UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

October 29, 2008

Mr. R.W. Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: INTERIM LETTER 5: CHAPTERS 19 AND 22 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE ESBWR DESIGN

Dear Mr. Borchardt:

During the 556th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 2008, we discussed Chapters 19 and 22 of the NRC staff's Safety Evaluation Report (SER) with open items related to the Economic Simplified Boiling Water Reactor (ESBWR) design certification application. Our ESBWR Subcommittee held meetings on June 3 and August 21-22, 2008, to discuss technical aspects of the ESBWR design as well as the open items and the combined license (COL) action items identified in these Chapters. During these meetings, we had the benefit of discussions with representatives of the NRC staff and General Electric-Hitachi Nuclear Energy (GEH). We also had the benefit of the documents referenced. We previously commented on Chapters 2, 5, 8, 11, 12, and 17 in our November 20, 2007, letter, on Chapters 9, 10, 13, and 16 in our March 20, 2008, letter, on Chapters 4, 6, 15, 18, and 21 in our May 23, 2008, letter, and on Chapter 3 in our July 21, 2008, letter. Our reviews have not addressed security matters and their impact on ESBWR design.

CONCLUSIONS AND RECOMMENDATIONS

1. We await the staff's completion of the review of the ESBWR Probabilistic Risk Analysis (PRA) Revision 3 prior to evaluating the adequacy of the PRA for ESBWR Design Certification.
2. The bases for the assumption that passive ESBWR systems have a very low failure probability are currently incomplete. A better exposition of GEH analysis and a more systematic evaluation of the relevant uncertainties are required.
3. The technical basis for the failure probability estimates for the digital instrumentation and control (DI&C) systems should be provided.
4. Specific issues need to be clarified to ensure the functionality of the Basemat-internal Melt Arrest and Coolability device as a 'defense-in-depth' measure for severe accident conditions.
5. We will review the resolution of open items in SER Chapters 19 and 22 during future meetings.

BACKGROUND

The ESBWR is a direct-cycle power conversion system with natural circulation cooling in the reactor vessel under normal operation. It has a passive emergency core cooling system that operates without the need for emergency alternating current power systems or operator actions within the first 72 hours following a reactor transient or accident.

At the request of the staff, we have agreed to review the staff's SER on a chapter-by-chapter basis to identify technical issues that merit further consideration, thereby aiding effective resolution of any concerns, as well as assisting in the timely completion of the review of the ESBWR design certification application. Accordingly, the staff has provided at this time SER Chapters 19 and 22 with open items for our review.

DISCUSSION

The ESBWR design certification application was accepted formally by the staff in December 2005. Since that time, revisions to the Design Certification Document (DCD) have been issued, with the most recent being DCD Revision 5 and the PRA Revision 3. These revisions have included updates to the overall design and modifications that address the staff's requests for additional information originating from the staff's review of DCD Revision 4 as well as expanded analyses and correction of errors in the PRA.

Chapters 1-21 of PRA Revision 3 are identical to those of PRA Revision 2. The updates to the PRA are described in Chapter 22 but documentation of the updated logic diagrams was not provided. Our preliminary review of the PRA models in Revision 2 identified logic errors, inconsistencies, and lack of fidelity to the design descriptions in selected fault trees for the Gravity-Driven Cooling System (GDCS) and the Isolation Condenser System (ICS). Additional preliminary review of PRA models also identified errors and omissions in the modeling of physical and functional dependencies through the integrated event tree models (e.g., GDCS deluge valves success criteria and models, and anticipated transient without scram impacts from Standby Liquid Control System injection line breaks). Discussions with GEH indicate that many of the errors we identified in Revision 2 have been addressed. The GEH PRA analysts in their discussions of the PRA models demonstrated a thorough knowledge of the design and its details. However, the available Revision 3 PRA documentation provided to us does not contain sufficient detail for us to fully confirm the current status of the models. The staff is aware of these issues and will soon conduct an on-site audit of the PRA Revision 3 and supporting documentation. After the staff has completed its review, we will review the adequacy of the PRA for ESBWR design certification.

The analysts have made a number of simplifying assumptions about which components and causes for failure will be addressed. For example, possible causes for spurious closure of manual valves are systematically omitted from the models. The models include contributions from unplanned maintenance on active equipment in selected systems, but the models do not account consistently for equipment unavailability due to unplanned maintenance. The PRA also does not account for unavailability of safety system equipment due to the planned, coordinated work that may be performed during plant power operation, which is allowed by the current Technical Specifications. GEH explained that such simplifications are deliberate decisions by

PRA analysts. Limited sensitivity studies have been performed to examine the potential risk impacts from individual issues. However, the integrated impacts from these omissions remain unclear.

The ESBWR design is incomplete and includes new equipment for which there is no operational experience. Therefore, some of the PRA models and data are generic and cannot yet be design-specific. In the one case where substantial test data have been collected to develop an appropriate failure rate for a new component design, the wet-well vacuum breakers, the approach and assumptions have not been justified.

The comparisons between MAAP and TRACG analyses that were performed as part of sensitivity studies to establish PRA success criteria have demonstrated the adequacy of MAAP for the thermal-hydraulic calculations needed to support the ESBWR PRA. The sensitivity studies are helpful in addressing the robustness of passive safety systems. GEH has attempted to address uncertainty in thermal-hydraulic performance of the passive systems through building conservatism into the success criteria; i.e., a "minimum"+1 approach. However, it is not clear to us how the "minimum" numbers were obtained. If the "minimum" is not the true minimum, the "minimum"+1 may not represent conservatism that can be used to address uncertainty.

