May 31, 2002

Kevin Borton, Licensing Manager
Exelon Generation
200 Exelon Way
Kennett Square, Pennsylvania 19348

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) ON HIGH TEMPERATURE MATERIALS GRAPHITE; CONTROL OF CHEMICAL ATTACK; AND DESIGN CODES AND STANDARDS FOR THE PEBBLE BED MODULAR REACTOR (PBMR)

Dear Mr. Borton:

The U.S. Nuclear Regulatory Commission's (NRC's) objectives for the Pebble Bed Modular Reactor (PBMR) pre-application review are to obtain information from Exelon on the PBMR design and its technical bases in order to: (1) identify significant technical issues, safety issues and policy issues and, (2) identify a path for resolution of the issues. Achieving these objectives is expected to enhance the effectiveness and efficiency of the staff's review of an actual PBMR license application, and to provide guidance to Exelon that is useful in the preparation of an application.

Since June 2001, the NRC staff has conducted periodic public meetings with the Exelon Generation Company (Exelon) and the U.S. Department of Energy (DOE) to receive presentations and obtain information on a range of technical and programmatic topics supporting the PBMR pre-application review. These periodic meetings provided a starting point for obtaining information from Exelon on the PBMR design and its technical bases and for identifying significant issues for which staff resolution guidance would be pursued.

Early in the pre-application review, the staff requested Exelon to document the information that had been informally presented in these meetings and to formally submit it for staff review. Accordingly, between October 2001 and March 2002 Exelon formally submitted the requested documents as technical "white papers." As shown in Enclosure 1, Exelon submitted white papers for most of the fourteen technical and programmatic topics that were presented at the public meetings.

Exelon requested that the staff provide feedback on the technical, safety or policy issues, including staff questions related to each of the submitted technical white papers and associated presentations. The white papers, including any updates and formal responses to staff identified issues and questions, were to provide the primary basis for the staff's pre-application review findings, conclusions, positions and guidance.

The purpose of this letter is to provide the staff's feedback on technical, safety or policy issues in terms of requests for additional information (RAIs) on selected technical white papers and the associated meeting presentations. The selected white papers (and number designations) are: (1) "Graphite Presentation to USNRC in Support of PBMR Pre-Application Activities"; (2) "Control of Chemical Attack in the PBMR"; (3) "Summary of PBMR Design Codes and Standards"; and (4) "RPV and Connecting Piping."

Enclosure 2 contains the RAIs for each of the four white papers. The RAIs for each paper have been grouped into one of two categories. The RAIs in Category 1 are those that are considered relevant to either policy issues or significant safety or technical issues that are the focus of the PBMR pre-application review. Category 2 RAIs involve technical issues for which responses would be required if a PBMR license application were to be submitted. Additionally, one RAI in Category 2 is considered significant to support the development of the NRC's infrastructure of tools, data and expertise that would be needed to conduct a PBMR license application review. This RAI has been identified with an asterisk (*) and any response would be of most benefit to the staff if provided in advance of a license application. All RAIs in Enclosure 2 have been identified by white paper number and RAI category.

The staff recognizes Exelon's announced plans to end its participation in the PBMR project in South Africa when the current PBMR feasibility study is completed, and that Exelon plans to terminate its PBMR pre-application review activities with the staff. Therefore, the staff understands that in most cases Exelon does not plan to respond to the enclosed RAIs. Even so, we believe that there is a mutual desire to address and document the current PBMR pre-application review work in manner which would be of benefit to the staff, to others who might seek to resume PBMR pre-application review activities and to interested stakeholders. Therefore, the staff is transmitting the enclosed RAIs to formally document the results of the staff's review of these white paper topics to date and to place them on the public record.

It is requested that you review the enclosed RAIs and respond as to whether or when the requested information will be provided by Exelon to the NRC.

The reporting and/or record keeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Please contact me (301-415-7499) or Stuart Rubin (301-415-7480) if you have any questions on this request.

Sincerely,

/RA/

Farouk Eltawila, Director
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Project No. 713

Enclosures: As stated

cc w/encls:
Standard Service List Addresses

Letter dated: 5/31/02

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) ON HIGH TEMPERATURE MATERIALS GRAPHITE; CONTROL OF CHEMICAL ATTACK; AND DESIGN CODES AND STANDARDS FOR THE PEBBLE BED MODULAR REACTOR (PBMR)

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Letter dated: 5/31/02

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Table 1: PBMR Pre-Application Review Technical and Programmatic Topics Presented and Documented by Exelon

Meeting Date	Meeting Presentation Technical Topic	Exelon Technical White Paper: Title of Paper	Transmittal Date	No.
Jun 12-13, 2001	Fuel Overview -Design, Manufacturing, QC and Qualification	PBMR Nuclear Fuel	11/16/01[1]	8
		Fuel Fabrication Quality Control Measures and Performance Monitoring Plans for PBMR Fuel	1/31/02	9
Jul 17-18, 2001	Design Codes and Standards	Summary of PBMR Design Codes and Standards	10/30/01	3
		RPV and Connecting Piping - White Paper	12/17/01	4
	Fuel Irradiation Program	(See Technical White Paper No.10)	3/18/02	-
Aug 15-16, 2001	Analytical Codes and Software Control	PBMR Analytical (Computer) Codes Data Table	10/30/01	5
	Fuel Design Logic	None	11/16/01	-
	Core Design	PBMR Design and Heat Removal Preliminary Description	3/04/02	6
	Heat Removal			
Oct 25, 2001	High Temperature Materials Graphite	Graphite Presentation to USNRC in Support of PBMR Pre-application Activities	10/23/01	1
	Control of Chemical Attack	Control of Chemical Attack in the PBMR	10/23/01	2
	Systems Design Approach and Status	None	N/A	-
	High Temperature Materials	None	N/A	-
Nov 29-30, 2001	Operational Modes and States	PBMR Operational Modes and States	11/27/01	7
	Testing Requirements for a Combined License	Testing Requirements for Issuance of a Combined License	11/27/01	11
Mar 28 2002	Fuel Qualification Test Program	Pebble Bed Modular Reactor Fuel Qualification Test Program	3/18/02	10

[1]Paper was later withdrawn.

References:

1. Slides of the July 18, 2001, presentation by Vijay Nilekani, Exelon, “PBMR Design Codes and Standards.”

2. Letter from K. F. Borton, Exelon, to the Document Control Desk, “Documents Supporting the October 25, 2001, Pre-Application Meeting Regarding the Pebble Bed Modular Reactor (PBMR),” dated October 23, 2001, with the following attachments:

   Attachment 1, titled “Graphite Presentation to USNRC in Support of PBMR Pre-Application Activities.”

   Attachment 2, titled, “Control of Chemical Attack in the PBMR Presentation to USNRC in support of PBMR Pre-Application Activities.”

3. Slides of the October 25, 2001, presentation by Mark A. Davies, Exelon, “Pebble Bed Modular Reactor High Temperature Materials Graphite.”

4. Slides of the October 25, 2001, presentation by Mark A. Davies, Exelon, “Pebble Bed Modular Reactor High Temperature Materials.”

5. Letter from K.F. Borton, Exelon, to the Document Control Desk, “Summary of Preapplication Presentations Regarding the PBMR,” dated October 30. 2001, with the following attachments:

   Attachment 1: Summary of the PBMR design codes and standards information presented to the NRC on July 18, 2001.

   Attachment 2: PBMR Analytical (computer) Codes Data Table presented to the NRC on August 16, 2001.

6. Letter from K.F. Borton, Exelon, to the Document Control Desk, “Information Regarding Materials and Design Codes to be Used for the RPV and Connecting Piping for the PBMR,” dated December 17, 2001, with an attached PBMR document, “RPV and Connecting Piping - White Paper,” Rev. 1, dated December 5, 2001.

7. Letter from K.F. Borton, Exelon, to the Document Control Desk, “Response to NRC letter dated September 26, 2001, Regarding the PBMR Technical Information Availability,” dated November 15, 2001.

8. Exelon Presentation, “Process for Selection of Licensing Basis Events for the PBMR,” Fred Silady, July 17, 2001.

White Paper No. 1:
Graphite Presentation to USNRC in Support of PBMR Pre-application Activities

REQUEST FOR ADDITIONAL INFORMATION
PEBBLE BED MODULAR REACTOR

HIGH TEMPERATURE MATERIALS, GRAPHITE

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General Comments

Category 1 RAIs are considered relevant to either potential policy issues or significant safety issues or technical issues that are the focus of the PBMR pre-application review.

1.1.1. The staff understands that the irradiation database for the graphite materials is very limited and will not cover the irradiation requirements of the PBMR for a design plant life of 35 fpy. Furthermore, most of the test data are derived from graphite materials exposed to conditions of relatively low fluence and temperature. Exelon indicated that several semi-empirical relationships are developed based on such data. Exelon plans to use those empirical relationships to assess the graphite performance at PBMR conditions. The staff has concerns regarding the use of these semi-empirical relationships to extrapolate the material properties for conditions of high fluence and temperature, because the extrapolated material properties may not be conservative when the relationships are based only on the low fluence/low temperature test data. To ensure there is adequate safety margin in component designs, the graphite material properties used in design need to be verified or validated by testing of the specific graphite materials selected for manufacturing the reactor components.

Category 2 RAIs are considered relevant to safety issues or technical issues for which responses would be required at the time of or before a PBMR license application is submitted.

1.2.1. The information provided is of very general nature and does not contain detailed stress analysis of graphite components, including its nonlinear behavior, for normal and transient conditions of PBMR service. Specific life expectancy estimations under expected PBMR service are lacking. In the white paper, several design and analysis concepts from UK, Germany, and Japan were briefly introduced, without providing a list of references for readers to find out further details. It will be more useful to provide a list of references from where the information was extracted.

In particular, on topics of:

1. Component Performance Assessment (UK, Japan, Germany Approaches on Pages 19-22)
2. Performance Assessment Criteria (UK, Germany Approaches on Pages 22-23)
3. Irradiation Induced Creep (UK creep model, and Russian and US creep laws on Pages 17-18)
4. Material Properties - Irradiated Behavior
   a) Figure 2--Graphite Damage Mechanism (after Simmons [4]) on Page 10
   b) Figure 3-- Crystal Dimension Changes (after Kelly [6]) on Page 11

1.2.2. In general, the data presented and the statements made in the white paper do not indicate whether they apply to an air, helium, or some other atmosphere such as CO_2 in the case of British data. The type of atmosphere could potentially and, perhaps, significantly affect the behavior of graphite at high temperatures and irradiation. In general, for all data presented for review it should be indicated what the atmosphere was under which the data were collected, or a justification given why data collected in another atmosphere is applicable. The effects of potential contamination and detrimental impurities in helium on the properties used for design of graphite components should also be addressed.

