors, when the fuel is enriched greater than 4 percent by weight of 235U (as given in 10 CFR 51.52(a)(2) to a maximum of 5 percent, and when the fuel irradiation is greater than 33,000 megawatt-days per metric ton (as given in 10 CFR 51.52(a)(3) to a maximum of 60,000 megawattdays per metric ton, it has been shown that the environmental cost contributions are either unchanged or may in fact be reduced from those summarized in Table S-4 (Baker et al. 1988; 53 FR 30355 ; NRC 1996). The impacts of transportation of fuel irradiated to 62,000 megawatt-days per metric ton have also been considered and found to be bounded by those summarized in Table S-4 (NRC 1999, 64 FR 48496) The reviewer should instruct the reviewers for ESRP 7.4 to adopt this table as representing the environmental impacts of radioactive materials transportation accidents. (4) When any of the above criteria are not met, expand the analysis of the required data to the level necessary to provide sufficient data to support a subsequent impact analysis that would supplement the impact data of Table S-4. The reviewer should notify the reviewers for ESRP 7.4 that Table S-4 cannot					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	be used and that a supplemental impact assessment will be required. (5) When required, a full description and detailed analysis of the impacts of transporting fuel and waste to and from the reactor should include the following: • Description of the method(s) used to estimate routine (incident-free) radiological impacts, including impacts to populations and maximally-exposed individuals. Ensure the methods used are defensible (e.g., use the latest version of RADTRAN (SNL 2007)). • Specification of input parameters and sources used in the impact assessment. Review the parameters and source documents to ensure they are defensible. Where assumptions are used to fill in missing or highly uncertain data, ensure the assumptions are bounding and reasonable; i.e., the assumptions tend to overstate transportation impacts yet are not so conservative that they could mask the true environmental impacts of the reactor and lead to invalid conclusions. • Presentation of results, including population doses, maximally-exposed individual doses, and health effects for transportation crews and the general public. (6) Perform a confirmatory analysis of the supplemental transportation impact assessment. The confirmatory analysis is intended to assess and confirm the basis and conclusions of the applicant's supplemental transportation impact assessment. Document the confirmatory analysis in the EIS.					
5.8 (Draft Rev. 0, March	Socioeconomic Impacts					

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2000)						
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
5.8.1 (Draft Rev. 0, March 2000)	Physical Impacts of Station Operation					
	Acceptance criteria for noise, dust, air pollution, and visual aesthetics are based on meeting the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.71 and 10 CFR 51.45 as related to the potential significance of physical impacts of station operations.					
	The ER must comply with the requirements of 29 CFR 1910, "Occupational and Health Standards," with respect to noise, dust, and air pollution.					
	The ER must comply with the requirements of 40 CFR 50-90 as related to National Primary and Secondary Air Quality Standards.					
	The ER must comply with the requirements of Clean Air Act of 1970, as amended, as related to air quality during plant operations.					
	The ER must comply with the requirements of Occupational Safety and Health Act, Noise Provision, 39 Federal Register 10518, Department of Labor, OSHA (May 29, 1971) with respect to noise pollution standards.					
	Regulatory guidance and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to economic and social impact of plant operations.					
	The reviewer's analysis of operational impacts on the community will be linked to the environmental reviews directed by ESRPs 2.1, 2.2.1, 2.5.1, and 2.5.2; all of ESRP Chapter 3.0; and ESRPs 5.3.3, 5.5.1, and 5.5.2 to ensure that the environmental factors most likely to be impacted by proposed plant operation are adequately addressed. To evaluate the information presented in the applicant's environmental report, the reviewer should take the following steps: (1) Identify the people, buildings, roads, and recreational facilities that could be affected for each potential impact. • Determine the - sensitive use patterns (e.g., hospitals, residences, recreational areas, viewsheds) - allowable limits of impacts, where available. • Consider impacts from noise, air pollution, and visual intrusion. (2) Identify the potential operational impacts on these elements and predict the extent and magnitude of the impacts. Impacts may be described in qualitative terms if the effect on the community is expected to be small. (3) If adverse impacts can be predicted, conduct a more detailed analysis and, where practical, make quantitative estimates of the magnitude of the impacts.					

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	(4) Consult with the reviewer for ESRP 4.1.1 to identify the construction features that are expected to have operational impacts (e.g., access roads). If operational impacts are projected to be temporary extensions of the construction impact, this may be noted, and no further analysis will be needed. (5) Consult with the reviewers of both ESRPs 3.7 and 4.4.1 to complete the analysis of visual impacts, with emphasis on the identification of measures and controls (e.g., screening) to mitigate the impacts determined to be adverse. (6) Identify those proposed design features and operating procedures that can be expected to mitigate the physical impacts. Means available for mitigation include • drift and noise eliminators • air pollution control devices • landscaping for visual screening. (7) Become familiar with the provisions of standards, guides, and agreements pertinent to the operational impacts of nuclear power stations. (8) Consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies to verify that current applicable regulations and guides are available. For example, consult • the EPA for ambient air quality standards and air pollutant levels • the Occupational Safety and Health Administration guidelines and standards applicable to facility operation.					

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	(9) Verify that the applicant has made commitments to comply with these applicable regulations and guides. (10) Examine proposed operation activities in light of recognized "good practice." The term "good practice" as used here refers to those activities that tend to mitigate noise levels and adverse physical impacts on the community.					
5.8.2 (Draft Rev. 0, March 2000)	Social and Economic Impacts of Station Operation					
	Acceptance criteria are for including socioeconomic impacts during operations based on meeting the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.45(c) with respect to analysis of socioeconomic data.					
	The ER must comply with the requirements of 10 CFR 51.45(d) and 51.71(d) with respect to the socioeconomic impacts of plant operations analyses required in the development of the ER and EIS.					
	The ER must comply with the requirements of 10 CFR 52.18 with respect to reviewing applications for early site permits.					
	The ER must comply with the requirements of 10 CFR 52.81 with respect to reviewing applications for combined licenses.					
	Regulatory positions and specific criteria to meet the regulations identified are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976) with respect to benefits and costs to nearby populations during operations.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The reviewer's analysis of the social and economic impacts of operation should be linked to the environmental descriptions provided by the reviewer for ESRP 2.5.2 and the construction impact assessments of ESRP 4.4.2. The reviewer should ensure that those environmental factors most likely to be impacted by operation of the proposed plant are described in sufficient detail to permit assessment of the predicted impacts. To evaluate this information, the reviewer should take the following steps: (1) Identify and analyze components of the regional and community social, political, and economic systems that would be potentially impacted. (2) Determine, from the full scope of potential impacts, those that are minor and those that are likely to be sufficiently important to require detailed analysis. • Generally, operating impacts other than those related to tax revenues will be less than the corresponding impacts of construction. • It may not be necessary to re-address impacts determined to be minor by the reviewer for ESRP 4.4.2. (3) Where practical, develop quantitative measures of identified adverse impacts. (4) Consult with the reviewers for ESRPs 5.1.1 and 5.2.2 to determine if any of the impacts identified under these sections are projected to be of sufficient social or economic consequence to be examined further under this plan. (5) Categorize impacts into those resulting directly from plant operation and those resulting from the requirements of the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	operating staff using the following procedure: • Analyze the social and economic impacts directly associated with plant operation, as follows: - Determine by jurisdiction the tax revenues derived from station operation. - Predict the physical demands placed on local public facilities and services (e.g., fire, police, sewer and water) by plant operation and compare these demands with existing facilities and services. - In consultation with appropriate reviewers, determine if any impacts identified under landuse or water-use impacts require further analysis regarding social and economic consequences. • Analyze the socioeconomic impacts associated with the operating staff, as follows: - Determine the operating staff requirements by predicting the number of workers originating from within the region and the number of in-migrants. - Predict the geographic distribution of in-migrants. - Estimate the overall impact of in-migrants and procurements of goods and services on regional income, employment, and population. - Estimate the flow of tax revenues generated by the operational payroll and induced economic activity.					

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	• Describe any unique changes predicted to occur in the social and political structure and character of impacted communities, labor force mobility, and residential choices and describe the mechanisms available to these communities to plan for and accommodate change induced by plant operation. Include the socioeconomic effects in any analysis of potential plant accident scenarios. • Consider the following types of socioeconomic impacts: labor force mobility and residential choices; impacts linked to changes in visual quality; and impacts from changes in tourism and recreation.					
5.8.3 (Draft Rev. 1, July 2007)	Environmental Justice Impacts					
	The acceptance criteria for environmental justice impacts are based on the relative requirement of the following:					
	The ER must comply with the requirements of 10 CFR 51.45(c) with respect to analysis of socioeconomic data.					
	NRC specific policy on treatment of environmental justice matters can be found in "Policy Statement on the Treatment of Environmental Justice Matters in NRC Regulatory and Licensing Actions." Federal Register,69 FR 52040, August 24, 2004.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The Council on Environmental Quality (CEQ) guidance for addressing environmental justice (CEQ 1997) is not binding, but should be followed as appropriate.					
	The guidelines for specific information requirements for					

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	environmental justice determinations are described in Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Issues, Appendix D to Office of Nuclear Reactor Regulation (NRR) NRR Office Instruction LIC-203, NRC Office of Nuclear Rector Regulation, Washington, D.C. (NRC 2004). NRR Office Office Instruction LIC-203 is revised periodically. Obtain the latest revision for current guidance. Information submitted by the applicant is adequate and meets the 10 CFR 51.45 requirements and NRR guidelines if it permits the identification of potential disproportionate and negative impacts on minority and low-income populations as required in that guidance.					
	The ER must comply with the requirements of Regulatory Guide 4.7, Rev. 2., General Site Suitability Criteria for Nuclear Power Stations (NRC 1998a), which specifies the avoidance of disproportionately high and adverse impacts on minority and low-income populations during plant siting.					
	The kinds of data and information required will be affected by site- and station-specific factors, and the degree of detail should be modified according to the anticipated magnitude of the potential impact. The data-requirements analysis should generally be the same for any type of environmental review that requires the preparation of an environmental report (ER). (1) Determine which impacts are likely to be of concern and, therefore, what environmental impact areas should be discussed. (2) Contact the lead staff responsible for ESRP 2.5.4 and ESRPs 5.1 through 5.8 to determine whether the appropriate impact areas are being discussed					

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	(3) Contact the lead staff responsible for ESRP 7.1 to obtain a description of potential accidents. (4) Examine the record of the National Environmental Policy Act (NEPA) public scoping process to determine whether appropriate environmental impact areas are being discussed with respect to environmental justice. ESRP 2.5.4 in particular discusses specific efforts that may have been made to interview representatives of minority communities and other regional contacts (such as social service agencies) having specific knowledge about the locations, resource dependencies, customs and practices, and pre-existing health and socioeconomic conditions of minority and low-income populations in the region. The results of this additional outreach, if any, should also be evaluated. (5) Contact the responsible personnel of each affected State for sites located on or near State boundaries, or where transmission line routes, access corridors, or offsite areas pass through more than one State. Supplemental data obtained from other individuals and organizations may be useful in determining the completeness of the applicant's identification of minority and low-income populations. (6) Analyze the potential impacts on minority and low-income populations. (a) briefly describe pathways by which any environmental impact during operations may interact with cultural or economic facts that may result in disproportionate environmental impacts on minority and low-income populations. If there are none, so state,					

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	and provide a brief discussion of why the potential pathways do not result in impact. (b) assess (qualitative or quantitative, as appropriate) the degree to which each minority or low-income population would disproportionately experience adverse human health or environmental impacts during operations as compared with impacts on the general population in the impacted area. (c) assess the degree to which each minority and low-income population would disproportionately receive any benefits compared with the general population. (d) assess (qualitative or quantitative, as appropriate) the significance or potential significance of such environmental impacts on each minority and low-income population. Significance is determined by considering the disproportionate exposure, multiple-hazard conditions, and cumulative hazard conditions outlined in the Environmental Justice: Guidance Under the National Environmental Policy Act (CEQ 1997). (e) discuss any mitigative measures for which credit is being taken to reduce environmental justice concerns. (f) When alternative sites are being evaluated, similar reviews should be conducted for each site, using reconnaissance-level data (see ESRP 9.3).					
5.9 (Draft Rev. 0, March 2000)	Decommissioning					
	Acceptance criteria for the analysis and evaluation of decommissioning are based on the relevant requirements of					

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	the following:					
	The ER must comply with the requirements of 10 CFR 50.33 with respect to preparing a decommissioning funding plan report on how funds will be available to radiologically decommission the facility.					
	The ER must comply with the requirements of 10 CFR 50.75 with respect to the requirements for reasonable assurance that funds will be available to radiologically decommission the facility, and with respect to the minimum amounts required to demonstrate such assurance.					
	The ER must comply with the requirements of 10 CFR 52.77 with respect to requirements for a combined license, including the information specified in 10 CFR 50.33 to provide a decommissioning funding plan report showing how reasonable assurance will be given that funds will be available to radiologically decommission the facility.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance (NRC 1999), which provides review procedures for verifying the information submitted by a new applicant for an operating license in the form of a report indicating how reasonable assurance will be given that funds will be available to radiologically decommission the facility.					
	NRC regulations do not require the applicant to submit detailed plans for decommissioning plans and, in the absence of such plans, no detailed analysis of decommissioning is necessary. However, applicants for operating licenses (10 CFR 50.33[k]) and combined licenses (10 CFR 52.77) must include as part of their application a report that contains a certification that financial assurance for decommissioning will be provided in an amount					

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	that may be more, but not less, than the amount stated in the table in 10 CFR 50.75(c)(1). The reviewer should take the following steps: (1) The reviewer should ensure that the applicant has submitted the report required by 10 CFR 50.33(k) and specified in 10 CFR 50.75(b)(1). (2) The reviewer should coordinate with the reviewer of the Decommissioning Funding Assurance in the Generic Issues and Environmental Projects Branch to ensure that the appropriate review is being or has been made and to obtain the cost estimate for decommissioning the proposed facility (if available) or the amount from the table in 10 CFR 50.75(c)(1) as well as the means for financial assurance.					
5.10 (Draft Rev. 1, July 2007)	Measures and Controls to Limit Adverse Impacts During Operation					
	Acceptance criteria for the summary of the measures to monitor and control potentially adverse impacts of operation are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50.36b with respect to environmental conditions for an NRC license or permit for the protection of the nonaquatic environment. Such conditions can cover reporting, recordkeeping, and monitoring.					
	The ER must comply with the requirements of 10 CFR 51, Appendix A to Subpart A, with respect to discussion of alternatives and mitigating measures to avoid or minimize adverse impacts.					
	The ER must comply with the requirements of 10 CFR 52.24 with respect to issuing early site permits containing the conditions and					

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	limitations as the Commission deems appropriate and necessary.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the measures planned to reduce undesirable effects of station operation.					
	The reviewer's analysis should include identification and tabulation of operational impacts requiring mitigation, identification of the applicant's commitments that limit and control these impacts, and comparison of the applicant's commitments with impacts requiring mitigation. The reviewer should take the following steps: (1) Identify and tabulate the operational impacts (see the reviewers for ESRPs 5.1 through 5.8) that are of sufficient severity to need mitigation, i.e., measures and controls to limit the impact. (2) List the applicant's commitments for mitigating the impact. (3) Identify, based on consultation with appropriate staff reviewers, the applicant's commitments that will satisfy the staff's concerns for mitigation. (4) When it is determined that there are no appropriate applicant commitments to control or limit an adverse impact, the staff should consult with reviewers for the appropriate ESRPs 5.1.1 through 5.8.3, the reviewers for ESRPs 9.4.1 through 9.4.3, and the EPM to identify mitigation measures. Note those impacts for					

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	which no appropriate measures and controls to limit the impact can be identified. (5) Prepare a table similar to Table 5.10-1 to compare potentially adverse operational impacts with the applicant's commitments for measures and controls to limit the impacts. Identify adverse impacts that cannot be mitigated or for which mitigation is not practical. (6) Confirm that the operational impacts, when considered on a site-specific basis, are adverse and should be mitigated. • Make this determination through consultation with the appropriate reviewers for ESRPs 5.1.1 through 5.8.3. • Take into account experience gained from the review of operational data from other plants having similar impacts. • Ensure that adequate documentation is available to support the staff conclusions with respect to the nature and severity of those impacts requiring mitigation. (7) Confirm that the available measures and controls to limit each impact have been evaluated to verify that a practical level of mitigation can be achieved by these methods and controls. • Confirm that each measure and control is reasonable, i.e., involves methods and techniques that are appropriate and achievable on a site-specific basis. • Confirm that the measures and controls are specific and unambiguous, and are structured so that their application and results can be verified through subsequent field reviews and inspections.					

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	(8) Confirm that environmental, economic, and social costs of the available measures and controls have been balanced against the benefits expected. • Consult with the appropriate benefit-cost reviewers in conducting this portion of the evaluation. Benefit-cost reviews cannot be used as a basis for noncompliance with NRC regulations. • When mitigation techniques do not lead to an improvement in the overall benefit-cost ratio, and if mitigation is not required by law, the impact may be accepted without mitigation and considered in the overall project benefit-cost balancing. (9) Document any operations-related reporting, recordkeeping, and monitoring requirements that should be included in any environmental protection plan attached to the proposed license or permit.					
5.11 (Draft Rev. 0, July 2007)	Cumulative Impacts Related to Station Operation.					
	Acceptance criteria for the summary of cumulative impacts associated with proposed operational activities are the following:					
	The ER must comply with the requirements of 10 CFR 51.10(a) with respect to NRC policy to voluntarily take account, subject to certain conditions, of the regulations of CEQ implementing NEPA. The CEQ regulations specify that an EIS discuss cumulative impacts [40 CFR 1508.25(c)(3)].					
	Regulatory positions and specific criteria to meet the regulations identified above are as follows:					

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	The ER must comply with the requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976) with respect to the inclusion in an application of an assessment of (1) cumulative and projected long-term effects from the point of view that each generation is trustee of the environment for each succeeding generation, and (2) any cumulative buildup of radionuclides in the environment.					
	The reviewer's analysis should include identification and tabulation of potentially adverse cumulative impacts associated with operation of the proposed plant. The reviewer should take the following steps: (1) Identify past, present, and known future Federal, non-Federal, and private actions that could have meaningful cumulative impacts with the proposed action. Review of the aggregate effects of past actions is needed to the extent that the review provides information regarding the proposed action (CEQ 2005). (2) Identify the geographic area to be considered in evaluating cumulative impacts. CEQ guidance is to use natural ecological or sociocultural boundaries (CEQ 1997). Possible geographic areas that could be used to determine the appropriate geographic area for a cumulative impact analysis are in Table 2-2 of CEQ (1997). (3) Identify and tabulate the cumulative impacts associated with operation of the proposed plant. Input should be obtained from the reviewers for ESRPs 4.1 through 4.5. CEQ guidance is that agencies should focus on cumulative impact information that is relevant to reasonably foreseeable significant adverse impacts, is essential to a reasoned choice among alternatives, and can be obtained without exorbitant cost (CEQ 2005). Cumulative effects					

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	may result from the accumulation of similar effects or the synergistic interaction of different effects (CEQ 1997).					
6.0 (Draft Rev. 1, July 2007)	Environmental Measurements and Monitoring Programs					Exclude, Administrative
6.1 (Draft Rev. 0, March, 2000)	Thermal Monitoring					
	Acceptance criteria for the thermal programs on the proposed sites are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to defining activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction activities.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to NPDES permit conditions for discharges, including storm water discharges.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole source aquifer.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of Federal, State, local, regional, and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts of striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S. Supreme Court granted the States additional authority to limit hydrological alterations beyond the States' role in regulating water rights.					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of ERs, including hydrology, water-use, and water-quality issues.					
	The regulatory position necessary to meet the objective identified above requires documentation of consultations with NPDES authority.					

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	The reviewer should consider the following separate but related aspects of the applicant's thermal monitoring program: • Preapplication Monitoring. The program of field monitoring and data collection is used to support the applicant's thermal descriptions. • Preoperational Monitoring. The program of thermal monitoring establishes a baseline for identifying and assessing environmental impacts resulting from plant operation. • Operational Monitoring. The program of thermal monitoring establishes changes in water temperature resulting from plant operation. Each of these aspects is discussed in greater detail in the sections that follow.					
	<u>Preapplication Monitoring</u> Information from the applicant's preapplication monitoring program is used to aid in the description of the baseline water temperature. Generally, data are needed on a seasonal basis and should be sufficient to characterize seasonal variations throughout an annual cycle. Long-term trends may be established using regional data; the reviewer may rely on input from other sources (e.g., Federal, State, regional, local, or Native American tribal agencies) for these data. The reviewer should analyze the available data to determine that they are adequate to support the environmental descriptions of ESRP 2.3.1 and the impact analyses of ESRPs 5.2 and 5.3.2. The following factors should be considered in the analysis: • the location and number of monitoring stations as					

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	required to consider the following factors: - bathymetric characteristics in the vicinity of the site - type of cooling system employed and its probable operating modes - transient hydrological parameters in the vicinity of the site - vertical and horizontal temperature and salinity structure in the vicinity of the site. • the sampling frequency and times to ensure that important temporal variations (e.g., tidal variations) are adequately monitored • the duration of monitoring programs • the data analysis procedures. • data quality objectives (if any)					
	Preoperational Monitoring The preoperational monitoring program supplements any preapplication monitoring in providing a baseline water temperature database. Discussion of the applicant's preoperational monitoring plan should mention the following: • the average and extreme extent and enclosed surface area of the limiting excess temperature isotherms as established by the NPDES permitting agency, by comparison with background and baseline data • temperatures at positions appropriate to define the extent of the mixing zones (proposed or established) • time temperature relationships at biological monitoring					

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	stations • any other parameters required by the NPDES permitting agency • data quality objectives (if any).					
	Operational Monitoring The operational monitoring program is designed to establish changes in water temperature resulting from plant operation. NPDES permitting agencies will specify operational monitoring requirements. The reviewer should describe the status of NPDES permit consultations and NPDES permit renewal.					
6.2 (Draft Rev. 0, March 2000)	Radiological Monitoring					
	Acceptance criteria for the radiological environmental measurements and monitoring programs are found in the following:					
	The ER must comply with the requirements of Regulatory Guide 4.1, Rev.1, Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants (NRC 1975), with respect to establishing a program for monitoring radioactive materials from a reactor in the environment.					
	The ER must comply with the requirements of Regulatory Guide 4.15, Rev.1, Quality Assumptions for Radiological Monitoring Programs (Normal Operations)—Effluent Streams and the Environment (NRC 1979a), with respect to establishing an appropriate quality assurance program for the radiological environmental monitoring program.					
	The ER must comply with the requirements of Radiological Assessment Branch Technical Position regarding Radiological					

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	Environmental Monitoring Programs, Rev. 1, Radiological Assessment Branch Technical Position (NRC 1979b).					
	The following analysis procedures include a review of the applicant's proposed preoperational radiological environmental monitoring program and the applicant's operational monitoring program, as appropriate. The preoperational program should establish (or may have established) the baseline from which subsequent identification and assessment of radiological environmental impacts resulting from plant operation can be made. (1) Compare the applicant's proposed program (including quality assurance) with the basic criteria of Regulatory Guides 4.1 and 4.15 and the recommended program elements of the NRC Branch Technical Position, "An Acceptable Radiological Environmental Monitoring Program." The discussion of radiological environmental monitoring from the Branch Technical Position document is reproduced in Table 6.2-1. (2) Consider the following factors in the analysis: • If a preoperational program is under consideration, it should be based on the development of baseline data for important pathways and the anticipated types and quantities of radionuclides to be released from the plant. - The purposes of the premonitoring program are to measure background levels and their variations along the anticipated critical pathways in the area surrounding the station; to train personnel; and to evaluate procedures, equipment, and technique. - The preoperational monitoring program should be					

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	initiated 2 years before plant operation. (See Table 6.2-1 to this ESRP for recommended program durations.) The elements (sampling media and type of analysis) of both preoperational and operational programs should be essentially the same. • If an operational program is under consideration, it should be based on baseline data already developed for important pathways and types and quantities of radionuclides released from the plants. - The program should be developed from baseline data that have already been obtained. - Consider adjustments being proposed by the applicant, based on operating experience. - The program should provide baseline data to evaluate the possibility of buildup of long-lived radionuclides in the environment and to identify potential physical and biological sites of radionuclide accumulation. - The program should establish the baseline from which correlations between levels of radiation and radioactivity in the environment and radioactive releases from plant operation may be made. • The monitoring program should include a documented quality assurance program. (3) Consult with the reviewer for ESRP 5.4.1 to identify the significant pathways of radiological impact to man and biota, e.g.,					

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	food, recreational use, water use. (4) Consult with the reviewer for ESRP 3.5 to determine the locations of effluent release points and orientation of the plant and any radioactive material storage locations. (5) Consult with the reviewers for ESRPs 2.3.1, 2.3.2, and 2.7 to analyze the relationship between the proposed (or actual) effluent release point locations and the proposed (or actual) water and air sampling locations from the standpoint of detection of potential (or actual) buildup of radioactive materials from effluents. Use the site visit to observe the location of proposed (or actual) sampling and measuring locations relative to potential (or actual) radiological impact pathways. (6) Determine whether sufficient and adequate information has been provided to analyze and evaluate the proposed radiological environmental monitoring program and, if it has, whether the proposed program will accomplish the stated goals and objectives. (7) Consult with the reviewers for ESRP 2.3.1 and 2.3.2 to verify that ground and surface-water sampling points are located to best detect potential (or actual) concentrations of radioactive materials associated with liquid effluents. (8) Consult with the reviewer for ESRP 2.7 to verify that air monitoring and sample points are located to best detect potential (or actual) concentrations of radioactive materials from airborne effluents. Determine whether sampling frequency and duration, sample size, and lower limits of detection are appropriate for the pathway being monitored.					

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	(9) If the program is judged to deviate from these criteria, identify program additions or modifications, including changes in locations, additions of sampling and measurement stations, or deletion of some measurements. (10) When a preoperational program is being considered, ensure that (a) each important pathway of radiological impact to man will be monitored (b) each monitoring program element will accumulate meaningful baseline data from which subsequent operational radiological impacts may be determined and controlled. (11) When changes to an operational program are being considered, evaluate the technical merit of the applicant's justification.					
6.3 (Draft Rev. 0, March, 2000)	Hydrological Monitoring					
	Acceptance criteria for the review of thermal monitoring programs are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with respect to conditions, limitations, and restrictions on construction activities.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with respect to procedures on floodplain and wetlands protection.					

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	The ER must comply with the requirements of 40 CFR 122 with respect to NPDES permit conditions for discharges including storm water discharges.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole source aquifer.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of Federal, State, regional, local, and Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	Compliance with environmental quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act (CWA) is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action that are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts of striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					

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	Because water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S. Supreme Court granted the States additional authority to limit hydrological alterations beyond the States' role in regulating water rights.					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of Environmental Reports including hydrology, water-use, and water-quality issues.					
	The regulatory position necessary to meet the objective identified above requires documentation of consultations with National Pollution Discharge Elimination System (NPDES) administrative authority, and/or water rights regulatory authority.					
	The reviewer should consider the following separate but related aspects of the applicant's hydrological monitoring program: • Preapplication Monitoring. The program of field monitoring and data collection is used to support the applicant's baseline hydrological descriptions. • Construction Monitoring. The program of hydrological monitoring to control anticipated impacts from site preparation and construction and to detect any unexpected impacts arising from these activities may include preconstruction monitoring to establish a baseline for assessing the subsequent impacts of site preparation and construction. This monitoring will be needed only in unusual circumstances when specific adverse impacts are predicted.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• Preoperational Monitoring. The program of hydrological monitoring establishes a baseline for identifying and assessing environmental impacts resulting from plant operation. • Operational Monitoring. The program of hydrological monitoring establishes the impacts of operation of the plant and detects any unexpected impacts arising from plant operation. Each of these aspects is discussed in greater detail below. If available, documentation of data quality objectives should be reviewed.					
	<u>Preapplication Monitoring</u> Information from the applicant's preapplication monitoring program is used to aid in the assessment of site acceptability and to support the staff's database as needed to identify surface-water or groundwater system impacts that could result from construction and operation of the proposed plant. Generally, data are needed on a seasonal basis and should be sufficient to characterize seasonal variations throughout at least one annual cycle. The reviewer should analyze the available data to determine that they are adequate to support the environmental descriptions of ESRP 2.3 and the impact analyses of ESRPs 4.2, 5.2, 5.3.1, and 5.3.2. The following factors should be considered in the analysis: • the location and number of monitoring stations (and wells) as required to consider the following factors: - bathymetric characteristics of surface waters in the site vicinity					

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	- soil and groundwater system characteristics in the site vicinity - the type of cooling system employed and its operating modes - type of sanitary and chemical waste retention method - transient hydrological and meteorological parameters in the site vicinity. • the sampling frequency and times to ensure that important temporal variations (e.g., tidal variations and intense rainfall) are adequately monitored • the duration of monitoring programs • the sediment transport characteristics.					
	<u>Construction Monitoring</u> Construction monitoring will be required when specific adverse impacts are predicted (e.g., impact due to dewatering, increased turbidity). The reviewer should determine these predicted impacts from the ESRP 4.2 and 4.3 reviews and should analyze the proposed monitoring programs associated with these predicted impacts.					
	<u>Preoperational Monitoring</u> The preoperational monitoring program is designed to provide the database necessary for evaluating any hydrologic changes arising from operation of the proposed plant. The applicant's preoperational monitoring plan should be analyzed to determine if adequate baseline data will be available to assess the					

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	following: • the alteration of surface-water flow fields in the site vicinity • alteration of groundwater flow (e.g., saltwater intrusion) • impact of sanitary and chemical waste-retention methods on groundwater quality • alteration of sediment transport • alteration of floodplains or wetlands.					
	Operational Monitoring The operational monitoring program is designed to establish the impacts of operation of the plant and to detect any unexpected impacts arising from plant operation. Operational monitoring may be required by permitting agencies.					
6.4 (Draft Rev. 0, March, 2000)	Meteorological Monitoring					
	Acceptance criteria for the onsite meteorological measurements program are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 50, Appendix I, with respect to meteorological data used in determining compliance with numerical guides for doses to meet the criterion of "as low as is reasonably achievable" (ALARA).					
	The ER must comply with the requirements of 10 CFR 51.45(c) with respect to meteorological data provided to aid the Commission in its development of an independent analysis.					
	The ER must comply with the requirements of 10 CFR 51.50 with respect to keeping records of environmental data.					

