SRXB Input to RAI related to "Control of Chemical Attack in the PBMR Presentation to USNRC In Support of PBMR Pre-application Activities"

In the subject presentation the claim is made 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

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 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 depended 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 of the subject report, we request the following additional information:

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

2. 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.

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

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

G/1

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.

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

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

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

8. 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.

9. 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 the subject presentation. 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.

10. For the beyond the design basis case in the subject presentation 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.

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

12. Page 3: What sources of water exist in the PBMR and how are these sources mitigated from entering the primary system?

13. Page 6: What is the time scale for detecting water ingress leaks and isolating the leaks?

SRXB Comments and RAIs on the PBMR White Paper - High-Temperature Materials Graphite

The subject White Paper consists mainly of sections and figures copied from the short summary report "Graphite for High-Temperature Reactors", EPRI, August 2001, prepared B. Marsden of AEA Technology in England. These sections are annotated with comments as to the relevance of the material to PBMR conditions. This material together with the handout "Pebble Bed Modular Reactor High Temperature Materials Graphite" by Mark A Davies, October 2001 will be the context of what follows. In general, the data exhibited and the statements made in the above do not indicate whether they apply to a He atmosphere or some other atmosphere such as CO_2 in the case of British data. It is well known that the type of atmosphere affects 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, ora justification given why data collected in another atmosphere is applicable.

Nuclear Graphite Manufacture:

a) On page 37 of the handout the statement is made "Suitable Nuclear Grade Graphites can be determined by appropriate choice of manufacturing process parameters". How can you be sure that you can make such a determination, what appears to be a priori, in light of the fact that there is no data beyond "turnaround" for PBMR conditions?

b) PBMR has chosen Sigri Great Lakes as the preferred supplier for the graphite reflector. A table is show 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 conditions of interest in PBMR?

The Damage Process:

a) 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?

b) 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."

i) Do you mean that 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 the handout implies that there is significant material to material variability with regard to turnaround and the fluence level at the material exhaustion limit. What are the mathematical expressions used to characterize the material behavior of the relevant graphites; and how do you plan to estimate the coefficients which distinguish the behaviors between particular graphites?

ii) 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?

iii) Do you account for cyclic fatigue, such as due to thermal striping in the plenums?

iv) 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?

c) Is the expression on page 18 of the White Paper for creep or the creep rate?

i) The expression includes primary and secondary creep terms. Is there a tertiary creep term in the case of PBMR conditions?

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

iii) Under what conditions, if any, does material exhaustion supersede the creep limit?

Component Performance Assessment:

a) 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?

b) The recommended failure criteria for PBMR graphite components give on page 22 of the White Paper is consistent with how many effective full power years of PBMR operation? Are these criteria affected by the reported premature cracking in AGRs (Nucleonics Week, January 24, 2002)?

c) 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?

Additional Questions:

Page 2

1. Under the adequate cooling of the pebble bed section it is stated that "care must be taken in the design to ensure that the core barrel does exceed temperature limits due to natural circulation of hot gas from the core." Assuming that it was actually meant that the core barrel should NOT exceed the temperature limits, please provide the temperature limits for the core barrel and other critical components.

Page 3

2. It is stated that the outlet plenum is designed so that the differential temperature within the gas leaving the care is less than 60 degrees C. What type of stresses does this temperature differential produce?

3. Please explain how the geometry requirement for the RCSS has a significant impact on the structural performance of the blocks during the lifetime of the plant.

4. Please provide additional details of the method for vertically stepping the top blocks of the bottom reflector.

Page 14

5. It is stated that PBMR will ignore the concept of equivalent temperature. Does this imply that PBMR does not intend to use the other data or will a correlation between the previous data and PBMR be supplied?

Page 15

6. Please provide additional details on what is meant by "family" characteristics and how ATR-2E and VQMB graphite relate to PBMR graphite.

7. What is the temperature range for which the fluence has been extrapolated? Please provide justification for extrapolating the data.

Page 17

8. 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?

Page 23

9. 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.

10. What are the typical parameter ranges for material exhaustion to occur?

General note

11. Provide verification and validation for all the equations and relationships used to support the PBMR graphite relationships.