1.2.3. A case should be adequately presented why the data on 'old' graphites can be expected to support and predict the behavior of 'new' and yet-to-be manufactured graphites, whose raw materials for manufacture are different, for PBMR application. The need for additional materials behavior data for the specific grade of graphite selected for the PBMR reflector has been recognized and an irradiation program proposed. However, the proposed irradiation program needs to be reconsidered in light of what can reasonably be achieved in high dose irradiation experiments.

(a) Will the MTR program obtain information on the effect of potential air/moisture ingress on properties used in the design of graphite components?

(b) When would this MTR program proposed by Exelon be completed?

(c) In determining the design properties of irradiated graphites, the specimens sizes should conform to ASTM or other standard and be comparable to the sizes of on unirradiated graphite test coupons for valid comparison. How do the available capsule designs and irradiation facilities affect the standard specimen sizes for testing?

(d) Will data relating to carbon/carbon composites be generated in the MTR program?

1.2.4. The white paper should provide information on the verification and validation for all the equations and relationships used to support the PBMR graphite properties and their variability.

Specific Comments

1.2.5. On page 2 of Attachment to Reference 2, it is stated that the top reflector is suspended from the core barrel top plate by means of carbon-carbon composite tie rods. Exelon should provide detailed information regarding the mechanical and physical properties of the carbon-carbon composite materials, method of fabrication and installation, and the geometry of mechanical joints that connects the graphite top reflector to the core barrel top plate. The details of design criteria including materials property requirements and nondestructive evaluation procedures should be identified and discussed. If a threaded configuration is designed for the mechanical joint, other details such a lubricants or surface treatments and pre-load provisions should be provided. Exelon should also discuss the bases for selecting this material for tie rods and provide the service experience of the carbon-carbon composite in an environment similar to PBMR.

1.2.6. On page 3 of Attachment to Reference 2, It is stated that the outlet plenum is designed so that the differential temperature within the gas leaving the core is less than 60 ^{o}C. What type and magnitude of stresses does this temperature differential produce in the bottom reflector?

1.2.7. On page 3 of Attachment to Reference 2, please explain how the geometry requirement for the reactivity control shutdown system (RCSS) has a significant impact on the structural performance of the blocks during the lifetime of the plant. Are the RCSS channel graphite sleeves replaceable?

1.2.8. On page 3 of Attachment to reference 2, please provide additional details of the method for vertically stepping the top blocks of the bottom reflector.

1.2.9. The table on page 4 of Attachment to Reference 2 contains data of temperature and fluence level at various locations inside the PBMR reactor. Please provide details regarding how these data are derived. If these data are analytically calculated, please identify the procedures, assumptions and models/computer codes that are used for the calculations. If you plan to use those data for design of PBMR reactor components, provide the following information (i) the scatter bands and/or uncertainties that are associated with the calculated results, (ii) your plan to verify/validate the calculated results by testing in PBMR configuration and operating conditions.

1.2.10. On page 6 of Attachment to reference 2, for the graphite properties data presented, please provide information on how these data relate to (a) properties specifications for the various graphite components used in the PBMR, (b) manufacturing specifications for the specific graphite components of the PBMR that ensures the requirements of properties specifications, and (c) national or international standards used to test, measure, and analyze the properties specified in properties specifications. Additional information unique to the application in the PBMR environment and condition should also be provided.

1.2.11. On page 6 of Attachment to reference 2, it is mentioned that PBMR has chosen Sigri Great Lakes as the preferred supplier for the graphite reflector. A table is shown of some of physical properties of the unirradiated graphites. Does data exist for these products for irradiated conditions? In particular, if these data exist, what is their relation to the PBMR service conditions?

1.2.12. In the properties table, given on page 6 of Attachment to Reference 2, what are the significant properties for design? What are their test variability? What is their billet-to-billet, and within billet variability? How do manufacturing methods affect them? How are the properties variations accounted for, including the impurity levels?

How do the properties listed in the Table on page 6 of Attachment to Reference 2 reflect the properties of ATR-2E and VQMB graphites mentioned on page 15 of Attachment 2?

1.2.13. On page 13 of Attachment to Reference 2, it is not clear what measure of fast neutron dose is to be used in the analysis and prediction of graphite behavior at high temperatures in PBMR. If the integrated flux above 0.18 Mev is used, what is the basis for this choice? If EDND is used, what is the evidence that the Thompson-Wright damage function and the standard nickel flux in DIDO are appropriate for PBMR conditions?

1.2.14. On page 14 of Attachment to Reference 2, it is stated that PBMR will ignore the concept of equivalent temperature. Until such time as better higher temperature, high fluence data become available. When will such data become available? How will the data be analyzed for applicability to PBMR temperatures and fluences? Does this imply that PBMR does not intend to use the other data or will a justifiable correlation between the previous data and PBMR be supplied?

1.2.15. On page 15 of Attachment to Reference 2, it is stated that "It is assumed that for the type of graphite to be used in PBMR, in the temperature and fluence range of interest, the graphite behavior is consistent, i.e. the material properties, when irradiated in a similar flux, may be described by mathematical equations, which are functions of irradiation temperature and dose."

(a) Do the mathematical equations for all graphites of interest to PBMR and the graphites used in existing and past reactors have the same analytic form with regard to temperature and fluence? The figure on page 19 of Reference 3 implies that there is significant material to material variability with regard to turnaround and the fluence level at the material exhaustion limit. What mathematical expressions are used to characterize the material behavior of the relevant graphites? How are the coefficients, which distinguish the behaviors between particular graphites, estimated?

(b) In computing the predicted state at end of service life based on the mathematical models, the fluence is a monotonic function of time while the temperature is not. How will the temperature be quantified so as to represent the correct damage contribution? How will the uncertainty in the prediction be computed? How will the distinction between fixed and random effects be made?

(c) How are the effects of cyclic fatigue, such as due to thermal striping in the plenums, accounted for?

(d) What is the basis for assuming that in the temperature and fluence of interest, the graphite behavior is consistent? What uncertainties are included in this assumption? How are the uncertainties accounted for?

1.2.16. On page 15 of Attachment to Reference 2, what is the temperature range for which the fluence has been extrapolated? Please provide justification for extrapolating the data.

1.2.17. On page 17 of Attachment to Reference 2, what is the basis for assuming that the Poisson's ratio is independent of material direction and irradiation when there is evidence which suggests otherwise?

1.2.18. The expression on page 18 of Attachment to Reference 2 seems to represent the cumulative creep strain of primary and secondary creep.

(a) The influence of creep *rate* seems to have been accounted for by the "creep modulus" factor. How are the modulus "structure term", s, and the "pinning term", p, determined for primary and secondary creep?

(b) Is there a tertiary creep term which would be applicable under PBMR conditions?

(c) What is the limiting creep level (irradiation and thermal) for the PMBR design? Does it vary with temperature and fluence?

(d) Is stress rupture possible due to mechanisms other than creep under PBMR environment?

(e) Under what conditions does material exhaustion supersede the creep limit?

1.2.19. On page 19 of Attachment to Reference 2, graphite is identified correctly as a brittle material, but there is no discussion of the use of fracture mechanics in the design criteria. Please address this issue, especially in considering estimation of expected lifetime of graphite components under PBMR environment.

1.2.20. On page 19 of Attachment to Reference 2, regarding graphite performance assessment, please provide additional information on the conversion of strength data obtained between different test methods, such as bend, tension, etc. State the minimum number of specimens required to obtain convergent values of Weibull modulus.

1.2.21. On page 20 of Attachment to Reference 2, in the table which relates the safety factor to failure probability, are these data independent of temperature and fluence? What are the effects of gaseous and particulate wear, erosion and corrosion on these values?

1.2.22. In discussing failure criterion on page 22 of Attachment to Reference 2, since fast neutron irradiation rapidly increases the strength of graphite due to pinning of the dislocations in the basal planes for PBMR conditions which is a more limiting failure criterion, one based on unirradiated graphite properties with the standard safety factors, or one derived using the UK proposed Griffith failure criterion and taking into account irradiated graphite properties. In particular, when considered for a fixed level of confidence and in the context of a modified Weibull distribution and a probability of crack initiation of 10^{-4} as proposed in the ASME code? (SRXB)

1.2.23. The recommended failure criterion of 10^{-4} is given for PBMR graphite components give on page 22 of Attachment to Reference 2. How many effective full power years of PBMR operation does this criterion represent? Are these criteria affected by the reported premature cracking in AGRs (Nucleonics Week, January 24, 2002)?

1.2.24. Fuel pebbles are surrounded by graphite and are spherical in shape. These spherical fuel pebbles are expected to be in piles that can extend anywhere from 30 to 50 feet. Under no helium flow condition and high temperatures, these pebbles can come together in point contact, subject to high contact point stresses due to dead load from the pile above. The contact areas of the spheres can deform plastically from dead weight stresses. How is this effect taken into account? Since the effect of scaling up the reactor size will affect this behavior non-linearly, is there any study being planned to address this issue?

1.2.25. The graphite spheres and fuel spheres are continuously moving through the core cavity and the transfer channels. The repeated motion of the spheres under load could cause material loss at the surface of the spheres and the supporting graphite components. Therefore, the staff recommends that the wear/abrasion property of the graphite materials used for manufacturing components of reflectors, graphite spheres and fuel spheres be evaluated and specified in the materials or components specifications to ensure structural integrity of the components as well as to minimize the graphite dust in the reactor environment. If Exelon believes that this is not necessary, provide the basis and justification, including supporting test data.

1.2.26. What are the effects of graphite dust being potentially carried away by helium getting deposited with the formation of agglomerated condensate on piping, thus clogging the flow path?

1.2.27. On page 21 of Attachment to Reference 2 that has a table on the variation of statistical distributions for failure stress,

(a) The test population seems to be limited for Weibull modulus determination. What sample size and population are required for testing graphite to obtain reliable and repeatable Weibull modulus?

(b) Do the strength data represent room temperature strength of specimens irradiated at the indicated temperatures?

