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NGNP and Hydrogen Production Preconceptual Design Report

SPECIAL STUDY 20.6: NGNP LICENSING AND PERMITTING STUDY

Revision 0

APPROVALS

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ACRONYMS

Abbreviation or Acronym	Definition
ABB-CE	Asea Brown Boveri-Combustion Engineering
ACQR	Air Quality Control Region
ACRS	Advisory Committee on Reactor Safeguards
AD	Advanced Design
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANPR	Advance Notice of Proposed Rulemaking (from NRC)
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
AVR	Arbeitsgemeinschaft Versuchsreaktor (German for Jointly-operated Prototype Reactor)
BDBE	Beyond Design Basis Event
BOD	Basis of Design
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Uranium Reactor
CFR	Code of Federal Regulations
CP	Construction Permit
COL	Combined License
CSWTF	Central Sanitary Wastewater Treatment Facility
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Design Certification
DCD	Design Control Document
DID	Defense-In-Depth
DOE	Department of Energy (US)
EIS	Environmental Impact Statement
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
EPA	Environmental Protection Agency
EPOL	Experimental Plate-out Loop

Abbreviation or Acronym	Definition
EPP	Environmental Permitting Plan
EPSS	Environmental Permitting Status Summary
Eskom	Eskom Holdings Limited – RSA
ESP	Early Site Permit
FFTF	Fast Flux Test Facility
FHSS	Fuel Handling and Storage System
FMEA	Failure Modes and Effects Analysis
FSER	Final Safety Evaluation Report (issued by NRC)
FSV	Fort St. Vrain
GSA	Gas Storage Area
HAZOP	Hazard and Operability (study)
HPB	Helium Pressure Boundary
HTF	Helium Test Facility
HTGR	High-Temperature Gas-Cooled Reactor
HTR	High-Temperature Reactor
HTTR	High-Temperature Test Reactor
HVAC	Heating, Ventilation and Air-Conditioning
IAEA	International Atomic Energy Agency
IDEQ	Idaho Department of Environmental Quality
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory
IET	Integral Effects Test
IPOF	Isopiestic Plate-out Facility
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
LBE	Licensing Basis Event
LBT	License by Test
LRB	Licensing Review Basis
LWA	Limited Work Authorization
LWR	Light Water Reactor
MDEP	Multinational Design Evaluation Program
MHTGR	Modular High Temperature Gas Reactor
MPS	Main Power System
ND	No Discharge

Abbreviation or Acronym	Definition
NEPA	National Environmental Policy Act
NESHAP	National Emission Standards for Hazardous Air Pollutants
NGNP	Next Generation Nuclear Plant
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission (USA)
NUREG	Nuclear Regulations (from NRC)
OL	Operating License
OSHA	Occupational Safety and Health Administration
PBMR	Pebble Bed Modular Reactor
PBMR (Pty) Ltd	Pebble Bed Modular Reactor Company (Pty) Ltd (Republic of South Africa)
PHA	Process Hazards Analysis
PIUS	Process Inherent Ultimate Safety
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PRISM	Power Reactor Innovative Small Module
PSER	Pre-application Safety Evaluation Report
PSM	Process Safety Management
PSS	Process Steam System
R&D	Research and Development
RCRA	Resource Conservation and Recovery Act
RIRIP	Risk-Informed Regulation Implementation Plan
RMP	Risk Management Plan
SDA	Standard Design Approval
SECY	Letter to the Secretary of the U.S. Nuclear Regulatory Commission
SER	Safety Evaluation Report
SET	Separate Effects Test
SPCC	Spill Prevention Control and Countermeasures
SSC	Structures, Systems and Components
SWPPP	Storm Water Management Pollution Protection Plan
TBD	To Be Determined
TEDE	Total Effective Dose Equivalent
TLRC	Top Level Regulatory Criteria
TRISO	TRIple-coated ISOtropic

Abbreviation or Acronym	Definition
USA	United States of America
UST	Underground Storage Tank
V&V	Verification and Validation

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APPENDICES

20.6 NGNP LICENSING AND PERMITTING STUDY

SUMMARY AND CONCLUSIONS

The purpose of this special study is to address specific tasks related to NGNP licensing, as given in the Statement of Work (Ref. 20.6-1). Pertinent NRC regulations and the corresponding industry experience were reviewed and recommendations have been made. The key recommendations from this study are:

- Build on PBMR (Pty) Ltd–NRC pre-application interactions, including risk-informed, performance-based methods,
- Adopt an NGNP licensing strategy, based on Part 52, to obtain an Early Site Permit with an embedded Limited Work Authorization followed by a Combined License. Maintain a Part 50 fallback strategy for a two-step license pending the success of the pre-application interactions.
- Use License-By-Test as warranted by expected benefits to achieve timely full-power operation of the NGNP, and Design Certification for the Commercial plant.
- Establish and demonstrate the licensing requirements commensurate with the chosen hydrogen production design(s) reflecting separation distance and facility interactions stemming from such design(s).

EPA, State and local permitting are not expected to present any significant licensing impediment for the NGNP.

It is also recommended that (1) NRC progress on licensing rulemakings (i.e., Part 50, Part 52) be followed and results incorporated into the NGNP Licensing Strategy during the conduct of Activity 15, (2) licensing research and development needs specific to the NGNP safety analysis evaluation models be identified as the basic design is developed, and (3) the recently received "site selection" report for the New Production Reactor site at INL be reviewed to identify any limiting environmental conditions.

The above recommendations and the actions identified in ACTIONS RECOMMENDED FOR OTHER ACTIVITIES are inputs to Activity 15, whose major objective is the development and recommendation of an overall licensing strategy for the NGNP, including approximate cost estimates and schedule impacts. This approach will support development and receipt of a Design Certification for follow-on NGNP commercial plants due to the valuable precedents established in the NGNP licensing.

INTRODUCTION

The purpose of this special study is to address the following specific tasks related to NGNP licensing, as given in the Statement of Work (Ref. 20.6-1):

- Task 20.6.1 - Licensing under Part 50 vs. Part 52,
- Task 20.6.2 - Feasibility of Mixed Licensing Approach (Part 52 ESP and Part 50 CP/OL),
- Task 20.6.3 - Feasibility of Using New Advanced Reactor Licensing Framework (to become Part 53),
- Task 20.6.4 - Practicality of "License by Test" Licensing Method,
- Task 20.6.5 - Licensing of an Integrated Nuclear Power/Hydrogen Plant,
- Task 20.6.6 - Method for Integration of Probabilistic Risk Assessment (PRA) Techniques During Design Phase, and
- Task 20.6.7 - EPA/State Permits for Integrated Nuclear Power / Hydrogen Plant.

This study assumes that a reliable and high-quality supply of fuel spheres will be available for the NGNP at the time it is needed. However, this study does not address the regulations related to the fuel production facility. Similarly, this study assumes that spent fuel will be stored on-site and does not include transportation to an offsite location for either storage or reprocessing. These are topics for future study.

Furthermore, this study does not directly address compliance with DOE regulations and orders, but since the scope of DOE regulations (e.g., 10 CFR 830, 10 CFR 835) overlaps that of NRC regulations, the DOE regulations should be reviewed to identify any compliance issues relative to NRC regulations.

The results herein draw from the experience of the members of the Westinghouse-led Team. This Team has substantial experience with operating LWRs, design certification of ALWRs, past HTGR licensing activities and past and ongoing PBMR (Pty) Ltd licensing activities.

The following sections summarize the results of work performed for each of the tasks identified in the Statement of Work. Presentation slides prepared in the course of this special study are presented in Appendices 20.6.B and 20.6.C.

20.6.1 LICENSING UNDER PART 50 VS. PART 52

The purpose of this section is to summarize and compare the Part 50 and Part 52 nuclear plant licensing processes. Both the current licensing process under 10 CFR Part 50 and the newer process under 10 CFR Part 52 are relevant to NGNP licensing. While these regulations were developed by the NRC based on principally Light Water Reactor (LWR) experience, other reactor types such as High-Temperature Gas Reactors (HTGRs) are not excluded. In fact, NRC has licensed HTGRS (e.g., Peach Bottom-1 and Fort St. Vrain) under much earlier versions of Part 50. NRC has also conducted substantial review for the Modular HTGR which demonstrates how a HTGR would be licensed with more contemporary editions of Part 50 (see NUREG-1338 for a summary, Reference 20.6.1-1). This is significant since the technical provisions of Part 50 underpin the requirements of Part 52 as well.

The following sections summarize both sets of regulations and provide a comparison, including pros and cons based on past experience and current expectations for the NGNP. This work reflects the substantive licensing experience gained in support of the PBMR (Pty) Ltd design certification pre-application interactions with the NRC and the experience of Exelon Generation Company during their COL pre-application interactions with NRC.

Types of Licenses

The regulations in 10 CFR Part 50 are promulgated by the Nuclear Regulatory Commission pursuant to the Atomic Energy Act of 1954, as amended, and by Title II of the Energy Reorganization Act of 1974, to provide for the licensing of production and utilization facilities. The NRC is authorized to issue licenses for two classes of facilities: (1) medical therapy, research, and development and (2) industrial or commercial. The NRC regulations describing these licenses are quoted below:

> *'§ 50.20 Two classes of licenses.*
> *Licenses will be issued to named persons applying to the Commission therefore, and will be either class 104 or class 103.*
>
> *§ 50.21 Class 104 licenses; for medical therapy and research and development facilities.*
>
> *A class 104 license will be issued, to an applicant who qualifies, for any one or more of the following: to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use.*
>
> *(a) A utilization facility for use in medical therapy; or*
>
> *(b)(1) A production or utilization facility the construction or operation of which was licensed pursuant to subsection 104b of the Act prior to December 19, 1970;*

(2) A production or utilization facility for industrial or commercial purposes constructed or operated under an arrangement with the Administration entered into under the Cooperative Power Reactor Demonstration Program, except as otherwise specifically required by applicable law; and

(3) A production or utilization facility for industrial or commercial purposes, when specifically authorized by law.

(c) A production or utilization facility, which is useful in the conduct of research and development activities of the types specified in section 31 of the Act, and which is not a facility of the type specified in paragraph (b) of this section or in § 50.22.

[21 FR 355, Jan. 19, 1956, as amended at 31 FR 15145 Dec. 2, 1966; 35 FR 19659, Dec. 29, 1970; 38 FR 11446, May 8, 1973; 43 FR 6924, Feb. 17, 1978]

§ 50.22 Class 103 licenses; for commercial and industrial facilities.

A class 103 license will be issued, to an applicant who qualifies, for any one or more of the following: To transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use a production or utilization facility for industrial or commercial purposes; Provided, however, That in the case of a production or utilization facility which is useful in the conduct of research and development activities of the types specified in section 31 of the Act, such facility is deemed to be for industrial or commercial purposes if the facility is to be used so that more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training.

[38 FR 11446, May 8, 1973, as amended at 43 FR 6924, Feb. 17, 1978]'

10 CFR Part 52 includes the Part 50 regulations by reference, but provides for the issuance of a single combined license (COL, a combined construction and operating license). In addition, new optional concepts of an early site permit (ESP) and plant design certification (DC) are introduced. The intent of the ESP and DC is to provide complete substantive reviews of safety issues and to receive NRC approvals before the initiation of plant construction. Although Part 52 can generally be viewed as a one-step process because NRC issues only one license, there is significant flexibility in the sequence of applying for an ESP, a DC, and/or a COL.

Implications for the NGNP

Subsection 104 (Medical Therapy and Research & Development licenses) of the AEA-1954 allows the NRC to issue a license for a commercial utilization facility using the minimum amount of regulation. The following differences in language are noted:

Subsection 103.b (Commercial)	Subsection 104.c (R&D/Medical)
The Commission shall issue such licenses…….to persons……who agree to make available to the Commission such technical information and data concerning activities under such licenses as the Commission may determine necessary to promote the common defense and security and to protect the health and safety of the public….	……The Commission is directed to impose only such minimum amount of regulation of the licensee as the Commission finds will permit the Commission to fulfill its obligations under this Act to promote the common defense and security and to protect the health and safety of the public and permit the conduct of widespread and diverse research and development.

However, the above Section 50.22 of NRC regulations indicates that a utilization facility, such as the NGNP, that uses more than 50% of its output for sale or commercial distribution would be licensed not as a research facility (with minimal regulation) but as a commercial facility under Section 103.

Moreover, since the NGNP is to provide a substantial basis for follow-on NGNP commercial plants and since it is judged that a Subsection 103 commercial facility application would provide a more-applicable precedent for the NGNP commercial plant than would a Subsection 104 research and development facility application, it is appropriate to apply for a Subsection 103 commercial facility license for the NGNP. In addition, applying for a Subsection 103 license does not preclude application of the "license by test" concept to individual components or systems on a case-by-case basis (further discussed in Section 20.6.4).

20.6.1.1 LICENSING UNDER 10 CFR PART 50

20.6.1.1.1 Background

Today's commercial operating power plants were licensed by the NRC using the "two-step" process of 10 CFR Part 50. Figure 20.6.1-1 highlights the initial step of obtaining a Construction Permit (CP) followed by the issuance of an Operating License (OL). As indicated in Figure 20.6.1-1, the CP review provides a second opportunity for design review by NRC staff and a second review by the public. While experience has shown that design or operational changes may be merited after the start of construction under a Part 50 license due to design evolution with concurrent engineering and design or adverse industry experience (e.g., TMI-related changes), the fact remains that such changes after construction start have proven to be very expensive in terms of capital cost and schedule stretch-out. Nonetheless, Part 50 remains the basis for today's commercial operating plants. Part 50 also remains an option for licensing of new plants.

Obtaining a CP and OL through the 10 CFR Part 50 process may be appropriate for the NGNP, if (1) not enough design information is available to support an acceptable application under Part 52 or (2) future work on the NGNP project schedule shows that time is of the essence to begin permanent construction.

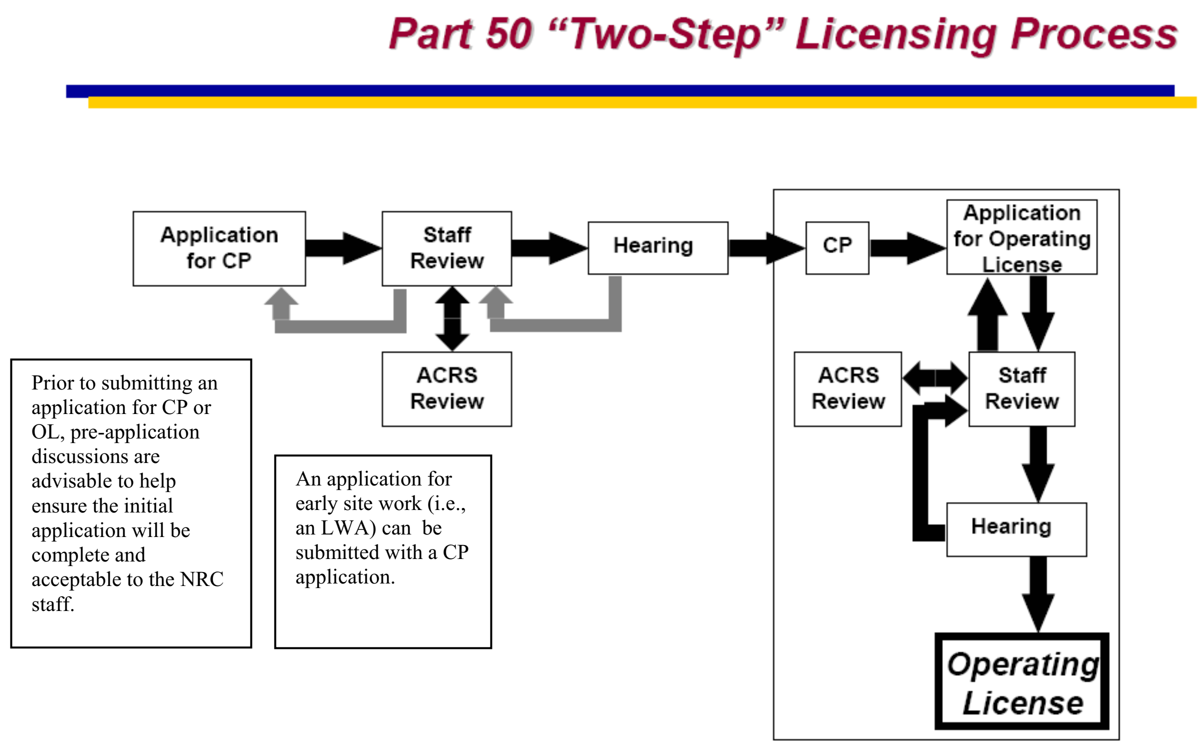

Figure 20.6.1-1: Part 50 "Two-Step" Licensing Process

20.6.1.1.2 Licensing Processes

The Part 50 license process and its requirements are provided in 10 CFR 50; however, many other regulations must be addressed as well. While there may be some flexibility as to whether detailed information must be provided during the CP stage or during the CP review, the following list of the more significant regulations indicates the broad extent of regulations that must be addressed by the applicant.

- 10 CFR 2 — Rules of Practice for Domestic Licensing Proceedings and Orders
- 10 CFR 20 — Standards for Protection Against Radiation
- 10 CFR 50 — Domestic Licensing of Production and Utilization Facilities
- 10 CFR 51 — Environmental Protection Regulations for Domestic Licensing
- 10 CFR 73 — Physical Protection of Plants and Materials
- 10 CFR 74 — Material Control and Accounting of Special Nuclear Material

- 10 CFR 75 Safeguards on Nuclear Material
- 10 CFR 95 Security Clearance and Safeguarding of Restricted Data
- 10 CFR 100 Reactor Site Criteria
- 10 CFR 140 Financial Protection
- 10 CFR 171 Annual Fees
- 29 CFR 1910 Subpart H – Occupational Safety and Health
- 40 CFR 50 - 99 Subchapter C, Air Programs
- 40 CFR 100 – 149 Subchapter D, Water Programs
- 40 CFR 190-197 Subpart F, Radiation Protection Programs
- 40 CFR 239-299 Subchapter I, Solid Wastes
- 40 CFR 400-471 Subchapter N, Effluent Guidelines and Standards
- 40 CFR 1500-1518 Council on Environmental Quality (Environmental Impact Statements)
- 16 U.S.C. 792 Federal Power Act

In addition to the above, a license applicant will have to address NRC policy statements that cover the manner in which regulatory requirements are implemented. The following list identifies existing policy statements which will likely be implemented for the NGNP licensing effort. In addition, for a new reactor licensing effort such as the NGNP a number of new policy statements may be developed to address new regulatory issues such as those listed in Sections 20.6.1.3 and 20.6.2.3.

- Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (August 1985)
- Regulation of Advanced Nuclear Power Plants; Statement of Policy (July 1986)
- Policy Statement on Safety Goals (August 1986)
- Nuclear Power Plant Standardization; Policy Statement (September 1987)
- Policy Statement on Use of Probabilistic Risk Assessment Methods in Nuclear Activities (August 1995)

As already indicated above, the 10 CFR Part 50 licensing process first requires a CP, then an OL. If it is desired to perform early site preparation work, the CP can be preceded by a Limited Work Authorization. Details of each step are provided in the following sub-sections.

20.6.1.1.2.1 Limited Work Authorization

Under Part 50, Section 50.10(e), the NRC can authorize an applicant to begin certain site preparation work, but excluding work on structures, systems or components which would prevent or mitigate the consequences of postulated accidents. An Environmental Impact Statement (EIS) that meets 10 CFR 51 must be issued prior to authorizing early site work.

Based partly on experience with the industry over the past several years, including initial Early Site Permit (ESP) reviews, proposed revisions to Sections 2.101(9), 50.10, 51.49, 51.76, and 52.25 were published (Federal Register, October 17, 2006) to more clearly define what work could be performed as part of a Limited Work Authorization (LWA) and the conditions for such applications, including EISs. The basic intent of the proposed revisions is (1) to remove the prohibition against "commencement of construction" prior to NRC approval and instead (2) require an LWA for only those activities which have a reasonable nexus to radiological health and safety and/or common defense and security. The net effect of the revisions is to clearly define the activities for which an LWA is requested and decrease the number of early site activities requiring NRC approval. Complete applications for an LWA can be submitted by applicants for a CP, an ESP, or a COL or the holder of an ESP. Applicants for an ESP that desired authority to perform early site work could so request and the permission, if forthcoming, would be granted as part of the ESP itself (a separate LWA would not be issued). Applicants for a CP, an ESP, or a COL (i.e., excluding the holders of an ESP) have the option of submitting the LWA application in two parts [presumably based on the availability of site information and the project schedule], with not more than 12 months between submittal of part one and part two.

The proposed Section 50.10 defines the term "construction" for any structure, system or component of a facility required by the regulations to be included in a site safety analysis report or a final safety analysis report to include:

- Excavation
- Sub-surface preparation including driving of piles
- Installation of foundations and placement of concrete
- In-place fabrication or erection
- Testing

The definition of "construction" excludes and, therefore does not require an LWA, for the following items:

- Changes for temporary use of land for public recreation
- Site exploration, including borings and other pre-construction monitoring
- Site preparation, including items such as clearing, grading, drainage installation, and temporary roads
- Installation of fencing and other access control measures
- Construction of temporary support buildings
- Construction of permanent service facilities
- Procurement or manufacture of components, or manufacture of a nuclear reactor under a manufacturing license
- Construction of buildings to be used for activities other than operation of a facility

20.6.1.1.2.2 Construction Permit

Summarizing from 10 CFR Part 50, a CP for the construction of a production or utilization facility will be issued prior to the issuance of an OL if the application is otherwise acceptable, and will be converted upon due completion of the facility and Commission action into an OL as provided in 10 CFR Part 50.

The Commission may issue a CP if the Commission finds that (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public; (2) such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the standard safety analysis report; (3) safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and that (4) on the basis of the foregoing, there is reasonable assurance that, (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration site criteria, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

A CP will constitute an authorization to the applicant to proceed with construction, but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit. The applicant, at his option, may request such approvals in the CP or, from time to time, by amendment of his CP. The Commission may, in its discretion, incorporate in any CP provisions requiring the applicant to furnish periodic reports of the progress and results of research and development programs designed to resolve safety questions.

Any CP will be subject to the limitation that a license authorizing operation of the facility will not be issued by the Commission until (1) the applicant has submitted to the Commission, by amendment to the application, the complete final safety analysis report, portions of which may be submitted and evaluated from time to time, and (2) the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the requirements of the license and the regulations in Chapter 1 of 10 CFR.

20.6.1.1.2.3 Operating License

In determining that a 10 CFR Part 50 OL will be issued to an applicant, the Commission will be guided partly by the following considerations (from 10 CFR 50.57):

- " (1) Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
- (2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
- (3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter; and
- (4) The applicant is technically and financially qualified to engage in the activities authorized by the operating license in accordance with the regulations in this chapter. However, no finding of financial qualification is necessary for an electric utility applicant for an operating license for a utilization facility of the type described in § 50.21(b) or § 50.22.
- (5) The applicable provisions of Part 140 of this chapter have been satisfied; and
- (6) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public."

20.6.1.1.3 Considerations for Licensing the NGNP

The 10 CFR Part 50 process is well proven and broadly understood within the commercial industry. It has advantages and disadvantages relative to NGNP licensing, which are highlighted in Table 20.6.1-1.

Table 20.6.1-1: 10 CFR 50 Licensing Approach

10 CFR 50	PROS	CONS
	Well proven	Requires NRC review and public hearing for both CP & OL; Therefore, susceptible to delays from additional regulatory review and public intervention.
	Understood by industry and NRC	Susceptible to design changes even after substantial construction has been completed.
	Achieves the NRC obligation to ensure safe performance.	Can be excessively costly in terms of regulatory review time, design changes, and construction schedule stretch-out.
	Can start regulatory review with a lesser degree of design detail – which can allow (1) design and analysis to be developed with NRC input and (2) construction to be initiated earlier than might be possible under Part 52.	NRC and public are not bound by the approval of a particular plant when reviewing future plants of the same design.
	Review for single plant may be easier and less costly (relative to review for a design to be certified).	Misses opportunity to develop and demonstrate the feasibility of a “one- step” COL – which is needed for follow-on commercial plants.
		Does not address the questions of how to define LBEs for non-LWRs or for process heat or cogeneration type of plants.
		Depends on applying deterministic requirements from LWR power plants.

20.6.1.2 LICENSING UNDER 10 CFR PART 52

20.6.1.2.1 Background

In 1989 the NRC completed work on a final rule, 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants”. By publishing this rule, the NRC sought to facilitate increased standardization of reactor designs (and commensurate improvements in safety) as well as permit an applicant a more streamlined and efficient process for seeking approval of a reactor design relative to the Part 50 licensing process. In a March 13, 2006 Federal Register Notice, the NRC proposed changes that would clarify the applicability of various requirements to each of the Part 52 licensing processes (i.e., early site permit, standard design approval, standard design certification, combined license, and manufacturing license). Public comments were received and in October 2006 the NRC staff recommended to the Commission that a final rule be published. Additionally, in an October 17, 2006 Federal Register Notice, supplemental proposed revisions to Parts 50 and 52 were announced, the purpose of which is to make the licensing process even more efficient, especially in regards to the issuance of a Limited Work Authorization for early site preparation work. Part 52 allows an applicant to obtain pre-approval of an essentially complete plant design prior to committing to the construction of a plant. This rule reduces licensing uncertainty by allowing most siting and design issues to be resolved up front.

