September 27, 2011

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NRC Project #0748

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

SUBJECT: Contract No. DE-AC07-05ID14517 — Next Generation Nuclear Plant Project Submittal — Response to Nuclear Regulatory Commission Request for Additional Information Letter No. 004 Regarding Next Generation Nuclear Plant Project High Temperature Materials White Paper — NRC Project # 0748

Consistent with the actions identified in "NGNP Licensing Strategy – Report to Congress," dated August 2008, the purpose of this letter is to submit responses to the subject U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information regarding the subject Next Generation Nuclear Plant (NGNP) Project white paper. The enclosure contains the NGNP Project's responses for those Requests for Additional Information (RAIs) received in NRC RAI Letter Number 004 (Request for Additional Information No's. 5901, 5898, 5800, 5899, and 5900), dated July 25, 2011.

The NRC licensing process encourages early interactions to identify and resolve policy, regulatory, and key technical issues related to the proposed facility. Conducting effective interactions with the NRC is a critical part of the NGNP licensing strategy because the early resolution of issues can significantly impact the preparation of an acceptable license application, the subsequent application review schedule, and the ultimate deployment of the NGNP. This NGNP Porject response to the NRC's RAIs represents one in a series of submittals that address priority licensing topics related to establishing High Temperature Gas-Cooled Reactor (HTGR) regulatory requirements using the process outlined in the Licensing Strategy.

Following NRC Staff review of these RAI responses, and pending resolution of associated follow-on questions, the NGNP Project requests that the NRC provide feedback and documentation of its review in a format that will facilitate resolution of key design, safety, and licensing issues on the topic of high temperature materials that can be used as a firm basis for the preparation of future HTGR license application(s).

If you have any questions, please contact me at (208) 526-6063 or James Kinsey, Director, NGNP Regulatory Affairs at (208) 569-6751.

Sincerely,

G. A. Gibbs

Greg Gibbs, Project Director
Next Generation Nuclear Plant Project

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NRO

Enclosure:

1. NGNP Response to NRC RAI No's. 5901, 5898, 5800, 5899, and 5900, Revision 0.

References:

(a) "Next Generation Nuclear Plant – High Temperature Materials White Paper," June 25, 2010, CCN 221269

(b) NRC RAI Letter Number 004 (Request for Additional Information No's. 5901, 5898, 5800, 5899, and 5900), dated July 25, 2011

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Consolidated Draft RAI Responses for High Temperature Materials

Acronym List

AGC	Advanced Graphite Creep
AGR	Advanced Gas-cooled Reactor
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor
BPV	boiler and pressure vessel
CFRC	carbon fiber reinforced carbon
CSC	core structure ceramic
CTE	coefficient of thermal expansion
DDN	design data need
Dido	Materials Test Reactor (UK)
DOE	Department of Energy
dpa	displacements-per- atom
DPP	demonstration power plant
EDF	Electricité De France
EDN	equivalent Dido nickel
FSV	Fort St. Vrain
GA	General Atomics
GDC	general design criteria
HFR	high flux reactor
HTGR	high temperature gas-cooled reactor
HTM	high temperature material
HTR	high temperature reactor
HTR-PM	High Temperature Reactor-Pebble-bed Module (Chinese HTGR)
HTS	heat transport system
HTTR	High Temperature Test Reactor

IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
ISFS	independent spent fuel storage
JAEA	Japan Atomic Energy Agency
JAERI	Japanese Atomic Energy Research Institute, Tokai-Mura Japan
JNM	Journal of Nuclear Materials
KTA	Kerntechnische Ausschuss (German Standards Agency)
LBE	licensing basis event
LLC	limited liability company
LWR	light water reactor
MTR	materials test reactor
MQP	Material Qualification Plan
NDE	nondestructive examination
NEUP	Nuclear Energy University Program
NGNP	Next Generation Nuclear Plant
NIMS	National Institute for Materials Science
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
OSTI	Office of Science and Technical Information
PBMR	Pebble Bed Modular Reactor
PIE	post irradiation examination
PIRT	phenomena identification and ranking table
POF	probability of failure
Q/A	questions and answers
QA	quality assurance
R&D	research and development
RAI	request for additional information
RI-PB	risk informed and performance based

RIT	reactor inlet temperature
RPV	reactor pressure vessel
SCC	stress corrosion cracking
SGCC	Subgroup on Graphite Core Components (ASME)
SRC	structural reliability class
THTR	Thorium High-Temperature Reactor

Consolidated Draft RAIs Responses for High Temperature Materials

General RAIs No. 5901 Revision 0

RAI GEN-1: What are the effects of degradation and failure of sealants, gaskets, attachments to high temperature metal piping, etc. degradation, debris on high temperature metallic materials, graphite, and C-C and other composite materials?

Response GEN-1:
This question is pertinent to the issue of material compatibility and must be addressed during review of the design for the NGNP reactor. The NGNP design has not yet advanced to the level of detail at which these features have been defined. Therefore, this topic was not included within the outcome objectives summarized in Section 5 of the white paper. These potential degradation effects will be addressed as a part of the future license application process.

RAI GEN-2: What are the effects of insulation lift-off and consequent debris generation on high temperature metallic materials, graphite, and C-C and other composite materials?

Response GEN-2:
Specific effects of insulation lift-off and/or debris generation must be assessed once the reference design, technical specifications, components, and material selections have been established. Therefore, this topic was not included within the outcome objectives summarized in Section 5 of the white paper.

In presently proposed NGNP designs, this issue is limited to the hot duct assembly and upper plenum shroud, which are the only identified internal components employing internal insulation. As can be seen by referring to the response to RAI HTM-3, the generation of insulation-derived debris from the hot duct assembly would require a major failure of the internal liner of the hot duct assembly. For the upper plenum shroud, the generation of insulation-derived debris would require major failure of metallic cover plates.

RAI GEN-3: What type of specific information exists or needed to address aging management issues tailored to specific postulated or known degradation mechanism? This information is especially important to select the appropriate inspection methods, periods, and examination area.

Response GEN-3:
Specific aging issues for the graphite and metallic components are being actively pursued within the Graphite and High Temperature Materials Research & Development (R&D) Programs (References 1-3). Specific aging issues include irradiation induced dimensional changes and creep within graphite components, creep-fatigue and creep within metallic components, long-term environmental degradation of metallic and graphite components, and loss of mechanical strength and fracture resistance in graphite due to long term irradiation dose. Information from these NGNP programmatic studies will determine the appropriate mitigation and inspection measures required to assure the safety case of the high temperature gas-cooled reactor (HTGR). The NGNP program is not directly assessing the aging issues of composite systems

but is directing work being performed at university programs under the Nuclear Energy University Program (NEUP).

However, specific aging monitoring programs, monitoring methods, periods, and examination areas of components can only be determined once the reference design, technical requirements, components, and material selections have been established. Aging management will be addressed as a part of the future license application process.

References:

1. PLN-2497, 2010, "Graphite Technology Development Plan," Rev 1, October 2010.
2. PLN-2803, 2010, "Next Generation Nuclear Plant Pressure Vessel Materials Research and Development Plan," Rev. 1, July 2010.
3. PLN-2804, 2010, "Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan," Rev. 1, September 2010.

RAI GEN-4: The white paper does not address decommissioning issues related to NGNP HTGR graphite core components and other ceramic and carbon-carbon composites. Provide information regarding decommissioning of these materials.

Response GEN-4:
Decomissioning was not considered in the Materials High Temperature White Paper and was not an objective of this white paper. This topic will be addressed in the future licensing application process.

RAI GEN-5: In LWRs, the integrity of the reactor coolant pressure boundary is assured through a defense-in-depth approach, such limits on allowable identified reactor coolant leakage. Leakage through the pressure boundary is not allowed during power production. How will such defense-in-depth approach be applied to metallic pressure boundary components for the NGNP? In other words, how will GDC 30 be met?

Response GEN-5:
The programmatic defense-in-depth approach is addressed in INL/EXT-09-17139, "Next Generation Nuclear Plant Defense-in-Depth Approach." In general, NGNP defense-in-depth relies partially on a functional containment approach utilizing multiple physical barriers to radionuclide release, including the Helium Pressure Boundary. The Helium Pressure Boundary will be designed and fabricated to ASME Boiler and Pressure Vessel Code Section III requirements for metallic pressure boundary components. Details will be addressed during development of the detailed design and during the future license application process.

RAI GEN-6: The staff is aware of design data need (DDN) documents, which were generated by Westinghouse, AREVA, and GA for graphite. These vendors had analyzed the graphite PIRT results and their conceptual designs and identified several topical areas where more information is needed. This white paper does not provide any information on these documents, DOE's evaluation of these documents, and provide a nexus between the potential vendor's identified DDNs and DOE-INL's research which would supposedly provide such data to the designer. Provide a discussion of specific data needs identified by these DDNs. The staff is particularly interested in graphite spalling, especially during the later stages of reactor operation.

Response GEN-6:
The High Temperature Materials White Paper, as well as all of the other NGNP white papers, was written in a manner to be generic with regard to HTGR technology (i.e., independent of variations among potential reactor suppliers in design details and technology development needs). This approach is appropriate at this stage of the NGNP Project, with no decision having yet been made regarding which reactor supplier will conduct the final design of the NGNP. Accordingly, specific DDNs are not discussed in any of the NGNP white papers.

In general, the data needs and additional research and development needs identified in the PIRT reviews are consistent with the plans (Reference 1). Graphite oxidation from a steam incursion is being addressed through oxidation activities in the plan. Dust generation and tribology studies are being performed in university research programs under the NEUP program. The effects or probability of fracture failure leading to spallation of graphite components is being addressed under the multi-axial strength and failure mechanism development sections of the graphite technology development program.

Reference
1. PLN-2497, 2010, "Graphite Technology Development Plan," Rev 1, October 2010.

RAI GEN-7: In Section 2.8, it is stated that "Further development of the regulatory infrastructure will be desirable in support of follow-on commercial plants." However, there does not appear to be any further discussion within the white paper describing what infrastructure development is considered desirable to support a review of a HTGR commercial plant license application. Given that this regulatory infrastructure development could be beneficial for NGNP licensing, provide additional discussion of specific needs

Response GEN-7:
By stating that further regulatory infrastructure development to support modular HTGR licensing is desirable, the NGNP Project is referring to a future risk informed and performance based regulatory infrastructure. The project also believes that the ongoing regulatory gap analysis will support the proposed materials selection and qualification process proposed in this white paper.

Furthermore, as stated in Section 2.8 of the white paper – "*The candidate materials being considered for primary HTGR components are generally commercially available and are in use in high temperature applications in other industries and, in some cases, have been used in HTGR applications in the U.S. and other countries.*"

However, specific requirements will be addressed once the reference design, technical requirements, components, and material selections have been established. Therefore, details will be addressed during development of the detailed design and during the future license application process.

RAI GEN-8: In general, references have not been provided in the white paper. The staff needs corresponding technical citations for conducting an informed review. Provide a comprehensive list of references, and incorporate this information into any future revision of the white paper.

Response GEN-8:
Although some references are provided in Section 6 of the white paper, additional references that support technical citations have been provided in responses to RAIs. These references will be collected and will be provided in the next revision of the white paper, as appropriate. The additional reference materials will be those that are already publically available.

High Temperature Metals RAIs No. 5898 Revision 0

NGNP's General Response for High Temperature Metals RAIs

ASME has recently completed development of the Section III, Division 5 Rules for Construction of Nuclear Facility Components that apply to High Temperature Reactors, including HTGRs (Reference 1). According to the ASME Web site, release of the Division 5 Code is scheduled for October 31, 2011. In part, the new Division 5 Code incorporates updated versions of ASME Code Cases N-201-5 and N-499-2, which are referenced at several locations within the High Temperature Materials White Paper, notably including Section 2.7, which summarizes the status of ASME Code development, and Section 5.1, which identifies the outcome objectives for metallic materials. As the new Division 5 rules are published, it is anticipated that the referenced Code Cases will be annulled.

For this reason, Section 2.7, Section 5.1 and additional sections in the High Temperature Materials White Paper that identify these specific Code Cases will be modified to acknowledge the anticipated evolution from the present Code Cases to the new Division 5 rules. In the RAI responses below, references to Code Cases N-201-5 and N-499-2 should be understood in the above context. The modification to the white paper will be completed upon resolution of the issues raised in the RAIs.

Reference:

1. http://www.asme.org/products/codes---standards/bpvc-iii-5---2011-bpvc-section-iii-rules-for-const

RAI HTM-1: In Section 3.2.1.2, it is stated that "However, given the present extensive database and the large material thicknesses involved, oxidation effects are not expected to be significant, making the need for new data unlikely." What information is currently available on the "internal oxidation", and oxidation of grain boundaries, in addition to surface oxidation for these materials? How are the important performance-related properties affected by potential long-term internal oxidation?

Response HTM-1:
The properties of SA-508/533 are measured in air for the Code; therefore, the majority of properties that have been determined and tabulated include the effects of an oxidizing environment, including any potential internal oxidation. The reactor pressure vessel (RPV), which sees the highest vessel temperatures, normally operates at temperatures comparable to those seen by LWR vessels (in the range of 300°C in the NGNP steam cycle designs—see response to RAI HTM-2). The exterior of the vessel is exposed to air; the internal environment is the primary helium coolant, which normally contains low levels of oxidants (typically ≤10ppm total). Bounding transients seen by the RPV are limited by design to <540°C, the Code Case N-499-2 limit, and are of low frequency and limited duration.

RAI HTM-2: Regarding the Section 3.2.1.3 discussion of experience with SA-508/533, what information is available on the potential carburization due to the presence of potential carbon/graphite dust in the HTGR?

Response HTM-2:
Based on exposures of low alloy steel materials such as SA-508/533 to simulated HTGR environments, neither significant carburization nor decarburization was observed at temperatures below about 450°C (Reference 1). In the designs of interest to this the High Temperature Materials White Paper, the normal operating temperature of the vessel system is driven by the reactor inlet temperature (RIT), which is much lower, typically in the range of 300°C. For example, in recent prismatic steam cycle designs, the RIT ranges from 290–325°C; for the pebble bed HTR-Modul 200, the RIT is 250°C. The actual temperature of the vessel would be still lower by some 25–50°C, depending upon the details of the design.

No data specific to SA-508/533-carbon reaction/diffusion couples are known to exist. However, given the modest temperature and robust structure, no significant degree or effect of carburization would be expected.

Reference:
1. P. L. Rittenhouse, "Creep and Corrosion Behaviour of Ferritic Steels in a Simulated Steam-Cycle HTR Environment, High Temperature Materials Programme," HTMP Report No. 4, March 1977.

RAI HTM-3: In Section 3.2.2.2, it is stated that "Material selection and qualification for the hot duct liner is based mainly on high temperature strength, corrosion resistance, and time dependent stress effects such as creep and stress rupture." What technical considerations would be required with regard to erosion and erosion/corrosion effects due to potential carbonaceous dust and insulation debris flowing along with the helium coolant? What flow-induced vibration effects should be considered for the hot duct liner? Why is not the cross "vessel" not a duct? Will this cross "vessel" also contain hot duct liner?

Response HTM-3:
To provide a context for the responses to this and subsequent questions (i.e., RAIs HTM-5, HTM-7, and HTM-28), Figure 1 shows the overall arrangement of a nuclear heat supply system that is typical of the concepts of interest to this white paper. The reactor is contained in one vessel and the primary heat exchanger is contained in a second vessel, with the cross vessel providing a connection between the two. The primary heat exchanger is located to the side and below the level of the reactor to minimize natural circulation of hot gas to the primary heat exchanger during pressurized conduction cooldown events, when active heat removal with forced circulation is not available.

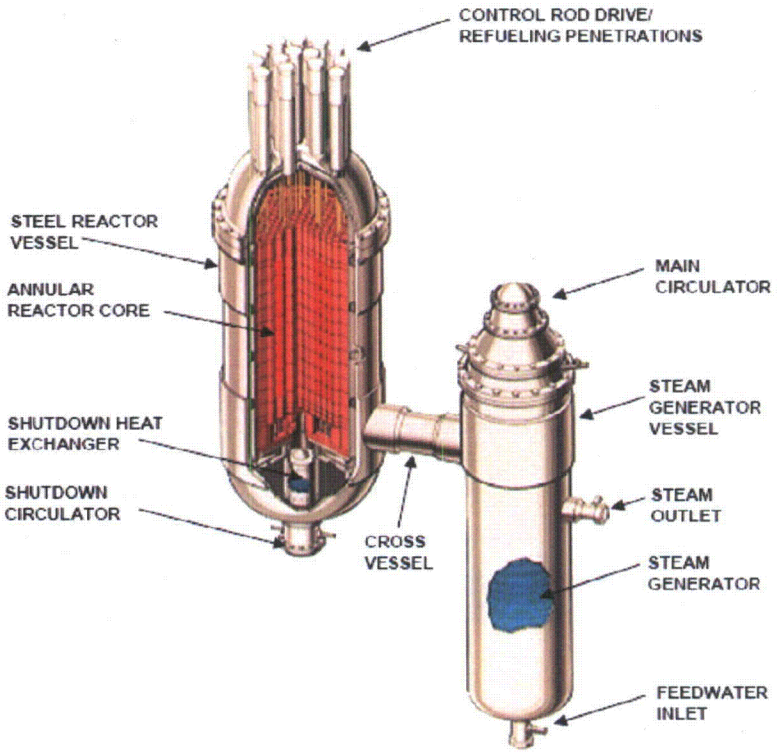

Figure 1. Typical Nuclear Heat Supply System Arrangement.

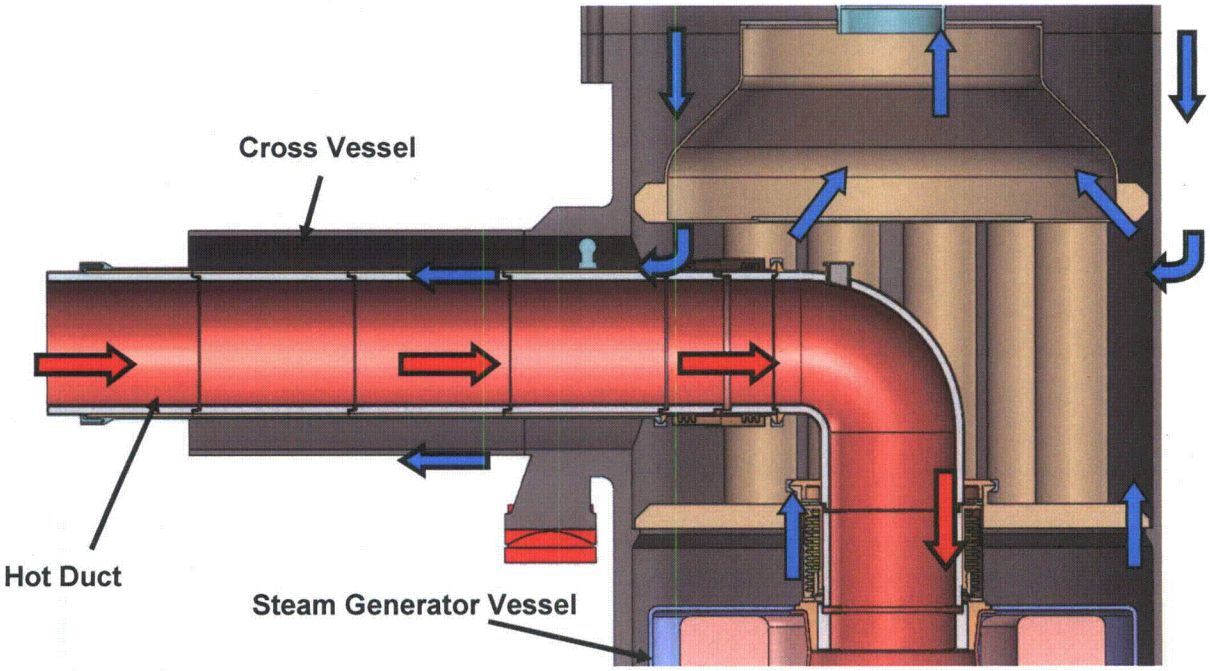

Figure 2. Typical Hot Gas Duct Arrangement within Cross Vessel.