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	The ER must comply with the requirements of 10 CFR 52.17(a)(1) with respect to describing the meteorological characteristics of the proposed site in an early site permit application.					
	The ER must comply with the requirements of 10 CFR 100.10(c)(2) with respect to data collected for use in characterizing meteorological conditions of the site and surrounding area.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Section C of Regulatory Guide 1.23, Onsite Meteorological Programs (NRC 1972), contains specific criteria for an acceptable meteorological- measurement system.					
	Appendix A of ESRP 2.7 describes an acceptable format for submission of meteorological data to NRC. Data may be submitted on magnetic tape or other media.					
	Section C.4 of Regulatory Guide 1.111, Rev. 1, Methods for Estimating Atmospheric transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors (NRC 1977), contains guidance on summarization of meteorological measurements for use with models.					
	Regulatory Guide 1.21, Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluens from Light-Water-Cooled Nuclear Power Plants (NRC 1974), contains guidance on summarization of meteorological data for submission with reports of releases of radioactive materials in gaseous effluents.					
	The reviewer should verify that sufficient information has been provided to adequately assess the onsite meteorological					

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	measurements program and other data-collection programs used by the applicant to (1) describe local and regional atmospheric transport and diffusion characteristics, (2) ensure environmental protection, and (3) provide an adequate meteorological database for evaluation of the effects of plant operation. The reviewer should consider the following separate but related aspects of the applicant's meteorological monitoring program: • Preapplication Monitoring. The program of field monitoring and data collection is used to support the applicant's meteorological descriptions. • Site Preparation and Construction Monitoring. This is the proposed program of meteorological monitoring to control anticipated impacts from site preparation and construction and to detect any unexpected impacts arising from these activities. This program may include preconstruction monitoring to establish a baseline for assessing the subsequent impacts of site preparation and construction. This monitoring will be needed only in unusual circumstances when specific adverse impacts are predicted. • Preoperational Monitoring. The program of meteorological monitoring establishes a baseline for identifying and assessing environmental impacts resulting from plant operation. • Operational Monitoring. The program of meteorological monitoring establishes a baseline for use in evaluation of the environmental impacts of plant operation. In terms of onsite meteorological instrumentation, the reviewer should ensure that the basic meteorological parameters measured by instrumentation at all sites include wind direction					

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	and wind speed at two levels, ambient air temperature difference between two levels, temperature, and atmospheric moisture at height(s) representative of water-vapor release (at sites at which large quantities of water vapor are emitted during plant operation). Guidance on meteorological data to be used as input to atmospheric dispersion modeling and assessment is given in Regulatory Guides 1.111 and 1.21. With these considerations in mind, the reviewer should evaluate instrument siting, meteorological sensors, and the recording of their output, instrument surveillance, data acquisition and reduction, and data screening.					
	<u>Instrument Siting</u> The reviewer should compare instrument types, heights, and locations to the recommendations of Regulatory Guide 1.23, Sections C.1 and C.2, as follows: (Note: Additional guidance on instrument siting may be found in ANSI/ANS-2.5, "American National Standard for Determining Meteorological Information at Nuclear Power Sites," and in International Atomic Energy Agency Safety Series No. 50-5G-S3, "Atmospheric Dispersion in Nuclear Power Plant Siting" [IAEA 1980].) (1) Evaluate local exposure of instruments, as follows: (a) Examine the local exposure of the wind and temperature sensors to ensure that the measurements will represent the general site area after plant construction. • Determine whether the tower that supports the sensors will influence the wind or temperature measurements.					

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	• Keep the following guidelines in mind: - Professional experience and studies have shown that wind sensors should be mounted on booms so that the sensors are at least one (and preferably two or more) tower widths away from an open latticed tower and at least two stack or tower widths away from a stack or closed tower. - For temperature sensors, mounting booms need not be as long as those for wind direction sensors, but the sensors must be unaffected by thermal radiation from the tower itself. - No temperature sensors may be mounted directly on stacks or closed towers. - Mounting booms for all sensors should be oriented normal to the prevailing wind at the site. (b) Determine whether the terrain at or near the base of the tower will unnaturally affect the wind or temperature measurements. • Evaluate the heat reflection characteristics of the surface underlying the meteorological tower (grass, soil, gravel, paving, etc.) to ensure that localized influences on measurements are minimal. • Examine the position, size, and materials used in the construction of the recorder shelter and the proximity and heights of nearby trees and structures, including exhaust stream plumes, for potential localized influence on the measurements.					

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	(2) Evaluate the general exposure of instruments as follows: (a) Verify that the tower position(s) will allow the instrumentation to provide measurements that represent the overall site meteorology without plant structure interference. (b) Determine and evaluate the representativeness of the locations, as follows: • Examine topographic maps that have been modified to show the finished plant grade and features. • Conduct a site visit. • Use professional judgment on airflow patterns. (c) Examine the plant structure layout, including structure heights and potential influence on meteorological measurements, using the following guidelines: • For no discernible influence on measurements, towers should be located at least ten obstruction heights away from major obstructions. • For towers located more than five obstruction heights from major obstructions, the influence should be minimal. • Tower locations within five obstruction heights should be analyzed on a case-by-case basis.					
	<u>Meteorological Sensors</u> The reviewer should evaluate meteorological sensors as follows: (1) Evaluate sensor type and performance specifications.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(a) Consider manufacturers' specifications, performance analyses, and operating experience for these sensors in evaluating their accuracy and potential for acceptable data recovery. (b) Use standardized evaluations and operational experience reports contained in research papers. Guidance for sensor evaluation is found in Regulatory Guide 1.23 and Atmospheric Science and Power Production (Randerson 1984). (2) Determine the suitability of the specific type of sensor for use in the environmental conditions expected to occur at the site, by considering the range of wind conditions and the ability of the sensors to withstand corrosion, blowing sand, salt, air pollutants, birds, and insects. (3) If the sensors are new and unique, consult a meteorological instrumentation expert (e.g., National Oceanic and Atmospheric Administration—Idaho National Engineering and Environmental Laboratory [NOAA–INEEL]) to complete the analysis.					
	Recording of Meteorological Sensor Output The reviewer should evaluate the recording of the sensor output as follows: (1) Evaluate the methods of recording (e.g., digital or analog, instantaneous or average engineering units or raw voltages) and recording equipment, including performance specifications and location of the equipment. Consider manufacturers' specifications and operating experience for the recorders when considering accuracy and the potential for acceptable data recovery.					

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	(2) Review the controlled environmental conditions in which the recorders are kept (instrument shelter or control room) for adequacy in accordance with the manufacturers' specifications. Confirm the ability to obtain a direct readout from the recorders in situ during routine inspection of systems so that the reviewer will be able to relate the recorder output directly to what the sensor should be seeing. Some specific recommendations are contained in Regulatory Guide 1.23, Section C.3.					
	<u>Instrumentation Surveillance</u> When evaluating instrumentation surveillances, the reviewer should do the following: • Review the inspection, maintenance, and calibration procedures and their frequency. - Compare the surveillance procedures and the frequency of attention that the instrumentation systems receive with operating experience at this site and at other sites with similar instrumentation to determine if acceptable data recovery with acceptable accuracy is likely throughout the duration of the meteorological program. - Review calibration reports and results to determine sensor stability and accuracy over the period of data collection. Guidelines for acceptable accuracy and acceptable data recovery are specified in Regulatory Guide 1.23, Sections C.4 and 5. Any deviations from Regulatory Guide 1.23 must be justified.					
	<u>Data Acquisition and Reduction</u>					

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	To evaluate data acquisition and reduction, the reviewer should take the following steps: (1) Review the procedures, including both hardware and software, for data acquisition and reduction. Because there are many methods of acquiring data from meteorological measurement systems, the review procedure varies. The following basic components of the program should be reviewed: • accuracy of measuring in units of direct measurement and their precision • accuracy in conversion of direct measurement units to meteorological units • accuracies involved in frequency and mode (instantaneous or average) of sampling • time over which system outputs are averaged for final data disposition and accuracy of these data. (2) Because the instrument accuracy recommendations of Regulatory Guide 1.23 refer to overall system accuracy for instantaneous recorded values or time-averaged values, assess the overall system accuracy in addition to the component (sensor, recorder, and reduction) accuracies. The assessment should consist primarily of using statistical procedures for compound errors based on sensor accuracy, recorder accuracy, conversion of units accuracy, frequency and mode of sampling, and for error reduction by averaging.					
	<u>Data Screening</u> In addition to the checks and calibration of the onsite meteorological instruments,					

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	(1) Screen the recorded meteorological data to evaluate the data quality. (2) Review the data screening programs and program output to determine data quality, data validity, and data recovery rate. Examples of data screening programs are contained in NUREG-0917, Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data (Snell 1982).					
6.5 (Draft Rev. 0, March 2000)	Ecological Monitoring					
	Acceptance criteria for the ecological monitoring programs are based on meeting the intent of the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
6.5.1 (Draft Rev. 0	Terrestrial Ecology and Land Use					
	Acceptance criteria for the review of terrestrial environmental measurements and monitoring programs are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51.50 with respect to conditions and monitoring requirements for protecting the non-aquatic environment related to the issuance of a construction permit, operating license, or combined license.					
	The ER must comply with the requirements of 10 CFR 51.71(c) with respect to the status of compliance with environmental requirements.					
	The ER must comply with the requirements of Coastal Zone					

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	Management Act of 1972 with respect to natural resources and land or water use of the coastal zone.					
	The ER must comply with the requirements of Endangered Species Act of 1973 with respect to identifying and monitoring endangered species.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act of 1958 with respect to consideration of fish and wildlife resources in the planning of development projects that affect water resources.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), details the means by which the applicant should collect the baseline data presented in other sections and should describe the applicant's plans and programs for monitoring the environmental impacts of site preparation, station construction, and station operation. The reviewer should ensure that the applicant's plans for measurement of conditions before site preparation include all environmental parameters that must subsequently be monitored during station operation, as well as during site preparation and station construction.					
	Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998), states that the ecological systems and biota at potential sites and their environs should be sufficiently well known to allow reasonably certain predictions that there would be no significant impacts to the terrestrial ecology associated with the construction or operation of a nuclear-power station at the site. The reviewer should ensure that the applicant's monitoring program is capable of identifying important species or ecological systems and					

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	detecting whether station construction and operation would have any deleterious impacts on these resources.					
	Regulatory Guide 4.11, Rev. 1, Terrestrial Environmental Studies for Nuclear Power Stations (NRC 1977), contains technical information for the design and execution of environmental monitoring studies, the results of which may be appropriate for inclusion in the applicant's environmental report. The reviewer should ensure that the appropriate results are included in the environmental report (ER).					
	ANSI/ANS-18.5-1982 contains guidance and a rationale for performing terrestrial ecological monitoring at each stage of the licensing process and for specific power plant designs. The type, frequency, duration, and magnitude of impacts to terrestrial biota vary with power plant location, design, and methods of construction and operation. Thus, the reviewer should ensure that the applicant's proposed monitoring programs include study of those ecological variables that will most likely be impacted by the construction and operation of the individual power plant.					
	The reviewer should consider the following general stages of the applicant's terrestrial ecology monitoring program:					
	<u>Preapplication Monitoring</u> The program of terrestrial ecological field monitoring is used to support the applicant's descriptions of the terrestrial ecological environment. Preapplication monitoring is needed to support applications for early site permits, construction permits, operating licenses, and combined licenses. Information from the applicant's preapplication monitoring program is used to aid in the assessment of site suitability and to					

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	support the staff's database as needed to identify and evaluate potential impacts to the terrestrial environment that could result from construction or operation of the proposed project. Generally, data are needed on a seasonal basis and should be sufficient to characterize seasonal variations throughout at least one annual cycle. Additional data may be needed on a site-specific basis. (1) Evaluate the preapplication monitoring program to determine that it is adequate to support the environmental descriptions of ESRP 2.4.1. These data should cover the following: • the distribution and abundance of "important" species and habitats. Critical life history information should include parameters such as feeding areas, wintering areas, and migration routes to the extent that the proposed project is expected to affect these parameters. • descriptions of any modifications that may contribute to the existing patterns of plant and animal communities, including agricultural practices, the development of cooling ponds and reservoirs, cooling towers, transmission corridors, and access routes. Except under unusual circumstances, no specific land-use monitoring will be required.					
	<u>Site Preparation and Construction Monitoring</u> This monitoring is appropriate for applications for a construction permit or a combined license and is the proposed program of terrestrial environmental monitoring to control anticipated impacts from site preparation and facility construction. Construction monitoring will be required only when specific					

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	adverse impacts are predicted and when conscientious construction practices coupled with systematic inspection is insufficient to prevent adverse impacts. (1) Determine predicted impacts from the ESRPs 4.1.1, 4.1.2, and 4.3.1. (2) Analyze the proposed monitoring programs associated with these predicted impacts to determine if adequate impact assessment is possible and to determine that adequate mitigation programs can be selected if needed.					
	<u>Preoperational Monitoring</u> A program of terrestrial environmental monitoring may be necessary to establish a baseline for identifying and assessing the environmental impacts to terrestrial biota resulting from plant operation. Preoperational monitoring programs should be evaluated for applications for an operating license or a combined license. The applicant's preoperational monitoring plan should build on the preapplication monitoring program and the site preparation and construction monitoring. The program should be complementary, and if possible, integrated with environmental monitoring conducted in the vicinity of the power station by other agencies not supported by the applicant. The program should be statistically sound and designed to provide an adequate baseline so that the operational monitoring program can detect expected impacts with a degree of confidence commensurate with the risks and costs involved. Where consistent with construction planning, two or more consecutive years of data collection should be planned, and the program should demonstrate a logical extension of both the preapplication and site-preparation					

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	monitoring programs and should be integrated with any required construction monitoring programs. (1) Analyze the program to determine if adequate baseline data will be provided to allow assessment of the following parameters: • for closed-cycle cooling facilities, drift and vapor plume impacts regarding vegetation growth and habitat modification as it affects animals • bird collisions with plant structures or transmission lines and towers • any impacts on "important" species and habitats.					
	<u>Operational Monitoring</u> A program of terrestrial ecological monitoring may be necessary to establish a baseline for use and evaluation of the environmental impacts of continued plant operation. It continues the studies conducted during preoperational monitoring. An operational monitoring program should be included with an application for an operating license, for a combined license, and for license renewal applications. Operational monitoring programs may not be fully developed at the time of applying for a construction permit.					
	<u>General</u> When evaluating the above four types of monitoring programs, the following features should be considered: (1) Ensure that the applicant has, to the extent feasible, described the general scope and objectives of its intended programs and has provided a tentative listing of parameters that it believes should be monitored. The application should include					

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	• the duration over which the parameters will be monitored • provisions for updating the program (included in the applicant's ER). (2) Establish whether adequate data will be provided as outlined above. If the monitoring programs are judged to be inadequate or to include unnecessary elements, the reviewer should evaluate potential additions and deletions. (3) Consider the following features for each of the four types of monitoring programs: • The continuity of design, i.e., each monitoring program should build upon the methodology and informational outputs of the previous program. • The relationship to environmental monitoring conducted by other agencies in the vicinity of the power station should be described. • The bases and objective of each element of the monitoring program should be clearly stated, as well as its relationship to the overall environmental monitoring program. • If outputs of a preceding monitoring program or project demonstrate no significant impacts, then provisions to study such effects in successive monitoring programs should be reduced or deleted. • The program should allow for periodic modification based on the results of previous monitoring to ensure that the current monitoring effort is sufficient and justified when compared to a current assessment of the effects that plant construction and/or operation are having on the environment.					

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	• The intensity of sampling required for each anticipated impact should be commensurate with the degree of impact expected. The reviewer should balance the potential impacts of any sampling program against the potential benefits when making this evaluation. • Measurement and sampling methods, e.g., sampling locations and equipment, the pattern, frequency, and duration of sampling and sample size should be described. • Statistical validity, including the mean, standard deviation, confidence limits, and sample size should be clearly indicated. • If population-dynamics models were used in the impact analyses, determine if sampling data are available to support the model. If not, suggest such sampling if verification of the model is necessary.					
6.5.2 (Draft Rev. 0	Aquatic Ecology					
	Acceptance criteria for the review of aquatic environmental measurements and monitoring programs are based on the relevant requirements of the following regulations:					
	The ER must comply with the requirements of 10 CFR 51.50 with respect to conditions and monitoring requirements for protecting the environment related to the issuance of a construction permit, operating license, or combined license.					
	The ER must comply with the requirements of 10 CFR 51.71(c) with respect to the status of compliance with environmental requirements.					
	The ER must comply with the requirements of Coastal Zone Management Act of 1972 with respect to natural resources, and land or water use of the coastal zone.					
	The ER must comply with the requirements of Endangered					

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	Species Act of 1973 with respect to identifying and monitoring endangered species.					
	The ER must comply with the requirements of Federal Water Pollution Control Act Amendments of 1972 with respect to restoration and maintenance of the chemical, physical, and biological integrity of water resources.					
	The ER must comply with the requirements of Fish and Wildlife Coordination Act of 1958 with respect to consideration and monitoring of fish and wildlife resources and the planning of development projects that affect water resources.					
	The ER must comply with the requirements of Marine Mammal Protection Act of 1972 with respect to the protection of marine mammals.					
	The ER must comply with the requirements of Marine Protection, Research, and Sanctuaries Act of 1972 with respect to dumping of dredged material into the ocean and monitoring marine resources during construction.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), details the means by which the applicant collected the baseline data presented in other sections and should describe the applicant's plans and programs for monitoring the environmental impacts of site preparation, station construction, and station operation. The reviewer should ensure that the applicant's plans for measurement of conditions prior to site preparation include all environmental parameters that must subsequently be monitored during station operation, as well as during site preparation and station construction.					
	Regulatory Guide 4.7, General Site Suitability for Nuclear					

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	Power Stations (NRC 1998), contains guidance that ecological systems and biota at potential sites and their environs be sufficiently well known to allow reasonably certain predictions that there would be no significant impacts to the aquatic ecology associated with the construction or operation of a nuclear power station at the site. The reviewer should ensure that the applicant's monitoring program is capable of identifying important species or ecological systems and detecting whether station construction and operation have any deleterious impacts on these resources.					
	Regulatory Guide 4.11, Rev. 1, Terrestrial Environmental Studies for Nuclear Power Stations (NRC 1977), contains technical information for the design and execution of environmental monitoring studies, the results of which may be appropriate for inclusion in the applicant's ER. The reviewer should ensure that the appropriate results are included in the ER.					
	The program analysis involves the review of the following separate but related aspects of the applicant's aquatic-ecology monitoring program:					
	<u>Preapplication Monitoring</u> The program of aquatic field monitoring is used to support the applicant's descriptions of the aquatic ecological environment. Preapplication monitoring is needed to support applications for early site permits, construction permits, operating licenses, and combined licenses. The applicant's preapplication monitoring program is used to aid in the assessment of site suitability and to support the staff's database as needed to identify and evaluate potential impacts to					

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	the aquatic environment that would result from construction and operation of the proposed project. Generally, data are needed on a seasonal basis and should be sufficient to characterize seasonal variations throughout at least one annual cycle. Additional data (e.g., spawning periods for "important" species) may be needed on a site-specific basis. • Evaluate the preapplication monitoring program to determine that it is adequate to support the environmental descriptions in ESRP 2.4.2. These data should cover the following: - the distribution and abundance of "important" species and habitats. Critical life history information should include parameters such as spawning areas, nursery grounds, food habits, feeding areas wintering areas, and migration routes to the extent that the proposed project is expected to affect these parameters. - descriptions of any modifications that may contribute to the existing patterns of plant and animal communities such as dams, dredging, clearing of stream banks, etc.					
	Site Preparation and Construction Monitoring This monitoring is appropriate for applications for a construction permit or a combined license, and is the proposed program of aquatic environmental monitoring to control anticipated impacts from site preparation and plant construction. Construction monitoring will be required only when specific adverse impacts are predicted and when conscientious construction practices coupled with systematic inspection is insufficient. When evaluating site preparation and construction monitoring,					

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	• determine the predicted impacts from the output of the environmental reviews of ESRPs 4.2 and 4.3.2 • analyze the proposed monitoring programs associated with these predicted impacts to determine if adequate impact assessment is possible and that adequate mitigation programs can be selected if needed.					
	<u>Preoperational Monitoring</u> A program of aquatic environmental monitoring may be necessary to establish a baseline for identifying and assessing the environmental impacts to aquatic biota resulting from plant operation. Preoperational monitoring programs should be evaluated for applications for an operating license or a combined license. Any necessary preoperational monitoring will ordinarily be defined in the NPDES permit. When evaluating preoperational monitoring, analyze the available data to determine that they are adequate to support the environmental descriptions in ESRP 2.4.2, being sure to consider the following: • the location and value of commercial and sport fisheries by species, season, and catch • the distribution and abundance of "important" fish, shellfish, and other invertebrates including benthos. Critical life history information should include spawning areas, nursery grounds, feeding areas, wintering areas, and migration routes. • endangered or threatened species that are known or expected to be present, together with any specific habitat requirements or community interrelationships • the physical, chemical, and biological factors known to					

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	influence the distribution and relative abundance of "important" species • station features and operations that contribute to the existing patterns of plant and animal communities, and that may increase the presence and abundance of nuisance organisms.					
	<u>Operational Monitoring</u> A program of aquatic ecological monitoring may be necessary to establish a baseline for use and evaluation of the environmental impacts of continued plant operation. It continues the studies conducted during preoperational monitoring. Operational monitoring programs should be evaluated for applications for an operating license or a combined license. Any necessary operational monitoring program will be covered under the relevant NPDES permit.					
	<u>General</u> When evaluating these four types of monitoring programs, the following features should be considered: (1) Ensure that the applicant has, to the extent feasible, described the general scope and objectives of its intended programs and provided a tentative listing of parameters that it believes should be monitored. • The application should include the time period over which the parameters will be monitored. • Provisions for updating the program (included in the applicant's ER). (2) Establish whether data will be provided as outlined above. Where the monitoring programs are judged to be inadequate or					

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	to include unnecessary elements, the reviewer should evaluate potential additions and deletions. (3) Consider the following features for each of the four types of monitoring programs: • the continuity of design, i.e., each monitoring program builds upon the methodology and informational outputs of the previous program • the relationship to environmental monitoring conducted by other agencies in the vicinity of the power station • the bases and objective of each element of the monitoring program, as well as its relationship to the overall environmental monitoring program • data from an earlier monitoring program or project. Where data demonstrate no significant impacts, then provisions to study such effects in successive monitoring programs should be reduced or deleted. • The program should allow for periodic modification based on the results of previous monitoring to ensure that the current monitoring effort is sufficient and justified when compared with a current assessment of the effects that plant construction and/or operation are having on the environment. • The intensity of sampling necessary for each anticipated impact should be commensurate with the degree of impact expected. The reviewer should balance the potential impacts of any sampling program against the potential benefits when making this evaluation. • measurement and sampling methods, e.g., sampling locations and equipment; the pattern, frequency, and duration of sampling; and sample size to measure					

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	anticipated impacts • statistical validity, including the mean, standard deviation, and confidence limits. Sample size should be clearly indicated. • If population dynamics models are used in the impact analyses, determine if sampling data are available to support the model and, if they are not available, suggest such sampling if verification of the model is necessary.					
6.6 (Draft Rev. 0	Chemical Monitoring					
	Acceptance criteria for the review of chemical monitoring programs are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 33 CFR 322 with respect to definition of activities requiring permits.					
	The ER must comply with the requirements of 33 CFR 330, Appendix A, with regard to conditions, limitations, and restrictions on construction activities.					
	The ER must comply with the requirements of 40 CFR 6, Appendix A, with regard to procedures on floodplain and wetlands protection.					
	The ER must comply with the requirements of 40 CFR 122 with respect to NPDES permit conditions for discharges including storm-water discharges.					
	The ER must comply with the requirements of 40 CFR 227 with respect to criteria for evaluating environmental impacts.					
	The ER must comply with the requirements of 40 CFR 149 with respect to possible supplemental restrictions on waste disposal and water use in or above a sole source aquifer.					
	The ER must comply with the requirements of 40 CFR 165 with respect to pesticide disposal.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the requirements of 40 CFR 403 with respect to chemical effluents.					
	The ER must comply with the requirements of 40 CFR 423 with respect to effluent limitations on existing and new point sources.					
	The ER must comply with the requirements of Federal, State, regional, local, and affected Native American tribal water laws and water rights.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Compliance with environmental-quality standards and requirements of the Federal Water Pollution Control Act (FWPCA), commonly referred to as the Clean Water Act, is not a substitute for and does not negate the requirement for NRC to weigh the environmental impacts of the proposed action, including any degradation of water quality, and to consider alternatives to the proposed action, which are available for reducing the adverse impacts. If an environmental assessment of aquatic impacts is available from the permitting authority, the NRC will consider the assessment in its determination of the magnitude of the environmental impacts in striking an overall benefit-cost balance. When no such assessment of aquatic impacts is available from the permitting authority, the NRC (possibly in conjunction with the permitting authority and other agencies having relevant expertise) will establish its own impact determination.					
	Since water quality and water supply are interdependent, changes in water quality must be considered simultaneously with changes in water supply. In Jefferson County PUD #1 vs. Department of Ecology (U.S. Supreme Court Case), the U.S. Supreme Court granted the States additional authority to limit					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	hydrological alterations beyond the States' role in regulating water rights.					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the format and content of environmental reports, including hydrology, water-use, and water-quality issues.					
	Documentation of consultations with NPDES administrative agency is necessary to meet the objectives identified above.					
	In this analysis, the reviewer should consider the following separate but related aspects of the applicant's water-quality monitoring program:					
	<u>Preapplication Monitoring</u> The applicant's preapplication monitoring program aids in the assessment of site suitability and supports the staff's description of potential environmental impacts that would result from construction and operation of the proposed facility. Generally, data are needed on a seasonal basis, and descriptions should be sufficient to characterize seasonal variations throughout an annual cycle. The data provided should support the environmental descriptions of hydrology, water use, water quality, aquatic ecology, and plant water supply given in ESRP Chapters 2.0 and 3.0.					
	<u>Construction Monitoring</u> A construction monitoring program may be required by the NPDES administrative agency to provide the data necessary to assess water-quality changes resulting from construction of the proposed project. The time frame for sampling each water-quality parameter should be appropriate for the period of					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	expected change and should include preconstruction monitoring when it is necessary to establish a baseline.					
	<u>Preoperational Monitoring</u> If preapplication monitoring data have not provided an adequate water-quality baseline, a preoperational monitoring program may be required by the NPDES administrative agency. Such a program should provide an adequate baseline so that the operational monitoring program can detect such changes with a degree of confidence commensurate with the risks and costs involved. When consistent with construction planning, two or more consecutive years of data collection should be planned, and the program should demonstrate a logical extension of both the preapplication and site preparation and construction monitoring programs. The reviewer should analyze the ability of the proposed program to characterize the water quality at the site and in the vicinity and thus provide a baseline for the identification and measurement of water-quality changes from station operation.					
	<u>Operational Monitoring</u> The applicant's operational monitoring program identifies changes in water quality resulting from plant operation. Operational monitoring programs update estimates of various effluent treatment systems' effectiveness and provide real time warnings of any failures in the effluent treatment systems. The reviewer should describe the operational monitoring system in terms of the NPDES permitting agency's monitoring requirements. The reviewer should also describe the status of NPDES permit consultations and NPDES permit renewal.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	In evaluating these monitoring programs, the reviewer should take the following steps: (1) Consider whether sufficient and adequate data to accomplish the goals of the monitoring programs will be provided. (a) If the monitoring programs are judged to be inadequate or to include unnecessary elements, consider modifications. (b) Ensure that all such recommendations are consistent with NRC policy and requirements established by the EPA or other State agencies responsible for the NPDES permit. (2) Verify that the following features are described for each of the programs: • Each monitoring program should build upon the methodology and data of the previous program. • If data from an earlier monitoring program or project demonstrate no significant changes in a water-quality parameter, provisions to study such parameters in successive monitoring programs should be reduced or deleted. • The intensity of sampling required for each water-quality parameter should be commensurate with the degree of impact expected. • Sampling equipment, pattern, frequency, duration, and number of samples should be adequate to measure water-quality parameters. • Statistical reliability, including the mean, standard deviation, and confidence limits, should be described. • Data quality objectives, if any, should be described.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• Quality assurance procedures should be described.					
6.7 (Draft Rev. 0	Summary of Monitoring Programs					
	Acceptance criteria for the review of efforts to limit adverse impacts during operation are based on the relevant requirements of the following:					
	The ER must comply with the requirements of 10 CFR 51, Appendix A, with respect to discussion of alternatives and mitigating measures to avoid or minimize adverse impacts.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), contains guidance on the measures planned to reduce undesirable effects of station operation.					
	Regulatory Guide 4.8, Rev. 0, Preparation of Environmental Technical Specifications for Nuclear Power Plants (NRC 1975), contains guidance on the environmental-surveillance program.					
	Regulatory Guide 4.15, Rev. 1, Quality Assumptions for Radiological Monitoring Programs (NRC 1979), contains guidance on quality of measurement results.					
	The reviewer's analysis should consist of identification and tabulation of the applicant's existing and proposed monitoring programs during site preparation and construction during the preoperational and operational stages, as appropriate. When considering this analysis, the reviewer should use the following steps: (1) Prepare a table listing the applicant's existing or proposed monitoring programs by general subject. Provide sufficient					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	program details (e.g., instrumentation, location, sampling frequency) to allow adequate program description. (2) Prepare a summary table describing the combined monitoring program suitable for inclusion in the environmental impact statement (EIS). Identify those program elements that have been defined in response to requirements of other agencies, e.g., NPDES permit conditions. (3) If final program details for preoperational- and operational monitoring programs are not available at the time of the environmental review, tabulate the general program requirements and specify the date or time period when final program details should be available.					
7.0 (Draft Rev. 0, March, 2000)	Environmental Impacts of Postulated Accidents Involving Radioactive Materials					Exclude, Administrative
7.1 (Draft Rev. 0, March, 2000)	Design Basis Accidents					
	The ER must comply with the relevant requirements of 10 CFR 50.34 with respect to the applications for construction permits and operating licenses. This includes an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility.	Refer to the table on 10 CFR.				
	The ER must comply with the relevant requirements of 10 CFR 52.17 with respect to applications for early site permits.	Refer to the table on 10 CFR.				
	The ER must comply with the relevant requirements of 10 CFR	Refer				

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	52.79 for combined licenses with regard to requirements in 10 CFR 50.34 for the analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility.	to the table on 10 CFR.				
	The ER must comply with the relevant requirements of Regulatory Guide 1.3, Rev. 2, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (NRC 1974), with respect to evaluating the potential radiological consequences of a loss-of-coolant accident for boiling-water reactors (BWRs).	Refer to the table on RGs.				
	The ER must comply with the relevant requirements of Regulatory Guide 1.4, Rev. 1, Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (NRC 1973), with respect to evaluating the potential radiological consequences of a loss-of-coolant accident for pressurized-water-reactors (PWRs).	Refer to the table on RGs.				
	The ER must comply with the relevant requirements of Regulatory Guide 1.70, Rev. 3, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition (NRC 1978) with respect to analyses of DBAs other than loss-of-coolant Accidents.	Refer to the table on RGs.				
	The ER must comply with the relevant requirements of Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (NRC 1982) with respect to information on dispersion models.	Refer to the table on RGs.				
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976) with respect to the calculation of χ/Q values for determining offsite dose consequences from postulated accidents.	Refer to the table on RGs.				