20.6.1.2.2 Licensing Processes

The three subparts of Part 52 which are the most relevant for this special study are:

- Early Site Permit (with or without a Limited Work Authorization)
- Design Certification
- Combined License

Part 52 has been called a "one-step" licensing process because the license is issued for both construction and operation by NRC following completion of the COL hearing (subject to satisfactory completion of the Inspection, Test, Analysis and Acceptance Criteria (ITAAC)). Application of the Early Site Permit and Design Certification subparts of Part 52 is optional and DC could either precede or follow the COL. Figure 20.6.1-2 illustrates a sequence of an ESP and COL, followed by DC. Prior to submitting an application for an ESP, a DC or a COL, pre-application discussions are advisable to help ensure the initial application will be complete and acceptable to the NRC staff and that the NRC staff has adequate time to prepare for any application associated with a first-of-a-kind reactor. Moreover, an application for early site work (i.e., an LWA) can be submitted in conjunction with either an ESP or a COL.

20.6.1.2.2.1 Limited Work Authorization

The proposed Limited Work Authorization process summarized in Section 20.6.1.1.2.1 may also be used in conjunction with Part 52, as specifically stated in Section 52.25 of the forthcoming revision to Part 52 (see the October 17, 2006 proposed rulemaking).

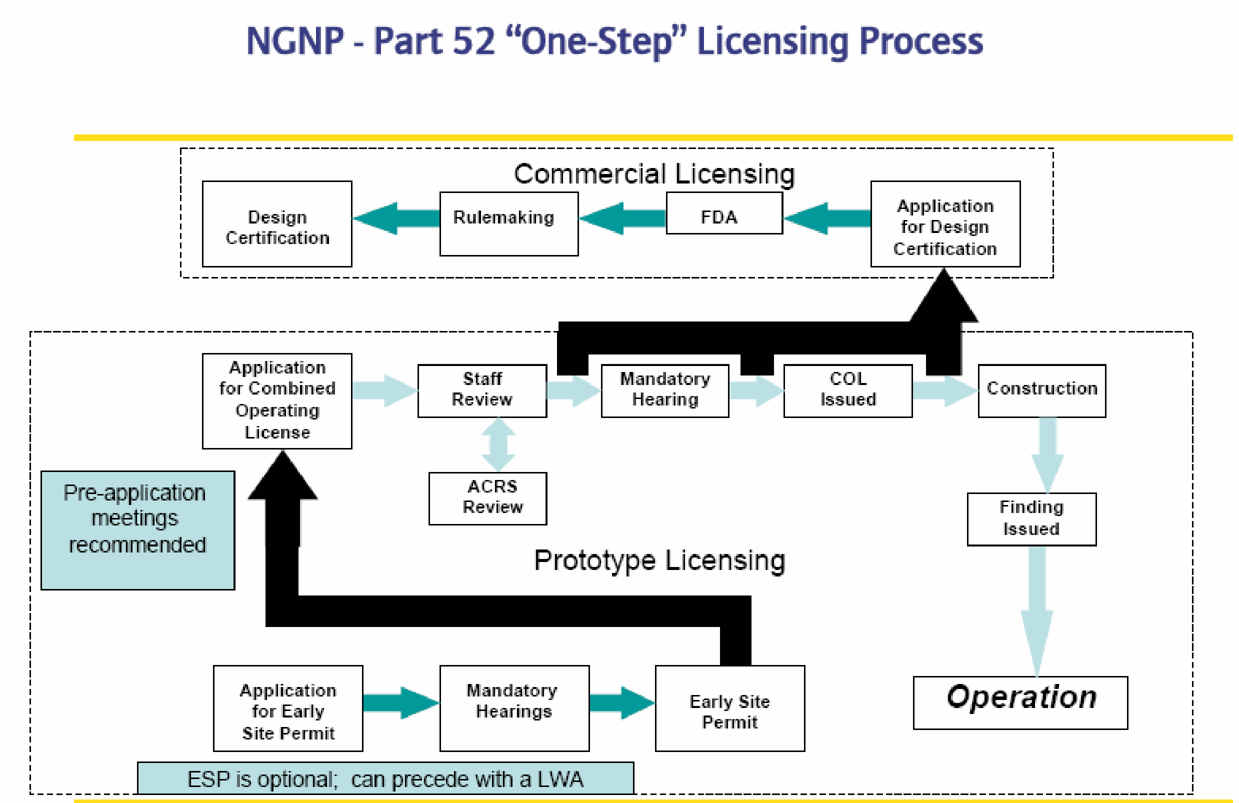

Figure 20.6.1-2: Part 52 "One-Step" Licensing Process

20.6.1.2.2.2 Early Site Permit

10 CFR 52 Subpart A sets out the requirements and procedures applicable to NRC issuance of early site permits (ESPs) for approval of a site or sites for one or more nuclear power facilities separate from the filing of an application for a CP or COL.

The intent of an ESP is to allow a licensee to select and license a site prior to filing an application for a CP (under Part 50) or a COL (under Part 52). The ESP application must include a description and safety assessment of the site containing the following information:

- Site description, site boundaries and characterization information including site boundaries
- Seismic, meteorological, hydrological, hydro-geological, and geological characteristics of the proposed site, and aquatic and terrestrial ecosystems
- Number, type, and thermal power level of facilities at the site
- Transmission corridors

- Site safety assessment
- Nearby industrial facilities
- Projected population profiles
- Thermal and radiological effluents and environmental impact report
- Emergency planning information
- Site restoration (or "site redress") plan. This plan is needed if early site work is conducted and the ESP expires before being referenced in a CP or COL.

An ESP is subject to the usual NRC procedural requirements. This includes an acceptance review (docketing), public hearings pursuant to NEPA requirements, and referral to the Advisory Committee on Reactor Safeguards (ACRS). As further noted in Section 20.6.1.2.2, during any subsequent COL review the NRC will assess the extent of the ESP NEPA coverage and determine whether significant changes have occurred since the ESP was issued. Fees for NRC review time apply in the normal manner. The duration of an ESP is at least 10 years but not more than 20 years from the date of issuance. Even so, an ESP does not expire if it has been referenced by a CP application, an OL application based on the CP, or a COL application. If the ESP is not referenced in such applications and the expiration date is approaching, a renewal application may be submitted.

The ESP describes the characteristics of the specific site being considered. If a DC exists and is to be referenced in the COL for that specific site, it will be necessary to review the ESP site characteristics to determine whether they fall within the site envelope described in the DC. If exceptions to the DC site envelope are required, the DC will have to be revised or the exception will have to be identified in the COL. In either case, the NRC staff will perform their technical review of the exception and the public will have the opportunity to comment thereon. The NRC experience with ESP reviews for three existing LWR sites (expected total review time of approximately 48 months) has materially assisted the NRC and industry establishing what is needed to complete the process. Having substantively completed these reviews, it is expected that future ESP application reviews would require less time, especially if it is expected that there will be limited public objection (e.g., the INL site). Hence it is estimated that the NGNP ESP review would be processed in approximately 30-36 months.

An ESP has the major advantage of reducing initial project risk related to site approval. An ESP also provides an opportunity for partial site preparation through an embedded LWA (see Section 20.6.1.1.2.1). This latter feature reduces schedule risk and potentially lowers total project cost. Considering the work already performed for the expected INL site for the NGNP and the assumption that there will be little or no serious intervener objections to construction of a reactor at this site, site approval risk is judged to be low. Use of an ESP for the NGNP will facilitate later licensing reviews by clearing up site issues before the subsequent licensing reviews are started. Another benefit of an ESP is to establish a foundation of requirements and precedents useful to subsequent NGNP Commercial plant ESP applications.

20.6.1.2.2.3 Design Certification

A Standard Design Approval (SDA) – sometimes called a final design approval - under 10 CFR 52, Appendix O is a prerequisite for certification of a standard design. The NRC estimates that it would take approximately 30 - 48 months for the technical review, depending on items such as the uniqueness of the design and the need for testing, and an additional 12 months for the rulemaking (42 - 60 months total).

The DC rule for a specific design (issued as an appendix to Part 52) references a Design Control Document (DCD) which is prepared and maintained by the applicant. A COL applicant who references a DC Rule would be required to incorporate the DCD information into its own plant-specific application. Once issued, a DC is valid for 15 years. Changes to the DC rule would be subject to formal rulemaking, while changes to (or departures from) the DCD can be made by the DC holder in accordance with change control processes spelled out in the DC rule.

The procedure for certifying a standard design is performed under Subpart B of 10 CFR Part 52 and is carried out in two stages: technical and administrative (see Figure 20.6.1-3). The technical review stage starts when the application is filed. Initially an acceptance review of the application is performed to ensure the application is of the requisite completeness and quality that the NRC needs to be able to perform its review. This stage continues with detailed technical reviews by the NRC staff and the ACRS and ends with the issuance of a Final Safety Evaluation Report (FSER) that discusses the staff's conclusions related to the acceptability of the design. The FSER provides the bases for issuance of the Standard Design Approval (SDA).

The administrative review stage begins with the publication of a Federal Register notice that initiates rulemaking, in accordance with 10 CFR 52.51, "Administrative Review of Applications," and includes a proposed DC rule. The rulemaking culminates either with the denial of the application or the issuance of a DC rule.

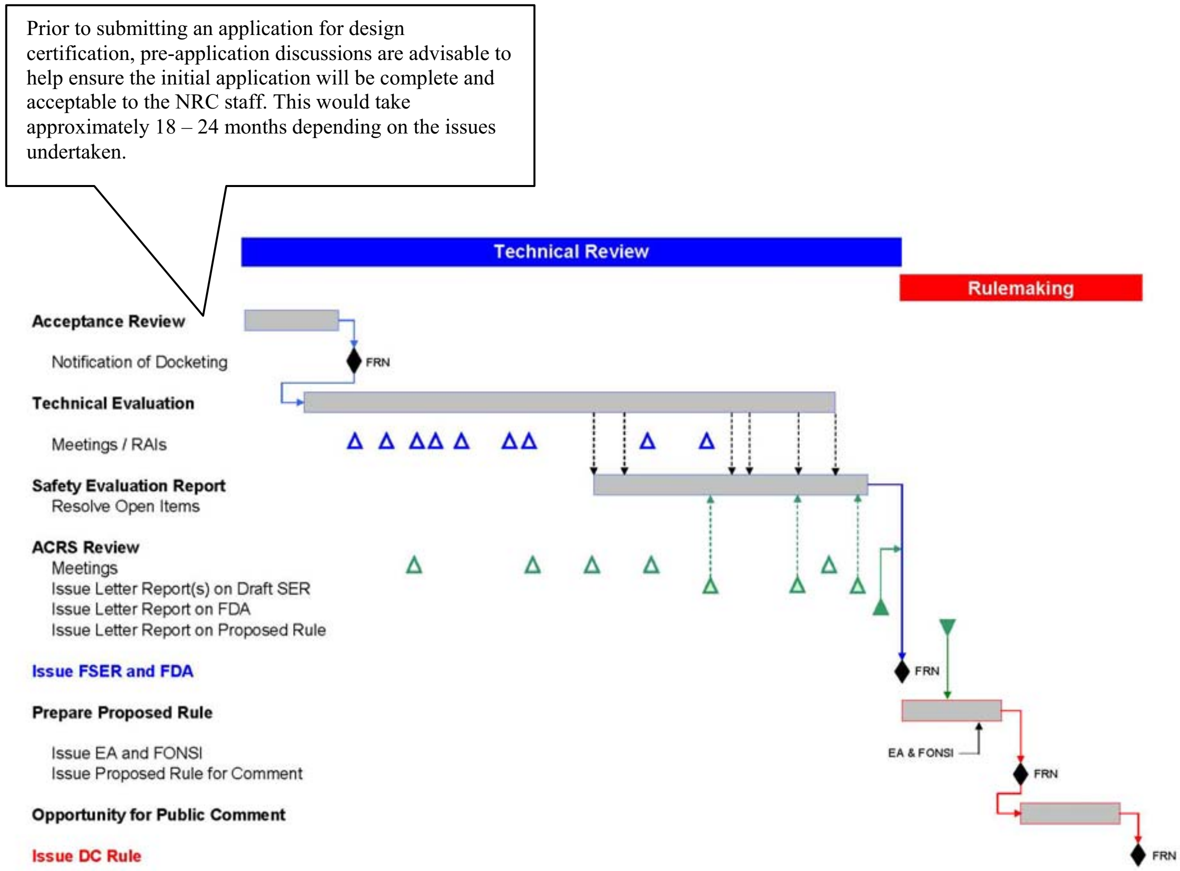

Figure 20.6.1-3: Design Certification Review Process under 10 CFR Part 52

20.6.1.2.2.4 ITAAC Requirements

An essential new feature of the Part 52 licensing process is the development of ITAAC (Inspections, Tests, Analyses, and Acceptance Criteria). ITAAC are the set of criteria by which the NRC will determine whether the plant has been constructed and is able to be operated in compliance with the certified design. Without ITAAC, there could be no one-step licensing. When the DC precedes the COL, the design-specific ITAAC are submitted as part of Tier 1 of the DC application. When the COL precedes the DC, the design-specific ITAAC are submitted as of the COL application. In either case, site-specific ITAAC are submitted as part of the COL application.

The COL applicant must provide ITAAC information as part of its COL application. The amount of information provided by the COL applicant will depend on the ITAAC information developed at the DC stage (if a DC is referenced) and will include additional information needed to complete safety reviews on operational programs.

In addition to the ITAAC that correspond to the certified design, the applicant for a COL must provide ITAAC for site-specific design features and which are outside the scope of the certified design.

In Section 20.6.1.3, Issues for Resolution, two additional 10 CFR Part 52 concepts related to ITAAC are addressed. The first is "Programmatic ITAAC" while the second is "Emergency Planning ITAAC."

20.6.1.2.2.5 Combined License

10 CFR 52 Subpart C sets out the requirements and procedures applicable for Commission issuance of COLs for nuclear power facilities. An application for a COL under this subpart may, but need not, reference a DC issued under Subpart B or an ESP issued under Subpart A, or both.

Unless otherwise specifically provided for in this subpart, all provisions of Part 50 and its appendices applicable to holders of CPs for nuclear power reactors also apply to holders of COLs. Similarly, all provisions of Part 50 and its appendices applicable to holders of OLs also apply to holders of COLs issued under this subpart, once the Commission has made the findings required that the initial duration of the license under Part 52 may not exceed 40 years.

The Nuclear Energy Institute has provided draft guidance to COL applicants assuming an ESP + DC + COL licensing process (NEI report #04-01, Revision E, dated October 5, 2005). During 2006, the NRC and industry developed through a series of workshops new guidance that would advance prior guidance found in Regulatory Guide 1.70. The draft guidance document DG-1145 is being finalized and is expected to be issued in early 2007. These guidance documents should be reviewed in detail when developing a more detailed licensing plan for the NGNP.

During the COL review, the NRC will assess the extent of the ESP NEPA coverage and determine whether significant changes have occurred since the ESP was issued. Depending on the results of their review, NRC will prepare either a draft EA or a draft EIS (for more detail, see NRC's November 16, 2006, summary of discussions held at the October 25, 2006, public meeting on the Part 52 rulemaking).

20.6.1.2.3 Process Considerations for Licensing the NGNP

The principal characteristics of licensing under Part 52 are highlighted in Table 20.6.1-2. Similar to Part 50, the technical and economic assessment processes are reasonably well understood. Once a COL is issued, construction and startup can proceed at the licensee's discretion. Part 52 is a new approach, but builds on the strengths of Part 50 while eliminating many Part 50 weaknesses.

Table 20.6.1-2: 10 CFR 52 Licensing Approach

10 CFR 52	PROS	CONS
	Reasonably well understood in the context of LWRs.	Not developed for gas-cooled reactors.
	Combines CP & OL. The completion of the NRC design review prior to the start of construction should eliminate most regulatory-driven design changes after construction is started.	Requires early submittal of an essentially complete plant design and information to close safety issues, including interface requirements for plant design and corresponding ITAAC.
	Allows pursuit of an ESP prior to selection of a specific design, thereby permitting early resolution of site-related issues.	
	Incorporates a formal ITAAC process that requires resolution of safety issues prior to issuance of the COL, and aids in limiting the likelihood of new regulatory issues during construction.	
	Design documents are similar to Part 50 (i.e., uses SER, PRA)	
	Public participation is primarily early (before construction). Petitions challenging a final COL must meet higher acceptance standards similar to those applied to licensed operating reactors	

20.6.1.3 ISSUES FOR RESOLUTION

The process in 10 CFR Part 52 is being revised as described in the March 13, 2006 Federal Register Notice and in the October 17, 2006, supplemental proposed rulemaking on LWAs. Both of these rulemaking actions are expected to be completed in early 2007 and would need to be evaluated as to their impact on NGNP licensing (to be done as part of Activity 15 if NRC actions are completed before work on Activity 15 concludes).

As part of Activity 15, the NGNP licensing schedule needs to be developed, including interfaces with NGNP design development activities and other expected NRC licensing actions such as the reviews of expected LWR COL applications. Licensing of the NGNP should benefit from coming after licensing of the next new LWRs. The LWR COL applications (referencing an ESP and a DC) are expected in 2007 – 2008 time frame with NRC approval approximately 30 - 36 months later (i.e., approval in the ~2010 - 2011 time frame). The review for a COL without referencing a DC would require a longer time, approximately 36 – 60 months, depending on the design and completeness of the application.

In addition, due to the uniqueness of the NGNP, many hundreds of questions and issues will likely be raised by the NRC staff during its review of the NGNP license application. Below are two lists that provide examples of major technical and policy issues that could be raised during either the pre-application planning review or during the technical review itself and that could affect the development of the NGNP licensing program.

The lists below are based on experience gained from considerable interactions with the NRC on topics relevant to HTGR licensing, beginning in the 1980's with the MHTGR review and continuing to today. Recent, key inputs have been provided from meetings and submittals associated with the PBMR (Pty) Ltd design certification pre-application program. These issues would likely be raised regardless of whether it is decided to pursue regulatory approval under Part 50 or Part 52.

"Policy" issues:

- Method for integration of PRA into the design process
    - Compliance with NRC guidance (e.g., Regulatory Guide 1.200)
    - Risk goals and compliance with the NRC Safety Goal Policy
    - Treatment of uncertainties
    - Interfaces with plant operations (e.g., living PRA, Design Reliability Assurance Program, Operational Reliability Assurance Program)
- Method for Licensing Basis Event selection
- Method for analysis of accidents and severe accidents
    - Use of confinement vs. LWR containment
    - Design criteria, if any, for aircraft impact (see NRC rulemaking on revisions to 10 CFR 73 and input from industry via a Nuclear Energy Institute letter dated December 8, 2006)
    - Specification of the mechanistic accident source term for NGNP particle fuel elements

  - Compliance with NUREG-0654/FEMA REP-1, Emergency Response Plans

- Method for safety classification of structures, systems, and components

- Method for evaluation of the Defense-in-Depth principle for non-LWR designs

- Method for meeting the NRC expectation of enhanced safety in new nuclear plants

- Method for verification & validation of analytical methods and computer codes

- The manner, if any, in which the NRC review of the NGNP license application will be coordinated with the NRC's ongoing development of the technology-neutral licensing framework (proposed Part 53)

- The need and content for a pre-application document (e.g., memorandum of understanding or Licensing Review Basis document) which identifies issues associated with regulatory requirements, policies, and guidance, and generally specifies how the staff and applicant will address them (see Section 20.6.3.4.1 for more discussion).

  - The need and content of a document that provides a detailed outline of the application to NRC (i.e., application for CP, OL, ESP, DC, COL)

  - Agreement with the NRC staff on the process to be used to develop appropriate regulatory criteria for the NGNP based on current LWR requirements and MHTGR experience.

  - The manner in which new issues (e.g., see Section 20.6.3.4.2) will be identified and addressed.

  - Identification of NRC regulations which are not applicable to a non-LWR and other regulations for which exemptions (per Section 50.12) will be required.

- The extent to which NRC (and/or the EPA and OSHA) reviews the process hazards analysis (PHA) for the hydrogen production facility and places requirements on the NGNP, including for example, the requirements for storing 10,000 pounds or more of hydrogen (40 CFR 68.95 and 29 CFR 1910.119)

- The method for improvement of plant physical security though design of plant layout, structures, systems, and components (e.g., SECY-06-0204, "Proposed Rulemaking – Security Assessment Requirements for New Nuclear Power Reactor Designs," dated September 28, 2006)

- Method for Regulatory Treatment of Operational Programs in the COL Process (previously known as "programmatic ITAAC") – depending on the level of detail in the COL application per SECY-05-0197 and its Staff Requirements Memorandum, dated February 22, 2006.

- Method for implementing Emergency Planning ITAAC – per SECY-06-0019 and Nuclear Energy Institute guidance document #NEI 04-01.

"Technical" review issues:

- Detailed implementation of the above policy issues
- Materials selection and their supporting industry codes and standards
- Human Factors Engineering and design of the control room
- Fuel design and qualification
- Fuel manufacturing quality assurance
- Startup and test program – scope and scheduling
- In-service inspection and testing program
- Decommissioning costs and funding
- Implementation of the NRC Construction Inspection Program for new reactors (e.g., Staff Requirements Memorandum for SECY-06-0041, dated April 21, 2006.
- Incorporation of Lessons Learned from current and anticipated ESP and COL reviews (e.g., ACRS letter #ACRSR-2213, dated September 22, 2006 on review of ESP applications).
- Confirmation of a Quality Assurance Program that ensures compliance with NRC requirements and guidance (e.g., 10 CFR 50, Appendix B and ASME NQA-1 2000).

These policy and technical licensing issues, that will have to be resolved for the NGNP, have embedded needs for research and development that are being defined as part of the PBMR (Pty) Ltd design certification pre-application program. Examples are the need for fuel qualification data and the need for separate effects tests to support the verification & validation of safety analysis evaluation models.

20.6.1.4 RECOMMENDATIONS

It is recommended that the NRC rulemakings related to the above licensing processes be followed as part of Activity 15 and that new developments and understandings be incorporated into the NGNP Licensing Strategy.

Recommendations in regards to which licensing options should be investigated further (in Activity 15) and adopted for the NGNP are presented in Sections 20.6.2.

20.6.2 FEASIBILITY OF MIXED LICENSING APPROACH (PART 52 ESP AND PART 50 CP/OL)

20.6.2.1 BACKGROUND

The previous sections provided an overview of the process and issues associated with both 10 CFR 50 and 10 CFR 52 licensing approaches. While it may be possible to select just one of these approaches and be able to have confidence in licensing the NGNP, it is also feasible to consider a mixed Part 50/52 licensing strategy.

The following two major categories of mixed approaches were considered:

- Basically Part 50 --- Use of Part 50 for a CP and OL, with an optional use of Part 52 for an ESP and with an optional use of a limited work authorization for early site preparation.
- Basically Part 52 --- Use of Part 52 for COL submittal, with various mixes of including or not including the ESP and DC as separate submittals and including or not including an optional use of a limited work authorization for early site preparation.

20.6.2.2 POTENTIAL APPROACHES FOR LICENSING THE NGNP

The licensing process options available for the NGNP are summarized below. Each of these options has different economic and schedule considerations and risks. Comments below have been limited to general matters with each option. Additional discussion of cost, schedule, and risk issues for the recommended option will be developed in Activity 15.

- CP + OL (Part 50 only)
    - Option Defined --- CP + OL without use of ESP and with or without an optional LWA.
    - Benefits & Best Use --- Well proven; possibly good match for NGNP if substantial design information is not available at the time of the application and there is a need to move to the construction stage.
    - Comment --- The most significant risk of this approach is that the OL review provides an opportunity for the NRC staff or the public to raise new issues and request design changes after plant construction is initiated.

- ESP + CP + OL
  - Option Defined --- Part 50 CP + OL, plus the use of Part 52 for an ESP. This option can be exercised with or without an optional LWA prior to issuance of the ESP.
  - Benefits & Best Use --- This approach provides early site approval and allows early site work. This approach also establishes a potentially useful template and precedent for ESP applications for commercial NGNP-like plants.
  - Comment --- For the NGNP, it is expected that there will be limited public objection to construction and operation at the selected INL site. In this case, given the site characterization work already performed for the INL site, the ESP should be accomplished with little or no risk. If it turns out that there is public opposition, the use of the ESP would be even more appropriate since it provides the opportunity for resolving site issues as early as possible.
- ESP + DC + COL (Part 52 only)
  - Option Defined --- Part 52 ESP, DC, and COL with or without an optional LWA prior to issuance of the ESP.
  - Benefits & Best Use --- Lower risk licensing process – which essentially prohibits design changes and intervener delays after construction is started.
  - Comment --- For the NGNP, the development of a DC requires substantial design detail (i.e., enough design detail to resolve all regulatory issues) to be documented in an application, reviewed and certified by the NRC in order to be referenced in a COL. This is impractical, even if portions are done in parallel, since the R&D programs for the NGNP may require changes that would introduce design, operations or safety considerations that would have to be inserted into the application. Furthermore, the purpose of a DC is to fix the conditions of the design and operations for future standardized replicates of the referenced DC plant. The basic nature of the NGNP is to be a prototype that develops insights into design and safety issues for subsequent designs and consequently, even if a DC were completed, the new insights and features would cause the "fixed" DC to be re-opened for NRC and public scrutiny, negating one of the most important risk reduction features of a DC.
- ESP + COL (design review is part of COL)
  - Option Defined --- Part 52 ESP and COL, but design information is not submitted for DC prior to COL submittal. This option can be exercised with or without an optional LWA prior to issuance of the ESP.