GEH has addressed thermal-hydraulic uncertainty through sensitivity studies. A better exposition of the failure modes and the effects of uncertainties on passive system reliability is needed to increase our confidence in their results. For example, it is not clear that the sensitivity studies have addressed the full range of uncertainty in the thermal-hydraulic variables.

During our reviews of the Office of Nuclear Regulatory Research program on DI&C systems, we have commented that it is premature to estimate failure probabilities (Ref. 8). Rather, we have urged the staff to focus on a systematic identification of the failure modes for these DI&C systems. The failure probability estimates for these DI&C systems are provided in Table 5.2-3 of the ESBWR PRA, Revision3.

- Failure probability (digital trip module fails to function) = 6.00E-04

- Common-cause failure probability of two trip modules = 1.111E-05

- Common-cause failure probability of three trip modules = 1.111E-06

The technical basis for these probabilities should be provided.

The Basemat-internal Melt Arrest and Coolability (BiMAC) device is a novel defense-in-depth core retention design to provide for long-term core debris coolability for severe accident management. GEH provided extensive documentation in regard to steady-state heat transfer test data and analysis (scaled as ½ scale and ¼ segment size) of the BiMAC device and its ability to remove core debris decay heat in the drywell cavity. The scaling laws for this steady-state experiment need further explanation to ensure that the tests are adequate to demonstrate the applicability of the behavior at full scale. In addition, the onset of flow instabilities (static and/or dynamic) may inhibit local coolant flow and convective cooling, thereby compromising burnout limits. GEH will provide additional information to help assess the likelihood of such instabilities.

The initial core melt transient deposition that initiates operation of the BiMAC device could pose problems for the long-term operability of the BiMAC device. It is not clear what the composition of the sacrificial material above the cooling tubes is to be and whether it will be able to handle the high transient heat flux that will occur when melt pours onto a localized region, (this deposition could be complicated by a metallic melt pour and/or a large pour rate). The GEH documentation does not seem to provide any review of past molten core-concrete interaction (MCCI) experiments for transient core melt pouring and melt spreading behavior in order to bound this initial transient heat load and its effects on the BiMAC device. GEH documentation does not provide an analysis of an initial asymmetric pour that would inhibit melt spreading and possibly cause an excessive heat flux damaging the BiMAC device near its corners. Finally, the GEH analysis does not seem to consider ex-vessel steam explosions as a mechanism to damage the BiMAC downcomer feed tubes along the vertical walls. Asymmetric melt pours into the water pool, after initial melt deposition and deluge valve actuation, could result in ex-vessel steam explosions that could 'crimp' the BiMAC downcomer tubes and thus affect long-term coolability. These specific issues need to be clarified to be able to assess the functionality of the BiMAC device as a 'defense-in-depth' measure for severe accident conditions.

We will review the resolution of open items in SER Chapters 19 and 22 during future meetings.

Sincerely,

/RA/

William J. Shack
Chairman

REFERENCES

1. Memorandum from David B. Matthews, Director, Division of New Reactor Licensing (DNRL), Office of New Reactors (NRO), to Frank P. Gillespie, Executive Director, Advisory Committee on Reactor Safeguards and Advisory Committee on Nuclear Waste, dated May 16, 2008, transmitting SER with Open Items for Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation" Regarding the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification (ML080850462).

2. Memorandum from David B. Matthews, Director, Division of New Reactor Licensing (DNRL), Office of New Reactors (NRO), to Frank P. Gillespie, Executive Director, Advisory Committee on Reactor Safeguards and Advisory Committee on Nuclear Waste, dated May 8, 2008, transmitting SER with Open Items for Chapter 22, "Regulatory Treatment of Nonsafety Systems" Regarding the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification (ML080940713).

3. Letter from James C. Kinsey, Project Manager, ESBWR Licensing, GEH, to NRC, dated February 22, 2007, transmitting ESBWR Design Control Document, Revision 3 (ML070660561).

4. Letter from William J. Shack, Chairman, Advisory Committee on Reactor Safeguards, to Luis A. Reyes, Executive Director for Operations, dated November 20, 2007, "Interim Letter: Chapters 2, 5, 8, 11, 12, and 17 of the NRC Staff's Safety Evaluation Report With Open Items Related to the Certification of the ESBWR Design" (ML073070006).

5. Letter from William J. Shack, Chairman, Advisory Committee on Reactor Safeguards, to Luis A. Reyes, Executive Director for Operations, dated March 20, 2008, "Interim Letter Chapters 9, 10, 13, and 16 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design" (ML080670596).

6. Letter from William J. Shack, Chairman, Advisory Committee on Reactor Safeguards, to R. W. Borchardt, Executive Director for Operations, dated May 23, 2008, "Interim Letter 3: Chapters 4, 6, 15, 18, and 21 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design" (ML081330447).

7. Letter from William J. Shack, Chairman, Advisory Committee on Reactor Safeguards, to R. W. Borchardt, Executive Director for Operations, dated July 21, 2008, "Interim Letter 4: Chapter 3 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design" (ML081930777).

8. Letter from William J. Shack, Chairman, Advisory Committee on Reactor Safeguards, to Dale E. Klein, Chairman, U. S. Nuclear Regulatory Commission, dated April 29, 2008, "Digital Instrumentation and Control Systems Interim Staff Guidance" (ML081050636).

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1. Memorandum from David B. Matthews, Director, Division of New Reactor Licensing (DNRL), Office of New Reactors (NRO), to Frank P. Gillespie, Executive Director, Advisory Committee on Reactor Safeguards and Advisory Committee on Nuclear Waste, dated May 16, 2008, transmitting SER with Open Items for Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation" Regarding the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification (ML080850462).
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Letter to R. William Borchardt, EDO, NRC, from William J. Shack, Chairman, ACRS, dated October 29, 2008

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