Report:

RECONCILIATION OF PHENOMENA IDENTIFICATION RANKING TABLES AND PRECONCEPTUAL DESIGN REPORTS

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REVISED DESIGN INTEGRATION REVIEW TEAM REPORT
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1.0 INTRODUCTION

As a part of their preparation to perform future safety assessment and verification for the Next Generation Nuclear Plant (NGNP) design, the U.S. Nuclear Regulatory Commission (NRC) has performed a Phenomena Identification and Ranking Tables (PIRT) effort. The PIRT efforts have been performed by panels of experts from NRC, Department of Energy (DOE) national laboratories, and the nuclear industry at large. The thrust of the PIRT efforts was to identify phenomena of concern and related needs in six key areas likely to pose challenges, in terms of NRC technical review and verification:

- Accidents and Thermal Fluids
- Reactor Physics and Neutronics (including criticality calculations and experiments)
- Fuel Performance, Fission Product Transport, and Dose
- High-Temperature Materials (metallic)
- Graphite Materials
- Process Heat for Hydrogen Production

From NRC's point of view, the purpose of the PIRT efforts has been to identify significant needs they will have for analytical tools and data that will enable them to perform their regulatory function in reviewing the license application of the selected NGNP design.

This report document describes an effort undertaken by the Battelle Energy Alliance (BEA) Design Integration and Review Team, as part of the NGNP Project, to reconcile the PIRTs with the three competing NGNP designs. At this stage of the NGNP project, the design side of the reconciliation was represented by the pre-conceptual design reports (PCDRs) submitted in 2007 by AREVA, General Atomics, and Westinghouse.

The detailed PIRT exercises have been documented in NUREG/CR-6844 (July 2004) and NUREG/CR-6944 (March 2008). That data is further analyzed and reduced in the "*Next Generation Nuclear Plant Gap Analysis Report*" (July 2008), drafted by Oak Ridge National Laboratory under contract to NRC. Although the Gap Analysis document is under NRC review and will later be published as a NUREG, the draft represented a reasonable summary of the PIRTs and was consulted as part of this reconciliation effort.

The overall objective of the reconciliation task, consistent with NRC's PIRT and gap analysis objectives, was to subject the PCDRs to detailed examination against the PIRTs, and determine whether NRC's projected needs have been recognized in the PCDRs, either as ongoing or planned R&D efforts. This comparison of "what is needed" to "what is already available or planned" was seen as a way of identifying previously unknown NGNP project risks that require mitigation. Filling of these significant gaps in the knowledge base will ensure the availability of information and analysis tools required for NRC to adequately assess NGNP safety characteristics. More generally, this process of risk assessment and mitigation will help to identify and avoid unexpected impacts on the NGNP technical success, project cost, and project schedule.

2.0 METHODOLOGY

2.1 General description of methodology

NGNP project management directed each of the three competing reactor designers (AREVA, General Atomics, and Westinghouse) to perform a reconciliation of their R&D plans, as identified in their respective pre-conceptual design reports (PCDRs), against NRC's PIRT effort. The purpose of this request was to verify that:

- R&D needs have been addressed comprehensively
- No significant risks and related project costs and schedule durations are being omitted

NGNP project management also determined that a Design Integration Review Team should perform this reconciliation internally. The Team consisted of the following resources:

- Richard Garrett – NGNP Engineering Director and Team Leader
- Mark Holbrook – NGNP Licensing Engineer
- George Ghonma – Regulatory Consultant
- William Mangiante – Engineering/Regulatory Consultant
- Richard Hobbins – R&D Consultant

Supporting resources include:

- James Kinsey – NGNP Licensing Director
- Phillip Mills – NGNP Engineering Deputy Director
- David Petti - NGNP Technology Development Director
- John Collins – NGNP Lead Systems Engineer, Technology Development

The task undertaken by the Team was to compare the PIRTs, as summarized in ORNL's gap analysis, against the most current R&D plans reflected in the PCDRs submitted by AREVA, General Atomics, and Westinghouse. Subsequent to that evaluation and review of PCDRs, the NGNP internal R&D organization performed a second tier of review in which they incorporated their first-hand knowledge of the R&D planned directly by DOE and BEA for the NGNP Project. These plans include such efforts as the Component Test Facility (CTF), planned for construction at the Idaho National Laboratory (INL) site, and related analyses and experiments. While the NGNP R&D organization provided input on all of the identified PIRT issues, their main purpose was to report on the exceptions identified earlier in the PIRT/PCDR reconciliation process…i.e., needs that the reactor design vendors seemingly had not addressed.

The Design Integration Review Team's overarching objectives were to determine whether the significant PIRT items are being addressed by the overall NGNP R&D program (consisting of the three reactor designers and the internal Project R&D organization) and to identify any items that might be absent from the R&D already performed or planned by NGNP and its contractors.

2.2 Role of data collection and summary tables in methodology

The task of comparing the summarized PIRTs against the PCDRs was formulated in a task plan, which is Appendix A to this document. Once the task was underway, one of the first activities was to extract an inventory of PIRT issues from the summarized PIRTs, so that all work would be performed against a standardized set of issues identified by NRC. Eventually, that inventory of issues was incorporated into the data collection and summary tables, which are described below.

The first stage of assessment work was distributed among several team members, who performed detailed examinations of their assigned PCDRs or PCDR sections. Separate sets of worksheet tables were prepared for data collection. Each set of data collection tables was dedicated to one of the three reactor designers, and each set was divided into the six major PIRT technical areas. During the data collection phase, assessment personnel examined and extracted information from their assigned PCDRs that was relevant to the various NRC issues, and entered that information into the tables.

The data collection tables show:

- Summarized PIRT issues identified by NRC
- Excerpts extracted from the PCDRs that demonstrate that the issues are being addressed and/or provide information that is otherwise relevant
- Established Design Data Needs (DDNs) that are relevant to the identified issues
- Comments as to whether the identified issues were being addressed in the PCDRs

The data collection tables represent the first step in determining whether the summarized PIRTs are addressed by the PCDRs.

Further description and presentation of the data collection tables are contained in section 3.0 of this report, "*Comparison of Summarized PIRTs against Vendor PCDRs*"**.**

A set of summary tables was prepared to reflect the joint efforts of review and analysis performed by the team members on the three PCDRs. Like the data collection tables, the summary tables were divided into the six major PIRT technical areas. The summary tables were produced so that: a) the three vendors' applicable efforts could be viewed side-by-side, and b) the NGNP R&D organization could easily provide its incremental input to the comments/conclusions column, as agreed in the task plan.

The summary tables show:

- Summarized PIRT issues identified by NRC
- Excerpts extracted from each of the three PCDRs that demonstrate that the issues are being addressed and/or provide information that is otherwise relevant
- Assessment comments provided by the Team members
- Incremental responses provided by the NGNP R&D organization (see section 2.3)

The summary tables represent the second step in determining whether the summarized PIRTs are addressed by the PCDRs.

Further description and presentation of the summary tables are contained in section 3.0 of this report, "*Comparison of Summarized PIRTs against Vendor PCDRs*"**.**

2.3 Role of NGNP Project R&D input in methodology

The work outputs of each PIRT/PCDR comparison effort were documented in the data collection and summary tables. At that point, the Team members had addressed any perceived inadequacies, relative to the ability of R&D performed or planned by the reactor designers to meet the needs identified by NRC. Exceptions (i.e., gaps not addressed) are shown in the *Comments/Conclusions* columns of the data collection and summary tables in bold italics. The draft summary tables were then submitted for review by the NGNP internal R&D organization, which was charged with determining whether the issues, particularly those not being addressed by the reactor designers, would be addressed by Project R&D efforts. Input received from NGNP Project R&D was incorporated into the final versions of the summary tables.

The NGNP Project R&D input represents the third step in determining whether the summarized PIRTs are addressed by the PCDRs.

A description of the feedback provided by the NGNP R&D organization is presented in section 4.0 of this report, "*NGNP Project R&D Input*".

2.4 Role of conclusion tables in methodology

The overall flowdown of assessment information in this task is:

- Origination of assessment information in the data collection tables by the Design Integration Review Team members. Data collection tables are attached as separate files at the end of section 3.1 of this report.
- Combination of assessment information in the summary tables for further consideration and input by NGNP Project R&D. Summary tables are attached as separate files at the end of section 3.2 of this report.
- Coalescence of assessment information in the conclusion tables, with a final statement about each of the identified issues. Conclusion tables are included in section 5.0 of this report.

The conclusion tables present the identified issues, a compilation, by reactor designer, of comments indicating whether the issues are addressed in the PCDR, and each reactor designer's relevant DDNs. The conclusion tables also contain the input provided by the NGNP internal Project R&D organization, for each identified issue. The final (i.e., right-hand) column in the conclusion tables contains an answer to the question: "*Is this item being addressed or does it pose a new risk?*" That column contains an overall assessment of whether the issue is being addressed, and also an assessment by organization (AREVA, GA, WEC, and NGNP R&D).

The conclusion tables represent the fourth and final step in determining whether the summarized PIRTs are addressed by the PCDRs.

Further description and presentation of the conclusion tables are contained in section 5.0 of this report, "*Conclusions*".

2.5 PIRT/PCDR reconciliation task deliverables

The deliverables for this task consist of:

- The data collection, summary, and conclusion tables that were used to amass most of the information regarding the reconciliation of the PIRTs and PCDRs, and
- This report document, which presents the Team's work process and conclusions in comparing the summarized PIRTs vs. R&D efforts performed or currently planned by the NGNP Project and its contractors.

3.0 COMPARISON OF SUMMARIZED PIRTS AGAINST VENDOR PCDRs

3.1 Data collection tables

Described and attached in this section are the data collection tables. There are three sets of data collection tables, each corresponding to the PIRT/PCDR assessment performed for one of the reactor vendors, labeled as follows:

- Tables 1A (AREVA) through 1F (AREVA) – Data collection tables corresponding to assessment of the summarized PIRTs vs. the NGNP PCDR generated by AREVA.

- Tables 1A (GA) through 1F (GA) – Data collection tables corresponding to assessment of the summarized PIRTs vs. the NGNP PCDR generated by General Atomics.

- Tables 1A (WEC) through 1F (WEC) – Data collection tables corresponding to assessment of the summarized PIRTs vs. the NGNP PCDR generated by Westinghouse.

Each of the three sets of data collection tables are arranged in accordance with the six PIRT technical areas, and reflect the comparisons made by the Design Integration Review Team members between the needs identified by NRC and the R&D performed or planned by the NGNP reactor design contractors.

The data collection tables display columns that contain:

- *Identifying Item Numbers*. These numbers were established for referencing convenience and are specific to this report.

- *NRC Needs/Issues Identified in the PIRTs*. Per the task plan, these were extracted from ORNL's Gap Analysis document. They were also checked against NRC's issued PIRT documents to ensure that significant issues are captured. The Gap Analysis should be consulted if the reader requires more detailed context to understand the issue as it is represented in the data collection tables.

- *Applicable R&D or Already-Identified Solution*. These are excerpts from the PCDRs that provide a measure of proof that the PIRT items cited are being addressed in R&D that is ongoing or already performed, or have been recognized for future R&D. In most cases they are quotes from the PCDRs. In a few cases in which a direct quote would be unmanageably long for presentation in a table, a summary statement is used. Each entry in this column contains a reference to the PCDR section from which it is extracted, or reference to another document if appropriate. The term "already-identified solution" applies to instances in which NRC's concern is already being addressed in the design, or R&D that has already been performed.

- Related Design Data Needs (DDNs). These were extracted from the PCDRs to the extent possible, and also from a master list of DDNs maintained by the NGNP Project staff. Related DDNs were included in the input tables because they are indicators of areas of needed information already identified by the designers that will require some form of technology development, component verification, or process verification.

- *Comments/conclusions* made by the Design Integration Review Team members, specifying whether the identified needs are being met and/or planned for, or have not been addressed or recognized. Information in the Comments/Conclusions column indicating a need not being met is shown in bold italics in the tables.

Electronic versions of the data collection tables are attached here:

C:\Documents and Settings\bill.mangiant **Design Integration Review Team Data Collection Tables AREVA**	C:\Documents and Settings\bill.mangiant **Design Integration Review Team Data Collection Tables General Atomics**	C:\Documents and Settings\bill.mangiant **Design Integration Review Team Data Collection Tables Westinghouse**

3.2 Summary tables

Described and attached in this section are the summary tables. There is a single set of summary tables, labeled Table 2A through 2F.

The set of six summary tables is arranged in accordance with the six PIRT technical areas, reflecting the comparisons made by the Design Integration Review Team members between the needs identified by NRC and the R&D performed or planned by all three NGNP reactor design contractors.

The summary tables display columns that contain:

- *Identifying Item Numbers*. Same identifiers as used in the data collection tables.

- *NRC Needs/Issues Identified in the PIRTs*. Same needs/issues as in the data collection tables.

- *Applicable R&D or Already-Identified Solution*. A compilation of the PCDR excerpts contained in the data collection tables, arranged side-by-side for purposes of review and comparison.

- *Comments/conclusions* made by the Design Integration Review Team members. All of the comments/conclusions contained in the data collection tables are displayed in this column, for purposes of compilation. Each entry is preceded by a bullet “Based on review of the _____ PCDR”, to identify whether the comment/conclusion relates to the NGNP PCDR generated by AREVA, General Atomics, or Westinghouse.

- *NGNP R&D Response.* The set of summary tables went through an intermediate step of being submitted to the NGNP Project internal R&D organization for their input, which is included in the *Comments/conclusions* column of the summary tables. NGNP Project R&D provided input on every identified issue.

Electronic versions of the summary tables are attached here:

C:\Documents and Settings\bill.mangiant

Design Integration Review Team Summary Tables

4.0 PROJECT R&D INPUT

As indicated in sections 2.3 and 3.2, appropriate information from the data collection tables (reflecting the Design Integration Review Team comparisons of the summarized PIRT issues against the AREVA, General Atomics and Westinghouse PDCRs) was compiled in the summary tables, which were then submitted to the NGNP Project internal R&D organization for their input.

The NGNP Project R&D organization maintains a database of R&D program plans organized into the following areas:

- Fuel Design and Fabrication
- Fuel Irradiation Testing
- Fuel Accident Condition Testing
- Fuel Performance Analysis Modeling
- Fuel Fission Product Transport Modeling
- Waste
- Graphite
- Methods
- High Temperature

That database contains NRC-identified research need descriptions, R&D program numbers, and also indications of whether the planned R&D efforts consist of analyses, experiments, or both. Thus, the R&D organization was in a position to provide relevant feedback corresponding to the issues that are the basis for this PIRT/PCDR reconciliation, and indeed they did provide input on every issue. In many instances, they were able to confirm that R&D was ongoing or planned in areas where R&D coverage was not apparent from the PCDR review.

The NGNP Project R&D input is included in the *Comments/conclusions* column of the summary tables, which are provided in section 3.2 of this report.

At this stage of gathering and processing assessment information, the Design Integration Review Team had developed reconciliation data on all of the summarized PIRT issues from the three PCDRs and from the NGNP Project R&D organization. The Team was then in a position to draw final conclusions on whether R&D efforts required to satisfy NRC's identified needs are in fact ongoing or planned. Those conclusions are presented in section 5.0 of this report.

5.0 CONCLUSIONS

5.1 Characterization of the PIRTs

Clearly, the NRC considers that they will face challenges in performing the safety review and design verification of the NGNP that they have not faced before, in their role as regulator of commercial power reactors in the U.S. Having received their charter, along with DOE, to develop a licensing strategy for the NGNP under the Energy Policy Act of 2005, NRC has been proactive in identifying phenomena and issues that they believe will be of concern to them in the regulatory process. They have identified these items in a general, philosophical way as analytical tools and data that they will need to perform their regulatory function, without necessarily being specific about how these tools and data will be made available and where they will come from.

Since virtually all of the development activity for NGNP is being performed by DOE and its contractors (including direct national laboratory resources and the three commercial reactor designers), it is reasonable to assume that much of the information required by NRC will come from the NGNP Project effort, with the possible exception that NRC may need to develop some confirmatory analysis codes on their own. Even in those cases, one might expect that the confirmatory codes would be based on NGNP-developed codes used as templates. Hence, a reconciliation of the Phenomena Identification and Ranking Tables (PIRTs) generated by NRC, with the Pre-conceptual Design Reports (PCDRs) generated by the NGNP Project, was seen as prudent to ensure that the NGNP Project is on track to satisfy NRCs anticipated needs.

The PIRTs (or even the PIRTs as they are summarized in the ORNL Gap Analysis document) represent a large body of information, however significant themes emerge from that information, some because they are visited and re-visited in the course of the documents. In this report, the issues have been encapsulated. For uniformity, they have been repeated in the data collection, summary, and conclusion tables, so the reader may see them at the same level of detail in all of the tables. For purposes of offering a further reduction of data at this concluding section of the reconciliation report, the issues identified by NRC in the six major technical areas of the PIRTs are characterized as follows:

- *Accidents and Thermal Fluids* – In this technical area, NRC identified analytical and modeling needs and issues associated with understanding the NGNP design and performance, specific phenomena related to the NGNP fuel, reactor, and safety systems, as well as databases that will be required for purposes of performing independent confirmatory analyses.

- *Reactor Physics and Neutronics* (including criticality calculations and experiments) - In this technical area, NRC identified specific reactor phenomena such as monitoring of temperature in the core, various aspects of reactivity control, accounting for variations in fuel production, and the importance of reactor physics testing to confirm the acceptability of unique core configurations in the NGNP designs.

- *Fuel Performance, Fission Product Transport, and Dose* - In this technical area, NRC identified fuel and fission product transport issues associated with safety documentation needs, development, validation and verification of analytical codes, general needs for information on materials and components, safety classification and function of NGNP design features, the containment vs. confinement issue, and the need for integrated testing of structures, systems and components.

- *High-Temperature Materials (metallic)* - In this technical area, NRC identified specific physical materials data needs for high temperature metallic and composite components, uniquely important phenomena such as emissivity of metallic surfaces, effects of aging and irradiation on insulation and in-vessel structures, potential failure of components such as control rods, isolation valves and the intermediate heat exchanger (IHX), development of structural mechanics codes to model flaw initiation and propagation in new materials (i.e., not yet codified by ASME), and resolution of potential field fabrication issues with the larger vessels.

- *Graphite Materials* - In this technical area, NRC identified issues associated with establishing ASME-approved grades of graphite, generating the necessary materials characterization and performance data to confirm standard graphite specifications, developing "whole core" analytical models to predict stress states in graphite components, conducting research to understand standard degradation effects such as irradiation and more unique effects such as air/water ingress, accounting for statistical variations during fabrication, and understanding the effects of moving surface interactions (tribology) of graphite components in a high temperature helium environment.

- *Process Heat for Hydrogen Production* - In this technical area, NRC declared that they needed to know more specifics about the design of the hydrogen production systems to develop more solid impressions and needs; however they offered a few preliminary concerns. These included possible impact of oxygen and hydrogen gases (escaped from the H2 facility) on the reactor facility, potential failure modes of safety grade equipment caused by overpressurization or corrosion initiated in the H2 facility, and potential inadvertent reactivity addition and/or chemical attack due to water ingress, if a steam generator is included in the design.

5.2 Items that were not fully addressed and may represent project risk

After assessing the summarized PIRTs against the AREVA, General Atomics, and Westinghouse PCDRs, and receiving input from the NGNP internal R&D organization, the Design Integration Review Team was able to conclude that the vast majority of NRC's identified issues are being addressed by R&D efforts that are either planned or ongoing. Conclusions on whether NRC's identified needs were being met were based on the totality of information available from the three reactor vendor PCDRs (since no design down-select has yet been made), and the additional information provided by NGNP R&D. The conclusion tables follow in their entirety in section 5.4.

The exception items – those that were not fully addressed in the PCDRs and might therefore represent previously unidentified risk to the project – are extracted from the conclusion tables as follows:

- *Item A-1 from "Accident and Thermal Fluids", dealing with core coolant bypass flow phenomena in normal operation* – This item is, for the most part, being addressed, however R&D for instrumentation required to measure in-core temperatures was noted as a soft spot.

- *Item B-4 from "Reactor Physics and Neutronics", dealing with high-temperature in-core neutron detectors and the ability to detect abnormal power distributions* – Again, this item is, for the most part, being addressed, however instrumentation R&D is noted as a soft spot. In this case, instrumentation refers to high temperature in-core neutron detectors required to map power distributions.

- *Item B-5 from "Reactor Physics and Neutronics", dealing with potential control rod misalignments and effects on power tilting* - This item has not been addressed in specific terms. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses.

- *Item B-6 from "Reactor Physics and Neutronics", dealing with potential steam/water ingress events* - This item is being addressed, however, for the most part, the position seems to be that the phenomenon will not be analyzed directly, but the design will demonstrate that water ingress is not a credible event. The incredibility argument is typically a difficult one to take with NRC. If this argument continues to be used, the final NGNP design should be capable of demonstrating convincingly that the event is prevented by specific design features.

- *Item B-7 from "Reactor Physics and Neutronics", dealing with the issue of graphite moderator producing a "harder" thermal-neutron energy spectrum that may cause difficulties with reactivity control* - Although all 3 reactor designers have strategies for reactivity control, this item is not being addressed specifically. This does not seem to be a high risk item, but is a safety/criticality analysis issue that may require some study and sensitivity analysis.

- *Item B-8, dealing with the potential for variations in fuel characteristics to cause difficulties in accurately calculating fuel and moderator temperature coefficients* - This item is not being addressed specifically. However, indications are that the more specific parametric analyses that might be required to satisfy NRC's future needs are within current capabilities.

- *Item B-9, dealing with safety qualification of operators and equipment in an extended ATWS event* - This item is not being addressed. However, the ATWS event and the safety functions required by it are well established in the licensing process. This would appear to be a safety analysis issue that will be handled in later stages of the project without the need for R&D.

- *Item C-1 from "Fuel Performance and Fission Product Transport and Dose", dealing with general safety analysis and safety documentation needs* - Most of the subparts of this item are being addressed. The area not being addressed is comprised of the parts of the issue that deal with hold-up and attenuation of fission products on specific surfaces in the primary circuit components and in the reactor building, related safety analysis assumptions, and preservation of those assumptions with Technical Specifications. This represents a risk to the project in terms of securing a successful safety evaluation from NRC.

- *Item C-2 from "Fuel Performance and Fission Product Transport and Dose", dealing with model development and V&V* - As in item C-1, most parts of this item are covered with anticipated R&D, however there are vulnerabilities dealing with the assumptions and modeling of the hold-up and attenuation of fission products on specific surfaces in the primary circuit components and in the reactor building. Item C-2 adds the issue of reactions between the fission products with the reactor building materials, which will depend on the confinement design. These vulnerabilities represent risks to the project in terms of securing a successful safety evaluation from NRC.

- *Item C-3 from "Fuel Performance and Fission Product Transport and Dose", dealing with materials and component data* - About half of the subparts of this item are being addressed, with the balance dealing again with the uncertainties surrounding assumptions and modeling of the hold-up and attenuation of fission products on specific surfaces in the primary circuit components and in the reactor building. This item adds (to the issues already raised in items C-1 and C-2) concerns over possible contamination events, and formation and stability of the surface films that will be depended upon to retain fission products. These concerns represent risks to the project in terms of securing a successful safety evaluation from NRC.

- *Item C-4 from "Fuel Performance and Fission Product Transport and Dose", dealing with reactor component and containment/confinement configuration, and their roles in the safety case* - Again, the themes of modeling and budgeting fission product hold-up at each step along the transport pathway emerge in this item, and are not apparently addressed in current R&D efforts or planning. These concerns represent risks to the project in terms of securing a successful safety evaluation from NRC.

- *Item D-4 from "High Temperature Materials (metallic)", dealing with effects on insulation* - This item is being addressed; however it appears to have been addressed thus far only marginally. Issues over insulation degradation and subsequent degradation on plant safety performance are typically of great concern to NRC. These concerns represent risks to the project in terms of securing a successful safety evaluation from NRC.

- *Item D-11 from "High Temperature Materials (metallic)", dealing with the need for flaw assessment procedures, particularly for pressure boundary crack initiation and propagation* – Generally speaking, this item is being addressed, in that there is recognition that better and more material-specific structural mechanics models are required for the NGNP. However, the recognition seems to be weak, in comparison to discussion of other types of modeling codes that are required for the project. This almost surely revolves around the uncertainties associated with materials selection and the possible lack of qualification and ASME codification (depending on the materials chosen). This issue of materials selection, qualification, and structural mechanics modeling would appear to be a significant risk to overall project success.

- *Item D-15 from "High Temperature Materials (metallic)", dealing with anticipated oversized reactor pressure vessel and the potential need for some degree of site fabrication* - This item is being addressed; however, the matter of alloy selection, which will impact vessel fabrication, is a core issue that does not appear to have been settled yet. This issue would appear to be a significant risk to overall project success.

- *Item E-9 from “Graphite”, dealing with potential blockage of fuel element block or reactivity control block due to graphite spalling* - This item is being addressed weakly at this stage of the design effort. This concern represents a risk to the project in terms of securing a successful safety evaluation from NRC.

- *Item F-1 from “Process Heat for Hydrogen”, dealing with impact of escaping gases such as H2 and O2 on the integrity of reactor systems, structures and components* - Indications are that this item has been addressed only preliminarily thus far. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses.

- *Item F-2 from “Process Heat for Hydrogen”, dealing with selection of helium as the heat transfer fluid, failure of the IHX, and possible subsequent overpressurization of the secondary side of the plant* - Indications are that this item has been addressed only preliminarily thus far. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses.

- *Item F-3 from “Process Heat for Hydrogen”, dealing with failure of the IHX and subsequent damage to safety-related SSCs, initiated by corrosive chemicals from the H2 production facility* - Indications are that this item has been addressed only preliminarily thus far. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses.

- *Item F-4 from “Process Heat for Hydrogen”, dealing with steam generator failures as a possible cause of steam/water ingress events that result in damage to nuclear fuel and graphite components* - This issue seems to have been handled thus far in terms of a non-credibility type of argument with regard to water ingress events. These are typically not easy arguments to make with NRC. These concerns represent risks to the project in terms of securing a successful safety evaluation from NRC. As mentioned above in relation to item B-6, the final NGNP design should be capable of demonstrating convincingly that steam/water ingress events are prevented by specific design features.

- *Item F-5 from “Process Heat for Hydrogen”, dealing with loss of pressurized coolant inventory from the intermediate loop leading to a loss of primary reactor heat sink, leading to IHX failure and loss of reactor primary system coolant* - Indications are that this item has been addressed only preliminarily thus far. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses.

For additional information, Appendix B provides an *Abbreviated Results Table*, which is a compact version of some of the information drawn from the conclusion tables. The Appendix B table is divided into the six major technical areas, and displays item numbers, abbreviated issue descriptions, and answers to the question: *“Is this item being addressed?”* For each issue, that question is answered for AREVA, GA, WEC, NGNP R&D, and the overall project. In most cases, the answer to this question is a straight yes or no, with a few exceptions in which the issue was not applicable, or in which the coverage of the issue was partial. The Appendix B table provides the reader with the opportunity to view each of the categories quickly, and to get a capsule view of how the responses to issues are distributed.

5.3 Risk areas that may require planning for further technology development

Based on the exception items identified and described in section 5.2, the risks identified by this assessment can be further summarized as follows:

- Insufficient development of instrumentation required for measurement of in-core temperatures and mapping of power distributions. (Items A-1 and B-4)

- Insufficient knowledge of potential control rod failures/misalignments and their effects on power tilting. (Item B-5)

- Insufficient knowledge of potential for and consequences of steam/water ingress accidents; possible over-reliance on an "incredibility" position for disposing of these events. (Items B-6 and F-4)

- Insufficient knowledge of potential reactivity control challenges that may be posed by graphite moderation producing a "harder" thermal energy spectrum than typically experienced with light water moderation. (Item B-7)

- Uncertainties in calculating fuel and moderator temperature coefficients associated with variations in fuel characteristics. (Item B-8)

- Insufficient knowledge of safety qualification requirements for operator and equipment in an extended ATWS event. (Item B-9)

- Uncertainties associated with modeling and preserving assumptions associated with fission product holdup, particularly on surfaces of primary circuit components and in the reactor building. (Items C-1, C-2, C-3, and C-4)

- Insufficient knowledge concerning the degradation of insulation due to long periods of exposure to high temperature and irradiation, and further effect of the degraded insulation on plant performance. (Item D-4)

- Insufficient plans for development of structural mechanics codes required to establish integrity and qualification of high temperature metallic components. (Item D-11)

- Insufficient plans to develop new qualified methods for partial site fabrication of oversized RPV. (Item D-15)

- Insufficient knowledge of potential blockage of fuel element or reactivity control blocks due to graphite spalling. (Item E-9)

- Insufficient knowledge of hazards associated with hydrogen production facility. (Items F-1, F-2, F-3, F-4 and F-5)

These risk items all represent areas in which additional technology development efforts, represented by additional Design Data Needs (DDNs), may be required.

5.4 Conclusion tables

The conclusion tables, labeled 3A through 3F, are presented in their entirety on the pages that follow. This set of tables addresses all of the summarized PIRT issues. The conclusion tables represent the culmination of information gathered and assessments performed in the data collection tables and summary tables, as described in sections 2, 3 and 4 of this report.

The conclusion tables display columns for the identified issues, a summary of comments indicating whether the issues are addressed in the PCDRs, relevant DDNs, and input provided by the NGNP Project R&D organization. The final column in the conclusion tables contains an answer to the question: "*Is this item being addressed or does it pose a new risk?*" That question is answered from an overall perspective, and also from the individual perspectives of the AREVA PCDR, the GA PCDR, the WEC PCDR, and the input provided by the NGNP R&D organization.

TABLE 3A – ACCIDENTS AND THERMAL FLUIDS - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
A-1	*Core-Coolant Bypass Flow Phenomena (Normal Operation)* • Overcome difficulties in estimating bypass flow • More complete understanding and accounting of related design features such as fuel blocks (PMR) and core barrel configurations • In-core temperature testing • Parametric analysis of gap configurations to bound questions associated with gap and bypass flows	• *Based on review of AREVA PCDR*: The needs for refined analyses to better understand the core bypass flow phenomenon, and core monitoring instrumentation and testing, have been recognized in the AREVA PCDR. • *Related AREVA DDNs*:: AREVA-011 thru AREVA-022 (modeling codes); AREVA-002 (core barrel); AREVA-030, and AREVA-034 (instrumentation and testing) • *Based on review of GA PCDR*: The needs for refined analyses to better understand the core bypass flow phenomenon, and core monitoring instrumentation and testing, have been recognized in the General Atomics PCDR. • *Related GA DDNs*: C.11.01.01 thru C.11.03.11) , C.11.04.03 (core instrumentation and testing); C.11.03.31 (core instrumentation validation); C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development) • *Based on review of WEC PCDR:* There is no indication that this item has been specifically addressed in the WEC PCDR. • *Related WEC DDNs*: (None) • *NGNP R&D Response*: Experiments and analysis planned in R&D program. Instrumentation R&D is needed. No real funding (except small amount for university) available.	*Overall*: This item is being addressed. Instrumentation R&D is noted as a soft spot. *By Organization*: AREVA - Addressed GA - Addressed WEC - Not Addressed NGNP R&D - Addressed
A-2	*Effective Core Thermal Conductivity* • For prismatic cores – Make available dose and temperature-dependent graphite thermal properties (especially thermal conductivity) to the NRC T/F code suite, to account for large uncertainties as well as for characterization of annealing effects during long-term heat-up D-LOFC accidents. • For pebble bed cores - Also considerable error bounds in effective core thermal conductivity as a function of both temperature and irradiation. Existing correlations available are empirical, but PBMR project has an experimental facility to be used to refine the database.	• *Based on review of AREVA PCDR:* The needs for determining the properties of graphite materials, including thermal conductivity, have been recognized in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-001 thru AREVA-003 (metallics); AREVA-004 thru AREVA-007 (ceramics); AREVA-008 and AREVA-009 (graphites); AREVA-010 (core internal structure) • *Based on review of GA PCDR*: The needs for determining the properties of graphite materials, including thermal conductivity, have been recognized in the General Atomics PCDR. • *Related GA DDNs*: C.11.03.16 (graphite thermal properties data); C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development); C.16.00.01 thru C.16.00.06 (RCCS) • *Based on review of WEC PCDR*: Indications are that this item has been addressed in the WEC PCDR, and will be further addressed as the project progresses. • *Related WEC DDNs*: NHSS-01-03 (fuel graphite testing); NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: R&D planned in Graphite program to cover this.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
A-3	*Afterheat Correlations* • Peak fuel temperatures in the D-LOFC	• *Based on review of AREVA PCDR*: There is no indication that this item has been specifically addressed in the AREVA PCDR.	*Overall*: This item is being addressed.

TABLE 3A – ACCIDENTS AND THERMAL FLUIDS - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	accident are very sensitive to the afterheat (vs. time) to the same extent as they are to the core thermal conductivity function. Afterheat correlations are sensitive to fuel type and burn-up histories. Tracking fuel histories during operation can be challenging, and afterheat validation data is more difficult to obtain for long times after shutdown.	• *Related AREVA DDNs*: AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR*: This phenomenon and the need for improved models have been addressed in the General Atomics PCDR. • *Related GA DDNs*: C.07.02.01 thru C.07.02.09 (fuel performance data); C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development); C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS); C.16.00.01 thru C.16.00.06 (RCCS) • *Based on review of WEC PCDR*: There is general discussion of limiting of peak fuel temperatures; however there is no indication that this item has been specifically addressed in the WEC PCDR. • *Related WEC DDNs*: NHSS-01-02 (fuel heating tests for accident conditions) • *NGNP R&D Response*: No R&D planned here. It would appear that sensitivity analysis could adequately deal with this.	*By Organization*: AREVA – Not Addressed GA - Addressed WEC – Not Addressed NGNP R&D - Addressed
A-4	*Core Effective Pressure Drop* • Standardized and well-documented correlations for core pressure drop; conformation data may be needed for low-flow cases to better characterize flow distribution and plume formation (for the P-LOFC) and in-core airflow distributions during air ingress accidents. • PBR - parametric analyses using established ranges of different packing fractions to define a performance envelope.	• *Based on review of AREVA PCDR:* There is no indication that this item has been specifically addressed in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR:* This phenomenon and related efforts to date have been addressed in the General Atomics PCDR. • *Related GA DDNs*: C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development) • *Based on review of WEC PCDR:* It appears that the PBR will have the required test capability, but there is no indication in the WEC PCDR that parametric analyses based on packing fractions have been or will be addressed. • Related WEC DDNs: (None) • *NGNP R&D Response*: Covered in Methods plan.	*Overall*: This item is being addressed. *By Organization*: AREVA – Not Addressed GA - Addressed WEC – Not Addressed NGNP R&D - Addressed
A-5	*RCCS Performance during LOFC* • Simulate RCCS safety functions in detail, with its predominantly radiant heat transfer coupling to the RPV and other heat transfer mechanisms within the reactor cavity. RCCS functions include maintaining the reactor cavity liner concrete temperature below prescribed limits, preventing the RPV peak temperature from exceeding limits	• *Based on review of AREVA PCDR:* The needs for modeling and simulation code development described in this item have been recognized in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR:* This phenomenon, related efforts to date, and the need for modeling and simulation codes have been addressed in the General Atomics PCDR. • *Related GA DDNs*: C.16.00.01 thru C.16.00.06 (RCCS)	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC – Not Addressed NGNP R&D – Not Addressed

TABLE 3A – ACCIDENTS AND THERMAL FLUIDS - CONCLUSIONS

Item	NRC Need/Issue Identified	Design Integration Review Team Comments/Conclusions	Is this Item being addressed or does it pose a new risk?
	during LOFC events, and minimizing parasitic heat losses during normal operation. • Models may be needed to simulate large pressure pulses in D-LOFC accidents that could damage the RCCS, reducing cooling and/or opening up another release path for air or water ingress to the reactor cavity, and perhaps for FPT out to the environment.	• *Based on review of WEC PCDR:* The need for performance verification has been recognized. However, it is impossible to determine if safety functions will be modeled in detail or if large pressure pulses will be simulated in the WEC PCDR. • *Related WEC DDNs*: (None) • *NGNP R&D Response:* R&D is planned to validate heat transfer in RCCS. Impact of pressure pulse on RCCS performance is open issue. (It could be done, but not currently in the plan.)	
A-6	*Fuel Performance Models* • Aspects of maximum fuel temperature plus time-at-temperature histories (critical limiting factors) for all fuel regions provide inputs to fuel failure models, to determine source terms and dose-vs.-frequency estimates. • Chemical reactions in air or water ingress accidents, which depend on temperature and should be included in the T/F codes. Especially for fast transients, detailed temperature profiles of the fuel and graphite should be taken into account for thermal stress calculations.	• *Based on review of AREVA PCDR*: The needs for modeling and simulation code development described in this item have generally been recognized in the AREVA PCDR. However, it is not possible to determine whether these codes will include: - time-at-temperature histories for all fuel regions - chemical reactions in water ingress accidents (AREVA seems to have determined that these are not credible events) - detailed temperature profiles of fuel and graphite in fast transients • *Related AREVA DDNs*: AREVA-014, AREVA-016, and AREVA-022 (computer codes) • *Based on review of GA PCDR*: The needs for modeling and simulation code development described in this item have been recognized in the General Atomics PCDR. • *Related GA DDNs*: C.07.01.01 thru C.07.01.07 (fuel fabrication); C.07.02.01 thru C.07.02.09 (fuel performance data); C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); N.07.05.01 thru N.07.05.14 (integrity testing of fuel and graphites) • *Based on review of WEC PCDR*: The needs for modeling and code development have been recognized in the WEC PCDR. • *Related WEC DDNs*: NHSS-01-02 (fuel heating testing) • *NGNP R&D Response*: R&D's Fuel program has this type of model as a key element.	*Overall*: This item is being addressed. *By Organization*: AREVA – Partially Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
A-7	*Air Ingress Phenomena* • With little or no detail available about the confinement, only generalized studies and experiments would be practical. Bounding analytical studies could be useful in determining positive and negative features of proposed design characteristics. The major	• *Based on review of AREVA PCDR*: The need for greater understanding of the air ingress phenomenon has been recognized in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-028 and AREVA-031 (SCS and RCCS) • *Based on review of GA PCDR*: The need for greater understanding of the air ingress phenomenon has been recognized in the General Atomics PCDR. • Related GA DDNs: C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); N.07.05.02 thru	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed

TABLE 3A – ACCIDENTS AND THERMAL FLUIDS - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	features of general interest would be quantification of long-term air in-leakage into the confinement, and the mixing and stratification characteristics of gases in prototypical cavities within the confinement.	N.07.05.05 (graphite and other oxidation rates in air); C.11.03.23 (graphite oxidation) • *Based on review of WEC PCDR*: The need for greater understanding of the air ingress phenomenon has been recognized in the WEC PCDR. • Related WEC DDNs: (None) • *NGNP R&D Response*: This is part of Methods program.	NGNP R&D - Addressed
A-8	*Long-term analysis need* - Comprehensive suite of verified and validated accident simulation codes (core thermal-fluids, core neutronics, whole-plant transient behavior, confinement analysis, and chemical reactions), agreed-upon accident cases for regulatory acceptance, and robust supporting databases that NRC can use for independent confirmatory analysis of candidate plant and confinement designs and options.	• *Based on review of AREVA PCDR*: Long-term analysis needs for computer code development have been recognized in the AREVA PCDR. Databases have been addressed by AREVA in terms of candidate alloys and fuel materials. • *Related AREVA DDNs*: AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR*: Long-term analysis needs for computer code development have been recognized in the General Atomics PCDR. Databases required for confirmatory use by NRC have not been addressed in the General Atomics PCDR. • *Related GA DDNs*: C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development) • *Based on review of WEC PCDR*: Long-term analysis needs for computer code development, and supporting databases, are either completed, underway, or planned for the future, as discussed in the WEC PCDR. • *Related WEC DDNs*: NHSS-01-01 (Data to extend the irradiated fuels qualification database); NHSS-01-02 (Extend heat up data under accident conditions); NHSS-01-03 (Extend temperature-fluence envelope of fuel graphite); NHSS-02-01 and NHSS-02-02 (Extend irradiated materials qualification database for Reflector Graphite); HTS-01-1 thru HTS-01-19 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics) • *NGNP R&D Response*: This piece of work is covered in Methods part of R&D program.	*Overall*: This item is being addressed. *By Organization*: AREVA – Partially Addressed GA – Partially Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3B – REACTOR PHYSICS AND NEUTRONICS - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
B-1	Time-dependence and spatial distribution of decay heat as a major factor in determining maximum fuel temperature during a D-LOFC.	• *Based on review of AREVA PCDR*: The needs for modeling and simulation code development described in this item have been recognized in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR*: The needs for modeling and simulation code development described in this item have been recognized in the General Atomics PCDR. • *Related GA DDNs*: C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development); C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS); C.16.00.01 thru C.16.00.06 (RCCS) • *Based on review of WEC PCDR*: The needs for fuel testing and modeling to determine fuel temperature in a D-LOFC are recognized in the WEC PCDR. • *Related WEC DDNs*: NHSS-01-01 thru NHSS-01-03 (fuel testing) • *NGNP R&D Response*: All the codes will have spatial and time dependence in them. No experimental plans for obtaining such data.	*Overall*: This item is being addressed. *By Organization*: AREVA – Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
B-2	Control and shutdown rod worth and reserve shutdown worth as required for hot and cold shutdown.	• *Based on review of AREVA PCDR*: The needs for understanding control and reserve shutdown capability, as described in this item, is recognized in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR*: The needs for understanding control and reserve shutdown capability, as described in this item, is recognized in the General Atomics PCDR. • *Related GA DDNs*: C.11.01.03 and C11.01.04 (control rod and reserve shutdown design verification); C.11.03.22 (reserve shutdown pellet process development); C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS); C.16.00.01 thru C.16.00.06 (RCCS) • *Based on review of WEC PCDR*: The need for determining/validating rod worths has been recognized in the WEC PCDR, and work is either completed or is in progress. • *Related WEC DDNs*: (None) • *NGNP R&D Response*: No experimental work planned. Better analytic tools are under development to be able to calculate shutdown and control rod worths. There are adequate benchmarks available for both design options.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
B-3	Sudden positive reactivity insertion due to pebble core compaction (packing fraction) due to earthquake.	• *NA to AREVA and GA PCDRs:* This is a PBR phenomenon and is not applicable to the PMR core. It does not apply to either AREVA or General Atomics designs.	*Overall*: This item is being addressed.

TABLE 3B – REACTOR PHYSICS AND NEUTRONICS - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *Based on review of WEC PCDR*: It appears that WEC has the test facilities to simulate a condition of increased packing density in the PBMR core; however there is no indication that this item has been specifically addressed in the WEC PCDR. • Related WEC DDNs: (None) • *NGNP R&D Response*: Methods program has looked at this and has tools to do this type of problem.	*By Organization*: AREVA – Not Applicable GA – Not Applicable WEC – Not Addressed NGNP R&D - Addressed
B-4	For tests at both PMRs and PBRs, consideration should be given (at least in the first core) to use of high-temperature in-core neutron detectors that can provide maps of axial and azimuthal power distributions and core-inner-to-outer-radius power tilts; these detectors would likely be located only in the inner and outer reflectors rather than in the core, due to temperature and connection limitations. • PMR concern - Whether improper axial-loading of fuel blocks during refueling can lead to an undetected power distribution anomaly and result in excessive operating fuel temperatures. • PBR concern - Radial and azimuthal power distributions in the mixed-fuel pebble bed are not well known, and there are indications from melt-wire tests conducted in the AVR (Germany) suggesting that pebbles near the walls of the reflector experienced unexpectedly high fuel temperatures.	• *Based on review of AREVA PCDR*: The need for core monitoring instrumentation has been recognized in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-030, and AREVA-034 (instrumentation and testing) • *Based on review of GA PCDR*: The need for core monitoring instrumentation has been recognized in the General Atomics PCDR. • Related GA DDNs: C.11.01.01 thru C.11.03.11) , C.11.04.03 (core instrumentation and testing); C.11.03.31 (core instrumentation validation); C.11.04.03 (neutron detector service equipment design) • *Based on review of WEC PCDR*: There is discussion of instrumentation and monitoring, however there is no indication that in-core instrumentation has been specifically addressed in the WEC PCDR. • Related WEC DDNs: (None) • *NGNP R&D Response*: R&D is needed. No funding yet except a small amount allocated to university grant program.	*Overall*: This item is being addressed. Instrumentation R&D is noted as a soft spot. *By Organization*: AREVA - Addressed GA - Addressed WEC – Not Addressed NGNP R&D - Addressed
B-5	In both the PMR and PBR, control rod misalignments in the outer reflector during operation would result in azimuthal power tilting that could cause xenon-135-induced oscillations when the misalignment is corrected; however, this needs to be verified by analysis and confirmed by test.	• *Based on review of AREVA PCDR*: There is no indication that this item has been specifically addressed in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-030 (testing of control rod drive system) • *Based on review of GA PCDR*: The issue of control rod misalignment has been addressed in the General Atomics PCDR with descriptions of the design features that maintain control rod alignment. • *Related GA DDNs*: C.11.01.03 (control rod design verification); C.11.03.02 thru C.11.03.06 (control rod failure modes and integrity); 11.03.24 (control rod high temp materials properties)	*Overall*: This item has not been addressed in specific terms. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses. *By Organization*: AREVA – Not Addressed

TABLE 3B – REACTOR PHYSICS AND NEUTRONICS - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *Based on review of WEC PCDR*: The general need to understand xenon oscillations was identified. However, there were no specifics in the WEC PCDR that mentioned outer reflector control rod misalignments leading to azimuthal power tilting and possible xenon oscillations. • *Related WEC DDNs*: (None) • *NGNP R&D Response*: No plans in R&D. Appears to be a design issue.	GA – Not Addressed WEC – Not Addressed NGNP R&D – Not Addressed
B-6	Replacing helium with a hydrogen-bearing compound such as in a steam/water ingress event may produce a pronounced positive reactivity. Steam/water ingress tends to have a positive reactivity effect due to increased neutron moderation and reduced neutron leakage.	• *Based on review of AREVA PCDR*: The need for greater understanding of the water ingress phenomenon has been recognized in the AREVA PCDR. • *Related AREVA DDNs*: (None) • *Based on review of GA PCDR*: The need for greater understanding of the water ingress phenomenon has been recognized in the General Atomics PCDR. • *Related GA DDNs*: C.11.01.03 and C11.01.04 (control rod and reserve shutdown design verification) • *Based on review of WEC PCDR*: There is discussion of reactivity control, and also of steam generator design to prevent water ingress; however there is no indication that this item has been specifically addressed in the WEC PCDR. • Related WEC DDNs: (None) • *NGNP R&D Response*: Current R&D does not address water ingress given very low probability of occurrence. Would change if steam generator is part of primary system.	*Overall*: This item is being addressed. For the most part, the position seems to be that the phenomenon will not be analyzed directly, but the design will demonstrate that water ingress is not a credible event. The incredibility argument is typically a difficult one to take with NRC. If this argument continues to be used, the final NGNP design should be capable of demonstrating convincingly that the event is prevented by specific design features. *By Organization*: AREVA - Addressed GA - Addressed WEC – Not Addressed NGNP R&D – Not Addressed
B-7	With a higher atomic mass moderator such as carbon, the mean thermal energy of neutrons will be higher than that for hydrogen bound with oxygen in water; that is, graphite will tend to produce a "harder" thermal-neutron energy spectrum than would water-moderated systems. Thus, the moderator temperature-dependent reactivity coefficient (MTC) in both PMR and PBR depends upon the change of thermal-neutron energy spectrum with temperature, with possibly large effects on reactivity. Concerns are for effects on core transient	• *Based on review of AREVA PCDR*: The AREVA PCDR has addressed AREVA's design strategies for reactivity control and neutron control, as features of the design. AREVA has not specifically addressed NRC's concern over the "harder thermal-neutron energy spectrum" and its "possibly large effects on reactivity". • *Related AREVA DDNs*: AREVA-030 (RCCS); AREVA-031 (neutron control system drive mechanism) • *Based on review of GA PCDR*: The General Atomics PCDR has addressed the General Atomics design strategies for reactivity control and neutron control as features of the design, and has stated that all credible reactivity addition events can be controlled. General Atomics has not specifically addressed NRC's concern over the "harder thermal-neutron energy spectrum" and its "possibly large effects on reactivity". • *Related GA DDNs*: C.11.01.03 and C11.01.04 (control rod and reserve shutdown design verification); C.16.00.01 thru C.16.00.06 (RCCS)	*Overall*: Although all 3 reactor designers have strategies for reactivity control, this item is not being addressed specifically. This does not seem to be a high risk item, but is a safety/criticality analysis issue that may require some study and sensitivity analysis. *By Organization*:

TABLE 3B – REACTOR PHYSICS AND NEUTRONICS - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	behavior and passive safety shutdown characteristics.	• *Based on review of WEC PCDR*: There are discussions of the reactivity control systems and inherent ability of the core to resist increased reactivity, however there is no indication that "harder thermal neutron energy spectrum" and its "possibly large effects on reactivity" have been addressed in the WEC PCDR. • *Related WEC DDNs*: (None) • *NGNP R&D Response*: Unclear what is meant here. With H_2O ingress, spectrum will soften and the additional moderation by steam will cause reactivity increase. Without water, both systems exhibit a large negative temperature coefficient.	AREVA – Not Addressed GA – Not Addressed WEC – Not Addressed NGNP R&D – Not Addressed
B-8	Variations in fuel enrichments, kernel diameters, coatings, and density of packing (PMR vs. PBR) must be accounted for in calculating the neutron reaction self-shielding effects in both the resonance or epithermal region and the thermal region of the neutron energy spectrum, to properly calculate the Doppler fuel temperature coefficient of reactivity and the MTC.	• *Based on review of AREVA PCDR*: There is no indication that this item has been specifically addressed in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-014 (fuel performance modeling and codes) • *Based on review of GA PCDR*: The treatment of fuel production in terms of establishing standard statistically based specifications has been addressed in the General Atomics PCDR. It is not apparent that the phenomena cited in this item have been used as a basis for that specification. • *Related GA DDNs*: C.07.01.01 thru C.07.01.07 (fuel fabrication); C.07.02.01 thru C.07.02.09 (fuel performance data) • *Based on review of WEC PCDR*: There are discussions of reactivity control and fuel fabrication; however there is no indication that this item regarding variations has been specifically addressed in the WEC PCDR. • *Related WEC DDNs*: (None) • *NGNP R&D Response*: It appears NRC wants to understand how small variations in all fuel parameters (thickness, density, packing fraction, etc.) impact physics parameters. This can be done by any of the vendors or by INL tools.	*Overall*: This item is not being addressed specifically. However, indications are that the more specific parametric analyses that might be required to satisfy NRC's future needs are within current capabilities. *By Organization*: AREVA – Not Addressed GA – Not Addressed WEC – Not Addressed NGNP R&D – Not Addressed
B-9	Due to concerns over control rod drive reliability and re-criticality after Xenon-135 decay, the plant operator retains the safety function of achieving long-term hot and cold shutdown during an extended ATWS; and the equipment used by the operator to carry out this safety function, whether located in the control room or in a remote location, must be appropriately qualified to execute that safety function.	• *Based on review of AREVA PCDR*: In the AREVA PCDR, this item appears to be addressed in the design. All appropriate systems appear to be safety grade. However, there is no indication that re-criticality following xenon decay in an ATWS event has been specifically addressed in the AREVA PCDR. • *Related AREVA DDNs*: AREVA-006 (control rod sheaths) • *Based on review of GA PCDR*: In the General Atomics PCDR, this item appears to be addressed in the design. However, there is no indication that re-criticality following xenon decay in an ATWS event has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs*: C.11.01.03 and C11.01.04 (control rod and reserve shutdown design verification); C.11.03.02 thru C.11.03.06 (control rod failure modes and integrity); 11.03.24 (control rod high temp materials properties)	*Overall*: This item is not being addressed. However, the ATWS event and the safety functions required by it are well established in the licensing process. This would appear to be a safety analysis issue that will be handled in later stages of the project without the need for R&D. *By Organization*: AREVA – Not Addressed

TABLE 3B – REACTOR PHYSICS AND NEUTRONICS - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *Based on review of WEC PCDR*: There is no indication that this item has been specifically addressed in the WEC PCDR. • *Related WEC DDNs*: (None) • *NGNP R&D Response*: Design issue – not R&D	GA – Not Addressed WEC – Not Addressed NGNP R&D – Not Addressed
B-10	The uniqueness of configuration (tall, thin annular core) of current PMR and PBR designs and high operating temperatures require detailed reactor physics testing of the first unit as a function of core burnup, and of the start-ups of the second and perhaps third cycles. Attention should be paid to the instrumentation needs for these tests since neutron sensors must be both distributed and inter-calibrated to infer power distributions. Neutron detectors used in test measurements should also be sensitive enough to measure reactivity and changes in flux levels and distributions.	• *Based on review of AREVA PCDR*: Needs for testing and instrumentation are recognized in the AREVA PCDR. In the PCDR, the tall core shape is actually used repeatedly by AREVA as a design feature that will tend to slow down the plant response to transients and accidents. • *Related AREVA DDNs*: AREVA-001 thru AREVA-010 (materials testing); AREVA-023 thru AREVA-034 (system and component testing) • *Based on review of GA PCDR*: Needs for analytical codes, instrumentation, and testing relating to core monitoring are addressed in the General Atomics PCDR. General Atomics credits the core geometry as one of the design features that will help to control the type of phenomenon of concern to NRC. • *Related GA DDNs*: C.11.01.01 thru C.11.03.11) , C.11.04.03 (core instrumentation and testing); C.11.03.31 (core instrumentation validation); C.11.04.03 (neutron detector service equipment design verification) • *Based on review of WEC PCDR*: There are discussions of Nuclear Heat Supply System instrumentation, and of the general PBMR control philosophy; however, there is no indication that this item regarding detailed reactor physics testing and associated instrumentation has been specifically addressed in the WEC PCDR. • *Related WEC DDNs*: (None) • *NGNP R&D Response*: Initial core test; not R&D, planned in the near future.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC – Not Addressed NGNP R&D – Addressed

<table>
<tr><th colspan="4">TABLE 3C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - CONCLUSIONS</th></tr>
<tr><th>Item</th><th>NRC Need/Issue Identified</th><th>Design Integration Review Team Comments/Conclusions</th><th>Is this Item being addressed or does it pose a new risk?</th></tr>
<tr><td>C-1</td><td>General Safety Analysis/Safety Document Needs
• Comprehensive description of the NGNP safety philosophy, a listing of the components involved, and the conditions under which these components are expected to perform their safety functions.
• Explanation of how this philosophy meets the defense-in-depth approach and, in particular, answers to the following:
○ Will the components that perform a safety function (retain FPs) be classified as safety-related components, with the imposition of equipment qualification, in-service inspections, and/or Technical Specifications LCOs and SRs?
○ How will aging issues be addressed? If the safety function of a component is to retain FPs on its surface during adverse conditions, how can it be ensured that this function can be retained for long periods (decades), despite the possible presence of other long-term surface degradation mechanisms?
○ Will the surface state of a non-replaceable or difficult-to-replace component be reactivated by chemical action or cleaning during its service life?
• A sound basis for the selection of the physical models and the data for these</td><td>• Based on review of AREVA PCDR: The AREVA PCDR has recognized most of the safety analysis/safety document needs detailed in this item, with the exception of the following:
- Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.
- The fuel database must be developed, as well as fuel-failure models and fuel material properties (both measurable and process controlled).
• Related AREVA DDNs: AREVA-001 thru AREVA-010 (materials testing); AREVA-011 thru AREVA-022 (modeling codes); AREVA-023 thru AREVA-040 (system and component testing)

• Based on review of GA PCDR: The General Atomics PCDR has recognized most of the safety analysis/safety document needs detailed in this item, with the exception of the following:
- How will aging issues be addressed? If the safety function of a component is to retain FPs on its surface during adverse conditions, how can it be ensured that this function can be retained for long periods (decades), despite the possible presence of other long-term surface degradation mechanisms?
- Will the surface state of a non-replaceable or difficult-to-replace component be reactivated by chemical action or cleaning during its service life?
- Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.
• Related GA DDNs: C.07.01.01 thru C.07.01.07 (fuel fabrication); C.07.02.01 thru C.07.02.09 (fuel performance data); C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development); C.11.04.04 thru C.11.04.06 (ISIs and Surveillances for reactor internals and core supports); C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS); C.16.00.01 thru C.16.00.06 (RCCS)

• Based on review of WEC PCDR: The WEC PCDR has recognized most of the safety analysis/safety documentation needs detailed in this item, with the exception of the following:
- Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.
• Related WEC DDNs: NHSS-01-01 thru NHSS-01-03 (fuel testing); NHSS-02-01 and NHSS-02-02 (graphite properties); HTS-01-01 thru HTS-01-019 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics); HTS-04-01 (high temp. ducts and insulation)

• NGNP R&D Response: Development of the Fuel database is planned.</td><td>Overall:
Most of the subparts of this item are being addressed. The area not being addressed is comprised of the parts of the issue that deal with hold-up and attenuation of fission products on specific surfaces in the primary circuit components and in the reactor building, related safety analysis assumptions, and preservation of those assumptions with Technical Specifications. This represents a risk to the project in terms of securing a successful safety evaluation from NRC.

By Organization:
AREVA – Partially Addressed
GA – Partially Addressed
WEC – Partially Addressed
NGNP R&D – Partially Addressed</td></tr>
</table>

TABLE 3C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	models must be justified. • The materials to be used and their sensitivity on the transport case must be identified. • Once the actual reactor design is available, the transport pathways that result from the accident conditions must be identified, along with the relevant models and data needed for the resulting calculations. • Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified. • The fuel database must be developed, as well as fuel-failure models and fuel material properties (both measurable and process controlled).		
C-2	*Model Development and V&V* - Physical models and the supporting mathematical methods, addressing: • Nuclides of interest • Fission product release from the fuel • Diffusion, adsorption, and desorption in graphite and fuel matrix materials • Adsorption, desorption, and in-diffusion in reactor system metals • Chemical and physical forms of the FPs in the coolant • Tritium transport models • Aerosols and dusts that plate-out on reactor system components and their mobility • Fission product reactions with the	• *Based on review of AREVA PCDR*: The AREVA PCDR has recognized the needs for most of the model development and V&V detailed in this item, with the following exceptions: - Fission product reactions with the confinement building materials - Determination of the safety function of each subsystem and the level of FPT attenuation required. - Determination of level of sensitivity to component uncertainties and how this reflects on the physical models. - Estimation of difficulty in obtaining the data and conducting the testing to support the safety case. • *Related AREVA DDNs:* AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR*: The General Atomics PCDR has recognized the needs for most of the model development and V&V detailed in this item, with the following exceptions: - Fission product reactions with the confinement building materials - Determination of the safety function of each subsystem and the level of FPT attenuation required. (Safety functions are specified, but level of FPT attenuation is not addressed.) - Determination of level of sensitivity to component uncertainties and how this reflects on the physical models. - Estimation of difficulty in obtaining the data and conducting the testing to support the safety case.	*Overall*: As in item C-1, most parts of this item are covered with anticipated R&D, however there are vulnerabilities dealing with the assumptions and modeling of the hold-up and attenuation of fission products on specific surfaces in the primary circuit components and in the reactor building. This item adds the issue of reactions between the fission products with the reactor building materials, which will depend on the confinement design. These vulnerabilities represent risks to the project in terms of securing a successful safety evaluation from NRC.

TABLE 3C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	confinement building materials • Reactions of the reactor system components and fission products with air or steam • Plume models that transport the released material beyond the reactor building • Determination of the safety function of each subsystem and the level of FPT attenuation required. • Determination of level of sensitivity to component uncertainties and how this reflects on the physical models. • Estimation of difficulty in obtaining the data and conducting the testing to support the safety case. • Scoping of how V&V can be performed.	• *Related GA DDNs:* C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); C.11.02 and C.11.03 (reactor vessel, core, and hot duct); N.13.01 and N.13.02 (primary heat transport system and IHX); C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS); C.16.00.01 thru C.16.00.06 (RCCS); C.31.01.01 and C.31.01.02 (reactor protection system) • *Based on review of WEC PCDR*: The WEC PCDR has recognized most of the needs detailed in this item, with the exception of the following: - Fission product reactions with the confinement building materials - Reactions of the reactor system components and fission products with air or steam - Plume models that transport the released material beyond the reactor building - Determination of the safety function of each subsystem and the level of FPT attenuation required. - Determination of level of sensitivity to component uncertainties and how this reflects on the physical models. - Estimation of difficulty in obtaining the data and conducting the testing to support the safety case. • *Related WEC DDNs:* NHSS-01-01 (Data to extend the irradiated fuels qualification database); NHSS-01-02 (Extend heat up data under accident conditions); NHSS-01-03 (Extend temperature-fluence envelope of fuel graphite) • *NGNP R&D Response*: This area of R&D is part of Fuel program. Specifics depend on design of confinement and its safety role. Most is covered or planned in Fuel program.	*By Organization*: AREVA – Partially Addressed GA – Partially Addressed WEC – Partially Addressed NGNP R&D – Partially Addressed
C-3	*Materials/Component Data* - Relevant data on materials or components over the range of interest and data uncertainties (single effects testing), including the following: • Graphite transport property and air/steam erosion data specific to the design material. • Metal alloy data specific to the design material. • Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product. • Data on helium impurities that will likely set the oxygen potential of the system, and the species to be included in an analysis. • Data associated with component aging: surface qualities of the reactor system	• *Based on review of AREVA PCDR*: The AREVA PCDR has recognized most of the needs for materials and component data detailed in this item, with the exception of the following: - Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product. - Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit. - Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. - If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties. - Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior. • *Related AREVA DDNs:* AREVA-001 thru AREVA-010 (materials testing); AREVA-023 thru AREVA-040 (system and component testing) • *Based on review of GA PCDR*: The General Atomics PCDR has recognized most of the needs for materials and component data detailed in this item, with the exception of the following: - Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit.	*Overall*: About half of the subparts of this item regarding materials and component data are being addressed, with the balance dealing again with the uncertainties surrounding assumptions and modeling of the hold-up and attenuation of fission products on specific surfaces in the primary circuit components and in the reactor building. This item adds concerns over possible contamination events, and formation and stability of the surface films that will be depended upon to retain fission products. These concerns represent risks to the project in terms of securing a successful safety evaluation from NRC.

TABLE 3C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	components after many years of operation. • Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit. • Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. • If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties. • Data regarding turbine or power conversion components that may have to be decontaminated prior to maintenance (initial collection of FPs while in the reactor circuit; decontamination of components; new surface state of the component after decontamination). • Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior,	- Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. - If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties. - Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior. • *Related GA DDNs:* C.07.01 and C.07.02 (fuel fabrication and performance); C.07.03 (fission product transport); C.07.04 (core corrosion data); C.11.01 (neutron control materials and testing); C.11.02 and C.11.03 (reactor internals, hot duct and core); C.12.01 (materials properties for reactor vessel); N.13.01 and N.13.02 (various IHX tests and materials research); C.14.01 and 14.04 (various SCS tests and materials research); C.16.00.01, C.16.00.02, and C.16.00.05 (RCCS emissivity, testing, conductivity); C.21.01.05 (fuel handling test) • *Based on review of WEC PCDR*: The WEC PCDR has addressed some of the issues associated with this item, with the exception of the following: - Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product. - Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit. - Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. - If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties. - Data regarding turbine or power conversion components that may have to be decontaminated prior to maintenance (initial collection of FPs while in the reactor circuit; decontamination of components; new surface state of the component after decontamination). - Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior • *Related WEC DDNs:* NHSS-01-01 thru NHSS-01-03 (fuel testing); NHSS-02-01 and NHSS-02-02 (graphites); HTS-01-01 thru HTS-01-019 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics); HTS-04-01 (high temp. ducts and insulation); HPS-03-01 thru HPS-03-04 (feed purification) • *NGNP R&D Response*: Most is covered in Fuel R&D program. The following items depend on how much credit is going to be taken for plateout which is still a subject of debate: - Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product. - Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit. - Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. - If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component,	*By Organization*: AREVA – Partially Addressed GA – Partially Addressed WEC – Partially Addressed NGNP R&D – Partially Addressed

TABLE 3C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		these materials may have to undergo testing for transport properties. - Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior.	
C-4	Reactor component and confinement/containment configuration and their relative roles in the safety case • Respective roles of the reactor circuit and containment or confinement system must be known before their modeling adequacy can be determined. • Estimate of source and budgeting of FP holdup among the fuel form, reactor circuit components, mobile elements such as dust, and the reactor building, as a means of focusing components to be emphasized in analysis. • Determination of transport pathway, goals for FP retention at each step in the pathway, local (accident) operating environment at each step of the pathway.	• *Based on review of AREVA PCDR*: The definition of roles has been included in the AREVA PCDR. The need for computer code development to understand fission product transport and distribution has been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR*: The definition of roles has been included in the General Atomics PCDR. The need for computer code development to understand fission product transport and distribution has been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); C.11.02 and C.11.03 (reactor vessel, core, and hot duct); C.12.01 (materials properties for reactor vessel); C.16.00.03 (integrated performance of RCCS) • *Based on review of WEC PCDR*: The part of this item addressing the general understanding of roles of reactor building and equipment has been addressed, and an experimental facility intended to research issues of plateout and dust are described. However, the central issue of budgeting and modeling fission product hold-up among specific design features has not been addressed in the WEC PCDR. • *Related WEC DDNs:* (None) • *NGNP R&D Response*: Safety approval issue - not R&D.	*Overall*: Again, the themes of modeling and budgeting fission product hold-up at each step along the transport pathway emerge in this item, and are not apparently addressed in current R&D efforts or planning. These concerns represent risks to the project in terms of securing a successful safety evaluation from NRC. *By Organization*: AREVA – Partially Addressed GA – Partially Addressed WEC – Partially Addressed NGNP R&D – Not Addressed
C-5	Computational software or other methods for determining the quantitative results • Data collection and proof that the selected model is adequate under all the normal and accident conditions of interest. Need to know that model envelops releases, and have reasonable proof that the model predicts an upper limit. • Need to have a description of the physical models and the reactor configuration, showing that the models are appropriate for the conditions of interest. • Need to have the data required for the models: single-effects data for each	• *Based on review of AREVA PCDR*: Needs for computer model development and testing have been recognized in the AREVA PCDR. Reactor configuration is available in the PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-010 (materials testing); AREVA-011 thru AREVA-022 (modeling codes); AREVA-023 thru AREVA-040 (system and component testing) • *Based on review of GA PCDR*: Needs for computer model development and testing have been recognized in the General Atomics PCDR. Reactor configuration is available in the PCDR. • *Related GA DDNs:* C.07.02.07 (fuel testing); C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); C.11.01.11 (neutron control assembly test); C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development, core testing); C.11.04.04 thru C.11.04.06 (ISIs and Surveillances for reactor internals and core supports); N.13.02.06 thru N.13.02.09 (IHX tests); C.14.01.02 thru C.14.01.04 C.14.04.01 thru 14.04.05, and 14.04.07 (SCS tests); C.16.00.02 (RCCS test); C.21.01.05 (fuel handling test) • *Based on review of WEC PCDR*: The WEC PCDR has recognized the needs for computer model development and	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	material and component acquired under individual testing, and integral data designed to show that the codes get the correct answer for a complete system under the conditions of interest.	supporting testing. Reactor configuration is available in the PCDR. • *Related WEC DDNs:* NHSS-01-01 (Data to extend the irradiated fuels qualification database); NHSS-01-02 (Extend heat up data under accident conditions; NHSS-01-03 (Extend temperature-fluence envelope of fuel graphite); NHSS-02-01 and NHSS-02-02 (Extend irradiated materials qualification database for Reflector Graphite); HTS-01-1 thru HTS-01-19 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics) • *NGNP R&D Response*: This need is generally covered by Methods program and relevant parts of Fuel program.	
C-6	Integral testing over a wide range of conditions to support the development of computational methods and the quantification of the data and associated uncertainties • Attempt to use existing data from past programs to the degree appropriate. • Planning of any in-pile loop program would require a complete description of the normal operating environment and of the accidents, along with any scaling factors. Extensive modeling will be necessary to design the loop and determine off-normal conditions that the loop can be expected to simulate. Model predictions (with the previously collected single-effects data) will need to be made.	• *Based on review of AREVA PCDR*: Needs for testing have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-010 (materials testing); AREVA-023 thru AREVA-040 (system and component testing) • *Based on review of GA PCDR*: Needs for testing have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.07.02.07 (fuel testing); C.11.01.11 (neutron control assembly test); C.11.03.45 and C.11.03.46 (core crossflow and core fluctuation tests); C.11.04.04 thru C.11.04.06 (ISIs and Surveillances for reactor internals and core supports); N.13.02.06 thru N.13.02.09 (IHX tests); C.14.01.02 thru C.14.01.04 C.14.04.01 thru 14.04.05, and 14.04.07 (SCS tests); C.16.00.02 (RCCS test); C.21.01.05 (fuel handling test) • *Based on review of WEC PCDR*: The WEC PCDR has at many levels recognized the indicated needs for testing. • *Related WEC DDNs:* (None) • *NGNP R&D Response*: Agree with NRC comment. Factored into thinking of Methods and Fuel integral testing.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3D - HIGH TEMPERATURE MATERIALS (METALLIC) - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
D-1	*Physical Materials Data* - Requirements for physical aspects to be included in modeling high-temperature metallic components: • Inelastic materials behavior for materials, times, and temperatures for very high temperature structures (e.g., creep, fatigue, creep-fatigue). • Adequacy and applicability of current ASME Code allowables with respect to service times and temperatures for operational stresses. • Adequacy and applicability of current state of high-temperature design methodology (e.g., constitutive models, complex loading, failure criteria, flaw assessment methods). • Effects of product form and section thickness. • Joining methods including welding, diffusion bonding, and issues associated with dissimilar materials in structural components. • Effects of irradiation on materials strength, ductility, and toughness. • Degradation mechanisms and inspectability. • Oxidation, carburization, decarburization, and nitriding of metallic components in impure helium and helium-nitrogen. • Micro-structural stability during long-term aging in environment. • Effects of short and long term on mechanical properties (e.g., tensile, fatigue, creep, creep-fatigue, ductility, toughness). • High-velocity erosion/corrosion.	• *Based on review of AREVA PCDR*: The AREVA PCDR has captured most of the needs detailed in this item, with the following exceptions: - Micro-structural stability during long-term aging in environment. - High-velocity erosion and corrosion. - Compatibility with heat-transfer media and reactants for hydrogen generation. • *Related AREVA DDNs:* AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification); AREVA-013 (improvement/development of “other” codes) • *Based on review of GA PCDR*: The General Atomics PCDR has captured most of the needs detailed in this item, with the following exceptions: - Degradation mechanisms and inspectability. - Micro-structural stability during long-term aging in environment. - High-velocity erosion and corrosion. - Development and stability of surface layers on RPV and core barrel affecting emissivity • *Related GA DDNs:* C.11.02 and C.11.03 (reactor internals, hot duct and core); C.11.04.04 and C.11.04.05 (design verification for metallic reactor internals and core supports); C.12.01 (materials properties for reactor vessel, including heavy sections); N.13.01 and N.13.02 (various IHX tests and materials research); N.41.01.01, N.41.01.02, N.41.01.03, and N.41.02.01 (materials data for high heat power conversion system components) • *Based on review of WEC PCDR*: The needs for materials data identified in this item have been addressed in the WEC PCDR, and some of the R&D has already been performed. • *Related WEC DDNs:* NHSS-01-01 thru NHSS-01-03 (fuel testing); NHSS-02-01 and NHSS-02-02 (graphites); HTS-01-01 thru HTS-01-019 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics); HTS-04-01 (high temp. ducts and insulation) • *NGNP R&D Response*: All covered by Materials R&D program.	*Overall*: This item is being addressed. *By Organization*: AREVA – Partially Addressed GA – Partially Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3D - HIGH TEMPERATURE MATERIALS (METALLIC) - CONCLUSIONS

Item	NRC Need/Issue Identified	Design Integration Review Team Comments/Conclusions	Is this Item being addressed or does it pose a new risk?
	• Rapid oxidation of graphite and carbon-carbon composites during air-ingress accidents. • Compatibility with heat-transfer media and reactants for hydrogen generation. • Development and stability of surface layers on RPV and core barrel affecting emissivity.		
D-2	*Physical Materials Data (Composites)* - Requirements for physical aspects to be included in modeling high-temperature structural composites, such as carbon-carbon or silicon carbide–silicon carbide: • Effects of composite component selection and infiltration method. • Effects of architecture and weave. • Materials properties up to and including very high temperatures (e.g., strength, fracture, creep, corrosion, thermal shock resistance). • Effects of irradiation on materials strength and dimensional stability. • Fabrication scaling processes. • Adequacy and validation of design methods. • Degradation mechanisms and inspectability.	• *Based on review of AREVA PCDR*: In general, the AREVA PCDR recognizes the needs for greater understanding of materials characteristics and behavior of composites. However, there is no indication that the following topics from this item are considered: - Effects of composite component selection and infiltration method. - Effects of architecture and weave. - Fabrication scaling processes. - Adequacy and validation of design methods. • *Related AREVA DDNs:* AREVA-004 thru AREVA-007 (testing/verification of ceramics, including composites) • *Based on review of GA PCDR*: Although a program to address issues associated with composite materials is not specifically addressed, it is included by reference as a part of the program to qualify graphites, in the General Atomics PCDR. • *Related GA DDNs:* C.11.01.03 (control rod design verification); C.11.03.02 thru C.11.03.06 (control rod failure modes and integrity); C.11.03.24 (control rod high temp materials properties) • *Based on review of WEC PCDR*: The WEC PCDR has recognized the need for composite physical materials data. • *Related WEC DDNs:* HTS-02-01 (Review existing technology); HTS-02-02 (Materials properties database); HTS-02-03 (Design Methods); HTS-02-04 (Performance verification); HTS-02-05 (Manufacturing technology); HTS-02-06 (Codes and Standards) • *NGNP R&D Response*: Composites are in Materials R&D plan. The plan will address these issues.	*Overall*: This item is being addressed. *By Organization*: AREVA – Partially Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
D-3	Compromise of RPV surface emissivity due to loss of desired surface layer properties. Compromise of emissivities of in-vessel surfaces.	• *Based on review of AREVA PCDR*: Need for greater understanding of the surface emissivity material characteristic has been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification); AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR*: The importance of surface emissivities, including analytical efforts performed to date, have been recognized in the General Atomics PCDR.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed

TABLE 3D - HIGH TEMPERATURE MATERIALS (METALLIC) - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *Related GA DDNs:* N.11.02.16 (reactor internals emissivity); C.12.01.06 (reactor vessel emissivity); C.16.00.01 thru C.16.00.06 (RCCS) • *Based on review of WEC PCDR*: The need to explore issues relating to surface emissivity has been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-019 (IHX metallics) • *NGNP R&D Response*: Covered in Materials R&D program.	WEC - Addressed NGNP R&D - Addressed
D-4	Effects on insulation • Aging fatigue and environmental degradation of insulation materials (debris plugging). • Environmental and irradiation degradation/thermal instability of fibrous insulation	• *Based on review of AREVA PCDR*: There is no indication that this item has been specifically addressed. The AREVA PCDR has recognized the need for R&D regarding aging of materials, but has not addressed these specific issues on insulation. • *Related AREVA DDNs:* (None) • *Based on review of GA PCDR*: There is no indication that this item has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* N.11.02.14 (fibrous insulation properties); N.11.02.15 (hard ceramic insulation properties); N.13.02.07 (IHX insulation tests); C.14.04.01 (SHE insulation tests) • *Based on review of WEC PCDR*: The need to explore issues relating to high temperature ducts and insulation has been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-019 (IHX metallics); HTS-04-01 (high temp ducts and insulation) • *NGNP R&D Response*: Mentioned in Materials R&D plan. No current work underway.	*Overall*: This item is being addressed; however, it appears to have been addressed thus far only marginally. Issues over insulation degradation and subsequent degradation on plant safety performance are typically of great concern to NRC. These concerns represent risks to the project in terms of securing a successful safety evaluation from NRC. *By Organization*: AREVA – Not Addressed GA – Not Addressed WEC - Addressed NGNP R&D - Addressed
D-5	Primary boundary failures in compact IHX (roles of design methods, manufacturing controls, inspection/testing).	• *Based on review of AREVA PCDR*: Indication in the AREVA PCDR is that this item is being addressed in the design. Also, AREVA has addressed improvement of design methods in section 19.2.4 (beginning on p. 290) and proposed a main component fabrication strategy in section 21.1.27 (p. 320). • *Related AREVA DDNs:* AREVA-003 (testing and codification of IHX materials); AREVA-024 and AREVA-025 (testing of IHX) • *Based on review of GA PCDR*: The need for design verification testing of critical components such as the IHX has been recognized in the General Atomics PCDR. • *Related GA DDNs:* N.13.02.01 thru N.13.02.09 (IHX materials and design)	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3D - HIGH TEMPERATURE MATERIALS (METALLIC) - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *Based on review of WEC PCDR*: The need to explore issues relating to the IHX, including the limited lifetime of the component, has been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-019 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX composites) • *NGNP R&D Response*: Part of CTF testing of PCHE.	
D-6	Control rod insertion failures (role of structural design methods for composites).	• *Based on review of AREVA PCDR*: Need for control rod material qualification is recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-006 (control rod sheath materials); AREVA-030 (testing of neutron control drive mechanism) • *Based on review of GA PCDR*: The needs to further characterize, test and qualify the composite material selected for control rod assemblies have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.11.01.03 (control rod design verification); C.11.03.02 thru C.11.03.06 (control rod failure modes and integrity); C.11.03.24 (control rod high temp materials properties) • *Based on review of WEC PCDR*: Not applicable to the WEC PCDR. There is no indication that WEC intends to use composite materials for control rod system components. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-015 (IHX metallics) • *NGNP R&D Response*: Part of Composites R&D plan.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC – Not Applicable NGNP R&D - Addressed
D-7	Irradiation induced creep of in-vessel metallic structures.	• *Based on review of AREVA PCDR*: The need to better understand the phenomenon of creep has been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification) • *Based on review of GA PCDR*: The need to better understand the phenomenon of creep has been recognized in the General Atomics PCDR. • *Related GA DDNs:* N.11.02.11 (irradiation effects on metallic reactor internals) • *Based on review of WEC PCDR*: The need to explore materials issues relating metallic structures, including those associated with the RPV and IHX, has been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-015 (IHX metallics) • *NGNP R&D Response*: Part of Materials R&D plan for IHX materials.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
D-8	Core radial restraint failure (role of structural design and fabrication for composites).	• *Based on review of AREVA PCDR*: There is no indication that this item has been specifically addressed in AREVA's PCDR.	*Overall*: This item is being addressed.

TABLE 3D - HIGH TEMPERATURE MATERIALS (METALLIC) - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *Related AREVA DDNs:* AREVA-007 (testing and codification of composites) • *Based on review of GA PCDR*: There is no indication that this item has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* C.11.02.01 (core support strength data); C.11.02.11 and C.11.02.12 (helium, temp. and irradiation effects on metallic reactor internals); C.11.04.05 and C.11.04.06 (core support design verification) • *Based on review of WEC PCDR*: The need to explore issues relating to the core restraints has been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-015 (IHX metallics) • *NGNP R&D Response*: Should be addressed in Composites R&D plan.	*By Organization*: AREVA – Not Addressed GA – Not Addressed WEC - Addressed NGNP R&D - Addressed
D-9	Isolation and other valve failures (self-welding, galling, seizing)	• *Based on review of AREVA PCDR*: Need for isolation valve qualification is recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-026 (testing of isolation valves) • *Based on review of GA PCDR*: The applications and need for addressing issues associated with isolation valves have been addressed in the General Atomics PCDR. • *Related GA DDNs:* C.14.01.04 (shutdown circulator loop shutoff valve test); N.42.02.01 and N.42.02.02 (sec heat transport isolation valves) • *Based on review of WEC PCDR*: The need to explore issues relating to valve failures has been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-015 (IHX metallics) • *NGNP R&D Response*: No current R&D planned. Part of CTF testing.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
D-10	Initiate development of the data and models needed by ASME Boiler and Pressure Vessel (B&PV) Code Subcommittees to formulate time-dependent failure criteria that will ensure adequate life and safety for metallic materials in the NGNP. These include obtaining the data necessary to develop experimentally based constitutive models for the NGNP construction materials, which are the foundation of the inelastic design analyses specifically required by	• *Based on review of AREVA PCDR*: Needs for ASME code development and supporting structural mechanics models have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification); AREVA-013 (improvement/development of “other” codes) • *Based on review of GA PCDR*: The needs for developing structural models and ASME Code qualification for high temperature metallic materials have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.07.04 (core corrosion data); C.11.02 and C.11.03 (reactor internals, hot duct and core); C.11.04.04 and C.11.04.05 (design verification for metallic reactor internals and core supports); C.12.01 (materials properties for reactor vessel); N.13.01 and N.13.02 (various IHX tests and materials research)	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3D - HIGH TEMPERATURE MATERIALS (METALLIC) - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	ASME B&PV Sect. III Division I Subsection NH.	• *Based on review of WEC PCDR*: The need to explore issues relating to the qualification of NGNP metallics under approved ASME Code Cases has been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-019 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics) • *NGNP R&D Response*: Part of Materials R&D plan.	
D-11	Safety assessments dependent on time-dependent flaw growth and the resulting leak rates from postulated pressure-boundary breaks will require a flaw assessment procedure capable of reliably predicting crack-induced failures, as well as the size and growth of the resulting opening in the pressure boundary.	• *Based on review of AREVA PCDR*: Need for structural mechanics models, including flaw assessment, have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-013 (improvement/development of “other” codes) • *Based on review of GA PCDR*: There is no indication that this item has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* N.12.01.01 thru N.12.01.03 (RPV materials) • *Based on review of WEC PCDR*: The need to explore issues relating to the mechanical properties of metallic pressure boundary materials has been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-019 (IHX metallics) • *NGNP R&D Response*: Unclear.	*Overall*: Generally speaking, this item is being addressed, in that there is recognition that better and more material-specific structural mechanics models are required for the NGNP. However, the recognition seems to be weak, in comparison to discussion of other types of modeling codes that are required for the project. This almost surely revolves around the uncertainties associated with materials selection and the possible lack of qualification and ASME codification (depending on the materials chosen). This issue of materials selection, qualification, and structural mechanics modeling would appear to be a significant risk to overall project success. *By Organization*: AREVA - Addressed GA – Not Addressed WEC - Addressed NGNP R&D – Not Addressed
D-12	Materials data and extrapolation procedures must be developed and guidance provided to ensure that allowable operation period and range of stress and temperature for materials of construction are extended to meet the proposed operating temperatures and lifetimes. Creep-fatigue rules are an	• *Based on review of AREVA PCDR*: Needs for greater understanding of materials characteristics, related ASME Code efforts, scale-up of significant metal components, and development of structural mechanics codes have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification); AREVA-013 (improvement/development of “other” codes)	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed

TABLE 3D - HIGH TEMPERATURE MATERIALS (METALLIC) - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	area of particular concern for the materials and temperatures of interest and must be updated and validated. (example concern: RPV long-term thermal aging)	• *Based on review of GA PCDR*: The needs for developing structural models and ASME Code qualification for high temperature metallic materials have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.11.02 and C.11.03 (reactor internals, hot duct and core); C.11.04.04 and C.11.04.05 (design verification for metallic reactor internals and core supports); C.12.01 (materials properties for reactor vessel); N.13.01 and N.13.02 (various IHX tests and materials research) • *Based on review of WEC PCDR*: The need to explore issues relating to extended lifetimes and more severe conditions anticipated for metallic components has been recognized in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites); HTS-01-01 thru HTS-01-19 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics); HTS-04-01 (high temp ducts and insulation) • *NGNP R&D Response*: Covered in Materials R&D plan.	GA - Addressed WEC - Addressed NGNP R&D - Addressed
D-13	Since IHX sections must operate at the full exit temperature of the reactor, effort should be initiated to obtain data supporting the determination of the metallurgical stability and environmental resistance of IHX materials in anticipated impure helium coolant environments for the lifetimes anticipated.	• *Based on review of AREVA PCDR*: The need for high purity helium is addressed in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification); AREVA-027 (helium purification system) • *Based on review of GA PCDR*: The need for high purity helium is addressed in the General Atomics PCDR in the design. • *Related GA DDNs:* N.13.02.01 (effects of helium and temp on IHX) • *Based on review of WEC PCDR*: The WEC PCDR has addressed this topic. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-015 (IHX metallics) • *NGNP R&D Response*: R&D underway in Materials R&D program.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
D-14	Work should be initiated to quantify crack initiation and propagation in the IHX due to creep, creep-fatigue, and aging. These materials-related phenomena related to the IHX were identified for potentially contributing to FP release at the site boundary.	• *Based on review of AREVA PCDR*: Needs for greater understanding of materials characteristics and associated component testing have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification); AREVA-024 and AREVA-025 (IHX testing) • *Based on review of GA PCDR*: The need to better understand the phenomenon of creep has been recognized in the General Atomics PCDR. • *Related GA DDNs:* N.13.02.01 thru N.13.02.09 (IHX materials) • *Based on review of WEC PCDR*: The WEC PCDR has addressed this topic. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-015 (IHX metallics)	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3D - HIGH TEMPERATURE MATERIALS (METALLIC) - CONCLUSIONS

Item	NRC Need/Issue Identified	Design Integration Review Team Comments/Conclusions	Is this Item being addressed or does it pose a new risk?
		• *NGNP R&D Response*: Testing is underway in Materials R&D program.	
D-15	Specific issues must be addressed for RPVs that are too large for shop fabrication and transportation. Validated procedures for on-site welding, PWHT, and inspections must be developed for the materials of construction. For vessels using materials other than those typical of LWR construction to enable operation at higher temperatures, confirmation of their fabricability (especially, effects of forging size and weldability) and data on their irradiation resistance is needed. Three materials-related phenomena related to the RPV fabrication and operation were identified for potentially contributing to FP release at the site boundary, particularly for 9Cr–1 Mo–V steels capable of higher-temperature operation: crack initiation and subcritical crack growth, process control to avoid material degradation during field fabrication, and property control in heavy sections.	• *Based on review of AREVA PCDR*: Needs for resolution of issues associated with heavy sections, materials characteristics and feasibility of using the 9Cr-1Mo alloy, and fabrication of large vessels have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 (testing and codification for vessel materials) • *Based on review of GA PCDR*: The General Atomics PCDR contains options for fabricating the reactor vessel, including forging or welding together sections of rolled plate. There is no indication that the need to research issues relating to vessels too large for shop fabrication has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* N.12.01.02 (RPV heavy sections) • *Based on review of WEC PCDR*: The need to explore issues relating to possible RPV fabrication activities has been recognized in the WEC PCDR. • *Related WEC DDNs:* (None) • *NGNP R&D Response*: 9Cr-1 Mo is not primary candidate for RPV. SA 503/533 is primary candidate. These issues are well know for 9Cr-1 Mo.	*Overall*: This item is being addressed; however, the matter of alloy selection, which will impact vessel fabrication, is a core issue that does not appear to have been settled yet. This issue would appear to be a significant risk to overall project success. *By Organization*: AREVA - Addressed GA – Not Addressed WEC - Addressed NGNP R&D - Addressed
D-16	For high-temperature metals technology, there is a need for analytical models, in particular for developing time-dependent design criteria for complex structures, along with verification by structural testing. ASME Code-approved simplified methods have not yet been proven and are not permitted for compact IHX components. Analytical modeling of carbon-carbon composite behavior would be useful in developing approved methods for designing, proof testing, model standard testing, validation tests, and probabilistic methods of design. Scalability and fabrication issues must be addressed, including large-scale structures (meters in diameter), as well as smaller structures.	• *Based on review of AREVA PCDR*: Needs for improved high temperature metals technology, ASME code-approved materials designations, structural mechanics codes describing materials behavior and characteristics, and resolution of large scale fabrication strategies have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification); AREVA-004 thru AREVA-007 (testing/verification of ceramics, including composites) • *Based on review of GA PCDR*: The needs for analytical models and ASME Code qualification of high temperature metals have been recognized in the General Atomics PCDR. • *Related GA DDNs:* N.13.02.01 thru N.13.02.09 (IHX materials) • *Based on review of WEC PCDR*: The needs to explore issues relating to establishing mechanical properties, design methods, and supporting new ASME Code Cases for metallic components have been recognized in the WEC PCDR. • *Related WEC DDNs:* HTS-01-17 thru HTS-01-19 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics) • *NGNP R&D Response*: Part of Materials R&D program.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3E – GRAPHITE - CONCLUSIONS

Item	NRC Need/Issue Identified	Design Integration Review Team Comments/Conclusions	Is this Item being addressed or does it pose a new risk?
E-1	Lack of confirmatory data for the grades of graphite selected by potential NGNP vendors. This situation has occurred because: • Graphite grades used in prior HTGRs are no longer available, and thus development of new grades has been required. • Increased temperature of the NGNP compared to prior graphite-moderated reactors. • In the case of the PBR, the larger neutron dose that the core components will experience compared to that of previous HTGRs licensed in the United States.	• *Based on review of AREVA PCDR*: Needs for development of updated, code-approved graphite materials designations have been recognized in the AREVA PCDR, and some of the R&D has been performed. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials) • *Based on review of GA PCDR*: The need for ASME Code qualification of graphites has been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites); HTS-02-01 thru HTS-02-06 (IHX ceramics) • *NGNP R&D Response*: These data are key outputs from Graphite program.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
E-2	Lack of consensus codes and standards. Efforts are under way through the ASME to develop a consensus design code for graphite core components, but to date a useable code has not been approved. ASTM test standards exist for many of the physical properties of concern to the reactor designer, but further work is required, especially in the area of small (irradiation) specimen test methods.	• *Based on review of AREVA PCDR*: Needs for development of approved ASME codes for graphite have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials) • *Based on review of GA PCDR*: The need for ASME Code qualification of graphites has been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites); HTS-01-17 thru HTS-01-19 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics) • *NGNP R&D Response*: This is part of Graphite R&D program.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
E-3	Theoretical models for the effects of neutron damage on the properties of graphite have been developed, however, these models need modification for the new graphites and will need to be extended to higher temperatures and/or higher neutron doses.	• *Based on review of AREVA PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials); AREVA-013 (improvement/development of "other" codes)	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed

TABLE 3E – GRAPHITE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	V&V of theoretical models will require generation of experimental data on the effect of neutron irradiation on properties.	• *Based on review of GA PCDR*: The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response:* This is part of Graphite program.	GA - Addressed WEC - Addressed NGNP R&D - Addressed
E-4	Uncertainties in the temperature and dose received by a component; the severity of temperature and dose gradients in a component; the rate of dimensional change in the specific graphite used in a given design; the extent to which stresses are relieved by irradiation-induced creep; and the extent of changes in key physical properties such as elastic moduli, thermal conductivity, coefficient of thermal expansion, compound to make the prediction of component stress levels, and hence decisions regarding component lifetime and replacement schedules, very imprecise.	• *Based on review of AREVA PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials); AREVA-013 (improvement/development of “other” codes) • *Based on review of GA PCDR*: The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics, development of structural mechanics models, and irradiation tests to determine component lifetime and replacement schedules are recognized in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: This is part of Graphite program.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
E-5	Whole-core models are required that can predict the stress states of graphite components within the core. Such models should be capable of taking inputs such as temperature and neutron dose and calculating the dimensional change, creep, thermal conductivity, etc., from established theoretical models. Reliable stress-state predictions as a function of reactor life would enable reactor operators and regulators to provide NDE guidance and make decisions regarding inspection intervals and core block replacement.	• *Based on review of AREVA PCDR*: Needs for improved reactor analysis computer models have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-011 thru AREVA-022 (modeling codes) • *Based on review of GA PCDR*: Reactor core analyses performed to date and the need for further analytical models have been described in the General Atomics PCDR. • *Related GA DDNs:* C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development); C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: The WEC PCDR has addressed this issue.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3E – GRAPHITE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: Mainly vendor scope, but Graphite R&D program has activity in whole-core modeling.	
E-6	Basic research should be conducted to strengthen the understanding and modeling capability of the displacement damage process in graphite. In addition, in graphite technology, there is a need for analytical models for oxidation, changes in physical properties, irradiation induced dimensional change, and irradiation creep. They could be developed to feed into a structural integrity model for the graphite core which would be used for core design and safety assessment.	• *Based on review of AREVA PCDR*: Needs for improved structural mechanics computer models have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials); AREVA-013 (improvement/development of “other” codes) • *Based on review of GA PCDR*: The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport); C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development); C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: Part of Graphite program	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
E-7	Irradiation induced change in the coefficient of thermal expansion, including effects of creep strain.	• *Based on review of AREVA PCDR*: Needs for further knowledge of the phenomena described in this item have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials); AREVA-013 (improvement/development of “other” codes) • *Based on review of GA PCDR*: The needs for further materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: Part of Graphite program scope.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
E-8	Irradiation induced change in mechanical properties such as strength and toughness,	• *Based on review of AREVA PCDR*: Needs for further knowledge of the material characteristics described in this item	*Overall*:

TABLE 3E – GRAPHITE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
	including the effect of creep strain.	have been recognized in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials); AREVA-013 (improvement/development of “other” codes) • *Based on review of GA PCDR*: The needs for further materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Related GA DDNs:* C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: Part of Graphite program scope.	This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed
E-9	Blockage of coolant channel in a fuel element block or reactivity control block due to graphite failure and/or graphite spalling.	• *Based on review of AREVA PCDR*: There is no indication that this item has been specifically addressed in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-014 (fuel performance modeling and codes); AREVA-008 thru AREVA-010 (graphite qualification) • *Based on review of GA PCDR*: There is no indication that this item has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* C.07.02.01 thru C.07.02.09 (fuel performance data); C.11.03.11 thru C.11.03.23 (graphite materials characterization); C.11.03.42 (control rod channel flow data) • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models, including implementation of failure models, are recognized in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: Design issue- no R&D.	*Overall*: This item is being addressed weakly at this stage of the deign effort. This concern represents a risk to the project in terms of securing a successful safety evaluation from NRC. *By Organization*: AREVA – Not Addressed GA – Not Addressed WEC - Addressed NGNP R&D – Not Addressed
E-10	Statistical variation of non-irradiated properties, due to forming, processing, raw materials, and formulation.	• *Based on review of AREVA PCDR*: This item has been addressed in the AREVA PCDR for statistical control and sampling related to fuel fabrication, but not for other graphite components. • *Related AREVA DDNs:* AREVA-014 (fuel performance modeling and codes) • *Based on review of GA PCDR*: The need for statistical control is addressed in the General Atomics PCDR for fuel, but it is not addressed for other graphites.	*Overall*: This item is being addressed. *By Organization*: AREVA – Partially Addressed GA – Partially Addressed

TABLE 3E – GRAPHITE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *Related GA DDNs:* C.07.01.01 thru C.07.01.07 (fuel fabrication); C.07.02.01 thru C.07.02.09 (fuel performance data) • *Based on review of WEC PCDR*: This item has been addressed in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites); HTS-02-01 thru HTS-02-06 (IHX ceramics) • *NGNP R&D Response*: Part of Graphite program.	WEC - Addressed NGNP R&D - Addressed
E-11	Ability to develop generic specifications that will ensure consistency of graphite quality over the lifetime of the reactor fleet, including for replacement components.	• *Based on review of AREVA PCDR*: There is no indication that this item has been specifically addressed in the AREVA PCDR. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials) • *Based on review of GA PCDR*: The General Atomics PCDR includes generic specifications for fuel quality, but not for other graphite components. • *Related GA DDNs:* C.07.01.01 thru C.07.01.07 (fuel fabrication); C.07.02.01 thru C.07.02.09 (fuel performance data); C.11.03.11 thru C.11.03.23 (graphite materials characterization) • *Based on review of WEC PCDR*: This item has been addressed in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: Part of Graphite scope.	*Overall*: This item is being addressed. *By Organization*: AREVA – Not Addressed GA – Partially Addressed WEC - Addressed NGNP R&D - Addressed
E-12	Tribology (effects of moving surface interactions) of graphite in helium environment, including potentially impure helium environment (examples: surfaces sticking together; surfaces wearing on each other to generate dust, etc.)	• *Based on review of AREVA PCDR*: In the AREVA PCDR, a helium purification system has been incorporated into the design to ensure the purity of the helium environment, and the need for improved knowledge of tribology has been recognized. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials) • *Based on review of GA PCDR*: In the General Atomics PCDR, a helium purification system has been incorporated into the design to ensure the purity of the helium environment. There is no indication that phenomena associated with materials tribology have been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* C.11.02.10 and C.11.02.13 (effects of helium on reactor internals and hot duct); C.11.03.11 thru C.11.03.23 (graphite materials characterization); N13.01.01 (effects of helium on primary heat transport); N13.02.01 (effects of helium on IHX); N.14.01.06 and 14.04.12 (effects of helium on SCS); N.42.02.01 (effects of helium on secondary transport) • *Based on review of WEC PCDR*: This item has been addressed in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites)	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA – Partially Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3E – GRAPHITE - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		• *NGNP R&D Response*: Part of Graphite program.	
E-13	Impact of degradation of thermal conductivity on fuel temperature limits.	• *Based on review of AREVA PCDR*: In the AREVA PCDR, this item has been addressed in the design and by studies already performed. • *Related AREVA DDNs:* AREVA-008 thru AREVA-010 (testing and codification of graphite materials) • *Based on review of GA PCDR*: This phenomenon has been recognized and quantified in the General Atomics PCDR. • *Related GA DDNs:* C.07.02.04 (fuel compact thermophysical properties); C.11.03.16 (graphite thermal properties data) • *Based on review of WEC PCDR*: This item has been addressed in the WEC PCDR. • *Related WEC DDNs:* NHSS-02-01 and NHSS-02-02 (graphites) • *NGNP R&D Response*: Part of Graphite scope.	*Overall*: This item is being addressed. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D - Addressed

TABLE 3F – PROCESS HEAT FOR HYDROGEN - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
F-1	Cold oxygen (O2) and other heavy-gas accidental releases from the process plant that can flow from the chemical plant to the nuclear plant (depending upon wind, relative plant elevations, and nuclear plant air intakes) and potentially impact the integrity of reactor systems, structures, and components (SSCs). All of the proposed processes for production of hydrogen start with water, and thus all of the processes will produce oxygen as a byproduct of hydrogen production. Oxygen is the one common chemical safety issue that can impact nuclear plant safety. At high oxygen concentrations, many "noncombustible" materials become combustible and the potential for spontaneous combustion increases. Increased oxygen levels at the reactor can compromise the functioning of safety equipment.	• *Based on review of AREVA PCDR*: Specific design of the hydrogen production facility was outside of the AREVA PCDR assigned scope, and there is no indication that this item has been specifically addressed. • *Related AREVA DDNs:* (None) • *Based on review of GA PCDR*: There is no indication that this item has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* N.44.01, N.44.02, N44.03, N44.04, N45.03, and N45.04 (hydrogen production) • *Based on review of WEC PCDR*: Indications are that WEC has addressed this item in the PCDR and will address it in more detail, via process hazards analysis, as the project progresses. • *Related WEC DDNs:* (None) • *NGNP R&D Response*: Separation of reactor and H_2 plant should reduce this concern.	*Overall*: Indications are that this item has been addressed only preliminarily thus far. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses. *By Organization*: AREVA – Not Addressed GA – Not Addressed WEC - Addressed NGNP R&D – Not Addressed
F-2	Failure of the IHX leading to potential damage to safety-related SSCs in the reactor due to blow-down effects from large mass transfer and over-pressurization of either secondary or primary side. The impact of the IHX failure depends upon the selection of the heat transfer fluid in the secondary heat transport loop. Helium is the leading candidate for the heat transport loop, but no final decisions have been made. If helium is used, the helium inventory in the secondary loop may be greater than the inventory in the reactor; thus, any leak in the IHX can significantly increase the total helium inventory involved in any reactor depressurization event.	• *Based on review of AREVA PCDR*: Indications are that this item has been addressed in the AREVA PCDR in the design, and will be further addressed with design improvements as the project progresses. • *Related AREVA DDNs:* AREVA-002 (IHX materials testing); AREVA-024 and AREVA-025 (IHX testing) • *Based on review of GA PCDR*: There is no indication that this item has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* N.13.02.01 thru N.13.02.09 (IHX) • *Based on review of WEC PCDR*: Indications are that this item has been addressed in the WEC PCDR, and will be further addressed with design improvements as the project progresses. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-12 (IHX metallics); HTS-02-01 thru HTS-02-06 (IHX ceramics) • *NGNP R&D Response*: Safety analysis issue – no R&D impact.	*Overall*: Indications are that this item has been addressed only preliminarily thus far. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses. *By Organization*: AREVA - Addressed GA – Not Addressed WEC - Addressed NGNP R&D – Not Addressed
F-3	Failure of the process heat exchanger (PHX) leading to potential damage to safety-related SSCs in the reactor, due to fuel and primary system corrosion from the introduction of	• *Based on review of AREVA PCDR*: There is no indication that this item has been specifically addressed in the AREVA PCDR. • *Related AREVA DDNs:* (None)	*Overall*: Indications are that this item has been addressed only preliminarily thus far. However, from the

TABLE 3F – PROCESS HEAT FOR HYDROGEN - CONCLUSIONS

Item	NRC Need/Issue Identified	Design Integration Review Team Comments/Conclusions	Is this Item being addressed or does it pose a new risk?
	corrosive process plant chemicals leaking down the process heat transport line and failing the IHX.	• *Based on review of GA PCDR*: There is no indication that this item has been specifically addressed in the General Atomics PCDR. • *Related GA DDNs:* N.45.04.02 (HTE heat exchangers) • *Based on review of WEC PCDR*: Indications are that this item has been addressed in the WEC PCDR, and will be further addressed with design improvements as the project progresses. • *Related WEC DDNs:* HTS-01-01 thru HTS-01-12 (IHX metallics); HPS-04-03 thru HPS-04-07 (hydrogen production) • *NGNP R&D Response*: Safety analysis issue – No R&D impact. (*Note:* PHX is tertiary with respect to the reactor.)	standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses. *By Organization*: AREVA – Not Addressed GA – Not Addressed WEC - Addressed NGNP R&D – Not Addressed
F-4	Steam generator failures leading to the introduction of steam/water into the primary system, potentially causing a reactivity spike and chemical attack of the TRISO fuel particle coatings and graphite. Some hydrogen production processes, such as high-temperature electrolysis, require steam as a process feedstock; thus, the high-temperature reactor may be required to provide high-temperature steam.	• *Based on review of AREVA PCDR*: The AREVA PCDR has proposed development of a white paper to provide discussion of water ingress events, including steam generator tube leaks. • *Related AREVA DDNs:* AREVA-040 (steam cycle testing) • *Based on review of GA PCDR*: The General Atomics PCDR indicates that this is not a likely event, and consequences would be acceptable if it did occur. • *Related GA DDNs:* N.07.05.07 thru N.07.05.10 (graphite and other corrosion rates due to water); C.11.03.18 and C.11.03.19 (graphite corrosion data and methods validation); C.11.03.23 (graphite oxidation for postulated accidents); N.45.04.01 (HTE steam generator/superheater) • *Based on review of WEC PCDR*: The WEC PCDR indicates that this issue scenario is not probable for the PBMR design. Nonetheless, significant R&D is planned to ensure reliability of the steam generator. • *Related WEC DDNs:* SG-01-01 thru SG-01-17 (steam generator) • *NGNP R&D Response*: Design dependent. Currently steam generation is tertiary so such an event cannot happen.	*Overall*: This issue seems to have been handled thus far in terms of a non-credibility type of argument with regard to water ingress events. These are typically not easy arguments to make with NRC. These concerns represent risks to the project in terms of securing a successful safety evaluation from NRC. *By Organization*: AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D – Addressed
F-5	Loss of the pressurized coolant inventory from the intermediate loop leading to a loss of primary reactor heat sink and the potential for hydrodynamic forces on the IHX leading to IHX failure and loss of reactor primary system coolant.	• *Based on review of AREVA PCDR*: This item has been addressed in the AREVA PCDR in the design. • *Related AREVA DDNs:* AREVA-002 (IHX materials testing); AREVA-024 and AREVA-025 (IHX testing); AREVA-028 (SCS); AREVA-031 (RCCS) • *Based on review of GA PCDR*: This item has been addressed in the General Atomics PCDR in the design. • *Related GA DDNs:* N.13.02.01 thru N.13.02.09 (IHX); C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS); C.16.00.01 thru C.16.00.06 (RCCS) • *Based on review of WEC PCDR*: Indications are that this item has been addressed in the WEC PCDR, and will be further	*Overall*: Indications are that this item has been addressed only preliminarily thus far. However, from the standpoint of risk, it is reasonable to expect that it will be addressed fully as the design progresses. *By Organization*:

TABLE 3F – PROCESS HEAT FOR HYDROGEN - CONCLUSIONS			
Item	**NRC Need/Issue Identified**	**Design Integration Review Team Comments/Conclusions**	**Is this Item being addressed or does it pose a new risk?**
		addressed with design improvements as the project progresses. • *Related WEC DDNs:* HTS-01-01 through HTS-01-12 (IHX metallics) • NGNP R&D Response: Safety analysis issue – no R&D impact.	AREVA - Addressed GA - Addressed WEC - Addressed NGNP R&D – Not Addressed

6.0 INFORMATION SOURCES

6.1 R. N. Morris et al., *TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents—Main Report*, U.S. Nuclear Regulatory Commission, NUREG/CR-6844, Vol. 1, July 2004.