- Benefits & Best Use --- Allows the use of Part 52 to submit a COL application after the design is substantially finished, but not yet tested. May be a viable option for the NGNP plant.

- Comment 1 --- For the NGNP plant, an ESP is recommended to help ensure site approval is achieved as soon as possible and to provide schedule margin if unanticipated issues were raised. In addition, inclusion of an LWA request in the ESP application is recommended in order to permit early start of construction and, correspondingly, decrease construction schedule risk.

- Comment 2 --- For the current version of Part 52 concerns were expressed that a COL could not be issued for a prototype reactor since such new designs must reference operating experience for the same or similar designs – and by definition that experience does not exist for a prototype reactor. However, via the March 13, 2006, and the October 17, 2006, Federal Register Notices on proposed changes to Parts 50, 52 and others, the NRC would be able to issue a COL for a prototype, with potential conditions to ensure safety during startup and testing. This revision, if approved by the NRC, would make a COL application much more attractive for the NGNP plant and would facilitate implementation of "license by test" for certain systems or components.

- Comment 3 --- Submitting a COL without a DC is recommended for the NGNP, due to (1) the expectation that substantive reactor and safety system design detail will be available through the PBMR (Pty) Ltd and NGNP programs, (2) the desire to set standard application requirements and regulatory precedents for follow-on DC applications for NGNP Commercial plants, and (3) the expectation that design information required for DC, and not available from the PBMR (Pty) Ltd program, would be developed as part of the NGNP program.

- DC + COL (site approval is part of COL)

  - Option Defined --- Part 52 DC and COL, but site approval is delayed until the end of the COL process. This option can be exercised with or without an optional LWA.

  - Benefits & Best Use --- Allows a single application for a site and design that references an already-approved DC. While this approach does not make sense for the NGNP due to the lack of existing DC when the license application is prepared, it could be used later for an NGNP Commercial plant. For commercial plant applications, this can produce the shortest licensing schedule (the environmental and technical reviews are concurrent), although it would be of higher risk until there is broad experience and acceptance of a given design that has been validated by repeated earlier licensing and operational successes.

    - Comment --- To avoid NGNP Commercial plant delays, it would be advisable to pursue DC as soon as needed information is available from the designer and the experience of operating the NGNP plant.

- COL only (design and site approvals part of COL)
    - Option Defined --- Part 52 COL, but with complete environmental and design information being submitted with the COL application rather than separately in advance. This option can be exercised with or without an optional LWA.
    - Benefits & Best Use --- This option can produce the shortest licensing schedule (the environmental and technical reviews are concurrent), although it would be high risk as substantially complete plant design work is required before submittal. This option might be beneficial for an application that requires a mature reactor design (e.g., previously certified) coupled in a new way to a process application at a complex site. In this case, the presumption is that the environmental siting conditions are more, or equally, complex to the nuclear issues.
    - Comment --- The above comment for the option of DC + COL applies to this option as well. Importantly, as stated in Comment 2 above, the possibility of NRC issuing a COL for a prototype plant makes this COL option more feasible.

20.6.2.3 ISSUES FOR RESOLUTION

This section highlights issues that need to be resolved, in addition to those in Section 20.6.1.3, if the 10 CFR Part 52 process is selected:

- Evolving process --- Although the risk is expected to be low, Part 52 is still a new and unproven process for nuclear plants other than LWR plants.
- Need to submit design information early --- "One-step" licensing is a significant strength of Part 52; but, to implement this process, the plant design would need to be completed earlier than under Part 50.
- Ability to develop ITAAC for the NGNP --- Since ITAAC are a requirement for DCs and COLs, a set of comprehensive and accurately defined ITAAC and the groundrules for developing them are essential.

20.6.2.4 RECOMMENDATIONS

The following is recommended:

- Incorporate the preferred licensing option (namely, applying under Part 52 for an ESP with an embedded LWA, followed by a COL) into an overall NGNP Licensing Strategy, but maintain the option for a Part 50 strategy for a two-step license pending the success of the pre-application interactions.

This recommendation is based on:

- enabling site work to be started as soon as possible,
- the NRC's proposed revisions to the regulations which facilitate the use of LWAs and which allow issuance of a COL for a prototype plant,
- the expectation that substantive reactor and safety system design detail will be available for the NGNP at the time a COL application is prepared,
- the observation that there would be less schedule risk for the NGNP if a two-step licensing review (CP plus OL) were avoided,
- the expectation that design information required for DC, and not available from the PBMR (Pty) Ltd program, would be developed as part of the NGNP program,
- the realization that future detailed studies of the NGNP schedule could demonstrate the need to start construction earlier than could be supported using the COL-only licensing option, and
- the desire to set standard application requirements and regulatory precedents for follow-on DC applications for NGNP Commercial plants

20.6.3 FEASIBILITY OF USING NEW ADVANCED REACTOR LICENSING FRAMEWORK (PROPOSED NEW 10 CFR PART 53)

20.6.3.1 BACKGROUND

Initial reactor safety requirements in the 1950's and 1960's were technology neutral. With the advent in the late1960's and early 1970's of large Light Water Reactor (LWR) designs, the Atomic Energy Commission (AEC) and, later, the Nuclear Regulatory Commission (NRC) refocused its requirements development efforts almost entirely toward the promulgation of LWR-specific regulations. The regulations and subsequent guidance (Safety Guides, Regulatory Guides, Standard Review Plans, etc.) that were put forth in this period were not only technology-specific but also prescriptive in nature. That is, the underlying approach was to use deterministic methods and assumptions to analyze the safety of nuclear reactors and to establish the conditions for specific requirements that safety-related structures, systems, and components (SSCs) were required to meet.

Following the Three Mile Island accident, the focus on reactor safety started to change from the deterministic to a more risk-informed and performance based framework of regulations. The use of a structured probabilistic risk assessment (PRA) for evaluating the risks to the public (initially highlighted in WASH-1400, *Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants* [Ref. 20.6-3-1]) was advanced.

Beginning in the mid-1980's, several NRC policy statements were published that form the high-level background in which today's reactor safety regulations are proposed, developed and implemented. Table 20.6.3-1 provides a list of relevant policy statements and their objectives. When taken together, these policy statements provide the foundation in which risk-informed methodologies are to be considered when establishing the safety and design bases for advanced nuclear plants.

Table 20.6.3-1: NRC Policy Statements on Safety Performance Expectations for Advanced Reactor Designs

Policy Statement	Objectives
Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (Federal Register Vol. 50, No. 153, pg 32138-32150, August 8,1985)	States the expectation that new plants are to achieve a higher standard of severe accident safety performance than prior designs
Regulation of Advanced Nuclear Power Plants; Statement of Policy (Federal Register, Vol. 51, No. 130, pg. 24643-14648, July 8, 1986)	States the expectation that advanced reactors provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions.

Safety Goals for the Operations of Nuclear Power Plants; Policy Statement (Federal Register, Vol. 51, No. 149, pp.28044-28049, August 4, 1986; republished with corrections, Vol. 51, No. 160, pg. 30028-30023, August 21, 1986)	Establishes two qualitative safety goals, which are supported by two quantitative objectives.
Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement (Federal Register, Vol. 60, No. 158, pg. 42622-42629, August 16, 1995)	States the expectation that the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC deterministic approach and supports the NRC traditional DID philosophy. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices.

During the 1990s, the NRC advanced the implementation of risk-informed methods for both operating reactors and for Advanced LWRs that were undergoing DC reviews.

The transition away from a purely deterministic approach to the use of risk-informed and performance based methods has been slow and incremental, reflecting the large body of regulations in place for the fleet of existing LWRs. The transition is continuing and the development of the proposed Part 53 reflects the extension of risk-informed methods to a technology-neutral approach to reactor regulation. The following sections not only summarize these developments, but also provide the basis for the judgment that much work remains to complete the proposed Part 53 rulemaking and that the schedule for completing the rulemaking is uncertain.

The summary in the following sections reflects a significant amount of document review and NRC meeting attendance carried out during approximately the past two years in support of the PBMR design certification pre-application interactions with the NRC.

20.6.3.2 INCREMENTAL MOVEMENT TOWARD A RISK-INFORMED, PERFORMANCE-BASED REGULATORY FRAMEWORK

In SECY-98-300, *Options for Risk-Informed Revisions to 10 CFR Part 50 - "Domestic Licensing of Production and Utilization Facilities"* [Ref. 20.6.3-2], the NRC staff proposed a phased implementation strategy for incorporating risk-informed attributes into the regulations. Three options were presented for Commission consideration:

Option 1 – Terminate any further efforts to add risk-informed attributes to Part 50 technical requirements. Then ongoing rulemaking activities, which have risk informed elements, could continue but no new initiatives would be pursued.

Option 2 – Address implementing changes to the regulatory scope for SSCs needing special treatment. Under this option a graded approach to classifying SSCs in terms of quality (e.g., quality assurance, environmental qualification, technical specifications, etc.) would be established. This option did not address changing the design of the plant or the design-basis accidents.

Option 3 – Identify changes to specific regulatory requirements. This option was stated as ranging from a broad, complete rewrite of Part 50 to a more limited scope effort, focusing on regulations considered to have 'the most significant potential for improving safety and efficiency and reducing unnecessary burden". No specific rulemaking proposals were identified pending further study by the NRC staff.

In its Staff Requirements Memorandum on SECY-98-300 [Ref. 20.6.3-3], the Commission approved implementation of Options 1 and 2, adding that the scope of the Maintenance Rule (§50.65) be changed to conform to the risk-informed regulatory framework being developed as part of Option 2. The Commission also approved the staff's recommendation to study Option 3. Once the staff completed its study, they were to provide the Commission recommendations on specific regulatory changes that should be pursued. Policy issues that were to be considered included:

- Voluntary vs. mandatory conformance with modified Part 50
- Industry pilot studies with selected exemptions to Part 50
- Modification of the scope of the Maintenance Rule [§§50.65(a)(3) and (a)(4)]
- Clarification of staff authority for applying risk-informed decision making

Significant progress has been made since SECY-98-300 in finalizing rulemaking efforts to risk-inform selected regulations under Part 50 and to establish a set of guidance documents suitable for their implementation. Ongoing activities are tracked in the NRC semi-annual Risk-Informed Regulation Implementation Plan (RIRIP). Table 20.6.3-2 provides several examples of risk-informed regulations and guidance that either have been developed or are continuing to be developed since the mid-1990's.

Table 20.6.3-2: Examples of Areas Where Risk-Informed Attributes Have Been Incorporated into 10 CFR Part 50 Requirements and Guidance

General
RG 1.174 An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis (Rev 1, November 2002) RG 1.176 An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance (Rev 0, August 1998) RG 1.200 An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Draft Rev 1, August 2006)
§50.36 Technical Specifications
RG 1.177 An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications (Rev 0, August 1998)
§50.44 Combustible Gas Control for Nuclear Power Reactors
§50.46a Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors (Draft Final Rule)
§50.48 Fire Protection
RG 1.205 Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants (Rev 0, May 2006)
§50.55a Codes and Standards
RG 1.175 An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing (Rev 0, August 1998) RG 1.178 An Approach for Plant-Specific, Risk-informed Decisionmaking: Inservice Inspection of Piping (Rev 1, September 2003)
§50.65 Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
RG 1.182 Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants (Rev 0, May 2000)
§50.67 Accident Source Term
RG 1.183 Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors (Rev 0, July 2000)
§50.69 Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors
RG 1.201 Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance (Rev 1, May 2006)

NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (revision is currently ongoing)
DG-1146 A Performance-Based Approach To Define the Site-Specific Earthquake Ground Motion – Alternative to RG 1.165 (Draft, October 2006)

These risk-informed regulations and guidance documents were developed or are being developed based on an LWR-specific basis and in anticipation of use for license amendments on a backfit basis for existing LWRs. Consequently, much of the guidance needs to be interpreted or reevaluated in order to be determined to be applicable to advanced, non-LWR reactor designs.

20.6.3.3 DEVELOPMENT OF A NEW ADVANCED REACTOR LICENSING FRAMEWORK (PROPOSED PART 53)

In SECY-06-0007 [Ref. 20.6.3-4] the NRC staff proposed:

> *'…to achieve the Commission's direction to make a risk-informed and performance-based revision to 10 CFR Part 50 by creating a completely new risk-informed and performance-based Part 50 (to be called Part 53) that is applicable to all reactor technologies. The development of this new Part 53 will integrate safety, security, and preparedness. This approach will ensure that the reactor regulations, and staff processes and programs, are built on a unified safety concept and are properly integrated so that they complement one another.'*

The Commission in its Staff Requirements Memorandum on SECY-06-0007 [Ref. 20.6.3-5] approved the staff's recommendation to publish an Advance Notice of Proposed Rulemaking (ANPR). Comments on the ANPR were to be completed by December 2006 and the staff is to provide its recommendation on whether and, if so, how to proceed with rulemaking by May 2007.

The ANPR on *Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors* was published in the Federal Register on May 4, 2006 [Ref. 20.6.3-6]. In the Statement of Considerations accompanying the ANPR, it was noted:

> *'The Nuclear Regulatory Commission (NRC) is considering modifying its approach to develop risk-informed and performance-based requirements applicable to nuclear power reactors. The NRC is considering an approach that, in addition to the ongoing effort to revise some specific regulations to make them risk-informed and performance-based, would establish a comprehensive set of risk-informed and performance-based requirements applicable for all nuclear power reactor technologies as an alternative to*

current requirements. This new rule would take advantage of operating experience, lessons learned from the current rulemaking activities, advances in the use of risk-informed technology, and would focus NRC and industry resources on the most risk-significant aspects of plant operations to better ensure public health and safety. The set of new alternative requirements would be intended primarily for new power reactors although they would be available to existing reactor licensees.'

The ANPR also noted that the NRC "…plans to continue the ongoing efforts to revise specific regulations in 10 CFR Part 50 as described in SECY-98-300…"

20.6.3.3.1 Elements of the Proposed New Part 53

A summary of the NRC staff's expectations for a new Part 53 can be found in the ANPR's accompanying draft guidance document, NUREG-1860, Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50 (Draft Working Report) [Ref. 20.6.3-7]. This NUREG is an "information document" and does not represent a fully-reviewed or approved NRC staff position. The report, expected to be finalized in December 2006 or early 2007, is to be used in support of the NRC staff's recommendation to the Commission on whether and how to proceed with rulemaking (due in May 2007). The Executive Summary of NUREG-1860 is presented in Appendix 20.6.A.

Two tasks were identified in the ANPR: (1) development of a technical basis for a risk-informed and performance-based Part 53 and (2) development of specific regulations for Part 53. At this stage, the NRC did not propose new regulations but sought feedback on a series of questions into the basis of the proposal itself as well as several policy issues associated with advanced non-LWRs. Areas of inquiry included:

A. Plan

B. Integration of Safety, Security, and Emergency Preparedness

C. Level of Safety

D. Integrated Risk

E. ACRS Views on Level of Safety and Integrated Risk

F. Containment Functional Performance Standards

G. Technology-Neutral Framework

H. Defense-in-Depth

I. Single Failure Criterion

J. Continue Individual Rulemakings to Risk-Inform 10 CFR Part 50

20.6.3.3.2 Industry Feedback

The development of a risk-informed, technology-neutral framework will be ground-breaking. It is not believed that a rigorous and workable regulatory rule framework can be successfully developed as a theoretical exercise. Absent the pilot application of the framework to an actual ALWR and non-LWR designs, there will be many unknowns and many unidentified issues, and less confidence that the regulations will achieve their objectives.

Feedback to the NRC on the proposed Part 53 rule has been provided to the NRC via written submittals and workshops by organizations, including the American Society for Mechanical Engineers, the American Nuclear Society, the Nuclear Energy Institute, PBMR (Pty) Ltd, and Westinghouse.

Feedback has been in favor of pursuing the development of the rule. However, concerns were identified related to (1) the use of LWR-derived concepts by the NRC staff in developing examples of how risk-informed requirements might be developed for non-LWR designs, (2) the time required for development and implementation of a fully technology-neutral set of regulations and guidance and (3) the need for additional development and implementation of PRA methods on an actual design (e.g., a gas-cooled reactor) in order to gain insight into resolution of numerous issues for non-LWRs such as implementation of Defense-in-Depth, and implementation of a "confinement system" rather than a pressure-retaining containment.

Additionally, and more substantially, PBMR (Pty) Ltd established a formal design certification pre-application project with the NRC in April 2005 for their gas reactor design. As part of the scope of pre-application, PBMR (Pty) Ltd has developed and submitted a series of four submittals (so called "white papers"), that provide a full discussion of how to develop a risk-informed, performance-based design and licensing approach for the PBMR design. These papers taken together provide the first instance of a real application describing risk-informed techniques for advancing the PBMR design as recommended by industry. The timely review of these papers by the NRC will benefit the NGNP and materially assist in solidifying the licensing strategy for the NGNP.

20.6.3.4 ISSUES FOR RESOLUTION

In comments on the ANPR, industry recommended that, prior to spending substantial effort in developing the details of such a regulatory framework, the NRC identify one or more proof-of-concept applications, review and approve the application(s) using risk-informed technology-neutral concepts developed by the applicant and accepted by the NRC, and then develop the details of Part 53 based upon the lessons learned from review of the application(s). This approach is similar to the approach used by the NRC to develop 10 CFR §50.69 *Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors*, in which the NRC reviewed and approved a risk-informed exemption for the South Texas Project, and then used the lessons-learned from that review in developing the details of the regulation.

Appendix E to NUREG-1860, Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50 (Draft Working Report) [Ref. 20.6.3-7] describes a preliminary example of a risk-informed selection process for licensing basis events (LBEs) and the classification of safety significant systems, structures and components (SSCs). Industry comments on the ANPR point out that this example is not an appropriate basis for development of a risk-informed technology-neutral rule. First, the review was performed for an LWR, and therefore does not represent a true test of a regulation that is intended to transform the NRC current LWR-biased regulations into technology-neutral regulations. More importantly, the review was theoretical and did not get to the detail required to fully understand the ramifications of a risk-informed approach for an actual design. It did not involve the review of a real application with a live applicant, and therefore did not raise and cannot have raised all of the types of issues that will arise from the review of an actual application. Therefore, the review in Appendix E to NUREG-1860 does not represent a true test of whether a risk-informed technology neutral regulatory framework is workable.

20.6.3.4.1 Establishment of an Agreed Upon Set of Requirements by Which an Advanced Reactor Can Be Licensed

Fort St. Vrain was licensed under the existing Part 50 rule set and HTGRs could also be licensed under Part 50 if needed. The lack of regulations and guidance clearly applicable to gas-cooled reactors, however, would lead to delays in review and extended negotiations over exemptions, departures from requirements, etc. Nonetheless, the existing rule set generally remains workable for gas reactors based on previous MHTGR and Exelon PBMR experience as well as current PBMR (Pty) Ltd pre-application interactions with the NRC. This concern can be partially ameliorated through the development of a Licenisng Review Basis (LRB) document, which could be called a memorandum of understanding, during the early stages of a licensing program.

The development of an LRB can be used by license applicants and the NRC to address the approach to be used to resolve technical and other regulatory issues. A summary of Commission discussions relative to the need for development of LRB documents for advanced reactor designs can be found in several NRC staff reports to the Commission:

- SECY-90-362, October 24, 1990: This Commission paper presents NRC staff comments and recommendations concerning the need to develop a licensing review basis (LRB) document for the passive advanced light water reactor (LWR) designs and for future advanced designs (ADs). The comments of the staff pertain to the following: 1) the completion of the Asea Brown Boveri-Combustion Engineering (ABB-CE) System 80+ LRB document; 2) the views of ABB-CE, General Electric Company, and Westinghouse Electric Corporation; 3) the views of the Nuclear Management Resources Council; and 4) the need for LRB documents for future ADs. Due to Commission guidance, initiatives are underway in individual subject areas that can be used in place of a formal LRB document for passive advanced LWRs. These initiatives will accomplish all of the major goals that the NRC initially intended to achieve through the LRB concept. The NRC can address other concerns individually as they develop. However, the Commission may still wish to develop an LRB document for other ADs that differ significantly from current LWR technology. It is recommended that the Commission 1) approve the staff recommendation that a formal LRB document will not be required for the AP-600 and SBWR reviews; 2) approve the staff recommendation that LRB documents for the MHTGR, PRISM, PIUS, and CANDU are needed; and 3) note that the staff will incorporate Commission comments when received and finalize the ABB-CE LRB for Commission approval.

- SECY-91-419, "The Need for Licensing Review Basis Documents for the Power Reactor Innovative Small Module (PRISM), Modular High Temperature Gas-Cooled Reactor (MHTGR), Canadian Deuterium Uranium 3 (CANDU 3), and Process Inherent Ultimate Safety (PIUS) Projects", December 30, 1991; and SRM dated January 21, 1992.

- SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and Canadian Deuterium Uranium Reactor (CANDU) 3 Designs and Their Relationship to Current Regulatory Requirements," April 8, 1993 (revised April 28, 1993); and SRM dated July 30, 1993.

- SECY-95-299, "Issuance of the Draft of the Final Pre-application Safety Evaluation Report (PSER) for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," December 19, 1995; and SRM dated February 13, 1996.

This last SECY paper summarizes the staff's review and findings from its pre-application review of the Modular High-Temperature Gas-Cooled Reactor (MHTGR) [Ref. 20.6.3-8]. This report, NUREG-1338, concluded that additional work would be needed to establish a set of regulatory requirements for licensing review of the MHTGR design.

In its pre-application review on the submittal of an application for a Combined Operating License (COL) for the PBMR [Ref. 20.6.3-9], Exelon Generation Company outlined a screening process which could be used for the NGNP project to categorize existing requirements as applicable, partially applicable, or not applicable (see Figure 20.6.3-1). Such a process can also help identify areas where new requirements specific to the NGNP design may be needed.

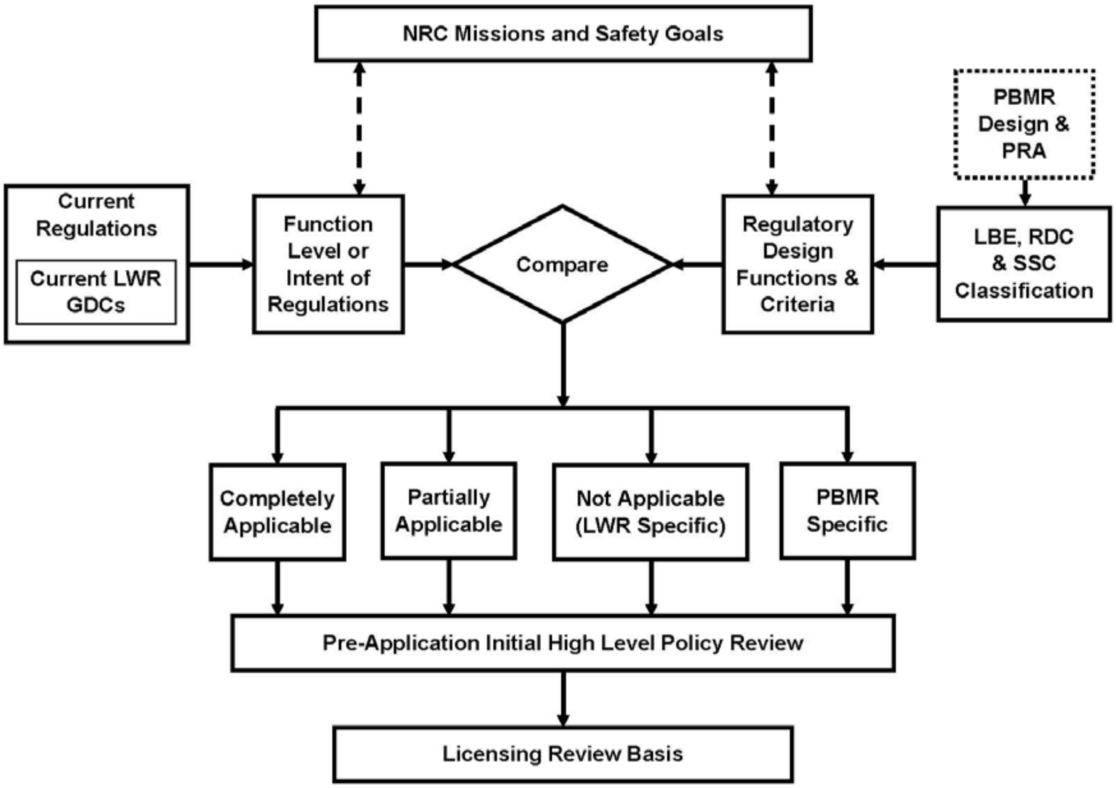

Figure 20.6.3-1: Example of Risk-Informed Screening Process for Identification of a Licensing Review Basis for the NGNP Design

The Exelon proposal included two steps: The initial step would be a screening of the existing regulatory framework for applicability and the second step would be to refine the focus of the applicable regulations using available design and risk-informed insights to develop the proposed regulatory requirements set. The acceptability of the end product (i.e., a licensing review basis) would need to be determined with the NRC during the pre-application review period. The expected outcome would then be to create early agreements where possible and a greater confidence that the license application will provide the proper information for an efficient and effective regulatory review.