(c) What method was used for determining strength? Tension, 3-point, 4- point bend, etc? Was the test conducted according to ASTM Standard Test procedure? If it is 4-point bend, were the tests conducted with 1/4-pt. loading or 1/3-pt loading? What were the specimen sizes? What was the surface condition? (Machined, polished, etc.)

(d) Do the data represent ATR-2E or VQMB graphite?

(e) What is the factor σ_o in the table? If it is the minimum strength parameter in the 3-parameter Weibull distribution, then what are the mean strength values?

1.2.28. In the MTR program outlined on page 23 of Attachment to Reference 2, what specific nondestructive tests are planned? Why? How will the results be utilized in the design of graphite components?

1.2.29. Attachment 1 of Reference 2 addresses PBMR probabilistic design concepts for high-temperature graphite components. The discussion of performance assessment criteria provided appears to be too general and does not give enough details to explain the basis for formulating the criteria. For example, the section listed a set of loads to be considered without further elaborating key considerations which would form the basis for defining these loads. The section also refers to the use of a probability of failure of 10^{-4} figure without clear definition of the term and discussion of a rationale for the selection of the probability number which is relevant to PBMR specific design considerations. A more detailed discussion is needed on selection of specific failure probability values commensurate with acceptable risk (for example, provide justification for the proposed selection of 10^{-4} as the failure probability value for safety critical components whose failure would result in significant risk). Also discuss the allowable safety factors selected for mean applied stresses such as, safety factor of 2.5 for high-risk components, 1.5 for mean stresses in tension and compression loadings, and 1 for local bending or local tension. The selection of particular values for failure probability and safety factors for the components needs to be justified. In this regard, Exelon should provide a discussion of its risk-based or risk-informed aspect of structural design considerations applicable to the PBMR.

The white paper also seems to recommend the use of a power law to modify the strength of irradiated graphite with n=0.5 or 1 depending on the change in modulus of elasticity. Considering that there are little data at very high irradiation doses and temperatures of interest to the PBMR, the selection of particular values for failure probability and safety factors for the components need to be justified based on NRC guidance on plant risk goals. These justifications should be provided to staff for comments and guidance in the early stages of components' design and evaluation.

(a) How are uncertainties and sensitivity of the involved variables analyzed and accounted for in estimating failure probability?

(b) How is the estimate of the risk of failure of the test coupon translated into estimating the risk of failure of the graphite component under actual PBMR service conditions? How are such risk estimations affected by the varying manufacturing processes used for different graphite components?

1.2.30. On page 23 of Attachment to Reference 2. it is stated that the sleeves are designed to be replaceable to cater for a dropped rod. Please provide additional details on how a dropped rod will require a sleeve replacement.

1.2.31. On page 24 of Attachment to Reference 2, it has been stated that, "...there is sufficient data to justify graphite performance for at least 15 fpy...". The technical information presented in the white paper is insufficient to support this statement. Also, since the PBMR design is planned for greater than 15 fpy, provide justification regarding the bases for such design and replacement time criteria.

1.2.32. On page 24 and 25 of Attachment to Reference 2, provide rationale for the time of replacement of graphite core blocks and reflectors. What is "part way through life"?

1.2.33. On page 25 of Attachment to Reference 2 the statement is made "Suitable Nuclear Grade Graphites can be determined by appropriate choice of manufacturing process parameters". Please provide adequate support for this determination in light of the fact that there is no data beyond "turnaround" for PBMR conditions?

1.2.34. Insufficient gas mixing at the outlet plenum can lead to high temperature differentials and resulting distortion of graphite components. Describe how design is considering sufficient mixing to avoid temperature differential distortions and cracking due to thermal fatigue in graphite components.

1.2.35. The information provided on graphite components is generally lacking with respect to detailed analysis of potential operational hazards of PBMR. Specifically, there is no categorization of the specific properties and discussions relevant to the operational safety of the PBMR. For example, if the option of replacement of the inner reflectors is pursued, are there scenarios in which a reflector brick can be dropped? If yes, what damage if any can result and how will the graphite brick be retrieved?

1.2.36. Small absorber sphere (SAS) falling freely (under gravity) into the holes in the side reflectors have no shock absorber. Spheres are falling on the order of 20 meters. What is the effect on the spheres and on the side or bottom reflector blocks from this impact?

1.2.37. On page 23 of Attachment to Reference 2, what are the typical parameter ranges for material exhaustion to occur? How are design and operational safety of the PBMR ensured from a consideration of these property ranges? What defense-in-depth measures are available for mitigation of this hazard?

1.2.38. Information has not been provided regarding detailed considerations of potential air/water ingress, such as the effects on degradation of graphite components including oxidation, strength, creep and fatigue. Such information would be required for a thorough evaluation of the proposed design.

1.2.39. Virtually no information has been provided of the behavior of graphite under expected off-normal and hypothetical accident conditions. Please analyze potential hazards under these conditions and present information on the stresses, temperatures, and atmospheric contamination (purity of helium compromised) and the residence time under such conditions. Provide information on how performance reliability of graphite components in ensuring adequate safety under these conditions would be assured.

1.2.40. In the preliminary design, the cylindrical carbon reflector structure is proposed to be radially supported by “brackets” from the metal core barrel. Creep and fracturing of the carbon could lead to load distribution away from the brackets. Has there been any consideration given to such a phenomenon?

1.2.41. Based on the material presented on high-temperature reactor internals in the presentation material, a list of the mechanical and thermal transient conditions that the graphite structures will experience over the life of the plant should be provided.

1.2.42. In the proposed design of the core internal structure, the ceramic reflector structure consists of high quality graphite blocks that have holes in them for control rods and it is necessary to retain alignment through vertically arranged blocks. These blocks are supported vertically via a dowel system, and azimuthally via a radial keying system. Given that the graphite blocks will be subjected to high radiation and high temperature, the blocks may experience creep and loss of fracture toughness due to possible cracking. As a consequence, the seismic adequacy of these structures needs to be demonstrated through a feasibility study to establish stress and deformation limits. Furthermore, these structures are considerably taller than any existing design; consequently, the nonlinear response during an earthquake (in both the horizontal and the vertical directions), the effects of forced, or lack of, helium flow, and any flow induced vibratory loads could potentially all combine to contribute to detrimental consequences. The evaluation of structural integrity under these conditions requires that substantial margins be established to assure functionality, given the uncertainty in graphite properties under these conditions. Provide a plan of action to ensure that a future design will be acceptable for above considerations.

1.2.43. Page 19 of Attachment 1 to Reference 2 states that graphite is stronger in compression than in tension.

(a) Provide justification why the proposed factors of safety for tension should not be greater than for compression.

(b) The same page states that in the UK, it has been the practice to carry out failure tests on full size components in all the predicted loading modes. Indicate if PBMR will carry out such tests. If not, provide justification for not performing such tests.

1.2.44. Page 22 of Attachment 1 to Reference 2 states that computer programs MARC or ABAQUS will be used for visco-elastic finite element analysis. Attachment 2 of Reference 5 does not list these programs.

(a) Indicate the reason for the discrepancy.

(b) These programs will be used to perform best estimate calculations and performance evaluations in accordance with a German approach. Provide a discussion of this design aspect.

(c) The same page states that specially written material models will be written to be used with these programs, and these will be validated against other codes developed in the US, UK or Germany. How will these be validated against actual test data?

1.2.45. It is stated in Attachment 1 to Reference 2 that finite element codes will be used to model the linear visco-elastic behavior of the graphite material without elaborating on the basis for the visco-elastic material assumption and its related material constitutive laws. Discussion of adequate bench-marking of finite element codes to be used for PBMR specific analyses should be provided. The Attachment does not provide a discussion of the potential need for additional experimental or testing efforts to enhance the material data base and support the formulation of the performance assessment criteria specific for the PBMR. Exelon is requested to provide a brief discussion to address this concern and a more elaborate justification in the application.

White Paper No. 2:
Control of Chemical Attack in the PBMR

REQUEST FOR ADDITIONAL INFORMATION
PEBBLE BED MODULAR REACTOR

CONTROL OF CHEMICAL ATTACK

References:

2. Letter from K. F. Borton, Exelon, to the Document Control Desk, “Documents Supporting the October 25, 2001, Pre-Application Meeting Regarding the Pebble Bed Modular Reactor (PBMR),” dated October 23, 2001, with the following attachments:

   Attachment 1, titled “Graphite Presentation to USNRC in Support of PBMR Pre-Application Activities.”

   Attachment 2, titled, “Control of Chemical Attack in the PBMR Presentation to USNRC in support of PBMR Pre-Application Activities.”

6. Letter from K.F. Borton, Exelon, to the Document Control Desk, “Information Regarding Materials and Design Codes to be Used for the RPV and Connecting Piping for the PBMR,” dated December 17, 2001, with an attached PBMR document, “RPV and Connecting Piping - White Paper,” Rev. 1, dated December 5, 2001.

Discussion and Questions Related to System Analysis

System Characteristics With Potential to Impact Air Ingress

It is argued that the PBMR reactor is protected from air ingress by the so-called "diving bell" characteristic. Helium is lighter than air and the flowpaths to the core enter the reactor vessel at the bottom; consequently, the helium will tend to stay inside the reactor vessel, preventing air ingress. The argument continues that air ingress to the graphite core via natural circulation is delayed until sufficient air diffuses into the core. Discussions of this phenomena generally refer to a condition where the geometry is like an inverted manometer with the helium, which is significantly lighter than air, trapped in the upper part and air diffusing in from below.

The attached figure shows relative system elevations (not necessarily to scale) and locations of various heat sinks (coolers) of the PBMR piping. Note, the actual piping of the PBMR is pipe within a pipe (concentric) with hot gas on the inside and cooler gas on the outside. The attached figure was an attempt to lay this out in a linear fashion to better visualize the flow path and elevations.

A potential concern is that differential heating and cooling, perhaps combined with a "loop seal" formed by dense air in a low point in the piping could create a mechanism for induced flow of air into the core. Unexpected behavior of "loop seals", relatively simple piping configuration problems, and unanalyzed temperature differences had been major factors in the TMI accident, the Brown's Ferry partial scram failure events, and other risk significant events. Given the limited experience with the gas flow characteristics of the PBMR reactor, lessons learned from past experience could be relevant.

It is not obvious that the "diving bell" characteristic or the inverted manometer with equal pressure at the bottom openings of the hot leg and cold leg is always the correct assumption. Given the complicated piping system of inner and outer pipes, and the large elevation differences within the piping system, these factors need to be addressed.