6.2 S. J. Ball and S. E. Fisher, *Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)—Volume 1: Main Report*, NUREG/CR-6944, Vol. 1 (ORNL/TM-2007/147, Vol. 1), Oak Ridge National Laboratory, March 2008.

6.3 S.J. Ball, *Next Generation Nuclear Plant Gap Analysis Report* (ORNL/TM-2007/228), Oak Ridge National Laboratory, July 2008.

6.4 *NGNP with Hydrogen Production Preconceptual Design Studies Report*, Document #12-9051191-000, AREVA NP Inc., June 2007.

6.5 *NGNP and Hydrogen Production Preconceptual Design Studies Report*, Document #911107, General Atomics, July 2007.

6.6 *NGNP and Hydrogen Production Preconceptual Design Studies Report*, Document # NGNP-01-RPT-001 through NGNP-18-RPT-001, Westinghouse Electric Company, LLC, May 2007.

6.7 *Next Generation Nuclear Plant Pre-Conceptual Design Report*, INL/EXT-07-12967, Rev. 1, Idaho National Laboratory/Battelle Energy Alliance, November 2007.

7.0 APPENDIX A - Design Integration and Review Team Task Description and Plan

Next Generation Nuclear Plant Project
Design Integration Review Team

TASK DESCRIPTION AND PLAN – Phase One – Revision as of 9/02/08

1. Introduction – PIRTs and Gap Analysis performed for NRC analytical and data needs

As a part of their preparation to perform future safety assessment and verification for the Next Generation Nuclear Plant (NGNP) design, the U.S. Nuclear Regulatory Commission (NRC) has performed a Phenomena Identification and Ranking Tables (PIRT) effort. From NRC's point of view, the purpose of this effort is to identify needs they will have for analytical tools and data that will enable them to perform their regulatory function. The detailed PIRT exercises have been documented in NUREG/CR-6844 (July 2004) and NUREG/CR-6944 (March 2008). That data is further analyzed and reduced to an actionable level in the "Next Generation Nuclear Plant Gap Analysis Report" (ORNL/TM-2007/228, July 2008).

An excerpt from the Executive Summary of the Gap Analysis Report (bolding added):

"*This report follows up on recent NRC-sponsored phenomena identification and ranking table (PIRT) exercises for the next generation nuclear plant (NGNP) and is intended to identify the significant "gaps" between what is needed and what is already available to NRC to adequately assess NGNP safety characteristics. Building on the PIRT efforts, this task goes a step further by incorporating evaluations of accident sequences and risk to determine important gaps in the knowledge base and further to recommend how these gaps might be addressed. Report sections are typically organized to first provide a background and summary of what is needed, next identify what data and tools are available, and finally describe the gaps. This information is of interest to NRC assessments of the confirmatory research and development (R&D) needs for NGNP licensing."*

In the PIRT and subsequent Gap Analysis, NRC/ORNL subdivided the NGNP into six areas that are likely to pose challenges in terms of technical review and verification. Those categories are as follows:

1. Accidents and thermal fluids.
2. Reactor physics and neutronics (including criticality calculations and experiments).
3. Fuel performance and fission product transport (FPT) and dose.
4. High-temperature materials (metallic).
5. Graphite.
6. Process heat for hydrogen production.

2. Design Integration Review Team – Checking current Contractor and Project R&D plans against recent Gap Analysis

NGNP project management has requested that the three competing reactor designers (Areva, General Atomics, Westinghouse) perform a reconciliation of their R&D plans, as identified in their respective Pre-conceptual Design Reports, against NRC's PIRT effort. The purpose of this request is to verify that: 1) R&D needs have been addressed comprehensively, and 2) no significant risks and related project costs are being omitted. While these inputs from the three contractors are expected in the near future, NGNP project management has also determined that a Design Integration Review Team should perform this reconciliation internally. The Team consists of the following resources:

- Richard Garrett – NGNP Engineering Director and Team Leader
- Mark Holbrook – NGNP Licensing Engineer
- George Ghonma – Regulatory Consultant
- Bill Mangiante – Engineering/Regulatory Consultant
- Dick Hobbins - NGNP Technology Development Representative

Supporting resources include:

- Jim Kinsey – NGNP Licensing Director
- Phil Mills – NGNP Engineering Deputy Director
- Dave Petti - NGNP Technology Development Director
- John Collins – NGNP Lead Systems Engineer, Technology Development

The task undertaken by the Team will be to compare the Gap Analysis against the most current R&D plans submitted by Areva, General Atomics, Westinghouse, and (by exception) the NGNP internal R&D organization. The Team will determine whether the items in the Gap Analysis are being addressed by the overall NGNP R&D program, and identify any items that might be absent from the Project's R&D plans.

3. Task Methodology

The Team members will each evaluate the Gap Analysis document in comparison to a portion of the currently planned NGNP R&D efforts, with work responsibilities distributed as follows:

- Comparison of Gap Analysis vs. Areva R&D Plan - Mangiante
- Comparison of Gap Analysis vs. General Atomics R&D Plan - Mangiante
- Comparison of Gap Analysis vs. Westinghouse R&D Plan – Holbrook/Ghonma
- Determine, by exception, whether any unaddressed NRC gaps are accounted for by internal R&D programs – Petti/Hobbins
- Incorporation of all inputs into final report – Mangiante
- Review and approval of final report – Garrett, Kinsey, Petti

The work output of each comparison effort will consist of a table similar to Table 1, which includes columns for Gap Analysis items and the planned R&D efforts. The Team member will perform the comparison for his assigned R&D Plan(s), and provide comments if applicable for any specific Gap Item/Applicable R&D Program. The Team member will also draw conclusions to the degree that they are possible from this effort, specifically addressing any perceived inadequacies that may exist, relative to the ability of planned R&D to meet needs identified by NRC/ORNL. The Team member may provide his comments and/or conclusions on text separate from the table, if that is most efficient for him. A separate set of worksheets like Table 1 will be filled out for R&D planned by Areva, General Atomics, and Westinghouse. The individual Table 1 comparison inputs will be summarized as shown in Table 2. If any exceptions (i.e., gaps not addressed) are identified from the comparisons of "gaps" to those 3 R&D plans, the exceptions will also be reviewed vs. the NGNP internal R&D plan, to see if they are addressed there. Any gaps so determined to be associated with NGNP internal R&D efforts will be described in the Team's report deliverable.

The Gap Analysis document defined actual gaps in 4 of the 6 major areas identified. The two categories with no identified gaps will be handled as follows:

- In the Reactor Physics and Neutronics area, the Gap Analysis document indicated that there were no significant gaps to identify, but it did identify "Major Phenomena of Interest" that required particular attention. These are called out as such in Table 1, and the Team members will enter R&D efforts opposite these issues if such R&D efforts are planned.

- The Gap Analysis document identified no gaps in the Process Heat for Hydrogen area, because the authors felt they did not have enough information on the design to identify gaps. In this particular area, the Team members again will use the "Major Phenomena of Interest" items described in the Gap Analysis, as these items capture the PIRT items of highest concern to NRC as documented in NUREG/CR-6944.

4. Deliverables and Schedule

The deliverable for this task will be a report documenting the Team's work process and conclusions in comparing the Gap Analysis vs. R&D efforts currently planned by the NGNP Project and its contractors. The report will contain the individual comparison efforts as illustrated in Table 1, the summary of comparisons as illustrated in Table 2, and descriptive text that will explain the task and summarize any conclusions that are drawn. A draft of this report will be submitted no later than September 8, 2008, so that it is available for the upcoming NGNP safety basis review. The remainder of the schedule is laid out below.

Schedule Milestone Dates:

- Start Task – 8/11/08
- Complete comparison and provide comparison table and comments/conclusions as inputs for draft report – 9/5/08
- Incorporate all available inputs into draft report – 9/8/08
- Incorporate DDN #s into input tables – 9/12/08
- Receive inputs on exceptions from NGNP R&D organization – 9/12/08 and 9/19/08
- Complete review draft of report deliverable and distribute for management review – 9/19/08
- Receive management review comments – 9/26/08
- Issue final report with review comments incorporated – 9/30/08

Note: Important to keep in mind - NRC's needs for analysis tools and data for performing the safety review of NGNP are not precisely the same as the greater NGNP Project needs for designing, advancing technologies, constructing, and performing safety analysis for the NGNP.

- NRC's need for data will presumably be fulfilled by the NGNP Project, and should represent a subset of NGNP's total data output (example: materials characteristics data on high temperature metals and graphite).

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- NRC's need for analytical tools may be similar to NGNP's, but may be fulfilled by other outlets so NRC can perform independent verification (example: safety analysis codes).

- NGNP may perform a significant amount of R&D that is essential to the project but beyond NRC's needs for safety review (example: operational and commercial R&D).

However, NRC and NGNP R&D needs should have a lot in common, and thus, this task should produce useful correlations, and insights as to whether planned NGNP R&D efforts have deficiencies that have not been identified previously and need to be addressed.

Table 1 – Worksheet for Performing Comparisons between Gap Analysis and Planned _________ R&D Efforts

Item	NRC Needs/Issues Identified	Applicable R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
A-1, A-2, etc.	**Accidents and Thermal Fluids** • (Gap A) • (Gap B) • Etc.			
B-1, B-2, etc.	**Reactor physics and neutronics** • (Phenomena of Interest A) • Etc.			
C-1, C-2, etc.	**Fuel performance and fission product transport (FPT) and dose** • (Gap A) • (Gap B) • Etc.			
D-1, D-2, etc.	**High-temperature materials (metallic)** • (Gap A) • (Gap B) • Etc.			
E-1, E-2, etc.	**Graphite** • (Gap A) • (Gap B) • Etc.			
F-1, F-2, etc.	**Process heat for hydrogen production** • (Phenomena of Interest A) • Etc.			

(Notes: A separate set of worksheets like this one will be filled out for Areva, General Atomics, and Westinghouse. These tables will be presented in 8 ½" x 14" format.))

Table 2 – Summary of Comparisons between Gap Analysis and Planned NGNP R&D Efforts

Item	NRC Needs/Issues Identified	Applicable AREVA R&D or Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
A-1, A-2, etc.	**Accidents and Thermal Fluids** • (Gap A) • (Gap B) • Etc.				
B-1, B-2, etc.	**Reactor physics and neutronics** • (Phenomena of Interest A) • Etc.				
C-1, C-2, etc.	**Fuel performance and fission product transport (FPT) and dose** • (Gap A) • (Gap B) • Etc.				
D-1, D-2, etc.	**High-temperature materials (metallic)** • (Gap A) • (Gap B) • Etc.				
E-1, E-2, etc.	**Graphite** • (Gap A) • (Gap B) • Etc.				
F-1, F-2, etc.	**Process heat for hydrogen production** • (Phenomena of Interest A) • Etc.				

(Note: These tables will be presented in 11” x 17” format.)

8.0 APPENDIX B – Abbreviated Results Table

Appendix B – ABBREVIATED RESULTS TABLE

	ITEM	ABBREVIATED ISSUE DESCRIPTION	IS THIS ITEM BEING ADDRESSED?				
			AREVA	GA	WEC	NGNP-R&D	OVERALL
ACCIDENTS AND THERMAL FLUIDS	A-1	Core-Coolant Bypass Flow Phenomena (Normal Operation)	Yes	Yes	No	Yes	Yes
	A-2	Effective Core Thermal Conductivity	Yes	Yes	Yes	Yes	Yes
	A-3	Afterheat Correlations	No	Yes	No	Yes	Yes
	A-4	Core Effective Pressure Drop	No	Yes	No	Yes	Yes
	A-5	RCCS Performance during LOFC	Yes	Yes	No	No	Yes
	A-6	Fuel Performance Models	Partially	Yes	Yes	Yes	Yes
	A-7	Air Ingress Phenomena	Yes	Yes	Yes	Yes	Yes
	A-8	Long-term analysis need	Partially	Partially	Yes	Yes	Yes
REACTOR PHYSICS AND NEUTRONICS	B-1	Time-dependence and spatial distribution of decay heat	Yes	Yes	Yes	Yes	Yes
	B-2	Control and reserve shutdown worth	:Yes	Yes	Yes	Yes	Yes
	B-3	Sudden positive reactivity insertion due to pebble core compaction	NA	NA	No	Yes	Yes
	B-4	Use of high-temperature in-core neutron detectors to determine power anomalies	Yes	Yes	No	Yes	Yes
	B-5	Control rod misalignments resulting in power tilting and xenon-135-induced oscillations	No	No	No	No	No
	B-6	Positive reactivity insertion due to steam/water ingress	Yes	Yes	No	No	Yes
	B-7	“Harder” thermal-neutron energy spectrum with graphite than with water moderation	No	No	No	No	No
	B-8	Variations in fuel characteristics accounted for in calculating fuel temperature coefficient of reactivity and the MTC	No	No	No	No	No
	B-9	Operator and equipment must be qualified for safety functions during extended ATWS	No	No	No	No	No
	B-10	Uniquely tall, thin annular core and high operating temperatures require detailed reactor physics testing and instrumentation	Yes	Yes	No	Yes	Yes
FUE	C-1	General Safety Analysis/Safety Document Needs	Partially	Partially	Partially	Partially	Partially

Appendix B – ABBREVIATED RESULTS TABLE

	ITEM	ABBREVIATED ISSUE DESCRIPTION	IS THIS ITEM BEING ADDRESSED?				
			AREVA	GA	WEC	NGNP-R&D	OVERALL
	C-2	Model Development and V&V	Partially	Partially	Partially	Partially	Partially
	C-3	Materials/Component Data	Partially	Partially	Partially	Partially	Partially
	C-4	Role of reactor component and confinement/containment configuration in the safety case	Partially	Partially	Partially	No	Partially
	C-5	Computational software or other methods for determining the quantitative results	Yes	Yes	Yes	Yes	Yes
	C-6	Integral testing to support development of computational methods and data quantification	Yes	Yes	Yes	Yes	Yes
HIGH TEMPERATURE MATERIALS (METALLIC)	D-1	Physical Materials Data for high-temperature metallic components	Partially	Partially	Yes	Yes	Yes
	D-2	Physical Materials Data for high temperature structural composites	Partially	Yes	Yes	Yes	Yes
	D-3	Compromise of RPV surface emissivities due to loss of desired surface layer properties	Yes	Yes	Yes	Yes	Yes
	D-4	Aging and environmental effects on insulation	No	No	Yes	Yes	Yes
	D-5	Roles of design methods, manufacturing controls, inspection and testing in avoiding primary boundary failures in compact IHX	Yes	Yes	Yes	Yes	Yes
	D-6	Role of structural design methods for composites in avoiding control rod insertion failures	Yes	Yes	NA	Yes	Yes
	D-7	Irradiation induced creep of in-vessel metallic structures.	Yes	Yes	Yes	Yes	Yes
	D-8	Role of structural design and fabrication for composites in avoiding core radial restraint failure	No	No	Yes	Yes	Yes
	D-9	Isolation and other valve failures such as self-welding, galling and seizing	Yes	Yes	Yes	Yes	Yes
	D-10	Development of data and models needed for ASME Code qualification of metallic materials	Yes	Yes	Yes	Yes	Yes
	D-11	Development of a flaw assessment procedure for predicting crack-induced pressure boundary failures	Yes	No	Yes	No	Yes
	D-12	Development of materials data and extrapolation procedures for proposed operating temperatures and component lifetimes	Yes	Yes	Yes	Yes	Yes

Appendix B – ABBREVIATED RESULTS TABLE

	ITEM	ABBREVIATED ISSUE DESCRIPTION	IS THIS ITEM BEING ADDRESSED?				
			AREVA	GA	WEC	NGNP-R&D	OVERALL
	D-13	Development of data supporting IHX metallurgical stability and environmental resistance in potentially impure helium environment	Yes	Yes	Yes	Yes	Yes
	D-14	Quantify IHX crack initiation and propagation due to creep, creep-fatigue, and aging	Yes	Yes	Yes	Yes	Yes
	D-15	Address issues for RPVs too large for shop fabrication and transportation	Yes	No	Yes	Yes	Yes
	D-16	Develop time-dependent design criteria, analytical models, and verification testing for complex high temperature metallic compact IHX structures, addressing scalability and fabrication issues.	Yes	Yes	Yes	Yes	Yes
GRAPHITE	E-1	Confirmatory data for grades of graphite selected	Yes	Yes	Yes	Yes	Yes
	E-2	Consensus codes and standards for grades of graphi9te selected	Yes	Yes	Yes	Yes	Yes
	E-3	Extend and V&V theoretical models for effects of neutron damage on properties of graphite	Yes	Yes	Yes	Yes	Yes
	E-4	Graphite uncertainties: temperature and dose received by components; severity of temperature and dose gradients; rate of dimensional change; extent of changes in key physical properties	Yes	Yes	Yes	Yes	Yes
	E-5	Development of whole-core models to predict the stress states of graphite components within the reactor core	Yes	Yes	Yes	Yes	Yes
	E-6	Basic graphite properties research to strengthen understanding and modeling capability	Yes	Yes	Yes	Yes	Yes
	E-7	Irradiation induced change in graphite coefficient of thermal expansion, including effects of creep strain.	Yes	Yes	Yes	Yes	Yes
	E-8	Irradiation induced change in graphite mechanical properties such as strength and toughness	Yes	Yes	Yes	Yes	Yes
	E-9	Blockage of coolant channel in a fuel element block or reactivity control block due to graphite failure and/or graphite spalling	No	No	Yes	No	Yes
	E-10	Statistical variation of non-irradiated graphite properties due to forming, processing, raw materials, and formulation	Partially	Partially	Yes	Yes	Yes
	E-11	Development of generic graphite specifications to ensure consistent graphite quality	No	Partially	Yes	Yes	Yes

Appendix B – ABBREVIATED RESULTS TABLE

	ITEM	ABBREVIATED ISSUE DESCRIPTION	IS THIS ITEM BEING ADDRESSED?				
			AREVA	GA	WEC	NGNP-R&D	OVERALL
	E-12	Tribology (effects of moving surface interactions) of graphite in helium environment	Yes	Partially	Yes	Yes	Yes
	E-13	Impact of degradation of thermal conductivity on fuel temperature limits	Yes	Yes	Yes	Yes	Yes
PROCESS HEAT FOR HYDROGEN	F-1	Cold oxygen (O2) and other heavy-gas accidental releases from process plant impacting integrity of reactor SSCs	No	No	Yes	No	No
	F-2	Failure of IHX leading to potential damage to safety-related SSCs in reactor due to blow-down of helium in secondary loop	Yes	No	Yes	No	Yes
	F-3	Failure of the process heat exchanger (PHX) leading to potential damage to safety-related SSCs in reactor	No	No	Yes	No	No
	F-4	Steam generator failures leading to introduction of steam/water into primary system, causing reactivity spike and chemical attack fuel particle coatings and graphite	Yes	Yes	Yes	Yes	Yes
	F-5	Loss of pressurized coolant inventory from intermediate loop leading to loss of primary reactor heat sink, IHX failure, and loss of reactor primary system coolant	Yes	Yes	Yes	No	Yes

Table 1A (AREVA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
A-1	*Core-Coolant Bypass Flow Phenomena (Normal Operation)* • Overcome difficulties in estimating bypass flow • More complete understanding and accounting of related design features such as fuel blocks (PMR) and core barrel configurations • In-core temperature testing • Parametric analysis of gap configurations to bound questions associated with gap and bypass flows	(Sec 4.3.1, p. 42) – "The ***core bypass flow*** shall be maintained within an acceptable range which ensures a good compromise for the fuel temperature in ***normal and accidental conditions*** (existence of a minimum amount of bypass in lateral reflector)." (Sec. 6.1.1.7.1, p. 52) - ***Core Bypass Flow*** – "The major issue of the thermal-hydraulic design is the core bypass flow. It is directly related to the core thermal performance. In the core, the flow partitions itself among the coolant channels, the absorber element channels and the gaps between the columns of fuel and reflector blocks. The objective in the core flow design is to maximize flow through the coolant channels (which directly flow to where the power is being produced). This means ***minimizing flow through the gaps between columns,*** and limiting the flow in the absorber element channels to that needed to cool the absorbers when they are inserted. ***Refined analyses will need to be performed in the frame of the Conceptual Design to assess the value of core bypass flow and propose design improvements to minimize the bypass***." (Sec. 17.6, p. 240) – ***Initial Startup Operations and Testing*** – "…The schedule provides four years for this phase of plant operations. The first two years (2017 and 2018) is dedicated to non-nuclear system testing and turn-over including the standard system turn-over from construction to operations. The second two years (2019 and 2020) includes initial plant criticality. During this phase all safety systems will be examined and tested and several special licensing related tests is planned. This phase of the plant operation includes component dismantling and inspection and fuel examination." (Sec. 19.2.3, p. 289) – ***Instrumentation*** – "NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy. For neutron flux detectors some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired."	AREVA-011 thru AREVA-022 (modeling codes) AREVA-002 (core barrel) AREVA-030, and AREVA-034 (instrumentation and testing)	The needs for refined analyses to better understand the core bypass flow phenomenon, and core monitoring instrumentation and testing, have been recognized in the AREVA PCDR.

Table 1A (AREVA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
A-2	*Effective Core Thermal Conductivity* • For prismatic cores – Make available dose and temperature-dependent graphite thermal properties (especially thermal conductivity) to the NRC T/F code suite, to account for large uncertainties as well as for characterization of annealing effects during long-term heat-up D-LOFC accidents. • For pebble bed cores - Also considerable error bounds in effective core thermal conductivity as a function of both temperature and irradiation. Existing correlations available are empirical, but PBMR project has an experimental facility to be used to refine the database.	(Sec. 19.2, p. 281) – ***R&D Needs*** – "Materials development and qualification. This covers certain high-temperature steels, composites, and ***graphite*** selection/qualification." (Sec. 19.2.2.2, p. 285) – ***Ceramics*** – "No nuclear components or structures made of composites were used for the past HTRs or for other reactor concepts. The use of composites is driven by their high resistance to high or very high temperatures. An R&D program has been launched in the frame of Antares to explore the possible use of such materials inside the primary circuit. Thermal insulation, using composite materials, will be needed to provide thermal protection of metallic components which would otherwise be subjected to helium at very high temperatures. The R&D needs for applied composite materials (C/C or C/SiC composites) emphasizes qualification of material properties such as: 1. ***thermal-physical properties (thermal conductivity*** (K), coefficient of thermal expansion (CTE), heat capacity (Cp)), 2. mechanical properties including multiaxial strength, 3. fracture properties, 4. fatigue properties and 5. behavior in an oxidizing atmosphere and oxidation effects on properties." (Sec. 19.2.2.3, p. 286) - ***Graphite Materials*** – "Graphite, an essential structural material for the VHTR, will operate under significant irradiation conditions and requires a characterization in the range of expected temperatures. Nuclear grade graphite was used in past HTRs programs, amassing a substantial database. These grades are no longer available. An R&D program has been launched within Antares program to select the best candidates among the new available grades or to request the development of a new grade, and to acquire design data. Nuclear graded structural graphite (PCEA, NBG17 and/or NBG18) qualification includes: 1***. thermal-physical properties*** (***K***, CTE, Cp, emissivity), 2. mechanical properties including multiaxial strength, 3. fracture properties, 4. fatigue properties, 5. irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep, 6. behavior under oxidized atmosphere including oxidation effects on properties and 7. tribology. Due to schedule limits, it is recommended that graphite R&D be performed in two phases: preliminary and detailed… Development of ASME and ASTM codes and standards for graphite is essential for timely application graphite for NGNP reactor."	AREVA-001 thru AREVA-003 (metallics) AREVA-004 thru AREVA-007 (ceramics) AREVA-008 and AREVA-009 (graphites) AREVA-010 (core internal structure)	The needs for determining the properties of graphite materials, including thermal conductivity, have been recognized in the AREVA PCDR.

Table 1A (AREVA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		The pebble bed core portion of the item is not applicable to the PMR.		
A-3	*Afterheat Correlations* • Peak fuel temperatures in the D-LOFC accident are very sensitive to the afterheat (vs. time) to the same extent as they are to the core thermal conductivity function. Afterheat correlations are sensitive to fuel type and burn-up histories. Tracking fuel histories during operation can be challenging, and afterheat validation data is more difficult to obtain for long times after shutdown.	***There is no indication that this item has been specifically addressed.*** However, reactor system analysis software (MANTA, RELAP), neutronics software (MCNP, NEPHTYS, MONTENURNS, CABERNET), and fuel performance software (ATLAS) are addressed in section 19.2.4, including discussions of ***fuel burnup***.	AREVA-011 thru AREVA-022 (modeling codes)	***There is no indication that this item has been specifically addressed in the AREVA PCDR.***
A-4	*Core Effective Pressure Drop* • Standardized and well-documented correlations for core pressure drop; conformation data may be needed for low-flow cases to better characterize flow distribution and plume formation (for the P-LOFC) and in-core airflow distributions during air ingress accidents. • PBR - parametric analyses using established ranges of different packing fractions to define a performance envelope.	***There is no indication that this item has been specifically addressed.*** However, thermal-hydraulics software (STAR-CD) is addressed in section 19.2.4, and would include calculations of ***pressure in the core***. The pebble bed core portion of the item is not applicable to the PMR.	AREVA-011 thru AREVA-022 (modeling codes)	***There is no indication that this item has been specifically addressed in the AREVA PCDR.***
A-5	*RCCS Performance during LOFC* • Simulate RCCS safety functions in detail, with its predominantly radiant heat transfer coupling to the RPV and other heat transfer mechanisms within the reactor cavity. RCCS functions include maintaining the reactor cavity liner concrete temperature below prescribed limits, preventing the RPV peak temperature from exceeding limits during LOFC events, and minimizing parasitic heat losses during normal operation. • Models may be needed to simulate large pressure pulses in D-LOFC accidents that could damage the RCCS, reducing cooling and/or opening up another release path for air or water ingress to the reactor cavity, and perhaps for FPT out to the environment.	(Table 19-5, p. 290) – ***MANTA*** – “Calculation of main system parameters (temperature, pressure, flow rate) of the HTR plant during all transient (normal, abnormal) when the primary coolant flows in forced convection, in order to define plant operation and control and to provide load data for primary components. Possibility to calculate generalized natural convection.” (Code is fully applicable, needs validation) (Table 19-5, p. 290) - ***STAR-CD*** – “Determination of: 1) thermal loadings on the components (vessels, internals, fuel…) during normal or upset conditions, 2) the thermal behavior of the core, 3) the mixing inside the primary system, 4) heat losses and performances of components, 5) flow repartition across the components and 6) ***pressure shock waves***.” (Code is fully applicable, needs validation)	AREVA-011 thru AREVA-022 (modeling codes)	The needs for modeling and simulation code development described in this item have been recognized in the AREVA PCDR.
A-6	*Fuel Performance Models* • Aspects of maximum fuel temperature plus time-at-temperature histories (critical limiting factors) for all fuel regions provide inputs to fuel failure models, to determine source terms and dose-vs.-frequency	(Sec. 19.2.4.3, p. 292) – ***Thermal Hydraulics/Pneumatics Codes/STAR-CD*** – “Code development and qualification R&D needs are evaluated at a “High” Priority. • Development of ***graphite oxidation model for air ingress transients*** on reactor internal structures.	AREVA-014, AREVA-016, and AREVA-022 (computer codes)	The needs for modeling and simulation code development described in this item have generally been recognized in the AREVA PCDR. ***However, it is not possible to determine whether these codes will***

Table 1A (AREVA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	estimates. • Chemical reactions in air or water ingress accidents, which depend on temperature and should be included in the T/F codes. Especially for fast transients, detailed temperature profiles of the fuel and graphite should be taken into account for thermal stress calculations.	• Qualification of: ○ conduction cooldown models on representative geometry, materials and temperature, ○ turbulence and mixing on representative mock-ups in critical areas (lower and upper reactor plena, hot gas duct, core bypass, IHX collectors) and ○ graphite oxidation models with selected graphite grades in representative operating conditions. Several predecessor tests performed with different graphite grades at CEA and FZJ. NACOK experiments within the European RAPHAEL project (coupling of graphite models with thermo-fluid dynamic behavior) can be applied for STAR-CD qualification." (Sec. 19.2.4.4, p. 292) – ***Fuel Performance Models and Codes/ATLAS*** – "The R&D need for ATLAS development/modification is to improve the diffusion and the coatings corrosion modeling. For code qualification the heat-up experiments of irradiated fuel particles at relevant operating conditions (burnup, temperature, fluence) are required to anchor the developed code…qualification of ATLAS…includes two irradiation and heat up tests. In addition, there is an R&D need to develop the UCO models."		***include:*** • ***time-at-temperature histories for all fuel regions*** • ***chemical reactions in water ingress accidents (AREVA seems to have determined that these are not credible events)*** • ***detailed temperature profiles of fuel and graphite in fast transients***
A-7	*Air Ingress Phenomena* • With little or no detail available about the confinement, only generalized studies and experiments would be practical. Bounding analytical studies could be useful in determining positive and negative features of proposed design characteristics. The major features of general interest would be quantification of long-term air in-leakage into the confinement, and the mixing and stratification characteristics of gases in prototypical cavities within the confinement.	(Sec. 11.5.2.3, p. 178) – ***Air Ingress*** – The current state of knowledge of air ingress is provided in this section from an accident analysis point of view, with the following uncertainties identified: • Influence of conduction cooldown uncertainties • Benefit of primary circuit loop isolation strategies • Benefit of SCS actuation • Influence of air on fuel particles performances as well as on the radio-elements trapped in the graphite blocks • Onset of global natural convection and, particularly, the determination of the time when it starts • Consequences of CO release • Limitation of air available in the pressure boundary cavity by design and possible operator actions • For large breaks, the assumptions concerning the shutdown of the reactor and the main circulator have to be assessed in order to evaluate if their failure could drastically increase the consequences. If it is the case, these actions should be performed with a high reliability for practically eliminating their occurrence.	AREVA-028 and AREVA-031 (SCS and RCCS)	The need for greater understanding of the air ingress phenomenon has been recognized in the AREVA PCDR.

Table 1A (AREVA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		• Reliability and role of heat removal systems (Sec. 21.1.3, p. 315) – ***Air Ingress Assessment*** – "Air ingress events are a potential issue for all graphite moderated HTRs, due to the concerns associated with graphite oxidation. This issue is similar to the water ingress scenarios in that, while there is a credible technical issue which must be addressed in the course of the reactor design and safety analysis, there is also a large perception issue that is somewhat independent of the technical issues. An objective characterization of air ingress events is recommended in order to put these events in the proper context. The recommended assessment of air ingress events should include scenario definition, controlling phenomenon, potential consequences, and mitigation strategies. The objective is to provide a reasonable framework for the discussion and quantitative evaluation of these events."		
A-8	*Long-term analysis need* - Comprehensive suite of verified and validated accident simulation codes (core thermal-fluids, core neutronics, whole-plant transient behavior, confinement analysis, and chemical reactions), agreed-upon accident cases for regulatory acceptance, and robust supporting databases that NRC can use for independent confirmatory analysis of candidate plant and confinement designs and options.	(Sec. 19.2.4, p. 290) – ***Computer Codes and Methods Development and Validation –*** Included in this section are descriptions of R&D needs for computer codes addressing reactor system analysis, neutronics, thermal hydraulics/pneumatics, fuel performance, fission product transport, and structural mechanics.	AREVA-011 thru AREVA-022 (modeling codes)	Long-term analysis needs for computer code development have been recognized in the AREVA PCDR. Databases have been addressed by AREVA in terms of candidate alloys and fuel materials.

Table 1B (AREVA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
B-1	Time-dependence and spatial distribution of decay heat as a major factor in determining maximum fuel temperature during a D-LOFC.	(Sec. 11.5..2.1, p. 175) – ***Loss of Primary Forced Convection*** – Conduction Cooldown Events – This section contains AREVA's ***bounding D-LOFC*** (and limiting design basis event)***,*** referred to as a Depressurized Conduction Cooldown (DCC). The section describes the plant engineered safety features response to the event, indicating that the ***temperature increase is slow and peak temperatures for fuel and core support structures are limited***. (Sec. 19.2.4.2, p. 292) – ***Neutronics Codes/MONTEBURNS*** – "The R&D needs for MONTEBURNS are "High Priority." 1. Experimental results of fuel irradiation experiments (compacts or pebbles) at representative burnups, temperatures and fluences. 2. Experimental results of ***decay heat*** at short term (<100 hours) for representative fuel composition and burnup.	AREVA-011 thru AREVA-022 (modeling codes)	The needs for modeling and simulation code development described in this item have generally been recognized in the AREVA PCDR.
B-2	Control and shutdown rod worth and reserve shutdown worth as required for hot and cold shutdown.	(Sec. 19.2.4.2, p. 291) – ***Neutronics Codes/MCNP and NEPHTYS*** – "The R&D needs for both MCNP and NEPHTYS are of "High Priority." 1. The approach for qualification consists of comparing results against Monte-Carlo reference calculations and benchmarking against the few available experimental data (FSV, HTTR). Thus new dedicated critical experiments, with an asymptotic spectrum representative of the expected prismatic fuel assembly and core, with full access to pin-by-pin power distributions, and ***control rod and burnable poisons worths*** are needed. 2. Experimental data of neutronic characteristics (spectrum, fission and capture rates) at the interface between a prismatic fuel assembly and a graphite reflector assembly. Data from FSV and HTTR first criticality testing can be applicable to MCNP and NEPHTYS code qualification."	AREVA-011 thru AREVA-022 (modeling codes)	The need for understanding control and reserve shutdown capability, as described in this item, is recognized in the AREVA PCDR.
B-3	Sudden positive reactivity insertion due to pebble core compaction (packing fraction) due to earthquake.	PBR phenomena; not applicable to PMR core.	(Not applicable)	This is a PBR phenomenon and is not applicable to the PMR core.
B-4	For tests at both PMRs and PBRs, consideration should be given (at least in the first core) to use of high-temperature in-core neutron detectors that can provide maps of axial and azimuthal power distributions and core-inner-to-outer-radius power tilts; these detectors would likely be located only in the inner and outer reflectors rather than in the core, due to temperature and connection limitations. • PMR concern - Whether improper axial-loading of fuel blocks during refueling can lead to an undetected power distribution anomaly and result in excessive operating fuel temperatures.	(Sec. 6.1.3.2, P. 64) – ***Neutron Control Equipment*** – "Neutron control is effected using equipment for positioning the control rods and nuclear instrumentation. The primary components include Neutron Control Assemblies (NCA) and nuclear instrumentation. There are 24 NCAs, of which 18 are used for the 36 operating control rods in the outer reflector and 6 are used for the 12 startup control rods in the inner core. Each NCA contains 2 independent chain, wheel, gear and motor type control rod drives – one per control rod. A friction clutch between each motor and the drive mechanism is included to prevent overload. The nuclear instrumentation consists of ex-vessel ***neutron detectors***, source range detectors, and ***in-core flux mapping units***. During normal operation, the neutron flux levels are monitored by the ex-vessel neutron detectors, whose range overlaps with that of the source-range detectors. During startup and shutdown, the neutron flux levels are monitored using	AREVA-030, and AREVA-034 (instrumentation and testing)	The need for core monitoring instrumentation has been recognized in the AREVA PCDR.

Table 1B (AREVA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
	• PBR concern - Radial and azimuthal power distributions in the mixed-fuel pebble bed are not well known, and there are indications from melt-wire tests conducted in the AVR (Germany) suggesting that pebbles near the walls of the reflector experienced unexpectedly high fuel temperatures.	the source-range detectors. The in-core flux mapping units are used to verify ***axial flux profiles*** and confirm power stability." (Sec. 19.2.3, p. 289) – ***Instrumentation*** – "NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy. For ***neutron flux detectors*** some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired." The PBR portion of this item is not applicable to the PMR core.		
B-5	In both the PMR and PBR, control rod misalignments in the outer reflector during operation would result in azimuthal power tilting that could cause xenon-135-induced oscillations when the misalignment is corrected; however, this needs to be verified by analysis and confirmed by test.	***There is no indication that this item has been specifically addressed.***	AREVA-030 (testing of control rod drive system)	***There is no indication that this item has been specifically addressed in the AREVA PCDR.***
B-6	Replacing helium with a hydrogen-bearing compound such as in a steam/water ingress event may produce a pronounced positive reactivity. Steam/water ingress tends to have a positive reactivity effect due to increased neutron moderation and reduced neutron leakage.	(Sec. 11.5.2.4, p.179) – ***Water Ingress*** – Water ingress is treated from an accident analysis perspective in sec. 11.5.2.4, including the identification of positive reactivity insertion as an unresolved issue. Other unresolved issues are identified, including: • Benefit of start up of the SCS • Benefit of primary circuit loop isolation strategies • Impact of water on graphite structure and its heat transfer properties • Influence of water on fuel particles performances as well on the radio-elements trapped in the graphite blocks • Consequences of CO and H2 release • Limitation of water available to enter the pressure boundary • Impact of possible actuation of safety valve (primary and secondary) on potential radiological releases (Sec. 21.1.2, p. 315) – "Any decision to adopt a steam cycle HTR configuration increases the significance of ***water ingress events*** due to the potential for steam	(None)	The need for greater understanding of the water ingress phenomenon has been recognized in the AREVA PCDR.

Table 1B (AREVA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		generator leaks. This issue was successfully managed in previous operating HTRs. However, the possibility for water ingress continues to be perceived as a significant issue within the broader nuclear community. There are various reasons for this including misunderstanding of the source of water ingress in the Fort St. Vrain reactor, failure to appreciate the differences in steam generator technology between HTRs and LWRs, and unfamiliarity with the consequences and mitigation of water ingress in HTRs. Steam line breaks within the reactor building also must be considered for steam cycle concepts. Steam line breaks must be evaluated for building pressurization issues and for any impact on building venting and filter systems, if a vented confinement concept is used for the NGNP. A ***white paper should be developed addressing water ingress*** and steam line break events and their likely impact on NHS design. The intent is not necessarily to provide detailed analyses of such events. Rather the focus should be on describing the issues and concerns associated with each type of event, the potential significance of these events on operation, safety, and licensing, mitigation of these events including likely design features which might be utilized, and ***likely R&D*** that might be necessary to resolve any open issues."		
B-7	With a higher atomic mass moderator such as carbon, the mean thermal energy of neutrons will be higher than that for hydrogen bound with oxygen in water; that is, graphite will tend to produce a "harder" thermal-neutron energy spectrum than would water-moderated systems. Thus, the moderator temperature-dependent reactivity coefficient (MTC) in both PMR and PBR depends upon the change of thermal-neutron energy spectrum with temperature, with possibly large effects on reactivity. Concerns are for effects on core transient behavior and passive safety shutdown characteristics.	(Sec. 4.3.1, p. 43) – "The ***reactivity temperature coefficient*** shall be sufficiently negative to shutdown the nuclear chain reaction before an unacceptable fuel temperature is reached, and maintain the core in a safe state for a time offering the certainty to reliably introduce absorber elements." (Sec. 6.1.1.3, P. 49) – ***Core Reactivity Control*** – "The ***core reactivity*** is controlled by the ***core negative temperature coefficient*** and control rods, and possibly by lumped burnable poison located in the fuel assemblies. It is also complemented by the Reactor Reserve Shutdown System (RRSS). This system is used to shutdown the reactor and maintain it a sub-critical state if the rod system fails to trip the reactor." (Sec. 6.1.1.4, p. 49) – ***Reactivity Balance*** – "The core reactivity balance is presented in Table 6-2 for Beginning of Cycle (BOC) and End of Cycle (EOC) and includes the following items: • Reactivity due to equilibrium xenon. • ***Temperature reactivity effect*** (Doppler, moderator, and reflector) - their sum represents the cold to hot transition. • Reactivity due to burn-up, which is the excess reactivity required to achieve cycle lifetime. • Control rod worths. The sum of the xenon worth, the total temperature reactivity effect, and the burn-up reactivity yields a BOC required control rod worth of 19.2 %Δk/k. The total available worth is 24.9 %Δk/k, which is sufficient to cover stuck rod worth and shutdown	AREVA-030 (RCCS) AREVA-031 (neutron control system drive mechanism)	The AREVA PCDR has addressed AREVA's design strategies for reactivity control and neutron control, as features of the design. ***AREVA has not specifically addressed NRC's concern over the "harder thermal-neutron energy spectrum" and its "possibly large effects on reactivity".***

Table 1B (AREVA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		margin." (Sec. 6.1.3, p. 63) – ***Neutron Control*** – "The core reactivity is primarily controlled by the core negative temperature coefficient and control rods. In addition, the placement of fuel blocks having known reactivity based on burn-up (for irradiated fuel), initial enrichment levels and the possible inclusion of burnable poisons provide for further control of reactivity. Core reactivity control is complemented by the Reactor Reserve Shutdown System (RRSS) that will safely shutdown the reactor and maintain a subcritical state in the event that the control rods fail to operate during accident conditions." (Sec. 6.1.3.4, p. 65) – ***Neutron Control During Accident Conditions*** – "The detection of reactivity insertion events leads to reactor shutdown by automatic insertion of the control rods by the Reactor Protection System (RPS). In cases where the events are coupled with a loss of electrical power, the controls rods will drop into the core by gravity. The RRSS, can be manually actuated to achieve a diverse method of reactor shutdown, should control rod insertion not be accomplished. The two neutron absorbing systems are designed so that the insertion of either one of these systems ensures and maintains a subcritical state in all conditions. This includes, in particular, the reactivity due to the core cooling down to the coldest shutdown state combined with the xenon effect and the ***reactivity insertion*** due to the initiating event."		
B-8	Variations in fuel enrichments, kernel diameters, coatings, and density of packing (PMR vs. PBR) must be accounted for in calculating the neutron reaction self-shielding effects in both the resonance or epithermal region and the thermal region of the neutron energy spectrum, to properly calculate the Doppler fuel temperature coefficient of reactivity and the MTC.	***There is no indication that this item has been specifically addressed.*** However, the importance and uncertainties associated with fuel fabrication and consistent fuel quality are well-recognized. See section 15.0***, Fuel Strategy***, beginning on page 220. Also, the R&D aspects of fuel development and qualification (***fuel kernel, coating***, compact, QA, and mass production) are addressed in section 19.2.1, beginning on page 282.	AREVA-014 (fuel performance modeling and codes)	***There is no indication that this item has been specifically addressed in the AREVA PCDR.***
B-9	Due to concerns over control rod drive reliability and re-criticality after Xenon-135 decay, the plant operator retains the safety function of achieving long-term hot and cold shutdown during an extended ATWS; and the equipment used by the operator to carry out this safety function, whether located in the control room or in a remote location, must be appropriately qualified to execute that safety function.	(Sec 6.1.3.4, p. 65) – Neutron Control During Accident Conditions – "The detection of reactivity insertion events leads to reactor shutdown by automatic insertion of the control rods by the Reactor Protection System (RPS). In cases where the events are coupled with a loss of electrical power, the ***controls rods will drop into the core by gravity***. The (Reactor Reserve Shutdown System (RRSS), can be manually actuated to achieve a diverse method of reactor shutdown, should control rod insertion not be accomplished. The two neutron absorbing systems are designed so that the insertion of either one of these systems ensures and maintains a subcritical state in all conditions. This includes, in particular, the reactivity due to the core cooling down to the coldest shutdown state combined with the ***xenon effect*** and the reactivity insertion due to the initiating event."	AREVA-006 (control rod sheaths)	In the AREVA PCDR, this item appears to be addressed in the design. All appropriate systems appear to be safety grade. However, ***there is no indication that re-criticality following xenon decay in an ATWS event has been specifically addressed in the AREVA PCDR.***

Table 1B (AREVA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		(Sec. 11.5.2.6, p. 182) – Reactivity Excursion – "The detection of reactivity insertion events leads to reactor shutdown by automatic insertion of the control rods by the RPS. ***A second system, the RSS can also achieve the function. RSS is manually actuated.*** The two neutron absorbing systems are designed so that the insertion of at least one these systems ensures and maintains subcriticality in any conditions. This includes in particular the reactivity due to core cooling down to the coldest shutdown state combined with the ***xenon effect*** and the reactivity insertion due to the initiating event. If the reactivity insertion and the reactivity insertion speed are limited and if the ***reactor is not shut down***, the situation is potentially controllable though power, fuel temperature and helium temperature should rise. In particular, as power increases, fuel temperature rises rapidly and, due to Doppler effect, results in negative reactivity feedback. Heating of the graphite moderator and most of all of the reflectors occurs more slowly, and, as a consequence, the associated temperature feedbacks come relatively later." (Sec. 11.3.2.1, p. 166) – "The negative temperature coefficient of reactivity and neutronically-inert helium coolant inherently ***stabilize the heat generation during any situation in acceptable conditions before the occurrence of significant xenon effect***. There is a large grace period before unacceptable consequences of xenon effect occur. This allows active operation of the control rod system or the reserve shutdown system. The control rod system and the reserve shutdown system are both capable to shut down the reactor during any condition including xenon effect occurrence. ***Any situation which could not be mitigated by these provisions is practically eliminated by design***. If neither control rods nor reserve shutdown material are inserted, the temperature coefficient of reactivity will tend to shut down the reactor from any power level following loss of forced convection cooling, such that the RCCS alone can safely cool the core beyond 24 hrs after the initial shutdown. If neither control rods nor reserve shutdown material are inserted, the temperature coefficient of reactivity will tend to shut down the reactor from any power level following loss of forced convection cooling, such that the RCCS alone can safely cool the core beyond 24 hrs after the initial shutdown."		
B-10	The uniqueness of configuration (tall, thin annular core) of current PMR and PBR designs and high operating temperatures require detailed reactor physics testing of the first unit as a function of core burnup, and of the start-ups of the second and perhaps third cycles. Attention should be paid to the instrumentation needs for these tests since neutron sensors must be both distributed and inter-calibrated to infer power distributions. Neutron detectors used in test measurements should also be sensitive enough to measure reactivity and changes in flux levels and distributions.	(Sec. 10.2.7, p. 154) – ***NGNP Safety Testing*** –"As a prototype for a possible fleet of Generation IV commercial nuclear power plants, the NGNP is expected to demonstrate the plant's passive and inherent safety features through a series of tests emulating various anticipated operational occurrences and design basis events. Unique operation and control strategies are envisioned such that key measures, based on safety, component tolerances, system efficiencies, etc. can be identified to define the operational envelope expected for future licensing activities related to a commercial plant while providing sufficient protection of the plant staff, the public, and the investment in the various NGNP systems, structures, and components." (Sec. 10.2.8, p. 154) – ***High Temperature Testing*** – "To characterize the	AREVA-001 thru AREVA-010 (materials testing) AREVA-023 thru AREVA-034 (system and component testing)	Needs for testing and instrumentation are recognized in the AREVA PCDR. In the PCDR, the tall core shape is actually used repeatedly by AREVA as a design feature that will tend to slow down the plant response to transients and accidents.

Table 1B (AREVA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		performance of the processes associated with the HPPP as a function of temperature, the NGNP will be expected to provide helium temperatures in the range of 1000 – 1100 °C. To sustain such temperatures, the NGNP will provide only the power demand required by the HPPP and shutdown helium circulation in the power generation loops. This testing mode could also facilitate the study of as yet-to-be determined future missions of the NGNP plant that may require alternative components, materials, and/or fluids." Sec. 17.6, p. 240 – ***Initial Startup Operations and Testing*** – "The initial startup and testing is critical to the overall schedule performance of any nuclear plant. The NGNP prototype facility is no exception. As the prototype demonstration plant for the new generation of high temperature gas cooled reactors the NGNP initial startup operation and testing schedule is developed to achieve the following: • Component testing and turn-over • System functional testing and turn-over • Initial approach to criticality • Zero power operation • Power ascension including grid connection • Normal plant safety system tests (AOO tests) • Special licensing performance tests (DBA tests) • Commercial operability endurance tests • Component dismantling and examination • Fuel examination The schedule provides four years for this phase of plant operations. The first two years (2017 and 2018) is dedicated to non-nuclear system testing and turn-over including the standard system turn-over from construction to operations. The second two years (2019 and 2020) includes initial plant criticality. During this phase all safety systems will be examined and tested and several special licensing related tests is planned. This phase of the plant operation includes component dismantling and inspection and fuel examination." (Sec. 19.2.3, p. 289) – ***Instrumentation*** – "NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy. For neutron flux detectors some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 ºC. Monitoring accident conditions		

Table 1B (AREVA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired.”		

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
C-1	*General Safety Analysis/Safety Document Needs* • Comprehensive description of the NGNP safety philosophy, a listing of the components involved, and the conditions under which these components are expected to perform their safety functions. • Explanation of how this philosophy meets the defense-in-depth approach and, in particular, answers to the following: ○ Will the components that perform a safety function (retain FPs) be classified as safety-related components, with the imposition of equipment qualification, in-service inspections, and/or Technical Specifications LCOs and SRs? ○ How will aging issues be addressed? If the safety function of a component is to retain FPs on its surface during adverse conditions, how can it be ensured that this function can be retained for long periods (decades), despite the possible presence of other long-term surface degradation mechanisms? ○ Will the surface state of a non-replaceable or difficult-to-replace component be reactivated by chemical action or cleaning during its service life? • A sound basis for the selection of the physical models and the data for these models must be justified. • The materials to be used and their sensitivity on the transport case must be identified. • Once the actual reactor design is available, the transport pathways that result from the accident conditions must be identified, along with the relevant models and data needed for the resulting calculations. • Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.	(Sec. 11, pp. 160-188) - The ***safety philosophy, listing of the components involved, and the conditions under which these components are expected to perform their safety functions*** are described in Sec. 11. (Sec. 11, pp. 160-188) – The attainment of ***defense-in-depth*** is addressed throughout Sec. 11, and as an individual topic in Sec. 11.3.4, p. 171. (Sec. 13.2, p. 211) – ***In Service Inspection*** – "1. The NGNP design shall provide access to the helium pressure boundary to permit in service inspection as required by appropriate sections of the ASME B&PV Code. 2. Where cost effective, the design of systems and components shall incorporate those features required to implement on-line in service inspection. If the unit or major component must be removed from service, design features shall be included to accomplish the inspection during the power unit allotted planned outage time. 3. Plant piping design shall minimize the need for snubbers and restraints and shall ensure inspectability. 4. Design documentation shall include plans and procedures for conducting in service inspection and shall identify equipment necessary to conduct the inspection. The equipment vendor shall furnish the ISI equipment not commercially available. 5. An in-service inspection program shall be developed and maintained throughout the design process. The program shall include anticipated durations and worker-hours, including health physics, for isolating the equipment/system, preparing for and performing the inspections, and returning the equipment/ system to service. Physical and/or computer models shall be used to assess inspectability. 6. The plant design shall include those facilities and features required to set up the in-core fuel handling equipment for periodic inspections, maintenance, testing, and demonstrations of integrated equipment operation. Such inspection, maintenance, testing, and demonstrations shall not interfere with core refueling operations nor have an adverse effect on plant operation." (Sec. 15.0, p. 220) – "In addition to these requirements and values, ***expected fuel performance characteristics*** will eventually be defined by required plant radionuclide release performance under operational and accident conditions to meet regulatory offsite and worker dose limits. The ***limiting radionuclide releases*** associated with the key accident analyses ***have not yet been determined***. As such, the NGNP plant specific required fuel performance characteristics have not yet been defined." (Sec. 19.2.4, p. 290) – ***Computer Codes and Methods Development and Validation –*** Included in this section are descriptions of R&D needs for computer codes addressing reactor system analysis, neutronics, thermal	AREVA-001 thru AREVA-010 (materials testing) AREVA-011 thru AREVA-022 (modeling codes) AREVA-023 thru AREVA-040 (system and component testing)	The AREVA PCDR has recognized most of the safety analysis/safety document needs detailed in this item, ***with the exception of the following:*** • ***Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.*** • ***The fuel database must be developed, as well as fuel-failure models and fuel material properties (both measurable and process controlled).***

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	• The fuel database must be developed, as well as fuel-failure models and fuel material properties (both measurable and process controlled).	hydraulics/pneumatics, ***fuel performance, fission product transport***, and structural mechanics. Fuel performance code R&D is addressed in section 19.2.4.4, p. 292, and fission product transport code R&D is addressed in section 19.2.4.5, p. 293.		
C-2	*Model Development and V&V* - Physical models and the supporting mathematical methods, addressing: • Nuclides of interest • Fission product release from the fuel • Diffusion, adsorption, and desorption in graphite and fuel matrix materials • Adsorption, desorption, and in-diffusion in reactor system metals • Chemical and physical forms of the FPs in the coolant • Tritium transport models • Aerosols and dusts that plate-out on reactor system components and their mobility • Fission product reactions with the confinement building materials • Reactions of the reactor system components and fission products with air or steam • Plume models that transport the released material beyond the reactor building • Determination of the safety function of each subsystem and the level of FPT attenuation required. • Determination of level of sensitivity to component uncertainties and how this reflects on the physical models. • Estimation of difficulty in obtaining the data and conducting the testing to support the safety case. • Scoping of how V&V can be performed.	See item A-7 for needs relating to air ingress phenomenon. See items B-6 and F-4 for needs relating to water ingress phenomenon. (Sec. 19.2.4, p. 290) – ***Computer Codes and Methods Development and Validation –*** Included in this section are descriptions of R&D needs for computer codes addressing reactor system analysis, neutronics, thermal hydraulics/pneumatics, fuel performance, fission product transport, and structural mechanics. (Sec. 19.2.4.4, p. 292) – ***Fuel Performance Models and Codes*** – "The R&D need for ***ATLAS*** development/modification is to improve the diffusion and the coatings corrosion modeling. For code qualification the heat-up experiments of irradiated fuel particles at relevant operating conditions (burnup, temperature, fluence) are required to anchor the developed code…qualification of ATLAS…includes two irradiation and heat up tests. In addition, there is an R&D need to develop the UCO models." (Sec. 19.2.4.5, p. 293) – ***Fission Product Transport*** – "The R&D needs of the ***FP Transport code*** include development of models for: • assessment of product activation in the primary circuit (in particular tritium and 14C), • radio-contamination distribution in the primary circuit, making distinction between circulating activity, plated out / deposited activity and purification system, for both normal operation and accidental situations, • radio-contamination releases outside the primary pressure boundary and • radio-contamination releases in the environment during accident scenarios. …It is also recommended to develop a mechanical analysis code for the NHS."	AREVA-011 thru AREVA-022 (modeling codes)	The AREVA PCDR has recognized the needs for most of the model development and V&V detailed in this item, ***with the following exceptions:*** • ***Fission product reactions with the confinement building materials*** • ***Determination of the safety function of each subsystem and the level of FPT attenuation required.*** • ***Determination of level of sensitivity to component uncertainties and how this reflects on the physical models.*** • ***Estimation of difficulty in obtaining the data and conducting the testing to support the safety case.***
C-3	*Materials/Component Data* - Relevant data on materials or components over the range of interest and data uncertainties (single effects testing), including the following: • Graphite transport property and air/steam erosion data specific to the design material. • Metal alloy data specific to the design material. • Data regarding transport properties sensitive to material surface conditions and chemical form of the fission	See item C-6 for extensive treatment of component and system testing. See item D-1 for materials data relating to metallic materials. See item E-1 for materials data relating to graphite materials. (Sec. 7.7.1, p. 105) - ***Helium Purification Train*** – "The primary functions of the	AREVA-001 thru AREVA-010 (materials testing) AREVA-023 thru AREVA-040 (system	The AREVA PCDR has recognized most of the needs for materials and component data detailed in this item, ***with the exception of the following:*** • ***Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product.***

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	product. • Data on helium impurities that will likely set the oxygen potential of the system, and the species to be included in an analysis. • Data associated with component aging: surface qualities of the reactor system components after many years of operation. • Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit. • Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. • If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties. • Data regarding turbine or power conversion components that may have to be decontaminated prior to maintenance (initial collection of FPs while in the reactor circuit; decontamination of components; new surface state of the component after decontamination). • Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior,	Purification Train are: • Removal of chemical and particulate contaminants from the primary coolant • Supply of purified helium to appropriate systems Since helium is used as the primary coolant, a ***helium purification system is required to provide the necessary degree of helium purity***. Oxidizing contaminants, in particular, may not exceed predetermined limits established in the specification. In detail, the helium purification system has the following functions: • Removal of particulate and gaseous contaminants from the primary coolant to maintain design values, in particular for H2O, CO, CO2, N2, H2, CH4 • Removal of tritium • Removal of other radioactive contaminants from the helium, especially before transfer to the purified gas store (Xe, Kr, Ar) • Start up purification of the primary system before initial start up and after inspections and maintenance • Purification of newly delivered helium" (Sec. 13.3.2, p. 213) – ***Power Conversion System*** (PCS) – "No precise information on the PCS maintainability has been produced during the pre-conceptual phase and this task will be performed in the next phases of the project. However, because of the NGNP indirect cycle design, radionuclides contamination of the PCS components are expected to low to nonexistent, therefore, PCS maintainability would be similar to the standard industry practice for ***non-contaminated turbomachinery and combined cycle components***." (Sec. 19.2.1, p. 283) – ***"(Fuel) Compact*** fabrication using thermosetting resins has been developed and demonstrated on a laboratory scale. However, ***currently-available materials have not been irradiated and performance under relevant environment has not been demonstrated***. …priority of this R&D need was evaluated as "High. …The other three compact R&D needs include: 1. Testing to confirm compact pressures and temperatures in order to minimize fuel damage. 2. Development of the heat treating process to ensure complete graphitization of the matrix material. 3. Perform irradiation tests on compacts to demonstrate performance for nominal and off-nominal operating conditions." (Sec. 19.2.1, p. 283) – "…3. Irradiation testing will be required to confirm that fuel performance matches performance from the laboratory/pilot facilities."	and component testing)	• ***Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit.*** • ***Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions.*** • ***If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties.*** • ***Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior.***

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		(Sec. 19.2.2, p. 284) – ***Materials Development and Qualification*** – "The materials R&D needs will focus on testing and qualification of the key materials commonly used in very high-temperature designs. The materials R&D will address the materials needed for the VHTR reactor, power conversion unit, intermediate heat exchanger (IHX), and associated balance of plant. (Sec. 19.2.2.1, p. 283) – ***Metallic Materials*** – "The materials R&D needs will focus on ***testing and qualification of the key materials*** commonly used in very high-temperature designs. The materials R&D will address the materials needed for the ***VHTR reactor, power conversion unit, intermediate heat exchanger (IHX), and associated balance of plant***."		
C-4	Reactor component and confinement/containment configuration and their relative roles in the safety case • Respective roles of the reactor circuit and containment or confinement system must be known before their modeling adequacy can be determined. • Estimate of source and budgeting of FP holdup among the fuel form, reactor circuit components, mobile elements such as dust, and the reactor building, as a means of focusing components to be emphasized in analysis. • Determination of transport pathway, goals for FP retention at each step in the pathway, local (accident) operating environment at each step of the pathway.	(Sec 9.2.1, p. 136) Reactor Building (***containment/confinement***) and its functions are described in section 9.2.1, starting on p. 136. (Sec. 10, pp. 147-160) - Nuclear System (***reactor circuit***) design and operation are described in section 10, pp. 147-160. (Sec. 17.3.3, p. 238) – "Of paramount importance to the project for the timely receipt of the LWA and CP milestones is the demonstration of the safety basis for the NGNP, albeit on a preliminary level at that time. In particular, and setting the project at risk, is the ***resolution of the containment issue***. Will a pressure retaining containment structure, similar to that used in LWRs, be required for the NGNP? Or, will a confinement arrangement as proposed for the NGNP Preconceptual design be acceptable? Feeding into this debate is the reliability of the particle fuel for the HTR's safety case and the need for containment are intimately linked through fuel reliability." (Sec. 19.2.4.5, p. 293) – Other Codes/***Fission Product (FP) Transport*** – "The R&D needs of the FP Transport code include development of models for: • assessment of ***product activation in the primary circuit*** (in particular tritium and 14C), • ***radio-contamination distribution in the primary circuit***, making distinction between circulating activity, plated out / deposited activity and purification system, for both normal operation and accidental situations, • radio-contamination releases ***outside the primary pressure boundary*** and • radio-contamination releases in the environment during accident scenarios. It is also recommended to develop a mechanical analysis code for the NHS."	AREVA-011 thru AREVA-022 (modeling codes)	The definition of roles has been included in the AREVA PCDR. The need for computer code development to understand fission product transport and distribution has been recognized in the AREVA PCDR.
C-5	Computational software or other methods for determining	See item C-2 for description of identified R&D efforts for computer codes/models.	AREVA-001 thru AREVA-	Needs for computer model development and testing have been recognized in the AREVA

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	the quantitative results • Data collection and proof that the selected model is adequate under all the normal and accident conditions of interest. Need to know that model envelops releases, and have reasonable proof that the model predicts an upper limit. • Need to have a description of the physical models and the reactor configuration, showing that the models are appropriate for the conditions of interest. • Need to have the data required for the models: single-effects data for each material and component acquired under individual testing, and integral data designed to show that the codes get the correct answer for a complete system under the conditions of interest.	See item C-6 for description of identified R&D effort for component and integral testing.	010 (materials testing) AREVA-011 thru AREVA-022 (modeling codes) AREVA-023 thru AREVA-040 (system and component testing)	PCDR. Reactor configuration is available in the PCDR.
C-6	Integral testing over a wide range of conditions to support the development of computational methods and the quantification of the data and associated uncertainties • Attempt to use existing data from past programs to the degree appropriate. • Planning of any in-pile loop program would require a complete description of the normal operating environment and of the accidents, along with any scaling factors. Extensive modeling will be necessary to design the loop and determine off-normal conditions that the loop can be expected to simulate. Model predictions (with the previously collected single-effects data) will need to be made.	(Sec. 10.2.7, p. 154) – ***NGNP Safety Testing*** –"As a prototype for a possible fleet of Generation IV commercial nuclear power plants, the NGNP is expected to demonstrate the plant's passive and inherent safety features through a series of tests emulating various anticipated operational occurrences and design basis events. Unique operation and control strategies are envisioned such that key measures, based on safety, component tolerances, system efficiencies, etc. can be identified to define the operational envelope expected for future licensing activities related to a commercial plant while providing sufficient protection of the plant staff, the public, and the investment in the various NGNP systems, structures, and components." (Sec. 10.2.8, p. 154) – ***High Temperature Testing*** – "To characterize the performance of the processes associated with the HPPP as a function of temperature, the NGNP will be expected to provide helium temperatures in the range of 1000 – 1100 °C. To sustain such temperatures, the NGNP will provide only the power demand required by the HPPP and shutdown helium circulation in the power generation loops. This testing mode could also facilitate the study of as yet-to-be determined future missions of the NGNP plant that may require alternative components, materials, and/or fluids." (Sec. 17.6, p. 240) – ***Initial Startup Operations and Testing*** – "The initial startup and testing is critical to the overall schedule performance of any nuclear plant. The NGNP prototype facility is no exception. As the prototype demonstration plant for the new generation of high temperature gas cooled reactors the NGNP initial startup operation and testing schedule is developed to achieve the following: • Component testing and turn-over	AREVA-001 thru AREVA-010 (materials testing) AREVA-023 thru AREVA-040 (system and component testing)	Needs for testing have been recognized in the AREVA PCDR.

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		• System functional testing and turn-over • Initial approach to criticality • Zero power operation • Power ascension including grid connection • Normal plant safety system tests (AOO tests) • Special licensing performance tests (DBA tests) • Commercial operability endurance tests • Component dismantling and examination • Fuel examination The schedule provides four years for this phase of plant operations. The first two years (2017 and 2018) is dedicated to non-nuclear system testing and turn-over including the standard system turn-over from construction to operations. The second two years (2019 and 2020) includes initial plant criticality. During this phase all safety systems will be examined and tested and several special licensing related tests is planned. This phase of the plant operation includes component dismantling and inspection and fuel examination." (Sec. 19.2, p. 281) – ***R&D Needs*** - "Fuel development and qualification, particularly ***irradiation and testing of compacts*** and mass production processes." (Sec. 19.2, p. 282) – ***R&D Needs*** – "Components testing. A large (10 MW) ***helium test loop*** is required for prototype tests of components." (Sec. 19.2.3, p. 287) – ***Circulators*** – "Circulators up to 4 MWe have already operated in HTR reactors. The test program is dedicated to component qualification during the commissioning phase rather than as an R&D task. Planned tests include: 1. Air tests of the impeller (at scale 0.2 to 0.4). 2. Helium tests of magnetic and catcher bearings. 3. Tests of the circulator shutoff valve. 4. Full scale integrated tests." (Sec. 19.2.3, p. 287) – ***IHXs*** – "The R&D inputs are based on two IHX concepts: Tubular IHX for 193 MWt power conversion and Plate IHX for 60 MWt loads for hydrogen plant loop. Small test facilities up to 1 MWt are available. Large test facilities of about 10 MWt will need to be designed and built." (Sec. 19.2.3, p. 287) – ***Tubular IHX*** – "The Tubular IHX design is based on the		

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		extrapolation of past German experience. NGNP requirements lead to high temperature operation with an innovative secondary fluid mixture of helium and nitrogen. Risk D-012 identifies feasibility concerns on module size, temperature level, corrosion/nitriding, manufacturing and assembly (which are not state of the art). Tubular IHX R&D needs include: 1. Tests to confirm fabrication feasibility (tube bending, tube welding, nozzles on hot header, ISIR and assembly, etc). 2. Corrosion and nitriding tests on base and coated materials in a representative environment. 3. Fabrication of representative IHX mock-ups from thermo-hydraulic and manufacturing point of views. 4. Testing in representative helium and helium-nitrogen environments is recommended. The current plan is to use a full scale mock-up for component qualification. The need for intermediate testing on sub-scale mock-ups is deemed unnecessary provided that manufacturing issues are sufficiently addressed." (Sec. 19.2.3, p. 288) – ***Plate IHX*** – "The feasibility of the plate IHX is a concern and a reduced lifetime is expected. Primary concerns are temperature level, corrosion, manufacturing, and thermal mechanical resistance…The plate IHX R&D needs include: 1. Development of visco-plastic model (material data-base to be completed). 2. Corrosion tests on base and coated materials in a representative environment. 3. Development of manufacturing techniques (fusion welding, diffusion bonding, brazing and forming). 4. Tests on representative IHX mock-ups from both thermo-hydraulic and manufacturing point of views (diffusion bonding, brazing, ISIR). A three step approach is recommended for component qualification, these are: 1. tests in air with sub-scale mock-ups, 2. tests in helium with sub-scale mock-ups (about 1 MWt test loop). These tests will provide a basis for recommendations on which type of concept should be used for the NGNP, and 3. final qualification on a full scale mock-up (at least for the channels and the plates) on a large test facility (around 10 MWt)." (Sec. 19.2.3, p. 288) – ***Isolation Valves*** – "A hot gas isolation valve was designed during the German HTR development program and tested in the KVK test facilities. The corresponding valve was designed for operation in helium at 900 °C and is very close to what is envisioned for the VHTR.		

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		The two qualification steps are: 1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc. 2. Full scale mock-up tests in a relevant helium-nitrogen environment. These tests should cover: 1. manufacturing parameters, 2. depressurization tests, 3. pressure loss, heat loss, support tube temperature tests in a relevant helium-nitrogen environment, 4. leak tightness tests of the valve, 5. closing and opening and 6. fatigue and creep-fatigue of specific areas." (Sec. 19.2.3, p. 288) – ***Fuel Handling System*** – "Currently the Fuel Server portion of the Fuel Handling System requires the most development. The remainder of the Fuel Handling System components, including the Fuel Elevator, Adaptor Plate and Fuel Handling Machine, has been demonstrated at the Fort St. Vrain reactor. In addition, the HTTR reactor utilized a similar set of components. Due to its "Low" priority, the Fuel Server system will be designed during the program. Testing of the Fuel Server system, beyond initial component testing, will be incorporated into the Fuel Handling System development testing program." (Sec. 19.2.3, p. 289) – ***Reactor Cavity Cooling System*** - "Use of an un-insulated reactor vessel coupled with a water-cooled panel heat exchanger as a core cooling mechanism for accident conditions has not been demonstrated. The basic components of the system are fairly common and well understood. Proper design and sizing of the system will require a demonstrated understanding of key heat transfer parameters for the vessel wall and panel surfaces. Determination of the heat transfer characteristics of the proposed surfaces for the reactor vessel and the panel heat exchanger will need to be accomplished. A large scale demonstration of the capability of the RCCS to remove reactor decay heat is recommended. ..Currently there is facility available at ANL which can accommodate a large scale demonstration of the RCCS." (Sec. 19.2.3, p. 289) – ***Hot Gas Duct*** – "The reference design for the primary and secondary hot gas duct is the Vee-shaped metallic concept. This design appears to be compatible with the core expected outlet temperature, subject to demonstrating that no significant hot streaks occur. The ceramic concept will be retained as a fall back option. The hot gas duct qualification should be performed in three steps: 1. Elementary tests to characterize the fiber conditions, assembly techniques,		

Table 1C (AREVA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
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		spacers, etc. 2. Sub-scale mock-up tests, about 1 MWt in helium if possible, to validate fiber specification and ceramic spacer specification. 3. Full scale mock-up tests, around 10 MWt. These tests should at least cover 1. depressurization tests, 2. pressure loss, heat loss, temperature of the support tube (in helium), 3. leak tightness tests of connections 4. fatigue and creep-fatigue tests (e.g., bellows, Vee-shape spacers, etc). …In the first stages of the design, tests should cover both the metallic and ceramic concepts." (Sec. 19.2.3, p. 289) – ***Instrumentation*** – "NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy. For neutron flux detectors some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired."		

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
D-1	*Physical Materials Data* - Requirements for physical aspects to be included in modeling high-temperature metallic components: • Inelastic materials behavior for materials, times, and temperatures for very high temperature structures (e.g., creep, fatigue, creep-fatigue). • Adequacy and applicability of current ASME Code allowables with respect to service times and temperatures for operational stresses. • Adequacy and applicability of current state of high-temperature design methodology (e.g., constitutive models, complex loading, failure criteria, flaw assessment methods). • Effects of product form and section thickness. • Joining methods including welding, diffusion bonding, and issues associated with dissimilar materials in structural components. • Effects of irradiation on materials strength, ductility, and toughness. • Degradation mechanisms and inspectability. • Oxidation, carburization, decarburization, and nitriding of metallic components in impure helium and helium-nitrogen. • Micro-structural stability during long-term aging in environment. • Effects of short and long term on mechanical properties (e.g., tensile, fatigue, creep, creep-fatigue, ductility, toughness). • High-velocity erosion/corrosion. • Rapid oxidation of graphite and carbon-carbon composites during air-ingress accidents. • Compatibility with heat-transfer media and reactants for hydrogen generation. • Development and stability of surface layers on RPV and core barrel affecting emissivity.	See item A-7 for data relating to air ingress phenomena. (Sec. 6.2.4.2.2, p. 75) – ***Impact and Toughness Properties*** - "Impact tests have been performed at - 20°C and 0°C on products purchased in Europe. Impact tests were also performed at other temperatures in order to determine the Charpy V transition curves. For the Charpy V at -20°C, the target in Europe was 40 J minimum for the average value of the three specimens and 28 J minimum for individual test results. These values were met for rolled and forged plates with thicknesses from 20 to 200 mm." (Sec. 6.2.4.2.3, p. 75) - ***Creep*** - "Creep test programs underway are mainly dedicated to defining the negligible creep domain. They are aimed at improving the knowledge of creep properties at moderate temperatures (< 500°C), including the effect of the post weld heat treatment. Negligible creep is also a topic which has been studied in the context of the ASME/DOE Gen IV material project (AREVA NP as the lead contractor). This work also covers creep-fatigue of mod 9Cr1Mo." (Sec. 6.2.4.2.4, p. 75) – ***Effect of Aging*** - "From available data on modified 9Cr1Mo, it can be expected that there should not be any significant aging effect below 480°C. Nevertheless thermal treatments with increasing duration up to more than 25,000 hrs at 450°C, 475°C and 500°C have been started to confirm this conclusion. Base material, heat affected zone, and weld metal samples are included in the test program. The present status after 10,000 hrs at 500°C indicates no shift in the ductile brittle transition temperature (DBTT)." (Sec. 6.2.4.2.5, p. 75) – ***Effect of Irradiation*** - "Irradiations have been carried at Joint Research Center in Petten on both base metal and weld metal (150 mm thick welded joint). No significant shifts in mechanical properties and ductile-brittle transition temperatures have been observed for the expected end-of-life fluence of the reactor pressure vessel." (Sec. 6.2.4.2.6, p. 75) – ***Corrosion in Helium Environment*** - "For temperatures below 450°C, expected carburization in impure helium environment will be a very slow process affecting only the surface layers of the vessel wall. For temperatures expected during off-normal situations (about 550°C), no problems are expected due to their short durations. A test program is however necessary to confirm the performance of mod 9Cr1Mo in representative HTR conditions." (Sec. 6.2.4.2.7, p. 76) – ***Weldability*** - "A significant R&D program has been launched	AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification) AREVA-013 (improvement/ development of "other" codes)	The AREVA PCDR has captured most of the needs detailed in this item, ***with the following exceptions:*** • ***Micro-structural stability during long-term aging in environment.*** • ***High-velocity erosion/corrosion.*** • ***Compatibility with heat-transfer media and reactants for hydrogen generation.***

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		to demonstrate the weldability of heavy section products. Initial tests carried out with GTAW (Gas Tungsten Arc Welding) process had shown hot cracking. It was shown later on that proper selection of filler material could eliminate this problem. The welding program covered the main welding processes likely to be used, namely SAW (Submerged Arc Welding), GTAW and SMAW (Shielded Metal Arc Welding) processes. The available results are very encouraging, showing acceptable mechanical properties and no cracks. Optimization is still necessary to achieve the required impact test values for the range of post weld heat treatment temperatures investigated, in particular for SAW and SMAW processes. Further activities are envisioned on GMAW (Gas Metal Arc Welding or MIG) process, which uses a filler material similar to that used for GTAW and is suitable for automatic on-site welding in horizontal position, with a larger deposit rate compared to GTAW." (Sec. 6.2.4.2.8, p. 76) – ***Emissivity*** – "Understanding of radiative heat transfer is of prime importance in the evaluation of the temperatures of the fuel, reactor vessel, and metallic internals, particularly during conduction cooldown situations. Measurements have been carried out to define emissivity values, not only for the Reactor Pressure Vessel material but also for the metallic and graphite internals. Tests have been carried out for the range of temperatures covering normal to off-normal situations and taking into account specimens with surface conditions representative of the RPV at the beginning and end of life." (Sec. 6.2.4.2.9, p. 76) – ***Codes and Standards*** – "Mod 9Cr1Mo is presently covered by ASME Section III Subsection NB for temperatures below 700°F (371°C). Rules have been introduced in Subsection NH (2004 edition) to include mod 9Cr1Mo for higher temperatures. Rules are presently limited to plates and small size forgings, and revision is necessary to extend the rules to heavy section plates and forgings and extend the stress allowables to cover a 60-year design life. Other necessary Code improvements concern the definition of negligible creep conditions and the improvement of creep-fatigue design rules." (Sec. 6.4, p. 90) "…***corrosion and nitriding*** are a concern at such high temperatures and it is recommended investigating the possibility of protecting the hottest parts of the IHX with a coating. Further R&D will be also required to confirm the material behavior at such temperatures and provide necessary information in the context the material and component qualification program." See item A-7 for information on current state of and uncertainties associated with ***air ingress*** phenomenon. (Sec. 13.1.14, p. 210) – "The NHS module shall be designed to allow all components		

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		within the helium pressure boundary to be removed and reinstalled to ***make possible inspection***, repair and replacement. A trade study to determine the method of removal and replacement of components within the primary pressure boundary, based on the degree of difficulty, time and cost and the projected probability of occurrence shall be completed and documented by completion of preliminary design." (Sec. 19.2, p. 281) – ***R&D Needs*** – "Materials development and qualification. This covers certain ***high-temperature steels,*** composites, and graphite selection/qualification." (Sec. 19.2, p. 282) – ***R&D Needs*** – "Power Conversion System. This work covers ***nitriding*** tests and improvement of blade performance." (Sec. 19.2.2.1, p. 284) – **Metallic Materials** – "For Mod 9Cr1Mo steel the R&D needs of "High Priority" include mechanical ***properties on heavy section products (base and weld metal), effects of aging and radiation, corrosion in helium environment, weldability, emissivity, negligible creep conditions and creep fatigue.*** A specific test program on representative ***plates and forgings (including welded joints)*** will be required for component qualification." (Sec. 19.2.2.1, p. 284) – ***Metallic Materials*** – "Mod 9Cr1Mo is covered by the ASME code up to 371ºC in Subsection NB and beyond 371ºC in Subsection NH. Subsection NH does not currently cover heavy section products and needs to be updated to cover specific aspects of Mod 9Cr1Mo. Actions have already been launched in the context of the DOE/ASME Gen IV material project to provide basis for code development. R&D efforts to support this codification should be continued. In view of past experience in gas cooled reactor, alloy 800H is a prime candidate for metallic internals operating in cold helium. Moreover, efforts are in progress to extend its coverage up to 850°C in ASME III-NH. …For 800H alloy the R&D needs include: 1. Emissivity measurement under likely representative state of surface (as machined and oxidized after machining) and 2. Corrosion behavior under representative primary helium environment. For extension of alloy 800H coverage in ASME III-NH the following items are needed: 1. Long term tests at temperature higher than 760°C, 2. Tensile tests at temperature higher than 870°C and 3. Extension to cover 60 years lifetime. Two available nickel-based super alloys (In617 and Haynes 230) have been selected as structural materials for the IHX: In617 (NiCr22Co12Mo), which has been widely studied in the early 80's for HTR application and Haynes 230 (NiCr22W14), which		

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		has been developed more recently but it exhibits better corrosion resistance. An extensive research program has been launched in France within the framework of the ANTARES program to evaluate mechanical properties, thermal stability, and corrosion resistance in the temperature range of 700 °C to 1000 °C for extended periods." (Sec. 19.2.4, p. 293) – ***Structural Mechanics*** – "The main tools for structural analysis exist, but specific modeling and correlations for NGNP geometry and materials have to be developed. This work includes: 1) incorporation of ***constitutive laws*** for materials and developing numerical models 2) seismic modeling of a block-type core 3) fluid structure interaction and flow-induced-vibration methodology, and 4) leak-before-break methodology." Sec. 19.2.5, p. 293) – Power Conversion System – "***Nitriding*** of metals will occur when exposed to ***hot nitrogen***. This nitriding process tends to embrittle metals which could lead to failures of turbine blades and pressure boundaries such as boiler tubes, gas shells, etc. The need to experimentally determine the degree of nitriding that occurs in potential PCS materials, and to quantify the effects of temperature on nitriding, has been identified. This R&D need is not only for turbomachinery, but also for IHX (Tube) and Brayton cycle gas duct."		
D-2	*Physical Materials Data (Composites)* - Requirements for physical aspects to be included in modeling high-temperature structural composites, such as carbon-carbon or silicon carbide–silicon carbide: • Effects of composite component selection and infiltration method. • Effects of architecture and weave. • Materials properties up to and including very high temperatures (e.g., strength, fracture, creep, corrosion, thermal shock resistance). • Effects of irradiation on materials strength and dimensional stability. • Fabrication scaling processes. • Adequacy and validation of design methods. • Degradation mechanisms and inspectability.	(Sec. 13.1.14, p. 210) – "The NHS module shall be designed to allow all components within the helium pressure boundary to be removed and reinstalled to ***make possible inspection***, repair and replacement. A trade study to determine the method of removal and replacement of components within the primary pressure boundary, based on the degree of difficulty, time and cost and the projected probability of occurrence shall be completed and documented by completion of preliminary design." (Sec. 19.2, p. 281) – ***R&D Needs*** – "Materials development and qualification. This covers certain high-temperature steels, ***composites***, and graphite selection/qualification." (Sec. 19.2.2.2, p. 285) - Ceramics – "The use of ***composites*** is driven by their high resistance to high or very high temperatures. An R&D program has been launched in the frame of Antares to explore the possible use of such materials inside the primary circuit. Thermal insulation, using composite materials, will be needed to provide thermal protection of metallic components which would otherwise be subjected to helium at very high temperatures….The R&D needs for applied composite materials (C/C or C/SiC composites) emphasizes qualification of ***material properties*** such as:	AREVA-004 thru AREVA-007 (testing/verification of ceramics, including composites)	In general, the AREVA PCDR recognizes the needs for greater understanding of materials characteristics and behavior of composites. ***However, there is no indication that the following topics from this item are considered:*** • ***Effects of composite component selection and infiltration method.*** • ***Effects of architecture and weave.*** • ***Fabrication scaling processes.*** • ***Adequacy and validation of design methods.***

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		1. thermal-physical properties (thermal conductivity (K), coefficient of thermal expansion (CTE), heat capacity (Cp)), 2. mechanical properties including multiaxial strength, 3. fracture properties, 4. fatigue properties and 5. behavior in an oxidizing atmosphere and oxidation effects on properties. In addition, for thermal insulation, ceramic ***materials qualification*** should be for: 1. thermal-physical properties (K, CTE, Cp) and 2. behavior under oxidation. No ***control rods*** made of composites were used for past HTRs, or for other reactor concepts….Other composites such as C/SiC are also envisioned. An R&D program has been launched in the frame of Antares to explore the possibility of employing such composites for the control rods. SiC/SiC composites are not considered mature enough to meet the NGNP 2018 schedule. Additional tests for control rod ceramic materials include: 1. irradiation effects on properties including ***irradiation induced dimensional change and irradiation induced creep*** and 2. tribology."		
D-3	Compromise of RPV surface emissivity due to loss of desired surface layer properties. Compromise of emissivities of in-vessel surfaces.	(Sec. 6.2.4.2.8, p. 76) – ***Emissivity*** – "Understanding of radiative heat transfer is of prime importance in the evaluation of the temperatures of the fuel, reactor vessel, and metallic internals, particularly during conduction cooldown situations. Measurements have been carried out to define emissivity values, not only for the Reactor Pressure Vessel material but also for the metallic and graphite internals. ***Tests have been carried out*** for the range of temperatures covering normal to off-normal situations and taking into account specimens with surface conditions representative of the RPV at the beginning and end of life." (Sec. 19.2.2.1, p. 284) – Metallic Materials – "For Mod 9Cr1Mo steel the R&D needs, of "High Priority," include mechanical properties on heavy section products (base and weld metal), effects of aging and radiation, corrosion in helium environment, weldability, ***emissivity***, negligible creep conditions and creep fatigue. A specific test program on representative plates and forgings (including welded joints) will be required for component qualification. ..For 800H alloy the R&D needs include: 1. ***Emissivity*** measurement under likely representative state of surface (as machined and oxidized after machining)…" (Sec. 19.2.2.3, p. 286) – Graphite Materials – "Graphite, an essential structural material for the VHTR, will operate under significant irradiation conditions and requires a characterization in the range of expected temperatures. Nuclear grade graphite was used in past HTRs programs, amassing a substantial database. These grades are no longer available. An R&D program has been launched within Antares	AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification) AREVA-011 thru AREVA-022 (modeling codes)	Need for greater understanding of the surface emissivity material characteristic has been recognized in the AREVA PCDR.

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		program to select the best candidates among the new available grades or to request the development of a new grade, and to acquire design data…Nuclear graded structural graphite (PCEA, NBG17 and/or NBG18) qualification includes: 1. thermal-physical properties (K, CTE, Cp, ***emissivity***)…" (Sec. 19.2.4, p. 290) – ***Computer Codes and Methods Development and Validation –*** Included in this section are descriptions of R&D needs for computer codes addressing reactor system analysis, neutronics, thermal hydraulics/pneumatics, fuel performance, fission product transport, and structural mechanics.		
D-4	Effects on insulation • Aging fatigue and environmental degradation of insulation materials (debris plugging). • Environmental and irradiation degradation/thermal instability of fibrous insulation	***There is no indication that this item has been specifically addressed.*** However, effects of aging are addressed in section 6.2.4.2.4, and R&D that will address aging of metallic materials is addressed in section 19.2.2.1.	(None)	***There is no indication that this item has been specifically addressed. The AREVA PCDR has recognized the need for R&D regarding aging of materials, but has not addressed these specific issues on insulation.***
D-5	Primary boundary failures in compact IHX (roles of design methods, manufacturing controls, inspection/testing).	(Sec. 11.5.2.5, p. 180) - ***IHX Failure*** – Sec. 11.5.2.5 indicates that the key factors against radiological release from the IHX will be low fuel failure rate during normal operation (resulting in low activity in the primary circuit), high purity helium provided by the Helium Purification System, and slow and limited evolution of fuel temperature during any accident (resulting in limited fuel failures during accidents).	AREVA-003 (testing and codification of IHX materials) AREVA-024 and AREVA-025 (testing of IHX)	Indication in the AREVA PCDR is that this item is being addressed in the design. Also, AREVA has addressed improvement of design methods in section 19.2.4 (beginning on p. 290) and proposed a main component fabrication strategy in section 21.1.27 (p. 320).
D-6	Control rod insertion failures (role of structural design methods for composites).	(Sec. 6.5, p. 90) – "The selection of composite materials as ***control rod*** cladding requires significant R&D actions to qualify this component and facilitate its approval by the Regulator."	AREVA-006 (control rod sheath materials) AREVA-030 (testing of neutron control drive mechanism)	Need for control rod material qualification is recognized in the AREVA PCDR.
D-7	Irradiation induced creep of in-vessel metallic structures.	(Sec. 19.2.2.1, P. 284) – Metallic Materials – "For Mod 9Cr1Mo steel the R&D needs, of "High Priority," include mechanical properties on heavy section products (base and weld metal), effects of aging and radiation, corrosion in helium environment,	AREVA-001 thru AREVA-003, and	The need to better understand the phenomenon of creep has been recognized in the AREVA PCDR.

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		weldability, emissivity, negligible ***creep conditions*** and ***creep fatigue***. A specific test program on representative plates and forgings (including welded joints) will be required for component qualification… In617 and Haynes 230 R&D needs, of "Medium Priority," have been identified to address the following issues: 1. baseline mechanical property data, including ***creep-fatigue*** data…"	AREVA-010 (metallic materials testing and codification)	
D-8	Core radial restraint failure (role of structural design and fabrication for composites).	***There is no indication that this item has been specifically addressed.***	AREVA-007 (testing and codification of composites)	***There is no indication that this item has been specifically addressed in AREVA's PCDR.***
D-9	Isolation and other valve failures (self-welding, galling, seizing)	(Sec. 6.5, p. 91) – ***Hot Isolation Valves*** – "Such type of component has already been qualified in the context of the former German HTR program but it will need to be checked that the environment proposed on the secondary side will not justify significant design adaptations." (Sec. 19.2.3, p. 288) – ***Isolation Valves*** – "A hot gas isolation valve was designed during the German HTR development program and tested in the KVK test facilities. The corresponding valve was designed for operation in helium at 900 °C and is very close to what is envisioned for the VHTR. The two qualification steps are: 1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc. 2. Full scale mock-up tests in a relevant helium-nitrogen environment. These tests should cover: 1. manufacturing parameters, 2. depressurization tests, 3. pressure loss, heat loss, support tube temperature tests in a relevant helium-nitrogen environment, 4. leak tightness tests of the valve, 5. closing and opening and 6. fatigue and creep-fatigue of specific areas."	AREVA-026 (testing of isolation valves)	Need for isolation valve qualification is recognized in the AREVA PCDR.
D-10	Initiate development of the data and models needed by ASME Boiler and Pressure Vessel (B&PV) Code Subcommittees to formulate time-dependent failure criteria that will ensure adequate life and safety for metallic materials in the NGNP. These include obtaining the data necessary to develop experimentally based constitutive models for the NGNP construction materials, which are the foundation of the inelastic design analyses specifically	See item D-1 for R&D description of ***ASME Code*** efforts and development of ***structural mechanics codes***.	AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification)	Needs for ASME code development and supporting structural mechanics models have been recognized in the AREVA PCDR.

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	required by ASME B&PV Sect. III Division I Subsection NH.		AREVA-013 (improvement/ development of "other" codes)	
D-11	Safety assessments dependent on time-dependent flaw growth and the resulting leak rates from postulated pressure-boundary breaks will require a flaw assessment procedure capable of reliably predicting crack-induced failures, as well as the size and growth of the resulting opening in the pressure boundary.	See item D-1 for description of R&D efforts for ***structural mechanics codes***, including ***leak-before-break methodology***.	AREVA-013 (improvement/ development of "other" codes)	Need for structural mechanics models, including flaw assessment, have been recognized in the AREVA PCDR.
D-12	Materials data and extrapolation procedures must be developed and guidance provided to ensure that allowable operation period and range of stress and temperature for materials of construction are extended to meet the proposed operating temperatures and lifetimes. Creep-fatigue rules are an area of particular concern for the materials and temperatures of interest and must be updated and validated. (example concern: RPV long-term thermal aging)	See item D-1 information on ***materials properties, ASME Code efforts***, and development of ***structural mechanics codes.*** (Sec. 6.4, p. 89) – "Based on past experience in Germany (full scale mock up tested in the KVK helium loop) and Japan (HTTR), a high temperature tubular IHX is deemed feasible at the following conditions: • Helium/helium heat exchanger • Effectiveness 90 % • T = 850°C and with some limited periods in operation up to 950°C • Limited pressure difference in operation < 3 bars • Lifetime 20 to 30 years • IHX module power around 150 MWth. The proposed 193 MWth tubular IHXs will require an increase of the number and length of tubes which ***should be achievable through design improvements***. The extension to 900°C design temperature should be obtained by a reduction of ***design life*** to 20 years." (Sec. 6.4, p. 90) – "For the ***compact IHX*** proposed for the heat transport to H2 plant, significant R&D and design work is still required to obtain a design able to operate at 900°C (or above). Operating conditions are however less demanding (reduced pressure transients and He environment on the secondary side) and it is currently considered that such a concept can be implemented, subject to limiting the design life to 5 years. This life reduction is acceptable due to limited cost impact on the overall plant and due to the fact that the availability required on the H2 plant side should not be as large as that required on the Power Conversion side."	AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification) AREVA-013 (improvement/ development of "other" codes)	Needs for greater understanding of materials characteristics, related ASME Code efforts, scale-up of significant metal components, and development of structural mechanics codes have been recognized in the AREVA PCDR.
D-13	Since IHX sections must operate at the full exit temperature of the reactor, effort should be initiated to obtain data	(Sec. 7.7.1, p. 105) - ***Helium Purification Train*** – "The primary functions of the	AREVA-001 thru AREVA-	The need for high purity helium is addressed

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	supporting the determination of the metallurgical stability and environmental resistance of IHX materials in anticipated impure helium coolant environments for the lifetimes anticipated.	Purification Train are: • Removal of chemical and particulate contaminants from the primary coolant • Supply of purified helium to appropriate systems Since helium is used as the primary coolant, a ***helium purification system is required to provide the necessary degree of helium purity***. Oxidizing contaminants, in particular, may not exceed predetermined limits established in the specification. In detail, the helium purification system has the following functions: • Removal of particulate and gaseous contaminants from the primary coolant to maintain design values, in particular for H2O, CO, CO2, N2, H2, CH4 • Removal of tritium • Removal of other radioactive contaminants from the helium, especially before transfer to the purified gas store (Xe, Kr, Ar) • Start up purification of the primary system before initial start up and after inspections and maintenance • Purification of newly delivered helium"	003, and AREVA-010 (metallic materials testing and codification) AREVA-027 (helium purification system)	in the AREVA PCDR.
D-14	Work should be initiated to quantify crack initiation and propagation in the IHX due to creep, creep-fatigue, and aging. These materials-related phenomena related to the IHX were identified for potentially contributing to FP release at the site boundary.	See item C-6 for component testing efforts and item D-1 for metallic materials efforts.	AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification) AREVA-024 and AREVA-025 (IHX testing)	Needs for greater understanding of materials characteristics and associated component testing have been recognized in the AREVA PCDR.
D-15	Specific issues must be addressed for RPVs that are too large for shop fabrication and transportation. Validated procedures for on-site welding, PWHT, and inspections must be developed for the materials of construction. For vessels using materials other than those typical of LWR construction to enable operation at higher temperatures, confirmation of their fabricability (especially, effects of forging size and weldability) and data on their irradiation resistance is needed. Three materials-related phenomena related to the RPV fabrication and operation were identified for potentially contributing to FP release at the site boundary, particularly for 9Cr–1 Mo–V steels capable of higher-temperature	Sec. 6.2.4.2.1 (p. 75) "Significant efforts have been made to perform characterizations on representative ***heavy section products***. Metallographic evaluations performed so far indicate a good homogeneity throughout the thickness. R&D actions presently underway are based on two products recently purchased: • a forged plate, 200 mm thick, supplied by Japan Steel Work • a rolled plate, 140 mm thick, supplied by Industeel. Tensile tests performed on the 200 mm forged plate in the temperature range 20°C-600°C indicated that yield strengths are higher than the ASME minimum values. Concerning ultimate tensile strength, the data obtained from the 200 mm forged plate are slightly lower than ASME values but further evaluation should be performed to clarify if the difference should be attributed to a product effect or to the definition of	AREVA-001 (testing and codification for vessel materials)	Needs for resolution of issues associated with heavy sections, materials characteristics and feasibility of using the 9Cr-1Mo alloy, and fabrication of large vessels have been recognized in the AREVA PCDR.

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	operation: crack initiation and subcritical crack growth, process control to avoid material degradation during field fabrication, and property control in heavy sections.	ultimate tensile strength (as a reminder, ASME design values should not be considered as true minima). It is also to be mentioned that actions are underway in the context of the ASME/DOE Gen IV material project (actions led by the University of Dayton Research Institute). Activities concern the update of stress allowables for mod 9 Cr1Mo, covering the effect of product form and extension of stress allowables to a 60-year design life." (Sec. 6.4, p. 90) – "Welding of mod 9Cr-1Mo is also an issue but weldability actions carried out by AREVA in the past few years indicate that welding of heavy section products should be fully achievable (even though optimization of welding products and welding parameters is still required)." (Sec. 21.1.7, p. 317) – ***Confirm Selection of 9Cr-1Mo RPV Material*** – "Modified 9Cr-1Mo steel provides significant performance advantages for the reactor pressure vessel material including high temperature capability and improved irradiation resistance compared to SA508. However, 9Cr-1Mo is not an established reactor vessel material, and its use will require development in terms of procurement, fabrication, qualification, and code acceptance. Therefore, a more detailed study should be planned and implemented to amplify, refine, and elaborate the factors in the assessment and selection of 9Cr-1Mo steel for the primary pressure vessels (e.g., forging, fabrication, procurement, codification). This study must distinguish perception from reality regarding the fabrication difficulties associated with 9Cr-1Mo. Attention must be given to the relative schedule risks associated with 9Cr-1Mo compared to SA508 for HTR applications against the relative associated performance advantages." (Sec. 21.1.27, p. 320) - **Main Component Fabrication Strategy – "**In parallel with the INL site heavy component transportation issues study, a fabrication strategy for main components should be developed. This study should include identification of potential suppliers, assessments of on-site versus off-site fabrication issues, and comparison of relative costs."		
D-16	For high-temperature metals technology, there is a need for analytical models, in particular for developing time-dependent design criteria for complex structures, along with verification by structural testing. ASME Code-approved simplified methods have not yet been proven and are not permitted for compact IHX components. Analytical modeling of carbon-carbon composite behavior would be useful in developing approved methods for designing, proof testing, model standard testing, validation tests, and probabilistic methods of design. Scalability and fabrication issues must be addressed, including large-scale structures (meters in	(Sec. 6.4, p. 90) – "For the ***compact IHX*** proposed for the heat transport to H2 plant, significant R&D and design work is still required to obtain a design able to operate at 900°C (or above). Operating conditions are however less demanding (reduced pressure transients and He environment on the secondary side) and it is currently considered that such a concept can be implemented, subject to limiting the design life to 5 years. This life reduction is acceptable due to limited cost impact on the overall plant and due to the fact that the availability required on the H2 plant side should not be as large as that required on the Power Conversion side." See item D-15 for information on ***main component fabrication strategy***.	AREVA-001 thru AREVA-003, and AREVA-010 (metallic materials testing and codification) AREVA-004	Needs for improved high temperature metals technology, ASME code-approved materials designations, structural mechanics codes describing materials behavior and characteristics, and resolution of large scale fabrication strategies have been recognized in the AREVA PCDR.

Table 1D (AREVA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	diameter), as well as smaller structures.	See item D-1 for R&D description of ***ASME Code*** efforts, and needs for ***structural mechanics codes.***	thru AREVA-007 (testing/verification of ceramics, including composites)	

Table 1E (AREVA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
E-1	Lack of confirmatory data for the grades of graphite selected by potential NGNP vendors. This situation has occurred because: • Graphite grades used in prior HTGRs are no longer available, and thus development of new grades has been required. • Increased temperature of the NGNP compared to prior graphite-moderated reactors. • In the case of the PBR, the larger neutron dose that the core components will experience compared to that of previous HTGRs licensed in the United States.	(Sec. 19.2, p. 281) – ***R&D Needs*** – "Materials development and qualification. This covers certain high-temperature steels, composites, and ***graphite*** selection/qualification." (Sec. 19.2.2.3, p. 286) - ***Graphite Materials*** – "Graphite, an essential structural material for the VHTR, will operate under significant irradiation conditions and requires a characterization in the range of expected temperatures. Nuclear grade graphite was used in past HTRs programs, amassing a substantial database. ***These grades are no longer available***. An R&D program has been launched within Antares program to select the best candidates among the new available grades or to request the development of a new grade, and to acquire design data. Nuclear graded structural graphite (PCEA, NBG17 and/or NBG18) qualification includes: 1. thermal-physical properties (K, CTE, Cp, emissivity), 2. mechanical properties including multiaxial strength, 3. fracture properties, 4. fatigue properties, 5. irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep, 6. behavior under oxidized atmosphere including oxidation effects on properties and 7. tribology. Due to schedule limits, it is recommended that graphite R&D be performed in two phases: preliminary and detailed… Development of ASME and ASTM codes and standards for graphite is essential for timely application graphite for NGNP reactor." The portion of this item that is specific to the PBR is not applicable to the PMR.	AREVA-008 thru AREVA-010 (testing and codification of graphite materials)	Needs for development of updated, code-approved graphite materials designations have been recognized in the AREVA PCDR, and some of the R&D has been performed.
E-2	Lack of consensus codes and standards. Efforts are under way through the ASME to develop a consensus design code for graphite core components, but to date a useable code has not been approved. ASTM test standards exist for many of the physical properties of concern to the reactor designer, but further work is required, especially in the area of small (irradiation) specimen test methods.	(Sec. 6.1.2.3, p. 62) – Design Code – "Rules for nonmetallic materials are presently under preparation in the context of the ***ASME Subgroup on Graphite Core Components***." (Sec. 6.4, p. 90) – "There is no feasibility issue associated to the mechanical design of graphite core components. Feasibility lies more on the availability of material properties of the new grades envisioned for VHTR design (in particular properties of irradiated material) and on the ***availability of design rules approved by ASME Code Committee and by the Regulator***."	AREVA-008 thru AREVA-010 (testing and codification of graphite materials)	Needs for development of approved ASME codes for graphite have been recognized in the AREVA PCDR.
E-3	Theoretical models for the effects of neutron damage on the properties of graphite have been developed, however, these models need modification for the new graphites and will	See item E-1 for description of graphite R&D efforts. (Sec. 19.2.4, p. 293) – ***Structural Mechanics*** – "The main tools for structural	AREVA-008 thru AREVA-010 (testing	Needs for greater definition of materials characteristics and development of structural mechanics models are recognized

Table 1E (AREVA) – GRAPHITE - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
	need to be extended to higher temperatures and/or higher neutron doses. V&V of theoretical models will require generation of experimental data on the effect of neutron irradiation on properties.	analysis exist, but specific modeling and correlations for NGNP geometry and materials have to be developed. This work includes: 1) incorporation of ***constitutive laws*** for materials and developing numerical models 2) seismic modeling of a block-type core 3) fluid structure interaction and flow-induced-vibration methodology, and 4) leak-before-break methodology."	and codification of graphite materials) AREVA-013 (improvement/ development of "other" codes)	in the AREVA PCDR.
E-4	Uncertainties in the temperature and dose received by a component; the severity of temperature and dose gradients in a component; the rate of dimensional change in the specific graphite used in a given design; the extent to which stresses are relieved by irradiation-induced creep; and the extent of changes in key physical properties such as elastic moduli, thermal conductivity, coefficient of thermal expansion, compound to make the prediction of component stress levels, and hence decisions regarding component lifetime and replacement schedules, very imprecise.	See Item E-1 for description of graphite R&D efforts. See item E-3 regarding efforts for a structural mechanics code/model.	AREVA-008 thru AREVA-010 (testing and codification of graphite materials) AREVA-013 (improvement/ development of "other" codes)	Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the AREVA PCDR.
E-5	Whole-core models are required that can predict the stress states of graphite components within the core. Such models should be capable of taking inputs such as temperature and neutron dose and calculating the dimensional change, creep, thermal conductivity, etc., from established theoretical models. Reliable stress-state predictions as a function of reactor life would enable reactor operators and regulators to provide NDE guidance and make decisions regarding inspection intervals and core block replacement.	See item E-3 regarding efforts for a structural mechanics code/model. (Sec. 19.2.4.1, p. 291) – ***Reactor System Analysis Code/MANTA*** – "Global validation of MANTA currently consists of code-to-code benchmarking: comparisons with CATHARE from CEA (France), LEDA from EDF (France), ASURA from MHI (Japan), REALY2 from GA (USA) and RELAP5-3D from INL (USA) have already shown good agreement. Qualification against experimental data is also progressing (EVO loop, HE-FUS3 loop and PBMM). Nevertheless additional benchmarks against experimental data are required. Some facilities that could provide valuable data have been identified: namely, HTTR reactor in Japan, HTR10 reactor in China, SBL-30 loop in the USA (SNL). The qualification of component models will follow from the qualification tests of the components. The core model qualification follows from comparison with other codes and with experimental results. Further, experimental data from HTTR and HTR-10 safety tests and from SBL-30 loop is required." (Sec. 19.2.4.1, p. 291) – ***Reactor System Analysis Code/RELAP*** – "The U.S. DOE sponsors RELAP5 code development at the INL. It is expected that this support will continue. Development needs are highlighted in the report INEEL/EXT-04-02993.	AREVA-011 thru AREVA-022 (modeling codes)	Needs for improved reactor analysis computer models have been recognized in the AREVA PCDR.