This approach has been endorsed and included in the PBMR (Pty) Ltd pre-application project as well. This project is an on-going design review that meets the industry objective of having a demonstration of principle for the new framework. Due to substantial similarities in fundamental regulatory issues that are applicable to all small, inherently safe gas reactors, the NGNP licensing strategy and the PBMR licensing strategy share a large number of generic issues that can be resolved with greater confidence through parallel trial programs. Furthermore, greater regulatory efficiency and effectiveness can be achieved, as well as greater licensing certainty for gas reactors in general, through coordinated pursuit of generic issues with the NRC.

20.6.3.4.2 Policy Issues Associated with Advanced Non-LWR Designs

Over the past several years, the NRC staff has issued the following reports to the Commission which discuss the status of issues that will likely have to be addressed before completion of the Part 53 rulemaking:

- SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," March 28, 2003; and SRM dated June 26, 2003.
- SECY-04-0157, "Status of Staff's Proposed Regulatory Structure for New Plant Licensing and Potentially New Policy Issues," August 30, 2004.
- SECY-05-0006, "Second Status Paper on the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing," January 7, 2005.
- SECY-05-0130, "Policy Issues Related to New Plant Licensing and Status of the Technology-Neutral Framework for New Plant Licensing," July 21, 2005; and SRM dated September 14, 2005.

Issues such as those in the above SECY reports will likely also have to be addressed during the review of the NGNP license application.

20.6.3.5 RECOMMENDATIONS

The development of the expected Part 53 technology-neutral framework is a work in progress; staff recommendations to the Commission are not due until May 2007 and the final rule would not be published until approximately the 2012 time frame. Based on industry comments to date, it can be judged that a long time frame will be needed for completion of the rule and its implementation and the expected 2012 time frame may be further delayed due to the scope of the issues and competing NRC priorities. It is clear that the time-frame for the rule is inconsistent with the development of a licensing application for the NGNP under Part 52, since there will not be a final rule against which the application can be reviewed. Moreover, the review of the first application for a particular reactor design type under Part 53 will be especially difficult since there is no experience that shows how compliance of an application can be evaluated against the expected technology-neutral regulatory requirements. Therefore, implementation of Part 53 is not recommended.

Nonetheless, it might be advisable to work with the NRC on development of Part 53 even though the NGNP might be using the Part 52 licensing process and underlying technical requirements as they essentially exist today. It is important to recognize that the absence of the new rule does not prevent the use of risk-informed and performance-based techniques to establish design and safety requirements for the NGNP. This approach is fully endorsed by the NRC today in numerous policy statements and is highly encouraged under the existing regulatory framework.

20.6.4 PRACTICALITY OF "LICENSE BY TEST" LICENSING METHOD

20.6.4.1 BACKGROUND

The existing understanding for license by test (LBT) of full scale, prototype, nuclear power plants dates back to test reactors at INL, the Fast Flux Test Facility (FFTF), Clinch River and early HTGR demo plants. These plants predated the current Nuclear Regulatory Commission and Department of Energy. At that time the Atomic Energy Commission (AEC) was responsible for licensing nuclear power plants and the licensing regulations were significantly less specific than the current licensing regulations. In the same time frame the AEC licensed the Peach Bottom-1 and Fort St. Vrain gas cooled reactors, both of which operated for a number of years. While licensing of these gas cooled reactors may provide some precedence for current licensing, a recent study of Fort St. Vrain operational experience (Ref. 20.6.4-1) noted:

> *"In conclusion, the study notes that it would be very difficult to draw any generalizations from FSV [Fort St. Vrain] licensing and operations that can be applied to the licensing of future gas-cooled reactors without careful consideration of the specific circumstances that were applicable to FSV…."*

Much more insight into the potential practicality of "license by test" for current HTGR plants can be gained from relevant NRC regulations and more recent NRC licensing reviews of HTGRs, as discussed further below. While DOE regulations do exist for licensing nuclear power plants (e.g., References 20.6.4-2, 20.6.4-3, etc.), the scope of this review will focus on relevant NRC regulations because 1) Section 651 of the Energy Policy Act of 2005 (Ref. 20.6.4-4) designated that the NRC "shall have licensing and regulatory authority for any reactor authorized under this subsection" and 2) the DOE orders are high level regulations subject to broad interpretation which ultimately would not provide an equivalent high level of commercial licensing certainty.

While, as stated above, the LBT concept is generically permitted by the NRC, it has not been defined in regulatory documents. However, there is a significant amount of regulatory documentation addressing the testing of prototype plants that can be interpreted as LBT when they are applied to full scale prototypes. The rest of this section provides relevant examples from a variety of regulatory documents that can be directly applied to full scale prototype nuclear power plants including non-LWR plants where applicable. From Reference 20.6.4-7 that documents the "Final Rule: Early Site Permits; Standard Design Certifications and Combined Licenses Policy Statement for Nuclear Power Reactors":

> *"Prototype testing is likely to be required for certification of advanced non-light-water designs because these revolutionary designs use innovative means to accomplish their safety functions, such as passive decay heat removal and reactivity control, which have not been licensed and operated in the United States."*

The following requirements from the above Federal Register notice were incorporated into 10 CFR 52 Section 52.47:

> *"(b)(2)(i) Certification of a standard design which differs significantly from the light water reactor designs described in paragraph (b)(l) of this section or utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions will be granted only if:*
> *(A) (1) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;*
> *(2) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;*
> *(3) 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; and*
> *(4) The scope of the design is complete except for site-specific elements such as the service water intake structures and the ultimate heat sink; or*
> *(B) There has been acceptable testing of an appropriately sited, full-size, prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If the criterion in paragraph (b)(2)(i)(A)(4) of this section is not met, the testing of the prototype must demonstrate that the non-certified portion of the plant cannot significantly affect the safe operation of the plant."*

In its March 13, 2006 Federal Register Notice on the Part 52 proposed rulemaking, the NRC proposed the following changes to 10 CFR 50 Section 50.43(e):

> *"(e) Applications for a design certification, combined license, manufacturing license, or OL 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 testing period."*

The statement of considerations accompanying this proposed rulemaking notes that:

"if a prototype plant is used to comply with the testing requirements, the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to compensate for any uncertainties associated with the performance of the new or innovative safety features in the prototype plant. Although the NRC stated that it favors the use of prototypical demonstration facilities and that prototype testing is likely to be required for certification of advanced non-light-water designs (see Policy Statement at 51 FR 24646; July 8, 1986, and Section II of the final rule (54 FR 15372; April 18, 1989) on 10 CFR part 52), this revised proposed rule would not require the use of a prototype plant for qualification testing. Rather, this proposed rule would provide that if a prototype plant is used to qualify an advanced reactor design, then additional requirements may be required for licensing the prototype plant to compensate for any uncertainties with the unproven safety features. Also, the prototype plant could be used for commercial operation."

§52.47(b)(2)(i)(B) and the proposed revision to 50.43(e) provide the regulatory criteria that would be the basis for establishing test criteria. If LBT were going to be applied, §52.47(b)(2)(i)(A) would also relevant because of the potential that significant aspects of the PBMR and NGNP designs meet or will meet the criteria of Part A and; therefore, do not have to be certified by LBT.

In the Draft Pre-application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (Ref. 20.6.4-8), the staff provided more detailed guidance on the application of prototype (i.e., LBT) testing criteria for HTGRs. In Sections 1.8, 3.2.1.1 and 3.2.2.3 of the report, the staff stated that the MHTGR has the capability to demonstrate:

1) "By test the significant safety features and, performance of the plant over a wide range of events,"

2) "Via testing on the first-of-a-kind or prototype plant, that reasonable assurance will exist that these [innovative] features will prevent or accommodate accidents," and

3) "Fission-product-retention capability of the design via a testing program utilizing a full-size prototype plant consisting of at least one reactor module and the associated systems, structures, and components necessary to demonstrate safety."

"Such testing should be done at an isolated site, such as the National Reactor Testing Station, and the prototype plant should conform to the same regulations and standards as the design to be certified. The testing program should generate plant performance data sufficient to validate safety-analysis analytical models.

"In addition, for acceptance of a design without a containment building, these criteria would require demonstration via a full-size prototype test at an isolated site of the fission product-retention capability of the design. Requiring such demonstration testing is

> considered necessary to compensate for removal of the traditional (and testable) containment building. Such testing will help ensure that licensed plants of that design have adequate fission-product retention."

In Section 14 (Prototype-Plant Testing) of Ref. 20.6.4-8, the staff further clarifies the potential for license by test before completing the design certification:

> "However, based on judgments of the adequacy of existing operating experience, the novel design features proposed, and the status of the present technology base, the staff requires that testing and operation of a prototype test reactor, located at an isolated site, be mandatory before design certification. … The testing program would not intentionally risk damage to the plant, such as elevating reactor-vessel temperatures into the service level C domain."

While this guidance is more detailed than the Part 52 rule, it is compatible with the rule.

Via Reference 20.6.4-9 the staff informed the Commission of the staff's procedure for determining the need for a prototype or other demonstration facility for the advanced reactor designs.

> "The advanced reactor designs may need testing ranging from basic research and development (R&D) up to a full-size prototype plant in order to demonstrate that these designs are sufficiently mature to be certified.
>
> "…The prototype could include additional safety features to protect the public, the plant staff, and the plant itself from the consequences of unanticipated failures during the testing period. The function of each system in the prototype must accurately represent the function specified in the final design in order to justify the design for certification under 10 CFR Part 52. In addition to physically constructing the prototype, the applicant must design the testing program to test the full range of design features and safety claims associated with the plant. Some features may not be testable in the prototype without damaging and possibly destroying the plant, resulting in consequences that are unacceptable. For these features and design functions, the prototype test must be performed at partial power levels or be supplemented with other types of tests (e.g., special features tests or component tests) to validate the behavior of the design without the extreme consequences that could result if the feature were tested in the full-size plant. The applicant would need a comprehensive testing program and a program for ensuring safety while the uncertainties of the plant are being tested. The prototype for an advanced reactor design may need some additional safety features to compensate for the uncertainties in the design that the prototype is intended to test. However, the applicant would have to insure that the additional safety features would not affect the test program. For example, if a design is proposed without a containment, the ability of such a plant to protect the public would be very uncertain if the safety systems failed and a release occurred. Therefore, the prototype might be built at an isolated site that would minimize the threat of exposure to the public from atmospheric dispersion of accidental releases, or

the prototype could be built inside a containment designed to capture any release from the plant under all postulated conditions. New designs with less diversity and redundancy in safety systems or with boundaries that rely on highly reliable equipment, may require extra trains or components that can be used if the reliability of the system or component is not as high as expected. The backup system or component, which is only intended for the prototype, could be used to perform the function if the primary equipment were to fail. In such tests, if the backup equipment were used, it would indicate a failure of the plant design, the assumptions, or the reliability of the equipment. Therefore, the safety claim and the design would not be sufficient for the NRC staff to certify the new design under 10 CFR Part 52"

The NRC provides more detail of what would be expected, if a prototype, demonstration plant were used to generate the research and development information required to support the licensing of the plant in Appendix G, Section G.2.2 of NUREG-1860, "Framework for Development of a Risk Informed, Performance Based Alternative to 10 CFR Part 50" (Ref. 20.6.4-10).

"Use of Prototype Testing

"New plants may also propose the use of a demonstration plant, in lieu of conducting extensive research and development. In this case, the demonstration plant would be used to demonstrate the safety of the design in lieu of a series of separate research and development efforts. If such an approach is to be accepted, the applicant would need to address:

"• What would be the objective of the test program:
- Which aspects of plant safety can be addressed by demonstration plant testing?
- Which types of analytical tools could be validated?
- What phenomena could be addressed?

"• What would be the scope of the test program:
- How would the test program be selected?
- Would it be conducted during initial startup only?
- How would plant aging, irradiation, burnup effects be tested?
- Would tests cover the full range of the accidents or only partial ranges, with the remainder done by analysis?
- What instrumentation would be required?

"• Are any special provisions needed in case the tests do not go as planned (e.g., containment, EP, has to be on a remote site, DOE site, etc.)?
"• How would equipment reliability assumptions be verified?
"• What acceptance criteria would be necessary (e.g., scope, treatment of uncertainties)?
"• Would there be any limitations on future design changes?
"• If the initial demonstration plant is to be licensed, how would this be accomplished?

"Also, documentation for the test program results needs to be specified."

While NUREG-1860 is currently a working draft, the NRC has referenced it extensively in supporting their effort to develop 10 CFR Part 53 (see Section 20.6.3) and it has been the basis for NRC sponsored workshops on the Advanced Notice of Proposed Regulation for 10 CFR 53. NUREG-1860, or a later revision, is expected to become the NRC basis for implementing 10 CFR Part 53. Since NUREG-1860 is being developed to address advanced plants (including non-LWR plants), much of the licensing approach developed for 10 CFR Part 53 will also be applicable to the licensing approach for 10 CFR Parts 50 or 52.

An indication of the relationship of license-by-test to this discussion of prototype testing is provided in an earlier version of NUREG-1860 (Ref. 20.6.4-11). In that version, the NRC provided the following introduction to the "prototype" test program:

> "New plants may propose the use of a license-by-test approach, in lieu of conducting extensive research and development. The use of a license by test approach results primarily from the new technologies and reactor designs that could be proposed in the future (e.g., HTGRs, modular reactor designs), whereby one module could be built and used to demonstrate the safety of the design in lieu of a series of separate research and development efforts."

The change from "license-by-test" to "prototype testing" is an indication of the fact that "prototype testing" is already mentioned in regulation, while "license-by-test" is not.

The regulatory documentation cited above provides a detailed description of what needs to be provided to the NRC to support full scale prototype testing (i.e., LBT). The following sections describe the proposed licensing process and approaches that could support the potential application of LBT to the NGNP. It is also noted that a successful LBT program on the NGNP would provide information in support of a Design Certification application for an NGNP Commercial plant.

20.6.4.2 LICENSING PROCESSES

The LBT concept for the NGNP would initially be limited to proposed separate effects and/or potentially integrated testing on the plant, as constructed, for the purpose of generating safety performance, margin analysis, and uncertainty management information that had not been established by previous separate effects or prototype testing in the PBMR (Pty) Ltd or NGNP research and development programs. It should be noted that if these programs provide all information needed to close regulatory review issues on a timely basis, LBT would not be required for the NGNP and this plant would not be designated as a "prototype" for regulatory evaluation purposes (discussed further below). However, to help ensure that full-power operation of the NGNP is achieved in a timely manner, LBT tests could be performed during the normal commissioning of the plant with adequate restrictions to assure safe plant operation while the tests are being performed and evaluated. In addition, successful completion of these tests would provide information necessary to support Design Certification of the NGNP Commercial plant.

All of the licensing approaches considered in this report will require NRC review and approval of the evaluation models (EMs) used to perform the licensing analyses for NGNP. Regulatory Guide 1.203, "Transient and Accident Analysis Methods" (Ref. 20.6.4-12) describes a current process that the NRC considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. The Regulatory Position section describes the multi-step Evaluation Model Development and Assessment Process (EMDAP) for developing and assessing Evaluation Models, and provides guidance on related subjects, such as quality assurance, documentation, general purpose codes, and a graded approach to the process. The EMDAP approach for approving EMs applies to all of the licensing approaches described earlier in this report.

The extent that LBT may be included in this process depends upon the need to use LBT to develop sufficient test data to gain approval of the EMs. As identified above, 10 CFR §52.47(b)(2)(i)(A) allows that, if sufficient data exists at the time of Design Certification (DC) approval to adequately validate and verify (V&V) the EMs used in the DC documentation, additional prototype (e.g., LBT) tests are not required. The same logic would apply to the licensing of the NGNP.

20.6.4.3 POTENTIAL APPROACHES FOR LICENSING THE NGNP

The proposed approach for licensing NGNP with respect to LBT is to use the EMDAP process described in Reference 20.6.4-12 for each of the EMs included in the COL application and to determine what LBTs are required to gain approval of the EMs and subsequently approval of the COL.

During the implementation of EMDAP, Step 4 requires the development of Phenomena Identification and Ranking Tables (PIRTs) by an independent panel of experts. The results of the PIRTs are to identify 1) those phenomena that need to be included in the relevant EMs and 2) the extent of the test data required to adequately V&V the model used to analyze the phenomena during the transients and accidents addressed in the application documentation. Currently a broad group of international organizations including experts from vendors, regulators, national laboratories and academia are working together to develop a comprehensive set of PIRTs applicable to all HTGR designs.

Subsequent EMDAP steps then review the existing separate effects tests (SETs) and integral effects tests (IETs) to assess what additional data, if any, is required to adequately support the EM V&V. LBT can then be considered as one means of generating any needed additional data. Therefore, the initial extent of LBT will only be what is necessary to gain regulatory approval of the EMs used to perform the licensing analyses for the NGNP. Accident conditions would only be tested to the extent that the test transients 1) do not adversely impact the safety of the plant and 2) do not cause any significant economic damage to the plant. Extrapolation of the plant response to limiting accident conditions would then be analytically determined by the approved EMs. This may require the development of data from other separate effect and integral effect test facilities, as necessary, to assure the various EMs can adequately determine transient/accident consequences.

In an attempt to limit the need for LBT based tests, numerous on-going SET and IET programs are currently underway with the goal of developing adequate data to validate EMs for all of the NGNP engineering phenomena important to safety.

Examples of large scale test facilities include the PBMR (Pty) Ltd Helium Test Facility (HTF) that provides steady state and transient tests of functionality in the operating helium environment, and the PBMR (Pty) Ltd Heat Transfer Test Facility that will determine the heat transfer properties of packed graphite pebble beds with heat generation under various cooling conditions.

In addition, numerous part scale test programs have been completed or are underway to provide data that will support the validation of design assumptions and safety codes. These part scale tests include:

1. The PBMR (Pty) Ltd Pebble Bed Micro-model Test Facility that provides scaled system response of PBMR to transients and accidents.
2. The ASTRA critical facility provides experimental investigation of neutronics characteristics of a reactor with geometrical characteristics similar to the PBMR reactor.
3. The NACOK facility used to study air ingress transients.
4. The PBMR (Pty) Ltd Plate-out Test Facility consisting of an Experimental Plate-out Loop (EPOL) and an Isopiestic Plate-out Facility (IPOF).
5. TRISO fuel data from existing test facilities in Germany and other planned tests.

In addition, there is the potential that other gas cooled reactor plants could be in operation before completion of the NGNP Commercial plant DC effort. In this case, operating plant data could be used to help support approval of the EMs used in the DC documentation.

20.6.4.4 ISSUES FOR RESOLUTION

The LBT concept for NGNP will initially be limited to proposed separate effects and/or integrated testing on the plant, as constructed, for the purpose of generating safety performance, margin analysis, uncertainty management, etc. information 1) that has not been established by previous separate effects or prototype testing and 2) that is required to support Design Certification. Therefore, the initial issue that needs to be resolved is the determination of what additional test data is needed to adequately support approval of the EMs used in the COL documentation. This will require completion of the EMDAP PIRTs and comparison of the identified data requirements against the existing and expected data from existing and proposed test facilities.

Implementation of the EMDAP process would result in the determination of what additional data is required to adequately support approval of the EMs. Then a cost/benefit/risk study, in conjunction with the NRC review process, would be performed to determine what data should be generated by LBT. The studies would evaluate the tradeoff with required instrumentation, design changes, and potentially power level limitations. The end result would be to use LBT selectively only when justified by risks and benefits. Finally, any LBT would more than likely be included in the ITAAC, increasing the steps toward achieving full power operation.

The current evaluations of post-accident consequences discussed in Reference 20.6.4-13 are limited because demonstration of robust performance of the TRISO fuel pellets has not yet been verified by additional tests or experiments. While relevant fuel tests are planned, results will not be available for several years. Consequently, there are uncertainties in the fuel performance and source term to be used. Therefore, LBT with appropriate initial limits on operating power could be used to help generate fuel performance data needed to support full power operation.

Reference 20.6.4-13 identifies that the greatest equipment development uncertainties for the co-generation plant are associated with the intermediate heat exchanger between the reactor and secondary coolant system, and the hot gas valves that isolate the nuclear side from the power conversion system (for an indirect cycle) or the hydrogen production plant in case of disturbances. While LBT could be used to confirm operation of these components, scaled prototype tests could be performed to limit the risk of unacceptable performance of the full size components.

While the NGNP reactor does not necessarily need to be instrumented and initially operated to fully support LBT, the high-level function and requirements in Sections 3.5 (Include Provisions for Future Testing) and 3.6 (Enable Demonstration of Energy Products and Processes) of Ref. 20.6.4-14 will probably require equivalent instrumentation. This instrumentation is required to ensure that sufficient flexibility is provided in the design to allow future research and development testing programs to be conducted. In addition, with appropriate instrumentation, the NGNP can provide data needed for verification and validation of codes used for licensing and design certification of future evolutionary VHTR plants.

20.6.4.5 RECOMMENDATIONS

It is recommended that License-By-Test be used as warranted by expected benefits to achieve (1) timely full-power operation of the NGNP and (2) design certification for the follow-on NGNP Commercial plant.

20.6.5 LICENSING OF AN INTEGRATED NUCLEAR POWER / HYDROGEN PLANT

20.6.5.1 BACKGROUND

While the 2005 Energy Act (Ref. 20.6.5-1) directs the DOE to manage construction of a facility that will enable research and development on advanced reactors of the type selected and on alternative approaches for reactor-based production of hydrogen, it further directs that 1) "the Nuclear Regulatory Commission shall have licensing and regulatory authority for any reactor authorized under this subsection" and 2) "the Commission shall give priority to the licensing of a utilization facility that is collocated with a hydrogen production facility." The Act does not directly address the regulatory criteria that apply to the hydrogen production facility. This could give the NRC some latitude in implementing the OSHA Occupational Safety and Health Standards (e.g., 1910 Subpart H - Hazardous Materials; 1910.103 – Hydrogen, 1910.106 - Flammable and Combustible Liquids, etc.) that would apply to a hydrogen plant.

Commissioner Jeffrey S. Merrifield provided the following overview of the NRC's responsibilities in a presentation at the 3rd International Topical Meeting on High Temperature Reactor Technology (Ref. 20.6.5-2):

> "The case of hydrogen production, however, seems to be a somewhat more complicated question. For many members of the U.S. public, the use of hydrogen is equated with the Hindenburg dirigible disaster dating back to 1937. Now I am not suggesting that this is my view. I am quite aware that we have moved far forward in our ability to safely produce and utilize hydrogen. Nonetheless, when the use of nuclear power is tied to hydrogen production, one receives the predictable questions regarding the safety of having these two technologies side by side, and one cannot disregard the fact that hydrogen production facilities have equally stringent safety and fire protection requirements. The fact remains, however, that nuclear safety regulators such as myself will have to ask the hard questions regarding these issues because our public and our Congressional and Parliamentary overseers will clearly expect us to answer these issues in a clear manner and with a sound technical basis."

While there are no NRC regulations directly applicable to a cogeneration plant, there is some precedent on licensing such a plant. Although it was not built, licensing of the Midland Plant Units 1 &2 did progress through the generation of the NRC Safety Evaluation Report (SER). The Midland plant was a dual function plant, providing both electrical power and process steam. The process steam system (PSS) would deliver tertiary steam to Dow Chemical Co. at the site boundary by utilizing steam to steam re-boilers that separate the secondary and tertiary steam systems. Process steam would therefore be separated from the primary system by two stages of heat exchangers. As an additional safety system a process steam radiation monitoring program was to be implemented.

In the SER (Ref. 20.6.5-3) the staff stated:

"The monitoring program provides assurance of timely detection of statistically significant radioactivity in the process steam.

"The process steam is produced in a tertiary heat exchanger. Therefore, there are three physical barriers between the radioactivity in the core and the process steam: the fuel rod cladding, the steam generator, and the tertiary heat exchanger. It is considered unlikely that all three barriers will simultaneously fail leading to measurable amounts of radioactivity in the process steam. From the system description in the FSAR, the staff has determined that the system is capable of detecting and annunciating the presence of reactor-produced radioactivity in the process steam system and discontinuing supplying the process steam to Dow prior to exceeding the limits specified in the Technical Specifications.

"The applicant has evaluated the effect of the process steam system on each of the Chapter 15 events. The staff concurs with the applicant's assessment that the process steam will not adversely affect the reactor system."

Basically the staff approved the Midland plant application by determining that:

1) The reactor system would meet appropriate Technical Specifications regarding impact on the DOW process heat system, and

2) The DOW process steam system would not adversely affect the reactor system.

Currently, the NRC recognizes a similar generic approach in NUREG-1860 (Reference 20.6.5-4) when it provides a comparison of the proposed technology-neutral framework against IAEA Safety Standard NS-R-1:

"Power Plants used for cogeneration, heat generation or desalination shall be designed to prevent radioactive material from the nuclear plant to the desalination or district heating unit under all conditions."

Even though NUREG-1860 also acknowledges that this approach is not included in the proposed framework, still a similar generic licensing process could be applied to the hydrogen plant part of the integrated licensing process.

In the process of establishing licensing regulations and criteria for an integrated nuclear power / hydrogen plant, the NRC could take advantage of similar licensing processes already developed by other international regulatory bodies.

In Reference 20.6.5-5 the German Reactor Safety Commission describes the "special experiences (they) have gained in the project "AVR Reconstruction for Process Heat Applications Demonstration" with heat in the form of hot helium of 950 ˚C with the result of a positive recommendation for licensing, given by an Advisory Council, consisting of members of the Reactor Safety Commission, established by the Federal Minister of the Interior of the Federal Republic of Germany.