Category 1 RAIs are considered relevant to either potential policy issues or significant safety issues or technical issues that are the focus of the PBMR pre-application review.

The following questions relate to potential air ingress scenarios:

2.1.1. What primary system rupture locations present the potential for the most air ingress to the core?

2.1.2. Following a rupture of the primary system, temperatures within the primary system could range from very high in the core to near ambient within heat exchangers (coolers). Is air ingress increased in some scenarios due to these temperature differentials?

2.1.3. Cold air collecting at a low point in the power conversion unit (PCU) and creating a "loop seal" might result in a configuration where cooling of the helium in the PCU would result in a mechanism to displace helium from the core more quickly than would be the case with simple diffusion. Have all potential scenarios related to heat sinks and "loop seals" been considered when analyzing for air ingress?

2.1.4. Have the implications of a break in the core conditioning system (CCS) piping been thoroughly considered, including the impact of the heat exchangers in the CCS?

2.1.5. What are the potential impacts on air ingress to the core of pressure changes within the citadel as a result of opening or closing rupture disks and blow out panels following a pressure boundary rupture accident? Does the elevation of the rupture disc and blow out panels impact air ingress potential?

Given a pipe break, forced circulation could introduce air into the core quickly. Once air is introduced into the core, the "diving bell" characteristic is no longer applicable. Continuation of pumping action by the system turbo-compressors following a system rupture could potentially introduce air during turbo-compressor coast-down. Also, spurious equipment operation or inappropriate operator actions could pump air through the system through inadvertent start of the start-up blower system (SBS) or CCS electrically powered blowers. Design features to prevent these scenarios may be needed.

2.1.6. The formation of a chimney effect airflow path through the reactor core has not been considered. Is this an incredible scenario, why?

Category 2 RAIs are considered relevant to safety issues or technical issues for which responses would be required at the time of or before a PBMR license application is submitted.

Questions related to potential equipment operation:

2.2.1. What is the potential for introduction of air to the core following a system rupture due to continued pumping action due to coast-down of the turbo-compressors.

2.2.2. What is the impact of inadvertent or spurious operation of pumps and valves in the power conversion unit on potential air ingress scenarios?

2.2.3. What interlocks and permissives prevent start of the startup blower system helium circulator?

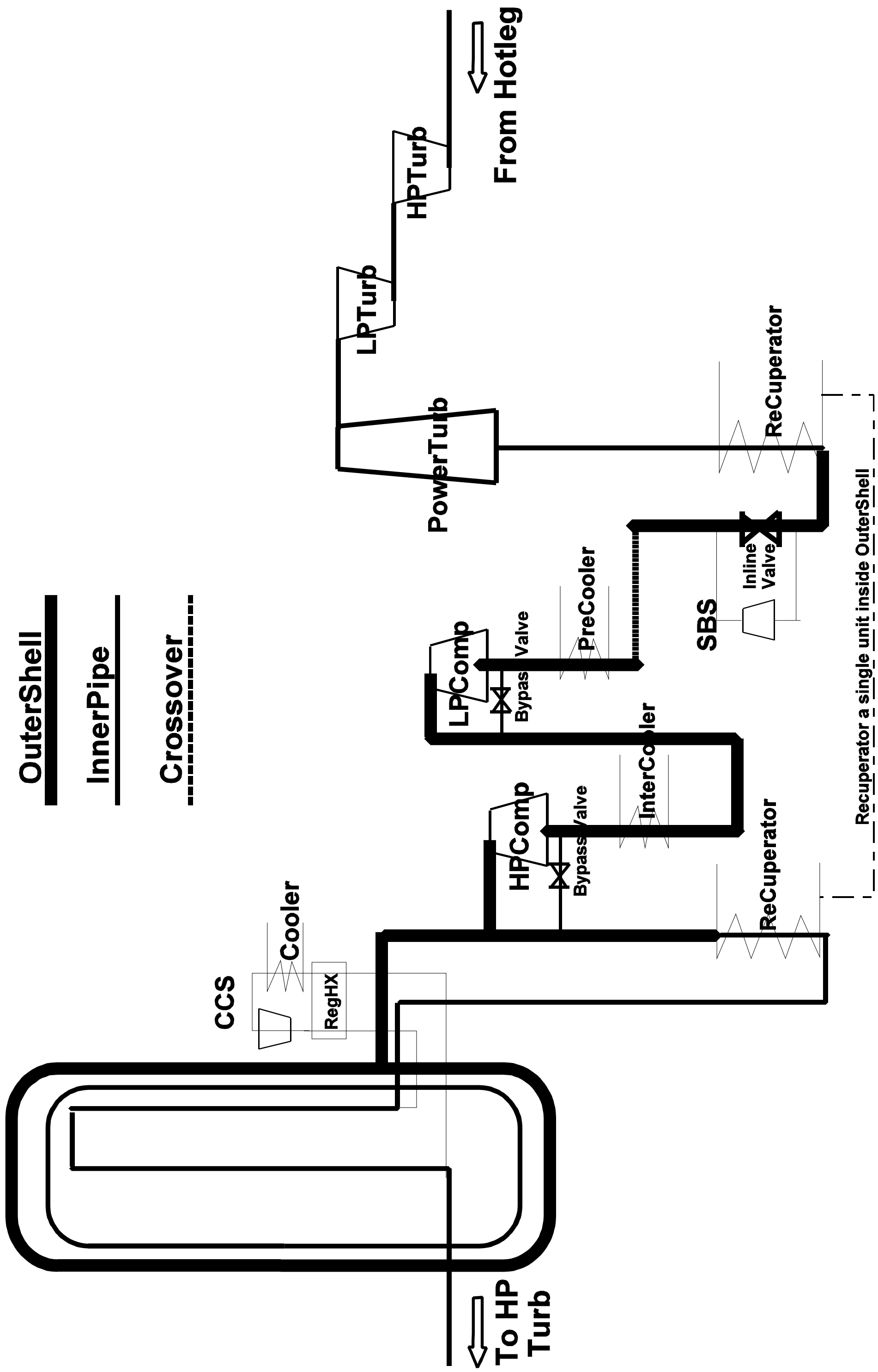
2.2.4. What interlocks and permissives prevent start of the core conditioning system helium circulator?

System Characteristics With Potential to Impact Water Ingress

Water is present in the primary system turbo-compressor coolers and the CCS heat exchanger. During normal operation, the helium pressure is higher than the water pressure in the coolers so that leakage would occur from the helium system to the water system. Also, the elevation of the turbo-compressor coolers is below the active part of the core. However, previous experience with Ft. St. Vrain water ingress events indicates that the potential for water ingress may still need to be addressed.

2.2.5. What operating limits are placed on primary system water vapor or humidity measures; and what instrumentation is available in the primary system to detect water ingress?

2.2.6. What provisions exist during low helium pressure conditions in the primary system to prevent or limit water ingress in the event of rupture of cooler or heat exchanger water line?



OuterShell
InnerPipe
Crossover
From Hotleg
HPTurb
LPTurb
PowerTurb
ReCuperator
Inline Valve
SBS
PreCooler
LPComp
Bypass Valve
InterCooler
HPComp
Bypass Valve
ReCuperator
Recuperator a single unit inside OuterShell
Cooler
CCS
RegHX
To HP Turb

Discussions and Questions Related to Materials Considerations

I. Graphite:

In the white paper provided as Attachment 2 to the October 23, 2001 letter (Reference 2), Exelon claims that the PBMR design constrains the external dose to below 100 µS/hr. This claim is supported by two arguments:

A. The PBMR design is sufficiently different to preclude the events that have lead to chemical attack of graphite in the past in operating reactors. Such as:

i) Water ingress experienced by AVR and Fort St. Vrain, and

ii) Air ingress experienced at Windscale and Chernobyl.

B. The PBMR design has taken account of the potential problem of water or air ingress through operating procedures and design features:

i) Water ingress is limited by operating procedures such as ensuring that the water circuits are not activated until the gas pressure exceeds the water pressure by a suitable margin.

ii) Air ingress in the case of small pipe breaks is limited by physical considerations and result in a maximum corrosion fraction of 0.00005 of the graphite content of the RPV; for medium breaks both physical considerations and operator procedures result in a maximum corrosion fraction of 0.002 of the graphite content of the RPV; for large breaks, beyond the design basis, the assumption that the total inventory of air in the reactor building passes through the reactor results in the oxidation of < 0.01 of the graphite content of the RPV.

The first argument (A), based on a comparison with other reactor designs, is peripheral to the issue.

The second argument (B) is based on integral values of the maximum corrosion fraction, which are difficult to assess in light of the fact that corrosion is a local phenomenon dependent on the local environment of the graphite. Moreover, the validity of the quoted estimated corrosion fractions is impossible to judge without a description and validation of the basic corrosion models which quantify the *potential* for corrosion over the parameter space of interest.

Thus, to judge the results reported in Attachment 2 to Reference 2, we request the following additional information:

2.2.7. Identify the gas/graphite reactions germane to the air and water ingress analyses of PBMR.

2.2.8. Rank them in order of importance (high, medium, low) and identify those deemed sufficiently important to include in the analyses and those that were not and why.

2.2.9. Identify (if any) catalysts for the reactions used in the analyses.

2.2.10. Give the mathematical description of the models used to compute the oxidation rates. In particular:

a) How do you model the low temperature regimes, where reaction rates are slow and the oxygen can penetrate the graphite in depth, thereby, reducing strength without an apparent change in geometry?

b) Similarly, how do you model the high temperature regimes where chemical reactivity is high and all oxygen penetrating the laminar sublayer of the gas flowing past the hot graphite surface reacts immediately at the surface. Here the geometry of the graphite changes without damaging the material in depth.

2.2.11. Discuss the validation of the correlations and the uncertainty associated with the prediction over the parameters of interest in the PBMR.

2.2.12. Give the sensitivity of the prediction to the environmental variables in the correlation.

2.2.13. What assumptions are made about the reactivity of the graphite and fuel matrix carbon? What oxidation kinetics data are used? Are the data applicable to the graphite that will be used in the PBMR? Similarly, will the new fuel matrix carbon behave in an identical fashion to the previous German material given that there will inevitably be changes in the ingredients?