Table 1E (AREVA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		Validation beyond that identified in this report and consistent with that planned for MANTA should be pursued.”		
E-6	Basic research should be conducted to strengthen the understanding and modeling capability of the displacement damage process in graphite. In addition, in graphite technology, there is a need for analytical models for oxidation, changes in physical properties, irradiation induced dimensional change, and irradiation creep. They could be developed to feed into a structural integrity model for the graphite core which would be used for core design and safety assessment.	See item E-3 regarding efforts for a structural mechanics code/model.	AREVA-008 thru AREVA-010 (testing and codification of graphite materials) AREVA-013 (improvement/ development of “other” codes)	Needs for improved structural mechanics computer models have been recognized in the AREVA PCDR.
E-7	Irradiation induced change in the coefficient of thermal expansion, including effects of creep strain.	See description on graphite R&D needs from Sec. 19.2.2.3, p. 286, under item E-1.	AREVA-008 thru AREVA-010 (testing and codification of graphite materials) AREVA-013 (improvement/ development of “other” codes)	Needs for further knowledge of the phenomena described in this item have been recognized in the AREVA PCDR.
E-8	Irradiation induced change in mechanical properties such as strength and toughness, including the effect of creep strain.	See description on graphite R&D needs from Sec. 19.2.2.3, p. 286, under item E-1.	AREVA-008 thru AREVA-010 (testing and codification of graphite materials) AREVA-013 (improvement/ development of “other”	Needs for further knowledge of the material characteristics described in this item have been recognized in the AREVA PCDR.

Table 1E (AREVA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
			codes)	
E-9	Blockage of coolant channel in a fuel element block or reactivity control block due to graphite failure and/or graphite spalling.	***There is no indication that this item has been specifically addressed.*** However, in Table 11-2, "Preliminary List of DBE Initiating Events", "***single fuel channel blockage***" is listed as a preliminary initiator for a design basis event.	AREVA-014 (fuel performance modeling and codes) AREVA-008 thru AREVA-010 (graphite qualification)	***There is no indication that this item has been specifically addressed in the AREVA PCDR.***
E-10	Statistical variation of non-irradiated properties, due to forming, processing, raw materials, and formulation.	This item has not been fully addressed for all graphite components. However, the following references exist for statistical control and sampling related to fuel fabrication: (Sec. 15.1.2.3, p. 224) – Compact Fabrication – "A predetermined number of compacts are destructively evaluated to ensure the lot of compacts ***meet the fuel specification on a statistical basis***. Once the ***chemical and physical attributes*** of the compacts has been confirmed, the lot will be certified and released for fuel block fabrication." (Sec. 15.3, p. 232) – Fuel Qualification Plan – "The second sequence is designed to provide the data used to qualify the fuel for use in the NGNP plant. A quantity of fuel would be fabricated, irradiated, and inspected that would yield the ***statistics required to demonstrate that the fuel supports the plant safety case***. This fuel would also be fabricated in the pilot line. It is envisioned that several batches would be made and blended to form a homogeneous lot upon which the results would be based. This process will be used to as closely as possible reflect anticipated commercial scale fabrication techniques. The ***statistical basis and acceptance criteria for the test will reflect this processing technique***." The R&D aspects of ***fuel development and qualification*** (fuel kernel, coating, compact, QA, and mass production) are addressed in section 19.2.1, beginning on page 282.	AREVA-014 (fuel performance modeling and codes)	This item has been addressed in the AREVA PCDR for statistical control and sampling related to fuel fabrication, ***but not for other graphite components.***
E-11	Ability to develop generic specifications that will ensure consistency of graphite quality over the lifetime of the reactor fleet, including for replacement components.	***There is no indication that this item has been specifically addressed.***	AREVA-008 thru AREVA-010 (testing and codification of graphite	***There is no indication that this item has been specifically addressed in the AREVA PCDR.***

Table 1E (AREVA) – GRAPHITE - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
			materials)	
E-12	Tribology (effects of moving surface interactions) of graphite in helium environment, including potentially impure helium environment (examples: surfaces sticking together; surfaces wearing on each other to generate dust, etc.)	(Sec. 7.7.1, p. 105) – ***Helium Purification Train*** – "The primary functions of the Purification Train are: • Removal of chemical and particulate contaminants from the primary coolant • Supply of purified helium to appropriate systems Since helium is used as the primary coolant, a ***helium purification system is required to provide the necessary degree of helium purity***. Oxidizing contaminants, in particular, may not exceed predetermined limits established in the specification. In detail, the helium purification system has the following functions: • Removal of particulate and gaseous contaminants from the primary coolant to maintain design values, in particular for H_2O, CO, CO_2, N_2, H_2, CH_4 • Removal of tritium • Removal of other radioactive contaminants from the helium, especially before transfer to the purified gas store (Xe, Kr, Ar) • Start up purification of the primary system before initial start up and after inspections and maintenance • Purification of newly delivered helium" (Sec. 19.2.2, p. 284) – Materials Development and Qualification – "…the VHTR design relies on contact conditions between different materials (metal to metal, graphite to ceramics, ceramics to metal, etc.) and R&D actions have to be performed to assess the contact conditions to avoid unexpected situations (bonding, wear, etc). As an example, the core support to reactor vessel interface is currently assumed to be a sliding interface. R&D actions are required to make sure that the helium environment (together with the contact pressure) is not likely to create a bonding effect between the alloy 800H and the 9CR1Mo materials. ***Tribology tests are needed on expected couples of materials in representative VHTR conditions***."	AREVA-008 thru AREVA-010 (testing and codification of graphite materials)	In the AREVA PCDR, a helium purification system has been incorporated into the design to ensure the purity of the helium environment, and the need for improved knowledge of tribology has been recognized.
E-13	Impact of degradation of thermal conductivity on fuel temperature limits.	(Sec. 20.2.1, p. 301) – Commercial Plant Power Level – "Modular VHTR's rely on conduction and thermal radiation in their passive safety features for decay heat removal. The thermal performance of the plant during a loss of active cooling is dominated by four items: the geometry of the plant, the thermal energy stored in the core at the beginning of the event, and energy (the decay heat) that is generated inside the core, and the ***heat transfer properties of the core (graphite)***. AREVA performed parametric studies to evaluate the sensitivity of the results of the limiting design basis accident (depressurized conduction cool down) to the key influencing parameters; namely, core power level, core inlet temperature, and ***graphite conductivity*** in terms of an equivalent change in reactor power. The results of the study support the conclusion that a maximum reactor thermal power rating of 565 MWth should be acceptable while allowing some margin for uncertainties. Based on the above, the commercial VHTR module should be designed to operate at the	AREVA-008 thru AREVA-010 (testing and codification of graphite materials)	In the AREVA PCDR, this item has been addressed in the design and by studies already performed.

Table 1E (AREVA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		maximum safe power level; and, based on the AREVA's evaluation of plant safety limits, that maximum power level is 565 MWth." Additional information discussing parametric studies, including thermal conductivity of graphite, can be found on page 26 of Appendix B2 of the PCDR. This information indicates that the sensitivity of peak temperature to variations in thermal conductivity of graphite is relatively low.		

Table 1F (AREVA) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
F-1	Cold oxygen (O2) and other heavy-gas accidental releases from the process plant that can flow from the chemical plant to the nuclear plant (depending upon wind, relative plant elevations, and nuclear plant air intakes) and potentially impact the integrity of reactor systems, structures, and components (SSCs). All of the proposed processes for production of hydrogen start with water, and thus all of the processes will produce oxygen as a byproduct of hydrogen production. Oxygen is the one common chemical safety issue that can impact nuclear plant safety. At high oxygen concentrations, many "noncombustible" materials become combustible and the potential for spontaneous combustion increases. Increased oxygen levels at the reactor can compromise the functioning of safety equipment.	(Sec. 11.6, p. 184) – ***Distance between facilities*** is recognized as a key aspect of collocating the reactor and hydrogen production plant,. Also, in section 11.6.2, page 188, ***oxygen is recognized as a hazard*** in the sense that it increases the hydrogen explosion risk. However specific design of the hydrogen production facility was outside of AREVA's assigned scope, and ***there is no indication that this item has been specifically addressed.***	(None)	Specific design of the hydrogen production facility was outside of the AREVA PCDR assigned scope, and ***there is no indication that this item has been specifically addressed.***
F-2	Failure of the IHX leading to potential damage to safety-related SSCs in the reactor due to blow-down effects from large mass transfer and over-pressurization of either secondary or primary side. The impact of the IHX failure depends upon the selection of the heat transfer fluid in the secondary heat transport loop. Helium is the leading candidate for the heat transport loop, but no final decisions have been made. If helium is used, the helium inventory in the secondary loop may be greater than the inventory in the reactor; thus, any leak in the IHX can significantly increase the total helium inventory involved in any reactor depressurization event.	(Sec. 11.3.2.5, p. 169) – "The unique design consideration accommodating the ***Hydrogen Production Pilot Plant*** (HPPP) is the primary ***coolant circuit*** dedicated for this application. It is unique in that it is sized for the smaller amount of energy needed for this application relative to that for power generation, about 60 MWth. It is expected that the ***performance of equipment and components in this circuit will be of high research and development interest***. Frequent inspection, maintenance, and design modification is anticipated." (Sec. 11.5.2.5, p. 180) – ***IHX Failure*** – "As an indirect cycle design, nuclear heat generated in the reactor core of AREVA's NGNP concept is transmitted to the power conversion or process heat system via an IHX. Failure of the IHX is any breach of the physical boundary between the primary and secondary circuits. ***AREVA's NGNP concept is designed with zero pressure differential across the pressure boundary common to both the primary and secondary circuits. As such, fluid exchange between the circuits following an IHX failure is driven by momentum and diffusion phenomena, rather than pressure.*** The main safety issue is the confinement of radiological content. AREVA's NGNP concept employs two separate IHX designs (i.e., plate type IHX supporting the HPPP process heat application and tube-type IHX supporting power generation). While the likelihood of failure is considered to be smaller with the tube-type design, the radiological consequences of an IHX failure are independent of IHX-type. The primary defense against a radiological release is the maintenance of low activity in the primary circuit….IHX failure detection is achieved by activity detection in the secondary side of the IHX combined with an overpressure of the primary helium compared to the secondary gas during normal operation. In case of IHX failure detection, the following actions are to be taken: • Heat generation control	AREVA-002 (IHX materials testing) AREVA-024 and AREVA-025 (IHX testing)	Indications are that this item has been addressed in the AREVA PCDR in the design, and will be further addressed with design improvements as the project progresses.

Table 1F (AREVA) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		• Control rods insertion (by automatic action) as abnormal parameter value is detected (note that in case of combination with loss of electrical power, control rods drop by gravity in the core), • Reserve Shutdown System insertion by operator action, if control rod insertion fails. • IHX isolation valves are closed. Isolation valves are implemented on the secondary circuit, close to the IHX vessel. • Automatic primary circulator trip on effected loop or, if the effected loop cannot be identified, on all loops. • Heat removal after reactor shutdown • SDHRS, if the effected IHX can be isolated, • SCS startup ensuring decay heat removal through primary helium forced convection. This start-up should be manual since large duration should be available. (If necessary, e.g., for reliability purpose, automatic startup may be defined). • RCCS with passive heat removal capacity. Efforts to depressurize the primary system and, thus, further minimize leakage to the secondary will also be considered."		
F-3	Failure of the process heat exchanger (PHX) leading to potential damage to safety-related SSCs in the reactor, due to fuel and primary system corrosion from the introduction of corrosive process plant chemicals leaking down the process heat transport line and failing the IHX.	Specific design of the hydrogen production facility was outside of AREVA's assigned scope, and they have addressed the design in a general way, mainly to identify and characterize required interfaces with the nuclear plant. ***There is no indication that this item has been specifically addressed.***	(None)	***There is no indication that this item has been specifically addressed in the AREVA PCDR.***
F-4	Steam generator failures leading to the introduction of steam/water into the primary system, potentially causing a reactivity spike and chemical attack of the TRISO fuel particle coatings and graphite. Some hydrogen production processes, such as high-temperature electrolysis, require steam as a process feedstock; thus, the high-temperature reactor may be required to provide high-temperature steam.	The AREVA design uses a ***steam generator*** as part of the Power Conversion System. (Sec. 8.3.1, p. 119) – "The gas turbine exhaust contains significant residual heat most of which is transferred to tertiary water/steam in the Heat Recovery Steam Generator (HRSG). The steam generated in the HRSG drives the HP and LP steam turbine to drive its generator to produce electricity." Table 11-2, p. 174, identifies ***steam generator tube ruptures*** among the preliminary list of initiating events for design basis events, and Section 11.5.2.4, p. 179, identifies a leak from a steam generator combined with failure of an IHX as a potential event of water ingress into the primary circuit, with the consequences being ***reactivity insertion*** and combustible gas control. Section 11.5.2.4 identifies the following ***uncertainties that must be resolved*** to ensure that this event can be mitigated: • the benefit of start up of the SCS • the benefit of primary circuit loop isolation strategies • the impact of water on graphite structure and its heat transfer properties	AREVA-040 (steam cycle testing)	The AREVA PCDR has proposed development of a white paper to provide discussion of water ingress events, including steam generator tube leaks.

Table 1F (AREVA) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		• the influence of water on fuel particles performances as well on the radio-elements trapped in the graphite blocks • the consequences of CO and H2 release • the limitation of water available to enter the pressure boundary • the impact of possible actuation of safety valve (primary and secondary) on potential radiological releases. (Sec. 21.1.2, p. 315) – "Any decision to adopt a steam cycle HTR configuration increases the significance of water ingress events due to the potential for ***steam generator leaks***. This issue was successfully managed in previous operating HTRs. However, the possibility for water ingress continues to be perceived as a significant issue within the broader nuclear community. There are various reasons for this including misunderstanding of the source of water ingress in the Fort St. Vrain reactor, failure to appreciate the differences in steam generator technology between HTRs and LWRs, and unfamiliarity with the consequences and mitigation of water ingress in HTRs. Steam line breaks within the reactor building also must be considered for steam cycle concepts. Steam line breaks must be evaluated for building pressurization issues and for any impact on building venting and filter systems, if a vented confinement concept is used for the NGNP. A ***white paper should be developed addressing water ingress and steam line break events*** and their likely impact on NHS design. The intent is not necessarily to provide detailed analyses of such events. Rather the focus should be on describing the issues and concerns associated with each type of event, the potential significance of these events on operation, safety, and licensing, mitigation of these events including likely design features which might be utilized, and ***likely R&D*** that might be necessary to resolve any open issues."		
F-5	Loss of the pressurized coolant inventory from the intermediate loop leading to a loss of primary reactor heat sink and the potential for hydrodynamic forces on the IHX leading to IHX failure and loss of reactor primary system coolant.	(Sec. 11.3.3.2, p. 170) – Heat Removal after Shutdown Function • "The Shutdown Cooling System (SCS) implemented inside the reactor vessel: ***This system can operate even if the secondary circuit and the primary forced helium circulation are not available.*** SCS is designed for achieving this function in pressurized and depressurized conditions. The SCS is made by a circulator and a heat exchanger transferring the decay heat from the core to a water circuit. • ***In case of failure of these systems***, the decay heat is transferred from the reactor vessel to the Reactor Cavity Cooling System (RCCS) mainly by radiation. The RCCS consists of two independent and redundant trains operating in natural circulation. During any conditions, its function is to maintain acceptable temperature of the reactor cavity concrete and the vessel support devices." Also see item F-2 for additional information from section 11.5.2.5, which discusses	AREVA-002 (IHX materials testing) AREVA-024 and AREVA-025 (IHX testing) AREVA-028 (SCS) AREVA-031 (RCCS)	This item has been addressed in the AREVA PCDR in the design.

Table 1F (AREVA) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		various modes of IHX failures, recovery modes, and event mitigations.		

Table 1A (GA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
A-1	*Core-Coolant Bypass Flow Phenomena (Normal Operation)* • Overcome difficulties in estimating bypass flow • More complete understanding and accounting of related design features such as fuel blocks (PMR) and core barrel configurations • In-core temperature testing • Parametric analysis of gap configurations to bound questions associated with gap and bypass flows	See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3.1.2.2, p. 3-27) - ***Bypass Flow Reduction*** – "Fuel temperatures can be reduced by reducing bypass flow. Bypass flow is defined as any flow that bypasses the coolant holes of the fuel elements. As shown in Figure 3.1-20, ***bypass flow channels include gaps between fuel columns and leakage between/from PSR blocks***. For the reference GT-MHR core design, approximately 3% of the flow is supplied to the control-rod channels, which have ***orifices to minimize bypass flow*** while also maintaining adequate cooling for the control rods. Composite-clad control rods require little or no cooling, which helps reduce the bypass flow fraction. Bypass flow can also be ***reduced by using graphite sealing keys*** below the active core to provide additional flow resistance for bypass flow occurring between fuel columns. ***Lateral restraint devices and sealing tubes in the PSR riser channels*** can reduce the leakage flow between/from the PSR blocks…FES has analyzed the flow distribution in the reactor vessel using a 3-D, 120°-sector ANSYS model (Figure 3.1-23). For the reference GT-MHR design (Figure 3.1-8), the bypass flow fraction is approximately 0.20. As shown in Table 3.1-9, routing the inlet flow through the PSR increases the bypass flow fraction to 0.37, primarily because of the relatively large lateral pressure gradients between the inlet flow path and reactor core. Adding sealing sleeves and lateral restraints reduces the bypass flow fraction to 0.14. Adding sealing keys at the bottom of the core further reduces the bypass flow fraction to 0.10. Reducing the bypass flow fraction from 0.20 to 0.10 reduces peak fuel temperatures by approximately 50°C. The FES results are consistent with recent calculations performed by OKBM for their design concept shown in Figure 3.1-10. The OKBM design also includes sealing sleeves in the coolant riser paths and lateral restraints to reduce bypass flow. As indicated in Figure 3.1-24, OKBM also estimates the bypass flow fraction to be approximately 0.10. FES has also performed flow distribution calculations using the FLOWNET flow network code." (sec. 3.1.2.2, p. 3-60) – "Design modifications to reduce bypass flow include sealing sleeves in the PSR risers, lateral restraints, and sealing keys below the core. Independent analyses by FES and OKBM indicate that bypass flow can be reduced to about 10%. The ***reactor internals design should be developed in more detail and include these modifications***." (Sec. 3.10.1.3, p. 3-200) - ***Nuclear Power Instrumentation*** – "Power-range ***ex-core neutron detectors*** will be placed in six detector wells, equally spaced around the reactor vessel. Each well will extend from the lower region of the reactor core to the refueling floor. Access to the detector wells will be from the refueling floor. Neutron detection equipment will include Intermediate and Power Range Monitoring Channels, and Source Range Detector Assemblies and Monitoring Channels. The	C.11.01.01 thru C.11.03.11) , C.11.04.03 (core instrumentation and testing) C.11.03.31 (core instrumentation validation) C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development)	The needs for refined analyses to better understand the core bypass flow phenomenon, and core monitoring instrumentation and testing, have been recognized in the General Atomics PCDR.

Table 1A (GA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable General Atomics R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		latter are retractable ***in-core devices***, and entail a significant NGNP development effort. "		
A-2	*Effective Core Thermal Conductivity* • For prismatic cores – Make available dose and temperature-dependent graphite thermal properties (especially thermal conductivity) to the NRC T/F code suite, to account for large uncertainties as well as for characterization of annealing effects during long-term heat-up D-LOFC accidents. • For pebble bed cores - Also considerable error bounds in effective core thermal conductivity as a function of both temperature and irradiation. Existing correlations available are empirical, but PBMR project has an experimental facility to be used to refine the database.	See item E-1 for information relating to graphite materials program. (Sec. 3.1.2.2, p. 3-43) – "A 30-deg. sector ANSYS model was used to analyze both low-pressure conduction cooldown (LPCC) and high-pressure conduction cooldown (HPCC) events. In order to reduce vessel temperatures during these accidents, the reactor internal design was modified to include a 100-mm layer of carbon insulation on the outer radial boundary of the PSR…A key parameter for these calculations is the ***graphite thermal conductivity***, which decreases with damage caused by neutron irradiation. For these studies, calculations were performed using both irradiated and unirradiated graphite properties. Calculations were also performed assuming ***annealing of irradiation damage*** as the graphite temperature increases according to the GA model for H-451 graphite. Full recovery from irradiation damage is assumed to occur at temperatures greater than 1300°C. The ANSYS model shown in Figure 3.1-41 was used to calculate the effective thermal conductivity of the graphite blocks. ***Other key parameters that affect heat transfer to the RCCS are the emissivities of the PSR, core barrel, RPV, and RCCS panels***…"	C.11.03.16 (graphite thermal properties data) C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development) C.16.00.01 thru C.16.00.06 (RCCS)	The needs for determining the properties of graphite materials, including thermal conductivity, have been recognized in the General Atomics PCDR.
A-3	*Afterheat Correlations* • Peak fuel temperatures in the D-LOFC accident are very sensitive to the afterheat (vs. time) to the same extent as they are to the core thermal conductivity function. Afterheat correlations are sensitive to fuel type and burn-up histories. Tracking fuel histories during operation can be challenging, and afterheat validation data is more difficult to obtain for long times after shutdown.	See item A-8 for information relating to development of analytical models. (Sec. 5.1.3, p. 5-7) – Accident/Transient Analysis – "In terms of safety consequences, the bounding accidents for the NGNP are a loss of flow leading to a high pressure conduction cooldown (HPCC) and loss of coolant leading to a low pressure conduction cooldown (LPCC). The HPCC event is typically initiated by trip of the PCS. The RPS automatically initiates a reactor trip on low flow or TM trip. The system pressure quickly equilibrates at about 5 MPa as the TM coasts down. Because the system remains at high pressure, the decay heat is more uniformly distributed within the core and vessel than during a LPCC event. The LPCC event is typically initiated by a small primary coolant leak, causing the system to depressurize to atmospheric pressure. The RPS automatically initiates a reactor trip on low coolant pressure. For both events, the SCS fails to start and decay heat is removed by thermal radiation and natural convection from the reactor vessel to the RCCS. These events have been analyzed in detail for the GT-MHR, and the results have shown that ***peak fuel temperatures*** remain below the design goal of 1600°C, and the temperatures for the vessel and other safety-related SSCs also remain below acceptable limits. ***Using an ATHENA model, these events were re-analyzed using the NGNP initial conditions***. Figure 5.1-3 shows the calculated peak fuel temperatures for the HPCC and LPCC events. For the LPCC event, the peak fuel	C.07.02.01 thru C.07.02.09 (fuel performance data) C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development) C.14.01.01 thru C.14.01.06, C.14.04.01	This phenomenon and the need for improved models have been addressed in the General Atomics PCDR.

Table 1A (GA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		temperature is 1525°C and occurs about 60 h following initiation of the event. For the HPCC event, the peak fuel temperature is 1349°C and occurs about 50 h following initiation of the event. As shown in Figure 5.1-4, the calculated peak vessel temperatures for the HPCC and LPCC events were approximately 478°C and 517°C, respectively. For both events, the peak vessel temperatures occurred about 72 h following initiation of the event. ***These results are consistent with previous results for the GT-MHR and show that the H2-MHR should retain the passive safety characteristics of the GT-MHR."***	thru C.14.04.12 (SCS) C.16.00.01 thru C.16.00.06 (RCCS)	
A-4	*Core Effective Pressure Drop* • Standardized and well-documented correlations for core pressure drop; conformation data may be needed for low-flow cases to better characterize flow distribution and plume formation (for the P-LOFC) and in-core airflow distributions during air ingress accidents. • PBR - parametric analyses using established ranges of different packing fractions to define a performance envelope.	In section 3.1.2.2, from pages 3-17 through 3-22, references are made to organizations that have performed pertinent core pressure drop analyses, and the computer codes used, including ATHENA. The pebble bed core portion of this item is not applicable to the PMR.	C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development)	This phenomenon and related efforts to date have been addressed in the General Atomics PCDR.
A-5	*RCCS Performance during LOFC* • Simulate RCCS safety functions in detail, with its predominantly radiant heat transfer coupling to the RPV and other heat transfer mechanisms within the reactor cavity. RCCS functions include maintaining the reactor cavity liner concrete temperature below prescribed limits, preventing the RPV peak temperature from exceeding limits during LOFC events, and minimizing parasitic heat losses during normal operation. • Models may be needed to simulate large pressure pulses in D-LOFC accidents that could damage the RCCS, reducing cooling and/or opening up another release path for air or water ingress to the reactor cavity, and perhaps for FPT out to the environment.	See item A-8 for information relating to development of analytical models. (Sec. 3.1.2.2, p. 3-59) – "…The RCCS heat removal rate could be increased during both normal operation and transient if the flow rate could be increased or the local heat-transfer coefficients within the RCCS could be increased. Increasing the RCCS stack height will increase the natural convection flow rate. However, as shown in Figure 3.1-55, there is only a slight reduction in peak vessel temperature (~6°C) for RCCS stack heights over the range 10 to 40 m. ***RCCS design optimization should be assessed in more detail during the next design phase***." (Sec. 3.5, p. 3-99) – Reactor Cavity Cooling System – "The system is required to operate continuously in all modes of plant operation to support normal operation, and, ***if forced cooling is lost, it functions to remove decay heat to ensure investment and safety protection.*** The RCCS consists of a cooling panel which includes cold downcomers and hot risers and is located inside the reactor cavity surrounding the reactor vessel. Connected to the cooling panel are the concentric hot and cold ducts which connect the panel to the inlet/outlet structure." (Sec. 3.5.2, p. 3-100) – ***RCCS Operation*** – "The RCCS is designed to remove ~4 MWt when the primary cooling circuit is either pressurized or depressurized. The RCCS is not required to remove decay heat during normal operation. However, since the system is passive, the system removes some parasitic heat during normal power operation, and removes some decay heat during normal shutdown because of the	C.16.00.01 thru C.16.00.06 (RCCS)	This phenomenon, related efforts to date, and the need for modeling and simulation codes have been addressed in the General Atomics PCDR.

Table 1A (GA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable General Atomics R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		difference in the reactor vessel temperature and the outside air temperature."		
A-6	*Fuel Performance Models* • Aspects of maximum fuel temperature plus time-at-temperature histories (critical limiting factors) for all fuel regions provide inputs to fuel failure models, to determine source terms and dose-vs.-frequency estimates. • Chemical reactions in air or water ingress accidents, which depend on temperature and should be included in the T/F codes. Especially for fast transients, detailed temperature profiles of the fuel and graphite should be taken into account for thermal stress calculations.	See item A-8 for information relating to development of analytical models. (Sec. 3.1.2.2, p. 3-29) - ***Fuel-Element Modifications***. "The ***thermal performance of the graphite fuel element can be improved*** by reducing the temperature rise from the bulk coolant to the fuel compact centerline. This can be accomplished by reducing the diameters of the coolant holes and fuel compacts. This modified design is referred to as a 12-row block because the number of rows of fuel holes across the flats of the hexagonal block was increased from 10 to 12 (excluding boundary rows). Figure 3.1-25 shows the conventional 10-row block design and the 12-row block design. Parameters for the 10-row and 12-row block designs are given in Table 3.1-10. For the 12-row block design, the minimum web thickness between the fuel and coolant holes was kept the same as the 10-row block for structural/strength considerations. As shown in Figure 3.1-26, the 12-row block design can reduce peak fuel temperatures by 30°C to 40°C, which can allow for reduction of the coolant inlet temperature. The higher flow resistance for the 12-row block is compensated for by the lower flow rate associated with a lower inlet temperature." (Sec. 3,1,2,2, p. 3-34) – ***Fuel Management Strategies*** – "Previous studies have shown that power distributions can be flattened if a concept referred to as fuel placement is used. With this concept, each column contains both new and old fuel in alternating layers at the beginning of an equilibrium cycle. In effect, fuel placement reduces the "age" component of power peaking. As shown in Figure 3.1-27, the fuel placement refueling scheme can reduce the peak column-averaged power factor by about 6%. Because the viscosity of helium increases with temperature, columns with higher peaking factors will tend to receive less flow, which further increases peak fuel temperatures. ***Flattening the power distribution among fuel columns will reduce flow variations and help to reduce peak fuel temperatures***. As part of their work with GA on nuclear hydrogen development, KAERI has been investigating a similar refueling scheme for a 3-batch core (Figure 3.1-28). The KAERI concept uses 9 fuel elements (slightly longer than standard) per column to facilitate a 3-batch shuffling scheme, and adds 6 additional columns (108 fuel columns) to reduce the average power density by 5.6%. KAERI has performed ***3-dimensional physics calculations*** to evaluate this concept, using 12% enriched fissile fuel only and zoning the particle packing fraction to reduce radial peaking factors. Figure 3.1-29 shows the calculated power distribution (at end of cycle when peaking factors were the highest) and Figure 3.1-30 shows the ***flow distribution calculated*** by GA using the POKE code. For these calculations, the bypass flow fraction was assumed to be 0.10 for each column. Figure 3.1-31 shows the ***calculated core temperature distributions*** for the 10-row and 12-row block designs with a coolant outlet temperature of 950°C and the coolant inlet temperature reduced to 490°C. Because of the relatively flat power and flow distributions, the calculated peak fuel temperature is below 1250°C, even with the	C.07.01.01 thru C.07.01.07 (fuel fabrication) C.07.02.01 thru C.07.02.09 (fuel performance data) C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport) N.07.05.01 thru N.07.05.14 (integrity testing of fuel and graphites)	The needs for modeling and simulation code development described in this item have been recognized in the General Atomics PCDR.

Table 1A (GA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		reduced inlet temperature and coolant flow rate. Only about 20% to 30% of the fuel is predicted to be above 1000°C, which helps limit release of Ag-110m and other noble metallic fission products.” (Sec. 3.1.2.2, p. 3-60) – “The block-core design provides great flexibility to optimize power distributions using fuel shuffling schemes. Scoping studies show fuel shuffling can significantly reduce power peaking factors and flatten flow distributions. ***More detailed assessments of fuel shuffling should be performed, including coupled physics/thermal analyses and assessing the impact of control-rod movemen***t. An additional 30°C to 40°C margin for peak fuel temperatures can be obtained using a modified, 12-row block design, which could allow for further reduction in the coolant inlet temperature. ***More detailed assessments of this concept include manufacturability, structural/stress analyses, and impacts on fuel costs***.” (Sec. 3.1.4.3, p. 3-73) – ***Radionuclide Transport Mechanisms*** – “Radionuclide transport is modeled in the fuel kernel, the particle coatings, fuel-compact matrix, fuel-element graphite, primary coolant circuit, and Reactor Building. [IAEA 1997] provides an excellent overview and an extensive bibliography of radionuclide transport mechanisms. The transport of radionuclides from the location of their birth through the various material regions of the core to their release into the helium coolant is a relatively complicated process. The principal steps and pathways are shown schematically in Figure 3.1-66. Also for certain classes of radionuclides, some steps are eliminated (e.g., noble gases are not diffusively released from intact TRISO particles and are not significantly retarded by the compact matrix or fuel-element graphite). While the actual radionuclide transport phenomena in the core can be very complex, the basic approach for modeling these phenomena is to treat radionuclide transport as a solid-state diffusion problem with various modifications and/or additions to account for the effects of irradiation and heterogeneities in the core materials…” (Sec. 3.1.4.4, p. 3-76) – ***Fuel Quality and Performance Requirements*** – “…The fuel and reactor core are to be designed such that there is at least a 50% probability that the radionuclide releases will be less than the Maximum Expected criteria, and at least a 95% probability that the releases will be less than the Design criteria. The logic for deriving these fuel requirements is illustrated in Figure 3.1-68. Top-level requirements for the NGNP are defined by both the regulators and the users. Lower-level requirements are then systematically derived using a systems-engineering approach. With this approach, the radionuclide control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on radionuclide releases from the VLPC, reactor vessel, and reactor core are successively derived. Fuel failure criteria are in turn derived from the allowable core release limits. Finally, the required as-manufactured fuel attributes are		

Table 1A (GA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		derived from the in-reactor fuel-failure criteria, with consideration of achievable values based on existing fuel manufacturing experience, thereby providing a logical basis for the fuel quality specifications…The maximum allowable release fractions for 30.2-yr Cs-137 and 249.8-d Ag-110m are included in Table 3.1-16 because these nuclides are expected to be the ***strongest contributors to worker dose***, based on previous assessments of radionuclide plateout distributions and plant-maintenance requirements." (Sec. 5.1.1.3, p. 5-6) - ***Control of Chemical Attack* – "**Chemical attack on fuel particles and on the graphite core structure can result from air or water ingress into the primary system. ***Steps have been taken to prevent ingress of contaminants, and consequences are expected to be acceptable if they occur. The likelihood of water entering the primary system is limited by the absence of high pressure and high energy sources of water in proximity to the primary system***. Under normal operating conditions, all water coolers and heat exchangers operate at lower pressures than the pressure of the primary coolant with which they exchange heat. In the event of a cooler or heat exchanger leak, primary coolant helium would leak out into the secondary cooling water until pressures equilibrate. Then the rate of ingress of sub-cooled water would be small, as water tries to enter the primary system by diffusion and gravity. The amount of water that could enter is limited to the inventory of water in the secondary coolant circuit located above the elevation of the leak. Most of the sub-cooled water that could enter the power conversion vessel would remain at the bottom of the vessel. Very little of it would become entrained in the helium coolant and be transported to the core. Core cooling can still be provided by either the PCS or the SCS, and would limit the potential for chemical attack. If core cooling is not available, the potential of water transport to the core would still be limited. The sub-cooled water will not flash to steam unless the primary coolant helium pressure is below the water saturation pressure, which may occur only when the reactor is operating at a low power level. The reaction rate of water and core graphite will be negligible. The reaction of steam and graphite is slow and endothermic and therefore is not self-sustaining."		
A-7	*Air Ingress Phenomena* • With little or no detail available about the confinement, only generalized studies and experiments would be practical. Bounding analytical studies could be useful in determining positive and negative features of proposed design characteristics. The major features of general interest would be quantification of long-term air in-leakage into the confinement, and the mixing and stratification characteristics of gases in prototypical cavities within the confinement.	See item A-6 for information presented on air ingress. Also in section 5.1.1.3, p. 5-6: "The likelihood of a breach of the primary coolant boundary, such that ***air ingress*** becomes a concern, is limited by the high integrity associated with pressure vessels and the limited size of penetrations. In the event of a breach, primary helium coolant would leak out until inside and outside pressures equilibrated. Then, the rate of air ingress would be small, as air tries to enter the breach primarily by natural circulation and diffusion at the same time as helium coolant, which it is displacing, tries to exit through the same hole. Large air ingress rates would require an implausible scenario of two concurrent breaches of an ASME Code Section III vessel in order to set up an effective circulation path. However, even	C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport) N.07.05.02 thru	The need for greater understanding of the air ingress phenomenon has been recognized in the General Atomics PCDR.

Table 1A (GA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		in that circumstance, air flow would be restricted by the flow resistance characteristics in the core (e.g., cooling channel high length-to-diameter ratio). Finally, the amount of air is limited by the size of the low leakage below grade containment building. As a result, the overall heat of reaction of air with graphite remains small relative to core decay heat. Also, any air that enters the primary coolant must react with graphite elements and fuel compact matrices before it can reach and chemically react with the embedded refractory-coated fuel particles. "	N.07.05.05 (graphite and other oxidation rates in air) C.11.03.23 (graphite oxidation)	
A-8	*Long-term analysis need* - Comprehensive suite of verified and validated accident simulation codes (core thermal-fluids, core neutronics, whole-plant transient behavior, confinement analysis, and chemical reactions), agreed-upon accident cases for regulatory acceptance, and robust supporting databases that NRC can use for independent confirmatory analysis of candidate plant and confinement designs and options.	(Sec. 7.2.1.1, p. 7-6) – ***Design Methods Development and Validation*** – "The design methods for analyzing prismatic HTGRs were first developed to support the design and licensing of FSV and the large HTGRs in the 1970s. A brief summary status of the prismatic core design methods is presented below. ***Most of the design methods used for the analysis of the plant systems, structures and components are commercially available design tools,*** such as ANSYS, SINDA/FLUENT, RELAP5, Pro/E, etc. GA's reactor physics codes were originally developed from the basic neutron transport and diffusion theory equations. These methods were adapted to high-temperature, graphite moderated systems to allow calculation of temperature-dependent graphite scattering kernels, and the development of fine group cross sections for graphite systems from point-wise data (e.g., ENDF/B, JEF, and JENDL data sets). These nuclear design methods have been benchmarked against other industry standard codes, such as MCNP, and integral test data from operating HTGRs and critical experiments with generally good agreement. While the ***experimental data used for nuclear code V&V*** are considered reliable, some of the older data and, in particular, the international data ***may not have an adequate QA pedigree to be accepted by the NRC without some confirmatory testing***. The basic approach for performing core thermal/fluid flow analyses for prismatic HTGRs was also established to support the design of FSV and the large HTGRs in the 1970s, and a number of codes were written at GA for that purpose. Although the analytical tools have evolved and the computational capabilities have improved enormously with modern computers, the basic analytical approach is still valid. Future core thermal/flow analysis for normal operation and accidents will be performed with industry standard codes, such as ANSYS and RELAP5, and various commercial CFD codes as required. Design methods have also been developed to predict the various fuel performance and radionuclide transport phenomena in HTGRs in order to generate source terms for plant design and safety analysis. The accuracy of these design methods have been assessed by comparing code predictions with data from operating reactors and integral test data from various experimental programs. ***In general, the uncertainties in the predicted source terms are large. These design methods are adequate for predicting source terms during NGNP conceptual design, but they will need to be upgraded during preliminary design and validated prior to completion of final design***. A number of core structural analysis codes were developed at GA during the past three decades and used extensively for core design and safety analysis.	C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport) C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development)	Long-term analysis needs for computer code development have been recognized in the General Atomics PCDR. ***Databases required for confirmatory use by NRC have not been addressed in the General Atomics PCDR.***

Table 1A (GA) – ACCIDENTS AND THERMAL FLUIDS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		However, future core structural analysis, including seismic analysis, will be performed with ANSYS and ANSYS/DYNA3D. ***Improved constitutive equations for graphite along with improved material property data will be required.”*** (Sec. 7.2.3.7, p. 7-16) – ***Design Methods Development and Validation*** – “An extensive code development and validation program is presented in the NGNP Design Methods Development and Validation Research and Development Program Plan. The emphasis is heavily upon core nuclear and thermal/fluid flow computational methods. Design methods for predicting coated-particle fuel performance and fission product transport are not addressed. Instead, the Plan states that the AGR Fuel Program will provide the necessary design methods for those applications. While the AGR Plan does include development of improved component models, etc., it does not include scope for developing ***advanced computational tools for full-core performance analysis or for predicting RN transport throughout the plant, and tritium transport is not addressed at all.*** The GA Team’s perspective is that the emphasis of the current NGNP methods development plan is misguided. At least for prismatic MHRs, the currently available computational tools for core nuclear analysis and thermal/fluid flow analysis are largely adequate for NGNP conceptual and preliminary design. The traditional GA design methods for analyzing prismatic HTGRs, which were first developed to support the design and licensing of FSV and the large HTGRs in the 1970s, are still available. However, for nuclear analysis, the traditional codes have been largely supplanted by industry standard codes, such as DIF3D and MCNP; and for thermal, flow, and structural analyses, commercial codes, such as ANSYS, RELAP5, SINDA/FLUENT, and CFX, are already being used routinely by the GA Team. In contrast, the ***design methods for predicting fuel performance and fission product transport are in need of modernization and upgrade*** to support NGNP design and licensing.”		

Table 1B (GA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
B-1	Time-dependence and spatial distribution of decay heat as a major factor in determining maximum fuel temperature during a D-LOFC.	(Sec. 5.1.3., p. 5-7) – ***Accident/Transient Analysis*** - "The bounding design basis events (DBEs) for the NGNP will be a loss of flow leading to a high pressure conduction cooldown (HPCC) and loss of coolant leading to a ***low pressure conduction cooldown*** (LPCC). The HPCC event is typically initiated by trip of the PCS. The RPS automatically initiates a reactor trip on low flow or turbomachine trip. ***Because the system remains at high pressure, the decay heat is more uniformly distributed within the core and vessel than during a LPCC event***. The LPCC event is typically initiated by a small primary coolant leak, causing the system to depressurize to atmospheric pressure. The RPS automatically initiates a reactor trip on low coolant pressure. For both events, the SCS fails to start and ***decay heat is removed by thermal radiation and natural convection from the reactor vessel to the RCCS***). ***These events have been analyzed in detail*** for a MHR operating with a reactor outlet coolant temperature of 950°C, and the results show that peak fuel temperatures remain below the design goal of 1600°C, and the temperatures for the vessel and other safety-related SSCs also remain below acceptable limits. For the LPCC event, the peak fuel temperature is 1525°C and occurs about 60 hours following initiation of the event. For the HPCC event, the peak fuel temperature is 1349°C and occurs about 50 hours following initiation of the event. The calculated peak vessel temperatures for the HPCC and LPCC events are approximately 478°C and 517°C, respectively. For both events, the peak vessel temperatures occurred about 72 h following initiation of the event."	C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development) C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS) C.16.00.01 thru C.16.00.06 (RCCS)	The needs for modeling and simulation code development described in this item have been recognized in the General Atomics PCDR.
B-2	Control and shutdown rod worth and reserve shutdown worth as required for hot and cold shutdown.	(Sec. 3.1.2.1, p. 3-12) – "The active core consists of 102 fuel columns in three annular rings with 10 fuel blocks per fuel column, for a total of 1020 fuel blocks in the active core. As shown in Figure 3.1-7, the core is designed with 120-degree symmetry and the ***control rods*** are also operated symmetrically. The outer reflector contains 36 control rods, arranged as 12 groups with 3 rods per group. There are 4 control-rod groups in the active core, again with 3 rods per group. The core also contains 18 channels for insertion of ***Reserve Shutdown Control*** (RSC) material (in the form of boronated pellets) in the event the control rods become inoperable. During operation, control rods in the active core are completely withdrawn, and only the control rods in the outer reflector are used for control. This control method precludes damage to the in-core control rods during loss-of-coolant accidents. A control rod design using a carbon-carbon composite for the cladding material is being evaluated that would allow the in-core rods (or control rods located in the inner reflector) to be used during normal operation, which will provide greater flexibility for flattening the radial power distribution and provide some additional margin for maintaining fuel temperatures and fuel performance within acceptable limits." (Sec. 3.10.2.2.2, p. 3-213) – ***Reactor Power and Temperature Control*** – "This previously developed control scheme is used for NGNP steady or transitory Reactor	C.11.01.03 and C11.01.04 (control rod and reserve shutdown design verification) C.11.03.22 (reserve shutdown pellet process development) C.14.01.01 thru C.14.01.06, C.14.04.01 thru	The needs for understanding control and reserve shutdown capability, as described in this item, is recognized in the General Atomics PCDR.

Table 1B (GA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		Exit temperature control. The control uses an outer temperature control loop, feeding an inner reactor flux control loop, and connected to a Control Rod Drive System as depicted in Figure 3.10-4. ***Control rod withdrawal/insertion sequencing*** is based on selective "one-at-a-time" rod withdrawal or insertion from predetermined control rod is applied through inclusion of total reactor mass flow rate to adjust for reactor core thermal "time-constant" variation over a wide range of reactor flow rate. This is not shown in Figure 3.10-4, but it is based on the sum of the two primary flow measurements which are shown. This scheme allows consistent ***"tight" adjustment*** of reactor power through the operating range in spite of the large core thermal effects which are characteristic of HTGR reactors." (Sec. 3.1.3, p. 3-64) – "The ***reserve shutdown control material*** is of the same composition as that for the control rods, except the B4C granules and graphite matrix are formed into cylindrical pellets with rounded ends and a diameter of 14 mm. The B4C granules are coated with dense PyC to prevent oxidation during off-normal events. The pellets are stored in hoppers located above the reactor core in both the both the inner and outer neutron control assemblies." (Sec. 5.1.1.3, p. 5-5) – ***Control of Heat Generation*** – "***Control rods*** drop by gravity into the core upon loss of electrical power. An automatic positive control action can cause the rods to drop, or the event itself may cut the power supply. It is an advantage that the rods need not be powered in. In addition, the NGNP has a ***redundant and diverse system*** to drop boronated graphite pellets by gravity into designated fuel element channels for reactivity control equivalent to rod insertion. Initiation of the latter system requires a positive control signal and an active response. If both the control rod and the reserve shutdown systems fail, (i.e., if neither control rods nor reserve shutdown material are inserted into the core), the temperature coefficient of reactivity will shut down the reactor from any power level following loss of cooling. As an example, given that no additional positive reactivity is inserted, core power will be reduced to shutdown levels by negative temperature coefficient alone, such that the RCCS alone can safely cool the core for more than 30 h after the initial shutdown. A test conducted at the AVR in Germany supports analysis which shows that following this initial shutdown, a gradual core temperature increase and negative reactivity addition will occur, with the core stabilizing at a low power level at which the heat generation rate matches the core cooling capability of the passive heat sinks. This is a stable, safe condition that can be maintained until corrective actions are taken to insert the control rods or drop reserve shutdown control material into the core to affect a full shutdown and to allow the reactor to be taken to cold shutdown condition."	C.14.04.12 (SCS) C.16.00.01 thru C.16.00.06 (RCCS)	
B-3	Sudden positive reactivity insertion due to pebble core compaction (packing fraction) due to earthquake.	PBR phenomenon; not applicable to PMR core.	(Not applicable)	This is a PBR phenomenon and is not applicable to the PMR core.

Table 1B (GA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
B-4	For tests at both PMRs and PBRs, consideration should be given (at least in the first core) to use of high-temperature in-core neutron detectors that can provide maps of axial and azimuthal power distributions and core-inner-to-outer-radius power tilts; these detectors would likely be located only in the inner and outer reflectors rather than in the core, due to temperature and connection limitations. • PMR concern - Whether improper axial-loading of fuel blocks during refueling can lead to an undetected power distribution anomaly and result in excessive operating fuel temperatures. • PBR concern - Radial and azimuthal power distributions in the mixed-fuel pebble bed are not well known, and there are indications from melt-wire tests conducted in the AVR (Germany) suggesting that pebbles near the walls of the reflector experienced unexpectedly high fuel temperatures.	See item A-1 for information on in-core and ex-core monitoring instrumentation. See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3.1.3, p. 3-61) – Neutron Control System – "The neutron control system design is the same as that for the GT-MHR. ***The system components consist of inner and outer neutron control assemblies, neutron source, source range detector assemblies, ex-vessel neutron detector assemblies, and the in-core flux mapping system***…During normal operation, the neutron flux levels are monitored by 6 symmetrically spaced ex-vessel fission chamber thermal neutron detectors. The signals from these detectors interface with the automatic control and protection systems to operate the control rod drives or the reserve shutdown control equipment. Three fission chamber source-range detectors are used to monitor neutron flux during startup and shutdown. These detectors are symmetrically spaced in reentrant penetrations located in the bottom head of the reactor vessel. These penetrations extend into vertical channels in the reflector elements near the bottom of the core. ***The in-core flux mapping system consists of movable detectors in the central column of the inner reflector and in the outer permanent reflectors***. The system enters from a housing located above the reactor vessel and vertically traverses down through the core to the bottom reflectors. The system contains two independent fission chambers and a single thermocouple."	C.11.01.01 thru C.11.03.11) , C.11.04.03 (core instrumentation and testing) C.11.03.31 (core instrumentation validation) C.11.04.03 (neutron detector service equipment design)	The need for core monitoring instrumentation has been recognized in the General Atomics PCDR.
B-5	In both the PMR and PBR, control rod misalignments in the outer reflector during operation would result in azimuthal power tilting that could cause xenon-135-induced oscillations when the misalignment is corrected; however, this needs to be verified by analysis and confirmed by test.	See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3.1.3, p. 3-61) – "The control rod guide tubes extend from the gamma shielding downward through the top head of the reactor vessel and upper plenum shroud to the upper core restraint elements. The ***guide tubes provide a clear passage for the control rods*** as they are inserted into and withdrawn from the core. All neutron control assemblies are equipped with two independent control rod drive units. The control rod drive equipment is located in the upper part of the neutron control assembly. The equipment consists of a DC torque motor, a 60:1 speed reducer, and a cable storage drum, all of which are mounted on a metal frame. The control rod is ***lowered and raised with a flexible high-nickel alloy cable***. Figure 3.1-58 shows the control rod design. The neutron absorber material consists of B4C granules uniformly dispersed in a graphite matrix and formed into annular compacts. The boron is enriched to 90 weight percent B-10 and the compacts contain 40 weight percent B4C. The compacts have an inner diameter of 52.8 mm, an outer diameter of 82.6 mm, and are enclosed in Incoloy 800H canisters for structural support. Alternatively, carbon-fiber reinforced carbon (C-C) composite canisters may be used for structural support. The control rod consists of a string of 18 canisters with sufficient ***mechanical flexibility to accommodate any postulated offset*** between elements, even during a seismic event."	C.11.01.03 (control rod design verification) C.11.03.02 thru C.11.03.06 (control rod failure modes and integrity) 11.03.24 (control rod high temp materials properties)	The issue of control rod misalignment has been addressed in the General Atomics PCDR with descriptions of the design features that maintain control rod alignment.

Table 1B (GA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
B-6	Replacing helium with a hydrogen-bearing compound such as in a steam/water ingress event may produce a pronounced positive reactivity. Steam/water ingress tends to have a positive reactivity effect due to increased neutron moderation and reduced neutron leakage.	See item A-6 for description of design features and controls associated with prevention of chemical attack by water ingress. See item B-7 for information relating to reactivity control. See item C-6 for a description of the test program. (Sec. 3.4.1, p. 3-97) – "During normal operation of the reactor system, the SCS operates in a standby mode. During this mode, a small amount of cold leg helium leaks (back flows) through the closed shutdown valve and flows opposite the normal flow direction through the SCS circulator and over the SCS heat exchanger tubes. In this mode the circulator is not operating, but the SCS cooling water system supplies a small amount of water flow to the heat exchanger. This water flow prevents thermal shock when the SCS switches to an active cooling mode, but also results in a parasitic heat loss of up to 1.3 MWt during normal operation. Therefore, the standby-mode water flow must be set as low as possible without resulting in one or both of the following adverse conditions: (a) boiling and/or (b) static instability due to the large hydrostatic head in the heat exchanger. ***During standby mode, the primary coolant helium pressure is higher than the SCS water pressure, in order to prevent water ingress into the reactor system during normal operation***. The SCS is manually switched from standby mode to an active cooling mode at the discretion of an operator."	C.11.01.03 and C11.01.04 (control rod and reserve shutdown design verification)	The need for greater understanding of the water ingress phenomenon has been recognized in the General Atomics PCDR.
B-7	With a higher atomic mass moderator such as carbon, the mean thermal energy of neutrons will be higher than that for hydrogen bound with oxygen in water; that is, graphite will tend to produce a "harder" thermal-neutron energy spectrum than would water-moderated systems. Thus, the moderator temperature-dependent reactivity coefficient (MTC) in both PMR and PBR depends upon the change of thermal-neutron energy spectrum with temperature, with possibly large effects on reactivity. Concerns are for effects on core transient behavior and passive safety shutdown characteristics.	(Sec. 3.1.1.1, p. 3-1) – "The fuel for the GT-MHR consists of microspheres of uranium oxycarbide that are coated with multiple layers of pyrolytic carbon (pyrocarbon) and silicon carbide. The GT-MHR core is designed to use a blend of two different particle types; a fissile particle that is enriched to 19.8% U-235 and fertile particle with natural uranium (NU, enrichment of 0.7% U-235). ***The fissile/fertile loading ratio is varied with location in the core, in order to optimize reactivity control***, minimize power peaking, and maximize fuel cycle length. The GT-MHR coated particle design parameters are given in Table 3.1-2. The fissile and fertile particle designs are somewhat different, with the fertile particle having a larger kernel and a thinner buffer coating layer. Preliminary core physics calculations performed by INL for an NGNP prismatic block MHR suggest that the reactor may be able to utilize a single fuel particle design, with the fuel particles potentially having different U-235 enrichments. However, ***more detailed calculations are needed*** to confirm that a single fuel particle design provides adequate core design flexibility." (Sec. 5.1.1.1, p. 5-2) – "The NGNP reactor core is designed to have a ***negative temperature coefficient of reactivity***. This characteristic means that as the reactor gets hotter, the change in temperature alone will reduce reactor power. ***For all***	C.11.01.03 and C11.01.04 (control rod and reserve shutdown design verification) C.16.00.01 thru C.16.00.06 (RCCS)	The General Atomics PCDR has addressed the General Atomics design strategies for reactivity control and neutron control as features of the design, and has stated that all credible reactivity addition events can be controlled. ***General Atomics has not specifically addressed NRC's concern over the "harder thermal-neutron energy spectrum" and its "possibly large effects on reactivity".***

Table 1B (GA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		credible reactivity addition events, the negative temperature coefficient can limit reactor power. (Sec. 5.1.1.3, p. 5-5) – ***Control of Heat Generation*** – "Control rods drop by gravity into the core upon loss of electrical power. An automatic positive control action can cause the rods to drop, or the event itself may cut the power supply. It is an advantage that the rods need not be powered in. In addition, the NGNP has a redundant and diverse system to drop boronated graphite pellets by gravity into designated fuel element channels for ***reactivity control*** equivalent to rod insertion. Initiation of the latter system requires a positive control signal and an active response. If both the control rod and the reserve shutdown systems fail, (i.e., if neither control rods nor reserve shutdown material are inserted into the core), the ***temperature coefficient of reactivity will shut down the reactor*** from any power level following loss of cooling. As an example, given that no additional positive reactivity is inserted, core power will be reduced to shutdown levels by negative temperature coefficient alone, such that the RCCS alone can safely cool the core for more than 30 h after the initial shutdown. A test conducted at the AVR in Germany supports analysis which shows that following this initial shutdown, a gradual core temperature increase and negative reactivity addition will occur, with the core stabilizing at a low power level at which the heat generation rate matches the core cooling capability of the passive heat sinks. This is a ***stable, safe condition*** that can be maintained until corrective actions are taken to insert the control rods or drop reserve shutdown control material into the core to affect a full shutdown and to allow the reactor to be taken to cold shutdown condition." (Sec. 7.4, p. 7-25) – ***Response to Accident Tests*** - "These tests are intended to demonstrate the inherent response characteristics of the reactor module. Four basic categories of events are proposed: (1) ***reactivity transients***, (2) pressurized cool down, (3) water ingress, and (4) depressurized cool down. These categories cover the performance of the key systems which provide safety and investment protection."		
B-8	Variations in fuel enrichments, kernel diameters, coatings, and density of packing (PMR vs. PBR) must be accounted for in calculating the neutron reaction self-shielding effects in both the resonance or epithermal region and the thermal region of the neutron energy spectrum, to properly calculate the Doppler fuel temperature coefficient of reactivity and the MTC.	See item B-7 for information relating to reactivity control. See item E-11 for GA's draft generic fuel specification. (Sec. 3.1.4.4, p. 3-76) – ***Fuel Quality and Performance Requirements*** – "…The fuel and reactor core are to be designed such that there is at least a 50% probability that the radionuclide releases will be less than the Maximum Expected criteria, and at least a 95% probability that the releases will be less than the Design criteria. The logic for deriving these fuel requirements is illustrated in Figure 3.1-68. Top-level requirements for the NGNP are defined by both the regulators and the users. Lower-level requirements are then systematically derived using a systems-engineering	C.07.01.01 thru C.07.01.07 (fuel fabrication) C.07.02.01 thru C.07.02.09 (fuel performance	The treatment of fuel production in terms of establishing standard statistically-based specifications has been addressed in the General Atomics PCDR. ***It is not apparent that the phenomena cited in this item have been used as a basis for that specification***.

Table 1B (GA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable General Atomics R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		approach. With this approach, the radionuclide control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on radionuclide releases from the VLPC, reactor vessel, and reactor core are successively derived. Fuel failure criteria are in turn derived from the allowable core release limits. Finally, the required as-manufactured fuel attributes are derived from the in-reactor fuel-failure criteria, with consideration of achievable values based on existing fuel manufacturing experience, thereby providing a logical basis for the fuel quality specifications…The maximum allowable release fractions for 30.2-yr Cs-137 and 249.8-d Ag-110m are included in Table 3.1-16 because these nuclides are expected to be the ***strongest contributors to worker dose***, based on previous assessments of radionuclide plateout distributions and plant-maintenance requirements." Sec. 7.2.3.1, beginning on p. 7-9, provides information on ***technology development required*** for the General Atomics Fuel Fission Products Program, including Fuel Process Development and Fuel Materials Qualification.	data)	
B-9	Due to concerns over control rod drive reliability and re-criticality after Xenon-135 decay, the plant operator retains the safety function of achieving long-term hot and cold shutdown during an extended ATWS; and the equipment used by the operator to carry out this safety function, whether located in the control room or in a remote location, must be appropriately qualified to execute that safety function.	(Sec. 5.1.1.3, p. 5-5) – ***Control of Heat Generation*** – "***Control rods*** drop by gravity into the core upon loss of electrical power. An automatic positive control action can cause the rods to drop, or the event itself may cut the power supply. It is an advantage that the rods need not be powered in. In addition, the NGNP has a ***redundant and diverse system*** to drop boronated graphite pellets by gravity into designated fuel element channels for reactivity control equivalent to rod insertion. Initiation of the latter system requires a positive control signal and an active response. ***If both the control rod and the reserve shutdown systems fail***, (i.e., if neither control rods nor reserve shutdown material are inserted into the core), the temperature coefficient of reactivity will shut down the reactor from any power level following loss of cooling." In the General Atomics PCDR, this item appears to be addressed in the design, however, ***there is no indication that re-criticality following xenon decay in an ATWS event has been specifically addressed in the General Atomics PCDR.***	C.11.01.03 and C11.01.04 (control rod and reserve shutdown design verification) C.11.03.02 thru C.11.03.06 (control rod failure modes and integrity) 11.03.24 (control rod high temp materials properties)	In the General Atomics PCDR, this item appears to be addressed in the design, however, ***there is no indication that re-criticality following xenon decay in an ATWS event has been specifically addressed in the General Atomics PCDR.***
B-10	The uniqueness of configuration (tall, thin annular core) of current PMR and PBR designs and high operating temperatures require detailed reactor physics testing of the first unit as a function of core burnup, and of the start-ups of the second and perhaps third cycles. Attention should be	See item A-1 for information on in-core and ex-core monitoring instrumentation. See item A-8 for a description of Design Methods Development and Validation.	C.11.01.01 thru C.11.03.11) , C.11.04.03 (core	Needs for analytical codes, instrumentation, and testing relating to core monitoring are addressed in the General Atomics PCDR. General Atomics credits the core geometry as one of the design features that will help to

Table 1B (GA) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	paid to the instrumentation needs for these tests since neutron sensors must be both distributed and inter-calibrated to infer power distributions. Neutron detectors used in test measurements should also be sensitive enough to measure reactivity and changes in flux levels and distributions.	See item B-4 for information on instrumentation of Neutron Control System. See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 5.1.1.1, p. 5-2) - Core Geometry and Size – "The annular core geometry, ***limited core diameter***, low thermal power rating, and low power density of the NGNP assure sufficient decay heat removal to an ultimate heat sink by the natural processes of radiation, conduction, and convection, to ***preclude any significant particle coating failure or radionuclide release*** under all conditions of loss of forced cooling or loss of coolant pressure."	instrumentation and testing) C.11.03.31 (core instrumentation validation) C.11.04.03 (neutron detector service equipment design verification)	control the type of phenomenon of concern to NRC.

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
C-1	*General Safety Analysis/Safety Document Needs* • Comprehensive description of the NGNP safety philosophy, a listing of the components involved, and the conditions under which these components are expected to perform their safety functions. • Explanation of how this philosophy meets the defense-in-depth approach and, in particular, answers to the following: ○ Will the components that perform a safety function (retain FPs) be classified as safety-related components, with the imposition of equipment qualification, in-service inspections, and/or Technical Specifications LCOs and SRs? ○ How will aging issues be addressed? If the safety function of a component is to retain FPs on its surface during adverse conditions, how can it be ensured that this function can be retained for long periods (decades), despite the possible presence of other long-term surface degradation mechanisms? ○ Will the surface state of a non-replaceable or difficult-to-replace component be reactivated by chemical action or cleaning during its service life? • A sound basis for the selection of the physical models and the data for these models must be justified. • The materials to be used and their sensitivity on the transport case must be identified. • Once the actual reactor design is available, the transport pathways that result from the accident conditions must be identified, along with the relevant models and data needed for the resulting calculations. • Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.	See item A-8 for a description of Design Methods Development and Validation. See item C-6 for information relating to the test program. See item D-1 for information relating to metallic materials characterization. See item E-1 for information relating to graphite materials characterization. (Sec. 1.5.1, p. 1-18) – "Both PMRs and PBRs can use UCO fuel, and by doing so would benefit from lower fuel costs because of the higher fuel burnup obtainable with UCO fuel relative to UO2 fuel. However, the economic penalty associated with use of UO2 fuel would be greater for a PMR than a PBR because this would necessitate a shorter refueling cycle, thereby reducing reactor availability. Also, it is not clear that a PMR loaded with UO2 fuel could operate for an extended period of time with a core outlet coolant temperature of 950°C because of the potential for kernel migration in UO2 fuel exposed to high thermal gradients. The capability of PBRs to use UO2 fuel, which has a ***more extensive irradiation and safety testing database*** than UCO fuel, could potentially make licensing a pebble bed NGNP somewhat less difficult than licensing a prismatic block NGNP. However, this advantage would not extend to a follow-on commercial pebble bed VHTR because it is ***expected that UCO fuel will have been qualified and be available*** for use by the time a commercial VHTR is built." (Sec. 3.1.4, p. 3-64) – "For modular gas-cooled reactor designs, a ***hallmark philosophy*** has been adopted since the early 1980s to design the plant such that radionuclides would be retained in the core during normal operation and postulated accidents. The key to achieving this safety goal is the reliance upon ceramic-coated fuel particles for primary fission product containment at their source, along with passive cooling to assure that the integrity of the coated particles is maintained even if the normal active cooling systems were permanently disrupted. This design philosophy has been carried forward for all subsequent MHR designs, including the NGNP. Fuel performance and radionuclide control in gas-cooled reactors is discussed in detail in numerous publications, including [IAEA 1997], [Hanson 2002], and [Hanson 2004a]. As is discussed in detail in Section 5.1.1.1, the radionuclide containment system for the NGNP, which reflects a ***defense-in-depth philosophy***, is comprised of multiple barriers to limit radionuclide release from the core to the environment to insignificant levels during normal operation and postulated accidents. The five principal release barriers are: (1) the fuel kernel; (2) the particle coatings (particularly the SiC coating); (3) the fuel element structural graphite; (4) the primary coolant pressure boundary; and (5) the Reactor Building/containment structure. The	C.07.01.01 thru C.07.01.07 (fuel fabrication) C.07.02.01 thru C.07.02.09 (fuel performance data) C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport) C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development) C.11.04.04 thru C.11.04.06 (ISIs and Surveillances for reactor internals and core supports)	The General Atomics PCDR has recognized most of the safety analysis/safety document needs detailed in this item, ***with the exception of the following:*** • ***How will aging issues be addressed? If the safety function of a component is to retain FPs on its surface during adverse conditions, how can it be ensured that this function can be retained for long periods (decades), despite the possible presence of other long-term surface degradation mechanisms?*** • ***Will the surface state of a non-replaceable or difficult-to-replace component be reactivated by chemical action or cleaning during its service life?*** • ***Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.***

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	• The fuel database must be developed, as well as fuel-failure models and fuel material properties (both measurable and process controlled).	most important of these barriers to fission product release from the core is the silicon carbide and pyrocarbon coatings of each fuel particle. Both the SiC and PyC coatings provide a barrier to the release of fission gases. The SiC coating acts as the primary barrier to the release of metallic fission products because of the low solubilities and diffusion coefficients of fission metals in SiC…" (Sec. 3.10, p. 3-197) – ***Plant Operation and Control Systems*** – "The unique features of the MHR assure the general public inherent protection against fission product release from the reactor core. In addition, the inclusion of the safety-related Reactor Protection System (RPS) and the non-safety-related Investment Protection System (IPS) in the NGNP specifically provide the "***defense in depth***" design strategy that is required for modern reactor plants. Other design areas related to a complete "defense in depth" protection strategy are the Essential AC Electric System and the Essential DC Electric System. Also, systems such as the Reactor System contain end-action hardware to perform safety-related and non-safety actions. The Plant Control, Data, and Instrumentation System (PCDIS) provides normal control and instrumentation functions, and also provides overall integration of the control and protection functions into a combined plant control system. This system provides normal (main loop) cooling if possible following a reactor trip, broadening "***defense in depth***" design features by making the SCS or RCCS less likely to be used for reactor cooling." ***Sections 3.10.1.1, p. 3-198 (RPS/IPS), and 3.10.2.1, p. 3-208 (PCDIS) also contain recommendations for further development and improvement of the design of these systems*."** (Sec. 5.1.1, p. 5-1) – ***Key Inherent Safety Features and Design Provisions*** – "A ***defense-in-depth approach*** to safety has always been used in the design of MHRs including the NGNP. The philosophy of defense-in depth includes prevention of accidents by requiring reliable operating systems capable of handling anticipated operational occurrences. It nevertheless assumes these systems could fail and thus requires that certain functions be fulfilled to prevent and mitigate consequences of those failures. The ultimate goal is to ensure that plant operation will have negligible impact on the health and safety of the public under a comprehensive, extensive range of expected and postulated conditions. A key feature of defense-in-depth is the provision of multiple barriers to the release of fission products and systems which protect these barriers. Furthermore, these systems are capable of functioning despite credible failures, by being redundant, independent, and diverse. The assurance of safety is thereby vested in multiple, independent safety provisions, no one of which is relied upon excessively. Analysis of design-basis events (DBEs) and beyond-design basis events (BDBEs) early in the design process is a means of identifying and providing ways to further enhance plant safety.. Finally, contingency measures are provided in the event that fission products are released anyway. Defense-in-depth is comprehensive, covering aspects of human involvement (e.g., administrative controls, quality assurance, human factors engineering, training, etc.) to assure the	C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS) C.16.00.01 thru C.16.00.06 (RCCS)	

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable General Atomics R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		accuracy and sufficiency of the design, construction, and operation of the plant." (***Additional information follows on inherent safety characteristics, specific design features, multiple barriers to release of fission products, accident prevention and mitigation***.) (Sec. 5.2.1.1, p. 5-13) – "After a CP is issued by the NRC, the applicant must, if it did not as part of the original application, submit a Final Safety Analysis Report (FSAR) to support its application for an Operating License (OL). Typically, after the CP is issued by the NRC and construction is underway, the Licensee begins developing the FSAR. During this period of the Project, the PSAR is revised and updated to reflect the evolving plant design as well as ***operational aspects*** (e.g., procedures, ***Technical Specifications***, Human Factors, Emergency Planning, Security, Programs, etc.) that were not available, or needed by the NRC during the early phase of the Project for the issuance of the CP. The FSAR describes the final design of the facility as well as its operational and emergency procedures. The NRC then prepares a Final Safety Evaluation Report (FSER) for the OL, and the ACRS makes an independent evaluation and presents its advice to the Commission."		
C-2	*Model Development and V&V* - Physical models and the supporting mathematical methods, addressing: • Nuclides of interest • Fission product release from the fuel • Diffusion, adsorption, and desorption in graphite and fuel matrix materials • Adsorption, desorption, and in-diffusion in reactor system metals • Chemical and physical forms of the FPs in the coolant • Tritium transport models • Aerosols and dusts that plate-out on reactor system components and their mobility • Fission product reactions with the confinement building materials • Reactions of the reactor system components and fission products with air or steam • Plume models that transport the released material beyond the reactor building • Determination of the safety function of each subsystem and the level of FPT attenuation required. • Determination of level of sensitivity to component uncertainties and how this reflects on the physical	See item A-6 for information on air and water ingress See item A-8 for a description of Design Methods Development and Validation. See item C-5 for description from Sec. 7.2.3.1, p. 7-13, ***Radionuclide Transport.*** Tritium transport , radionuclide transport in the containment, and other fission product tests are identified as needs. See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3.1.4.3, p. 3-73) - ***Radionuclide Transport Mechanisms*** - "Radionuclide transport is modeled in the fuel kernel, the particle coatings, fuel compact matrix, fuel-element graphite, primary coolant circuit, and Reactor Building. [IAEA 1997] provides an excellent overview and an extensive bibliography of radionuclide transport mechanisms. The transport of radionuclides from the location of their birth through the various material regions of the core to their release into the helium coolant is a relatively complicated process. The principal steps and ***pathways*** are shown schematically in Figure 3.1-66. Also for certain classes of radionuclides, some steps are eliminated (e.g., noble gases are not diffusively released from intact TRISO particles and are not significantly retarded by the compact matrix or fuel element graphite). While the actual radionuclide transport phenomena in the core can be very complex, the basic approach for modeling these phenomena is to treat radionuclide transport as a solid-state diffusion problem with various modifications and/or additions to account for the effects of irradiation and heterogeneities in the core	C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport) C.11.02 and C.11.03 (reactor vessel, core, and hot duct) N.13.01 and N.13.02 (primary heat transport system and IHX) C.14.01.01 thru	The General Atomics PCDR has recognized the needs for most of the model development and V&V detailed in this item, ***with the following exceptions:*** • ***Fission product reactions with the confinement building materials*** • Determination of the safety function of each subsystem and the level of FPT attenuation required. ***(Safety functions are specified, but level of FPT attenuation is not addressed.)*** • ***Determination of level of sensitivity to component uncertainties and how this reflects on the physical models.*** • ***Estimation of difficulty in obtaining the data and conducting the testing to support the safety case.***

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	models. • Estimation of difficulty in obtaining the data and conducting the testing to support the safety case. • Scoping of how V&V can be performed.	materials…The transport of volatile fission metals in fuel-compact matrix and graphite is also modeled as transient diffusion processes. It is assumed that sorption equilibrium prevails in the gap between the fuel compact and the fuel hole surface of the fuel block. At the coolant boundary, the mass flux from the surface into the flowing coolant is given by the product of a convective mass transfer coefficient and the concentration gradient between the equilibrium desorption pressure and the mixed-mean concentration in the coolant. Diffusion coefficients and sorption isotherms have been determined experimentally for a number of nuclear graphites and matrix materials [IAEA 1997]. The ***transport and deposition of condensable radionuclides from the flowing helium coolant to fixed surfaces in the primary coolant circuit*** is essentially a convective mass transfer problem. Usually, deposition is conceived as a two-step process: (1) gaseous diffusion to the wall and (2) a wall effect, typically an adsorption process. The latter step is necessary because numerous experiments have shown that, under certain circumstances, graphitic and metallic surfaces have a limited capacity to sorb certain radioactive species. The sorptivity of metals for volatile fission products is typically a function of surface oxidation state and temperature. The wall effect may be simply an adsorption process whereby the active sites are confined to the surface. Alternatively, there are some data suggesting that certain radionuclides, principally Ag isotopes, may penetrate into the bulk of metallic components. The condensable radionuclides that are ***plated out*** in the primary circuit may be partially re-entrained and released to the Reactor Building during rapid depressurization transients. A potentially significant removal mechanism, especially during rapid depressurizations, is mechanical ***re-entrainment of deposited particulate matter contaminated by plateout and/or spallation of friable surface films***; this mechanical re-entrainment is traditionally referred to as “liftoff”. Empirical liftoff models have been developed by correlating the fractional re-entrainment of plated out fission products measured in blowdown tests with the shear ratio (the ratio of the wall shear during a depressurization transient to that during normal operation). The VLPC of the NGNP is expected to be a significant barrier to the release of condensable radionuclides to the environment during accident conditions. Consequently, the ***natural removal mechanisms, including condensation, gravitational settling, and turbulent deposition are modeled***.” (Sec. 5.1.2, p. 5-7) – ***Safety Related Systems, Structures, and Components*** – The key safety features and safety functions of the NGNP have been described above. On the basis of prior safety assessments, the major systems, structures, and components which are relied upon to perform one or more safety functions (e.g., ensuring safe shutdown and protection of the primary coolant pressure boundary) or are otherwise relied upon to meet the siting dose criteria have been identified. The set of plant features proposed to be classified as safety-related is comprised of the following: • Reactor System, including neutron control assemblies, ex-vessel neutron detectors, the reactor internals, reactor core, and fuel	C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS) C.16.00.01 thru C.16.00.06 (RCCS) C.31.01.01 and C.31.01.02 (reactor protection system)	

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		• Vessel System, including the ASME Section III vessels and pressure relief • RCCS, including the entire system as required for removal of residual heat • RPS, including all sensors, control logic, and housings supporting safety trips and wells which are part of the Reactor Service Building • Essential AC and DC power systems Consistent with the simple yet robust safety design approach, only a relatively modest number of systems, structures, and components (SSCs) are important in ensuring public health and safety. Equally important, these SSCs reflect the utilization of passive features. Thus, not only is susceptibility to failures in power systems, moving parts, and operator error reduced by the NGNP safety systems, but the operating staff's maintenance and ISI burdens are minimized."		
C-3	*Materials/Component Data* - Relevant data on materials or components over the range of interest and data uncertainties (single effects testing), including the following: • Graphite transport property and air/steam erosion data specific to the design material. • Metal alloy data specific to the design material. • Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product. • Data on helium impurities that will likely set the oxygen potential of the system, and the species to be included in an analysis. • Data associated with component aging: surface qualities of the reactor system components after many years of operation. • Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit. • Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. • If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties.	See item C-2 for information relating to radionuclide mechanisms and transport modeling. See item C-6 for a description of the Initial Testing and Inspection Program. See Item D-1 for materials data relating to metallic materials. See item E-1 for materials data on graphite materials. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine. (Sec. 3.1.4.1, p. 3-70) - ***Diffusive Release Through Intact Coatings –*** **"**Based on previous irradiation testing and post-irradiation heating, SiC is not very retentive of Ag (and possibly other noble metals) at high temperatures. The Ag-110m ***transports through the primary cooling circuit and deposits on the cooler wetted surfaces***, which could impact operations and maintenance activities. The plateout activity is also a potential source of ***radioactivity release during hypothetical accidents involving a rapid loss of coolant***, when the shear forces during depressurization are sufficiently high to remove some of the deposited activity. Figure 3.1-64 shows the breakthrough time as a function of temperature for Ag diffusing through a 35-μm SiC layer. For temperatures above 1000°C, the breakthrough time is less than 100 days, which is well below the fuel residence time of 850 days. As discussed in Section 3.1.4, limiting the release of Ag to acceptable levels is largely accomplished through optimization of the nuclear and thermal hydraulic design of the reactor core."	C.07.01 and C.07.02 (fuel fabrication and performance) C.07.03 (fission product transport) C.07.04 (core corrosion data) C.11.01 (neutron control materials and testing) C.11.02 and C.11.03 (reactor internals, hot duct and core) C.12.01 (materials properties for reactor vessel)	The General Atomics PCDR has recognized most of the needs for materials and component data detailed in this item, ***with the exception of the following:*** • ***Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit.*** • ***Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions.*** • ***If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties.*** • ***Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior.***

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	• Data regarding turbine or power conversion components that may have to be decontaminated prior to maintenance (initial collection of FPs while in the reactor circuit; decontamination of components; new surface state of the component after decontamination). • Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior,	Section 3.6, beginning on page 3-101, presents a detailed description of the Power Conversion System. With regard to ***decontamination***, section 3.6.1.1, p. 3-101, points out that one of the key design features of the PCS is the use of electromagnetic bearings, which eliminate the possibility of ***lubricant ingress*** into the primary circuit. Section 3.6.2.2.2, p. 3-121, addresses radioactive decontamination of compressors as a potential risk issue. Section 3.6.2.3.2, p. 3-125, addresses radioactive contamination plate-out on turbine materials, causing brittleness and corrosion, as a potential risk issue. Section 3.6.2.4 addresses potential contamination of the electric generator with the following statement: "Radioactive contamination and high temperatures will be managed by enclosing the generator in a separate compartment at a pressure slightly above the rest of the PCU and with cooling to avoid subjecting the generator to undue temperatures." (Sec. 3.9.1, p. 3-189) – ***Primary Coolant Purification System*** – "This subsystem provides a means to remove circulating impurities from the primary coolant helium, and to transfer those impurities to the radioactive liquid and gas waste systems of the facility. A separate regeneration section within this subsystem is used to remove the impurities that accumulate in the purification subsystem adsorbers. The regeneration section is operated periodically under automatic control whenever regeneration is required. The primary coolant helium purification subsystem consists of two separate, independent, but identical trains of components as shown in Figure 3.9-1. All of the components that make up the trains are mechanically passive in nature; however, the adsorber elements become radioactive as the removed impurities are concentrated within the various media. Each purification train must therefore be located in a shielded vault to minimize personnel exposure to radiation. Helium purification is accomplished by routing a small side stream of helium from the primary coolant system through a series of purification components. ***These components remove the following chemical impurities: Br, I, H2O, CO, CO2, H2 (including Tritium), N2, O2, H2S, Kr, Xe, CH4, and other hydrocarbons***."	N.13.01 and N.13.02 (various IHX tests and materials research) C.14.01 and 14.04 (various SCS tests and materials research) C.16.00.01, C.16.00.02, and C.16.00.05 (RCCS emissivity, testing, conductivity) C.21.01.05 (fuel handling test)	
C-4	Reactor component and confinement/containment configuration and their relative roles in the safety case • Respective roles of the reactor circuit and containment or confinement system must be known before their modeling adequacy can be determined. • Estimate of source and budgeting of FP holdup among the fuel form, reactor circuit components, mobile elements such as dust, and the reactor building, as a means of focusing components to be emphasized in analysis. • Determination of transport pathway, goals for FP retention at each step in the pathway, local (accident) operating environment at each step of the pathway.	See item C-2 for information relating to radionuclide mechanisms and transport modeling. (Sec. 4.1, p. 4-1) – ***Reactor Building*** – "The RB for the NGNP 600-MWt reactor is classified as a ***vented low-pressure containment*** (VLPC). The RB is approximately 30-m (100-ft) wide by 50-m (165-ft) long. The RB consists of a below-grade multi-celled, embedded structure and the RCCS inlet/outlet structures, both of which are constructed of cast-in-place reinforced concrete. The degree of embedment was selected to serve a number of objectives, including reduced cost and complexity of construction, ease of operation, minimization of shielding, and good seismic performance. The below-grade location provides significant design benefits including grade level access for refueling, reduction of seismic effects, and protection from external events."	C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport) C.11.02 and C.11.03 (reactor vessel, core,	The definition of roles has been included in the General Atomics PCDR. The need for computer code development to understand fission product transport and distribution has been recognized in the General Atomics PCDR.

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
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		(Sec. 5.1.1.2, p. 5-4) - ***Primary Coolant Pressure Boundary*** – "The fourth release barrier is the primary coolant pressure boundary. This barrier is provided by the steel pressure vessels, which will be designed and constructed to ASME Section III Division 1 requirements. The chemically inert helium coolant minimizes corrosion and eliminates the need for the complications of steel internal cladding. The entire reactor module is protected by the underground RB from external events and is conservatively designed to accommodate internal events. The helium purification train is very effective at removing long-lived fission gases and contaminates from the primary coolant. However, for short-lived fission gases, the dominant removal mechanism is radioactive decay, and for the condensable fission products, the ***dominant removal mechanism is deposition, or plateout, on the various helium-wetted surfaces in the primary circuit***." (Sec. 5.1.1.2, p. 5-4) – ***Containment*** – "The reinforced concrete, vented low-pressure containment is the fifth barrier to the release of radionuclides. It is a normally closed space, located below grade. It is equipped with a vent that opens if the pressure inside the containment exceeds its design set point, releasing mass and energy associated with a blow down and protecting the integrity of the building and the RCCS. Even if the vent opens, natural removal mechanisms (including radioactive decay, condensation, fallout, and ***plateout***) reduce the concentration of radionuclides in the containment atmosphere, reducing the offsite releases. While the vent allows the release of radionuclides released promptly, the release of associated gases early in the event eliminates the driving pressure that could transport the delayed source term out of the building. After release of the initial blow-down energy pulse, the vent is designed to close for containment of radionuclides that might diffuse out of the fuel during time-at-temperature conditions. Robust design features protect the containment function from degradation by external events. Inclusion of a broad spectrum of DBEs protects the containment function from damage by internal events."	and hot duct) C.12.01 (materials properties for reactor vessel) C.16.00.03 (integrated performance of RCCS)	
C-5	Computational software or other methods for determining the quantitative results • Data collection and proof that the selected model is adequate under all the normal and accident conditions of interest. Need to know that model envelops releases, and have reasonable proof that the model predicts an upper limit. • Need to have a description of the physical models and the reactor configuration, showing that the models are appropriate for the conditions of interest. • Need to have the data required for the models: single-effects data for each material and component acquired	See item A-8 for a description of Design Methods Development and Validation. See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3, p. 3-1) – Plant Technical Description – "This Section provides a technical ***description of the entire NGNP plant***, including the nuclear systems, the Power Conversion System (PCS), the Heat Transport System (HTS), the hydrogenproduction facilities, the Helium Services System, the Plant Operation and Control System, and the balance of plant (BOP). The nuclear systems include the Reactor System, the Vessel System, the Shutdown Cooling System (SCS), the Fuel Handling System, and the Reactor Cavity Cooling System (RCCS)." (***Following***	C.07.02.07 (fuel testing) C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport)	Needs for computer model development and testing have been recognized in the General Atomics PCDR. Reactor configuration is available in the PCDR.

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	under individual testing, and integral data designed to show that the codes get the correct answer for a complete system under the conditions of interest.	***subsections describe components, systems and configurations in detail.)*** (Sec. 7.2.3.1, p. 7-13) – ***Radionuclide Transport*** – "As indicated in the PPMP, there is a substantial risk that the RN transport work scope included in the AGR Plan will be inadequate to support NGNP design and licensing. This problem has been exacerbated by chronic funding shortfalls for the AGR Fuel Program; consequently, no experimental work in the RN transport area has been initiated to date with the exception that the driver fuel has been fabricated for irradiation tests AGR-3 and AGR-4. In fact, no experimental work on RN transport outside of the core is planned until FY12. The significant RN transport issues identified with the AGR Plan are summarized below. • A ***series of fission product transport tests in an in-pile loop are needed*** in order to generate the integral test data necessary to validate the predicted source terms for the NGNP. The AGR Plan contains tasks to construct an in-pile loop and to perform an in-pile test program. However, the design and construction of the loop are not initiated until FY13. The technical feasibility of constructing such a facility (presumably in the ATR) and the associated costs and schedule must be established far earlier if the design methods for predicting RN transport in the primary circuit are to be validated before the end of NGNP final design. The cost and schedule estimates for loop design and construction appear to be very optimistic. • The AGR Plan does not address tritium transport (perhaps, in part, because it is a generic development plan which does not focus on a specific reactor design). ***Tasks to characterize tritium retention in the core and tritium permeation through heat exchanger materials*** need to be added to address NGNP DDNs. • The ***AGR Plan does not address RN transport in the VLPC***. It only includes an evaluation of the extent to which the experimental water-reactor database for radionuclide transport in high-pressure containment buildings might be applicable to the VLPC. A recent evaluation concluded that these data are of limited value for refining and independently validating the design methods used to predict radionuclide transport in VLPCs because the radionuclide concentrations and the physical and chemical forms in the two systems are too different. As a result, new DDNs have been identified that the AGR Fuel Program needs to address."	C.11.01.11 (neutron control assembly test) C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development, core testing) C.11.04.04 thru C.11.04.06 (ISIs and Surveillances for reactor internals and core supports) N.13.02.06 thru N.13.02.09 (IHX tests) C.14.01.02 thru C.14.01.04 C.14.04.01 thru 14.04.05, and 14.04.07 (SCS tests) C.16.00.02 (RCCS test)	

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
			C.21.01.05 (fuel handling test)	
C-6	Integral testing over a wide range of conditions to support the development of computational methods and the quantification of the data and associated uncertainties • Attempt to use existing data from past programs to the degree appropriate. • Planning of any in-pile loop program would require a complete description of the normal operating environment and of the accidents, along with any scaling factors. Extensive modeling will be necessary to design the loop and determine off-normal conditions that the loop can be expected to simulate. Model predictions (with the previously collected single-effects data) will need to be made.	(Sec. 7.4, p. 7-24) – ***Initial Testing and Inspection Program*** – "A testing and inspection program is proposed to be carried out at the start of NGNP operations. The testing and inspection program, as currently envisioned, is expected to be performed over a period of approximately one year prior to startup and two years following startup. The general objective of the testing, beyond qualification of the facility for power operation, is to effectively compress the operating time by inducing events that would not normally be expected to occur during a two year operating period, to support the following NGNP Project objectives: • Demonstrating the basis for commercialization of the nuclear system, the hydrogen production facility, and the power conversion concept. Essential elements of this objective include: ○ Demonstrating that the requisite reliability and capacity factor can be achieved over an extended period of operation. ○ Demonstrating normal O&M activities including activities required during major outages or equipment replacement or maintenance as well as O&M that might be required in the event of major equipment failures. • Establishing the basis for licensing the commercial version of NGNP by the NRC. This will be achieved in major part through licensing the prototype by NRC and initiating the process for certification of the nuclear system design. The proposed testing and inspections to be performed are divided into the following categories: ***Preoperational Tests*** – These tests address the capability of selected SSCs to meet performance requirements, to the extent they can be tested outside of full plant service conditions. Successful completion of preoperational tests demonstrates that individual system performance is acceptable and the plant is ready for hot functional tests. The preoperational tests and inspections to be performed will be specified in the SSC System Design Description (SDD) documents ***Baseline In-service Inspection*** – These are pre-operational tests of all the in-service-inspections (ISI) to be performed through out the plant's lifetime. These tests provide baseline data for comparison with future in-service inspection results. ***Hot Functional Tests*** – In these tests, the nuclear heat supply facility (the reactor primary system) will be operated at full power reactor gas inlet temperature, flow, and helium pressure with heat supplied by motoring the helium compressor and IHX	C.07.02.07 (fuel testing) C.11.01.11 (neutron control assembly test) C.11.03.45 and C.11.03.46 (core crossflow and core fluctuation tests) C.11.04.04 thru C.11.04.06 (ISIs and Surveillances for reactor internals and core supports) N.13.02.06 thru N.13.02.09 (IHX tests) C.14.01.02 thru C.14.01.04 C.14.04.01 thru 14.04.05, and 14.04.07	Needs for testing have been recognized in the General Atomics PCDR.

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		circulator. The tests will provide data on flow performance through out the primary system (pressures, temperatures, vibrations, etc) as well as functional testing of all monitoring instrumentation. In addition, a first check on vessel heat and temperature management and operation of the RCCS will be provided. ***Fuel Loading*** – As fuel loading progresses, neutron flux monitoring results can be compared with predictions. ***Startup Tests*** – Startup testing includes pre-critical, low power, and power ascension testing. Following verification of the core physics design, power is increased in steps to full power operation. Plant operating parameters will be verified to be within design limits, and response to load changes, transition of loads between the PCS and the hydrogen production plants and reactor trips will be demonstrated throughout the power ascension program. ***Performance Tests*** – These tests will subject the plant to less frequent events expected to occur during normal operation including power PCS trip, loss of secondary system flow or pressure, etc. ***Response to Accident Tests*** – These tests are intended to demonstrate the inherent response characteristics of the reactor module. Four basic categories of events are proposed: (1) reactivity transients, (2) pressurized cool down, (3) water ingress, and (4) depressurized cool down. These categories cover the performance of the key systems which provide safety and investment protection ***Post Test Inspections and Maintenance Demonstrations*** – Following the completion of the above testing at power operating conditions, a shutdown would be scheduled for performance of inspections and to demonstrate major maintenance operations. Inspections would be performed of all the systems to ascertain any abnormal effects of the above tests. Major maintenance operations would be demonstrated such as refueling, reflector replacement, performance of remote ISI operations, and removal and replacement of major equipment items such as a TM rotor, IHX heat transfer element, major hydrogen production equipment and other plant items not designed for the life of the plant. Although preliminary planning indicates that the response to ***accident testing*** will comprise only a small fraction of the total testing interval, the tests are a major element of the total program. The tests to be performed have been developed based on a preliminary evaluation, and will be adjusted based on further evaluation of design and licensing issues as the project proceeds. The ***ability to demonstrate the response to low probability events in a full scale plant without damage which***	(SCS tests) C.16.00.02 (RCCS test) C.21.01.05 (fuel handling test)	

Table 1C (GA) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		would preclude subsequent long term operation is a key feature of the modular helium-cooled reactor. Demonstrating this capability is a vital element in the successful development of a commercial plant which is economically competitive, and generally accepted by utility/users, the financial community, and general public." (Sec. 7.2.3.5, p. 7-16) **-** ***Design Verification & Support Programs –*** **"**The base technology for designing most MHR SSCs derives from five decades of international R&D programs combined with the design, construction, and operation of seven He-cooled reactors. Nevertheless, there are ***design-specific features of some SSCs that will require design verification by testing*** with semi-scale mockups or with actual prototypical components. Such testing is referred to herein as design verification and support (DV&S). The current NGNP and NHI technology development programs are largely generic because there is no reference NGNP design. Many fundamental design selections have yet to be made, e.g., reactor core type, IHX configuration, hydrogen production process, etc. Consequently, the current TDPs do not address DV&S DDNs to a significant degree. When the reference NGNP design is chosen, additional TDPs will need to be prepared that address the DV&S DDNs for key SSCs. It is expected that new design-specific TDPs will include plans for the Reactor System, Vessel System, RCCS, etc. Additional validation of the nuclear design methods will probably be needed for licensing the MHR design because of its annular core, which uses reflector control rods, and because of its reliance on inherent safety features in contrast to engineered safeguards. Conduct of new critical experiments, especially at elevated temperatures, will be problematic because no test facility currently exists in the U.S. A viable option would be to perform the tests in a foreign facility."		

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable General Atomics R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
D-1	*Physical Materials Data* - Requirements for physical aspects to be included in modeling high-temperature metallic components: • Inelastic materials behavior for materials, times, and temperatures for very high temperature structures (e.g., creep, fatigue, creep-fatigue). • Adequacy and applicability of current ASME Code allowables with respect to service times and temperatures for operational stresses. • Adequacy and applicability of current state of high-temperature design methodology (e.g., constitutive models, complex loading, failure criteria, flaw assessment methods). • Effects of product form and section thickness. • Joining methods including welding, diffusion bonding, and issues associated with dissimilar materials in structural components. • Effects of irradiation on materials strength, ductility, and toughness. • Degradation mechanisms and inspectability. • Oxidation, carburization, decarburization, and nitriding of metallic components in impure helium and helium-nitrogen. • Micro-structural stability during long-term aging in environment. • Effects of short and long term on mechanical properties (e.g., tensile, fatigue, creep, creep-fatigue, ductility, toughness). • High-velocity erosion/corrosion. • Rapid oxidation of graphite and carbon-carbon composites during air-ingress accidents. • Compatibility with heat-transfer media and reactants for hydrogen generation. • Development and stability of surface layers on RPV and core barrel affecting emissivity.	Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine. (Sec. 3.1.2.2, p. 3-16) – "Other design modifications that have been investigated include modifications to the reactor internal design to reduce bypass flow and modifications to the fuel-element design to enhance heat transfer. In addition, fuel shuffling strategies have been investigated that can reduce power peaking factors. ***These modifications can provide additional margin for fuel temperatures during normal operation, and may allow additional reduction of the coolant inlet temperature, such that SA-533/SA-508 steel (used for LWR reactor vessels) could be used for the NGNP reactor vessel***." (Sec. 3.1.4.1, p. 3-65) – ***Fuel Failure Mechanisms*** – "A number of failure mechanisms have been observed during irradiation testing and post-irradiation heating of coated-particle fuels, including pressure-vessel failure, kernel migration, and ***corrosion of the SiC layer by fission products… corrosion of the SiC layer by fission products is a key factor for determining limitations on fuel temperatures."*** (Sec. 3.2.1, p. 3-79) – ***Selection of Vessel Materials*** – "The reference GT-MHR design selected 9Cr-1Mo-V steel for the reactor vessel. Although this material was developed for Liquid Metal Fast Breeder Reactor applications, its American Society of Mechanical Engineers (ASME) code qualification had not been completed. ***GA material specialists have recommended against using 9Cr-1Mo-V steel*** for the NGNP, primarily due to expected welding difficulties and lack of manufacturing and operating experience. Although the primary coolant temperature for the NGNP is higher than that for the GT MHR, the ***alternative studies discussed in Section 3.1.2.2 indicate the reactor vessel temperatures can be maintained within limits that allow selection of a vessel material having temperature limits lower than 9Cr-1Mo-V steel***." Sec. 3.2.2, p. 3-80) – ***Reactor Vessel*** – "The ***material selected for the reactor vessel for the NGNP pre-conceptual design is 2¼Cr-1Mo steel***. As discussed in Section 3.1.2.2, a design alternative to incorporate cooling of the reactor vessel is being considered, which could potentially lower reactor vessel temperatures to a level that would ***allow use of proven light water reactor vessel materials (e.g., SA508/SA533 steel***). The reactor vessel design parameters are given in Table 3.2-1. The manufacturer of LWR vessels makes considerable use of SA508 forgings. GA	C.11.02 and C.11.03 (reactor internals, hot duct and core) C.11.04.04 and C.11.04.05 (design verification for metallic reactor internals and core supports) C.12.01 (materials properties for reactor vessel, including heavy sections) N.13.01 and N.13.02 (various IHX tests and materials research) N.41.01.01, N.41.01.02, N.41.01.03, and N.41.02.01 (materials data for high heat power conversion system	The General Atomics PCDR has captured most of the needs detailed in this item, ***with the following exceptions:*** • ***Degradation mechanisms and inspectability.*** • ***Micro-structural stability during long-term aging in environment.*** • ***High-velocity erosion/corrosion.*** • ***Development and stability of surface layers on RPV and core barrel affecting emissivity***

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable General Atomics R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		has had discussions with two reactor vessel manufactures concerning NGNP vessel fabrication, specifically Japan Steel Works (JSW) and DOOSAN Heavy Industries and Construction (DOOSAN). The current maximum cylindrical forging size is limited to 8.2 m diameter. ***As an alternative approach to forgings, GA material experts suggest manufacturing the reactor vessel from rolled plate***, or a combination of rolled plant and forgings. Manufacturing schemes for both the forgings (seam plan) and rolled plate designs for the reactor vessel as provided by DOOSAN are shown in Figures 3.2-2 and 3.2-3, respectively." (Sec. 3.2.3, p. 3-82) – "***The NGNP hot duct material will be a high temperature alloy (e.g., Incoloy 800H, Hastelloy-XR, or Inconel 617***). The cross vessel is a cylindrical vessel designed and fabricated according to Section III of the ASME Code. It has an inner diameter of 2.29 m, a wall thickness of 7.62 cm, and is approximately 2.86 m in length. ***The material selected for this cross vessel for the NGNP pre-conceptual design is 2¼Cr-1Mo steel***. As discussed in Section 3.1.2.2, a design alternative to incorporate cooling of the reactor vessel is being considered, which could potentially lower reactor vessel temperatures to a level that would ***allow use of proven light water reactor vessel materials (e.g., SA508/SA533 steel***). If this alternative is selected, the cross vessel would also likely be manufactured using the same material." (Sec. 3.2.4, p. 3-83) ***– Power Conversion Vessel – "The material selected for the PCS vessel for the NGNP preconceptual design is SA508/SA533 steel***. However, if further evaluation concludes that higher temperature material is necessary, then 2¼Cr - 1Mo would be used for the PCS vessel as well as the reactor vessel. The PCS vessel has an inner diameter of 7.5 m, a wall thickness of 152 mm, and is approximately 35.2 m in height. Details of the PCS vessel are given on Table 3.2-2." (Sec. 3.2.5, p. 3-84) – ***IHX Vessel*** – "The IHX vessel is a pressure boundary for the primary helium coolant and will be designed according to Section III of the ASME Code. ***The material selected for the IHX vessel for the NGNP preconceptual design is 2¼Cr-1Mo steel***. The IHX vessel may include a ceramic fiber insulation layer on inside surfaces to maintain operating temperatures within the material temperature limits. The vessel has an inner diameter of 3.81 m and is approximately 16 m in height." (Sec. 5.1.1.2, p. 5-4) – ***Primary Coolant Pressure Boundary*** – "The fourth release barrier is the primary coolant pressure boundary. This barrier is provided by the steel pressure vessels, which will be designed and constructed to ASME Section III Division 1 requirements. The chemically inert helium coolant minimizes corrosion and eliminates the need for the complications of steel internal cladding. The entire reactor module is protected by the underground RB from external events and is	components)	

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		conservatively designed to accommodate internal events. The ***helium purification train is very effective at removing long-lived fission gases and contaminates from the primary coolant***. However, for short-lived fission gases, the dominant removal mechanism is radioactive decay, and for the condensable fission products, the dominant removal mechanism is deposition, or plateout, on the various helium-wetted surfaces in the primary circuit." (Sec. 7.2.1.1, p. 7-5) – ***High Temperature Materials*** – "Structural metals will be used throughout the primary coolant circuit of the NGNP, including the reactor internals, hot ducts, and heat exchangers. When the first HTGRs were designed, it was obvious that the metallic components would operate at high temperature and that some would be exposed to high neutron doses as well. The environmental aspect that was not fully anticipated until the first prototype HTGRs were operated was the extent to which the reactor primary coolant chemistry could vary. The design of the reactor metal components is based on the ASME Code with conservative reductions in Code allowables based on existing data relative to environmental effects on the various alloys. Since the early 1960s, numerous test programs and experiments have been conducted in support of metals technology for HTGRs. Extensive laboratory testing, using a range of temperatures and helium impurity levels, has been carried out in the U.S., Europe, and Japan over the past three decades to verify the performance of a variety of high-temperature materials in helium environments expected for HTGR systems. Test materials included wrought alloys such as 2¼Cr-1Mo steel, Alloy 800H, Hastelloy X, Inconel 617 (IN 617) and other metals. ***The greatest materials challenge for NGNP design will be to qualify a metal for the IHX which can operate at 950°C with a long lifetime (IN 617 is the leading candidate)***. The Japanese HTTR has an IHX made of Hastelloy XR. This IHX has been designed to operate at 950°C with a lifetime of 10 years." (Sec. 7.2.1.2, p. 7-7) – ***Hydrogen Production/SI Process*** – "The highly corrosive nature of chemical streams in the SI process has led to significant research work in the area of materials compatibility. Early screenings showed that alloys of tantalum appeared suitable, and ***current work is exploring long-term performance and corrosion resistance of materials*** stressed or machined in ways that materials of construction for larger scale plants will experience." (Sec. 7.2.3.2, p. 7-14) **- *Structural Materials R&D Program* – "**The objective of the NGNP Materials R&D Program [NGNP Materials Program 2005] is to provide the essential materials R&D needed to support the design and licensing of the NGNP, excluding the hydrogen plant. The most important products of the program will be qualified nuclear graphite for the reactor core and high temperature metals for use throughout the nuclear heat source, PCS, primary HTS, and balance of plant. The GA Team perspective on the graphite and metals program is briefly summarized		

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		below…***High Temperature Metals* -** The metals program described in [NGNP Materials Program 2005] is evaluating a large number of alloys for high temperature applications throughout the Reactor System, PCS, and Primary HTS. With an important exception, the planned program appears responsive to the structural metals DDNs defined herein for a prismatic NGNP, but from the GA Team's perspective it may be excessive. Because the reference NGNP design has not been chosen, the current materials R&D program is necessarily a generic program. Once the reference design is determined, the metals R&D program needs to be focused on a relatively few alloys (e.g., a prime and a backup alloy for each application). To that end, a ***comprehensive, stand-alone metals TDP should be prepared*** that defines the entire scope (test matrices, etc.), schedule, and cost of the planned program. A high-priority task will be to complete qualification of IN 617 for an IHX operating at 950°C. An important deficiency in the current metals R&D program is that it does not include turbine blade alloys. There is considerable incentive to develop and qualify a turbine-blade alloy that can be used without blade cooling at 950°C with an acceptable service life20. The turbine blade alloy R&D program should emphasize helium effects as well as thermal fatigue, and the threshold concentrations and temperatures for possible ***corrosion of turbine alloys by radionuclide plateout (Te, Cs, Ag) should be investigated***." (Sec. 7.2.3.3, p. 7-15) ***- Energy Transfer Technology Program –* "**The GA Team understands that an Energy Transfer TDP will be prepared [PPMP 2006]. Presumably, it will emphasize the ***design and qualification of an IHX*** capable of operating at 950°C for long life times (several decades). While some DDNs related to the IHX are generic (e.g., the materials DDNs that will be addressed by the materials R&D program), other DDNs are design specific (e.g., printed circuit vs. helical coil, etc.); consequently, a ***reference conceptual design for the IHX*** is urgently needed to provide direction and priority to the energy transfer R&D programs. This Energy Transfer TDP will also need to address DDNs related to ***process heat exchangers (hydrogen plants), piping insulation, isolation valves, and high temperature circulators."***		
D-2	*Physical Materials Data (Composites)* - Requirements for physical aspects to be included in modeling high-temperature structural composites, such as carbon-carbon or silicon carbide–silicon carbide: • Effects of composite component selection and infiltration method. • Effects of architecture and weave. • Materials properties up to and including very high temperatures (e.g., strength, fracture, creep, corrosion, thermal shock resistance). • Effects of irradiation on materials strength and	The ***composite material-related concerns stated in this item are not specifically addressed in the PCDR***. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, graphite components, and the turbine. ***Appendix B to the PCDR contains the NGNP schedule. Per that schedule, composites will be codified as part of the overall ASME/ASTM codification effort for graphites***. The structural materials R&D program for graphites is described in section 7.2.3.2, p. 7-14. (Sec. 3.1.2, p. 3-12) – Reactor Core and Internals Design – "A control rod design	C.11.01.03 (control rod design verification) C.11.03.02 thru C.11.03.06 (control rod failure modes and integrity)	Although a program to address issues associated with composite materials is not specifically addressed, it is included by reference as a part of the program to qualify graphites, in the General Atomics PCDR.