COMMENTS ON "SUMMARY OF PEBBLE BED MODULAR REACTOR (PBMR) DESIGN CODES AND STANDARDS"

By letter dated October 30, 2001, Exelon Generation requested the NRC staff to provide feedback on Exelon's presentation of PBMR design codes and standards for Pre-application. The EEIB staff has reviewed the "Electrical and Instrumentation & Control" Section statement which states:

> *The primary standards for the nuclear safety-related systems, such as Reactor Protection System (RPS), Post-Event Instrumentation (PEI), associated neutronic instrumentation, and RPS & PEI Human Machine Interfaces (HMI), are Institute of Electrical and Electronics Engineers, IEEE Standard (Std) 603, 1998 and IEEE Std 7-4.3.2, 1993. Other applicable IEEE sub-references used are, IEEE std 308, IEEE Std 344, IEEE 577 and IEEE 1023 for RPS/PEI HMI only. Guidance from NUREG 0800 will also be used. It is noted that 10 CFR 50.55a refers to IEEE 603, 1991 edition; the PBMR project is working to the 1998 edition; This will need to be addressed with the NRC.*
>
> *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 controls will be designed in accordance with international Electrotechnical Commission (IEC) standards. The international Electrotechnical Commission, based in Geneva, Switzerland, is affiliated with the International Organization for Standardization - ISO, and is endorsed by 14 countries, including US, UK and Germany.*
>
> *NUREG 0700, "Human System Interface Review Guidelines" is the primary input to Control Room Design (excluding RPS & PEI HMI). Other applicable references include NUREG 0711, NUREG CR5908, NUREG CR6105, and NUREG CR 6146 and other relevant NRC issued CR's. Guidance from NUREG 0800, Chapters 13 and 18, will also be used. It was noted that the Conrtol Room design is in preliminary stage at this time.*

Guidance from NUREG 0737 and other appropriate guidance will be used for the radiological monitoring system, seismic monitoring system, etc., which are in the preliminary design stage.

IEC Standards will be used in the design of the 50 Hz electrical power systems. IEEE Standards will be used in the design of the 60 Hz electrical power systems.

STAFF COMMENTS ON PBMR DESIGN CODES AND STANDARDS (I&C AREA)

1. 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.

2. 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 describe 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. Most Light Water Reactors refueling cycles are less than 2 years. The PBMR outage cycle is 6 years. Instrument drift during such long operating period would be a concern. The instrument setpoint methodology should take that consideration.

4. 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.

5. 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. These issues are outside the EEIB review scope. That should be reviewed by Human Factor reviewer (IOLB).

6. IEEE Std 308, "IEEE standard Criteria for Class 1E Power Systems", IEEE Std 344, "Seismic Qualification of Class 1E Equipment", and IEEE Std 577, "Requirement for Reliability Analysis in the Design of Operation of Safety System", and radiological monitoring system are outside the I&C review scope.

COMMENTS ON "SUMMARY OF PEBBLE BED MODULAR REACTOR (PBMR) DESIGN CODES AND STANDARDS," BY ELECTRICAL POWER SECTION

By letter dated October 30, 2001, Exelon Generation requested the NRC staff to provide feedback on Exelon's presentation of PBMR design codes and standards for Pre-application. The Electrical Section of EEIB has reviewed it and provided the following comments:

> *The primary standards for the nuclear safety-related systems are: IEEE Std. 308-1991,"Criteria for Class 1E Systems," IEEE Std. 323-1990, "Qualifying Class 1E Equipment," IEEE Std 384-1992, " Independence of Class 1E Equipment and Circuits," IEEE Std, 484-1996, "Installation Design, and Installation of Vented Lead-Acid batteries," IEEE Std. 603-1991, " Criteria for Safety Systems, and IEEE Std. 741-1997, "Protection of Class 1E systems and Equipment." Other applicable IEEE sub-references used are, IEEE std 317-1983, "Electric Penetration Assemblies in Containment Structures," IEEE Std 344-1993, "Seismic Qualification of Electric Equipment, " IEEE Std 379-1983, "Type Test of Class 1E Electric Cables, Field Splices, and Connections," IEEE Std 450-1995, "Practice for Maintenance, Testing, and Replacement of Vented Lead Acid Batteries," IEEE Std. 650-1990, "Qualification of Class 1E Battery Chargers and Inverters," IEEE Std 665-1987, 'Generating Station Grounding," IEEE Std 765, "Preferred Power Supply," and IEEE Std. 944-1986, "Application and Testing of Uninterruptible Power Supplies." The station blackout (SBO) coping capability to conform to 10 Codes of Federal Regulation 50.63 will also be used. Guidance from NUREG 0800 Chapter 8 will also be used. Most Light Water Reactors refueling cycles are ≤ 2 years and the PBMR outage cycle is 6 years.*
>
> *IEC Standards will be used in the design of the 50 Hz electrical power systems. IEEE Standards will be used in the design of the 60 Hz electrical power systems.*
>
> *All the standards addressed above have not had a current endorsement. We shall address them as we proceed with our review*