A typical arrangement illustrating the concepts associated with the hot gas duct is shown in Figure 2. Note that the hot gas duct is located within and is concentric to the cross vessel. During normal operation, the high-temperature, lower pressure helium leaving the reactor (shown with the red arrows) flows within the hot gas duct, and the cool, higher pressure helium (blue arrows) returning from the steam generator and circulator flows in the annulus between the hot gas duct and the cross vessel. While details vary in specific designs, the hot gas duct functionally comprises three layers:

1) an outer layer that serves as an internal pressure boundary, with the maximum pressure difference being the pressure rise across the circulator;
2) an insulating layer that prevents regenerative heat exchange between reactor outlet and reactor inlet streams; and
3) an inner liner (not pressure tight) that retains the insulation and defines the flow path for the high-temperature helium leaving the reactor.
4)

Since the higher pressure is on the outside of the hot gas duct, any leakage would be from outside (cold) to inside (hot).

In response to the specific questions of this RAI, the cross vessel is a part of the vessel system that comprises the major part of the primary helium pressure boundary. It is designed as a vessel in accordance with the rules of ASME Section III, Class 1, just as are the reactor vessel and the steam generator (SG) vessel. The cross vessel has internals just as the reactor vessel and SG vessel; it encloses the hot gas duct, of which the liner is the innermost layer.

The thickness selected for the inner liner of the hot gas duct will include an allowance for corrosion and erosion that has not yet been established. The corrosion allowance will be based on the reactor outlet temperature and the primary coolant chemistry specification, in conjunction with materials corrosion data. The allowance for erosion will take into account the temperature, velocity, and geometry of the flow path; the number of dust and debris particles per unit of flow volume; and particle size, density, and hardness.

In general, the allowances for corrosion and erosion are not expected to be large relative to the thickness of the liner. Provisions will be included to visually inspect the interior of the hot gas duct as a means of detecting unanticipated deterioration. While the hot gas duct is designed for the plant life, it can be replaced, if necessary.

The potential for flow induced vibration and its effects on the hot gas duct liner will be addressed by analysis. The effects of vibration on the insulating properties of the hot gas duct will be characterized by testing.

RAI HTM-4: Section 3.2.2: For the use of Alloy 800H as core support component, what technical issues need to be addressed or known to have been resolved with respect to the effects of oxidation and erosion, and erosion/corrosion reducing the potential load bearing capacity in shear?

Response HTM-4:
Erosion and corrosion can reduce the wall thickness of components, and oxidation can affect the surface of materials by various means including degradation of the grain boundaries, thereby reducing the strength of the affected volume of surface material. Hence, the effects of

erosion and corrosion on the axial, shear, bending, and torsion load carrying capacity of core support components need to be taken into account during the design process. The ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (construction rules for core support structures) provides clear guidance. The first sentence of NG-3121 indicates:

> Material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provision made for these effects during the design or specified life of the structure by a suitable increase in or addition to the thickness of the base metal over that determined by the design formulas.

The NGNP Project is currently investigating the potential loss of load bearing area for 800H materials under quasi-static environmental conditions to support the quantification of this effect. Quantification of erosion/corrosion effects will be determined later after further design details become known.

Note that the core support component for which Alloy 800H is a candidate material is the core support structure, which comprises the core support floor and the core barrel. The core support structure encloses and supports the core graphite structures, including the fuel. During normal operation, the flow path is such that the temperatures seen by the core support structure are governed by the reactor inlet temperature, which is in the range of 300°C (see response to RAI HTM-2). At these temperatures, corrosion would be minimal, even if corrosive elements were available in the helium coolant. As noted above, the potential effects of erosion will be evaluated and will be addressed by a thickness allowance, if necessary; however, such effects are expected to be minimal in the inlet region.

RAI HTM-5: In Section 3.2.2.2 it is stated that "Studies sponsored by ASME Standards and Technology, LLC have determined that there is currently sufficient information available to extend Code qualification to 850°C for a maximum use temperature for 500,000 hours design life for lower temperatures." What information is available for Alloy 800H containing potential cracks, especially when used as hot-gas-duct liner? Are there crack growth rate data available under the conditions of expected design loads and service environments to provide assurance that potential crack growth would be within the tolerance limits, if any?

Response HTM-5:
Although the NGNP technology and development program is expecting to generate Alloy 800H crack growth data at a later date, those testing plans are not included in current NGNP Project documentation. Once these test plans have been formalized and documented, that information then can be shared with the NRC.

Note that since the hot gas duct liner does not serve a pressure retaining function, extensive cracking would be required to result in major failure of that component.

RAI HTM-6: In Section 3.2.2.2, a discussion has been provided on the bimetallic weld between the Alloy 800H tubing and ferritic steel tubing. It is mentioned that "However, special consideration will need to be given to corrosion questions, especially where alternating wet-dry conditions might exist." It is not clear what this special consideration is. From the LWR operating experience, the staff might potentially expect data on potential stress corrosion

cracking (SCC) as well as material loss due to corrosion, and the effects of wet-dry cycling on SCC.

Response HTM-6:
The word "special" will be deleted in the next update of the white paper. Corrosion is a normal consideration in the water-side environment of a steam generator, and alternate wetting and drying implies the potential for local concentration effects that have posed difficulties in LWR steam generators. It is noted that Alloy 800 is now a preferred tube material in German LWRs and has had excellent service experience in the more difficult two-phase wetted environment of the LWR steam generator shell side. In the HTGR steam generators, water is on the tube side, which avoids crevice concentration effects.

RAI HTM-7: In the Section 3.2.2.2 discussion of the hot duct liner, it is stated that "the hot duct liner will be under minimal stress so that the high temperature strength and time dependent stress effects that drive the 760°C (1400°F) limit may not directly apply. Further, as already noted earlier, data already available would support operation at even higher temperatures. During a pressurized conduction cool down event, the hot duct liner may exceed the current code temperature and, if so, would require further evaluation at these high temperatures for qualification." The staff notes that because of thermal transients, the potential for thermal cracking of the hot duct liner cannot be ruled out, especially in regions where changes in cross section geometry occur. What is the effect of the presence of pre-existing or operation-induced cracks on these available data?

Response HTM-7:
Refer to the response to RAI HTM-3 for a summary description of the hot gas duct.

The loads imposed on the hot gas duct liner (both thermal and structural) will be assessed using established thermal/structural analysis methods and available data for Alloy 800H, which have been established to minimize the incidence of failures, such as cracks. The liner is not a pressure boundary and, by design, will see minimal loads during normal operation and during other events within the duty cycle. While the possibility of locally high temperatures during pressurized conduction cooldown events (loss of forced cooling by both the main Heat Transport System (HTS) and the independent Shutdown Cooling System) has not been ruled out, the following are noted:

- Temperatures significantly exceeding those seen during normal operation are unlikely, since the main circulator includes a flow-assisted check valve that passively closes when the HTS circulator stops. The check valve is required for proper operation of the shutdown cooling system used for maintenance; however, it is not required for passive heat removal via the path through the reactor vessel. Also, the HTS design is configured to prevent natural circulation through the hot gas duct to the steam generator during conduction cooldown events.
- Pressurized conduction cooldown events are rare events with a correspondingly low number of cycles. A typical design duty cycle might include five such events for evaluation purposes.

- Since there is no active circulation, the hot gas duct liner has no assigned function during pressurized conduction cooldown events, and the presence or absence of cracks would have no bearing on the outcome of these events.

RAI HTM-8: In Section 3.2.2.2, it is stated that "At temperatures below 900°C (1652°F), extended operation studies in impure helium gas have shown that Alloy 800H develops chromia scales along with significant internal oxidation of aluminum and subsurface depletion of Cr. However, it must be noted that the corrosion behavior is quite sensitive to levels and ratios of active impurity species and this must be considered in the evaluation process."

The formation of corrosion-induced scale (oxidation product of the corrosion reaction) could be a potential regulatory concern. The adhesion characteristics of the heavy metal oxide scale could potentially dictate the potential for peeling of such scales over time due to a variety of reasons including the effect of the flowing coolant. Such peeling could result in the coolant gas acting as the carrier of the activated heavy metal radionuclide in the stream. What information is currently known or planned to be investigated in this regard?

Response HTM-8:
The corrosion behavior of Alloy 800H is dependent upon time, temperature and coolant chemistry. These considerations have been extensively studied in prior HTGR research and development (R&D) efforts for the range of conditions expected for the HTGR reactor design (Reference 1).

In the case of components normally exposed to the reactor inlet conditions (core support structures, inlet plenum thermal shield components, control rod drive components), the normal operating temperature of the inlet helium is in the range of 300°C (see response to RAI HTM-2), and no significant corrosion would be expected in these inlet areas. While higher temperatures might be seen as a result of certain licensing basis events (LBEs) involving conduction cooldown, they are both rare and of limited duration. Given these considerations, no significant degradation of inlet area metallic components is expected in most cases, and none is expected for structural components (e.g., core support structure). The one potential exception is control rod components if they are made of metallic materials. While the degradation would not be sufficient to affect the functions of control rod insertion and reactor shutdown, it may be necessary to replace the control rods following such rare events.

By contrast, the hot gas duct liner and certain hot end steam generator components (including the tubes) will be exposed to the reactor outlet temperature of 750°C during normal operation. Temperatures during LBEs would not significantly exceed the normal operation temperature and would be of limited duration. Data from prior R&D efforts indicate that, with proper chemistry control during normal operation, corrosion rates are acceptable at these temperatures and would be addressed through the normal application of a corrosion allowance.

Cobalt is the alloying element that would be of greatest potential concern for activation if it were carried through the core in the coolant stream. This element is in low concentration in Alloy 800H and has not been found to be present in significant quantity in the oxide scale. The microchemistry of the oxide scale resulting from environmental interaction with helium containing varying levels of impurities is being characterized in the NGNP Technology Development Program (References 2 - 4).

Reference:

1. Advanced Gas Cooled Nuclear Reactor Materials Evaluation and Development Program Quarterly Progress Reports, July 1, 1980 through September 30, 1983, DOE-ET-3402 – 51, -54, -57, -64, -67, -71, -73, -83, -85, -87, and –90.
2. PLN-2497, 2010, "Graphite Technology Development Plan," Rev 1, October 2010.
3. PLN-2803, 2010, "Next Generation Nuclear Plant Pressure Vessel Materials Research and Development Plan," Rev. 1, July 2010.
4. PLN-2804, 2010, "Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan," Rev. 1, September 2010.

RAI HTM-9: In Section 3.2.2.2, a statement is made that "Measurements of emissivity will be required after oxidation in air and helium to determine the most appropriate values." Similar statements could be found throughout the white paper for other properties and situations. As currently written, the white paper does not provide a roadmap to access these needed information.

Provide to the staff the status of addressing these information needs, including description of the programs and organizations that are addressing these needs. Also, discuss how the results are expected to inform the design, or confirm the existing design concepts.

Response HTM-9:
The NGNP High Temperature Materials Program has two documents that describe the NGNP high temperature material technology development needs [References 1, 2]. These two documents describe in detail all of the known issues that need to be addressed in the area of high temperature applications for metallic materials. They are updated whenever significant new information becomes available from the design teams.

During the next revision of the white paper, Section 3.2.2.2 (page 18) will be revised to provide clarification as follows:

> *Ensuring the performance of Alloy 800H components under accident conditions requires consideration of thermal properties such as emissivity and thermal diffusivity, as these are integral to the material's ability to meet applicable design requirements during a conduction cooldown event. Emissivity values after oxidation in air and helium are the most appropriate values. These experiments have been carried out at University of Wisconsin-Madison in a Nuclear Energy Research Initiative project and confirmed by results at University of Missouri-Columbia. The values agree with the expectation that essentially all chromium oxide scales have similar emissivity values. Nominal values of thermal diffusivity are available in ASME Section II, Part D at temperatures up to 815°C (1499°F); additional data are being determined by the NGNP Technology Development Program.*

References:

1. Plan PLN-2803, "Next Generation Nuclear Plant Pressure Vessel Materials Research and Development," Rev. 1, July 2010.
2. PLN-2804, "Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan," Rev. 1, September 2010.

RAI HTM-10: The discussion in Section 3.2.2.3 describes operating experience with Alloy 800H during operation of various HTGRs around the world. However, it is not clear to the staff that the history described is at full designed power output of the facilities. Provide additional information regarding the operational histories and how that experience is expected to be applicable to NGNP. What limitations are involved in extrapolating this operating experience to NGNP?

Response HTM-10:
The introductory paragraph to Section 3.2.2.3 was provided as background only to highlight the many hours of operation experienced in HTGR plants without notable issues with heat exchanger or steam generator materials. Detailed histories of the heat exchanger operating conditions that would allow direct extrapolation are not presently available to the NGNP Project.

RAI HTM-11: Section 3.2.2.3 states that "Extensive studies evaluating the effects of operating temperature on the performance of Alloy 800H have been performed." However, it is not clear if these studies were performed at the NGNP expected loads and atmosphere. Describe how these data will be extrapolated for the HTGR conditions, especially over long periods of reactor operation.

Response HTM-11:
Previous studies form part of the foundation for ASME's current provisions for the use of Alloy 800H up to 760°C (1400°F). The values in the Code that resulted from these experiments define the allowable stresses in the design. In addition, there are extensive data for these properties for a range of stresses at temperatures through 1000°C (1832°F). The influence of atmosphere has been examined for some properties, e.g., creep for up to 125,000 hours in the most likely HTGR helium chemistry. The Code allowable for design using 800H will be extended by ASME to a range of 850–900°C and up to 500,000 hours in air. This time and temperature will define the design limits for this alloy.

Note that in the present NGNP designs, Alloy 800H components in the reactor inlet area (see response to RAI HTM-8) will normally see temperatures close to the reactor inlet temperature and are expected to operate in the time-independent materials properties range except during limited duration LBEs (see response to RAI HTM-2). The Hot Gas Duct liner and certain steam generator components, including the tubing, will normally be exposed to reactor outlet conditions (see response to RAI HTM-3).

During the next revision of the white paper, Section 3.2.2.3 (page 18) will be revised to provide clarification as the following:

> *Extensive studies evaluating the effects of operating temperature on the performance of Alloy 800H have been performed. The minimum creep rate versus stress at 593–760°C (1100–1400°F) was determined using regression analysis. Fatigue behavior of Alloy 800H has been evaluated from room temperature to 760°C (1400°F) and low-cycle and high-cycle fatigue data were taken at 760°C (1400°F). These studies form part of the foundation for ASME's current provisions for the use of Alloy 800H up to 760°C (1400°F). The values in the Code that resulted from these experiments define the allowable stresses in the design. In addition, there are extensive data for these properties for a range of stresses at temperatures through 1000°C (1832°F).*

RAI HTM-12: Section 3.2.2.4 states that, for Alloy 800 H "..,the [ASME] code does not address other key requirements of the design of these components, such as the emissivity, corrosion resistance, thermal aging, and irradiation effects." What efforts are currently underway in the ASME Code development for this apparent deficiency? How does the NGNP project plan to address this deficiency?

Response HTM-12:
The ASME Code covers Alloy 800H in terms of high temperature strength and time dependent stress effects, such as creep and stress rupture. The standard atmosphere for generating data incorporated in the Code is laboratory air. Thus, the Code does not address other key requirements of the design of these components, such as the emissivity, corrosion resistance, thermal aging, and irradiation effects. All of these potential influences on the properties are being addressed in the NGNP Technology Development Program.

During the next revision of the white paper, Section 3.2.2.4 (page 19) will be revised to provide clarification as the following:

> *The ASME Code covers Alloy 800H in terms of high temperature strength and time dependent stress effects, such as creep and stress rupture. The standard atmosphere for generating data incorporated in the Code is laboratory air. Thus, the Code does not address other key requirements of the design of these components, such as the emissivity, corrosion resistance in the helium environment thermal aging, and irradiation effects. These potential influences on the properties are being addressed in the NGNP Technology Development Program.*

RAI HTM-13: In Section 3.2.3.2, it is stated that "An alternative under consideration is using a high temperature composite for the control rods." However, no information was provided in the white paper on this topic. Provide additional discussion of the materials being considered so that the staff can compare and evaluate the options.

Response HTM-13:
With regard to the specific question, if metallic alloy properties are not sufficient based on detailed design analysis, composite control rods may be considered. Both carbon fiber/carbon matrix and SiC fiber/SiC matrix composites have been suggested, however, there is no present activity under the NGNP Project to qualify these materials. Possible use of these materials is addressed in Section 3.5 of the High Temperature Materials White Paper.

At this time, the NGNP Project is not considering the use of Alloy X/XR at the current core outlet temperatures. Section 3.2.3 will be deleted from the whitepaper, as will other areas of the whitepaper that address Alloy X/XR.

RAI HTM-14: In Section 3.2.3.1, it is stated that "However, all of the components being considered are composed of relatively thick sections, so that the overall effects of corrosion are likely to be minimal." Given the absence of data on internal oxidation and its effect on significant properties, the selection of "relatively thick sections" may or may not be effective in mitigating progressive degradation due to potential (stress corrosion/fatigue) cracking. Provide additional justification for the claimed expected effect of corrosion.

Response HTM-14:
At this time, the NGNP Project is not considering the use of Alloy X/XR at the current core outlet temperatures. In the next revision, Section 3.2.3 will be deleted from the white paper, as will other areas of the white paper that address Alloy X/XR.

RAI HTM-15: In Section 3.2.3.3 states that "Testing of rupture time variation with applied stress showed that Alloy X would not rupture at 7 MPa (1000 psi) and 871°C (1600°F) during the 60-year life of the plant. Test data also indicated that the creep rate at 871°C (1600°F) would be insignificant. These studies may later form part of the foundation for ASME allowing use of Alloy X at up to 871°C (1600°F) during normal operation." Considering the possibility of micro and macro cracks after some duration of reactor operation, what creep and rupture data exist to confirm the applicability of such data in the presence of cracks?