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	dimensional stability. • Fabrication scaling processes. • Adequacy and validation of design methods. • Degradation mechanisms and inspectability.	using a ***carbon-carbon composite for the cladding material*** is being evaluated that would allow the in-core rods (or control rods located in the inner reflector) to be used during normal operation, which will provide greater flexibility for flattening the radial power distribution and provide some additional margin for maintaining fuel temperatures and fuel performance within acceptable limits." (Sec. 3.1.2.2, p. 3-27) – Bypass Flow Reduction – "Fuel temperatures can be reduced by reducing bypass flow. Bypass flow is defined as any flow that bypasses the coolant holes of the fuel elements. As shown in Figure 3.1-20, bypass flow channels include gaps between fuel columns and leakage between/from PSR blocks. For the reference GT-MHR core design, approximately 3% of the flow is supplied to the control-rod channels, which have orifices to minimize bypass flow while also maintaining adequate cooling for the control rods. ***Composite-clad control rods require little or no cooling, which helps reduce the bypass flow fraction***." (Sec. 3.1.3, p. 3-61) – Neutron Control System – "The neutron absorber material consists of B4C granules uniformly dispersed in a graphite matrix and formed into annular compacts. The boron is enriched to 90 weight percent B-10 and the compacts contain 40 weight percent B4C. The compacts have an inner diameter of 52.8 mm, an outer diameter of 82.6 mm, and are enclosed in Incoloy 800H canisters for structural support. ***Alternatively, carbon-fiber reinforced carbon (C-C) composite canisters may be used for structural support***. The control rod consists of a string of 18 canisters with sufficient mechanical flexibility to accommodate any postulated offset between elements, even during a seismic event."	C.11.03.24 (control rod high temp materials properties)	
D-3	Compromise of RPV surface emissivity due to loss of desired surface layer properties. Compromise of emissivities of in-vessel surfaces.	(Sec. 3.1.2.2, p. 3-43) – "A 30-deg. sector ANSYS model was used to analyze both low-pressure conduction cooldown (LPCC) and high-pressure conduction cooldown (HPCC) events. In order to reduce vessel temperatures during these accidents, the reactor internal design was modified to include a 100-mm layer of carbon insulation on the outer radial boundary of the PSR…A key parameter for these calculations is the ***graphite thermal conductivity***, which decreases with damage caused by neutron irradiation. For these studies, calculations were performed using both irradiated and unirradiated graphite properties. Calculations were also performed assuming ***annealing of irradiation damage*** as the graphite temperature increases according to the GA model for H-451 graphite. Full recovery from irradiation damage is assumed to occur at temperatures greater than 1300°C. The ANSYS model shown in Figure 3.1-41 was used to calculate the effective thermal conductivity of the graphite blocks. ***Other key parameters that affect heat transfer to the RCCS are the emissivities of the PSR, core barrel, RPV, and RCCS panels***…" General recommendations for a high temperature metals R&D program are addressed in section 7.2.3.2, page 7-14.	N.11.02.16 (reactor internals emissivity) C.12.01.06 (reactor vessel emissivity) C.16.00.01 thru C.16.00.06 (RCCS)	The importance of surface emissivities, including analytical efforts performed to date, have been recognized in the General Atomics PCDR.

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
D-4	Effects on insulation • Aging fatigue and environmental degradation of insulation materials (debris plugging). • Environmental and irradiation degradation/thermal instability of fibrous insulation	(Sec. 7.2.3.3, p. 7-15) ***- Energy Transfer Technology Program –*** "The GA Team understands that an Energy Transfer TDP will be prepared [PPMP 2006]. Presumably, it will emphasize the design and qualification of an IHX capable of operating at 950°C for long life times (several decades). While some DDNs related to the IHX are generic (e.g., the materials DDNs that will be addressed by the materials R&D program), other DDNs are design specific (e.g., printed circuit vs. helical coil, etc.); consequently, a reference conceptual design for the IHX is urgently needed to provide direction and priority to the energy transfer R&D programs. This Energy Transfer TDP will also need to address DDNs related to process heat exchangers (hydrogen plants), ***piping insulation***, isolation valves, and high temperature circulators." ***There is no indication that this item has been specifically addressed.***	N.11.02.14 (fibrous insulation properties) N.11.02.15 (hard ceramic insulation properties) N.13.02.07 (IHX insulation tests) C.14.04.01 (SHE insulation tests)	***There is no indication that this item has been specifically addressed in the General Atomics PCDR.***
D-5	Primary boundary failures in compact IHX (roles of design methods, manufacturing controls, inspection/testing).	See item C-6 for a description of the Initial Testing and Inspection Program.	N.13.02.01 thru N.13.02.09 (IHX materials and design)	The need for design verification testing of critical components such as the IHX has been recognized in the General Atomics PCDR.
D-6	Control rod insertion failures (role of structural design methods for composites).	(TDP Sec. 3.3.1, p. 64) – Core Graphites – "The use of carbon/carbon (C/C) ***composites is proposed for several subcomponents in the control rod assembly***. The selection was based on limited data from ORNL's work on irradiated C/C composite for fusion energy applications. ***C/C composite, therefore, needs to be further characterized by testing and its compatibility in the reactor environment needs to be assessed before it can be qualified for use in the NGNP***."	C.11.01.03 (control rod design verification) C.11.03.02 thru C.11.03.06 (control rod failure modes and integrity) C.11.03.24 (control rod high temp materials	The needs to further characterize, test and qualify the composite material selected for control rod assemblies have been recognized in the General Atomics PCDR.

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
			properties)	
D-7	Irradiation induced creep of in-vessel metallic structures.	Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	N.11.02.11 (irradiation effects on metallic reactor internals)	The need to better understand the phenomenon of creep has been recognized in the General Atomics PCDR.
D-8	Core radial restraint failure (role of structural design and fabrication for composites).	(Sec. 3.1.2.1, p. 3-14) – "The ***lower graphite core support assembly*** consists of two layers of hexagonal elements support pedestals for the fuel and reflector columns that form the lower plenum, and the lower plenum floor, which consists of a layer of graphite elements and two layers of ceramic elements that insulate the metallic core support from the hot helium in the lower plenum. The ***upper core restraint*** elements have the same hexagonal cross sections as the graphite elements below them and are one-half the height of a standard fuel element. Dowel/socket connections are used to align the core-restraint elements with the graphite blocks. The core restraint elements are also keyed to each other and to the core barrel. The upper core restraint blocks provide stability during refueling and maintain relatively uniform and small gaps between columns during operation. The metallic core support includes a floor section and a core barrel that are welded together. The metallic core support is supported both vertically and laterally by the reactor vessel. The upper plenum shroud is a welded, continuous dome that rests on top of the core barrel to form the upper plenum. The upper plenum shroud includes penetrations for inserting control rods and reserve shutdown material, for refueling, and for core component replacement." ***There is no indication that this item has been specifically addressed.***	C.11.02.01 (core support strength data) C.11.02.11 and C.11.02.12 (helium, temp. and irradiation effects on metallic reactor internals) C.11.04.05 and C.11.04.06 (core support design verification)	***There is no indication that this item has been specifically addressed in the General Atomics PCDR.***
D-9	Isolation and other valve failures (self-welding, galling, seizing)	(Sec. 3.7.2.1, p. 3-149) – "Secondary HTS Piping and Isolation Valves - It is expected that the secondary heat transport loop will have ***three isolation valves on each leg*** – two near the IHX and one near the PHX. Isolation valves are necessary to prevent the propagation of events in either the NGNP reactor or hydrogen production plant from affecting the other. ***Double isolation valves on the hot leg and cold leg sides of the IHX allow these isolation valves to be part of the primary coolant pressure boundary and part of the containment building boundary***. Isolation valves are also necessary to perform maintenance on the heat transport loop. Figure 3.7-5 presents a diagram of a potential high temperature isolation valve (HTIV) being developed for use on HTTR by the Japan Atomic Energy Agency (JAEA). For HTTR, a ½ scale prototype of the HTIV has been tested. The valve, as shown in Figure 3.7-5, is an angle valve with internal glass wool insulation. The rod body and seat were made of Hastelloy X and the seat had a coating metal of Stellite No. 6 and 30 wt% Cr_3C_2. The casing of the valve was made of carbon steel which was limited to 350°C due to the internal insulation. Testing was performed at 4.0 MPa and 900°C."	C.14.01.04 (shutdown circulator loop shutoff valve test) N.42.02.01 and N.42.02.02 (sec heat transport isolation valves)	The applications and need for addressing issues associated with isolation valves have been addressed in the General Atomics PCDR.

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		(Sec. 7.2.3.3, p. 7-15) ***- Energy Transfer Technology Program –*** **"**The GA Team understands that an Energy Transfer TDP will be prepared [PPMP 2006]. Presumably, it will emphasize the design and qualification of an IHX capable of operating at 950°C for long life times (several decades). While some DDNs related to the IHX are generic (e.g., the materials DDNs that will be addressed by the materials R&D program), other DDNs are design specific (e.g., printed circuit vs. helical coil, etc.); consequently, a reference conceptual design for the IHX is urgently needed to provide direction and priority to the energy transfer R&D programs. This Energy Transfer TDP will also need to address DDNs related to process heat exchangers (hydrogen plants), piping insulation***, isolation valves,*** and high temperature circulators."		
D-10	Initiate development of the data and models needed by ASME Boiler and Pressure Vessel (B&PV) Code Subcommittees to formulate time-dependent failure criteria that will ensure adequate life and safety for metallic materials in the NGNP. These include obtaining the data necessary to develop experimentally based constitutive models for the NGNP construction materials, which are the foundation of the inelastic design analyses specifically required by ASME B&PV Sect. III Division I Subsection NH.	See item D-1 for information relating to ASME code development, and for information relating to technology development efforts required for high temperature metals.	C.07.04 (core corrosion data) C.11.02 and C.11.03 (reactor internals, hot duct and core) C.11.04.04 and C.11.04.05 (design verification for metallic reactor internals and core supports) C.12.01 (materials properties for reactor vessel) N.13.01 and N.13.02 (various IHX tests and materials research)	The needs for developing structural models and ASME Code qualification for high temperature metallic materials have been recognized in the General Atomics PCDR.

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
D-11	Safety assessments dependent on time-dependent flaw growth and the resulting leak rates from postulated pressure-boundary breaks will require a flaw assessment procedure capable of reliably predicting crack-induced failures, as well as the size and growth of the resulting opening in the pressure boundary.	***There is no indication that this item has been specifically addressed.*** However, general recommendations for a high temperature metals R&D program are addressed in section 7.2.3.2, page 7-14.	N.12.01.01 thru N.12.01.03 (RPV materials)	***There is no indication that this item has been specifically addressed in the General Atomics PCDR.***
D-12	Materials data and extrapolation procedures must be developed and guidance provided to ensure that allowable operation period and range of stress and temperature for materials of construction are extended to meet the proposed operating temperatures and lifetimes. Creep-fatigue rules are an area of particular concern for the materials and temperatures of interest and must be updated and validated. (example concern: RPV long-term thermal aging)	See item D-1 for information on efforts to develop structural models and ASME Code qualification for high temperature metallic materials. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine. The topic of creep is addressed for the reactor vessel in section 3.1.2.2, beginning on page 3-15.	C.11.02 and C.11.03 (reactor internals, hot duct and core) C.11.04.04 and C.11.04.05 (design verification for metallic reactor internals and core supports) C.12.01 (materials properties for reactor vessel) N.13.01 and N.13.02 (various IHX tests and materials research)	The needs for developing structural models and ASME Code qualification for high temperature metallic materials have been recognized in the General Atomics PCDR.
D-13	Since IHX sections must operate at the full exit temperature of the reactor, effort should be initiated to obtain data supporting the determination of the metallurgical stability and environmental resistance of IHX materials in anticipated impure helium coolant environments for the lifetimes anticipated.	(Sec. 3.9.1, p. 3-189) – ***Primary Coolant Purification System*** – "This subsystem provides a means to remove circulating impurities from the primary coolant helium, and to transfer those impurities to the radioactive liquid and gas waste systems of the facility. A separate regeneration section within this subsystem is used to remove the impurities that accumulate in the purification subsystem adsorbers. The regeneration section is operated periodically under automatic control whenever regeneration is required. The primary coolant helium purification subsystem consists of two separate, independent, but identical trains of components as shown in Figure 3.9-1. All of the components that make up the trains are mechanically passive in nature; however, the	N.13.02.01 (effects of helium and temp on IHX)	The need for high purity helium is addressed in the General Atomics PCDR in the design.

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		adsorber elements become radioactive as the removed impurities are concentrated within the various media. Each purification train must therefore be located in a shielded vault to minimize personnel exposure to radiation. Helium purification is accomplished by routing a small side stream of helium from the primary coolant system through a series of purification components. ***These components remove the following chemical impurities: Br, I, H2O, CO, CO2, H2 (including Tritium), N2, O2, H2S, Kr, Xe, CH4, and other hydrocarbons*.**"		
D-14	Work should be initiated to quantify crack initiation and propagation in the IHX due to creep, creep-fatigue, and aging. These materials-related phenomena related to the IHX were identified for potentially contributing to FP release at the site boundary.	See item D-1 for information on efforts to develop structural models for high temperature metallic materials. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine. The topic of creep is addressed for the IHX in section 1.5.3, page 1-21.	N.13.02.01 thru N.13.02.09 (IHX materials)	The need to better understand the phenomenon of creep has been recognized in the General Atomics PCDR.
D-15	Specific issues must be addressed for RPVs that are too large for shop fabrication and transportation. Validated procedures for on-site welding, PWHT, and inspections must be developed for the materials of construction. For vessels using materials other than those typical of LWR construction to enable operation at higher temperatures, confirmation of their fabricability (especially, effects of forging size and weldability) and data on their irradiation resistance is needed. Three materials-related phenomena related to the RPV fabrication and operation were identified for potentially contributing to FP release at the site boundary, particularly for 9Cr–1 Mo–V steels capable of higher-temperature operation: crack initiation and subcritical crack growth, process control to avoid material degradation during field fabrication, and property control in heavy sections.	(Sec. 3.2.2, p. 3-80) – ***Reactor Vessel*** – "The manufacturer of LWR vessels makes considerable use of SA508 forgings. GA has had discussions with two reactor vessel manufactures concerning NGNP vessel fabrication, specifically Japan Steel Works (JSW) and DOOSAN Heavy Industries and Construction (DOOSAN). The current maximum cylindrical forging size is limited to 8.2 m diameter. As an alternative approach to forgings, ***GA material experts suggest manufacturing the reactor vessel from rolled plate***, ***or a combination of rolled plant and forgings***. Manufacturing schemes for both the forgings (seam plan) and rolled plate designs for the reactor vessel as provided by DOOSAN are shown in Figures 3.2-2 and 3.2-3, respectively."	N.12.01.02 (RPV heavy sections)	The General Atomics PCDR contains options for fabricating the reactor vessel, including forging or welding together sections of rolled plate. ***There is no indication that the need to research issues relating to vessels too large for shop fabrication has been specifically addressed in the General Atomics PCDR.***
D-16	For high-temperature metals technology, there is a need for analytical models, in particular for developing time-dependent design criteria for complex structures, along with verification by structural testing. ASME Code-approved simplified methods have not yet been proven and are not permitted for compact IHX components. Analytical modeling of carbon-carbon composite behavior would be useful in developing approved methods for designing, proof testing, model standard testing, validation tests, and probabilistic methods of design. Scalability and fabrication issues must be addressed, including large-scale structures (meters in	See item A-8 for information relating to development of analytical models. See item D-1 for information relating to ASME code approval of high temperature metals.	N.13.02.01 thru N.13.02.09 (IHX materials)	The needs for analytical models and ASME Code qualification of high temperature metals have been recognized in the General Atomics PCDR.

Table 1D (GA) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	diameter), as well as smaller structures.			

Table 1E (GA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
E-1	Lack of confirmatory data for the grades of graphite selected by potential NGNP vendors. This situation has occurred because: • Graphite grades used in prior HTGRs are no longer available, and thus development of new grades has been required. • Increased temperature of the NGNP compared to prior graphite-moderated reactors. • In the case of the PBR, the larger neutron dose that the core components will experience compared to that of previous HTGRs licensed in the United States.	(Sec. 7.2.1.1, p. 7-5) – ***High Temperature Materials*** – "The design of the NGNP graphite components is based on a considerable international body of graphite data. In the early 1970's, a near-isotropic, petroleum coke based graphite, designated Grade H-451, was developed by Great Lakes Carbon, and numerous test programs and experiments were conducted to characterize its behavior. H-451 was used successfully in FSV reloads, and it was the reference fuel element graphite for the NP-MHTGR. Unfortunately, this graphite is ***no longer commercially available***, and a ***priority task for the NGNP technology program is to identify and qualify a replacement having comparable properties***. The component models and material property data for designing graphite components are documented and controlled in the GA Graphite Design Data Manual. It is planned to use these data in the conceptual design (and perhaps preliminary design) of the NGNP core until a replacement for H-451 graphite is characterized." (Sec. 7.2.3.2, p. 7-15) **-** ***Structural Materials R&D Program*** **– "**The objective of the NGNP Materials R&D Program [NGNP Materials Program 2005] is to provide the essential materials R&D needed to support the design and licensing of the NGNP, excluding the hydrogen plant. The most important products of the program will be qualified nuclear graphite for the reactor core and high temperature metals for use throughout the nuclear heat source, PCS, primary HTS, and balance of plant. The GA Team perspective on the graphite and metals program is briefly summarized below…***Graphite Program*** **-** The graphite program described in [NGNP Materials Program 2005] is evaluating at least 16 nuclear graphites and fuel-element matrix materials from at least four international graphite vendors. The current focus of the program is the graphite irradiation capsule AGC-1 which is intended to provide irradiation creep design and dimensional change data on candidate graphites for the NGNP program. Creep data will be obtained for six major graphite grades: H-451 and IG-110, both of which are included as reference graphites, and four new grades, PCEA, NBG-17, NBG-18, and IG-430. In addition, AGC-1 contains ten minor grades of graphite. A comprehensive, stand-alone graphite TDP is needed to define the entire scope, schedule and cost of the planned program. The planned program is probably responsive to the graphite DDNs defined herein for a prismatic NGNP, but from the GA Team's perspective, it may be excessive. The graphite service conditions in a prismatic VHTR are not demanding (e.g., fast neutron fluence to the fuel element <5 x 1021 n/cm2, E >0.18 Mev). Previously qualified H-451 for fuel and reflector elements and Stackpole 2020 for the core support structure have adequate material properties. The primary need is to identify and qualify a replacement for H-451. The recommended approach is to use AGC 1 as a screening capsule to identify the lowest-cost graphite with properties comparable to H-451 and then to perform supplemental testing to establish a correspondence between the behavior of the replacement graphite and the extensive H-451 experience base. The GA Team considers the qualification of a replacement graphite for H-451 to be a high priority,	C.11.03.11 thru C.11.03.23 (graphite materials characterizatio n)	The need for ASME Code qualification of graphites has been recognized in the General Atomics PCDR.

Table 1E (GA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		but a low risk task." The portion of this item that is specific to the PBR is not applicable to the PMR.		
E-2	Lack of consensus codes and standards. Efforts are under way through the ASME to develop a consensus design code for graphite core components, but to date a useable code has not been approved. ASTM test standards exist for many of the physical properties of concern to the reactor designer, but further work is required, especially in the area of small (irradiation) specimen test methods.	See item E-1 for information relating to ASME Code qualification of graphites.	C.11.03.11 thru C.11.03.23 (graphite materials characterization)	The need for ASME Code qualification of graphites has been recognized in the General Atomics PCDR.
E-3	Theoretical models for the effects of neutron damage on the properties of graphite have been developed, however, these models need modification for the new graphites and will need to be extended to higher temperatures and/or higher neutron doses. V&V of theoretical models will require generation of experimental data on the effect of neutron irradiation on properties.	See item A-8 for information relating to development of analytical models. See item E-1 for information relating to ASME Code qualification of graphites, and for structural materials R&D relating to graphite.	C.11.03.11 thru C.11.03.23 (graphite materials characterization)	The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the General Atomics PCDR.
E-4	Uncertainties in the temperature and dose received by a component; the severity of temperature and dose gradients in a component; the rate of dimensional change in the specific graphite used in a given design; the extent to which stresses are relieved by irradiation-induced creep; and the extent of changes in key physical properties such as elastic moduli, thermal conductivity, coefficient of thermal expansion, compound to make the prediction of component stress levels, and hence decisions regarding component lifetime and replacement schedules, very imprecise.	See item E-1 for information relating to ASME Code qualification of graphites, and for structural materials R&D relating to graphite. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	C.11.03.11 thru C.11.03.23 (graphite materials characterization)	The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the General Atomics PCDR.
E-5	Whole-core models are required that can predict the stress states of graphite components within the core. Such models should be capable of taking inputs such as temperature and neutron dose and calculating the dimensional change, creep, thermal conductivity, etc., from established theoretical models. Reliable stress-state predictions as a function of reactor life would enable reactor operators and regulators to provide NDE guidance and make decisions regarding inspection intervals and core block replacement.	See item A-8 for information relating to development of analytical models. Section 3.1.2, Reactor Core and Internals Design (pages 3-12 through 3-61), contains detailed descriptions of the software used to model the reactor system thus far. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport) C.11.03.41 thru C.11.03.46, C.11.03.51 and	Reactor core analyses performed to date and the need for further analytical models have been described in the General Atomics PCDR.

Table 1E (GA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
			C.11.03.52 (core physics data development) C.11.03.11 thru C.11.03.23 (graphite materials characterization)	
E-6	Basic research should be conducted to strengthen the understanding and modeling capability of the displacement damage process in graphite. In addition, in graphite technology, there is a need for analytical models for oxidation, changes in physical properties, irradiation induced dimensional change, and irradiation creep. They could be developed to feed into a structural integrity model for the graphite core which would be used for core design and safety assessment.	See item A-8 for information relating to development of analytical models. See item E-1 for information relating to materials characterization and qualification of graphites. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	C.07.03.01 thru C.07.03.07, C.07.03.09 thru C.07.03.22 (fission product transport) C.11.03.41 thru C.11.03.46, C.11.03.51 and C.11.03.52 (core physics data development) C.11.03.11 thru C.11.03.23 (graphite materials characterization)	The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the General Atomics PCDR.
E-7	Irradiation induced change in the coefficient of thermal expansion, including effects of creep strain.	See item E-1 for information relating to materials characterization and qualification of graphites. Concerns over the creep phenomenon due to operation at high temperatures and	C.11.03.11 thru C.11.03.23 (graphite	The needs for further materials characterization and qualification of graphites have been recognized in the General Atomics PCDR.

Table 1E (GA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	materials characterization)	
E-8	Irradiation induced change in mechanical properties such as strength and toughness, including the effect of creep strain.	See item E-1 for information relating to materials characterization and qualification of graphites. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	C.11.03.11 thru C.11.03.23 (graphite materials characterization)	The needs for further materials characterization and qualification of graphites have been recognized in the General Atomics PCDR.
E-9	Blockage of coolant channel in a fuel element block or reactivity control block due to graphite failure and/or graphite spalling.	***There is no indication that this item has been specifically addressed.*** See item B-7 for information relating to reactivity control.	C.07.02.01 thru C.07.02.09 (fuel performance data) C.11.03.11 thru C.11.03.23 (graphite materials characterization) C.11.03.42 (control rod channel flow data)	***There is no indication that this item has been specifically addressed in the General Atomics PCDR.***
E-10	Statistical variation of non-irradiated properties, due to forming, processing, raw materials, and formulation.	(Sec. 3.1.4.4, p. 3-76) – ***Fuel Quality and Performance Requirements*** – "…The fuel and reactor core are to be designed such that there is at least a 50% probability that the radionuclide releases will be less than the Maximum Expected criteria, and at least a 95% probability that the releases will be less than the Design criteria. The logic for deriving these fuel requirements is illustrated in Figure 3.1-68. Top-level requirements for the NGNP are defined by both the regulators and the users. Lower-level requirements are then systematically derived using a systems-engineering approach. With this approach, the radionuclide control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on radionuclide releases from the VLPC, reactor vessel, and reactor core are successively derived. Fuel failure criteria are in turn derived from the allowable core release limits. Finally, the required as-manufactured fuel attributes are	C.07.01.01 thru C.07.01.07 (fuel fabrication) C.07.02.01 thru C.07.02.09 (fuel performance	The need for statistical control is addressed in the General Atomics PCDR for fuel, ***but it is not addressed for other graphites.***

Table 1E (GA) – GRAPHITE - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable General Atomics R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		derived from the in-reactor fuel-failure criteria, with consideration of achievable values based on existing fuel manufacturing experience, thereby providing a logical basis for the fuel quality specifications…The maximum allowable release fractions for 30.2-yr Cs-137 and 249.8-d Ag-110m are included in Table 3.1-16 because these nuclides are expected to be the ***strongest contributors to worker dose***, based on previous assessments of radionuclide plateout distributions and plant-maintenance requirements." General Atomics discusses, in the PCDR, fuel fabrication a number of times in logistical terms, recommending initial fuel supplies from foreign vendors, and construction of a fuel fabrication facility in Idaho for the longer term (section 1.2.1, page 1-3).. Uniform quality of fuel or any other graphite materials is not addressed in the PCDR.	data)	
E-11	Ability to develop generic specifications that will ensure consistency of graphite quality over the lifetime of the reactor fleet, including for replacement components.	(Sec. 3.1.1.2, p. 3-10) – "GA prepared draft ***fuel product specifications*** to define the property requirements for the ***kernels, coated particles, and fuel compacts***. The requirements were written to be consistent with an NFI fuel particle design in order to utilize NFI's existing fuel manufacturing capability to the greatest extent possible, thereby avoiding a significant fuel R&D program. The NFI extended burnup fuel particle design was selected rather than the reference High Temperature Engineering Test Reactor (HTTR) fuel particle design because this fuel particle is designed for irradiation to higher burnup and is more consistent with the reference German fuel particle design. Table 3.1-5 summarizes the physical properties of two NFI fuel particle designs and compares them to the reference German particle and to the Advanced Gas Reactor (AGR) reference fuel particle as defined in the preliminary AGR fuel product specification [AGR Fuel Spec. 2004]. The primary implications of this approach are that the kernel will be UO2 (rather than UCO), the ***U-235 enrichment will be 10%*** (as opposed to the effective U-235 enrichment of about 10.8% for the GT-MHR initial core), the fuel compacts will be made using the HTTR matrix material, and the particle packing fraction in the fuel compacts is limited to about 30%. The fuel quality requirements written into the draft fuel product specification are consistent with the desired quality level for NGNP as specified in the preliminary AGR fuel product specification."	C.07.01.01 thru C.07.01.07 (fuel fabrication) C.07.02.01 thru C.07.02.09 (fuel performance data) C.11.03.11 thru C.11.03.23 (graphite materials characterizatio n)	The General Atomics PCDR includes generic specifications for fuel quality, ***but not for other graphite components***.
E-12	Tribology (effects of moving surface interactions) of graphite in helium environment, including potentially impure helium environment (examples: surfaces sticking together; surfaces wearing on each other to generate dust, etc.)	(Sec. 3.9.1, p. 3-189) – ***Primary Coolant Purification System*** – "This subsystem provides a means to remove circulating impurities from the primary coolant helium, and to transfer those impurities to the radioactive liquid and gas waste systems of the facility. A separate regeneration section within this subsystem is used to remove the impurities that accumulate in the purification subsystem adsorbers. The regeneration section is operated periodically under automatic control whenever regeneration is required. The primary coolant helium purification subsystem consists of two separate,	C.11.02.10 and C.11.02.13 (effects of helium on reactor internals and hot duct)	In the General Atomics PCDR, a helium purification system has been incorporated into the design to ensure the purity of the helium environment. ***There is no indication that phenomena associated with materials tribology have been specifically addressed in the General***

Table 1E (GA) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		independent, but identical trains of components as shown in Figure 3.9-1. All of the components that make up the trains are mechanically passive in nature; however, the adsorber elements become radioactive as the removed impurities are concentrated within the various media. Each purification train must therefore be located in a shielded vault to minimize personnel exposure to radiation. Helium purification is accomplished by routing a small side stream of helium from the primary coolant system through a series of purification components. ***These components remove the following chemical impurities: Br, I, H2O, CO, CO2, H2 (including Tritium), N2, O2, H2S, Kr, Xe, CH4, and other hydrocarbons***." ***There is no indication that phenomena associated with materials tribology have been specifically addressed.***	C.11.03.11 thru C.11.03.23 (graphite materials characterizatio n) N13.01.01 (effects of helium on primary heat transport) N13.02.01 (effects of helium on IHX) N.14.01.06 and 14.04.12 (effects of helium on SCS) N.42.02.01 (effects of helium on secondary transport)	***Atomics PCDR.***
E-13	Impact of degradation of thermal conductivity on fuel temperature limits.	(Sec. 3.1.2.2, p. 3-44) – "The reduction in graphite thermal conductivity with irradiation results in a peak fuel temperature increase of approximately 100°C. Accounting for thermal annealing of the irradiation damage reduces peak fuel temperatures by approximately 30°C. However, the effect of irradiation on graphite thermal conductivity has little impact on peak vessel temperatures."	C.07.02.04 (fuel compact thermophysical properties) C.11.03.16 (graphite thermal properties data)	This phenomenon has been recognized and quantified in the General Atomics PCDR.

Table 1F (GA) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
F-1	Cold oxygen (O2) and other heavy-gas accidental releases from the process plant that can flow from the chemical plant to the nuclear plant (depending upon wind, relative plant elevations, and nuclear plant air intakes) and potentially impact the integrity of reactor systems, structures, and components (SSCs). All of the proposed processes for production of hydrogen start with water, and thus all of the processes will produce oxygen as a byproduct of hydrogen production. Oxygen is the one common chemical safety issue that can impact nuclear plant safety. At high oxygen concentrations, many "noncombustible" materials become combustible and the potential for spontaneous combustion increases. Increased oxygen levels at the reactor can compromise the functioning of safety equipment.	(Sec. 2.3, p. 2-15) – "The distance between the hydrogen plants and the reactor building is 90 meters based on the results of an INL engineering evaluation of the necessary separation distance [INL 2006]. No earthen berm or blast suppression barrier is considered necessary between the hydrogen production facilities and the reactor with a separation distance of 90 meters because the reactor s below rade. However, the hydrogen production facilities are circumvented by a low berm, which serves as a chemical spill retention barrier." (Sec. 5.2.2.3, p. 5-37) – "There is no currently anticipated inherent excessive risk in the thermo-chemical production of hydrogen that would preclude licensing of the NGNP or licensing of associated commercial-scale hydrogen productions plants based on the hydrogen production processes demonstrated in the NGNP. An attractive feature of the GT-MHR plant for electricity production is siting flexibility, because no plan for public evacuation is required as the result of the MHR's passive-safety features. For a commercial scale H2-MHR, a potential issue that requires further evaluation is whether or not a public evacuation plan is required because of potential accidents that could cause chemical releases from the SI hydrogen plant. However, chemical releases should not impact the passive safety of the reactor system." The General Atomics PCDR also contains general performance statements that the hydrogen production system will be designed to have no adverse impact on the primary system. ***There is no indication that this item has been specifically addressed.***	N.44.01, N.44.02, N44.03, N44.04, N45.03, and N45.04 (hydrogen production)	***There is no indication that this item has been specifically addressed in the General Atomics PCDR.***
F-2	Failure of the IHX leading to potential damage to safety-related SSCs in the reactor due to blow-down effects from large mass transfer and over-pressurization of either secondary or primary side. The impact of the IHX failure depends upon the selection of the heat transfer fluid in the secondary heat transport loop. Helium is the leading candidate for the heat transport loop, but no final decisions have been made. If helium is used, the helium inventory in the secondary loop may be greater than the inventory in the reactor; thus, any leak in the IHX can significantly increase the total helium inventory involved in any reactor depressurization event.	The General Atomics PCDR contains general performance statements that the hydrogen production system will be designed to have no adverse impact on the primary system. ***There is no indication that this item has been specifically addressed.***	N.13.02.01 thru N.13.02.09 (IHX)	***There is no indication that this item has been specifically addressed in the General Atomics PCDR.***
F-3	Failure of the process heat exchanger (PHX) leading to potential damage to safety-related SSCs in the reactor, due to fuel and primary system corrosion from the introduction of	See item F-1 for an excerpt from the General Atomics preliminary hazards analysis for the hydrogen production system. The General Atomics PCDR also contains general performance statements that the hydrogen production system will be	N.45.04.02 (HTE heat	***There is no indication that this item has been specifically addressed in the***

Table 1F (GA) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	corrosive process plant chemicals leaking down the process heat transport line and failing the IHX.	designed to have no adverse impact on the primary system. ***There is no indication that this item has been specifically addressed.***	exchangers)	***General Atomics PCDR.***
F-4	Steam generator failures leading to the introduction of steam/water into the primary system, potentially causing a reactivity spike and chemical attack of the TRISO fuel particle coatings and graphite. Some hydrogen production processes, such as high-temperature electrolysis, require steam as a process feedstock; thus, the high-temperature reactor may be required to provide high-temperature steam.	See item B-7 for information relating to reactivity control. (Sec. 5.1.1.3, p. 5-6) - **Control of Chemical Attack – "**Chemical attack on fuel particles and on the graphite core structure can result from air or water ingress into the primary system. ***Steps have been taken to prevent ingress of contaminants, and consequences are expected to be acceptable if they occur. The likelihood of water entering the primary system is limited by the absence of high pressure and high energy sources of water in proximity to the primary system***. Under normal operating conditions, all water coolers and heat exchangers operate at lower pressures than the pressure of the primary coolant with which they exchange heat. In the event of a cooler or heat exchanger leak, primary coolant helium would leak out into the secondary cooling water until pressures equilibrate. Then the rate of ingress of sub-cooled water would be small, as water tries to enter the primary system by diffusion and gravity. The amount of water that could enter is limited to the inventory of water in the secondary coolant circuit located above the elevation of the leak. Most of the sub-cooled water that could enter the power conversion vessel would remain at the bottom of the vessel. Very little of it would become entrained in the helium coolant and be transported to the core. Core cooling can still be provided by either the PCS or the SCS, and would limit the potential for chemical attack. If core cooling is not available, the potential of water transport to the core would still be limited. The sub-cooled water will not flash to steam unless the primary coolant helium pressure is below the water saturation pressure, which may occur only when the reactor is operating at a low power level. The reaction rate of water and core graphite will be negligible. The reaction of steam and graphite is slow and endothermic and therefore is not self-sustaining."	N.07.05.07 thru N.07.05.10 (graphite and other corrosion rates due to water) C.11.03.18 and C.11.03.19 (graphite corrosion data and methods validation) C.11.03.23 (graphite oxidation for postulated accidents) N.45.04.01 (HTE steam generator/superheater)	The General Atomics PCDR indicates that this is not a likely event, and consequences would be acceptable if it did occur.
F-5	Loss of the pressurized coolant inventory from the intermediate loop leading to a loss of primary reactor heat sink and the potential for hydrodynamic forces on the IHX leading to IHX failure and loss of reactor primary system coolant.	(Sec. 5.1.1.3, p. 5-5) – ***Control of Heat Removal*** – "Reactor cooling can be accomplished by the PCS, the SCS, the HTS, or by passive cooling through the reactor vessel to the RCCS. The PCS, which operates during power generation, provides primary shutdown cooling. PCS cooling capability is an active system. The SCS is designed specifically for residual heat removal in the event that the PCS is unavailable. In the NGNP, the HTS is another active system that can be used to remove heat from the reactor core. In the event the PCS and SCS are unavailable, the core design ensures passive residual heat removal capability. The limited core diameter, limited power density, and unique core assembly configuration (annular with a large length-to-diameter ratio) limit core and fuel temperatures during passive cooling. The RCCS, which is independent and diverse from the PCS and SCS in fundamental ways, acts to keep structures, including the reactor vessel and containment building, within allowable temperature limits. The RCCS is totally	N.13.02.01 thru N.13.02.09 (IHX) C.14.01.01 thru C.14.01.06, C.14.04.01 thru C.14.04.12 (SCS)	This item has been addressed in the General Atomics PCDR in the design.

Table 1F (GA) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		passive under accident conditions. Reactor heat is transferred through the reactor vessel walls to RCCS cooling panels by conduction, natural convection and radiation heat transfer; the vessel walls are uninsulated to facilitate this process. The RCCS air cooling loops are naturally circulating. With RCCS cooling, core temperatures peak after about 2 days and cool within several days to below 1100°C. Even if the RCCS were not available for some reason, heat from the reactor vessel walls would be transferred through the inoperable RCCS panels to the containment building itself and ultimately to the earth surrounding it. This cooling capability is also totally passive. It is not in the design basis and is not necessary to meet any safety requirements or quantitative safety goal, but exists as an inherent feature, enhancing the safety of the NGNP. With core cooling in this mode, core temperatures peak at about the same level as with RCCS cooling, but cool more slowly thereafter. The NGNP vessel system has a unique safety function in support of core cooling systems. LWR vessels must confine primary coolant (i.e., water) at all times, at least so that the core will remain covered. However, while containing the helium coolant is an important vessel function for the NGNP, ***sufficient core cooling can be provided even if the helium coolant is lost.*** “	C.16.00.01 thru C.16.00.06 (RCCS)	

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
A-1	*Core-Coolant Bypass Flow Phenomena (Normal Operation)* • Overcome difficulties in estimating bypass flow • More complete understanding and accounting of related design features such as fuel blocks (PMR) and core barrel configurations • In-core temperature testing • Parametric analysis of gap configurations to bound questions associated with gap and bypass flows	See item A-8 for information relating to modeling activities. See item C-6 for information regarding testing. See Item E-1 for graphite materials information, including core blocks. (Sec. 4.2.10, p. 4-68 thru 4-76) - *NHSS Control and Instrumentation System* – “The NHSS Control and Instrumentation System comprise only of equipment found in the Nuclear Heat Supply Building and all operator interaction is performed in the Central Control & Supervisory System (CCSS) as described in the Section 9: Balance of Plant Systems. The primary systems comprising the NHSS Control and Instrumentation system are the Operational Control System (OCS), the Equipment Protection System (EPS) and the Reactor Protective System (RPS). The OCS monitors and controls the NHSS systems throughout their normal operating range. The EPS detects operating regimes or operating conditions that may be harmful to NHSS equipment, and takes appropriate action to prevent or minimize potential damage. The RPS automatically initiates RUS protection whenever pre-established set points are exceeded…. The RPS consists of three subsystems: Reactor Trip System (RTS), Post Event Instrumentation System (PEI) and the Manual Diverse Shutdown System (MDSS). The RPS provides functions to prevent exceeding predefined safe operating limits and to provide information to operators in the event of nuclear accidents. The Protection System is implemented using a Class 1E qualified digital platform that is capable of performing logical operations as well as algorithmic processing. The same platform is used for both the RTS and the PEI applications. The MDSS is a hard-wired system that allows tripping the breakers without dependence on any software. All portions of the RPS are treated as Class 1E Structures, Systems and Components (SSC)…. The RTS automatically prevents operation of the RUS in an unsafe condition by shutting down the RUS whenever predetermined operating limits are approached, or when Design Basis Accident conditions are detected. The operating limits are selected, based on initial conditions (RUS power and outlet temperature) assumed in the safety analysis…. The MDSS is a hardwired system that enables operators to manually initiate reactor trip, RCS and RSS activation functions, from both the Main Station Control Room and the PEMRR. The MDSS controls are supported by monitoring instrumentation associated with the PEI system or the Plant Computer displays to provide the operator the capability to know when to take the appropriate manual action. For each reactor trip function (RS and RSS activation), a set of three switches that are individual and redundant are provided in the Main Station Control Room and a set of three switches that are individual and redundant are provided in the PEMRR. Thus, a total of 4 sets of three switches each comprise the MDSS for the protection system.”	(None)	***There is no indication that this item has been specifically addressed in the WEC PCDR.***

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		(Sec. 4.4.2, p. 4-84) – *Future Studies/Flow Requirements* – "…Assess the effect of the non-symmetrical flow in the riser channels on the temperature distribution of the Side Reflector and core barrel assembly (CBA). This is due to the lack of riser channels above the outlet pipes. Assess the flow distribution in the outlet slots due to the changes in the outlet plenum for mal-distribution. ***Assess the sealing of the riser channels in the bottom reflector to limit direct core bypass flow. PLOFC flows need to be assessed to ensure that no hot gas flows down the risers and overheats the reactor pressure vessel (RPV).*** The higher heat loss to the RCCS caused by the higher RPV temperature needs to be considered." (Sec. 16.9.5, p. 16-105) – *Design and Evaluation Model Verification and Validation* – "In order to analyze various aspects of the integrated NGNP, the ***software currently used for the PBMR–DPP design will be enhanced and extended to have an integrated core neutronic/thermal hydraulic analysis tool*** coupled to the Power Conversion and Hydrogen Production Systems models for simulating normal and off-design conditions. This tool will allow for steady-state and transient analyses of the integrated NGNP plant and will enable operational and control studies. Various verification and validation activities are required to ensure that this tool provide accurate results. Aspects that will require V&V are: 1. The input data used for these models and calculations will also need to be verified to ensure accurate results. Examples of data required are state of the art cross section data that will reduce uncertainty and calculation margins as well as input data for the Hydrogen Production System and heat transfer correlations used in the evaluation models. 2. Calculation model verification and validation for various phenomena and performance conditions. Specific aspects that need to be addressed in HTR cores are: 1. Non-local heat generation in HTR cores - An issue that is contributing to uncertainties in the coupled neutronics and thermal-hydraulics of pebble bed reactors design is the treatment of the so-called non-local heat generation contributors. These include: a. Heat generation due to γ-radiation and neutron moderation; b. ***Heat redistribution in the reflector regions due to the bypass flows***; c. Non-local γ-power in the reflector during a depressurized loss of coolant (DLOFC) event. 2. Cold critical experiment – Design techniques and methodologies implemented in the design codes need to be validated. A cold critical experiment will provide the opportunity to measure many parameters that		

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		are calculated. 3. Dust modeling and associated impact on plant performance and safety also need to be addressed."		
A-2	*Effective Core Thermal Conductivity* • For prismatic cores – Make available dose and temperature-dependent graphite thermal properties (especially thermal conductivity) to the NRC T/F code suite, to account for large uncertainties as well as for characterization of annealing effects during long-term heat-up D-LOFC accidents. • For pebble bed cores - Also considerable error bounds in effective core thermal conductivity as a function of both temperature and irradiation. Existing correlations available are empirical, but PBMR project has an experimental facility to be used to refine the database.	See item C-6 for information on testing and test facilities. The WEC technology development report, in Section 16.2.2 ***identified the need to extend the temperature-fluence envelope for the fuel graphite.*** The needed R&D includes irradiation of graphite spheres at a temperature and to a fluence level applicable to the NGNP, plus post-irradiation examination and analysis. (Section 16.2.3.1, p. 16-38) ***Samples for investigation and irradiation will be cut from pressed graphite spheres provided for the test***. These samples will be cut parallel and perpendicular to the extrusion direction. Following irradiation, the following characteristics will be measured: • Geometrical size • Mass • Calculation of sample density • Measurement of sample density • Sample porosity • ***Thermal conductivity in the range 20 up to Irradiation Temperature*** • Electric conductivity in the range 20 up to Irradiation Temperature • Thermal coefficient of linear expansion in the range 20 up to Irradiation Temperature • Dynamic Young's modulus • Compression strength • Ultimate bending strength • Optical ceramography • Uranium and thorium content The above measured characteristics will be compared to values obtained during pre-irradiation characterization.	NHSS-01-03 (fuel graphite testing) NHSS-02-01 and NHSS-02-02 (graphites)	Indications are that this item has been addressed in the WEC PCDR, and will be further addressed as the project progresses.
A-3	*Afterheat Correlations* • Peak fuel temperatures in the D-LOFC accident are very sensitive to the afterheat (vs. time) to the same extent as they are to the core thermal conductivity function. Afterheat correlations are sensitive to fuel type and burn-up histories. Tracking fuel histories during operation can be challenging, and afterheat validation	(Sec. 2.1.1, p. 2-13) – ***Commercial Plant Summary Description*** - The reference PBMR H2 PHP fuel is TRISO-coated UO_2 fuel particles embedded in the spherical pebble fuel elements. The pebbles are circulated through the core to effect on-line refueling which is compatible with continuous process industries. The fuel cycle adapts Low Enriched Uranium (LEU) to achieve optimal burnup and overall core and fuel performance. A high degree of safety is achieved without reliance on prompt operator actions and startup of standby equipment by the use of passive design	NHSS-01-02 (fuel heating tests for accident conditions)	There is general discussion of limiting of peak fuel temperatures; however ***there is no indication that this item has been specifically addressed in the WEC PCDR.***

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
	data is more difficult to obtain for long times after shutdown.	features. The design limits peak fuel temperatures during normal operation and during the long-duration loss of forced circulation accidents such that radionuclide retention within the fuel is maintained.		
A-4	*Core Effective Pressure Drop* • Standardized and well-documented correlations for core pressure drop; conformation data may be needed for low-flow cases to better characterize flow distribution and plume formation (for the P-LOFC) and in-core airflow distributions during air ingress accidents. • PBR - parametric analyses using established ranges of different packing fractions to define a performance envelope.	(Section 16.2.1.1.2, p. 16-20) – "The HTTF facility will consist of a number of smaller test sections that will be used for separate effects tests and a main test section that will be used to perform integrated effects tests. The smaller test sections will consist of a scaled down pebble bed and a number of duct-type sections packed with pebbles to represent pebble bed sections with predetermined homogeneous porosities. The main test section will represent an annular pebble bed and it will have the capability to heat the pebble bed (made up of graphite pebbles) and to characterize the heat transfer behavior of such a pebble bed. It is envisaged that the HTTF shall fulfill the needs for tests to characterize the following main phenomena required for simulating heat transfer in a pebble bed: • Pebble bed effective conductivity, which is a combined coefficient representing conductive and radiation heat transfer, required at the pebble bed center region and wall regions respectively. • Convection heat transfer coefficient required at the pebble bed center region and wall regions respectively. • ***Pebble bed pressure drop correlation.*** • Braiding effect correlation, which defines the mixing effect of gas flowing through a pebble bed. • Natural convection heat transfer coefficient."	(None)	It appears that the PBR will have the required test capability, but ***there is no indication in the WEC PCDR that parametric based on packing fractions have been or will be addressed.***
A-5	*RCCS Performance during LOFC* • Simulate RCCS safety functions in detail, with its predominantly radiant heat transfer coupling to the RPV and other heat transfer mechanisms within the reactor cavity. RCCS functions include maintaining the reactor cavity liner concrete temperature below prescribed limits, preventing the RPV peak temperature from exceeding limits during LOFC events, and minimizing parasitic heat losses during normal operation. • Models may be needed to simulate large pressure pulses in D-LOFC accidents that could damage the RCCS, reducing cooling and/or opening up another release path for air or water ingress to the reactor cavity, and perhaps for FPT out to the environment.	See item A-8 for information on model development. Section 3.2.1.8, p. 16-33 mentions that FOAK design and performance verification is in progress, including features of the RCCS. Section 4.2.4, pp. 4-49 thru 4-52, provides a detailed description of the Reactor Cavity Cooling System (RCCS), including functions, requirements, layout, interfaces, and operation. (Sec. 16.2.1.9, p. 16-33) – *Balance of Plant R&D* – "Verification of the RCCS system's passive mode operation via analysis. The RCCS will not be qualified by testing and for this reason require two independent analysis to be done. The analysis codes selected are RELAP and SPECTRA. The configuration of the RCCS is such that is not equivalent to the normal LWR scenarios and require some research in order to model the system correctly."	(None)	The need for performance verification has been recognized. ***However, it is impossible to determine if safety functions will be modeled in detail or if large pressure pulses will be simulated in the WEC PCDR.***
A-6	*Fuel Performance Models* • Aspects of maximum fuel temperature plus time-at-	(Sec. 5.2.4.2, p. 5-17) – *Testing and Qualification (Fuel)* – "Even though test results from the German pebble-bed reactor program are available, and is the basis for the PBMR DPP, expected operational parameters specified for the NGNP was not	NHSS-01-02 (fuel heating	The needs for modeling and code development have been recognized in the

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION				
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	temperature histories (critical limiting factors) for all fuel regions provide inputs to fuel failure models, to determine source terms and dose-vs.-frequency estimates.
	• Chemical reactions in air or water ingress accidents, which depend on temperature and should be included in the T/F codes. Especially for fast transients, detailed temperature profiles of the fuel and graphite should be taken into account for thermal stress calculations.
		envisaged during PBMR efforts or the German program. Therefore to ensure the safe application of PBMR DPP based fuel in the NGNP program, the following ***parameters and/or aspects are considered to be important in terms of fuel testing and qualification*** must be considered:
		• Expected/specified normal operating condition parameters
		    ○ Maximum fuel temperatures
		    ○ Percentage burn-up
		    ○ ***Percentage fission product release at specified temperatures***
		• Expected/specified adverse (accident) operating condition parameters
		    ○ Maximum temperatures
		    ○ Percentage burn-up
		    ○ ***Percentage fission product release at specified temperatures***
		• Statistical requirements of tests and qualification samples to ensure confident and safe application thereof for design and further operational specification refinements
		• Pre-designate procedures and facilities for pre- and post-irradiation tests to support qualifications”
		
		(Section 16.2.1.5, p. 16-30) – *Fuel Qualification* – “PBMR (Pty) Ltd has embarked on an intensive fuel qualification programmed to ensure that the quality of their manufactured fuel is similar to the German LEU-TRISO fuel. This program is currently underway and will ***qualify the fuel in terms of physical properties, maximum fuel temperatures, percentage burn-up and fission product release for both normal operating and accident conditions***. Statistical requirements of tests and qualification samples will also be investigated to ensure confident and safe application thereof for design and further operational specification refinements. Part of the qualification program will consist of irradiating a number of fuel spheres, containing a statistically significant number of coated particles, to full PBMR irradiation requirements in a material testing reactor.”
		
		(Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* – “***Corrosion/Oxidation Models for CFD (Air Ingress analysis)*** are planned. Implementation of chemical reactions in commercial CFD codes for analysis of air ingress consequences during postulated accident conditions. Will include validation against NACOK experimental results.”
		
		(Section 16.2.2, p. 16-35) – “DDN NHSS-01-02 specifies data to correspondingly extend the heat up data pertaining to accident conditions. R&D for the fuel itself comprises ***irradiation of fuel samples at the higher temperature*** applicable to the
			testing)	WEC PCDR.

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		NGNP, post irradiation examination and subsequent heat up of some samples to simulate accident conditions, plus corresponding modeling and analysis." (Section 16.2.3.1, p. 16-38) – "The remaining 11 fuel spheres, after the irradiation testing, will be subjected to heating tests simulating DLOFC transient temperatures, nominally 1800°C (rounded to nearest hundred degrees from DDN NHSS-01-02). Following heating tests, all heated fuel spheres will be visually examined again and their fission product inventories measured. One heated fuel sphere will be deconsolidated to provide coated particles for ceramography and fission product distribution measurements."		
A-7	*Air Ingress Phenomena* • With little or no detail available about the confinement, only generalized studies and experiments would be practical. Bounding analytical studies could be useful in determining positive and negative features of proposed design characteristics. The major features of general interest would be quantification of long-term air in-leakage into the confinement, and the mixing and stratification characteristics of gases in prototypical cavities within the confinement.	(Section 16.2.1.1.6, p. 16-26) – "NACOK stands for Natural Convection with Corrosion. The main section of this facility is made up of a vertical channel of 300 mm x 300 mm and 7.5 m tall. The experimental channel is composed of sections representing a bottom reflector, sphere packing (pebble bed) and a top reflector. The experimental set-up was designed to be able to represent different breaks in pipes connecting to the reactor. Breaks can be created that simulate the coaxial duct (reactor outlet pipe), the defueling chute at the bottom of the reactor and the fuelling line at the top of the reflector. By a sectional design, different core heights can also be simulated. All sections of the experimental channel and of the return pipe can be heated to accident-relevant temperatures. At different positions, the local gas compositions can be measured." (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* – "***Corrosion/Oxidation Models for CFD (Air Ingress analysis)*** are planned. Implementation of chemical reactions in commercial CFD codes for analysis of air ingress consequences during postulated accident conditions. Will include validation against NACOK experimental results."	(None)	The need for greater understanding of the air ingress phenomenon has been recognized in the WEC PCDR.
A-8	*Long-term analysis need* - Comprehensive suite of verified and validated accident simulation codes (core thermal-fluids, core neutronics, whole-plant transient behavior, confinement analysis, and chemical reactions), agreed-upon accident cases for regulatory acceptance, and robust supporting databases that NRC can use for independent confirmatory analysis of candidate plant and confinement designs and options.	(Sec. 8.3.3, p. 8-82) – *Design Basis Transient Study* – "An engineering study is recommended to identify and ***analyze transient cases*** that could effect the design requirements of the PCS with respect to ensuring safety of the Nuclear Heat Supply System (NHSS) and Heat Transport System (HTS). Demonstration cases and commercial configurations will be assessed, to ***ensure that the NHSS, HTS, and HPS function within the design basis envelopes through the assumed transient conditions***." (Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a ***full scope PRA that addresses all internal and external hazards, including those associated with the HPF, will be developed***. During the conceptual design of the NGNP, a Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health	NHSS-01-01 (Data to extend the irradiated fuels qualification database) NHSS-01-02 (Extend heat up data under accident conditions.	Long-term analysis needs for computer code development, and supporting databases, are either completed, underway, or planned for the future, as discussed in the WEC PCDR.

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION				
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		Administration (OSHA) requirements. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Section 16.2.1.2, p. 16-29) – *Neutronics Design Tools* • Completed: Detailed ***core neutronic design and shutdown system analysis***. Establish the core layout, control system, neutronic behavior in steady state and transients as well as safety and fuel performance analysis. • In Progress: Tools for ***initial criticality, startup and run in phase; V&V of legacy codes***; engineering and training simulator; and analysis of reactivity transients. • Future: ***Integrated core neutronic/radiation/fuel performance code development***. Future knowledge and expert base as the product of core competency established; important for plant optimization, licensing in other markets, client support, reducing of calculational margins. • Future: ***Verified high temperature cross section libraries and measurements***. Important to establish state of the art high temperature cross section data; needed for new methods and codes; reduction of margins and uncertainty. (Sec. 16.2.1.6, p. 16-30) – *Core Structural Ceramics R&D* – Completed program under the PBMR-Specific Materials Test Reactor Program to conduct supplemental irradiations of NBG-18 to ***verify consistency with the established database for similar graphites***. (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* • Completed: ***Systems CFD software for Thermo-Fluids design*** of VHTR system. Development of M-Tech Industrial's Flownet network Thermo-Hydraulics code into the commercial Flownex software product. Includes component and reactor models to facilitate simulation of thermo-hydraulic steady state and transient behavior of indirect and direct power conversion cycles. R&D included V&V of the code to nuclear industry standards. • In Progress: ***CFD Models for Thermo-Fluids behavior*** of Pebble Bed CFD. Implementation of models for flow and heat transfer in pebble beds for use in detailed CFD reactor models used for reactor design. Includes coding and V&V of these models as User Defined Functions (UDFs) in commercial CFD codes. • Also In Progress: Engineering design ***tools for predicting distortion behavior and failure of irradiated graphite materials*** such as core blocks; and discrete element modeling of interaction between graphite reflector structures and fuel spheres. • Planned: Corrosion/oxidation ***models for air ingress*** consequences during	NHSS-01-03 (Extend temperature-fluence envelope of fuel graphite) NHSS-02-01 and NHSS-02-02 (Extend irradiated materials qualification database for Reflector Graphite) HTS-01-1 thru HTS-01-19 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX ceramics)	

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		postulated accident conditions. Section 16.2.1.11, p. 16-35) – *Safety Analysis* • Mostly Completed: ***V&V of commercial codes***. V&V of commercial codes as per regulatory requirements to nuclear industry standards; these V&V include comparison with benchmark calculations and/or test results. • In Progress - ***Identification of initiating events*** from FMECA/HAZOP processes; identification of postulated initiating events and establishment of accident scenarios and required analysis and assumptions. • Completed - ***Establishment of adequate conservative assumptions, analysis methodologies and processes to address all uncertainties*** to provide a justifiable safety case. • In progress - ***Establishment of a best estimate methodology*** including integrated accident analysis codes, assumptions, V&V, acquisition of plant data and licensing processes. • In Progress – ***Establishment of confinement modeling capability***. Contracting analysis work, confinement modeling assumptions, methodology; selection, purchasing and application of codes. • In Progress – ***Source term analysis***. Establishment of core release rates; activation of dust, contaminants and erosion products; selection and distribution of radionuclide's in reactor and power conversion unit; activity release mechanisms and radio nuclide transport; establishment of activity and release mechanisms of solid and liquid waste systems; (Sec. 16.2.2, p. 16-35) – *Design Data Needs (Nuclear Heat Supply System)* – "Three DDNs have been identified pertaining to the NGNP Fuel. The first of these DDNs (NHSS-01-01) identifies the need for data to ***extend the irradiated fuels qualification database*** from the temperature-burnup envelope of the PBMR Demonstration Power Plant (DPP) to that of the PBMR NGNP. The second DDN (NHSS-01-02) specifies data to correspondingly extend the heat up data pertaining to accident conditions. The third DDN (NHSS-01-03) provides for an extension of the temperature-fluence envelope of the Fuel Graphite to that required by the NGNP. In all three cases, the extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature." (Sec. 16.2.2, p. 16-35) – *Design Data Needs (Nuclear Heat Supply System)* – "Two DDNs (NHSS-02-01 and NHSS-02-02) have been identified to ***extend the irradiated materials qualification database for Reflector Graphite*** from the temperature-fluence envelope of the PBMR DPP to that of the PBMR NGNP. The extension of		

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature. The corresponding R&D comprises irradiation of graphite samples at low and high temperatures, plus post-irradiation examination and analysis." (Sec. 16.2.3.1, p. 16-36) – *Fuel Qualification R&D Plan* – "Two supplemental irradiation tests are planned for the PBMR NGNP to ***extend the database supporting the fuel*** for the PBMR DPP. The first of these irradiation tests, which is proposed to start in FY2009 will use pre-production fuel. The second, proposed to start in 2012 will use actual production fuel." (Sec. 16.2.4, p. 16-42) - *Core Structural Ceramics Reflector Graphite R&D* – "In parallel, the NGNP Program at INL is embarking on a graphite development effort that addresses multiple product forms (including NBG-18) and applications (including the PBMR). The INL program places particular emphasis on the understanding of fundamental graphite characteristics that would, ideally, allow the characterization of new coke and/or graphite sources without the need for an extensive irradiation program. To the extent that the INL program addresses NBG-18 and that manufacturing and QA systems development are generic, there is a potential to accelerate the INL effort and reduce its cost by utilizing applicable results of the PBMR DPP development work that would otherwise be duplicated. From the PBMR perspective, there is a potential to ***expand the database supporting NBG-18*** and, potentially, to reduce the scope of surveillance, testing, inspection and maintenance (STIM) required as a basis for operation of the PBMR DPP. Further potential benefits are access to multiple qualified vendors for follow-on PBMR commercial deployments and easing the burden associated with qualification of new graphite sources. In order to take mutual advantage of PBMR's ongoing program to qualify SGL graphite plus INL's and PBMR's mutual interests to cooperate on graphite qualification with SGL and Graftek, efforts are underway to develop a collaborative program. In the interim, a preliminary scope, cost and schedule for R&D activities addressing the Reflector Graphite DDNs for the PBMR NGNP have been developed." (Sec. 16.3.1, p. 16-50) – *Design Data Needs (Heat Transport Facility)* – "The final DDNs supporting the metallic IHX, HTS-01-18 and HTS-01-19, are established to ***provide the underlying database supporting NGNP-specific code cases for the IHX material and design***, respectively. There is a potential that such code cases would also be applicable to early commercial plants, pending formal implementation within the ASME Code. For ceramic/composite IHXs, six placeholder DDNs (HTS-02-01 through HTS-02-06) have been identified, as both the DDN's and the associated R&D activities will need further development during conceptual design. The first DDN provides for a review of existing technology that is potentially applicable to the		

Table 1A (WEC) – ACCIDENTS AND THERMAL FLUIDS – DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		development of a ceramic IHX. The anticipated result of the corresponding R&D effort will be the selection of one or more materials and/or heat exchanger technologies for further development. The second DDN specifies the need for a ***materials property database*** for the selected materials. The third DDN addresses the need for design methods, while the fourth identifies requirements for performance verification. The fifth and sixth DDNs address manufacturing technology and the development of codes and standards…. The R&D activities pertaining to DDNs HTS-01-01 through HTS-01-06 provide for the extended qualification of the current reference IHX material, Alloy 617. This extended qualification is required due to the demanding operating conditions that will be seen by the IHX, plus the small grain size that is expected to be required for compact heat exchangers as they are characterized by very thin heat transfer surface cross-sections. As described in DDN HTS-01-01 (Section 6.3.1), an initial effort is required to further develop the specification for Alloy 617 and to establish a reference for characterization. ***Included in this effort, is a review of the current database for this material, consultation with material vendors and consideration of a controlled specification variant, Alloy 617CCA, that potentially decreases the range of uncertainties with respect to properties.*** The conclusion of this effort will be procurement of materials to be used for subsequent testing."		

Table 1B (WEC) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
B-1	Time-dependence and spatial distribution of decay heat as a major factor in determining maximum fuel temperature during a D-LOFC.	See item A-8 for information associated with model development. (Sec. 2.1.1, p. 2-13) – ***Commercial Plant Summary Description*** - The reference PBMR H2 PHP fuel is TRISO-coated UO_2 fuel particles embedded in the spherical pebble fuel elements. The pebbles are circulated through the core to effect on-line refueling which is compatible with continuous process industries. The fuel cycle adapts Low Enriched Uranium (LEU) to achieve optimal burnup and overall core and fuel performance. A high degree of safety is achieved without reliance on prompt operator actions and startup of standby equipment by the use of passive design features. ***The design limits peak fuel temperatures during normal operation and during the long-duration loss of forced circulation accidents*** such that radionuclide retention within the fuel is maintained. (Sec. 5.2.4.2.4, p. 5-21) – *PBMR Fuel Irradiation Program* – "Following from the above conclusions, it is proposed to extend the existing German database by performing additional irradiation and heating tests using PBMR fuel spheres from a qualified fuel manufacturing line. The proposed PBMR irradiation program will consist of irradiating a number of fuel spheres, containing a statistically significant number of coated particles, to full PBMR irradiation requirements in a material testing reactor. Irradiation will be performed in such a way that temperatures encountered during normal cycling of fuel spheres through the PBMR core are simulated until PBMR irradiation targets are reached. A number of simulated PLOFC temperature transients will be superimposed on the normal temperature cycles for a sufficient number of fuel spheres to ensure statistic validity. Following the completion of the irradiation test, a sufficient number of ***irradiated fuel spheres will be subjected to heating tests using temperature cycles simulating the expected DLOFC temperature*** transient for the PBMR. The PBMR start-up core consists of a mixture of start-up fuel spheres and graphite spheres. In addition to irradiation test on fuel spheres, irradiation tests will also be performed on matrix graphite spheres manufactured on the same manufacturing line as fuel spheres. The reason for this is that some matrix graphite properties cannot be measured on fuel spheres containing coated particles." (Sec. 5.2.4.2.5, p. 5-22) – *Normal Operation and Accident Conditions Testing and Qualification* – "Generally, during normal and accident conditions operation of TRISO particle containing fuel spheres, three variables must be taken into consideration. These are in order of importance; temperature, burn-up, and fast neutron dose. Deviations of these variables from acceptable, specified levels are an indication of process parameters where possible fuel failure and pending upset process conditions (accident conditions which may include the release of fission products) may arise, and must be considered during testing and qualification efforts."	NHSS-01-01 thru NHSS-01-03 (fuel testing)	The needs for fuel testing and modeling to determine fuel temperature in a D-LOFC are recognized in the WEC PCDR.

Table 1B (WEC) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		(Sec. 5.3, pp. 5-39 thru 5-45) – Fuel Design Development Needs (DDNs) – DDNs and the efforts planned to fulfill them are described for the PBMR fuel, including: • Fuel Irradiation Tests for Normal Operational Conditions (Sec. 5.3.1.3.1) • Fuel Heating Tests for Accident Conditions (Sec. 5.3.1.3.2) • Fuel Graphite Irradiation Tests (Sec. 5.3.1.3.3) (Section 16.2.2, p. 16-34) - Planned: Corrosion/Oxidation Models for CFD (Air Ingress analysis). Implementation of chemical reactions in commercial CFD codes for analysis of air ingress consequences ***during postulated accident conditions***. Will include validation against NACOK experimental results. (Section 16.2.2, p. 16-34) - In Progress: Identification of licensing basis events; and required analysis; and V&V. Identification of initiating events from FMECA/HAZOP processes; identification of postulated initiating events and ***establishment of accident scenarios and required analysis and assumptions.***		
B-2	Control and shutdown rod worth and reserve shutdown worth as required for hot and cold shutdown.	(Sec. 4.2.1.5, p. 4-29) – *Reactivity Control System (RCS)* – "The ***Reactivity Control System (RCS) is used to control the reactivity in the fuel core, to quickly shut the reactor down and to keep it in a shutdown mode***….The RCS consists of 24 identical control rods. The control rods are grouped into 12 control rods and another group of 12 shutdown rods. The control system moves each group alternatively to have the rods inserted to an equal depth into the side reflector. The only difference between shutdown rods and normal operation control rods is the length of the chain, with the control rods only traveling in the top part of the reflector, and the shutdown rods capable of traveling in the bottom part of the reflector. Each rod consists of six segments containing absorber material in the form of sintered B_4C rings between two coaxial cladding tubes. Gaps between the cladding tubes and B_4C rings prevent constraint forces from arising due to radiation-induced swelling of the B_4C. Pressure equalizing openings expose the B_4C to the coolant gas to avoid any pressure build-up. The RCS consists of the following major subsystems/components: • RCS Control Rod Drive Mechanism (CRDM) consisting of the chain drive, chain container and scram shock absorber which functions as the primary shock absorber. The purpose of the CRDM is to translate rotational movement into linear movement. • Rod and chain, with the rods absorbing neutrons and the chain connects the chain drive to the control rod. • RCS secondary shock absorber, which prevent damage to the control rod and the core structures ceramics following a chain failure. • RCS drive motor, which keep the control rod in position and move the control rod up and down. During a power failure the control rod will fall down the control rod	(None)	The need for determining/validating rod worths has been recognized in the WEC PCDR, and work is either completed or is in progress.

Table 1B (WEC) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		channel under gravitational force. • RCS control rod guide tubes connect the CRDM housing to the Core Structure and serve as a guide for the Control Rod. During the anticipated operating modes of the Reactor Unit System, the RCS is required to raise and lower the control rods and hold them steady in any position over their entire range of travel. The control rod and shutdown rod positioning is commanded by the Operational Control System (OCS). Control- or shutdown rod insertion (scram) action is initiated by the Reactor Protection System (RPS), which overrides the OCS. During full power operation, both banks of control rods (24) are inserted into the upper third of the core. ***During hot shutdown both banks are moved simultaneously down to the middle third of the reactor. Cold shutdown will be achieved if bank 1 remains in position, while bank 2 continues to the fully inserted position.*** Details of possible variations will be calculated during the conceptual design phase. When power is cut to the drive motors (scram activation), the rods are inserted by gravity. During this event, the drop velocity of the RCS units is limited to a pre-determined value." (Sec. 4.2.1.6, p. 4-32) – *Reserve Shutdown System (RSS)* – "***The purpose of the Reserve Shutdown System (RSS) is to maintain the reactor in a subcritical state during shutdown***….. The RSS consists of eight units that can insert Small Absorber Spheres into the eight borings of the central reflector. Small Absorber Spheres are typically inserted to shut the reactor down to 'cold' conditions for maintenance operations. When inserted, the RSS keeps the reactor subcritical to an average core temperature of 100ºC or less. The RSS neutronic function is thus to act as an absorber in the lower part of the reactor that is out of reach of the solid control rods. The presence of the small absorber spheres creates a negative reactivity which ensures subcriticality….. When shutdown is required, the valves of the small absorber spheres storage units are opened to allow the small absorber spheres to flow under gravity into the central reflector borings. The small absorber spheres are removed from the channels (all eight channels are removed at the same time) and transported back via the sphere return pipe to the feeder bin by means of a gas transport system. The feeder bin distributes the small absorber spheres to the eight small absorber spheres storage containers. Gas flow from the FHSS blower fluidizes and moves the small absorber spheres. During small absorber spheres transport, the FHSS does not transport fuel. The FHSS is isolated from the reactor and the RSS switches to small absorber spheres transport mode. The small absorber spheres units, interfacing with the RPV and CB, operate under the same pressure and temperature as the reactor, and therefore small absorber spheres can only be transported at gas temperatures amenable to the valves and other components wetted by gas flow." (Section 16.1.1.4.2, p. 24) The ASTRA critical facility represents a cylindrical side		

Table 1B (WEC) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		reflector consisting of graphite blocks with an octagon shaped core in the centre and a solid cylindrical centre column. The core is filled with fuel spheres and absorber spheres. Control rods, shutdown rods and a single regulating rod are situated in the first set of blocks closest to the core in the side reflector. ***This allows different critical configurations to check single control rod reactivity worth's or different combinations to look at interference (or shadowing) effects and permits better V&V of control rod models and methods used in the analysis tools***. (Section 16.2.1.2, p.16-29) Completed: Detailed core neutronic design and shutdown system analysis. Establish the core layout, ***control system, neutronic behavior in steady state and transients as well as safety and fuel performance analysis.*** (Section 16.2.1.2, p.16-29) In progress: Establishment of analysis capability for ***reactivity transients***. Establish the know-how of ***performing control rod withdrawal,*** xenon oscillations, SSE, thermal transients, etc. (Section 16.2.1.7, p. 16-32) In progress: Control of Reactor. Understanding of the correct methods to control the flux and Reactor Outlet Temperature, as well as ***estimating the reactivity and the shutdown margin***.		
B-3	Sudden positive reactivity insertion due to pebble core compaction (packing fraction) due to earthquake.	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). (Sec. 16.2.1.1.2.2, p. 16-20) – *Heat Transfer Test Facility* – "The HTTF facility will consist of a number of smaller test sections that will be used for separate effects tests and a main test section that will be used to perform integrated effects tests. The smaller test sections will consist of a scaled down pebble bed and a number of duct-type sections packed with pebbles to represent pebble bed sections with predetermined homogeneous porosities. The main test section will represent an annular pebble bed and it will have the capability to heat the pebble bed (made up of graphite pebbles) and to characterize the heat transfer behavior of such a pebble bed. It is envisaged that the HTTF shall fulfill the needs for tests to characterize the following main phenomena required for simulating heat transfer in a pebble bed: • Pebble bed effective conductivity, which is a combined coefficient representing conductive and radiation heat transfer, required at the pebble bed centre region and wall regions respectively. • Convection heat transfer coefficient required at the pebble bed centre region and wall regions respectively. • Pebble bed pressure drop correlation. • Braiding effect correlation, which defines the mixing effect of gas flowing	(None)	It appears that WEC has the test facilities to simulate a condition of increased packing density in the PBMR core; however ***there is no indication that this item has been specifically addressed in the WEC PCDR.***

Table 1B (WEC) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		through a pebble bed. • Natural convection heat transfer coefficient." (Sec. 16.2.1.1.6, p. 16-25) – "NACOK stands for Natural Convection with Corrosion. The main section of this facility is made up of a vertical channel of 300 mm x 300 mm and 7.5 m tall. The experimental channel is composed of sections representing a bottom reflector, sphere packing (pebble bed) and a top reflector. The experimental set-up was designed to be able to represent different breaks in pipes connecting to the reactor. Breaks can be created that simulate the coaxial duct (reactor outlet pipe), the defueling chute at the bottom of the reactor and the fuelling line at the top of the reflector. By a sectional design, different core heights can also be simulated. All sections of the experimental channel and of the return pipe can be heated to accident-relevant temperatures. At different positions, the local gas compositions can be measured." (Sec. 16.2.1.1.7, p. 16-27) – "The SANA test facility consists of a heated pebble bed inside a furnace to simulate the thermal conditions of an HTGR-core. Different heater configurations are possible but Figure 16.2-11 shows a schematic of the test facility with a single central heating element. The diameter of the pebble bed is 1.5 m and the height is 1.0 m. The overall height of the facility is 3.2 m and the maximum heating capacity of the single central heating element is 35 kW. The top and bottom of the facility are well-insulated while the outside of the furnace is open to atmosphere. More than 50 steady-state as well as some transient tests were carried out in the facility. In these experiments all the main parameters of a pebble bed were varied, such as pebble material, pebble diameter, gas type, heating power and heating geometry."		
B-4	For tests at both PMRs and PBRs, consideration should be given (at least in the first core) to use of high-temperature in-core neutron detectors that can provide maps of axial and azimuthal power distributions and core-inner-to-outer-radius power tilts; these detectors would likely be located only in the inner and outer reflectors rather than in the core, due to temperature and connection limitations. • PMR concern - Whether improper axial-loading of fuel blocks during refueling can lead to an undetected power distribution anomaly and result in excessive operating fuel temperatures. • PBR concern - Radial and azimuthal power distributions in the mixed-fuel pebble bed are not well known, and there are indications from melt-wire tests conducted in the AVR (Germany) suggesting that pebbles near the	See item A-1 for a discussion of instrumentation provided for the PBMR Nuclear Heat Supply System.	(None)	There is discussion of instrumentation and monitoring, however ***there is no indication that in-core instrumentation has been specifically addressed in the WEC PCDR.***

Table 1B (WEC) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
	walls of the reflector experienced unexpectedly high fuel temperatures.			
B-5	In both the PMR and PBR, control rod misalignments in the outer reflector during operation would result in azimuthal power tilting that could cause xenon-135-induced oscillations when the misalignment is corrected; however, this needs to be verified by analysis and confirmed by test.	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). (Section 16.2.1.2, p.16-29) - In progress: Establishment of analysis capability for reactivity transients. Establish the know-how of performing control rod withdrawal, xenon oscillations, SSE, thermal transients, etc.	(None)	The general need to understand xenon oscillations was identified. ***However, there were no specifics in the WEC PCDR that mentioned outer reflector control rod misalignments leading to azimuthal power tilting and possible xenon oscillations.***
B-6	Replacing helium with a hydrogen-bearing compound such as in a steam/water ingress event may produce a pronounced positive reactivity. Steam/water ingress tends to have a positive reactivity effect due to increased neutron moderation and reduced neutron leakage.	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). See item F-4 for information relating to steam generator design and design development. This addresses water ingress from the standpoint of the efforts that are being dedicated to ensure steam generator reliability and separation from the primary system.	(None)	There is discussion of reactivity control, and also of steam generator design to prevent water ingress; however ***there is no indication that this item has been specifically addressed in the WEC PCDR.***
B-7	With a higher atomic mass moderator such as carbon, the mean thermal energy of neutrons will be higher than that for hydrogen bound with oxygen in water; that is, graphite will tend to produce a "harder" thermal-neutron energy spectrum than would water-moderated systems. Thus, the moderator temperature-dependent reactivity coefficient (MTC) in both PMR and PBR depends upon the change of thermal-neutron energy spectrum with temperature, with possibly large effects on reactivity. Concerns are for effects on core transient behavior and passive safety shutdown characteristics.	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). (Section 11.2.2, p.11-12) - The NHSS also has a negative temperature coefficient, which results in the reactivity and consequently the neutronic power to counteract temperature changes. The NHSS is therefore to a large extent self-regulating and minimum control interaction is required to maintain the reactor outlet temperature at a given value.	(None)	There are discussions of the reactivity control systems and inherent ability of the core to resist increased reactivity, however t***here is no indication that "harder thermal neutron energy spectrum" and its "possibly large effects on reactivity" have been addressed in the WEC PCDR.***
B-8	Variations in fuel enrichments, kernel diameters, coatings, and density of packing (PMR vs. PBR) must be accounted for in calculating the neutron reaction self-shielding effects in both the resonance or epithermal region and the thermal region of the neutron energy spectrum, to properly calculate the Doppler fuel temperature coefficient of reactivity and the MTC.	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). (Sec. 5.2.5, pp. 5-25 thru 5-34) – *Fuel Supply and Fabrication* – This section provides substantial detail on the manufacturing process intended for the PBMR fuel.	(None)	There are discussions of reactivity control and fuel fabrication; however ***there is no indication that this item regarding variations has been specifically addressed in the WEC PCDR.***
B-9	Due to concerns over control rod drive reliability and re-criticality after Xenon-135 decay, the plant operator retains the safety function of achieving long-term hot and cold shutdown during an extended ATWS; and the equipment used by the operator to carry out this safety function, whether located in the control room or in a remote location,	***There is no indication that this item has been specifically addressed.***	(None)	***There is no indication that this item has been specifically addressed in the WEC PCDR.***

Table 1B (WEC) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	must be appropriately qualified to execute that safety function.			
B-10	The uniqueness of configuration (tall, thin annular core) of current PMR and PBR designs and high operating temperatures require detailed reactor physics testing of the first unit as a function of core burnup, and of the start-ups of the second and perhaps third cycles. Attention should be paid to the instrumentation needs for these tests since neutron sensors must be both distributed and inter-calibrated to infer power distributions. Neutron detectors used in test measurements should also be sensitive enough to measure reactivity and changes in flux levels and distributions.	See item C-6 for discussion of PBMR test facilities. (Sec. 4.2.10, p. 4-68 thru 4-76) - *NHSS Control and Instrumentation System* – "The NHSS Control and Instrumentation System comprise only of equipment found in the Nuclear Heat Supply Building and all operator interaction is performed in the Central Control & Supervisory System (CCSS) as described in the Section 9: Balance of Plant Systems. The primary systems comprising the NHSS Control and Instrumentation system are the Operational Control System (OCS), the Equipment Protection System (EPS) and the Reactor Protective System (RPS). The OCS monitors and controls the NHSS systems throughout their normal operating range. The EPS detects operating regimes or operating conditions that may be harmful to NHSS equipment, and takes appropriate action to prevent or minimize potential damage. The RPS automatically initiates RUS protection whenever pre-established set points are exceeded…. The RPS consists of three subsystems: Reactor Trip System (RTS), Post Event Instrumentation System (PEI) and the Manual Diverse Shutdown System (MDSS). The RPS provides functions to prevent exceeding predefined safe operating limits and to provide information to operators in the event of nuclear accidents. The Protection System is implemented using a Class 1E qualified digital platform that is capable of performing logical operations as well as algorithmic processing. The same platform is used for both the RTS and the PEI applications. The MDSS is a hard-wired system that allows tripping the breakers without dependence on any software. All portions of the RPS are treated as Class 1E Structures, Systems and Components (SSC)…. The RTS automatically prevents operation of the RUS in an unsafe condition by shutting down the RUS whenever predetermined operating limits are approached, or when Design Basis Accident conditions are detected. The operating limits are selected, based on initial conditions (RUS power and outlet temperature) assumed in the safety analysis…. The MDSS is a hardwired system that enables operators to manually initiate reactor trip, RCS and RSS activation functions, from both the Main Station Control Room and the PEMRR. The MDSS controls are supported by monitoring instrumentation associated with the PEI system or the Plant Computer displays to provide the operator the capability to know when to take the appropriate manual action. For each reactor trip function (RS and RSS activation), a set of three switches that are individual and redundant are provided in the Main Station Control Room and a set of three switches that are individual and redundant are provided in the PEMRR. Thus, a total of 4 sets of three switches each comprise the MDSS for the protection system." (Sec. 11.2.2, p. 11-12) – *NGNP Integrated Control Philosophy* – "Controllability and transient performance of the NGNP are ultimately determined by the dynamic characteristics of the NHSS, HTS, HPS and PCS. The NGNP demonstration plant	(None)	There are discussions of Nuclear Heat Supply System instrumentation, and of the general PBMR control philosophy; however, ***there is no indication that this item regarding detailed reactor physics testing and associated instrumentation has been specifically addressed in the WEC PCDR.***

Table 1B (WEC) – REACTOR PHYSICS AND NEUTRONICS - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		has the following dominant system characteristics: • The thermal response of the NHSS is slow, since the graphite-moderated core has a large thermal capacity relative to its heat generation and removal rates. However, the NHSS is the most critical system and will govern the overall control philosophy. • The advantage is that the large thermal capacity of the core allows relatively fast load changes of the system without requiring fast response from the core. In principle, the energy stored in the core can be tapped or additional energy can be stored, with minimum core temperature changes. ***The NHSS also has a negative temperature coefficient, which results in the reactivity and consequently the neutronic power to counteract temperature changes.*** The NHSS is therefore to a large extent self-regulating and minimum control interaction is required to maintain the ROT at a given value. • Another advantage is that the HTS can also be controlled easily by controlling the speed of the PHTS and SHTS circulators. A loss of outside electric load, PCS trip or HPS trip will result in temperature changes in the PCS, which will propagate to the SHTS, PHTS and the NHSS. Temperature changes can be monitored and controlled by manipulating the mass flow rates through the PCS, SHTS and PHTS. Thus, in principle the NGNP consists of an inherently stable and slow acting NHSS coupled to a stable HTS, HPS and PCS that will require active control to remain stable under all anticipated operating scenarios.”		

Table 1C (WEC) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
C-1	*General Safety Analysis/Safety Document Needs* • Comprehensive description of the NGNP safety philosophy, a listing of the components involved, and the conditions under which these components are expected to perform their safety functions. • Explanation of how this philosophy meets the defense-in-depth approach and, in particular, answers to the following: ○ Will the components that perform a safety function (retain FPs) be classified as safety-related components, with the imposition of equipment qualification, in-service inspections, and/or Technical Specifications LCOs and SRs? ○ How will aging issues be addressed? If the safety function of a component is to retain FPs on its surface during adverse conditions, how can it be ensured that this function can be retained for long periods (decades), despite the possible presence of other long-term surface degradation mechanisms? ○ Will the surface state of a non-replaceable or difficult-to-replace component be reactivated by chemical action or cleaning during its service life? • A sound basis for the selection of the physical models and the data for these models must be justified. • The materials to be used and their sensitivity on the transport case must be identified. • Once the actual reactor design is available, the transport pathways that result from the accident conditions must be identified, along with the relevant models and data needed for the resulting calculations. • Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.	See item A-8 for discussion of modeling efforts. Section 2.2.2, pp. 2-36 thru 2-38, *Regulatory Requirements for the NGNP*, provides a detailed enumeration and description of the regulatory/safety requirements that will be met by the PBMR/NGNP, including ***high level safety goals, site level consequence and environmental limits, dose limits for anticipated operational occurrences, dose limits for design basis events, safety goals for beyond design basis events, state and local regulatory requirements, and industry codes and standards***. (Sec. 4, pp. 4-9 thru 4-85) – *Nuclear Heat Supply System* – This section contains ***detailed descriptions of the reactor system, layout, roles of components, and supporting systems. Fuel design is addressed*** in section 4.2.1.1, p. 4-17, and throughout section 5, *Reactor Fuel*. (Sec. 11.2.2, p. 11-12) – *NGNP Integrated Control Philosophy* – "Controllability and transient performance of the NGNP are ultimately determined by the dynamic characteristics of the NHSS, HTS, HPS and PCS. The NGNP demonstration plant has the following dominant system characteristics: • The ***thermal response of the NHSS is slow, since the graphite-moderated core has a large thermal capacity*** relative to its heat generation and removal rates. However, the NHSS is the most critical system and will govern the overall control philosophy. • The advantage is that the large thermal capacity of the core allows relatively fast load changes of the system without requiring fast response from the core. In principle, the energy stored in the core can be tapped or additional energy can be stored, with minimum core temperature changes. The ***NHSS also has a negative temperature coefficient, which results in the reactivity and consequently the neutronic power to counteract temperature changes***. The NHSS is therefore to a large extent self-regulating and minimum control interaction is required to maintain the ROT at a given value. • Another advantage is that the HTS can also be controlled easily by controlling the speed of the PHTS and SHTS circulators. A loss of outside electric load, PCS trip or HPS trip will result in temperature changes in the PCS, which will propagate to the SHTS, PHTS and the NHSS. Temperature changes can be monitored and controlled by manipulating the mass flow rates through the PCS, SHTS and PHTS. Thus, in principle the NGNP consists of an inherently stable and slow acting NHSS coupled to a stable HTS, HPS and PCS that will require active control to remain	NHSS-01-01 thru NHSS-01-03 (fuel testing) NHSS-02-01 and NHSS-02-02 (graphite properties) HTS-01-01 thru HTS-01-019 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX ceramics) HTS-04-01 (high temp. ducts and insulation)	The WEC PCDR has recognized most of the safety analysis/safety documentation needs detailed in this item, ***with the exception of the following:*** • ***Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.***

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	• The fuel database must be developed, as well as fuel-failure models and fuel material properties (both measurable and process controlled).	stable under all anticipated operating scenarios." (Sec. 12.1.2, p. 12-12) – *Maintenance requirements for Systems and Components* - The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. General requirements for each system include that the design provides ***access to the pressure boundaries to permit in-service inspection as required by appropriate sections of the ASME B&PV Code***, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible. (Sec. 12.3.1.4, p. 12-23) – *Reactor Cavity Cooling System* - The RCCS is designed for life of the plant, thus ***no scheduled maintenance other than In-Service-Inspection*** (ISI) is envisaged. Provision is made in the design for 'as required' inspection and repair of the RCCS. Special tools will be developed up to the point of having a basic design in place, so that should a repair become necessary, the required equipment can be procured at short notice. As far as possible, 'off-the-shelf' equipment is used. (Sec. 14, p. 14-7) – *Safety: Summary and Conclusions* -."The ***safety design philosophy is to apply the principles of defense-in-depth at a fundamental level in which a diverse combination of inherent reactor characteristics, passive design features and Structures, Systems and Components (SSCs), active engineered systems, and operator actions are deployed to maintain the integrity of robust passive barriers to radionuclide release***. The reactor-specific key safety functions are derived in a top-down manner with the objective of protecting the integrity of the multiple barriers to radionuclide release; these include the control of heat generation, control of heat removal, control of chemical attack, maintenance of core and reactor geometry, and maintenance of the reactor building structural integrity. A fundamental aspect of the safety design philosophy is to provide the capability to perform safety functions first through the selection of inherent reactor characteristics and engineered systems that operate on passive design principles and then to support these safety functions with combinations of diverse active engineered systems and operator actions. The safety design approach for the NGNP is derived from a risk-informed and performance-based model of defense-in-depth. This approach recognizes three major elements: Plant Capability Defense-in-Depth, Programmatic Defense-in-Depth, and a Risk-Informed Evaluation of Defense-in-Depth. These three elements enable the examination of a plant's defense-in-depth capability from different perspectives including those of: • Designing the plant and the capabilities of its SSCs that perform safety functions		

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		• Defining the programs that ensure the plant will be built as designed and will operate safely throughout the plant lifetime while preserving the intended defense-in-depth capabilities. • Evaluating how the plant performs its safety functions in the prevention and mitigation of accidents and determining the adequacy of defense-in-depth. The NGNP safety design approach is framed in terms of ***reactor-specific safety functions*** that were developed from the top goal of retaining the inventory of radionuclides primarily within the fuel and then considering the specific functions that when satisfied would ***protect the integrity of the fuel and other radionuclide transport barriers.*** The required safety functions include those to: • Control heat generation (reactivity) • Control heat removal • Control chemical attack • Maintain core and reactor vessel geometry • Maintain reactor building structural integrity The safety evaluation for the NGNP will be performed using a risk-informed and performance-based approach. The key elements of this technology-neutral approach include: (1) the use of accident frequency vs. radiological dose criteria that are derived from current U.S. licensing requirements, referred to as Top Level Regulatory Criteria (TLRC), (2) use of a full-scope Probabilistic Risk Assessment (PRA) to select the Licensing Basis Events (LBEs), (3) development of reactor-specific functions, selection of the corresponding safety-related SSCs, and their regulatory design criteria, (4) deterministic design conditions and special treatment requirements for the safety-related SSCs, and (5) a risk-informed evaluation of defense-in-depth." (Sec. 14.2.4, p. 14-16) – *Selection of Design Features to Perform Safety Functions* – "The NGNP safety design is based on meeting the following objectives that specifically incorporate the defense-in-depth approach described above: • Provide safe, economic and reliable nuclear heat to the HPS and the PCS, which produce hydrogen and electricity respectively • Select compatible fuel, moderator, & coolant with inherent safety characteristics • Utilize proven technologies to the maximum extent practical • Design reactor with inherent characteristics and passive safety features sufficient to protect the public as the primary strategy for Plant Capability Defense-in-Depth • Supplement with active design features and SSCs for investment protection		

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		and as a secondary strategy for Plant Capability Defense-in-Depth Important inherent characteristics of the NGNP design include: • Ceramic-coated pebble fuel • Capability to maintain integrity at high temperatures • Chemically compatible with coolant and moderator • Graphite moderator • Capability to maintain integrity at high temperatures • High thermal heat capacity • Chemically compatible with fuel and coolant • Large neutron migration length for neutron stability • Helium coolant • Single phase over all normal and accident conditions • Chemically and neutronically inert • Low stored thermal energy In addition to these inherent characteristics, the NGNP has both passive and active ***design features to perform defense-in-depth functions***, as discussed below. The NGNP safety design approach is to provide inherent characteristics and passive SSCs that are sufficient to protect the public and to meet the Top Level Requirements and to provide the primary strategy for Plant Capability Defense-in-Depth, and then to provide additional active SSCs to provide additional levels of defense-in-depth as well as to meet user requirements for plant availability and investment protection. A summary of the inherent characteristics and passive SSCs that are available to support each required safety function, as well as the additional active SSCs that support these functions is provided….” (Sec. 15.1.1.1, p. 15-11) – *Fuel Type and Form* – “The PBMR core consists of fuel elements containing uranium dioxide coated particles that generate heat by means of fission reactions. The fuel pebble consists of uranium dioxide coated fuel particles and matrix graphite pressed into a spherical shape. A fuel sphere is divided into two regions. The inner spherical region is known as the fuel region, while the outer shell surrounding the fuel region is known as the fuel-free region. The fuel region of each fuel sphere contains a large number of evenly dispersed spherical particles known as coated particles in which the fuel is contained while there are no coated particles in the fuel-free region. The design of the coated particles and fuel sphere is summarized in…” (Note: additional information follows in this section that provides details on the fuel design, materials, fabrication, transportation, receipt, onsite		

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		handling, and loading into the reactor core.) (Sec. 16.2.1.1.5, p. 16-23) – *Experimental Plate-out Facility* – "The purpose of tests done in the Experimental Plate-out Test Facility (POTF) is to obtain representative PBMR material plate-out parameters and, if possible (to be determined by a second feasibility study), to ***obtain graphite dust and fission product interaction data***." (Note: This is followed with descriptions of the Experimental Plate-out Loop and the Isopiestic Plate-out facility.) (Sec. 16.2.2, p. 16-35) – *Design Data Needs (Nuclear Heat Supply System)* – "Three DDNs have been identified pertaining to the NGNP Fuel. The first of these DDNs (NHSS-01-01) identifies the need for data to ***extend the irradiated fuels qualification database*** from the temperature-burnup envelope of the PBMR Demonstration Power Plant (DPP) to that of the PBMR NGNP. The second DDN (NHSS-01-02) specifies data to correspondingly extend the heat up data pertaining to accident conditions. The third DDN (NHSS-01-03) provides for an extension of the temperature-fluence envelope of the Fuel Graphite to that required by the NGNP. In all three cases, the extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature." (Sec. 16.2.3.1, p. 16-36) – *Fuel Qualification R&D Plan* – "Two supplemental irradiation tests are planned for the PBMR NGNP to ***extend the database supporting the fuel*** for the PBMR DPP. The first of these irradiation tests, which is proposed to start in FY2009 will use pre-production fuel. The second, proposed to start in 2012 will use actual production fuel."		
C-2	*Model Development and V&V* - Physical models and the supporting mathematical methods, addressing: • Nuclides of interest • Fission product release from the fuel • Diffusion, adsorption, and desorption in graphite and fuel matrix materials • Adsorption, desorption, and in-diffusion in reactor system metals • Chemical and physical forms of the FPs in the coolant • Tritium transport models • Aerosols and dusts that plate-out on reactor system components and their mobility • Fission product reactions with the confinement building	See item A-8 for model development information. (Sec. 4, pp. 4-9 thru 4-85) – *Nuclear Heat Supply System* – This section contains detailed descriptions of the reactor system, layout, roles of components, and supporting systems. ***Fuel design is addressed in section 4.2.1.1, p. 4-17, and throughout section 5, Reactor Fuel***. (Sec. 5.2.2.1.4, p. 5-14) – *Silicon Carbide Layer* – "***The production of fuel spheres having coated particles with intact SiC layers and the assurance that these layers will remain intact under all foreseeable fuel core conditions form the primary barrier to the release of radiation from NGNP***. When SiC is deposited from methyltrichlorsilane under the correct conditions, a layer of nearly 100% theoretical density is obtained. At high temperatures, the ILTI and OLTI layers partially lose their ability to contain cesium, silver and strontium. The purpose of the SiC layer is to prevent the release of these fission products into the graphite matrix,	NHSS-01-01 (Data to extend the irradiated fuels qualification database) NHSS-01-02 (Extend heat up data under accident conditions. NHSS-01-03	The WEC PCDR has recognized most of the needs detailed in this item, ***with the exception of the following:*** • ***Fission product reactions with the confinement building materials*** • ***Reactions of the reactor system components and fission products with air or steam*** • ***Plume models that transport the released material beyond the reactor building*** • ***Determination of the safety function of each subsystem and the level of FPT attenuation required.***

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	materials
	• Reactions of the reactor system components and fission products with air or steam
	• Plume models that transport the released material beyond the reactor building
	• Determination of the safety function of each subsystem and the level of FPT attenuation required.
	• Determination of level of sensitivity to component uncertainties and how this reflects on the physical models.
	• Estimation of difficulty in obtaining the data and conducting the testing to support the safety case.
	• Scoping of how V&V can be performed.
		and then into the reactor helium stream. The SiC layer thus acts as the principal pressure and fission product retention barrier in the coated particle. The coated particle structure results in the SiC layer being kept under compression as long as possible by its interaction with the ILTI and OLTI pyrocarbon layers as described above. The silicon carbide layer has a thickness of approximately 35 µm and a density greater than approximately 3.2 gm/cm^{3}. ***The production of fuel spheres having coated particles with intact SiC layers and the assurance that these layers will remain intact under all foreseeable fuel core conditions is the cornerstone of the safety design approach in which the fuel is the primary barrier to the release of radionuclides***."
		
		(Sec. 5.2.4.2, p. 5-17) – *Testing and Qualification (Fuel)* – "Even though test results from the German pebble-bed reactor program are available, and is the basis for the PBMR DPP, expected operational parameters specified for the NGNP was not envisaged during PBMR efforts or the German program. Therefore to ensure the safe application of PBMR DPP based fuel in the NGNP program, the following ***parameters and/or aspects are considered to be important in terms of fuel testing and qualification*** must be considered:
		• Expected/specified normal operating condition parameters
		    ○ Maximum fuel temperatures
		    ○ Percentage burn-up
		    ○ ***Percentage fission product release at specified temperatures***
		• Expected/specified adverse (accident) operating condition parameters
		    ○ Maximum temperatures
		    ○ Percentage burn-up
		    ○ ***Percentage fission product release at specified temperatures***
		• Statistical requirements of tests and qualification samples to ensure confident and safe application thereof for design and further operational specification refinements
		• Pre-designate procedures and facilities for pre- and post-irradiation tests to support qualifications"
		
		(Sec. 6.4.2.7, p. 6-100) – *Tritium Transport* – "At the operating temperatures of the NGNP, the ***transport of tritium*** through intact metallic heat exchangers will be enhanced. Whether this is a major or minor concern has yet to be determined and will depend on a number of technical factors, such as the materials selected for the IHX and PCHX, the diffusion coefficients associated with tritium transport through the selected materials, the chemistry of the PHTS and SHTS helium and the effectiveness of the PHTS and SHTS helium service systems in removing tritium. A further issue relates to establishing the appropriate limits on tritium transport to the
			(Extend temperature-fluence envelope of fuel graphite)	• ***Determination of level of sensitivity to component uncertainties and how this reflects on the physical models.***
				• ***Estimation of difficulty in obtaining the data and conducting the testing to support the safety case.***

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		product hydrogen and/or steam, which, presumably would further depend on their intended use. This future study will consider these technical and regulatory issues as input to Conceptual Design." (Sec. 14.3, p. 14-21) – *Risk-Informed Performance-based Safety Evaluation* – "In support of the pre-application interactions with the Nuclear Regulatory Commission (NRC), leading to the U.S. design certification of the PBMR, a risk-informed and performance-based approach has been proposed as described more fully in References…This proposal is a proactive response to NRC policies on the expanded use of PRA methods to in the licensing process as well as its Advanced Reactor Policy. This approach builds upon the risk-informed licensing approach that was developed by the Department of Energy (DOE) for the MHTGR in the 1980's and the more recent experience with the Exelon-proposed licensing approach for the PBMR. The approach is also consistent with the basic elements of the Technology-Neutral Framework that is under development by the NRC in support of new plant licensing. The key elements of this technology-neutral approach include: (1) the use of accident frequency vs. radiological dose criteria that are derived from current U.S. licensing requirements, referred to as Top Level Regulatory Criteria (TLRC), (2) use of a full-scope PRA to select the LBEs, (3) ***development of reactor-specific functions, selection of the corresponding safety-related Systems, Structures, and Components (SSCs), and their regulatory design criteria***, (4) ***deterministic design conditions and special treatment requirements for the safety-related SSCs***, and (5) a risk informed evaluation of defense-in-depth as described in the previous section. The relationships among these elements are described in…" (Sec. 16.2.1.1.5, p. 16-23) – *Experimental Plate-out Facility* – "The purpose of tests done in the Experimental Plate-out Test Facility (POTF) is to ***obtain representative PBMR material plate-out parameters and, if possible (to be determined by a second feasibility study), to obtain graphite dust and fission product interaction data***." (Note: This is followed with descriptions of the Experimental Plate-out Loop and the Isopiestic Plate-out facility.)		
C-3	*Materials/Component Data* - Relevant data on materials or components over the range of interest and data uncertainties (single effects testing), including the following: • Graphite transport property and air/steam erosion data specific to the design material. • Metal alloy data specific to the design material. • Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product. • Data on helium impurities that will likely set the oxygen	See item C-1 for fuels materials characterization and qualification information. See item D-1 for metallic materials characterization and qualification information. See item E-1 for graphite materials characterization and qualification information. Sec. 4.2.1.2, pp. 4-19 thru 4-22, contains a detailed description of the reactor core barrel assembly, including required functions, ***materials specifications***, methods of assembly, and interfaces with other components and systems. Sec. 4.2.1.3, pp. 4-23 thru 4-27, contains a detailed description of the reactor core ceramic structures (top,	NHSS-01-01 thru NHSS-01-03 (fuel testing) NHSS-02-01 and NHSS-02-02 (graphites) HTS-01-01	The WEC PCDR has addressed some of the issues associated with this item, ***with the exception of the following:*** • ***Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product.*** • ***Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other***

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	potential of the system, and the species to be included in an analysis. • Data associated with component aging: surface qualities of the reactor system components after many years of operation. • Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit. • Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. • If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties. • Data regarding turbine or power conversion components that may have to be decontaminated prior to maintenance (initial collection of FPs while in the reactor circuit; decontamination of components; new surface state of the component after decontamination). • Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior,	bottom, side and central reflectors), including required functions, ***materials specifications***, and interfaces with other components and systems. Sec. 4.2.1.4, pp. 4-27 thru 4-29, contains a detailed description of the Reactor Pressure Vessel, including required functions, materials specifications, and interfaces with other components and systems. (Sec. 4.2.7.2, p. 4-60) – *Helium Purification System* – "The Helium Purification System (HPURS) is ***used to provide the required degree of helium purity***. High purity coolant is required in order to minimize corrosion and contamination in the PHTS and SHTS. This is done by bleeding off a partial flow of helium from the PHTS and SHTS. The extraction point is from the highest pressure points, i.e. the PHTS and SHTS circulator discharges within the HTS. This flow is tapped off constantly during operation of the plant. The ***HPS removes chemical gaseous contaminants from the primary coolant within the PHTS*** by the use of, catalysts, adsorbers and the manipulation of helium temperature extracted from the PHTS and SHTS. The required helium purity levels will be confirmed during the conceptual design." (Sec. 16.2.1.1.5, p. 16-23) – *Experimental Plate-out Facility* – "The purpose of tests done in the Experimental Plate-out Test Facility (POTF) is to ***obtain representative PBMR material plate-out parameters and, if possible (to be determined by a second feasibility study), to obtain graphite dust and fission product interaction data***." (Note: This is followed with descriptions of the Experimental Plate-out Loop and the Isopiestic Plate-out facility.)	thru HTS-01-019 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX ceramics) HTS-04-01 (high temp. ducts and insulation) HPS-03-01 thru HPS-03-04 (feed purification)	***(decontamination?) fluid is introduced into the reactor circuit.*** • ***Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions.*** • ***If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties.*** • ***Data regarding turbine or power conversion components that may have to be decontaminated prior to maintenance (initial collection of FPs while in the reactor circuit; decontamination of components; new surface state of the component after decontamination).*** • ***Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior***
C-4	Reactor component and confinement/containment configuration and their relative roles in the safety case • Respective roles of the reactor circuit and containment or confinement system must be known before their modeling adequacy can be determined. • Estimate of source and budgeting of FP holdup among the fuel form, reactor circuit components, mobile elements such as dust, and the reactor building, as a means of focusing components to be emphasized in analysis. • Determination of transport pathway, goals for FP retention at each step in the pathway, local (accident) operating environment at each step of the pathway.	See item A-8 for model development information. (Sec. 1.9.2.2, p. 1-31) – *Future Studies* • *PBMR Reactor Building Requirements, Functions and Features* – "***Prepare a white paper on the requirements, functions, and design features of the PBMR reactor building.*** Clearly identify all requirements, including performance, costs, investment protection and safety. Perform an analysis of the safety considerations of the PBMR designed with a confinement versus an LWR-type containment. Identify benefits and adverse consequences of selecting an LWR-type containment. Consider the citadel concept of the DPP for either alternative. • *Dust Control* - Prepare a ***white paper on dust control*** for the NGNP NHSS. Determine whether it is necessary to test the IHX for dust blockage and release	(None)	The part of this item addressing the general understanding of roles of reactor building and equipment has been addressed, and an experimental facility intended to research issues of plateout and dust are described. ***However, the central issue of budgeting and modeling fission product hold-up among specific design features has not been addressed in the WEC PCDR.***

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		during accidents. Consider similarities and differences with the DPP and its approach to dust management." (Sec. 4, pp. 4-9 thru 4-85) – *Nuclear Heat Supply System* – This section contains detailed descriptions of the ***reactor system, layout, roles of components, and supporting systems. Fuel design is addressed*** in section 4.2.1.1, p. 4-17, and throughout section 5, *Reactor Fuel*. (Sec. 10.2.1.1, p. 10-21 thru 10-23) – *Nuclear Heat Supply Building* - "Confinement requirements are identified in Section 4 with the Reactor Building HVAC system. The HVAC system and internal partitions are designed maintain separation of highly contaminated or potentially contaminated compartments from non contaminated or less contaminated compartments. ***The building is designed to prevent release of all liquids to the environment. The building is designed to withstand the pressurization effects of a major helium coolant depressurization. The building is also designed for tornado wind loads and missiles, in accordance with USNRC regulatory requirements identified in…The potential and consequences of airplane impact are also considered. The potential for volcanic ash intrusion is addressed in the design of the NHSS HVAC system as discussed in Section 4. Fire area separation is as determined in the fire hazard analysis to be completed in the preliminary design phase and uses minimum 3 hour separation walls. This building is required to be fully functional in response to design basis events as required in Sections 4, 14 and 17. Key functions include structural integrity to maintain reactor core geometry, and confinement of radiological materials as required, to ensure safety of the public***…. The reactor pressure vessel is supported by the reactor cavity walls. The reactor cavity structure is a thick reinforced concrete cylinder. Shielding requirements primarily drive the dimensions of this interior structure….From a functional perspective, the ***confinement zone around the citadel forms an essential part of the confinement system. This confinement zone is sealed sufficiently to limit leakage of airborne radiological material***. Leakage from the confinement zone through doors, penetrations and hatches is limited to ensure that within this space can be maintained by the HVAC system at an air pressure of less than atmospheric such that air flow is inward, and any air exchange with the environment is through filters….Adjacent cavities within the citadel are not interconnected during normal operation. In the event of a depressurization of the helium piping pressure boundary, the pressure transients are relieved via the Pressure Relief System (PRS). Burst panels within the PRS open during postulated accidents when the over-pressure in the citadel and confinement zone exceeds the set limit of the burst panels. The release goes up the depressurization shaft, through filters and is released in an upward direction."		