A paper titled "European Research and Development on HTGR Process Heat Applications" presented in Reference 20.6.5-6, describes the current commissioning phase in Japan of the High Temperature Test Reactor (HTTR) combined nuclear/chemical facility. The paper describes scaled/prototype testing and other preparatory tests that include the examination of the permeation behavior of hydrogen and tritium. It also includes an approach to a safety analysis for the HTTR connected with a steam reformer unit that gives special attention to the potential development of a detonation pressure wave as the result of an inadvertent release of natural gas from the LNG storage tank into the environment and its ignition. The JAERI calculations simulating the impact of a methane vapor cloud explosion on the HTTR have shown that no significant influence on the reactor building is expected.

Both of these programs would provide the NRC the opportunity to include international cooperation in developing the licensing process for the integrated nuclear power / hydrogen plant. Involving German and Japanese regulators in this licensing process would be a natural extension of the NRC sponsored Multinational Design Evaluation Program (MDEP).

The remainder of this section provides background information on the unique aspects of siting a hydrogen production plant in close proximity to an HTGR.

Chapter 8, "Safety Risks of a Large-Scale Hydrogen Application" of Reference 20.6.5-7 provides technical information on the atmospheric dispersion of hydrogen and the combustion behavior of hydrogen culminating in the determination of the required distance between the location of a gas leakage and the object to be protected, which takes account of the evolving flammable atmosphere as well as of the pressure and heat wave resulting from a possible ignition. Figure 8-14 of this report provides the safe distance versus amount of hydrogen storage from a variety of regulatory sources.

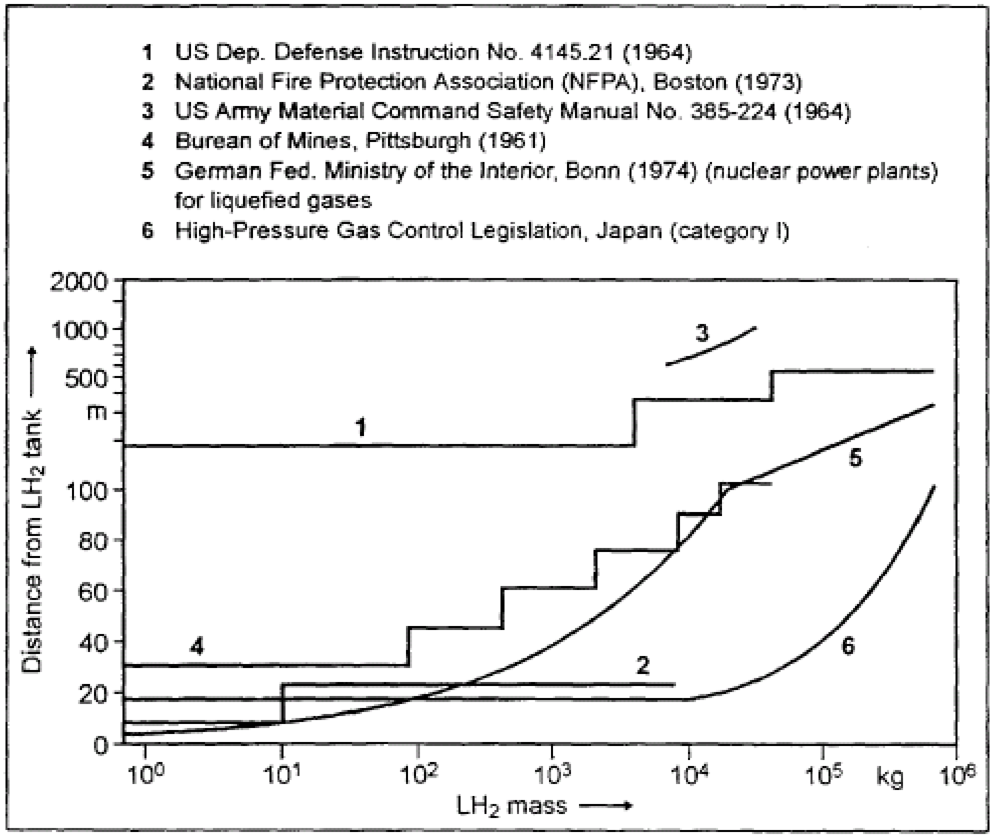

Fig. 8-14: Safety distances diagram (please note the scale change on the ordinate!) #1 and #3 from [40], #2 and #6 from [4], #4 from [130], #5 from [36]

Figure 20.6.5-1: Safety Distances Diagram
[Figure 8-14, imported from report #IAEA-TECDOC-1085, Ref. 20.6.5-7]

As seen from this figure, a distance in the range of 300m to 500m essentially bounds all of the regulations for storage of up to ~10^4 kg of hydrogen. The distance can be significantly reduced for lower mass storage of hydrogen. In addition, mitigating design features (e.g., blast shields, igniters, explosion resistant storage tanks, etc.) could reduce the required safe distance.

To take advantage of the reduced distance would require regulatory approval of an appropriate safety distance methodology and a plant design that potentially moved the storage of large quantities of hydrogen to a more remote location, leaving only the in-process hydrogen and a smaller storage facility at the hydrogen plant site. An example of a proposed methodology that would justify a separation distance of approximately 100 m is provided in INL/EXT-05-00137, "Separation Requirements for a Hydrogen Production Plant and High-Temperature Nuclear Reactor" (Ref. 20.6.5-8)

Section 8.7 "Safety and Risk Assessment for Hydrogen Containing Systems", of Reference 20.6.5-7, provides a discussion of the PHOEBUS safety concept that includes the determination of conceivable accident scenarios and the description of a hydrogen gas warning system.

Section 10.3 "Concept of a Nuclear Hydrogen Economy", of Reference 20.6.5-7, discusses the important points that "further safety-related work is necessary on tritium and hydrogen transportation in the process heat system; process gas cloud explosion hazards inside/outside the reactor building, response of containment structures, … improvement of material and structural design by examination of long-term creep behavior and high-temperature corrosion properties of reference materials and the new super-alloys."

Section 3.2.3 of Reference 20.6.5-7, "Hydrogen and Lithium Behavior," states the following:

> "Under conditions of nuclear process heat plants operation, hydrogen and tritium exhibit high mobility at 950 ºC causing permeation through the walls of the heat exchanging components. Tritium produced in the primary circuit could permeate into the product gas in the secondary circuit and eventually create a radiation problem to the consumer. In the opposite direction, hydrogen from the secondary circuit could diffuse into the primary circuit to cause corrosion of the fuel elements and of the graphite liner in the core with the formation of methane."
>
> …
>
> "Possibilities of reducing the permeation streams are the formation of oxide layers or, as active measures, an improved design of the gas purification system or even the construction of an intermediate circuit between the primary and secondary circuit to serve as a sink for both hydrogen and tritium."

The discussion in Section 3.2.6 of Reference 20.6.5-7 with respect to gas explosion research states that "precautions must be taken to minimize the risk of a fire or gas explosion such as avoidance of explosive gas ingress, proper detection devices, inerting, sufficient safety distances, inerting, appropriate layout of secondary coolant boundary, explosion-proofed wall, plant isolation valves."

In addition, "hydrogen has long been recognized to have a deleterious effect on some metals by changing their physical properties [e.g., embrittlement]." [Reference 20.6.5-7, Section 8.3.1.1]

The above information provides insight on the unique aspects of siting a hydrogen production plant in close proximity to an HTGR that should be considered in conjunction with implementing the OSHA Occupational Safety and Health Standards (e.g., 1910 Subpart H - Hazardous Materials; 1910.103 – Hydrogen, 1910.106 - Flammable and Combustible Liquids, etc.) that would apply to a hydrogen plant. Other safety issues will be identified during the design and licensing process through, for example, the identification of LBEs as stated in Sections 20.6.5.3 and 20.6.5.4.

20.6.5.2 LICENSING PROCESSES

The licensing processes for an integrated nuclear power / hydrogen plant would start with the regulations that apply to the nuclear plant (e.g., 10 CFR Parts 50, 52 and/or 53, including all the embedded regulations) and the hydrogen plant (e.g., 1910 Subpart H - Hazardous Materials; 1910.103 – Hydrogen, 1910.106 - Flammable and Combustible Liquids, etc.). Based on the integrated plant design submitted, the initial licensing processes would be expanded to identify what part of the regulations for one of the plants would have to be applied to the other plant.

Examples of potential crossover regulations include:

1. If operator actions over a period of time are required to safely shutdown the hydrogen plant following an accident resulting in radiation release, nuclear regulatory criteria for control room operation may apply to the hydrogen plant. In addition, the hydrogen plant design may have to include capability to safely shut down the plant from the protected control room.

2. Depending upon the accident scenarios evaluated for the hydrogen plant, releases of hydrogen and subsequent potential conflagrations, explosions, etc. may require specific design features for the protection of nuclear plant personnel.

3. Plant layouts/design requirements must be identified (e.g., turbine orientation, barriers/blast shields, storage tank location/size, plant operating limits, required separation/isolation of interfacing systems, etc.) that assure failures in these structures, systems and components do not adversely impact either plant. These regulatory requirements should be established for the NGNP in a manner such that they become precedents for the NGNP Commercial plant design and the corresponding Design Certification program.

4. Limitations on radioactive releases from the reactor plant to the hydrogen plant through the interfacing systems (implemented by specified design criteria on the interfacing systems and related Technical Specifications) may be more restrictive than current nuclear regulations due to requirements to meet OSHA/EPA standards.

20.6.5.3 POTENTIAL APPROACHES FOR LICENSING THE NGNP

Licensing of an integrated nuclear power / hydrogen plant must address the regulations that apply to each plant plus consideration of potential interactions between the plants. Essentially each plant design should initially meet its own regulatory criteria. In addition, the transient and accident consequences of potential interfacing events should have sufficiently limited impact on the other plant so that the combined regulations for both plants are met. Finally, depending on normal plant operation, safety system operation, transient/accident consequences and required operator actions, some regulatory criteria for one plant may apply to the other plant. This will require significant cooperation and clarification among various regulatory authorities (e.g., NRC, OSHA, DOE, etc.). The basic licensing approaches for the hydrogen plant should be obtained from the relevant regulatory authorities (e.g., DOE, OSHA, etc.) that have previously licensed similar plants.

The approach to establish (1) the potential interfacing transients and accidents that need to be addressed as part of the basic licensing approach for each plant and (2) the potential crossover regulations should start with the determination of the interfacing transient and accidents that must be considered in the licensing processes. The determination of these licensing basis events (LBEs) would follow the parallel approaches described in References 20.6.5-8 and 20.6.5-9. Basically, Failure Modes and Effects Analysis (FMEA) studies would be used to identify potential initiating events and system failures in the interfacing systems. Then, using the plant PRA to determine risk-informed sequences, Fault Trees and Event Trees would be used to define potential LBEs versus probability of occurrence.

Then the EMDAP from Regulatory Guide 1.203 (Ref. 20.6.5-10) would be implemented to develop, validate and verify one or more Evaluation Models (EMs) to analyze the limiting transients and accidents. In the process implementing EMDAP for these LBEs, a Phenomena Identification and Ranking Table (PIRT) would be generated to identify what important phenomena need to be included in the EM. The PIRT will also identify the scope of required test data (generated by scale model tests or potential verification via license by test) needed to adequately support the V&V of the integrated system EM(s). The resulting approved EM(s) will then be used to analyze the appropriate interfacing LBEs to show that all of the acceptance criteria for the combined regulations are met.

20.6.5.4 ISSUES FOR RESOLUTION

The major issue of licensing an integrated nuclear power / hydrogen plant will be the resolution of the various regulations that apply to each plant, plus the identification and resolution of the cross-over regulations that will address any potential interactions between the plants. This will require significant cooperation and clarification among various regulatory authorities (e.g., NRC, OSHA, DOE, etc.). The NGNP project will support this effort by identifying potential cross-over regulations and proposing related acceptance criteria.

The licensing safety analyses will have to address integrated safety issues. Identification of integrated safety issues will be based on applying the methods of References 20.6.5-8 and 20.6.5-9 to identify relevant, integrated LBEs and then evaluating these LBEs by implementing EMDAP from Regulatory Guide 1.203. A unique integrated safety issue must address the separation/isolation of the nuclear power source from the hydrogen production facility. This will require the development of an appropriate cross-over regulation and associated EMs/acceptance criteria based on the methodology provided in Reference 20.6.5-8 and/or the regulations identified in Figure 8-14 of Reference 20.6.5-7

Section 20.6.4.4 notes that the greatest equipment development uncertainties for the co-generation plant are associated with the intermediate heat exchanger (IHX), and the hot gas valves that isolate the nuclear side from the power conversion system. The significant licensing issues associated with these components are 1) the limitation of potential tritium migration across the IHX interface so that the maximum amount of tritium released from the integrated NGNP facilities or found in drinking water does not exceed EPA standards and 2) the ability of the hot gas valves to adequately isolate the hydrogen plant, thereby ensuring that worker and public dose limits for the integrated NGNP and hydrogen production facilities do not exceed NRC regulatory limits.

Finally, as indicated in Section 20.6.5.2, the design characteristics of the NGNP that are specified to resolve regulatory issues should be developed in a manner to establish design and licensing precedents for the NGNP Commercial plant.

20.6.5.5 RECOMMENDATIONS

The recommendations are to:

- Establish the specific licensing requirements for the chosen hydrogen production design with conceptual separation distance and facility interactions that establish precedents for the NGNP Commercial plant, and
- Further develop the licensing approaches and issues described above (Sections 20.6.5.3 and 20.6.5.4) as the NGNP design is developed.

20.6.6 METHOD FOR INTEGRATION OF PROBABILISTIC RISK ASSESSMENT (PRA) TECHNIQUES DURING DESIGN PHASE

20.6.6.1 BACKGROUND

Aspects of the previous HTGR/MHTGR PRA development efforts have been incorporated into the current PRA effort for the PBMR and are considered applicable to the NGNP. It is recommended that PRA methods similar to those described in the PBMR (Pty) Ltd pre-application program be adopted for the NGNP.

The following descriptions review the history of the previous HTGR, MHTGR, and PBMR (Pty) Ltd PRA efforts and summarize their conclusions.

NRC policy, stated in the mid-1980's, is that advanced reactor designs should be more risk-informed. An HTGR risk-informed licensing approach was initially presented to the NRC as part of the MHTGR design review which was presented to the NRC in the late 1980's and early 1990's. The results of the extensive NRC review of the MHTGR design are documented in NUREG-1338.

Exelon Generation Company picked up on the expanding use of risk-methods and described a licensing approach for the PBMR design in its COL pre-application interactions with the NRC. The results of NRC's review of Exelon's licensing proposal are documented in a March 26, 2002 letter from the NRC (Ref 20.6.6-10).

The current PBMR design expands on the MHTGR and Exelon work, filling in areas where questions were raised and further refining the methods. Additionally, PBMR (Pty) Ltd is able to take advantage of several developments since the Exelon submittal that have occurred in the understanding and application of risk methods. Notably, a number of policy issues related to the use of PRA methods were presented to the Commission in SECY-03-0047. The PBMR (Pty) Ltd approach, summarized below, provides an up-to-date perspective on PRA developments and expectations, and reflects the benefit of white papers submitted to the NRC (Refs. 20.6.6-6, 20.6.6-7, 20.6.6-8, and 20.6.6-9) and meetings and discussions with NRC staff in the 2005 – 2006 time-frame.

20.6.6.2 INTEGRATION OF PRA TECHNIQUES FOR THE PBMR DESIGN

A full-scope, all-modes PRA is being developed that allows for a logical and structured method to evaluate the overall safety characteristics of the PBMR plant and provide input to the risk-informed evaluation and selection of design features." It is recommended that a similar PRA method be implemented for the NGNP. The following aspects comprise the PRA:

- The potential sources of release of radioactive material, including the sources in the reactor core and Main Power System (MPS), process systems, and Fuel Handling and Storage System (FHSS).

- All planned operating and shutdown modes, including plant configurations expected for planned maintenance, tests and inspections.

- A full range of potential causes of initiating events, including internal plant hardware failures, human operator and staff errors, internal plant hazards such as internal fires and floods, and external plant hazards such as seismic events, transportation accidents and any nearby industrial facility accidents.

- Event sequences that cover a reasonably complete set of combinations of failures and successes of SSCs and operator actions in the performance of safety functions. These event sequences will be defined in sufficient detail to characterize mechanistic source terms and off-site radiological consequences comparable to an LWR Level 3 PRA.

- Quantification of the frequencies and radiological consequences of each of the significant event sequences modeled in the PRA. This quantification includes mean point estimates and an appropriate quantification of uncertainty in the form of uncertainty probability distributions that account for quantifiable sources of uncertainty in the accident frequencies, mechanistic source terms, and off-site radiological consequences. Additionally, an appropriate set of sensitivity analyses will be performed to envelope sources of uncertainty that are not quantifiable.

- Recognizing the modular aspects of the PBMR design, the PRA will define event sequences that impact reactor modules independently, as well as those that impact two or more reactor modules concurrently. The frequencies will be calculated on a per-plant-year basis, and the consequences will consider the number of reactor modules and sources that are involved in the definition of the mechanistic source terms.

- In order to support the development of regulatory design criteria, the PRA will be capable of evaluating the cause and effect relationships between design characteristics and risk, and of supporting a structured evaluation of sensitivities to examine the risk impact of adding and removing selected design capabilities, and setting and adjusting SSC reliability requirements.

20.6.6.2.1 Elements of the PRA

The scope of the PRA to support this risk-informed approach is comprehensive, complete, and comparable to a full-scope, all modes, Level 3 PRA for an LWR covering a full set of internal and external events. However, due to the inherent features of the PBMR design, the approach to modeling initiating events and event sequences can be simplified in terms of size and complexity in comparison to that in an LWR PRA model. These simplifications will not diminish the quality of the PRA, and will facilitate the capability to perform effective independent peer reviews as needed to meet ASME and ANS PRA standards' peer review requirements.

The PBMR PRA has been organized into elements that are consistent with the way in which PRA elements have been defined in the ASME [Ref. 20.6.6-1] and ANS PRA Standards [Ref. 20.6.6-2, 20.6.6-3, and 20.6.6-4], and Regulatory Guide 1.200 [Ref. 20.6.6-5]. The PRA elements, which may be considered building blocks of the PRA models, are shown in Figure 20.6.6-1 and include:

- Definition of Plant Operating States
- Initiating Events Analysis
- Event Sequence Development
- Success Criteria Development
- Thermal and Fluid Flow Analysis
- Systems Analysis
- Data Analysis
- Human Reliability Analysis
- Internal Flooding Analysis
- Internal Fire Analysis
- Seismic Risk Analysis
- Other External Events Analysis
- Event Sequence Frequency Quantification
- Mechanistic Source Term Analysis
- Radiological Consequence Analysis
- Risk Integration and Interpretation of Results
- Peer Review

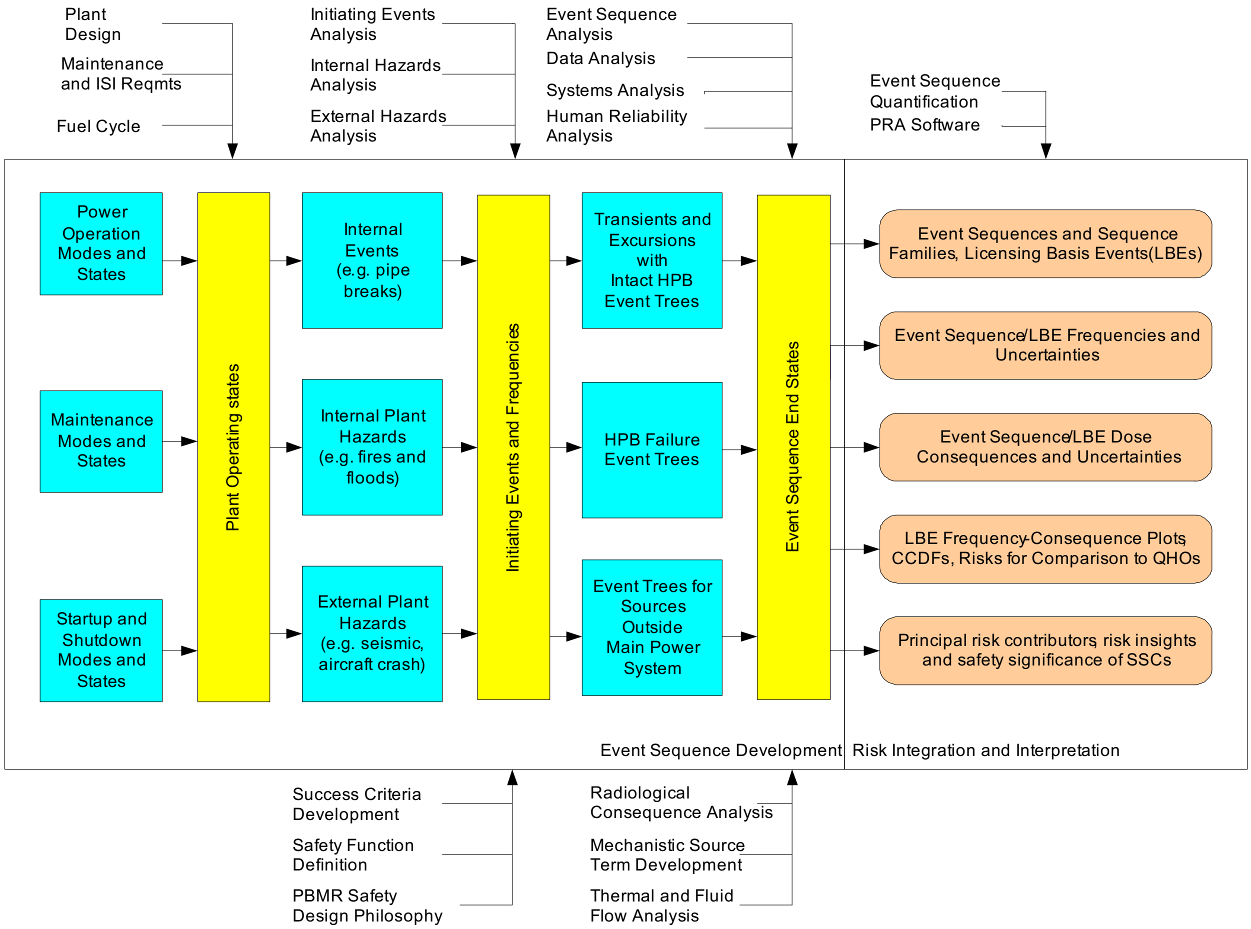

Figure 20.6.6-1: Overview of PBMR PRA Model Elements

In the development of the PBMR design, probabilistic and deterministic safety assessments have developed concurrently and in an integrated fashion. It is recommended that a similar development method be implemented for the NGNP. The safety design philosophy itself is rooted in deterministic safety assessment principles. Key design parameters such as the core size and shape, power density, reactor cavity configuration, the fuel particle design and manufacturing specification, are based on the principle of preventing core damage and large releases from the fuel using deterministic methods and means. Important aspects of the PBMR (Pty) Ltd safety design philosophy, such as the importance placed on inherent and passive means to implement safety functions, are based on sound deterministic design principles. The design calculations that were made to establish these parameters were based on a conservative deterministic engineering analysis for a set of enveloping events and boundary conditions, and in accordance with the defense-in-depth (DID) philosophy.

The systematic selection of initiating events and the deterministic analysis are performed in an integral fashion. The applicable knowledge that is available to support the selection of possible initiating events for both the probabilistic and deterministic safety analysis is applied for this purpose. This knowledge base is systematically developed by application of Failure Modes and Effects Analysis, Hazard and Operability (HAZOP) investigations, and by reviews of lists of events that have been considered for other HTGRs, GCRs and LWRs. The need for a systematic, comprehensive, and reproducible set of initiating events is viewed to be fundamental to both the probabilistic as well as the traditional deterministic approaches to the selection of LBEs.

20.6.6.2.2 Selection of Licensing Basis Events (LBEs)

The risk-informed licensing approach proposed for the PBMR includes the definition of Top Level Regulatory Criteria (TLRC) that provide frequency and dose limits for the LBEs, and in this respect determine what must be met for licensing approval. The selection of the LBEs answers the question of when the TLRC are to be met. The spectrum of potential accidental radioactive releases from the PBMR plant is divided into three regions of a scenario frequency versus consequence chart. The regions include those associated with:

- Anticipated Operational Occurrences (AOOs) are those conditions of plant operation which are expected to occur one or more times during the life of the plant. Current plants were licensed to operate for an initial 40-year period; however, with the advent of license renewal, operating licenses of conventional plants have been increased for some plants by 20-year increments. Therefore, a conservative value of 1 x 10^{-2} is used to establish the lower bound of the AOO region. For this region, 10 CFR Part 20 provides the applicable criteria, as it specifies the numerical guidance to assure that releases of radioactive material to unrestricted areas during normal reactor operations, including AOOs, are maintained As Low As Reasonably Achievable (ALARA).