Response HTM-15:
At this time, the NGNP Project is not considering the use of Alloy X/XR at the current core outlet temperatures. In the next revision, Section 3.2.3 will be deleted from the white paper, as will other areas of the white paper that address Alloy X/XR.

RAI HTM-16: Section 3.2.3.4 states that "The ASME Code does not currently address key requirements of the design of Alloy X/XR components such as corrosion resistance, thermal aging effects, irradiation effects, high temperature strength, and time dependent stress effects such as creep and stress rupture, so the qualification of these materials will require further evaluation." Does NGNP research address these evaluations? If so, what is the status of these evaluations?

Response HTM-16:
At this time, the NGNP Project is not considering the use of Alloy X/XR at the current core outlet temperatures. In the next revision, Section 3.2.3 will be deleted from the white paper, as will other areas of the white paper that address Alloy X/XR.

RAI HTM-17: Section 3.2.4.2 does not list fatigue as one of the important considerations for modified 9Cr-1Mo alloy as a material for core support structure. However, limited information is provided in Section 3.2.4.3 on this issue. Provide additional information regarding the potential limitations involved in extrapolation of these limited data to 60-year behavior.

Response HTM-17:
Section 3.2.4.2 should have listed fatigue as an important consideration for designing components made from modified 9Cr-1Mo alloy. During the next revision, the white paper will be corrected to include fatigue in Section 3.2.4.2. The newly approved ASME Boiler and Pressure Vessel Code, Section III, Division 5 provides references to the fatigue curve for modified 9Cr-1Mo alloy (for elevated temperature evaluations) via Appendix T in Subsection NH of Division 1. This strain range-allowable cycles curve (Fig. T-1420-1E) contains data out to 10^8 cycles.

RAI HTM-18: In section 3.2.5, and other places throughout the white paper, a 60-year design has been proposed for several components. Reference is also made for data availability for definite time periods, such as, for example, 300,000 hr. How many hours do you anticipate that the 60 year life represents?

Response HTM-18:
The equivalent number of operational hours associated with a 60-year design lifetime is considered to be 500,000 hours. This value is obtained by determining the total number of hours over a 60-year time interval and assuming a 95% availability factor.

$$(60 \text{ years})\times(365.25 \text{ days/year})\times(24 \text{ hours/day})(.95) = 499662 \text{ hours} \approx 500{,}000 \text{ hours}$$

The 95% availability factor equates to about three months of shutdown every five years, which is conservative for time of use evaluations.

RAI HTM-19: There are apparently contradictory statements in Section 3.2.4.2. While comparing 300,000 hour data availability, a statement is made that "Whether this [behavior] remains true for the HTGR 60-year design life will be evaluated during qualification." However, while considering thermal aging, data are available for only 75,000 hours, while a statement is made "Yield strength, ultimate tensile strength, and ductility are not significantly affected. Therefore, thermal aging of this material is not expected to be an issue." The staff needs some clarification on this apparent discrepancy in concepts.

Response HTM-19:
For the HTGR design, the core support structure/core barrel is the only component for which Modified 9Cr-1Mo is being considered. In this application, the normal operating temperature is expected to governed by the reactor inlet temperature (in the range of 300°C), and will be in the time-independent materials properties range. Operation at elevated temperatures during rare LBEs involving conduction cooldown would be of limited duration and is expected to be within the range of the ASME Section III coverage. Also, see response to RAI HTM-2.

RAI HTM-20: Section 3.2.4.2 states that "No particular corrosion concerns are expected for Modified 9Cr-1Mo at the service temperatures of the HTGR." Provide justification for this statement, including possible effects of internal oxidation, and describing any research to evaluate this aspect for 9Cr-1Mo alloy.

Response HTM-20:
Modified 9Cr-1Mo is not currently specified for a pressure retaining application in the present steam cycle NGNP designs. Regarding its potential use in the core support structure (including the core support floor and core barrel), service temperatures during normal operation would be governed by the reactor inlet temperature (in the range of 300°C), except during rare LBEs involving conduction cooldown. During such events, temperatures could be elevated for limited durations (see response to RAI HTM-2). It is further noted that this alloy is a chromia former, as are Alloys 800H and 617. For these reasons, corrosion is not a particular concern. Also, see response to RAI HTM-2. Additional discussion of this topic is further provided in References 1 and 2.

References:
1. Shah V. N., et al., Argonne National Laboratory, *Review and Assessment of Codes and Proceduresfor HTGR Components;* NUREG/CR-6816; June 2003.
2. Natesan K., et al., Argonne National Laboratory, *Materials Behavior in HTGR Environments;* ANL-02/37 NUREG/CR-6824; February 2003.

RAI HTM-21: As mentioned in Section 3.2.4.3, the use of 9Cr-1Mo alloy in the fossil fuel industry has had qualified success due to the lack and/or implementation of quality assurance in the production of the components made from this material. It is stated that "Special care must be taken during processing, fabrication, and installation to create and maintain the proper microstructure to obtain the desired material properties. It is not currently possible to insure that the steel is properly heat treated through-thickness by means of nondestructive examination. The necessity for pre-weld and post-weld heat treatment makes onsite fabrication of components from this steel problematic." Provide a discussion of how these issues will be addressed in the intended nuclear application.

Response HTM-21:
There are currently no plans to consider the use of Modified 9Cr-1Mo for the current HTGR steam cycle concept in a pressure-retaining application. For most of the core internals and support structures where Modified 9Cr-1Mo is being considered, the fabrication would not be performed on-site. Where onsite fabrication is necessary, it would be performed to the same quality standards as shop fabrication, including heat treatments. If Modified 9Cr-1Mo is selected for consideration for the core barrel, the wall thickness would be thin enough that current fabrication techniques would be used. The weld and inspection techniques would need to be developed and qualified.

For potential future applications requiring thick section fabrication of components using Modified 9Cr-1Mo, a thorough qualification program would be required. Such components are not planned for the current HTGR steam cycle concept.

In the next update of the High Temperature Materials White Paper, the sentence in the second paragraph of Section 3.2.4.3 will be deleted to provide clarification;

> *It is not currently possible to insure that the steel is properly heat treated through-thickness by means of nondestructive examination.*

While technically true, NDE is not the basis for such determinations. Rather it is accomplished by design and analysis of the heat treatment arrangement and confirmed by the monitoring of temperatures during heat treatment.

RAI HTM-22: At the end of Section 3.2.4.3 it is stated that "Fracture toughness is good and relatively constant with a KJQ value of ~275 MPa[m]1/2 from room temperature through 200 °C (392 °F); irradiation to 3 dpa reduces KJQ to ~100 MPa[m]1/2, but this is still a substantial value." This statement is potentially arguable since the expected operating temperatures are potentially much higher. In addition, "substantial" is a relative and subjective term, so the staff is unable to ascertain what the minimum value should be required for a given level of confidence in component performance. Therefore, provide additional discussion addressing these issues.

Response HTM-22:
Fracture toughness requirements for LWR components are addressed in the ASME Code, Section III, and, for pressure retaining components, in 10 CFR 50, Appendix G (Fracture Toughness Requirements), which further references Section XI of the ASME Code. The same requirements (including the requirement to account for the effects of neutron irradiation) are judged to be applicable to the modular HTGR. Since Modified 9Cr-1Mo is not being considered

for pressure retaining components in HTGR, concerns with this material related to startup and pressurization are not applicable in the present HTGR steam cycle designs. In the case of core support components, loads are relatively low and temperatures modest during normal operation (in the range of 300°C; see the response to RAI HTM-2). Note also that fracture toughness improves with temperature over the operating range of interest.

RAI HTM-23: In section 3.2.5.2, fatigue is not listed as one of the important considerations for 2.25Cr-1Mo alloy as a material for cold-end steam generator tubing. Provide justification for not including fatigue in these considerations.

Response HTM-23:
A discussion of fatigue should have been provided in Section 3.2.5.2. Therefore, during the next revision of the white paper, Section 3.2.5.2 (page 24) will be revised as follows:

> *For the cold-end steam generator tubing, material selection and qualification is based on high temperature strength, thermal aging effects, time dependent stress effects, thermal conductivity, fatigue resistance, and corrosion resistance.*

RAI HTM-24: In the Section 3.2.5.2 discussion of the use of 2.25Cr-1Mo alloy for cold-end steam generator tubing, it is stated that "Thermal conductivity is integral to assessing the ability of tubing to transfer heat efficiently from the primary helium gas to the secondary side water. Measurements will be required after oxidation in air and helium to determine whether degradation of heat transfer properties will need to be taken into account." However, it was also stated in Section 3.2.5.1 that such tubing "will be exposed to helium and water during normal operation". Provide a rationale as to why such tests are not required after oxidation in water environment.

Response HTM-24:
A discussion of oxidation in water should have been provided in Section 3.2.5.2. Therefore, during the next revision of the white paper, Section 3.2.5.2 (page 24) will be revised as follows:

> *Thermal conductivity is integral to assessing the ability of tubing to transfer heat efficiently from the primary helium gas to the secondary side water. Measurements will be required after oxidation in air, secondary water or steam, and helium to determine whether degradation of heat transfer properties will need to be taken into account.*

RAI HTM-25: In Section 3.2.5.2, regarding the formation of protective layer for continued corrosion possibility, it is stated that "In the HTGR cold-end steam generator tubing, the water temperature will be about 400°C (752°F), which should cause the formation of a protective layer of Fe_3O4." Provide information on the adhesion capacity and lift-off characteristics of this protective (apparently in-situ formed) coating for the NGNP operating environment.

Response HTM-25:
There is extensive experience with the use of this steel in fossil boiler applications and more limited experience in HTGRs and sodium cooled reactors. The native oxide has shown good adherence under the conditions encountered in these prior applications. The tenacity of this oxide may require experimental verification in either thermal cycling or flow testing if further design information suggests conditions that are significantly outside the base of experience (not expected).

In addition, during the next revision of the white paper, Section 3.2.5.2 (page 24) will be revised to provide clarification as follows:

> *In the HTGR cold-end steam generator tubing, the water temperature will be about 400°C (752°F), which experience shows will cause the formation of a protective layer of Fe_3O_4.*

RAI HTM-26: In Section 3.2.5.3, and in other places in the white paper, it is stated that "The Japanese HTTR has operated for over 10 years." However, this statement is vague, and does not adequately characterize the HTTR operating experience for evaluation of material performance. Provide additional information regarding the HTTR operating history, including actual years of full design power reactor operation, the cumulative fluence (dose), and the process parameters (i.e., temperatures, pressures, chemical environment) experienced by the various materials and components referenced in the white paper to provide some meaningful comparison.

Response HTM-26:
The statement regarding HTTR was provided as background information. There is currently no intention to utilize HTTR materials operating data to proceed with design work for the NGNP Project. With regard to the 2.25Cr-1Mo material, existing data as available in the ASME BPV Code and other databases will be utilized for HTGR design. Also, as discussed in Section 3.2.5.4, the Code does not address key requirements of the design components, such as corrosion resistance and thermal aging effects. Therefore, additional research & development effort will be required.

RAI HTM-27: In Section 3.2.6.2, fatigue is not listed as one of the important considerations for 316H stainless steel as a material for core barrel assembly. Provide justification for not including fatigue in these considerations.

Response HTM-27:
A discussion of the potential effect of fatigue should have been provided in Section 3.2.6.2. Therefore, during the next revision of the white paper, Section 3.2.6.2 (page 25) will be revised to provide the following:

> *Material selection criteria for the core barrel assembly are dominated by high temperature strength, resistance to fatigue and vibration, thermal conductivity, and resistance to oxidation and neutron irradiation.*

RAI HTM-28: A potential issue for materials used for hot duct liner is the possible chemical reaction with the material it is adjacent to it. For example, if the hot duct design involves tube-in-tube configuration with the inner tube consisting of the so-called "hot duct liner" and the outer tube of material not experiencing very high temperature with ceramic insulation in-between, then potentially two issues could be envisioned. First, in areas where the ceramic insulation is not in contact with the inner liner, those areas would not dissipate heat via conduction as readily as those areas where contact with the ceramic insulation exists, which could lead to the formation of hot spots in the inner liner. Over time, creep could occur due to material softening. Second, in areas where contact with ceramic insulation occurs, the constituents of the ceramic insulation could thermodynamically react with the liner material resulting in corrosion, pitting, cracking, and

eventually stress corrosion cracking. If a crack breaks through the liner, the ceramic insulation could be loosened due to high coolant flow and ceramic insulation could be carried as debris with the coolant, which could also lead to erosion, erosion corrosion of the liner and other contacting components. Provide information on how these potential issues are addressed in NGNP research and design.

Response HTM-28:
Refer to the response to RAI HTM-3 for a description of the hot gas duct.

The interior of the hot gas duct liner will, by design, operate at the reactor outlet temperature during normal operation when active flow is present; thus, the formation of insulation-related "hot spots" is not a relevant issue. As noted in the response to RAI HTM-7 the stresses seen by the liner are minimal by design, and the thermal structural characteristics of the liner will be addressed by analysis.

The compatibility of the liner and insulation material will be addressed by testing where data are not presently available. Testing of the insulating characteristics of the liner, including vibration testing and cyclic thermal testing are needed to confirm the expected performance of the insulation system and to identify any potential for insulation degradation resulting from the plant design duty cycle. Details regarding the above will be addressed during detailed design efforts and the future license application process.

RAI HTM-29: In Section 3.2.2.4 and other locations, reference is made to German Standard KTA 3221. For example, Section 3.2.24 states that "German Standard KTA 3221 allows use of Alloy 800H up to 1000°C (1832°F)." However, the rationale for this allowance does not appear to have been provided in the text in the white paper. Section 3.2.2.4 states that "An ASME and DOE joint effort is currently underway to obtain the basis of the KTA 3221 draft standard, including information on the quality assurance program under which the data were collected." Provide additional information which to describe the basis for the KTA 3221 standard and justify its applicability to NGNP.

Response HTM-29:
The NGNP Project does not plan to use the German Standard KTA 3221 for design purposes. It was only a draft standard that was never finalized. However, the underlying data, which includes work at Petten that was undertaken in support of the HTGR programs in Europe, are being used in an ongoing ASME/DOE Generation IV Reactor Material Project task (identified as Task 13) being managed by ASME Standards Technology, LLC. In the work on Task 13, the ASME material experts have made an effort to identify the sources of the data and to show how one source stands relative to another. Generally, the ASME material experts found that the Petten data produced more conservative stress values than the original US-produced values or the Japanese National Institute for Materials Science (NIMS) values. Any incorporation of this data will be subject to the strenuous review and approval process of the ASME BPV II Standards Committee on Materials. These issues are expected to be discussed in a final report on Task 13, anticipated to be available by the end of 2011.

RAI HTM-30: Section 4.1.4 states that "In order for a metallic material and, more generally, any structural material to be considered for use in the HTGR, it must be qualified for the appropriate service conditions and environment. In this usage, qualification implies that the material has

been evaluated, based on a set of experimental data sufficient to reliably describe its behavior, and found to be able to meet the requirements placed upon it by the design for conditions of operation." Please state how the Code Cases mentioned in Section 4.1.3, (Code Case N-201-5 and Section III, Subsection NH) will assist in qualifying these metallic structural materials (as indicated in Section 5.1).

Response HTM-30:
The definition proposed for the use of the word "qualification" from Section 4.1.4 and repeated in the RAI is consistent with the intent of its use in the white paper. In the next revision of the White Paper, Section 5.1 will be further updated to make it clear that the ASME Code is not proposed as the exclusive basis for qualification. This will be done in conjunction with the changes noted in NGNP's General Response for Metals RAIs at the beginning of the Metals RAI section.

A further requirement for successful licensing is that the qualification findings must also be acknowledged by the NRC in cases where safety is an issue. Acceptance of the mentioned Code Cases by the NRC would provide a basis for regulatory compliance for evaluating the modular HTGR materials and associated application functions and requirements. While NRC acceptance of these Code Cases would significantly reduce the burden on the designer, it is acknowledged that evaluation against the Code and/or Code Cases alone is not sufficient.

Graphite RAIs No. 5800 Revision 0

RAI G-1: Section 3.1. Are there data or will data be generated for the compressive strength of graphite as functions of cumulative dose and temperature? How will such data be used in design?

Response G-1:
The NGNP Technology Development Program includes compression testing of nonirradiated samples as part of the billet characterization program. The use of nonirradiated graphite compression properties is consistent with the ASME graphite design rules (see paragraph below) and is conservative within the range of the NGNP application. As further validation, the irradiation program will include compression testing of a limited number of samples. The response to RAI G-25 provides additional discussion regarding the use of nonirradidated graphite data in the ASME code.

Currently, the NGNP Project is developing a document to further elucidate the ASME graphite design rules. The document will provide the background theories used in the ASME design rules and their application in code assessments (e.g., compressive strength requirements, use of tensile strength for determination of probability of failure requirements for various structural reliability classes, how the graphite component lifetime is determined using the design code). The plan is for this document to be an addendum to the updated white paper in the future. If the document becomes available prior to the update to the High Temperature Materials White Paper, it will be provided as a reference.

The ASME graphite design rules require the Design Specification to include key graphite material property requirements for graphite core components. For a list of required nonirradiated and irradiated mechanical property requirements for use with the ASME assessment methods, see ASME, Section III, Division 5, Subsection HH, Subpart A and Subpart A Appendixes.

During the next revision of the white paper, Section 3.1 (page 13) will be revised to provide clarification regarding the ASME design process as follows:

> *The above information would then be used to develop the component design and performance specification, which will state the component material and mechanical requirements under normal and accident conditions. For example, the use of ASME design rules requires a Design Specification that includes key material property requirements for graphite core components. The ASME graphite design rules provide a list of required nonirradiated and irradiated mechanical property requirements for use with the ASME graphite design rules.*

RAI G-2:
Deleted by the NRC.

RAI G-3:
Deleted by the NRC.

RAI G-4: The first paragraph of Section 3.3 briefly outlines graphite qualification testing. The completeness of the oxidation data may be questionable since in-pore diffusion data may not be available for reactor operating conditions. The significance of the diffusion characteristics as a function of oxidation depth needs to be established to understand the uncertainties in the bulk oxidation data, obtained in the kinetics region alone. How will planned graphite qualification testing ensure adequate understanding of oxidation?

Response G-4:
Oxidation occurs in three primary regimes: (1) kinetically controlled regime, (2) in-pore diffusion controlled regime; and (3) boundary-layer diffusion controlled regime. Understanding the phenomena that control all three regimes is important for predicting overall oxidation behavior. The kinetically controlled regime is characterized by lower temperatures, essentially unlimited supply of oxidants, limited diffusion of oxidants into the graphite microstructure, and nearly uniform oxidation with depth into the graphite. The in-pore diffusion controlled regime is characterized by intermediate temperatures with both kinetics and diffusion through the graphite pores controlling the overall oxidation rate. As a result, the level of oxidation (sometimes referred to as "burnoff") decreases with depth into the graphite. The boundary-layer diffusion controlled regime occurs at higher temperatures and oxidants are reacted before they can diffuse to any significant depth into the graphite. As a result, the overall oxidation rate is essentially independent of kinetics and diffusion within the graphite, and the graphite ablates at the surface. The transition temperatures for these regimes will depend on the specific grade of graphite, the oxidant (e.g., oxygen vs. water vapor), and the flow regime. The phenomena that control all three regimes are well understood and models have been developed that seamlessly account for all three regimes. These models have been verified and validated to a limited extent using data from previous experiments.