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Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		(Sec. 16.2.1.1.5, p. 16-23) – *Experimental Plate-out Facility* – "The purpose of tests done in the Experimental Plate-out Test Facility (POTF) is to ***obtain representative PBMR material plate-out parameters and, if possible (to be determined by a second feasibility study), to obtain graphite dust and fission product interaction data***." (Note: This is followed with descriptions of the Experimental Plate-out Loop and the Isopiestic Plate-out facility.)		
C-5	Computational software or other methods for determining the quantitative results • Data collection and proof that the selected model is adequate under all the normal and accident conditions of interest. Need to know that model envelops releases, and have reasonable proof that the model predicts an upper limit. • Need to have a description of the physical models and the reactor configuration, showing that the models are appropriate for the conditions of interest. • Need to have the data required for the models: single-effects data for each material and component acquired under individual testing, and integral data designed to show that the codes get the correct answer for a complete system under the conditions of interest.	See item A-8 for model development information. See Item C-6 for information on testing programs. (Sec. 4, pp. 4-9 thru 4-85) – *Nuclear Heat Supply System* – This section contains ***detailed descriptions of the reactor system, layout, roles of components, and supporting systems. Fuel design is addressed*** in section 4.2.1.1, p. 4-17, and throughout section 5, *Reactor Fuel*. (Sec. 14.3, p. 14-21) – *Risk-Informed Performance-based Safety Evaluation* – "In support of the pre-application interactions with the Nuclear Regulatory Commission (NRC), leading to the U.S. design certification of the PBMR, a risk-informed and performance-based approach has been proposed as described more fully in References…. This proposal is a proactive response to NRC policies on the expanded use of PRA methods to in the licensing process as well as its Advanced Reactor Policy. This approach builds upon the risk-informed licensing approach that was developed by the Department of Energy (DOE) for the MHTGR in the 1980's and the more recent experience with the Exelon-proposed licensing approach for the PBMR. The approach is also consistent with the basic elements of the Technology-Neutral Framework that is under development by the NRC in support of new plant licensing. The key elements of this technology-neutral approach include: (1) the use of accident frequency vs. radiological dose criteria that are derived from current U.S. licensing requirements, referred to as Top Level Regulatory Criteria (TLRC), (2) use of a full-scope PRA to select the LBEs, ***(3) development of reactor-specific functions, selection of the corresponding safety-related Systems, Structures, and Components (SSCs), and their regulatory design criteria, (4) deterministic design conditions and special treatment requirements for the safety-related SSCs***, and (5) a risk informed evaluation of defense-in-depth as described in the previous section. The relationships among these elements are described in…" (Sec. 16.2.1.5, p.16-30) – *Fuel Qualification* – "PBMR (Pty) Ltd has embarked on an intensive fuel qualification programmed to ensure that the quality of their manufactured fuel is similar to the German LEU-TRISO fuel. This program is currently underway and will ***qualify the fuel in terms of physical properties, maximum fuel temperatures, percentage burn-up and fission product release***	NHSS-01-01 (Data to extend the irradiated fuels qualification database) NHSS-01-02 (Extend heat up data under accident conditions. NHSS-01-03 (Extend temperature-fluence envelope of fuel graphite) NHSS-02-01 and NHSS-02-02 (Extend irradiated materials qualification database for Reflector Graphite) HTS-01-1 thru HTS-01-19 (IHX metallics)	The WEC PCDR has recognized the needs for computer model development and supporting testing. Reactor configuration is available in the PCDR.

Table 1C (WEC) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		for both normal operating and accident conditions. Statistical requirements of tests and qualification samples will also be investigated to ensure confident and safe application thereof for design and further operational specification refinements. Part of the qualification program will consist of irradiating a number of fuel spheres, containing a statistically significant number of coated particles, to full PBMR irradiation requirements in a material testing reactor." (Sec. 16.2.1.5, p. 16-30) – *Core Structural Ceramics Qualification* – "For the PBMR DPP, Core Structural Ceramics include the Reflector Graphite that establishes the core geometry, Carbon Fiber Reinforced Carbon (CFRC) components associated with the core lateral restraints and tie rods supporting the upper reflector, and ceramic components used to provide thermal insulation below the core."	HTS-02-01 thru HTS-02-06 (IHX ceramics)	
C-6	Integral testing over a wide range of conditions to support the development of computational methods and the quantification of the data and associated uncertainties • Attempt to use existing data from past programs to the degree appropriate. • Planning of any in-pile loop program would require a complete description of the normal operating environment and of the accidents, along with any scaling factors. Extensive modeling will be necessary to design the loop and determine off-normal conditions that the loop can be expected to simulate. Model predictions (with the previously collected single-effects data) will need to be made.	See item A-8 for model development information. (Sec 11.1.2, p. 11-10) – *Overview of the NGNP Plant Simulator* – "The NGNP ***Plant Simulator will provide realistic, simulated plant and control responses*** in a computing environment. The Plant Simulator aims to develop, test, verify and validate simulation models and modules as well as control algorithms and strategies for the NGNP. An interface will be provided, which will aid the testing, verification and validation of the Operational Control System (OCS). An Operator Training Simulator (OTS) will be provided, which in turn provides for the development, testing, verification and validation of training exercises for trainee operators. The Plant Simulator will also function as a constituent component of the OTS for the training and licensing of operators. Another function of the Plant Simulator will be to test and design control philosophies before plant construction and commissioning to ensure safe plant operation. Furthermore, the Plant Simulator will be able to predict the impact of modifications on the plant after commissioning, train plant operators and test their skill levels before they operate the actual plant." (Sec. 11.5.2, p. 11-30) – *Future Studies* • "Develop integrated simulation tool - A need exists for a ***comprehensive computational model of the NGNP plant***. An integrated simulation tool needs to be developed in order to set-up an integrated model of the NGNP including the NHSS, HTS, PCS, HPS and BOP systems. This tool should be able to model the performance of the actual plant in steady state mode as well as during transitions and transient events. The tool should also be to model proposed control strategies to verify the adequacy the integrated control philosophy. The model will also serve to develop plant simulations for planning of operations and for operator training. • Update modes diagram - ***The modes diagram should be updated and expanded in the conceptual design phase***. The primary transitions and transient events should be identified together with all the required control functions. The	(None)	The WEC PCDR has at many levels recognized the indicated needs for testing.

Table 1C (WEC) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		integrated control philosophy should be evaluated and optimized during the conceptual design phase. • Simulate steady state, transitions and transient events - The above-mentioned simulation tool will be used to simulate steady state conditions, transitions and expected transients. The model should also be used to evaluate the integrated control philosophy and control functions of the different systems during the conceptual design phase to serve as input to component design in the basic and detail design phases. • Specific transitions and transient events for early investigation - Various specific transitions and transient events are identified for early investigation: • Simulate the start-up transitions in order to determine the operating conditions of the different systems and components of the NGNP during start-up. • ***Test the integrated control strategy*** to investigate interdependencies among the different systems. • Determine equipment protection requirements from above-mentioned transient analyses results. • Investigate the HPS conditioning and start-up transitions with specific focus on the operating conditions in the decomposition reactor. • Another future study is required to determine the level of detail in which the different NGNP systems need to be modeled in order to achieve accurate predictions of the actual NGNP performance within realistic computation time within the Plant Simulator." Section 16.2, *Nuclear Heat Supply System*, pages 16-18 through 16-28, contains a detailed description of the test facilities to be used for the PBMR, including: • The Pebble Bed Micro Model, a fully functional power conversion model to demonstrate concept and control. • The Heat Transfer Test Facility, which will be used o validate correlations currently used to ***model the heat transfer and fluid flow phenomena*** required for integrated simulation of the pebble bed core, via a comprehensive set of separate effects tests; and to validate different simulation methodologies applied in integrated models that represent the entire pebble bed core, via a ***comprehensive set of integrated effects tests***. The HTTF facility will consist of a number of smaller test sections that will be used for separate effects tests and a main test section that will be used to perform integrated effects tests. The smaller test sections will consist of a scaled down pebble bed and a number of duct-type sections packed with pebbles to represent pebble bed sections with predetermined homogeneous porosities. The main test section will represent an annular pebble bed and it will have the capability to heat the pebble bed (made up of graphite pebbles) and to characterize the heat transfer behavior of such a		

Table 1C (WEC) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		pebble bed. It is expected that the HTTF will fulfill the needs for tests to characterize several of the main phenomena required for simulating heat transfer in a pebble bed. • The Helium Test Facility, which was designed to ***develop and test components and sub-systems of the PBMR Main Support Systems*** (MSS) and other PBMR Components. It is a risk mitigation initiative for testing these components and systems in a helium environment at high pressure and high temperature but without nuclear radiation. • The ASTRA Facility, the purpose of which is to ***perform tests in the experimental investigation of neutronics characteristics*** of a reactor with geometrical characteristics similar to the PBMR reactor. The facility represents a cylindrical side reflector consisting of graphite blocks with an octagon shaped core in the centre and a solid cylindrical centre column. The core is filled with fuel spheres and absorber spheres. Control rods, shutdown rods and a single regulating rod are situated in the first set of blocks closest to the core in the side reflector. This allows different critical configurations to check single control rod reactivity worth's or different combinations to look at interference (or shadowing) effects and permits better V&V of control rod models and methods used in the analysis tools. • The Experimental Plate-out Loop, which is based on the design of the German Laminar Loop. In the facility, a radioactive source will deliver fission products into a warm gas stream of helium at 9 bars. The stream will then pass through a tube of a material under investigation. The tube will be heated in axial section to determine the effect of temperature on fission product deposition. The gas stream will leave the tube to be cooled and filtered where after it starts with the route again. • The Isopiestic Plate-out Facility, a laboratory type set-up for the investigation of plate-out parameters in a static environment. In the facility, a radioactive source will be placed within a container filled with helium. A vacuum is created and the subsequent deposition of fission products on a required material specimen is monitored. • The Natural Convection with Corrosion Facility. The main section of this facility is made up of a vertical channel of 300 mm x 300 mm and 7.5 m tall. The experimental channel is composed of sections representing a bottom reflector, sphere packing (pebble bed) and a top reflector. The experimental set-up was designed to be able to ***represent different breaks in pipes connecting to the reactor***. Breaks can be created that simulate the coaxial duct (reactor outlet pipe), the defueling chute at the bottom of the reactor and the fuelling line at the top of the reflector. By a sectional design, different core heights can also be simulated. All sections of the experimental channel and of the return pipe can be heated to accident-relevant temperatures. At different positions, the local gas compositions can be measured.		

Table 1C (WEC) – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		• The SANA test facility, which consists of a heated pebble bed inside a furnace to simulate the thermal conditions of an HTGR-core. Different heater configurations are possible but the PCDR shows a schematic of the test facility with a single central heating element. The diameter of the pebble bed is 1.5 m and the height is 1.0 m. The overall height of the facility is 3.2 m and the maximum heating capacity of the single central heating element is 35 kW. The top and bottom of the facility are well-insulated while the outside of the furnace is open to atmosphere. ***More than 50 steady-state as well as some transient tests were carried out in the facility. In these experiments all the main parameters of a pebble bed were varied, such as pebble material, pebble diameter, gas type, heating power and heating geometry.***		

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
D-1	*Physical Materials Data* - Requirements for physical aspects to be included in modeling high-temperature metallic components: • Inelastic materials behavior for materials, times, and temperatures for very high temperature structures (e.g., creep, fatigue, creep-fatigue). • Adequacy and applicability of current ASME Code allowables with respect to service times and temperatures for operational stresses. • Adequacy and applicability of current state of high-temperature design methodology (e.g., constitutive models, complex loading, failure criteria, flaw assessment methods). • Effects of product form and section thickness. • Joining methods including welding, diffusion bonding, and issues associated with dissimilar materials in structural components. • Effects of irradiation on materials strength, ductility, and toughness. • Degradation mechanisms and inspectability. • Oxidation, carburization, decarburization, and nitriding of metallic components in impure helium and helium-nitrogen. • Micro-structural stability during long-term aging in environment. • Effects of short and long term on mechanical properties (e.g., tensile, fatigue, creep, creep-fatigue, ductility, toughness). • High-velocity erosion/corrosion. • Rapid oxidation of graphite and carbon-carbon composites during air-ingress accidents. • Compatibility with heat-transfer media and reactants for hydrogen generation. • Development and stability of surface layers on RPV and core barrel affecting emissivity.	See item A-8 for information associated with model development. See item D-2 for information associated with composite materials. Sec. 4.2.1.2, pp. 4-19 thru 4-22, contains a ***detailed description of the reactor core barrel assembly, including required functions, materials specifications, methods of assembly***, and interfaces with other components and systems. Sec. 4.2.1.3, pp. 4-23 thru 4-27, contains a detailed description of the reactor core ceramic structures (top, bottom, side and central reflectors), including required functions, materials specifications, and interfaces with other components and systems. Sec. 4.2.1.4, pp. 4-27 thru 4-29, contains a ***detailed description of the Reactor Pressure Vessel, including required functions, materials specifications***, and interfaces with other components and systems. (Sec. 4.2.1.4, p. 4-28) – *Reactor Pressure Vessel (RPV)* – "The RPV is required to withstand all the normal operating conditions over the lifetime of the reactor and all the abnormal conditions for the specified number of occurrences, ***without any degradation of its ability to perform its nuclear and non-nuclear functions***. The RPV shall be designed for a nominal working pressure of 9.0 MPa, and a design pressure of 9.7 MPa. ***The RPV shall be designed and constructed to ASME III, Division I, Subsection NB and Code Case N-499-2***…. The RPV contains the RUS components and parts of the FHSS. ***The RPV is manufactured from carbon steel SA 533 Type B Class 1 for plates, SA 508 Type 3 Class 1 for forgings and SA 540 Grade B24 Class 3 for bolts.*** The RPV consists of a main cylindrical section with hemispherical upper and lower heads. The upper head is bolted to the cylindrical section and incorporates penetrations for the mechanisms of the FHSS, RCS, RSS and for instrumentation. An opening is provided in the centre of the upper head to allow access to the Core Structures for reflector replacement. The RPV has a maximum external diameter of approximately 6.8 m and its total length is approximately 30 m. The ***lower head is welded to the main cylindrical section***, and will have openings for fuel discharge, the RSS absorption spheres discharge and an access opening intended for use only during initial installation operations. The nozzle forgings of the Core Outlet Pipe and the vessel supports are attached to the lower reinforced part of the cylindrical section. This part is reinforced to withstand reactor vessel support loads. The RPV supports provide vertical as well as bottom horizontal support. Localized shielding may be attached to structures inside the RPV top part to reduce activation of the RPV head. Additional reinforcement is provided at the level of the upper attachment points for the upper seismic restraints. The nozzle forgings of the Core Inlet Pipe are attached to the top part of the RPV…. The RPV will be subjected to 9.0 MPa during normal operation. The ***reactor inlet helium flow will be controlled to ensure that the RPV temperature is within the***	NHSS-01-01 thru NHSS-01-03 (fuel testing) NHSS-02-01 and NHSS-02-02 (graphites) HTS-01-01 thru HTS-01-019 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX ceramics) HTS-04-01 (high temp. ducts and insulation)	The needs for materials data identified in this item have been addressed in the WEC PCDR, and some of the R&D has already been performed.

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		embrittlement data for the RPV material*.” (Sec. 4.4.2, p. 4-85) – *Future Studies/Metallic Structural Requirements* – “The ASME design codes used for design of the RPV and CBA are written for 40 year plant life. ***The creep data in the code needs to be extended to 60 years for the NGNP project. The effect of changes in the reactor inlet temperature, on the RPV and CBA due to operational changes in the cycle needs to be assessed. The pressure differential over the CBA (top to bottom) caused by the inlet flow through the restraints needs to be assessed, since it adds downwards force onto the CBA support.” (Sec. 6.3.1, p. 6-35) – *Intermediate Heat Exchanger – Metallic* – “The intermediate heat exchanger (IHX) is a critical high-temperature component of the NGNP. To attain the cost and performance goals of the NGNP and related commercial process heat plants, ***a plate-type compact heat exchanger, which is characterized by thin metal cross-sections, has been selected as the initial reference for the NGNP design. The reference material for the IHX is Alloy 617 and Alloy 230 has been identified as a backup material***. Given the demanding operating conditions for the IHX, it has been concluded that a parallel development of advanced (ceramic and/or composite) heat exchangers should be pursued in parallel with the reference design. The DDNs identified in Section 6.3.1 support the reference metallic IHX concept…. ***A total of 19 DDNs have been identified for the metallic IHX. These DDNs address materials characterization and qualification, development of methods and criteria for design and analysis and performance verification. In addition, DDNs are identified to support NGNP-specific ASME Code Cases for the IHX materials and design***.” (Note: DDNs are listed below, as they appear in Table 6.3-1.) • Establish Reference Specifications for Alloy 617 • Thermal/Physical and Mechanical Properties of Alloy 617 • Welding and As-Welded Properties of Materials of Alloy 617for Compact Heat Exchangers • Aging Effects of Alloy 617 • Environmental Effects of Impure Helium on Alloy 617 • Influence of Grain Size on Materials Properties on Alloy 617 • Establish Reference Specifications for Alloy 230 • Thermal/Physical and Mechanical Properties of Alloy 230 • Welding and As-Welded Properties of Materials of Alloy 230for Compact Heat Exchangers • Aging Effects of Alloy 230 • Environmental Effects of Impure Helium on Alloy 230 • Influence of Grain Size on Materials Properties on Alloy 230 • Methods for Thermal/Fluid Modeling of Plate-Type Compact Heat Exchangers		

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		• Methods for Stress/Strain Modeling of Plate-Type Compact Heat Exchangers • Criteria for Structural Adequacy of Plate-Type Compact Heat Exchangers at Very High Temperatures • Methods for Performance Modeling of Plate-Type Compact Heat Exchangers • IHX Performance Verification • Data Supporting Materials Code Case • Data Supporting Design Code Case (Sec. 8.3.3, p. 8-43) – *Helium Environment Effects on 2-1/4Cr-1Mo* - "The primary coolant contains impurities which may cause material corrosion in the form of ***oxidation, decarburization and carburization***. At the design temperatures of the components, carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep fatigue and crack growth. ***Another factor that must be considered along with decarburization or carburization is the change in microstructure due to thermal aging***. In the tests performed to date (with pretest aging exposures up to 34,000 hr and creep tests up to 100,000 hr), no influence of testing environment was seen on the creep properties up to I% strain. Also, ***relatively minimal effect of thermal aging on the creep rupture and rupture ductility was seen at these low strain levels. Previous studies indicate the effect of NGNP primary coolant chemistry on the tensile properties of this material is negligible***. The general effect of thermal aging treatments is to decrease the yield and ultimate tensile strengths. The magnitude of the effect increases with aging temperature and time. Available information on the creep fatigue behavior of 2-1/4 Cr-1 Mo steel have indicated that this ***material tested in helium environment has improved fatigue life for all test weld forms in comparison with tests in air except for the tensile hold only tests***. Some limited test data obtained on 2-1/4 Cr-1 Mo weldments suggests that these weldments might be as strong as the base metal. Review of available data indicated that low-oxygen environments, including NGNP helium improve the continuous cycling fatigue behavior of this material over the temperature range 914°F (490°C) to 1100°F (593°C)…. ***Data are needed to confirm that exposure to impure primary coolant helium at appropriate temperature does not reduce the selected design mechanical properties of 2-1/4 Cr-1 Mo base metal and its weldments below the values of the ASME Code Subsection NH***." (Sec. 8.3.3, p. 8-46) – *Helium Environment Effects on 800H* – "The primary coolant contains impurities which may cause material corrosion in the form of ***oxidation, decarburization and carburization***. At the design temperatures of operation of the Alloy 800H components, carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent		

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		carbon transport may significantly affect surface sensitive properties such as fatigue, creep fatigue and crack growth. ***Another factor that must be considered along with decarburization or carburization is the change in microstructure due to thermal aging***. Extensive data is available on the degree of carburization/decarburization and oxidation of Alloy 800H as a function of temperature, impurity levels, and exposure time in simulated HTGR primary coolant helium. Based on these data and the given impurity levels of the NGNP, and the temperatures of operation of steam generator components, carburization of Alloy 800H is expected to be minimal over the life of the plant. The effect of impure helium environments on the creep rupture properties of Alloy 800H were evaluated for times exceeding 30,000 h. Comparative creep tests in air and in helium for the same heats of Alloy 800H were reviewed and except for two isolated data points obtained at 1400°F (760°C), all the data measured fall well within the scatter bands for air creep-rupture data. Tests were performed on commercial heats of Alloy 800H thermally aged in the temperature range of 1000° to 1500°F (538° to 816°C) for times up to 30,000 h. ***Age-hardening was observed at 1000°, 1100° and 1200°F (538°, 593° and 649°C), resulting in increases in the yield and ultimate tensile strengths of the material with some reduction in ductility***. The presence of the HTGR helium environment had no discernible effect on the stress behavior during low-cycle fatigue, high-cycle fatigue or creep-fatigue testing performed on Alloy 800H to date. The ***low-cycle fatigue life at 1200°F (650°C) was significantly increased compared with that in air.*** Fracture toughness data available to date indicate the room temperature tensile properties and CVN energy for this material aged 10,000 h at 593°C (1100°F) are significantly different than values for material aged for the same length of time at other temperatures. On the other hand, the J1c measured on material aged 10,000 h at 593°C (1100°F) is significantly lower than that measured on material aged for the same length of time at the other temperatures….***Data are needed to confirm that exposure to impure primary coolant (helium) at appropriate temperature does not reduce the selected design mechanical properties of Alloy 800H base metal and its weldments below the values of the ASME Code Subsection NH***." (Sec. 8.3.3, p. 8-74) – *Bi-metallic Weld Structural Integrity* – "The Steam Generator tube bundle design incorporates BMWs between the Alloy 800H Finishing Superheater (FSH) tubing and the 2 ¼ Cr-1Mo Evaporator/Economizer/Superheater (EES) tubing. The existing service data base for BMWs between 2 ¼ Cr-1Mo and austenitic stainless steel with nickel based fillers has been used to establish an upper limit temperature criterion for BMW operation of 480^{o} C. Data are needed to confirm the adequacy of the upper temperature limit of 480 ^{o} C and the creep-fatigue properties of the BMW and to verify the adequacy of design analysis methods…. Experience with superheater and reheater BMWs in fossil-fired power plants has indicated that at temperatures above 565^{o}C the creep-fatigue capabilities of the weldments can be compromised by the growth of large carbides at the weld/ferritic steel interface in such weldments. ***Data are needed to confirm the adequacy of the temperature margin in the design and to confirm the fatigue strength of the***		

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		welds at the lower temperatures using prototypical tubes, tube materials, weld filler materials, and operating conditions*.” (Sec. 12.1.1, p. 12-11) – *Maintainability* – “Plant outages are scheduled at 5-year intervals. The ***minimum lifetime of IHX A of 10 years and the requirement of maximum 6-year intervals between maintenance and inspection on other SSC’s are the main constraints in deciding on intervals between scheduled outages. Scheduled outages will last maximum 30 days for outages scheduled after 5 years and 15 years. The 10 year outage requires the replacement of the IHX A and will last a maximum of 50 days. The 20 year outage requires the replacement of ceramic core structures and will last a maximum of 180 days. The scheduled maintenance outages are repeated in 20 year cycles.” (Sec. 12.1.2, p. 12-12) – *Maintenance Requirements for Systems and Components* – “The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. ***General requirements for each system include that the design provides access to the pressure boundaries to permit in-service inspection as required by appropriate sections of the ASME B&PV Code***, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible.” (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – “These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. ***Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies***.” (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* - Planned: ***Corrosion/oxidation models for air ingress consequences during postulated accident conditions***. (Sec. 16.3.1, p. 16-50) – *Design Data Needs (Heat Transport Facility)* – “The final DDNs supporting the metallic IHX, HTS-01-18 and HTS-01-19, are established to		

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Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		provide the underlying database supporting NGNP-specific code cases for the IHX material and design, respectively. There is a potential that such code cases would also be applicable to early commercial plants, pending formal implementation within the ASME Code. For ceramic/composite IHXs, six placeholder DDNs (HTS-02-01 through HTS-02-06) have been identified, as both the DDN's and the associated R&D activities will need further development during conceptual design. The first DDN provides for a review of existing technology that is potentially applicable to the development of a ceramic IHX. The anticipated result of the corresponding R&D effort will be the selection of one or more materials and/or heat exchanger technologies for further development. The second DDN specifies the need for a ***materials property database*** for the selected materials. The third DDN addresses the need for design methods, while the fourth identifies requirements for performance verification. The fifth and sixth DDNs address manufacturing technology and the development of codes and standards…. The R&D activities pertaining to DDNs HTS-01-01 through HTS-01-06 provide for the extended qualification of the current reference IHX material, Alloy 617. This extended qualification is required due to the demanding operating conditions that will be seen by the IHX, plus the small grain size that is expected to be required for compact heat exchangers as they are characterized by very thin heat transfer surface cross-sections. As described in DDN HTS-01-01 (Section 6.3.1), an initial effort is required to further develop the specification for Alloy 617 and to establish a reference for characterization. ***Included in this effort, is a review of the current database for this material, consultation with material vendors and consideration of a controlled specification variant, Alloy 617CCA, that potentially decreases the range of uncertainties with respect to properties.*** The conclusion of this effort will be procurement of materials to be used for subsequent testing." (Sec. 16.3.2, pp. 16-50 through 16-59) – This section provides a ***detailed plan for IHX R&D, including metallic and composite materials.***		
D-2	*Physical Materials Data (Composites)* - Requirements for physical aspects to be included in modeling high-temperature structural composites, such as carbon-carbon or silicon carbide–silicon carbide: • Effects of composite component selection and infiltration method. • Effects of architecture and weave. • Materials properties up to and including very high temperatures (e.g., strength, fracture, creep, corrosion, thermal shock resistance). • Effects of irradiation on materials strength and dimensional stability.	(Sec. 6.3.2, pp. 6-75 thru 6-88) – *Intermediate Heat Exchanger – Ceramic/Composite* – This section contains a detailed description of the WEC approach to establishing an engineering design basis for the composite materials to be used in the IHX. The ***overall approach will be to determine appropriate materials, develop design and manufacturing methods, establish and verify characteristics through testing, and develop an ASME code case for the composite materials***. DDNs are also included in the section, covering the following: • Perform a review of existing technology • Develop a physical properties database • Develop design methods • Conduct performance verification testing • Develop manufacturing technology	HTS-02-01 (Review existing technology) HTS-02-02 (Materials properties database) HTS-02-03 (Design	The WEC PCDR has recognized the need for composite physical materials data.

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	• Fabrication scaling processes. • Adequacy and validation of design methods. • Degradation mechanisms and inspectability.	• Support development of an ASME III Code Case for materials and design (Sec. 12.1.1, p. 12-11) – *Maintainability* – "Plant outages are scheduled at 5-year intervals. The minimum lifetime of IHX A of 10 years and the requirement of maximum 6-year intervals between maintenance and ***inspection*** on other SSC's are the main constraints in deciding on intervals between scheduled outages. Scheduled outages will last maximum 30 days for outages scheduled after 5 years and 15 years. The 10 year outage requires the replacement of the IHX A and will last a maximum of 50 days. The 20 year outage requires the replacement of ceramic core structures and will last a maximum of 180 days. The scheduled maintenance outages are repeated in 20 year cycles." (Sec. 12.1.2, p. 12-12) – *Maintenance requirements for Systems and Components* – "The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. General requirements for each system include that the design provides access to the pressure boundaries to permit ***in-service inspection*** as required by appropriate sections of the ASME B&PV Code, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible." (Sec. 16.3.2, pp. 16-50 through 16-59) – This section provides a detailed plan for IHX R&D, ***including metallic and composite materials***.	Methods) HTS-02-04 (Performance verification) HTS-02-05 (Manufacturing technology) HTS-02-06 (Codes and Standards)	
D-3	Compromise of RPV surface emissivity due to loss of desired surface layer properties. Compromise of emissivities of in-vessel surfaces.	(Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. ***Emissivity testing of various surface treatments*** and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies."	HTS-01-01 thru HTS-01-019 (IHX metallics)	The need to explore issues relating to surface emissivity has been recognized in the WEC PCDR.
D-4	Effects on insulation • Aging fatigue and environmental degradation of insulation materials (debris plugging). • Environmental and irradiation degradation/thermal instability of fibrous insulation	(Sec. 6.3.4, p. 6-92) – *High Pressure Ducts and Insulation* – "High-temperature ducts and insulation are utilized within the PHTS and SHTS pressure boundary piping to direct helium flow from the reactor to IHX A (nominally at 950ºC), from IHX A to IHX B (nominally at 760oC) and from the secondary IHX A outlet to the PCHX and SG (nominally at 900ºC). ***The high-temperature ducts and insulation are contained within pressure boundary pipes of low-alloy (SA 508/533) steel that are designed to ASME Section III requirements.*** On this basis, the pressure boundary	HTS-01-01 thru HTS-01-019 (IHX metallics) HTS-04-01 (high temp	The need to explore issues relating to high temperature ducts and insulation has been recognized in the WEC PCDR.

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		pipes must be maintained at 371ºC or less, based on Section III requirements for normal operational service conditions. The current design, in common with the PBMR-DPP high-temperature piping and ducts, is to provide both passive internal insulation and active cooling of the PHTS pressure boundary piping containing the hottest fluids (Reactor to IHX A and IHX A to IHX B). The IHX A vessel is also to be actively cooled. However, the highest temperature in the NGNP PHTS nominally exceeds that of the DPP by 50ºC and available coolant flow for active cooling is at 350ºC vs. ~120ºC in the DPP. ***These factors imply incremental challenges for the ducts and insulation***. The high temperature piping connecting the IHX A secondary outlet to the PCHX and helium mixing-chamber (nominally at 900ºC) and the piping connecting the PCHX to the helium-mixing chamber (nominally at 659ºC) will initially be evaluated with passive insulation systems only and with no active cooling. Other piping sections will operate at 350ºC or less and no DDNs have been identified for these components…. One DDN has been identified for the development of the high-temperature ducts and insulation systems. ***This DDN addresses insulation systems, hot duct liner characterization, metallic materials selection and qualification and performance verification of the hot duct piping and other hot piping prototypes***…. The proposed approach is to experimentally verify and validate the materials and design selections associated with key high-temperature ducts and insulation, using fabricated prototypes. This ***proposed approach will include the evaluation of mechanical properties, helium environmental effects, aging effects and micro-structural changes on the metallic alloys selected; long term validation of the acceptability and continued effectiveness of the insulation systems; and long term evaluation of the prototype piping systems to maintain acceptable levels of performance under operational and off normal conditions***. One key objective is to support the potential selection of more economic designs with passive internal insulation." (Sec. 8.3.3, p. 8-56) – *Insulation Verification Test* – "***Thermal and mechanical performance of the insulation located in the active flow region (i.e., helium inlet plenum and outer shroud) of the steam generator needs to be verified***. The concerns are the ***possibility of insulation becoming loose during operation and blocking helium flow*** areas and the difficulty with accessibility for maintenance or alteration once the steam generator is installed….A considerable amount of literature is available relative to high temperature insulation physical and thermophysical properties. A variety of insulations are available in special forms to meet specific service requirements…. Physical and operational characteristics of insulation are required. Specific data needed would be relative ***to thermal cycling of fibrous insulation, effects of mechanical and acoustic vibrations, and effects of flow and thermal gradients***. These tests produce temperature data for certain critical components of the steam generator and verify the proposed thermal barrier for the life of the plant. Additional test data relative to any destructive impact on insulation due to vibrations and sliding contacting surfaces, as needed, would be obtained…. The selected design approach is to perform testing of different critical regions under	ducts and insulation)	

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		simulated environment conditions. Thermal performance of the insulation can be obtained by analysis; however, analysis alone is not sufficient to assure the mechanical performance of the insulation. ***Performing the described tests is the only way of checking the mechanical performance of the insulation***." (Sec. 16.3.5, p. 16-61) – *High Temperature Ducts and Insulation* – "Materials development is not presently anticipated for the high-temperature ducts and insulation. Requirements for metallic materials are expected to be enveloped by the metallic IHX development activities described in Section 16.3.2. Insulation will be adapted from the PBMR-DPP and/or prior German experience. This assumption regarding materials development will be revisited during conceptual design in conjunction with the ducts and insulation special study outlined in Section 6.4.2….The R&D program required to validate the performance of the high-temperature ducts and insulation comprises a series of performance and environmental tests that will subject prototypical duct and insulation components, or parts thereof, to conditions representative of those found in the HTS. The principal objective of these tests is to verify the adequacy of thermal insulation and its stability under HTS environmental conditions. ***Examples of envisioned tests include determining effective thermal conductivity of representative assemblies in helium, environmental and aging tests in helium and the effects of vibration***. At this point, large-scale flow tests are not envisioned, since the NGNP will obtain input from the PBMR-DPP, which utilizes an actively cooled design in the high-temperature sections. The latter will facilitate validation of analysis methods to be applied. The scope of the ducts and insulation R&D activities will be revisited during conceptual design in conjunction with the ducts and insulation special study outlined in Section 6.4.2."		
D-5	Primary boundary failures in compact IHX (roles of design methods, manufacturing controls, inspection/testing).	See item D-1 for information on metallic IHX and item D-2 for information on composite IHX. (Sec. 12.1.1, p. 12-11) – *Maintainability* – "Plant outages are scheduled at 5-year intervals. ***The minimum lifetime of IHX A of 10 years and the requirement of maximum 6-year intervals between maintenance and inspection on other SSC's are the main constraints in deciding on intervals between scheduled outages***. Scheduled outages will last maximum 30 days for outages scheduled after 5 years and 15 years. The 10 year outage requires the replacement of the IHX A and will last a maximum of 50 days. The 20 year outage requires the replacement of ceramic core structures and will last a maximum of 180 days. The scheduled maintenance outages are repeated in 20 year cycles." (Sec. 12.1.2, p. 12-12) – *Maintenance requirements for Systems and Components* – "The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen	HTS-01-01 thru HTS-01-019 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX composites)	The need to explore issues relating to the IHX, including the limited lifetime of the component, has been recognized in the WEC PCDR.

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. ***General requirements for each system include that the design provides access to the pressure boundaries to permit in-service inspection as required by appropriate sections of the ASME B&PV Code***, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible."		
D-6	Control rod insertion failures (role of structural design methods for composites).	***Not applicable to the WEC PCDR. There is no indication that WEC intends to use composite materials for control rod system components.*** See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS).	HTS-01-01 thru HTS-01-015 (IHX metallics)	Not applicable to the WEC PCDR. There is no indication that WEC intends to use composite materials for control rod system components.
D-7	Irradiation induced creep of in-vessel metallic structures.	See item D-1 for information on metallic IHX. (Sec. 4.4.2, p. 4-85) – *Future Studies/Metallic Structural Requirements* – "The ASME design codes used for design of the RPV and CBA are written for 40 year plant life. ***The creep data in the code needs to be extended to 60 years for the NGNP project. The effect of changes in the reactor inlet temperature, on the RPV and CBA due to operational changes in the cycle needs to be assessed.*** The pressure differential over the CBA (top to bottom) caused by the inlet flow through the restraints needs to be assessed, since it adds downwards force onto the CBA support." (Sec. 16.2.1.3, p. 16-29) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. ***Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies***."	HTS-01-01 thru HTS-01-015 (IHX metallics)	The need to explore materials issues relating metallic structures, including those associated with the RPV and IHX, has been recognized in the WEC PCDR.
D-8	Core radial restraint failure (role of structural design and fabrication for composites).	(Sec. 16.2.1.6, p. 16-30) – *Core Structural Ceramics Qualification* – "For the PBMR DPP, Core Structural Ceramics include the Reflector Graphite that establishes the core geometry, ***Carbon Fiber Reinforced Carbon (CFRC) components associated with the core lateral restraints and tie rods supporting the upper reflector***, and ceramic components used to provide thermal insulation below the core. A summary of the Core Structural Ceramics R&D supporting the PBMR DPP is provided in Table 16.2-3."	HTS-01-01 thru HTS-01-015 (IHX metallics)	The need to explore issues relating to the core restraints has been recognized in the WEC PCDR.

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
D-9	Isolation and other valve failures (self-welding, galling, seizing)	(Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a ***full scope PRA that addresses all internal and external hazards***, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. ***Testing include: Self-welding in a Helium atmosphere*** on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies." (Sec. 16.2.1.8, p. 16-31) – *First Of a Kind Components R&D* – ***Valves: Manufacturability and performance verification in helium.***	HTS-01-01 thru HTS-01-015 (IHX metallics)	The need to explore issues relating to valve failures has been recognized in the WEC PCDR.
D-10	Initiate development of the data and models needed by ASME Boiler and Pressure Vessel (B&PV) Code Subcommittees to formulate time-dependent failure criteria that will ensure adequate life and safety for metallic materials in the NGNP. These include obtaining the data necessary to develop experimentally based constitutive models for the NGNP construction materials, which are the foundation of the inelastic design analyses specifically required by ASME B&PV Sect. III Division I Subsection NH.	See item A-8 for information on model development. See items D-1 and D-2 for information on development of metallic and composite materials and design characterization and methods, ***including requirements to support new ASME Code Cases***. (Sec. 4.2.1.4, p. 4-28) – *Reactor Pressure Vessel (RPV)* – "The RPV is required to withstand all the normal operating conditions over the lifetime of the reactor and all the abnormal conditions for the specified number of occurrences, without any degradation of its ability to perform its nuclear and non-nuclear functions. The RPV shall be designed for a nominal working pressure of 9.0 MPa, and a design pressure of 9.7 MPa. The ***RPV shall be designed and constructed to ASME III, Division I, Subsection NB and Code Case N-499-2***." (Sec. 4.4.2, p. 4-85) – *Future Studies/Metallic Structural Requirements* – "The ***ASME design codes used for design of the RPV and CBA are written for 40 year plant life. The creep data in the code needs to be extended to 60 years for the NGNP project***. The effect of changes in the reactor inlet temperature, on the RPV and CBA due to operational changes in the cycle needs to be assessed. The pressure	HTS-01-01 thru HTS-01-019 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX) ceramics)	The need to explore issues relating to the qualification of NGNP metallics under approved ASME Code Cases has been recognized in the WEC PCDR.

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		differential over the CBA (top to bottom) caused by the inlet flow through the restraints needs to be assessed, since it adds downwards force onto the CBA support." (Sec. 16, p. 16-13) – *Intermediate Heat Exchanger* – "The IHX is a critical component of the NGNP and a fundamental enabling technology for high-temperature process heat applications in general. The IHX requires a significant development effort, largely in the materials area, to demonstrate that a design can be developed for the high temperatures, pressures and transients expected for the NGNP. Parallel efforts are recommended that address the most promising metallic and ceramic materials. The ***results will support an ASME code case and a final design for the IHX that will need to be prototyped and tested***." (Sec. 16.3.1, p. 16-50) – *Design Data Needs (Heat Transport Facility)* – "The final DDNs supporting the metallic IHX, HTS-01-18 and HTS-01-19, are established to ***provide the underlying database supporting NGNP-specific code cases for the IHX material and design***, respectively. There is a potential that such code cases would also be applicable to early commercial plants, pending formal implementation within the ASME Code. For ceramic/composite IHXs, six placeholder DDNs (HTS-02-01 through HTS-02-06) have been identified, as both the DDN's and the associated R&D activities will need further development during conceptual design. The first DDN provides for a review of existing technology that is potentially applicable to the development of a ceramic IHX. The anticipated result of the corresponding R&D effort will be the selection of one or more materials and/or heat exchanger technologies for further development. The second DDN specifies the need for a materials property database for the selected materials. The third DDN addresses the need for design methods, while the fourth identifies requirements for performance verification. The fifth and sixth DDNs address manufacturing technology and the development of codes and standards…. The R&D activities pertaining to DDNs HTS-01-01 through HTS-01-06 provide for the extended qualification of the current reference IHX material, Alloy 617. This extended qualification is required due to the demanding operating conditions that will be seen by the IHX, plus the small grain size that is expected to be required for compact heat exchangers as they are characterized by very thin heat transfer surface cross-sections. As described in DDN HTS-01-01 (Section 6.3.1), an initial effort is required to further develop the specification for Alloy 617 and to establish a reference for characterization. Included in this effort, is a review of the current database for this material, consultation with material vendors and consideration of a controlled specification variant, Alloy 617CCA, that potentially decreases the range of uncertainties with respect to properties. The conclusion of this effort will be procurement of materials to be used for subsequent testing."		
D-11	Safety assessments dependent on time-dependent flaw	See items D-1 and D-2 for information on development of metallic and composite	HTS-01-01	The need to explore issues relating to the

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	growth and the resulting leak rates from postulated pressure-boundary breaks will require a flaw assessment procedure capable of reliably predicting crack-induced failures, as well as the size and growth of the resulting opening in the pressure boundary.	materials and design characterization and methods. (Section 16.2.1.1.6, p. 16-26) – "NACOK stands for Natural Convection with Corrosion. The main section of this facility is made up of a vertical channel of 300 mm x 300 mm and 7.5 m tall. The experimental channel is composed of sections representing a bottom reflector, sphere packing (pebble bed) and a top reflector. The experimental set-up was designed to be able to represent different breaks in pipes connecting to the reactor. Breaks can be created that simulate the coaxial duct (reactor outlet pipe), the defueling chute at the bottom of the reactor and the fuelling line at the top of the reflector. By a sectional design, different core heights can also be simulated. All sections of the experimental channel and of the return pipe can be heated to accident-relevant temperatures. At different positions, the local gas compositions can be measured." (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies."	thru HTS-01-019 (IHX metallics)	mechanical properties of metallic pressure boundary materials has been recognized in the WEC PCDR.
D-12	Materials data and extrapolation procedures must be developed and guidance provided to ensure that allowable operation period and range of stress and temperature for materials of construction are extended to meet the proposed operating temperatures and lifetimes. Creep-fatigue rules are an area of particular concern for the materials and temperatures of interest and must be updated and validated. (example concern: RPV long-term thermal aging)	See items D-1 and D-2 for information on development of metallic and composite materials and design characterization and methods, including requirements to support new ASME Code Cases. (Sec. 4.2.1.4, p. 4-28) – *Reactor Pressure Vessel* – "***The RPV is required to withstand all the normal operating conditions over the lifetime of the reactor and all the abnormal conditions for the specified number of occurrences, without any degradation of its ability to perform its nuclear and non-nuclear functions***. The RPV shall be designed for a nominal working pressure of 9.0 MPa, and a design pressure of 9.7 MPa. The RPV shall be designed and constructed to ASME III, Division I, Subsection NB and Code Case N-499-2." (Sec. 4.4.2, p. 4-85) – *Future Studies/Metallic Structural Requirements* - The ASME design codes used for design of the RPV and CBA are written for 40 year plant life. ***The creep data in the code needs to be extended to 60 years for the NGNP project.*** The effect of changes in the reactor inlet temperature, on the RPV and CBA due to operational changes in the cycle needs to be assessed. The pressure	NHSS-02-01 and NHSS-02-02 (graphites) HTS-01-01 thru HTS-01-19 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX ceramics) HTS-04-01 (high temp	The need to explore issues relating to extended lifetimes and more severe conditions anticipated for metallic components has been recognized in the WEC PCDR.

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		differential over the CBA (top to bottom) caused by the inlet flow through the restraints needs to be assessed, since it adds downwards force onto the CBA support. (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). ***High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies***." (Sec. 16.3.1, p. 16-50) – *Design Data Needs (Heat Transport Facility*) – "The final DDNs supporting the metallic IHX, HTS-01-18 and HTS-01-19, are established to provide the underlying database supporting NGNP-specific code cases for the IHX material and design, respectively. There is a potential that such code cases would also be applicable to early commercial plants, pending formal implementation within the ASME Code. For ceramic/composite IHXs, six placeholder DDNs (HTS-02-01 through HTS-02-06) have been identified, as both the DDN's and the associated R&D activities will need further development during conceptual design. The first DDN provides for a review of existing technology that is potentially applicable to the development of a ceramic IHX. The anticipated result of the corresponding R&D effort will be the selection of one or more materials and/or heat exchanger technologies for further development. The ***second DDN specifies the need for a materials property database for the selected materials.*** The third DDN addresses the need for design methods, while the fourth identifies requirements for performance verification. The fifth and sixth DDNs address manufacturing technology and the development of codes and standards…. The ***R&D activities pertaining to DDNs HTS-01-01 through HTS-01-06 provide for the extended qualification of the current reference IHX material***, Alloy 617. This extended qualification is required due to the demanding operating conditions that will be seen by the IHX, plus the small grain size that is expected to be required for compact heat exchangers as they are characterized by very thin heat transfer surface cross-sections. As described in DDN HTS-01-01 (Section 6.3.1), an initial effort is required to further develop the specification for Alloy 617 and to establish a reference for characterization. Included in this effort, is a review of the current database for this material, consultation with material vendors and consideration of a controlled specification variant, Alloy 617CCA, that potentially decreases the range of uncertainties with respect to properties. The conclusion of this effort will be procurement of materials to be used for subsequent testing."	ducts and insulation)	

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
D-13	Since IHX sections must operate at the full exit temperature of the reactor, effort should be initiated to obtain data supporting the determination of the metallurgical stability and environmental resistance of IHX materials in anticipated impure helium coolant environments for the lifetimes anticipated.	See items D-1 and D-2 for information on development of IHX metallic and composite materials and design characterization and methods, including requirements to support new ASME Code Cases. (Sec. 4.2.7.2, p. 4-60) – *Helium Purification System* – "The ***Helium Purification System (HPURS) is used to provide the required degree of helium purity. High purity coolant is required in order to minimize corrosion and contamination in the PHTS and SHTS***. This is done by bleeding off a partial flow of helium from the PHTS and SHTS. The extraction point is from the highest pressure points, i.e. the PHTS and SHTS circulator discharges within the HTS. This flow is tapped off constantly during operation of the plant. The HPS removes chemical gaseous contaminants from the primary coolant within the PHTS by the use of, catalysts, adsorbers and the manipulation of helium temperature extracted from the PHTS and SHTS. The ***required helium purity levels will be confirmed during the conceptual design***." (Sec. 16.3.2, pp. 16-50 through 16-59) – ***This section provides a detailed plan for IHX R&D, including metallic and composite materials***.	HTS-01-01 thru HTS-01-015 (IHX metallics)	The WEC PCDR has addressed this topic.
D-14	Work should be initiated to quantify crack initiation and propagation in the IHX due to creep, creep-fatigue, and aging. These materials-related phenomena related to the IHX were identified for potentially contributing to FP release at the site boundary.	See items D-1 and D-2 for information on development of IHX metallic and composite materials and design characterization and methods, including requirements to support new ASME Code Cases. (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, ***creep and fatigue testing*** in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies." (Sec. 16.3.2, pp. 16-50 through 16-59) – ***This section provides a detailed plan for IHX R&D, including metallic and composite materials.***	HTS-01-01 thru HTS-01-015 (IHX metallics)	The WEC PCDR has addressed this topic.
D-15	Specific issues must be addressed for RPVs that are too large for shop fabrication and transportation. Validated procedures for on-site welding, PWHT, and inspections must be developed for the materials of construction. For vessels using materials other than those typical of LWR	(Sec. 1.9.2.1, p. 1-30) – *Future Studies: Major Equipment Transportation Trade Study/Major Component Field Fabrication* – "***Detailed studies of the: (1) transportation routes and size constraints for transport of large components or sub-components such as the reactor and steam generator and (2) potential modularization for major components are recommended early in the***	(None)	The need to explore issues relating to possible RPV fabrication activities has been recognized in the WEC PCDR.

Table 1D (WEC) – HIGH TEMPERATURE MATERIALS (METALLIC) - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	construction to enable operation at higher temperatures, confirmation of their fabricability (especially, effects of forging size and weldability) and data on their irradiation resistance is needed. Three materials-related phenomena related to the RPV fabrication and operation were identified for potentially contributing to FP release at the site boundary, particularly for 9Cr–1 Mo–V steels capable of higher-temperature operation: crack initiation and subcritical crack growth, process control to avoid material degradation during field fabrication, and property control in heavy sections.	***conceptual design phase. This latter study will assess schedule and cost advantages and disadvantages of final assembly of major items at or near the site.*** These studies will influence the design of access roads and a rail spur shown on the site plan and plot plan as well as plans for modification to other roads in the vicinity of the INL site. " (Sec. 12.1.2, p. 12-12) – *Maintenance requirements for Systems and Components* – "The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. General requirements for each system include that the design ***provides access to the pressure boundaries to permit in-service inspection as required by appropriate sections of the ASME B&PV Code, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible.***"		
D-16	For high-temperature metals technology, there is a need for analytical models, in particular for developing time-dependent design criteria for complex structures, along with verification by structural testing. ASME Code-approved simplified methods have not yet been proven and are not permitted for compact IHX components. Analytical modeling of carbon-carbon composite behavior would be useful in developing approved methods for designing, proof testing, model standard testing, validation tests, and probabilistic methods of design. Scalability and fabrication issues must be addressed, including large-scale structures (meters in diameter), as well as smaller structures.	See item A-8 for information on model development. See items D-1 and D-2 for information on development of IHX metallic and composite materials and design characterization and methods, including requirements to support new ASME Code Cases. (Sec. 16.3.2, pp. 16-50 through 16-59) – ***This section provides a detailed plan for IHX R&D, including metallic and composite materials.***	HTS-01-17 thru HTS-01-19 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX ceramics)	The needs to explore issues relating to establishing mechanical properties, design methods, and supporting new ASME Code Cases for metallic components have been recognized in the WEC PCDR.

Table 1E (WEC) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
E-1	Lack of confirmatory data for the grades of graphite selected by potential NGNP vendors. This situation has occurred because: • Graphite grades used in prior HTGRs are no longer available, and thus development of new grades has been required. • Increased temperature of the NGNP compared to prior graphite-moderated reactors. • In the case of the PBR, the larger neutron dose that the core components will experience compared to that of previous HTGRs licensed in the United States.	Sec. 4.2.1.2, pp. 4-19 thru 4-22, contains a detailed description of the reactor core barrel assembly, including required functions, materials specifications, methods of assembly, and interfaces with other components and systems. ***Sec. 4.2.1.3, pp. 4-23 thru 4-27, contains a detailed description of the reactor core ceramic structures*** (top, bottom, side and central reflectors), including required functions, materials specifications, and interfaces with other components and systems. Sec. 4.2.1.4, pp. 4-27 thru 4-29, contains a detailed description of the Reactor Pressure Vessel, including required functions, materials specifications, and interfaces with other components and systems. (Sec. 4.3.1, p. 4-77) – *Reactor Graphite/Core Structure Ceramics* – "The PBMR NGNP Core Structure Ceramics (CSC) comprise the non-metallic components enclosed within the core barrel and its underlying support structure, plus the additional non-metallic components that form and support the top reflector assembly. The components of the CSC are principally manufactured from graphite. Certain components of the CSC, specifically the ***Reflector Graphite components that are adjacent to or near the core, operate in a harsh environment where they are subjected to high neutron fluences at high temperatures.*** The reliable operation of the CSC is important for the continued safe and economic operation of the plant as a whole. Thus, the qualification of the CSC, and particularly the Reflector Graphite, for the relevant conditions is of importance…. The physical design of the CSC for the PMBR NGNP is essentially identical to that of the PBMR Demonstration Power Plant (DPP); however, the operating conditions vary, as the PBMR NGNP has both a lower reactor inlet temperature and high reactor outlet temperature. On that basis, the DDNs for the PBMR NGNP are incremental to those already addressed within the PBMR-Specific Materials Test Reactor Program (PSMP)….To accommodate the expanded operating range of the PBMR NGNP, two DDNs have been identified for the Reflector Graphite, addressing the low and high temperature regimes, respectively. These DDNs provide for acquiring the data necessary to achieve a comparable level of qualification for the PBMR NGNP as is presently planned for the PBMR DPP." (Sec. 16.2.1.6, p. 16-30) – *Core Structural Ceramics R&D* – Completed program under the PBMR-Specific Materials Test Reactor Program to conduct supplemental irradiations of NBG-18 to ***verify consistency with the established database for similar graphites***. (Section 16.2.1.6, p. 16-31) The PBMR-Specific Materials Test Reactor Program (PSMP) is structured to provide data supporting startup and initial operation of the PBMR to the first planned outage period (6 years) as input to the licensing process. Completion of the PSMP is planned for the timeframe of plant startup in 2012. The	NHSS-02-01 and NHSS-02-02 (graphites) HTS-02-01 thru HTS-02-06 (IHX ceramics)	Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed.

Table 1E (WEC) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		results of the PSMP will be confirmed through surveillance, testing, inspection and maintenance activities over the plant operating lifetime. ***R&D programs relating to qualification of core structural ceramics are as follows:*** • Complete: Reflector graphite specification (NBG-18 graphite). • Complete: Reflector graphite manufacturing process. • Complete: Reflector graphite QA process. • Complete: Characterization of reflector graphite unirradiated properties. • In-progress: Conduct supplemental irradiations of NBG-18 to verify consistency with the established database for similar graphites. • Planned: Perform component-specific tests to characterize and quality carbon fiber reinforced carbon (CFRC) components. • Planned: Characterization of insulation materials, including, where appropriate, irradiation to modest fluence levels. (Sec. 16.2.1.10, p. 16-34) – *Engineering Design Tools R&D* – Table 16.2-7 describes planned and in-progress ***efforts for developing irradiated material behavior models (e.g. graphite blocks); graphite corrosion/oxidation models for air ingress models; and discrete element modeling to simulate behavior of graphite bodies in contact with each other*** (e.g. block reflector structures and fuel spheres). (Section 16.2.2, p. 16-35) - *Design Data Needs (Nuclear Heat Supply System)* – "Three DDNs have been identified pertaining to the NGNP Fuel. The first of these DDNs (NHSS-01-01) identifies the need for data to extend the irradiated fuels qualification database from the temperature-burnup envelope of the PBMR Demonstration Power Plant (DPP) to that of the PBMR NGNP. The second DDN (NHSS-01-02) specifies data to correspondingly extend the heat up data pertaining to accident conditions. The third DDN (NHSS-01-03) provides for an extension of the temperature-fluence envelope of the Fuel Graphite to that required by the NGNP. In all three cases, the extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature. The corresponding R&D for the fuel itself comprises irradiation of fuel samples at the higher temperature applicable to the NGNP, post irradiation examination and subsequent heat up of some samples to simulate accident conditions, plus corresponding modeling and analysis. For the Fuel Graphite, the R&D comprises the irradiation of graphite spheres at a temperature and to a fluence level applicable to the NGNP, plus post-irradiation examination and analysis. ***Two DDNs (NHSS-02-01 and NHSS-02-02) have been identified to extend the irradiated materials qualification database for Reflector Graphite from the temperature-fluence envelope of the PBMR DPP to that of the PBMR NGNP***. The extension of PBMR DPP data is required due to		

Table 1E (WEC) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature. The corresponding R&D comprises ***irradiation of graphite samples at low and high temperatures, plus post-irradiation examination and analysis***." (Sec. 16.2.4, p. 16-42) - *Core Structural Ceramics Reflector Graphite R&D* – "In parallel, the NGNP Program at INL is embarking on a graphite development effort that addresses multiple product forms (including NBG-18) and applications (including the PBMR). The INL program places particular emphasis on the understanding of fundamental graphite characteristics that would, ideally, allow the characterization of new coke and/or graphite sources without the need for an extensive irradiation program. To the extent that the INL program addresses NBG-18 and that manufacturing and QA systems development are generic, there is a potential to accelerate the INL effort and reduce its cost by utilizing applicable results of the PBMR DPP development work that would otherwise be duplicated. From the PBMR perspective, there is a potential to ***expand the database supporting NBG-18*** and, potentially, to reduce the scope of surveillance, testing, inspection and maintenance (STIM) required as a basis for operation of the PBMR DPP. Further potential benefits are access to multiple qualified vendors for follow-on PBMR commercial deployments and easing the burden associated with qualification of new graphite sources. In order to take mutual advantage of PBMR's ongoing program to qualify SGL graphite plus INL's and PBMR's mutual interests to cooperate on graphite qualification with SGL and Graftek, efforts are underway to develop a collaborative program. In the interim, a preliminary scope, cost and schedule for R&D activities addressing the Reflector Graphite DDNs for the PBMR NGNP have been developed."		
E-2	Lack of consensus codes and standards. Efforts are under way through the ASME to develop a consensus design code for graphite core components, but to date a useable code has not been approved. ASTM test standards exist for many of the physical properties of concern to the reactor designer, but further work is required, especially in the area of small (irradiation) specimen test methods.	See item E-1 for materials characterization and qualification efforts. (Sec. 16.2.3.1, p. 16-38) – *Fuel Graphite Irradiation Tests* - "Samples for investigation and irradiation will be cut from pressed graphite spheres provided for the test. These samples will be cut parallel and perpendicular to the extrusion direction. Following irradiation, the following characteristics will be measured: • ***Geometrical size*** • ***Mass*** • ***Calculation of sample density*** • ***Measurement of sample density*** • ***Sample porosity*** • ***Thermal conductivity in the range 20 up to Irradiation Temperature*** • ***Electric conductivity in the range 20 up to Irradiation Temperature*** • ***Thermal coefficient of linear expansion in the range 20 up to Irradiation***	NHSS-02-01 and NHSS-02-02 (graphites) HTS-01-17 thru HTS-01-19 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX ceramics)	Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed.

Table 1E (WEC) – GRAPHITE - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		Temperature • ***Dynamic Young's modulus*** • ***Compression strength*** • ***Ultimate bending strength*** • ***Optical ceramography*** • ***Uranium and thorium content*** The above measured characteristics will be compared to values obtained during pre-irradiation characterization."		
E-3	Theoretical models for the effects of neutron damage on the properties of graphite have been developed; however, these models need modification for the new graphites and will need to be extended to higher temperatures and/or higher neutron doses. V&V of theoretical models will require generation of experimental data on the effect of neutron irradiation on properties.	(Sec. 16.2.1.10, p. 16-34) – *Engineering Design Tools R&D* – Table 16.2-7 describes planned and in-progress efforts for developing ***irradiated material behavior models (e.g. graphite blocks); graphite corrosion/oxidation models for air ingress models; and discrete element modeling to simulate behavior of graphite bodies in contact with each other*** (e.g. block reflector structures and fuel spheres). (Section 16.2.2, p. 16-35) Two DDNs (NHSS-02-01 and NHSS-02-02) have been identified to ***extend the irradiated materials qualification database for Reflector Graphite from the temperature-fluence envelope of the PBMR DPP to that of the PBMR NGNP***. The extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature. ***The corresponding R&D comprises irradiation of graphite samples at low and high temperatures, plus post-irradiation examination and analysis***.	NHSS-02-01 and NHSS-02-02 (graphites)	Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed.
E-4	Uncertainties in the temperature and dose received by a component; the severity of temperature and dose gradients in a component; the rate of dimensional change in the specific graphite used in a given design; the extent to which stresses are relieved by irradiation-induced creep; and the extent of changes in key physical properties such as elastic moduli, thermal conductivity, coefficient of thermal expansion, compound to make the prediction of component stress levels, and hence decisions regarding component lifetime and replacement schedules, very imprecise.	See items E1 and E2 for information on graphite materials characterization and behavior, including measurement of physical changes and properties. (Sec. 4.4.2, p. 4-84) – *Future Studies/Core Structural Ceramics (CSC)* – "The ***effect of the higher power level on the life of the CSC needs to be investigated***. It may require more frequent CSC replacements. ***Thermal stresses in the bottom reflector blocks due to the temperature gradient between the inlet and outlet flow need to be assessed. The effect of the increased amount of abrasion of the reflectors due to the higher amount of spheres circulated per day also needs further investigation***." (Sec. 16.2.4.1, p. 16-42) – *Core Structure Ceramics* – Reflector Graphite R&D – "***Two supplemental irradiation series are planned*** for the PBMR NGNP to extend the database supporting the PBMR DPP. One, corresponding to DDN NHSS-02-01, is at low temperature, nominally proposed at 350ºC. The other, corresponding to DDN NHSS-02-02, is at high-temperature, nominally proposed at 950ºC. For both of these series, the initial step is planning for the irradiation test program and	NHSS-02-01 and NHSS-02-02 (graphites)	Needs for greater definition of materials characteristics, development of structural mechanics models, and irradiation tests to determine component lifetime and replacement schedules are recognized in the WEC PCDR.

Table 1E (WEC) – GRAPHITE - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		preparation of the graphite samples and capsules. An initial irradiation is proposed at both temperatures to an intermediate fluence level that corresponds, as a minimum, to the maintenance outage interval of the PBMR NGNP, 5 years. The data provided by this irradiation will be sufficient to support initial operation of the PBMR NGNP and is planned in support of the construction and operating license application. A ***more extended irradiation is planned at both temperatures to confirm or establish the target design life for the replaceable reflector***. The latter is planned for completion prior to initial startup activities.”		
E-5	Whole-core models are required that can predict the stress states of graphite components within the core. Such models should be capable of taking inputs such as temperature and neutron dose and calculating the dimensional change, creep, thermal conductivity, etc., from established theoretical models. Reliable stress-state predictions as a function of reactor life would enable reactor operators and regulators to provide NDE guidance and make decisions regarding inspection intervals and core block replacement.	See item A-8 for information on development of modeling tools. (Sec. DDN NHSS-02-01, p. 4-80) – *Extended Properties of Irradiated Graphite at Low Temperatures* – “Within the PSMP envelope depicted, the initial design of the PBMR DPP graphite structures is based on historical irradiation test data. ***This data is summarized and used to generate the model for irradiated properties that is used for the design….This model will be validated and extended by making use of the material irradiation test data*** generated as a result of the PSMP. The proposed approach is to obtain the data necessary to ***extend the present PSMP envelope to lower temperatures*** and, thus, to qualify the Reflector Graphite for NGNP operational conditions.” (Sec. DDN NHSS-02-02, p. 4-82) – *Extended Properties of Irradiated Graphite at High Temperatures* – “Within the PSMP envelope depicted, the initial design of the PBMR DPP graphite structures is based on historical irradiation test data. ***This data is summarized and used to generate the model for irradiated properties that is used for the design. This model will be validated and extended by making use of the material irradiation test data*** generated as a result of the PSMP…. The proposed approach is to obtain the data necessary to ***extend the present PSMP envelope to high temperatures and fluence levels*** and, thus, to qualify the Reflector Graphite for NGNP operational conditions.” (Sec. 12.1.1, p. 12-11) – *Maintainability* – “Plant outages are scheduled at 5-year intervals. The minimum lifetime of IHX A of 10 years and the requirement of ***maximum 6-year intervals between maintenance and inspection*** on other SSC’s are the main constraints in deciding on intervals between scheduled outages. Scheduled outages will last maximum 30 days for outages scheduled after 5 years and 15 years. The 10 year outage requires the replacement of the IHX A and will last a maximum of 50 days. The 20 year outage requires the ***replacement of ceramic core structures*** and will last a maximum of 180 days. The scheduled maintenance outages are repeated in 20 year cycles.” (Sec. 12.3.1.1.1, p. 12-20) – *Core Structures and Reactor Pressure Vessel* – “The	NHSS-02-01 and NHSS-02-02 (graphites)	The WEC PCDR has addressed this topic.

Table 1E (WEC) – GRAPHITE - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions
		reactor core structure and Reactor Pressure Vessel (RPV) are designed not to require any maintenance or access during normal operation. Maintenance that will occur in 20 year intervals, when the side and central reflectors are replaced. ***Replacement of the reflectors*** will require special tools as the initial core structure installation and commissioning process is in a 'clean' (non-nuclear) environment, and personnel access is not a significant limitation. The core structure assembly tools will then later undergo further development for use in a radiological contaminated environment to perform the core structure refit. The initial installation activities would help to evaluate and validate the replacement concept. A system will be conceptualized to be able to perform a core structure refit during the reflector replacement outage. This entails dismantling the top end of the reactor (such as removing the RSS assemblies), and removing the centre column. The transport and storage of the core structure's used graphite blocks are accommodated within the site infrastructure." (Sec. 12.5.2.4, p. 12-38) – *Replacement of the Core Structures Ceramic (CSC)* – "The requirement of 500 MWt power will inevitably lead to higher neutron fluence in the ceramic core structures. The ***lifetime of the graphite structures in the core, especially the central reflector, needs to be investigated for the high fluences*** encountered in the NGNP. The accessibility of the ceramic core structures also need to be addressed, as the removal of the RPV head is a complicated undertaking."		
E-6	Basic research should be conducted to strengthen the understanding and modeling capability of the displacement damage process in graphite***.*** In addition, in graphite technology, there is a need for analytical models for oxidation, changes in physical properties, irradiation induced dimensional change, and irradiation creep. They could be developed to feed into a structural integrity model for the graphite core which would be used for core design and safety assessment.	See items A-8 and E-5 for information on development of modeling tools. See item E1 for information on graphite materials characterization and behavior. (Sec. 16.9.5, p. 16-105) – *Fundamental Properties of Nuclear Graphite* – "The properties of nuclear-grade graphite, and the changes in those properties as a function of temperature and fluence, are known to vary, depending upon the source of raw materials and the specific details of the processing steps used in its manufacture. The graphite R&D program proposed for the PBMR NGNP (Section 16.3) comprises the minimum incremental enabling technology requirements relative to the PBMR DPP. However, a more ***fundamental understanding of graphite properties as a function of raw material sources and processing would be highly desirable*** in the context of an expanding commercial market for High-Temperature Gas Cooled Reactors (HTGRs). Through such improved understanding, the need for expensive and time-consuming graphite qualification programs based on irradiation capsules might be avoided or at least reduced in scope. Further, a ***more fundamental understanding of graphite properties might provide the basis for enhancing the life span of reflector graphite***. The expanded graphite development program, outlined by INL [Ref 16-4], appears to provide the initial steps toward such an improved fundamental understanding. As already noted in Section 16.2, there is a significant potential for collaboration	NHSS-02-01 and NHSS-02-02 (graphites)	Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR.

Table 1E (WEC) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		between INL/NGNP in the PBMR in this area, and steps are underway to affect such collaboration."		
E-7	Irradiation induced change in the coefficient of thermal expansion, including effects of creep strain.	See items E1 and E2 for information on graphite materials characterization and behavior, including measurement of physical changes and properties.	NHSS-02-01 and NHSS-02-02 (graphites)	Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR.
E-8	Irradiation induced change in mechanical properties such as strength and toughness, including the effect of creep strain.	See items E1 and E2 for information on graphite materials characterization and behavior, including measurement of physical changes and properties.	NHSS-02-01 and NHSS-02-02 (graphites)	Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR.
E-9	Blockage of coolant channel in a fuel element block or reactivity control block due to graphite failure and/or graphite spalling.	(Sec. 16.2.1.6, p. 16-31) – *Core Structural Ceramics R&D* – "The most demanding Core Structural Ceramics R&D activities supporting the PBMR DPP center upon the Reflector Graphite. The Reflector Graphite components, and specifically the replaceable reflector components immediately adjacent to the core, are the only Core Structural Ceramic components for which fluence-related life limits must be established. The PBMR-Specific Materials Test Reactor Program (PSMP) is structured to provide data supporting startup and initial operation of the PBMR to the first planned outage period (6 years) as input to the licensing process. Completion of the PSMP is planned for the timeframe of plant startup in 2012. The results of the PSMP will be confirmed through surveillance, testing, inspection and maintenance activities over the plant operating lifetime." (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* – "In Progress: Irradiated Graphite material behavior models. Implementation of non-linear material behavior models for use with commercial FEA codes in ***predicting distortion and material properties of graphite blocks under irradiation. Includes implementation of graphite failure models***."	NHSS-02-01 and NHSS-02-02 (graphites)	Needs for greater definition of materials characteristics, and development of structural mechanics models, including implementation of failure models, are recognized in the WEC PCDR.
E-10	Statistical variation of non-irradiated properties, due to forming, processing, raw materials, and formulation.	(Section 16.2.1.6, p.16-31) Completed: Reflector Graphite Unirradiated Properties. Characterize the unirradiated properties of NBG-18 graphite. (Sec. 16.9.5, p. 16-105) – *Fundamental Properties of Nuclear Graphite* - The ***properties of nuclear-grade graphite, and the changes in those properties as a function of temperature and fluence, are known to vary, depending upon the source of raw materials and the specific details of the processing steps used in its manufacture.*** The graphite R&D program proposed for the PBMR NGNP (Section 16.3) comprises the minimum incremental enabling technology requirements relative to the PBMR DPP. However, a more fundamental understanding of graphite properties as a function of raw material sources and processing would be highly desirable in the context of an expanding commercial market for High-Temperature Gas Cooled Reactors (HTGRs). Through such improved understanding, the need for expensive and time-consuming graphite qualification programs based on irradiation	NHSS-02-01 and NHSS-02-02 (graphites) HTS-02-01 thru HTS-02-06 (IHX ceramics)	This item has been addressed in the WEC PCDR.

Table 1E (WEC) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		capsules might be avoided or at least reduced in scope. Further, a more fundamental understanding of graphite properties might provide the basis for enhancing the life span of reflector graphite. The expanded graphite development program, outlined by INL [Ref 16-4], appears to provide the initial steps toward such an improved fundamental understanding. As already noted in Section 16.2, there is a significant potential for collaboration between INL/NGNP in the PBMR in this area, and steps are underway to affect such collaboration.		
E-11	Ability to develop generic specifications that will ensure consistency of graphite quality over the lifetime of the reactor fleet, including for replacement components.	(Section 16.2.1.6, p.16-31) ***Completed: Reflector graphite specification. Establish the specification for procurement of NBG-18 graphite***.	NHSS-02-01 and NHSS-02-02 (graphites)	This item has been addressed in the WEC PCDR.
E-12	Tribology (effects of moving surface interactions) of graphite in helium environment, including potentially impure helium environment (examples: surfaces sticking together; surfaces wearing on each other to generate dust, etc.)	(Sec. 4.2.7.2, p. 4-60) – *Helium Purification System* - The Helium Purification System (HPURS) is ***used to provide the required degree of helium purity***. High purity coolant is required in order to minimize corrosion and contamination in the PHTS and SHTS. This is done by bleeding off a partial flow of helium from the PHTS and SHTS. The extraction point is from the highest pressure points, i.e. the PHTS and SHTS circulator discharges within the HTS. This flow is tapped off constantly during operation of the plant. The ***HPS removes chemical gaseous contaminants from the primary coolant within the PHTS*** by the use of, catalysts, adsorbers and the manipulation of helium temperature extracted from the PHTS and SHTS. The required helium purity levels will be confirmed during the conceptual design. Sec. 16.2.1.10, p. 16-34) – *Engineering Design Tools R&D* – Table 16.2-7 describes planned and in-progress efforts for developing irradiated material behavior models (e.g. graphite blocks); graphite corrosion/oxidation models for air ingress models; and ***discrete element modeling to simulate behavior of graphite bodies in contact with each other (e.g. block reflector structures and fuel spheres).***	NHSS-02-01 and NHSS-02-02 (graphites)	This item has been addressed in the WEC PCDR.
E-13	Impact of degradation of thermal conductivity on fuel temperature limits.	(Sec. 16.2.3.1, p. 16-38) – *Fuel Graphite Irradiation Tests* - "Samples for investigation and irradiation will be cut from pressed graphite spheres provided for the test. These samples will be cut parallel and perpendicular to the extrusion direction. Following irradiation, the following characteristics will be measured: • Geometrical size • Mass • Calculation of sample density • Measurement of sample density • Sample porosity • ***Thermal conductivity in the range 20 up to Irradiation Temperature*** • Electric conductivity in the range 20 up to Irradiation Temperature • Thermal coefficient of linear expansion in the range 20 up to Irradiation	NHSS-02-01 and NHSS-02-02 (graphites)	This item has been addressed in the WEC PCDR.

Table 1E (WEC) – GRAPHITE - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		Temperature • Dynamic Young's modulus • Compression strength • Ultimate bending strength • Optical ceramography • Uranium and thorium content The above measured characteristics will be compared to values obtained during pre-irradiation characterization."		

Table 1F (WEC) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
F-1	Cold oxygen (O2) and other heavy-gas accidental releases from the process plant that can flow from the chemical plant to the nuclear plant (depending upon wind, relative plant elevations, and nuclear plant air intakes) and potentially impact the integrity of reactor systems, structures, and components (SSCs). All of the proposed processes for production of hydrogen start with water, and thus all of the processes will produce oxygen as a byproduct of hydrogen production. Oxygen is the one common chemical safety issue that can impact nuclear plant safety. At high oxygen concentrations, many "noncombustible" materials become combustible and the potential for spontaneous combustion increases. Increased oxygen levels at the reactor can compromise the functioning of safety equipment.	(Sec. 2.2.1.3.2, p. 2-32) – *Hydrogen Production System (HPS) Configuration* – "The HPS facility shall be separated from the NHSS consistent with commercial plant economic and risk tradeoffs. The ***interfaces between the HPS and the NHSS shall be designed to ensure that failures or upset conditions in the HPS do not result in failures or adverse impacts to the NHSS***." (Sec. 2.2.1.3.4, p. 2-33) – *HPS Safety & Licensing* – "The ***hydrogen production facilities, including the conversion, storage, and distribution systems, shall comply with the requirements of 29 CFR 1910.103, Occupational Safety and Health Standards (OSHA), Subpart H - Hazardous Materials, Hydrogen. In the event that the HPS facility also produces and stores significant quantities of oxygen, the requirements of 29 CFR 1910.104, Oxygen, shall be applied. The design, operation and maintenance of the HPS shall comply with 29CFR 1910.119, "Process Safety Management of Highly Hazardous Chemicals"."*** (Sec. 7) - WEC's focus during pre-conceptual design was on the hybrid sulfur (HyS) method for hydrogen production. Clearly, this technology produces oxygen as an intended end product, and assumptions are made in the pre-conceptual design that the oxygen will be purified and delivered through a pipeline to the plant boundary, for final transportation and sale. (Sec. 7.2.1.2,, p. 7-45) - "*Safety*" - Indicates that the ***HyS system will meet NFPA and OSHA requirements, and eventually will be subjected to a process hazards analysis***. (Sec. 7.2.1.4, p. 7-45) - "*Environmental Interface*" - ***Allows for the possibility that the oxygen may be vented rather than sold, indicating only that the O2 would need to be purified to meet emissions standards if this method of disposition were chosen***. (Sec. 7.2.1.4, p. 7-49) - "*Interface with Workers and the Public*" – "Other potential hazards that must be considered during future design phases are the possibility of hot oxygen, SO_2, SO_3 and sulfuric acid vapor leaks. Hot oxygen presents a serious fire hazard. Many substances that will not burn in air will burn vigorously in a stream of hot oxygen. The sulfur-containing gases are much denser than air and can form dangerous clouds, especially in confined spaces. The plant contains large amounts of liquid sulfuric acid in a range of concentrations. All of these liquids are dangerous if spilled. ***In future design phases attention must be paid to prevention and handling of spills***."	(None)	Indications are that WEC has addressed this item in the PCDR and will address it in more detail, via process hazards analysis, as the project progresses.

Table 1F (WEC) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		(Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, ***a full scope PRA that addresses all internal and external hazards***, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a ***Process Hazards Assessment (PHA) for the Hydrogen Production Facility*** (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP."		
F-2	Failure of the IHX leading to potential damage to safety-related SSCs in the reactor due to blow-down effects from large mass transfer and over-pressurization of either secondary or primary side. The impact of the IHX failure depends upon the selection of the heat transfer fluid in the secondary heat transport loop. Helium is the leading candidate for the heat transport loop, but no final decisions have been made. If helium is used, the helium inventory in the secondary loop may be greater than the inventory in the reactor; thus, any leak in the IHX can significantly increase the total helium inventory involved in any reactor depressurization event.	WEC considered metallic and ceramic technologies. For the metallic IHX, several ***DDNs were identified to provide characterization and qualification of materials (Alloy 617 and Alloy 230). Quality research in these areas would reduce the probability if IHX failure***….The DDNs for ceramics are more developmental in nature but should have an influence on the probability of IHX failure should a ceramic IHX be used. (Sec. 11.2.2.4, p. 11-14) – *Helium Pressure Control* – "The pressure in the PHTS and SHTS will be controlled by utilizing the Pressure Control System, which is a subsystem of the Helium Service System (HSS). The ***HSS will govern and control the amount of helium (pressure) in the PHTS and SHTS according to predefined pressure set-points for the PHTS and SHTS***. During normal operation the plant will generally operate at the rated pressure levels and very little pressure control actions are envisioned. However, during plant start-up, transitions and transient events, pressure control is an important control function. The HSS will be responsible for controlling the pressure in such a way that the pressure differentials across the components in the HTS (especially the IHX, PCHX and SG) remain within specified limits to ***avoid operation of the components outside their design envelopes***. The HSS will control the pressure inside the HTS by injecting/extracting helium from/to a higher/lower pressure source." (Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a ***full scope PRA that addresses all internal and external hazards,*** including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Sec. 16.3.2, pp. 16-50 through 16-59) – This section provides a ***detailed plan for IHX R&D***, including metallic and composite materials.	HTS-01-01 thru HTS-01-12 (IHX metallics) HTS-02-01 thru HTS-02-06 (IHX ceramics)	Indications are that this item has been addressed in the WEC PCDR, and will be further addressed with design improvements as the project progresses.
F-3	Failure of the process heat exchanger (PHX) leading to potential damage to safety-related SSCs in the reactor, due	***Development for metallic materials in the process coupling heat exchanger (PCHX) will be covered by the metallic IHX developmental research***.	HTS-01-01 through HTS-	Indications are that this item has been addressed in the WEC PCDR, and will be

Table 1F (WEC) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
	to fuel and primary system corrosion from the introduction of corrosive process plant chemicals leaking down the process heat transport line and failing the IHX.	(Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a full scope PRA that addresses all internal and external hazards, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a ***Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements***. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Sec. 16.4.1, p. 16-65) – *Design Data Needs (Hydrogen Production Facility)* – "Note that two reference materials have been identified for the IHX, Alloy 617 and Alloy 230. These same materials will be tested for the PCHX design."	01-12 (IHX metallics) HPS-04-03 thru HPS-04-07 (hydrogen production)	further addressed with design improvements as the project progresses.
F-4	Steam generator failures leading to the introduction of steam/water into the primary system, potentially causing a reactivity spike and chemical attack of the TRISO fuel particle coatings and graphite. Some hydrogen production processes, such as high-temperature electrolysis, require steam as a process feedstock; thus, the high-temperature reactor may be required to provide high-temperature steam.	The WEC PCDR indicates that this ***scenario is not applicable to the WEC design because the scenario is not probable***. The S/G in the WEC design does not interface with the primary heat transport system. Instead, it interfaces with the helium-filled secondary heat transport system. This provides isolation from the primary system. (Sec. 8, p. 8-8) - *Power Conversion System: Summary and Conclusions* – "The Steam Generator has been identified as a developmental component based on prior design development experience for other High Temperature Gas Reactors (HTGR) applications. The requirements, configuration, materials and design features of this component require that a number of Design Data Needs (DDNs) be satisfied for successful design, manufacturing, delivery and long term operation of the prototype and follow-on components. ***Eighteen items of need are identified for the Steam Generator***…. It is recommended that a future study be conducted that will evaluate alternative approaches for the steam generator including more conventional designs (e.g. refractory lined, U tube) compared to the once through helical type SG proposed in the preconceptual design. Single vs. multiple trains will be evaluated. The results of the study will establish a path forward for design development of the steam generator." (Note: The 18 DDNs for the steam generator are enumerated and described in Table 8-20, page 8-40.) (Sec. 10.2.5, p. 10-27) – *Steam Generator Building* – "Major systems and components include the Secondary Heat Transport System helium piping, the PCS steam generator, steam generator supports, feedwater and main steam system piping, and main steam safety valves. ***Liquid secondary containment is required in the event of a major liquid spill or leakage of PCS feedwater. Physical separation is required between this building and the NHSS Building***. The NHSB exterior walls provide a barrier designed for security, fire and potential pressure loads	SG-01-01 thru SG-01-17 (steam generator)	The WEC PCDR indicates that this issue scenario is not probable for the PBMR design. Nonetheless, significant R&D is planned to ensure reliability of the steam generator.

Table 1F (WEC) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION

Item	NRC Need/Issue Identified	Applicable Westinghouse R&D *or* Already-Identified Solution	Related DDNs	Comments/Conclusions

		due to pipe rupture in this building. This building and its contents are not credited in the mitigation of design basis events affecting the NHSS. In addition, the ***building or its contents do not interact with the NHSS building in a manner that compromises the safety functions of the NHSS***."
		
		(Sec 11.5.2, p. 11-30) – Steam *Generator Operating Conditions* – "The SG will operate at helium inlet temperatures of up to 900°C in certain modes of plant operation. A future study is needed to investigate the effect that extended period of operation at these conditions will have on the SG design."
		
		(Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a full scope PRA that addresses all internal and external hazards, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a ***Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements***. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP."
		
		(Sec. 16.5.1, p. 16-95) – The ***Design Data Needs associated with the steam generator*** are shown in Table 16.5-1 and are categorized as follows:
		• Materials - indicates that material related data is needed to support long term operation in the specified helium and secondary environments.
		• Methods Development, Verification and Validation (MDV&V) - this grouping is further categorized as:
		    o Performance- indicates that configuration specific testing is needed to support final design.
		    o Design - configuration specific testing or mock-ups to support final design of the demonstration unit.
		    o Fabrication - configuration specific mock-ups to qualify techniques for prototype fabrication.		

F-5	Loss of the pressurized coolant inventory from the intermediate loop leading to a loss of primary reactor heat sink and the potential for hydrodynamic forces on the IHX leading to IHX failure and loss of reactor primary system coolant.	***Development for metallic materials in the secondary heat transport system is expected to be enveloped by the metallic IHX developmental research.*** (Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a ***full scope PRA that addresses all internal and external hazards***, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a ***Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements***. This preliminary PHA will help	HTS-01-01 through HTS-01-12 (IHX metallics)	Indications are that this item has been addressed in the WEC PCDR, and will be further addressed with design improvements as the project progresses.

Table 1F (WEC) – PROCESS HEAT FOR HYDROGEN - DATA COLLECTION				
Item	**NRC Need/Issue Identified**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Related DDNs**	**Comments/Conclusions**
		establish the specific safety design requirements and criteria for safe operation of the NGNP." (Sec. 16.3.2, pp. 16-50 through 16-59) – This section provides a ***detailed plan for IHX R&D, including metallic and composite materials.***		

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
A-1	*Core-Coolant Bypass Flow Phenomena (Normal Operation)* • Overcome difficulties in estimating bypass flow • More complete understanding and accounting of related design features such as fuel blocks (PMR) and core barrel configurations • In-core temperature testing • Parametric analysis of gap configurations to bound questions associated with gap and bypass flows	(Sec 4.3.1, p. 42) – "The ***core bypass flow*** shall be maintained within an acceptable range which ensures a good compromise for the fuel temperature in ***normal and accidental conditions*** (existence of a minimum amount of bypass in lateral reflector)." (Sec. 6.1.1.7.1, p. 52) - ***Core Bypass Flow*** – "The major issue of the thermal-hydraulic design is the core bypass flow. It is directly related to the core thermal performance. In the core, the flow partitions itself among the coolant channels, the absorber element channels and the gaps between the columns of fuel and reflector blocks. The objective in the core flow design is to maximize flow through the coolant channels (which directly flow to where the power is being produced). This means ***minimizing flow through the gaps between columns,*** and limiting the flow in the absorber element channels to that needed to cool the absorbers when they are inserted. ***Refined analyses will need to be performed in the frame of the Conceptual Design to assess the value of core bypass flow and propose design improvements to minimize the bypass***." (Sec. 17.6, p. 240) – ***Initial Startup Operations and Testing*** – "…The schedule provides four years for this phase of plant operations. The first two years (2017 and 2018) is dedicated to non-nuclear system testing and turn-over including the standard system turn-over from construction to operations. The second two years (2019 and 2020) includes initial plant criticality. During this phase all safety systems will be examined and tested and several special licensing related tests is planned. This phase of the plant operation includes component dismantling and inspection and fuel examination." (Sec. 19.2.3, p. 289) – ***Instrumentation*** – "NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy. For neutron flux detectors some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 °C. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired."	See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3.1.2.2, p. 3-27) - ***Bypass Flow Reduction*** – "Fuel temperatures can be reduced by reducing bypass flow. Bypass flow is defined as any flow that bypasses the coolant holes of the fuel elements. As shown in Figure 3.1-20, ***bypass flow channels include gaps between fuel columns and leakage between/from PSR blocks***. For the reference GT-MHR core design, approximately 3% of the flow is supplied to the control-rod channels, which have ***orifices to minimize bypass flow*** while also maintaining adequate cooling for the control rods. Composite-clad control rods require little or no cooling, which helps reduce the bypass flow fraction. Bypass flow can also be ***reduced by using graphite sealing keys*** below the active core to provide additional flow resistance for bypass flow occurring between fuel columns. ***Lateral restraint devices and sealing tubes in the PSR riser channels*** can reduce the leakage flow between/from the PSR blocks…FES has analyzed the flow distribution in the reactor vessel using a 3-D, 120°-sector ANSYS model (Figure 3.1-23). For the reference GT-MHR design (Figure 3.1-8), the bypass flow fraction is approximately 0.20. As shown in Table 3.1-9, routing the inlet flow through the PSR increases the bypass flow fraction to 0.37, primarily because of the relatively large lateral pressure gradients between the inlet flow path and reactor core. Adding sealing sleeves and lateral restraints reduces the bypass flow fraction to 0.14. Adding sealing keys at the bottom of the core further reduces the bypass flow fraction to 0.10. Reducing the bypass flow fraction from 0.20 to 0.10 reduces peak fuel temperatures by approximately 50°C. The FES results are consistent with recent calculations performed by OKBM for their design concept shown in Figure 3.1-10. The OKBM design also includes sealing sleeves in the coolant riser paths and lateral restraints to reduce bypass flow. As indicated in Figure 3.1-24, OKBM also estimates the bypass flow fraction to be approximately 0.10. FES has also performed flow distribution calculations using the FLOWNET flow network code." (sec. 3.1.2.2, p. 3-60) – "Design modifications to reduce bypass flow include sealing sleeves in the PSR risers, lateral restraints, and sealing keys below the core. Independent analyses by FES and OKBM indicate that bypass flow can be reduced to about 10%. The ***reactor internals design should be developed in more detail and include these modifications***." (Sec. 3.10.1.3, p. 3-200) - ***Nuclear Power Instrumentation*** – "Power-range ***ex-core neutron detectors*** will be placed in six detector wells, equally spaced around the reactor vessel. Each well will extend from the lower region of the reactor core to the refueling floor. Access to the detector wells will be from the refueling floor. Neutron detection equipment will include Intermediate and Power Range	See item A-8 for information relating to modeling activities. See item C-6 for information regarding testing. See Item E-1 for graphite materials information, including core blocks. (Sec. 4.2.10, p. 4-68 thru 4-76) - *NHSS Control and Instrumentation System* – "The NHSS Control and Instrumentation System comprise only of equipment found in the Nuclear Heat Supply Building and all operator interaction is performed in the Central Control & Supervisory System (CCSS) as described in the Section 9: Balance of Plant Systems. The primary systems comprising the NHSS Control and Instrumentation system are the Operational Control System (OCS), the Equipment Protection System (EPS) and the Reactor Protective System (RPS). The OCS monitors and controls the NHSS systems throughout their normal operating range. The EPS detects operating regimes or operating conditions that may be harmful to NHSS equipment, and takes appropriate action to prevent or minimize potential damage. The RPS automatically initiates RUS protection whenever pre-established set points are exceeded…. The RPS consists of three subsystems: Reactor Trip System (RTS), Post Event Instrumentation System (PEI) and the Manual Diverse Shutdown System (MDSS). The RPS provides functions to prevent exceeding predefined safe operating limits and to provide information to operators in the event of nuclear accidents. The Protection System is implemented using a Class 1E qualified digital platform that is capable of performing logical operations as well as algorithmic processing. The same platform is used for both the RTS and the PEI applications. The MDSS is a hard-wired system that allows tripping the breakers without dependence on any software. All portions of the RPS are treated as Class 1E Structures, Systems and Components (SSC)…. The RTS automatically prevents operation of the RUS in an unsafe condition by shutting down the RUS whenever predetermined operating limits are approached, or when Design Basis Accident conditions are detected. The operating limits are selected, based on initial conditions (RUS power and outlet temperature) assumed in the safety analysis…. The MDSS is a hardwired system that enables operators to manually initiate reactor trip, RCS and RSS activation functions, from both the Main Station Control Room and the PEMRR. The MDSS controls are supported by monitoring instrumentation associated with the PEI system or the Plant Computer displays to provide the operator the capability to know when to take the appropriate manual action. For each reactor trip function (RS and RSS activation), a set of three switches that are individual and redundant are provided in the Main Station Control Room and a set of three switches that are individual and redundant are provided in the PEMRR. Thus, a total of 4 sets of three switches each comprise the MDSS for the protection system."	• Based on review of AREVA PCDR: The needs for refined analyses to better understand the core bypass flow phenomenon, and core monitoring instrumentation and testing, have been recognized in the AREVA PCDR. • Based on review of GA PCDR: The needs for refined analyses to better understand the core bypass flow phenomenon, and core monitoring instrumentation and testing, have been recognized in the General Atomics PCDR. • Based on review of WEC PCDR: ***There is no indication that this issue has been specifically addressed in the WEC PCDR.*** • NGNP R&D Response: Experiments and analysis planned in R&D program. Instrumentation R&D is needed. No real funding (except small amount for university) available.

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
			Monitoring Channels, and Source Range Detector Assemblies and Monitoring Channels. The latter are retractable ***in-core devices***, and entail a significant NGNP development effort. “	(Sec. 4.4.2, p. 4-84) – *Future Studies/Flow Requirements* – “…Assess the effect of the non-symmetrical flow in the riser channels on the temperature distribution of the Side Reflector and core barrel assembly (CBA). This is due to the lack of riser channels above the outlet pipes. Assess the flow distribution in the outlet slots due to the changes in the outlet plenum for mal-distribution. ***Assess the sealing of the riser channels in the bottom reflector to limit direct core bypass flow. PLOFC flows need to be assessed to ensure that no hot gas flows down the risers and overheats the reactor pressure vessel (RPV).*** The higher heat loss to the RCCS caused by the higher RPV temperature needs to be considered.” (Sec. 16.9.5, p. 16-105) – *Design and Evaluation Model Verification and Validation* – “In order to analyze various aspects of the integrated NGNP, the ***software currently used for the PBMR–DPP design will be enhanced and extended to have an integrated core neutronic/thermal hydraulic analysis tool*** coupled to the Power Conversion and Hydrogen Production Systems models for simulating normal and off-design conditions. This tool will allow for steady-state and transient analyses of the integrated NGNP plant and will enable operational and control studies. Various verification and validation activities are required to ensure that this tool provide accurate results. Aspects that will require V&V are: 1. The input data used for these models and calculations will also need to be verified to ensure accurate results. Examples of data required are state of the art cross section data that will reduce uncertainty and calculation margins as well as input data for the Hydrogen Production System and heat transfer correlations used in the evaluation models. 2. Calculation model verification and validation for various phenomena and performance conditions. Specific aspects that need to be addressed in HTR cores are: 1. Non-local heat generation in HTR cores - An issue that is contributing to uncertainties in the coupled neutronics and thermal-hydraulics of pebble bed reactors design is the treatment of the so-called non-local heat generation contributors. These include: a. Heat generation due to γ-radiation and neutron moderation; b. ***Heat redistribution in the reflector regions due to the bypass flows***; c. Non-local γ-power in the reflector during a depressurized loss of coolant (DLOFC) event. 2. Cold critical experiment – Design techniques and methodologies implemented in the design codes need to be validated. A cold critical experiment will	

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				provide the opportunity to measure many parameters that are calculated. 3. Dust modeling and associated impact on plant performance and safety also need to be addressed."	
A-2	*Effective Core Thermal Conductivity* • For prismatic cores – Make available dose and temperature-dependent graphite thermal properties (especially thermal conductivity) to the NRC T/F code suite, to account for large uncertainties as well as for characterization of annealing effects during long-term heat-up D-LOFC accidents. • For pebble bed cores - Also considerable error bounds in effective core thermal conductivity as a function of both temperature and irradiation. Existing correlations available are empirical, but PBMR project has an experimental facility to be used to refine the database.	(Sec. 19.2, p. 281) – ***R&D Needs*** – "Materials development and qualification. This covers certain high-temperature steels, composites, and ***graphite*** selection/qualification." (Sec. 19.2.2.2, p. 285) – ***Ceramics*** – "No nuclear components or structures made of composites were used for the past HTRs or for other reactor concepts. The use of composites is driven by their high resistance to high or very high temperatures. An R&D program has been launched in the frame of Antares to explore the possible use of such materials inside the primary circuit. Thermal insulation, using composite materials, will be needed to provide thermal protection of metallic components which would otherwise be subjected to helium at very high temperatures. The R&D needs for applied composite materials (C/C or C/SiC composites) emphasizes qualification of material properties such as: 1. ***thermal-physical properties (thermal conductivity*** (K), coefficient of thermal expansion (CTE), heat capacity (Cp)), 2. mechanical properties including multiaxial strength, 3. fracture properties, 4. fatigue properties and 5. behavior in an oxidizing atmosphere and oxidation effects on properties." (Sec. 19.2.2.3, p. 286) - ***Graphite Materials*** – "Graphite, an essential structural material for the VHTR, will operate under significant irradiation conditions and requires a characterization in the range of expected temperatures. Nuclear grade graphite was used in past HTRs programs, amassing a substantial database. These grades are no longer available. An R&D program has been launched within Antares program to select the best candidates among the new available grades or to request the development of a new grade, and to acquire design data. Nuclear graded structural graphite (PCEA, NBG17 and/or NBG18) qualification includes: 1***. thermal-physical properties*** (***K***, CTE, Cp, emissivity), 2. mechanical properties including multiaxial strength, 3. fracture properties, 4. fatigue properties, 5. irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep, 6. behavior under oxidized atmosphere including oxidation effects on properties and 7. tribology. Due to schedule limits, it is recommended that graphite R&D be performed in two phases: preliminary and detailed… Development of ASME and ASTM codes and standards for	See item E-1 for information relating to graphite materials program. (Sec. 3.1.2.2, p. 3-43) – "A 30-deg. sector ANSYS model was used to analyze both low-pressure conduction cooldown (LPCC) and high-pressure conduction cooldown (HPCC) events. In order to reduce vessel temperatures during these accidents, the reactor internal design was modified to include a 100-mm layer of carbon insulation on the outer radial boundary of the PSR…A key parameter for these calculations is the ***graphite thermal conductivity***, which decreases with damage caused by neutron irradiation. For these studies, calculations were performed using both irradiated and unirradiated graphite properties. Calculations were also performed assuming ***annealing of irradiation damage*** as the graphite temperature increases according to the GA model for H-451 graphite. Full recovery from irradiation damage is assumed to occur at temperatures greater than 1300°C. The ANSYS model shown in Figure 3.1-41 was used to calculate the effective thermal conductivity of the graphite blocks. ***Other key parameters that affect heat transfer to the RCCS are the emissivities of the PSR, core barrel, RPV, and RCCS panels***…"	See item C-6 for information on testing and test facilities. The WEC technology development report, in Section 16.2.2 identified the need to extend the temperature-fluence envelope for the fuel graphite. The needed R&D includes irradiation of graphite spheres at a temperature and to a fluence level applicable to the NGNP, plus post-irradiation examination and analysis. (Section 16.2.3.1, p. 16-38) Samples for investigation and irradiation will be cut from pressed graphite spheres provided for the test. These samples will be cut parallel and perpendicular to the extrusion direction. Following irradiation, the following characteristics will be measured: • Geometrical size • Mass • Calculation of sample density • Measurement of sample density • Sample porosity • ***Thermal conductivity in the range 20 up to Irradiation Temperature*** • Electric conductivity in the range 20 up to Irradiation Temperature • Thermal coefficient of linear expansion in the range 20 up to Irradiation Temperature • Dynamic Young's modulus • Compression strength • Ultimate bending strength • Optical ceramography • Uranium and thorium content The above measured characteristics will be compared to values obtained during pre-irradiation characterization.	• *Based on review of AREVA PCDR:* The needs for determining the properties of graphite materials, including thermal conductivity, have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for determining the properties of graphite materials, including thermal conductivity, have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: Indications are that this item has been addressed in the WEC PCDR, and will be further addressed as the project progresses. • *NGNP R&D Response*: R&D planned in Graphite program to cover this.