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Accidents are categorized as "design basis" or "severe." The DBAs are accidents that the plant is designed specifically to accommodate. The evaluation of DBAs is performed for the NRCGs SER using conservative assumptions.					
	When analyzing doses calculated to result from DBAs, the reviewer should do the following: (1) Examine the applicant's descriptions of accidents considered (as given in the ER) and compare them with the descriptions of accidents given in Appendix A of this ESRP (as taken from Chapter 15 of the SRP) to ensure that all accidents with anticipated offsite-dose consequences have been considered. (a) Coordinate with the reviewer of SRP Chapter 15 to ensure that all appropriate accidents have been identified. (b) Verify that the applicant provides a justification (included in the EIS) for not estimating the consequences of any accident given in Appendix A to this ESRP. (2) Examine the applicant's estimated doses for the appropriate accidents given in Chapter 15 of the SRP. Ensure that the applicant used a 50th percentile χ/Q value that was based on onsite meteorological data, or 10% of the levels given in Regulatory Guide 1.3 or Regulatory Guide 1.4, to represent more realistic dispersion conditions than assumed in the safety evaluation. (3) Determine that the calculation of dose consequences resulting from a DBA to verify that the applicant's proposed exclusion area and low-population-zone distances are adequate to provide a high degree of protection of the public from a variety of potential plant accidents. For construction permit holders before January 10, 1997, a low population zone should be of such a size that an individual located at any point on its outer					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	boundary who is exposed to the radioactive cloud resulting from the release during the entire period of the passage would not receive a total radiation dose to the whole-body in excess of 0.25 sievert (25 rem) or a total radiation dose in excess of 3 sieverts (300 rem) to the thyroid from exposure to iodine (10 CFR 100.11). For all other October 1999 7.1-5 NUREG-1555 applicants, the current siting regulations require an exclusion area of such a size that an individual located for any 2-hour period at the exclusion area boundary would receive a dose that would not be in excess of 0.25 sievert (25 rem) total effective dose equivalent (TEDE). A license to operate the facility would not be granted if the calculated exposures exceed the dose-guideline values.					
7.2 (Draft Rev. 1, July 2007)	Severe Accidents					
	The ER must comply with the relevant requirements of 10 CFR 51.45 with respect to the requirement to address alternatives.					
	The ER must comply with the relevant requirements of 10 CFR 51.50(b) with respect to applications for early site permits.					
	The ER must comply with the relevant requirements of 10 CFR 51.50(c) with respect to applications for combined licenses.					
	Severe accidents are those involving multiple failures of equipment or function and, therefore, the likelihood of occurrence is lower for severe accidents than for design basis accidents, but the consequences of such accidents may be higher. The environmental consequences of severe accidents are estimated using acceptable methodology (such as the MACCS2 code package; Chanin and Young [1997]. The risks for specific accident types are defined as the product of the probability of that type of accident occurring multiplied by the estimated consequences for that type of accident.					
	When analyzing doses calculated to result from severe					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	accidents, the reviewer should do the following: (1) Obtain copies of Chapter 19 of the SER for the reactor design and/or of Chapter 19 of the Design Control Document, Tier 2 (DCD), if the application references a certified design or design undergoing certification. (2) If the application references a certified reactor design or a design undergoing certification, (a) Consult with the reviewer for Chapter 19 of the SER and/or Design Control Document to determine if the information given in the ER on which the applicant's analysis is based is appropriate (release sequences, core damage frequencies, and source terms). Determine if the ER includes release sequences for both internally-initiated and externally-initiated events. Estimates of the core damage frequencies of externally-initiated events are typically provided in the design certification probabilistic risk assessment documentation and should be considered within the sever accident assessment. NUREG-1742 (NRC 2001) provides insights on externally-initiated events for current generation reactors. Otherwise (b) Consult with the reviewer for Chapter 19 of the SAR to determine if the information given in the ER on which the applicant's analysis is based is appropriate (release sequences, core damage frequencies, and source terms). Determine if the ER includes release sequences for both internally-initiated and externally-initiated events. It may be necessary to review other information related to potential releases including core inventory estimates (e.g., from the ORIGEN-ARP code [Bowman and Leal					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	1998]) and estimates of release fractions (e;g; from the RADTRAD code [Humphreys et al. 1998; Bixler and Erickson 1999]), which are used as input to the severe accident consequence assessment. (3) In consultation with the reviewer of the SER and/or DCD, determine if the method (computer code) used to evaluate the environmental consequences is appropriate and that it evaluates consequences to a distance of 80 km (50 mi). If the method used for the consequence assessment is not currently approved or endorsed by NRC, then the method should be evaluated in detail, or the consequences of severe accidents should be evaluated using a method approved or endorsed by the NRC and the results should be compared with the results of the consequence assessment calculated by the applicant. (4) Consult with the reviewers of ESRPs 2.2.1 and 2.2.3 to determine if land fraction and land-use characterization (farm land, etc) used as input to the consequence assessment methodology used to support the ER severe accident analysis are appropriate and consistent with land-use information used elsewhere in the ER. (5) Consult with the reviewer of ESRP 2.3.1 to determine if proposed site is over or near a sole source aquifer. (6) Consult with the reviewer of ESRP 2.3.2 to determine if water-use input to the consequence assessment methodology used to support the ER severe accident analysis is appropriate and consistent with water-use information used elsewhere in the ER (e.g., does the ER include a list of public surface water users within 80 km (50 mi) of the site?)					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(7) Consult with the reviewer of ESRP 2.5 to determine if economic input (land values, relocation costs, cleanup costs, etc) to the consequence assessment methodology used to support the ER severe accident analysis is appropriate. (8) Consult with the reviewer of ESRP 2.5.1 to determine if demographic input to the consequence assessment methodology used to support the ER severe accident analysis is appropriate and consistent with demographic information presented elsewhere in the ER. (9) Consult with the reviewer of ESRP 2.7 to determine if meteorological data input to the consequence assessment methodology used to support the ER severe accident analysis is appropriate and consistent with meteorological information used elsewhere in the ER. (Check to see that the same meteorological data were used for to develop input to evaluation of radiological impacts of normal operations, design-basis accidents, and severe accidents.) (10) Evaluate the protective actions considered by the applicant in its consequence assessment. Were protective actions properly considered? (11) Evaluate the applicant's analysis of consequences associated with the groundwater pathway. Compare the applicant's analysis with the analysis for generic sites presented in NUREG-0440 (NRC 1978) and the analyses for actual sites presented in NUREG-1437 (NRC 1996). (12) Compare severe accident dose risks with the Commission's Safety Goals (NRC 1986) and with the doses estimated for normal operations.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
7.3 (Draft Rev. 1, July 2007)	Severe Accident Mitigation Alternatives					
	The ER must comply with the relevant requirements of the U.S. Court of Appeals decision in Limerick Ecology Action v. NRC 869 F.2d 719 (3rd Cir. 1989) with respect to the requirement that the NRC include consideration of certain SAMAs in environmental impact reviews performed under Section 102(2)(c) of NEPA as part of operating-license applications.					
	The ER must comply with the relevant requirements of 10 CFR 51.45 with respect to the requirement to address alternatives.					
	The ER must comply with the relevant requirements of 10 CFR 51.50(b) with respect to applications for early site permits.					
	The ER must comply with the relevant requirements of 10 CFR 51.50(c) with respect to applications for combined licenses.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are provided in the following:					
	The ER must comply with the relevant requirements of Interim Policy Statement, "Power Plants—Nuclear Power Plant Accident Considerations under NEPA" (1980) with respect to the early consideration of either additional features or other actions that would prevent or mitigate the consequences of serious accidents.					
	The ER must comply with the relevant requirements of "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" (1985) with respect to probabilistic risk assessments for new reactor designs and new plant applications.					
	The ER must comply with the relevant requirements of Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants" (1986) with respect to the safety goals for nuclear power plants.					
	The ER must comply with the relevant requirements of SECY-91-229 (NRC 1991a), which presents alternative courses of action					

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	and the staff*s recommendations concerning the treatment of the SAMA issues to be considered under NEPA as they relate to the certification of standard plant designs, including evolutionary, passive, and advanced reactors.					
	The ER must comply with the relevant requirements of NUREG/BR-0058, Rev. 4 (NRC 2004), although not directly applicable to new reactors, establishes a framework for evaluation of SAMAs including estimation of values and impacts for design alternatives and the "dollars per person-rem" conversion factors.					
	The ER must comply with the relevant requirements of NUREG/BR-0184 (NRC 1997a) with respect to the value impact methodology.					
	The ER must comply with the relevant requirements of Generic Letter 88-20 (NRC 1988) with respect to the performance of an IPE at operating plants for severe-accident vulnerabilities.					
	The ER must comply with the relevant requirements of Generic Letter 88-20, Supplement 3 (NRC 1990), with respect to accident prevention and mitigation features identified in the Containment Performance Improvement Program that may be valid for consideration in the review of SAMA.					
	The ER must comply with the relevant requirements of Generic Letter 88-20, Supplement 4 (NRC 1991b) with respect to conducting an individual plant examination for externally initiated events.					
	The ER must comply with the relevant requirements of Regulatory Guides 1.174 (NRC 2002) and 1.200 (NRC 2007) with respect to general concepts in use and evaluation of probabilistic risk assessments for risk-informed decisions.					
	The ER should also show completeness and reasonableness, also with respect to the following: (1) the identification of SAMAs applicable to the plant or design under consideration, (2) the estimation of core damage frequency reduction and averted					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	person-rem for each SAMA, (3) the estimation of cost for each SAMA, (4) the ranking of value-impact screening criteria to identify SAMAs for further consideration, and (5) the final disposition of promising SAMAs.					
	In addition to the above, the reviewer for ESRP 7.3 should be familiar with industry guidance on SAMA analysis for inclusion in environmental reports submitted with applications for operating license renewal (NEI 2005).					
	When evaluating SAMAs, the reviewer should do the following: (1) Be familiar with analyses previously performed and with the potential process and design alternatives, if any, in previous studies, including the following: • Limerick (NRC 1989) • Watts Bar (NRC 1995) • System 80+ (NRC 1997b) • the Advanced Boiling Water Reactor (ABWR) (NRC 1997c) • the GESSAR II (NRC 1985b) • the Containment Improvement Program • Generic Environmental Impact Statement for License Renewal (NRC 1996) and site-specific supplements. • AP1000 (NRC 2004). (2) Evaluate the applicant's methods for identifying the potential mitigation alternatives. If the applicant used an alternative methodology to a probabilistic risk assessment approach to assess potential SAMAs (e.g., a margins-based approach to evaluate external events initiated by fires or seismic activity), the staff evaluation should be appropriately modified. For example, the synergistic effects of mitigation alternatives that reduce risks for internally initiated events that also provide a benefit for					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	mitigation of externally initiated events should be considered. Alternative benefit-cost approaches are appropriate when a margins method has been used to screen external events. (a) Determine if this set of potential design alternatives and procedural modifications represents a reasonable range of preventive and mitigative alternatives. (b) Verify that the applicant's list of potential SAMAs includes a reasonable range of applicable SAMAs derived from consideration of previous analyses and based on insights from the Level 1 and Level 2 portions of the applicant's PRA or IPE/IPEEE. (3) Evaluate the applicant's basis for estimating the degree to which various alternatives would reduce risk (expressed as a reduction in core damage frequency or in terms of person-rem averted). In performing its independent assessment, the staff may make bounding assumptions to determine the magnitude of the potential risk reduction for each SAMA. (4) Evaluate whether the applicant's cost estimates for each SAMA are reasonable and compare the cost estimates with estimates developed elsewhere (e.g., using previous SAMA evaluations or using accepted cost-estimation tools). (5) Evaluate the benefit-cost comparison to determine if it is consistent with the benefit-cost balance criteria and methodology given in NUREG/BR-0184 (NRC 1997a) and NUREG/BR-0058, Rev. 4 (NRC 2004), and further analyze any SAMAs that are within a decade of the NUREG/BR-0058, Rev. 2, or NUREG/CR-6349 (Mubayi et al. 1995) benefit-cost criteria to ensure that a sufficient margin is present to account for uncertainties in					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	assumptions used to determine the cost and benefit estimates. The benefit-cost criterion in NUREG/BR-0058 is $200,000 per person-sievert averted ($2000 per person-rem averted) for health effects. In addition, a criterion of $300,000 per person-sievert averted ($3000 per person-rem averted) is given in NUREG/CR-6349 (Mubayi et al. 1995) for offsite damage and other related costs for severe accidents. (6) Subject any SAMAs that remain following the screening given above to further probabilistic and deterministic considerations, including a qualitative assessment of the following: • the impact of additional benefits that could accrue for the SAMA if it would be effective in reducing risk from certain external events, as well as internal events • the effects of improvements already made at the plant • any operational disadvantage associated with the potential SAMA.					
7.4 (Draft Rev. 0, March, 2000)	Transportation Accidents					
	The ER must comply with the relevant requirements of 10 CFR 51.52(a) with respect to the design and operational parameters related to the transportation of fuel and waste to and from the reactor.					
	Regulatory guidelines and specific criteria necessary to meet the regulations identified above are as follows:					
	There are no regulatory positions specific to this ESRP. However, there are generic determinations of environmental effects of transportation of fuel with enrichment to 5% uranium-235 by weight irradiated to a maximum of 62,000 megawatt days per ton, provided that the fuel is shipped more than 5 years after discharge from the reactor (NRC 1996, NRC 1999a, 64 FR					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	48496).					
	If the reviewer of ESRP 3.8 determines that the proposed transportation of radioactive materials complies with the provisions of paragraph (a) of 10 CFR 51.52, no further analysis is needed. An additional analysis of transportation accidents should be made when the reviewer of ESRP 3.8 determines that the proposed project does not comply with the following provisions of 10 CFR 51.52(a)(5): Unirradiated fuel is shipped to the reactor by truck; irradiated fuel is shipped from the reactor by truck, rail, or barge; and radioactive waste other than irradiated fuel is shipped from the reactor by truck or rail.					
	When transportation modes differing from the above requirement are proposed, the reviewer should do the following: (1) Prepare an analysis of the modes as they apply to the proposed transportation of new fuel, irradiated fuel, and radioactive wastes. (a) Conduct the analysis to determine whether the proposed transportation modes can result in environmental risks greater than those summarized in the “Accidents in Transport” section of Table S-4 (in 10 CFR 51.52). (b) When it is obvious that the proposed modes do not represent an increased environmental risk, do the following: • Terminate the analysis. • Prepare a statement to the effect that the proposed transportation modes are within the scope of Table S-4. (c) When this is not the case, do the following:					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• Consider the accident probabilities and accident statistics for each proposed transportation mode. • Compare these data with the probabilities and statistics considered in WASH-1238 and Supplement. • Determine to what extent the differences will affect the accident data of Table S-4. (2) When the proposed transportation of radioactive materials does not comply with the provisions of 10 CFR 51.52(a)(1-4), determine the extent to which transportation accidents involving radioactive materials represent an increased probability of risk to the general public over those risks shown in Table S-4, and determine whether this increase is significant. Generic determinations of environmental effects of transportation of fuel with enrichment to 5% uranium-235 by weight irradiated to a maximum of 62,000 megawatt days per ton, provided that the fuel is shipped more than 5 years after discharge from the reactor (NRC 1996, NRC 1999a, 64 FR 48496). These determinations were that the environmental impacts of the transport of irradiated fuel having these characteristics are bounded by the impacts listed in Table S-4. (a) If the increased risk can be shown to be significant, evaluate the possibility of the use of those transportation modes described in 10 CFR 51.52(a)(5). (b) If it is not possible to use these modes, seek other modes that project a lower risk. (c) Ensure that estimated transportation distances for spent fuel have been considered in determining any increased probability of risks.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
8.0 (Draft Rev. 0, March, 2000)	Need for Power					Exclude, Administrative
8.1 (Draft Rev. 1, July 2007)	Description of Power System					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A(4), with respect to discussion of the no-action alternative in NRC EISs.					
	The ER must comply with the relevant requirements of 10 CFR 51.71(d) with respect to weighing the costs and benefits of the proposed action and reasonable alternatives.					
	The ER must comply with the relevant requirements of 10 CFR 51.75(b) and (c) with respect to applications for early site permits and combined licenses, respectively.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to a description of the existing power system.					
	If an independent review of the description of the power system is to be conducted by NRC staff in lieu of using a review prepared by affected States and/or regions or ISO, the procedures discussed below should be followed. These procedures also may be used by the reviewer as an aid in evaluating studies, forecasts and resource plans prepared by others. (1) Obtain the required information for this analysis from • the applicant's environmental report • the applicant's annual report					

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	• data filed by the applicant with FERC, the applicable State public utility commission, and/or the applicable State facility siting authorities, and data and studies filed by the applicant with the relevant NERC reliability council and regional transmission operator (2) Examine the geographical boundaries of the applicant's service area, the power pool or regional transmission organization (if applicable), and the NERC electric reliability region and wholesale power market of which the applicant is a part. Determine the probable competitors for the proposed facility using whatever reputable power market analysis is available, including NERC region reliability assessments and regional transmission organization plans and interconnection requests. (a) Identify major electrical load centers on the map of the relevant service area and transmission paths and constraints to them from the proposed plant location. (b) Examine the current population and the number and types of customers in the relevant service area and fraction with access to competitive retail power suppliers and rates of "choice" within them. (3) Identify the appropriate NERC electric reliability council region. (a) Examine any pertinent power pool and regional transmission operator agreements and reliability studies. (b) Examine the applicant's major power purchases/sales with neighboring utility companies and retail power suppliers.					

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	(c) Examine any wheeling or diversity interchange agreements and any current or proposed intertie agreements. (4) Ensure that the information and data derived from the analysis are adequate to serve as a basis for characterizing the applicant's service and market areas and relevant regional relationships. (a) Identify any unusual features that affect subsequent evaluations of the need for power (e.g., large industrial customers or a noncontiguous service area). (b) Ensure that these features are accounted for and have been explained.					
8.2 (Draft Rev. 0, March 2000)	Power Demand					
	The ER must comply with the relevant requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	There are no regulatory positions specific to this ESRP.					
	The material to be prepared for ESRP 8.2 is informational in nature, and no specific analysis of data is required.					
8.2.1 (Draft Rev. 1, July 2007)	Power and Energy Requirements					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A(4), with respect to discussion of the no-action alternative in NRC environmental impact statements (EISs).					
	The ER must comply with the relevant requirements of 10 CFR					

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	51.71(d) with respect to weighing the costs and benefits of the proposed action and reasonable alternatives.					
	The ER must comply with the relevant requirements of 10 CFR 51.75(b) and (c) with respect to applications for early site permits, combined licenses, construction permits, and operating licenses, respectively.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to electrical demand and projections.					
	If an independent review of power and energy requirements is needed by NRC staff in lieu of using a review prepared by affected States and/or regions, the procedures discussed below should be followed. These procedures also may be used by the reviewer as an aid in evaluating forecasts prepared by others. These procedures assume that the applicant is a traditional utility. Industry best practice may evolve as a result of deregulation. The reviewer should be aware of, and use, industry best practice where possible. In this context, industry best practice is defined by methods used by leading consultants in the field, the Energy Information Administration (EIA), federal power marketing administrations such as the Bonneville Power Administration and including the Tennessee Valley Authority, and leading state and regional power planning organizations, such as in California, New York, and Wisconsin and the Northwest Power and Conservation Council. (1) Analyze the historical data and forecasts of demand factors					

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	for completeness and agreement with other forecasts, emphasizing the forecasted growth in kWh sales in the context of retail electricity prices. Growth rates during periods of flat or declining real retail power prices should be expected to be higher than during periods when prices are increasing. (2) Analyze the forecasting methodologies employed to the extent needed to reach conclusions regarding their acceptability. Relevant factors to consider include the following: • price of electricity and elasticity of demand • energy efficiency and energy substitution including on-site power production from renewables, combined heat and power, etc. • price of alternative fuels • income • economic activity • number of customers • weather • saturation levels of electricity using devices • treatment of uncertainty. (3) Consider how the demand influencing factors are taken into account. If scientific methodologies are employed, determine if they pass standard tests of acceptability (e.g., statistical tests of significance). (4) Analyze any parameter estimates (e.g., price and income elasticities) obtained by the applicant's methodologies to determine the degree to which they agree with other estimates that are generally available for the relevant region from federal					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(e.g., EIA), State, or regional sources. Compare the applicant's latest projections with those made earlier for the same or overlapping time periods. Consider the reasons forecasts for overlapping periods differ. (5) Evaluate the applicant's forecasts and the data and methodology used to make these forecasts and reach one of the following conclusions: (a) The applicant's forecast and all data and methodologies are verified by the staff analyses, and the reviewer concludes that the methodology, underlying assumptions, and results are similar to those that would have been used and obtained by the staff. (b) The applicant's forecasts, methodologies, and data used cannot be verified by the staff using the stated review procedures. In this case, the staff should identify where problems in the review occurred and request additional information.					
8.2.2 (Draft Rev. 1, July 2007)	Factors Affecting Growth of Demand					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A(4), with respect to discussion of the no-action alternative in NRC environmental impact statements (EISs).					
	The ER must comply with the relevant requirements of 10 CFR 51.71(d) with respect to analysis of alternatives and to weighing the costs and benefits of the proposed action and reasonable alternatives.					
	The ER must comply with the relevant requirements of 10 CFR 51.75(b) and (c) with respect to applications for early site permits and combined licenses, respectively.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to electrical demand and projections.					
	If an independent review of need for power is needed by NRC staff in lieu of using a review prepared by affected States and/or regions, the procedures discussed below should be followed. These procedures also may be used by the reviewer as an aid in evaluating forecasts prepared by others. The procedures assume a traditional utility. Industry best practice may evolve as a result of deregulation of the utiliy industry. The reviewer should be aware of, and use, industry best practice where possible. In this context, industry best practice is defined by methods used by leading consultants in the field, the Energy Information Administration (EIA), federal power marketing administrations such as the Bonneville Power Administration and including the Tennessee Valley Authority, and leading state and regional power planning organizations, such as in California, New York, and Wisconsin and the Northwest Power and Conservation Council. Economic and Demographic Trends (1) Analyze the applicant's estimates of the effects of economic, employment, and demographic trends on the applicant's projected growth of electricity demand in the relevant service area. Growth in demand typically follows patterns of growth in population, employment, and income. (2) Obtain or prepare independent forecasts for the economic					

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	and demographic variables identified by the applicant as affecting the rate of growth of electricity demand within the relevant service area. (3) Consider additional variables when it appears that they could affect electricity demand growth. In particular, consider trends in manufacturing employment, out-sourcing, and growth in service industries in relation to energy intensive manufacturing. Forecasts prepared for service areas other than those to be served by the applicant may be used when in the reviewer's judgment they are sufficiently similar to provide a meaningful comparison. (4) For each variable used by the applicant, (a) Compare the applicant's projected growth rates with growth rates developed or obtained by the reviewer. (b) Identify differences. (c) Analyze significant differentials as they contribute either positive or negative effects to the applicant's forecasted growth rate of electricity demand. (5) Compare the historic growth of these variables with the forecasted growth rates, and identify differences as positive or negative influences on projected electricity demand growth.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Energy Efficiency and Substitution (1) Estimate the importance of energy efficiency and substitution in the relevant service area by preparing an estimate of the effect of these factors on projected kilowatt-hour (kWh) sales and peak demand in the relevant service area for the proposed initial year of plant operation (first unit). Consider power production from renewables by customers (including thermal uses such as the use of ground source heat pumps in place of conventional air conditioners, passive solar designs for heating and cooling, and building integrated solar and wind power) and combined heat and power. (a) Contrast this estimate with that of the applicant. (b) Note any significant differences between the two estimates. (c) Calculate the annual compound growth rate in kWh sales and peakload for the last 15 years and compute the increase or decrease in growth rates during the period. Consider historic and projected future electricity growth rates in conjunction with comparable trends and forecasts for retail electricity prices. (2) Identify those elements that could have contributed to diminished growth during the historic period and in the forecast period. The list should include the following • increases in energy efficiency including changes in building and appliance codes • higher prices of electricity and tariffs that encourage conservation and demand reduction					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• economic recession • milder than usual weather. (3) Estimate the relative effects of energy efficiency, price, recession, and weather on diminished growth using the following analyses: (a) Compare the real rate of change in the average price of a kWh of electricity in the service area in the last 15 years and contrast with the real rate of change nationally. (b) Compute the real rate of change in the gross regional product for the relevant service area (or geographic approximation) in the last 15 years with the real rate of increase in gross national product. (c) Review peakload growth in the last 15 years (adjusted for temperature) and discuss positive or negative effects on observed growth rate. (4) Consider the effect of substitution on growth using the following analyses: (a) Review the importance of oil and gas in the relevant service area relative to their availability. Consider any curtailments or denials to new customers (residential, industrial, and commercial) if they exist. Determine the relevant service area's dependence on fossil fuels and the ratio between demand and available supply. (b) Identify trends in new homes (all-electric versus other),					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	purchases of new appliances (electric versus other), and shifts in industrial energy and commercial energy requirements. Determine if electricity is capturing or losing an increasing share of the new and replacement market, and the reasons for the increasing or decreasing share. (5) Determine the extent to which the future substitution between electrical energy and fuels such as oil and natural gas may tend to increase or decrease the demand for electric power and thus offset or reinforce the impacts of energy efficiency measures. (6) Consider any estimates developed by the applicant with respect to the impact of substitution on realized growth rate and determine any adjustments to growth forecasts that may have been made to reflect the substitution. (7) Consider the following factors as they contribute to electricity demand growth: (a) the extent to which technological breakthroughs, government legislation and subsidies, and large energy efficiency investments may provide greater energy efficiency savings than have been experienced in the past paying particular attention to building, appliance, and equipment energy efficiency codes and standards including voluntary programs such as Energy Star and Leadership in Energy and Environmental Design (LEED). (b) the extent to which energy sources (e.g., synthetic natural gas, hydrogen) or energy conversion systems (e.g., renewable power systems and geothermal and solar space heating and cooling systems) currently under development may reasonably					

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	be expected to compete with or significantly reduce the use of electricity. Consult with the reviewer of ESRP 9.2 to complete this portion of the review. (c) the possibility that long-term savings may not be particularly significant if new electricity uses are introduced (e.g., increased availability of plug-in hybrid vehicles). (d) similarly, the possibility that improvements in energy efficiency would result in offsetting electricity savings and thus, decreased use of electric power. (e) the possibility of "double counting" energy savings (e.g., energy efficiency is an economic response and some conservation will be included in price factors, although specific conservation programs, including building codes and standards, will be additive). Price and Rate Structure (1) Determine how and to what extent the applicant has considered price response in demand forecasts. (a) Where the applicant has developed and/or used an econometric model, identify the applicant's price elasticities, forecasted growth rates for the price of electricity, and treatment of price competition. (b) Obtain independent forecasts of growth in the real price of					

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	electricity. (c) Compare these forecasts with the treatment of price in the applicant's analysis. (2) Consider the effects of price competition and alternative rate structures that would moderate load growth or reshape load curves. (a) Consider alternative rate structures such as peakload pricing, inverted rates, marginal cost pricing, and flattened rates. Also consider rate and utility programs that promote use of renewable power, such as green power tariffs that either substitute power from renewable sources for conventional supplies or aggregate supplemental payments by consumers to invest in new renewable power resources. (b) Analyze the relevant region's present attempts and future plans to improve the system load factor via rate restructuring (e.g., higher tail rate during peak periods and demand charges that are based on maximum demand) or valley filling from new electricity uses, such as off-peak charging of vehicle batteries. (c) Estimate anticipated effects on annual electricity consumption and peakload demand. (3) Determine to what extent economic, employment, and demographic trends, energy efficiency and substitution, open competition, and price and rate structure are likely to affect the rate of growth of electrical demand. This determination should be					

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	based on the following information: • the effect of economic and demographic variables on the expected growth of electricity demand with particular emphasis in the aging of existing residents and in-migration of new ones • the effect of energy efficiency improvements and substitution on projected kWh sales and peak demand, especially the impacts from building, appliance, and equipment energy efficiency codes and standards • the effect of price competition with other fuels and on-site generating options and the growth in the real price of electricity on the expected growth of electricity demand • the capability of present and proposed rate structures to promote load management, customer site generation via net metering, and substitution of renewable power for conventional generation. (4) Ensure that the data and analyses submitted by the applicant are accurate and in sufficient detail to allow one to conclude that the forecast submitted by the applicant properly reflects the factors listed above. (a) If the reviewer concludes that the applicant has taken reasonable account of these factors in its forecast, the reviewer can endorse the applicant's forecast. (b) If the reviewer determines by analysis that adequate consideration has not been given to the factors listed above, however the forecast demand is consistent with independent forecasts (see ESRP 8.2.1) that do include these factors, the					

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	reviewer can endorse the applicant's forecast.					
8.3 (Draft Rev. 1, July 2007)	Power Supply					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A(4), with respect to discussion of the no-action alternative in NRC environmental impact statements (EISs).					
	The ER must comply with the relevant requirements of 10 CFR 51.71(d) with respect to analysis of alternatives and to weighing the costs and benefits of the proposed action and reasonable alternatives.					
	The ER must comply with the relevant requirements of 10 CFR 51.75(b) and (c) with respect to applications for early site permits and combined licenses, respectively.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to descriptions of the power system additions, retirements, etc.					
	If an independent review of the need for power is to be conducted by NRC staff in lieu of using a review prepared by affected States and/or regions, the procedures discussed below should be followed. These procedures also may be used by the reviewer as an aid in evaluating resource plans prepared by others. These procedures assume a traditional utility. Industry best practice may evolve as a result of deregulation of the utility industry. The reviewer should be aware of, and use, industry best practice where possible. In this context, industry best practice is defined by methods used by leading consultants in the field, the Energy Information Administration (EIA), federal power marketing administrations such as the Bonneville Power					

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	Administration and including the Tennessee Valley Authority, and leading state and regional power planning organizations, such as in California, New York, and Wisconsin and the Northwest Power and Conservation Council. Current best practice includes development of resource supply curves that rank from low to high prospective supply options (including energy efficiency as a supply option) on the basis of cost (typically net present value) with respective potential quantities of energy and power (see Northwest Power and Conservation Council power plans for a detailed description). Supply curves should facilitate staff comparison of supply options because some resources are inherently limited in terms of capacity and may, therefore, not be adequate substitutes for large central baseload generating plants. Reviews of both applicant materials and materials from others used to verify the applicant's submission need to address need for power in the context of both the utility service area, if the proposed plant is dedicated to utility demand, and the larger regional market where surplus power from the proposed plant could be sold or power from other sources purchased to displace the need for the proposed plant. The following procedures should be applied in an analysis of each of these regions. (1) Segregate the regional plants by fuel type and consider the present and future availability of the indicated fuel. (a) Identify any factors (e.g., air quality regulations or forced outages of long duration) that have affected past plant availability or capacity factor. (b) Consider how these factors may affect planned availability or					

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	capacity factor. (2) Relate the applicant's definitions of baseload, intermediate, and peaking plants to other accepted uses of these terms. Where the applicant's designations do not conform to accepted uses, determine the reason for the differences. (3) Analyze the region's present and planned generation mix in light of the region's present and planned purchases and sales (firm and nonfirm) of power and energy. (a) Include nonfirm purchases and sales of power when considering the capability of the relevant region's power system. (b) Include firm sales and purchases of power when considering the applicant's peakload responsibility. (c) Consider the relevant region's and applicant's role as either a net purchaser or net seller. (d) Quantify shifts in the relevant region's and applicant's position over time, i.e., whether the region and applicant are becoming more dependent or less dependent on purchasing power from or selling power to other systems. (e) Identify and determine the reasons for any unusual purchases or sales that have occurred. Pay particular attention to "load islands" and other transmission constraints. (f) Consider the possibility of a reduction in overall capacity					

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	requirements for the region that could be accomplished by the wheeling and pooling of power and more efficient wholesale power market operations, such as locational pricing. (g) Consider expected trends towards distributed and self-generation by consumers, such as from combined heat and power projects, building integrated renewable such as solar photovoltaic, small wind turbines, and low temperature geothermal generators. In particular, consider state and federal policies facilitating development of these resources including tax and other incentives, renewable portfolio requirements, net metering requirements, and utility programs to reduce peak demand, especially programs that encourage customers to operate customer owned generation during peak demand periods. (4) Where the relevant region plans deratings, redesignations, or retirements (whose total is 200 MW or more) within approximately 2 years before or after the proposed date of commercial operation of the proposed project, determine the reasons for such a change. (a) Determine the reasons for all 100-MW or larger unit redesignations or retirements. (b) Analyze the historical, present, and projected ratio of baseload capacity to total capacity and determine reasons for any large variations in this ratio over time. (5) Determine whether					

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	• the description of present and planned capacity correctly identifies baseload, intermediate, and peaking units and that planned additions are reasonable. • the description of present and planned purchases and sales of power and energy correctly identifies the applicant's capabilities to sell or need to purchase. • plans for redesignation or re-rating of generating capacity have been explained and are reasonable. • the proposed baseload fraction of the applicant's total capacity is appropriate.					
8.4 (Draft Rev. 1, July 2007)	Assessment of Need for Power					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A(4), with respect to discussion of the no-action alternative in NRC environmental impact statements (EISs).					
	The ER must comply with the relevant requirements of 10 CFR 51.71(d) with respect to analysis of alternatives and to weighing the costs and benefits of the proposed action and reasonable alternatives.					
	The ER must comply with the relevant requirements of 10 CFR 51.75(b) and (c) with respect to applications for early site permits and combined licenses, respectively.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the need for new capacity.					
	If an independent review of need for power is to be conducted by NRC staff in lieu of using a review prepared by affected States and/or regions or other independent third-party, the procedures					

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	discussed below should be followed. These procedures also may be used by the reviewer as an aid in evaluating forecasts prepared by others. The procedures assume a traditional utility. Industry best practice may evolve in response to deregulation of the utility industry. The reviewer should be aware of, and use, industry best practice where possible. In this context, best practice is defined by methods used by leading consultants in the field, the Energy Information Administration (EIA), federal power marketing administrations such as the Bonneville Power Administration and including the Tennessee Valley Authority, and leading state and regional power planning organizations, such as California, New York, and Wisconsin and the Northwest Power and Conservation Council. Current best practice includes development of resource supply curves that rank from low to high prospective supply options (including energy efficiency as a supply option) on the basis of cost (typically net present value) with respective potential quantities of energy and power (see Northwest Power and Conservation Council power plans for a detailed description). Supply curves should facilitate staff comparison of supply options because some resources are inherently limited in terms of capacity and may, therefore, not be adequate substitutes for large central baseload generating plants. (1) Calculate baseload demand as that portion of forecasted kilowatt-hour (kWh) sales occurring at loads equal to or less than average load. (a) Forecasted growth in the relevant region(s) as a range: • The forecasted growth rates of kWh sales in this analysis should include at least the applicant's mid-					

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	range, high, low, 75th percentile, and 25th percentile forecasts, and the forecast ranges developed by the affected State and/or region or NRC staff (ESRP 8.2.1). • If the range of reasonable forecasts developed or adopted by the staff (the 25th percentile to 75th percentile range) encompasses the applicant's forecasts of the 25th to 75th percentile range, perform the analysis using the NRC range. • If the range of relevant regional forecasts developed or adopted by the NRC staff is encompassed by in the applicant's 25th percentile to 75th percentile range, perform the analysis using the applicant's range. • If the two ranges partially overlap or one is lower, use the lower of the two ranges. (b) In any case, analyze • reasons for differences between the applicant's forecast and the forecast developed or adopted by the staff • the implications for baseload demand of the extreme value forecasts. (2) Analyze the power supply data (e.g., capacity factors, variable costs, and redesignations) and estimate the baseload capacity of the system using the evaluation of ESRP 8.3. (3) Compare the supply of baseload capacity with the demand for baseload capacity for the first 3 years of commercial operation of all proposed units. (4) Identify the reserve margin requirements currently in					

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	acceptance for the service area and identify the organization responsible for establishing this requirement. (a) Determine if the reserve margin requirements at the time the proposed units are scheduled to begin operation are different from the current reserve margin requirements. (b) Contact the appropriate regional reliability council, other regional bodies, power pools, and FERC to compare this reserve margin requirement with requirements recommended by these organizations. (5) Calculate the region's accredited generating capacity (i.e., total installed capacity plus nonfirm purchases and less nonfirm sales) for the period extending from 1 year preceding commercial operation of the proposed first unit to the 3rd year of commercial operation of the proposed last unit. (6) Calculate peakload responsibility based on the growth rates for peakload demand calculated for ESRP 8.2.1. (7) For reviews requiring additional staff analysis, calculate peakload responsibility based on forecasted growth rates for peakload demand. (a) Determine these by contrasting the applicant's projected range of growth rates for system peakload with the range of growth rates developed or adopted by the staff for the system peak.					