For oxidation of graphite with air in the kinetically-controlled regime, ASTM has recently developed a standard for measuring the kinetic parameters (ASTM D-7542). General Atomics has used similar procedures to measure the kinetic parameters for oxidation of grades H-327 and H-451 graphite by oxygen and water vapor. Previous data for H-451 graphite have shown that the oxidation rate peaks at an intermediate level of burnoff (~20–40%) in the kinetically controlled regime, and models have been developed to account for this burnoff dependence. The kinetic rate expressions are different for oxidation by oxygen and water vapor. For oxygen, the kinetic rate generally has a power-law dependence with oxygen concentration, with the exponent ranging from 0.5–1.0. For water vapor, the kinetic rate generally follows a Langmuir-Henschelwood mechanism. In this mechanism, the rate is proportional to water vapor concentration at low concentrations, is independent of water vapor concentration at high concentrations, and is inhibited by the reaction product (hydrogen).

In the in-pore diffusion controlled regime, the effective diffusivity is an additional parameter that must be measured in order to predict the overall oxidation rate. The effective diffusivity also depends on burnoff, and correlations have been developed to account for this effect. Typically, graphite specimens that have been oxidized in the kinetically-controlled regime to different levels of burnoff (uniform burnoff throughout the specimen) are used for effective diffusivity measurements.

In the boundary-layer diffusion controlled regime, the parameter of primary interest is the effective mass transfer coefficient of the oxidant across the boundary layer, accounting for counter diffusion of reaction products.

An ongoing activity in NGNP Graphite Technology Development Plan (Reference 1) will provide the data to determine the parameters described above and their effects on the rate of oxidation. Further, various university research grants have been awarded in this area with the intent to determine the effects of oxidation on graphite performance, specifically at normal operating conditions. These data will be combined with the programmatic data generated from the NGNP graphite research and development program to ensure an adequate understanding of oxidation mechanisms in all three oxidation regimes.

As discussed above, there are oxidation data and models for previous grades of graphite, including grades H-327 and H-451 used in the Ft. St. Vrain reactor. The NGNP program has completed a literature review of available oxidation reports and literature from past reactor development programs.

Note that the potential for significant oxidation would be limited to rare and limited duration LBEs. As stated in Section 3.3.7.1 of the white paper, "Oxidation of graphite components must also be considered, however, its influence on component strength and, hence, structural integrity is not expected to be significant for events within the design basis." This is due to the limited time for oxidation during one of these events (seconds to minutes for water ingress; hours to a few days for air ingress), and further mitigated by the limitations on oxidant ingress that are integral to the respective designs.

Reference:

1. PLN-2497, 2010, "Graphite Technology Development Plan," Rev 1, October 2010.

RAI G-5: Section 3.3.1 states that the fuel cycle for a prismatic reactor is expected to be about 18 months long, with about half of the fuel elements replaced at the end of that cycle. Since the fuel compacts are packed inside the fuel element block, does this mean that the fuel element blocks are replaced at 18-month intervals? Are the fuel "elements" the same as fuel "blocks?" What experience exists currently in replacing fuel blocks in a safe manner, without affecting the structural integrity of adjacent blocks or other core components, which could potentially be more brittle due to irradiation hardening and damage? During the lifetime contemplated for the reactor, how will the irradiated blocks be stored to ensure their long term integrity? What industry demonstrated methods of safe disposition currently exist for these irradiated blocks?

Response G-5:
Based on multiple questions in RAI G-5, questions and answers (Q/A) have been separated, as given below:

Q. Since the fuel compacts are packed inside the fuel element block, does this mean that the fuel element blocks are replaced at 18-month intervals? Are the fuel "elements" the same as fuel "blocks"?

A. Yes, the fuel "elements" are the same as fuel "blocks" and consist of graphite blocks loaded with fuel compacts and burnable poison compacts, which are sealed inside the graphite

block. In typical prismatic reactor fuel cycles, half of the fuel elements are replaced at intervals ranging from approximately 18–24 months. Thus, individual fuel blocks may reside in the core for up to four years.

Q. What experience exists currently in replacing fuel blocks in a safe manner, without affecting the structural integrity of adjacent blocks or other core components, which could potentially be more brittle due to irradiation hardening and damage?

A. The Fort St. Vrain reactor was successfully refueled on three occasions and was defueled for decommissioning with no damage to fuel or reflector elements. No prismatic reactor graphite core components experience irradiation doses exceeding turnaround, thus avoiding doses approaching the ASME cohesive limit.

Q. During the lifetime contemplated for the reactor, how will the irradiated blocks be stored to ensure their long term integrity?

A. For fuel being directly removed from the reactor temporary storage and cooling of used fuel elements can utilizes dry wells within pools of circulating water. Design options are also available for utilizing natural-convection air-cooling for this system. After one to two years of temporary storage and cooling, the fuel elements are transferred to long-term dry storage within the Nuclear Island of the plant site.

The non-fueled graphite core components in Fort St. Vrain were disposed as low level waste. Used graphite fuel blocks (graphite blocks containing fuel compacts) are stored on the Fort St. Vrain site at the Independent Spent Fuel Storage (ISFS) installation operated by the Department of Energy under NRC Docket Number 72-09. This is a dry well storage system utilizing natural-convection air-cooling. The fuel blocks are being stored in the interim until Department of Energy has identified final deposition for the fuel.

Q. What industry-demonstrated methods of safe disposition currently exist for these irradiated blocks?

A. Safe disposition of Modular Helium Reactor spent fuel elements has been evaluated [1] and it was concluded that these fuel elements are a robust waste form for permanent disposal in a geologic repository. If a strategy were adopted to separately dispose of fuel compacts and the graphite blocks (to reduce high-level waste volume), the blocks themselves would meet the current requirements for Class C low-level waste, based on a conservative activation analysis. Additionally, spent nuclear fuel from Fort St. Vrain and Peach Bottom Cores 1 and 2 are included in the DOE application for geological storage of spent nuclear fuel at the Yucca Mountain repository under NRC Docket 63-001.

During the next revision of the white paper, Section 3.3.1 (page 29) will be revised to provide clarification regarding the ASME design process as the following:

> *Both design concepts employ permanent and replaceable reflector components, the permanent sections usually being the peripheral reflector regions exposed to a much lower fluence and irradiation temperature. In prismatic designs, refueling outages occur at approximately 18-month intervals, during which one-half of the fuel elements are*

replaced. The refueling interval and fuel element replacement schedule are presently governed by fuel cycle considerations. The inner reflector elements in prismatic designs would typically be replaced at 6-year intervals during one of the refueling outages, leaving only the permanent reflectors to last the lifetime of the plant. In general, the lifetime and replacement schedules of replaceable reflector elements are dependent on the accumulated fast neutron fluence. As a result, elements adjacent to the active core will have shorter lifetimes. Previous assessments performed in the 1980s for the steam-cycle modular HTGR (based on stress and thermal analyses) have shown the expected lifetimes can range from 3 years (for outer reflector blocks with control rods adjacent to the active core) to 10 years (for standard reflector elements further away from the active core), with an overall average replacement schedule of approximately 6 years. For the NGNP demonstration plant, these lifetimes will be re-evaluated based on the ASME graphite design rules that are under development.

Reference:

1. General Atomics, 2002, "Assessment of GT-MHR Spent Fuel Characteristics and Repository Performance," PC-000502, Rev. 0, April 2002.

RAI G-6: Section 3.3.1 states that inner reflector elements will be replaced about every 6 years: What is the technical basis for replacement at 6-year intervals? What in-service inspections and evaluations are planned to ensure any potential degradation during this 6-year interval is detected and understood so that the elements meet design requirements? Is it anticipated that reflector blocks may be returned to the reactor? If so, what criteria, including internal cracking, will be used to justify their continued service?

Response G-6:
The replacement schedule is an initial estimate based upon the expected dose levels at this location in the core. The criterion behind this estimate is to keep the dose below levels that approach turnaround to ensure that the graphite components remain in a compressive stress state, thus minimizing the probability of fracture. However, this initial estimate may change based upon the results from NGNP graphite irradiation experiments, wherein the onset of turnaround will be determined as a function of dose for each graphite grade that has been identified as a potential candidate for NGNP.

Inspection of replaceable reflector elements removed from the reactor during refueling operations will provide additional information that will augment the replacement schedule decisions based upon irradiation experiments performed by the NGNP Technology Development Program.

During the next revision of the white paper, Section 3.3.1 (page 29) will be revised as shown in the response to RAI G-5 to provide clarification regarding the ASME design process.

RAI G-7: Section 3.3.1. Describe the operating experience lessons learned from reflect block replacement in gas reactors. For example, how does the potential irradiation-induced brittleness affect safe removal of the block without damage to the core, such as impact of falling chunks from replacement blocks, sidewall friction to adjacent blocks, etc.?

Response G-7:
The Fort St. Vrain reactor was successfully refueled on three occasions and was defueled for decommissioning with no damage to fuel or reflector elements. The replaceable reflector blocks were removed and replaced in the same manner and using the same equipment as the fuel elements. The fuel elements and reflector blocks were designed with appropriate dimensions and tolerances to provide the small gaps needed for installation and removal of these components. At shutdown, irradiation-induced shrinkage serves to slightly widen these gaps. During the cold shutdown conditions for refueling, the gaps were generally somewhat wider than they were at the beginning of the fuel cycle. Prismatic graphite core components receive doses well short of the cohesive life limit, therefore spalling is not expected.

Reference 1 provides a summary of Fort St. Vrain operating and maintenance experience. The Fort St. Vrain fuel handling system demonstrated that refueling could be accomplished quickly and efficiently with no damage to fuel or reflector elements and with minimal worker dose.

In the case of pebble bed reactors, the core would be unloaded prior to replacement of reflector components.

Reference:
1. W.A. Simon, A.J. Kennedy, and D.W. Warembourg, 1992, "The Fort St. Vrain Power Station Operating and Maintenance Experience," GA-A21080, General Atomics, San Diego, CA.

RAI G-8: Section 3.3.1 describes projected graphite component peak temperature during accident conditions as approximately 1100°C for pebble bed design and around 1400°C for prismatic design. What data exist for thermal conductivity and irradiation-induced expansion and contraction at these temperatures for short duration (pulse) exposure? Discuss the effect of these factors on the NGNP safety case, especially for the degradation of metallic components due to sudden creep into the tertiary regime potentially resulting in rupture of pressure boundary and release of nuclide inventory?

Response G-8:
By way of clarification, the transients associated with the noted elevated temperatures are rare design basis events that would evolve over periods of hours to a few days. There are no short duration or "pulse" exposures associated with these events.

In part, the question asks whether there are material property changes that could preclude the insulating graphite blocks from thermally protecting the surrounding metallic components during a thermal transient under postulated accident conditions. In response, the principal effect of such elevated temperatures would be a tendency to anneal the effects of accumulated irradiation damage. There are no anticipated thermal annealing effects from such thermal transients that would change the thermal conductivity or the physical dimensions of the graphite components to the extent that the metallic components could become thermally degraded from direct exposure to the hot outlet gas. Further, during such events, there would be no neutron flux, so no irradiation-induced changes (i.e., dimensional changes) would result.

Any thermally induced changes to the graphite core components will be evaluated from experience from previous nuclear graphite components as well as data from current irradiation

testing under the NGNP Project. Data from the NGNP Graphite R&D program and the core design will be utilized to model the likely thermal conductivity and dimensional changes

The actual thermal property changes will be obtained from the NGNP Graphite Technology Development Program. The graphite irradiation program will be measuring the specific thermal property changes in a number of graphite types proposed for use in the HTGR design. However, in general, thermal conductivity values during thermal transient events will actually be conservative since the core temperature will increase above normal operating conditions and the irradiation damage inside the graphite core components will tend to be annealed out of the material, allowing the conductivity to increase from values that were established during normal operating conditions. The change of the thermal expansion property (i.e., coefficient of thermal expansion) is a little more complex since it is dependent upon the dose received for each individual component. In general, the thermal expansion has been observed to increase slightly for graphite that has received low dose levels but to decrease significantly for those graphite components that have received larger doses. The specific thermal expansion changes to the graphite type(s) selected for the NGNP design will be determined from the NGNP graphite irradiation program. Based upon the data from the graphite irradiation program and the core design, no metallic components would be exposed to these maximum core temperatures. See the response to RAI HTM-8 for further discussion related to metallic components.

RAI G-9:
Deleted by the NRC.

RAI G-10: Section 3.3.2 states that, "Until the graphite code is published and accepted by the NRC, graphite material selection will focus on existing design and operating experience with both past and currently available grades of reactor graphite." Provide a description of any operating experience for currently available grades of reactor graphite. Describe any regulatory body-certified design experience for currently available grades of reactor graphite.

Response G-10:
Until the graphite code is published and accepted by the NRC, graphite material selection will focus on existing design and operating experience to guide the selection of new grades of nuclear grade graphite and/or the inclusion of currently utilized nuclear grade graphites. Graphite grade selection will consider graphite manufacturers' recommendations on proposed analog grades of graphite to replace discontinued grades or newly developed grades to replace past nuclear grade graphites.

Below are references to the HTTR program and the JAEA website where the public documents related to the operating experience with IG-110 and PGX can be downloaded. The OSTI website can be used to search for other historical HTGR reactor program documents for grades of previously used graphite that are no longer available. The British Magnox and AGR reactors are not representative of U.S. HTGR designs. The Fort St. Vrain and Peach Bottom 1 reactors were DOE demonstration reactors operating under a Class 104 NRC license. Other foreign HTGRs were research or demonstration reactors.

During the next revision of the white paper, Section 3.3.2 (page 29) will be revised to provide clarification regarding the graphite grade selection process as follows:

Material selection for the graphite components will be based on the same general principles discussed in Section 3.1. One significant difference is that, up until now, only minimal guidance has been available from established regulatory requirements or the ASME Code regarding the use of these materials. This situation is expected to evolve, since a consensus ASME code on graphite component design for HTGRs has been prepared by the ASME Subgroup on Graphite Core Components and is expected to be published in 2011 (see Section 3.3.5 below). As explained in Section 3.3.5, the addition of nuclear grade graphite to the ASME Code would be ideal, but it is not required for nuclear grade graphite selection. Until the graphite code is published and accepted by the NRC, graphite material selection will focus on existing design and operating experience to guide the selection of new grades of nuclear grade graphite and the inclusion of past nuclear grade graphites. Graphite grade selection will consider graphite manufacturers' recommendations on proposed analog grades of graphite to replace discontinued grades or newly developed grades to replace past nuclear grade graphites.

References:

1. JAEA website: http://jolissrch-inter.tokai-sc.jaea.go.jp/search/servlet/interSearch?LANG=en
2. Osti: http://www.osti.gov/bridge/
3. JAERI-M 86-192, IAEA Specialists Meeting on Graphite Component Structural Design, Sanokawa, Konomo, dated September 8-11, 1986
4. JAEA-Research-2009-042, Draft of standard for graphite core components in High Temperature Gas-cooled Reactor, Taiju; Eto, Motokuni; Kunimoto, Eiji; Shiozawa, Shusaku; Sawa, Kazuhiro; Oku, Tatsuo; Maruyama, Tadashi, dated January, 2010
5. JAERI-M 91-154, An Explication of design data of the graphite structural design code for core support components of High Temperature Engineering Test Reactor, Ishihara, Masahiro; Iyoku, Tatsuo; Shiozawa, Shusaku, dated September, 1991
6. JAERI-M 7647, Temperature Irradiation Effect on the Mechanical Properties of HTGR Graphites, Tatsuo Oko, Motokuni Eto, Katsuo Fujisaki, dated March, 1978
7. JAERI 1332, Design of High Temperature Engineering Test Reactor (HTTR), Shinzo Saito, et. al., dated September 1994

RAI G-11: Based on information provided in 3.3.2, it appears that the technical basis for graphite selection is being developed for currently available grades of reactor graphite. If this understanding is correct, what basis is used for the NGNP graphite core component design?

Response G-11:

Until the graphite code is published and accepted by NRC, graphite material selection will focus on existing design and operating experience to guide the selection of new grades of nuclear grade graphite and the inclusion of current utilized nuclear grade graphites. Graphite grade selection will consider graphite manufacturers' recommendations on proposed analog grades of graphite to replace discontinued grades or on newly developed grades to replace past nuclear grade graphites.

RAI G-12: Section 3.3.2 states that, "Fabrication experience and technical maturity are additional selection criteria that must be considered." Explain how these aspects will be considered, and what criteria will be used, including their technical basis.

Response G-12:
The quote from Section 3.3.2 states that graphite grades that have more the fabrication experience and technical maturity will be produced with improved consistency and fewer defects (disparate flaws, density variations, mixing issues, etc.). Additionally the graphite grades selected must satisfy ASTM D-7219-08, "Standard Specification for Isotropic and Near-isotropic Nuclear Graphites." The fabrication maturity was used by the NGNP Graphite Technology Development Program in part to define which graphite types were "major" grades and which were "minor" grades. The criterion used to distinguish major from minor grades was whether the grade had experience with at least one full sized batch fabrication run. Those grades that had a full-sized fabrication run were considered mature and ready for use within a reactor core and were designated as a major grade. Graphite types that only had limited fabrication batches or even laboratory process runs were not considered mature enough for consideration as a major grade.

RAI G-13: Section 3.3.3 states, "A key requirement in prismatic designs is the need for fine grained graphite, with its correspondingly higher strength, for the fuel elements to ensure an adequate number of grains across the thickness of the graphite webs between the fuel compacts and the coolant holes. Relatively high thermal stresses are generated in the thin graphite ligaments between the coolant and fuel channels in these elements. Ideally, a ligament should be no thinner than 10-times the maximum grain size." Describe how this statement is guided by design code?

Response G-13:
The new ASME graphite design code does not currently provide guidance in selecting graphite grades for use in components with thin ligaments. Such features are, therefore, the responsibility of the respective designers. The NGNP Project will petition the ASME Subgroup on Graphite Core Components to include guidance for selection of graphite grades for core components incorporating thin ligaments.