2.2.14. How do these issues (3, 4, 5, 6,7) differ with respect to grades of graphite used in the PBMR design?.

2.2.15. Where in the PBMR do the highest reaction rates take place during steady state operation assuming the nominal design basis trace concentrations of reactive materials. At those locations give the graphite loss rate, the temperatures, the total graphite loss over the reactor residence time of the graphite, and the gas concentrations at those locations.

2.2.16. Where in the PBMR do the maximum reaction rates take place during the design basis transients that result in water or air ingress and were assumed in Attachment 2 of Reference 2? At those locations, as a function of time over the duration of the transient, give the graphite loss rate, the temperatures, the total graphite loss, and the gas concentrations.

2.2.17. For the beyond the design basis case in Attachment 2 of Reference 2, give as a function of core height and function of time the radial core-average values for the graphite loss rate, the fuel element temperature, the total graphite loss, and the gas concentrations.

2.2.18. Page 1of Attachment 2, Reference 2: Irrespective of quantity, what other chemicals can corrode the graphite which exist in the PBMR design?

2.2.19. Page 3 of Attachment 2, Reference 2: What sources of water exist in the PBMR and how are these sources mitigated from entering the primary system?

2.2.20. Page 6 of Attachment 2, Reference 2: What is the time scale for detecting water ingress leaks and isolating the leaks?

2.2.21. What consideration is given to the behavior of graphite dust potentially generated from the attrition of pebbles and the reflector graphite? Are graphite and carbon dust considered at all? What clean up system is in place for dust collection?

II. Metals:

2.2.22. Page 7 of attachment 2 to Reference 2 states that "All parts of the Main Power System (besides pipes with < 65 mm diameter) are designed and manufactured to the ASME III code, thus making large failures improbable." We note that failure of pipes designed and manufactured to the ASME Code Section III requirements is not improbable and in fact have occurred many times in pipes of light water reactors designed to ASME Section III. Failures have occurred by: thermal fatigue, thermal fatigue due to fluid stratification in the pipes, stress corrosion cracking, and by corrosion/erosion. What design considerations and steps is PBMR taking to avoid failures by fatigue, stress corrosion cracking and corrosion/erosion? How are the effects of sensitization, thermal aging, and of the helium environment containing impurities, including oxygen, being addressed? How are locations in different pipes with potential for stratification and for Corrosion/erosion being identified? How are locations where crevices exist being identified, and how is the potential for impurity concentrations and material degradation being evaluated? In addition discuss the effects of the environment, including the presence of oxygen, on oxidation, carburization, decarburization, and loss of strength and of fatigue and creep resistance.

<u>**Discussion and Questions Related to Human Factors**</u>

<u>Human Performance Issues</u>

The human actions discussed below are described in Attachment 2 of Reference 2 and are related to rupture panels, blow out panels, rupture disks, and pipe breaks. Relevant passages are quoted below. Specific human actions are indicated by <u>*underlined italics*</u>.

a. "**Medium size breaks**. .... A break of that size will cause the failure of rupture panels in the relief shaft designed to protect the containment.... The vents are designed to automatically close after the event and include a second <u>*manually operated closure*</u>" (p. 9). "**Larger Breaks Beyond the Design Basis**. For events beyond the design basis ... the large breaks are designed to vent through blow out panels in the top of the citadel. These will open at a higher pressure than the rupture discs for the smaller breaks. A series of trellised vents are situated in the sidewall of the upper floor and will allow blow off of the released coolant gas. They are not made to close automatically although at least partial closure is expected" (p. 10).

b. "**Mitigating Strategies**. .... For medium size leaks the vents will be closed and the HVAC restarted thus allowing cleaning of the remaining air in the building. Some surface contamination will obviously exist after the event, but <u>*entry into affected areas with the proper protective clothing*</u> is regarded as standard procedure..." (p. 10). "4. Summary. .... "Large breaks ... have acceptable risk due to limitations on air flow through the core, the low susceptibility to the coated particle to corrosion at the expected prevailing temperatures and the very high achievability of <u>*break closure by manual means*</u>" (p. 11).

c. "For large breaks the mitigation strategies are similar. Assuming that core isolation was unsuccessful, the <u>*actions of the operators will be to close the opening*</u> with a suitable material, which will be available in the module building or nearby, <u>*and/or to add inert gas to the building, citadel, and/or core*</u>" (p. 11).

<u>Information Requests</u>

2.2.23. For each of the scenarios described in Section 1 above, identify and describe the relevant human actions. What specific operator actions are required to respond to pressure boundary ruptures at different locations? Do operator actions vary depending on the location and size of the primary system rupture location? Have all the relevant functions been appropriately allocated to either equipment or to personnel? (See NUREG-0711, Chapter 4, "Element 3 - Functional Requirements Analysis and Function Allocation.")

2.2.24. For each of the scenarios described in Section 1 above answer the following questions. What information is required by the personnel? Will procedures exist for these actions? (See NUREG-0711, Chapter 9, "Element 8 - Procedure Development.") How does the person know that he/she is supposed to perform the task? How much time is available to perform the task? How does the person know that he/she has performed the task completely and satisfactorily? Will the task require donning protective clothing? Describe the ingress/egress paths taken by the personnel to accomplish the tasks. Will the task require special tools or materials? (See NUREG-0711, Chapter 5, "Element 4 - Task Analysis.")

2.2.25. What instrumentation is available to operators to determine the status of system pressure boundaries? What instrumentation is available in the primary system to detect water ingress? Where is the instrumentation located? How does the operator access the information? (See NUREG-0711, Chapter 8, "Element 7 - Human-System Interface Design.")

2.2.26. How many and what kinds of people are required to perform the tasks required to respond to the scenarios described in Section 1 above? (See NUREG-0711, Chapter 5, "Element 5 - Staffing.") What training will be required of personnel to perform the actions to respond to the scenarios? (See NUREG-0711, Chapter 10, "Element 9 - Training Development Program.")

2.2.27. What are the potential improper operator actions in response to air ingress accidents which could result in larger volumes of air ingress? Could these actions include inadvertent start of the CCS or SBS blowers? What is the potential for increased air ingress if the operator responds to the wrong scenario? What is the impact of inadvertent operation of pumps and valves in the power conversion unit on potential air ingress scenarios? (See NUREG-0711, Chapter 7, "Element 6 - Human Reliability Analysis.")

2.2.28. Describe the program that will be followed to verify that personnel can successfully perform the tasks required to respond to the scenarios? (See NUREG-0711, Chapter 11, "Element 10 - Human Factors Verification and Validation.")

2.2.29. Describe the effect of these events on the other modular reactors on site. Describe the effect of these events on the tasks and the workload of the other crew members.

<u>Sources for Supporting Human Performance Guidance</u>.

General guidance on providing information concerning human actions is provided in the following documents. The general guidance in these documents needs to be tailored to the specific actions that are the subject of this memorandum.

a. USNRC. Office of Nuclear Reactor Regulation. "Standard Review Plan." NUREG-0800. Chapter 18, "Human Factors Engineering." (April, 1966).

b. USNRC. Office of Nuclear Reactor Regulation. "NRC Information Notice 97-78: Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times." (October 23, 1997).

c. USNRC. Office of Nuclear Reactor Regulation. "Human Factors Engineering Program Review Model." NUREG-0711. (1994).

d. Higgins, J.C., and J.M. O'Hara. 2000. "Proposed Approach for Reviewing Changes to Risk-Important Human Actions." NUREG/CR-6689. Prepared by Brookhaven National Laboratory for USNRC.

e. USNRC. Office of Nuclear Reactor Research. "Draft Regulatory Guide DG-1110 (Proposed Revision 1 to Regulatory Guide 1.174) An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." (June 2001).

White Paper No. 3:
Summary of PBMR Design Codes and Standards

REQUEST FOR ADDITIONAL INFORMATION
PEBBLE BED MODULAR REACTOR

DESIGN CODES AND STANDARDS

References:

1. Slides of the July 18, 2001, presentation by Vijay Nilekani, Exelon, "PBMR Design Codes and Standards."

2. Letter from K. F. Borton, Exelon, to the Document Control Desk, "Documents Supporting the October 25, 2001, Pre-Application Meeting Regarding the Pebble Bed Modular Reactor (PBMR)," dated October 23, 2001, with the following attachments:

   Attachment 1, titled "Graphite Presentation to USNRC in Support of PBMR Pre-Application Activities."

   Attachment 2, titled, "Control of Chemical Attack in the PBMR Presentation to USNRC in support of PBMR Pre-Application Activities."

5. Letter from K.F. Borton, Exelon, to the Document Control Desk, "Summary of Pre-application Presentations Regarding the PBMR," dated October 30. 2001, with the following attachments:

   Attachment 1: Summary of the PBMR design codes and standards information presented to the NRC on July 18, 2001.

   Attachment 2: PBMR Analytical (computer) Codes Data Table presented to the NRC on August 16, 2001.

6. Letter from K.F. Borton, Exelon, to the Document Control Desk, "Information Regarding Materials and Design Codes to be Used for the RPV and Connecting Piping for the PBMR," dated December 17, 2001, with an attached PBMR document, "RPV and Connecting Piping - White Paper," Rev. 1, dated December 5, 2001.

7. Letter from K.F. Borton, Exelon, to the Document Control Desk, "Response to NRC letter dated September 26, 2001, Regarding the PBMR Technical Information Availability," dated November 15, 2001.

8. Exelon Presentation, "Process for Selection of Licensing Basis Events for the PBMR," Fred Silady, July 17, 2001.

Category 1 RAIs are considered relevant to either potential policy issues or significant safety issues or technical issues that are the focus of the PBMR pre-application review.

<u>Question Related to Design Code Applicability and Quality Control Assurance</u>

3.1.1. Exelon has indicated that they may request an exemption to In-Service Inspection (ISI) of safety components because of the limited accessibility of components in their proposed design (Reference 5, attachment 1). Regulation requires that important components must be designed so that they are accessible for inservice inspection. Alternatively, continuous online monitoring of such components could be considered and planned for in the design. In addition, continuous monitoring of structural integrity should be considered in light of the long outage cycle of 6 years for the PBMR. The NRC staff is aware that Exelon has indicated details for the ISI plan will be available by September 2002 (Reference 7).

Category 2 RAIs are considered relevant to safety issues or technical issues for which responses would be required at the time of or before a PBMR license application is submitted.