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		graphite is essential for timely application graphite for NGNP reactor." The pebble bed core portion of the item is not applicable to the PMR.			
A-3	*Afterheat Correlations* • Peak fuel temperatures in the D-LOFC accident are very sensitive to the afterheat (vs. time) to the same extent as they are to the core thermal conductivity function. Afterheat correlations are sensitive to fuel type and burn-up histories. Tracking fuel histories during operation can be challenging, and afterheat validation data is more difficult to obtain for long times after shutdown.	***There is no indication that this item has been specifically addressed.*** However, reactor system analysis software (MANTA, RELAP), neutronics software (MCNP, NEPHTYS, MONTENURNS, CABERNET), and fuel performance software (ATLAS) are addressed in section 19.2.4, including discussions of ***fuel burnup***.	See item A-8 for information relating to development of analytical models. (Sec. 5.1.3, p. 5-7) – Accident/Transient Analysis – "In terms of safety consequences, the bounding accidents for the NGNP are a loss of flow leading to a high pressure conduction cooldown (HPCC) and loss of coolant leading to a low pressure conduction cooldown (LPCC). The HPCC event is typically initiated by trip of the PCS. The RPS automatically initiates a reactor trip on low flow or TM trip. The system pressure quickly equilibrates at about 5 MPa as the TM coasts down. Because the system remains at high pressure, the decay heat is more uniformly distributed within the core and vessel than during a LPCC event. The LPCC event is typically initiated by a small primary coolant leak, causing the system to depressurize to atmospheric pressure. The RPS automatically initiates a reactor trip on low coolant pressure. For both events, the SCS fails to start and decay heat is removed by thermal radiation and natural convection from the reactor vessel to the RCCS. These events have been analyzed in detail for the GT-MHR, and the results have shown that ***peak fuel temperatures*** remain below the design goal of 1600°C, and the temperatures for the vessel and other safety-related SSCs also remain below acceptable limits. ***Using an ATHENA model, these events were re-analyzed using the NGNP initial conditions***. Figure 5.1-3 shows the calculated peak fuel temperatures for the HPCC and LPCC events. For the LPCC event, the peak fuel temperature is 1525°C and occurs about 60 h following initiation of the event. For the HPCC event, the peak fuel temperature is 1349°C and occurs about 50 h following initiation of the event. As shown in Figure 5.1-4, the calculated peak vessel temperatures for the HPCC and LPCC events were approximately 478°C and 517°C, respectively. For both events, the peak vessel temperatures occurred about 72 h following initiation of the event. ***These results are consistent with previous results for the GT-MHR and show that the H2-MHR should retain the passive safety characteristics of the GT-MHR."***	(Sec. 2.1.1, p. 2-13) – ***Commercial Plant Summary Description*** - The reference PBMR H2 PHP fuel is TRISO-coated UO_2 fuel particles embedded in the spherical pebble fuel elements. The pebbles are circulated through the core to effect on-line refueling which is compatible with continuous process industries. The fuel cycle adapts Low Enriched Uranium (LEU) to achieve optimal burnup and overall core and fuel performance. A high degree of safety is achieved without reliance on prompt operator actions and startup of standby equipment by the use of passive design features. The design limits peak fuel temperatures during normal operation and during the long-duration loss of forced circulation accidents such that radionuclide retention within the fuel is maintained.	• *Based on review of AREVA PCDR*: ***There is no indication that this item has been specifically addressed in the AREVA PCDR.*** • *Based on review of GA PCDR*: This phenomenon and the need for improved models have been addressed in the General Atomics PCDR. • *Based on review of WEC PCDR*: There is general discussion of limiting of peak fuel temperatures; however ***there is no indication that this item has been specifically addressed in the WEC PCDR.*** • *NGNP R&D Response*: No R&D planned here. It would appear that sensitivity analysis could adequately deal with this.
A-4	*Core Effective Pressure Drop* • Standardized and well-documented correlations for core pressure drop; conformation data may be needed for low-flow cases to better characterize flow distribution and plume formation (for the P-LOFC) and in-core airflow distributions during air ingress accidents. • PBR - parametric analyses using established ranges of different packing fractions to define a performance envelope.	***There is no indication that this item has been specifically addressed.*** However, thermal-hydraulics software (STAR-CD) is addressed in section 19.2.4, and would include calculations of ***pressure in the core***. The pebble bed core portion of the item is not applicable to the PMR.	In section 3.1.2.2, from pages 3-17 through 3-22, references are made to organizations that have performed pertinent core pressure drop analyses, and the computer codes used, including ATHENA. The pebble bed core portion of this item is not applicable to the PMR.	(Section 16.2.1.1.2, p. 16-20) – "The HTTF facility will consist of a number of smaller test sections that will be used for separate effects tests and a main test section that will be used to perform integrated effects tests. The smaller test sections will consist of a scaled down pebble bed and a number of duct-type sections packed with pebbles to represent pebble bed sections with predetermined homogeneous porosities. The main test section will represent an annular pebble bed and it will have the capability to heat the pebble bed (made up of graphite pebbles) and to characterize the heat transfer behavior of such a pebble bed. It is envisaged that the HTTF shall fulfill the needs for tests	• *Based on review of AREVA PCDR:* ***There is no indication that this item has been specifically addressed in the AREVA PCDR.*** • *Based on review of GA PCDR:* This phenomenon and related efforts to date have been addressed in the

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				to characterize the following main phenomena required for simulating heat transfer in a pebble bed: • Pebble bed effective conductivity, which is a combined coefficient representing conductive and radiation heat transfer, required at the pebble bed center region and wall regions respectively. • Convection heat transfer coefficient required at the pebble bed center region and wall regions respectively. • ***Pebble bed pressure drop correlation.*** • Braiding effect correlation, which defines the mixing effect of gas flowing through a pebble bed. • Natural convection heat transfer coefficient."	General Atomics PCDR. • *Based on review of WEC PCDR:* It appears that the PBR will have the required test capability, but ***there is no indication in the WEC PCDR that parametric based on packing fractions have been or will be addressed.*** • *NGNP R&D Response*: Covered in Methods plan.
A-5	*RCCS Performance during LOFC* • Simulate RCCS safety functions in detail, with its predominantly radiant heat transfer coupling to the RPV and other heat transfer mechanisms within the reactor cavity. RCCS functions include maintaining the reactor cavity liner concrete temperature below prescribed limits, preventing the RPV peak temperature from exceeding limits during LOFC events, and minimizing parasitic heat losses during normal operation. • Models may be needed to simulate large pressure pulses in D-LOFC accidents that could damage the RCCS, reducing cooling and/or opening up another release path for air or water ingress to the reactor cavity, and perhaps for FPT out to the environment.	(Table 19-5, p. 290) – ***MANTA*** – "Calculation of main system parameters (temperature, pressure, flow rate) of the HTR plant during all transient (normal, abnormal) when the primary coolant flows in forced convection, in order to define plant operation and control and to provide load data for primary components. Possibility to calculate generalized natural convection." (Code is fully applicable, needs validation) (Table 19-5, p. 290) - ***STAR-CD*** – "Determination of: 1) thermal loadings on the components (vessels, internals, fuel…) during normal or upset conditions, 2) the thermal behavior of the core, 3) the mixing inside the primary system, 4) heat losses and performances of components, 5) flow repartition across the components and 6) ***pressure shock waves***." (Code is fully applicable, needs validation)	See item A-8 for information relating to development of analytical models. (Sec. 3.1.2.2, p. 3-59) – "…The RCCS heat removal rate could be increased during both normal operation and transient if the flow rate could be increased or the local heat-transfer coefficients within the RCCS could be increased. Increasing the RCCS stack height will increase the natural convection flow rate. However, as shown in Figure 3.1-55, there is only a slight reduction in peak vessel temperature (~6°C) for RCCS stack heights over the range 10 to 40 m. ***RCCS design optimization should be assessed in more detail during the next design phase***." (Sec. 3.5, p. 3-99) – Reactor Cavity Cooling System – "The system is required to operate continuously in all modes of plant operation to support normal operation, and, ***if forced cooling is lost, it functions to remove decay heat to ensure investment and safety protection.*** The RCCS consists of a cooling panel which includes cold downcomers and hot risers and is located inside the reactor cavity surrounding the reactor vessel. Connected to the cooling panel are the concentric hot and cold ducts which connect the panel to the inlet/outlet structure." (Sec. 3.5.2, p. 3-100) – ***RCCS Operation*** – "The RCCS is designed to remove ~4 MWt when the primary cooling circuit is either pressurized or depressurized. The RCCS is not required to remove decay heat during normal operation. However, since the system is passive, the system removes some parasitic heat during normal power operation, and removes some decay heat during normal shutdown because of the difference in the reactor vessel temperature and the outside air temperature."	See item A-8 for information on model development. Section 3.2.1.8, p. 16-33 mentions that FOAK design and performance verification is in progress, including features of the RCCS. Section 4.2.4, pp. 4-49 thru 4-52, provides a detailed description of the Reactor Cavity Cooling System (RCCS), including functions, requirements, layout, interfaces, and operation. (Sec. 16.2.1.9, p. 16-33) – *Balance of Plant R&D* – "Verification of the RCCS system's passive mode operation via analysis. The RCCS will not be qualified by testing and for this reason require two independent analysis to be done. The analysis codes selected are RELAP and SPECTRA. The configuration of the RCCS is such that is not equivalent to the normal LWR scenarios and require some research in order to model the system correctly."	• *Based on review of AREVA PCDR:* The needs for modeling and simulation code development described in this item have been recognized in the AREVA PCDR. • *Based on review of GA PCDR:* This phenomenon, related efforts to date, and the need for modeling and simulation codes have been addressed in the General Atomics PCDR. • *Based on review of WEC PCDR:* The need for performance verification has been recognized. ***However, it is impossible to determine if safety functions will be modeled in detail or if large pressure pulses will be simulated in the WEC PCDR.*** • *NGNP R&D Response:* R&D is planned to validate heat transfer in RCCS. Impact of pressure pulse on RCCS performance is open issue. (It could be done, but not currently in the plan.)

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY					
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A-6	*Fuel Performance Models* • Aspects of maximum fuel temperature plus time-at-temperature histories (critical limiting factors) for all fuel regions provide inputs to fuel failure models, to determine source terms and dose-vs.-frequency estimates. • Chemical reactions in air or water ingress accidents, which depend on temperature and should be included in the T/F codes. Especially for fast transients, detailed temperature profiles of the fuel and graphite should be taken into account for thermal stress calculations.	(Sec. 19.2.4.3, p. 292) – ***Thermal Hydraulics/Pneumatics Codes/STAR-CD*** – "Code development and qualification R&D needs are evaluated at a "High" Priority. • Development of ***graphite oxidation model for air ingress transients*** on reactor internal structures. • Qualification of: ○ conduction cooldown models on representative geometry, materials and temperature, ○ turbulence and mixing on representative mock-ups in critical areas (lower and upper reactor plena, hot gas duct, core bypass, IHX collectors) and ○ graphite oxidation models with selected graphite grades in representative operating conditions. Several predecessor tests performed with different graphite grades at CEA and FZJ. NACOK experiments within the European RAPHAEL project (coupling of graphite models with thermo-fluid dynamic behavior) can be applied for STAR-CD qualification." (Sec. 19.2.4.4, p. 292) – ***Fuel Performance Models and Codes/ATLAS*** – "The R&D need for ATLAS development/modification is to improve the diffusion and the coatings corrosion modeling. For code qualification the heat-up experiments of irradiated fuel particles at relevant operating conditions (burnup, temperature, fluence) are required to anchor the developed code…qualification of ATLAS…includes two irradiation and heat up tests. In addition, there is an R&D need to develop the UCO models."	See item A-8 for information relating to development of analytical models. (Sec. 3.1.2.2, p. 3-29) - ***Fuel-Element Modifications***. "The ***thermal performance of the graphite fuel element can be improved*** by reducing the temperature rise from the bulk coolant to the fuel compact centerline. This can be accomplished by reducing the diameters of the coolant holes and fuel compacts. This modified design is referred to as a 12-row block because the number of rows of fuel holes across the flats of the hexagonal block was increased from 10 to 12 (excluding boundary rows). Figure 3.1-25 shows the conventional 10-row block design and the 12-row block design. Parameters for the 10-row and 12-row block designs are given in Table 3.1-10. For the 12-row block design, the minimum web thickness between the fuel and coolant holes was kept the same as the 10-row block for structural/strength considerations. As shown in Figure 3.1-26, the 12-row block design can reduce peak fuel temperatures by 30°C to 40°C, which can allow for reduction of the coolant inlet temperature. The higher flow resistance for the 12-row block is compensated for by the lower flow rate associated with a lower inlet temperature." (Sec. 3,1,2,2, p. 3-34) – ***Fuel Management Strategies*** – "Previous studies have shown that power distributions can be flattened if a concept referred to as fuel placement is used. With this concept, each column contains both new and old fuel in alternating layers at the beginning of an equilibrium cycle. In effect, fuel placement reduces the "age" component of power peaking. As shown in Figure 3.1-27, the fuel placement refueling scheme can reduce the peak column-averaged power factor by about 6%. Because the viscosity of helium increases with temperature, columns with higher peaking factors will tend to receive less flow, which further increases peak fuel temperatures. ***Flattening the power distribution among fuel columns will reduce flow variations and help to reduce peak fuel temperatures***. As part of their work with GA on nuclear hydrogen development, KAERI has been investigating a similar refueling scheme for a 3-batch core (Figure 3.1-28). The KAERI concept uses 9 fuel elements (slightly longer than standard) per column to facilitate a 3-batch shuffling scheme, and adds 6 additional columns (108 fuel columns) to reduce the average power density by 5.6%. KAERI has performed ***3-dimensional physics calculations*** to evaluate this concept, using 12% enriched fissile fuel only and zoning the particle packing fraction to reduce radial peaking factors. Figure 3.1-29 shows the calculated power distribution (at end of cycle when peaking factors were the highest) and Figure 3.1-30 shows the ***flow distribution calculated*** by GA using the POKE code. For these calculations, the bypass flow fraction was assumed to be 0.10 for each column. Figure 3.1-31 shows the ***calculated core temperature distributions*** for the 10-row and 12-row block designs with a coolant outlet temperature of 950°C and the coolant inlet temperature reduced to 490°C. Because of the relatively flat power and flow distributions, the calculated	(Sec. 5.2.4.2, p. 5-17) – *Testing and Qualification (Fuel)* – "Even though test results from the German pebble-bed reactor program are available, and is the basis for the PBMR DPP, expected operational parameters specified for the NGNP was not envisaged during PBMR efforts or the German program. Therefore to ensure the safe application of PBMR DPP based fuel in the NGNP program, the following ***parameters and/or aspects are considered to be important in terms of fuel testing and qualification*** must be considered: • Expected/specified normal operating condition parameters ○ Maximum fuel temperatures ○ Percentage burn-up ○ ***Percentage fission product release at specified temperatures*** • Expected/specified adverse (accident) operating condition parameters ○ Maximum temperatures ○ Percentage burn-up ○ ***Percentage fission product release at specified temperatures*** • Statistical requirements of tests and qualification samples to ensure confident and safe application thereof for design and further operational specification refinements • Pre-designate procedures and facilities for pre- and post-irradiation tests to support qualifications" (Section 16.2.1.5, p. 16-30) – *Fuel Qualification* – "PBMR (Pty) Ltd has embarked on an intensive fuel qualification programmed to ensure that the quality of their manufactured fuel is similar to the German LEU-TRISO fuel. This program is currently underway and will ***qualify the fuel in terms of physical properties, maximum fuel temperatures, percentage burn-up and fission product release for both normal operating and accident conditions***. Statistical requirements of tests and qualification samples will also be investigated to ensure confident and safe application thereof for design and further operational specification refinements. Part of the qualification program will consist of irradiating a number of fuel spheres, containing a statistically significant number of coated particles, to full PBMR irradiation requirements in a material testing reactor." (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* – "***Corrosion/Oxidation Models for CFD (Air Ingress analysis)*** are planned. Implementation of chemical reactions in commercial CFD codes for analysis of air ingress consequences during postulated accident conditions. Will include validation against NACOK experimental results." (Section 16.2.2, p. 16-35) – "DDN NHSS-01-02 specifies data to correspondingly extend the heat up data pertaining	• *Based on review of AREVA PCDR*: The needs for modeling and simulation code development described in this item have generally been recognized in the AREVA PCDR. ***However, it is not possible to determine whether these codes will include:*** ***- time-at-temperature histories for all fuel regions*** ***- chemical reactions in water ingress accidents (AREVA seems to have determined that these are not credible events)*** ***- detailed temperature profiles of fuel and graphite in fast transients*** • *Based on review of GA PCDR*: The needs for modeling and simulation code development described in this item have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The needs for modeling and code development have been recognized in the WEC PCDR. • *NGNP R&D Response*: R&D's Fuel program has this type of model as a key element.

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY

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			peak fuel temperature is below 1250°C, even with the reduced inlet temperature and coolant flow rate. Only about 20% to 30% of the fuel is predicted to be above 1000°C, which helps limit release of Ag-110m and other noble metallic fission products." (Sec. 3.1.2.2, p. 3-60) – "The block-core design provides great flexibility to optimize power distributions using fuel shuffling schemes. Scoping studies show fuel shuffling can significantly reduce power peaking factors and flatten flow distributions. ***More detailed assessments of fuel shuffling should be performed, including coupled physics/thermal analyses and assessing the impact of control-rod movemen***t. An additional 30°C to 40°C margin for peak fuel temperatures can be obtained using a modified, 12-row block design, which could allow for further reduction in the coolant inlet temperature. ***More detailed assessments of this concept include manufacturability, structural/stress analyses, and impacts on fuel costs***." (Sec. 3.1.4.3, p. 3-73) – ***Radionuclide Transport Mechanisms*** – "Radionuclide transport is modeled in the fuel kernel, the particle coatings, fuel-compact matrix, fuel-element graphite, primary coolant circuit, and Reactor Building. [IAEA 1997] provides an excellent overview and an extensive bibliography of radionuclide transport mechanisms. The transport of radionuclides from the location of their birth through the various material regions of the core to their release into the helium coolant is a relatively complicated process. The principal steps and pathways are shown schematically in Figure 3.1-66. Also for certain classes of radionuclides, some steps are eliminated (e.g., noble gases are not diffusively released from intact TRISO particles and are not significantly retarded by the compact matrix or fuel-element graphite). While the actual radionuclide transport phenomena in the core can be very complex, the basic approach for modeling these phenomena is to treat radionuclide transport as a solid-state diffusion problem with various modifications and/or additions to account for the effects of irradiation and heterogeneities in the core materials…" (Sec. 3.1.4.4, p. 3-76) – ***Fuel Quality and Performance Requirements*** – "…The fuel and reactor core are to be designed such that there is at least a 50% probability that the radionuclide releases will be less than the Maximum Expected criteria, and at least a 95% probability that the releases will be less than the Design criteria. The logic for deriving these fuel requirements is illustrated in Figure 3.1-68. Top-level requirements for the NGNP are defined by both the regulators and the users. Lower-level requirements are then systematically derived using a systems-engineering approach. With this approach, the radionuclide control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on radionuclide releases from the VLPC, reactor vessel, and reactor core are successively derived. Fuel failure criteria are in turn derived from the	to accident conditions. R&D for the fuel itself comprises ***irradiation of fuel samples at the higher temperature*** applicable to the NGNP, post irradiation examination and subsequent heat up of some samples to simulate accident conditions, plus corresponding modeling and analysis." (Section 16.2.3.1, p. 16-38) – "The remaining 11 fuel spheres, after the irradiation testing, will be subjected to heating tests simulating DLOFC transient temperatures, nominally 1800°C (rounded to nearest hundred degrees from DDN NHSS-01-02). Following heating tests, all heated fuel spheres will be visually examined again and their fission product inventories measured. One heated fuel sphere will be deconsolidated to provide coated particles for ceramography and fission product distribution measurements."	

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			allowable core release limits. Finally, the required as-manufactured fuel attributes are derived from the in-reactor fuel-failure criteria, with consideration of achievable values based on existing fuel manufacturing experience, thereby providing a logical basis for the fuel quality specifications…The maximum allowable release fractions for 30.2-yr Cs-137 and 249.8-d Ag-110m are included in Table 3.1-16 because these nuclides are expected to be the ***strongest contributors to worker dose***, based on previous assessments of radionuclide plateout distributions and plant-maintenance requirements." (Sec. 5.1.1.3, p. 5-6) - ***Control of Chemical Attack –*** "Chemical attack on fuel particles and on the graphite core structure can result from air or water ingress into the primary system. ***Steps have been taken to prevent ingress of contaminants, and consequences are expected to be acceptable if they occur. The likelihood of water entering the primary system is limited by the absence of high pressure and high energy sources of water in proximity to the primary system***. Under normal operating conditions, all water coolers and heat exchangers operate at lower pressures than the pressure of the primary coolant with which they exchange heat. In the event of a cooler or heat exchanger leak, primary coolant helium would leak out into the secondary cooling water until pressures equilibrate. Then the rate of ingress of sub-cooled water would be small, as water tries to enter the primary system by diffusion and gravity. The amount of water that could enter is limited to the inventory of water in the secondary coolant circuit located above the elevation of the leak. Most of the sub-cooled water that could enter the power conversion vessel would remain at the bottom of the vessel. Very little of it would become entrained in the helium coolant and be transported to the core. Core cooling can still be provided by either the PCS or the SCS, and would limit the potential for chemical attack. If core cooling is not available, the potential of water transport to the core would still be limited. The sub-cooled water will not flash to steam unless the primary coolant helium pressure is below the water saturation pressure, which may occur only when the reactor is operating at a low power level. The reaction rate of water and core graphite will be negligible. The reaction of steam and graphite is slow and endothermic and therefore is not self-sustaining."		
A-7	*Air Ingress Phenomena* • With little or no detail available about the confinement, only generalized studies and experiments would be practical. Bounding analytical studies could be useful in determining positive and negative features of proposed design characteristics. The major features of general interest would be quantification of long-term air in-leakage into the confinement, and the mixing and stratification characteristics of gases in prototypical cavities within the confinement.	(Sec. 11.5.2.3, p. 178) – ***Air Ingress*** – The current state of knowledge of air ingress is provided in this section from an accident analysis point of view, with the following uncertainties identified: • Influence of conduction cooldown uncertainties • Benefit of primary circuit loop isolation strategies • Benefit of SCS actuation • Influence of air on fuel particles performances as well as on the radio-elements trapped in the graphite blocks • Onset of global natural convection and, particularly, the determination of the time when it starts	See item A-6 for information presented on air ingress. Also in section 5.1.1.3, p. 5-6: "The likelihood of a breach of the primary coolant boundary, such that ***air ingress*** becomes a concern, is limited by the high integrity associated with pressure vessels and the limited size of penetrations. In the event of a breach, primary helium coolant would leak out until inside and outside pressures equilibrated. Then, the rate of air ingress would be small, as air tries to enter the breach primarily by natural circulation and diffusion at the same time as helium coolant, which it is displacing, tries to exit through the same hole. Large air ingress rates would require an implausible scenario of two concurrent breaches of an ASME Code Section III vessel in	(Section 16.2.1.1.6, p. 16-26) – "NACOK stands for Natural Convection with Corrosion. The main section of this facility is made up of a vertical channel of 300 mm x 300 mm and 7.5 m tall. The experimental channel is composed of sections representing a bottom reflector, sphere packing (pebble bed) and a top reflector. The experimental set-up was designed to be able to represent different breaks in pipes connecting to the reactor. Breaks can be created that simulate the coaxial duct (reactor outlet pipe), the defueling chute at the bottom of the reactor and the fuelling line at the top of the reflector. By a sectional design, different core heights can also be simulated. All sections of the experimental channel and of the return pipe can be heated to accident-relevant temperatures. At different positions, the	• *Based on review of AREVA PCDR*: The need for greater understanding of the air ingress phenomenon has been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The need for greater understanding of the air ingress phenomenon has

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		• Consequences of CO release • Limitation of air available in the pressure boundary cavity by design and possible operator actions • For large breaks, the assumptions concerning the shutdown of the reactor and the main circulator have to be assessed in order to evaluate if their failure could drastically increase the consequences. If it is the case, these actions should be performed with a high reliability for practically eliminating their occurrence. • Reliability and role of heat removal systems (Sec. 21.1.3, p. 315) – ***Air Ingress Assessment*** – "Air ingress events are a potential issue for all graphite moderated HTRs, due to the concerns associated with graphite oxidation. This issue is similar to the water ingress scenarios in that, while there is a credible technical issue which must be addressed in the course of the reactor design and safety analysis, there is also a large perception issue that is somewhat independent of the technical issues. An objective characterization of air ingress events is recommended in order to put these events in the proper context. The recommended assessment of air ingress events should include scenario definition, controlling phenomenon, potential consequences, and mitigation strategies. The objective is to provide a reasonable framework for the discussion and quantitative evaluation of these events."	order to set up an effective circulation path. However, even in that circumstance, air flow would be restricted by the flow resistance characteristics in the core (e.g., cooling channel high length-to-diameter ratio). Finally, the amount of air is limited by the size of the low leakage below grade containment building. As a result, the overall heat of reaction of air with graphite remains small relative to core decay heat. Also, any air that enters the primary coolant must react with graphite elements and fuel compact matrices before it can reach and chemically react with the embedded refractory-coated fuel particles. "	local gas compositions can be measured." (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* – "***Corrosion/Oxidation Models for CFD (Air Ingress analysis)*** are planned. Implementation of chemical reactions in commercial CFD codes for analysis of air ingress consequences during postulated accident conditions. Will include validation against NACOK experimental results."	been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The need for greater understanding of the air ingress phenomenon has been recognized in the WEC PCDR. • *NGNP R&D Response*: This is part of Methods program.
A-8	*Long-term analysis need* - Comprehensive suite of verified and validated accident simulation codes (core thermal-fluids, core neutronics, whole-plant transient behavior, confinement analysis, and chemical reactions), agreed-upon accident cases for regulatory acceptance, and robust supporting databases that NRC can use for independent confirmatory analysis of candidate plant and confinement designs and options.	(Sec. 19.2.4, p. 290) – ***Computer Codes and Methods Development and Validation –*** Included in this section are descriptions of R&D needs for computer codes addressing reactor system analysis, neutronics, thermal hydraulics/pneumatics, fuel performance, fission product transport, and structural mechanics.	(Sec. 7.2.1.1, p. 7-6) – ***Design Methods Development and Validation*** – "The design methods for analyzing prismatic HTGRs were first developed to support the design and licensing of FSV and the large HTGRs in the 1970s. A brief summary status of the prismatic core design methods is presented below. ***Most of the design methods used for the analysis of the plant systems, structures and components are commercially available design tools,*** such as ANSYS, SINDA/FLUENT, RELAP5, Pro/E, etc. GA's reactor physics codes were originally developed from the basic neutron transport and diffusion theory equations. These methods were adapted to high-temperature, graphite moderated systems to allow calculation of temperature-dependent graphite scattering kernels, and the development of fine group cross sections for graphite systems from point-wise data (e.g., ENDF/B, JEF, and JENDL data sets). These nuclear design methods have been benchmarked against other industry standard codes, such as MCNP, and integral test data from operating HTGRs and critical experiments with generally good agreement. While the ***experimental data used for nuclear code V&V*** are considered reliable, some of the older data and, in particular, the international data ***may not have an adequate QA pedigree to be accepted by the NRC without some confirmatory testing***. The basic approach for performing core thermal/fluid flow analyses for prismatic HTGRs was also established to support the design of FSV and the large HTGRs in the 1970s, and a number of codes were written at GA for that purpose. Although the analytical tools have evolved and the computational capabilities have improved enormously with modern computers, the basic analytical	(Sec. 8.3.3, p. 8-82) – *Design Basis Transient Study* – "An engineering study is recommended to identify and ***analyze transient cases*** that could effect the design requirements of the PCS with respect to ensuring safety of the Nuclear Heat Supply System (NHSS) and Heat Transport System (HTS). Demonstration cases and commercial configurations will be assessed, to ***ensure that the NHSS, HTS, and HPS function within the design basis envelopes through the assumed transient conditions***." (Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a ***full scope PRA that addresses all internal and external hazards, including those associated with the HPF, will be developed***. During the conceptual design of the NGNP, a Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Section 16.2.1.2, p. 16-29) – *Neutronics Design Tools* • Completed: Detailed ***core neutronic design and shutdown system analysis***. Establish the core layout, control system, neutronic behavior in steady state and transients as well as safety and fuel performance analysis. • In Progress: Tools for ***initial criticality, startup and***	• *Based on review of AREVA PCDR*: Long-term analysis needs for computer code development have been recognized in the AREVA PCDR. ***Databases have been addressed by AREVA in terms of candidate alloys and fuel materials.*** • *Based on review of GA PCDR*: Long-term analysis needs for computer code development have been recognized in the General Atomics PCDR. ***Databases required for confirmatory use by NRC have not been addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: Long-term analysis needs for computer code

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
			approach is still valid. Future core thermal/flow analysis for normal operation and accidents will be performed with industry standard codes, such as ANSYS and RELAP5, and various commercial CFD codes as required. Design methods have also been developed to predict the various fuel performance and radionuclide transport phenomena in HTGRs in order to generate source terms for plant design and safety analysis. The accuracy of these design methods have been assessed by comparing code predictions with data from operating reactors and integral test data from various experimental programs. ***In general, the uncertainties in the predicted source terms are large. These design methods are adequate for predicting source terms during NGNP conceptual design, but they will need to be upgraded during preliminary design and validated prior to completion of final design***. A number of core structural analysis codes were developed at GA during the past three decades and used extensively for core design and safety analysis. However, future core structural analysis, including seismic analysis, will be performed with ANSYS and ANSYS/DYNA3D. ***Improved constitutive equations for graphite along with improved material property data will be required.”*** (Sec. 7.2.3.7, p. 7-16) – ***Design Methods Development and Validation*** – “An extensive code development and validation program is presented in the NGNP Design Methods Development and Validation Research and Development Program Plan. The emphasis is heavily upon core nuclear and thermal/fluid flow computational methods. Design methods for predicting coated-particle fuel performance and fission product transport are not addressed. Instead, the Plan states that the AGR Fuel Program will provide the necessary design methods for those applications. While the AGR Plan does include development of improved component models, etc., it does not include scope for developing ***advanced computational tools for full-core performance analysis or for predicting RN transport throughout the plant, and tritium transport is not addressed at all.*** The GA Team’s perspective is that the emphasis of the current NGNP methods development plan is misguided. At least for prismatic MHRs, the currently available computational tools for core nuclear analysis and thermal/fluid flow analysis are largely adequate for NGNP conceptual and preliminary design. The traditional GA design methods for analyzing prismatic HTGRs, which were first developed to support the design and licensing of FSV and the large HTGRs in the 1970s, are still available. However, for nuclear analysis, the traditional codes have been largely supplanted by industry standard codes, such as DIF3D and MCNP; and for thermal, flow, and structural analyses, commercial codes, such as ANSYS, RELAP5, SINDA/FLUENT, and CFX, are already being used routinely by the GA Team. In contrast, the ***design methods for predicting fuel performance and fission product transport are in need of modernization and upgrade*** to support NGNP design and licensing.”	***run in phase; V&V of legacy codes***; engineering and training simulator; and analysis of reactivity transients. • Future: ***Integrated core neutronic/radiation/fuel performance code development***. Future knowledge and expert base as the product of core competency established; important for plant optimization, licensing in other markets, client support, reducing of calculational margins. • Future: ***Verified high temperature cross section libraries and measurements***. Important to establish state of the art high temperature cross section data; needed for new methods and codes; reduction of margins and uncertainty. (Sec. 16.2.1.6, p. 16-30) – *Core Structural Ceramics R&D* – Completed program under the PBMR-Specific Materials Test Reactor Program to conduct supplemental irradiations of NBG-18 to ***verify consistency with the established database for similar graphites***. (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* • Completed: ***Systems CFD software for Thermo-Fluids design*** of VHTR system. Development of M-Tech Industrial's Flownet network Thermo-Hydraulics code into the commercial Flownex software product. Includes component and reactor models to facilitate simulation of thermo-hydraulic steady state and transient behavior of indirect and direct power conversion cycles. R&D included V&V of the code to nuclear industry standards. • In Progress: ***CFD Models for Thermo-Fluids behavior*** of Pebble Bed CFD. Implementation of models for flow and heat transfer in pebble beds for use in detailed CFD reactor models used for reactor design. Includes coding and V&V of these models as User Defined Functions (UDFs) in commercial CFD codes. • Also In Progress: Engineering design ***tools for predicting distortion behavior and failure of irradiated graphite materials*** such as core blocks; and discrete element modeling of interaction between graphite reflector structures and fuel spheres. • Planned: Corrosion/oxidation ***models for air ingress*** consequences during postulated accident conditions. Section 16.2.1.11, p. 16-35) – *Safety Analysis* • Mostly Completed: ***V&V of commercial codes***. V&V of commercial codes as per regulatory requirements to nuclear industry standards; these V&V include comparison with benchmark calculations and/or test results. • In Progress - ***Identification of initiating events*** from FMECA/HAZOP processes; identification of postulated initiating events and establishment of accident scenarios and required analysis and assumptions. • Completed - ***Establishment of adequate***	development, and supporting databases, are either completed, underway, or planned for the future, as discussed in the WEC PCDR. • *NGNP R&D Response*: This piece of work is covered in Methods part of R&D program.

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
				conservative assumptions, analysis methodologies and processes to address all uncertainties to provide a justifiable safety case. • In progress - ***Establishment of a best estimate methodology*** including integrated accident analysis codes, assumptions, V&V, acquisition of plant data and licensing processes. • In Progress – ***Establishment of confinement modeling capability***. Contracting analysis work, confinement modeling assumptions, methodology; selection, purchasing and application of codes. • In Progress – ***Source term analysis***. Establishment of core release rates; activation of dust, contaminants and erosion products; selection and distribution of radionuclide's in reactor and power conversion unit; activity release mechanisms and radio nuclide transport; establishment of activity and release mechanisms of solid and liquid waste systems; (Sec. 16.2.2, p. 16-35) – *Design Data Needs (Nuclear Heat Supply System)* – "Three DDNs have been identified pertaining to the NGNP Fuel. The first of these DDNs (NHSS-01-01) identifies the need for data to ***extend the irradiated fuels qualification database*** from the temperature-burnup envelope of the PBMR Demonstration Power Plant (DPP) to that of the PBMR NGNP. The second DDN (NHSS-01-02) specifies data to correspondingly extend the heat up data pertaining to accident conditions. The third DDN (NHSS-01-03) provides for an extension of the temperature-fluence envelope of the Fuel Graphite to that required by the NGNP. In all three cases, the extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature." (Sec. 16.2.2, p. 16-35) – *Design Data Needs (Nuclear Heat Supply System)* – "Two DDNs (NHSS-02-01 and NHSS-02-02) have been identified to ***extend the irradiated materials qualification database for Reflector Graphite*** from the temperature-fluence envelope of the PBMR DPP to that of the PBMR NGNP. The extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature. The corresponding R&D comprises irradiation of graphite samples at low and high temperatures, plus post-irradiation examination and analysis." (Sec. 16.2.3.1, p. 16-36) – *Fuel Qualification R&D Plan* – "Two supplemental irradiation tests are planned for the PBMR NGNP to ***extend the database supporting the fuel*** for the PBMR DPP. The first of these irradiation tests, which is proposed to start in FY2009 will use pre-production fuel. The second, proposed to start in 2012 will use actual production fuel."	

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				(Sec. 16.2.4, p. 16-42) - *Core Structural Ceramics Reflector Graphite R&D* – "In parallel, the NGNP Program at INL is embarking on a graphite development effort that addresses multiple product forms (including NBG-18) and applications (including the PBMR). The INL program places particular emphasis on the understanding of fundamental graphite characteristics that would, ideally, allow the characterization of new coke and/or graphite sources without the need for an extensive irradiation program. To the extent that the INL program addresses NBG-18 and that manufacturing and QA systems development are generic, there is a potential to accelerate the INL effort and reduce its cost by utilizing applicable results of the PBMR DPP development work that would otherwise be duplicated. From the PBMR perspective, there is a potential to ***expand the database supporting NBG-18*** and, potentially, to reduce the scope of surveillance, testing, inspection and maintenance (STIM) required as a basis for operation of the PBMR DPP. Further potential benefits are access to multiple qualified vendors for follow-on PBMR commercial deployments and easing the burden associated with qualification of new graphite sources. In order to take mutual advantage of PBMR's ongoing program to qualify SGL graphite plus INL's and PBMR's mutual interests to cooperate on graphite qualification with SGL and Graftek, efforts are underway to develop a collaborative program. In the interim, a preliminary scope, cost and schedule for R&D activities addressing the Reflector Graphite DDNs for the PBMR NGNP have been developed." (Sec. 16.3.1, p. 16-50) – *Design Data Needs (Heat Transport Facility)* – "The final DDNs supporting the metallic IHX, HTS-01-18 and HTS-01-19, are established to ***provide the underlying database supporting NGNP-specific code cases for the IHX material and design***, respectively. There is a potential that such code cases would also be applicable to early commercial plants, pending formal implementation within the ASME Code. For ceramic/composite IHXs, six placeholder DDNs (HTS-02-01 through HTS-02-06) have been identified, as both the DDN's and the associated R&D activities will need further development during conceptual design. The first DDN provides for a review of existing technology that is potentially applicable to the development of a ceramic IHX. The anticipated result of the corresponding R&D effort will be the selection of one or more materials and/or heat exchanger technologies for further development. The second DDN specifies the need for a ***materials property database*** for the selected materials. The third DDN addresses the need for design methods, while the fourth identifies requirements for performance verification. The fifth and sixth DDNs address manufacturing technology and the development of codes and standards…. The R&D activities pertaining to DDNs HTS-01-01 through HTS-01-06 provide for the extended qualification of the current reference IHX material, Alloy 617. This extended qualification is required due to the demanding operating conditions that will be seen by the IHX, plus the small grain	

TABLE 2A – ACCIDENTS AND THERMAL FLUIDS – SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
				size that is expected to be required for compact heat exchangers as they are characterized by very thin heat transfer surface cross-sections. As described in DDN HTS-01-01 (Section 6.3.1), an initial effort is required to further develop the specification for Alloy 617 and to establish a reference for characterization. ***Included in this effort, is a review of the current database for this material, consultation with material vendors and consideration of a controlled specification variant, Alloy 617CCA, that potentially decreases the range of uncertainties with respect to properties.*** The conclusion of this effort will be procurement of materials to be used for subsequent testing.”	

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
B-1	Time-dependence and spatial distribution of decay heat as a major factor in determining maximum fuel temperature during a D-LOFC.	(Sec. 11.5..2.1, p. 175) – ***Loss of Primary Forced Convection*** – Conduction Cooldown Events – This section contains AREVA's ***bounding D-LOFC*** (and limiting design basis event)***,*** referred to as a Depressurized Conduction Cooldown (DCC). The section describes the plant engineered safety features response to the event, indicating that the ***temperature increase is slow and peak temperatures for fuel and core support structures are limited***. (Sec. 19.2.4.2, p. 292) – ***Neutronics Codes/MONTEBURNS*** – "The R&D needs for MONTEBURNS are "High Priority." 1. Experimental results of fuel irradiation experiments (compacts or pebbles) at representative burnups, temperatures and fluences. 2. Experimental results of ***decay heat*** at short term (<100 hours) for representative fuel composition and burnup.	(Sec. 5.1.3., p. 5-7) – ***Accident/Transient Analysis*** - "The bounding design basis events (DBEs) for the NGNP will be a loss of flow leading to a high pressure conduction cooldown (HPCC) and loss of coolant leading to a ***low pressure conduction cooldown*** (LPCC). The HPCC event is typically initiated by trip of the PCS. The RPS automatically initiates a reactor trip on low flow or turbomachine trip. ***Because the system remains at high pressure, the decay heat is more uniformly distributed within the core and vessel than during a LPCC event***. The LPCC event is typically initiated by a small primary coolant leak, causing the system to depressurize to atmospheric pressure. The RPS automatically initiates a reactor trip on low coolant pressure. For both events, the SCS fails to start and ***decay heat is removed by thermal radiation and natural convection from the reactor vessel to the RCCS***). ***These events have been analyzed in detail*** for a MHR operating with a reactor outlet coolant temperature of 950°C, and the results show that peak fuel temperatures remain below the design goal of 1600°C, and the temperatures for the vessel and other safety-related SSCs also remain below acceptable limits. For the LPCC event, the peak fuel temperature is 1525°C and occurs about 60 hours following initiation of the event. For the HPCC event, the peak fuel temperature is 1349°C and occurs about 50 hours following initiation of the event. The calculated peak vessel temperatures for the HPCC and LPCC events are approximately 478°C and 517°C, respectively. For both events, the peak vessel temperatures occurred about 72 h following initiation of the event."	See item A-8 for information associated with model development. (Sec. 2.1.1, p. 2-13) – ***Commercial Plant Summary Description*** - The reference PBMR H2 PHP fuel is TRISO-coated UO_2 fuel particles embedded in the spherical pebble fuel elements. The pebbles are circulated through the core to effect on-line refueling which is compatible with continuous process industries. The fuel cycle adapts Low Enriched Uranium (LEU) to achieve optimal burnup and overall core and fuel performance. A high degree of safety is achieved without reliance on prompt operator actions and startup of standby equipment by the use of passive design features. ***The design limits peak fuel temperatures during normal operation and during the long-duration loss of forced circulation accidents*** such that radionuclide retention within the fuel is maintained. (Sec. 5.2.4.2.4, p. 5-21) – *PBMR Fuel Irradiation Program* – "Following from the above conclusions, it is proposed to extend the existing German database by performing additional irradiation and heating tests using PBMR fuel spheres from a qualified fuel manufacturing line. The proposed PBMR irradiation program will consist of irradiating a number of fuel spheres, containing a statistically significant number of coated particles, to full PBMR irradiation requirements in a material testing reactor. Irradiation will be performed in such a way that temperatures encountered during normal cycling of fuel spheres through the PBMR core are simulated until PBMR irradiation targets are reached. A number of simulated PLOFC temperature transients will be superimposed on the normal temperature cycles for a sufficient number of fuel spheres to ensure statistic validity. Following the completion of the irradiation test, a sufficient number of ***irradiated fuel spheres will be subjected to heating tests using temperature cycles simulating the expected DLOFC temperature*** transient for the PBMR. The PBMR start-up core consists of a mixture of start-up fuel spheres and graphite spheres. In addition to irradiation test on fuel spheres, irradiation tests will also be performed on matrix graphite spheres manufactured on the same manufacturing line as fuel spheres. The reason for this is that some matrix graphite properties cannot be measured on fuel spheres containing coated particles." (Sec. 5.2.4.2.5, p. 5-22) – *Normal Operation and Accident Conditions Testing and Qualification* – "Generally, during normal and accident conditions operation of TRISO particle containing fuel spheres, three variables must be taken into consideration. These are in order of importance; temperature, burn-up, and fast neutron dose. Deviations of these variables from acceptable, specified levels are an indication of process parameters where possible fuel failure and pending upset process conditions (accident conditions which may include the release of fission products) may arise, and must be considered during testing and	• *Based on review of AREVA PCDR*: The needs for modeling and simulation code development described in this item have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for modeling and simulation code development described in this item have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The needs for fuel testing and modeling to determine fuel temperature in a D-LOFC are recognized in the WEC PCDR. • *NGNP R&D Response*: All the codes will have spatial and time dependence in them. No experimental plans for obtaining such data.

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				qualification efforts." (Sec. 5.3, pp. 5-39 thru 5-45) – Fuel Design Development Needs (DDNs) – DDNs and the efforts planned to fulfill them are described for the PBMR fuel, including: • Fuel Irradiation Tests for Normal Operational Conditions (Sec. 5.3.1.3.1) • Fuel Heating Tests for Accident Conditions (Sec. 5.3.1.3.2) • Fuel Graphite Irradiation Tests (Sec. 5.3.1.3.3) (Section 16.2.2, p. 16-34) - Planned: Corrosion/Oxidation Models for CFD (Air Ingress analysis). Implementation of chemical reactions in commercial CFD codes for analysis of air ingress consequences ***during postulated accident conditions***. Will include validation against NACOK experimental results. (Section 16.2.2, p. 16-34) - In Progress: Identification of licensing basis events; and required analysis; and V&V. Identification of initiating events from FMECA/HAZOP processes; identification of postulated initiating events and ***establishment of accident scenarios and required analysis and assumptions.***	
B-2	Control and shutdown rod worth and reserve shutdown worth as required for hot and cold shutdown.	(Sec. 19.2.4.2, p. 291) – ***Neutronics Codes/MCNP and NEPHTYS*** – "The R&D needs for both MCNP and NEPHTYS are of "High Priority." 1. The approach for qualification consists of comparing results against Monte-Carlo reference calculations and benchmarking against the few available experimental data (FSV, HTTR). Thus new dedicated critical experiments, with an asymptotic spectrum representative of the expected prismatic fuel assembly and core, with full access to pin-by-pin power distributions, and ***control rod and burnable poisons worths*** are needed. 2. Experimental data of neutronic characteristics (spectrum, fission and capture rates) at the interface between a prismatic fuel assembly and a graphite reflector assembly. Data from FSV and HTTR first criticality testing can be applicable to MCNP and NEPHTYS code qualification."	(Sec. 3.1.2.1, p. 3-12) – "The active core consists of 102 fuel columns in three annular rings with 10 fuel blocks per fuel column, for a total of 1020 fuel blocks in the active core. As shown in Figure 3.1-7, the core is designed with 120-degree symmetry and the ***control rods*** are also operated symmetrically. The outer reflector contains 36 control rods, arranged as 12 groups with 3 rods per group. There are 4 control-rod groups in the active core, again with 3 rods per group. The core also contains 18 channels for insertion of ***Reserve Shutdown Control*** (RSC) material (in the form of boronated pellets) in the event the control rods become inoperable. During operation, control rods in the active core are completely withdrawn, and only the control rods in the outer reflector are used for control. This control method precludes damage to the in-core control rods during loss-of-coolant accidents. A control rod design using a carbon-carbon composite for the cladding material is being evaluated that would allow the in-core rods (or control rods located in the inner reflector) to be used during normal operation, which will provide greater flexibility for flattening the radial power distribution and provide some additional margin for maintaining fuel temperatures and fuel performance within acceptable limits." (Sec. 3.10.2.2.2, p. 3-213) – ***Reactor Power and Temperature Control*** – "This previously developed control scheme is used for NGNP steady or transitory Reactor Exit temperature control. The control uses an outer temperature control loop, feeding an inner reactor flux control loop, and connected to a Control Rod Drive System as depicted in Figure 3.10-4. ***Control rod withdrawal/insertion sequencing*** is based on selective "one-at-a-time" rod	(Sec. 4.2.1.5, p. 4-29) – *Reactivity Control System (RCS)* – "The ***Reactivity Control System (RCS) is used to control the reactivity in the fuel core, to quickly shut the reactor down and to keep it in a shutdown mode***....The RCS consists of 24 identical control rods. The control rods are grouped into 12 control rods and another group of 12 shutdown rods. The control system moves each group alternatively to have the rods inserted to an equal depth into the side reflector. The only difference between shutdown rods and normal operation control rods is the length of the chain, with the control rods only traveling in the top part of the reflector, and the shutdown rods capable of traveling in the bottom part of the reflector. Each rod consists of six segments containing absorber material in the form of sintered B_4C rings between two coaxial cladding tubes. Gaps between the cladding tubes and B_4C rings prevent constraint forces from arising due to radiation-induced swelling of the B_4C. Pressure equalizing openings expose the B_4C to the coolant gas to avoid any pressure build-up. The RCS consists of the following major subsystems/components: • RCS Control Rod Drive Mechanism (CRDM) consisting of the chain drive, chain container and scram shock absorber which functions as the primary shock absorber. The purpose of the CRDM is to translate rotational movement into linear movement. • Rod and chain, with the rods absorbing neutrons and the chain connects the chain drive to the control rod. • RCS secondary shock absorber, which prevent damage to the control rod and the core structures ceramics following a chain failure.	• *Based on review of AREVA PCDR*: The needs for understanding control and reserve shutdown capability, as described in this item, is recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for understanding control and reserve shutdown capability, as described in this item, is recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The need for determining/validating rod worths has been recognized in the WEC PCDR, and work is either completed or is in progress. • *NGNP R&D Response*: No experimental work

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
			withdrawal or insertion from predetermined control rod is applied through inclusion of total reactor mass flow rate to adjust for reactor core thermal "time-constant" variation over a wide range of reactor flow rate. This is not shown in Figure 3.10-4, but it is based on the sum of the two primary flow measurements which are shown. This scheme allows consistent ***"tight" adjustment*** of reactor power through the operating range in spite of the large core thermal effects which are characteristic of HTGR reactors." (Sec. 3.1.3, p. 3-64) – "The ***reserve shutdown control material*** is of the same composition as that for the control rods, except the B4C granules and graphite matrix are formed into cylindrical pellets with rounded ends and a diameter of 14 mm. The B4C granules are coated with dense PyC to prevent oxidation during off-normal events. The pellets are stored in hoppers located above the reactor core in both the both the inner and outer neutron control assemblies." (Sec. 5.1.1.3, p. 5-5) – ***Control of Heat Generation*** – "***Control rods*** drop by gravity into the core upon loss of electrical power. An automatic positive control action can cause the rods to drop, or the event itself may cut the power supply. It is an advantage that the rods need not be powered in. In addition, the NGNP has a ***redundant and diverse system*** to drop boronated graphite pellets by gravity into designated fuel element channels for reactivity control equivalent to rod insertion. Initiation of the latter system requires a positive control signal and an active response. If both the control rod and the reserve shutdown systems fail, (i.e., if neither control rods nor reserve shutdown material are inserted into the core), the temperature coefficient of reactivity will shut down the reactor from any power level following loss of cooling. As an example, given that no additional positive reactivity is inserted, core power will be reduced to shutdown levels by negative temperature coefficient alone, such that the RCCS alone can safely cool the core for more than 30 h after the initial shutdown. A test conducted at the AVR in Germany supports analysis which shows that following this initial shutdown, a gradual core temperature increase and negative reactivity addition will occur, with the core stabilizing at a low power level at which the heat generation rate matches the core cooling capability of the passive heat sinks. This is a stable, safe condition that can be maintained until corrective actions are taken to insert the control rods or drop reserve shutdown control material into the core to affect a full shutdown and to allow the reactor to be taken to cold shutdown condition."	• RCS drive motor, which keep the control rod in position and move the control rod up and down. During a power failure the control rod will fall down the control rod channel under gravitational force. • RCS control rod guide tubes connect the CRDM housing to the Core Structure and serve as a guide for the Control Rod. During the anticipated operating modes of the Reactor Unit System, the RCS is required to raise and lower the control rods and hold them steady in any position over their entire range of travel. The control rod and shutdown rod positioning is commanded by the Operational Control System (OCS). Control- or shutdown rod insertion (scram) action is initiated by the Reactor Protection System (RPS), which overrides the OCS. During full power operation, both banks of control rods (24) are inserted into the upper third of the core. ***During hot shutdown both banks are moved simultaneously down to the middle third of the reactor. Cold shutdown will be achieved if bank 1 remains in position, while bank 2 continues to the fully inserted position.*** Details of possible variations will be calculated during the conceptual design phase. When power is cut to the drive motors (scram activation), the rods are inserted by gravity. During this event, the drop velocity of the RCS units is limited to a pre-determined value." (Sec. 4.2.1.6, p. 4-32) – *Reserve Shutdown System (RSS)* – "***The purpose of the Reserve Shutdown System (RSS) is to maintain the reactor in a subcritical state during shutdown***….. The RSS consists of eight units that can insert Small Absorber Spheres into the eight borings of the central reflector. Small Absorber Spheres are typically inserted to shut the reactor down to 'cold' conditions for maintenance operations. When inserted, the RSS keeps the reactor subcritical to an average core temperature of 100ºC or less. The RSS neutronic function is thus to act as an absorber in the lower part of the reactor that is out of reach of the solid control rods. The presence of the small absorber spheres creates a negative reactivity which ensures subcriticality….. When shutdown is required, the valves of the small absorber spheres storage units are opened to allow the small absorber spheres to flow under gravity into the central reflector borings. The small absorber spheres are removed from the channels (all eight channels are removed at the same time) and transported back via the sphere return pipe to the feeder bin by means of a gas transport system. The feeder bin distributes the small absorber spheres to the eight small absorber spheres storage containers. Gas flow from the FHSS blower fluidizes and moves the small absorber spheres. During small absorber spheres transport, the FHSS does not transport fuel. The FHSS is isolated from the reactor and the RSS switches to small absorber spheres transport mode. The small absorber spheres units, interfacing with the RPV and CB, operate under the same pressure and temperature as the reactor, and therefore small absorber spheres can only be transported at gas temperatures amenable to the valves and other components wetted by	planned. Better analytic tools are under development to be able to calculate shutdown and control rod worths. There are adequate benchmarks available for both design options.

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
				gas flow." (Section 16.1.1.4.2, p. 24) The ASTRA critical facility represents a cylindrical side reflector consisting of graphite blocks with an octagon shaped core in the centre and a solid cylindrical centre column. The core is filled with fuel spheres and absorber spheres. Control rods, shutdown rods and a single regulating rod are situated in the first set of blocks closest to the core in the side reflector. ***This allows different critical configurations to check single control rod reactivity worth's or different combinations to look at interference (or shadowing) effects and permits better V&V of control rod models and methods used in the analysis tools***. (Section 16.2.1.2, p.16-29) Completed: Detailed core neutronic design and shutdown system analysis. Establish the core layout, ***control system, neutronic behavior in steady state and transients as well as safety and fuel performance analysis.*** (Section 16.2.1.2, p.16-29) In progress: Establishment of analysis capability for ***reactivity transients***. Establish the know-how of ***performing control rod withdrawal,*** xenon oscillations, SSE, thermal transients, etc. (Section 16.2.1.7, p. 16-32) In progress: Control of Reactor. Understanding of the correct methods to control the flux and Reactor Outlet Temperature, as well as ***estimating the reactivity and the shutdown margin***.	
B-3	Sudden positive reactivity insertion due to pebble core compaction (packing fraction) due to earthquake.	PBR phenomena; not applicable to PMR core.	PBR phenomenon; not applicable to PMR core.	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). (Sec. 16.2.1.1.2.2, p. 16-20) – *Heat Transfer Test Facility* – "The HTTF facility will consist of a number of smaller test sections that will be used for separate effects tests and a main test section that will be used to perform integrated effects tests. The smaller test sections will consist of a scaled down pebble bed and a number of duct-type sections packed with pebbles to represent pebble bed sections with predetermined homogeneous porosities. The main test section will represent an annular pebble bed and it will have the capability to heat the pebble bed (made up of graphite pebbles) and to characterize the heat transfer behavior of such a pebble bed. It is envisaged that the HTTF shall fulfill the needs for tests to characterize the following main phenomena required for simulating heat transfer in a pebble bed: • Pebble bed effective conductivity, which is a combined coefficient representing conductive and radiation heat transfer, required at the pebble bed centre region and wall regions respectively. • Convection heat transfer coefficient required at the pebble bed centre region and wall regions	• *NA to AREVA and GA PCDRs* This is a PBR phenomenon and is not applicable to the PMR core. It does not apply to either AREVA or General Atomics designs. • *Based on review of WEC PCDR*: It appears that WEC has the test facilities to simulate a condition of increased packing density in the PBMR core; however ***there is no indication that this item has been specifically addressed in the WEC PCDR.*** • *NGNP R&D Response*: Methods program has looked at this and has tools to do this type of problem.

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY

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				respectively. • Pebble bed pressure drop correlation. • Braiding effect correlation, which defines the mixing effect of gas flowing through a pebble bed. • Natural convection heat transfer coefficient." (Sec. 16.2.1.1.6, p. 16-25) – "NACOK stands for Natural Convection with Corrosion. The main section of this facility is made up of a vertical channel of 300 mm x 300 mm and 7.5 m tall. The experimental channel is composed of sections representing a bottom reflector, sphere packing (pebble bed) and a top reflector. The experimental set-up was designed to be able to represent different breaks in pipes connecting to the reactor. Breaks can be created that simulate the coaxial duct (reactor outlet pipe), the defueling chute at the bottom of the reactor and the fuelling line at the top of the reflector. By a sectional design, different core heights can also be simulated. All sections of the experimental channel and of the return pipe can be heated to accident-relevant temperatures. At different positions, the local gas compositions can be measured." (Sec. 16.2.1.1.7, p. 16-27) – "The SANA test facility consists of a heated pebble bed inside a furnace to simulate the thermal conditions of an HTGR-core. Different heater configurations are possible but Figure 16.2-11 shows a schematic of the test facility with a single central heating element. The diameter of the pebble bed is 1.5 m and the height is 1.0 m. The overall height of the facility is 3.2 m and the maximum heating capacity of the single central heating element is 35 kW. The top and bottom of the facility are well-insulated while the outside of the furnace is open to atmosphere. More than 50 steady-state as well as some transient tests were carried out in the facility. In these experiments all the main parameters of a pebble bed were varied, such as pebble material, pebble diameter, gas type, heating power and heating geometry."	
B-4	For tests at both PMRs and PBRs, consideration should be given (at least in the first core) to use of high-temperature in-core neutron detectors that can provide maps of axial and azimuthal power distributions and core-inner-to-outer-radius power tilts; these detectors would likely be located only in the inner and outer reflectors rather than in the core, due to temperature and connection limitations. • PMR concern - Whether improper axial-loading of fuel blocks during refueling can lead to an undetected power distribution anomaly and result in excessive operating fuel temperatures. • PBR concern - Radial and azimuthal power distributions in the mixed-fuel pebble bed are not well known, and there are indications from melt-wire tests	(Sec. 6.1.3.2, P. 64) – ***Neutron Control Equipment*** – "Neutron control is effected using equipment for positioning the control rods and nuclear instrumentation. The primary components include Neutron Control Assemblies (NCA) and nuclear instrumentation. There are 24 NCAs, of which 18 are used for the 36 operating control rods in the outer reflector and 6 are used for the 12 startup control rods in the inner core. Each NCA contains 2 independent chain, wheel, gear and motor type control rod drives – one per control rod. A friction clutch between each motor and the drive mechanism is included to prevent overload. The nuclear instrumentation consists of ex-vessel ***neutron detectors***, source range detectors, and ***in-core flux mapping units***. During normal operation, the neutron flux levels are monitored by the ex-vessel neutron detectors, whose range overlaps with that of the source-range detectors. During startup and shutdown, the neutron flux levels are monitored using the source-range detectors. The in-core flux mapping units are used to verify ***axial flux profiles*** and confirm power stability."	See item A-1 for information on in-core and ex-core monitoring instrumentation. See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3.1.3, p. 3-61) – Neutron Control System – "The neutron control system design is the same as that for the GT-MHR. ***The system components consist of inner and outer neutron control assemblies, neutron source, source range detector assemblies, ex-vessel neutron detector assemblies, and the in-core flux mapping system***…During normal operation, the neutron flux levels are monitored by 6 symmetrically spaced ex-vessel fission chamber thermal neutron detectors. The signals from these detectors interface with the automatic control and protection systems to operate the control rod drives or the reserve shutdown control equipment. Three fission chamber source-range detectors are used to monitor neutron flux during	See item A-1 for a discussion of instrumentation provided for the PBMR Nuclear Heat Supply System.	• *Based on review of AREVA PCDR*: The need for core monitoring instrumentation has been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The need for core monitoring instrumentation has been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: There is discussion of instrumentation and

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY

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	conducted in the AVR (Germany) suggesting that pebbles near the walls of the reflector experienced unexpectedly high fuel temperatures.	(Sec. 19.2.3, p. 289) – ***Instrumentation*** – "NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy. For ***neutron flux detectors*** some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired." The PBR portion of this item is not applicable to the PMR core.	startup and shutdown. These detectors are symmetrically spaced in reentrant penetrations located in the bottom head of the reactor vessel. These penetrations extend into vertical channels in the reflector elements near the bottom of the core. ***The in-core flux mapping system consists of movable detectors in the central column of the inner reflector and in the outer permanent reflectors***. The system enters from a housing located above the reactor vessel and vertically traverses down through the core to the bottom reflectors. The system contains two independent fission chambers and a single thermocouple."		monitoring, however ***there is no indication that in-core instrumentation has been specifically addressed in the WEC PCDR.*** • *NGNP R&D Response*: R&D is needed. No funding yet except a small amount allocated to university grant program.
B-5	In both the PMR and PBR, control rod misalignments in the outer reflector during operation would result in azimuthal power tilting that could cause xenon-135-induced oscillations when the misalignment is corrected; however, this needs to be verified by analysis and confirmed by test.	***There is no indication that this item has been specifically addressed.***	See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3.1.3, p. 3-61) – "The control rod guide tubes extend from the gamma shielding downward through the top head of the reactor vessel and upper plenum shroud to the upper core restraint elements. The ***guide tubes provide a clear passage for the control rods*** as they are inserted into and withdrawn from the core. All neutron control assemblies are equipped with two independent control rod drive units. The control rod drive equipment is located in the upper part of the neutron control assembly. The equipment consists of a DC torque motor, a 60:1 speed reducer, and a cable storage drum, all of which are mounted on a metal frame. The control rod is ***lowered and raised with a flexible high-nickel alloy cable***. Figure 3.1-58 shows the control rod design. The neutron absorber material consists of B4C granules uniformly dispersed in a graphite matrix and formed into annular compacts. The boron is enriched to 90 weight percent B-10 and the compacts contain 40 weight percent B4C. The compacts have an inner diameter of 52.8 mm, an outer diameter of 82.6 mm, and are enclosed in Incoloy 800H canisters for structural support. Alternatively, carbon-fiber reinforced carbon (C-C) composite canisters may be used for structural support. The control rod consists of a string of 18 canisters with sufficient ***mechanical flexibility to accommodate any postulated offset*** between elements, even during a seismic event."	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). (Section 16.2.1.2, p.16-29) - In progress: Establishment of analysis capability for reactivity transients. Establish the know-how of performing control rod withdrawal, xenon oscillations, SSE, thermal transients, etc.	• *Based on review of AREVA PCDR*: ***There is no indication that this item has been specifically addressed in the AREVA PCDR.*** • *Based on review of GA PCDR*: The issue of control rod misalignment has been addressed in the General Atomics PCDR with descriptions of the design features that maintain control rod alignment. • *Based on review of WEC PCDR*: The general need to understand xenon oscillations was identified. ***However, there were no specifics in the WEC PCDR that mentioned outer reflector control rod misalignments leading to azimuthal power tilting and possible xenon oscillations.*** • *NGNP R&D Response*:

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY

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					No plans in R&D. Appears to be a design issue.
B-6	Replacing helium with a hydrogen-bearing compound such as in a steam/water ingress event may produce a pronounced positive reactivity. Steam/water ingress tends to have a positive reactivity effect due to increased neutron moderation and reduced neutron leakage.	(Sec. 11.5.2.4, p.179) – ***Water Ingress*** – Water ingress is treated from an accident analysis perspective in sec. 11.5.2.4, including the identification of positive reactivity insertion as an unresolved issue. Other unresolved issues are identified, including: • Benefit of start up of the SCS • Benefit of primary circuit loop isolation strategies • Impact of water on graphite structure and its heat transfer properties • Influence of water on fuel particles performances as well on the radio-elements trapped in the graphite blocks • Consequences of CO and H2 release • Limitation of water available to enter the pressure boundary • Impact of possible actuation of safety valve (primary and secondary) on potential radiological releases (Sec. 21.1.2, p. 315) – "Any decision to adopt a steam cycle HTR configuration increases the significance of ***water ingress events*** due to the potential for steam generator leaks. This issue was successfully managed in previous operating HTRs. However, the possibility for water ingress continues to be perceived as a significant issue within the broader nuclear community. There are various reasons for this including misunderstanding of the source of water ingress in the Fort St. Vrain reactor, failure to appreciate the differences in steam generator technology between HTRs and LWRs, and unfamiliarity with the consequences and mitigation of water ingress in HTRs. Steam line breaks within the reactor building also must be considered for steam cycle concepts. Steam line breaks must be evaluated for building pressurization issues and for any impact on building venting and filter systems, if a vented confinement concept is used for the NGNP. A ***white paper should be developed addressing water ingress*** and steam line break events and their likely impact on NHS design. The intent is not necessarily to provide detailed analyses of such events. Rather the focus should be on describing the issues and concerns associated with each type of event, the potential significance of these events on operation, safety, and licensing, mitigation of these events including likely design features which might be utilized, and ***likely R&D*** that might be necessary to resolve any open issues."	See item A-6 for description of design features and controls associated with prevention of chemical attack by water ingress. See item B-7 for information relating to reactivity control. See item C-6 for a description of the test program. (Sec. 3.4.1, p. 3-97) – "During normal operation of the reactor system, the SCS operates in a standby mode. During this mode, a small amount of cold leg helium leaks (back flows) through the closed shutdown valve and flows opposite the normal flow direction through the SCS circulator and over the SCS heat exchanger tubes. In this mode the circulator is not operating, but the SCS cooling water system supplies a small amount of water flow to the heat exchanger. This water flow prevents thermal shock when the SCS switches to an active cooling mode, but also results in a parasitic heat loss of up to 1.3 MWt during normal operation. Therefore, the standby-mode water flow must be set as low as possible without resulting in one or both of the following adverse conditions: (a) boiling and/or (b) static instability due to the large hydrostatic head in the heat exchanger. ***During standby mode, the primary coolant helium pressure is higher than the SCS water pressure, in order to prevent water ingress into the reactor system during normal operation***. The SCS is manually switched from standby mode to an active cooling mode at the discretion of an operator."	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). See item F-4 for information relating to steam generator design and design development. This addresses water ingress from the standpoint of the efforts that are being dedicated to ensure steam generator reliability and separation from the primary system.	• *Based on review of AREVA PCDR*: The need for greater understanding of the water ingress phenomenon has been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The need for greater understanding of the water ingress phenomenon has been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: There is discussion of reactivity control, and also of steam generator design to prevent water ingress; however ***there is no indication that this item has been specifically addressed in the WEC PCDR.*** • *NGNP R&D Response*: Current R&D does not address water ingress given very low probability of occurrence. Would change if steam generator is part of primary system.
B-7	With a higher atomic mass moderator such as carbon, the mean thermal energy of neutrons will be higher than that for hydrogen bound with oxygen in water; that is, graphite will tend to produce a "harder" thermal-neutron energy spectrum than would water-moderated systems. Thus, the moderator temperature-dependent reactivity coefficient (MTC) in both PMR and PBR depends upon the change of thermal-neutron energy spectrum with	(Sec. 4.3.1, p. 43) – "The ***reactivity temperature coefficient*** shall be sufficiently negative to shutdown the nuclear chain reaction before an unacceptable fuel temperature is reached, and maintain the core in a safe state for a time offering the certainty to reliably introduce absorber elements." (Sec. 6.1.1.3, P. 49) – ***Core Reactivity Control*** – "The ***core reactivity*** is controlled by the ***core negative temperature***	(Sec. 3.1.1.1, p. 3-1) – "The fuel for the GT-MHR consists of microspheres of uranium oxycarbide that are coated with multiple layers of pyrolytic carbon (pyrocarbon) and silicon carbide. The GT-MHR core is designed to use a blend of two different particle types; a fissile particle that is enriched to 19.8% U-235 and fertile particle with natural uranium (NU, enrichment of 0.7% U-235). ***The fissile/fertile loading ratio is varied with location in the core, in order to optimize reactivity control***, minimize power peaking, and maximize fuel cycle length. The GT-MHR coated particle design	See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). (Section 11.2.2, p.11-12) - The NHSS also has a negative temperature coefficient, which results in the reactivity and consequently the neutronic power to counteract temperature changes. The NHSS is therefore to a large extent self-regulating and minimum control interaction is required to	• *Based on review of AREVA PCDR*: The AREVA PCDR has addressed AREVA's design strategies for reactivity control and neutron control, as features of the design. ***AREVA has not specifically addressed***

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	temperature, with possibly large effects on reactivity. Concerns are for effects on core transient behavior and passive safety shutdown characteristics.	***coefficient*** and control rods, and possibly by lumped burnable poison located in the fuel assemblies. It is also complemented by the Reactor Reserve Shutdown System (RRSS). This system is used to shutdown the reactor and maintain it a sub-critical state if the rod system fails to trip the reactor." (Sec. 6.1.1.4, p. 49) – ***Reactivity Balance*** – "The core reactivity balance is presented in Table 6-2 for Beginning of Cycle (BOC) and End of Cycle (EOC) and includes the following items: • Reactivity due to equilibrium xenon. • ***Temperature reactivity effect*** (Doppler, moderator, and reflector) - their sum represents the cold to hot transition. • Reactivity due to burn-up, which is the excess reactivity required to achieve cycle lifetime. • Control rod worths. The sum of the xenon worth, the total temperature reactivity effect, and the burn-up reactivity yields a BOC required control rod worth of 19.2 %Δk/k. The total available worth is 24.9 %Δk/k, which is sufficient to cover stuck rod worth and shutdown margin." (Sec. 6.1.3, p. 63) – ***Neutron Control*** – "The core reactivity is primarily controlled by the core negative temperature coefficient and control rods. In addition, the placement of fuel blocks having known reactivity based on burn-up (for irradiated fuel), initial enrichment levels and the possible inclusion of burnable poisons provide for further control of reactivity. Core reactivity control is complemented by the Reactor Reserve Shutdown System (RRSS) that will safely shutdown the reactor and maintain a subcritical state in the event that the control rods fail to operate during accident conditions." (Sec. 6.1.3.4, p. 65) – ***Neutron Control During Accident Conditions*** – "The detection of reactivity insertion events leads to reactor shutdown by automatic insertion of the control rods by the Reactor Protection System (RPS). In cases where the events are coupled with a loss of electrical power, the controls rods will drop into the core by gravity. The RRSS, can be manually actuated to achieve a diverse method of reactor shutdown, should control rod insertion not be accomplished. The two neutron absorbing systems are designed so that the insertion of either one of these systems ensures and maintains a subcritical state in all conditions. This includes, in particular, the reactivity due to the core cooling down to the coldest shutdown state combined with the xenon effect and the ***reactivity insertion*** due to the initiating event."	parameters are given in Table 3.1-2. The fissile and fertile particle designs are somewhat different, with the fertile particle having a larger kernel and a thinner buffer coating layer. Preliminary core physics calculations performed by INL for an NGNP prismatic block MHR suggest that the reactor may be able to utilize a single fuel particle design, with the fuel particles potentially having different U-235 enrichments. However, ***more detailed calculations are needed*** to confirm that a single fuel particle design provides adequate core design flexibility." (Sec. 5.1.1.1, p. 5-2) – "The NGNP reactor core is designed to have a ***negative temperature coefficient of reactivity***. This characteristic means that as the reactor gets hotter, the change in temperature alone will reduce reactor power. ***For all credible reactivity addition events, the negative temperature coefficient can limit reactor power***." (Sec. 5.1.1.3, p. 5-5) – ***Control of Heat Generation*** – "Control rods drop by gravity into the core upon loss of electrical power. An automatic positive control action can cause the rods to drop, or the event itself may cut the power supply. It is an advantage that the rods need not be powered in. In addition, the NGNP has a redundant and diverse system to drop boronated graphite pellets by gravity into designated fuel element channels for ***reactivity control*** equivalent to rod insertion. Initiation of the latter system requires a positive control signal and an active response. If both the control rod and the reserve shutdown systems fail, (i.e., if neither control rods nor reserve shutdown material are inserted into the core), the ***temperature coefficient of reactivity will shut down the reactor*** from any power level following loss of cooling. As an example, given that no additional positive reactivity is inserted, core power will be reduced to shutdown levels by negative temperature coefficient alone, such that the RCCS alone can safely cool the core for more than 30 h after the initial shutdown. A test conducted at the AVR in Germany supports analysis which shows that following this initial shutdown, a gradual core temperature increase and negative reactivity addition will occur, with the core stabilizing at a low power level at which the heat generation rate matches the core cooling capability of the passive heat sinks. This is a ***stable, safe condition*** that can be maintained until corrective actions are taken to insert the control rods or drop reserve shutdown control material into the core to affect a full shutdown and to allow the reactor to be taken to cold shutdown condition." (Sec. 7.4, p. 7-25) – ***Response to Accident Tests*** - "These tests are intended to demonstrate the inherent response characteristics of the reactor module. Four basic categories of events are proposed: (1) ***reactivity transients***, (2) pressurized cool down, (3) water ingress, and (4) depressurized cool down. These categories cover the performance of the key systems which provide safety and investment protection."	maintain the reactor outlet temperature at a given value.	***NRC's concern over the "harder thermal-neutron energy spectrum" and its "possibly large effects on reactivity".*** • *Based on review of GA PCDR*: The General Atomics PCDR has addressed the General Atomics design strategies for reactivity control and neutron control as features of the design, and has stated that all credible reactivity addition events can be controlled. ***General Atomics has not specifically addressed NRC's concern over the "harder thermal-neutron energy spectrum" and its "possibly large effects on reactivity".*** • *Based on review of WEC PCDR*: There are discussions of the reactivity control systems and inherent ability of the core to resist increased reactivity, however t***here is no indication that "harder thermal neutron energy spectrum" and its "possibly large effects on reactivity" have been addressed in the WEC PCDR.*** • *NGNP R&D Response*: Unclear what is meant here. With H_2O ingress, spectrum will soften and the additional moderation by steam will cause reactivity increase. Without water, both systems exhibit a large negative temperature coefficient.
B-8	Variations in fuel enrichments, kernel	***There is no indication that this item has been***	See item B-7 for information relating to reactivity control.	See item B-2 for information associated with the Reactivity	• *Based on review of*

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
	diameters, coatings, and density of packing (PMR vs. PBR) must be accounted for in calculating the neutron reaction self-shielding effects in both the resonance or epithermal region and the thermal region of the neutron energy spectrum, to properly calculate the Doppler fuel temperature coefficient of reactivity and the MTC.	***specifically addressed.*** However, the importance and uncertainties associated with fuel fabrication and consistent fuel quality are well-recognized. See section 15.0***, Fuel Strategy***, beginning on page 220. Also, the R&D aspects of fuel development and qualification (***fuel kernel, coating***, compact, QA, and mass production) are addressed in section 19.2.1, beginning on page 282.	See item E-11 for GA's draft generic fuel specification. (Sec. 3.1.4.4, p. 3-76) – ***Fuel Quality and Performance Requirements*** – "…The fuel and reactor core are to be designed such that there is at least a 50% probability that the radionuclide releases will be less than the Maximum Expected criteria, and at least a 95% probability that the releases will be less than the Design criteria. The logic for deriving these fuel requirements is illustrated in Figure 3.1-68. Top-level requirements for the NGNP are defined by both the regulators and the users. Lower-level requirements are then systematically derived using a systems-engineering approach. With this approach, the radionuclide control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on radionuclide releases from the VLPC, reactor vessel, and reactor core are successively derived. Fuel failure criteria are in turn derived from the allowable core release limits. Finally, the required as-manufactured fuel attributes are derived from the in-reactor fuel-failure criteria, with consideration of achievable values based on existing fuel manufacturing experience, thereby providing a logical basis for the fuel quality specifications…The maximum allowable release fractions for 30.2-yr Cs-137 and 249.8-d Ag-110m are included in Table 3.1-16 because these nuclides are expected to be the ***strongest contributors to worker dose***, based on previous assessments of radionuclide plateout distributions and plant-maintenance requirements." Sec. 7.2.3.1, beginning on p. 7-9, provides information on ***technology development required*** for the General Atomics Fuel Fission Products Program, including Fuel Process Development and Fuel Materials Qualification.	Control System (RCS) and the Reserve Shutdown System (RSS). (Sec. 5.2.5, pp. 5-25 thru 5-34) – *Fuel Supply and Fabrication* – This section provides substantial detail on the manufacturing process intended for the PBMR fuel.	*AREVA PCDR*: ***There is no indication that this item has been specifically addressed in the AREVA PCDR.*** • *Based on review of GA PCDR*: The treatment of fuel production in terms of establishing standard statistically based specifications has been addressed in the General Atomics PCDR. ***It is not apparent that the phenomena cited in this item have been used as a basis for that specification***. • *Based on review of WEC PCDR*: There are discussions of reactivity control and fuel fabrication; however ***there is no indication that this item regarding variations has been specifically addressed in the WEC PCDR.*** • *NGNP R&D Response*: It appears NRC wants to understand how small variations in all fuel parameters (thickness, density, packing fraction, etc.) impact physics parameters. This can be done by any of the vendors or by INL tools.
B-9	Due to concerns over control rod drive reliability and re-criticality after Xenon-135 decay, the plant operator retains the safety function of achieving long-term hot and cold shutdown during an extended ATWS; and the equipment used by the operator to carry out this safety function, whether located in the control room or in a remote location, must be appropriately qualified to execute that safety function.	(Sec 6.1.3.4, p. 65) – Neutron Control During Accident Conditions – "The detection of reactivity insertion events leads to reactor shutdown by automatic insertion of the control rods by the Reactor Protection System (RPS). In cases where the events are coupled with a loss of electrical power, the ***controls rods will drop into the core by gravity***. The (Reactor Reserve Shutdown System (RRSS), can be manually actuated to achieve a diverse method of reactor shutdown, should control rod insertion not be accomplished. The two neutron absorbing systems are designed so that the insertion of either one of these systems ensures and maintains a subcritical state in all conditions. This includes, in particular, the reactivity due to the core cooling down to the coldest shutdown state combined with	(Sec. 5.1.1.3, p. 5-5) – ***Control of Heat Generation*** – "***Control rods*** drop by gravity into the core upon loss of electrical power. An automatic positive control action can cause the rods to drop, or the event itself may cut the power supply. It is an advantage that the rods need not be powered in. In addition, the NGNP has a ***redundant and diverse system*** to drop boronated graphite pellets by gravity into designated fuel element channels for reactivity control equivalent to rod insertion. Initiation of the latter system requires a positive control signal and an active response. ***If both the control rod and the reserve shutdown systems fail***, (i.e., if neither control rods nor reserve shutdown material are inserted into the core), the temperature coefficient of reactivity will shut down the	***There is no indication that this item has been specifically addressed.***	• *Based on review of AREVA PCDR*: In the AREVA PCDR, this item appears to be addressed in the design. All appropriate systems appear to be safety grade. However, ***there is no indication that re-criticality following xenon decay in an ATWS event has been specifically addressed in the AREVA PCDR.***

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		the ***xenon effect*** and the reactivity insertion due to the initiating event." (Sec. 11.5.2.6, p. 182) – Reactivity Excursion – "The detection of reactivity insertion events leads to reactor shutdown by automatic insertion of the control rods by the RPS. ***A second system, the RSS can also achieve the function. RSS is manually actuated.*** The two neutron absorbing systems are designed so that the insertion of at least one these systems ensures and maintains subcriticality in any conditions. This includes in particular the reactivity due to core cooling down to the coldest shutdown state combined with the ***xenon effect*** and the reactivity insertion due to the initiating event. If the reactivity insertion and the reactivity insertion speed are limited and if the ***reactor is not shut down***, the situation is potentially controllable though power, fuel temperature and helium temperature should rise. In particular, as power increases, fuel temperature rises rapidly and, due to Doppler effect, results in negative reactivity feedback. Heating of the graphite moderator and most of all of the reflectors occurs more slowly, and, as a consequence, the associated temperature feedbacks come relatively later." (Sec. 11.3.2.1, p. 166) – "The negative temperature coefficient of reactivity and neutronically-inert helium coolant inherently ***stabilize the heat generation during any situation in acceptable conditions before the occurrence of significant xenon effect***. There is a large grace period before unacceptable consequences of xenon effect occur. This allows active operation of the control rod system or the reserve shutdown system. The control rod system and the reserve shutdown system are both capable to shut down the reactor during any condition including xenon effect occurrence. ***Any situation which could not be mitigated by these provisions is practically eliminated by design***. If neither control rods nor reserve shutdown material are inserted, the temperature coefficient of reactivity will tend to shut down the reactor from any power level following loss of forced convection cooling, such that the RCCS alone can safely cool the core beyond 24 hrs after the initial shutdown. If neither control rods nor reserve shutdown material are inserted, the temperature coefficient of reactivity will tend to shut down the reactor from any power level following loss of forced convection cooling, such that the RCCS alone can safely cool the core beyond 24 hrs after the initial shutdown."	reactor from any power level following loss of cooling." In the General Atomics PCDR, this item appears to be addressed in the design, however, ***there is no indication that re-criticality following xenon decay in an ATWS event has been specifically addressed in the General Atomics PCDR.***		• *Based on review of GA PCDR*: In the General Atomics PCDR, this item appears to be addressed in the design. However, ***there is no indication that re-criticality following xenon decay in an ATWS event has been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: ***There is no indication that this item has been specifically addressed in the WEC PCDR.*** • *NGNP R&D Response*: Design issue – not R&D
B-10	The uniqueness of configuration (tall, thin annular core) of current PMR and PBR designs and high operating temperatures require detailed reactor physics testing of the first unit as a function of core burnup, and of the start-ups of the second and perhaps third cycles. Attention should be paid to the instrumentation needs for these tests since neutron sensors must be both distributed and inter-calibrated to infer power distributions. Neutron detectors used in test measurements	(Sec. 10.2.7, p. 154) – ***NGNP Safety Testing*** –"As a prototype for a possible fleet of Generation IV commercial nuclear power plants, the NGNP is expected to demonstrate the plant's passive and inherent safety features through a series of tests emulating various anticipated operational occurrences and design basis events. Unique operation and control strategies are envisioned such that key measures, based on safety, component tolerances, system efficiencies, etc. can be identified to define the operational envelope expected for future licensing activities related to a commercial plant while providing sufficient protection of the	See item A-1 for information on in-core and ex-core monitoring instrumentation. See item A-8 for a description of Design Methods Development and Validation. See item B-4 for information on instrumentation of Neutron Control System.	See item C-6 for discussion of PBMR test facilities. (Sec. 4.2.10, p. 4-68 thru 4-76) - *NHSS Control and Instrumentation System* – "The NHSS Control and Instrumentation System comprise only of equipment found in the Nuclear Heat Supply Building and all operator interaction is performed in the Central Control & Supervisory System (CCSS) as described in the Section 9: Balance of Plant Systems. The primary systems comprising the NHSS Control and Instrumentation system are the Operational	• *Based on review of AREVA PCDR*: Needs for testing and instrumentation are recognized in the AREVA PCDR. In the PCDR, the tall core shape is actually used repeatedly by AREVA as a design feature that will tend to slow down the plant response to transients and

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
	should also be sensitive enough to measure reactivity and changes in flux levels and distributions.	plant staff, the public, and the investment in the various NGNP systems, structures, and components." (Sec. 10.2.8, p. 154) – ***High Temperature Testing*** – "To characterize the performance of the processes associated with the HPPP as a function of temperature, the NGNP will be expected to provide helium temperatures in the range of 1000 – 1100 °C. To sustain such temperatures, the NGNP will provide only the power demand required by the HPPP and shutdown helium circulation in the power generation loops. This testing mode could also facilitate the study of as yet-to-be determined future missions of the NGNP plant that may require alternative components, materials, and/or fluids." Sec. 17.6, p. 240 – ***Initial Startup Operations and Testing*** – "The initial startup and testing is critical to the overall schedule performance of any nuclear plant. The NGNP prototype facility is no exception. As the prototype demonstration plant for the new generation of high temperature gas cooled reactors the NGNP initial startup operation and testing schedule is developed to achieve the following: • Component testing and turn-over • System functional testing and turn-over • Initial approach to criticality • Zero power operation • Power ascension including grid connection • Normal plant safety system tests (AOO tests) • Special licensing performance tests (DBA tests) • Commercial operability endurance tests • Component dismantling and examination • Fuel examination The schedule provides four years for this phase of plant operations. The first two years (2017 and 2018) is dedicated to non-nuclear system testing and turn-over including the standard system turn-over from construction to operations. The second two years (2019 and 2020) includes initial plant criticality. During this phase all safety systems will be examined and tested and several special licensing related tests is planned. This phase of the plant operation includes component dismantling and inspection and fuel examination." (Sec. 19.2.3, p. 289) – ***Instrumentation*** – "NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy. For neutron flux detectors some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. For temperature measurements the standard thermocouples used in nuclear plants today are capable of	See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 5.1.1.1, p. 5-2) - Core Geometry and Size – "The annular core geometry, ***limited core diameter***, low thermal power rating, and low power density of the NGNP assure sufficient decay heat removal to an ultimate heat sink by the natural processes of radiation, conduction, and convection, to ***preclude any significant particle coating failure or radionuclide release*** under all conditions of loss of forced cooling or loss of coolant pressure."	Control System (OCS), the Equipment Protection System (EPS) and the Reactor Protective System (RPS). The OCS monitors and controls the NHSS systems throughout their normal operating range. The EPS detects operating regimes or operating conditions that may be harmful to NHSS equipment, and takes appropriate action to prevent or minimize potential damage. The RPS automatically initiates RUS protection whenever pre-established set points are exceeded…. The RPS consists of three subsystems: Reactor Trip System (RTS), Post Event Instrumentation System (PEI) and the Manual Diverse Shutdown System (MDSS). The RPS provides functions to prevent exceeding predefined safe operating limits and to provide information to operators in the event of nuclear accidents. The Protection System is implemented using a Class 1E qualified digital platform that is capable of performing logical operations as well as algorithmic processing. The same platform is used for both the RTS and the PEI applications. The MDSS is a hard-wired system that allows tripping the breakers without dependence on any software. All portions of the RPS are treated as Class 1E Structures, Systems and Components (SSC)…. The RTS automatically prevents operation of the RUS in an unsafe condition by shutting down the RUS whenever predetermined operating limits are approached, or when Design Basis Accident conditions are detected. The operating limits are selected, based on initial conditions (RUS power and outlet temperature) assumed in the safety analysis…. The MDSS is a hardwired system that enables operators to manually initiate reactor trip, RCS and RSS activation functions, from both the Main Station Control Room and the PEMRR. The MDSS controls are supported by monitoring instrumentation associated with the PEI system or the Plant Computer displays to provide the operator the capability to know when to take the appropriate manual action. For each reactor trip function (RS and RSS activation), a set of three switches that are individual and redundant are provided in the Main Station Control Room and a set of three switches that are individual and redundant are provided in the PEMRR. Thus, a total of 4 sets of three switches each comprise the MDSS for the protection system." (Sec. 11.2.2, p. 11-12) – *NGNP Integrated Control Philosophy* – "Controllability and transient performance of the NGNP are ultimately determined by the dynamic characteristics of the NHSS, HTS, HPS and PCS. The NGNP demonstration plant has the following dominant system characteristics: • The thermal response of the NHSS is slow, since the graphite-moderated core has a large thermal capacity relative to its heat generation and removal rates. However, the NHSS is the most critical system and will govern the overall control philosophy. • The advantage is that the large thermal capacity of the core allows relatively fast load changes of the system without requiring fast response from the core. In principle, the energy stored in the core can be tapped or additional energy can be stored, with minimum core temperature changes. ***The NHSS also has a negative***	accidents. • *Based on review of GA PCDR*: Needs for analytical codes, instrumentation, and testing relating to core monitoring are addressed in the General Atomics PCDR. General Atomics credits the core geometry as one of the design features that will help to control the type of phenomenon of concern to NRC. • *Based on review of WEC PCDR*: There are discussions of Nuclear Heat Supply System instrumentation, and of the general PBMR control philosophy; however, ***there is no indication that this item regarding detailed reactor physics testing and associated instrumentation has been specifically addressed in the WEC PCDR.*** • *NGNP R&D Response*: Initial core test, not R&D, planned in the near future.

TABLE 2B – REACTOR PHYSICS AND NEUTRONICS - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
		measuring operating temperatures up to 1200 ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired.”		***temperature coefficient, which results in the reactivity and consequently the neutronic power to counteract temperature changes.*** The NHSS is therefore to a large extent self-regulating and minimum control interaction is required to maintain the ROT at a given value. • Another advantage is that the HTS can also be controlled easily by controlling the speed of the PHTS and SHTS circulators. A loss of outside electric load, PCS trip or HPS trip will result in temperature changes in the PCS, which will propagate to the SHTS, PHTS and the NHSS. Temperature changes can be monitored and controlled by manipulating the mass flow rates through the PCS, SHTS and PHTS. Thus, in principle the NGNP consists of an inherently stable and slow acting NHSS coupled to a stable HTS, HPS and PCS that will require active control to remain stable under all anticipated operating scenarios.”	

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D Efforts *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
C-1	*General Safety Analysis/Safety Document Needs* • Comprehensive description of the NGNP safety philosophy, a listing of the components involved, and the conditions under which these components are expected to perform their safety functions. • Explanation of how this philosophy meets the defense-in-depth approach and, in particular, answers to the following: o Will the components that perform a safety function (retain FPs) be classified as safety-related components, with the imposition of equipment qualification, in-service inspections, and/or Technical Specifications LCOs and SRs? o How will aging issues be addressed? If the safety function of a component is to retain FPs on its surface during adverse conditions, how can it be ensured that this function can be retained for long periods (decades), despite the possible presence of other long-term surface degradation mechanisms? o Will the surface state of a non-replaceable or difficult-to-replace component be reactivated by chemical action or cleaning during its service life? • A sound basis for the selection of the physical models and the data for these models must be justified. • The materials to be used and their sensitivity on the transport case must be identified. • Once the actual reactor design is available, the transport pathways that result from the accident conditions must be identified, along with the relevant models and data needed for the resulting calculations. • Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified. • The fuel database must be developed, as	(Sec. 11, pp. 160-188) - The ***safety philosophy, listing of the components involved, and the conditions under which these components are expected to perform their safety functions*** are described in Sec. 11. (Sec. 11, pp. 160-188) – The attainment of ***defense-in-depth*** is addressed throughout Sec. 11, and as an individual topic in Sec. 11.3.4, p. 171. (Sec. 13.2, p. 211) – ***In Service Inspection*** – "1. The NGNP design shall provide access to the helium pressure boundary to permit in service inspection as required by appropriate sections of the ASME B&PV Code. 2. Where cost effective, the design of systems and components shall incorporate those features required to implement on-line in service inspection. If the unit or major component must be removed from service, design features shall be included to accomplish the inspection during the power unit allotted planned outage time. 3. Plant piping design shall minimize the need for snubbers and restraints and shall ensure inspectability. 4. Design documentation shall include plans and procedures for conducting in service inspection and shall identify equipment necessary to conduct the inspection. The equipment vendor shall furnish the ISI equipment not commercially available. 5. An in-service inspection program shall be developed and maintained throughout the design process. The program shall include anticipated durations and worker-hours, including health physics, for isolating the equipment/system, preparing for and performing the inspections, and returning the equipment/ system to service. Physical and/or computer models shall be used to assess inspectability. 6. The plant design shall include those facilities and features required to set up the in-core fuel handling equipment for periodic inspections, maintenance, testing, and demonstrations of integrated equipment operation. Such inspection, maintenance, testing, and demonstrations shall not interfere with core refueling operations nor have an adverse effect on plant operation." (Sec. 15.0, p. 220) – "In addition to these requirements and values, ***expected fuel performance characteristics*** will eventually be defined by required plant radionuclide release performance under operational and accident conditions to meet regulatory offsite and worker dose limits. The ***limiting radionuclide releases*** associated with the key accident analyses ***have not yet been determined***. As such, the NGNP plant specific required fuel performance characteristics have not yet been defined." (Sec. 19.2.4, p. 290) – ***Computer Codes and Methods Development and Validation –*** Included in this section are descriptions of R&D needs for computer codes addressing reactor system analysis, neutronics, thermal hydraulics/pneumatics, ***fuel performance, fission product transport***, and structural mechanics. Fuel performance	See item A-8 for a description of Design Methods Development and Validation. See item C-6 for information relating to the test program. See item D-1 for information relating to metallic materials characterization. See item E-1 for information relating to graphite materials characterization. (Sec. 1.5.1, p. 1-18) – "Both PMRs and PBRs can use UCO fuel, and by doing so would benefit from lower fuel costs because of the higher fuel burnup obtainable with UCO fuel relative to UO2 fuel. However, the economic penalty associated with use of UO2 fuel would be greater for a PMR than a PBR because this would necessitate a shorter refueling cycle, thereby reducing reactor availability. Also, it is not clear that a PMR loaded with UO2 fuel could operate for an extended period of time with a core outlet coolant temperature of 950°C because of the potential for kernel migration in UO2 fuel exposed to high thermal gradients. The capability of PBRs to use UO2 fuel, which has a ***more extensive irradiation and safety testing database*** than UCO fuel, could potentially make licensing a pebble bed NGNP somewhat less difficult than licensing a prismatic block NGNP. However, this advantage would not extend to a follow-on commercial pebble bed VHTR because it is ***expected that UCO fuel will have been qualified and be available*** for use by the time a commercial VHTR is built." (Sec. 3.1.4, p. 3-64) – "For modular gas-cooled reactor designs, a ***hallmark philosophy*** has been adopted since the early 1980s to design the plant such that radionuclides would be retained in the core during normal operation and postulated accidents. The key to achieving this safety goal is the reliance upon ceramic-coated fuel particles for primary fission product containment at their source, along with passive cooling to assure that the integrity of the coated particles is maintained even if the normal active cooling systems were permanently disrupted. This design philosophy has been carried forward for all subsequent MHR designs, including the NGNP. Fuel performance and radionuclide control in gas-cooled reactors is discussed in detail in numerous publications, including [IAEA 1997], [Hanson 2002], and [Hanson 2004a]. As is discussed in detail in Section 5.1.1.1, the radionuclide containment system for the NGNP, which reflects a ***defense-in-depth philosophy***, is comprised of multiple barriers to limit radionuclide release from the core to the environment to insignificant levels during normal operation and postulated accidents. The five principal release barriers are: (1) the fuel kernel; (2) the particle coatings (particularly the SiC coating); (3) the fuel element structural graphite; (4) the primary coolant pressure boundary; and (5) the Reactor	See item A-8 for discussion of modeling efforts. Section 2.2.2, pp. 2-36 thru 2-38, *Regulatory Requirements for the NGNP*, provides a detailed enumeration and description of the regulatory/safety requirements that will be met by the PBMR/NGNP, including ***high level safety goals, site level consequence and environmental limits, dose limits for anticipated operational occurrences, dose limits for design basis events, safety goals for beyond design basis events, state and local regulatory requirements, and industry codes and standards***. (Sec. 4, pp. 4-9 thru 4-85) – *Nuclear Heat Supply System* – This section contains ***detailed descriptions of the reactor system, layout, roles of components, and supporting systems. Fuel design is addressed*** in section 4.2.1.1, p. 4-17, and throughout section 5, *Reactor Fuel*. (Sec. 11.2.2, p. 11-12) – *NGNP Integrated Control Philosophy* – "Controllability and transient performance of the NGNP are ultimately determined by the dynamic characteristics of the NHSS, HTS, HPS and PCS. The NGNP demonstration plant has the following dominant system characteristics: • The ***thermal response of the NHSS is slow, since the graphite-moderated core has a large thermal capacity*** relative to its heat generation and removal rates. However, the NHSS is the most critical system and will govern the overall control philosophy. • The advantage is that the large thermal capacity of the core allows relatively fast load changes of the system without requiring fast response from the core. In principle, the energy stored in the core can be tapped or additional energy can be stored, with minimum core temperature changes. The ***NHSS also has a negative temperature coefficient, which results in the reactivity and consequently the neutronic power to counteract temperature changes***. The NHSS is therefore to a large extent self-regulating and minimum control interaction is required to maintain the ROT at a given value. • Another advantage is that the HTS can also be controlled easily by controlling the speed of the PHTS and SHTS circulators. A loss of outside electric load, PCS trip or HPS trip will result in temperature changes in the PCS, which will propagate to the SHTS, PHTS and the NHSS. Temperature changes can be monitored and controlled by manipulating the mass flow rates through the PCS, SHTS and PHTS. Thus, in principle the NGNP consists of an inherently stable and slow acting NHSS coupled to a stable HTS, HPS and PCS that will require active control to remain stable under all anticipated operating scenarios." (Sec. 12.1.2, p. 12-12) – *Maintenance requirements for*	• *Based on review of AREVA PCDR*: The AREVA PCDR has recognized most of the safety analysis/safety document needs detailed in this item, ***with the exception of the following:*** ***- Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.*** ***- The fuel database must be developed, as well as fuel-failure models and fuel material properties (both measurable and process controlled).*** • *Based on review of GA PCDR*: The General Atomics PCDR has recognized most of the safety analysis/safety document needs detailed in this item, ***with the exception of the following:*** ***- How will aging issues be addressed? If the safety function of a component is to retain FPs on its surface during adverse conditions, how can it be ensured that this function can be retained for long periods (decades), despite the possible presence of other long-term surface degradation mechanisms?*** ***- Will the surface state of a non-replaceable or difficult-to-replace component be reactivated by chemical action or cleaning during its service life?***

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D Efforts *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
	well as fuel-failure models and fuel material properties (both measurable and process controlled).	code R&D is addressed in section 19.2.4.4, p. 292, and fission product transport code R&D is addressed in section 19.2.4.5, p. 293.	Building/containment structure. The most important of these barriers to fission product release from the core is the silicon carbide and pyrocarbon coatings of each fuel particle. Both the SiC and PyC coatings provide a barrier to the release of fission gases. The SiC coating acts as the primary barrier to the release of metallic fission products because of the low solubilities and diffusion coefficients of fission metals in SiC…” (Sec. 3.10, p. 3-197) – ***Plant Operation and Control Systems*** – “The unique features of the MHR assure the general public inherent protection against fission product release from the reactor core. In addition, the inclusion of the safety-related Reactor Protection System (RPS) and the non-safety-related Investment Protection System (IPS) in the NGNP specifically provide the “***defense in depth***” design strategy that is required for modern reactor plants. Other design areas related to a complete “defense in depth” protection strategy are the Essential AC Electric System and the Essential DC Electric System. Also, systems such as the Reactor System contain end-action hardware to perform safety-related and non-safety actions. The Plant Control, Data, and Instrumentation System (PCDIS) provides normal control and instrumentation functions, and also provides overall integration of the control and protection functions into a combined plant control system. This system provides normal (main loop) cooling if possible following a reactor trip, broadening “***defense in depth***” design features by making the SCS or RCCS less likely to be used for reactor cooling.“ ***Sections 3.10.1.1, p. 3-198 (RPS/IPS), and 3.10.2.1, p. 3-208 (PCDIS) also contain recommendations for further development and improvement of the design of these systems.*****”** (Sec. 5.1.1, p. 5-1) – ***Key Inherent Safety Features and Design Provisions*** – “A ***defense-in-depth approach*** to safety has always been used in the design of MHRs including the NGNP. The philosophy of defense-in depth includes prevention of accidents by requiring reliable operating systems capable of handling anticipated operational occurrences. It nevertheless assumes these systems could fail and thus requires that certain functions be fulfilled to prevent and mitigate consequences of those failures. The ultimate goal is to ensure that plant operation will have negligible impact on the health and safety of the public under a comprehensive, extensive range of expected and postulated conditions. A key feature of defense-in-depth is the provision of multiple barriers to the release of fission products and systems which protect these barriers. Furthermore, these systems are capable of functioning despite credible failures, by being redundant, independent, and diverse. The assurance of safety is thereby vested in multiple, independent safety provisions, no one of which is relied upon excessively. Analysis of design-basis events (DBEs) and beyond-design basis events (BDBEs) early in the design process is a means of identifying and providing ways to further enhance plant safety.. Finally, contingency measures are provided in the event that fission products are released anyway. Defense-in-depth is comprehensive,	*Systems and Components* - The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. General requirements for each system include that the design provides ***access to the pressure boundaries to permit in-service inspection as required by appropriate sections of the ASME B&PV Code***, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible. (Sec. 12.3.1.4, p. 12-23) – *Reactor Cavity Cooling System* - The RCCS is designed for life of the plant, thus ***no scheduled maintenance other than In-Service-Inspection*** (ISI) is envisaged. Provision is made in the design for ‘as required’ inspection and repair of the RCCS. Special tools will be developed up to the point of having a basic design in place, so that should a repair become necessary, the required equipment can be procured at short notice. As far as possible, ‘off-the-shelf’ equipment is used. (Sec. 14, p. 14-7) – *Safety: Summary and Conclusions* - .”The ***safety design philosophy is to apply the principles of defense-in-depth at a fundamental level in which a diverse combination of inherent reactor characteristics, passive design features and Structures, Systems and Components (SSCs), active engineered systems, and operator actions are deployed to maintain the integrity of robust passive barriers to radionuclide release***. The reactor-specific key safety functions are derived in a top-down manner with the objective of protecting the integrity of the multiple barriers to radionuclide release; these include the control of heat generation, control of heat removal, control of chemical attack, maintenance of core and reactor geometry, and maintenance of the reactor building structural integrity. A fundamental aspect of the safety design philosophy is to provide the capability to perform safety functions first through the selection of inherent reactor characteristics and engineered systems that operate on passive design principles and then to support these safety functions with combinations of diverse active engineered systems and operator actions. The safety design approach for the NGNP is derived from a risk-informed and performance-based model of defense-in-depth. This approach recognizes three major elements: Plant Capability Defense-in-Depth, Programmatic Defense-in-Depth, and a Risk-Informed Evaluation of Defense-in-Depth. These three elements enable the examination of a plant’s defense-in-depth capability from different perspectives including those of: • Designing the plant and the capabilities of its SSCs that perform safety functions • Defining the programs that ensure the plant will be built as designed and will operate safely throughout the plant lifetime while preserving the intended defense-in-depth capabilities. • Evaluating how the plant performs its safety	***- Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.*** • *Based on review of WEC PCDR*: The WEC PCDR has recognized most of the safety analysis/safety documentation needs detailed in this item, ***with the exception of the following:*** ***- Technical Specifications for the maximum acceptable FP loading of key components must be determined along with practical methods of ensuring that the levels can be determined during normal operation. A recovery plan for handling and recovering from exceeding the limits should be identified.*** • *NGNP R&D Response*: Development of the Fuel database is planned.