During the next revision of the white paper, Section 3.3.3 (pages 30-31) will be revised to provide clarification regarding the ASME design process as the following:

> *"Graphite material selection criteria further stem from the functional requirements of the graphite core components for the specific reactor type (pebble or prismatic). These key functional requirements will be described in the Design Specification. A key function not currently addressed in the ASME graphite design code is the selection of graphite grades for use in graphite components with thin ligaments, presently the responsibility of the respective designers. NGNP will petition the ASME Subgroup on Graphite Core Components to include guidance for selection of graphite grades for core components incorporating thin ligaments.*
>
> *In the remaining paragraphs of this section, examples of graphites used in foreign reactors are presented as background information providing a historical summary of nuclear graphite manufacturing used in the foreign reactor programs. The introduction of these foreign high temperature gas reactors is not intended to provide insight to NGNP component sizes or to provide claims of manufacturability of future NGNP graphite core components."*

RAI G-14: In Section 3.3.3, it is stated that, 'The little available high fluence data for IG-110 at 600 °C (1112 °F) indicates similar dimensional change behavior similar to that of historic coarser grained materials, such as ATR-2E. Irradiated properties data for IG-110 tends to be presented without directional orientation on the basis that the material is isotropic; however, this is an aspect that needs some verification, since unirradiated properties measured for IG-110 in different orthogonal directions can show some variation." Provide information on the behavior of these materials (for comparison verification) for the NGNP operating conditions of temperature, fluence, atmosphere (coolant chemistry) and residence time. What information is available for these materials under accident conditions?

Response G-14:
The NGNP Graphite Technology Development Program will provide data on the behavior of directional orientation material property values for isotropic graphite types (i.e., IG-110, etc.) for the NGNP operating conditions of temperature, fluence, atmosphere (coolant chemistry) and residence time. These data will be generated from the AGC irradiation experiment, and some data will also be obtained from the as-fabricated baseline testing program once it is complete. Additional historical data on isotropic graphite types will be provided, if necessary, once a design has been established and an application is submitted. Finally, specifications and definitions for isotropic and near isotropic graphites are addressed in ASTM D7219 "Standard Specification for Isotropic and Near-isotropic Nuclear Graphites". This specification covers the classification, processing, and properties of nuclear grade graphite billets with dimensions sufficient to meet the designer's requirements for fuel elements and reflector blocks in a high temperature gas-cooled reactor.

RAI G-15: In Section 3.3.3, on page 31, the last paragraph contains a discussion on the manufacturability of graphite core components, dictated by the component size. For comparison purposes, provide relative size information for the various designs cited in this white paper, compared to the contemplated NGNP HTGR.

Response G-15:
The conceptual design of the HTGR is still underway, and a final decision regarding the HTGR design has not yet been made. Accordingly, the requested comparison cannot be made at this time. The reactor designer will use the key requirements in a design specification to select a graphite grade, considering manufacturability, grain size, and graphite costs, to achieve compliance with the ASME graphite design rules. See the response to RAI G-13.

RAI G-16: Section 3.3.4, Table 2 lists the properties of various graphite grades. Are these average properties? Are these properties originating from the manufacturer's data? Do they originate from the same type of fabrication and sources of raw materials as those being currently investigated at INL? Provide a discussion of how variation in material properties affect the NGNP design.

Response G-16
Properties from Table 2 are average values and are from public manufacturer-published source material. The fabrication process for each graphite type currently under investigation in the NGNP Graphite Technology Development Program will be the same or very similar [1].

The raw materials used to fabricate the graphite will be different, resulting in possible very small changes to the material properties between different batches. This variation between batches is one of the main research activities for the NGNP graphite program. Billets from different batches (and different raw material sources) will be compared to determine the variation in material properties as a function of batch. In addition, the irradiation testing program will use samples from different batches to ascertain the effects of raw material variation on irradiation induced material property changes. These changes will be determined from the NGNP Graphite Technology Development Program once irradiation testing is completed.

Reference:

1. PLN-2497, 2010, "Graphite Technology Development Plan," Revision 1, Idaho National Laboratory, October 2010.

RAI G-17: In Section 3.3.4, it is stated that, "More anisotropic grades not precisely meeting the D7219-08 degree of isotropy requirement may still be applied in HTGR applications, provided due diligence is paid to material behavior under irradiation, operating conditions (fluence-temperature), and design considerations. These grades can be applied to lower fluence regions as recommended by ASTM D7301-08, which would typically apply to the outer reflector regions." What is the basis for this apparent deviation from ASTM D7219-08 and ASTM D7301-08 requirements for near-isotropic graphite specifications for nuclear core components?

Response G-17:
More anisotropic grades not precisely meeting the ASTM D7219-08 degree of isotropy requirement may still be applied in HTGR applications, in low fluence conditions where changes to mechanical properties due to fluence exposure do not challenge the functional requirements of the graphite component. ASTM D7301-08 permits the use of graphite not meeting the isotropy requirement of ASTM D7219-08 for use in components where irradiation induced dimensional changes at low fluences do not challenge the functionality of the component and maintain the assigned Structural Reliability Class (SRC) requirements (see the ASME addendum to the updated white paper on the definition of Structural Reliability Class).

In the next revision of the white paper, Section 3.3.4 (page 32) will be revised to provide clarification regarding the ASTM standards as follows:

> *Importantly, ASTM D7219-08 specifies a range of physical and mechanical properties for isotropic grades that allows for a broad range of nuclear graphite grades as far as raw materials, forming method, purity level, and actual properties are concerned. In low-fluence conditions, where changes to mechanical properties due to fluence exposure do not challenge the functional requirements of the graphite component and the assigned Structural Reliability Class requirements defined in Section 3.3.5 can be maintained, alternate materials may be considered in accordance with ASTM D7301-08.*

RAI G-18: In Section 3.3.4, it is stated that, "It is important to recognize that the degree of isotropy only serves as an initial indicator of the graphite behavior under irradiation. End-product isotropy is influenced by raw material, grain size, forming method, and heat treatment." The reviewer understood that materials data after irradiation have been or will be generated using "end-product" specimens. Is this understanding correct? Furthermore, what specifically is

noteworthy in this paragraph compared to what has been stated earlier regarding graphite manufacture in Section 3.3.3?

Response G-18
"End-product" specimens will be used in the irradiation program. This paragraph repeats the behavior of isotropic materials previously discussed. It will therefore be deleted in the next revision to the white paper.

The following paragraph in Section 3.3.4 will be deleted to provide clarity to this section:

> *It is important to recognize that the degree of isotropy only serves as an initial indicator of the graphite behavior under irradiation. End-product isotropy is influenced by raw material, grain size, forming method, and heat treatment. Graphite billets can be fabricated by extrusion, isostatic molding, or vibration molding. Extrusion tends to yield graphites that are less isotropic and less dimensionally stable under irradiation than molded graphites, although isotropy can be improved remarkably through careful control of raw material and processing. Isostatic-molded graphite is commonly available in smaller sizes than extruded grades, while vibration molding is available for larger block sizes.*

RAI G-19: In Table 3 it is stated that, "The strength reserves offered by the material must exceed the allowable operating component stresses." Explain what is meant by "strength reserves?" The response should address how these reserves compare with safety factor for design margin, and how they correlate with the probability of failure requirements for various structural reliability classes in the proposed ASME code for design of graphite core components, and discuss what assurance is provided that these "strength reserves" will remain during reactor operating life.

Response G-19:
The term "strength reserves" is not a concept used in the ASME code. Therefore, during next revision of the white paper, the use of this term will be deleted. Currently, the NGNP Project is developing a document to further elucidate the ASME graphite design rules. The document will provide the background theories used in the ASME design rules and their application in code assessments (e.g., compressive strength requirements, use of tensile strength for determination of probability of failure requirements for various structural reliability classes, how the graphite component lifetime is determined using the design code). The plan is for this document to be an addendum to the updated white paper in the future. If the document becomes available prior to the update to the white paper, it will be provided as a reference material.

RAI G-20: In Table 3, it is stated that "higher strengths are achievable with isostatic-molded, fine grain graphite, but these typically possess lower fracture toughness." In Section 3.3.3, it is stated that "A key requirement in prismatic designs is the need for fine grained graphite, with its correspondingly higher strength, for the fuel elements to ensure an adequate number of grains across the thickness of the graphite webs between the fuel compacts and the coolant holes. Relatively high thermal stresses are generated in the thin graphite ligaments between the coolant and fuel channels in these elements. Ideally, a ligament should be no thinner than 10-times the maximum grain size." Provide an explanation of the relationship between these two statements with respect to the expected or desired functionality requirement.

Response G-20:
The new ASME graphite design code does not presently address requirements for designs with thin ligaments. The evaluation of such features is, therefore, the responsibility of the respective designers. NGNP will petition the ASME Subgroup on Graphite Core Components to address requirements for thin ligaments (also see response to RAI G-13).

RAI G-21: In Section 3.3.4.2, under the heading "Process Variables", it is stated that "the particle size distribution is generally classified by maximum grain size into coarse-, medium-, and fine-grain material." Since size distribution denotes a range of particle sizes existing in a material, clarify the classification by "maximum grain size." Also, describe what is meant by "volatile carbon artifacts."

Response G-21:
Graphite grades are categorized by the market they serve (i.e., electrode or nuclear grade graphite) and then classified by grain size. Nuclear grade graphite is made from a mixture of solid filler particles (graphite flour) and liquid binder (tar pitch) which is then heated (baked) to form a solid green body form by solidifying the liquid binder material. The liquid binder solidifies at these baking temperatures forming a solid, carbon-rich phase and short chained hydrocarbon gases comprising the "volatiles". The solid phase stays in the porous body binding the graphitic flour together while the gaseous volatiles (e.g., short-chained hydrocarbons) are driven out of the green body during this baking stage. "Volatile carbon artifacts" refers to these gaseous volatiles.

Specific graphite grain size definitions are listed in ASTM D7219 "Standard Specification for Isotropic and Near-isotropic Nuclear Graphites." For nuclear graphite, the maximum grain size is 1.68 mm as specified in D7219.

In addition, "volatile carbon artifacts" will be replaced with "volatile components" in Section 3.3.4.2.

RAI G-22:
Deleted by the NRC.

RAI G-23: In Section 3.3.4.2, under the heading "Process Variables, clarify what is meant by "furnace limitations. "Since graphitization is said to occur through the passage of current through the billet at very high current, is it the difficulty in having the needed electrical resistance of the billet for graphitization?

Response G-23:
"Furnace limitations" refers to the physical size limitations of the furnaces. These are large graphite components, which take a great deal of energy to fabricate, and as such the furnaces need to be rather large. As long as the furnace size meets the component requirements this is not an important issue. It is expected the reactor designer would select a graphite manufacturer and graphite grade for which the furnace size does not limit the component's size.

RAI G-24: Regarding item 6 under the heading "Process Variables" in Section 3.3.4.2, do the baking and impregnation occur at different or same location. If at different locations, what

considerations need to be addressed to maintain the integrity and geometry of the blocks during the process cycle?

Response G-24:
Graphite manufacturing processes are controlled by and proprietary to the manufacturers. The NGNP Project does not have access to this information and therefore cannot comment on manufacturing processes and facilities. Questions regarding the graphite manufacturers' processes and facilities should be directed to the specific graphite manufacturer and issues should be addressed during the design and licensing application phase.

The graphite manufacturers must have the knowledge and confidence in their proprietary production process to demonstrate that their graphite meets the requirements of the ASME code. The ASME code requires the final graphite billets be supplied by a manufacturer with a quality assurance program accepted by the ASME.

RAI G-25: In Table 4 of Section 3.3.5.2, clarify the definition of probability of failure. For example, is it the probability of failure of a single brick per a reactor year of operation or the probability of failure based on the distribution of expected strength of a graphite core component, when subject to operational stress at any given time of operation? Is this probability of failure based on irradiated data? If based on non-irradiated data, discuss why this usage is considered conservative.

Response G-25
The ASME graphite design rules define the probability of failure as crack initiation in a finite volume of the graphite component based on the assessment procedures of the ASME graphite design rules. The evaluation is applied throughout lifetime operation.

The prediction of crack initiation does not necessary mean imminent component failure. It is the designer's responsibility to evaluate fractures predicted by the ASME analysis procedures as fluence accumulates, and to determine if the component has lost or still maintain its functionality. The ASME graphite design rules define peak equivalent stress as the stress state at a point in the component determined using irradiated thermal properties and local irradiation strains and creep. This stress is used in calculating the probability of failure. The tensile and compressive strength used in the ASME code is collected on nonirradiated specimens. This is conservative for, as the graphite acquires fluence, both strengths increase until a limiting value of fluence defined by the cohesive life limit, is reached. The graphite core designer takes into account this behavior in designing the reactor's components. Currently, the NGNP Project is developing a document to further elucidate the ASME graphite design rules. The document will provide the background theories used in the ASME design rules and their application in code assessments (e.g., compressive strength requirements, use of tensile strength for determination of probability of failure requirements for various structural reliability classes, how the graphite component lifetime is determined using the design code). The plan is for this document to be an addendum to the updated white paper in the future. If the document becomes available prior to the update to the white paper, it will be provided as a reference material.

In addition, Table 4, Maximum probability of failure for each safety class, will be revised to clarify the maximum probability of failure for the respective structural reliability classes as follows:

Table 4. Maximum probability of failure for each safety class.

Structural Reliability Class	Maximum Probability of Failure
SRC-1	1.0E-4
SRC-2	1.0E-4 nonirradiated and 1.0E-2 for irradiated components
SRC-3	1.0E-2

RAI G-26: In Section 3.3.5.2, it is stated that "Another deviation from past ASME metallic codes is allowance of cracks in the graphite components. The rules require the core designer to demonstrate through analyses or testing that cracked graphite core components can maintain their assigned safety function and that the graphite component is remotely retrievable when cracks of a specified size and orientation are present." However, the staff is not able to find this provision in the draft code. The current draft code allows for defined probability of failure values for several structural reliability classes for graphite core components. The graphite core assembly itself should maintain its required geometry and be able to perform its intended functions.

Section 3.3.5.2 also states that "The code also deviates from the ASME standard practice of defining primary, secondary, and membrane stresses." However, NRC staff is unable to confirm this statement from reading the draft Code. Provide clarification, reconciling provisions of the draft ASME Code with the white paper description.

Response G-26:
ASME Section III, Division 5 provides the following clarifications related to Section 3.3.5.2:

ASME Section III, Division 5, Subsection HH, Subpart A-3100(b)
The design approach selected is semi-probabilistic, based on the variability in the strength data of the graphite grade. Due to the nature of the material, it is not possible to ensure absolute reliability, expressed as an absence of cracks, of Graphite Core Components. This is reflected in the setting of Probability of Failure (POF) targets. Also note that due to the complex nature of the loadings of graphite components in a reactor combined with the possibility of disparate failures of material due to undetectable manufacturing defects, the Probability of Failure values used as design targets may not be precisely accurate predictions of the rate of cracking of components in service. The Designer is required to evaluate the effects of cracking of individual Graphite Core Components in the course of the design of the Graphite Core Assembly and ensure that the assembly is damage tolerant.

ASME Section III, Division 5, Subsection HH, Subpart A-3330(b)
(b) arrange the Graphite Core Components comprising the Graphite Core Assembly so that cracking of individual Graphite Core Components does not detrimentally affect the connections between the Graphite Core Components and thus impede the functionality of the Graphite Core Assembly;

ASME Section III, Division 5, Subsection HH, Subpart A-3214.7
The combined stress is the sum of all of the components of stress at a point. In design, it is customary to distinguish between primary and secondary stresses. These are defined as follows:

- Primary stress is any normal stress or a shear stress developed by an imposed loading that is necessary to satisfy the laws of equilibrium of external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. Primary stresses that considerably exceed the material strength will result in failure. A thermal stress is not classified as a primary stress.

- Secondary stress is a normal stress or a shear stress developed by the constraint of adjacent material or by self-constraint of the structure. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions may satisfy the conditions that cause the stress to occur.

Due to the brittle nature of graphite, no distinction is made between primary and secondary stresses for the purpose of assessment to these rules. Combined stress is thus the combination of primary and secondary stress.

During the next revision of the white paper, Section 3.3.5.2 (page 36) will be revised to provide clarification regarding the ASME Code as follows:

> *"The code also deviates from the ASME standard practice of defining primary and secondary stresses. Due to the brittle nature of graphite, no distinction is made between primary and secondary stresses for the purpose of assessment to ASME graphite design rules. Combined stress is thus the combination of primary and secondary stresses. Irradiation induced stresses and mechanical stress concentrations are the largest contributors to stresses in graphite core components. These irradiation-induced stresses follow thermal and neutron fast fluence gradients within the block. These stress gradients are the critical stresses in the graphite blocks. Stress concentrations arise from keyways and dowels in the graphite components producing the highest mechanical stresses. Other mechanical stresses (combined membrane, bending and peak stresses) are of lesser magnitude, but are included in the calculation of stresses at a point."*

RAI G-27:
Deleted by the NRC.

RAI G-28: It is stated in Section 3.3.5.2 that "the mechanical stresses (primary, secondary, and membrane) do not challenge the graphite core components strength margins." However, it would appear that the geometry and placement of the core components, and the presence of notches, keyways, and dowels could potentially create stress concentration effects different than primary, secondary, and membrane stress. Arguably, these stresses need to be combined with those from the irradiation induced strain and the resulting stresses to arrive at the combined stress on the component. Provide a discussion addressing these considerations.

Response G-28:
The statement that the mechanical stresses "do not challenge the graphite core components strength margins" was included to highlight the fact that mechanical stresses are typically not governing relative to stresses induced by fluence and thermal effects. Regardless, the term "strength margins" is not congruent with the ASME code.

Per the ASME graphite design rules, all irradiation induced loads as well as all mechanical loads have to be included in determining the stress at a point. The stress state at a point is used in determining the probability of failure (Reference 1). This paper identifies the graphite strength margins by providing anticipated stresses in graphite components and graphite strengths in prior HTGR designs.

In the next revision of the white paper, Section 3.3.5.2 (page 36) will be revised as given in the response to RAI G-26.

Reference:
1. B. T. Kelly, 1978, "Radiation Damage in Graphite and Its Relevance to Reactor Design", Progress in Nuclear Energy, v. 2, 1978.

RAI G-29: In Section 3.3.6.1, projected service conditions for the pebble bed reactor has been provided. Provide similar information for a prismatic reactor design.

Response G-29:
Based on work done in 2007 during pre-conceptual design for the NGNP Project, the maximum fast neutron fluence for the prismatic permanent reflector blocks is less than 4×10^{20} n/cm^2 (E > 0.18 MeV). The maximum fast neutron fluence for replaceable reflector blocks (and fuel elements) is less than 4×10^{21} n/cm^2 (E > 0.18 MeV) as of HTGR conceptual design. During normal operation and LBEs, the maximum graphite temperature in the fuel elements is approximately the same as the maximum fuel temperature. For normal operation, the maximum time-averaged fuel temperature is < 1250°C. The minimum graphite temperature during normal operation is approximately the same as the coolant inlet temperature.

During depressurized conduction cooldown LBEs, the maximum peak temperature is ~1600°C and occurs in a very small region of the core (adjacent to the inner graphite reflector of the annular core and typically in Layers 4 and 5 of the 10 layers of fuel elements) for at most a day or two.