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	The same rules for comparison apply as for annual kWh sales: • If the range of reasonable forecasts developed or adopted by the staff encompasses the applicant's forecast, the reviewer should perform the analysis using the developed or adopted forecast. • If the range of forecasts falls below the applicant's forecast(s), the reviewer should use the staff forecasts. (8) For each estimate of peakload responsibility and for each year under consideration, calculate reserve margin as Based on the reserve margins and the projections for baseload demand, determine the timespan representing the probable dates when plant capacity will initially be needed. (9) Prepare an analysis of the costs and benefits of not having sufficient and timely capacity additions and also the costs and benefits of adding capacity too soon. (a) For these purposes, assume the applicant's proposed date of commercial operation of all proposed units and consider the effects of the load materializing 3 years earlier than this date and 3 years later than this date. (b) The 6-year timespan may be shifted if conditions specific to the service area suggest this to be appropriate. Treatment of this subject should include, at a minimum,					

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	participation by the socioeconomic and benefit-cost reviewers. (10) If a need-for-power analysis conducted by or for one or more relevant regions affected by the proposed plant concludes there is a need for new generating capacity, that finding should be given great weight provided that the analysis was systematic, comprehensive, subject to confirmation, and responsive to forecast uncertainty. This source may be the most appropriate if the proposed plant is not planned to serve a traditional utility load or as a retail power supplier in a specific region, but is expected to provide power as a merchant plant to a regional wholesale power market. In this case, the analysis of the relevant market should include an assessment of competitors to the proposed plant. If no such analysis is available, determine whether the projected peakload responsibility plus the reserve requirement exceeds the total accredited generating capacity and, absent special circumstances, these findings justify the conclusion that new capacity is warranted. Although this criterion does not show a need for baseload capacity, it does demonstrate a need for new capacity that is independent of type. This criterion, coupled with an affirmative indication that there is a need for baseload capacity, justifies a baseload addition within the timespan determined by the reviewer's forecast analysis. (11) If these criteria cannot be met, it may still be possible that the proposed facility will be needed on some other basis. The analysis should be summarized in a table similar to Table 8.4-3.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Additional considerations include the following: • the relevant region's need to diversify sources of energy (e.g., using a mix of nuclear fuel and coal for baseload generation) • the potential to reduce the average cost of electricity to consumers • the nationwide need to reduce reliance on imported petroleum • the case of a significant benefit-cost advantage being associated with plant operation before system demand for the plant capacity develops. (This will require the reviewer's benefit-cost evaluation of the consequences of not having sufficient baseload capacity or of adding this capacity too soon.) If none of the above criteria can be satisfied, it may be concluded that there is no need for additional baseload generating capability on the scale represented by the applicant's proposal during the timespan considered.					
9.0 (Draft Rev. 0, March, 2000)	Alternatives to the Proposed Action					Exclude, Administrative
9.1 (Draft Rev. 0, March, 2000)	No-Action Alternative					
	Acceptance criteria for the review of the no-action alternative is based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A to subpart A, with respect to including analysis of alternatives to the proposed action.					

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	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to discussing the effect of no action on increasing generating capacity.					
	The reviewer should establish the validity of the forecast data for the energy consequences expected as a result of not building the proposed facility and taking no alternative actions, such as the development of alternative energy sources, or the use of conservation measures.					
9.2 (Draft Rev. 0, March 2000)	Energy Alternatives					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the relevant requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
9.2.1 (Draft Rev. 1, July 2007)	Alternatives Not Requiring New Generating Capacity					
	Acceptance criteria for the review of alternatives not requiring new generating capacity are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.71(d) and 10 CFR 51, Appendix A to Subpart A, with respect to including analysis of alternatives to the proposed action in					

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	the EIS.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the analysis of alternatives to adding new generating capacity.					
	The analysis includes two separate evaluations: the first of power purchases and reactivation and the second of energy efficiency. Projections by Federal, State, regional, local, and affected Native American tribal agencies energy planners may be the most useful source of capacity and demand information available. The reviewer should consult current NRC policies regarding these evaluations for alternative analyses. The extent of this analysis should be determined by the amount and cost of capacity available through combinations of purchases of power and reactivating or extending the service life of plants within the relevant regional system. To make this determination, the reviewer should conduct a brief initial analysis following the procedures in the following subsections to identify the probable amount of electrical generating capacity available.					
	<u>Power Purchases</u> The reviewer should determine if excess generating capacity (capacity beyond reserve margin requirements) will be available for extended periods of time from other sources. The time period to be considered for determining this availability should cover a 6-year period starting with the expected first year of commercial					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	operation of the proposed project. Excess generating capacity of these utilities and/or systems should be summed and compared with the capacity need established by the reviewer of ESRP 8.4. If sufficient excess capacity has been identified to warrant continuation of this review, the reviewer should do the following: (1) Determine if adequate transmission line interties exist for the efficient transfer of this power. (2) Determine the administrative structure of the current generating supply system in the relevant regional grid and the applicant's relationship to this structure in terms of current and projected power supply. Full account should be taken of nondiscriminatory access rules as promulgated by the Federal Energy Regulatory Commission (FERC). (3) Consult with the reviewer for ESRP 3.7 to identify existing transmission lines and corridors within the region. (4) If transmission lines and interties are not available, make general estimates of the costs to construct and maintain such lines and estimates of the environmental impacts associated with their construction and maintenance.					
	<u>Plant Reactivation or Extended Service Life</u> To review the relevant regional (e.g., power pool, power marketing area, major utility service area) inventory of the available generating plants, the reviewer should do the following: (1) Identify plants now deactivated but potentially operable.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(2) Identify plants scheduled for retirement during the period extending from the date of application through the 6th year of commercial operation of the proposed project. In considering alternatives, the reviewer should be guided by FERC practice to define relevant markets as those utilities and power generators directly interconnected to the applicant (first-tier markets). For each first-tier market, FERC considers all utilities interconnected to the first-tier utility and all utilities interconnected to the applicant as competitors in that relevant market. Thus, the competitors usually are assumed to include the second-tier utilities that can reach the market by virtue of the applicant's open-access transmission tariff. FERC admits that the open-access rule (61 Federal Register 21540) may lead to consideration of an area broader in scope than the first-tier and second-tier markets currently considered. However, evidence of transmission constraints may circumscribe the scope of the relevant market. FERC permits applicants and intervenors to argue that the market is broader or narrower than that offered by second-tier utilities. The argument must be more than open access and involves transmission constraints and cumulative transmission costs. When sufficient capacity is identified to warrant further analysis, the reviewer should review the estimate of the environmental and operating costs associated with the use of these plants. Factors to be considered in preparing these cost estimates should include the • capital costs needed to reactivate retired plants and to upgrade existing plants, when necessary, to comply with current standards					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• operating costs, including costs associated with meeting current environmental standards (these costs should be adjusted to account for reduced availability factors where applicable) • environmental costs, including the environmental impacts associated with alternative-energy sources.					
	<u>Conservation (Energy Efficiency)</u> The reviewer's analysis of conservation (increased energy efficiency) as an alternative to construction of the proposed plant should be based on the analysis and evaluation of conservation and substitution received from the reviewer for ESRP 8.2.2. Except for unusual circumstances, no additional review should be required to complete this portion of this ESRP, since the reviewers for ESRP 8.2.2 and 8.4, in the process of analyzing and evaluating the need for the plant, should make a determination that conservation is or is not a practical alternative to the proposed plant. The reviewer should consult with and assist the reviewer for ESRP 8.2.2 in analyzing the effects of conservation on the need for the plant and to prepare data for inclusion in this section of the EIS. The reviewer does not need to analyze the potential for conservation if the applicant is proposing to build a merchant plant to sell electric power on the open market and did not address the potential for conservation in the ER (Exelon Generation Co., LLC 2005). The reviewer should review the relevant regional (e.g., power pool, power marketing area, major utility service area) summation of the total amount of alternative electrical generating capacity available through a combination of purchased power and the reactivation and extended service life of plants within the regional system. If this combined capacity is insufficient to meet the capacity needs through the 6th year of commercial operation					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	of the proposed project, the reviewer may conclude that this alternative is not feasible. Where sufficient capacity is available, the reviewer should consider whether there are any factors unique to the relevant regional system that could prevent the reactivation or extended service life of existing units or the purchase of power from other systems. The reviewer should ensure that cost data associated with this alternative, including purchases of power, transmission line costs, capital/operating costs and environmental compliance costs of reactivated and extended service life plants, are available and can be compared with the costs of the proposed project. These cost data should be used by the reviewer for ESRP 9.2.3. Where sufficient electrical generating capacity is available to meet the need established by the reviewers for ESRP Chapter 8.0, and the costs of the alternative are reasonable when compared to costs of the proposed project, the reviewer of ESRP 9.2.1 should provide this assessment to the reviewer of ESRP 9.2.3. However, when costs of this alternative are significantly greater than costs of the proposed project, the reviewer, after consulting with the reviewers for ESRP 10.4, may conclude that the alternative is not practical. When the reviewer has determined that the alternatives of conservation, power plant reactivation and life extension, and power import have been adequately described and explored, this information should be included in the environmental impact statement (EIS) and communicated to the reviewer of ESRP 9.2.3 for analysis of alternatives.					
9.2.2 (Draft Rev. 1, July	Alternatives Requiring New Generating Capacity					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
2007)						
	Acceptance criteria for the review of alternatives requiring new generating capacity are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.71(d) and 10 CFR 51, Appendix A to Subpart A, with respect to the need to discuss alternatives to the proposed action in the EIS.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the analysis of alternatives requiring new generating capacity.					
	The reviewer should review the alternative energy sources and combinations of sources available to the applicant, and categorize them as either competitive or noncompetitive with the proposed project. A competitive alternative is one that is feasible and compares favorably with the proposed project in terms of environmental and health impacts. If the proposed project is intended to supply baseload power, a competitive alternative would also need to be capable of supplying baseload power. A competitive alternative could be composed of combinations of individual alternatives. (1) For competitive alternatives, the reviewer should ensure that the energy source or system meets the following criteria: • The energy conversion technology should be developed, proven, and available in the relevant region.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• The alternative energy source should provide generating capacity substantially equivalent to the capacity need established by the reviewer of ESRP 8.4. • The capacity should be available within the timeframe determined for the proposed project. • Use of the energy source is in accord with national policy goals for energy use. • Federal, State, or local regulations do not prohibit or restrict the use of the energy source. • There are no unusual environmental impacts or exceptional costs associated with the energy source that would make it impractical. • The reviewer should ensure that the following energy sources have been considered by the applicant: - wind - geothermal - natural gas - hydropower - municipal solid wastes - biomass - coal - photovoltaic cells - solar thermal power - wood waste - energy crops - other advanced systems (e.g. fuel cells, synthetic fuels, etc.). • The reviewer should ensure that all alternative energy sources available have been evaluated using the criteria listed above to determine if the alternatives can be considered competitive with the proposed project.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(2) For noncompetitive alternatives, the reviewer should ensure that the statements dismissing these alternatives are appropriately referenced, applied to the relevant regional system, and that the reasons for rejecting these alternatives have been provided. (3) For alternative energy sources, the reviewer should evaluate the applicant's or regional authority's analysis of each energy source to determine that it describes the source plant combination in sufficient detail to enable the reviewer of ESRP 9.2.3 to compare the environmental and social costs of this alternative with the proposed project. Specific analytical procedures should depend on the alternative. The reviewer should evaluate the analysis procedure in consultation with the reviewers of ESRP 9.2.3 (for analysis requirements) and ESRP Chapter 2.0 (for environmental descriptions and socioeconomic data). (4) For the alternatives considered competitive, the reviewer should ensure that there are suitable sites for an alternative plant and should determine the general characteristics of such a site-plant combination. The results of this analysis should be used by the reviewer of ESRP 9.2.3 in determining the impacts and costs (environmental, health, capital and operating costs, etc.) of the alternative and comparing them with the impacts and costs of the proposed project. Based on an appropriate site (this may include the proposed nuclear plant site) and the energy sources identified, the reviewer should consider the following: • distance from the fuel sources to the plant, probable transportation means, and mileages for each transportation means					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• average daily fuel requirements based on the installed capacity need determined by the reviewer for ESRP 8.4 and the heat content • need for fuel pretreatment (e.g., washing), if any, including the volumes of materials (water) required, the quantities of wastes produced, and means of waste disposal. Also include estimated effects of fuel source preparation on fuel characteristics, quantities of water required, and quantities of wastes produced. • in the case of coal or other solids as the preferred alternative to the proposed project, need for combustion-product solid waste disposal, including the quantities of wastes produced and disposal methods and locations for deposition of solid waste • need for flue-gas desulfurization, the process to be used, and (on an average daily basis), the raw material inputs and byproduct and/or waste product outputs and means of waste disposal • average daily atmospheric releases of carbon dioxide (CO_2) and pollutants of concern regulated under the Clean Air Act (including total suspended particulates [TSP], sulfur oxides [SOx], and nitrogen oxides [NOx]. (5) For alternatives that have been determined to be competitive, the reviewer should ensure that sufficient data are available to permit the reviewer of ESRP 9.2.3 to compare the environmental impacts and costs of these alternatives with costs of the proposed project. (6) For each alternative established as noncompetitive, a brief statement should be prepared describing or identifying the alternative and the basis for the staff's conclusion that it was					

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	noncompetitive.					
9.2.3 (Draft Rev. 1, July 2007)	Assessment of Alternative Energy Sources and Systems					
	Acceptance criteria for the review of energy alternatives are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 40 CFR 1502.14 with respect to "alternatives including the proposed action."					
	The ER must comply with the relevant requirements of 10 CFR 51.71(d) and 10 CFR 51, Appendix A to Subpart A with respect to the need to discuss alternatives to the proposed action in the EIS.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the analysis of alternative energy sources.					
	The analysis of competitive alternatives is a two-step process: (1) comparing estimated environmental impacts and health effects, and (2) considering estimated economic costs. To accomplish this, the reviewer should (1) Compare estimated environmental impacts and health effects for the proposed project and each competitive alternative. (2) Consider the economic costs of each competitive alternative deemed to be environmentally preferable to the proposed action. This analysis should be conducted in consultation with appropriate ESRP 10.4 reviewers. Assistance from these					

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	reviewers will be needed to establish the economic-cost data that should be used to develop a benefit-cost comparison with the baseline proposed project. For some costs, a range of costs may be preferable to a point value, particularly when there is considerable uncertainty in the data. To the extent practical, the analysis should be made with the objective of presenting the cost comparisons in tabular form. (3) Compile a tabular summary of the staff's characterization of the environmental and health impacts of the proposed action and the competitive alternative(s) (see Table 9.2.3-1 for an example). The characterization should use NRC's SMALL/MODERATE/LARGE characterizations as set out in the Introduction to NUREG-1555. Input for the characterizations should be obtained from the ESRP Ch. 4 and 5 reviewers and the reviewers of ESRP 9.2.1 and 9.2.2. (4) The economic cost data to be analyzed for competitive alternatives deemed to be environmentally preferable to the proposed action are the estimated costs of supplying electrical energy services over the expected life of the proposed project. The data should span 40 years unless there are unique factors that apply to the specific competitive alternative(s) under review. In the case of options involving generation, the 40-year levelized cost should be analyzed at appropriate plant capacity factors. The cost comparison between uranium and the alternative fuel should be developed in a tabular form such as shown in Table 9.2.3-2. The reviewer should review the applicant's cost calculations and ensure that they are reasonable. The other tables provided in this ESRP include worksheets that can assist in this evaluation.					
9.3 (Draft Rev.	Alternative Sites					

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1, July 2007)						
	Acceptance criteria for the review of alternative sites are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.45 with respect to the contents of an ER and the need to discuss alternatives.					
	The ER must comply with the relevant requirements of 10 CFR 51.50(b) or (c) with respect to the evaluation of alternative sites in the ER for early site permit or combined license applications, respectively.					
	The ER must comply with the relevant requirements of 10 CFR 51.71 with respect to discussion of alternatives in draft NRC environmental impact statements.					
	The ER must comply with the relevant requirements of 10 CFR 51.75(b) or (c) with respect to review of early site permit or combined license applications, respectively, to determine whether there is any obviously superior alternative to the site proposed.					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A, with respect to alternatives including the proposed action.					
	The ER must comply with the relevant requirements of Federal, State, local, and Native American Tribal laws and regulations affecting the siting of new energy facilities.					
	Regulatory positions and specific criteria necessary to meet the acceptance criteria include:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to selecting suitable plant sites.					
	The ER must comply with the relevant requirements of Office of Nuclear Reactor Regulation (NRR), Office Instruction No. LIC-					

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	203, Revision 1, "Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Issues."					
	The ER must comply with the relevant requirements of Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998), with respect to evaluating site selection in terms of ecological systems, biota, and environmental justice.					
	This review should accomplish the following objectives: (1) a brief description and evaluation of the applicant's site selection process, (2) presentation of the basis for the staff analysis, and (3) presentation of staff conclusions regarding alternatives to the proposed site. The fact that State authorities have approved the environmental acceptability of a site or a project after extensive and thorough environmentally sensitive hearings is properly entitled to "substantial weight" in this review. The staff 's analysis of alternative sites is a critical element of the environmental review. Under the general guidance and direction of the EPM, the reviewer(a) should analyze the applicant's site selection process and procedures. The subsections that follow explain the review process for (1) the ROI, (2) the candidate areas, (3) the potential sites, (4) the candidates sites, and (5) the proposed and alternative sites, in turn. The overall goal of the review is to understand the applicant's site-selection methodology so that an eventual evaluation can be made of the reasonableness and capability of this process to identify candidate sites that are among the best that can reasonably be found in the ROI. The reviewer's evaluation of the individual elements of the applicant's site-selection process should include consideration of both the process (i.e., methodology) used by the applicant and					

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	the reasonableness of the product (e.g., potential sites) identified by that process. After the candidate sites have been identified, the review involves a two-part sequential test for obvious superiority. The first stage of the test determines whether there are environmentally preferred sites among the alternative sites. The second stage of the test considers economics, technology, and institutional factors among the environmentally preferred sites to see if any is obviously superior to the proposed site. If there is no environmentally preferred or obviously superior site, the proposed site prevails. If an environmentally preferred site is found, the reviewer should consult with the Environmental Project Manager (EPM). A staff conclusion that an alternative site is obviously superior to the applicant's proposed site would normally lead to a recommendation that the application be denied. The following general guidance is provided for the reviewer in arriving at conclusions: • The reviewer should determine if the applicant has employed a practicable site-selection process with the principal objective of identifying candidate sites that would be among the best that could reasonably be found for the proposed plant. This standard implies that all such candidate sites should be licensable (which includes consideration of whether other necessary Federal, State, and local permits could be obtained). The reviewer should determine if the applicant's proposed site was selected from this list of candidate sites. The reviewer should determine whether the reconnaissance-level information used throughout the					

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	site-selection process was complete enough and of sufficient depth commensurate with the level of screening to support the decisions that were made. • The reviewer should determine if the applicant's candidate sites represent among the best that could reasonably have been found within the ROI, and if they do not, should request further information from the applicant. If the sites are among the best that could be found, the reviewer should determine if any such site is environmentally preferable to the applicant's proposed site. When such a determination is made, the reviewer should conduct a benefit-cost balance and comparison of the estimated costs (environmental, economic, and time) of completing construction of the proposed plant at the proposed site and at the environmentally preferable site or sites. The reviewer should use the results of this benefit-cost balance to determine if any environmentally preferable site can be shown to be obviously superior to the applicant's proposed site. The reviewer should use the following specific guidance during the review:					
	<u>Objectives and Procedures</u> The reviewer should ensure that the applicant's site-selection process was based on a documented procedure that includes as a minimum those elements described below.					
	<u>Region of Interest</u> Review and analyze the ROI selected by the applicant so that an eventual evaluation of the appropriateness (e.g., in terms of geographical, demographic, legal, regulatory, and institutional restrictions) of the selected region can be made.					

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	The ROI is typically selected based on geographic boundaries (e.g., the State in which the proposed site is located) or the relevant service area for the proposed plant. In cases where the proposed plant would not have a service area, the applicant should define a reasonable ROI and provide a justification. The ROI must be more extensive if environmental diversity would be substantially improved or if candidate sites do not meet initial threshold criteria (including the site criteria in 10 CFR 100), and added geographic areas likely would not increase costs substantially. The ROI may be smaller if sufficient environmental diversity exists, threshold criteria are satisfied, and costs would be exorbitant for considering sites outside the State or relevant service area. The reviewer should ensure that the selected ROI has been adequately described and that its boundaries are consistent with those factors outlined in the preceding paragraph. In making this determination, the reviewer should consider (1) how the applicant's ROI compares with the available geographical area, (2) the extent of and basis for restrictions to the ROI because of siting constraints, and (3) whether the ROI is consistent with the major load centers to be supplied by the proposed plant. As a general rule, the plant should be located at a site in the area of the load center or centers that the plant is to serve over its lifetime. The reviewer should determine if the selected ROI will permit such siting and that potentially desirable candidate areas have not been excluded on the basis of an arbitrarily defined ROI.					
	<u>Process for Identifying Candidate Area(s)</u> Review and analyze the candidate area(s) selected by the applicant so that an eventual evaluation of the appropriateness					

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	(e.g., in terms of safety considerations, prohibited areas, geographic or engineering restrictions, and environmental restrictions) of the selected candidate area(s) can be made. The candidate area(s) are a subset of the ROI, after unsuitable areas in the ROI are removed from consideration. Reasons that areas may be unsuitable include: • proximity to major centers of population density • lack of existing infrastructure (e.g., roads, railroads) • lack of a suitable cooling water source • distance to transmission lines, substations, or load centers • unsuitable topographic features (e.g., mountains, marshes, fault lines) • potential to impact valuable agricultural, residential, or industrial areas • potential to impact dedicated land-use areas (e.g., parks, historical sites, wilderness areas, testing grounds) • conflicts with land-use planning programs or other restrictions established by State, county, or local governments The applicant's process to identify candidate areas should consider these and other reasonable attributes in order to identify areas that are, or are not, potentially suitable for siting a new nuclear power plant. Only the determining characteristics of the identified areas need be presented in the ER. For example, if an area has no suitable cooling water source, then the area would be considered unsuitable and the other factors listed above need not be considered. This step of the site selection process is performed at a high level with the purpose of					

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	quickly identifying areas within the ROI that would not be suitable for the siting of a new nuclear power plant.					
	<u>Process for Identifying Potential Sites</u> Analyze the reconnaissance-level information available on Geographical, Environmental, and Siting Information System (GEn&SIS) or from other sources used in this portion of the site-selection process so that an eventual evaluation can be made of whether the information is adequate for the level of screening for which it is used. Review the potential sites identified by the applicant so that an eventual evaluation can be made with respect to (a) adequacy of the site-identification process, and (b) consistency with the applicant's criteria for site selection. The process used to identify potential sites considers attributes similar to those used in the process of identifying candidate areas. However, in general this step in the process requires a somewhat more detailed look at those criteria. In addition, in many cases the applicant will use the inverse of the attributes listed above, looking for positive rather than negative attributes. So, for example, the applicant would be looking for locations in the candidate area(s) that have ample water, are close to transmission facilities and load centers, have infrastructure in place, etc. However, negative attributes at a specific location (e.g., seismicity, threatened and endangered species) likely will also be used to de-select some sites. The goal of this step in the process is not to identify every potential site in the candidate area(s). Depending on the size of the candidate area(s), trying to identify all possible sites would yield an unworkable number of possible locations. However, the					

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	staff needs to determine whether the applicant used a logical process that would reasonably be expected to produce a list of the best potential sites in the candidate area(s).					
	<u>Process for Identifying Candidate Sites</u> Analyze the reconnaissance-level information available on Geographical, Environmental, and Siting Information System (GEn&SIS) or from other sources used in this portion of the site-selection process so that an eventual evaluation can be made of whether the information is adequate for the level of screening for which it is used. Analyze the applicant's candidate sites to the level needed to conclude that they are or are not potentially licensable sites and to identify the potential environmental impacts (adverse and beneficial) attributable to each site that would be used (a) by the applicant to select the proposed site, and (b) by the reviewer to determine the possible existence of an obviously superior site. The reviewer should analyze the applicant's site selection criteria from the viewpoint of their applicability to a wide variety of candidate sites and their value in permitting comparisons of potential impacts. The reviewer should determine if the selection process used to identify candidate sites was adequate. Sites may be selected on the basis of a screening process to identify unacceptable areas (e.g., population density) or on the basis of positive attributes. A table similar to Table 9.3-1 may be used by the reviewer to document the process of candidate site selection and screening. The reviewer should ensure that factors identified below have been considered by the applicant.					