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D Efforts *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
			covering aspects of human involvement (e.g., administrative controls, quality assurance, human factors engineering, training, etc.) to assure the accuracy and sufficiency of the design, construction, and operation of the plant." (***Additional information follows on inherent safety characteristics, specific design features, multiple barriers to release of fission products, accident prevention and mitigation***.) (Sec. 5.2.1.1, p. 5-13) – "After a CP is issued by the NRC, the applicant must, if it did not as part of the original application, submit a Final Safety Analysis Report (FSAR) to support its application for an Operating License (OL). Typically, after the CP is issued by the NRC and construction is underway, the Licensee begins developing the FSAR. During this period of the Project, the PSAR is revised and updated to reflect the evolving plant design as well as ***operational aspects*** (e.g., procedures, ***Technical Specifications***, Human Factors, Emergency Planning, Security, Programs, etc.) that were not available, or needed by the NRC during the early phase of the Project for the issuance of the CP. The FSAR describes the final design of the facility as well as its operational and emergency procedures. The NRC then prepares a Final Safety Evaluation Report (FSER) for the OL, and the ACRS makes an independent evaluation and presents its advice to the Commission."	functions in the prevention and mitigation of accidents and determining the adequacy of defense-in-depth. The NGNP safety design approach is framed in terms of ***reactor-specific safety functions*** that were developed from the top goal of retaining the inventory of radionuclides primarily within the fuel and then considering the specific functions that when satisfied would ***protect the integrity of the fuel and other radionuclide transport barriers.*** The required safety functions include those to: • Control heat generation (reactivity) • Control heat removal • Control chemical attack • Maintain core and reactor vessel geometry • Maintain reactor building structural integrity The safety evaluation for the NGNP will be performed using a risk-informed and performance-based approach. The key elements of this technology-neutral approach include: (1) the use of accident frequency vs. radiological dose criteria that are derived from current U.S. licensing requirements, referred to as Top Level Regulatory Criteria (TLRC), (2) use of a full-scope Probabilistic Risk Assessment (PRA) to select the Licensing Basis Events (LBEs), (3) development of reactor-specific functions, selection of the corresponding safety-related SSCs, and their regulatory design criteria, (4) deterministic design conditions and special treatment requirements for the safety-related SSCs, and (5) a risk-informed evaluation of defense-in-depth." (Sec. 14.2.4, p. 14-16) – *Selection of Design Features to Perform Safety Functions* – "The NGNP safety design is based on meeting the following objectives that specifically incorporate the defense-in-depth approach described above: • Provide safe, economic and reliable nuclear heat to the HPS and the PCS, which produce hydrogen and electricity respectively • Select compatible fuel, moderator, & coolant with inherent safety characteristics • Utilize proven technologies to the maximum extent practical • Design reactor with inherent characteristics and passive safety features sufficient to protect the public as the primary strategy for Plant Capability Defense-in-Depth • Supplement with active design features and SSCs for investment protection and as a secondary strategy for Plant Capability Defense-in-Depth Important inherent characteristics of the NGNP design include: • Ceramic-coated pebble fuel	

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				• Capability to maintain integrity at high temperatures • Chemically compatible with coolant and moderator • Graphite moderator • Capability to maintain integrity at high temperatures • High thermal heat capacity • Chemically compatible with fuel and coolant • Large neutron migration length for neutron stability • Helium coolant • Single phase over all normal and accident conditions • Chemically and neutronically inert • Low stored thermal energy In addition to these inherent characteristics, the NGNP has both passive and active ***design features to perform defense-in-depth functions***, as discussed below. The NGNP safety design approach is to provide inherent characteristics and passive SSCs that are sufficient to protect the public and to meet the Top Level Requirements and to provide the primary strategy for Plant Capability Defense-in-Depth, and then to provide additional active SSCs to provide additional levels of defense-in-depth as well as to meet user requirements for plant availability and investment protection. A summary of the inherent characteristics and passive SSCs that are available to support each required safety function, as well as the additional active SSCs that support these functions is provided ...” (Sec. 15.1.1.1, p. 15-11) – *Fuel Type and Form* – “The PBMR core consists of fuel elements containing uranium dioxide coated particles that generate heat by means of fission reactions. The fuel pebble consists of uranium dioxide coated fuel particles and matrix graphite pressed into a spherical shape. A fuel sphere is divided into two regions. The inner spherical region is known as the fuel region, while the outer shell surrounding the fuel region is known as the fuel-free region. The fuel region of each fuel sphere contains a large number of evenly dispersed spherical particles known as coated particles in which the fuel is contained while there are no coated particles in the fuel-free region. The design of the coated particles and fuel sphere is summarized in.….” (Note: additional information follows in this section that provides details on the fuel design, materials, fabrication, transportation, receipt, onsite handling, and loading into the reactor core.) (Sec. 16.2.1.1.5, p. 16-23) – *Experimental Plate-out Facility* – “The purpose of tests done in the Experimental Plate-out Test Facility (POTF) is to obtain representative PBMR material plate-out parameters and, if possible (to be determined by a second feasibility study), to ***obtain graphite dust and fission product interaction data***.” (Note: This is followed with descriptions of the Experimental	

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				Plate-out Loop and the Isopiestic Plate-out facility.) (Sec. 16.2.2, p. 16-35) – *Design Data Needs (Nuclear Heat Supply System)* – "Three DDNs have been identified pertaining to the NGNP Fuel. The first of these DDNs (NHSS-01-01) identifies the need for data to ***extend the irradiated fuels qualification database*** from the temperature-burnup envelope of the PBMR Demonstration Power Plant (DPP) to that of the PBMR NGNP. The second DDN (NHSS-01-02) specifies data to correspondingly extend the heat up data pertaining to accident conditions. The third DDN (NHSS-01-03) provides for an extension of the temperature-fluence envelope of the Fuel Graphite to that required by the NGNP. In all three cases, the extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature." (Sec. 16.2.3.1, p. 16-36) – *Fuel Qualification R&D Plan* – "Two supplemental irradiation tests are planned for the PBMR NGNP to ***extend the database supporting the fuel*** for the PBMR DPP. The first of these irradiation tests, which is proposed to start in FY2009 will use pre-production fuel. The second, proposed to start in 2012 will use actual production fuel."	
C-2	*Model Development and V&V* - Physical models and the supporting mathematical methods, addressing: • Nuclides of interest • Fission product release from the fuel • Diffusion, adsorption, and desorption in graphite and fuel matrix materials • Adsorption, desorption, and in-diffusion in reactor system metals • Chemical and physical forms of the FPs in the coolant • Tritium transport models • Aerosols and dusts that plate-out on reactor system components and their mobility • Fission product reactions with the confinement building materials • Reactions of the reactor system components and fission products with air or steam • Plume models that transport the released material beyond the reactor building • Determination of the safety function of each subsystem and the level of FPT attenuation required. • Determination of level of sensitivity to component uncertainties and how this reflects on the physical models.	See item A-7 for needs relating to air ingress phenomenon. See items B-6 and F-4 for needs relating to water ingress phenomenon. (Sec. 19.2.4, p. 290) – ***Computer Codes and Methods Development and Validation –*** Included in this section are descriptions of R&D needs for computer codes addressing reactor system analysis, neutronics, thermal hydraulics/pneumatics, fuel performance, fission product transport, and structural mechanics. (Sec. 19.2.4.4, p. 292) – ***Fuel Performance Models and Codes*** – "The R&D need for ***ATLAS*** development/modification is to improve the diffusion and the coatings corrosion modeling. For code qualification the heat-up experiments of irradiated fuel particles at relevant operating conditions (burnup, temperature, fluence) are required to anchor the developed code…qualification of ATLAS…includes two irradiation and heat up tests. In addition, there is an R&D need to develop the UCO models." (Sec. 19.2.4.5, p. 293) – ***Fission Product Transport*** – "The R&D needs of the ***FP Transport code*** include development of models for: • assessment of product activation in the primary circuit (in particular tritium and 14C), • radio-contamination distribution in the primary circuit, making distinction between circulating activity, plated out /	See item A-6 for information on air and water ingress See item A-8 for a description of Design Methods Development and Validation. See item C-5 for description from Sec. 7.2.3.1, p. 7-13, ***Radionuclide Transport.*** Tritium transport, radionuclide transport in the containment, and other fission product tests are identified as needs. See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3.1.4.3, p. 3-73) - ***Radionuclide Transport Mechanisms*** - "Radionuclide transport is modeled in the fuel kernel, the particle coatings, fuel compact matrix, fuel-element graphite, primary coolant circuit, and Reactor Building. [IAEA 1997] provides an excellent overview and an extensive bibliography of radionuclide transport mechanisms. The transport of radionuclides from the location of their birth through the various material regions of the core to their release into the helium coolant is a relatively complicated process. The principal steps and ***pathways*** are shown schematically in Figure 3.1-66. Also for certain classes of radionuclides, some steps are eliminated (e.g., noble gases are not diffusively released from intact TRISO particles and are not significantly retarded by the compact matrix or fuel element graphite). While the actual radionuclide transport phenomena in the core can be very complex, the basic approach for modeling these	See item A-8 for model development information. (Sec. 4, pp. 4-9 thru 4-85) – *Nuclear Heat Supply System* – This section contains detailed descriptions of the reactor system, layout, roles of components, and supporting systems. ***Fuel design is addressed in section 4.2.1.1, p. 4-17, and throughout section 5, Reactor Fuel***. (Sec. 5.2.2.1.4, p. 5-14) – *Silicon Carbide Layer* – "***The production of fuel spheres having coated particles with intact SiC layers and the assurance that these layers will remain intact under all foreseeable fuel core conditions form the primary barrier to the release of radiation from NGNP***. When SiC is deposited from methyltrichlorsilane under the correct conditions, a layer of nearly 100% theoretical density is obtained. At high temperatures, the ILTI and OLTI layers partially lose their ability to contain cesium, silver and strontium. The purpose of the SiC layer is to prevent the release of these fission products into the graphite matrix, and then into the reactor helium stream. The SiC layer thus acts as the principal pressure and fission product retention barrier in the coated particle. The coated particle structure results in the SiC layer being kept under compression as long as possible by its interaction with the ILTI and OLTI pyrocarbon layers as described above. The silicon carbide layer has a thickness of approximately 35 µm and a density greater than approximately 3.2 gm/cm^3. ***The production of fuel spheres having coated particles with intact SiC layers and the assurance that these layers will remain intact under all foreseeable fuel core conditions is the cornerstone of***	• *Based on review of AREVA PCDR*: The AREVA PCDR has recognized the needs for most of the model development and V&V detailed in this item, ***with the following exceptions:*** ***- Fission product reactions with the confinement building materials*** ***- Determination of the safety function of each subsystem and the level of FPT attenuation required.*** ***- Determination of level of sensitivity to component uncertainties and how this reflects on the physical models.*** ***- Estimation of difficulty in obtaining the data and conducting the testing to support the safety case.*** • *Based on review of GA PCDR*: The General Atomics PCDR has recognized the needs

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	• Estimation of difficulty in obtaining the data and conducting the testing to support the safety case.
	• Scoping of how V&V can be performed.
		deposited activity and purification system, for both normal operation and accidental situations,
		• radio-contamination releases outside the primary pressure boundary and
		• radio-contamination releases in the environment during accident scenarios.
		…It is also recommended to develop a mechanical analysis code for the NHS.”
			phenomena is to treat radionuclide transport as a solid-state diffusion problem with various modifications and/or additions to account for the effects of irradiation and heterogeneities in the core materials…The transport of volatile fission metals in fuel-compact matrix and graphite is also modeled as transient diffusion processes. It is assumed that sorption equilibrium prevails in the gap between the fuel compact and the fuel hole surface of the fuel block. At the coolant boundary, the mass flux from the surface into the flowing coolant is given by the product of a convective mass transfer coefficient and the concentration gradient between the equilibrium desorption pressure and the mixed-mean concentration in the coolant. Diffusion coefficients and sorption isotherms have been determined experimentally for a number of nuclear graphites and matrix materials [IAEA 1997]. The ***transport and deposition of condensable radionuclides from the flowing helium coolant to fixed surfaces in the primary coolant circuit*** is essentially a convective mass transfer problem. Usually, deposition is conceived as a two-step process: (1) gaseous diffusion to the wall and (2) a wall effect, typically an adsorption process. The latter step is necessary because numerous experiments have shown that, under certain circumstances, graphitic and metallic surfaces have a limited capacity to sorb certain radioactive species. The sorptivity of metals for volatile fission products is typically a function of surface oxidation state and temperature. The wall effect may be simply an adsorption process whereby the active sites are confined to the surface. Alternatively, there are some data suggesting that certain radionuclides, principally Ag isotopes, may penetrate into the bulk of metallic components. The condensable radionuclides that are ***plated out*** in the primary circuit may be partially re-entrained and released to the Reactor Building during rapid depressurization transients. A potentially significant removal mechanism, especially during rapid depressurizations, is mechanical ***re-entrainment of deposited particulate matter contaminated by plateout and/or spallation of friable surface films***; this mechanical re-entrainment is traditionally referred to as “liftoff”. Empirical liftoff models have been developed by correlating the fractional re-entrainment of plated out fission products measured in blowdown tests with the shear ratio (the ratio of the wall shear during a depressurization transient to that during normal operation). The VLPC of the NGNP is expected to be a significant barrier to the release of condensable radionuclides to the environment during accident conditions. Consequently, the ***natural removal mechanisms, including condensation, gravitational settling, and turbulent deposition are modeled***.”
			
			(Sec. 5.1.2, p. 5-7) – ***Safety Related Systems, Structures, and Components*** – The key safety features and safety functions of the NGNP have been described above. On the basis of prior safety assessments, the major systems, structures, and components which are relied upon to perform one or more safety functions (e.g., ensuring safe shutdown and protection of the primary coolant pressure boundary) or are otherwise relied upon to meet the siting
				the safety design approach in which the fuel is the primary barrier to the release of radionuclides.”
				
				(Sec. 5.2.4.2, p. 5-17) – *Testing and Qualification (Fuel)* – “Even though test results from the German pebble-bed reactor program are available, and is the basis for the PBMR DPP, expected operational parameters specified for the NGNP was not envisaged during PBMR efforts or the German program. Therefore to ensure the safe application of PBMR DPP based fuel in the NGNP program, the following ***parameters and/or aspects are considered to be important in terms of fuel testing and qualification*** must be considered:
				• Expected/specified normal operating condition parameters
				  ○ Maximum fuel temperatures
				  ○ Percentage burn-up
				  ○ ***Percentage fission product release at specified temperatures***
				• Expected/specified adverse (accident) operating condition parameters
				  ○ Maximum temperatures
				  ○ Percentage burn-up
				  ○ ***Percentage fission product release at specified temperatures***
				• Statistical requirements of tests and qualification samples to ensure confident and safe application thereof for design and further operational specification refinements
				• Pre-designate procedures and facilities for pre- and post-irradiation tests to support qualifications”
				
				(Sec. 6.4.2.7, p. 6-100) – *Tritium Transport* – “At the operating temperatures of the NGNP, the ***transport of tritium*** through intact metallic heat exchangers will be enhanced. Whether this is a major or minor concern has yet to be determined and will depend on a number of technical factors, such as the materials selected for the IHX and PCHX, the diffusion coefficients associated with tritium transport through the selected materials, the chemistry of the PHTS and SHTS helium and the effectiveness of the PHTS and SHTS helium service systems in removing tritium. A further issue relates to establishing the appropriate limits on tritium transport to the product hydrogen and/or steam, which, presumably would further depend on their intended use. This future study will consider these technical and regulatory issues as input to Conceptual Design.”
				
				(Sec. 14.3, p. 14-21) – *Risk-Informed Performance-based Safety Evaluation* – “In support of the pre-application interactions with the Nuclear Regulatory Commission (NRC), leading to the U.S. design certification of the PBMR, a risk-informed and performance-based approach has been proposed as described more fully…. This proposal is a
					for most of the model development and V&V detailed in this item, ***with the following exceptions:***
					- Fission product reactions with the confinement building materials
					- Determination of the safety function of each subsystem and the level of FPT attenuation required. (Safety functions are specified, but level of FPT attenuation is not addressed.)
					- Determination of level of sensitivity to component uncertainties and how this reflects on the physical models.
					- Estimation of difficulty in obtaining the data and conducting the testing to support the safety case.
					
					• *Based on review of WEC PCDR*:
					The WEC PCDR has recognized most of the needs detailed in this item, ***with the exception of the following:***
					- Fission product reactions with the confinement building materials
					- Reactions of the reactor system components and fission products with air or steam
					- Plume models that transport the released material beyond the reactor building
					- Determination of the safety function of each subsystem and the level of FPT attenuation required.
					- Determination of level of sensitivity to component uncertainties and how this reflects on the physical models.
					- Estimation of difficulty in obtaining the data and

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			dose criteria have been identified. The set of plant features proposed to be classified as safety-related is comprised of the following: • Reactor System, including neutron control assemblies, ex-vessel neutron detectors, the reactor internals, reactor core, and fuel • Vessel System, including the ASME Section III vessels and pressure relief • RCCS, including the entire system as required for removal of residual heat • RPS, including all sensors, control logic, and housings supporting safety trips and wells which are part of the Reactor Service Building • Essential AC and DC power systems Consistent with the simple yet robust safety design approach, only a relatively modest number of systems, structures, and components (SSCs) are important in ensuring public health and safety. Equally important, these SSCs reflect the utilization of passive features. Thus, not only is susceptibility to failures in power systems, moving parts, and operator error reduced by the NGNP safety systems, but the operating staff's maintenance and ISI burdens are minimized."	proactive response to NRC policies on the expanded use of PRA methods to in the licensing process, as well as its Advanced Reactor Policy. This approach builds upon the risk-informed licensing approach that was developed by the Department of Energy (DOE) for the MHTGR in the 1980's and the more recent experience with the Exelon-proposed licensing approach for the PBMR. The approach is also consistent with the basic elements of the Technology-Neutral Framework that is under development by the NRC in support of new plant licensing….The key elements of this technology-neutral approach include: (1) the use of accident frequency vs. radiological dose criteria that are derived from current U.S. licensing requirements, referred to as Top Level Regulatory Criteria (TLRC), (2) use of a full-scope PRA to select the LBEs, (3) ***development of reactor-specific functions, selection of the corresponding safety-related Systems, Structures, and Components (SSCs), and their regulatory design criteria***, (4) ***deterministic design conditions and special treatment requirements for the safety-related SSCs***, and (5) a risk informed evaluation of defense-in-depth as described in the previous section. The relationships among these elements are described in…." (Sec. 16.2.1.1.5, p. 16-23) – *Experimental Plate-out Facility* – "The purpose of tests done in the Experimental Plate-out Test Facility (POTF) is to ***obtain representative PBMR material plate-out parameters and, if possible (to be determined by a second feasibility study), to obtain graphite dust and fission product interaction data***." (Note: This is followed with descriptions of the Experimental Plate-out Loop and the Isopiestic Plate-out facility.)	***conducting the testing to support the safety case.*** • *NGNP R&D Response*: This area of R&D is part of Fuel program. Specifics depend on design of confinement and its safety role. Most is covered or planned in Fuel program.
C-3	*Materials/Component Data* - Relevant data on materials or components over the range of interest and data uncertainties (single effects testing), including the following: • Graphite transport property and air/steam erosion data specific to the design material. • Metal alloy data specific to the design material. • Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product. • Data on helium impurities that will likely set the oxygen potential of the system, and the species to be included in an analysis. • Data associated with component aging: surface qualities of the reactor system components after many years of operation. • Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit.	See item C-6 for extensive treatment of component and system testing. See item D-1 for materials data relating to metallic materials. See item E-1 for materials data relating to graphite materials. (Sec. 7.7.1, p. 105) - ***Helium Purification Train*** – "The primary functions of the Purification Train are: • Removal of chemical and particulate contaminants from the primary coolant • Supply of purified helium to appropriate systems Since helium is used as the primary coolant, a ***helium purification system is required to provide the necessary degree of helium purity***. Oxidizing contaminants, in particular, may not exceed predetermined limits established in the specification. In detail, the helium purification system has the following functions: • Removal of particulate and gaseous contaminants from the primary coolant to maintain design values, in particular for H2O, CO, CO2, N2, H2, CH4 • Removal of tritium • Removal of other radioactive contaminants from the	See item C-2 for information relating to radionuclide mechanisms and transport modeling. See item C-6 for a description of the Initial Testing and Inspection Program. See Item D-1 for materials data relating to metallic materials. See item E-1 for materials data on graphite materials. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine. (Sec. 3.1.4.1, p. 3-70) - ***Diffusive Release Through Intact Coatings –*** "Based on previous irradiation testing and post-irradiation heating, SiC is not very retentive of Ag (and possibly other noble metals) at high temperatures. The Ag-110m ***transports through the primary cooling circuit and deposits on the cooler wetted surfaces***, which could impact operations and maintenance activities. The plateout	See item C-1 for fuels materials characterization and qualification information. See item D-1 for metallic materials characterization and qualification information. See item E-1 for graphite materials characterization and qualification information. Sec. 4.2.1.2, pp. 4-19 thru 4-22, contains a detailed description of the reactor core barrel assembly, including required functions, ***materials specifications***, methods of assembly, and interfaces with other components and systems. Sec. 4.2.1.3, pp. 4-23 thru 4-27, contains a detailed description of the reactor core ceramic structures (top, bottom, side and central reflectors), including required functions, ***materials specifications***, and interfaces with other components and systems. Sec. 4.2.1.4, pp. 4-27 thru 4-29, contains a detailed description of the Reactor Pressure Vessel, including required functions, materials specifications, and interfaces with other components and systems. (Sec. 4.2.7.2, p. 4-60) – *Helium Purification System* – "The Helium Purification System (HPURS) is ***used to provide***	• *Based on review of AREVA PCDR*: The AREVA PCDR has recognized most of the needs for materials and component data detailed in this item, ***with the exception of the following:*** ***- Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product.*** ***- Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit.*** ***- Data on surface films for long-term growth and friability, since they are relevant to FP holdup***

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	• Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions. • If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties. • Data regarding turbine or power conversion components that may have to be decontaminated prior to maintenance (initial collection of FPs while in the reactor circuit; decontamination of components; new surface state of the component after decontamination). • Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior,	helium, especially before transfer to the purified gas store (Xe, Kr, Ar) • Start up purification of the primary system before initial start up and after inspections and maintenance • Purification of newly delivered helium" (Sec. 13.3.2, p. 213) – ***Power Conversion System*** (PCS) – "No precise information on the PCS maintainability has been produced during the pre-conceptual phase and this task will be performed in the next phases of the project. However, because of the NGNP indirect cycle design, radionuclides contamination of the PCS components are expected to low to nonexistent, therefore, PCS maintainability would be similar to the standard industry practice for ***non-contaminated turbomachinery and combined cycle components***." (Sec. 19.2.1, p. 283) – ***"(Fuel) Compact*** fabrication using thermosetting resins has been developed and demonstrated on a laboratory scale. However, ***currently-available materials have not been irradiated and performance under relevant environment has not been demonstrated***. …priority of this R&D need was evaluated as "High. …The other three compact R&D needs include: 1. Testing to confirm compact pressures and temperatures in order to minimize fuel damage. 2. Development of the heat treating process to ensure complete graphitization of the matrix material. 3. Perform irradiation tests on compacts to demonstrate performance for nominal and off-nominal operating conditions." (Sec. 19.2.1, p. 283) – "…3. Irradiation testing will be required to confirm that fuel performance matches performance from the laboratory/pilot facilities." (Sec. 19.2.2, p. 284) – ***Materials Development and Qualification*** – "The materials R&D needs will focus on testing and qualification of the key materials commonly used in very high-temperature designs. The materials R&D will address the materials needed for the VHTR reactor, power conversion unit, intermediate heat exchanger (IHX), and associated balance of plant. (Sec. 19.2.2.1, p. 283) – ***Metallic Materials*** – "The materials R&D needs will focus on ***testing and qualification of the key materials*** commonly used in very high-temperature designs. The materials R&D will address the materials needed for the ***VHTR reactor, power conversion unit, intermediate heat exchanger (IHX), and associated balance of plant***."	activity is also a potential source of ***radioactivity release during hypothetical accidents involving a rapid loss of coolant***, when the shear forces during depressurization are sufficiently high to remove some of the deposited activity. Figure 3.1-64 shows the breakthrough time as a function of temperature for Ag diffusing through a 35-µm SiC layer. For temperatures above 1000°C, the breakthrough time is less than 100 days, which is well below the fuel residence time of 850 days. As discussed in Section 3.1.4, limiting the release of Ag to acceptable levels is largely accomplished through optimization of the nuclear and thermal hydraulic design of the reactor core." Section 3.6, beginning on page 3-101, presents a detailed description of the Power Conversion System. With regard to ***decontamination***, section 3.6.1.1, p. 3-101, points out that one of the key design features of the PCS is the use of electromagnetic bearings, which eliminate the possibility of ***lubricant ingress*** into the primary circuit. Section 3.6.2.2.2, p. 3-121, addresses radioactive decontamination of compressors as a potential risk issue. Section 3.6.2.3.2, p. 3-125, addresses radioactive contamination plate-out on turbine materials, causing brittleness and corrosion, as a potential risk issue. Section 3.6.2.4 addresses potential contamination of the electric generator with the following statement: "Radioactive contamination and high temperatures will be managed by enclosing the generator in a separate compartment at a pressure slightly above the rest of the PCU and with cooling to avoid subjecting the generator to undue temperatures." (Sec. 3.9.1, p. 3-189) – ***Primary Coolant Purification System*** – "This subsystem provides a means to remove circulating impurities from the primary coolant helium, and to transfer those impurities to the radioactive liquid and gas waste systems of the facility. A separate regeneration section within this subsystem is used to remove the impurities that accumulate in the purification subsystem adsorbers. The regeneration section is operated periodically under automatic control whenever regeneration is required. The primary coolant helium purification subsystem consists of two separate, independent, but identical trains of components as shown in Figure 3.9-1. All of the components that make up the trains are mechanically passive in nature; however, the adsorber elements become radioactive as the removed impurities are concentrated within the various media. Each purification train must therefore be located in a shielded vault to minimize personnel exposure to radiation. Helium purification is accomplished by routing a small side stream of helium from the primary coolant system through a series of purification components. ***These components remove the following chemical impurities: Br, I, H2O, CO, CO2, H2 (including Tritium), N2, O2, H2S, Kr, Xe, CH4, and other hydrocarbons***."	***the required degree of helium purity***. High purity coolant is required in order to minimize corrosion and contamination in the PHTS and SHTS. This is done by bleeding off a partial flow of helium from the PHTS and SHTS. The extraction point is from the highest pressure points, i.e. the PHTS and SHTS circulator discharges within the HTS. This flow is tapped off constantly during operation of the plant. The ***HPS removes chemical gaseous contaminants from the primary coolant within the PHTS*** by the use of, catalysts, adsorbers and the manipulation of helium temperature extracted from the PHTS and SHTS. The required helium purity levels will be confirmed during the conceptual design." (Sec. 16.2.1.1.5, p. 16-23) – *Experimental Plate-out Facility* – "The purpose of tests done in the Experimental Plate-out Test Facility (POTF) is to ***obtain representative PBMR material plate-out parameters and, if possible (to be determined by a second feasibility study), to obtain graphite dust and fission product interaction data***." (Note: This is followed with descriptions of the Experimental Plate-out Loop and the Isopiestic Plate-out facility.)	***during accident conditions.*** ***- If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties.*** ***- Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior.*** • *Based on review of GA PCDR*: The General Atomics PCDR has recognized most of the needs for materials and component data detailed in this item, ***with the exception of the following:*** ***- Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit.*** ***- Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions.*** ***- If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for***

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D Efforts *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
					transport properties. ***- Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior.*** • *Based on review of WEC PCDR*: The WEC PCDR has addressed some of the issues associated with this item, ***with the exception of the following:*** ***- Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product.*** ***- Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit.*** ***- Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions.*** ***- If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties.*** ***- Data regarding turbine or power conversion components that may have to be decontaminated prior to maintenance (initial collection of FPs while in the reactor circuit;***

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY

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					decontamination of components; new surface state of the component after decontamination). ***- Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior*** • *NGNP R&D Response*: Most is covered in Fuel R&D program. The following items depend on how much credit is going to be taken for plateout which is still a subject of debate: *- Data regarding transport properties sensitive to material surface conditions and chemical form of the fission product.* *- Data that will help to determine the effects of operational upsets and unusual behavior that may occur if water, oil, or some other (decontamination?) fluid is introduced into the reactor circuit.* *- Data on surface films for long-term growth and friability, since they are relevant to FP holdup during accident conditions.* *- If a component is called upon to retain FPs during an accident, it effectively becomes part of the reactor safety system, and its long-term ability to retain FPs becomes a matter of concern. If FP retention is part of the function of a component, these materials may have to undergo testing for transport properties.* *- Since FP retention is sensitive to surface state and the chemical form of the FP, some means of predicting long-term stability of this retention behavior.*

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C-4	Reactor component and confinement/containment configuration and their relative roles in the safety case • Respective roles of the reactor circuit and containment or confinement system must be known before their modeling adequacy can be determined. • Estimate of source and budgeting of FP holdup among the fuel form, reactor circuit components, mobile elements such as dust, and the reactor building, as a means of focusing components to be emphasized in analysis. • Determination of transport pathway, goals for FP retention at each step in the pathway, local (accident) operating environment at each step of the pathway.	(Sec 9.2.1, p. 136) Reactor Building (***containment/confinement***) and its functions are described in section 9.2.1, starting on p. 136. (Sec. 10, pp. 147-160) - Nuclear System (***reactor circuit***) design and operation are described in section 10, pp. 147-160. (Sec. 17.3.3, p. 238) – "Of paramount importance to the project for the timely receipt of the LWA and CP milestones is the demonstration of the safety basis for the NGNP, albeit on a preliminary level at that time. In particular, and setting the project at risk, is the ***resolution of the containment issue***. Will a pressure retaining containment structure, similar to that used in LWRs, be required for the NGNP? Or, will a confinement arrangement as proposed for the NGNP Preconceptual design be acceptable? Feeding into this debate is the reliability of the particle fuel for the HTR's safety case and the need for containment are intimately linked through fuel reliability." (Sec. 19.2.4.5, p. 293) – Other Codes/***Fission Product (FP) Transport*** – "The R&D needs of the FP Transport code include development of models for: • assessment of ***product activation in the primary circuit*** (in particular tritium and 14C), • ***radio-contamination distribution in the primary circuit***, making distinction between circulating activity, plated out / deposited activity and purification system, for both normal operation and accidental situations, • radio-contamination releases ***outside the primary pressure boundary*** and • radio-contamination releases in the environment during accident scenarios. It is also recommended to develop a mechanical analysis code for the NHS."	See item C-2 for information relating to radionuclide mechanisms and transport modeling. (Sec. 4.1, p. 4-1) – ***Reactor Building*** – "The RB for the NGNP 600-MWt reactor is classified as a ***vented low-pressure containment*** (VLPC). The RB is approximately 30-m (100-ft) wide by 50-m (165-ft) long. The RB consists of a below-grade multi-celled, embedded structure and the RCCS inlet/outlet structures, both of which are constructed of cast-in-place reinforced concrete. The degree of embedment was selected to serve a number of objectives, including reduced cost and complexity of construction, ease of operation, minimization of shielding, and good seismic performance. The below-grade location provides significant design benefits including grade level access for refueling, reduction of seismic effects, and protection from external events." (Sec. 5.1.1.2, p. 5-4) - ***Primary Coolant Pressure Boundary*** – "The fourth release barrier is the primary coolant pressure boundary. This barrier is provided by the steel pressure vessels, which will be designed and constructed to ASME Section III Division 1 requirements. The chemically inert helium coolant minimizes corrosion and eliminates the need for the complications of steel internal cladding. The entire reactor module is protected by the underground RB from external events and is conservatively designed to accommodate internal events. The helium purification train is very effective at removing long-lived fission gases and contaminates from the primary coolant. However, for short-lived fission gases, the dominant removal mechanism is radioactive decay, and for the condensable fission products, the ***dominant removal mechanism is deposition, or plateout, on the various helium-wetted surfaces in the primary circuit***." (Sec. 5.1.1.2, p. 5-4) – ***Containment*** – "The reinforced concrete, vented low-pressure containment is the fifth barrier to the release of radionuclides. It is a normally closed space, located below grade. It is equipped with a vent that opens if the pressure inside the containment exceeds its design set point, releasing mass and energy associated with a blow down and protecting the integrity of the building and the RCCS. Even if the vent opens, natural removal mechanisms (including radioactive decay, condensation, fallout, and ***plateout***) reduce the concentration of radionuclides in the containment atmosphere, reducing the offsite releases. While the vent allows the release of radionuclides released promptly, the release of associated gases early in the event eliminates the driving pressure that could transport the delayed source term out of the building. After release of the initial blow-down energy pulse, the vent is designed to close for containment of radionuclides that might diffuse out of the fuel during time-at-temperature conditions. Robust design features protect the containment function from degradation by external events. Inclusion of a broad spectrum of DBEs protects the containment function from damage by internal events."	See item A-8 for model development information. (Sec. 1.9.2.2, p. 1-31) – *Future Studies* • *PBMR Reactor Building Requirements, Functions and Features* – "***Prepare a white paper on the requirements, functions, and design features of the PBMR reactor building.*** Clearly identify all requirements, including performance, costs, investment protection and safety. Perform an analysis of the safety considerations of the PBMR designed with a confinement versus an LWR-type containment. Identify benefits and adverse consequences of selecting an LWR-type containment. Consider the citadel concept of the DPP for either alternative. • *Dust Control* - Prepare a ***white paper on dust control*** for the NGNP NHSS. Determine whether it is necessary to test the IHX for dust blockage and release during accidents. Consider similarities and differences with the DPP and its approach to dust management." (Sec. 4, pp. 4-9 thru 4-85) – *Nuclear Heat Supply System* – This section contains detailed descriptions of the ***reactor system, layout, roles of components, and supporting systems. Fuel design is addressed*** in section 4.2.1.1, p. 4-17, and throughout section 5, *Reactor Fuel*. (Sec. 10.2.1.1, p. 10-21 thru 10-23) – *Nuclear Heat Supply Building* - "Confinement requirements are identified in Section 4 with the Reactor Building HVAC system. The HVAC system and internal partitions are designed maintain separation of highly contaminated or potentially contaminated compartments from non contaminated or less contaminated compartments. ***The building is designed to prevent release of all liquids to the environment. The building is designed to withstand the pressurization effects of a major helium coolant depressurization. The building is also designed for tornado wind loads and missiles, in accordance with USNRC regulatory requirements identified in…. The potential and consequences of airplane impact are also considered. The potential for volcanic ash intrusion is addressed in the design of the NHSS HVAC system as discussed in Section 4. Fire area separation is as determined in the fire hazard analysis to be completed in the preliminary design phase and uses minimum 3 hour separation walls. This building is required to be fully functional in response to design basis events as required in Sections 4, 14 and 17. Key functions include structural integrity to maintain reactor core geometry, and confinement of radiological materials as required, to ensure safety of the public***…. The reactor pressure vessel is supported by the reactor cavity walls. The reactor cavity structure is a thick reinforced concrete cylinder. Shielding requirements primarily drive the dimensions of this interior structure….From a functional perspective, the ***confinement zone around the citadel forms an essential part of the confinement system. This confinement zone is sealed***	• *Based on review of AREVA PCDR*: The definition of roles has been included in the AREVA PCDR. The need for computer code development to understand fission product transport and distribution has been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The definition of roles has been included in the General Atomics PCDR. The need for computer code development to understand fission product transport and distribution has been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The part of this item addressing the general understanding of roles of reactor building and equipment has been addressed, and an experimental facility intended to research issues of plateout and dust are described. ***However, the central issue of budgeting and modeling fission product hold-up among specific design features has not been addressed in the WEC PCDR.*** • *NGNP R&D Response*: Safety approval issue - not R&D.

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				sufficiently to limit leakage of airborne radiological material. Leakage from the confinement zone through doors, penetrations and hatches is limited to ensure that within this space can be maintained by the HVAC system at an air pressure of less than atmospheric such that air flow is inward, and any air exchange with the environment is through filters….Adjacent cavities within the citadel are not interconnected during normal operation. In the event of a depressurization of the helium piping pressure boundary, the pressure transients are relieved via the Pressure Relief System (PRS). Burst panels within the PRS open during postulated accidents when the over-pressure in the citadel and confinement zone exceeds the set limit of the burst panels. The release goes up the depressurization shaft, through filters and is released in an upward direction." (Sec. 16.2.1.1.5, p. 16-23) – *Experimental Plate-out Facility* – "The purpose of tests done in the Experimental Plate-out Test Facility (POTF) is to ***obtain representative PBMR material plate-out parameters and, if possible (to be determined by a second feasibility study), to obtain graphite dust and fission product interaction data***." (Note: This is followed with descriptions of the Experimental Plate-out Loop and the Isopiestic Plate-out facility.)	
C-5	Computational software or other methods for determining the quantitative results • Data collection and proof that the selected model is adequate under all the normal and accident conditions of interest. Need to know that model envelops releases, and have reasonable proof that the model predicts an upper limit. • Need to have a description of the physical models and the reactor configuration, showing that the models are appropriate for the conditions of interest. • Need to have the data required for the models: single-effects data for each material and component acquired under individual testing, and integral data designed to show that the codes get the correct answer for a complete system under the conditions of interest.	See item C-2 for description of identified R&D efforts for computer codes/models. See item C-6 for description of identified R&D effort for component and integral testing.	See item A-8 for a description of Design Methods Development and Validation. See item C-6 for a description of the Initial Testing and Inspection Program. (Sec. 3, p. 3-1) – Plant Technical Description – "This Section provides a technical ***description of the entire NGNP plant***, including the nuclear systems, the Power Conversion System (PCS), the Heat Transport System (HTS), the hydrogenproduction facilities, the Helium Services System, the Plant Operation and Control System, and the balance of plant (BOP). The nuclear systems include the Reactor System, the Vessel System, the Shutdown Cooling System (SCS), the Fuel Handling System, and the Reactor Cavity Cooling System (RCCS)." (***Following subsections describe components, systems and configurations in detail.)*** (Sec. 7.2.3.1, p. 7-13) – ***Radionuclide Transport*** – "As indicated in the PPMP, there is a substantial risk that the RN transport work scope included in the AGR Plan will be inadequate to support NGNP design and licensing. This problem has been exacerbated by chronic funding shortfalls for the AGR Fuel Program; consequently, no experimental work in the RN transport area has been initiated to date with the exception that the driver fuel has been fabricated for irradiation tests AGR-3 and AGR-4. In fact, no experimental work on RN transport outside of the core is planned until FY12. The significant RN transport issues identified with the AGR Plan are summarized below. • A ***series of fission product transport tests in an in-pile loop are needed*** in order to generate the integral	See item A-8 for model development information. See Item C-6 for information on testing programs. (Sec. 4, pp. 4-9 thru 4-85) – *Nuclear Heat Supply System* – This section contains ***detailed descriptions of the reactor system, layout, roles of components, and supporting systems. Fuel design is addressed*** in section 4.2.1.1, p. 4-17, and throughout section 5, *Reactor Fuel*. (Sec. 14.3, p. 14-21) – *Risk-Informed Performance-based Safety Evaluation* – "In support of the pre-application interactions with the Nuclear Regulatory Commission (NRC), leading to the U.S. design certification of the PBMR, a risk-informed and performance-based approach has been proposed as described more fully in…. This proposal is a proactive response to NRC policies on the expanded use of PRA methods to in the licensing process as well as its Advanced Reactor Policy. This approach builds upon the risk-informed licensing approach that was developed by the Department of Energy (DOE) for the MHTGR in the 1980's and the more recent experience with the Exelon-proposed licensing approach for the PBMR. The approach is also consistent with the basic elements of the Technology-Neutral Framework that is under development by the NRC in support of new plant licensing... The key elements of this technology-neutral approach include: (1) the use of accident frequency vs. radiological dose criteria that are derived from current U.S. licensing requirements, referred to as Top Level Regulatory Criteria (TLRC), (2) use of a full-scope PRA to select the LBEs, ***(3) development of reactor-specific functions, selection of the corresponding safety-related Systems, Structures, and Components (SSCs), and their***	• *Based on review of AREVA PCDR*: Needs for computer model development and testing have been recognized in the AREVA PCDR. Reactor configuration is available in the PCDR. • *Based on review of GA PCDR*: Needs for computer model development and testing have been recognized in the General Atomics PCDR. Reactor configuration is available in the PCDR. • *Based on review of WEC PCDR*: The WEC PCDR has recognized the needs for computer model development and supporting testing. Reactor configuration is available in the PCDR. • *NGNP R&D Response*: This need is generally covered by Methods

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D Efforts *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
			test data necessary to validate the predicted source terms for the NGNP. The AGR Plan contains tasks to construct an in-pile loop and to perform an in-pile test program. However, the design and construction of the loop are not initiated until FY13. The technical feasibility of constructing such a facility (presumably in the ATR) and the associated costs and schedule must be established far earlier if the design methods for predicting RN transport in the primary circuit are to be validated before the end of NGNP final design. The cost and schedule estimates for loop design and construction appear to be very optimistic. • The AGR Plan does not address tritium transport (perhaps, in part, because it is a generic development plan which does not focus on a specific reactor design). ***Tasks to characterize tritium retention in the core and tritium permeation through heat exchanger materials*** need to be added to address NGNP DDNs. • The ***AGR Plan does not address RN transport in the VLPC***. It only includes an evaluation of the extent to which the experimental water-reactor database for radionuclide transport in high-pressure containment buildings might be applicable to the VLPC. A recent evaluation concluded that these data are of limited value for refining and independently validating the design methods used to predict radionuclide transport in VLPCs because the radionuclide concentrations and the physical and chemical forms in the two systems are too different. As a result, new DDNs have been identified that the AGR Fuel Program needs to address."	***regulatory design criteria, (4) deterministic design conditions and special treatment requirements for the safety-related SSCs***, and (5) a risk informed evaluation of defense-in-depth as described in the previous section. The relationships among these elements are described in…." (Sec. 16.2.1.5, p.16-30) – *Fuel Qualification* – "PBMR (Pty) Ltd has embarked on an intensive fuel qualification programmed to ensure that the quality of their manufactured fuel is similar to the German LEU-TRISO fuel. This program is currently underway and will ***qualify the fuel in terms of physical properties, maximum fuel temperatures, percentage burn-up and fission product release for both normal operating and accident conditions***. Statistical requirements of tests and qualification samples will also be investigated to ensure confident and safe application thereof for design and further operational specification refinements. Part of the qualification program will consist of irradiating a number of fuel spheres, containing a statistically significant number of coated particles, to full PBMR irradiation requirements in a material testing reactor." (Sec. 16.2.1.5, p. 16-30) – *Core Structural Ceramics Qualification* – "For the PBMR DPP, Core Structural Ceramics include the Reflector Graphite that establishes the core geometry, Carbon Fiber Reinforced Carbon (CFRC) components associated with the core lateral restraints and tie rods supporting the upper reflector, and ceramic components used to provide thermal insulation below the core."	program and relevant parts of Fuel program.
C-6	Integral testing over a wide range of conditions to support the development of computational methods and the quantification of the data and associated uncertainties • Attempt to use existing data from past programs to the degree appropriate. • Planning of any in-pile loop program would require a complete description of the normal operating environment and of the accidents, along with any scaling factors. Extensive modeling will be necessary to design the loop and determine off-normal conditions that the loop can be expected to simulate. Model predictions (with the previously collected single-effects data) will need to be made.	(Sec. 10.2.7, p. 154) – ***NGNP Safety Testing*** –"As a prototype for a possible fleet of Generation IV commercial nuclear power plants, the NGNP is expected to demonstrate the plant's passive and inherent safety features through a series of tests emulating various anticipated operational occurrences and design basis events. Unique operation and control strategies are envisioned such that key measures, based on safety, component tolerances, system efficiencies, etc. can be identified to define the operational envelope expected for future licensing activities related to a commercial plant while providing sufficient protection of the plant staff, the public, and the investment in the various NGNP systems, structures, and components." (Sec. 10.2.8, p. 154) – ***High Temperature Testing*** – "To characterize the performance of the processes associated with the HPPP as a function of temperature, the NGNP will be expected to provide helium temperatures in the range of 1000 – 1100 °C. To sustain such temperatures, the NGNP will provide only the power demand required by the HPPP and shutdown helium circulation in the power generation loops. This testing mode could also facilitate the study of as yet-to-be determined future missions of the NGNP plant that may require alternative components, materials, and/or fluids."	(Sec. 7.4, p. 7-24) – ***Initial Testing and Inspection Program*** – "A testing and inspection program is proposed to be carried out at the start of NGNP operations. The testing and inspection program, as currently envisioned, is expected to be performed over a period of approximately one year prior to startup and two years following startup. The general objective of the testing, beyond qualification of the facility for power operation, is to effectively compress the operating time by inducing events that would not normally be expected to occur during a two year operating period, to support the following NGNP Project objectives: • Demonstrating the basis for commercialization of the nuclear system, the hydrogen production facility, and the power conversion concept. Essential elements of this objective include: ○ Demonstrating that the requisite reliability and capacity factor can be achieved over an extended period of operation. ○ Demonstrating normal O&M activities including activities required during major outages or equipment replacement or maintenance as well as O&M that might be required in the event of major equipment failures. • Establishing the basis for licensing the commercial version of NGNP by the NRC. This will be achieved in	See item A-8 for model development information. (Sec 11.1.2, p. 11-10) – *Overview of the NGNP Plant Simulator* – "The NGNP ***Plant Simulator will provide realistic, simulated plant and control responses*** in a computing environment. The Plant Simulator aims to develop, test, verify and validate simulation models and modules as well as control algorithms and strategies for the NGNP. An interface will be provided, which will aid the testing, verification and validation of the Operational Control System (OCS). An Operator Training Simulator (OTS) will be provided, which in turn provides for the development, testing, verification and validation of training exercises for trainee operators. The Plant Simulator will also function as a constituent component of the OTS for the training and licensing of operators. Another function of the Plant Simulator will be to test and design control philosophies before plant construction and commissioning to ensure safe plant operation. Furthermore, the Plant Simulator will be able to predict the impact of modifications on the plant after commissioning, train plant operators and test their skill levels before they operate the actual plant." (Sec. 11.5.2, p. 11-30) – *Future Studies* • "Develop integrated simulation tool - A need exists for a ***comprehensive computational model of the NGNP plant***.	• *Based on review of AREVA PCDR*: Needs for testing have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: Needs for testing have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The WEC PCDR has at many levels recognized the indicated needs for testing. • *NGNP R&D Response*: Agree with NRC comment. Factored into thinking of Methods and Fuel integral testing.

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D Efforts *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		(Sec. 17.6, p. 240) – ***Initial Startup Operations and Testing*** – "The initial startup and testing is critical to the overall schedule performance of any nuclear plant. The NGNP prototype facility is no exception. As the prototype demonstration plant for the new generation of high temperature gas cooled reactors the NGNP initial startup operation and testing schedule is developed to achieve the following: • Component testing and turn-over • System functional testing and turn-over • Initial approach to criticality • Zero power operation • Power ascension including grid connection • Normal plant safety system tests (AOO tests) • Special licensing performance tests (DBA tests) • Commercial operability endurance tests • Component dismantling and examination • Fuel examination The schedule provides four years for this phase of plant operations. The first two years (2017 and 2018) is dedicated to non-nuclear system testing and turn-over including the standard system turn-over from construction to operations. The second two years (2019 and 2020) includes initial plant criticality. During this phase all safety systems will be examined and tested and several special licensing related tests is planned. This phase of the plant operation includes component dismantling and inspection and fuel examination." (Sec. 19.2, p. 281) – ***R&D Needs*** - "Fuel development and qualification, particularly ***irradiation and testing of compacts*** and mass production processes." (Sec. 19.2, p. 282) – ***R&D Needs*** – "Components testing. A large (10 MW) ***helium test loop*** is required for prototype tests of components." (Sec. 19.2.3, p. 287) – ***Circulators*** – "Circulators up to 4 MWe have already operated in HTR reactors. The test program is dedicated to component qualification during the commissioning phase rather than as an R&D task. Planned tests include: 1. Air tests of the impeller (at scale 0.2 to 0.4). 2. Helium tests of magnetic and catcher bearings. 3. Tests of the circulator shutoff valve. 4. Full scale integrated tests." (Sec. 19.2.3, p. 287) – ***IHXs*** – "The R&D inputs are based on two IHX concepts: Tubular IHX for 193 MWt power conversion and Plate IHX for 60 MWt loads for hydrogen plant loop. Small test facilities up to 1 MWt are available. Large test facilities of about 10 MWt will need to be designed and built."	major part through licensing the prototype by NRC and initiating the process for certification of the nuclear system design. The proposed testing and inspections to be performed are divided into the following categories: ***Preoperational Tests*** – These tests address the capability of selected SSCs to meet performance requirements, to the extent they can be tested outside of full plant service conditions. Successful completion of preoperational tests demonstrates that individual system performance is acceptable and the plant is ready for hot functional tests. The preoperational tests and inspections to be performed will be specified in the SSC System Design Description (SDD) documents ***Baseline In-service Inspection*** – These are pre-operational tests of all the in-service-inspections (ISI) to be performed through out the plant's lifetime. These tests provide baseline data for comparison with future in-service inspection results. ***Hot Functional Tests*** – In these tests, the nuclear heat supply facility (the reactor primary system) will be operated at full power reactor gas inlet temperature, flow, and helium pressure with heat supplied by motoring the helium compressor and IHX circulator. The tests will provide data on flow performance through out the primary system (pressures, temperatures, vibrations, etc) as well as functional testing of all monitoring instrumentation. In addition, a first check on vessel heat and temperature management and operation of the RCCS will be provided. ***Fuel Loading*** – As fuel loading progresses, neutron flux monitoring results can be compared with predictions. ***Startup Tests*** – Startup testing includes pre-critical, low power, and power ascension testing. Following verification of the core physics design, power is increased in steps to full power operation. Plant operating parameters will be verified to be within design limits, and response to load changes, transition of loads between the PCS and the hydrogen production plants and reactor trips will be demonstrated throughout the power ascension program. ***Performance Tests*** – These tests will subject the plant to less frequent events expected to occur during normal operation including power PCS trip, loss of secondary system flow or pressure, etc. ***Response to Accident Tests*** – These tests are intended to demonstrate the inherent response characteristics of the reactor module. Four basic categories of events are proposed: (1) reactivity transients, (2) pressurized cool down, (3) water ingress, and (4) depressurized cool down. These categories cover the performance of the key systems	An integrated simulation tool needs to be developed in order to set-up an integrated model of the NGNP including the NHSS, HTS, PCS, HPS and BOP systems. This tool should be able to model the performance of the actual plant in steady state mode as well as during transitions and transient events. The tool should also be to model proposed control strategies to verify the adequacy the integrated control philosophy. The model will also serve to develop plant simulations for planning of operations and for operator training. • Update modes diagram - ***The modes diagram… should be updated and expanded in the conceptual design phase***. The primary transitions and transient events should be identified together with all the required control functions. The integrated control philosophy should be evaluated and optimized during the conceptual design phase. • Simulate steady state, transitions and transient events - The above-mentioned simulation tool will be used to simulate steady state conditions, transitions and expected transients. The model should also be used to evaluate the integrated control philosophy and control functions of the different systems during the conceptual design phase to serve as input to component design in the basic and detail design phases. • Specific transitions and transient events for early investigation - Various specific transitions and transient events are identified for early investigation: • Simulate the start-up transitions in order to determine the operating conditions of the different systems and components of the NGNP during start-up. • ***Test the integrated control strategy*** to investigate interdependencies among the different systems. • Determine equipment protection requirements from above-mentioned transient analyses results. • Investigate the HPS conditioning and start-up transitions with specific focus on the operating conditions in the decomposition reactor. • Another future study is required to determine the level of detail in which the different NGNP systems need to be modeled in order to achieve accurate predictions of the actual NGNP performance within realistic computation time within the Plant Simulator." Section 16.2, *Nuclear Heat Supply System*, pages 16-18 through 16-28, contains a detailed description of the test facilities to be used for the PBMR, including: • The Pebble Bed Micro Model, a fully functional power conversion model to demonstrate concept and control. • The Heat Transfer Test Facility, which will be used o validate correlations currently used to ***model the heat transfer and fluid flow phenomena*** required for integrated simulation of the pebble bed core, via a comprehensive set of separate effects tests; and to validate different simulation methodologies applied in integrated models that represent the entire pebble bed	

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D Efforts *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
		(Sec. 19.2.3, p. 287) – ***Tubular IHX*** – "The Tubular IHX design is based on the extrapolation of past German experience. NGNP requirements lead to high temperature operation with an innovative secondary fluid mixture of helium and nitrogen. Risk D-012 identifies feasibility concerns on module size, temperature level, corrosion/nitriding, manufacturing and assembly (which are not state of the art). Tubular IHX R&D needs include: 1. Tests to confirm fabrication feasibility (tube bending, tube welding, nozzles on hot header, ISIR and assembly, etc). 2. Corrosion and nitriding tests on base and coated materials in a representative environment. 3. Fabrication of representative IHX mock-ups from thermo-hydraulic and manufacturing point of views. 4. Testing in representative helium and helium-nitrogen environments is recommended. The current plan is to use a full scale mock-up for component qualification. The need for intermediate testing on sub-scale mock-ups is deemed unnecessary provided that manufacturing issues are sufficiently addressed." (Sec. 19.2.3, p. 288) – ***Plate IHX*** – "The feasibility of the plate IHX is a concern and a reduced lifetime is expected. Primary concerns are temperature level, corrosion, manufacturing, and thermal mechanical resistance…The plate IHX R&D needs include: 1. Development of visco-plastic model (material data-base to be completed). 2. Corrosion tests on base and coated materials in a representative environment. 3. Development of manufacturing techniques (fusion welding, diffusion bonding, brazing and forming). 4. Tests on representative IHX mock-ups from both thermo-hydraulic and manufacturing point of views (diffusion bonding, brazing, ISIR). A three step approach is recommended for component qualification, these are: 1. tests in air with sub-scale mock-ups, 2. tests in helium with sub-scale mock-ups (about 1 MWt test loop). These tests will provide a basis for recommendations on which type of concept should be used for the NGNP, and 3. final qualification on a full scale mock-up (at least for the channels and the plates) on a large test facility (around 10 MWt)." (Sec. 19.2.3, p. 288) – ***Isolation Valves*** – "A hot gas isolation valve was designed during the German HTR development program and tested in the KVK test facilities. The corresponding valve was designed for operation in helium at 900 °C and is very close to what is envisioned for the VHTR. The two qualification steps are:	which provide safety and investment protection ***Post Test Inspections and Maintenance Demonstrations*** – Following the completion of the above testing at power operating conditions, a shutdown would be scheduled for performance of inspections and to demonstrate major maintenance operations. Inspections would be performed of all the systems to ascertain any abnormal effects of the above tests. Major maintenance operations would be demonstrated such as refueling, reflector replacement, performance of remote ISI operations, and removal and replacement of major equipment items such as a TM rotor, IHX heat transfer element, major hydrogen production equipment and other plant items not designed for the life of the plant. Although preliminary planning indicates that the response to ***accident testing*** will comprise only a small fraction of the total testing interval, the tests are a major element of the total program. The tests to be performed have been developed based on a preliminary evaluation, and will be adjusted based on further evaluation of design and licensing issues as the project proceeds. The ***ability to demonstrate the response to low probability events in a full scale plant without damage which would preclude subsequent long term operation is a key feature of the modular helium-cooled reactor***. Demonstrating this capability is a vital element in the successful development of a commercial plant which is economically competitive, and generally accepted by utility/users, the financial community, and general public." (Sec. 7.2.3.5, p. 7-16) - ***Design Verification & Support Programs –*** **"**The base technology for designing most MHR SSCs derives from five decades of international R&D programs combined with the design, construction, and operation of seven He-cooled reactors. Nevertheless, there are ***design-specific features of some SSCs that will require design verification by testing*** with semi-scale mockups or with actual prototypical components. Such testing is referred to herein as design verification and support (DV&S). The current NGNP and NHI technology development programs are largely generic because there is no reference NGNP design. Many fundamental design selections have yet to be made, e.g., reactor core type, IHX configuration, hydrogen production process, etc. Consequently, the current TDPs do not address DV&S DDNs to a significant degree. When the reference NGNP design is chosen, additional TDPs will need to be prepared that address the DV&S DDNs for key SSCs. It is expected that new design-specific TDPs will include plans for the Reactor System, Vessel System, RCCS, etc. Additional validation of the nuclear design methods will probably be needed for licensing the MHR design because of its annular core, which uses reflector control rods, and because of its reliance on inherent safety features in contrast to engineered safeguards. Conduct of new critical experiments, especially at elevated temperatures, will be	core, via a ***comprehensive set of integrated effects tests***. The HTTF facility will consist of a number of smaller test sections that will be used for separate effects tests and a main test section that will be used to perform integrated effects tests. The smaller test sections will consist of a scaled down pebble bed and a number of duct-type sections packed with pebbles to represent pebble bed sections with predetermined homogeneous porosities. The main test section will represent an annular pebble bed and it will have the capability to heat the pebble bed (made up of graphite pebbles) and to characterize the heat transfer behavior of such a pebble bed. It is expected that the HTTF will fulfill the needs for tests to characterize several of the main phenomena required for simulating heat transfer in a pebble bed. • The Helium Test Facility, which was designed to ***develop and test components and sub-systems of the PBMR Main Support Systems*** (MSS) and other PBMR Components. It is a risk mitigation initiative for testing these components and systems in a helium environment at high pressure and high temperature but without nuclear radiation. • The ASTRA Facility, the purpose of which is to ***perform tests in the experimental investigation of neutronics characteristics*** of a reactor with geometrical characteristics similar to the PBMR reactor. The facility represents a cylindrical side reflector consisting of graphite blocks with an octagon shaped core in the centre and a solid cylindrical centre column. The core is filled with fuel spheres and absorber spheres. Control rods, shutdown rods and a single regulating rod are situated in the first set of blocks closest to the core in the side reflector. This allows different critical configurations to check single control rod reactivity worth's or different combinations to look at interference (or shadowing) effects and permits better V&V of control rod models and methods used in the analysis tools. • The Experimental Plate-out Loop, which is based on the design of the German Laminar Loop. In the facility, a radioactive source will deliver fission products into a warm gas stream of helium at 9 bars. The stream will then pass through a tube of a material under investigation. The tube will be heated in axial section to determine the effect of temperature on fission product deposition. The gas stream will leave the tube to be cooled and filtered where after it starts with the route again. • The Isopiestic Plate-out Facility, a laboratory type set-up for the investigation of plate-out parameters in a static environment. In the facility, a radioactive source will be placed within a container filled with helium. A vacuum is created and the subsequent deposition of fission products on a required material specimen is monitored. • The Natural Convection with Corrosion Facility. The main section of this facility is made up of a vertical	

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D Efforts *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
		1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc. 2. Full scale mock-up tests in a relevant helium-nitrogen environment. These tests should cover: 1. manufacturing parameters, 2. depressurization tests, 3. pressure loss, heat loss, support tube temperature tests in a relevant helium-nitrogen environment, 4. leak tightness tests of the valve, 5. closing and opening and 6. fatigue and creep-fatigue of specific areas." (Sec. 19.2.3, p. 288) – ***Fuel Handling System*** – "Currently the Fuel Server portion of the Fuel Handling System requires the most development. The remainder of the Fuel Handling System components, including the Fuel Elevator, Adaptor Plate and Fuel Handling Machine, has been demonstrated at the Fort St. Vrain reactor. In addition, the HTTR reactor utilized a similar set of components. Due to its "Low" priority, the Fuel Server system will be designed during the program. Testing of the Fuel Server system, beyond initial component testing, will be incorporated into the Fuel Handling System development testing program." (Sec. 19.2.3, p. 289) – ***Reactor Cavity Cooling System*** - "Use of an un-insulated reactor vessel coupled with a water-cooled panel heat exchanger as a core cooling mechanism for accident conditions has not been demonstrated. The basic components of the system are fairly common and well understood. Proper design and sizing of the system will require a demonstrated understanding of key heat transfer parameters for the vessel wall and panel surfaces. Determination of the heat transfer characteristics of the proposed surfaces for the reactor vessel and the panel heat exchanger will need to be accomplished. A large scale demonstration of the capability of the RCCS to remove reactor decay heat is recommended. ..Currently there is facility available at ANL which can accommodate a large scale demonstration of the RCCS." (Sec. 19.2.3, p. 289) – ***Hot Gas Duct*** – "The reference design for the primary and secondary hot gas duct is the Vee-shaped metallic concept. This design appears to be compatible with the core expected outlet temperature, subject to demonstrating that no significant hot streaks occur. The ceramic concept will be retained as a fall back option. The hot gas duct qualification should be performed in three steps: 1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc. 2. Sub-scale mock-up tests, about 1 MWt in helium if possible, to validate fiber specification and ceramic spacer specification. 3. Full scale mock-up tests, around 10 MWt.	problematic because no test facility currently exists in the U.S. A viable option would be to perform the tests in a foreign facility."	channel of 300 mm x 300 mm and 7.5 m tall. The experimental channel is composed of sections representing a bottom reflector, sphere packing (pebble bed) and a top reflector. The experimental set-up was designed to be able to ***represent different breaks in pipes connecting to the reactor***. Breaks can be created that simulate the coaxial duct (reactor outlet pipe), the defueling chute at the bottom of the reactor and the fuelling line at the top of the reflector. By a sectional design, different core heights can also be simulated. All sections of the experimental channel and of the return pipe can be heated to accident-relevant temperatures. At different positions, the local gas compositions can be measured. • The SANA test facility, which consists of a heated pebble bed inside a furnace to simulate the thermal conditions of an HTGR-core. Different heater configurations are possible but the PCDR shows a schematic of the test facility with a single central heating element. The diameter of the pebble bed is 1.5 m and the height is 1.0 m. The overall height of the facility is 3.2 m and the maximum heating capacity of the single central heating element is 35 kW. The top and bottom of the facility are well-insulated while the outside of the furnace is open to atmosphere. ***More than 50 steady-state as well as some transient tests were carried out in the facility. In these experiments all the main parameters of a pebble bed were varied, such as pebble material, pebble diameter, gas type, heating power and heating geometry.***	

TABLE 2C – FUEL PERFORMANCE AND FISSION PRODUCT TRANSPORT AND DOSE - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D Efforts *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
		These tests should at least cover 1. depressurization tests, 2. pressure loss, heat loss, temperature of the support tube (in helium), 3. leak tightness tests of connections 4. fatigue and creep-fatigue tests (e.g., bellows, Vee-shape spacers, etc). …In the first stages of the design, tests should cover both the metallic and ceramic concepts." (Sec. 19.2.3, p. 289) – ***Instrumentation*** – "NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy. For neutron flux detectors some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired."			

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
D-1	*Physical Materials Data* - Requirements for physical aspects to be included in modeling high-temperature metallic components: • Inelastic materials behavior for materials, times, and temperatures for very high temperature structures (e.g., creep, fatigue, creep-fatigue). • Adequacy and applicability of current ASME Code allowables with respect to service times and temperatures for operational stresses. • Adequacy and applicability of current state of high-temperature design methodology (e.g., constitutive models, complex loading, failure criteria, flaw assessment methods). • Effects of product form and section thickness. • Joining methods including welding, diffusion bonding, and issues associated with dissimilar materials in structural components. • Effects of irradiation on materials strength, ductility, and toughness. • Degradation mechanisms and inspectability. • Oxidation, carburization, decarburization, and nitriding of metallic components in impure helium and helium-nitrogen. • Micro-structural stability during long-term aging in environment. • Effects of short and long term on mechanical properties (e.g., tensile, fatigue, creep, creep-fatigue, ductility, toughness). • High-velocity erosion/corrosion. • Rapid oxidation of graphite and carbon-carbon composites during air-ingress accidents. • Compatibility with heat-transfer media and reactants for hydrogen generation. • Development and stability of surface layers on RPV and core barrel affecting emissivity.	See item A-7 for data relating to air ingress phenomena. (Sec. 6.2.4.2.2, p. 75) – ***Impact and Toughness Properties*** - "Impact tests have been performed at - 20°C and 0°C on products purchased in Europe. Impact tests were also performed at other temperatures in order to determine the Charpy V transition curves. For the Charpy V at -20°C, the target in Europe was 40 J minimum for the average value of the three specimens and 28 J minimum for individual test results. These values were met for rolled and forged plates with thicknesses from 20 to 200 mm." (Sec. 6.2.4.2.3, p. 75) - ***Creep*** - "Creep test programs underway are mainly dedicated to defining the negligible creep domain. They are aimed at improving the knowledge of creep properties at moderate temperatures (< 500°C), including the effect of the post weld heat treatment. Negligible creep is also a topic which has been studied in the context of the ASME/DOE Gen IV material project (AREVA NP as the lead contractor). This work also covers creep-fatigue of mod 9Cr1Mo." (Sec. 6.2.4.2.4, p. 75) – ***Effect of Aging*** - "From available data on modified 9Cr1Mo, it can be expected that there should not be any significant aging effect below 480°C. Nevertheless thermal treatments with increasing duration up to more than 25,000 hrs at 450°C, 475°C and 500°C have been started to confirm this conclusion. Base material, heat affected zone, and weld metal samples are included in the test program. The present status after 10,000 hrs at 500°C indicates no shift in the ductile brittle transition temperature (DBTT)." (Sec. 6.2.4.2.5, p. 75) – ***Effect of Irradiation*** - "Irradiations have been carried at Joint Research Center in Petten on both base metal and weld metal (150 mm thick welded joint). No significant shifts in mechanical properties and ductile-brittle transition temperatures have been observed for the expected end-of-life fluence of the reactor pressure vessel." (Sec. 6.2.4.2.6, p. 75) – ***Corrosion in Helium Environment*** - "For temperatures below 450°C, expected carburization in impure helium environment will be a very slow process affecting only the surface layers of the vessel wall. For temperatures expected during off-normal situations (about 550°C), no problems are expected due to their short durations. A test program is however necessary to confirm the performance of mod 9Cr1Mo in representative HTR conditions." (Sec. 6.2.4.2.7, p. 76) – ***Weldability*** - "A significant R&D program has been launched to demonstrate the weldability of heavy section products. Initial tests carried out with GTAW (Gas Tungsten Arc Welding) process had shown hot	Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine. (Sec. 3.1.2.2, p. 3-16) – "Other design modifications that have been investigated include modifications to the reactor internal design to reduce bypass flow and modifications to the fuel-element design to enhance heat transfer. In addition, fuel shuffling strategies have been investigated that can reduce power peaking factors. ***These modifications can provide additional margin for fuel temperatures during normal operation, and may allow additional reduction of the coolant inlet temperature, such that SA-533/SA-508 steel (used for LWR reactor vessels) could be used for the NGNP reactor vessel***." (Sec. 3.1.4.1, p. 3-65) – ***Fuel Failure Mechanisms*** – "A number of failure mechanisms have been observed during irradiation testing and post-irradiation heating of coated-particle fuels, including pressure-vessel failure, kernel migration, and ***corrosion of the SiC layer by fission products… corrosion of the SiC layer by fission products is a key factor for determining limitations on fuel temperatures."*** (Sec. 3.2.1, p. 3-79) – ***Selection of Vessel Materials*** – "The reference GT-MHR design selected 9Cr-1Mo-V steel for the reactor vessel. Although this material was developed for Liquid Metal Fast Breeder Reactor applications, its American Society of Mechanical Engineers (ASME) code qualification had not been completed. ***GA material specialists have recommended against using 9Cr-1Mo-V steel*** for the NGNP, primarily due to expected welding difficulties and lack of manufacturing and operating experience. Although the primary coolant temperature for the NGNP is higher than that for the GT MHR, the ***alternative studies discussed in Section 3.1.2.2 indicate the reactor vessel temperatures can be maintained within limits that allow selection of a vessel material having temperature limits lower than 9Cr-1Mo-V steel***." Sec. 3.2.2, p. 3-80) – ***Reactor Vessel*** – "The ***material selected for the reactor vessel for the NGNP pre-conceptual design is 2¼Cr-1Mo steel***. As discussed in Section 3.1.2.2, a design alternative to incorporate cooling of the reactor vessel is being considered, which could potentially lower reactor vessel temperatures to a level that would ***allow use of proven light water reactor vessel materials (e.g., SA508/SA533 steel***). The reactor vessel design parameters are given in Table 3.2-1. The manufacturer of LWR vessels makes considerable use of SA508 forgings. GA has had discussions with two reactor vessel manufactures concerning NGNP vessel fabrication,	See item A-8 for information associated with model development. See item D-2 for information associated with composite materials. Sec. 4.2.1.2, pp. 4-19 thru 4-22, contains a ***detailed description of the reactor core barrel assembly, including required functions, materials specifications, methods of assembly***, and interfaces with other components and systems. Sec. 4.2.1.3, pp. 4-23 thru 4-27, contains a detailed description of the reactor core ceramic structures (top, bottom, side and central reflectors), including required functions, materials specifications, and interfaces with other components and systems. Sec. 4.2.1.4, pp. 4-27 thru 4-29, contains a ***detailed description of the Reactor Pressure Vessel, including required functions, materials specifications***, and interfaces with other components and systems. (Sec. 4.2.1.4, p. 4-28) – *Reactor Pressure Vessel (RPV)* – "The RPV is required to withstand all the normal operating conditions over the lifetime of the reactor and all the abnormal conditions for the specified number of occurrences, ***without any degradation of its ability to perform its nuclear and non-nuclear functions***. The RPV shall be designed for a nominal working pressure of 9.0 MPa, and a design pressure of 9.7 MPa. ***The RPV shall be designed and constructed to ASME III, Division I, Subsection NB and Code Case N-499-2***…. The RPV contains the RUS components and parts of the FHSS. ***The RPV is manufactured from carbon steel SA 533 Type B Class 1 for plates, SA 508 Type 3 Class 1 for forgings and SA 540 Grade B24 Class 3 for bolts.*** The RPV consists of a main cylindrical section with hemispherical upper and lower heads. The upper head is bolted to the cylindrical section and incorporates penetrations for the mechanisms of the FHSS, RCS, RSS and for instrumentation. An opening is provided in the centre of the upper head to allow access to the Core Structures for reflector replacement. The RPV has a maximum external diameter of approximately 6.8 m and its total length is approximately 30 m. The ***lower head is welded to the main cylindrical section***, and will have openings for fuel discharge, the RSS absorption spheres discharge and an access opening intended for use only during initial installation operations. The nozzle forgings of the Core Outlet Pipe and the vessel supports are attached to the lower reinforced part of the cylindrical section. This part is reinforced to withstand reactor vessel support loads. The RPV supports provide vertical as well as bottom horizontal support. Localized shielding may be attached to structures inside the RPV top part to reduce activation of the RPV head. Additional reinforcement is provided at the level of the upper attachment points for the upper seismic restraints. The nozzle forgings of the Core Inlet Pipe are attached to	• *Based on review of AREVA PCDR*: The AREVA PCDR has captured most of the needs detailed in this item, ***with the following exceptions:*** ***- Micro-structural stability during long-term aging in environment.*** ***- High-velocity erosion and corrosion.*** ***- Compatibility with heat-transfer media and reactants for hydrogen generation.*** • *Based on review of GA PCDR*: The General Atomics PCDR has captured most of the needs detailed in this item, ***with the following exceptions:*** ***- Degradation mechanisms and inspectability.*** ***- Micro-structural stability during long-term aging in environment.*** ***- High-velocity erosion and corrosion.*** ***- Development and stability of surface layers on RPV and core barrel affecting emissivity*** • *Based on review of WEC PCDR*: The needs for materials data identified in this item have been addressed in the WEC PCDR, and some of the R&D has already been performed. • *NGNP R&D Response*: All covered by Materials R&D program.

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		cracking. It was shown later on that proper selection of filler material could eliminate this problem. The welding program covered the main welding processes likely to be used, namely SAW (Submerged Arc Welding), GTAW and SMAW (Shielded Metal Arc Welding) processes. The available results are very encouraging, showing acceptable mechanical properties and no cracks. Optimization is still necessary to achieve the required impact test values for the range of post weld heat treatment temperatures investigated, in particular for SAW and SMAW processes. Further activities are envisioned on GMAW (Gas Metal Arc Welding or MIG) process, which uses a filler material similar to that used for GTAW and is suitable for automatic on-site welding in horizontal position, with a larger deposit rate compared to GTAW.” (Sec. 6.2.4.2.8, p. 76) – ***Emissivity*** – “Understanding of radiative heat transfer is of prime importance in the evaluation of the temperatures of the fuel, reactor vessel, and metallic internals, particularly during conduction cooldown situations. Measurements have been carried out to define emissivity values, not only for the Reactor Pressure Vessel material but also for the metallic and graphite internals. Tests have been carried out for the range of temperatures covering normal to off-normal situations and taking into account specimens with surface conditions representative of the RPV at the beginning and end of life.” (Sec. 6.2.4.2.9, p. 76) – ***Codes and Standards*** – “Mod 9Cr1Mo is presently covered by ASME Section III Subsection NB for temperatures below 700°F (371°C). Rules have been introduced in Subsection NH (2004 edition) to include mod 9Cr1Mo for higher temperatures. Rules are presently limited to plates and small size forgings, and revision is necessary to extend the rules to heavy section plates and forgings and extend the stress allowables to cover a 60-year design life. Other necessary Code improvements concern the definition of negligible creep conditions and the improvement of creep-fatigue design rules.” (Sec. 6.4, p. 90) “…***corrosion and nitriding*** are a concern at such high temperatures and it is recommended investigating the possibility of protecting the hottest parts of the IHX with a coating. Further R&D will be also required to confirm the material behavior at such temperatures and provide necessary information in the context the material and component qualification program.” See item A-7 for information on current state of and uncertainties associated with ***air ingress*** phenomenon. (Sec. 13.1.14, p. 210) – “The NHS module shall be designed to allow all components within the helium pressure boundary to be removed and reinstalled to ***make possible inspection***, repair and replacement. A trade study to determine the method of removal and replacement of	specifically Japan Steel Works (JSW) and DOOSAN Heavy Industries and Construction (DOOSAN). The current maximum cylindrical forging size is limited to 8.2 m diameter. ***As an alternative approach to forgings, GA material experts suggest manufacturing the reactor vessel from rolled plate***, or a combination of rolled plant and forgings. Manufacturing schemes for both the forgings (seam plan) and rolled plate designs for the reactor vessel as provided by DOOSAN are shown in Figures 3.2-2 and 3.2-3, respectively.” (Sec. 3.2.3, p. 3-82) – “***The NGNP hot duct material will be a high temperature alloy (e.g., Incoloy 800H, Hastelloy-XR, or Inconel 617***). The cross vessel is a cylindrical vessel designed and fabricated according to Section III of the ASME Code. It has an inner diameter of 2.29 m, a wall thickness of 7.62 cm, and is approximately 2.86 m in length. ***The material selected for this cross vessel for the NGNP pre-conceptual design is 2¼Cr-1Mo steel***. As discussed in Section 3.1.2.2, a design alternative to incorporate cooling of the reactor vessel is being considered, which could potentially lower reactor vessel temperatures to a level that would ***allow use of proven light water reactor vessel materials (e.g., SA508/SA533 steel***). If this alternative is selected, the cross vessel would also likely be manufactured using the same material.” (Sec. 3.2.4, p. 3-83) **– *Power Conversion Vessel* – “*The material selected for the PCS vessel for the NGNP preconceptual design is SA508/SA533 steel*.** However, if further evaluation concludes that higher temperature material is necessary, then 2¼Cr - 1Mo would be used for the PCS vessel as well as the reactor vessel. The PCS vessel has an inner diameter of 7.5 m, a wall thickness of 152 mm, and is approximately 35.2 m in height. Details of the PCS vessel are given on Table 3.2-2.” (Sec. 3.2.5, p. 3-84) – ***IHX Vessel*** – “The IHX vessel is a pressure boundary for the primary helium coolant and will be designed according to Section III of the ASME Code. ***The material selected for the IHX vessel for the NGNP preconceptual design is 2¼Cr-1Mo steel***. The IHX vessel may include a ceramic fiber insulation layer on inside surfaces to maintain operating temperatures within the material temperature limits. The vessel has an inner diameter of 3.81 m and is approximately 16 m in height.” (Sec. 5.1.1.2, p. 5-4) – ***Primary Coolant Pressure Boundary*** – “The fourth release barrier is the primary coolant pressure boundary. This barrier is provided by the steel pressure vessels, which will be designed and constructed to ASME Section III Division 1 requirements. The chemically inert helium coolant minimizes corrosion and eliminates the need for the complications of steel internal cladding. The entire reactor module is protected by the underground RB from external events and is conservatively designed to accommodate internal events. The ***helium purification train is very effective at removing long-lived***	the top part of the RPV…. The RPV will be subjected to 9.0 MPa during normal operation. The ***reactor inlet helium flow will be controlled to ensure that the RPV temperature is within the embrittlement data for the RPV material***.” (Sec. 4.4.2, p. 4-85) – *Future Studies/Metallic Structural Requirements* – “The ASME design codes used for design of the RPV and CBA are written for 40 year plant life. ***The creep data in the code needs to be extended to 60 years for the NGNP project. The effect of changes in the reactor inlet temperature, on the RPV and CBA due to operational changes in the cycle needs to be assessed***. The pressure differential over the CBA (top to bottom) caused by the inlet flow through the restraints needs to be assessed, since it adds downwards force onto the CBA support.” (Sec. 6.3.1, p. 6-35) – *Intermediate Heat Exchanger – Metallic* – “The intermediate heat exchanger (IHX) is a critical high-temperature component of the NGNP. To attain the cost and performance goals of the NGNP and related commercial process heat plants, ***a plate-type compact heat exchanger, which is characterized by thin metal cross-sections, has been selected as the initial reference for the NGNP design. The reference material for the IHX is Alloy 617 and Alloy 230 has been identified as a backup material***. Given the demanding operating conditions for the IHX, it has been concluded that a parallel development of advanced (ceramic and/or composite) heat exchangers should be pursued in parallel with the reference design. The DDNs identified in Section 6.3.1 support the reference metallic IHX concept…. ***A total of 19 DDNs have been identified for the metallic IHX*** (*Error! Reference source not found.*)***. These DDNs address materials characterization and qualification, development of methods and criteria for design and analysis and performance verification. In addition, DDNs are identified to support NGNP-specific ASME Code Cases for the IHX materials and design***.” (Note: DDNs are listed below, as they appear in Table 6.3-1.) • Establish Reference Specifications for Alloy 617 • Thermal/Physical and Mechanical Properties of Alloy 617 • Welding and As-Welded Properties of Materials of Alloy 617for Compact Heat Exchangers • Aging Effects of Alloy 617 • Environmental Effects of Impure Helium on Alloy 617 • Influence of Grain Size on Materials Properties on Alloy 617 • Establish Reference Specifications for Alloy 230 • Thermal/Physical and Mechanical Properties of Alloy 230 • Welding and As-Welded Properties of Materials of Alloy 230for Compact Heat Exchangers • Aging Effects of Alloy 230 • Environmental Effects of Impure Helium on Alloy 230 • Influence of Grain Size on Materials Properties on Alloy	

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		components within the primary pressure boundary, based on the degree of difficulty, time and cost and the projected probability of occurrence shall be completed and documented by completion of preliminary design." (Sec. 19.2, p. 281) – ***R&D Needs*** – "Materials development and qualification. This covers certain ***high-temperature steels,*** composites, and graphite selection/qualification." (Sec. 19.2, p. 282) – ***R&D Needs*** – "Power Conversion System. This work covers ***nitriding*** tests and improvement of blade performance." (Sec. 19.2.2.1, p. 284) – **Metallic Materials** – "For Mod 9Cr1Mo steel the R&D needs of "High Priority" include mechanical ***properties on heavy section products (base and weld metal), effects of aging and radiation, corrosion in helium environment, weldability, emissivity, negligible creep conditions and creep fatigue.*** A specific test program on representative ***plates and forgings (including welded joints)*** will be required for component qualification." (Sec. 19.2.2.1, p. 284) – ***Metallic Materials*** – "Mod 9Cr1Mo is covered by the ASME code up to 371ºC in Subsection NB and beyond 371ºC in Subsection NH. Subsection NH does not currently cover heavy section products and needs to be updated to cover specific aspects of Mod 9Cr1Mo. Actions have already been launched in the context of the DOE/ASME Gen IV material project to provide basis for code development. R&D efforts to support this codification should be continued. In view of past experience in gas cooled reactor, alloy 800H is a prime candidate for metallic internals operating in cold helium. Moreover, efforts are in progress to extend its coverage up to 850°C in ASME III-NH. …For 800H alloy the R&D needs include: 1. Emissivity measurement under likely representative state of surface (as machined and oxidized after machining) and 2. Corrosion behavior under representative primary helium environment. For extension of alloy 800H coverage in ASME III-NH the following items are needed: 1. Long term tests at temperature higher than 760°C, 2. Tensile tests at temperature higher than 870°C and 3. Extension to cover 60 years lifetime. Two available nickel-based super alloys (In617 and Haynes 230) have been selected as structural materials for the IHX: In617 (NiCr22Co12Mo), which has been widely studied in the early 80's for HTR application and Haynes 230 (NiCr22W14), which has been developed more recently but it exhibits better corrosion resistance. An extensive research program has been launched in France within the framework of the ANTARES program to evaluate mechanical properties, thermal stability, and corrosion resistance in the temperature range of 700 °C to 1000 °C for extended	***fission gases and contaminates from the primary coolant***. However, for short-lived fission gases, the dominant removal mechanism is radioactive decay, and for the condensable fission products, the dominant removal mechanism is deposition, or plateout, on the various helium-wetted surfaces in the primary circuit." (Sec. 7.2.1.1, p. 7-5) – ***High Temperature Materials*** – "Structural metals will be used throughout the primary coolant circuit of the NGNP, including the reactor internals, hot ducts, and heat exchangers. When the first HTGRs were designed, it was obvious that the metallic components would operate at high temperature and that some would be exposed to high neutron doses as well. The environmental aspect that was not fully anticipated until the first prototype HTGRs were operated was the extent to which the reactor primary coolant chemistry could vary. The design of the reactor metal components is based on the ASME Code with conservative reductions in Code allowables based on existing data relative to environmental effects on the various alloys. Since the early 1960s, numerous test programs and experiments have been conducted in support of metals technology for HTGRs. Extensive laboratory testing, using a range of temperatures and helium impurity levels, has been carried out in the U.S., Europe, and Japan over the past three decades to verify the performance of a variety of high-temperature materials in helium environments expected for HTGR systems. Test materials included wrought alloys such as 2¼Cr-1Mo steel, Alloy 800H, Hastelloy X, Inconel 617 (IN 617) and other metals. ***The greatest materials challenge for NGNP design will be to qualify a metal for the IHX which can operate at 950°C with a long lifetime (IN 617 is the leading candidate)***. The Japanese HTTR has an IHX made of Hastelloy XR. This IHX has been designed to operate at 950°C with a lifetime of 10 years." (Sec. 7.2.1.2, p. 7-7) – ***Hydrogen Production/SI Process*** – "The highly corrosive nature of chemical streams in the SI process has led to significant research work in the area of materials compatibility. Early screenings showed that alloys of tantalum appeared suitable, and ***current work is exploring long-term performance and corrosion resistance of materials*** stressed or machined in ways that materials of construction for larger scale plants will experience." (Sec. 7.2.3.2, p. 7-14) - ***Structural Materials R&D Program*** **– "**The objective of the NGNP Materials R&D Program [NGNP Materials Program 2005] is to provide the essential materials R&D needed to support the design and licensing of the NGNP, excluding the hydrogen plant. The most important products of the program will be qualified nuclear graphite for the reactor core and high temperature metals for use throughout the nuclear heat source, PCS, primary HTS, and balance of plant. The GA Team perspective on the graphite and metals program is briefly summarized below…***High Temperature Metals -*** The metals program described in [NGNP Materials Program 2005] is evaluating a	230 • Methods for Thermal/Fluid Modeling of Plate-Type Compact Heat Exchangers • Methods for Stress/Strain Modeling of Plate-Type Compact Heat Exchangers • Criteria for Structural Adequacy of Plate-Type Compact Heat Exchangers at Very High Temperatures • Methods for Performance Modeling of Plate-Type Compact Heat Exchangers • IHX Performance Verification • Data Supporting Materials Code Case • Data Supporting Design Code Case (Sec. 8.3.3, p. 8-43) – *Helium Environment Effects on 2-1/4Cr-1Mo* - "The primary coolant contains impurities which may cause material corrosion in the form of ***oxidation, decarburization and carburization***. At the design temperatures of the components, carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep fatigue and crack growth. ***Another factor that must be considered along with decarburization or carburization is the change in microstructure due to thermal aging***. In the tests performed to date (with pretest aging exposures up to 34,000 hr and creep tests up to 100,000 hr), no influence of testing environment was seen on the creep properties up to I% strain. Also, ***relatively minimal effect of thermal aging on the creep rupture and rupture ductility was seen at these low strain levels. Previous studies indicate the effect of NGNP primary coolant chemistry on the tensile properties of this material is negligible***. The general effect of thermal aging treatments is to decrease the yield and ultimate tensile strengths. The magnitude of the effect increases with aging temperature and time. Available information on the creep fatigue behavior of 2-1/4 Cr-1 Mo steel have indicated that this ***material tested in helium environment has improved fatigue life for all test weld forms in comparison with tests in air except for the tensile hold only tests***. Some limited test data obtained on 2-1/4 Cr-1 Mo weldments suggests that these weldments might be as strong as the base metal. Review of available data indicated that low-oxygen environments, including NGNP helium improve the continuous cycling fatigue behavior of this material over the temperature range 914°F (490°C) to 1100°F (593°C)…. ***Data are needed to confirm that exposure to impure primary coolant helium at appropriate temperature does not reduce the selected design mechanical properties of 2-1/4 Cr-1 Mo base metal and its weldments below the values of the ASME Code Subsection NH***." (Sec. 8.3.3, p. 8-46) – *Helium Environment Effects on 800H* – "The primary coolant contains impurities which may cause material corrosion in the form of ***oxidation, decarburization and carburization***. At the design temperatures of operation	

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

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		periods." (Sec. 19.2.4, p. 293) – ***Structural Mechanics*** – "The main tools for structural analysis exist, but specific modeling and correlations for NGNP geometry and materials have to be developed. This work includes: 1) incorporation of ***constitutive laws*** for materials and developing numerical models 2) seismic modeling of a block-type core 3) fluid structure interaction and flow-induced-vibration methodology, and 4) leak-before-break methodology." Sec. 19.2.5, p. 293) – Power Conversion System – "***Nitriding*** of metals will occur when exposed to ***hot nitrogen***. This nitriding process tends to embrittle metals which could lead to failures of turbine blades and pressure boundaries such as boiler tubes, gas shells, etc. The need to experimentally determine the degree of nitriding that occurs in potential PCS materials, and to quantify the effects of temperature on nitriding, has been identified. This R&D need is not only for turbomachinery, but also for IHX (Tube) and Brayton cycle gas duct."	large number of alloys for high temperature applications throughout the Reactor System, PCS, and Primary HTS. With an important exception, the planned program appears responsive to the structural metals DDNs defined herein for a prismatic NGNP, but from the GA Team's perspective it may be excessive. Because the reference NGNP design has not been chosen, the current materials R&D program is necessarily a generic program. Once the reference design is determined, the metals R&D program needs to be focused on a relatively few alloys (e.g., a prime and a backup alloy for each application). To that end, a ***comprehensive, stand-alone metals TDP should be prepared*** that defines the entire scope (test matrices, etc.), schedule, and cost of the planned program. A high-priority task will be to complete qualification of IN 617 for an IHX operating at 950°C. An important deficiency in the current metals R&D program is that it does not include turbine blade alloys. There is considerable incentive to develop and qualify a turbine-blade alloy that can be used without blade cooling at 950°C with an acceptable service life20. The turbine blade alloy R&D program should emphasize helium effects as well as thermal fatigue, and the threshold concentrations and temperatures for possible ***corrosion of turbine alloys by radionuclide plateout (Te, Cs, Ag) should be investigated***." (Sec. 7.2.3.3, p. 7-15) **- *Energy Transfer Technology Program –*** **"**The GA Team understands that an Energy Transfer TDP will be prepared [PPMP 2006]. Presumably, it will emphasize the ***design and qualification of an IHX*** capable of operating at 950°C for long life times (several decades). While some DDNs related to the IHX are generic (e.g., the materials DDNs that will be addressed by the materials R&D program), other DDNs are design specific (e.g., printed circuit vs. helical coil, etc.); consequently, a ***reference conceptual design for the IHX*** is urgently needed to provide direction and priority to the energy transfer R&D programs. This Energy Transfer TDP will also need to address DDNs related to ***process heat exchangers (hydrogen plants), piping insulation, isolation valves, and high temperature circulators."***	of the Alloy 800H components, carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep fatigue and crack growth. ***Another factor that must be considered along with decarburization or carburization is the change in microstructure due to thermal aging***. Extensive data is available on the degree of carburization/decarburization and oxidation of Alloy 800H as a function of temperature, impurity levels, and exposure time in simulated HTGR primary coolant helium. Based on these data and the given impurity levels of the NGNP, and the temperatures of operation of steam generator components, carburization of Alloy 800H is expected to be minimal over the life of the plant. The effect of impure helium environments on the creep rupture properties of Alloy 800H were evaluated for times exceeding 30,000 h. Comparative creep tests in air and in helium for the same heats of Alloy 800H were reviewed and except for two isolated data points obtained at 1400°F (760°C), all the data measured fall well within the scatter bands for air creep-rupture data. Tests were performed on commercial heats of Alloy 800H thermally aged in the temperature range of 1000° to 1500°F (538° to 816°C) for times up to 30,000 h. ***Age-hardening was observed at 1000°, 1100° and 1200°F (538°, 593° and 649°C), resulting in increases in the yield and ultimate tensile strengths of the material with some reduction in ductility***. The presence of the HTGR helium environment had no discernible effect on the stress behavior during low-cycle fatigue, high-cycle fatigue or creep-fatigue testing performed on Alloy 800H to date. The ***low-cycle fatigue life at 1200°F (650°C) was significantly increased compared with that in air.*** Fracture toughness data available to date indicate the room temperature tensile properties and CVN energy for this material aged 10,000 h at 593°C (1100°F) are significantly different than values for material aged for the same length of time at other temperatures. On the other hand, the J1c measured on material aged 10,000 h at 593°C (1100°F) is significantly lower than that measured on material aged for the same length of time at the other temperatures….***Data are needed to confirm that exposure to impure primary coolant (helium) at appropriate temperature does not reduce the selected design mechanical properties of Alloy 800H base metal and its weldments below the values of the ASME Code Subsection NH***." (Sec. 8.3.3, p. 8-74) – *Bi-metallic Weld Structural Integrity* – "The Steam Generator tube bundle design incorporates BMWs between the Alloy 800H Finishing Superheater (FSH) tubing and the 2 ¼ Cr-1Mo Evaporator/Economizer/Superheater (EES) tubing. The existing service data base for BMWs between 2 ¼ Cr-1Mo and austenitic stainless steel with nickel based fillers has been used to establish an upper limit temperature criterion for BMW operation of 480^{o} C. Data are needed to confirm the adequacy of the upper temperature limit of 480 ^{o} C and	

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				the creep-fatigue properties of the BMW and to verify the adequacy of design analysis methods…. Experience with superheater and reheater BMWs in fossil-fired power plants has indicated that at temperatures above 565°C the creep-fatigue capabilities of the weldments can be compromised by the growth of large carbides at the weld/ferritic steel interface in such weldments. ***Data are needed to confirm the adequacy of the temperature margin in the design and to confirm the fatigue strength of the welds at the lower temperatures using prototypical tubes, tube materials, weld filler materials, and operating conditions***." (Sec. 12.1.1, p. 12-11) – *Maintainability* – "Plant outages are scheduled at 5-year intervals. The ***minimum lifetime of IHX A of 10 years and the requirement of maximum 6-year intervals between maintenance and inspection on other SSC's are the main constraints in deciding on intervals between scheduled outages***. Scheduled outages will last maximum 30 days for outages scheduled after 5 years and 15 years. The 10 year outage requires the replacement of the IHX A and will last a maximum of 50 days. The 20 year outage requires the replacement of ceramic core structures and will last a maximum of 180 days. The scheduled maintenance outages are repeated in 20 year cycles." (Sec. 12.1.2, p. 12-12) – *Maintenance Requirements for Systems and Components* – "The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. ***General requirements for each system include that the design provides access to the pressure boundaries to permit in-service inspection as required by appropriate sections of the ASME B&PV Code***, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible." (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. ***Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies***." (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* -	

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				Planned: ***Corrosion/oxidation models for air ingress consequences during postulated accident conditions***. (Sec. 16.3.1, p. 16-50) – *Design Data Needs (Heat Transport Facility)* – "The final DDNs supporting the metallic IHX, HTS-01-18 and HTS-01-19, are established to ***provide the underlying database supporting NGNP-specific code cases for the IHX material and design***, respectively. There is a potential that such code cases would also be applicable to early commercial plants, pending formal implementation within the ASME Code. For ceramic/composite IHXs, six placeholder DDNs (HTS-02-01 through HTS-02-06) have been identified, as both the DDN's and the associated R&D activities will need further development during conceptual design. The first DDN provides for a review of existing technology that is potentially applicable to the development of a ceramic IHX. The anticipated result of the corresponding R&D effort will be the selection of one or more materials and/or heat exchanger technologies for further development. The second DDN specifies the need for a ***materials property database*** for the selected materials. The third DDN addresses the need for design methods, while the fourth identifies requirements for performance verification. The fifth and sixth DDNs address manufacturing technology and the development of codes and standards…. The R&D activities pertaining to DDNs HTS-01-01 through HTS-01-06 provide for the extended qualification of the current reference IHX material, Alloy 617. This extended qualification is required due to the demanding operating conditions that will be seen by the IHX, plus the small grain size that is expected to be required for compact heat exchangers as they are characterized by very thin heat transfer surface cross-sections. As described in DDN HTS-01-01 (Section 6.3.1), an initial effort is required to further develop the specification for Alloy 617 and to establish a reference for characterization. ***Included in this effort, is a review of the current database for this material, consultation with material vendors and consideration of a controlled specification variant, Alloy 617CCA, that potentially decreases the range of uncertainties with respect to properties.*** The conclusion of this effort will be procurement of materials to be used for subsequent testing." (Sec. 16.3.2, pp. 16-50 through 16-59) – This section provides a ***detailed plan for IHX R&D, including metallic and composite materials.***	
D-2	*Physical Materials Data (Composites)* - Requirements for physical aspects to be included in modeling high-temperature structural composites, such as carbon-carbon or silicon carbide–silicon carbide: • Effects of composite component selection and infiltration method. • Effects of architecture and weave. • Materials properties up to and including very high temperatures (e.g., strength,	(Sec. 13.1.14, p. 210) – "The NHS module shall be designed to allow all components within the helium pressure boundary to be removed and reinstalled to ***make possible inspection***, repair and replacement. A trade study to determine the method of removal and replacement of components within the primary pressure boundary, based on the degree of difficulty, time and cost and the projected probability of occurrence shall be completed and documented by completion of preliminary design."	The ***composite material-related concerns stated in this item are not specifically addressed in the PCDR***. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, graphite components, and the turbine. ***Appendix B to the PCDR contains the NGNP schedule. Per that schedule, composites will be codified as part of the overall ASME/ASTM codification***	(Sec. 6.3.2, pp. 6-75 thru 6-88) – *Intermediate Heat Exchanger – Ceramic/Composite* – This section contains a detailed description of the WEC approach to establishing an engineering design basis for the composite materials to be used in the IHX. The ***overall approach will be to determine appropriate materials, develop design and manufacturing methods, establish and verify characteristics through testing, and develop an ASME code case for the composite materials***. DDNs are also included in the section, covering the following:	• *Based on review of AREVA PCDR*: In general, the AREVA PCDR recognizes the needs for greater understanding of materials characteristics and behavior of composites. ***However, there is no indication that the following topics from this***

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	fracture, creep, corrosion, thermal shock resistance). • Effects of irradiation on materials strength and dimensional stability. • Fabrication scaling processes. • Adequacy and validation of design methods. • Degradation mechanisms and inspectability.	(Sec. 19.2, p. 281) – ***R&D Needs*** – “Materials development and qualification. This covers certain high-temperature steels, ***composites***, and graphite selection/qualification.” (Sec. 19.2.2.2, p. 285) - Ceramics – “The use of ***composites*** is driven by their high resistance to high or very high temperatures. An R&D program has been launched in the frame of Antares to explore the possible use of such materials inside the primary circuit. Thermal insulation, using composite materials, will be needed to provide thermal protection of metallic components which would otherwise be subjected to helium at very high temperatures….The R&D needs for applied composite materials (C/C or C/SiC composites) emphasizes qualification of ***material properties*** such as: 1. thermal-physical properties (thermal conductivity (K), coefficient of thermal expansion (CTE), heat capacity (Cp)), 2. mechanical properties including multiaxial strength, 3. fracture properties, 4. fatigue properties and 5. behavior in an oxidizing atmosphere and oxidation effects on properties. In addition, for thermal insulation, ceramic ***materials qualification*** should be for: 1. thermal-physical properties (K, CTE, Cp) and 2. behavior under oxidation. No ***control rods*** made of composites were used for past HTRs, or for other reactor concepts….Other composites such as C/SiC are also envisioned. An R&D program has been launched in the frame of Antares to explore the possibility of employing such composites for the control rods. SiC/SiC composites are not considered mature enough to meet the NGNP 2018 schedule. Additional tests for control rod ceramic materials include: 1. irradiation effects on properties including ***irradiation induced dimensional change and irradiation induced creep*** and 2. tribology.”	***effort for graphites***. The structural materials R&D program for graphites is described in section 7.2.3.2, p. 7-14. (Sec. 3.1.2, p. 3-12) – Reactor Core and Internals Design – “A control rod design using a ***carbon-carbon composite for the cladding material*** is being evaluated that would allow the in-core rods (or control rods located in the inner reflector) to be used during normal operation, which will provide greater flexibility for flattening the radial power distribution and provide some additional margin for maintaining fuel temperatures and fuel performance within acceptable limits.” (Sec. 3.1.2.2, p. 3-27) – Bypass Flow Reduction – “Fuel temperatures can be reduced by reducing bypass flow. Bypass flow is defined as any flow that bypasses the coolant holes of the fuel elements. As shown in Figure 3.1-20, bypass flow channels include gaps between fuel columns and leakage between/from PSR blocks. For the reference GT-MHR core design, approximately 3% of the flow is supplied to the control-rod channels, which have orifices to minimize bypass flow while also maintaining adequate cooling for the control rods. ***Composite-clad control rods require little or no cooling, which helps reduce the bypass flow fraction***.” (Sec. 3.1.3, p. 3-61) – Neutron Control System – “The neutron absorber material consists of B4C granules uniformly dispersed in a graphite matrix and formed into annular compacts. The boron is enriched to 90 weight percent B-10 and the compacts contain 40 weight percent B4C. The compacts have an inner diameter of 52.8 mm, an outer diameter of 82.6 mm, and are enclosed in Incoloy 800H canisters for structural support. ***Alternatively, carbon-fiber reinforced carbon (C-C) composite canisters may be used for structural support***. The control rod consists of a string of 18 canisters with sufficient mechanical flexibility to accommodate any postulated offset between elements, even during a seismic event.”	• Perform a review of existing technology • Develop a physical properties database • Develop design methods • Conduct performance verification testing • Develop manufacturing technology • Support development of an ASME III Code Case for materials and design (Sec. 12.1.1, p. 12-11) – *Maintainability* – “Plant outages are scheduled at 5-year intervals. The minimum lifetime of IHX A of 10 years and the requirement of maximum 6-year intervals between maintenance and ***inspection*** on other SSCs are the main constraints in deciding on intervals between scheduled outages. Scheduled outages will last maximum 30 days for outages scheduled after 5 years and 15 years. The 10 year outage requires the replacement of the IHX A and will last a maximum of 50 days. The 20 year outage requires the replacement of ceramic core structures and will last a maximum of 180 days. The scheduled maintenance outages are repeated in 20 year cycles.” (Sec. 12.1.2, p. 12-12) – *Maintenance requirements for Systems and Components* – “The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. General requirements for each system include that the design provides access to the pressure boundaries to permit ***in-service inspection*** as required by appropriate sections of the ASME B&PV Code, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible.” (Sec. 16.3.2, pp. 16-50 through 16-59) – This section provides a detailed plan for IHX R&D, ***including metallic and composite materials***.	***item are considered:*** ***- Effects of composite component selection and infiltration method.*** ***- Effects of architecture and weave.*** ***- Fabrication scaling processes.*** ***- Adequacy and validation of design methods.*** • *Based on review of GA PCDR*: Although a program to address issues associated with composite materials is not specifically addressed, it is included by reference as a part of the program to qualify graphites, in the General Atomics PCDR. • *Based on review of WEC PCDR*: The WEC PCDR has recognized the need for composite physical materials data. • *NGNP R&D Response*: Composites are in Materials R&D plan. The plan will address these issues.
D-3	Compromise of RPV surface emissivity due to loss of desired surface layer properties. Compromise of emissivities of in-vessel surfaces.	(Sec. 6.2.4.2.8, p. 76) – ***Emissivity*** – “Understanding of radiative heat transfer is of prime importance in the evaluation of the temperatures of the fuel, reactor vessel, and metallic internals, particularly during conduction cooldown situations. Measurements have been carried out to define emissivity values, not only for the Reactor Pressure Vessel material but also for the metallic and graphite internals. ***Tests have been carried out*** for the range of temperatures covering normal to off-normal situations and taking into account specimens with surface conditions representative of the RPV at the beginning and end of life.” (Sec. 19.2.2.1, p. 284) – Metallic Materials – “For Mod 9Cr1Mo steel the R&D needs, of “High Priority,” include mechanical properties on heavy section products (base and weld metal), effects of aging and radiation, corrosion in helium environment, weldability, ***emissivity***, negligible	(Sec. 3.1.2.2, p. 3-43) – “A 30-deg. sector ANSYS model was used to analyze both low-pressure conduction cooldown (LPCC) and high-pressure conduction cooldown (HPCC) events. In order to reduce vessel temperatures during these accidents, the reactor internal design was modified to include a 100-mm layer of carbon insulation on the outer radial boundary of the PSR…A key parameter for these calculations is the ***graphite thermal conductivity***, which decreases with damage caused by neutron irradiation. For these studies, calculations were performed using both irradiated and unirradiated graphite properties. Calculations were also performed assuming ***annealing of irradiation damage*** as the graphite temperature increases according to the GA model for H-451 graphite. Full recovery from irradiation damage is assumed to occur at temperatures greater than 1300°C. The ANSYS model shown in Figure 3.1-41 was used to calculate the effective thermal conductivity of the graphite blocks. ***Other key***	(Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – “These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. ***Emissivity testing of various surface treatments*** and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies.”	• *Based on review of AREVA PCDR*: Need for greater understanding of the surface emissivity material characteristic has been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The importance of surface emissivities, including analytical efforts performed to date, have been recognized in the General Atomics PCDR.

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		creep conditions and creep fatigue. A specific test program on representative plates and forgings (including welded joints) will be required for component qualification. ..For 800H alloy the R&D needs include: 1. ***Emissivity*** measurement under likely representative state of surface (as machined and oxidized after machining)…" (Sec. 19.2.2.3, p. 286) – Graphite Materials – "Graphite, an essential structural material for the VHTR, will operate under significant irradiation conditions and requires a characterization in the range of expected temperatures. Nuclear grade graphite was used in past HTRs programs, amassing a substantial database. These grades are no longer available. An R&D program has been launched within Antares program to select the best candidates among the new available grades or to request the development of a new grade, and to acquire design data…Nuclear graded structural graphite (PCEA, NBG17 and/or NBG18) qualification includes: 1. thermal-physical properties (K, CTE, Cp, ***emissivity***)…" (Sec. 19.2.4, p. 290) – ***Computer Codes and Methods Development and Validation –*** Included in this section are descriptions of R&D needs for computer codes addressing reactor system analysis, neutronics, thermal hydraulics/pneumatics, fuel performance, fission product transport, and structural mechanics.	***parameters that affect heat transfer to the RCCS are the emissivities of the PSR, core barrel, RPV, and RCCS panels***…" General recommendations for a high temperature metals R&D program are addressed in section 7.2.3.2, page 7-14.		• *Based on review of WEC PCDR*: The need to explore issues relating to surface emissivity has been recognized in the WEC PCDR. • *NGNP R&D Response*: Covered in Materials R&D program.
D-4	Effects on insulation • Aging fatigue and environmental degradation of insulation materials (debris plugging). • Environmental and irradiation degradation/thermal instability of fibrous insulation	***There is no indication that this item has been specifically addressed.*** However, effects of aging are addressed in section 6.2.4.2.4, and R&D that will address aging of metallic materials is addressed in section 19.2.2.1.	(Sec. 7.2.3.3, p. 7-15) **- *Energy Transfer Technology Program –* "**The GA Team understands that an Energy Transfer TDP will be prepared [PPMP 2006]. Presumably, it will emphasize the design and qualification of an IHX capable of operating at 950°C for long life times (several decades). While some DDNs related to the IHX are generic (e.g., the materials DDNs that will be addressed by the materials R&D program), other DDNs are design specific (e.g., printed circuit vs. helical coil, etc.); consequently, a reference conceptual design for the IHX is urgently needed to provide direction and priority to the energy transfer R&D programs. This Energy Transfer TDP will also need to address DDNs related to process heat exchangers (hydrogen plants), ***piping insulation***, isolation valves, and high temperature circulators." ***There is no indication that this item has been specifically addressed.***	(Sec. 6.3.4, p. 6-92) – *High Pressure Ducts and Insulation* – "High-temperature ducts and insulation are utilized within the PHTS and SHTS pressure boundary piping to direct helium flow from the reactor to IHX A (nominally at 950ºC), from IHX A to IHX B (nominally at 760oC) and from the secondary IHX A outlet to the PCHX and SG (nominally at 900ºC). ***The high-temperature ducts and insulation are contained within pressure boundary pipes of low-alloy (SA 508/533) steel that are designed to ASME Section III requirements.*** On this basis, the pressure boundary pipes must be maintained at 371ºC or less, based on Section III requirements for normal operational service conditions. The current design, in common with the PBMR-DPP high-temperature piping and ducts, is to provide both passive internal insulation and active cooling of the PHTS pressure boundary piping containing the hottest fluids (Reactor to IHX A and IHX A to IHX B). The IHX A vessel is also to be actively cooled. However, the highest temperature in the NGNP PHTS nominally exceeds that of the DPP by 50ºC and available coolant flow for active cooling is at 350ºC vs. ~120ºC in the DPP. ***These factors imply incremental challenges for the ducts and insulation***. The high temperature piping connecting the IHX A secondary outlet to the PCHX and helium mixing-chamber (nominally at 900ºC) and the piping connecting the PCHX to the helium-mixing chamber (nominally at 659ºC) will initially be evaluated with passive insulation systems only and with no active cooling. Other piping sections will operate at 350ºC or less and no DDNs have been identified for these components…. One DDN has been identified for the development of the high-temperature ducts and insulation systems. ***This DDN***	• *Based on review of AREVA PCDR*: ***There is no indication that this item has been specifically addressed. The AREVA PCDR has recognized the need for R&D regarding aging of materials, but has not addressed these specific issues on insulation.*** • *Based on review of GA PCDR*: ***There is no indication that this item has been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: The need to explore issues relating to high temperature ducts and insulation has been recognized in the WEC PCDR.