- Design Basis Events (DBEs) encompass releases that are not expected to occur during the lifetime of a single nuclear power plant, but may be encountered during the lifetime of a population of nuclear power plants. Therefore, a value of 1 x 10^{-4} per plant-year is used to establish the lower bound of this region. For the DBE region, the 25 rem Total Effective Dose Equivalent (TEDE) criterion in 10 CFR §50.34a provides the quantitative dose guidance for accidental releases for siting a nuclear power plant to ensure that the surrounding population is adequately protected. The combination of the selected frequency limits and dose limits for the DBE region ensures that the NRC Safety Goal QHOs for individual risk of latent cancer fatality is met by several orders of magnitude for all event sequences within the DBE region.

- Beyond Design Basis Events (BDBEs) are improbable events that are not expected to occur during the lifetime of a large fleet of nuclear power plants. BDBEs are considered to assure that the risk to the public from low probability events is acceptable.

A composite frequency-consequence (F-C) chart depicting the three categories of LBEs for the PBMR plant is shown in Figure 20.6.6-2. The preliminary PRA results shown in the figure are from an earlier design for a 268 MWt PBMR (Pty) Ltd Demonstration Power Plant.

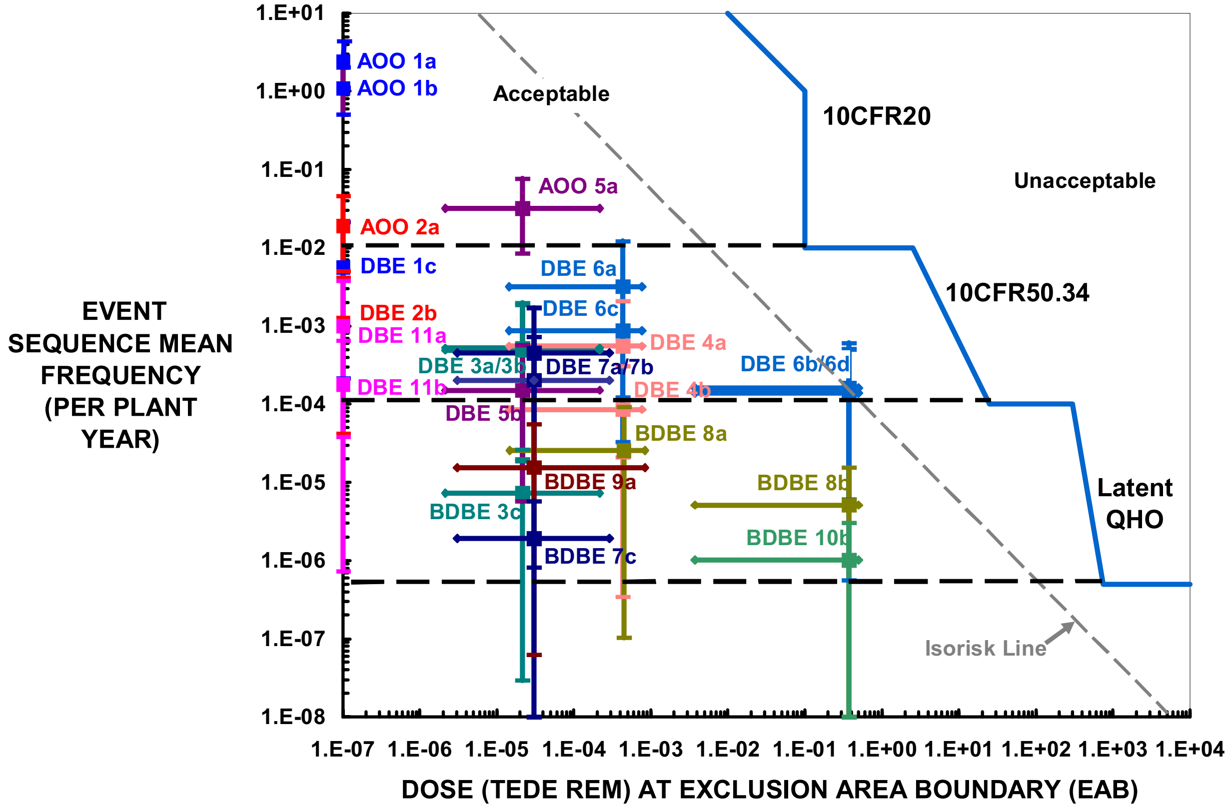

Figure 20.6.6-2: Frequency-Consequence Chart for all Three Categories of Licensing Basis Events

An examination of the entire frequency range and the identification of one or more of the TLRC as being applicable for each region provide assurance that the selected criteria are adequately established.

20.6.6.2.3 Safety Classification of Structures, Systems, and Components (SSCs)

Structures, Systems, and Components (SCCs) are classified relative to their safety significance to focus attention and resources on their design, construction, and operation commensurate with their safety significance. Safety functions needed to meet the TLRC are identified based on a review of the LBEs. Figure 20.6.6-3 illustrates the top-level functions with emphasis on the reactor sources, and includes functions needed for protection of both the public and on-site personnel.

As shown, the design includes functions for radionuclide retention within the fuel particles, fuel spheres, Helium Pressure Boundary (HPB), reactor building, and site. Not all of the functions in Figure 20.6.6-3 are required for each TLRC. Safety analyses have been performed to determine which are the required safety functions for the reactor sources, as identified as the minimum subset that is shaded, to keep the DBEs within the offsite dose limits of 10 CFR §50.34. The functions shown without shading are not required for public protection, but are included in the design to provide an element of defense-in-depth, and to meet user requirements for plant availability and investment protection. The required safety functions include those to:

- Maintain control of radionuclides
- Control heat generation (reactivity)
- Control heat removal
- Control chemical attack
- Maintain core and reactor vessel geometry
- Maintain reactor building structural integrity

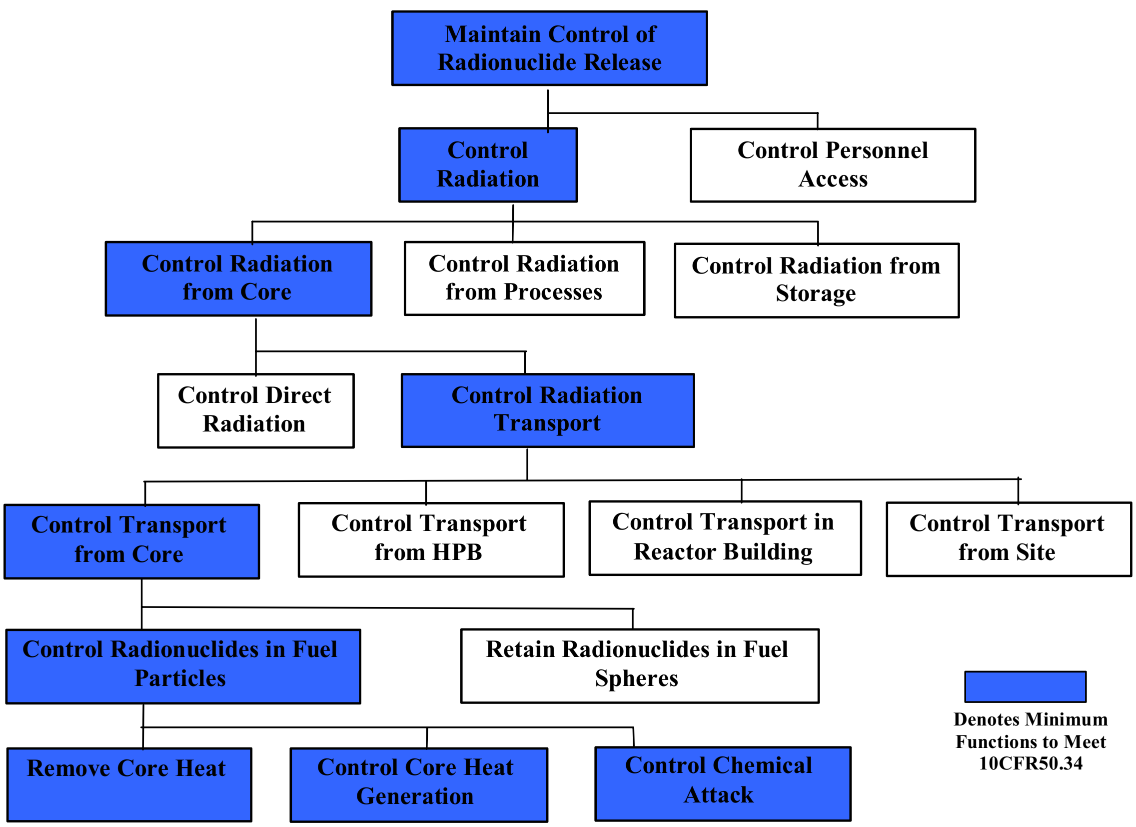

Figure 20.6.6-3: Safety Functions Needed during Licensing Basis Events to Meet Top Level Regulatory Criteria

Safety classification of SSCs is made in the context of the performance of SSCs with respect to specific safety functions during the spectrum of LBEs. The safety classification process and the corresponding special treatment control the frequencies and consequences of the

LBEs within the TLRC. The LBE frequencies are a function of the frequencies of initiating events from internal events, internal and external hazards, and the reliabilities and capabilities of the SSCs (including the operator) to prevent an initiating event from progressing to an accident, to mitigate the consequences of an accident, or both to prevent the former and mitigate the latter. In some cases, the initiating events are failures of SSCs themselves, in which case the reliability of each SSC in the prevention of the initiating event needs to be considered. In other cases, the initiating events represent challenges to an SSC in question, in which case the reliability of the SSC to perform a safety function in response to the initiating event needs to be considered. Finally, there are other cases in which the challenge to the SSC in question is defined by the combination of an initiating event and combinations of successes and failures of other SSCs in response to the initiating event. All of these cases are included in the PRA and represent the set of challenges presented to a specific SSC.

Special treatment requirements are also established for risk-significant SSCs. The purpose of special treatment requirements is twofold: First, special treatment helps ensure that the reliability and capability of each safety-related SSC are necessary and available in the prevention and mitigation of LBEs. The requirements for the reliability and capability of safety-related SSCs are derived from the frequencies and consequences of the LBEs that correspond to the SSCs in relation to the TLRC. Second, special treatment requirements increase the confidence that the safety-related SSCs will perform their safety functions in light of uncertainties about the reliabilities and capabilities of these SSCs. Hence, special treatment requirements help ensure that the frequencies and consequences of the LBEs fall within the TLRC as well as reduce the uncertainties about SSC reliability and performance in the context of the safety functions they perform in preventing and mitigating LBEs. The purpose of the special treatment is to increase the level of assurance that the SSCs will perform as predicted in the PRA under expected LBE conditions with the assessed uncertainties and in the DCA for conservative deterministic DBA conditions. As such, the special treatment requirements are an important element of defense-in-depth.

20.6.6.2.4 Defense-in-Depth Approach

Defense-in-depth (DID) is a crucial element of the overall safety of nuclear power plants. The principles of DID have been applied for the design, licensing, construction, operation and regulation of existing and advanced nuclear power plants.

PBMR (Pty) Ltd has adopted a risk-informed and performance-based approach to DID that recognizes three major elements: ***Plant Capability Defense-in-Depth***, ***Programmatic Defense-in-Depth***, and a ***Risk-Informed Evaluation of Defense-in-Depth***. This approach incorporates the concepts identified in previously published definitions of DID with clarifications that are necessary in order to apply these concepts to the PBMR. These three elements enable the examination of DID from different perspectives including those of:

- Designing the plant and specifying the capabilities of its SSCs in the performance of safety functions

- Defining the programs to ensure that the plant as-designed will be built and will operate safely throughout the lifetime of the plant and in a manner that preserves the DID capabilities intended in the design.
- Evaluating how the plant performs its safety functions in the prevention and mitigation of accidents in the context of a risk-informed and performance-based process in order to determine the adequacy and sufficiency of DID.

It is recognized that these elements of DID are not exclusive, but rather represent complementary and overlapping perspectives from which to apply the same underlying DID principles.

The current definitions and concepts of DID have evolved over a long period of time in designing and regulating the current fleet of light water reactor plants and have been modified in recent years to reflect the changes in philosophy brought about by risk-informed and performance-based regulation. The reason for having three major elements of DID is to organize our thinking in applying the underlying principles to the PBMR whose safety design philosophy differs in fundamental ways from that of an LWR. These elements are defined while recognizing that DID principles are applied in many areas of plant design, assurance, and regulation.

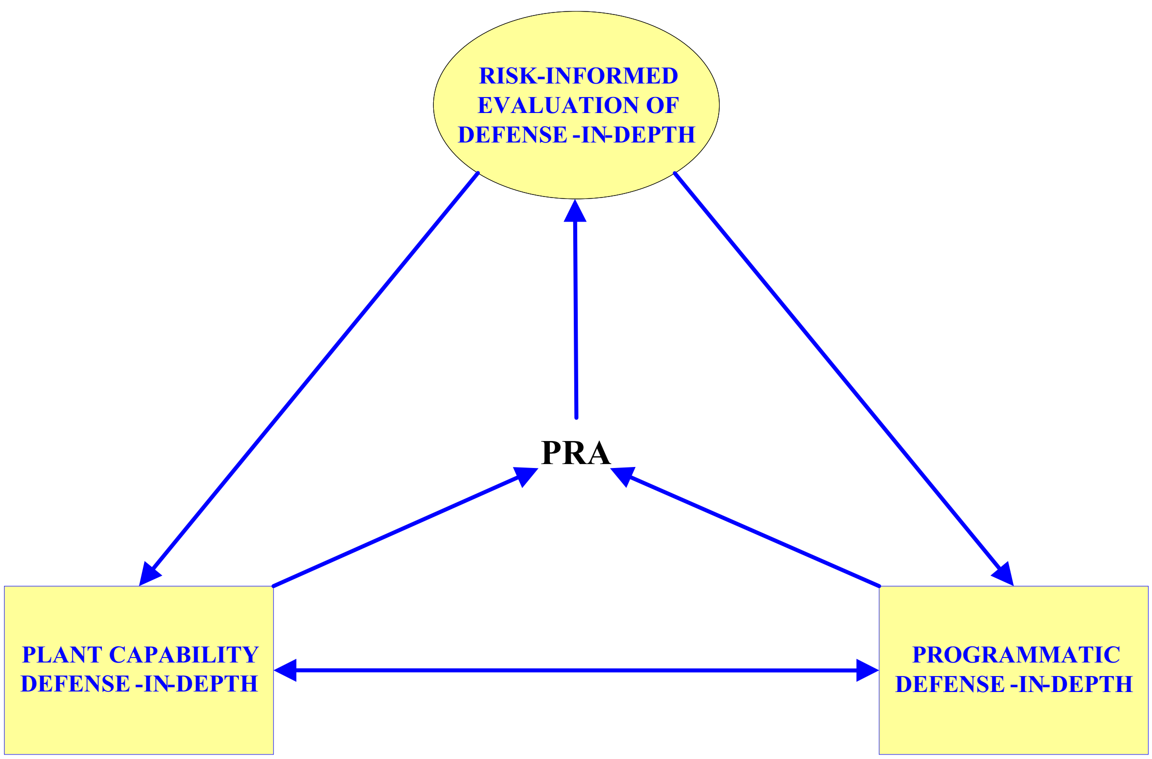

Figure 20.6.6-4: Elements of PBMR Approach to Defense-in-Depth

Plant Capability Defense-in-Depth reflects the decisions made by the designer to incorporate DID into the functional capability of the physical plant. These decisions include the use of multiple lines of defense and conservative design approaches for the barriers and SSCs performing safety functions associated with the prevention and mitigation of accidents. Thus, ***Plant Capability Defense-in-Depth*** includes the use of multiple barriers, diverse and redundant means to perform safety functions to protect the barriers, conservative design principles and safety margins, site selection, and other physical and tangible elements of the design that use multiple lines of defense and conservative design approaches to protect the public.

Programmatic Defense-in-Depth reflects the programmatic actions for designing, constructing, operating, testing, maintaining, and inspecting the plant so that there is a greater degree of assurance that the DID factored into the plant capabilities during the design stage is maintained throughout the life of the plant.

Risk-Informed Evaluation of Defense-in-depth is the structured use of information provided by the PRA to identify the roles of SSCs in the prevention and mitigation of accidents, to identify and evaluate uncertainties in the PRA results, to devise deterministic approaches to address these uncertainties, and to guide and provide risk insights to support deterministic judgments on the adequacy and sufficiency of DID. The event scenario models developed in the PRA provide an objective means of defining the roles that SSCs play in the prevention and mitigation of accidents.

An important aspect of the risk-informed evaluation of DID is a logical process for deciding the adequacy and sufficiency of the defense in depth reflected in the plant capabilities and assurance programs. Important feedback loops are shown in Figure 20.6.6-4 that represent the incorporation of risk insights into the development and enhancement of the plant capabilities and programs as the design and program development evolve.

In support of each of these elements of DID is a comprehensive PRA which helps ensure that all decision making in these processes are systematically evaluated in a comprehensive risk-informed manner. The PRA is based on the plant design and a specification of the capabilities of the plant SSCs in the performance of their functions, including the plant safety functions. The results of the PRA expose the characteristics of the ***Plant Capability Defense-in-Depth*** and are dependent on the safety margin and reliability of each SSC modeled in the PRA. The reliability of the SSCs responsible for the ***Plant Capability Defense-in-Depth*** is adequately assured by the elements of ***Programmatic Defense-in-Depth***. The PRA is called out separate from the DID elements in Figure 20.6.6-4 because information from the PRA is used to support the design, provide input to the formulation of process requirements, and provide information to evaluate the adequacy and sufficiency of these DID strategies. Conversely, the PRA itself provides a model of the plant capabilities and how the plant is operated and maintained under the programmatic controls, as part of the modeling and quantification of the scenarios. The PRA provides critical input to the identification and evaluation of the uncertainties that are addressed in the ***Plant Capability*** and ***Programmatic Defense-in-Depth*** elements. Hence the PRA is utilized in all the elements of the DID approach.

The PBMR (Pty) Ltd approach to DID is regarded as performance-based for several reasons. First, an objective perspective on the adequacy and sufficiency of DID is provided by comparing the frequencies and consequences of the LBEs and their uncertainties against the TLRC. Second, the plant capabilities include capabilities to monitor the plant performance against a set of parameters that confirm the safety operation of the plant. Third, the process of SSC safety classification and the definition of special treatment requirements provide a basis for monitoring the reliability and availability performance of the SSCs responsible for implementing safety functions. The level of special treatment applied to assure adequate reliability and capability of SSCs is commensurate with their risk significance. Hence, the approach is both performance-based and risk-informed.

20.6.6.3 SUMMARY

The following PRA concepts apply to NGNP:

- Previous HTGR/MHTGR PRA methodology is generally applicable. Since the 1980's, significant developments have been undertaken in advancing the use of PRA techniques and in establishing a set of recognized criteria for the preparation of comprehensive PRAs for advanced plant designs. For the PBMR design, these advancements were discussed with the NRC initially as part of the Exelon pre-application review in 2001-02 and currently as part of the PBMR design certification pre-application program.

- Four pre-application "white papers" on the development and use of a full-scope, all modes PRA for the PBMR (Pty) Ltd Design Certification pre-application program have been submitted to the NRC during 2006. It is anticipated that most or all of the information contained in these white papers would be applicable to the NGNP and would form the basis for development and implementation of the NGNP PRA.

- The PRA methodology developed for the PBMR plant can be a robust tool for determining the appropriate Licensing Basis Events (LBEs) for use in the design and safety analysis of the plant.

- This methodology can also be the basis for criteria for the safety classification of Structures, Systems, and Components (SSCs). The establishment of special treatment requirements for risk-significant SSCs then helps to ensure that the reliability and capability of each safety-related SSC are necessary and available in the prevention and mitigation of LBEs.

- A structured approach to evaluating the adequacy of Defense-in-Depth also aids in "tying together" the elements of a properly developed PRA with the safety design capability and program assurance elements of the plant design.

20.6.6.4 ISSUES FOR RESOLUTION

The development and implementation of PRA methods will be continued as part of the PBMR (Pty) Ltd pre-application meetings with the NRC. Therefore, at this time, there are no specific “issues for resolution” related to the integration of PRA methods into the design process for the NGNP.

20.6.6.5 RECOMMENDATIONS

It is recommended that methods similar to those for the PBMR (Pty) Ltd PRA program be adopted for the NGNP. The following approach is recommended:

- Build on PBMR (Pty) Ltd–NRC pre-application interactions, including risk-informed, performance-based methods, including:
  - Adoption of the PRA methodology for HTGR/MHTGR/PBMR to the NGNP to the maximum degree possible, including selection of LBEs, classification of SSCs, and implementation of the DID principles.
  - Build on specific PRA system models for HTGR/MHTGR/PBMR wherever model similarities between the plants allow. Develop new models that may be needed for the NGNP.

20.6.7 EPA/STATE PERMITS FOR INTEGRATED NUCLEAR POWER / HYDROGEN PLANT

20.6.7.1 BACKGROUND

This section highlights both the EPA and the Idaho Department of Environmental Quality (IDEQ) environmental permitting process for the NGNP. The environmental siting regulations for NGNP can be divided into federal and state requirements. The federal regulations are covered by the EPA and are implemented in conjunction with Environmental Policy Act (NEPA) and 10 CFR 51 compliance, while state regulations are set by the IDEQ for the NGNP. Local permits will likely be obtained from the INL infrastructure organization. Subsequent NGNP commercial plants would fall under state regulations of the state where the plant is sited.

20.6.7.2 SUMMARY

The NGNP is a demonstration of an advanced gas-cooled nuclear power reactor at a site on the reservation containing the Idaho National Laboratory (INL). The NGNP Environmental Permitting Plan (EPP) contains the philosophy, strategy, and schedule for obtaining the Federal, State, and local permits for the NGNP throughout the design process, prior to construction, and prior to operations. It is intended to be a living document which will guide the design team to design the facility to minimize the generation of radioactive and hazardous wastes, to minimize the releases of gaseous pollutants and liquid effluents, and to ensure that sensitive environmental habitats (e.g., wetlands) and threatened/endangered species are not significantly impacted. In this regard, actions and issues which will need to be resolved in the development of a complete Environmental Permitting Plan are listed in Table 20.6.7-1.

The EPP will ensure that permits that are required prior to procurement of construction materials and the commencement of construction will be obtained in a timely manner to avoid negative schedule impacts on the overall schedule. The EPP will also ensure that environmental permits required prior to operations will be obtained in a timely manner to avoid the situation of having the facility completed and ready for operation, but having these permits as a constraint on such operation. Experience with permitting for the Prototype is expected to be at least partly applicable to the first commercial plant, with adjustments as needed to handle differences in state and local requirements.

Table 20.6.7-1: Environmental Permitting Plan – Issues To Be Resolved

Item	Description
01	Environmental Permitting Status Summary (EPSS) will be developed when sufficient schedule information is available.
02	Establish whether an environmental compliance Basis of Design (BOD) will be developed.
03	Verify the extent of INL NGNP environmental controls.
04	Establish to what extent Environmental Protection Agency (EPA) has granted jurisdiction to Idaho Department of Environmental Quality (IDEQ) and in turn, to what extent IDEQ has granted jurisdiction directly to the INL environmental permitting authority.
05	Tie-ins to the existing INL infrastructure is the best option, but the availability and proximity of Central Sanitary Wastewater Treatment Facility (CSWTF) and domestic water lines needs to be established.
06	Need to identify the Air Quality Control Region (ACQR) that the facility will be located in and establish whether it is in an attainment area.
07	Need to establish all non-radionuclide gaseous emission points and emission levels.
08	Need to closely examine IDEQ air quality enabling regulations relative to potential emissions to determine the type of construction air permit needed.
09	Verify how IDEQ addresses diesel generator size and operations exemptions.
10	Establish whether a Concrete Batch Plant will be used during construction.
11	A routine release calculation using guidance in 40 CFR 61 and the CAP88-PC atmospheric transport and dispersion model is required to establish the basis for an exemption from a National Emission Standards for Hazardous Air Pollutants (NESHAP) Construction Permit.
12	Compare expected stored chemicals with the provisions of 40 CFR 302, 40 CFR 355, 40 CFR 68, and 29 CFR1910.119 to determine if a Risk Management Plan (RMP) and an Occupational Safety and Health Administration (OSHA) Process Safety Management (PSM) Plan are required.

Item	Description
13	The gases in the Gas Storage Area (GSA), or equivalent, need to be identified to ensure that no permits are required.
14	All liquid effluent waste streams need to be identified to finalize the National Pollutant Discharge Elimination System (NPDES) permitting requirements.
15	Need to establish if a tank to collect all low-level waste streams is part of the conceptual design. If so, need to identify whether it will be piped to another liquid effluent disposal stream.
16	Need to determine if storm water will be the only liquid effluent stream that will be routed to the INL NGNP outfall.
17	Need to ensure that the design routes external Heating Ventilation and Air Conditioning (HVAC) condensate to the National Pollutant Discharge Elimination System (CSWTF) and not to the storm water discharge.
18	Storm Water discharge piping and outfall location needs to be established.
19	Need to establish whether there will be no process water discharges to an NPDES outfall.
20	Need to establish whether 5 acres of total land area will be disturbed.
21	Need to determine if IDEQ permits a concrete washout impoundment basin with a No Discharge (ND) NPDES Permit or within the purview of a Storm Water Management Pollution Protection Plan (SWPPP).
22	Need to determine if the total quantity of buried diesel fuel in non- Underground Storage Tank (UST) vaults exceeds 42,000 pounds relative to Spill Prevention Control and Countermeasures (SPCC) Plan requirements.
23	Determine if an INL Site-wide SPCC Plan exists and whether it can accommodate the INL NGNP SPCC controls and countermeasures.
24	Need to establish whether the facility disturbs existing wetlands.
25	Need to establish whether a tie-in to the existing INL domestic water system is planned.