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	To be a candidate site, the following minimum criteria must be met: • Consumptive use of water should not cause significant adverse effects on other users. • The proposed action should not jeopardize Federal, State, and affected Native American tribal listed threatened, endangered, or candidates species or result in the destruction or adverse modification of critical habitat. • There should not be any potential significant impacts to spawning grounds or nursery areas of populations of important aquatic species on Federal, State, and affected Native American tribal lists. • Discharges of effluents into waterways should be in accordance with Federal, State, regional, local, and affected Native American tribal regulations and would not adversely impact efforts to meet water-quality objectives. • There should be no preemption of or adverse impacts on land specially designated for environmental, recreational, or other special purposes. • There would not be any potential significant impact on terrestrial and aquatic ecosystems, including wetlands, which are unique to the resource area. • There are no other significant issues that preclude the use of the site. The reviewer should determine if an adequate, well documented process for screening potential sites was employed, and that all potential sites were screened in a consistent manner. The reviewer should consider all screening criteria employed by the applicant in light of the objective of this process (i.e., to identify					

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	potentially licensable sites). The reviewer should compare the applicant's procedures with the recommendations of Regulatory Guide 4.7 and, when inconsistent, should coordinate with the EPM to determine the reasons for the variances. Based on reconnaissance level information, the reviewer should determine if the candidate sites identified by the screening process may be considered as potentially licensable and should also determine that the applicant's process provides reasonable assurance that potentially licensable candidate sites have not been omitted. Although there can be no specific criteria for determining that an adequate number of candidate sites have been identified, the reviewer should make such a determination, based on the ROI, the number of candidate areas, and the number and type of alternative sites evaluated by the applicant. In general, however, the identification of two or more different areas and three to five alternative sites in addition to the proposed site could be viewed as adequate.					
	<u>Proposed and Alternative Sites</u> The objective of this phase of the evaluation procedure is (1) to determine if the applicant has reasonably identified alternative sites(a), predicted the environmental impacts of construction and operation at these sites, and developed and used a logical, reproducible means of comparing sites that has led to the applicant's selection of the proposed site, and (2) to determine if any alternative site can be shown to be environmentally preferable, and if so, obviously superior to the applicant's proposed site. This analysis may be documented in a table such as Table 9.3.2, which records summary environmental information on each alternative site; the conclusion of environmental preferability for any sites; consideration of other factors; and any identification of an obviously superior site. Costs					

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	associated with alternative sites only need to be evaluated for alternative sites found to be environmentally preferable to the proposed site. Many of the following evaluation steps must be based on the reviewer's judgment. For these evaluations, the principal criterion will be the reasonableness of the applicant's data and procedures. The reviewer should make the following determinations:
	
	• Site Identification—The reviewer should determine that the alternative sites have been identified with sufficient precision to permit field inspections and to estimate environmental parameters. If the applicant is unable to provide precise alternative site boundaries, and if the reviewer determines that the reasons for this are valid, the reviewer should evaluate the general site area instead.
	• Environmental Descriptions—The reviewer should determine that environmental descriptions for the alternative sites are adequate to assess environmental impacts of plant construction and operation, and that the basic sources of information described below have been used to provide these data:
	    ○ review of the literature
	    ○ reports from Federal, State, regional, local, and affected Native American tribal agencies such as State geological agencies, EPA, U.S. Department of Agriculture, or county extension offices
	    ○ regional scientific, engineering, economic, and planning studies
	    ○ aerial photographs and topographic maps of candidate sites
	    ○ site-specific information from local citizens and					

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	from authorities associated with Federal, State, regional, local, and affected Native American tribal agencies, universities, and museums ○ onsite inspections (if any) by technical specialists. • Site Comparison by Applicant—The reviewer should determine that the applicant's final site-selection process is reasonable, makes full use of the candidate site data available, and presents the data in a manner that permits valid comparisons between sites. The objective of this evaluation of the applicant's process is not to determine that the applicant has selected the best site (since on the basis of previous evaluations, the reviewer has determined those candidate sites that can reasonably be expected to be licensable), but is to determine if any candidate site can be judged as environmentally preferable and, if so, obviously superior to the applicant's proposed site. The criterion for making this determination is that one or more important aspects, either singly or in combination, of a reasonably available alternative site are obviously superior to the corresponding aspects of the applicant's proposed site, and the alternative site does not have offsetting deficiencies. The reviewer should consider how the impact data used in the comparison were obtained, how they were applied to each candidate site, and how the comparisons between sites were made. As a general rule, the EPM and specific reviewers for key technical disciplines should make an onsite inspection of each alternative site identified in the application. Recognize that there will be special cases in which the proposed site was not selected on the basis of a					

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	systematic site-selection process. Examples include plants proposed to be constructed on the site of an existing nuclear power plant previously found acceptable on the basis of a NEPA review and/or demonstrated to be environmentally satisfactory on the basis of operating experience, and sites assigned or allocated to an applicant by a State government from a list of State-approved power-plant sites. For such cases, the reviewer should analyze the applicant's site-selection process only as it applies to candidate sites other than the proposed site, and the site-comparison process may be restricted to a site-by-site comparison of these candidates with the proposed site. The site selection process is the same for this case except for the fact that the proposed site is not selected from among the candidate sites based on a site by site comparison. • Site Comparison by Staff—The reviewer will use information regarding the environmental impacts of the proposed action at the proposed site that were developed in Chapters 4.0 and 5.0, and the reconnaissance level information available for the alternative sites, to perform an independent comparison of the proposed and alternative sites. The reviewer should consider the following topics in comparing the proposed and alternative sites: ○ hydrology, water quality, and water availability ○ aquatic biological resources, including wetlands, wetland buffers, essential fish habitat, and endangered species ○ terrestrial resources and land uses, including endangered species, and areas requiring					

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	special consideration ○ transmission corridors ○ socioeconomic factors, including aesthetics, archaeological and historic preservation, and environmental justice ○ population distribution and density ○ air quality ○ radiological and non-radiological health impacts ○ postulated accidents. In some cases the reviewer may find that certain impact categories may not vary among the proposed and alternative sites and, as a result, would not affect the evaluation of whether an alternative site is environmentally preferable to the proposed site. In these cases, impacts can be discussed generically. The reviewer should determine how environmental and health impact information was used by the applicant to predict site-specific impacts, and how the impacts were assembled for a site-to-site comparison. The reviewer will normally use the applicant's proposed plant and supporting system designs at the proposed and alternative sites for the purposes of the comparison. However, in some cases the reviewer may consider alternative systems at the alternative sites. For example, if the specific cooling water system design proposed by the applicant cannot be used at an alternative site, but there is a clearly feasible alternative cooling system design that would work, the reviewer should use the alternative cooling system design. This approach should be used sparingly and only in cases in which the proposed system cannot be used to maintain					

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	a manageable range of alternatives. This approach should only be used for systems that have a significant impact on the environment. If one or more environmentally preferable alternative sites are identified, then the reviewer must determine whether any environmentally preferable site is obviously superior to the proposed site. This portion of the evaluation brings into consideration factors other than the environmental impacts at the proposed and alternative sites. The factors to be considered include: o facility costs for any sites identified as being environmentally preferable o institutional constraints, as they affect site availability o additional public concerns. To the extent practical the reviewer should place the factors being considered into common terms (e.g., monetary cost or benefit). However, in a number of cases it won't be practical to do this and the reviewer will have to use judgment to reach a conclusion regarding whether an alternative site is obviously superior to the proposed site. In using judgment, the reviewer must document the basis for the conclusion so that it can be readily understood by those who will review the evaluation (e.g., a licensing board or the Commission). Because reviewer judgment is required for the decision that a site attribute is obviously superior, any such conclusion must be supported by the corresponding ESRP Chapters 2.0, 4.0, and 5.0 reviewers. The reviewer need not establish or confirm a relative ranking					

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	of candidate sites, but must determine by means of one-by-one comparisons that no alternative site is obviously superior to the proposed site. When the reviewer determines that an obviously superior site can be identified, the reviewer should consult with the applicant to determine the applicant's reasons (if not already known) for not selecting the obviously superior site. In addition, the reviewer should document the conclusion that an alternative site is obviously superior to the proposed site. Finally, the reviewer should alert the EPM to this finding. • Impact Predictions—The reviewer should determine that basic impact criteria (e.g., land use, water use) have been developed for each alternative site, using the environmental descriptions established by the applicant and considering the basic construction and operational parameters of the proposed plant. • Cost Data—If needed to determine whether an environmentally preferable alternative site is obviously superior to the proposed site, the reviewer should determine that economic-cost data associated with each alternative site have been provided, are reasonable, and permit comparison between the candidate sites. Analyze the candidate-site evaluation procedure in the detail needed to be able to make an eventual evaluation that no site within the appropriate study area can be judged (by this or by any other acceptable and accurate procedure based on reconnaissance level data) to be obviously superior to the applicant's proposed site.					

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9.4 (Draft Rev. 0, March 2000)	Alternative Plant and Transmission Systems					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the relevant requirements of 10 CFR 51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
9.4.1 (Draft Rev. 1, July 2007)	Heat Dissipation Systems					
	The analysis of alternative plant heat dissipation systems is a necessary step in the environmental impact statement (EIS) process. The acceptance criteria for this analysis are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.71with respect to the need to discuss alternatives in the environmental analysis.					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A, discussing alternatives to the proposed action.					
	The ER must comply with the relevant requirements of Fish and Wildlife Coordination Act of 1958.					
	The ER must comply with the relevant requirements of Marine Sanctuaries Act of 1972, as amended.					
	The ER must comply with the relevant requirements of Marine Mammal Protection Act, as amended.					
	The ER must comply with the relevant requirements of Coastal Zone Management Act of 1972, as amended.					
	The ER must comply with the relevant requirements of Federal Water Pollution Control Act.					

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	The ER must comply with the relevant requirements of 40 CFR 122 and 125 with respect to National Pollutant Discharge Elimination System (NPDES) permit conditions.					
	The ER must comply with the relevant requirements of Magnuson-Stevens Fishery Conservation and Management Act.					
	The ER must comply with the relevant requirements of Rivers and Harbors Appropriation Act of 1899.					
	The ER must comply with the relevant requirements of Endangered Species Act of 1973, as amended.					
	Regulatory positions and specific criteria necessary to meet the regulations as identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to alternative systems designs.					
	The ER must comply with the relevant requirements of LIC-203, Revision 1, Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Impacts (NRC 2004), with respect to NRC compliance with the Coastal Zone Management Act, the Endangered Species Act, and the Fish and Wildlife Coordination Act.					
	The "Memorandum of Understanding between the Corps of Engineers, U.S. Army, and the NRC for the Regulation of Nuclear Power Plants," 40 FR 60115, provides guidance with respect to the NRC exercising the primary responsibility in conducting environmental reviews and in preparing EISs for nuclear power stations. The Corps of Engineers should be consulted regarding (1) coastal erosion and other shoreline modifications, (2) siltation and sedimentation processes, (3) dredging activities and disposal of dredged materials, and (4) location of structures affecting navigable waters.					

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	The ER must comply with the relevant requirements of Federal, State, regional, local, and affected Native American tribal regulations, on water use, air and water quality, effluent discharge, and land use.					
	The principal objectives of this analysis are (1) to provide assistance to the reviewers for ESRP Chapters 4.0 and 5.0 concerned with construction or operational heat dissipation system impacts in identifying and verifying means to mitigate adverse impacts associated with the proposed heat dissipation system, and (2) to identify and analyze reasonable alternatives to the applicant's proposed system to the extent needed to rank them, from an environmental standpoint, as preferable or inferior to the applicant's proposed system. The depth of the analysis should be governed by the nature and magnitude of proposed heat dissipation system impacts predicted by the reviews of ESRP Chapters 4.0 and 5.0. If adverse impacts are predicted, the reviewers should coordinate in identifying and analyzing means to mitigate these impacts. The proposed system with any verified mitigation schemes (i.e., measures and controls to limit adverse impacts) should be the baseline system against which alternative heat dissipation systems are compared. The nature and adversity of the remaining unmitigated impacts for this baseline system should establish the level of analysis required in the review of alternative systems. This should permit staff evaluation and conclusions with respect to the environmental preference of these alternatives. When no adverse impacts have been predicted for the proposed system and the system will comply with the requirements of the CWA, the reviewer should conclude that there are no environmentally preferable heat dissipation-system alternatives.					

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	When environmentally preferable alternatives have been identified, the review should be expanded to consider the economic costs of any such alternative. This analysis should be done in consultation with appropriate ESRP 10.4 reviewers. Assistance from these reviewers should be requested to establish the economic-cost data to be used to develop a benefit-cost comparison with the baseline (proposed) heat dissipation system. The reviewer should consider the following classes of heat dissipation systems (additional systems, e.g., a combined tower/pond system, may be considered when site-specific conditions suggest that such a system would be environmentally preferable to the proposed system): • once through systems • closed cycle systems: - mechanical draft wet cooling towers (including circular towers) - natural draft cooling towers (including fan assisted towers) - wet dry cooling towers - dry cooling towers - cooling ponds - spray ponds. The reviewer should consider these alternatives for construction and operation at the applicant's proposed site. The analysis should include intake- and discharge-system environmental impacts (and economic costs) when these systems would need to be substantially different than those associated with the proposed heat dissipation system.					

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	The reviewer should conduct an initial environmental screening of each alternative heat dissipation system to eliminate those systems that are obviously unsuitable for use at the proposed site. Factors to be considered in this initial screening are land use (e.g., site size and terrain), water use (e.g., availability of cooling water), and legislative or regulatory restrictions. Economic factors should not be considered in this initial screening. Working through the EPM, the reviewer may consult with appropriate Federal and State agencies when needed to conduct this screening. The reviewer may also consult (through the EPM) with the appropriate administrative agencies to screen those alternatives that will not meet CWA requirements. The reviewer may establish other justifiable environmental bases for rejection of a given alternative. When the reviewer rejects an alternative, that alternative needs no further consideration other than the preparation of the reasons and justification for the rejection. The following procedure for developing the analysis of alternative heat dissipation systems considers both environmental and economic-cost factors. In following this procedure, the reviewer should initially consider only the environmental factors and should repeat the procedure for economic factors only for those alternatives shown to be environmentally preferable by the evaluation procedures of this ESRP. The analysis of those alternative heat dissipation systems not eliminated by the initial screening process should be based on the environmental and economic factors shown in Table 9.4.1-1. The reviewer should prepare a similar table for the heat dissipation systems under consideration, comparing each of the environmental and economic cost and benefit factors with those of the proposed heat dissipation system. Information for this table may be					

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	presented either in terms of absolute environmental and economic costs and benefits or as incremental costs and benefits referenced to the proposed system. Additional factors may be included when needed on a site- or system-specific basis. Preparation of this table should involve the following: (1) Land Use—Determine (1) the onsite land-use requirements of each system, (2) the practicality of heat dissipation system construction and operation within the specifics of site area, terrain, and the impacts of social and economic land-use costs, (3) the extent to which any system is sited on or results in modifications to the floodplain,(a) (4) any relevant wetlands or critical habitat issues, and (5) the impacts to terrestrial biota associated with system construction and operation. The reviewer should consult with the reviewers for ESRPs 2.2.1, 2.3.1, 4.1.1, 4.3.1, 5.1.1, and 5.3.3 to develop the comparative land-use and ecological impact data. (2) Water Use—Determine (1) the water-use requirements of each system, including intake requirements, water consumption, and intake/discharge water quality and quantity, (2) the practicality of this water use within the specifics of water availability and the impacts of present and known future water uses, and (3) the impacts of aquatic biota associated with system construction and operation. The reviewer should compare these data with characteristics of the proposed heat dissipation system. The economic cost of water consumed should be considered when these data are available. The reviewer should consult with the reviewers for ESRPs 2.3, 4.2.2, 4.3.2, 5.2.2, and 5.3 to develop the comparative water quality, water use, and ecological impact data. (3) Atmospheric Effects—Determine the predicted atmospheric					

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	effects of each alternative heat dissipation system (e.g., the extent and magnitude of cooling tower drift) and compare these effects with those of the proposed system. The reviewer should consult with the reviewers for ESRPs 2.7 and 5.3.3 to develop this comparison, which may be based on verified applicant supplied data or on independent staff estimations of atmospheric effects. (4) Thermal and Physical Effects—Estimate the predicted thermal and physical effects (e.g., thermal plumes, erosion, scouring) of each alternative heat dissipation system, and compare these effects with those of the proposed system. The reviewer should consult with the reviewers for ESRPs 2.3.1, 4.2.1, and 5.2.1 for assistance in making this comparison. (5) Noise Levels—Estimate operational noise levels for each of the alternatives and compare them with the predicted operating noise levels of the proposed system and with any Federal, State, regional, local, or affected Native American tribal restrictions. The reviewer should consider construction noise levels when these could be significant. (6) Aesthetics and Recreational Benefits—Compare the aesthetic impacts and potential recreational benefits of each alternative system with those of the proposed system. The reviewer should consult with the reviewers for ESRPs 2.5, 3.1, and 5.8 for assistance in making this comparison. (7) Operating and Maintenance Experience—Compare operating and maintenance experience of each alternative with the proposed system to develop a projected reliability factor for each system.					

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	(8) Generating Efficiency—Estimate the plant electrical generation efficiency for each alternative heat dissipation system and compare it with the generating efficiency using the proposed system. (9) Costs—Estimate the capital, operating, and maintenance costs for the proposed system and for each alternative considered. The reviewer should use these figures for economic-cost comparisons. The reviewer should determine if there are any site-specific factors that might affect the costs of any alternative and factor these additional costs into the comparison. (10) Other Considerations—When an alternative heat dissipation system will involve the use of intake or discharge systems that would be substantially different from the proposed system, repeat these procedures for both intake and discharge systems. This should supplement the appropriate environmental and economic-cost factors, as needed, to account for any differing intake and discharge system effects. The reviewer should consult with the reviewer for ESRP 9.4.2.					
	<u>General Considerations</u> The reviewer should ensure that each heat dissipation system alternative has been described in sufficient detail to enable an effective analysis and comparison of environmental impacts leading to a staff conclusion that the alternative system is environmentally preferable or inferior to the proposed system. For those alternatives determined to be environmentally preferable, the reviewer should ensure that economic-cost data are available in sufficient detail to enable the reviewer to conduct benefit-cost balance and comparisons with the proposed system					

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	leading to a final staff conclusion for heat dissipation-system consideration. The reviewer should also ensure that all comparisons are made on the basis of the proposed system as supplemented with those measures and controls to limit adverse impacts proposed by the applicant and concurred with by the staff. For those alternatives eliminated from consideration (1) on the basis of land-use, water-use, or legislative or regulatory requirements, or (2) because it is judged inferior to the proposed system, the reviewer should ensure that adequate documented justification for this action has been prepared. If a mitigation measure or alternative heat dissipation system is to be considered, determine that the measure or system being evaluated has a lesser overall environmental impact than the proposed system (i.e., is environmentally preferable). When this is true, the economic costs of mitigation or of the alternative could result in an improved project benefit-cost balance. When these criteria are met, the reviewer should verify those mitigation measures proposed by the reviewers for ESRP Chapters 4.0 and 5.0 or should consider an alternative heat dissipation system. The reviewer should be guided by the following general considerations: • Keep in mind that an environmental review of alternative heat dissipation systems, if conducted in the depth applied to the review of the proposed system, would be expected to find additional impacts and/or increased severity of the impacts already predicted for the alternative. The reviewer should allow for this when evaluating the comparative environmental impacts of each proposed alternative with those of the proposed system. • Ensure that the level of detail provided for each					

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	economic, environmental, and social cost estimate is commensurate with the level of importance of the related environmental impact. • Adjust the economic costs of each alternative system on the basis of equivalent generating capacity. • The evaluation of alternative heat dissipation systems may include consultation and coordination with those agencies responsible for NPDES administration. The reviewer may coordinate the evaluation of measures and controls to limit adverse impacts, or of alternatives to avoid adverse impacts (with the EPM as liaison), with NPDES administrators. When consulting with the EPA or with agencies of States having memoranda of understanding with NRC, the reviewer should ensure that the staff analyses and evaluations (1) are consistent with the details of these memoranda, and (2) will serve the needs of these agencies.					
	<u>Measures and Controls to Limit Adverse Impacts</u> When considering measures provided by the reviewers for ESRP Chapters 4.0 and 5.0 to mitigate adverse environmental impacts predicted for the proposed heat dissipation system, the reviewer's verification of the desirability of the measure should lead to the following conclusions: • The measure provides the desired mitigation and does not introduce other adverse environmental impacts not predicted for the proposed system. • The measure will result in an overall benefit-cost balance better than that of the proposed project. • The measure is not precluded by Federal, State, regional, local, or affected Native American tribal regulations, requirements, or ordinances.					

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	• The measure is consistent with NPDES requirements.					
	<u>Alternative Heat Dissipation Systems</u> The initial step in the evaluation of those alternative heat dissipation systems identified by the analysis procedure of this ESRP should be to categorize these systems as environmentally preferable or inferior to the proposed heat dissipation system as modified by measures and controls to limit adverse impacts. The following criteria should be applied to this evaluation: • When the reviewer determines that the proposed system (with mitigation measures, if necessary) will have no unavoidable adverse impacts and the system will comply with the requirements of the CWA, the reviewer should conclude that there are no environmentally preferable heat dissipation-system alternatives. • When the reviewer determines that the proposed heat dissipation system will meet CWA requirements, but is predicted to have unavoidable adverse environmental impacts, the reviewer should evaluate the identified alternative systems for potential environmental preference to the proposed system. The scope and extent of this evaluation should depend on the nature and magnitude of the proposed system's environmental impacts. An environmental review for the alternatives may be needed following the analysis and evaluation procedures of the appropriate ESRP Chapters 4.0 and 5.0. The following criteria apply to this evaluation: - Environmental preference will be established when an alternative can be shown to have no unavoidable adverse impacts and will meet CWA requirements.					

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	- Environmental preference may be established when an alternative that meets CWA requirements can be shown to have unavoidable adverse impacts that are less severe in both nature and magnitude than those of the proposed system. Determination of environmental preference under these conditions should involve consultation with the EPM and the appropriate ESRP Chapter 4.0 and 5.0 reviewers. This consultation should result in a joint determination of the status of any such alternative. - Environmental inferiority will be established when an alternative can be shown to have unavoidable adverse impacts that are more severe in both nature and magnitude than those of the proposed system, or that will not meet CWA requirements. When the reviewer determines that there are environmentally preferable alternatives to the proposed heat dissipation system, the reviewer should conduct those portions of the analysis instructions of this ESRP that deal with the economic costs of the alternative systems. • When environmentally preferable alternative heat dissipation systems have been identified, the reviewer should ensure that economic cost data have been developed for the alternatives and that these data are adequate for a benefit-cost balancing and comparison with the proposed system. This portion of the evaluation procedure should be conducted with the assistance of appropriate reviewers for ESRPs 10.4.1					

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	through 10.4.3. The reviewer should complete the economic and reliability portions of Table 9.4.1-1. On the basis of the completed table, the reviewer should balance and compare benefits and costs of the environmentally preferable alternative(s) with those of the proposed system. When an environmentally preferable alternative can be shown to have a higher benefit to cost ratio than the proposed system, the reviewer may conclude that the alternative should be considered an alternative to the proposed system. For those cases in which the benefits of the alternative are less than those of the proposed system or if economic costs are greater than those of the proposed system, a tentative conclusion that the alternative is superior lead to consultation with the EPM and with the appropriate ESRP Chapter 4.0 and 5.0 reviewers. If this consultation establishes that the benefit-cost balances of such alternatives are not superior to that of the proposed system, the alternatives should not receive further consideration. When alternatives have significantly decreased benefits or increased economic costs, they should be rejected for any further consideration as alternatives to the proposed systems.					
9.4.2 (Draft Rev. 1, July 2007)	Circulating Water Systems					
	Acceptance criteria for the review of alternatives to the proposed circulating water system are based on the relevant requirements of the following:					

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	The ER must comply with the relevant requirements of 10 CFR 51.71 with respect to the need to discuss alternatives in the environmental analysis.					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A, with respect to discussing alternatives to the proposed action.					
	The ER must comply with the relevant requirements of 40 CFR 122 and 125 with respect to National Pollutant Discharge Elimination System (NPDES) permit conditions.					
	The ER must comply with the relevant requirements of Federal Water Pollution Control Act.					
	The ER must comply with the relevant requirements of Coastal Zone Management Act of 1972, as amended.					
	The ER must comply with the relevant requirements of Endangered Species Act of 1973, as amended.					
	The ER must comply with the relevant requirements of Fish and Wildlife Coordination Act of 1958.					
	The ER must comply with the relevant requirements of Marine Mammal Protection Act, as amended.					
	The ER must comply with the relevant requirements of Marine Sanctuaries Act of 1972, as amended.					
	The ER must comply with the relevant requirements of Magnuson-Stevens Fishery Conservation and Management Act.					
	The ER must comply with the relevant requirements of Rivers and Harbors Appropriation Act of 1899.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to alternative systems designs.					
	The ER must comply with the relevant requirements of LIC-203,					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	Revision 1, Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Impacts (NRC 2004), with respect to NRC compliance with the Coastal Zone Management Act, the Endangered Species Act, and the Fish and Wildlife Coordination Act.					
	The "Memorandum of Understanding between the Corps of Engineers, U.S. Army, and the NRC for the Regulation of Nuclear Power Plants," 40 FR 60115, provides guidance with respect to the NRC exercising the primary responsibility in conducting environmental reviews and in preparing EISs for nuclear power stations. The Corps of Engineers should be consulted regarding (1) coastal erosion and other shoreline modifications, (2) siltation and sedimentation processes, (3) dredging activities and disposal of dredged materials, and (4) location of structures affecting navigable waters.					
	The ER must comply with the relevant requirements of Federal, State, regional, local, and affected Native American tribal regulations on water use, air and water quality, effluent discharge, and land use.					
	The principal objectives of this analysis procedure are (1) to provide assistance to those ESRP Chapter 4.0 and 5.0 reviewers concerned with construction or operational circulating water system impacts in identifying and verifying means to mitigate adverse impacts associated with the proposed circulating water systems, and (2) to identify and analyze reasonable alternatives to the applicant's proposed systems to the extent needed to rank them from an environmental standpoint as preferable or inferior to the applicant's proposed system. Tables 9.4.2-1 through 9.4.2-5 can be used or adapted to aid the review as appropriate. The depth of the analysis should be governed by the nature and magnitude of proposed circulating water system impacts					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	predicted by the ESRP Chapter 4.0 and 5.0 reviewers. When adverse impacts are predicted, the reviewer should coordinate with these reviewers in identifying and analyzing means to mitigate these impacts. The proposed system with any verified mitigation schemes (i.e., measures and controls to limit adverse impacts) should be the baseline system against which alternative circulating water systems will be compared. The nature and adversity of the remaining unmitigated impacts for this baseline system should establish the level of analysis required in the review of alternative systems to permit staff evaluation and conclusions with respect to the environmental preference of these alternatives. If no adverse impacts have been predicted for the proposed system and the system will comply with the requirements of the CWA, the reviewer should conclude that there are no environmentally preferable heat dissipation-system alternatives. When environmentally preferable alternatives have been identified, the review should be expanded to consider the economic costs of any such alternative. The reviewer should estimate the capital, operating, and maintenance costs for each circulating water system component considered and for each component of the proposed system. The reviewer should use these data to estimate total annual costs for each system and should use these annual costs for economic-cost comparisons. The reviewer should determine if there are any site-specific factors that might affect the costs of any alternative and should factor these increased or reduced costs into the comparison. As necessary, these cost estimates should consider allowances for additional maintenance costs when it can be shown (e.g., by operating experience) that system reliability will be lower than expected for the proposed system. This analysis should be done in consultation with appropriate reviewers for ESRPs 10.4.1					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	through 10.4.3. Assistance from these reviewers should be requested to establish the economic-cost data used to develop a benefit-cost comparison with the baseline (proposed) circulating water system. In this analysis, the reviewer should consider alternatives to the following components of the plant circulating water system: (1) intake systems (2) discharge systems (3) water supply (4) water treatment. The analysis should consider only those alternatives that are applicable at the proposed site and compatible with the proposed heat dissipation system. The following procedure for developing the analysis of alternative circulating water systems considers both environmental and economic cost factors. In following this procedure, the reviewer should initially consider only the environmental factors and should repeat the procedure for economic factors only for those alternatives shown to be environmentally preferable by the evaluation procedures of this ESRP.					
	<u>Initial Environmental Screening</u> The reviewer should consider the following factors in the initial environmental screening of each alternative circulating water system to eliminate those systems (or components) that are obviously unsuitable for use at the proposed site. Economic factors should not be considered in this initial screening. • plant water requirements					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	• site terrain and relationship to water bodies • water body geometry • other water use • ecological considerations • legislative or regulatory requirements. The following steps should be considered by the reviewer as part of the initial environmental screening procedures for each system: • Work through the EPM to consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies when needed to conduct this screening. • Consult the appropriate NPDES administrative agencies to screen for those alternatives that will not meet CWA requirements. • Establish any other justifiable environmental bases for rejection of a given alternative. When the reviewer rejects an alternative, that alternative needs no further consideration other than preparation of the reasons and justification for the rejection. (1) Intake Systems—To analyze alternative intake systems, the reviewer should perform the following steps: (a) Consult with the appropriate ESRP Chapter 4.0 and 5.0 reviewers to identify any mitigation measures or potentially superior alternative intake systems identified by these reviewers. (b) Consider the following classes of alternatives: • alternative intake systems (e.g., offshore vs. shoreline)					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• proposed system design modifications (e.g., reduced intake velocity, fish return system) • alternative locations of proposed system (e.g., up/downstream, alternative water bodies) • alternative procedures (e.g., screenwash operation, thermal defouling). (c) Consider the following environmental impacts and economic costs or factors for each mitigation measure and class of alternative: • construction impacts • impacts to aquatic ecology, including - entrapment - impingement - entrainment - other (site-specific) aquatic impacts. • water-use impacts, including physical impacts resulting from hydrologic alterations (e.g., breakwater construction) and impacts resulting from siting on the floodplain • compliance with Federal, State, regional, local, or affected Native American tribal regulations, requirements, or ordinances • capital cost, annual operating and maintenance costs, and total annual costs. (d) Compare the proposed system with those remaining classes of alternatives not eliminated in an initial screening: • Use a format similar to that shown in Table 9.4.2-1. • Inputs for this table may be either absolute costs and benefits or incremental costs and benefits referenced to					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	the proposed intake system. • Additional factors may be included on a site- or system-specific basis. (2) Discharge Systems—To analyze alternative discharge systems, the reviewer should perform the following steps: (a) Consult with the appropriate ESRP Chapter 4.0 and 5.0 reviewers to identify any mitigation measures or alternative discharge systems suggested by these reviewers. (b) Consider the following classes of alternatives: • alternative discharge systems (e.g., submerged offshore vs. shoreline) and discharge type (e.g., slot, multiport) • proposed system design modifications (e.g., modified discharge velocity, screens to prevent fish entry) • alternative locations of proposed discharge system (e.g., up/downstream, alternative water body). (c) Consider the following environmental impacts and economic costs or factors for each of the above classes of alternatives: • construction impacts • impacts to aquatic ecology • water-use impacts, including physical impacts of hydrological alterations and siting on the floodplain • compliance with Federal, State, regional, local, or affected Native American tribal regulations, requirements, or ordinances • capital costs, annual operating and maintenance costs, and total annual costs.					