<u>Discussion and Questions Related to Design of Reactor Vessel, Metallic Core Barrel, and Reactor Internal and Support Structures</u>

3.2.1. Attachment 1 to Reference 5 contains a summary of the PBMR Design Codes and Standards presentation of July 18, 2001. The first paragraph at the top of page 3 of the summary states, “The RPV Internal Core Barrel will be designed and fabricated in accordance with the ASME Code, Section III, Division 1, Subsection NG, *Core Support Structures,* 1998. ASME approved Code Case N-201 (1994), which permits temperatures up to 816 °C, (limited to pressures and durations) is used to address the higher temperatures experienced during a PLOFC event and a DLOFC event (720 °C).”

The 1998 Edition of the ASME Code, Section III, Division 1, Subsection NG, *Core Support Structures*, establishes the rules for materials, design, fabrication, and preparation of reports required in the manufacture and installation of core support structures. Subsection NG defines core support structures as those structures or parts of structures which are designed to provide direct support or restraint of the core within the reactor pressure vessel. Internal structures are defined as all structures within the reactor pressure vessel other than core support structures, fuel, blanket assemblies, control assemblies and instrumentation. The rules of Subsection NG apply to internal structures, when stipulated by the Certificate Holder manufacturing core supports. Subsection NG defines the loadings that shall be taken into account in the design of the core support structures, including seismic and the rupture of the main coolant pipe.

Based on the Subsection NG rules, the PBMR bottom and side graphite reflector structures are also core support structures, because they directly support or restrain the core. The Exelon letter only identified the core barrel as being designed and fabricated in accordance with Subsection NG.

Exelon also proposes to use Code Case N-201 as an alternative to the rules of Subsection NG for the design of the core barrel to accommodate service at elevated temperature during PLOFC and DLOFC events. Code Case N-201 covers the rules for construction of Section III, Subsection NG, *Core Support Structures*, for elevated temperature service. Part A of the Code Case extends the rules for restricted service at elevated temperature without explicit consideration of creep and stress-rupture. Part B of the Code Case altered the rules for service at elevated temperature to account for creep and stress rupture effects.

1. Provide clarification regarding the extent to which the rules of Subsection NG will be applied to the PBMR reactor vessel internals.

2. Identify core support structures and internal structures consistent with Subsection NG definitions, and describe how these rules will or will not be applied to each category of structures.

3. Subsection NG does not specifically address or prohibit the design and fabrication of graphite core support structures, nor does it provide the allowable physical properties for use in the design of structures with this material. If Exelon intends to use Subsection NG rules for the design of all the metallic defined core support structures, provide any alternative design rules or criteria for the graphite core support structures.

4. Exelon proposes to use Subsection NG in conjunction with Code Case N-201 to accommodate service at elevated temperature during PLOFC and DLOFC events.

   a) Identify the specific portions of the code case that will be used,

   b) Provide information with regard to: 1) the level, extent and duration of the elevated temperature and stress, 2) the structure location of the application, 3) consideration of environmental effects including water vapor and oxygen impurities in the helium gas, and 4) justification for the acceptability of any alternative design rules.

<u>Discussion and Questions Related to Civil, Structural and Seismic Analysis of the Containment Building</u>

3.2.2. Exelon stated in Reference 1 and in Reference 5 that they plan to reference the American Concrete Institute (ACI), ACI 349-97, "Code Requirements for Nuclear Safety Related Concrete Structures," and the American Institute of Steel Construction (AISC), AISC N690-1994, "American National Standard Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities" for the design and analysis of the PBMR civil structures. Exelon also listed a Department of Energy standard, DOE 1020-94, "Natural Phenomena Hazards Design and Evaluation Criteria for DOE Facilities," and a South African code (SABS 0160-1989) entitled "South African Standard, Code of Practice: General Procedures and Loads to be Adopted in the Design of Buildings," as documents they intend to reference in future licensing actions.

At present, the NRC staff has not endorsed or reviewed the DOE standard 1020-94 or the referenced South African code. Also, the ACI 349 code and the AISC N690 standard provide minimal guidance on acceptance criteria, design and use of concrete or steel structural modules in nuclear power plants. Furthermore, the ACI 349 code may not be applicable to the pressure retaining structures of the PBMR civil structures.

3.2.3. Exelon should review NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, 1987, Sections 3.8.1, “Concrete Containments;” 3.8.2, “Steel Containments;” and 3.8.4, “Other Seismic Category I Structures.” Provide a short description of the provisions in the referenced standards ASCE 7-98, US DOE Std.1020-94, and SABS 0160-1989, you plan to use for Civil-Structural design and construction. Provide the NRC staff with a comparison of the South African Code (SABS 0160 -1989) referenced in the PBMR Design Codes and Standards white paper with existing NRC guidance for the design and analysis of civil structures provided in NUREG-0800, Sections 3.8.1, 3.8.2, and 3.8.4.

3.2.4. On page 1 of attachment 1 to Reference 5 Exelon listed ACI 349-90 as the primary civil code they plan to use. In the PBMR Design Codes and Standards handout material provided at the July 18, 2001, meeting (Reference 1), Exelon stated that the ACI 349-97 code will be used. Exelon should clarify to the staff what version of the ACI 349 code they plan to use for the design of the PBMR civil structures.

3.2.5. Slides 5 through 7 of Reference 1 provide a list of civil, structural and seismic related codes and standards to be used for PBMR design. The list as presented appears to be incomplete. For reinforced or prestressed PBMR concrete elements which provide one or more intended PBMR containment functions, the ASME Code, Section III, Div. 2, “Code for Concrete Reactor Vessels and Containments” should apply. Explain the reason for the apparent exclusion of the noted ASME standard.

3.2.6. For the design of safety related steel structures inside and outside the envelope, the applicable standard would be ANSI/AISC N690-84 as endorsed in NUREG-0800, Appendix F of the Standard Review Plan (SRP) Section 3.8.4 (1996 Draft), or ANSI/AISC N690-94 with Supplement 1 (to be published in 2002). Note: Supplement 1 to ANSI/AISC N690-1994 incorporates the staff position in Appendix F of SRP 3.8.4 (1996 Draft), in addition to other safety relevant changes. Exelon will need to state that the design of safety related steel structures inside and outside the envelope will be designed to these standards. What type of structures (e.g. steel cages, concrete filled steel modules, precast modules) does Exelon plan to use in the PBMR modules and to what extent do the structures comply with or deviate from ACI 349 and AISC N690 existing guidance?

3.2.7. Exelon will need to address non destructive examination (NDE) of the modules and how quality will be assured during fabrication, transportation, and construction of the PBMR modules including roles and responsibilities of Exelon, the fabricator and the on-site constructor.

3.2.8. What will be the operating temperature in the containment structure? The staff is concerned that if the temperature is substantially higher than the limits specified in the ACI Code (150°F for long term, 200°F for short term and 300°F for local effects), the design should include a consideration for the reduction of concrete strength due to de-watering effect.

3.2.9. Exelon should review NUREG/CR-6486, “Assessment of Modular Construction for Safety-Related Structures of Advanced Nuclear Power Plants,” 1997. NUREG/CR-6486, provides an assessment of modular construction technical issues for the Westinghouse AP600, Combustion Engineering System 80+, and the General Electric Advanced Boiling Water Reactor.

3.2.10. Exelon should review NUREG/CR-6358, “Assessment of U. S. Industry Codes and Standards for Application to Advanced Nuclear Power Reactors,” 1995. NUREG/CR-6358, provides an assessment of industry consensus structural codes and standards prior to 1995.

3.2.11. With respect to industry codes and standards, NRC’s Standard Review Plan (NUREG-0800) Section 3.7 through 9 and the associated regulatory guides provide a comprehensive discussion of acceptable methods. Standards and Codes that are not discussed explicitly, can be used only when they are demonstrated by the applicant to be equivalent to those referenced in the Standard Review Plan. This demonstration is necessary for the NRC staff to draw its safety conclusion.

3.2.12.* The current design may not be applicable to a potential final license application for a final design. As the risk-informed design process of safety-significant components continues, seeking NRC staff review and comments on the intermediate design steps would be beneficial for expedient reviews of final designs.

3.2.13. In Reference 1, Exelon stated that US NRC guidance (NUREG-0800, RG 1.60, RG 1.122, RG 1.165, etc.), US DOE guidance (DOE-STD-1020-94, etc.), and IAEA guidance are to be used for the seismic analyses of PBMR. It is the staff’s understanding that the DOE guidance and IAEA guidance are much less stringent than those of the NRC guidance. If these two guidance criteria are to be used for licensing the PBMR in the US, Exelon should (1) compare these two sets of criteria with NRC guidelines, (2) identify differences between these two sets of criteria and NRC guidelines, (3) evaluate the significance of the differences identified, and (4) justify the applicability of these criteria for the PBMR design.

Discussion and Questions Related to Quality Group Designations

3.2.14. Briefly discuss the extent to which the design criteria of the plant structures, systems, and components important to safety meet the NRC “General Design Criteria for Nuclear Power Plants” specified in Appendix A to 10 CFR Part 50 for LWRs. For each criterion, a summary should be provided to show how the principal design features meet the criterion. Any exceptions to criteria should be identified and the justification for each exception should be discussed. From Regulatory Guide 1.70, Revision 3, Section 3.1, “Conformance with NRC General Design Criteria”, it is expected that some criteria unique to water cooled nuclear power plants may not be directly applicable to gas cooled nuclear power plants, such as the PBMR.

3.2.15. The described PBMR is stated to have no significant accident sequences involving loss of coolant or single failures of equipment whose function is to provide a heat removal function. However, a safety analysis of sufficient detail which demonstrates the necessary safety functions of components and systems is not provided. Without a complete understanding of these important functions under the most limiting combinations of conditions, an adequate assessment of the necessary quality group designations for components and systems cannot be made. In accordance with GDC 1, provide a safety analysis of sufficient detail to provide a determination of the proper quality group designations.

3.2.16. In accordance with GDC 2 of 10 CRF 50 Appendix A, provide an analysis of the quality group and seismic category of proposed PBMR structures, systems, and components to demonstrate their capability to withstand natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

Discussion and Questions Related to Design Code Applicability and Quality Control Assurance

3.2.17. Exelon should establish a quality assurance program for design, procurement, construction and operation of the proposed nuclear power plant that is consistent with the design bases specified in the license application. The program will have to meet the applicable requirements of Appendix B to 10 CFR Part 50 as determined for the regulatory frame work applied to the PBMR.