Item	Description
26	Need to determine if IDEQ has given INL infrastructure the authority for domestic water permitting at the INL site.
27	Need to establish whether buried diesel fuel tanks are double-walled or are vaults.
29	Need to determine groundwater level below plant grade to determine if a Groundwater Monitoring Plan is required.
30	Need to establish whether all INL NGNP wastes will be managed under the INL waste management infrastructure.
31	Need to establish whether any RCRA hazardous wastes will be generated during INL NGNP operations.
32	Need to establish that there is an INL Resource Conservation and Recovery Act (RCRA) Part B Site-wide Permit and that the INL NGNP processes do not involve RCRA waste treatment.
33	Need to determine if the Underground Piping Permit is part of the INL Site Clearance Permit Process.
34	Need to determine whether this demonstration reactor has gone through a down-selection process relative to a specific DOE site (i.e., Idaho) and relative to a specific technology.

20.6.7.3 ISSUES FOR RESOLUTION

As this report was being finalized, an environmental "site selection" report for the New Production Reactor was provided by INL (Environmental and Other Evaluations of Alternatives for Siting, Construction, and Operating New Production Reactor Capacity, DOE report #DOE/NO-0014, dated September 1992). This report should be reviewed as part of Activity 15.

Hazardous, radioactive, low level waste, and mixed waste handling, processing, storage, and shipment off site should all be given high priority to incorporate industry best practices. Much of INL's experience applies, but LWR operating experience over the last 40 years is also applicable. Special emphasis should be put on taking advantage of design features of the NGNP that minimize the generation of these types of wastes.

An initial list of site-specific hazards which should be considered in the NGNP safety analysis (chemical incidents from inside and outside the plant, acid spills, hydrogen storage detonations and deflagrations) is needed.

In addition, Table 20.6.7-1 identifies actions and issues which will need to be resolved in the development of a complete Environmental Permitting Plan.

20.6.7.4 RECOMMENDATIONS

The following actions are recommended:

- Review the site selection report mentioned in the first paragraph of Section 20.6.7.3.
- Continue reviewing and revising the EPP as the NGNP design and project schedule are developed.

LIST OF ASSUMPTIONS

The purpose of this section is to list the major assumptions for this Licensing and Permitting Special Study. The assumptions are listed below, along with a parenthetical reference to the section of this report where the assumption was identified.

- A reliable and high quality supply of fuel will be available when needed. Licensing of the fuel manufacturing facility is not addressed in this study (Section 20.6-B Introduction).

- Spent fuel will be stored on-site and does not include transportation of spent fuel to an offsite location for either storage or reprocessing, until there is a long term repository. (Section 20.6-B Introduction).

ACTIONS RECOMMENDED FOR OTHER ACTIVITIES

The purpose of this section is to provide a listing of actions that, along with recommendations stated throughout this report and in the next section, will be inputs to other Activities as listed below.

- Activity 15 – Licensing and Permitting
    - Review DOE orders to identify compliance issues relative to NRC regulations (20.6 – Introduction)
    - Follow NRC progress on licensing rulemakings that become available in early 2007 before Activity 15 work is concluded (Section 20.6.1).
    - Review NRC and industry positions on designing for aircraft impact and recommend criteria for NGNP (Section 20.6.1.3).
    - Review Nuclear Energy Institute report # NEI 04-01 and NRC draft Regulatory Guide DG-1145 for applicability to NGNP licensing (Section 20.6.1.2.2.5).
    - Identify an initial list of site-specific hazards which should be considered in the NGNP safety analysis (Section 20.6.7.3).

- Activity 16 – Economic Assessments: Provide licensing cost estimates.

- Activity 17 – Project Schedule: Provide estimated licensing schedules to the overall project schedule.

NGNP LICENSING RECOMMENDATIONS

This section provides recommendations related to the NGNP licensing process along with an identification of the section wherein the recommendation is presented in more detail. The more significant recommendations are identified as "key." These recommendations (and related actions listed in Section ACTIONS RECOMMENDED FOR OTHER ACTIVITIES) are inputs to the development of an NGNP Licensing Strategy in Activity 15.

- **Key:** Build on PBMR (Pty) Ltd–NRC pre-application interactions, including risk-informed, performance-based methods (Section 20.6.6.5).

- **Key:** Incorporate the recommended licensing option (application under Part 52 for an ESP with an embedded LWA, followed by a COL) into an overall NGNP Licensing Strategy (Section 20.6.2.4), but maintain a Part 50 strategy for a two-step license pending the success of the pre-application interactions.

- **Key:** Use License-By-Test as warranted by expected benefits to achieve timely full-power operation of the NGNP and design certification for the follow-on NGNP Commercial plant (Section 20.6.4.5).

- **Key:** Demonstrate hydrogen production capability with separation distance and facility interactions that establish precedents for the NGNP Commercial plant (Section 20.6.5.5).

- Follow NRC progress on licensing rulemakings (i.e., Part 50, Part 52) and incorporate results into the NGNP Licensing Strategy (Section 20.6.1.4).

- Identify licensing research and development needs specific to the NGNP safety analysis evaluation models as the basic design is developed (Section 20.6.4.5).

- Review the recently received "site selection" report for the New Production Reactor site at INL and identify any limiting environmental conditions (Section 20.6.7.4).

REFERENCES

REFERENCES FOR SECTION 20.6.1

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APPENDICES

APPENDIX 20.6.A – EXECUTIVE SUMMARY, NUREG-1860 (FRAMEWORK FOR A RISK-INFORMED, PERFORMANCE BASED ALTERNATIVE TO PART 50)

APPENDIX 20.6.B – DECEMBER 7, 2006, PRESENTATION SLIDES

APPENDIX 20.6.C – JANUARY 10-11, 2007, PRESENTATION SLIDES

APPENDIX 20.6.A – EXECUTIVE SUMMARY, NUREG-1860 (FRAMEWORK FOR A RISK-INFORMED, PERFORMANCE BASED ALTERNATIVE TO PART 50)

[The material below is quoted from NUREG-1860, July, 2006]

The purpose of this report is to document the technical basis to support the development of a risk-informed and performance-based process for the licensing of future nuclear power plants (NPP). As such, it documents an approach, scope and criteria that could be used by the NRC staff to develop a set of regulations that would serve as an alternative to 10 CFR 50 for licensing future NPPs. This alternative to 10 CFR 50 would have the following advantages:

- It would require a broader use of design specific risk information in establishing the licensing basis, thus better focusing the licensing basis, its safety analysis and regulatory oversight on those items most important to safety for that design.

- It would stress the use of performance as the metrics for acceptability, thus providing more flexibility to designers to decide on the design factors most appropriate for their design.

- It would be written to be applicable to any reactor technology, thus avoiding the time consuming and less predictable process of reviewing non-LWR designs against the LWR oriented 10 CFR 50 regulations, which requires case-by-case decisions (and possible litigation) on what 10 CFR 50 regulations are applicable and not applicable and where new requirements are needed.

- It would provide the foundation for technology-specific implementation, through the use of technology-specific implementing guidance in those areas unique to a specific technology.

The information contained in this report is intended to be applicable only to the licensing of commercial NPPs. Similar to 10 CFR 50, it covers the design, construction and operation phases of the plant lifecycle up to and including the initial stages of decommissioning (i.e., where spent fuel is still stored on-site). It covers the reactor, support systems, fuel handling and storage systems. The technical basis and process described in the report are directed toward the development of a stand alone set of requirements (containing technical as well as administrative items) that would be compatible and interface with the other existing parts of 10 CFR (e.g., Part 20, 51, 52, 73, 100, etc.) just as 10 CFR 50 is today. The approach taken in developing the technical basis and process is one that is a combination of deterministic and probabilistic elements and builds upon recent policy decisions by the Commission related to the use of a probabilistic approach in establishing the licensing basis.

At the highest level, the approach taken has as its goal developing a process and regulations that ensures that future NPPs achieve a level of safety at least as good as that defined by the Quantitative Health Objectives (QHOs) in the Commission's 1986 Safety Goal Policy

Statement. This is considered consistent with the Commission's 1986 Policy Statement on Advanced Reactors which states that the Commission expects advanced reactor designs will comply with the Commission's Safety Goal Policy Statement, and is discussed further in Chapter 3.

Defense-in-depth remains a fundamental part of the approach taken and has as its purpose applying deterministic principles to account for uncertainties. The defense-in-depth approach taken, at a high level, calls for:

- the application of a set of defense-in-depth principles that result in certain deterministic criteria; and
- multiple lines of defense against off-normal events and their consequences (called protective strategies).

The defense-in-depth principles, discussed in Chapter 4, address the various types of uncertainty (i.e., parameter, modeling and completeness) and require designs:

- consider intentional as well as inadvertent events;
- include accident prevention and mitigation capability;
- ensure key safety functions are not dependent upon a single element of design, construction, maintenance or operation;
- consider uncertainties in equipment and human performance and provide appropriate safety margin;
- provide alternative capability to prevent unacceptable releases of radioactive material; and
- be sited at locations that facilitate protection of public health and safety.

The protective strategies discussed in Chapter 5, address accident prevention and mitigation and consist of the following:

- physical protection (provides protection against intentional acts);
- maintaining stable operation (provides measures to reduce the likelihood of challenges to safety systems);
- protective systems (provides highly reliable equipment to respond to challenges to safety);

- maintaining barrier integrity (provides isolation features to prevent the release of radioactive material into the environment); and

- protective actions (provides planned activities to mitigate any impacts due to failure of the other strategies).

These protective strategies provide a high-level defense-in-depth structure which new designs would be required to have. In effect, they provide for successive lines of defense, each of which needs to be included in the design. A set of probabilistic criteria (Chapter 6) have been developed consistent with the Safety Goal QHOs that address:

- allowable consequences of event sequences versus their frequency;
- selection of event sequences which must be considered in the design; and
- safety classification of equipment.

The approach continues the practice of ensuring that the allowable consequences of events are matched to their frequency such that frequent events must have very low consequences and less frequent events can have higher consequences. This is expressed in the form of a frequency-consequence (F-C) curve as shown in Figure ES-1. The allowable consequences are based upon existing dose limits or doses necessary to meet the QHOs, as described in Chapter 6. Their correlation with event frequency is based upon guidance given in ICRP Publication 64, “Protection from Potential Exposure: A Conceptual Framework.” The consequences from each event sequence from the probabilistic risk assessment (PRA) and each event sequence selected as a licensing basis event (LBE - discussed below) must meet the F-C curve.

Frequency categories have been established to guide the selection of events which must be considered in the design. These frequency categories are:

- frequent events > 10^{-2}/yr
- infrequent events < 10^{-2}/yr but > 10^{-5}/yr
- rare events < 10^{-5}/yr but > 10^{-7}/yr

In all cases mean frequency values are to be used. These frequency categories define what event sequences must be considered in the licensing process. Within each of these frequency categories certain event sequences are chosen for more conservative deterministic analysis, including comparison to the F-C curve. These events are called LBEs and are generally those with the highest consequences for a given type of accident (e.g., reactivity insertion, loss of coolant, etc.). The purpose of the LBEs is to demonstrate the conservatism of the PRA analysis. In addition, a deterministic event, with a conservative source term is to be used for comparison with siting criteria. Chapter 6 provides additional descriptions of the event categories, the LBE selection and acceptance criteria, the deterministic event and analysis guidelines.

The safety classification of equipment is to follow a probabilistic approach whereby importance measures and other risk metrics are to be used to determine which equipment is safety significant and which is not. Equipment classified as safety significant would be subject to special treatment to ensure it can perform its safety function. Chapter 6 provides additional discussion on the safety classification process.

As discussed above, risk assessment will have a more prominent and fundamental role in the licensing process than it does today under 10 CFR 50, since the risk assessment will be an integral part of the design process and licensing analysis. Therefore, a high level of confidence is needed in the results of the risk assessment used to support licensing. In addition, under the risk-informed licensing approach, the risk assessment will need to be maintained up to date over the life of the plant, since it will be an integral part of decision-making with respect to operations (e.g., maintenance, plant configuration control) and plant modifications. Guidance on the scope and technical acceptability of the risk assessment needed to support this licensing approach is provided in Chapter 7.

In Chapter 8, the protective strategies are examined to identify what needs to be done to ensure the success of each one. Figure ES-2 illustrates the process used for this examination. The process starts with the development of a logic tree for each protective strategy which is used to develop a set of questions, the answers to which identify the topics the requirements must address to ensure the success of the protective strategy. This is supplemented by application of the defense-in-depth principles described above to each protective strategy to address uncertainties and utilization of the risk and design criteria developed in Chapter 6. The topics identified are organized by whether they apply to design, construction or operation and, where guidance related to the topic is provided in the framework, an appropriate reference is given. A similar process was also applied to the identification of topics for administrative requirements. The list of topics resulting from the process in Figure ES-2 is shown Table ES-1. The list of topics then forms the starting point for the development of requirements. Chapter 8 also provides guidance on how to develop the requirements, including utilizing a performance-based approach (i.e., following the guidelines in NUREG/BR-0303, "Guidance for Performance-Based Regulation") and using existing requirements in 10 CFR 50 where they are already technology-neutral (i.e., building upon existing requirements, as much as practical). A completeness check was also made by comparing the topics identified in Chapter 8 to other safety requirements (e.g., IAEA Standards, 10 CFR 50). The results of the completeness check are discussed in Chapter 8, and generally conclude that the topics included in Table ES-1 are reasonably complete. Finally, guidance regarding which of the requirements may need technology-specific guidance to support its implementation is provided in Chapter 8.

Table ES-1 Topics for requirements:

(A) Topics Common to Design, Construction and Operation

1) QA/QC
2) PRA scope and technical acceptability

(B) Physical Protection

1) General (10 CFR 73)
2) Perform security assessment integral with design
3) Security performance standards

(C) Good Design Practices

1) Plant Risk:
   - Frequency Consequence curve
   - QHOs (including integrated risk)
2) Criteria for selection of LBEs
3) LBE acceptance criteria:
   • frequent events (dose, plant damage)
   • infrequent events (dose, plant damage)
   • rare events (dose)
   • link to siting
4) Keep initiating events with potential to defeat two or more protective strategies <10-7/plant year
5) Criteria for safety classification and special treatment
6) Equipment Qualification
7) Analysis guidelines
   • realistic analysis, including failure assumptions
   • source term
8) Siting and site-specific considerations
9) Use consensus design codes and standards
10) Materials qualification
11) Provide 2 redundant, diverse, independent means for reactor shutdown and decay heat removal
12) Minimum - 2 barriers to FP release
13) Containment functional capability
14) No key safety function dependent upon a single human action or piece of hardware
15) Need to consider degradation and aging mechanisms in design
16) Reactor inherent protection (i.e., no positive power coefficient, limit control rod worth, stability, etc.)
17) Human factors considerations
18) Fire protection
19) Control room design
20) Alternate shutdown location
21) Flow blockage prevention
22) Specify reliability and availability goals consistent with PRA
   - Establish Reliability Assurance Program
   - Specify goals on initiating event frequency
23) Use of Prototype Testing
24) Research and Development
25) Combustible gas control
26) Coolant/water/fuel reaction control
27) Prevention of brittle fracture
28) Leak before break
29) I and C Systems
   • analog
   • digital
   • HMI

30) Criticality prevention
31) Protection of operating staff during accidents
32) Qualified analysis tools

(D) Good Construction Practices

1) Use accepted codes, standards, practices
2) Security
3) NDE
4) Inspection
5) Testing

(E) Good Operating Practices

1) Radiation protection during routine operation
2) Maintenance program
3) Personnel qualification
4) Training
5) Use of procedures
6) Use of simulators
7) Staffing
8) Aging management program
9) Surveillance, including materials surveillance program
10) ISI
11) Testing
12) Technical specifications, including environmental
13) Develop EOP and AM procedures integral with design
14) Develop EP integral with design
15) Monitoring and feedback
16) Work and configuration control
17) Living PRA
18) Maintain fuel and replacement part quality
19) Security

(F) Administrative

1) Standard format and content of applications
2) Change control process
3) Record keeping
4) Documentation control
5) Reporting
6) Monitoring and feedback:
   - plant performance
   - environmental releases [i.e., effluent releases to the environment]
   - testing results
7) Corrective action program
8) Backfitting
9) License amendments
10) Exemptions
11) Other legal and process items from 10CR50

APPENDIX 20.6.B – DECEMBER 7, 2006, PRESENTATION SLIDES

Special Study 20.6 – Licensing and Permitting

December 6-7, 2006
Stoughton, MA

Presentation Outline

- **Activity 20.6 Overview**
- **Task Status**
- **Summary**

Activity 20.6 Overview

- **Purpose:**
  - Perform a study to address the specific licensing tasks identified in the SOW and listed on the next slide
    - *provide related recommendations and their bases*
    - *Identify related concerns and open issues to be addressed in Activity 15*
  - Document results - input to the PCDR

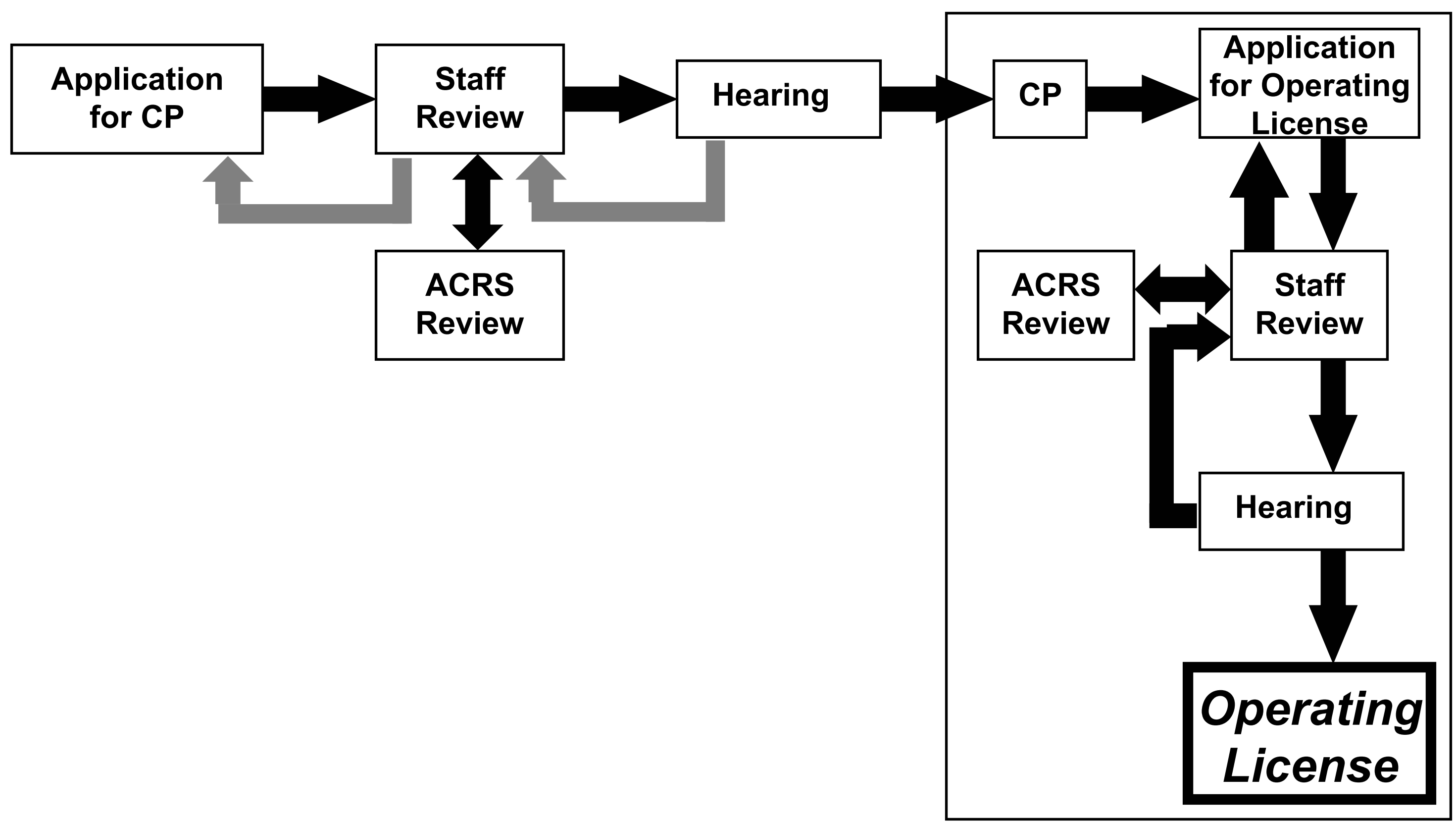
Activity Overview….

- **Specific Licensing Tasks:**
  - 2.6.1 - Licensing under Part 50 vs. Part 52
  - 2.6.2 - Feasibility of Mixed Licensing Approach (e.g., Part 52 ESP and Part 50 CP/OL)
  - 2.6.3 - Feasibility of Using New Advanced Reactor Licensing Framework (expected to become Part 53)
  - 2.6.4 - Practicality of “License by Test” Licensing Method
  - 2.6.5 - Licensing of an Integrated Nuclear Power/Hydrogen Plant
  - 2.6.6 - Integration of PRA Techniques During Design Phase
  - 2.6.7 - EPA/State Permits for Nuclear Power/Hydrogen Plant

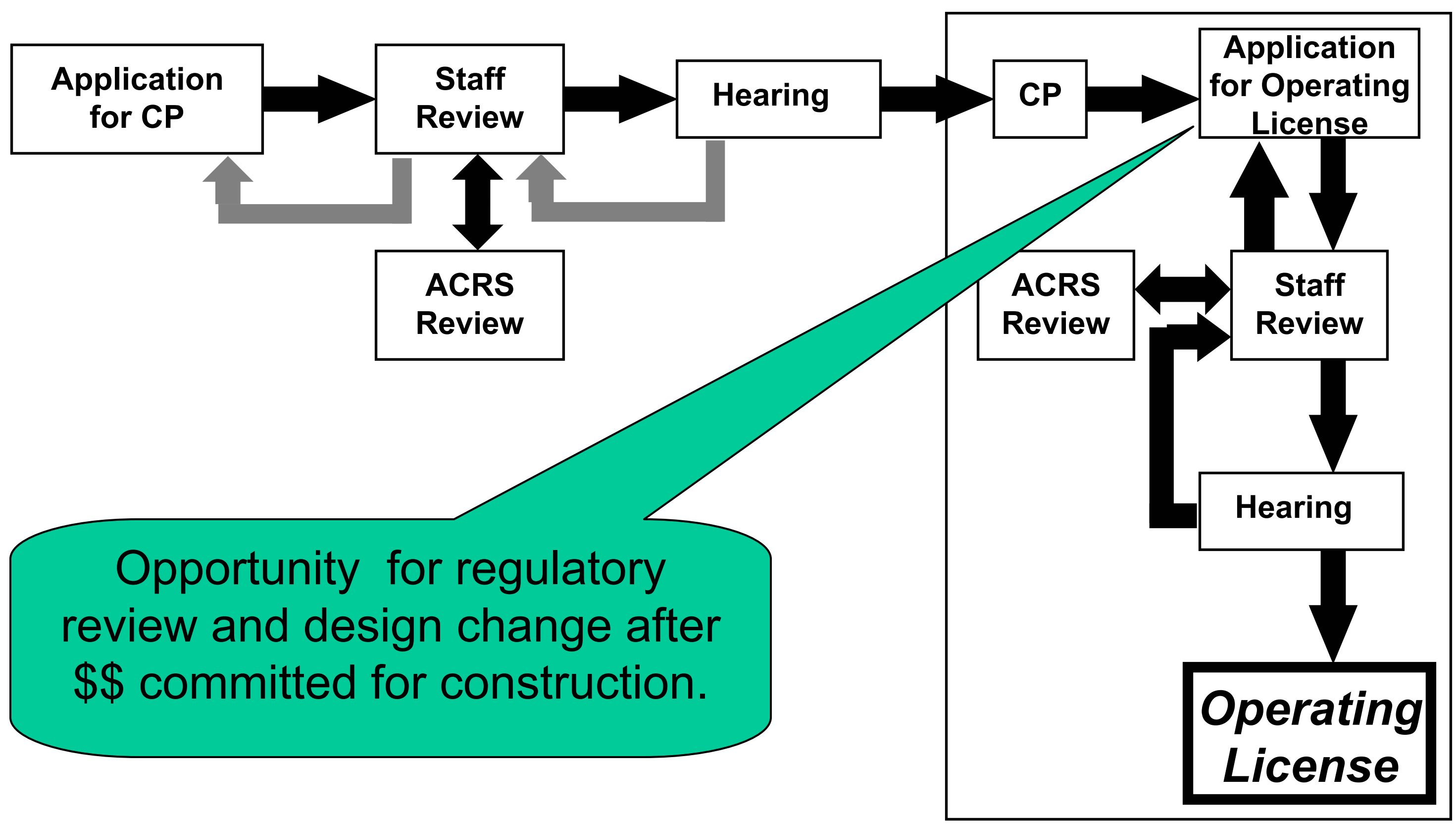
Task Status

- **Task 2.6.1 - Licensing under Part 50 vs. Part 52**
  - Part 50: CP/OL process by which today's commercial power plants were licensed
  - Part 52: Streamlined licensing process developed to address licensing delays experienced with Part 50
    - *Early Site Permit, Design Certification, Combined License*