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Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(d) Compare the proposed system with those remaining classes of alternatives not eliminated in an initial screening. Use a table format similar to that shown in Table 9.4.2-1. (3) Water Supply Systems—To analyze alternative water supplies, the reviewer should perform the following steps: (a) Consult with the appropriate ESRP Chapter 4.0 and 5.0 reviewers to identify any mitigation measures or alternative water supplies suggested by these reviewers. (b) Consider as potential alternative water sources those water bodies within reasonable proximity to the proposed plant site that are capable of supplying plant water needs. (c) When such water sources can be identified, compare them with the proposed water source using the following comparison factors: • water body location and description • estimated availability of water for plant use • restrictions (if any) on water use for power plant cooling • estimated aquatic, terrestrial, social, and environmental impacts associated with construction, operation, and maintenance of water transport systems from the water body to the plant • capital costs and operation and maintenance costs of the water transport system, including annual costs of water as delivered to the plant and costs associated with any necessary water treatment. (d) Use a format similar to that shown in Table 9.4.2-3 for this					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	comparison. Data for this table may be prepared either as absolute benefits and costs or as incremental benefits and costs referenced to the proposed water source. (4) Water Treatment System—To analyze water treatment systems, the reviewer should perform the following steps: (a) Consider alternatives on the basis of systems that avoid or minimize the use of chemicals, use lesser quantities of or less toxic chemicals, or do not discharge chemical wastes directly to the environment. (b) Unless an adverse impact attributable to the proposed plant service water treatment system has been identified, restrict this analysis to alternative circulating water treatment systems. (c) Consult with the reviewer for ESRP 3.3.3 to determine proposed water treatment systems and with the reviewer for ESRP 5.3.2.2 to determine potential impacts of discharged chemicals to aquatic biota. (d) Consider the following classes of alternatives: • alternative water treatment systems (e.g., mechanical vs. chemical) • modifications to the proposed system (e.g., alternative chemicals, alternative discharge points) • alternative operating procedures (e.g., shock treatment vs. continuous chemical addition, modified cooling tower concentration factors). (e) Determine the following environmental and economic costs or factors for each of the above classes of alternatives:					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• impacts to aquatic ecology (e.g., chemical toxicity) • land-use impacts (e.g., evaporation ponds) • water-use impacts (e.g., increased water use to achieve lower discharge chemical concentrations) • compliance with Federal, State, regional, local, or affected Native American tribal regulations, requirements, or ordinances • capital costs, annual operating and maintenance costs, and total annual costs. (f) Compare the proposed system with those remaining classes of alternatives not eliminated in an initial screening. Use a format similar to that shown in Tables 9.4.2-1 through 9.4.2-5.					
	<u>General Considerations</u> The reviewer should ensure that each circulating water system alternative has been described in sufficient detail to enable the reviewer to make an effective analysis and comparison of environmental impacts leading to a staff conclusion that the alternative system is environmentally preferable or inferior to the proposed system. For those alternatives determined to be environmentally preferable, the reviewer should ensure that economic-cost data are available in sufficient detail to enable the reviewer to conduct benefit-cost balancing and comparisons with the proposed system, leading to a final staff conclusion for circulating water system consideration. The reviewer should also ensure that all comparisons were made on the basis of the proposed system as supplemented with those measures and controls to limit adverse impacts proposed by the applicant and concurred with by the staff. For those alternatives eliminated from consideration (1) on the basis of land-use, water-use, or legislative or regulatory requirements, or (2) because it is judged					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	inferior to the proposed system, the reviewer should ensure that adequate documented justification for this action has been prepared. If a mitigation measure or alternative circulating water system is to be considered, the reviewer should determine that the measure or system being evaluated has a lesser overall environmental impact than the proposed system (i.e., is environmentally preferable). When this is true, the economic costs of mitigation or of the alternative could result in an improved projected benefit-cost balance. When these criteria are met, the reviewer should verify those mitigation measures proposed by the reviewers for ESRP Chapters 4.0 and 5.0 or should identify the need for an alternative circulating water system. The reviewer should be guided by the following general considerations: • The reviewer should keep in mind that an environmental review of alternative circulating water systems, if conducted in the depth applied to the review of the proposed system, would be expected to find additional impacts and/or increased severity of the impacts already predicted for the alternative. The reviewer should allow for this when evaluating the comparative environmental impacts of each proposed alternative with those of the proposed system. • The reviewer should ensure that the level of detail provided for each economic, environmental, and social cost estimate is commensurate with the level of importance of the related environmental impact. • The reviewer should adjust the economic costs of each alternative system on the basis of equivalent generating capacity.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• The evaluation of alternative circulating water systems may include consultation and coordination with those agencies responsible for NPDES administration. With the EPM as liaison, the reviewer should coordinate the evaluation of measures and controls to limit or avoid adverse impacts. When consulting through the EPM with the EPA, or with agencies of States that have memoranda of understanding with the NRC, the reviewer should ensure that the staff analyses and evaluations (1) are consistent with the details of these memoranda, (2) will serve the environmental impact statement needs of these agencies, and (3) are consistent with the requirements of the CWA.					
	<u>Measures and Controls to Limit Adverse Impacts</u> When considering measures identified by the reviewers for ESRP Chapters 4.0 and 5.0 to mitigate adverse environmental impacts predicted for the proposed circulating water system, the reviewer's verification of the desirability of the measure should lead to the following conclusions: • The measure provides the desired mitigation and does not introduce other adverse environmental impacts not predicted for the proposed system. • The measure will result in an overall benefit-cost balance better than that of the proposed project. • The measure is not precluded by Federal, State, regional, local, or affected Native American tribal regulations, requirements, or ordinances. • The measure is consistent with NPDES requirements.					
	<u>Alternative Circulating Water Systems</u>					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The initial step in evaluating those alternative intake systems, discharge systems, water supplies, or water treatment systems identified by the analysis procedure of this ESRP should be to categorize these systems as environmentally preferable or inferior to the proposed circulating water systems as modified by measures and controls to limit adverse impacts. The following criteria should be applied to this evaluation: • When the reviewer determines that the proposed system (with mitigation measures, if necessary) will have no unavoidable adverse impacts and the system will comply with the requirements of the CWA, the reviewer should conclude that there are no environmentally preferable alternatives. • When the reviewer determines that the proposed circulating water system will meet CWA requirements, but is predicted to have unavoidable adverse environmental impacts, the reviewer should evaluate the identified alternative systems for potential environmental preference to the proposed system. The scope and extent of this evaluation should depend on the nature and magnitude of the proposed system's environmental impacts. An environmental review for the alternatives may be needed following the analysis and evaluation procedures of the appropriate ESRP Chapters 4.0 and 5.0. The following criteria apply to this evaluation: - Environmental preference will be established when an alternative can be shown to have no unavoidable adverse impacts and will meet CWA requirements.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	- Environmental preference may be established when an alternative that meets CWA requirements can be shown to have unavoidable adverse impacts that are less severe in both nature and magnitude than those of the proposed system. Determination of environmental preference under these conditions should involve consultation with the EPM and the appropriate ESRP Chapters 4.0 and 5.0 reviewers. This consultation should result in a joint determination of the status of any such alternative. - Environmental inferiority will be established when an alternative can be shown to have unavoidable adverse impacts that are more severe in both nature and magnitude than those of the proposed system or that will not meet CWA requirements. When the reviewer determines that there are environmentally preferable alternatives to the proposed circulating water system, the reviewer should conduct those portions of the analysis instructions of this ESRP that deal with the economic costs of the alternative systems. • When environmentally preferable alternative circulating water systems have been identified, the reviewer should ensure that economic-cost data have been developed for the alternatives and that these data are adequate for a benefit-cost balancing and comparison with the proposed system. This portion of the evaluation procedure should be conducted with the assistance of appropriate reviewers for ESRPs					

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	10.4.1 through 10.4.3. The reviewer should complete the economic and reliability portions of Table 9.4.2-1. On the basis of the completed table, the reviewer should balance and compare benefits and costs of the environmentally preferable alternative(s) with those of the proposed system. When an environmentally preferable alternative can be shown to have a higher benefit to cost ratio than the proposed system, the reviewer may conclude that it should be considered as an alternative to the proposed system. For those cases in which benefits of the alternative are less than those of the proposed system or where economic costs are greater than those of the proposed system, a tentative conclusion that the alternative is superior should lead to consultation with the EPM and with the appropriate ESRP Chapter 4.0 and 5.0 reviewers. If this consultation establishes that the benefit-cost balances of such alternatives are not superior to that of the proposed system, the alternatives should not receive further consideration. When alternatives have significantly decreased benefits or increased economic costs, they should be rejected for any further consideration as alternatives to the proposed systems.					
9.4.3 (Draft Rev. 1, July 2007)	Transmission Systems					
	Acceptance criteria for the review of alternative transmission systems are based on the relevant requirements of the following:					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The ER must comply with the relevant requirements of 10 CFR 51.71(a) referring to 10 CFR 51.45(a)(3) with respect to the need to discuss alternatives in the environmental analysis.					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A, with respect to discussion of alternatives to the proposed action.					
	The ER must comply with the relevant requirements of 18 CFR Part 50 with respect to regulations for filing applications for permits to site interstate electric transmission facilities.					
	The ER must comply with the relevant requirements of Regulatory requirements specific for particular land types (see Table 4.1.2-1).					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to evaluation of alternative systems designs.					
	The ER must comply with the relevant requirements of Regulatory Guide 4.7, Rev. 2, General Site Suitability for Nuclear Power Stations (NRC 1998), with respect to site suitability guidelines.					
	The ER must comply with the relevant requirements of U.S. Nuclear Regulatory Commission, "Alternative Electrical Transmission Systems and Their Environmental Impact," NUREG-0316, August 1977 (NRC 1977), with respect to environmental impacts.					
	The principal objectives of this analysis procedure are (1) to provide assistance to those ESRP Chapter 4.0 and 5.0 reviewers concerned with identifying and verifying means to mitigate adverse impacts associated with the proposed transmission system, and (2) to identify and analyze reasonable alternatives to					

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	the applicant's proposed system to the extent needed to rank them, from an environmental standpoint, as preferable or not preferable to the applicant's proposed system. The analysis should consider only those alternatives applicable to and compatible with the proposed plant, the applicant's service area, and the regional transmission network. In this analysis, the reviewer should consider alternatives to transmission corridor routes. The reviewer should also ensure that due consideration has been given to the use of existing transmission line corridors as an alternative to the development of new corridors. The depth of the analysis should be governed by the nature and magnitude of proposed transmission-system impacts predicted by the ESRP Chapter 4.0 and 5.0 reviewers. When adverse impacts are predicted, the reviewer should coordinate with these reviewers in identifying and analyzing means to mitigate these impacts. The proposed system with any verified mitigation schemes (i.e., measures and controls to limit adverse impacts) should be the baseline system against which alternative transmission systems will be compared. The nature and adversity of the remaining unmitigated impacts for this baseline system should establish the level of analysis required in the review of alternative systems to permit staff evaluation and conclusions with respect to the environmental preference or equivalence of these alternatives. When no adverse impacts have been predicted for the proposed system, the review should be limited to an analysis of alternative transmission systems in the depth necessary to judge their environmental preferability to the applicant's proposed system. The reviewer should conduct an initial environmental screening of each alternative transmission system to eliminate those					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	systems that are obviously unsuitable for application to the proposed project. Economic factors should not be considered in this initial screening. Working through the EPM, the reviewer may consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies when needed to conduct this screening. When the reviewer rejects an alternative, that alternative needs no further consideration other than the preparation of the reasons and justification for the rejection. When environmentally preferable alternatives are identified, the review should be expanded to consider the economic costs of any such alternative. This analysis should be done in consultation with appropriate reviewers for ESRPs 10.4.1, 10.4.2, and 10.4.3. Assistance from these reviewers should be sought to establish the economic-cost data used to develop a benefit-cost comparison with the baseline (proposed) transmission system. The following procedure for developing the analysis of alternative transmission systems considers both environmental and economic-cost factors. In following this procedure, the reviewer should initially consider only the environmental factors, and should repeat the procedure for economic factors only for those alternatives shown to be environmentally preferable by the evaluation procedures of this ESRP. The analysis of those alternative transmission systems not eliminated by the initial screening process should be based on the environmental and economic factors shown in Table 9.4.3-1. The reviewer should prepare a similar table for each transmission system element under consideration, comparing each of the environmental and economic cost and benefit factors with those of the proposed transmission system element. Information for this table may be prepared either in terms of absolute environmental and economic					

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	costs and benefits, or as incremental costs and benefits referenced to the proposed system. Additional factors may be included when needed on a site- or system-specific basis as follows: (a) The reviewer's analysis of alternative corridor routes should be based on a comparison of those routes with the proposed routes described in ESRP 3.7. The comparison may be made for complete routes or for route segments, as appropriate, and should consider those factors listed under the heading "Data and Information Needs" in this ESRP. (b) The reviewer should consider both environmental and economic factors, using a tabular format similar to that shown in Table 9.4.3-1. The reviewer should consult with the reviewer for ESRP 3.7 and the appropriate ESRP Chapter 4.0 and 5.0 reviewers to establish construction and operation impacts for the proposed corridor routes. The reviewer's comparison of these data with those for the alternative corridors should involve the following: • Impacts—The reviewer should estimate the impacts that can be expected from development of alternative transmission corridors. The appropriate ESRP Chapter 4.0 and 5.0 reviewers should be consulted in making these estimates and in comparing these impacts with those predicted for the proposed corridor routes. • Economic Factors—The reviewer should estimate acquisition or right-of-way costs, clearing and construction costs, maintenance costs, and the costs to mitigate predicted environmental impacts for the proposed and alternative routes. Where there are appreciable differences in transmission line lengths, the					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	reviewer should estimate the loss in delivered electrical capacity due to transmission line losses. (c) The reviewer should consider alternative locations of auxiliary transmission system facilities only when the reviewers for ESRPs 4.1.2 or 5.1.2 advise relocating of such facilities. Using the guidance below, the reviewer should evaluate the applicant's process for identifying and selecting alternative transmission system routes to ensure that reasonable alternatives to the proposed routes have been considered. The reviewer should ensure that each transmission system alternative has been described in sufficient detail to enable the reviewer to make an effective analysis and comparison of environmental impacts leading to a staff conclusion that the alternative system is environmentally preferable, equivalent, or inferior to the proposed system. For those alternatives determined to be environmentally preferable, the reviewer should ensure that economic-cost data are available in sufficient detail to enable the reviewer to conduct benefit-cost balance and comparisons with the proposed system, leading to a final staff recommendation for transmission system consideration. The reviewer should also ensure that all comparisons are made on the basis of the proposed system, as supplemented with those measures and controls to limit adverse impacts proposed by the applicant and concurred with by the staff. For those alternatives eliminated from consideration on the basis of land use, water use, or legislative restrictions, the reviewer should ensure that adequate documented justification for this action has been prepared. (1) General Considerations					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(a) If a mitigation measure or alternative transmission system is being considered, the reviewer should determine first that the measure or system being evaluated has a lesser overall environmental impact than the proposed system (i.e., is environmentally preferable). When this is true, the economic costs of mitigation or of the alternative could result in an equivalent or improved project benefit-cost balance. When these criteria are met, the reviewer should verify that those mitigation measures proposed by the reviewers for ESRP Chapters 4.0 and 5.0 will meet the criteria as a feasible alternative transmission system. (b) The reviewer should keep in mind that an environmental review of alternative transmission systems, if conducted in the depth applied to the review of the proposed system, would be expected to find additional impacts and/or increased severity of the impacts already predicted for the alternative. The reviewer should allow for this when evaluating the comparative environmental impacts of each proposed alternative with those of the proposed system. (c) The reviewer should ensure that the level of detail provided for each economic, environmental, and social cost estimate is commensurate with the level of importance of the related environmental impact. (2) Measures and Controls to Limit Adverse Impacts (a) When considering measures identified by the reviewers for ESRP Chapters 4.0 and 5.0 to mitigate adverse environmental impacts predicted for the proposed transmission system, the reviewer's verification of the desirability of the measure should					

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	reach the following conclusions: • The measure provides the desired mitigation and does not introduce other adverse environmental impacts not predicted for the proposed system. • The measure will result in an overall benefit-cost balance equivalent to, or better than, that of the proposed project. • The measure is not precluded by Federal, State, regional, local, or affected Native American tribal regulations or ordinances. (3) Alternative Transmission Systems (a) The initial step in the evaluation of those alternative transmission systems identified by the analysis procedure of this ESRP should be to categorize these systems as environmentally preferable or inferior to the proposed transmission system as modified by measures and controls to limit adverse impacts. The following criteria should be applied to this evaluation: • When the reviewer determines that the proposed system (with mitigation measures, if necessary) will have no unavoidable adverse impacts and will comply with applicable Federal, State, regional, local, and affected Native American tribal regulations or requirements, the reviewer should conclude that there is no environmentally preferable transmission system alternative. • When the reviewer determines that the proposed transmission system will meet regulatory requirements, but is predicted to have unavoidable adverse environmental impacts, the reviewer should evaluate					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	the identified alternative systems for potential environmental preference to the proposed system. The scope and extent of this evaluation should depend on the nature and magnitude of the proposed system's environmental impacts. An environmental review of the alternatives may be required following the analysis and evaluation procedures of the appropriate ESRP Chapters 4.0 and 5.0. The following criteria apply to this evaluation: - Environmental preference will be established when an alternative can be shown to (1) have no unavoidable adverse impacts and (2) meet regulatory requirements. - Environmental preference may be established when an alternative that meets regulatory requirements can be shown to have unavoidable adverse impacts that are less severe in both nature and magnitude than those of the proposed system. Determination of environmental preference under these conditions should lead to consultation with the EPM and the appropriate ESRP Chapter 4.0 and 5.0 reviewers. This consultation should result in a joint determination of the status of any such alternative. When the reviewer determines that there are environmentally preferable alternatives to the proposed transmission system, the reviewer should conduct those portions of the analysis instructions of this ESRP that deal with the economic costs of the alternative systems. (b) When environmentally preferable alternative transmission systems have been identified, the reviewer should ensure that					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	economic cost data have been developed for the alternatives and that these data are adequate for a benefit-cost balance and comparison with the proposed system. This portion of the evaluation procedure should be conducted with the assistance of reviewers for ESRPs 10.4.1, 10.4.2, and 10.4.3. The reviewer should complete the economic factors portions of Table 9.4.3-1. On the basis of the completed table, the reviewer should balance and compare benefits and costs of the environmentally preferable alternative(s) with those of the proposed system. When an environmentally preferable alternative can be shown to have the same benefits as the proposed system with comparable reliability and at the same or lesser economic costs, the reviewer may conclude that the alternative should be considered as a replacement for the proposed system. For those cases in which benefits of the alternative are less than those of the proposed system (e.g., increased transmission losses or decreased system reliability) or where economic costs exceed those of the proposed system, a conclusion to further consider the alternative should lead to consultation with the Environmental Project Manager and with the appropriate ESRP Chapter 4.0 and 5.0 reviewers. If this conclusion establishes that the benefit-cost balances of such alternatives are no more than equivalent to the proposed system, the alternatives should not be considered further. When alternatives have significantly decreased benefits or increased economic costs, they should be rejected for any further consideration as replacements for the proposed system.					
10.0 (Draft Rev. 0, March, 2000)	Environmental Consequences of the Proposed Action					Exclude, Administrative

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
10.1 (Draft Rev. 0, March, 2000)	Unavoidable Adverse Environmental Impacts					
	Acceptance criteria for the evaluation of unavoidable adverse environmental impacts are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51, Appendix A, with respect to the identification of unavoidable adverse impacts to the environment.					
	Regulatory positions and specific criteria to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the content and presentation of material in an applicant's environmental report.					
	The reviewer's analysis and summary of adverse environmental impacts of construction and operation should be based on project design, construction, and operation (1) as proposed by the applicant and (2) which incorporates those measures and controls to limit adverse impacts that the staff consider appropriate. The reviewer should identify these impacts, organize them by environmental categories, and summarize each category for inclusion in the EIS. The following analysis procedure should be used: (1) Consult with the reviewers for ESRPs 4.6 and 5.10, and obtain a list of adverse environmental impacts from project construction and operation. (2) Organize these impacts as follows:					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(a) staff identified adverse impacts of construction and operation based on the project as proposed by the applicant (b) procedures and practices to mitigate or avoid these impacts (c) unavoidable adverse impacts that remain after all practical means to avoid or mitigate the impact have been taken. (3) Categorize the identified impacts according to the following format: • land use • hydrological and water use • ecological - terrestrial - aquatic • socioeconomic • radiological • atmospheric and meteorological • environmental justice. The categories may be further divided into construction and operational impacts if so desired. (4) Prepare a table summarizing the procedure followed in Steps 2 and 3 above, identifying the ESRP that provides details of the staff analysis. The table will describe the nature and magnitude of the impact (see Table 10.1-1 for example). (a) Determine the time scale of each impact (e.g., 4-6 months during construction, throughout the plant lifetime, indefinitely). (b) Identify (for subsequent use by the reviewer for ESRP 10.2)					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	any impacts that result in irreversible and irretrievable commitment of resources. (c) Include (for the reviewer for ESRP 10.3) those impacts that are to be considered short term or long term. (d) Consult with the appropriate ESRP Chapter 4.0 and 5.0 reviewers to ensure that adequate documentation, including applicant commitments to avoid adverse impacts, is available to support the staff conclusions regarding identification of each impact as adverse and unavailability of appropriate mitigating measures. (e) Ensure that each identified impact has been appropriately categorized. When a particular action or operation results in multiple impacts (e.g., access road construction and use may have impacts affecting land use, terrestrial ecology, and socioeconomics), ensure that the impacts are addressed in each appropriate category.					
10.2 (Draft Rev. 0, March 2000)	Irreversible and Irretrievable Commitments of Resources					
	Acceptance criteria for the evaluation of irreversible and irretrievable commitments of resources are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.45(b)(5) and 10 CFR 51, Appendix A to Subpart A, with respect to consideration of irreversible and irretrievable commitment of resources.					
	Regulatory positions and specific criteria to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the content and presentation of material in an applicant's ER.					
	The reviewer's analysis and summary of irreversible and irretrievable commitments of resources should consist of two sections: (1) irreversible environmental commitments (e.g., land-use productivity) predicted by the reviewers for ESRP Chapters 4.0 and 5.0, and (2) irretrievable material resources (e.g., steel) identified by the applicant as proposed for use in project construction and operation. The reviewer should identify these commitments and summarize them for inclusion in the EIS. The following analysis procedure should be used: (1) Consult with the reviewers for ESRP Chapters 4.0 and 5.0 and obtain a list of irreversible commitments of environmental resources based on the applicant's proposed project and the project with appropriate measures to limit and control adverse impacts. (2) Organize these commitments as follows: • staff identified commitments based on the project as proposed by the applicant • procedures and practices to minimize or avoid these commitments • unavoidable commitments that remain after all practical means to avoid or minimize the commitments have been taken. (3) Identify those materials (e.g., steel, concrete, uranium) that should be irretrievably committed during construction and operation of the plant.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	• Use the table format example shown in Table 10.2-1. • Analysis may be based on a standard (e.g., 1000 MWe) reactor size. • Modify the table on the basis of site- and plant-specific materials data supplied by the applicant. (4) Consult with the reviewer for ESRP 10.1 and with appropriate ESRP Chapters 4.0 and 5.0 reviewers to ensure that staff conclusions with respect to the irreversibility of environmental commitments are appropriate and can be supported. (5) Consider irreversible commitments as they may apply to the following categories: • land use • hydrological and water use • ecological - terrestrial - aquatic • socioeconomic • radiological • atmospheric and meteorological. (6) Ensure that the irretrievable commitments of material resources identified by the applicant are reasonable and consistent with the basic data of Table 10.2-1. (7) Ensure that any other material resources identified by the reviewers of ESRP Chapters 4.0 and 5.0 have been included. • Permanent resource commitments include land and uranium. • The generic table provided in 10 CFR 51.51 identifies					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	the environmental effects of the uranium fuel cycle for inclusion in the utilities' environmental report and provides information about uranium and related resources used in making nuclear fuel. (8) Ensure that the statement in the "Evaluation Findings" of this ESRP, with respect to uranium availability, has been updated to reflect current U.S. Department of Energy (DOE) resource analyses.					
10.3 (Draft Rev. 0, March, 2000)	Relationship Between Short Term Uses and Long Term Productivity of the Human Environment					
	Acceptance criteria for the evaluation of short term uses and long term productivity are based on the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.45(b)(4) and 10 CFR 51, Appendix A, with respect to consideration of the relationship between short-term uses and long-term productivity of the human environment.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to the content and presentation of material in an applicant's environmental report.					
	The reviewer's analysis of the relationship between short term uses and long term productivity should be based on the tabulation of unavoidable adverse environmental impacts and irreversible and irretrievable commitments of resources prepared by the reviewers for ESRPs 10.1 and 10.2 using the following					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	steps: (1) Consider that an occupation of land by plant structures for an indefinite period represents the maximum impact on long term productivity, unless other long term preemptions have been identified by these reviewers. (2) Identify through consultation with the appropriate ESRP Chapters 4.0 and 5.0 reviewers those other uses of the environment that will be precluded by plant construction and operation (e.g., loss of productive farmland) and that will classify these as either short term or long term preemptions. (3) Determine how any short term or long term benefits of the proposed project, as identified by appropriate ESRP Chapters 4.0 or 5.0 reviewers, affect any such preemptions. (4) As necessary, consult with appropriate Federal, State, regional, local, and affected Native American tribal agencies to make these determinations. (5) Evaluate the project's impact on short term use and long term productivity capabilities of the environment and determine if the EIS input statement given below is accurate and applicable.					
10.4 (Draft Rev. 0, March 2000)	Benefit-Cost Balance					
	The reviewer should ensure that the introductory paragraph prepared under this ESRP is consistent with the intent of the following regulation:					
	The ER must comply with the relevant requirements of 10 CFR					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	51.70(b) with respect to preparation of an EIS that is concise, clear, analytic, and written in plain language.					
10.4.1 (Draft Rev. 1, July 2007)	Benefits					
	Acceptance criteria of the analysis of benefits are based on meeting the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.45 and 51.71 with respect to the analyses required in the development of the environmental report (ER) and environmental impact statement (EIS).					
	The ER must comply with the relevant requirements of 10 CFR 51.50(b) with respect to reviewing applications for early site permits.					
	The ER must comply with the relevant requirements of 10 CFR 51.50(c) with respect to reviewing applications for combined licenses.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to information needs and formats for benefit-cost balancing.					
	To determine benefits, the reviewer should perform the following steps: (1) Ensure that all appropriate plant production benefits have been identified and quantified. Ensure that quantification of these benefits is correct and is consistent with the staff's findings in ESRP 8.4.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	(2) For other benefits, ensure the following: • All relevant benefits have been identified and established. • The quantification of each benefit is appropriate. • The relative significance of each benefit has been established and is appropriate to the impact. Benefits should be described in a tabular format similar to Table 10.4.1-1. Extra columns and rows may be added as needed. (3) Base benefits on the description of the project as proposed by the applicant, including post application modifications made by the applicant in response to staff assessments of measures to mitigate predicted environmental impacts. If staff identified alternatives have not been adopted by the applicant, identify and analyze benefits for the project as proposed by the applicant. (4) Identify and tabulate any other potential benefits of project construction and operation in consultation with appropriate ESRP reviewers. These potential benefits may include the following: • technical development • State and local tax revenues • incremental increase in regional productivity • enhancement of recreational values • enhancement of aesthetic values • environmental enhancement • creation and improvement of local roads or other facilities • intangible benefits (e.g., reduced dependence on					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	scarce fossil fuels). (5) Quantify benefits in monetary or other appropriate terms whenever possible and determine their significance on a political boundary or regional basis. When quantification of these benefits is not possible, make a qualitative assessment.					
10.4.2 (Draft Rev. 1, July 2007)	Costs					
	Acceptance criteria of the analysis of costs are based on meeting the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.45 and 51.71 with respect to the analyses required in the development of the environmental report (ER) and environmental impact statement (EIS).					
	The ER must comply with the relevant requirements of 10 CFR 51.50(b) with respect to reviewing applications for early site permits.					
	The ER must comply with the relevant requirements of 10 CFR 51.50(c) with respect to reviewing applications for combined licenses.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev. 2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to information needs and formats for a benefit-cost balancing.					
	The estimated internal and external costs to be considered by the reviewer should be based on the project as proposed by the					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	applicant and with modifications accepted by the applicant to mitigate and control predicted adverse impacts. Estimated internal and external costs should be described using the following procedures:					
	<u>Internal Costs Incurred by the Applicant</u> (1) Describe each identified internal estimated cost and outline the method used to obtain the described value (e.g., present worth cost). • Reference to other EIS sections may be made (when appropriate) to present the basis for the staff analysis. (2) List estimated costs. A sample format is shown in Table 10.4.2-1. • Where the information to be presented would be included in the table described in ESRP 10.4.3, the reviewer may reference that table instead of repeating the information in the table for this section. • Internal costs include capital costs (including the estimated capital cost of added transmission lines to support the proposed project even if the lines are not paid for by the applicant), operating and maintenance costs, fuel costs, and decommissioning costs. • Other costs may be classified as internal, when appropriate. • Express all internal costs, either provided by the applicant or estimated by the staff, in monetary terms. • For all internal costs, determine the present worth cost and levelized annual equivalent cost.					

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Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	- Express present worth costs in dollars of the first year of commercial operation of the first unit. - Express annual costs in dollars per year and mills per kilowatt-hour for the first year of commercial operation of the first unit. (3) Use methods and economic assumptions consistent with those used in the Chapter 9.4 ESRPs, and use the results of calculations presented by the reviewers of the Chapter 9.4 ESRPs when available. (a) Where plant capacity affects estimated internal costs, estimate the cost for both the high and low extremes of any range of plant-capacity factors assumed in the review for the Chapter 9.4 ESRPs. (b) Sum the present worth values of the internal costs to arrive at a total present worth internal cost of the proposed project.					
	<u>External Costs</u> (1) Estimate each external cost associated with an environmental impact, reference the corresponding environmental statement section, and describe or reference the method used to develop the cost data. • For quantified costs, show the relationship (significance) of the cost to the regional value of the impacted parameter. • Where costs cannot be quantified, estimate the significance of the cost as it relates to regional values. Cost data may be presented in tabular form or referenced to an					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	equivalent table (if provided) for ESRP 10.4.3. (2) Estimate the external costs of project construction and operation in consultation with the reviewers of ESRP Chapters 4.0, 5.0, 6.0, and 7.0. • Identify and tabulate each unmitigated adverse impact and estimate its cost. • Consider the costs of mitigated adverse impacts and appropriately assign these as internal or external costs. • Estimate costs in monetary or other appropriate terms whenever possible, and determine the significance of costs on a regional basis. - If monetary terms can be estimated , calculate them for the same time (year) selected for the internal-cost analysis. - If external costs cannot be quantified, present qualitative cost estimates for each such impact. The following typical cost terms (shown for a loss of offsite agricultural production) might be used: Monetary: "Annual loss of $4000.00 to soybean producers. The annual regional value of this crop to producers is $200,000.00." Quantitative: "Annual loss of 50 hectares of soybean cropland. The regional cropland used for soybean production averages 300 hectares." Qualitative: "MODERATE impact to regional production of					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	soybeans." (3) For estimated external costs, ensure the following: • Adverse impacts requiring mitigation or avoidance have been identified. • The estimated cost assigned to each impact is appropriate. • The relative significance of each estimated cost has been established and is appropriate to the impact. • Unavoidable adverse environmental impacts identified in ESRP 10.1 have been considered and assigned cost values, if appropriate. • All other external costs (e.g., resource commitments) not associated with an identified environmental impact have been considered. (4) Ensure that any transfer payment (e.g., tax) listed as a benefit in ESRP 10.4.1 has a corresponding cost considered in this section. (5) If estimated costs of measures and controls to mitigate, or alternatives to avoid, environmental impacts have been considered, ensure that all such costs have been presented in a manner that permits their comparison with corresponding costs of project elements as proposed by the applicant. (6) If environmentally preferrable alternatives have been identified, prepare a cost comparison for these alternatives and the applicant's proposal. • Reference the appropriate EIS section describing the impact and ESRP Chapter 9.4 analyses comparing					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	the applicant's proposal and the alternative(s).					
10.4.3 (Draft Rev. 0, March 2000)	Summary					
	Acceptance criteria for the review of the benefit-cost summary of the proposed action are based on meeting the relevant requirements of the following:					
	The ER must comply with the relevant requirements of 10 CFR 51.45(d) and 51.71(d) with respect to the analyses required in the development of the ER and EIS.					
	The ER must comply with the relevant requirements of 10 CFR 52.18 with respect to reviewing applications for early site permits.					
	The ER must comply with the relevant requirements of 10 CFR 52.81 with respect to reviewing applications for combined licenses.					
	The ER must comply with the relevant requirements of 10 CFR 51.95(c)(4) with respect to decision criteria for a record of decision.					
	Regulatory positions and specific criteria necessary to meet the regulations identified above are as follows:					
	The ER must comply with the relevant requirements of Regulatory Guide 4.2, Rev.2, Preparation of Environmental Reports for Nuclear Power Stations (NRC 1976), with respect to information needs and formats for benefit-cost balance.					
	This benefit-cost balancing should be conducted under the direction of those most knowledgeable of the entire project, the EPM, and the review team leader in concert with each individual reviewer. The reviewer may evaluate the benefit-cost balancing prepared by the applicant or may conduct such balancing independently.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	The reviewer should do the following steps: (1) Tabulate the benefits to be derived from the proposed project. This was done as a part of the review of ESRP 10.4.1. (2) Analyze the benefits in terms of megawatt-hours of electrical energy generated, megawatts of capacity, and less tangible benefits such as recreational or educational facilities resulting from the proposed project. Consider the benefits of the project as modified by suggested alternatives that have not been adopted by the applicant. (3) Tabulate the environmental costs of the project as proposed by the applicant, including all commitments made in the most recent amendments to the applicant's ER. This was done as a part of the review of ESRP 10.4.2. • This tabulation should include costs for each of the environmental impacts and other costs determined by the reviewer for ESRP 10.4.2. • This should also include the environmental costs of alternatives for measures and controls to mitigate adverse impacts that have not been adopted by the applicant. (4) Consider the following characteristics of each environmental cost in this analysis: • the environmental effect • the impact expected, quantified if possible • the relative significance of the cost and impact as compared to similar resources available in the region, quantified if possible.					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN

Section No./Rev.	Title/Requirement	Applicable?	Reg. or Guidance?		Add'l Reg. Needed?	Basis/Comment
	(5) Consider any environmentally preferable alternative identified by the reviewers for ESRPs 9.2 and 9.3 (which, if adopted by the staff, would imply recommended denial of the application to construct a nuclear power plant). • Provide the benefit-cost balance for these alternatives to determine if any may be considered as obviously superior to the proposed project. • Similarly assist the reviewers for ESRP 9.4 to determine if any environmentally preferable alternative plant or transmission-system component would have a benefit-cost balance that warrants its being recommended as an alternative to the proposed component. (6) Review the tabulation of the staff's assessments of the environmental costs and benefits of the project as proposed by the applicant and establish the reasonableness, accuracy, and completeness of the tabulation. This tabulation forms the baseline from which the acceptability of costs and benefits of additional requirements should be established. (7) Review the modifications identified by the staff in terms of absolute and relative environmental improvement and absolute and relative additional cost to the utility and community. (8) Express the environmental modifications and costs in various manners and units to ensure that the relative significance is expressed in the most useful perspective for decisionmaking. • If the environmental improvements are determined to be cost beneficial, note this in the tabulation along with any conditions to be included in the summary and					