Discussion and Questions Related to Electrical and Instrumentation & Control

3.2.18. IEEE Standard 603, 1991 edition vs IEEE Standard 603, 1998 edition.

10 CFR 50.55a(h) refers to IEEE Standard 603, 1991 edition. The criteria in the 1991 edition are, therefore, requirements that must be met. Exelon stated that the PBMR project is working to the 1998 edition. It is the staff's understanding that the purpose of 1998 revision is to clarify the application of this standard to computer-based safety systems and to advanced nuclear power generating station designs. The 1998 revision provides guidance for the treatment of electromagnetic interference (EMI) and radio-frequency interference (RFI), clarifies definitions and updates references. The staff has reviewed section by section on both IEEE Standard 603, 1991 edition and 1998 edition. The wording of criteria for safety systems are essentially the same. The staff considers that the 1998 edition is to clarify the existing criteria. There is no change to the existing criteria between 1991 edition and 1998 edition. Therefore, a design that meets the IEEE Standard 603, 1998 edition should satisfy 10 CFR 50.55a(h) requirements. Nevertheless, since the 1991 edition contains the requirement of 10 CFR 50.55a(h), Exelon should map the requirement of the 1998 edition to the requirement of the 1991 edition to show compliance with the regulation. An alternate option would be to petition for rule making to revise 10 CFR 50.55a(h).

3.2.19. For other standards referenced that do not have a current endorsement, NRC will address them in a similar fashion to IEEE Standard 603. IEEE Standard is unique in that it is referenced directly in the regulations, as opposed to Regulatory Guide, however the method of review will be similar.

3.2.20. Guidance from NUREG 0800 (Standard Review Plan (SRP))

NRC issued Standard Review Plan, Chapter 7 - Instrumentation and Controls, Revision 4 in June, 1997. The new SRP Chapter 7 and Appendix 7.0-A describes the overall review process for I&C systems including digital systems. The review scope covers I&C system design and implementation, including applicable life-cycle activities. Appendix 7.1-A describes Acceptance Criteria and Guidelines, Branch Technical Positions (BTP) and regulatory guides. Appendix 7.1-C provides guidance for evaluation of conformance to IEEE Standard 603 and IEEE Standard 7-4.3.2. Since the NRC staff will be using the guidance of SRP Chapter 7 for reviewing the pre-application of PBMR design, Exelon should specifically address that the PBMR I&C design uses guidance provided in NUREG 0800 Chapter 7, Revision 4.

3.2.21. Most Light Water Reactors refueling cycles are less than 2 years. The PBMR outage cycle is 6 years. This could present several issues. For example instrument drift during such long operating period would be a concern. The instrument setpoint methodology should take that consideration. Will the licencee's application address these issues in a systematic way?

3.2.22. Exelon stated that non-nuclear-safety-related systems for equipment protection will be designed in accordance with ANSI/ISA S84.01, 1996. Non-nuclear-safety-related systems for operational control will be designed in accordance with International Electrotechnical Commission (IEC) standards.

The SRP Section 7.7, "Control Systems" provides guidance for reviewing non-safety-related I&C systems. These systems are reviewed to ensure that they conform to the acceptance criteria and guidelines, that the controlled variables can be maintained within prescribed operating ranges, and the effects of operation or failure of these systems are bounded by the accident analyses in safety analysis report.

3.2.23. The Exelon statements cover many Human Interface System review guidelines and Control Room design guidelines such as IEEE Standard 1023, "Application for Human Factor Engineering", NUREG 0711, NUREG CR5908, NUREG CR6105, and NUREG CR6146. Is Exelon aware that NUREG 0700 and NUREG 0711 are in the process of being revised and the new versions will be released shortly?

3.2.24. The electrical and instrumentation & control section does not include any discussion on the multiple modular plant designs issues. Will Exelon be requesting any exceptions to current regulations or regulatory guidance associated with multiple plant implementation of the electrical or instrumentation & control? What, if any, will be the effects of shared or cross connected systems and changes in the levels or types of redundancy?

White Paper No. 4:
RPV and Connecting Piping - White Paper

REQUEST FOR ADDITIONAL INFORMATION
PEBBLE BED MODULAR REACTOR

DESIGN CODES AND STANDARDS

References:

1. Slides of the July 18, 2001, presentation by Vijay Nilekani, Exelon, “PBMR Design Codes and Standards.”

2. Letter from K. F. Borton, Exelon, to the Document Control Desk, “Documents Supporting the October 25, 2001, Pre-Application Meeting Regarding the Pebble Bed Modular Reactor (PBMR),” dated October 23, 2001, with the following attachments:

   Attachment 1, titled “Graphite Presentation to USNRC in Support of PBMR Pre-Application Activities.”

   Attachment 2, titled, “Control of Chemical Attack in the PBMR Presentation to USNRC in support of PBMR Pre-Application Activities.”

5. Letter from K.F. Borton, Exelon, to the Document Control Desk, “Summary of Pre-application Presentations Regarding the PBMR,” dated October 30. 2001, with the following attachments:

   Attachment 1: Summary of the PBMR design codes and standards information presented to the NRC on July 18, 2001.

   Attachment 2: PBMR Analytical (computer) Codes Data Table presented to the NRC on August 16, 2001.

6. Letter from K.F. Borton, Exelon, to the Document Control Desk, “Information Regarding Materials and Design Codes to be Used for the RPV and Connecting Piping for the PBMR,” dated December 17, 2001, with an attached PBMR document, “RPV and Connecting Piping - White Paper,” Rev. 1, dated December 5, 2001.

7. Letter from K.F. Borton, Exelon, to the Document Control Desk, “Response to NRC letter dated September 26, 2001, Regarding the PBMR Technical Information Availability,” dated November 15, 2001.

8. Exelon Presentation, “Process for Selection of Licensing Basis Events for the PBMR,” Fred Silady, July 17, 2001.

Category 1 RAIs are considered relevant to either potential policy issues or significant safety issues or technical issues that are the focus of the PBMR pre-application review.

<u>Discussion and Question Related to Civil, Structural and Seismic Analysis of the Containment Building</u>

4.1.1. As a result of defining the design basis accidents in Reference 8, if the structural envelope surrounding the reactor pressure vessel and connecting piping (Attachment 1 of Reference 5, and Reference 6) has to withstand pressures resulting from low probability design events in order to contain the radioactive material, the applicable code for designing and constructing the envelope would be either ASME Section III, Division 1, Subsection NE, "Rules for Class MC Containment Vessels," or ASME Section III, Division 2, Subsection CC, "Code for Concrete Reactor Vessels and Containments." The relevant NRC's regulatory documents are NUREG-0800, "Standard Review plan," Sections 3.8.1, 3.8.2, and Regulatory Guides 1.136 and 1.57. If the envelope need not be designed as a pressure retaining envelope, the concrete envelope can be designed using ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures," as augmented by Revision 2 of Regulatory Guide 1.142. In both cases, the envelope needs to be designed for the postulated external missiles (including aircraft impact). Does Exelon consider the Citadel to be a pressure or non-pressure retaining structure? Please include justification. State the applicable code to which the structural envelope will be designed.

<u>Discussion and Questions Related to Design of Piping and Pressure Vessels</u>

4.1.2. Design codes and material selection is addressed to some extent for RPV and connecting piping in the Exelon white papers, however design codes and material selection do not appear to be addressed for the hot piping and high temperature components of the PCU, for example, failure of turbine blades could produce projectiles which could impact the integrity of other components important to safety. Reference 6 indicates that Incoloy 800H, hot piping at the core outlet is at 900°C for 300,000 hrs, however the stress levels are not specified. What are the stress levels? Also, creep and fatigue life of hot piping and other high temperature components is not addressed. Exelon must specify the codes and standards to be used for creep and fatigue design of the hot piping and other high temperature components in the PBMR helium environment with consideration of impurities present in the gas stream, such as water vapor and oxygen.

4.1.3. Provide the basis and justification regarding designing the connecting piping as pressure vessels and not as pipes (Reference 6). Compare the differences in requirements pertaining to welding, fabrication, installation, pre-service inspection, in-service inspections and any others. Pressure vessels designed for the same pressure have much thicker walls than pipes. The thicker wall of the pressure vessel allows for crack detection before failure. The connecting piping can be considered a vessel if the wall thickness is sufficient. For typical piping wall thicknesses, cracks can propagate around much of the circumference and penetrate considerably through the wall before detection leading to a potential for DLOFC or PLOFC. Compare the failure probability of typical reactor pressure vessels to that for piping.

Category 2 RAIs are considered relevant to safety issues or technical issues for which responses would be required at the time of or before a PBMR license application is submitted.

<u>Discussion and Questions Related to Material Properties of Metallic Materials of Construction at High Temperatures</u>

4.2.1. Exelon plans to use ASME Code Cases N-499 and N-201 for the design of reactor pressure vessel (RPV) and the core support component of the core barrel, respectively for limited service at elevated temperature. NRC has not endorsed the use of these two code cases for any design applications. For the NRC to perform a proper review of these two code cases for design applications in the PBMR, Exelon should provide the basis and justification to demonstrate that it is conservative to design and fabricate the referenced components based on the requirements provided in the subject code cases. In particular, Exelon should address the effects of the bounding environmental condition, including irradiation and impurities in the helium such as oxygen, in addition to the effects of temperature and time on the mechanical and physical property values of the materials that are provided in the subject code cases. Test data should also be provided to support the discussion and conclusion. If relevant test data are not available, Exelon may propose a test program to verify or validate the assumed property values that are used for component design.

4.2.2. Exelon plans to use ASME Code Case N-499 for the design of reactor pressure vessel so that the selected vessel material will be allowed to operate for a limited period at elevated temperature. Code Case N-499 allows the selected vessel material to operate for limited periods in the temperature range of 371°C to 538°C. Exelon also identified that during the off-normal operating condition, the temperature of the reactor pressure vessel could be as high as 476°C. Since this temperature is only about 11% below the maximum temperature allowed by Code Case N-499, Exelon should provide detailed discussion regarding how the maximum temperature during off normal operating condition was calculated and the magnitude of uncertainties associated with the calculated results. Also discuss the results of sensitivity studies, if performed. The staff understands that the reported maximum temperatures are based on the thermo-hydraulic simulations of a “Reactor C” configuration and the PBMR design is still under development. Exelon needs to develop a plan to verify or validate the calculated results through experiments or demonstration plants when the design is finalized.