In the next revision of the white paper, Section 3.3.6.1 (page 36) will be revised to provide clarification regarding the service conditions as follows:

> *The fluence levels and irradiation temperatures seen by the various graphite reflector components are dependent upon both the reactor type and their locations within the reactor. Service conditions seen by graphite components within pebble bed reactors are typically more limiting, because of their longer design lives and the correspondingly higher fluence levels that will be accumulated by some of these components. Normal operating temperatures for the pebble bed graphite reflector range between 250 and 800°C (482 and 1472°F). The maximum temperatures that would be seen by the*

reflector graphite during certain LBEs are less than 1200°C (2192°F). Only components in close proximity to the pebble fuel are subjected to high fluence levels that may limit their lifetime to less than that of the design life of the plant as a whole, this maximum fluence being slightly above 1.1 × 10^{22} n.cm^{-2} EDN (equivalent Dido nickel) (15 dpa). The affected parts of these components do not serve a structural function.

The maximum fast neutron fluence for the prismatic permanent reflector blocks is less than 4 × 10^{20} n/cm^2 (E > 0.18 MeV) as of the NGNP pre-conceptual design. *The maximum fast neutron fluence for replaceable reflector blocks (and fuel elements) is less than 4 × 10^{21} n/cm^2 (E > 0.18 MeV). During normal operation and LBEs, the maximum graphite temperature in the fuel elements is approximately the same as the maximum fuel temperature. For normal operation, the maximum time-averaged fuel temperature is < 1250°C. The minimum graphite temperature during normal operation is approximately the same as the coolant inlet temperature.*

During depressurized conduction cooldown LBEs, the maximum peak temperature is ~1600°C and occurs in a very small region of the core (adjacent to the inner graphite reflector of the annular core and typically in Layers 4 and 5 of the 10 layers of fuel elements adjacent to the inner graphite reflector of the annular core) for at most a day or two.

RAI G-30: Section 3.3.6.2 describes properties for baseline characterization, including compressive strength and fatigue strength. However, these properties are not discussed in Section 3.3.6.3 regarding characterization of irradiated graphite. What bases exist to suggest that compressive strength and fatigue strength are not subjected to potential degradation due to irradiation?

Response G-30:
ASME analysis procedures use data from the Material Data Sheets collected by the designer. The conservatism in using nonirradiated test data is explained in the response to RAI G-25. Moreover, see Reference 1 for discussions on fatigue strength from irradiated material and Reference 2 for a discussion on irradiated graphite strength.

References:
1. Price, R. J., "Cyclic Fatigue of Near-Isotropic Graphite: Influence of Stress Cycle and Neutron Irradiation", *Carbon*, Vol 16, 1978.
2. Kelly, B. T, "Radiation Damage in Graphite and Reactor Design", *Progress in Nuclear Energy*, Vol 2, 1978.

RAI G-31: In Section 3.3.6.3, it is mentioned that the thermal conductivity of irradiated graphite depends on the fluence and that a secondary reduction in thermal conductivity occurs at high fluence range due to more advanced material degradation. Describe what is meant by "advanced material degradation?" For example, is it "pore generation", as mentioned in Appendix B, page 82? The mechanism for vacancy generation by displacement of atoms is understood. What is the mechanism for generation of pores? Is it physically possible to remove material by irradiation to form pores, or are these postulated mechanisms waiting for experimental evidence? If evidence exists in the form of transmission electron microscopy

observation, have they been verified to be consistent across different graphite types? What is the impact of such degradation on the need for removal of graphite blocks either for replacement, or for decommissioning and disposal?

Response G-31:

Advanced material degradation refers to the latter stages of graphite pore generation leading to eventual property degradation (including thermal conductivity) and structural failure in the graphite. The mechanism of pore generation is as follows. Neutron induced displacement damage in the graphite crystal causes simultaneous <c>-axis growth and <a>-axis shrinkage. Initially, <c>-axis expansion is accommodated within aligned cracks (porosity) contained in the graphite (e.g., "Mrozowski" or thermal cracks between graphitic basal layer planes, which form due to the anisotropic thermal shrinkage that occurs on cooling from processing temperatures).

Hence, the polycrystalline graphite exhibits overall volume shrinkage, dominated by the <a>-axis shrinkage. With further displacement damage the induced <c>-axis growth in the polycrystalline graphite fills the available aligned cracks/porosity and the polycrystalline graphite volume and dimensional behavior is increasingly dominated by crystal <c>-axis expansion. The rate of shrinkage of the polycrystalline graphite falls and eventually reaches zero. The polycrystalline graphite thus exhibits a volume "turnaround" phenomenon from bulk dimensional/volume shrinkage into growth. The mismatch of irradiation induced crystal strains causes the formation of local cracks or pores within the graphite structure (pore generation).

Hence, the mechanism of pore generation does not involve removal of material by irradiation. There is some microstructural evidence for induced changes in porosity (recent work at Manchester University, UK). However, much of the evidence for direct pore generation has not been published (German and UK data). Dimensional changes and physical property degradation at very large damage doses are explained by the mechanism of pore generation in graphite. This eventual structural and property degradation is recognized and limited in the ASME core component design code through the imposition of a temperature dependent cohesive life limit fluence (Reference 1) as the neutron fluence when +10% linear dimensional change (with-grain) has occurred in the components and the component may no longer be considered to provide a contribution to structural performance (HHA-3142.4). Further evidence of "pore generation" will become available through the NGNP Graphite Irradiation Program, where new X-ray tomography tools will be utilized to examine the pore structure of irradiated graphite. The mechanism of polycrystalline graphite crack/pore generation during irradiation is expected to be common to all NGNP candidate graphite grades, as is the form of the irradiation induced volume and dimensional change curves. However, the magnitude of the minimum volume shrinkage and the temperature dependent fluence at which the cohesive life limit is attained is expected to be somewhat different for the various candidate grades. Where the reactor design requires the graphite components to operate at high neutron doses and temperatures, data will be required to define the ASME code cohesive life limit for the graphite such that it has sufficient strength for replacement or decommissioning.

Reference:

1. ASME, 2010, "Boiler and Pressure Vessel Code," Sect. III, Division 5, Sub-Section H, article HA 3000.

RAI G-32: In Section 3.3.6.3, it is stated that "From a design perspective, the available strength reserve must be compared against the stress within the graphite component based on the fluence-temperature history of the component." Explain how this concept is handled in the design code. What criteria should be used for comparison and why? Describe the basis for accepting or rejecting a component for further service after such comparison.

Response G-32:
The ASME graphite design rules do not use the concept of "strength reserve" or "strength margins" as a measure of survivability or failure. The ASME graphite design rules define the probability of failure as crack initiation in a finite volume of the graphite component based on the assessment procedures of the ASME graphite design rules.

Crack initiation does not necessary mean imminent component failure. It will be the reactor designer's responsibility during the design activity to evaluate fractures predicted by the probabilistic analysis to exceed probabilistic thresholds as fluence accumulates, and to determine if the functionality of the component is lost or maintained.

Currently, the NGNP Project is developing a document to further elucidate the ASME graphite design rules. The document will provide the background theories used in the ASME design rules and their application in code assessments (e.g., compressive strength requirements, use of tensile strength for determination of probability of failure requirements for various structural reliability classes, how the graphite component lifetime is determined using the design code). The plan is for this document to be an addendum to the updated white paper in the future. If the document becomes available prior to the update to the white paper, it will be provided as a reference material.

During the next revision of the white paper, Section 3.3.6.3 (page 39) will be revised to provide clarification regarding the ASME code rules as follows:

> *Other design properties important to the evaluation of irradiation induced and mechanical loads and stresses in accordance with the ASME graphite design rules are found in Appendix II of Subsection HHA. ASME graphite design rules define the probability of failure as crack initiation in a finite volume of the graphite component based on the assessment procedures of the ASME graphite design rules. Crack initiation does not necessary mean imminent component failure. It is the designer's responsibility to evaluate fractures predicted by the probabilistic analysis to exceed probabilistic thresholds as fluence accumulates, and to determine if the functionality of the component is lost or maintained.*

RAI G-33: In Section 3.3.6.3, it is stated that "Evolution of creep strain with fluence in irradiated graphite (differential strain between stressed and unstressed) specimens is characterized by primary, secondary, and tertiary regimes, much like those observed in metals." However, this statement may not be accurate. While the traditional metallic creep is analyzed by the behavior of creep strain as a function of time at constant temperature and varying applied (tensile) stress or at constant applied (tensile) stress and varying temperature, the irradiation creep is analyzed as the strain as a function of cumulative irradiation dose at mostly constant applied (compressive) stress and temperature. The staff understands that the irradiation dose could be proportional to cumulative time. However, because of the 3-dimensional variation in the

temperature and dose, a brick could potentially experience, the concept may not be applicable. Provide additional justification for the white paper's description of creep strain evolution in graphite.

Response G-33:
Since the 3-dimensional temperature and dose profiles within the graphite components are design specific, the cumulative effect this has on graphite response will be addressed for all material properties once the design has been established. Once the proper conditions have been understood from the design, the variation in stress relief response (i.e., creep) as a function of this 3-dimensional temperature and dose variation can be effectively modeled based upon the results from the graphite irradiation program. The irradiation program will provide irradiation material property changes over a range of dose (0.5 – 7 dpa) at three nominal temperature levels (600, 900, and 1200°C) and three different stress levels (7 MPa, 14 MPa, and 21 MPa). Using these empirically determined values and an appropriate analytical creep model (currently being developed by the NGNP Graphite Program), the material response over a range of dose, stress, and temperature can be calculated for all positions within the graphite core. As the dose (or temperature) evolves over time the material response can be calculated up to a dose level of 7 dpa. Operation times or doses over an accumulated dose of 7 dpa will require additional data to verify the model and calculations

RAI G-34: In Section 3.3.6.3 it is stated that "There is evidence to suggest that the normalized creep strain (normalized to initial elastic strain) is similar for different grades of graphite, lending support to the theory that this creep behavior is not material grade specific. Additional creep data would be useful in supporting this position and extending its application for a broader fluence-temperature range and for a wider variety of nuclear graphite grades. This may help in rationalizing and minimizing the need for costly irradiation creep experiments for current reactor graphite grades or grades that may be developed in the future." Other sections of the white paper suggest that all graphite properties, normalized or not, depend strongly on the type of graphite and the way it is manufactured. While some dependencies can be mathematically cancelled out by "normalization", this does not mean, per se, that the irradiation creep behavior of the different types of graphite will be the same or similar and that irradiation experiments may not be needed for various types of graphite. Why should creep be different purely from the perspective of its microstructure influencing its mechanical behavior? Provide additional discussion and justification for the white paper description of normalized creep strain.

Response G-34:
The actual physical mechanisms controlling irradiation induced creep at an atomic and microstructural level in graphite are not fully understood. This statement merely confirms the limited understanding behind creep phenomenon and proposes that additional creep data be generated to answer some of the seemingly divergent data. The intent behind this statement is that the physics at the crystallite or atomic length scale appear to be similar for all graphite grades implying that the fundamental mechanisms are common to the graphite crystalline structure rather than occurring as a consequence of the fabrication process. However, the macroscopic effect of fabrication processes on creep rates can be significant (change in grain size, final density, extruded or iso-molded, graphitization temperature, etc.). This is assumed to be true for all other graphite material properties as well. New data will help determine what the fundamental mechanisms really are and how the fabrication process can influence the macroscopic response of the graphite under irradiation. Ascertaining the actual mechanisms

and how they can be influenced by the fabrication process will allow the scope of future graphite development and qualification programs to be correspondingly reduced. The desire is that with a better understanding of the actual physical mechanisms only confirmatory studies will be necessary for future graphite development.

The NGNP Graphite Program is investigating the various effects and mechanisms influencing creep (References 1-3). The measured creep rate as a function of dose/fluence and temperature for each major graphite grade is being investigated within the AGC irradiation program. Additionally, determination of the mechanisms responsible for irradiation induced creep is being pursued by the international graphite community (Reference 4). Specific creep data measurements for the NGNP major grades as well as further understanding of the mechanisms will be available from the Graphite Technology Development Program in the future.

References:

1. Analysis of irradiation creep experiments on nuclear reactor graphite Kelly, B.T.[1]; and Burchell, T.D.[1] Source: *Carbon*, v 32, n 1, p 119-125, 1994; DOI: 10.1016/0008-6223(94)90017-5; Publisher: Publ by Pergamon Press Inc.
2. Irradiation induced creep behavior of H-451 graphite Burchell, Timothy D. (Materials Science and Technology Division, Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN 37831-6088, United States) Source: *Journal of Nuclear Materials*, v 381, n 1-2, p 46-54, October 31, 2008.
3. Irradiation induced creep of graphite Burchell, T.D. (Materials Science and Engineering Division, Oak Ridge National Laboratory, Oak Ridge, United States); Murty, K.L.; Eapen, J. Source: *JOM*, v 62, n 9, p 93-99, September 2010.
4. IAEA, 2011, www.iaea.org/NuclearPower/Technology/CRP, accessed September 21, 2011.

RAI G-35: In Section 3.3.6.3, it is stated that a program is underway to demonstrate that properties measured on subsize specimens are valid when compared to those obtained on standard specimens. However, the validity of measurements on small samples could still be questionable in representing the "sampling" volume of graphite core component of the reactor. Provide additional justification for the planned use of subsize test specimens.

Response G-35:
The limited irradiation volume available within a nuclear materials test reactor precludes the use of standard sized test samples. Smaller sized sample specimens are thus required to obtain irradiation induced material property data. The issue of sub-sized test specimens is a recognized issue for the NGNP Project, and a number of activities are being conducted to determine any potential bias from utilizing samples with non-standard sizes. These activities include determining the effect of size (i.e., thickness) on thermal diffusivity measurements, the effect of grain size to sample test volume ratio for mechanical testing, the effect of sample size and geometry on fundamental frequency measurements of elastic properties, and sample oxidation rate changes as a function of test specimen size. Other scoping studies are either planned or anticipated in order to verify that the testing standards used for characterization are valid for these non-standard sized specimens. Results from these scoping study activities will be included in the data reports from the NGNP Graphite Technology Development once they are completed.

RAI G-36: Section 3.3.6.4 describes the PBMR Graphite Development Program. However, this program no longer exists. Revise the white paper to reflect the status of this program.

Response G-36:
The PBMR Program in South Africa has been terminated, and its design team was disbanded by the end of September 2010. All data have been turned over to the South African Government for archiving. The data are not presently available to the NGNP Project.

In the next revision of the white paper, Section 3.3.6.4 (page 40) will be modified as follows to clarify the status of the PBMR program:

PBMR Graphite Development Program

Prior to its termination in late-2010, the PBMR program in South Africa was working on the development of the Pebble Bed Modular Reactor for electricity and process energy. In support of the proposed PBMR Demonstration Power Plant (DPP), a detailed Materials Qualification Plan (MQP) was developed for NBG-18 graphite and partially implemented. The objectives of the MQP were two-fold:

- *To characterize the as-manufactured and irradiated properties of NBG-18 graphite as a basis for confirming its suitability for use in the PBMR and its compliance with the requirements established for the PBMR reflector components.*
- *To validate the PBMR analytical models for predicting the behavior of irradiated graphite.*

The approach taken in the MQP was to utilize the extensive historical database characterizing the irradiated properties of the earlier German ATR-2E and VQMB graphites, since those grades were believed to be similar to NBG-18. The MQP's irradiation test program would then have been used to validate the use of this existing database in developing analytical models for the initial design and structural analysis of the graphite core structures, and to supplement this database in areas where data were sparse or unavailable.

The overall strategy was for the irradiated properties database to be sufficiently complete by initial startup of the DPP to confirm or improve the accuracy of analytical models for graphite design, and to justify operation of the plant over a substantial portion of its life. Thereafter, the remainder of the irradiation test program would substantially lead the actual operation of the initial plants.

By the time of program termination, several batches of pre-production NBG-18 graphite had been acquired and characterized. The characterization of as-manufactured properties was effectively complete. The graphite irradiation tests were in the planning stage, but not started. All data have been turned over to the South African Government for archiving. The data are not presently available to the NGNP Project.

RAI G-37: In Section 3.3.6.4, under the heading "NGNP Graphite Development Program", it is stated that, "respectively, NBG-18 (coarse grain size, pitch coke, vibration molded) and PCEA (medium grain size, petroleum coke, extruded) graphites are considered to be grades most likely to meet the initial pebble-bed and prismatic design requirements." However, in Section 3.3.3, it is stated that "A key requirement in prismatic designs is the need for fine grained graphite, with its correspondingly higher strength, for the fuel elements to ensure an adequate number of grains across the thickness of the graphite webs between the fuel compacts and the coolant holes." Thus, Section 3.3.3 advocates the use of fine-grained iso-molding processing and material derived thereby. Clarify the implications of potentially different materials used for the same purpose. Are these materials interchangeable? Can one be substituted by the other for replacement?

Response G-37:
During the next revision of the white paper, NGNP Section 3.3.3 will be revised to clarify the statement as the following:

> *A key requirement in prismatic fuel elements is the need for finer grained graphite, with its correspondingly higher strength, to ensure an adequate number of grains across the thickness of the graphite webs between the fuel compacts and the coolant holes.*

Also, to clarify this section, the statement that iso-molded graphites need to be used for prismatic designs will be deleted. The only actual requirement is that the grain size be small enough to yield enough grains across the fuel element webs. The intent, of course, is that the graphite grain size must be significantly smaller than the web thickness dimensions to avoid the problem of having only one or two grains bridging the web thickness between fuel and coolant channels in the fuel element. If this basic requirement is met, then the materials can be interchangeable or substituted if they meet core design requirements (also see the responses to RAIs G-13 and G-20).

RAI G-38: In Section 3.3.6.4, it is stated that "Complete properties data need to be developed for the graphite(s) eventually selected for the NGNP. Once the baseline material properties for the selected graphite grade(s) have been established, irradiation induced property changes must then be determined, including the characterization of irradiation induced creep. Determining these properties are important data needed for the design to satisfy the safety-related functions identified in Section 3.3.6.1." Describe plans to provide sufficient data and information on the selected graphites for verification and confirmation by the NRC staff in review of the expected NGNP combined license application. If data are not expected to be available when that application is submitted, describe how the application will address this issue so that the NRC staff can conduct an informed evaluation.

Response G-38:
The NGNP Project plan to provide sufficient data and information on the selected graphite types is outlined in the plan (Reference 1). The NGNP Graphite Technology Development Plan (TDP) outlines the development activities to qualify the major graphite types and provides a timeline of when the data are expected to be available for review.

Reference:

1. PLN-2497, "Graphite Technology Development Plan," Rev 1, Idaho National Laboratory, October 2010.

RAI G-39: Appendix B states that "structural and dimensional changes in polygranular graphites are a function of both the crystallite dimensional changes and the graphite's texture." Describe what is meant by graphite's texture.? If texture is related to the surface structure of graphite, for example roughness, then what role does irradiation play to roughen the surface? Discuss the effect of roughness on coolant flow.

Response G-39:
Graphite texture simply refers to the alignment (or lack of alignment) of the individual crystallite grains in the graphite microstructure. Nuclear grade graphite types possess isotropic or near-isotropic microstructures which correspond to minimal texture. Texture occurs primarily from the green forming process (i.e., extrusion, vibramolding, or isomolding) with extrusion imposing the most texture on the microstructure and isomolding imposing little to no texture.