TABLE A1-16: NUREG-1555 - ENVIRONMENTAL REPORT STANDARD REVIEW PLAN						
Section No./Rev.	**Title/Requirement**	**Applicable?**	**Reg. or Guidance?**		**Add'l Reg. Needed?**	**Basis/Comment**
	conclusions section of the EIS. • If the environmental improvement is determined to be not cost-beneficial, the affected sections of the EIS should be written to reflect this conclusion. (9) After considering the benefit-cost aspects of the project, balance the benefits of the proposed project (tabulation of ESRP 10.4.1) against the total environmental costs (tabulation of ESRP 10.4.2) and reach a final conclusion as to the overall benefit-cost balance of the project.					
Appendix A	Guide to Relevant Environmental Standard Review Plans					
Appendix B	Review Responsibilities					

Table A1-17: Interim Staff Guidance						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
DC/COL-ISG-1 (Issued Final)	Interim Staff Guidance On Seismic Issues of High Frequency Ground Motion					
DC/COL-ISG-2 (Issued Final)	Interim Staff Guidance on Financial Qualifications of Applicants For Combined License Applications					Exclude. Administrative
DC/COL-ISG-3 (Issued Final)	PRA Information to Support Design Certification and Combined License Applications					
COL/ESP-ISG-4 (Issued Final)	Interim Staff Guidance on the Definition of Construction and on Limited Work Authorizations					
DC/COL-ISG-5 (Issued Final)	GALE86 Code for Calculation of Routine Radioactive Releases in Gaseous and Liquid Effluents to Support Design Certification and Combined License Applications					
DC/COL-ISG-6 (Issued Final)	Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications					
DC/COL-ISG-7 (Issued Final)	Assesment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures					
DC/COL-ISG-8 (Issued Final News Note Regarding Final Issuance)	Necessary Content of Plant-Specific Technical Specifications					
DC/COL-ISG-010 (Issued Final)	Review of Evaluation To Address Adverse Flow Effects in Equipment Other Than Reactor Internals					
DC/COL-ISG-011 (Issued Final)	Finalizing Licensing-basis Information					Exclude. Administrative
DC/COL-ISG-013 (Issued for Comments)	Interim Staff Guidance on NUREG-0800 Standard Review Plan Section 11.2 and Branch Technical Position 11-6 Assessing the Consequences of an Accidental Release of Radioactive Materials from Liquid Waste Tanks for Combined License					

Table A1-17: Interim Staff Guidance

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Applications Submitted under 10 CFR Part 52					
DC/COL-ISG-014 (Issued Final)	Assessing Ground Water Flow and Transport of Accidental Radionuclide Releases					
DC/COL-ISG-015 (Issued Final)	Post-Combined License Commitments					Exclude. Administrative
DC/COL-ISG-016 (Issued for Comments)	Staff Guidance on Interim Staff Guidance DC/COL-ISG-016 – Compliance With 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d)					
DC/COL-ISG-017 (Issued for Comments)	Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses					
DC/COL-ISG-018 (Issued for Comments)	Section 17.4 Reliability Assurance Program					
DC/COL-ISG-019 (Issued for Comments)	Gas Accumulation Issues in Safety Related Systems	NA				Exclude. Not applicable to the gas cooled HTGR
DC/COL-ISG-020 (Issued for Comments)	Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors					
DC/COL-ISG-021 (Issued for Comments)	Review of Nuclear Power Plant Designs using a Gas Turbine Driven Standby Emergency Alternating Current Power System	NA				Exclude. The HTGR does not rely on standby emergency AC power

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
GL 2006-02, (February 1, 2006)	Generic Letter "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power"					
GL 2003-01 (June 12, 2003)	Control Room Habitability					
SECY-88-0202, (July 15, 1988)	"Standardization of Advanced Reactor Designs," NRC. (ADAMS Accession No. ML051740706)					
SECY-88-0203, (July 15, 1988)	"Key Licensing Issues Associated with DOE-Sponsored Advanced Reactors," NRC. (ADAMS Accession No. ML051830035)					
SECY-88-0202 and SECY-88-0203 (September 19, 1988)	Staff Requirements Memorandum (SRM) for "Commission Action on the Key Licensing and Standardization Issues Associated with the DOE Advanced Reactor Concepts, NRC. (ADAMS Accession No. ML010650233)					
SECY-88-0202 and SECY-88-0203 (November 14, 1988)	SRM for "Standardization of Advanced Reactor Designs" and "Key Licensing Issues Associated with DOE Sponsored Advanced Reactor Designs," NRC. (ADAMS Accession No. ML010940239)					
SECY-91-0202 (July 2, 1991)	"Departures from Current Regulatory Requirements in Conducting Advanced Reactor Reviews," NRC. (ADAMS Accession No. ML051740732)					
SECY-93-0092 (April 8, 1993)	"Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," NRC. (ADAMS Accession No. ML040210725)					
SECY-93-092 (July 30, 1993)	SRM for "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," NRC. (ADAMS Accession No. ML003760774)					
NUREG-1368	"Pre-Application Safety Evaluation Report for the	NA				Exclude. Not applicable to the

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
(February 1994)	Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," NRC. (ADAMS Accession No. ML063410561)					HTGR, use NUREG-1338
SECY-95-035 (February 15, 1995)	"Reassessment of Fee Billing Practices and Fee Policy for Office of Nuclear Regulatory Research (RES) Activities Associated with Design Certification (DC) Applications," NRC. (ADAMS Accession No. ML023230188)					Exclude. Administrative
SECY-95-035 (March 10, 1995)	SRM for "Reassessment of Fee Billing Practices and Fee Policy for Office of Nuclear Regulatory Research (RES) Activities Associated with Design Certification (DC) Applications," NRC. (ADAMS Accession No. ML003756447)					Exclude. Administrative
NUREG-1338 (December 1995)	"Pre-Application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR) - Draft Copy of the Final Report," NRC. (ADAMS Accession No. ML052780519)					
Letter from the ACRS (May 19, 1999)	"The Role of Defense-In-Depth in a Risk-Informed Regulatory System." (ADAMS Accession No. ML091280427)					
Letter from Exelon Generation to NRC, (May 10, 2001)	"Regulatory Issues Related to the Pebble Bed Modular Reactor (PBMR)." (ADAMS Accession No. ML011420393)					
SECY-01-0207 (November 10, 2001)	"Legal and Financial Issues Related to Exelon's Pebble Bed Modular Reactor PBMR)," NRC. (ADAMS Accession No. ML012850139)					
SECY-01-207 (January 14, 2002)	SRM for "Legal and Financial Issues Related to Exelon's Pebble Bed Modular Reactor (PBMR)," Letter from the Advisory Committee on Reactor Safeguards, (ADAMS Accession No. ML020150058)					

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Industry White Paper (June 17, 2002)	"Integrated Approach to Modular Plant Licensing, Nuclear Energy Institute." (ADAMS Accession No. ML021970596)					
SECY-02-0139 (July 22, 2002)	"Plan for Resolving Policy Issues Related to Licensing Non-Light-Water Reactor Designs," NRC." (ADAMS Accession No. ML021790610)					
SECY-02-0176 (September 30, 2002)	"Proposed Rulemaking to Add New Section 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components" (WITS 199900061)," NRC. (ADAMS Accession No. ML022630164)					
SECY-02-0180 (October 7, 2002)	"Legal and Financial Policy Issues Associated with Licensing New Nuclear Power Plants," NRC." (ADAMS Accession No. ML023600088)					
SECY-03-0047 (March 28, 2003)	"Policy Issues Related to Licensing Non-Light-Water Reactor Designs," NRC." (ADAMS Accession No. ML030160002)					
SECY-02-180 (March 31, 2003)	SRM for "Staff Requirements – Legal and Financial Policy Issues Associated with Licensing New Nuclear Power Plants," NRC." (ADAMS Accession No. ML030900371)					
SECY-03-0059 (April 18, 2003)	"NRC's Advanced Reactor Research Program," NRC." (ADAMS Accession Nos. ML023310534, ML023310550, and ML023310540)					
SECY-03-0047 (June 26, 2003)	SRM for "Staff Requirements – Policy Issues Related to Licensing Non-Light-Water Reactor Designs," NRC." (ADAMS Accession No. ML031770124)					
Letter from the ACRC, (April 22, 2004)	"Options and Recommendations for Policy Issues Related to Licensing Non-Light-Water Reactor Design". (ADAMS Accession No. ML041250415)					
SECY-04-0103, (June 26, 2003)	"Status of Response to the June 26, 2003, Staff Requirements Memorandum on Policy Issues					

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Related to Licensing Non-Light-Water Reactor Designs," U.S. Nuclear on Policy Issues Related to Licensing Non-Light-Water Reactor Designs," NRC. (ADAMS Accession No. ML0411405211)					
SECY-04-0157 (August 30, 2004)	"Status of Staff's Proposed Regulatory Structure for New Plant Licensing and Potentially New Policy Issues," NRC. (ADAMS Accession No. ML042370388)					
SECY-05-0006 (January 7, 2005)	"Second Status Paper on the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing," NRC. (ADAMS Accession No. ML050120279)					
NRC Briefing, (April 5, 2005) (May 9, 2005.)	SRM for "Staff Requirements – Briefing on RES Programs, Performance, and Plans," NRC, (ADAMS Accession No. ML051290351)					
SECY-05-0120 (July 6, 2005)	"Security Design Expectations for New Reactor Licensing Activities," NRC. (ADAMS Accession No. ML051100233)					
SECY-05-0120) (September 9, 2005)	SRM for "Staff Requirements – Security Design Expectations for New Reactor Licensing Activities," NRC. (ADAMS Accession No. ML052520334)					
SECY-05-0130 (July 21, 2005)	"Policy Issues Related to New Plant Licensing and Status of the Technology-Neutral Framework for New Plant Licensing," NRC, (ADAMS Accession No. ML051670388)					
SECY-05-0130 (September 14, 2005)	SRM for "Policy Issues Related to New Plant Licensing and Status of the Technology-Neutral Framework for New Plant Licensing," NRC. (ADAMS Accession No. ML052570437)					
Letter from the ACRS	"Report on Two Policy Issues Related to New Plant Licensing," (ADAMS Accession No. ML052640580)					

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
(September 21, 2005)						
SECY-06-0007 (January 9, 2006)	"Staff Plan to Make a Risk-Informed and Performance-Based Revision to 10 CFR Part 50," NRC. (ADAMS Accession No. ML053420012)					
Letter from E. Wallace, Pebble Bed Modular Reactor (Pty) Ltd. to NRC (February 16, 2006)	"Submittal of PBMR Technical Description Document." (ADAMS Accession No. ML060540393)					
SECY-06-0007 (March 22, 2006)	SRM for "Staff Plan to Make a Risk-Informed and Performance-Based Revision to 10 CFR Part 50," NRC. (ADAMS Accession No. ML060810277)					
Letter from E. Wallace, Pebble Bed Modular Reactor (Pty) Ltd. to the NRC, (June 13, 2006)	"PBMR White Paper: PRA Approach," (ADAMS Accession No. ML061650404)					
SECY-07-0101 (June 14, 2007)	"Staff Recommendations Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50 (RIN 3150-AH81)," NRC. (ADAMS Accession Nos. ML070790236 and ML071010383)					
SECY-07-0101 (September 10, 2007)	SRM for "Staff Recommendations Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50 (RIN 3150-AH81)," NRC. (ADAMS Accession No. ML072530501)					
NUREG-1860 (December 2007)	"Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50," NRC. (ADAMS Accession Nos. ML073400763 and					

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	ML073400800)					
NRC memorandum (May 19, 2008)	"Anticipated Regulatory Issues Involving the Potential for Small Amounts of Tritium to Enter into Fluid Products Made with Nuclear Process Heat," NRC. (ADAMS Accession No. ML0813000685)					
COMSECY-08-0018 (June 16, 2008)	SRM for "Staff Requirements - Report to Congress on Next Generation Nuclear Plant (NGNP) Licensing Strategy," NRC. (ADAMS Accession No. ML081680501)					
NRC Meeting (August 7, 2008)	"Summary of Pre-Application Kickoff Meeting with NuScale Power Inc. on the NuScale Reactor Design and Proposed Licensing Activities," NRC. (ADAMS Accession No. ML082140161)					
SECY-08-0117 (August 7, 2008)	"Staff Approach to Verify Closure of Inspections, Tests, Analyses, and Acceptance Criteria and to Implement Title 10 CFR 52.99, 'Inspection During Construction,' and Related Portion of 10 CFR 52.103(G) on the Commission Finding," NRC. (ADAMS Accession Nos. ML081220237, ML081080355, ML081080392, and ML081080399)					
NRC Report to Congress (August 2008)	"Next Generation Nuclear Plant Licensing Strategy – A Report to Congress," NRC. (ADAMS Accession No. ML082290017)					
SECY-08-0130 (September 11, 2008)	"Updated Policy Statement on Regulation of Advanced Reactors," NRC. (ADAMS Accession No. ML082261489)					
SECY-08-0134 (September 12, 2008)	"Regulatory Structure for Spent Fuel Reprocessing – ABR Gap Analysis," NRC. (ADAMS Accession No. ML082110363)	NA				Exclude. Not applicable to the HTGR
SECY-08-0152 (October 15, 2008)	"Final Rule – Consideration of Aircraft Impacts for New Nuclear Power Reactors," NRC. (ADAMS Accession No. ML081050227)					

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
SECY-08-0117 (January 14, 2009)	SRM for "Staff Requirements – Staff Approach to Verify Closure of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) and to Implement Title 10 CFR 52.99, "Inspection During Construction," and Related Portion of 10 CFR 52.103(g) on the Commission Finding," NRC. (ADAMS Accession No. ML090140136)					
NRC Memorandum (February 12, 2009)	"Alternative Risk Metrics for New Light-Water Reactor Risk-Informed Applications," NRC. (ADAMS Accession No. ML090150636)					
Federal Register Volume 74, page 12735 (March 25, 2009)	"Variable Annual Fee Structure for Power Reactors,"					Exclude. Administrative
Letter from NEI (March 27, 2009)	"Transmission of Industry White Paper, "Risk Metrics for Operating New Reactors," for ACRS Review," (ADAMS Accession No. ML090900674)					
SECY-09-0056 (April 7, 2009)	"Staff Approach Regarding a Risk-Informed and Performance-Based Revision to Part 50 of Title 10 of the *Code of Federal Regulations* and Developing a Policy Statement on Defense-In-Depth for Future Reactors," NRC. (ADAMS Accession No. ML090360197)					
Letter from the ACRS (April 9, 2009)	"Draft Final Revision 2 to RG 1.200, 'An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Committee on Reactor Safeguards. (ADAMS Accession No. ML090930396)					
WCAP-17063-P and WCAP-	"Revising the EPZ for IRIS" (Proprietary) (ADAMS Accession No. ML091120356) and, "Revising the					

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
17063-NP (April 14, 2009)	EPZ for IRIS" (Non-Proprietary) (ADAMS Accession No. ML091120350), Westinghouse Electric Company.					
Federal Register, Volume 74, page 28112 (June 12, 2009)	"10 CFR Parts 50 and 52 – Consideration of Aircraft Impacts for New Nuclear Power Reactors; Final Rule."					
NRC Meeting (August 11, 2009)	"Summary of Pre-Application Kick-Off Meeting with Babcock & Wilcox on the mPower Reactor Design and Proposed Licensing Activities," NRC. (ADAMS Accession No. ML092170351)					
SECY-09-0119 (August 26, 2009)	"Staff Progress in Resolving Issues Associated with Inspections, Tests, Analyses, and Acceptance Criteria," NRC. (ADAMS Accession No. ML091980327)					
DOE Report DE-FOA-0000149 (September 18, 2009)	"Financial Assistance Funding Opportunity Announcement, Next Generation Nuclear Plant Program – Gas Cooled Reactor Design and Demonstration Projects," U.S. DOE					Exclude. Administrative
NRC Workshop (October 22, 2009)	"Summary of Workshop on Small- and Medium-Sized Nuclear Reactors (SMRs)," NRC. (ADAMS Accession No. ML092940138)					
SECY-09-0137 (September 23, 2009)	"Next Steps for Advance Notice of Proposed Rulemaking on Variable Annual Fee Structure for Power Reactors," NRC. (ADAMS Accession No. ML092660166)					Exclude. Administrative
SECY-09-0137 (October 13, 2009)	SRM for "Staff Requirements – Next Steps for Advance Notice of Proposed Rulemaking on Variable Annual Fee Structure for Power Reactors," NRC. (ADAMS Accession No. ML092861070)					Exclude. Administrative

Table A1-18: Generic Letters and SECY Documents

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
NRC Briefing (September 22, 2009)	SRM for "Staff Requirements – Periodic Briefing on New Reactor Issues – Progress in Resolving Issues Associated With Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC), NRC. (ADAMS Accession No. ML092890658)					

10 CFR 52.79(a)(17) requires that License Applications include information "with respect to compliance with technically relevant positions of the Three Mile Island requirements in Section 50.34(f) . . . " Some exceptions are provided. The source used for these requirements was NUREG-0737.

This table lists those requirements. For convenience, the appropriate section of 10 CFR 50.34(f) is listed in brackets following the TMI Action Plan ID number.

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
I.A.4.2 [10 CFR 50.34(f)(2)(i)]	Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs					
I.C.5 [10 CFR 50.34(f)(3)(i)]	Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.					
I.C.9 [10 CFR 50.34(f)(2)(ii)]	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures.					
I.D.1 [10 CFR 50.34(f)(2)(iii)]	Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts.					Exclude, current requirements address this as a part of Chapter 18 of the License Application.
I.D.2 [10 CFR 50.34(f)(2)(iv)]	Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of					

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.					
I.D.3 [10 CFR 50.34(f)(2)(v)]	Provide for automatic indication of the bypassed and operable status of safety systems.					Exclude. This requirement is now addressed as parts of other regulations and is generally addressed extensively throughout Chapter 7 of the License Application.
I.F.1 [10 CFR 50.34(f)(3)(ii)]	Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR Part 50 includes all structures, systems, and components important to safety.					
I.F.2 [10 CFR 50.34(f)(3)(iii)]	Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of “as- built” documentation; and (H) providing a QA role in design and analysis activities.					Administrative, Exclude from review, Applicable to all applicants.

Table A1-19: Three Mile Island Requirements (NUREG-0737)						
ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
II.B.2 [10 CFR 50.34(f)(2)(vii)]	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect equipment from the radiation environment.					
II.B.3 [10 CFR 50.34(f)(2)(viii)]	Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.					
II.B.8 [10 CFR 50.34(f)(1)(i)]	Levels 1 (Plant), 2 (Containment) & 3 (Site) PRAs to confirm meeting NRC Safety Goals.					Exclude, PRA is now the normal practice for License Applications.
II.B.8 [10 CFR 50.34(f)(1)(xii)]	Include a hydrogen control system that satisfies the requirements of 10 CFR 50.34 (f)(2)(ix). As a minimum consider hydrogen ignition and post-accident inerting.					
II.B.8 [10 CFR 50.34(f)(2)(ix)]	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage.					Exclude, this TMI item has been superseded by revisions to 10 CFR 50.44. Also, fuel-clad metal water reactions are not applicable to HTGR designs.
II.B.8 [10 CFR 50.34(f)(3)(iv)]	Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot					

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system.					
II.B.8 [10 CFR 50.34(f)(3)(v)]	Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8) (A)(1) Containment integrity will be maintained (i.e., for steel containment by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE- 3220, Service Level C Limits, except that evaluation of instablity is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions. (2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code,					

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	which are referenced in paragraphs (f)(3)(v)(A)(1) and (f)(3)(v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. (B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category. (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.					
II.D.1 [10 CFR 50.34(f)(2)(x)]	Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves … for all fluid conditions expected under operating conditions, transients and accidents . . . Consideration of ATWS conditions shall be included.					
II.D.3 [10 CFR 50.34(f)(2)(xi)]	Provide direct indication of relief and safety valve position (open or closed) in the control room.					

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
II.E.1.1 [10 CFR 50.34(f)(1)(ii)]	PWR Auxiliary Feedwater System evaluation.					Exclude, applicable to PWRs only.
II.E.1.2 [10 CFR 50.34(f)(2)(xii)]	Provide automatic and manual auxiliary feedwater (AFW) system initiation and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWRs only)					Exclude, applicable to PWRs only.
II.E.3.1 [10 CFR 50.34(f)(2)(xiii)]	Provide pressurizer heater power supply and associated motive and control power … . (Applicable to PWRs only)					Exclude, applicable to PWRs only.
II.E.4.1 [10 CFR 50.34(f)(3)(vi)]	For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.					
II.E.4.2 [10 CFR 50.34(f)(2)(xiv)]	Provide containment isolation systems that: (A) Ensure all non-essential systems are isolated automatically by the containment isolation system, (B) For each non-essential penetration (except instrument lines) have two isolation barriers in series, (C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal, (D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operations, and (E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.					
II.E.4.4 [10 CFR 50.34(f)(2)(xv)]	Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.					
II.E.5.1 [10 CFR	Establish a design criterion for the allowable number					Exclude, applicable to B&W

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
50.34(f)(2)(xvi)]	of actuation cycles of the ECCS and RPS consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only).					designed plants only.
II.F.1 [10 CFR 50.34(f)(2)(xvii)]	Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.					
II.F.2 [10 CFR 50.34(f)(2)(xviii)]	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as … a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's.					
II.F.3 [10 CFR 50.34(f)(2)(xix)]	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.					
II.G.1 [10 CFR 50.34(f)(2)(xx)]	Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: … . (Applicable to PWRs only)					Exclude, applicable to PWRs only.
II.J.3.1 [10 CFR 50.34 (f)(3)(vii)]	Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of					

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort.					
II.K.1.22 [10 CFR 50.34(f)(2)(xxi)]	Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only)					Exclude, applicable to BWRs only.
II.K.2.9 [10 CFR 50.34(f)(2)(xxii)]	Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only).					Exclude, applicable to B&W designed plants only.
II.K.2.10 [10 CFR 50.34(f)(2)(xxiii)]	Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on the loss of main feedwater and on turbine trip. (Applicable to B&W- designed plants only).					Exclude, applicable to B&W designed plants only.
II.K.3.2 [10 CFR 50.34(f)(1)(iv)]	Power Operated Relief Valves					Exclude, applicable to PWRs only.
II.K.3.13 [10 CFR 50.34(f)(1)(v)]	Separate High Pressure Coolant Injection (HPCI), High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) system initiation levels such that RCIC initiates at a higher water level than HPCI/HPCS.					Exclude, applicable to BWRs only.
II.K.3.16 [10	Perform a study to identify practical system					Exclude, applicable to BWRs

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
CFR 50.34(f)(1)(vi)]	modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only).					only.
II.K.3.18 [10 CFR 50.34(f)(1)(vii)]	Perform a feasibility and risk assessment study to determine the optimum Automatic Depressurization System (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWRs only)					Exclude, applicable to BWRs only.
II.K.3.21 [10 CFR 50.34(f)(1)(viii)]	Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present (Applicable to BWR's only).					Exclude, applicable to BWRs only.
II.K.3.23 [10 CFR 50.34(f)(2)(xxiv)]	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWRs only).					
II.K.3.24 [10 CFR 50.34(f)(1)(ix)]	Provide space cooling for RCIC, HPCI/HPCS systems for 2 hours following complete loss of offsite power. (Applicable to BWR's only).					Exclude, applicable to BWRs only.
II.K.3.28 [10 CFR 50.34(f)(1)(x)]	Ensure that ADS valves, accumulators and associated equipment will be capable of performing its intended functions during and following an accident.					Exclude, applicable to BWRs only.
II.K.3.45 [10 CFR 50.34(f)(1)(xi)]	Evaluate depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only)					Exclude, applicable to BWRs only.

Table A1-19: Three Mile Island Requirements (NUREG-0737)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
II.K.2.16 and II.K.3.25 [10 CFR 50.34(f)(1)(iii)]	Reactor Coolant Pump Seal damage.					
III.A.1.2 [10 CFR 50.34(f)(2)(xxv)]	Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near-site Emergency Operations Facility					
III.D.1.1 [10 CFR 50.34(f)(2)(xxvi)]	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.					
III.D.3.3 [10 CFR 50.34(f)(2)(xxvii)]	Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions.					
III.D.3.4 [10 CFR 50.34(f)(2)(xxviii)]	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term release, and make necessary design provisions to preclude such problems.					

10 CFR 52.79(a)(20) requires that the content of license applications include "proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design." This table is an inclusive list of Unresolved Safety Issues and Generic Safety Issues. Many of the issues have been resolved or have limited applicability, but are listed for completeness and for the information of the analyst performing the Regulatory Gap Analysis.

In addition, the Chernobyl issues related to graphite-moderated reactors are included. These issues were originally identified in NUREG-1251 and the plan for resolution was endorsed in SECY 89-081.

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
A-1	Water Hammer					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-3	Westinghouse Steam Generator Tube Integrity					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-4	CE Steam Generator Tube Integrity					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-5	B&W Steam Generator Tube Integrity					Exclude. This issue was resolved by implementation of other regulatory requirements.

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
A-6	Mark I Short-Term Program					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-7	Mark I Long-Term Program					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-8	Mark II Containment Pool Dynamic Loads Long-Term Program					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-9	ATWS					Exclude. This issue was resolved by implementation of other regulatory requirements. (10 CFR 50.62)
A-10	BWR Feedwater Nozzle Cracking					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-11	Reactor Vessel Materials Toughness					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-13	Snubber Operability Assurance					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-14	Flaw Detection					Exclude. This issue has been dropped.
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-16	Steam Effects on BWR Core Spray Distribution					Exclude. Applicable only to

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						Boiling Water Reactors.
A-17	Systems Interactions in Nuclear Power Plants					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-18	Pipe Rupture Design Criteria					Exclude. This issue has been dropped.
A-19	Digital Computer Protection System					
A-20	Impacts of the Coal Fuel Cycle Description					See Item B-72
A-21	Main Steam Line Break Inside Containment – Evaluation of Environmental Conditions for Equipment Qualification					Exclude. This issue has been dropped.
A-22	PWR Main Steam Line Break – Core, Reactor Vessel, and Containment Building Response					Exclude. This issue has been dropped.
A-23	Containment Leak Testing					
A-24	Qualification of Class 1E Safety-Related Equipment					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-25	Non-Safety Loads on Class 1E Power Sources					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-26	Reactor Vessel Pressure Transient Protection					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-27	Reload Applications					
A-28	Increase in Spent Fuel Pool Storage Capacity					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage					Exclude. This issue was resolved with no new requirements.
A-30	Adequacy of Safety- Related DC Power Supplies					Exclude. Issue integrated into Issue 128.

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
A-31	RHR Shutdown Requirements					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-32	Missile Effects					Exclude. Addressed in Items A-37, A-38 and B-68 (all of which have been dropped)
A-33	NEPA Review of Accident Risks					
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-35	Adequacy of Offsite Power Systems					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-36	Control of Heavy Loads Near Spent Fuel					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-37	Turbine Missiles					Exclude. This issue has been dropped.
A-38	Tornado Missiles					Exclude. This issue has been dropped.
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-40	Seismic Design Criteria					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-41	Long-Term Seismic Program					
A-42	Pipe Cracks in Boiling Water Reactors					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-43	Containment Emergency Sump Performance					Exclude. This issue was resolved by implementation of

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						other regulatory requirements.
A-44	Station Blackout					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-45	Shutdown Decay Heat Removal Requirements					Exclude. This issue was resolved with no new requirements.
A-46	Seismic Qualification of Equipment in Operating Plants					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-47	Safety Implications of Control Systems					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment					Exclude. This issue was resolved by implementation of other regulatory requirements.
A-49	Pressurized Thermal Shock					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-1	Environmental Technical Specifications					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-2	Forecasting Electricity Demand					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-3	Event Categorization					Exclude. This issue has been dropped.
B-4	ECCS Reliability					Exclude. Covered under one of the TMI Action Plan Items
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments					Exclude. This issue was resolved with no new requirements.

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
B-6	Loads, Load Combinations, Stress Limits					Exclude. Covered by Issue 119.1.
B-7	Secondary Accident Consequence Modeling					Exclude. This issue has been dropped.
B-8	Locking out of ECCS Power-Operated Valves					Exclude. This issue has been dropped.
B-9	Electrical Cable Penetrations of Containment					Exclude. This issue was resolved with no new requirements.
B-10	Behavior of BWR Mark III Containments					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-11	Subcompartment Standard Problems					
B-12	Containment Cooling Requirements (Non- LOCA)					Exclude. This issue was resolved with no new requirements.
B-13	Marviken Test Data Evaluation					
B-14	Study of Hydrogen Mixing Capability in Containment Post- LOCA					Exclude. Covered under Item A-48.
B-15	CONTEMPT Computer Code Maintenance					Exclude. This issue has been dropped.
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment					Exclude. Issue incorporated into Item A-18.
B-17	Criteria for Safety- Related Operator Actions					Exclude. This issue was resolved with no new requirements.
B-18	Vortex Suppression Requirements for Containment Sumps					Exclude. Issue incorporated into Item A-43.
B-19	Thermal-Hydraulic Stability					Exclude. This issue was resolved with no new requirements.
B-20	Standard Problem Analysis					

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
B-21	Core Physics					Exclude. This issue has been dropped.
B-22	LWR Fuel					Exclude. This issue has been dropped.
B-23	LMFBR Fuel					Exclude. This issue has been dropped.
B-24	Seismic Qualification of Electrical and Mechanical Equipment					Exclude. Issue is covered under Issue A-46.
B-25	Piping Benchmark Problems					
B-26	Structural Integrity of Containment Penetrations					Exclude. This issue was resolved with no new requirements.
B-27	Implementation and Use of Subsection NF					
B-28	Radionuclide/ Sediment Transport Program					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-29	Effectiveness of Ultimate Heat Sinks					
B-30	Design Basis Floods and Probability					
B-31	Dam Failure Model					Should be included in evaluation. Although dam failure models are not technology specific, adequacy of such modeling is an emerging issue with the NRC.
B-32	Ice Effects on Safety- Related Water Supplies					Exclude. Addressed in the evaluation of Issue 153.
B-33	Dose Assessment Methodology					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-34	Occupational Radiation Exposure Reduction					Exclude. This issue was covered under TMI Action Plan Item III.D.3.1, which was

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
						resolved with no new requirements being established.
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water-Cooled Power Reactors					
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-37	Chemical Discharges to Receiving Waters					
B-38	Reconnaissance Level Investigations					Exclude. This issue has been dropped.
B-39	Transmission Lines					Exclude. This issue has been dropped.
B-40	Effects of Power Plant Entrainment on Plankton					Exclude. This issue has been dropped.
B-41	Impacts on Fisheries					Exclude. This issue has been dropped.
B-42	Socioeconomic Environmental Impacts					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-43	Value of Aerial Photographs for Site Evaluation					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-45	Need for Power- Energy Conservation					Exclude. This issue is covered in Item B-2
B-46	Costs of Alternatives in Environmental Design					Exclude. This issue has been

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						dropped.
B-47	Inservice Inspection of Supports – Classes 1, 2, 3, and MC Components					Exclude. This issue has been dropped.
B-48	BWR Control Rod Drive Mechanical Failures					Exclude. This issue was resolved with no new requirements.
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-50	Post-Operating Basis Earthquake Inspection					
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components					Exclude. This item is covered by Item A-40.
B-52	Fuel Assembly Seismic and LOCA Responses					Exclude. This item is covered by Item A-2.
B-53	Load Break Switch					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-54	Ice Condenser Containments					Exclude. This issue was resolved with no new requirements.
B-55	Improved Reliability of Target Rock Safety Relief Valves					Exclude. This issue was resolved with no new requirements.
B-56	Diesel Reliability					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-57	Station Blackout					Exclude. This issue is covered in Item A-44.
B-58	Passive Mechanical Failures					Exclude. This issue was resolved with no new requirements.
B-59	(N-1) Loop Operation in BWRs and PWRs					Exclude. This issue does not

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						apply to the HTGR design.
B-60	Loose Parts Monitoring Systems					Exclude. This issue was resolved with no new requirements.
B-61	Allowable ECCS Equipment Outage Periods					Exclude. This issue was resolved with no new requirements.
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions					Exclude. This issue has been dropped.
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary					Exclude. This issue was resolved with no new requirements.
B-64	Decommissioning of Reactors					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-65	Iodine Spiking					Exclude. This issue has been dropped.
B-66	Control Room Infiltration Measurements					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-67	Effluent and Process Monitoring Instrumentation					Exclude. Covered by TMI Action Plan Item III.D.2.1.
B-68	Pump Overspeed During LOCA					Exclude. This issue has been dropped.
B-69	ECCS Leakage Ex-Containment					Exclude. Covered by one of the TMI Action Plan Items
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps					Exclude. This issue was resolved by implementation of other regulatory requirements.
B-71	Incident Response					Exclude. Covered in one of the TMI Action Plan Items

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
B-72	Health Effects and Life-Shortening from Uranium and Coal Fuel Cycles					
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel					Exclude. Covered in Item C-12.
C-1	Assurance of Continuous Long- Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment					Exclude. This issue was resolved by implementation of other regulatory requirements.
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure					Exclude. This issue was resolved with no new requirements.
C-3	Insulation Usage within Containment					Exclude. Addressed in the resolution of Issue A-43.
C-4	Statistical Methods for ECCS Analysis					Exclude. This issue was resolved by implementation of other regulatory requirements.
C-5	Decay Heat Update					Exclude. This issue was resolved by implementation of other regulatory requirements.
C-6	LOCA Heat Sources					Exclude. This issue was resolved with no new requirements.
C-7	PWR System Piping					Exclude. This issue was resolved with no new requirements.
C-8	Main Steam Line Leakage Control Systems					Exclude. This issue was resolved with no new requirements.
C-9	RHR Heat Exchanger Tube Failures					Exclude. This issue has been dropped.
C-10	Effective Operation of Containment Sprays in a LOCA					Exclude. This issue was resolved by implementation of other regulatory requirements.