4.2.3. A list should be provided of all materials used for the reactor pressure vessel and its appurtenances, core support structures, primary pressure boundary, connecting piping, and other components important to safety and the applicable material specifications, design stress and time at temperature and other environmental conditions. The identification of the grade or type and condition of the materials to be placed in service would also be required. If the code approved material specifications for the intended applications are not available, relevant material specifications should be developed following the format of American Society for Testing and Materials (ASTM) specifications. The subject specifications should be supported by the data and information as identified in ASME Code, Section III, Appendix IV, for approval of the new materials. Additional information unique to the application in the PBMR environment and condition should also be provided.

Discussion and Questions Related to Design of Reactor Vessel, Metallic Core Barrel, and Reactor Internal and Support Structures

4.2.4. Section 3 of Reference 6 indicates that the PBMR primary pressure boundary consists of the Reactor Pressure Vessel (RPV) and the Power Conversion Unit (PCU) pressure boundary. The PCU pressure boundary consists of a manifold vessel which is connected to a number of component vessels. Reference 6 indicates that the RPV will be designed to ASME Section III, Class 1 requirements. Indicate the design codes that will be used for the remaining component vessels that are part of the PBMR primary pressure boundary and discuss the basis for the selection for these design codes.

4.2.5. Section 5.3 of Reference 6 indicates that the connecting piping will be designed to ASME Code Class 2 requirements. The white paper further indicates that the material selected (SA 508 Grade 3, Class 2 and SA 533 Type B, Class 2) has a higher strength level than the Class 1 specification and therefore results in lower component masses due to reduced section thicknesses. Provide a comparison of the Class 1 and Class 2 ASME Code criteria for the above materials which illustrates how the use of ASME Class 2 criteria will result in components with reduced section thicknesses.

4.2.6. Detailed drawings should be provided showing typical dimensions and materials of the components of the core barrel and supports, the metallic core lateral restraints, the core graphite components of the top, side and bottom reflectors, and the graphite core supports. For example, the material composition of the dowels that hold the bottom reflector blocks in place relative to each other and relative to the bottom metallic structure would need to be specified.

4.2.7. Exelon should provide the design assumptions of loads, load combinations and material behavior in the PBMR helium environment with impurities including oxygen (creep, fatigue, fracture, loss of strength, loss of material), including material properties, constitutive relationships, acceptance criteria for allowable limits on stresses and strains and modeling assumptions for the structural design of reactor internals.

4.2.8. Sections 4.2, 4.3 and 4.4 of Reference 6 provide maximum temperature and duration parameters for both normal and off-normal operating conditions; however, no data related to pressure range parameters are included in these sections. Explain why pertinent pressure range parameters are not provided or needed in defining the applicable PBMR environments. That is, describe how Exelon is going to obtain the creep, fatigue, fracture, etc, material properties. Do these material properties take into consideration effects of irradiation, temperature, stress, time, the coolant gas of helium plus impurities including oxygen?

Discussion and Questions Related to Civil, Structural and Seismic Analysis of the Containment Building

4.2.9. Page 3 of Reference 1, titled “PBMR Integrated Design Process” should consider inclusion of potentially needed test based design verification tasks in the process chart.

4.2.10. References 1, 2 and 6 did not provide an overview of Exelon's approach for carrying out a deterministic analysis of an integrated PBMR model in order to ensure its seismic integrity. Because of the highly non-linear configurations and aging effects (due to irradiation, time, temperature, stress, impurities in the environment, etc) of the PBMR components including its graphite core, reflectors and supports as well as the non-ductile nature of the graphite, Exelon should provide a comprehensive descriptions of the PBMR seismic/structural analysis model, the basis for defining parameters used for the model, verification of the parameter used and applicable load combinations. The assumptions and limitations of finite element analysis codes to be used should also be discussed to demonstrate their applicability to the PBMR design. Additionally, because of the first-of-a-kind nature of the PBMR design configuration, the need for a test based verification of the PBMR's seismic response to pertinent loads in order to demonstrate the seismic design adequacy should be evaluated by Exelon.

4.2.11. From the meeting discussions and review of Reference 6, the staff finds that the modular concept of the PBMR is such that the plant structures, auxiliary systems, and supporting infrastructure can be constructed and operated on most sites and under the majority of environmental conditions anticipated around the world, i.e., standard design concept. Clearly define all the bounding parameters similar to those used for other standard plants such as ABWR, System 80+, and AP600. As a minimum, these parameters should include maximum ground water level, maximum flood level, bounding precipitation (rain, snow/ice, etc.), air temperatures (summer and winter), tornado (wind speed, maximum pressure differential, etc.), tornado missile spectra, average allowable static soil bearing capacity, lateral variability of soil layers, shear wave velocities (a spectrum of shear wave velocities of site conditions to envelop the majority of sites in the US), liquefaction potential of soil foundation, bounding earthquake ground motions (ground motion response spectra and/or ground motion time histories), fault displacement potential, etc.

<u>Discussion and Questions Related to Design of Piping and Pressure Vessels</u>

4.2.12. Section 5.3 of Reference 6 indicates that the connecting piping will be designed as a pressure vessel in accordance with NC 3300. Discuss the following aspects of the design of the connecting piping:

a. Indicate whether the connecting piping will contain any elbows, bends, tees or other standard piping fittings. If the piping will contain these components, discuss how they will be evaluated using the NC 3300 rules.

b. Discuss how the restraint of free end expansion of the piping will be evaluated using the rules of NC 3300.

4.2.13. Section 6.3 of Reference 6 indicates that a fatigue analysis of the connecting piping will be done in accordance with the rules of ASME III, Subsection NC - as applicable to vessels. Discuss the specific fatigue rules that will be used for this evaluation. Discuss what fatigue design curve will be used and how the effects of the environment including the presence of oxygen, strain rate, temperature, etc will be taken into account. Discuss any thermal or mechanical loads that could result in significant fatigue damage in the connecting piping.

4.2.14. The material of the outlet plenum seal connected to the 316 SS metallic core barrel (Reference 2, attachment 1) and the codes & standards that cover the design and properties of this seal where temperatures are ~900°C need to be specified. Insufficient gas mixing at the core outlet plenum can lead to high temperature differentials and resulting distortion and thermal fatigue of components. Describe how design is considering sufficient mixing to avoid temperature differential distortions.

4.2.15. Discuss the design of the connecting piping as it relates to 10 CFR Part 50, Appendix A, *Criterion 4 - Environmental and dynamic effects design bases*.

4.2.16. In the white paper provided in the December 17, 2001 letter (Reference 6), Exelon stated that the impurities in the helium gas could lead to either carburization or internal oxidation of the materials, which in turn would affect the creep and fatigue properties of the materials. Exelon also concluded that based on the test results that the helium impurities effects are only appreciable at high temperatures(> 550° C) and, therefore, Exelon expects no significant effects on the reactor pressure vessel and pressure boundary piping. Did these tests include all major impurities expected in the PBMR, including the presence of oxygen? In addition, the staff notes that the design of the hot pipes utilizes an internal sleeve, made of Incoloy 800H. The internal sleeve is exposed to a maximum temperature of 900°C during normal operating condition. Exelon should provide a detailed discussion regarding the effects of helium impurities, including oxygen, on the creep and fatigue properties of the internal sleeve and of heat exchanging metallic components. Exelon should identify any design measures and/or monitoring/inspection programs that will be implemented to mitigate the potential degradation of material properties. The hot piping at the core inlet is exposed to a maximum temperature of 600°C during off-normal operating condition for 100,000 hours (30% of life); the effects of helium impurities, including oxygen, on the material properties should also be discussed.

4.2.17. Thermal stresses in connecting pipes that are insulated by refractory or glass wool from the hot piping, utilizing an internal sleeve made of Incoloy 800H, which is thermally insulated from the outer pressure pipe, made of 16Mo3 (Reference 6), need to be evaluated. Crevices would naturally exist in the insulation or at joints, which raises the following questions: (1) What is the effect of gas migrating between spaces, and consequently causing hot spots and thermal stresses? (2) What happens to the concentration of chemicals/impurities trapped in the crevices? (3) How often is insulation replaced? (4) What is the potential impact on pipes of degradation of the casing and insulation, and of hot spots and deposition materials in the crevices? (5) Regarding the insulating material between Incoloy 800H and outer pressure pipe of 16Mo3: Have effects of breaks, gaps, and movement (e.g., caused by gas flow vibration) been evaluated?

<u>Discussion and Questions Related to Quality Group Designations</u>

4.2.18. The described PBMR has very high temperatures (i.e., > 370°C) throughout much of the primary system, such that it is expected that material creep and thermal fatigue could be significant. This may result in increased probability of failure of the pressure boundary components, compared to operational experience related to LWRs. Therefore, in accordance with General Design Criteria (GDC) 14, 15, 30, and 31 of 10 CRF 50 Appendix A, it would appear that the primary cooling system should be ASME Class 1. Provide a justification for not classifying all primary system components as ASME Class 1.

4.2.19. A part of the described PBMR design would consist of ASME Class 2 piping for conveying coolant to and from the reactor pressure vessel. This piping would connect directly to the Class 1 vessel without isolation valves or other barriers. However, it is necessary to provide some type of physical separation between the Class 2 and Class 1 components, since it is necessary to protect the pressure retaining integrity of the reactor vessel from a failure in the Class 2 piping. Provide justification for the selection of Class 2 for the connected primary system piping.

<u>Discussion and Questions Related to Leak Detection and Monitoring Programs</u>

4.2.20. Describe in detail the leak detection systems and/or monitoring program to identify leakage from the primary pressure boundary and the internal leakage from the sleeves of the hot piping and from the water/gas interface of the secondary water cooling systems. To ensure early detection of leakage, the sensitivity, response time, and monitoring frequency of the designed leakage detection systems/monitoring program should also be discussed. If leak before break (LBB) is planned in the design, discuss how the leak detection requirements are met. If a leak detection system is not planned in the plant design, provide justification why it is not needed, and also discuss how the design philosophy of defense in depth is met to ensure safe plant operation.

4.2.21. Reference 2, attachment 2 indicates leaking helium will be detected by the small quantities of radionuclides carried in the escaping gas before the helium pressure drops below the water pressure. Which radionuclides are being monitored? What is the effect of these radionuclides in the PCU system (for example, cobalt activation may be a concern due to continued exposure of the metallic components to the radionuclides over the operating lifetime of the plant)?