Graphite roughness is a direct function of the inherent porosity for each graphite type. The larger the pores inside the graphite material the rougher the surface will be when the component is machined. Irradiation should not affect the surface roughness of the graphite component.

Surface roughness of the graphite components does have an effect on the coolant flow and the extent of this effect is being investigated through university lead NEUP research projects.

RAI G-40: Appendix B describes the cohesive life limit. Provide additional definition of this limit. For example, is it a function of cumulative dose? How is the cohesive life limit affected by the temperature, atmosphere (coolant chemistry), and imposed stress? Can the cohesive limit creep-in before the expansion?

Response G-40:
The ASME Section III, Division 5, Subsection HH, Subpart HHA-3142.4 Graphite currently defines cohesive life limit as the following:

> *A temperature-dependent cohesive life limit fluence is to be defined for the graphite grade used for Graphite Core Components. Material that exceeds this fluence limit is considered to provide no contribution to the structural performance (stiffness and strength) of the Graphite Core Component. This fluence limit shall be set to the fluence at which the material experiences a +10% linear dimensional change in the with-grain direction. For full assessment (HHA-3230), this material shall not be included in the volume of the Graphite Core Component assessed.*

In addition, the NGNP Project will petition the ASME committee to provide more guidance on the cohesive life limit, taking into account coolant chemistry and irradiation induced creep.

RAI G-41: Appendix B: What difference, if any, exists between the irradiation creep when tensile stress is imposed versus the application of a compressive stress? What type of calculations and/or predictive relationships exist between the volumetric changes, considering

the two types of stresses? If mixed type of stress (e.g., biaxial tension or compression) exists, what would be their effect on irradiation creep and the resulting volumetric change in graphite? Discuss the importance of these factors for the NGNP design.

Response G-41:
Irradiation creep in graphite is more accurately described as "stress relaxation from irradiation induced strain". Since the strain (i.e., creep) is a function of the imposed stress state, the amount and rate of strain within the graphite material will depend closely upon the stress—tensile, compression, or biaxial. The most significant stresses imposed upon the components result from irradiation induced dimensional changes and on a smaller scale the compressive forces from stacking the components in the core. Under irradiation, nuclear grade graphite typically experiences macroscopic shrinkage until turnaround, where the material response switches to volumetric expansion. The irradiation induced stresses due to the dimensional changes are therefore primarily compressive in nature at lower doses with gradually increasing induced tensile stress states as the dose increases. These conditions are expected to induce a significantly complex stress state within the graphite ranging from pure tensile to pure compression and fully mixed stress states.

Common observations from previous irradiation creep experiments over the past 50 years have shown little difference in creep rate between loading the test samples in either compression or tension for lower irradiation doses, during dimensional shrinkage. However, the rate can change significantly from the applied stress state once turnaround has been achieved [1]. The tensile creep rate can be accelerated or reduced once turnaround is reached if the test samples are placed under tensile or compressive loading, respectively. The assumption is that these are conservative rate limits, and mixed stress states would tend to be in between these purely tensile loaded and compressive loaded samples.

The NGNP Graphite Program is investigating the various effects and mechanisms influencing creep. Within the AGC irradiation program the measured creep rate as a function of dose/fluence and temperature for each major graphite grade is being investigated for dose levels up to turnaround levels (0-7 dpa). Over this dose range, the samples can be loaded in compression or tension without differences. Dose levels of ~7 dpa are estimated to allow 12—15 years of active service for the graphite components receiving the highest flux levels for the current reactor designs. If the reactor design requires graphite components to receive doses exceeding 7dpa, additional tensile loaded irradiation experiments will be required to measure the creep rate of the major grades of graphite for dose levels past turnaround.

Additional information on this subject can be found in References 1-3.

References:
1. PLN-2497, "Graphite Technology Development Plan," Rev 1, Idaho National Laboratory, October 2010.
2. R.J. Price, 1981, "Irradiation Induced Creep in Graphite: A review," GA-A16402 UC-77, August 1981.
3. T.D. Burchell, 2009, "Irradiation Induced Creep in Graphite at High Temperature and Dose – A Revised Model," ORNL/TM-2008/098, February 13, 2009.

RAI G-42:
Deleted by the NRC.

RAI G-43: Appendix B states that specific heat is rather invariant with graphite type and also is not influenced by irradiation. However, there seem to be observations which indicate that irradiation affects specific heat, at least at low temperature. For example, see : (1) T. Iwata and M. Watanabe, "Increase in specific heat and possible hindered rotation of interstitial C2 molecules in neutron-irradiated graphite" Phys. Rev. B 81, 014105 (2010); (2) W. DeSorbo and W.W. Tyler, "Effect of Irradiation on the Low-Temperature Specific Heat of Graphite", J. Chem. Phys. 26, 244 (1957). Furthermore, different equations have been provided for different types of graphite in General Atomics "Graphite Handbook." In addition, that W. Windes, T. Burchell, and R. Bratton in "Graphite Technology Development Plan", INL/EXT-07-13165, p 23 (2007) state that "changes to the specific heat due to oxidation and/or irradiation will be compared to as-received values." Provide additional discussion of graphite specific heat characteristics, addressing information given in the references cited.

Response G-43:
As stated, the specific heat values of graphite are not expected to significantly change due to irradiation. The cited references specifically address low temperature irradiation testing where point defect populations and higher subsequent stored energy levels are expected to be significant with little recombination of point defects.

The NGNP Graphite Program will be measuring irradiation induced specific heat changes, if any, during post irradiation examination of the tested grades of graphite. As stated in the Graphite Technology Development Plan, there is minimal change expected from these samples exposed to high temperature irradiation. However, if significant changes are detected from the post irradiation examination (PIE) the graphite program will focus more attention on the measurements and mechanisms.

RAI G-44: Appendix B states that "emissivity of given graphite will depend on its surface condition and the environmental temperature." It is further stated that "The emissivity of nuclear graphite is not expected to change significantly with irradiation." Provide additional justification for the expected irradiation performance, including discussion of irradiation effects on graphite surface structure.

Response G-44:
The surface condition (i.e., roughness and/or deposits) are the only factors that could affect the thermal emissivity for graphite. It is anticipated that any changes to the graphite surface condition which would affect the emissivity will cause minimal changes since if the surface (i.e., material) is removed or altered the surface will still retain a pure black appearance with the same equivalent emissivity levels. The only mechanisms anticipated to affect changes to the emissivity would be exposure to the reactor environment causing surface roughness changes or small oxide deposits from impurities within the graphite or coolant environment forming on the surface. Any changes to the surface condition are chemical (e.g., oxidation of graphite), rather than irradiation induced, and, therefore, irradiation effects are expected to be marginal, if present at all.

Any surface roughness changes will minimally affect emissivity since any new surface exposed will still be graphite in nature and have the same emissivity values. In concept, for those graphite types with higher metallic impurity levels any oxidation from the coolant environment may produce small oxide deposits which would be formed on the outer surface of the graphite components and could affect the emissivity depending upon the purity levels of the graphite and coolant environment. However, nuclear grade graphite and the coolant chemistry in the NGNP core are expected to be very pure, minimizing any oxide deposits on the graphite surfaces even for long periods of time.

NGNP Project is working with various universities within the NEUP research program to measure any possible changes to the emissivity of graphite during exposure to the reactor core environment. If significant changes are discovered to actually occur, the program will perform additional activities to measure the extent of these changes (Reference 1).

Reference:

1. PLN-2497, 2010, "Graphite Technology Development Plan," Rev 1, Idaho National Laboratory, October 2010.

RAI G-45:
Deleted by the NRC.

RAI G-46: Section 3.3.6.4 states that "the prismatic HTGR design assumes that fuel and reflector blocks will be replaced well before turnaround." Clarify what is meant by "well before turnaround". What specific monitoring and inservice inspection will be used to know the dose limit "well before turnaround?"

Response G-46:
"Turnaround" refers to the fast neutron fluence at which the irradiation-induced dimensional change transitions from shrinkage to expansion. The turnaround fluence depends on the particular grade of graphite, the irradiation temperature, and can vary with grain orientation. Turnaround fluences will be determined as a part of the NGNP AGC program for NGNP Project graphite candidates.

The replacement of fuel elements will be governed by the fuel cycle. The replacement of reflector blocks will be guided by the data from the AGC program, supplemented by examination of replaced blocks. Also, see the response to RAI G-29 for projected fluences at replacement.

RAI G-47: In Section 3.3.6.4, it is stated that "test specimens from these grades were irradiated at 750 °C (1382 °F) up to high fluence (in excess of 20 dpa), beyond turnaround" in Petten irradiation program. However, results have not been provided in the white paper. Based on the results, describe how these grades compare in their volumetric change due to irradiation. Particularly for the prismatic design, apparently contradictory statements have been made about the merits of using fine-grained as well as medium-grained graphite for fuel elements. Provide additional information on the dimensional change behavior of these two grades.

Response G-47:
In the next revision of the white paper, Section 3.3.6.4 (page 43) will be modified to clarify that the Petten experiments were not irradiated to greater than 20 dpa. The NGNP AGC program will be producing irradiation data on both fine and medium grained-graphite in the future.
The following change will be made:

> *Test specimens from these grades were irradiated at 750°C (1382°F) up to fluences of approximately 10 dpa. A second phase of irradiation at 950°C (1742°F) up to fluences of between 12 and 14 dpa has been completed. These irradiations at HFR Petten (Netherlands) aim to provide irradiated properties data that can be used to compare irradiation behavior and post-irradiation properties of the different reactor grades available today. When the HFR Petten irradiation data are publically released, NGNP will compare the Petten data with NGNP irradiation data.*

RAI G-48: In Section 3.3.6.4, it is stated that "Graphite irradiation tests programs are also planned by China in support of the HTR-PM..." Describe what properties will be determined after irradiation using IG-110 graphite and when these data may become publicly available. Are there attempts to determine the extent of similarity of the Chinese irradiated specimens to those which are being irradiated in AGC program?

Response G-48:
The NGNP Project is not intending to use the Chinese test data for qualification of graphite materials, nor is the NGNP Project a participant in the Chinese project. Therefore, NGNP Project has no knowledge of the program beyond the fact that it exists. The discussion was provided as reference to future graphite testing that would be performed. The detailed Chinese test data are considered proprietary and are currently not available to the NGNP Project. Therefore, the data will not be used by the NGNP Project.

During the next revision of the white paper, Section 3.3.6.4 (page 43) will be modified to clarify the graphite irradiation program as follows:

> *Non-NGNP graphite irradiation tests programs are planned by China in support of the HTR-PM, a steam cycle pebble bed concept designed as a commercial follow-on to the HTR-10. The Chinese program is graphite-specific and covers the operating fluence-temperature envelope expected for the HTR-PM. The HTR-PM design is very similar to the German HTR Modul. However, unlike its German predecessor, which employed coarse grain, pitch coke nuclear graphite as reflector material (e.g., ATR-2E, ASR-1RS, PXA2N), the HTR-PM will employ fine-grained, isostatically-molded, petroleum-coke based IG-110 as the reflector graphite. This follows from the use of IG-11 for the HTR-10 graphite reflector. Data from the Chinese program are proprietary and not available to the NGNP Project and, thus, will not be used by the NGNP Project.*

RAI G-49: In Section 3.3.7.1 it is stated that "Oxidation of graphite components must also be considered, however, its influence on component strength and, hence, structural integrity is not expected to be significant for events within the design basis." What is the basis for this statement? For example, what evidence is currently available for the effect of simultaneous irradiation-induced graphite degradation and chemical reactivity degradation with the coolant constituents? Are they cumulative (separately additive) in nature? How are tensile,

compressive, and shear strength of graphite affected by the combined degradation mechanisms?

Response G-49:
The Designer will use the ASME code to account for oxidation in the design of graphite components. ASME Section III, Div. 5, Subsection HH, Subpart HHA 2131 states that the "Design Specification" shall contain the requirements for materials qualification, including envelopes for irradiation and oxidation. Further, HHA 2230 stipulates that data for oxidation effects shall be in the "Materials Data Sheets." Subpart HHA 3000 contains rules for design. HHA 3142 states that graphite shall be considered oxidized when the anticipated weight loss exceeds 1%. Plots are provided for the degradation of strength for several graphite types. HHA 3142 provides rules for designing with irradiated graphite. Changes in strength with irradiation dose are part of the required Materials Data Sheet (HHA Appendix II) but need only be provided if the designer chooses to take account of the increase in strength at low to moderate dose.

In practical terms, the operating specifications for the modular HTGR coolant will preclude the possibility of significant oxidation–related strength degradation during normal operation. In addition, significant degradation would not be expected for events within the design basis.

RAI G-50: In Section 3.3.7.2, "inherent fault tolerance" of the reflector is mentioned in several places. Provide a definition of this term, including discussion of defined engineering margins which are used to support statements in the white paper. Why does this fault tolerance provide safety confidence?

Response G-50:
Section 3.3.7.2, "Reliability and Integrity Management (RIM) Program and the Outcome Objective for RIM Program," as discussed in Section 5.2 Non-Metallic Materials, will be deleted from the white paper. The plant operational considerations presently discussed in Section 3.3.7.2 will be defined during the final design and licensing application phases.

The term "inherent fault tolerance" is not used in the ASME graphite design rules and, therefore, does not represent any concept of design safety. Its inclusion in the white paper was based on the judgment that the basic characteristics of the design make it tolerant to failures.

RAI G-51: In Section 3.3.7.2, visual inspection has been mentioned as a potential candidate for the RIM program. What are the effects of potential coating of cracks or other flaws by graphite dust and thus compromising possible detection? If flaws or cracks or crack-like defects are observed, what mechanisms are there for evaluating the safety significance of these observed "defects"? How will observance of these defects relate to the concept of "inherent fault tolerance" mentioned in this section?

Response G-51:
As stated in the response to RAI G-50, Section 3.3.7.2, "RIM Program" and the corresponding Outcome Objective, Item 3 in Section 5.2, "Non-Metallic Materials," will be deleted from the white paper.

RAI G-52: In Section 3.3.7.2, it is mentioned that "the designs of prismatic reactor concepts typically provide for the replacement of permanent graphite structures should unanticipated

degradation occur." What is the operational experience in replacing such rather tightly packed columns without potential damage to adjacent components and structures? Discuss how lessons learned from this experience will inform NGNP design and operation.

Response G-52:
Actual operating experience with commercial prismatic reactors is limited to Fort St. Vrain. The response to RAI G-7 addresses normal operation removal and replacement of fuel elements and replaceable reflectors in prism designs. Replacement of permanent reflectors in prism designs or replacement of any reflectors in pebble bed designs would be accomplished after first unloading the core. There is operational experience with initial placement of permanent reflectors; however, there is no operational experience with replacement of permanent reflectors in HTGRs.

As stated in the response to RAI G-50, Section 3.3.7.2, entitled, "RIM Program" and the corresponding Outcome Objective, Item 3 in Section 5.2, "Non-Metallic Materials," will be deleted from the white paper.

RAI G-53: In Section 4.2.1.1, it is stated that "Nuclear graphite has been successfully employed in the construction and operation of gas-cooled reactors, including HTGRs, for over 50 years." However, arguably, no such experience exists to extrapolate information for the expected operational conditions of the NGNP. For example, even though the British AGRs have been operating for a long time, their operational conditions are vastly different than the NGNP HTGR. Additionally, cracking in fuel element channels and control rod channels have been observed in the British AGRs since 2002, while such cracking was not initially expected. It appears that EDF Energy is providing revised safety case documents based on inspection data and the British regulators are continuously reviewing these documents to authorize continued reactor operation. The other HTGRs have not had field experience for such long term operation, including the Japanese HTTR and the Chinese HTR-10. Thus, the statement about "successful operation" is arguable. As has been stated in the white paper, properties information are lacking and are being generated currently for the newer grades of nuclear graphite. Provide additional discussion of the relevance and limitations of applying operating experience for graphite in other gas-cooled reactors to NGNP.

Response G-53:
The NGNP Project uses foreign reactor high temperature reactor program data only as background material and will not use this data for final design or licensing. The NGNP Project cannot comment on technical aspects of other foreign governments' or foreign countries' high temperature gas reactor programs, since these programs and reactors are not under NGNP Project control, nor will they be used to support licensing of the NGNP.

In the next revision of the white paper, Section 4.2.1.1 (page 63) will be modified to clarify ASME code for graphite structures as follows:

> *Nuclear graphite has been employed in the construction and operation of gas-cooled reactors, including HTGRs, for over 50 years. Up until this time no industrial design code or regulatory basis was in existence to assist in the regulatory approval process concerning graphite core structures. In the U.S., past HTGR designs were licensed as a*

DOE demonstration reactors with a Class 104 NRC license. An ASME code for graphite structures is presently being developed as ASME Section III, Division 5, Subsection HHA.

RAI G-54: Section 4.2.1.1 includes a discussion of a test program planned for deployment of the PBMR in South Africa. However, this project no longer exists, so the relevance of those plans to NGNP is questionable. Provide revisions to the white paper to reflect the current status of the PBMR project, including test programs which may not have been completed before the project was terminated.

Response G-54:
The following paragraphs in Section 4.2.1 (page 65) will be deleted in the next revision of the white paper, since, the PBMR program is no longer accessible by the NGNP Project, and the data will not be used by the NGNP Project for graphite qualification.

As an example, Figure 5 illustrates the relationship between the legacy German database and the PBMR Specific Materials Test Reactor Program irradiation conditions that were selected earlier for the proposed PBMR Demonstration Power Plant in South Africa. The solid blue line in the figure represents the projected temperature-fluence envelope at the end of service life for components that serve a structural function (SRC-1, as defined in Table 4), whereas the dotted red line denotes a similar envelope for the most highly-irradiated nonstructural components adjacent to the pebble fuel (SRC-2). As shown in Figure 5, the primary and secondary MTR data are designed to both confirm the applicability of the historical data and to supplement that data where required.

Finally, the proposed service life of the graphite components in the PBMR implies the need for a relatively lengthy MTR program. On this basis, the PBMR approach is to acquire MTR data for a significant portion of the service life prior to the start of the lead reactor. The balance of the MTR data would be acquired in such a manner that it substantially leads the actual operation of the reactor.

RAI G-55: In Section 4.2.1.3, it is stated that "The PIRT panel also concluded that theories that can explain graphite behavior have been postulated and, in many cases, shown to represent experimental data well." It is the NRC staff's understanding that phenomenological (curve-fitting) relationships developed using MTR data has worked reasonably well to predict graphite behavior, specifically for the British AGRs, but do not reflect operating conditions projected for the NGNP HTGR. Also, the NRC staff is not aware of any tested and proven theories which have explained the graphite behavior for all the graphite properties. British AGRs have experienced cracking in unexpected places, and in cracking modes that were not predicted. There is also some Japanese work which has developed theory for volume change due to irradiation; however, the staff is not aware of any operational experience which has supported such theory. Provide additional discussion of the relevance and limitations of applying current graphite behavior theories to NGNP.

Response G-55:
The graphite behavior theories discussed in the PIRT panel will not be used for final NGNP design or NGNP licensing. The ASME graphite core design rules use empirical mechanical and thermal data collected from qualified providers with known statistical confidence levels. The

relevance of these graphite behavior theories is moot, since the ASME graphite core design rules will be used.