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
C-11	Assessment of Failure and Reliability of Pumps and Valves					Exclude. This issue was resolved with no new requirements.
C-12	Primary System Vibration Assessment					Exclude. This issue was resolved with no new requirements.
C-13	Non-Random Failures					Exclude. Covered by A-17.
C-14	Storm Surge Model for Coastal Sites					Exclude. This issue has been dropped.
C-15	NUREG Report for Liquid Tank Failure Analysis					Exclude. This issue has been dropped.
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection					Exclude. This issue has been dropped.
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes					Exclude. This issue was resolved by implementation of other regulatory requirements.
D-1	Advisability of a Seismic Scram					Exclude. This issue has been dropped.
D-2	Emergency Core Cooling System Capability for Future Plants					Exclude. This issue has been dropped.
D-3	Control Rod Drop Accident					Exclude. This issue was resolved with no new requirements.
Issue 1	Failures in Air- Monitoring, Air- Cleaning, and Ventilating Systems					Exclude. This issue has been dropped.
Issue 2	Failure of Protective Devices on Essential Equipment					Exclude. This issue has been dropped.
Issue 3	Set Point Drift in Instrumentation					Exclude. This issue was resolved with no new requirements.
Issue 4	End-of-Life and Maintenance Criteria					Exclude. This issue was resolved with no new

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						requirements.
Issue 5	Design Check and Audit of Balance-of- Plant Equipment					Exclude. Addressed under one of the TMI Action Plan Items
Issue 6	Separation of Control Rod from its Drive and BWR High Rod Worth Events					Exclude. This issue was resolved with no new requirements.
Issue 7	Failures Due to Flow- Induced Vibrations					Exclude. This issue has been dropped.
Issue 8	Inadvertent Actuation of Safety Injection in PWRs					Exclude. Applicable only to Pressurized Water Reactors.
Issue 9	Reevaluation of Reactor Coolant Pump Trip Criteria					Exclude. Applicable only to Pressurized Water Reactors.
Issue 10	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges					Exclude. This issue has been dropped.
Issue 11	Turbine Disc Cracking					Exclude. This issue is covered by Item A-37.
Issue 12	BWR Jet Pump Integrity					Exclude. This issue was resolved with no new requirements.
Issue 13	Small-Break LOCA from Extended Overheating of Pressurizer Heaters					Exclude. This issue has been dropped.
Issue 14	PWR Pipe Cracks					Exclude. Applicable only to Pressurized Water Reactors.
Issue 15	Radiation Effects on Reactor Vessel Supports					Exclude. This issue was resolved with no new requirements.
Issue 16	BWR Main Steam Isolation Valve Leakage Control Systems					Exclude. This issue was resolved with no new requirements.
Issue 17	Loss of Offsite Power Subsequent to a LOCA					Exclude. This issue has been dropped.
Issue 18	Steam-Line Break with Consequential Small LOCA					Exclude. Applicable only to

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						Pressurized Water Reactors.
Issue 19	Safety Implications of Non-safety Instrument and Control Power Supply Bus					Exclude. Covered under A-47.
Issue 20	Effects of Electromagnetic Pulse on Nuclear Power Plants					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 21	Vibration Qualification of Equipment					Exclude. This issue was resolved with no new requirements.
Issue 22	Inadvertent Boron Dilution Events					Exclude. This issue was resolved with no new requirements.
Issue 23	Reactor Coolant Pump Seal Failures					Exclude. This issue was resolved with no new requirements.
Issue 24	Automatic ECCS Switchover to Recirculation					Exclude. This issue was resolved with no new requirements.
Issue 25	Automatic Air Header Dump on BWR Scram System					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 26	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power					Exclude. This issue is covered under Issue 17.
Issue 27	Manual vs. Automated Actions					Exclude. Covered under Item B-17
Issue 28	Pressurized Thermal Shock					Exclude. Covered under Item A-49.
Issue 29	Bolting Degradation or Failure in Nuclear Power Plants					
Issue 30	Potential Generator Missiles – Generator Rotor Retaining Rings					Exclude. This issue has been dropped.
Issue 31	Natural Circulation Cooldown					Exclude. This issue was

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						resolved by implementation of other regulatory requirements.
Issue 32	Flow Blockage in Essential Equipment Caused by Corbicula					Exclude. Combined and evaluated with Issue 51.
Issue 33	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power					Exclude. Covered in Item A-47.
Issue 34	RCS Leak					Exclude. This issue has been dropped.
Issue 35	Degradation of Internal Appurtenances in LWRs					Exclude. This issue has a low priority ranking.
Issue 36	Loss of Service Water					Exclude. This issue was resolved with no new requirements.
Issue 37	Steam Generator Overfill and Combined Primary and Secondary Blowdown					Exclude. Addressed as part of Issue A-47.
Issue 38	Potential Recirculation System Failure as a Consequence of Ingestion of Containment Paint Flakes or Other Fine Debris					Exclude. This issue has been dropped.
Issue 39	Potential for Unacceptable Interaction between the CRD System and Non-Essential Control Air System					Exclude. This issue is addressed in Issue 25.
Issue 40	Safety Concerns Associated with Pipe Breaks in the BWR Scram System					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 41	BWR Scram Discharge Volume Systems					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 42	Combination Primary/Secondary System LOCA					Exclude. Covered by one of the TMI Action Plan Items
Issue 43	Reliability of Air Systems					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 44	Failure of Saltwater Cooling System					Exclude. Remaining generic

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
						issue covered by Issue 43.
Issue 45	Inoperability of Instrumentation Due to Extreme Cold Weather					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 46	Loss of 125 Volt DC Bus					Exclude. This issue is covered by Issue 76.
Issue 47	The Loss of Offsite Power					Exclude. This issue was resolved with no new requirements.
Issue 48	LCO for Class 1E Vital Instrument Buses in Operating Reactors					Exclude. Integrated into the resolution of Issue 128.
Issue 49	Interlocks and LCOs for Class 1E Tie- Breakers					Exclude. Integrated into the resolution of Issue 128.
Issue 50	Reactor Vessel Level Instrumentation in BWRs					Exclude. This issue was resolved with no new requirements.
Issue 51	Proposed Requirements for Improving the Reliability of Open Cycle Service Water System					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 52	SSW Flow Blockage by Blue Mussels					Exclude. This issue has been combined with Issue 51.
Issue 53	Consequences of a Postulated Flow Blockage Incident in a BWR					Exclude. This issue has been dropped.
Issue 54	Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980					Exclude. Objectives of issue are met by one of the TMI Action Plan Items
Issue 55	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand					Exclude. This issue has been dropped.
Issue 56	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event					Exclude. Addressed under item A-47.
Issue 57	Effects of Fire Protection System Actuation on Safety- Related Equipment					Exclude. This issue was resolved with no new

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						requirements.
Issue 58	Containment Flooding					Exclude. This issue has been dropped.
Issue 59	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable					
Issue 60	Lamellar Tearing of Reactor Systems Structural Supports					Exclude. This issue is addressed as a subtask of Item A-12.
Issue 61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments					Exclude. This issue was resolved with no new requirements.
Issue 62	Reactor Systems Bolting Applications					Exclude. This issue was integrated into the resolution of Issue 29.
Issue 63	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis					Exclude. This issue has been dropped.
Issue 64	Identification of Protection System Instrument Sensing Lines					Exclude. This issue was resolved with no new requirements.
Issue 65	Probability of Core- Melt Due to Component Cooling Water System Failures					Exclude. Incorporated into the resolution of Issue 23.
Issue 66	Steam Generator Requirements					Exclude. This issue was resolved with no new requirements.
Issue 67	Steam Generator Staff Actions					Header row.
Issue 67.2.1	Integrity of Steam Generator Tube Sleeves					Exclude. Addressed in the resolution of Issue 135.
Issue 67.3.1	Steam Generator Overfill					Exclude. Issue is covered by Item A-47.
Issue 67.3.2	Pressurized Thermal Shock					Exclude. Addressed in Item A-49.

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Issue 67.3.3	Improved Accident Monitoring					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 67.3.4	Reactor Vessel Inventory Measurement					Exclude. Addressed by implementation of one of the TMI Action Plan Items
Issue 67.4.1	RCP Trip					Exclude. Issue covered by TMI Action Plan Item II.K.3(5).
Issue 67.4.2	Control Room Design Review					Exclude. This issue is covered by one of the TMI Action Plan Items
Issue 67.4.3	Emergency Operating Procedures					Exclude. This issue is covered by one of the TMI Action Plan Items
Issue 67.5.1	Reassessment of Radiological Consequences					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 67.5.2	Reevaluation of SGTR Design Basis					Exclude. This issue will be covered under Issue 67.5.1.
Issue 67.5.3	Secondary System Isolation					Exclude. This issue has been dropped.
Issue 67.6.0	Organizational Responses					Exclude. This issue is covered by one of the TMI Action Plan Items
Issue 67.7.0	Improved Eddy Current Tests					Exclude. Covered under Issue 135
Issue 67.8.0	Denting Criteria					Exclude. Covered under Issue 135
Issue 67.9.0	Reactor Coolant System Pressure Control					Exclude. Covered under Issue A-45.
Issue 67.10.0	Supplemental Tube Inspections					
Issue 68	Postulated Loss of Auxiliary Feedwater System					Exclude. Integrated into the

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture					resolution of Issue 124.
Issue 69	Make-Up Nozzle Cracking in B&W Plants					Exclude. This issue was resolved with no new requirements.
Issue 70	PORV and Block Valve Reliability					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 71	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety					Exclude. This issue has been dropped.
Issue 72	Control Rod Drive Guide Tube Support Pin Failures					Exclude. This issue has been dropped.
Issue 73	Detached Thermal Sleeves					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 74	Reactor Coolant Activity Limits for Operating Reactors					Exclude. This issue has been dropped.
Issue 75	Generic Implications of ATWS Events at the Salem Nuclear Plant					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 76	Instrumentation and Control Power Interactions					Exclude. This issue has been dropped.
Issue 77	Flooding of Safety Equipment Compartments by Backflow Through Floor Drains					Exclude. Integrated into the resolution of Issue A-17.
Issue 78	Monitoring of Fatigue Transient Limits for Reactor Coolant System					Exclude. This issue was resolved with no new requirements.
Issue 79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown					Exclude. This issue was resolved with no new requirements.
Issue 80	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II					Exclude. This issue was resolved with no new

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
	Containments					requirements.
Issue 81	Impact of Locked Doors and Barriers on Plant and Personnel Safety					Exclude LOW priority issue.
Issue 82	Beyond Design Basis Accidents in Spent Fuel Pools					Exclude. This issue was resolved with no new requirements.
Issue 83	Control Room Habitability					Exclude. This issue was resolved with no new requirements.
Issue 84	CE PORVs					Exclude. This issue was resolved with no new requirements.
Issue 85	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments					Exclude. This issue has been dropped.
Issue 86	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 87	Failure of HPCI Steam Line without Isolation					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 88	Earthquakes and Emergency Planning					Exclude. This issue was resolved with no new requirements.
Issue 89	Stiff Pipe Clamps					Exclude. LOW priority issue.
Issue 90	Technical Specifications for Anticipatory Trips					Exclude. This issue has been dropped.
Issue 91	Main Crankshaft Failures in Transamerica Delaval Emergency Diesel Generators					Exclude. This issue was resolved with no new requirements.
Issue 92	Fuel Crumbling During LOCA					Exclude. This issue has been dropped.
Issue 93	Steam Binding of Auxiliary Feedwater Pumps					Exclude. This issue was

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						resolved by implementation of other regulatory requirements.
Issue 94	Additional Low Temperature Overpressure Protection for Light Water Reactors					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 95	Loss of Effective Volume for Containment Recirculation Spray					Exclude. This issue was resolved with no new requirements.
Issue 96	RHR Suction Valve Testing					Exclude. Integrated into resolution of Issue 105.
Issue 97	PWR Reactor Cavity Uncontrolled Exposures					Exclude. Covered under one of the TMI Action Plan Items
Issue 98	CRD Accumulator Check Valve Leakage					Exclude. This issue has been dropped.
Issue 99	RCS/RHR Suction Line Valve Interlock on PWRs					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 100	Once-Through Steam Generator Level					Exclude. This issue has been dropped.
Issue 101	BWR Water Level Redundancy					Exclude. This issue was resolved with no new requirements.
Issue 102	Human Error in Events Involving Wrong Unit or Wrong Train					Exclude. This issue was resolved with no new requirements.
Issue 103	Design for Probable Maximum Precipitation					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 104	Reduction of Boron Dilution Requirements					Exclude. This issue has been dropped.
Issue 105	Interfacing Systems LOCA at LWRs					Exclude. This issue was resolved with no new

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						requirements.
Issue 106	Piping and the Use of Highly Combustible Gases in Vital Areas					Exclude. This issue was resolved with no new requirements.
Issue 107	Main Transformer Failures					Exclude. This issue has been dropped.
Issue 108	BWR Suppression Pool Temperature Limits					Exclude. Applicable only to Boiling Water Reactors.
Issue 109	Reactor Vessel Closure Failure					Exclude. This issue has been dropped.
Issue 110	Equipment Protective Devices on Engineered Safety Features					Exclude. This issue has been dropped.
Issue 111	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments					
Issue 112	Westinghouse RPS Surveillance Frequencies and Out- of-Service Times					Exclude. This is applicable only to Westinghouse PWR designs.
Issue 113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers					Exclude. This issue was resolved with no new requirements.
Issue 114	Seismic-Induced Relay Chatter					Exclude. This issue is addressed in the resolution of Issue A-46.
Issue 115	Enhancement of the Reliability of Westinghouse Solid State Protection System					Exclude. This issue was resolved with no new requirements.
Issue 116	Accident Management					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 117	Allowable Time for Diverse Simultaneous Equipment Outages					Exclude. This issue has been dropped.
Issue 118	Tendon Anchor Head Failure					Exclude. This issue was

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						resolved by implementation of other regulatory requirements.
Issue 119	Piping Review Committee Recommendations					Header row.
Issue 119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 119.2	Piping Damping Values					Exclude. This issue has been dropped.
Issue 119.3	Decoupling the OBE from the SSE					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 119.4	BWR Piping Materials					Exclude. Applicable only to Boiling Water Reactors.
Issue 119.5	Leak Detection Requirements					
Issue 120	On-Line Testability of Protection Systems					Exclude. This issue was resolved with no new requirements.
Issue 121	Hydrogen Control for Large, Dry PWR Containments					Exclude. This issue was resolved with no new requirements.
Issue 122	Davis-Besse Loss of All Feedwater Event of June 9, 1985 – Short-Term Actions					Header row.
Issue 122.1	Potential Inability to Remove Reactor Decay Heat.					Header row.
Issue 122.1.a	Failure of Isolation Valves in Closed Position.					Exclude. Integrated into the resolution of Issue 124.
Issue 122.1.b	Recovery of Auxiliary Feedwater.					Exclude. Integrated into resolution of Issues 122.2 and 124.
Issue 122.1.c	Interruption of Auxiliary Feedwater Flow.					Exclude. Integrated into resolution of Issue 124.
Issue 122.2	Initiating Feed-and- Bleed					Exclude. This issue was resolved with no new

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
						requirements.
Issue 122.3	Physical Security System Constraints.					Exclude. This issue has been dropped.
Issue 123	Deficiencies in the Regulations Governing DBA and Failure Criterion Suggested by the Davis-Besse Incident of June 9, 1985					Exclude. This issue has been dropped.
Issue 124	Auxiliary Feedwater System Reliability					Exclude. Applicable only to Pressurized Water Reactors.
Issue 125	Davis-Besse Loss of All Feedwater Event of June 9, 1985 – Long-Term Actions					Header row.
Issue 125.I.1	Availability of the Shift Technical Advisor					Exclude. This issue has been dropped.
Issue 125.I.2	PORV Reliability					Header row.
Issue 125.I.2.a	Need for a Test Program to Establish Reliability of the PORV.					Exclude. This issue is covered in Issue 70.
Issue 125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness.					Exclude. This issue is covered in Issue 70.
Issue 125.I.2.c	Need for Additional Protection Against PORV Failure.					Exclude. This issue has been dropped.
Issue 125.I.2.d	Capability of the PORV to Support Feed-and-Bleed.					Exclude. This issue is covered in Issue A-45.
Issue 125.I.3	SPDS Availability					Exclude. This issue was resolved with no new requirements.
Issue 125.I.4	Plant-Specific Simulator.					Exclude. This issue has been dropped.
Issue 125.I.5	Safety Systems Tested in All Conditions Required by DBA.					Exclude. This issue has been dropped.
Issue 125.I.6	Valve Torque, Limit, and Bypass Switch Settings.					Exclude. This issue has been dropped.
Issue 125.I.7	Operator Training Adequacy.					
Issue 125.I.7.a	Recover Failed Equipment.					Exclude. This issue has been

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						dropped.
Issue 125.I.7.b	Realistic Hands-On Training.					Exclude. This issue has been dropped.
Issue 125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center.					Exclude. This issue has been dropped.
Issue 125.II.1	Need for Additional Actions on AFW Systems.					Header row.
Issue 125.II.1.a	Two-Train AFW Unavailability.					Exclude. This issue has been dropped.
Issue 125.II.1.b	Review Existing AFW Systems for Single Failure.					Exclude. This issue is covered by Issue 124.
Issue 125.II.1.c	NUREG-0737 Reliability Improvements.					Exclude. This issue has been dropped.
Issue 125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants.					Exclude. This issue has been dropped.
Issue 125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems.					Exclude. This issue has been dropped.
Issue 125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure					Exclude. This issue has been dropped.
Issue 125.II.4	Thermal Stress of OTSG Components					Exclude. This issue has been dropped.
Issue 125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components.					Exclude. This issue has been dropped.
Issue 125.II.6	Reexamine PRA Estimates of Core Damage Risk from Loss of All Feedwater.					Exclude. This issue has been dropped.
Issue 125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break.					Exclude. This issue was resolved with no new requirements.
Issue 125.II.8	Reassess Criteria for Feed-and-Bleed Initiation.					Exclude. This issue has been dropped.
Issue 125.II.9	Enhanced Feed-and- Bleed Capability.					Exclude. This issue has been dropped.
Issue 125.II.10	Hierarchy of Impromptu Operator Actions.					Exclude. This issue has been

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						dropped.
Issue 125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater.					Exclude. This issue has been dropped.
Issue 125.II.12	Adequacy of Training Regarding PORV Operation.					Exclude. This issue has been dropped.
Issue 125.II.13	Operator Job Aids.					Exclude. This issue has been dropped.
Issue 125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally.					Exclude. This issue has been dropped.
Issue 126	Reliability of PWR Main Steam Safety Valves					Exclude. Applicable only to Pressurized Water Reactors.
Issue 127	Maintenance and Testing of Manual Valves in Safety- Related Systems					Exclude. LOW priority issue.
Issue 128	Electrical Power Reliability					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 129	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling.					Exclude. This issue has been dropped.
Issue 130	Essential Service Water Pump Failures at Multiplant Sites.					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 131	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse- Designed Plants.					Exclude. Applicable only to Westinghouse PWR designs.
Issue 132	RHR System Inside Containment.					Exclude. This issue has been dropped.
Issue 133	Update Policy Statement on Nuclear Plant Staff Working Hours.					
Issue 134	Rule on Degree and Experience Requirement.					Exclude. This issue was resolved with no new requirements.
Issue 135	Steam Generator and Steam Line Overfill.					Exclude. This issue was

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						resolved with no new requirements.
Issue 136	Storage and Use of Large Quantities of Cryogenic Combustibles On Site.					Exclude. This issue was resolved with no new requirements.
Issue 137	Refueling Cavity Seal Failure.					Exclude. This issue has been dropped.
Issue 138	Deinerting of BWR Mark I and Mark II Containments During Power Operations upon Discovery of RCS Leakage or a Train of a Safety System Inoperable					Exclude. This issue has been dropped.
Issue 139	Thinning of Carbon Steel Piping in LWRs.					Exclude. Applicable only to Light Water Reactors.
Issue 140	Fission Product Removal Systems.					Exclude. This issue has been dropped.
Issue 141	Large Break LOCA with Consequential SGTR.					Exclude. This issue has been dropped.
Issue 142	Leakage through Electrical Isolators in Instrumentation Circuits					Exclude. This issue was resolved with no new requirements.
Issue 143	Availability of Chilled Water Systems and Room Cooling					Exclude. This issue was resolved with no new requirements.
Issue 144	Scram without a Turbine/Generator Trip					Exclude. This issue has been dropped.
Issue 145	Actions to Reduce Common Cause Failures					Exclude. This issue was resolved with no new requirements.
Issue 146	Support Flexibility of Equipment and Components.					
Issue 147	Fire-Induced Alternate Shutdown/Control Room Panel Interactions.					
Issue 148	Smoke Control and Manual Fire-Fighting Effectiveness.					

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Issue 149	Adequacy of Fire Barriers.					Exclude. This issue has been dropped.
Issue 150	Overpressurization of Containment Penetrations.					Exclude. This issue has been dropped.
Issue 151	Reliability of Anticipated Transient Without Scram Recirculation Pump Trip in BWRs.					Exclude. This issue was resolved with no new requirements.
Issue 152	Design Basis for Valves that Might be Subjected to Significant Blowdown Loads.					Exclude. This issue has been dropped.
Issue 153	Loss of Essential Service Water in LWRs					Exclude. This issue was resolved with no new requirements.
Issue 154	Adequacy of Emergency and Essential Lighting.					Exclude. This issue has been dropped.
Issue 155	Generic Concerns Arising from TMI-2 Cleanup.					Header row.
Issue 155.1	More Realistic Source Term Assumptions					Exclude. Regulatory Guide 1.183 addresses alternate source terms.
Issue 155.2	Establish Licensing Requirements for Non- Operating Facilities					
Issue 155.3	Improve Design Requirements for Nuclear Facilities					Exclude. This issue has been dropped.
Issue 155.4	Improve Criticality Calculations					Exclude. This issue has been dropped.
Issue 155.5	More Realistic Severe Accident Scenario					Exclude. This issue has been dropped.
Issue 155.6	Improve Decontamination Regulations					Exclude. This issue has been dropped.
Issue 155.7	Improve Decommissioning Regulations					Exclude. This issue has been dropped.
Issue 156	Systematic Evaluation Program	_	_	_	_	Header row.
Issue 156.1.1	Settlement of Foundations and Buried Equipment					Exclude. This issue has been

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						dropped.
Issue 156.1.2	Dam Integrity and Site Flooding					Exclude. This issue has been dropped.
Issue 156.1.3	Site Hydrology and Ability to Withstand Floods					Exclude. This issue has been dropped.
Issue 156.1.4	Industrial Hazards					Exclude. This issue has been dropped.
Issue 156.1.5	Tornado Missiles					Exclude. This issue has been dropped.
Issue 156.1.6	Turbine Missiles					Exclude. This issue has been dropped.
Issue 156.2.1	Severe Weather Effects on Structures					Exclude. This issue has been dropped.
Issue 156.2.2	Design Codes, Criteria, and Load Combinations					Exclude. This issue has been dropped.
Issue 156.2.3	Containment Design and Inspection					Exclude. This issue has been dropped.
Issue 156.2.4	Seismic Design of Structures, Systems, and Components					Exclude. This issue has been dropped.
Issue 156.3.1.1	Shutdown Systems					Exclude. This issue has been dropped.
Issue 156.3.1.2	Electrical Instrumentation and Controls					Exclude. This issue has been dropped.
Issue 156.3.2	Service and Cooling Water Systems					Exclude. This issue has been dropped.
Issue 156.3.3	Ventilation Systems					Exclude. This issue has been dropped.
Issue 156.3.4	Isolation of High and Low Pressure Systems					Exclude. This issue has been dropped.
Issue 156.3.5	Automatic ECCS Switchover					Exclude. Covered in the resolution of Issue 24.
Issue 156.3.6.1	Emergency AC Power					Exclude. This issue has been

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						dropped.
Issue 156.3.6.2	Emergency DC Power					Exclude. This issue has been dropped.
Issue 156.3.8	Shared Systems					Exclude. This issue has been dropped.
Issue 156.4.1	RPS and ESFS Isolation					Exclude. Covered by the resolution of Issue 142.
Issue 156.4.2	Testing of the RPS and ESFS					Exclude. Covered by the resolution of Issue 120.
Issue 156.6.1	Pipe Break Effects on Systems and Components					Exclude. This issue was resolved with no new requirements.
Issue 157	Containment Performance					Exclude. This issue was resolved with no new requirements.
Issue 158	Performance of Safety-Related Power- Operated Valves under Design Basis Conditions					Exclude. This issue was resolved with no new requirements.
Issue 159	Qualification of Safety-Related Pumps While Running on Minimum Flow					Exclude. This issue has been dropped.
Issue 160	Spurious Actuations of Instrumentation upon Restoration of Power					Exclude. This issue has been dropped.
Issue 161	Use of Non-Safety- Related Power Supplies in Safety- Related Circuits					Exclude. This issue has been dropped.
Issue 162	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shutdown					Exclude. This issue has been dropped.
Issue 163	Multiple Steam Generator Tube Leakage					**This is identified as a HIGH priority issue and should be evaluated**.
Issue 164	Neutron Fluence in Reactor Vessel					Exclude. This issue has been dropped.

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
Issue 165	Spring-Actuated Safety and Relief Valve Reliability					Exclude. This issue was resolved with no new requirements.
Issue 166	Adequacy of Fatigue Life of Metal Components					Exclude. This issue was resolved with no new requirements.
Issue 167	Hydrogen Storage Facility Separation					Exclude. LOW priority issue.
Issue 168	Environmental Qualification of Electrical Equipment					Exclude. This issue was resolved with no new requirements.
Issue 169	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure					Exclude. This issue has been dropped.
Issue 170	Fuel Damage Criteria for High Burnup Fuel					Exclude. This issue was resolved with no new requirements.
Issue 171	ESF Failure from LOOP Subsequent to a LOCA					Exclude. This issue was resolved with no new requirements.
Issue 172	Multiple System Responses Program					Exclude. Covered in the resolution of Issue 106.
Issue 173	Spent Fuel Storage Pool	-	-	-	-	Header row.
Issue 173.A	Operating Facilities					Exclude. This issue was resolved with no new requirements.
Issue 173.B	Permanently Shutdown Facilities					Exclude. This issue was resolved with no new requirements.
Issue 174	Fastener Gaging Practices	-	-	-	-	Header row.
Issue 174.A	SONGS Employees' Concern					Exclude. This issue was resolved with no new requirements.
Issue 174.B	Johnson Gage Company Concern					Exclude. This issue was

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)						
ID	**Title/Requirement**	**Applicable**	**Reg or Guidance**	**Additional Design Info**	**Additional Reg Needed**	**Basis/Comment**
						resolved with no new requirements.
Issue 175	Nuclear Power Plant Shift Staffing					Exclude. This issue was resolved with no new requirements.
Issue 176	Loss of Fill-Oil in Rosemount Transmitters					Exclude. This issue was resolved with no new requirements.
Issue 177	Vehicle Intrusion at TMI					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 178	Effect of Hurricane Andrew on Turkey Point					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 179	Core Performance.					Exclude. This issue was resolved with no new requirements.
Issue 180	Notice of Enforcement Discretion.					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 181	Fire Protection					Exclude. This issue was resolved with no new requirements.
Issue 182	General Electric Extended Power Uprate					Exclude. This issue was resolved with no new requirements.
Issue 183	Cycle-Specific Parameter Limits in Technical Specifications					Exclude. This issue was resolved by implementation of other regulatory requirements.
Issue 184	Endangered Species					
Issue 185	Control of Recriticality Following Small-Break LOCAs in PWRs					Exclude. This issue was resolved with no new

Table A1-20: Unresolved and Generic Safety Issues (NUREG-0933)

ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
						requirements.
Issue 186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants					
Issue 187	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification					Exclude. This issue has been dropped.
Issue 188	Steam Generator Tube Leaks or Ruptures, Concurrent with Containment Bypass from Main Steam Line or Feedwater Line Breaches					Exclude. This issue was resolved with no new requirements.
Issue 189	Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident					Exclude. This is applicable only to relatively low volume small pressure suppression containment designs.
Issue 190	Fatigue Evaluation of Metal Components for 60-Year Plant Life					Exclude. This issue is applicable only to BWRs.
Issue 191	Assessment of Debris Accumulation on PWR Sump Performance					**This is identified as a HIGH priority issue and should be evaluated**.
Issue 192	Secondary Containment Drawdown Time					Exclude. This issue has been dropped.
Issue 193	BWR ECCS Suction Concerns					Exclude. Applicable to BWRs only.
Issue 194	Implications of Updated Probabilistic Seismic Hazard Estimates					
Issue 195	Hydrogen Combustion in Foreign BWR Piping					Exclude. This issue has been dropped.
Issue 196	Boral Degradation					Exclude. This issue was resolved with no new requirements.
Issue 197	Iodine Spiking Phenomena					Exclude. This issue has been dropped.
Issue 198	Hydrogen Combustion in PWR Piping					Exclude. This issue has been dropped.

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ID	Title/Requirement	Applicable	Reg or Guidance	Additional Design Info	Additional Reg Needed	Basis/Comment
Issue 199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States					
Issue 200	Tin Whiskers					Exclude. This issue has been dropped.
Issue 201	Small-Break LOCA and Loss of Offsite Power Scenario					Exclude. This issue has been dropped.
Issue 202	Spent Fuel Pool Leakage Limits					Exclude. This issue has been dropped.
Issue 203	Potential Safety Issues with Cranes that Lift Spent Fuel Casks					Exclude. This issue has been dropped.
CH6.1	Graphite-Moderated Reactors					
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR					
CH6.1B	Structural Graphite Experiments					
CH6.2	Assessment					