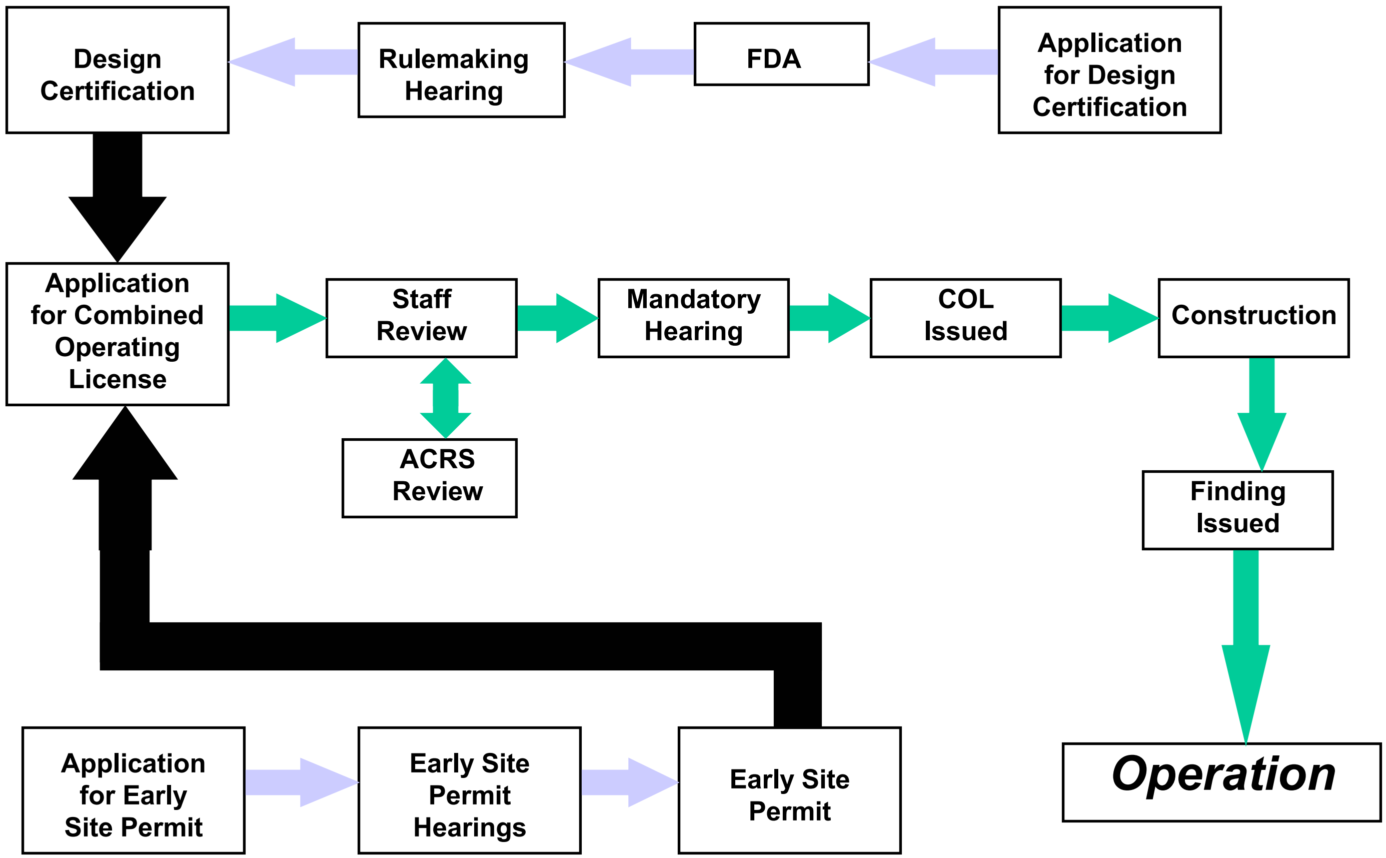
Part 50 "Two-Step" Licensing Process



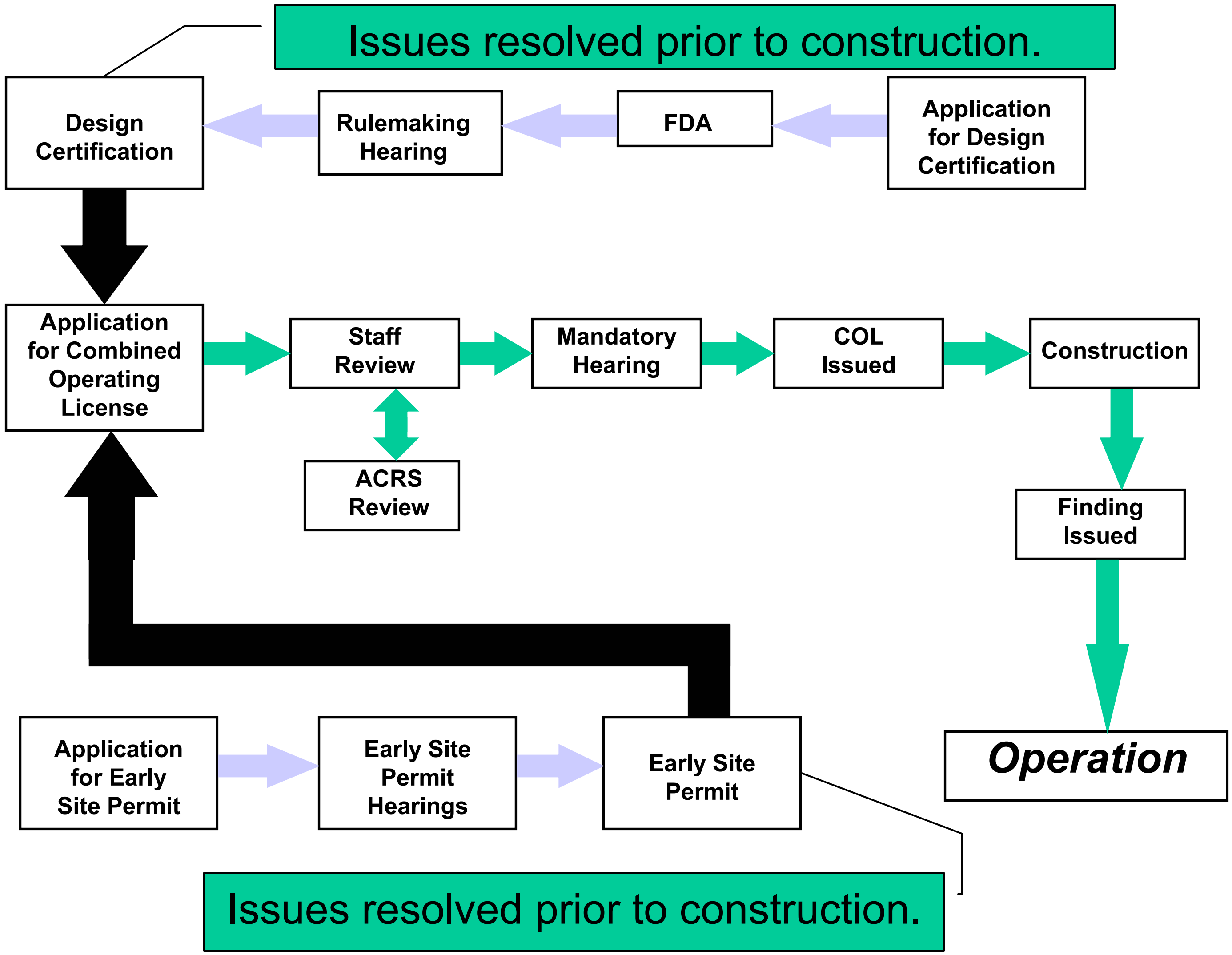
Part 50 "Two-Step" Licensing Process



Part 52 “One-Step” Licensing Process



Part 52 “One-Step” Licensing Process



Basic Requirements for an Early Site Permits

- **Site description and characterization information**
- **Number, type, and thermal power level of facilities**
- **Site safety assessment**
- **Near-by industrial facilities**
- **Projected population profiles**
- **Thermal and radiological effluents and environmental report**
- **Emergency planning information**
- **Site restoration plan**
  - if early site work is conducted and the Early Site Permit expires before being referenced in a construction permit or combined license.

Basic Requirements for Design Certifications

- **An essentially complete plant design must be provided, with level of detail to enable staff to close safety questions and confirm construction**
  - Deterministic design review, including
    - *Design information and safety analyses*
    - *Selected procurement and construction specifications*
  - Complete a detailed Probabilistic Safety Assessment & resolve all degraded core issues
  - Meet all current regulations, including post-TMI regulations
  - Resolve all Unresolved Safety Issues, Generic Safety Issues and other new technical issues
  - Testing and startup programs

Basic Requirements for Design Certification….

- **Since approval is granted before construction,**
  - Proposed tests, inspections, analyses, and acceptance criteria (ITAAC) which are necessary to provide assurance that the plant is built and will operate in accordance with the certified design must be provided.
  - Conceptual design descriptions (e.g., ultimate heat sink) must be provided for portions of the plant not within scope of certified design
- **Interface Requirements for conceptual design features and corresponding ITAAC must be provided**

Design Certification – Lessons Learned

- **Regulator has only one chance to review the design which would be constructed at several sites, so they ask more questions than they would for a Part 50 review**
  - ~1000 – 2000 questions for a Part 50 review
  - ~5000 – 10,000 questions for a Part 52 review
- **If issues not closed during NRC staff design review, action item would passed to the COL applicant**

Basic Requirements for Combined Licenses

- **Reference ESP or provide corresponding information**
  - If ESP referenced, show compliance with ESP parameters
- **Reference DC or provide corresponding information**
- **ITAAC for portions of facility not covered by those ITAAC in the DC**
- **Emergency plans**
  - Can reference related information in the ESP
  - Certification from local authorities of approval of emergency plans, or show good faith effort

Task Status….

- **Licensing under Part 50 vs. Part 52….**
  - PCDR input write-ups to include:
    - *Summary descriptions and comparison of Parts 50 and 52*
    - *List of regulatory review issues - resources are:*
      - NRC current pre-application review of PBMR for DC
      - NRC's previous pre-application review of PBMR for a COL (Exelon)
      - Current NGNP program

Task Status….

- **Task 2.6.2 - Feasibility of Mixed (Parts 50 and 52) Licensing Approach**
  - Industry has experience with these regulations
  - Options:
    - *Part 50 alone*
    - *ESP + DC + COL (Part 52 alone)*
    - *ESP + COL (design review is part of COL)*
    - *DC + COL (site approval is part of COL)*
    - *COL only (design and site approvals part of COL)*
    - *ESP + Part 50 CP and OL*
  - A Limited Work Authorization (early site preparation work) can be considered for each option

Task Status….

- **Task 2.6.3 - Feasibility of Using New Advanced Reactor Licensing Framework (expected Part 53)**
  - Technology-neutral licensing process; strongly risk-informed
    - *Penalty for using this expected rule may be delay NGNP project schedule*
      - Rule may not be finalized until the 2012 time frame
      - Implementing a generic rule for the first time, without experience of completed, risk-informed gas reactor design review would be difficult
    - *Benefit: current draft rule supports a stronger use of risk-informed evaluations relative to Parts 50 and 52.*

Preliminary Recommendation

- **Part 50 for NGNP prototype plant - Part 52 for NGNP commercial plants**
  - Design detail available for prototype
  - Plant configuration (prototype vs. commercial)
  - Startup and operation experience

- **Use of Part 53 by itself is not recommended, but should implement Part 53 risk-informed methods during design and review (Part 50) of the NGNP prototype**

Task Status….

- **Task 2.6.4 - Practicality of "License by Test" Licensing Method**
  - "License by Test" concept permitted by regulations, but implementation criteria & requirements are not defined
  - Current understanding based on
    - *Reactors at INL, FFTF, Clinch River and early HTGR demo plants*
    - *MHTGR experience with conceptualization*
  - Additional experience to be gained from the PBMR demonstration plant
  - Can be viewed as an effective substitute for design analysis & separate effects tests that would be expected prior to plant startup
    - *Tradeoff with required instrumentation, design changes, and potentially power level limitations*
  - Preliminary recommendation: use License by Test selectively when justified by risks & benefits

Task Status….

- **Task 2.6.5 - Licensing of an Integrated Nuclear Power/Hydrogen Plant**
  - Basic consideration: protect reactor from other processes (turbine generator, H2 plant)
    - *Implement design controls such as:*
      - turbine orientation
      - barriers vs. separation for H2 plant
    - *Perform PRA and/or FMEA analyses to identify adverse fluid and I&C interactions*
  - NGNP Prototype licensing needs to establish the basis for full-scale plant, for example:
    - *Required separation distance*
    - *Limits on non-nuclear operations in Hydrogen Plant*

Task Status….

- **Task 2.6.6 - Method for Integration of PRA Techniques During Design Phase**
  - Use “strongly” risk-informed approach
    - *Combination of deterministic design/analysis and PRA*
  - Fully integrate a complete PRA with deterministic design and analysis methods
    - *Licensing Basis Event selection*
    - *Classification of structures, systems, and components*
    - *Defense-in-Depth*
    - *Potential design feature and plant configuration selections*

Task Status….

- **Task 2.6.7 - EPA/State Permits for Integrated Nuclear Power/Hydrogen Plant**
  - ➢ Draft list of NGNP Prototype permits developed
    - – *Need to schedule dates for permits*
  - ➢ Report drafted, but have list of >30 issues & questions that need to be closed
    - – *We have the NPR INL Site Selection report - 1989*
      - References the “Site Characterization Report - 1985” and the “Siting Report – 1988”
      - Was EA, PEIS, or EIS subsequently issued?

Summary

- **Task 2.6.1 – Parts 50 and 52 description and comparison**
- **Task 2.6.2 – mixed licensing options:**
- **Task 2.6.3 – New Licensing Framework (Part 53)**
- **Recommendations:**
  - Part 50 for prototype and Part 52 for commercial unit
  - Use new Part 53 risk-informed methods in conjunction with Part 50
- **Task 2.6.4 – License by Test**
  - Keep option open; develop a plan
- **Task 2.6.5 – License combined H2 / power plant**
  - Use combination of design controls and analysis; identify potential reg. limits
- **Task 2.6.6 – Integrate PRA into design process**
  - Develop and implement a strongly risk-informed process
- **Task 2.6.7 – EPA/ State Permitting**
  - Need more information on specifics for INL site

APPENDIX 20.6.C – JANUARY 10-11, 2007, PRESENTATION SLIDES

Special Study 20.6 – Licensing and Permitting

Stanley Ritterbusch

Presentation Outline

- Objectives
- Summary of Recommendations
- Task Summary and Specific Recommendations
- Future Action

Objectives

- Perform a study to address the specific licensing tasks identified in the SOW
  - Provide related recommendations and their bases
  - Identify related concerns and open issues for inclusion in Activity 15

Licensing Study - Task List

- 2.6.1 - Licensing under Part 50 vs. Part 52
- 2.6.2 - Feasibility of Mixed Licensing Approach (e.g., Part 52 ESP and Part 50 CP/OL)
- 2.6.3 - Feasibility of Using New Advanced Reactor Licensing Framework (expected to become Part 53)
- 2.6.4 - Practicality of "License by Test" Licensing Method
- 2.6.5 - Licensing of an Integrated Nuclear Power/Hydrogen Plant
- 2.6.6 - Integration of PRA Techniques During Design Phase
- 2.6.7 - EPA/State Permits for Nuclear Power/Hydrogen Plant

Licensing Study – Summary of Recommendations

- Implement a licensing process for the NGNP Prototype with the following features:
  - Build on PBMR pre-application interaction, including risk-informed, performance-based methods
  - Application for Early Site Permit (with an embedded Limited Work Authorization)
  - Application for Part 52 Combined Construction and Operating License (COL), pending a successful PBMR pre-application program
  - Use License-by-Test selectively as warranted by expected benefits to achieve end result, namely:
    - Timely full power operation of the NGNP Prototype
    - Design Certification (DC) for the follow-on Commercial plant
  - Demonstrate H2 production with separation distance and facility interactions that establish commercial plant precedents

Licensing Study – Summary of Recommendations….

- Bases:
  - Site approval as early as possible
  - Operation of a prototype is not a pre-condition for issuance of a COL
  - Substantive design detail will be available for NGNP Prototype
  - Follow up with DC for use with NGNP Commercial plant licensing

- Develop the NGNP Prototype licensing strategy in Activity 15, including
  - Major activities
  - Cost impacts
  - Schedule impacts

Task Summary and Specific Recommendations

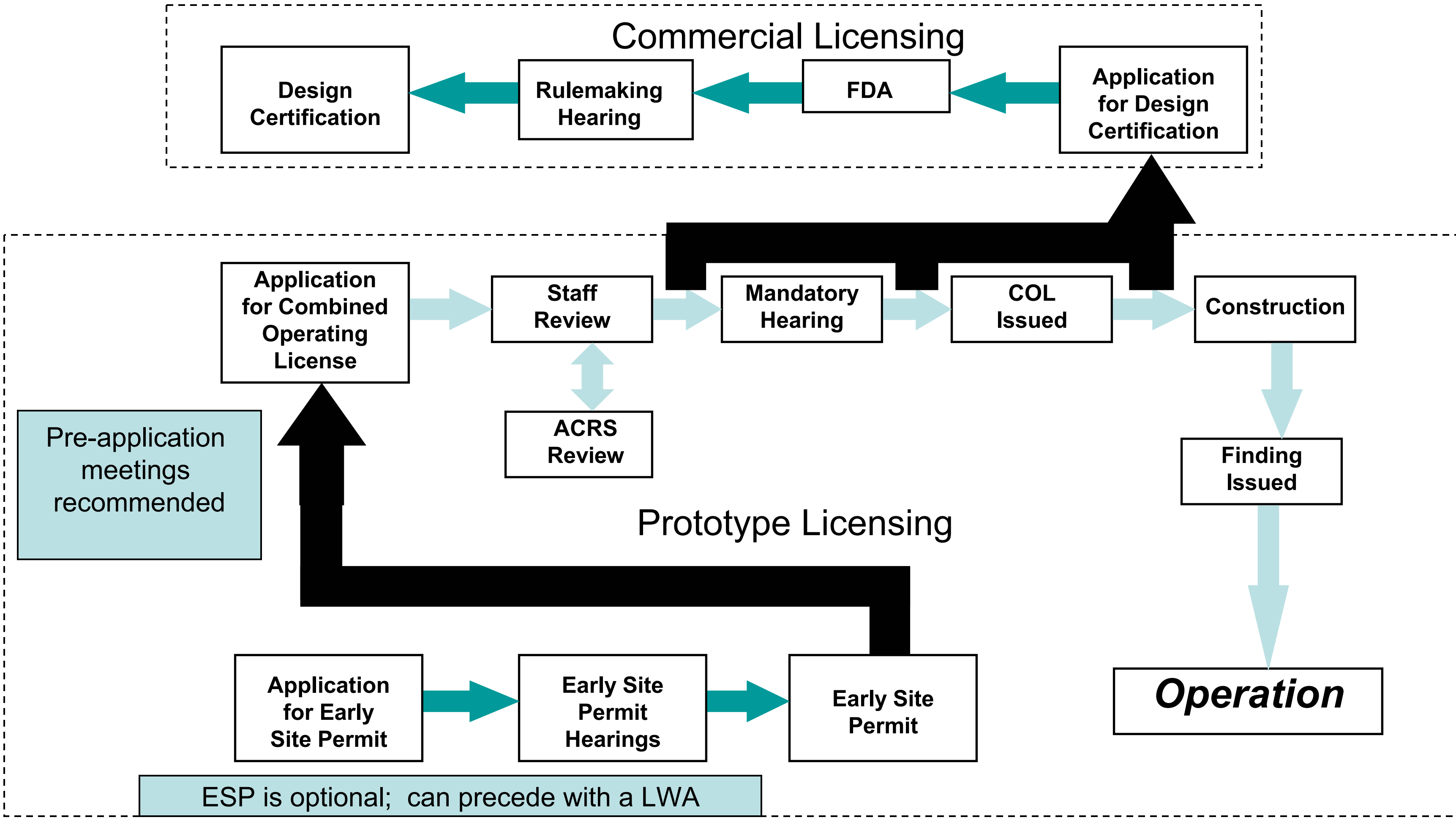
- Task 2.6.1 - Licensing under Part 50 vs. Part 52
  - Part 50: CP/OL process by which today's commercial power plants were licensed
    - Can start construction earlier than under Part 52 (which requires full design detail)
    - The NRC and the public have two opportunities for imposing new design changes
      - After initial design and after start of construction
      - Licensing and construction schedule delays

Task Summary and Specific Recommendations ....

- Part 52: Streamlined licensing process developed to address licensing delays experienced with Part 50
  - Early Site Permit, Combined License, and Design Certification

  - ESP allows early resolution of site issues
  - Provides only one opportunity for design review/changes
    - Before construction begins
  - Must have adequate design detail to meet regulations before design approval
    - Assuming successful PBMR pre-application program

NGNP - Part 52 "One-Step" Licensing Process



Task Summary and Specific Recommendations ….

- Task 2.6.2 - Feasibility of Mixed (Parts 50 and 52) Licensing Approach
  - It is feasible to mix some aspects of the two approaches
  - Use the following approach which uses only Part 52:
    - An Early Site Permit with an embedded Limited Work Authorization, followed by a COL
    - Implement risk-informed, performance-based methods (see task 2.6.6)
  - Bases:
    - Forthcoming regulation revisions (including LWAs, COL without prototype)
    - Starting site work as soon as possible
    - Less schedule risk
  - Establish the conditions of a contingency strategy for a Part 50 CP-OL approach if early construction is paramount for the integrated schedule
  - Incorporate the recommended licensing option into an overall NGNP licensing strategy

Task Summary and Specific Recommendations ….

- Task 2.6.3 - Feasibility of Using New Advanced Reactor Licensing Framework (expected Part 53)
  - Technology-neutral licensing process
    - Rule may not be finalized until the 2012 time frame at best
    - Implementing a generic rule for the first time, without experience of completed, risk-informed gas reactor design review would be difficult
    - Current draft rule supports a stronger use of risk-informed methods relative to Parts 50 and 52.
  - Implementation of Part 53 for the NGNP Prototype is not recommended
  - Further develop a risk-informed, performance-based design and licensing process for the NGNP Prototype, with input from the on-going Part 53 rulemaking

Task Summary and Specific Recommendations ….

- Task 2.6.4 - Practicality of "License by Test" Licensing Method
  - "License by Test" concept permitted by regulations, but implementation criteria & requirements are not defined.
  - Our definition:
    - Tests conducted on the constructed NGNP Prototype, that are not already part of the PBMR and NGNP R&D programs, to support development of safety analysis evaluation models
  - Can be viewed as an effective substitute for separate effects tests that would be expected prior to plant construction
    - Improved schedule - tradeoff with cost of required instrumentation, design changes, and potential power level limitations
  - Use License-by-Test selectively as warranted by expected benefits to achieve end result, namely:
    - Timely full power operation of NGNP Prototype
    - Commercial plant DC
  - With input from:
    - PBMR design certification verification & validation elements
    - NGNP research & development requirements, schedule, and cost

Task Summary and Specific Recommendations ….

- Task 2.6.5 - Licensing of an Integrated Nuclear Power/Hydrogen Plant
  - Protect reactor from other processes (turbine generator, H_2 plant)
    - Perform PRA and/or FMEA analyses to identify adverse transient thermodynamic, and/or control system interactions
    - Implement design controls such as:
      - turbine orientation
      - barriers and separation
  - Demonstrate H_2 production with separation distance and facility interactions that establish commercial plant precedents
  - As the NGNP Prototype design is developed, further develop the list of chemical plant hazards, design and operational considerations for licensing a nuclear power/hydrogen generation plant design

Task Summary and Specific Recommendations ….

- Task 2.6.6 - Method for Integration of PRA Techniques During Design Phase
  - PRA methods are maturing and increasingly being invoked in revised NRC regulations
    - Combination of deterministic and probabilistic methods
  - Based on PBMR pre-application activities, use risk-informed, performance-based approach as part of initial design work
    - Evaluate potential design feature and plant configuration selections
    - Licensing Basis Event selection
    - Classification of structures, systems, and components
    - Defense-in-Depth Evaluations

Task Summary and Specific Recommendations ….

- Task 2.6.7 - EPA/State Permits for Integrated Nuclear Power/Hydrogen Plant
  - Draft Environmental Permitting Plan (EPP) developed
    - Need to integrate with overall project schedule
    - 34 open issues and questions
  - Continue reviewing and revising the EPP as the NGNP design and project schedule are developed, include:
    - Review the NPR site selection report (Environmental and Other Evaluations of Alternatives for Siting, Constructing, and Operating NPR Capacity, 1992) for limiting conditions or constraints on NGNP

Future Action – Activity 15

- Develop a licensing strategy for the NGNP Prototype, in concert with the NGNP Commercial plant overall licensing strategy, based on the results, recommendations, and other actions identified in this special study.
- Incorporate results of evolving design studies into the licensing strategy.
- Given the engineering and R&D schedules, provide input to:
  - Activity 16 (Cost Estimates) on licensing costs
  - Activity 17 (Project Schedule) on licensing schedules.