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
				addresses insulation systems, hot duct liner characterization, metallic materials selection and qualification and performance verification of the hot duct piping and other hot piping prototypes*….* The proposed approach is to experimentally verify and validate the materials and design selections associated with key high-temperature ducts and insulation, using fabricated prototypes. This ***proposed approach will include the evaluation of mechanical properties, helium environmental effects, aging effects and micro-structural changes on the metallic alloys selected; long term validation of the acceptability and continued effectiveness of the insulation systems; and long term evaluation of the prototype piping systems to maintain acceptable levels of performance under operational and off normal conditions***. One key objective is to support the potential selection of more economic designs with passive internal insulation." (Sec. 8.3.3, p. 8-56) – *Insulation Verification Test* – "***Thermal and mechanical performance of the insulation located in the active flow region (i.e., helium inlet plenum and outer shroud) of the steam generator needs to be verified***. The concerns are the ***possibility of insulation becoming loose during operation and blocking helium flow*** areas and the difficulty with accessibility for maintenance or alteration once the steam generator is installed….A considerable amount of literature is available relative to high temperature insulation physical and thermophysical properties. A variety of insulations are available in special forms to meet specific service requirements…. Physical and operational characteristics of insulation are required. Specific data needed would be relative ***to thermal cycling of fibrous insulation, effects of mechanical and acoustic vibrations, and effects of flow and thermal gradients***. These tests produce temperature data for certain critical components of the steam generator and verify the proposed thermal barrier for the life of the plant. Additional test data relative to any destructive impact on insulation due to vibrations and sliding contacting surfaces, as needed, would be obtained…. The selected design approach is to perform testing of different critical regions under simulated environment conditions. Thermal performance of the insulation can be obtained by analysis; however, analysis alone is not sufficient to assure the mechanical performance of the insulation. ***Performing the described tests is the only way of checking the mechanical performance of the insulation***." (Sec. 16.3.5, p. 16-61) – *High Temperature Ducts and Insulation* – "Materials development is not presently anticipated for the high-temperature ducts and insulation. Requirements for metallic materials are expected to be enveloped by the metallic IHX development activities described in Section 16.3.2. Insulation will be adapted from the PBMR-DPP and/or prior German experience. This assumption regarding materials development will be revisited during conceptual design in conjunction with the ducts and insulation special study outlined in Section	• *NGNP R&D Response*: Mentioned in Materials R&D plan. No current work underway.

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				6.4.2….The R&D program required to validate the performance of the high-temperature ducts and insulation comprises a series of performance and environmental tests that will subject prototypical duct and insulation components, or parts thereof, to conditions representative of those found in the HTS. The principal objective of these tests is to verify the adequacy of thermal insulation and its stability under HTS environmental conditions. ***Examples of envisioned tests include determining effective thermal conductivity of representative assemblies in helium, environmental and aging tests in helium and the effects of vibration***. At this point, large-scale flow tests are not envisioned, since the NGNP will obtain input from the PBMR-DPP, which utilizes an actively cooled design in the high-temperature sections. The latter will facilitate validation of analysis methods to be applied. The scope of the ducts and insulation R&D activities will be revisited during conceptual design in conjunction with the ducts and insulation special study outlined in Section 6.4.2.”	
D-5	Primary boundary failures in compact IHX (roles of design methods, manufacturing controls, inspection/testing).	(Sec. 11.5.2.5, p. 180) - ***IHX Failure*** – Sec. 11.5.2.5 indicates that the key factors against radiological release from the IHX will be low fuel failure rate during normal operation (resulting in low activity in the primary circuit), high purity helium provided by the Helium Purification System, and slow and limited evolution of fuel temperature during any accident (resulting in limited fuel failures during accidents).	See item C-6 for a description of the Initial Testing and Inspection Program.	See item D-1 for information on metallic IHX and item D-2 for information on composite IHX. (Sec. 12.1.1, p. 12-11) – *Maintainability* – “Plant outages are scheduled at 5-year intervals. ***The minimum lifetime of IHX A of 10 years and the requirement of maximum 6-year intervals between maintenance and inspection on other SSCs are the main constraints in deciding on intervals between scheduled outages***. Scheduled outages will last maximum 30 days for outages scheduled after 5 years and 15 years. The 10 year outage requires the replacement of the IHX A and will last a maximum of 50 days. The 20 year outage requires the replacement of ceramic core structures and will last a maximum of 180 days. The scheduled maintenance outages are repeated in 20 year cycles.” (Sec. 12.1.2, p. 12-12) – *Maintenance requirements for Systems and Components* – “The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. ***General requirements for each system include that the design provides access to the pressure boundaries to permit in-service inspection as required by appropriate sections of the ASME B&PV Code***, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible.”	• *Based on review of AREVA PCDR*: Indication in the AREVA PCDR is that this item is being addressed in the design. Also, AREVA has addressed improvement of design methods in section 19.2.4 (beginning on p. 290) and proposed a main component fabrication strategy in section 21.1.27 (p. 320). • *Based on review of GA PCDR*: The need for design verification testing of critical components such as the IHX has been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The need to explore issues relating to the IHX, including the limited lifetime of the component, has been recognized in the WEC PCDR. • *NGNP R&D Response*: Part of CTF testing of PCHE.
D-6	Control rod insertion failures (role of structural design methods for composites).	(Sec. 6.5, p. 90) – “The selection of composite materials as ***control rod*** cladding requires significant R&D actions to	TDP Sec. 3.3.1, p. 64) – Core Graphites – “The use of carbon/carbon (C/C) ***composites is proposed for several***	***Not applicable to the WEC PCDR. There is no indication that WEC intends to use composite materials for control***	• *Based on review of AREVA PCDR*:

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		qualify this component and facilitate its approval by the Regulator."	***subcomponents in the control rod assembly***. The selection was based on limited data from ORNL's work on irradiated C/C composite for fusion energy applications. ***C/C composite, therefore, needs to be further characterized by testing and its compatibility in the reactor environment needs to be assessed before it can be qualified for use in the NGNP***."	***rod system components.*** See item B-2 for information associated with the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS).	Need for control rod material qualification is recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs to further characterize, test and qualify the composite material selected for control rod assemblies have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: Not applicable to the WEC PCDR. There is no indication that WEC intends to use composite materials for control rod system components. • *NGNP R&D Response*: Part of Composites R&D plan.
D-7	Irradiation induced creep of in-vessel metallic structures.	(Sec. 19.2.2.1, P. 284) – Metallic Materials – "For Mod 9Cr1Mo steel the R&D needs, of "High Priority," include mechanical properties on heavy section products (base and weld metal), effects of aging and radiation, corrosion in helium environment, weldability, emissivity, negligible ***creep conditions*** and ***creep fatigue***. A specific test program on representative plates and forgings (including welded joints) will be required for component qualification… In617 and Haynes 230 R&D needs, of "Medium Priority," have been identified to address the following issues: 1. baseline mechanical property data, including ***creep-fatigue*** data…"	Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	See item D-1 for information on metallic IHX. (Sec. 4.4.2, p. 4-85) – *Future Studies/Metallic Structural Requirements* – "The ASME design codes used for design of the RPV and CBA are written for 40 year plant life. ***The creep data in the code needs to be extended to 60 years for the NGNP project. The effect of changes in the reactor inlet temperature, on the RPV and CBA due to operational changes in the cycle needs to be assessed.*** The pressure differential over the CBA (top to bottom) caused by the inlet flow through the restraints needs to be assessed, since it adds downwards force onto the CBA support." (Sec. 16.2.1.3, p. 16-29) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. ***Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies***."	• *Based on review of AREVA PCDR*: The need to better understand the phenomenon of creep has been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The need to better understand the phenomenon of creep has been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The need to explore materials issues relating metallic structures, including those associated with the RPV and IHX, has been recognized in the WEC PCDR. • *NGNP R&D Response*:

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
					Part of Materials R&D plan for IHX materials.
D-8	Core radial restraint failure (role of structural design and fabrication for composites).	***There is no indication that this item has been specifically addressed.***	(Sec. 3.1.2.1, p. 3-14) – "The ***lower graphite core support assembly*** consists of two layers of hexagonal elements support pedestals for the fuel and reflector columns that form the lower plenum, and the lower plenum floor, which consists of a layer of graphite elements and two layers of ceramic elements that insulate the metallic core support from the hot helium in the lower plenum. The ***upper core restraint*** elements have the same hexagonal cross sections as the graphite elements below them and are one-half the height of a standard fuel element. Dowel/socket connections are used to align the core-restraint elements with the graphite blocks. The core restraint elements are also keyed to each other and to the core barrel. The upper core restraint blocks provide stability during refueling and maintain relatively uniform and small gaps between columns during operation. The metallic core support includes a floor section and a core barrel that are welded together. The metallic core support is supported both vertically and laterally by the reactor vessel. The upper plenum shroud is a welded, continuous dome that rests on top of the core barrel to form the upper plenum. The upper plenum shroud includes penetrations for inserting control rods and reserve shutdown material, for refueling, and for core component replacement." ***There is no indication that this item has been specifically addressed.***	(Sec. 16.2.1.6, p. 16-30) – *Core Structural Ceramics Qualification* – "For the PBMR DPP, Core Structural Ceramics include the Reflector Graphite that establishes the core geometry, ***Carbon Fiber Reinforced Carbon (CFRC) components associated with the core lateral restraints and tie rods supporting the upper reflector***, and ceramic components used to provide thermal insulation below the core. A summary of the Core Structural Ceramics R&D supporting the PBMR DPP is provided in Table 16.2-3."	• *Based on review of AREVA PCDR*: ***There is no indication that this item has been specifically addressed in AREVA's PCDR.*** • *Based on review of GA PCDR*: ***There is no indication that this item has been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: The need to explore issues relating to the core restraints has been recognized in the WEC PCDR. • *NGNP R&D Response*: Should be addressed in Composites R&D plan.
D-9	Isolation and other valve failures (self-welding, galling, seizing)	(Sec. 6.5, p. 91) – ***Hot Isolation Valves*** – "Such type of component has already been qualified in the context of the former German HTR program but it will need to be checked that the environment proposed on the secondary side will not justify significant design adaptations." (Sec. 19.2.3, p. 288) – ***Isolation Valves*** – "A hot gas isolation valve was designed during the German HTR development program and tested in the KVK test facilities. The corresponding valve was designed for operation in helium at 900 °C and is very close to what is envisioned for the VHTR. The two qualification steps are: 1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc. 2. Full scale mock-up tests in a relevant helium-nitrogen environment. These tests should cover: 1. manufacturing parameters, 2. depressurization tests, 3. pressure loss, heat loss, support tube temperature tests in a relevant helium-nitrogen environment, 4. leak tightness tests of the valve, 5. closing and opening and 6. fatigue and creep-fatigue of specific areas."	(Sec. 3.7.2.1, p. 3-149) – "Secondary HTS Piping and Isolation Valves - It is expected that the secondary heat transport loop will have ***three isolation valves on each leg*** – two near the IHX and one near the PHX. Isolation valves are necessary to prevent the propagation of events in either the NGNP reactor or hydrogen production plant from affecting the other. ***Double isolation valves on the hot leg and cold leg sides of the IHX allow these isolation valves to be part of the primary coolant pressure boundary and part of the containment building boundary***. Isolation valves are also necessary to perform maintenance on the heat transport loop. Figure 3.7-5 presents a diagram of a potential high temperature isolation valve (HTIV) being developed for use on HTTR by the Japan Atomic Energy Agency (JAEA). For HTTR, a ½ scale prototype of the HTIV has been tested. The valve, as shown in Figure 3.7-5, is an angle valve with internal glass wool insulation. The rod body and seat were made of Hastelloy X and the seat had a coating metal of Stellite No. 6 and 30 wt% Cr3C2. The casing of the valve was made of carbon steel which was limited to 350°C due to the internal insulation. Testing was performed at 4.0 MPa and 900°C." (Sec. 7.2.3.3, p. 7-15) ***- Energy Transfer Technology Program –*** "The GA Team understands that an Energy Transfer TDP will be prepared [PPMP 2006]. Presumably, it will emphasize the design and qualification of an IHX capable of operating at 950°C for long life times (several	(Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a ***full scope PRA that addresses all internal and external hazards***, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. ***Testing include: Self-welding in a Helium atmosphere*** on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies."	• *Based on review of AREVA PCDR*: Need for isolation valve qualification is recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The applications and need for addressing issues associated with isolation valves have been addressed in the General Atomics PCDR. • *Based on review of WEC PCDR*: The need to explore issues relating to valve failures has been recognized in the WEC PCDR. • *NGNP R&D Response*: No current R&D planned. Part of CTF testing.

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
			decades). While some DDNs related to the IHX are generic (e.g., the materials DDNs that will be addressed by the materials R&D program), other DDNs are design specific (e.g., printed circuit vs. helical coil, etc.); consequently, a reference conceptual design for the IHX is urgently needed to provide direction and priority to the energy transfer R&D programs. This Energy Transfer TDP will also need to address DDNs related to process heat exchangers (hydrogen plants), piping insulation***, isolation valves,*** and high temperature circulators.”	(Sec. 16.2.1.8, p. 16-31) – *First Of a Kind Components R&D – **Valves: Manufacturability and performance verification in helium.***	
D-10	Initiate development of the data and models needed by ASME Boiler and Pressure Vessel (B&PV) Code Subcommittees to formulate time-dependent failure criteria that will ensure adequate life and safety for metallic materials in the NGNP. These include obtaining the data necessary to develop experimentally based constitutive models for the NGNP construction materials, which are the foundation of the inelastic design analyses specifically required by ASME B&PV Sect. III Division I Subsection NH.	See item D-1 for R&D description of ***ASME Code*** efforts and development of ***structural mechanics codes***.	See item D-1 for information relating to ASME code development, and for information relating to technology development efforts required for high temperature metals.	See item A-8 for information on model development. See items D-1 and D-2 for information on development of metallic and composite materials and design characterization and methods, ***including requirements to support new ASME Code Cases***. (Sec. 4.2.1.4, p. 4-28) – *Reactor Pressure Vessel (RPV)* – “The RPV is required to withstand all the normal operating conditions over the lifetime of the reactor and all the abnormal conditions for the specified number of occurrences, without any degradation of its ability to perform its nuclear and non-nuclear functions. The RPV shall be designed for a nominal working pressure of 9.0 MPa, and a design pressure of 9.7 MPa. The ***RPV shall be designed and constructed to ASME III, Division I, Subsection NB and Code Case N-499-2***.” (Sec. 4.4.2, p. 4-85) – *Future Studies/Metallic Structural Requirements* – “The ***ASME design codes used for design of the RPV and CBA are written for 40 year plant life. The creep data in the code needs to be extended to 60 years for the NGNP project***. The effect of changes in the reactor inlet temperature, on the RPV and CBA due to operational changes in the cycle needs to be assessed. The pressure differential over the CBA (top to bottom) caused by the inlet flow through the restraints needs to be assessed, since it adds downwards force onto the CBA support.” (Sec. 16, p. 16-13) – *Intermediate Heat Exchanger* – “The IHX is a critical component of the NGNP and a fundamental enabling technology for high-temperature process heat applications in general. The IHX requires a significant development effort, largely in the materials area, to demonstrate that a design can be developed for the high temperatures, pressures and transients expected for the NGNP. Parallel efforts are recommended that address the most promising metallic and ceramic materials. The ***results will support an ASME code case and a final design for the IHX that will need to be prototyped and tested***.” (Sec. 16.3.1, p. 16-50) – *Design Data Needs (Heat Transport Facility)* – “The final DDNs supporting the metallic IHX, HTS-01-18 and HTS-01-19, are established to ***provide the underlying database supporting NGNP-specific***	• *Based on review of AREVA PCDR*: Needs for ASME code development and supporting structural mechanics models have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for developing structural models and ASME Code qualification for high temperature metallic materials have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The need to explore issues relating to the qualification of NGNP metallics under approved ASME Code Cases has been recognized in the WEC PCDR. • *NGNP R&D Response*: Part of Materials R&D plan.

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				code cases for the IHX material and design, respectively. There is a potential that such code cases would also be applicable to early commercial plants, pending formal implementation within the ASME Code. For ceramic/composite IHXs, six placeholder DDNs (HTS-02-01 through HTS-02-06) have been identified, as both the DDN's and the associated R&D activities will need further development during conceptual design. The first DDN provides for a review of existing technology that is potentially applicable to the development of a ceramic IHX. The anticipated result of the corresponding R&D effort will be the selection of one or more materials and/or heat exchanger technologies for further development. The second DDN specifies the need for a materials property database for the selected materials. The third DDN addresses the need for design methods, while the fourth identifies requirements for performance verification. The fifth and sixth DDNs address manufacturing technology and the development of codes and standards…. The R&D activities pertaining to DDNs HTS-01-01 through HTS-01-06 provide for the extended qualification of the current reference IHX material, Alloy 617. This extended qualification is required due to the demanding operating conditions that will be seen by the IHX, plus the small grain size that is expected to be required for compact heat exchangers as they are characterized by very thin heat transfer surface cross-sections. As described in DDN HTS-01-01 (Section 6.3.1), an initial effort is required to further develop the specification for Alloy 617 and to establish a reference for characterization. Included in this effort, is a review of the current database for this material, consultation with material vendors and consideration of a controlled specification variant, Alloy 617CCA, that potentially decreases the range of uncertainties with respect to properties. The conclusion of this effort will be procurement of materials to be used for subsequent testing.”	
D-11	Safety assessments dependent on time-dependent flaw growth and the resulting leak rates from postulated pressure-boundary breaks will require a flaw assessment procedure capable of reliably predicting crack-induced failures, as well as the size and growth of the resulting opening in the pressure boundary.	See item D-1 for description of R&D efforts for ***structural mechanics codes***, including ***leak-before-break methodology***.	***There is no indication that this item has been specifically addressed.*** However, general recommendations for a high temperature metals R&D program are addressed in section 7.2.3.2, page 7-14.	See items D-1 and D-2 for information on development of metallic and composite materials and design characterization and methods. (Section 16.2.1.1.6, p. 16-26) – “NACOK stands for Natural Convection with Corrosion. The main section of this facility is made up of a vertical channel of 300 mm x 300 mm and 7.5 m tall. The experimental channel is composed of sections representing a bottom reflector, sphere packing (pebble bed) and a top reflector. The experimental set-up was designed to be able to represent different breaks in pipes connecting to the reactor. Breaks can be created that simulate the coaxial duct (reactor outlet pipe), the defueling chute at the bottom of the reactor and the fuelling line at the top of the reflector. By a sectional design, different core heights can also be simulated. All sections of the experimental channel and of the return pipe can be heated to accident-relevant temperatures. At different positions, the local gas compositions can be measured.” (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – “These tests primarily provide material property data for design input and	• *Based on review of AREVA PCDR*: Need for structural mechanics models, including flaw assessment, have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: ***There is no indication that this item has been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: The need to explore issues relating to the mechanical properties of metallic

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
				verification of the effects of the anticipated operating environment. Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies."	pressure boundary materials has been recognized in the WEC PCDR. • *NGNP R&D Response*: Unclear.
D-12	Materials data and extrapolation procedures must be developed and guidance provided to ensure that allowable operation period and range of stress and temperature for materials of construction are extended to meet the proposed operating temperatures and lifetimes. Creep-fatigue rules are an area of particular concern for the materials and temperatures of interest and must be updated and validated. (example concern: RPV long-term thermal aging)	See item D-1 information on ***materials properties, ASME Code efforts***, and development of ***structural mechanics codes.*** (Sec. 6.4, p. 89) – "Based on past experience in Germany (full scale mock up tested in the KVK helium loop) and Japan (HTTR), a high temperature tubular IHX is deemed feasible at the following conditions: • Helium/helium heat exchanger • Effectiveness 90 % • T = 850°C and with some limited periods in operation up to 950°C • Limited pressure difference in operation < 3 bars • Lifetime 20 to 30 years • IHX module power around 150 MWth. The proposed 193 MWth tubular IHXs will require an increase of the number and length of tubes which ***should be achievable through design improvements***. The extension to 900°C design temperature should be obtained by a reduction of ***design life*** to 20 years." (Sec. 6.4, p. 90) – "For the ***compact IHX*** proposed for the heat transport to H2 plant, significant R&D and design work is still required to obtain a design able to operate at 900°C (or above). Operating conditions are however less demanding (reduced pressure transients and He environment on the secondary side) and it is currently considered that such a concept can be implemented, subject to limiting the design life to 5 years. This life reduction is acceptable due to limited cost impact on the overall plant and due to the fact that the availability required on the H2 plant side should not be as large as that required on the Power Conversion side."	See item D-1 for information on efforts to develop structural models and ASME Code qualification for high temperature metallic materials. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine. The topic of creep is addressed for the reactor vessel in section 3.1.2.2, beginning on page 3-15.	See items D-1 and D-2 for information on development of metallic and composite materials and design characterization and methods, including requirements to support new ASME Code Cases. (Sec. 4.2.1.4, p. 4-28) – *Reactor Pressure Vessel* – "***The RPV is required to withstand all the normal operating conditions over the lifetime of the reactor and all the abnormal conditions for the specified number of occurrences, without any degradation of its ability to perform its nuclear and non-nuclear functions***. The RPV shall be designed for a nominal working pressure of 9.0 MPa, and a design pressure of 9.7 MPa. The RPV shall be designed and constructed to ASME III, Division I, Subsection NB and Code Case N-499-2." (Sec. 4.4.2, p. 4-85) – *Future Studies/Metallic Structural Requirements* - The ASME design codes used for design of the RPV and CBA are written for 40 year plant life. ***The creep data in the code needs to be extended to 60 years for the NGNP project.*** The effect of changes in the reactor inlet temperature, on the RPV and CBA due to operational changes in the cycle needs to be assessed. The pressure differential over the CBA (top to bottom) caused by the inlet flow through the restraints needs to be assessed, since it adds downwards force onto the CBA support. (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). ***High Temperature oxidation, creep and fatigue testing in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies***." (Sec. 16.3.1, p. 16-50) – *Design Data Needs (Heat Transport Facility)* – "The final DDNs supporting the metallic IHX, HTS-01-18 and HTS-01-19, are established to provide	• *Based on review of AREVA PCDR*: Needs for greater understanding of materials characteristics, related ASME Code efforts, scale-up of significant metal components, and development of structural mechanics codes have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for developing structural models and ASME Code qualification for high temperature metallic materials have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The need to explore issues relating to extended lifetimes and more severe conditions anticipated for metallic components has been recognized in the WEC PCDR. • *NGNP R&D Response*: Covered in Materials R&D plan.

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				the underlying database supporting NGNP-specific code cases for the IHX material and design, respectively. There is a potential that such code cases would also be applicable to early commercial plants, pending formal implementation within the ASME Code. For ceramic/composite IHXs, six placeholder DDNs (HTS-02-01 through HTS-02-06) have been identified, as both the DDNs and the associated R&D activities will need further development during conceptual design. The first DDN provides for a review of existing technology that is potentially applicable to the development of a ceramic IHX. The anticipated result of the corresponding R&D effort will be the selection of one or more materials and/or heat exchanger technologies for further development. The ***second DDN specifies the need for a materials property database for the selected materials.*** The third DDN addresses the need for design methods, while the fourth identifies requirements for performance verification. The fifth and sixth DDNs address manufacturing technology and the development of codes and standards…. The ***R&D activities pertaining to DDNs HTS-01-01 through HTS-01-06 provide for the extended qualification of the current reference IHX material***, Alloy 617. This extended qualification is required due to the demanding operating conditions that will be seen by the IHX, plus the small grain size that is expected to be required for compact heat exchangers as they are characterized by very thin heat transfer surface cross-sections. As described in DDN HTS-01-01 (Section 6.3.1), an initial effort is required to further develop the specification for Alloy 617 and to establish a reference for characterization. Included in this effort, is a review of the current database for this material, consultation with material vendors and consideration of a controlled specification variant, Alloy 617CCA, that potentially decreases the range of uncertainties with respect to properties. The conclusion of this effort will be procurement of materials to be used for subsequent testing."	
D-13	Since IHX sections must operate at the full exit temperature of the reactor, effort should be initiated to obtain data supporting the determination of the metallurgical stability and environmental resistance of IHX materials in anticipated impure helium coolant environments for the lifetimes anticipated.	(Sec. 7.7.1, p. 105) - ***Helium Purification Train*** – "The primary functions of the Purification Train are: • Removal of chemical and particulate contaminants from the primary coolant • Supply of purified helium to appropriate systems Since helium is used as the primary coolant, a ***helium purification system is required to provide the necessary degree of helium purity***. Oxidizing contaminants, in particular, may not exceed predetermined limits established in the specification. In detail, the helium purification system has the following functions: • Removal of particulate and gaseous contaminants from the primary coolant to maintain design values, in particular for H2O, CO, CO2, N2, H2, CH4 • Removal of tritium • Removal of other radioactive contaminants from the helium, especially before transfer to the purified gas store (Xe, Kr, Ar) • Start up purification of the primary system before initial start up and after inspections and maintenance	(Sec. 3.9.1, p. 3-189) – ***Primary Coolant Purification System*** – "This subsystem provides a means to remove circulating impurities from the primary coolant helium, and to transfer those impurities to the radioactive liquid and gas waste systems of the facility. A separate regeneration section within this subsystem is used to remove the impurities that accumulate in the purification subsystem adsorbers. The regeneration section is operated periodically under automatic control whenever regeneration is required. The primary coolant helium purification subsystem consists of two separate, independent, but identical trains of components as shown in Figure 3.9-1. All of the components that make up the trains are mechanically passive in nature; however, the adsorber elements become radioactive as the removed impurities are concentrated within the various media. Each purification train must therefore be located in a shielded vault to minimize personnel exposure to radiation. Helium purification is accomplished by routing a small side stream of helium from the primary coolant system through a series of purification components. ***These components remove the following chemical impurities: Br, I, H2O, CO, CO2, H2 (including***	See items D-1 and D-2 for information on development of IHX metallic and composite materials and design characterization and methods, including requirements to support new ASME Code Cases. (Sec. 4.2.7.2, p. 4-60) – *Helium Purification System* – "The ***Helium Purification System (HPURS) is used to provide the required degree of helium purity. High purity coolant is required in order to minimize corrosion and contamination in the PHTS and SHTS***. This is done by bleeding off a partial flow of helium from the PHTS and SHTS. The extraction point is from the highest pressure points, i.e. the PHTS and SHTS circulator discharges within the HTS. This flow is tapped off constantly during operation of the plant. The HPS removes chemical gaseous contaminants from the primary coolant within the PHTS by the use of, catalysts, adsorbers and the manipulation of helium temperature extracted from the PHTS and SHTS. The ***required helium purity levels will be confirmed during the conceptual design***."	• *Based on review of AREVA PCDR*: The need for high purity helium is addressed in the AREVA PCDR. • *Based on review of GA PCDR*: The need for high purity helium is addressed in the General Atomics PCDR in the design. • *Based on review of WEC PCDR*: The WEC PCDR has addressed this topic. • *NGNP R&D Response*:

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		• Purification of newly delivered helium"	***Tritium), N2, O2, H2S, Kr, Xe, CH4, and other hydrocarbons***."	(Sec. 16.3.2, pp. 16-50 through 16-59) – ***This section provides a detailed plan for IHX R&D, including metallic and composite materials***.	R&D underway in Materials R&D program.
D-14	Work should be initiated to quantify crack initiation and propagation in the IHX due to creep, creep-fatigue, and aging. These materials-related phenomena related to the IHX were identified for potentially contributing to FP release at the site boundary.	See item C-6 for component testing efforts and item D-1 for metallic materials efforts.	See item D-1 for information on efforts to develop structural models for high temperature metallic materials. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine. The topic of creep is addressed for the IHX in section 1.5.3, page 1-21.	See items D-1 and D-2 for information on development of IHX metallic and composite materials and design characterization and methods, including requirements to support new ASME Code Cases. (Sec. 16.2.1.3, p. 16-30) – *Materials R&D* – "These tests primarily provide material property data for design input and verification of the effects of the anticipated operating environment. Testing include: Self-welding in a Helium atmosphere on a range of material combinations (coatings included), various temperatures and dynamic loading conditions. Emissivity testing of various surface treatments and coatings on low alloy ferritic materials (SA 533 Grade 3) and austenitic stainless steels (Type 316). High Temperature oxidation, ***creep and fatigue testing*** in a Helium atmosphere on Ni-base alloys (Inconel 738) and Stainless steels (Type 410). Irradiation and Post Irradiation Testing on pressure vessel material (SA 533 Grade 3) at a range of temperatures and fluencies." (Sec. 16.3.2, pp. 16-50 through 16-59) – ***This section provides a detailed plan for IHX R&D, including metallic and composite materials.***	• *Based on review of AREVA PCDR*: Needs for greater understanding of materials characteristics and associated component testing have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The need to better understand the phenomenon of creep has been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: The WEC PCDR has addressed this topic. • *NGNP R&D Response*: Testing is underway in Materials R&D program.
D-15	Specific issues must be addressed for RPVs that are too large for shop fabrication and transportation. Validated procedures for on-site welding, PWHT, and inspections must be developed for the materials of construction. For vessels using materials other than those typical of LWR construction to enable operation at higher temperatures, confirmation of their fabricability (especially, effects of forging size and weldability) and data on their irradiation resistance is needed. Three materials-related phenomena related to the RPV fabrication and operation were identified for potentially contributing to FP release at the site boundary, particularly for 9Cr–1 Mo–V steels capable of higher-temperature operation: crack initiation and subcritical crack growth, process control to avoid material degradation during field fabrication, and property control in heavy sections.	Sec. 6.2.4.2.1 (p. 75) "Significant efforts have been made to perform characterizations on representative ***heavy section products***. Metallographic evaluations performed so far indicate a good homogeneity throughout the thickness. R&D actions presently underway are based on two products recently purchased: • a forged plate, 200 mm thick, supplied by Japan Steel Work • a rolled plate, 140 mm thick, supplied by Industeel. Tensile tests performed on the 200 mm forged plate in the temperature range 20°C-600°C indicated that yield strengths are higher than the ASME minimum values. Concerning ultimate tensile strength, the data obtained from the 200 mm forged plate are slightly lower than ASME values but further evaluation should be performed to clarify if the difference should be attributed to a product effect or to the definition of ultimate tensile strength (as a reminder, ASME design values should not be considered as true minima). It is also to be mentioned that actions are underway in the context of the ASME/DOE Gen IV material project (actions led by the University of Dayton Research Institute). Activities concern the update of stress allowables for mod 9 Cr1Mo, covering the effect of product form and extension of stress allowables to a 60-year design life."	(Sec. 3.2.2, p. 3-80) – ***Reactor Vessel*** – "The manufacturer of LWR vessels makes considerable use of SA508 forgings. GA has had discussions with two reactor vessel manufactures concerning NGNP vessel fabrication, specifically Japan Steel Works (JSW) and DOOSAN Heavy Industries and Construction (DOOSAN). The current maximum cylindrical forging size is limited to 8.2 m diameter. As an alternative approach to forgings, ***GA material experts suggest manufacturing the reactor vessel from rolled plate***, ***or a combination of rolled plant and forgings***. Manufacturing schemes for both the forgings (seam plan) and rolled plate designs for the reactor vessel as provided by DOOSAN are shown in Figures 3.2-2 and 3.2-3, respectively."	(Sec. 1.9.2.1, p. 1-30) – *Future Studies: Major Equipment Transportation Trade Study/Major Component Field Fabrication* – "***Detailed studies of the: (1) transportation routes and size constraints for transport of large components or sub-components such as the reactor and steam generator and (2) potential modularization for major components are recommended early in the conceptual design phase. This latter study will assess schedule and cost advantages and disadvantages of final assembly of major items at or near the site.*** These studies will influence the design of access roads and a rail spur shown on the site plan and plot plan as well as plans for modification to other roads in the vicinity of the INL site. " (Sec. 12.1.2, p. 12-12) – *Maintenance requirements for Systems and Components* – "The maintenance requirements of the main components and systems of the Nuclear Heat Supply System (NHSS), Heat Transport System (HTS), the Hydrogen Production System (HPS) and the Power Conversion System (PCS) are given in the preceding section. General requirements for each system include that the design ***provides access to the pressure boundaries to permit in-service inspection as required by appropriate sections of the ASME B&PV Code, and to allow all components to be removed and reinstalled to make inspection, repair and replacement possible.***"	• *Based on review of AREVA PCDR*: Needs for resolution of issues associated with heavy sections, materials characteristics and feasibility of using the 9Cr-1Mo alloy, and fabrication of large vessels have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The General Atomics PCDR contains options for fabricating the reactor vessel, including forging or welding together sections of rolled plate. ***There is no indication that the need to research issues relating to vessels too large for shop fabrication has been***

TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		(Sec. 6.4, p. 90) – "Welding of mod 9Cr-1Mo is also an issue but weldability actions carried out by AREVA in the past few years indicate that welding of heavy section products should be fully achievable (even though optimization of welding products and welding parameters is still required)." (Sec. 21.1.7, p. 317) – ***Confirm Selection of 9Cr-1Mo RPV Material*** – "Modified 9Cr-1Mo steel provides significant performance advantages for the reactor pressure vessel material including high temperature capability and improved irradiation resistance compared to SA508. However, 9Cr-1Mo is not an established reactor vessel material, and its use will require development in terms of procurement, fabrication, qualification, and code acceptance. Therefore, a more detailed study should be planned and implemented to amplify, refine, and elaborate the factors in the assessment and selection of 9Cr-1Mo steel for the primary pressure vessels (e.g., forging, fabrication, procurement, codification). This study must distinguish perception from reality regarding the fabrication difficulties associated with 9Cr-1Mo. Attention must be given to the relative schedule risks associated with 9Cr-1Mo compared to SA508 for HTR applications against the relative associated performance advantages." (Sec. 21.1.27, p. 320) - **Main Component Fabrication Strategy – "**In parallel with the INL site heavy component transportation issues study, a fabrication strategy for main components should be developed. This study should include identification of potential suppliers, assessments of on-site versus off-site fabrication issues, and comparison of relative costs."			***specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: The need to explore issues relating to possible RPV fabrication activities has been recognized in the WEC PCDR. • *NGNP R&D Response*: 9Cr-1 Mo is not primary candidate for RPV. SA 503/533 is primary candidate. These issues are well know for 9Cr-1 Mo.
D-16	For high-temperature metals technology, there is a need for analytical models, in particular for developing time-dependent design criteria for complex structures, along with verification by structural testing. ASME Code-approved simplified methods have not yet been proven and are not permitted for compact IHX components. Analytical modeling of carbon-carbon composite behavior would be useful in developing approved methods for designing, proof testing, model standard testing, validation tests, and probabilistic methods of design. Scalability and fabrication issues must be addressed, including large-scale structures (meters in diameter), as well as smaller structures.	(Sec. 6.4, p. 90) – "For the ***compact IHX*** proposed for the heat transport to H2 plant, significant R&D and design work is still required to obtain a design able to operate at 900°C (or above). Operating conditions are however less demanding (reduced pressure transients and He environment on the secondary side) and it is currently considered that such a concept can be implemented, subject to limiting the design life to 5 years. This life reduction is acceptable due to limited cost impact on the overall plant and due to the fact that the availability required on the H2 plant side should not be as large as that required on the Power Conversion side." See item D-15 for information on ***main component fabrication strategy***. See item D-1 for R&D description of ***ASME Code*** efforts, and needs for ***structural mechanics codes.***	See item A-8 for information relating to development of analytical models. See item D-1 for information relating to ASME code approval of high temperature metals.	See item A-8 for information on model development. See items D-1 and D-2 for information on development of IHX metallic and composite materials and design characterization and methods, including requirements to support new ASME Code Cases. (Sec. 16.3.2, pp. 16-50 through 16-59) – ***This section provides a detailed plan for IHX R&D, including metallic and composite materials.***	• *Based on review of AREVA PCDR*: Needs for improved high temperature metals technology, ASME code-approved materials designations, structural mechanics codes describing materials behavior and characteristics, and resolution of large scale fabrication strategies have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for analytical models and ASME Code qualification of high temperature metals have been recognized in the General Atomics PCDR. • *Based on review of*

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
TABLE 2D - HIGH TEMPERATURE MATERIALS (METALLIC) - SUMMARY					
					WEC PCDR: The needs to explore issues relating to establishing mechanical properties, design methods, and supporting new ASME Code Cases for metallic components have been recognized in the WEC PCDR. • *NGNP R&D Response*: Part of Materials R&D program.

TABLE 2E – GRAPHITE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
E-1	Lack of confirmatory data for the grades of graphite selected by potential NGNP vendors. This situation has occurred because: • Graphite grades used in prior HTGRs are no longer available, and thus development of new grades has been required. • Increased temperature of the NGNP compared to prior graphite-moderated reactors. • In the case of the PBR, the larger neutron dose that the core components will experience compared to that of previous HTGRs licensed in the United States.	(Sec. 19.2, p. 281) – ***R&D Needs*** – "Materials development and qualification. This covers certain high-temperature steels, composites, and ***graphite*** selection/qualification." (Sec. 19.2.2.3, p. 286) - ***Graphite Materials*** – "Graphite, an essential structural material for the VHTR, will operate under significant irradiation conditions and requires a characterization in the range of expected temperatures. Nuclear grade graphite was used in past HTRs programs, amassing a substantial database. ***These grades are no longer available***. An R&D program has been launched within Antares program to select the best candidates among the new available grades or to request the development of a new grade, and to acquire design data. Nuclear graded structural graphite (PCEA, NBG17 and/or NBG18) qualification includes: 1. thermal-physical properties (K, CTE, Cp, emissivity), 2. mechanical properties including multiaxial strength, 3. fracture properties, 4. fatigue properties, 5. irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep, 6. behavior under oxidized atmosphere including oxidation effects on properties and 7. tribology. Due to schedule limits, it is recommended that graphite R&D be performed in two phases: preliminary and detailed… Development of ASME and ASTM codes and standards for graphite is essential for timely application graphite for NGNP reactor." The portion of this item that is specific to the PBR is not applicable to the PMR.	(Sec. 7.2.1.1, p. 7-5) – ***High Temperature Materials*** – "The design of the NGNP graphite components is based on a considerable international body of graphite data. In the early 1970's, a near-isotropic, petroleum coke based graphite, designated Grade H-451, was developed by Great Lakes Carbon, and numerous test programs and experiments were conducted to characterize its behavior. H-451 was used successfully in FSV reloads, and it was the reference fuel element graphite for the NP-MHTGR. Unfortunately, this graphite is ***no longer commercially available***, and a ***priority task for the NGNP technology program is to identify and qualify a replacement having comparable properties***. The component models and material property data for designing graphite components are documented and controlled in the GA Graphite Design Data Manual. It is planned to use these data in the conceptual design (and perhaps preliminary design) of the NGNP core until a replacement for H-451 graphite is characterized." (Sec. 7.2.3.2, p. 7-15) - ***Structural Materials R&D Program – "***The objective of the NGNP Materials R&D Program [NGNP Materials Program 2005] is to provide the essential materials R&D needed to support the design and licensing of the NGNP, excluding the hydrogen plant. The most important products of the program will be qualified nuclear graphite for the reactor core and high temperature metals for use throughout the nuclear heat source, PCS, primary HTS, and balance of plant. The GA Team perspective on the graphite and metals program is briefly summarized below…***Graphite Program -*** The graphite program described in [NGNP Materials Program 2005] is evaluating at least 16 nuclear graphites and fuel-element matrix materials from at least four international graphite vendors. The current focus of the program is the graphite irradiation capsule AGC-1 which is intended to provide irradiation creep design and dimensional change data on candidate graphites for the NGNP program. Creep data will be obtained for six major graphite grades: H-451 and IG-110, both of which are included as reference graphites, and four new grades, PCEA, NBG-17, NBG-18, and IG-430. In addition, AGC-1 contains ten minor grades of graphite. A comprehensive, stand-alone graphite TDP is needed to define the entire scope, schedule and cost of the planned program. The planned program is probably responsive to the graphite DDNs defined herein for a prismatic NGNP, but from the GA Team's perspective, it may be excessive. The graphite service conditions in a prismatic VHTR are not demanding (e.g., fast neutron fluence to the fuel element <5 x 1021 n/cm2, E >0.18 Mev). Previously qualified H-451 for fuel and reflector elements and Stackpole 2020 for the core support structure have adequate material properties. The primary need is to identify and qualify a replacement for H-451. The recommended approach is to use AGC 1 as a screening capsule to identify the lowest-cost graphite with properties comparable to H-451 and then to perform supplemental testing to establish a correspondence between the behavior of the replacement graphite and the extensive H-451 experience base. The GA Team considers	Sec. 4.2.1.2, pp. 4-19 thru 4-22, contains a detailed description of the reactor core barrel assembly, including required functions, materials specifications, methods of assembly, and interfaces with other components and systems. ***Sec. 4.2.1.3, pp. 4-23 thru 4-27, contains a detailed description of the reactor core ceramic structures*** (top, bottom, side and central reflectors), including required functions, materials specifications, and interfaces with other components and systems. Sec. 4.2.1.4, pp. 4-27 thru 4-29, contains a detailed description of the Reactor Pressure Vessel, including required functions, materials specifications, and interfaces with other components and systems. (Sec. 4.3.1, p. 4-77) – *Reactor Graphite/Core Structure Ceramics* – "The PBMR NGNP Core Structure Ceramics (CSC) comprise the non-metallic components enclosed within the core barrel and its underlying support structure, plus the additional non-metallic components that form and support the top reflector assembly. The components of the CSC are principally manufactured from graphite. Certain components of the CSC, specifically the ***Reflector Graphite components that are adjacent to or near the core, operate in a harsh environment where they are subjected to high neutron fluences at high temperatures.*** The reliable operation of the CSC is important for the continued safe and economic operation of the plant as a whole. Thus, the qualification of the CSC, and particularly the Reflector Graphite, for the relevant conditions is of importance…. The physical design of the CSC for the PMBR NGNP is essentially identical to that of the PBMR Demonstration Power Plant (DPP); however, the operating conditions vary, as the PBMR NGNP has both a lower reactor inlet temperature and high reactor outlet temperature. On that basis, the DDNs for the PBMR NGNP are incremental to those already addressed within the PBMR-Specific Materials Test Reactor Program (PSMP)….To accommodate the expanded operating range of the PBMR NGNP, two DDNs have been identified for the Reflector Graphite, addressing the low and high temperature regimes, respectively. These DDNs provide for acquiring the data necessary to achieve a comparable level of qualification for the PBMR NGNP as is presently planned for the PBMR DPP." (Sec. 16.2.1.6, p. 16-30) – *Core Structural Ceramics R&D* – Completed program under the PBMR-Specific Materials Test Reactor Program to conduct supplemental irradiations of NBG-18 to ***verify consistency with the established database for similar graphites***. (Section 16.2.1.6, p. 16-31) The PBMR-Specific Materials Test Reactor Program (PSMP) is structured to provide data supporting startup and initial operation of the PBMR to the first planned outage period (6 years) as input to the licensing process. Completion of the PSMP is planned for the	• *Based on review of AREVA PCDR*: Needs for development of updated, code-approved graphite materials designations have been recognized in the AREVA PCDR, and some of the R&D has been performed. • *Based on review of GA PCDR*: The need for ASME Code qualification of graphites has been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed. • *NGNP R&D Response*: These data are key outputs from Graphite program.

TABLE 2E – GRAPHITE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
			the qualification of a replacement graphite for H-451 to be a high priority, but a low risk task." The portion of this item that is specific to the PBR is not applicable to the PMR.	timeframe of plant startup in 2012. The results of the PSMP will be confirmed through surveillance, testing, inspection and maintenance activities over the plant operating lifetime. ***R&D programs relating to qualification of core structural ceramics are as follows:*** • Complete: Reflector graphite specification (NBG-18 graphite). • Complete: Reflector graphite manufacturing process. • Complete: Reflector graphite QA process. • Complete: Characterization of reflector graphite unirradiated properties. • In-progress: Conduct supplemental irradiations of NBG-18 to verify consistency with the established database for similar graphites. • Planned: Perform component-specific tests to characterize and quality carbon fiber reinforced carbon (CFRC) components. • Planned: Characterization of insulation materials, including, where appropriate, irradiation to modest fluence levels. (Sec. 16.2.1.10, p. 16-34) – *Engineering Design Tools R&D* – Table 16.2-7 describes planned and in-progress ***efforts for developing irradiated material behavior models (e.g. graphite blocks); graphite corrosion/oxidation models for air ingress models; and discrete element modeling to simulate behavior of graphite bodies in contact with each other*** (e.g. block reflector structures and fuel spheres). (Section 16.2.2, p. 16-35) - *Design Data Needs (Nuclear Heat Supply System)* – "Three DDNs have been identified pertaining to the NGNP Fuel. The first of these DDNs (NHSS-01-01) identifies the need for data to extend the irradiated fuels qualification database from the temperature-burnup envelope of the PBMR Demonstration Power Plant (DPP) to that of the PBMR NGNP. The second DDN (NHSS-01-02) specifies data to correspondingly extend the heat up data pertaining to accident conditions. The third DDN (NHSS-01-03) provides for an extension of the temperature-fluence envelope of the Fuel Graphite to that required by the NGNP. In all three cases, the extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature. The corresponding R&D for the fuel itself comprises irradiation of fuel samples at the higher temperature applicable to the NGNP, post irradiation examination and subsequent heat up of some samples to simulate accident conditions, plus corresponding modeling and analysis. For the Fuel Graphite, the R&D comprises the irradiation of graphite spheres at a temperature and to a fluence level applicable to the NGNP, plus post-irradiation examination and analysis. ***Two DDNs (NHSS-02-01 and NHSS-02-02) have been identified to extend the irradiated materials qualification database for Reflector***	

TABLE 2E – GRAPHITE - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
				Graphite from the temperature-fluence envelope of the PBMR DPP to that of the PBMR NGNP. The extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature. The corresponding R&D comprises ***irradiation of graphite samples at low and high temperatures, plus post-irradiation examination and analysis***." (Sec. 16.2.4, p. 16-42) - *Core Structural Ceramics Reflector Graphite R&D* – "In parallel, the NGNP Program at INL is embarking on a graphite development effort that addresses multiple product forms (including NBG-18) and applications (including the PBMR). The INL program places particular emphasis on the understanding of fundamental graphite characteristics that would, ideally, allow the characterization of new coke and/or graphite sources without the need for an extensive irradiation program. To the extent that the INL program addresses NBG-18 and that manufacturing and QA systems development are generic, there is a potential to accelerate the INL effort and reduce its cost by utilizing applicable results of the PBMR DPP development work that would otherwise be duplicated. From the PBMR perspective, there is a potential to ***expand the database supporting NBG-18*** and, potentially, to reduce the scope of surveillance, testing, inspection and maintenance (STIM) required as a basis for operation of the PBMR DPP. Further potential benefits are access to multiple qualified vendors for follow-on PBMR commercial deployments and easing the burden associated with qualification of new graphite sources. In order to take mutual advantage of PBMR's ongoing program to qualify SGL graphite plus INL's and PBMR's mutual interests to cooperate on graphite qualification with SGL and Graftek, efforts are underway to develop a collaborative program. In the interim, a preliminary scope, cost and schedule for R&D activities addressing the Reflector Graphite DDNs for the PBMR NGNP have been developed."	
E-2	Lack of consensus codes and standards. Efforts are under way through the ASME to develop a consensus design code for graphite core components, but to date a useable code has not been approved. ASTM test standards exist for many of the physical properties of concern to the reactor designer, but further work is required, especially in the area of small (irradiation) specimen test methods.	(Sec. 6.1.2.3, p. 62) – Design Code – "Rules for nonmetallic materials are presently under preparation in the context of the ***ASME Subgroup on Graphite Core Components***." (Sec. 6.4, p. 90) – "There is no feasibility issue associated to the mechanical design of graphite core components. Feasibility lies more on the availability of material properties of the new grades envisioned for VHTR design (in particular properties of irradiated material) and on the ***availability of design rules approved by ASME Code Committee and by the Regulator***."	See item E-1 for information relating to ASME Code qualification of graphites.	See item E-1 for materials characterization and qualification efforts. (Sec. 16.2.3.1, p. 16-38) – *Fuel Graphite Irradiation Tests* - "Samples for investigation and irradiation will be cut from pressed graphite spheres provided for the test. These samples will be cut parallel and perpendicular to the extrusion direction. Following irradiation, the following characteristics will be measured: • ***Geometrical size*** • ***Mass*** • ***Calculation of sample density*** • ***Measurement of sample density*** • ***Sample porosity*** • ***Thermal conductivity in the range 20 up to Irradiation Temperature*** • ***Electric conductivity in the range 20 up to***	• *Based on review of AREVA PCDR*: Needs for development of approved ASME codes for graphite have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The need for ASME Code qualification of graphites has been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*:

TABLE 2E – GRAPHITE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				Irradiation Temperature • ***Thermal coefficient of linear expansion in the range 20 up to Irradiation Temperature*** • ***Dynamic Young's modulus*** • ***Compression strength*** • ***Ultimate bending strength*** • ***Optical ceramography*** • ***Uranium and thorium content*** The above measured characteristics will be compared to values obtained during pre-irradiation characterization."	Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed. • *NGNP R&D Response*: This is part of Graphite R&D program.
E-3	Theoretical models for the effects of neutron damage on the properties of graphite have been developed, however, these models need modification for the new graphites and will need to be extended to higher temperatures and/or higher neutron doses. V&V of theoretical models will require generation of experimental data on the effect of neutron irradiation on properties.	See item E-1 for description of graphite R&D efforts. (Sec. 19.2.4, p. 293) – ***Structural Mechanics*** – "The main tools for structural analysis exist, but specific modeling and correlations for NGNP geometry and materials have to be developed. This work includes: 1) incorporation of ***constitutive laws*** for materials and developing numerical models 2) seismic modeling of a block-type core 3) fluid structure interaction and flow-induced-vibration methodology, and 4) leak-before-break methodology."	See item A-8 for information relating to development of analytical models. See item E-1 for information relating to ASME Code qualification of graphites, and for structural materials R&D relating to graphite.	(Sec. 16.2.1.10, p. 16-34) – *Engineering Design Tools R&D* – Table 16.2-7 describes planned and in-progress efforts for developing ***irradiated material behavior models (e.g. graphite blocks); graphite corrosion/oxidation models for air ingress models; and discrete element modeling to simulate behavior of graphite bodies in contact with each other*** (e.g. block reflector structures and fuel spheres). (Section 16.2.2, p. 16-35) – "Two DDNs (NHSS-02-01 and NHSS-02-02) have been identified to ***extend the irradiated materials qualification database for Reflector Graphite from the temperature-fluence envelope of the PBMR DPP to that of the PBMR NGNP***. The extension of PBMR DPP data is required due to the broader operating envelope of the PBMR NGNP, which has an increased power level, a lower reactor inlet temperature and higher reactor outlet temperature. ***The corresponding R&D comprises irradiation of graphite samples at low and high temperatures, plus post-irradiation examination and analysis***."	• *Based on review of AREVA PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: Needs for development of updated, code-approved graphite materials have been recognized in the WEC PCDR. Some of the R&D has been completed. • *NGNP R&D Response:* This is part of Graphite program.
E-4	Uncertainties in the temperature and dose received by a component; the severity of temperature and dose gradients in a component; the rate of dimensional change in the specific graphite used in a given design; the extent to which stresses are relieved by irradiation-induced creep; and the extent of changes in key physical properties such as elastic moduli, thermal conductivity, coefficient of thermal expansion, compound to make the prediction of component stress levels, and hence decisions regarding component lifetime	See Item E-1 for description of graphite R&D efforts. See item E-3 regarding efforts for a structural mechanics code/model.	See item E-1 for information relating to ASME Code qualification of graphites, and for structural materials R&D relating to graphite. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	See items E1 and E2 for information on graphite materials characterization and behavior, including measurement of physical changes and properties. (Sec. 4.4.2, p. 4-84) – *Future Studies/Core Structural Ceramics (CSC)* – "The ***effect of the higher power level on the life of the CSC needs to be investigated***. It may require more frequent CSC replacements. ***Thermal stresses in the bottom reflector blocks due to the temperature gradient between the inlet and outlet flow need to be assessed. The effect of the increased amount of abrasion of the reflectors due to the higher***	• *Based on review of AREVA PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the AREVA PCDR. • *Based on review of GA PCDR*:

TABLE 2E – GRAPHITE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
	and replacement schedules, very imprecise.			***amount of spheres circulated per day also needs further investigation***." (Sec. 16.2.4.1, p. 16-42) – *Core Structure Ceramics* – Reflector Graphite R&D – "***Two supplemental irradiation series are planned*** for the PBMR NGNP to extend the database supporting the PBMR DPP. One, corresponding to DDN NHSS-02-01, is at low temperature, nominally proposed at 350ºC. The other, corresponding to DDN NHSS-02-02, is at high-temperature, nominally proposed at 950ºC. For both of these series, the initial step is planning for the irradiation test program and preparation of the graphite samples and capsules. An initial irradiation is proposed at both temperatures to an intermediate fluence level that corresponds, as a minimum, to the maintenance outage interval of the PBMR NGNP, 5 years. The data provided by this irradiation will be sufficient to support initial operation of the PBMR NGNP and is planned in support of the construction and operating license application. A ***more extended irradiation is planned at both temperatures to confirm or establish the target design life for the replaceable reflector***. The latter is planned for completion prior to initial startup activities."	The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics, development of structural mechanics models, and irradiation tests to determine component lifetime and replacement schedules are recognized in the WEC PCDR. • *NGNP R&D Response*: This is part of Graphite program.
E-5	Whole-core models are required that can predict the stress states of graphite components within the core. Such models should be capable of taking inputs such as temperature and neutron dose and calculating the dimensional change, creep, thermal conductivity, etc., from established theoretical models. Reliable stress-state predictions as a function of reactor life would enable reactor operators and regulators to provide NDE guidance and make decisions regarding inspection intervals and core block replacement.	See item E-3 regarding efforts for a structural mechanics code/model. (Sec. 19.2.4.1, p. 291) – ***Reactor System Analysis Code/MANTA*** – "Global validation of MANTA currently consists of code-to-code benchmarking: comparisons with CATHARE from CEA (France), LEDA from EDF (France), ASURA from MHI (Japan), REALY2 from GA (USA) and RELAP5-3D from INL (USA) have already shown good agreement. Qualification against experimental data is also progressing (EVO loop, HE-FUS3 loop and PBMM). Nevertheless additional benchmarks against experimental data are required. Some facilities that could provide valuable data have been identified: namely, HTTR reactor in Japan, HTR10 reactor in China, SBL-30 loop in the USA (SNL). The qualification of component models will follow from the qualification tests of the components. The core model qualification follows from comparison with other codes and with experimental results. Further, experimental data from HTTR and HTR-10 safety tests and from SBL-30 loop is required." (Sec. 19.2.4.1, p. 291) – ***Reactor System Analysis Code/RELAP*** – "The U.S. DOE sponsors RELAP5 code development at the INL. It is expected that this support will continue. Development needs are highlighted in the report INEEL/EXT-04-02993. Validation beyond that identified in this report and consistent with that planned for MANTA should be pursued."	See item A-8 for information relating to development of analytical models. Section 3.1.2, Reactor Core and Internals Design (pages 3-12 through 3-61), contains detailed descriptions of the software used to model the reactor system thus far. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	See item A-8 for information on development of modeling tools. (Sec. DDN NHSS-02-01, p. 4-80) – *Extended Properties of Irradiated Graphite at Low Temperatures* – "Within the PSMP envelope depicted in…, the initial design of the PBMR DPP graphite structures is based on historical irradiation test data. ***This data is summarized and used to generate the model for irradiated properties that is used for the design….This model will be validated and extended by making use of the material irradiation test data*** generated as a result of the PSMP. The proposed approach is to obtain the data necessary to ***extend the present PSMP envelope to lower temperatures*** and, thus, to qualify the Reflector Graphite for NGNP operational conditions." (Sec. DDN NHSS-02-02, p. 4-82) – *Extended Properties of Irradiated Graphite at High Temperatures* – "Within the PSMP envelope depicted in…, the initial design of the PBMR DPP graphite structures is based on historical irradiation test data. ***This data is summarized and used to generate the model for irradiated properties that is used for the design. This model will be validated and extended by making use of the material irradiation test data*** generated as a result of the PSMP…. The proposed approach is to obtain the data necessary to ***extend the present PSMP envelope to high temperatures and fluence levels*** and, thus, to qualify the Reflector Graphite for NGNP operational conditions." (Sec. 12.1.1, p. 12-11) – *Maintainability* – "Plant outages are scheduled at 5-year intervals. The minimum lifetime of IHX A of 10 years and the requirement of ***maximum 6-year***	• *Based on review of AREVA PCDR*: Needs for improved reactor analysis computer models have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: Reactor core analyses performed to date and the need for further analytical models have been described in the General Atomics PCDR. • *Based on review of WEC PCDR*: The WEC PCDR has addressed this topic. • *NGNP R&D Response*: Mainly vendor scope, but Graphite R&D program has activity in whole-core modeling.

TABLE 2E – GRAPHITE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				intervals between maintenance and inspection on other SSCs are the main constraints in deciding on intervals between scheduled outages. Scheduled outages will last maximum 30 days for outages scheduled after 5 years and 15 years. The 10 year outage requires the replacement of the IHX A and will last a maximum of 50 days. The 20 year outage requires the ***replacement of ceramic core structures*** and will last a maximum of 180 days. The scheduled maintenance outages are repeated in 20 year cycles." (Sec. 12.3.1.1.1, p. 12-20) – *Core Structures and Reactor Pressure Vessel* – "The reactor core structure and Reactor Pressure Vessel (RPV) are designed not to require any maintenance or access during normal operation. Maintenance that will occur in 20 year intervals (possibly 15 year intervals, when the side and central reflectors are replaced. ***Replacement of the reflectors*** will require special tools as the initial core structure installation and commissioning process is in a 'clean' (non-nuclear) environment, and personnel access is not a significant limitation. The core structure assembly tools will then later undergo further development for use in a radiological contaminated environment to perform the core structure refit. The initial installation activities would help to evaluate and validate the replacement concept. A system will be conceptualized to be able to perform a core structure refit during the reflector replacement outage. This entails dismantling the top end of the reactor (such as removing the RSS assemblies), and removing the centre column. The transport and storage of the core structure's used graphite blocks are accommodated within the site infrastructure." (Sec. 12.5.2.4, p. 12-38) – *Replacement of the Core Structures Ceramic (CSC)* – "The requirement of 500 MWt power will inevitably lead to higher neutron fluence in the ceramic core structures. The ***lifetime of the graphite structures in the core, especially the central reflector, needs to be investigated for the high fluences*** encountered in the NGNP. The accessibility of the ceramic core structures also need to be addressed, as the removal of the RPV head is a complicated undertaking."	
E-6	Basic research should be conducted to strengthen the understanding and modeling capability of the displacement damage process in graphite. In addition, in graphite technology, there is a need for analytical models for oxidation, changes in physical properties, irradiation induced dimensional change, and irradiation creep. They could be developed to feed into a structural integrity model for the graphite core which would be used for core design and safety assessment.	See item E-3 regarding efforts for a structural mechanics code/model.	See item A-8 for information relating to development of analytical models. See item E-1 for information relating to materials characterization and qualification of graphites. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	See items A-8 and E-5 for information on development of modeling tools. See item E1 for information on graphite materials characterization and behavior. (Sec. 16.9.5, p. 16-105) – *Fundamental Properties of Nuclear Graphite* – "The properties of nuclear-grade graphite, and the changes in those properties as a function of temperature and fluence, are known to vary, depending upon the source of raw materials and the specific details of the processing steps used in its manufacture. The graphite R&D program proposed for the PBMR NGNP (Section 16.3) comprises the minimum incremental enabling technology requirements relative to the PBMR DPP. However, a more	• *Based on review of AREVA PCDR*: Needs for improved structural mechanics computer models have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for further analytical models and materials characterization and qualification of graphites have been recognized in the

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Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
				fundamental understanding of graphite properties as a function of raw material sources and processing would be highly desirable in the context of an expanding commercial market for High-Temperature Gas Cooled Reactors (HTGRs). Through such improved understanding, the need for expensive and time-consuming graphite qualification programs based on irradiation capsules might be avoided or at least reduced in scope. Further, a ***more fundamental understanding of graphite properties might provide the basis for enhancing the life span of reflector graphite***. The expanded graphite development program, outlined by INL [Ref 16-4], appears to provide the initial steps toward such an improved fundamental understanding. As already noted in Section 16.2, there is a significant potential for collaboration between INL/NGNP in the PBMR in this area, and steps are underway to affect such collaboration."	General Atomics PCDR. • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR. • *NGNP R&D Response*: Part of Graphite program
E-7	Irradiation induced change in the coefficient of thermal expansion, including effects of creep strain.	See description on graphite R&D needs from Sec. 19.2.2.3, p. 286, under item E-1.	See item E-1 for information relating to materials characterization and qualification of graphites. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	See items E1 and E2 for information on graphite materials characterization and behavior, including measurement of physical changes and properties.	• *Based on review of AREVA PCDR*: Needs for further knowledge of the phenomena described in this item have been recognized in the AREVA PCDR. • *Based on review of GA PCDR*: The needs for further materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR. • *NGNP R&D Response*: Part of Graphite program scope.
E-8	Irradiation induced change in mechanical properties such as strength and toughness, including the effect of creep strain.	See description on graphite R&D needs from Sec. 19.2.2.3, p. 286, under item E-1.	See item E-1 for information relating to materials characterization and qualification of graphites. Concerns over the creep phenomenon due to operation at high temperatures and accumulated irradiation, including the need for further technology development to fully understand mechanical properties, are documented throughout the PCDR, relating to the nuclear fuel, intermediate heat exchanger, reactor vessel, and turbine.	See items E1 and E2 for information on graphite materials characterization and behavior, including measurement of physical changes and properties.	• *Based on review of AREVA PCDR*: Needs for further knowledge of the material characteristics described in this item have been recognized in the AREVA PCDR.

TABLE 2E – GRAPHITE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
					• *Based on review of GA PCDR*: The needs for further materials characterization and qualification of graphites have been recognized in the General Atomics PCDR. • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models are recognized in the WEC PCDR. • *NGNP R&D Response*: Part of Graphite program scope.
E-9	Blockage of coolant channel in a fuel element block or reactivity control block due to graphite failure and/or graphite spalling.	***There is no indication that this item has been specifically addressed.*** However, in Table 11-2, "Preliminary List of DBE Initiating Events", "***single fuel channel blockage***" is listed as a preliminary initiator for a design basis event.	***There is no indication that this item has been specifically addressed.*** See item B-7 for information relating to reactivity control.	(Sec. 16.2.1.6, p. 16-31) – *Core Structural Ceramics R&D* – "The most demanding Core Structural Ceramics R&D activities supporting the PBMR DPP center upon the Reflector Graphite. The Reflector Graphite components, and specifically the replaceable reflector components immediately adjacent to the core, are the only Core Structural Ceramic components for which fluence-related life limits must be established. The PBMR-Specific Materials Test Reactor Program (PSMP) is structured to provide data supporting startup and initial operation of the PBMR to the first planned outage period (6 years) as input to the licensing process. Completion of the PSMP is planned for the timeframe of plant startup in 2012. The results of the PSMP will be confirmed through surveillance, testing, inspection and maintenance activities over the plant operating lifetime." (Section 16.2.1.10, p. 16-34) – *Engineering Design Tools* – "In Progress: Irradiated Graphite material behavior models. Implementation of non-linear material behavior models for use with commercial FEA codes in ***predicting distortion and material properties of graphite blocks under irradiation. Includes implementation of graphite failure models***."	• *Based on review of AREVA PCDR*: ***There is no indication that this item has been specifically addressed in the AREVA PCDR.*** • *Based on review of GA PCDR*: ***There is no indication that this item has been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: Needs for greater definition of materials characteristics and development of structural mechanics models, including implementation of failure models, are recognized in the WEC PCDR. • *NGNP R&D Response*: Design issue- no R&D.
E-10	Statistical variation of non-irradiated properties, due to forming, processing, raw materials, and formulation.	This item has not been fully addressed for all graphite components. However, the following references exist for statistical control and sampling related to fuel fabrication: (Sec. 15.1.2.3, p. 224) – Compact Fabrication – "A	(Sec. 3.1.4.4, p. 3-76) – ***Fuel Quality and Performance Requirements*** – "…The fuel and reactor core are to be designed such that there is at least a 50% probability that the radionuclide releases will be less than the Maximum Expected criteria, and at least a 95% probability that the	(Section 16.2.1.6, p.16-31) Completed: Reflector Graphite Unirradiated Properties. Characterize the unirradiated properties of NBG-18 graphite. (Sec. 16.9.5, p. 16-105) – *Fundamental Properties of Nuclear Graphite* - The ***properties of nuclear-grade***	• *Based on review of AREVA PCDR*: This item has been addressed in the AREVA PCDR for statistical control

TABLE 2E – GRAPHITE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		predetermined number of compacts are destructively evaluated to ensure the lot of compacts ***meets the fuel specification on a statistical basis***. Once the ***chemical and physical attributes*** of the compacts has been confirmed, the lot will be certified and released for fuel block fabrication." (Sec. 15.3, p. 232) – Fuel Qualification Plan – "The second sequence is designed to provide the data used to qualify the fuel for use in the NGNP plant. A quantity of fuel would be fabricated, irradiated, and inspected that would yield the ***statistics required to demonstrate that the fuel supports the plant safety case***. This fuel would also be fabricated in the pilot line. It is envisioned that several batches would be made and blended to form a homogeneous lot upon which the results would be based. This process will be used to as closely as possible reflect anticipated commercial scale fabrication techniques. The ***statistical basis and acceptance criteria for the test will reflect this processing technique***." The R&D aspects of ***fuel development and qualification*** (fuel kernel, coating, compact, QA, and mass production) are addressed in section 19.2.1, beginning on page 282.	releases will be less than the Design criteria. The logic for deriving these fuel requirements is illustrated in Figure 3.1-68. Top-level requirements for the NGNP are defined by both the regulators and the users. Lower-level requirements are then systematically derived using a systems-engineering approach. With this approach, the radionuclide control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on radionuclide releases from the VLPC, reactor vessel, and reactor core are successively derived. Fuel failure criteria are in turn derived from the allowable core release limits. Finally, the required as-manufactured fuel attributes are derived from the in-reactor fuel-failure criteria, with consideration of achievable values based on existing fuel manufacturing experience, thereby providing a logical basis for the fuel quality specifications…The maximum allowable release fractions for 30.2-yr Cs-137 and 249.8-d Ag-110m are included in Table 3.1-16 because these nuclides are expected to be the ***strongest contributors to worker dose***, based on previous assessments of radionuclide plateout distributions and plant-maintenance requirements." General Atomics discusses, in the PCDR, fuel fabrication a number of times in logistical terms, recommending initial fuel supplies from foreign vendors, and construction of a fuel fabrication facility in Idaho for the longer term (section 1.2.1, page 1-3).. Uniform quality of fuel or any other graphite materials is not addressed in the PCDR.	***graphite, and the changes in those properties as a function of temperature and fluence, are known to vary, depending upon the source of raw materials and the specific details of the processing steps used in its manufacture.*** The graphite R&D program proposed for the PBMR NGNP (Section 16.3) comprises the minimum incremental enabling technology requirements relative to the PBMR DPP. However, a more fundamental understanding of graphite properties as a function of raw material sources and processing would be highly desirable in the context of an expanding commercial market for High-Temperature Gas Cooled Reactors (HTGRs). Through such improved understanding, the need for expensive and time-consuming graphite qualification programs based on irradiation capsules might be avoided or at least reduced in scope. Further, a more fundamental understanding of graphite properties might provide the basis for enhancing the life span of reflector graphite. The expanded graphite development program, outlined by INL [Ref 16-4], appears to provide the initial steps toward such an improved fundamental understanding. As already noted in Section 16.2, there is a significant potential for collaboration between INL/NGNP in the PBMR in this area, and steps are underway to affect such collaboration.	and sampling related to fuel fabrication, ***but not for other graphite components.*** • *Based on review of GA PCDR*: The need for statistical control is addressed in the General Atomics PCDR for fuel, ***but it is not addressed for other graphites.*** • *Based on review of WEC PCDR*: This item has been addressed in the WEC PCDR. • *NGNP R&D Response*: Part of Graphite program.
E-11	Ability to develop generic specifications that will ensure consistency of graphite quality over the lifetime of the reactor fleet, including for replacement components.	***There is no indication that this item has been specifically addressed.***	(Sec. 3.1.1.2, p. 3-10) – "GA prepared draft ***fuel product specifications*** to define the property requirements for the ***kernels, coated particles, and fuel compacts***. The requirements were written to be consistent with an NFI fuel particle design in order to utilize NFI's existing fuel manufacturing capability to the greatest extent possible, thereby avoiding a significant fuel R&D program. The NFI extended burnup fuel particle design was selected rather than the reference High Temperature Engineering Test Reactor (HTTR) fuel particle design because this fuel particle is designed for irradiation to higher burnup and is more consistent with the reference German fuel particle design. Table 3.1-5 summarizes the physical properties of two NFI fuel particle designs and compares them to the reference German particle and to the Advanced Gas Reactor (AGR) reference fuel particle as defined in the preliminary AGR fuel product specification [AGR Fuel Spec. 2004]. The primary implications of this approach are that the kernel will be UO2 (rather than UCO), the ***U-235 enrichment will be 10%*** (as opposed to the effective U-235 enrichment of about 10.8% for the GT-MHR initial core), the fuel compacts will be made using the HTTR matrix material, and the particle packing fraction in the fuel compacts is limited to about 30%. The fuel quality requirements written into the draft fuel product specification are consistent with the desired quality level for NGNP as specified in the preliminary AGR fuel product specification."	(Section 16.2.1.6, p.16-31) ***Completed: Reflector graphite specification. Establish the specification for procurement of NBG-18 graphite***.	• *Based on review of AREVA PCDR*: ***There is no indication that this item has been specifically addressed in the AREVA PCDR.*** • *Based on review of GA PCDR*: The General Atomics PCDR includes generic specifications for fuel quality, ***but not for other graphite components***. • *Based on review of WEC PCDR*: This item has been addressed in the WEC PCDR. • *NGNP R&D Response*: Part of Graphite scope.
E-12	Tribology (effects of moving surface interactions) of graphite in helium environment,	(Sec. 7.7.1, p. 105) - ***Helium Purification Train*** – "The primary functions of the Purification Train are:	Sec. 3.9.1, p. 3-189) – ***Primary Coolant Purification System*** – "This subsystem provides a means to remove	(Sec. 4.2.7.2, p. 4-60) – *Helium Purification System* - The Helium Purification System (HPURS) is ***used to provide***	• *Based on review of AREVA PCDR*:

TABLE 2E – GRAPHITE - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
	including potentially impure helium environment (examples: surfaces sticking together; surfaces wearing on each other to generate dust, etc.)	• Removal of chemical and particulate contaminants from the primary coolant • Supply of purified helium to appropriate systems Since helium is used as the primary coolant, a ***helium purification system is required to provide the necessary degree of helium purity***. Oxidizing contaminants, in particular, may not exceed predetermined limits established in the specification. In detail, the helium purification system has the following functions: • Removal of particulate and gaseous contaminants from the primary coolant to maintain design values, in particular for H_2O, CO, CO_2, N_2, H_2, CH_4 • Removal of tritium • Removal of other radioactive contaminants from the helium, especially before transfer to the purified gas store (Xe, Kr, Ar) • Start up purification of the primary system before initial start up and after inspections and maintenance • Purification of newly delivered helium" (Sec. 19.2.2, p. 284) – Materials Development and Qualification – "…the VHTR design relies on contact conditions between different materials (metal to metal, graphite to ceramics, ceramics to metal, etc.) and R&D actions have to be performed to assess the contact conditions to avoid unexpected situations (bonding, wear, etc). As an example, the core support to reactor vessel interface is currently assumed to be a sliding interface. R&D actions are required to make sure that the helium environment (together with the contact pressure) is not likely to create a bonding effect between the alloy 800H and the 9CR1Mo materials. ***Tribology tests are needed on expected couples of materials in representative VHTR conditions***."	circulating impurities from the primary coolant helium, and to transfer those impurities to the radioactive liquid and gas waste systems of the facility. A separate regeneration section within this subsystem is used to remove the impurities that accumulate in the purification subsystem adsorbers. The regeneration section is operated periodically under automatic control whenever regeneration is required. The primary coolant helium purification subsystem consists of two separate, independent, but identical trains of components as shown in Figure 3.9-1. All of the components that make up the trains are mechanically passive in nature; however, the adsorber elements become radioactive as the removed impurities are concentrated within the various media. Each purification train must therefore be located in a shielded vault to minimize personnel exposure to radiation. Helium purification is accomplished by routing a small side stream of helium from the primary coolant system through a series of purification components. ***These components remove the following chemical impurities: Br, I, H_2O, CO, CO_2, H_2 (including Tritium), N_2, O_2, H_2S, Kr, Xe, CH_4, and other hydrocarbons***. ***There is no indication that phenomena associated with materials tribology have been specifically addressed.***	***the required degree of helium purity***. High purity coolant is required in order to minimize corrosion and contamination in the PHTS and SHTS. This is done by bleeding off a partial flow of helium from the PHTS and SHTS. The extraction point is from the highest pressure points, i.e. the PHTS and SHTS circulator discharges within the HTS. This flow is tapped off constantly during operation of the plant. The ***HPS removes chemical gaseous contaminants from the primary coolant within the PHTS*** by the use of, catalysts, adsorbers and the manipulation of helium temperature extracted from the PHTS and SHTS. The required helium purity levels will be confirmed during the conceptual design. Sec. 16.2.1.10, p. 16-34) – *Engineering Design Tools R&D* – Table 16.2-7 describes planned and in-progress efforts for developing irradiated material behavior models (e.g. graphite blocks); graphite corrosion/oxidation models for air ingress models; and ***discrete element modeling to simulate behavior of graphite bodies in contact with each other (e.g. block reflector structures and fuel spheres).***	In the AREVA PCDR, a helium purification system has been incorporated into the design to ensure the purity of the helium environment, and the need for improved knowledge of tribology has been recognized. • *Based on review of GA PCDR*: In the General Atomics PCDR, a helium purification system has been incorporated into the design to ensure the purity of the helium environment. ***There is no indication that phenomena associated with materials tribology have been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: This item has been addressed in the WEC PCDR. • *NGNP R&D Response*: Part of Graphite program.
E-13	Impact of degradation of thermal conductivity on fuel temperature limits.	(Sec. 20.2.1, p. 301) – Commercial Plant Power Level – "Modular VHTR's rely on conduction and thermal radiation in their passive safety features for decay heat removal. The thermal performance of the plant during a loss of active cooling is dominated by four items: the geometry of the plant, the thermal energy stored in the core at the beginning of the event, and energy (the decay heat) that is generated inside the core, and the ***heat transfer properties of the core (graphite)***. AREVA performed parametric studies to evaluate the sensitivity of the results of the limiting design basis accident (depressurized conduction cool down) to the key influencing parameters; namely, core power level, core inlet temperature, and ***graphite conductivity*** in terms of an equivalent change in reactor power. The results of the study support the conclusion that a maximum reactor thermal power rating of 565 MWth should be acceptable while allowing some margin for uncertainties. Based on the above, the commercial VHTR module should be designed to operate at the maximum safe power level; and, based on the AREVA's evaluation of plant safety limits, that maximum power level is 565 MWth."	(Sec. 3.1.2.2, p. 3-44) – "The reduction in graphite thermal conductivity with irradiation results in a peak fuel temperature increase of approximately 100°C. Accounting for thermal annealing of the irradiation damage reduces peak fuel temperatures by approximately 30°C. However, the effect of irradiation on graphite thermal conductivity has little impact on peak vessel temperatures."	(Sec. 16.2.3.1, p. 16-38) – *Fuel Graphite Irradiation Tests* - "Samples for investigation and irradiation will be cut from pressed graphite spheres provided for the test. These samples will be cut parallel and perpendicular to the extrusion direction. Following irradiation, the following characteristics will be measured: • Geometrical size • Mass • Calculation of sample density • Measurement of sample density • Sample porosity • ***Thermal conductivity in the range 20 up to Irradiation Temperature*** • Electric conductivity in the range 20 up to Irradiation Temperature • Thermal coefficient of linear expansion in the range 20 up to Irradiation Temperature • Dynamic Young's modulus	• *Based on review of AREVA PCDR*: In the AREVA PCDR, this item has been addressed in the design and by studies already performed. • *Based on review of GA PCDR*: This phenomenon has been recognized and quantified in the General Atomics PCDR. • *Based on review of WEC PCDR*: This item has been addressed in the WEC PCDR.

TABLE 2E – GRAPHITE - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
		Additional information discussing parametric studies, including thermal conductivity of graphite, can be found on page 26 of Appendix B2 of the PCDR. This information indicates that the sensitivity of peak temperature to variations in thermal conductivity of graphite is relatively low.		• Compression strength • Ultimate bending strength • Optical ceramography • Uranium and thorium content The above measured characteristics will be compared to values obtained during pre-irradiation characterization.”	• *NGNP R&D Response*: Part of Graphite scope.

TABLE 2F – PROCESS HEAT FOR HYDROGEN - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
F-1	Cold oxygen (O2) and other heavy-gas accidental releases from the process plant that can flow from the chemical plant to the nuclear plant (depending upon wind, relative plant elevations, and nuclear plant air intakes) and potentially impact the integrity of reactor systems, structures, and components (SSCs). All of the proposed processes for production of hydrogen start with water, and thus all of the processes will produce oxygen as a byproduct of hydrogen production. Oxygen is the one common chemical safety issue that can impact nuclear plant safety. At high oxygen concentrations, many "noncombustible" materials become combustible and the potential for spontaneous combustion increases. Increased oxygen levels at the reactor can compromise the functioning of safety equipment.	(Sec. 11.6, p. 184) – ***Distance between facilities*** is recognized as a key aspect of collocating the reactor and hydrogen production plant,. Also, in section 11.6.2, page 188, ***oxygen is recognized as a hazard*** in the sense that it increases the hydrogen explosion risk. However specific design of the hydrogen production facility was outside of AREVA's assigned scope, and ***there is no indication that this item has been specifically addressed.***	(Sec. 2.3, p. 2-15) – "The distance between the hydrogen plants and the reactor building is 90 meters based on the results of an INL engineering evaluation of the necessary separation distance [INL 2006]. No earthen berm or blast suppression barrier is considered necessary between the hydrogen production facilities and the reactor with a separation distance of 90 meters because the reactor s below grade. However, the hydrogen production facilities are circumvented by a low berm, which serves as a chemical spill retention barrier." (Sec. 5.2.2.3, p. 5-37) – "There is no currently anticipated inherent excessive risk in the thermo-chemical production of hydrogen that would preclude licensing of the NGNP or licensing of associated commercial-scale hydrogen productions plants based on the hydrogen production processes demonstrated in the NGNP. An attractive feature of the GT-MHR plant for electricity production is siting flexibility, because no plan for public evacuation is required as the result of the MHR's passive-safety features. For a commercial scale H2-MHR, a potential issue that requires further evaluation is whether or not a public evacuation plan is required because of potential accidents that could cause chemical releases from the SI hydrogen plant. However, chemical releases should not impact the passive safety of the reactor system." The General Atomics PCDR also contains general performance statements that the hydrogen production system will be designed to have no adverse impact on the primary system. ***There is no indication that this item has been specifically addressed.***	(Sec. 2.2.1.3.2, p. 2-32) – *Hydrogen Production System (HPS) Configuration* – "The HPS facility shall be separated from the NHSS consistent with commercial plant economic and risk tradeoffs. The ***interfaces between the HPS and the NHSS shall be designed to ensure that failures or upset conditions in the HPS do not result in failures or adverse impacts to the NHSS***." (Sec. 2.2.1.3.4, p. 2-33) – *HPS Safety & Licensing* – "The ***hydrogen production facilities, including the conversion, storage, and distribution systems, shall comply with the requirements of 29 CFR 1910.103, Occupational Safety and Health Standards (OSHA), Subpart H - Hazardous Materials, Hydrogen. In the event that the HPS facility also produces and stores significant quantities of oxygen, the requirements of 29 CFR 1910.104, Oxygen, shall be applied. The design, operation and maintenance of the HPS shall comply with 29CFR 1910.119, "Process Safety Management of Highly Hazardous Chemicals"."*** (Sec. 7) - WEC's focus during pre-conceptual design was on the hybrid sulfur (HyS) method for hydrogen production. Clearly, this technology produces oxygen as an intended end product, and assumptions are made in the pre-conceptual design that the oxygen will be purified and delivered through a pipeline to the plant boundary, for final transportation and sale. (Sec. 7.2.1.2,, p. 7-45) - "*Safety*" - Indicates that the ***HyS system will meet NFPA and OSHA requirements, and eventually will be subjected to a process hazards analysis***. (Sec. 7.2.1.4, p. 7-45) - "*Environmental Interface*" - ***Allows for the possibility that the oxygen may be vented rather than sold, indicating only that the O2 would need to be purified to meet emissions standards if this method of disposition were chosen***. (Sec. 7.2.1.4, p. 7-49) - "*Interface with Workers and the Public*" – "Other potential hazards that must be considered during future design phases are the possibility of hot oxygen, SO_2, SO_3 and sulfuric acid vapor leaks. Hot oxygen presents a serious fire hazard. Many substances that will not burn in air will burn vigorously in a stream of hot oxygen. The sulfur-containing gases are much denser than air and can form dangerous clouds, especially in confined spaces. The plant contains large amounts of liquid sulfuric acid in a range of concentrations. All of these liquids are dangerous if spilled. ***In future design phases attention must be paid to prevention and handling of spills***." (Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, ***a full scope PRA that***	• *Based on review of AREVA PCDR*: ***Specific design of the hydrogen production facility was outside of the AREVA PCDR assigned scope, and there is no indication that this item has been specifically addressed.*** • *Based on review of GA PCDR*: ***There is no indication that this item has been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: Indications are that WEC has addressed this item in the PCDR and will address it in more detail, via process hazards analysis, as the project progresses. • *NGNP R&D Response*: Separation of reactor and H_2 plant should reduce this concern.

TABLE 2F – PROCESS HEAT FOR HYDROGEN - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
				addresses all internal and external hazards, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a ***Process Hazards Assessment (PHA) for the Hydrogen Production Facility*** (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP."	
F-2	Failure of the IHX leading to potential damage to safety-related SSCs in the reactor due to blow-down effects from large mass transfer and over-pressurization of either secondary or primary side. The impact of the IHX failure depends upon the selection of the heat transfer fluid in the secondary heat transport loop. Helium is the leading candidate for the heat transport loop, but no final decisions have been made. If helium is used, the helium inventory in the secondary loop may be greater than the inventory in the reactor; thus, any leak in the IHX can significantly increase the total helium inventory involved in any reactor depressurization event.	(Sec. 11.3.2.5, p. 169) – "The unique design consideration accommodating the ***Hydrogen Production Pilot Plant*** (HPPP) is the primary ***coolant circuit*** dedicated for this application. It is unique in that it is sized for the smaller amount of energy needed for this application relative to that for power generation, about 60 MWth. It is expected that the ***performance of equipment and components in this circuit will be of high research and development interest***. Frequent inspection, maintenance, and design modification is anticipated." (Sec. 11.5.2.5, p. 180) – ***IHX Failure*** – "As an indirect cycle design, nuclear heat generated in the reactor core of AREVA's NGNP concept is transmitted to the power conversion or process heat system via an IHX. Failure of the IHX is any breach of the physical boundary between the primary and secondary circuits. ***AREVA's NGNP concept is designed with zero pressure differential across the pressure boundary common to both the primary and secondary circuits. As such, fluid exchange between the circuits following an IHX failure is driven by momentum and diffusion phenomena, rather than pressure.*** The main safety issue is the confinement of radiological content. AREVA's NGNP concept employs two separate IHX designs (i.e., plate type IHX supporting the HPPP process heat application and tube-type IHX supporting power generation). While the likelihood of failure is considered to be smaller with the tube-type design, the radiological consequences of an IHX failure are independent of IHX-type. The primary defense against a radiological release is the maintenance of low activity in the primary circuit….IHX failure detection is achieved by activity detection in the secondary side of the IHX combined with an overpressure of the primary helium compared to the secondary gas during normal operation. In case of IHX failure detection, the following actions are to be taken: • Heat generation control • Control rods insertion (by automatic action) as abnormal parameter value is detected (note that in case of combination with loss of electrical power, control rods drop by gravity in the core), • Reserve Shutdown System insertion by operator action, if control rod insertion fails. • IHX isolation valves are closed. Isolation valves are implemented on the secondary circuit, close to the IHX vessel. • Automatic primary circulator trip on effected loop or, if the effected loop cannot be identified, on all loops.	The General Atomics PCDR contains general performance statements that the hydrogen production system will be designed to have no adverse impact on the primary system. ***There is no indication that this item has been specifically addressed.***	WEC considered metallic and ceramic technologies. For the metallic IHX, several ***DDNs were identified to provide characterization and qualification of materials (Alloy 617 and Alloy 230). Quality research in these areas would reduce the probability if IHX failure***….The DDNs for ceramics are more developmental in nature but should have an influence on the probability of IHX failure should a ceramic IHX be used. (Sec. 11.2.2.4, p. 11-14) – *Helium Pressure Control* – "The pressure in the PHTS and SHTS will be controlled by utilizing the Pressure Control System, which is a subsystem of the Helium Service System (HSS). The ***HSS will govern and control the amount of helium (pressure) in the PHTS and SHTS according to predefined pressure set-points for the PHTS and SHTS***. During normal operation the plant will generally operate at the rated pressure levels and very little pressure control actions are envisioned. However, during plant start-up, transitions and transient events, pressure control is an important control function. The HSS will be responsible for controlling the pressure in such a way that the pressure differentials across the components in the HTS (especially the IHX, PCHX and SG) remain within specified limits to ***avoid operation of the components outside their design envelopes***. The HSS will control the pressure inside the HTS by injecting/extracting helium from/to a higher/lower pressure source." (Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a ***full scope PRA that addresses all internal and external hazards,*** including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Sec. 16.3.2, pp. 16-50 through 16-59) – This section provides a ***detailed plan for IHX R&D***, including metallic and composite materials.	• *Based on review of AREVA PCDR*: Indications are that this item has been addressed in the AREVA PCDR in the design, and will be further addressed with design improvements as the project progresses. • *Based on review of GA PCDR*: ***There is no indication that this item has been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: Indications are that this item has been addressed in the WEC PCDR, and will be further addressed with design improvements as the project progresses. • *NGNP R&D Response*: Safety analysis issue – no R&D impact.

TABLE 2F – PROCESS HEAT FOR HYDROGEN - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
		• Heat removal after reactor shutdown • SDHRS, if the effected IHX can be isolated, • SCS startup ensuring decay heat removal through primary helium forced convection. This start-up should be manual since large duration should be available. (If necessary, e.g., for reliability purpose, automatic startup may be defined). • RCCS with passive heat removal capacity. Efforts to depressurize the primary system and, thus, further minimize leakage to the secondary will also be considered.”			
F-3	Failure of the process heat exchanger (PHX) leading to potential damage to safety-related SSCs in the reactor, due to fuel and primary system corrosion from the introduction of corrosive process plant chemicals leaking down the process heat transport line and failing the IHX.	Specific design of the hydrogen production facility was outside of AREVA's assigned scope, and they have addressed the design in a general way, mainly to identify and characterize required interfaces with the nuclear plant. ***There is no indication that this item has been specifically addressed.***	See item F-1 for an excerpt from the General Atomics preliminary hazards analysis for the hydrogen production system. The General Atomics PCDR also contains general performance statements that the hydrogen production system will be designed to have no adverse impact on the primary system. ***There is no indication that this item has been specifically addressed.***	***Development for metallic materials in the process coupling heat exchanger (PCHX) will be covered by the metallic IHX developmental research***. (Sec. 14.5.2, p. 14-38) – *Future Studies* – “During the conceptual design phase, a full scope PRA that addresses all internal and external hazards, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a ***Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements***. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP.” (Sec. 16.4.1, p. 16-65) – *Design Data Needs (Hydrogen Production Facility)* – “Note that two reference materials have been identified for the IHX, Alloy 617 and Alloy 230. These same materials will be tested for the PCHX design.”	• *Based on review of AREVA PCDR*: ***There is no indication that this item has been specifically addressed in the AREVA PCDR.*** • *Based on review of GA PCDR*: ***There is no indication that this item has been specifically addressed in the General Atomics PCDR.*** • *Based on review of WEC PCDR*: Indications are that this item has been addressed in the WEC PCDR, and will be further addressed with design improvements as the project progresses. • *NGNP R&D Response*: Safety analysis issue – no R&D impact. (*Note:* PHX is tertiary with respect to the reactor.)
F-4	Steam generator failures leading to the introduction of steam/water into the primary system, potentially causing a reactivity spike and chemical attack of the TRISO fuel particle coatings and graphite. Some hydrogen production processes, such as high-temperature electrolysis, require steam as a process feedstock; thus, the high-temperature reactor may be required to provide high-temperature steam.	The AREVA design uses a ***steam generator*** as part of the Power Conversion System. (Sec. 8.3.1, p. 119) – “The gas turbine exhaust contains significant residual heat most of which is transferred to tertiary water/steam in the Heat Recovery Steam Generator (HRSG). The steam generated in the HRSG drives the HP and LP steam turbine to drive its generator to produce electricity.” Table 11-2, p. 174, identifies ***steam generator tube ruptures*** among the preliminary list of initiating events for design basis events, and Section 11.5.2.4, p. 179, identifies a leak from a steam generator combined with failure of an IHX as a potential event of water ingress into the primary circuit, with the consequences being ***reactivity insertion***	See item B-7 for information relating to reactivity control. (Sec. 5.1.1.3, p. 5-6) - **Control of Chemical Attack – “**Chemical attack on fuel particles and on the graphite core structure can result from air or water ingress into the primary system. ***Steps have been taken to prevent ingress of contaminants, and consequences are expected to be acceptable if they occur. The likelihood of water entering the primary system is limited by the absence of high pressure and high energy sources of water in proximity to the primary system***. Under normal operating conditions, all water coolers and heat exchangers operate at lower pressures than the pressure of the primary coolant with which they exchange heat. In the event of a cooler or	The WEC PCDR indicates that this ***scenario is not applicable to the WEC design because the scenario is not probable***. The S/G in the WEC design does not interface with the primary heat transport system. Instead, it interfaces with the helium-filled secondary heat transport system. This provides isolation from the primary system. (Sec. 8, p. 8-8) - *Power Conversion System: Summary and Conclusions* – “The Steam Generator has been identified as a developmental component based on prior design development experience for other High Temperature Gas Reactors (HTGR) applications. The requirements, configuration, materials and design features of this component require that a number of Design Data Needs	• *Based on review of AREVA PCDR*: The AREVA PCDR has proposed development of a white paper to provide discussion of water ingress events, including steam generator tube leaks. • *Based on review of GA PCDR*: The General Atomics PCDR indicates that this is not a likely event, and

TABLE 2F – PROCESS HEAT FOR HYDROGEN - SUMMARY

Item	NRC Need/Issue Identified	Applicable AREVA R&D *or* Already-Identified Solution	Applicable General Atomics R&D *or* Already-Identified Solution	Applicable Westinghouse R&D *or* Already-Identified Solution	Comments/Conclusions
		and combustible gas control. Section 11.5.2.4 identifies the following ***uncertainties that must be resolved*** to ensure that this event can be mitigated: • the benefit of start up of the SCS • the benefit of primary circuit loop isolation strategies • the impact of water on graphite structure and its heat transfer properties • the influence of water on fuel particles performances as well on the radio-elements trapped in the graphite blocks • the consequences of CO and H2 release • the limitation of water available to enter the pressure boundary • the impact of possible actuation of safety valve (primary and secondary) on potential radiological releases. (Sec. 21.1.2, p. 315) – "Any decision to adopt a steam cycle HTR configuration increases the significance of water ingress events due to the potential for ***steam generator leaks***. This issue was successfully managed in previous operating HTRs. However, the possibility for water ingress continues to be perceived as a significant issue within the broader nuclear community. There are various reasons for this including misunderstanding of the source of water ingress in the Fort St. Vrain reactor, failure to appreciate the differences in steam generator technology between HTRs and LWRs, and unfamiliarity with the consequences and mitigation of water ingress in HTRs. Steam line breaks within the reactor building also must be considered for steam cycle concepts. Steam line breaks must be evaluated for building pressurization issues and for any impact on building venting and filter systems, if a vented confinement concept is used for the NGNP. A ***white paper should be developed addressing water ingress and steam line break events*** and their likely impact on NHS design. The intent is not necessarily to provide detailed analyses of such events. Rather the focus should be on describing the issues and concerns associated with each type of event, the potential significance of these events on operation, safety, and licensing, mitigation of these events including likely design features which might be utilized, and ***likely R&D*** that might be necessary to resolve any open issues."	heat exchanger leak, primary coolant helium would leak out into the secondary cooling water until pressures equilibrate. Then the rate of ingress of sub-cooled water would be small, as water tries to enter the primary system by diffusion and gravity. The amount of water that could enter is limited to the inventory of water in the secondary coolant circuit located above the elevation of the leak. Most of the sub-cooled water that could enter the power conversion vessel would remain at the bottom of the vessel. Very little of it would become entrained in the helium coolant and be transported to the core. Core cooling can still be provided by either the PCS or the SCS, and would limit the potential for chemical attack. If core cooling is not available, the potential of water transport to the core would still be limited. The sub-cooled water will not flash to steam unless the primary coolant helium pressure is below the water saturation pressure, which may occur only when the reactor is operating at a low power level. The reaction rate of water and core graphite will be negligible. The reaction of steam and graphite is slow and endothermic and therefore is not self-sustaining."	(DDNs) be satisfied for successful design, manufacturing, delivery and long term operation of the prototype and follow-on components. ***Eighteen items of need are identified for the Steam Generator****….* It is recommended that a future study be conducted that will evaluate alternative approaches for the steam generator including more conventional designs (e.g. refractory lined, U tube) compared to the once through helical type SG proposed in the preconceptual design. Single vs. multiple trains will be evaluated. The results of the study will establish a path forward for design development of the steam generator." (Note: The 18 DDNs for the steam generator are enumerated and described in Table 8-20, page 8-40.) (Sec. 10.2.5, p. 10-27) – *Steam Generator Building* – "Major systems and components include the Secondary Heat Transport System helium piping, the PCS steam generator, steam generator supports, feedwater and main steam system piping, and main steam safety valves. ***Liquid secondary containment is required in the event of a major liquid spill or leakage of PCS feedwater. Physical separation is required between this building and the NHSS Building***. The NHSB exterior walls provide a barrier designed for security, fire and potential pressure loads due to pipe rupture in this building. This building and its contents are not credited in the mitigation of design basis events affecting the NHSS. In addition, the ***building or its contents do not interact with the NHSS building in a manner that compromises the safety functions of the NHSS***." (Sec 11.5.2, p. 11-30) – Steam *Generator Operating Conditions* – "The SG will operate at helium inlet temperatures of up to 900°C in certain modes of plant operation. A future study is needed to investigate the effect that extended period of operation at these conditions will have on the SG design." (Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a full scope PRA that addresses all internal and external hazards, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a ***Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements***. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Sec. 16.5.1, p. 16-95) – The ***Design Data Needs associated with the steam generator*** are shown in Table 16.5-1 and are categorized as follows: • Materials - indicates that material related data is needed to support long term operation in the specified helium and secondary environments. • Methods Development, Verification and Validation (MDV&V) - this grouping is further categorized as:	consequences would be acceptable if it did occur. • *Based on review of WEC PCDR*: The WEC PCDR indicates that this issue scenario is not probable for the PBMR design. Nonetheless, significant R&D is planned to ensure reliability of the steam generator. • *NGNP R&D Response*: Design dependent. Currently steam generation is tertiary so such an event cannot happen.

TABLE 2F – PROCESS HEAT FOR HYDROGEN - SUMMARY					
Item	**NRC Need/Issue Identified**	**Applicable AREVA R&D *or* Already-Identified Solution**	**Applicable General Atomics R&D *or* Already-Identified Solution**	**Applicable Westinghouse R&D *or* Already-Identified Solution**	**Comments/Conclusions**
				○ Performance- indicates that configuration specific testing is needed to support final design. ○ Design - configuration specific testing or mock-ups to support final design of the demonstration unit. ○ Fabrication - configuration specific mock-ups to qualify techniques for prototype fabrication.	
F-5	Loss of the pressurized coolant inventory from the intermediate loop leading to a loss of primary reactor heat sink and the potential for hydrodynamic forces on the IHX leading to IHX failure and loss of reactor primary system coolant.	(Sec. 11.3.3.2, p. 170) – Heat Removal after Shutdown Function • "The Shutdown Cooling System (SCS) implemented inside the reactor vessel: ***This system can operate even if the secondary circuit and the primary forced helium circulation are not available.*** SCS is designed for achieving this function in pressurized and depressurized conditions. The SCS is made by a circulator and a heat exchanger transferring the decay heat from the core to a water circuit. • ***In case of failure of these systems***, the decay heat is transferred from the reactor vessel to the Reactor Cavity Cooling System (RCCS) mainly by radiation. The RCCS consists of two independent and redundant trains operating in natural circulation. During any conditions, its function is to maintain acceptable temperature of the reactor cavity concrete and the vessel support devices." Also see item F-2 for additional information from section 11.5.2.5, which discusses various modes of IHX failures, recovery modes, and event mitigations.	(Sec. 5.1.1.3, p. 5-5) – ***Control of Heat Removal*** – "Reactor cooling can be accomplished by the PCS, the SCS, the HTS, or by passive cooling through the reactor vessel to the RCCS. The PCS, which operates during power generation, provides primary shutdown cooling. PCS cooling capability is an active system. The SCS is designed specifically for residual heat removal in the event that the PCS is unavailable. In the NGNP, the HTS is another active system that can be used to remove heat from the reactor core. In the event the PCS and SCS are unavailable, the core design ensures passive residual heat removal capability. The limited core diameter, limited power density, and unique core assembly configuration (annular with a large length-to-diameter ratio) limit core and fuel temperatures during passive cooling. The RCCS, which is independent and diverse from the PCS and SCS in fundamental ways, acts to keep structures, including the reactor vessel and containment building, within allowable temperature limits. The RCCS is totally passive under accident conditions. Reactor heat is transferred through the reactor vessel walls to RCCS cooling panels by conduction, natural convection and radiation heat transfer; the vessel walls are uninsulated to facilitate this process. The RCCS air cooling loops are naturally circulating. With RCCS cooling, core temperatures peak after about 2 days and cool within several days to below 1100°C. Even if the RCCS were not available for some reason, heat from the reactor vessel walls would be transferred through the inoperable RCCS panels to the containment building itself and ultimately to the earth surrounding it. This cooling capability is also totally passive. It is not in the design basis and is not necessary to meet any safety requirements or quantitative safety goal, but exists as an inherent feature, enhancing the safety of the NGNP. With core cooling in this mode, core temperatures peak at about the same level as with RCCS cooling, but cool more slowly thereafter. The NGNP vessel system has a unique safety function in support of core cooling systems. LWR vessels must confine primary coolant (i.e., water) at all times, at least so that the core will remain covered. However, while containing the helium coolant is an important vessel function for the NGNP, ***sufficient core cooling can be provided even if the helium coolant is lost.*** "	***Development for metallic materials in the secondary heat transport system is expected to be enveloped by the metallic IHX developmental research.*** (Sec. 14.5.2, p. 14-38) – *Future Studies* – "During the conceptual design phase, a ***full scope PRA that addresses all internal and external hazards***, including those associated with the HPF, will be developed. During the conceptual design of the NGNP, a ***Process Hazards Assessment (PHA) for the Hydrogen Production Facility (HPF) will be initiated in accordance with Occupational Safety and Health Administration (OSHA) requirements***. This preliminary PHA will help establish the specific safety design requirements and criteria for safe operation of the NGNP." (Sec. 16.3.2, pp. 16-50 through 16-59) – This section provides a ***detailed plan for IHX R&D, including metallic and composite materials.***	• *Based on review of AREVA PCDR*: This item has been addressed in the AREVA PCDR in the design. • *Based on review of GA PCDR*: This item has been addressed in the General Atomics PCDR in the design. • *Based on review of WEC PCDR*: Indications are that this item has been addressed in the WEC PCDR, and will be further addressed with design improvements as the project progresses. • NGNP R&D Response: Safety analysis issue – no R&D impact.

NEXT GENERATION NUCLEAR PLANT PROJECT INFORMATION INPUT SHEET

1. Document Information

Document ID: NGNP PIRT/PCDR Reconciliation Report
Revision ID: ______
Project Number: 23843
Document Title/Description: Reconciliation of Phenomena Identification Ranking Tables and Preconceptual Design Reports
Sub-Project No.: ______
Date of Record: 11/01/08
Document Author/Creator: Bill Mangiante
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