The second paragraph in Section 4.2.1.3 will be deleted, which includes the statement:

> *The PIRT panel also concluded that theories that can explain graphite behavior have been postulated and, in many cases, shown to represent experimental data well.*

Carbon Composite RAIs No. 5899 Revision 0

The NGNP Project plans to update Sections 3.5 and 4.2.3, Composites Materials, during the next revision of the white paper. Section 4.2.3 "Composites" will be deleted and Section 3.5 will be updated to clarify the purpose of the information being provided for this section. The information provided in this section is presented as background information providing a historical summary of nuclear composite development and manufacturing used in previous U.S. reactor designs as well as foreign reactor programs. The introduction of these foreign high temperature gas-cooled reactors is not intended to represent them as a basis for NGNP core component qualification. The following provides the intended change for Section 3.5:

> *Currently there has been no application of Carbon Fiber Reinforced Carbon (CFRC) composites in nuclear reactors, including HTGRs. Germany conducted some development work on CFRC components in the past, and the US NGNP program has addressed some current aspects of structural CFRC components for the NGNP design. However, the use of composite components are very design specific, with some reactor designs requiring components made from composite materials but other designs not requiring any composite components. Because specific composite components have not yet been selected for development within the NGNP Project, the project cannot provide specific information on areas for composite research and development.*
>
> *In addition, there are no established general design codes or standards addressing composite materials for HTGR applications that are analogous to the ASME Code for metallic or graphite components. However, there is a plan within ASME to address this matter through the ASME Subgroup on Graphite Core Components (SGCC). This subgroup has been officially sanctioned by the Board on Nuclear Codes and Standards as part of the BPV Code Section III infrastructure. The SGCC has concentrated its efforts to date on nuclear graphite (see Section 3.3.4); however, high temperature composites are also a part of the subgroup charter. The ASME Code section on nuclear graphite will be issued in 2011, and the expectation is that the composites code will be addressed thereafter.*
>
> *Given the current status, the proposed bases for qualification of composite components have not yet been developed. However, general information from past and present research and development programs for composite components is presented herein to provide a background understanding of the general issues that are involved in qualifying composites for use within HTGRs. The specific bases for qualification and regulatory review are to be developed later.*

In addition, the NGNP Project plans to update Section 5.2, Outcome Objectives for Nonmetallic Materials, based on above clarification

RAI COMP-1: Information presented in Section 3.5.2.1 is unclear, because the meaning and usage of qualitative adjectives used (i.e., high, low, extremely high, excellent, relative, moderate, etc.) is unclear. The text states such attributes should be "considered." Since these are relative terms, what bases are being used for comparison? Are the properties determination methods the same for these varying attributes, namely for the entire range or the methods could differ for different ranges for these properties?

Response COMP-1:
See the revised introduction regarding the status of the NGNP design and development of composite materials (i.e., NGNP's General Response for Carbon Composite RAIs).

Once a reference design for the NGNP core components have been established quantitative measurements/attributes will be presented in support of the composite components. These measured properties will be compared to the specification requirements established within the reference design criteria.

RAI COMP-2: Section 3.5.4.2: With respect to fatigue resistance, especially for tie-rods, straps, and other functional components, how will the acceptable fatigue limits be determined and applied? What type of modification to currently acceptable fatigue rule will be made?

Response COMP-2:
See the revised introduction regarding the status of the NGNP design and development of composite materials (i.e., NGNP's General Response for Carbon Composite RAIs).

However, in general, the topic of composite fatigue has been discussed within the ASME Subgroup on Graphite Core Components (SGCC). The SGCC faces a similar challenge in establishing rules and codes for these components since composites are usually fabricated/manufactured specifically to the requirements for each specific component. This makes development of a generic code for all composite components with different geometries, thermal requirements, mechanical needs, and chemical interactions particularly difficult. Specific requirements such as fatigue resistance will require a careful analysis of the component as well as the conditions the component is expected to experience. The SGCC is determining whether a general methodology can be imposed or whether this analysis should be the responsibility of the applicant.

RAI COMP-3: Section 3.5.4.2: What is the effect of oxidation on the fatigue strength of these composites? What information exists or needed to establish the effects of oxidation corrosion and potential cyclic fatigue of C-C and/other ceramic tie rods and functionally-related core internals, such as straps. Describe plans to demonstrate seismic reliability of such components via analytical and/or experimental programs?

Response COMP-3:
See the revised introduction regarding the status of the NGNP design and development of composite materials (i.e. NGNP's General Response for Carbon Composite RAIs).

However, in general, the topic of composite degradation and the effects of degradation on the composite performance are being investigated by the NGNP Project primarily through research grants to university research programs (i.e., the Nuclear Energy University Program [NEUP]). Long term composite degradation mechanisms and effects have been listed as areas of research interest in the NEUP program for a number of years, and some research, both at universities and in national labs, has been undertaken for this issue. In general, the phenomenon of fatigue corrosion in C-C composites for nuclear applications has not been specifically addressed to date. It is anticipated that some data will be available from other non-

nuclear applications such as C-C composite airplane braking pad development, composite aerospace applications, and other hi-tech applications utilizing composites.

Note that with the exception of the core outlet connection, which provides the transition between the core outlet plenum and hot gas duct, the identified components of interest would normally see low temperatures associated with the core inlet flow. Elevated temperature conditions associated with conduction cooldown LBEs would be limited in duration and, since there is no flow, would not impact the core outlet connection. The normal operating environment of the helium primary coolant would not be expected to result in significant oxidation.

RAI COMP-4: In Section 3.5.4.2, it is mentioned that fatigue tests were performed with component level tests for tie rods racetrack strap components. At what temperature and environment were these tests conducted, and how do those conditions compare to expected NGNP service conditions? Were such tests conducted with "reference" material to qualify the test?

Response COMP-4:
See the revised introduction regarding the status of the NGNP design and development of composite materials (i.e. NGNP's General Response for Carbon Composite RAIs).

The component-level tests mentioned in Section 3.5.4.2 were conducted by the PBMR Program in South Africa. As noted in the response to RAI G-36, the PBMR Program has been terminated and the specific conditions of these tests are not presently available to the NGNP Project.

COMP-5:
Deleted by the NRC.

RAI COMP-6: Section 3.5.4.2: How will composite components, such as tie rods and straps, be attached or bonded to other components? What are the structural requirement considerations for these linking or bonding " devices?

Response COMP-6:
See the revised introduction regarding the status of the NGNP design and development of composite materials (i.e. NGNP's General Response for Carbon Composite RAIs).

The tie rods and straps are mechanically attached. The specific details of the design were developed by PBMR, (Pty) Ltd., but are not available to the NGNP Project.

COMP-7:
Deleted by the NRC.

RAI COMP-8: Section 3.5.4.2: Discuss the need for and types of redundancy and/or diversity which will be considered for tie rods and straps.

Response COMP-8:
See the revised introduction regarding the status of the NGNP design and development of composite materials (i.e. NGNP's General Response for Carbon Composite RAIs). The specific

details of the design were developed by PBMR, (Pty) Ltd., but are not available to the NGNP Project.

Ceramic Insulation RAIs No. 5900 Revision 0

NGNP's General Response for Composite RAIs:

The NGNP Project plans to update Section 3.4, Ceramic Insulation Materials, during the next revision of the white paper. The update will clarify the purpose of the information being provided for this section. The information provided in this section is presented as background information providing a historical summary of nuclear ceramic insulation material development and manufacturing used in previous reactor designs including foreign reactor programs. The introduction of these foreign high temperature gas-cooled reactors is not intended to represent them as a basis for NGNP core component qualification. The following provides the intended change:

> *Graphite, the principal material used in HTGR core structures, has a relatively high thermal conductivity. The high conductivity of graphite is advantageous in terms of transport of heat from the fuel to the helium coolant during normal operation and via the Reactor Cavity Cooling System during certain LBEs. In some applications, however, it is desirable or necessary to control the flow of heat from the graphite core structures to adjacent metallic components (e.g., core support structure) to avoid excessive temperatures. Ceramic insulation may be used in conjunction with the graphite core structures to achieve this objective. Two classes of ceramic insulation have been used in HTGRs to date, baked carbon and fused or sintered quartz. While quartz-based materials provide a greater degree of insulation, baked carbon is often utilized, where practical, based on economic considerations and the similarity of its properties (e.g., neutronic properties, coefficient of thermal expansion) to those of the adjacent graphite core structures*
>
> *At present, specific ceramic insulation materials have not yet been selected for development within the NGNP Project.*
>
> *In addition, there are no established general design codes or standards addressing ceramic insulation materials for HTGR applications that are analogous to the ASME Code for metallic or graphite components. However, there is a plan within ASME to address this shortcoming through the ASME Subgroup on Graphite Core Components (SGCC). This subgroup has been officially sanctioned by the Board on Nuclear Codes and Standards as part of the BPV Code Section III infrastructure. The SGCC has concentrated its efforts to date on nuclear graphite (see Section 3.3.4); however, high temperature composites and manufactured carbon (ceramic insulation) materials are also a part of the subgroup charter. The ASME Code section on nuclear graphite will be issued in 2011, and the expectation is that the composites and manufactured carbon material codes will be addressed thereafter.*
>
> *Given the current status, the proposed bases for qualification of ceramic insulation components have not yet been developed. However, general information from past and present research and development programs is presented herein to provide a background understanding of the general issues that are involved in qualifying ceramic*

insulation components for use in HTGRs. The specific bases for qualification and regulatory review are to be developed later.

In this context, a discussion of baked carbon for use as core insulation is given in this section. In the subsections that follow, additional details are given regarding the manufacturing processes, properties, and prior uses of baked carbon. This is followed by a summary of the approach for the design and structural evaluation, as well as the representative bases for qualification of these insulation components.

In addition, the NGNP Project plans to update Section 5.2, Outcome Objectives for Nonmetallic Materials, based on the above clarification.

RAI INS-1: Table 7 in Section 3.4.2 provides properties data for carbon insulation and nuclear graphite. What ASTM material specification standard exists for carbon insulation materials for nuclear application to state nuclear grade carbon? Considering that the thermal conductivity of carbon insulation is lower than graphite, and thus potentially reducing resistance to thermal shock, what structural integrity requirements are contemplated for carbon insulation? Later, in Section 3.4.4, it is mentioned that Structural Reliability Class (SRC)-3 will be assigned to ceramic insulation components. What information is available on the strength of this insulation after multiple thermal shock loadings, and after irradiation?

Response INS-1:
See the revised introduction regarding the status of the NGNP design and development of ceramic insulation materials (i.e., NGNP's General Response for Composite RAIs).

Note, however, that the transients that would be seen by thermal insulation between the outer edges of the graphite reflector and the core barrel or between the graphite core structures and metallic core support would be comparable to those that would be seen by the graphite core structures, i.e., with rates measured in hours to days. For this reason, thermal shock is not expected to be a significant issue.

RAI INS-2: In Section 3.4.3, "nuclear grade ASR-0RB carbon" is discussed. What is the specification for this "nuclear grade" material? Is NBC-07 also a "nuclear grade" carbon? If so, what specification makes it a "nuclear grade"?

Response INS-2:
As discussed in the introduction of this section, only general information from past and present research and development programs is presented herein to provide a background understanding of the general issues that are involved in qualifying ceramic insulation components for use within HTGRs. The NGNP Graphite Technology Development Program does not have specifications for ASR-0RB or NBC-07 carbon, nor are the specifications outlined within ASTM standards or the ASME code. Presumably, there was a specification that was accepted by the Japanese regulatory authority for HTTR, but at the present time the NGNP Project is not specifically requiring the use of these materials in the reactor core, and no information is currently available to the NGNP Project.

RAI INS-3: In Section 3.4.3, it is stated that "Shaped carbon blocks are used at various locations in the core structures to insulate selected components from hot gas flow." What type

of thermal (heat transfer) analysis procedures are used to ensure adequate carbon thickness for the "shaped" carbon component to provide the required insulation? Does the thermal conductivity of carbon depend on temperature and neutron dose? If so, how this dependence accounted for in thermal calculations? Are such calculations based on consensus standard procedure?

Response INS-3:
See the revised introduction regarding the status of the NGNP design and development of ceramic insulation materials (i.e., NGNP's General Response for Composite RAIs).

However, in general, the location of these components (i.e., outside of the core) will result in minimal radiation dose levels to the material. It is anticipated that minimal property changes will result from this low neutron dose. However, this assumption will be verified through neutron analysis once the core design specifications have been drafted, and appropriate thermal analysis will be conducted.

RAI INS-4: In Section 3.4.3, it is stated that "the carbon blocks around the reflector, at the top of the reflector and the upper insulation layer below the reflector, have 5 wt% boron carbide added to the carbon to reduce neutron irradiation to the adjacent metallic components in these areas." How does the addition of boron carbide affect the thermal conductivity as a function of temperature? What is the time-dependency of thermal conductivity at temperature for the life time of the carbon component?

Response INS-4:
Section 3.4.3 provides a discussion of past experience with the use of baked carbon insulation in HTGRs, including the HTTR and HTR-10. The information provided in this section is primarily provided for historical purposes and not intended for direct application to the NGNP design. Past experience, based on thermal measurements taken by manufacturers, has shown that the addition of small amounts of boron carbide to ceramic insulting materials has minimal to no effect on the thermal conductivity of these materials. Additionally, in general, there is no indication that the addition of boron carbide to these materials significantly alters the lifetime thermal diffusivity/conductivity performance.

Further testing and data will be collected for qualifying the material once a baked carbon insulation material type(s) is selected for use.

RAI INS-5: In Section 3.4.3, it is stated that "the carbon insulation blocks were produced by the Lanzhou Carbon Works to ensure low thermal conductivity and good dimensional stability at high temperature." Clarify this statement to clearly identify what component or components are being described (e.g., the insulating block, or metallic structures) Why is dimensional stability important and to what extent? How do you assure that the good dimensional stability is maintained? What inspections would be conducted to assure this expected behavior?

Response INS-5:
See the revised introduction regarding the status of the NGNP design and development of ceramic insulation materials (i.e., NGNP's General Response for Composite RAIs).

However, in this general example the components being described are for insulating blocks at the bottom of the reactor. As such, they require low thermal conductivity/diffusivity properties to thermally protect the adjacent metallic components and high dimensional stability, since the support columns for the entire graphite core rest upon these insulating blocks and they must provide a very stable structure to ensure physical core stability.

RAI INS-6: In Section 3.4.4, it is stated that "the use of ceramic insulation in the CSC is restricted to areas where it is not exposed to significant fast neutron irradiation (e.g., <1018 n.cm-2 EDN) and, consequently, irradiation-induced changes in properties will be negligible." The value given for fast neutron irradiation appears to be misformatted. Is it correct to assume that it should appear as $<10^{18}$ n cm^{-2}? Throughout Section 3.4, clarify the various operating conditions, the specific materials to be used, and properties required to assure expected design performance.

Response INS-6:
The comment is correct. The intended value is 10^{18} n/cm^2 EDN (Equivalent DIDO Nickel). This dose corresponds to approximately 0.0013 dpa and illustrates that the fast neutron dose is very small for these components. This formatting error will be corrected, and the conversion will be added to Section 3.4.4 in the next revision of the white paper. The various operating conditions, the specific materials to be used, and properties required to assure expected design performance cannot be provided at this time because they will be determined by the final design.

RAI INS-7: Table 8 in Section 3.4.5, characterizes baked carbon insulation properties. Temperature-dependent friction coefficient in helium between graphite and carbon is one of the properties. This friction property between graphite and baked carbon insulation has not been identified for characterization in Sections 3.3.6.2 or 3.3.6.3 for graphite. Explain this discrepancy.

Response INS-7:
The friction coefficient between graphite and baked carbon would be required only if baked carbon is used. Thus, it is included in the baked carbon section.

RAI INS-8: In Section 3.4.5, is the PBMR procedure for determining equivalent boron content, mentioned in Table 8, publicly available? Provide a description of the procedure, and discuss any review and approval by a consensus standards organization, such as ASTM?

Response INS-8:
See the revised introduction regarding the status of the NGNP design and development of ceramic insulation materials (i.e., NGNP's General Response for Composite RAIs).

However, in general, there are ASTM standards, that specify the maximum boron content levels and how to determine the equivalent boron contents of nuclear materials. It is anticipated that these ASTM standards may be used when the NGNP reference design has been established or a designated designer identifies the specifications for ceramic insulating components. These standards include:

- ASTM D 7219, Standard Specification for Isotropic and Near-isotropic Nuclear Graphites

- ASTM C 1233, Practice for Determining Equivalent Boron Contents of Nuclear Materials.

As noted in the response to RAI G-36, the PBMR Program has been terminated. The PBMR procedure for determining equivalent boron content is not presently available to the NGNP Project.

RAI INS-9: Section 3.4: Cracking and continuously inter-connected porosity is a possibility for ceramic materials. Thus, while insulating characteristics may be functionally adequate, what is the possibility of the coolant flowing through this inter-connected porosity ("thermal streaking") which could potentially lead to hot spots continuously impinged by hot gas on the metallic strictures? If such thermal streaking is possible, then how can the design assure the structural integrity of pressure boundary material?

Response INS-9:
See the revised introduction regarding the status of the NGNP design and development of ceramic insulation materials (i.e., NGNP's General Response for Composite RAIs).

However, in general, fracture behavior of manufactured carbon and graphites, which depends greatly upon the defect microstructure, is currently being studied by NGNP graphite program (Reference 1) and the ASTM D02.F0 subcommittee on Manufactured Carbons and Graphite. The general observation is that the defect population within these manufactured carbon/graphite structures is not large or interconnected enough to pose a credible scenario where enough coolant gas flows through to create thermal streaking. The issue of bypass flows associated with fractures is essentially the same as stated for structural graphite components. The designer must demonstrate that coolant flows associated with credible fractures will not result in significant (i.e., sufficient to result in overheating of the metallic components) radial overheating from the active core, where heating is taking place, to the core periphery, where the core barrel or core support structure are located. Since heated coolant flows tend to involve cooler inlet helium (which is at higher pressure), it is not likely that this mechanism will pose a significant issue.

Reference:

1. PLN-2497, 2010, "Graphite Technology Development Plan," Rev 1, Idaho National Laboratory, October 2010.

RAI INS-10: Section 3.4 states that ceramic insulation will be adjacent to metallic components. How will reactor operating conditions contribute to thermodynamic and chemical reactions which may occur between these materials? What are the consequences resulting from potential degradation due to such reactions? If there is spalling of insulation refractory materials, as is typical in the case of refractory (thermal protection) use, what are the consequences? How will such degradation be detected and its effects mitigated?

Response INS-10:
See the revised introduction regarding the status of the NGNP design and development of ceramic insulation materials (i.e., NGNP's General Response for Composite RAIs).

Note, however, that the temperature of the metallic components in the protected regions are close to the reactor inlet temperature during normal operation (<300°C), and, thus, such reactions are unlikely. LBEs involving significantly elevated temperatures would be rare and of limited duration and location.