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FINAL REPLY:

William J. Shack, ACRS

TO:

Borchardt, EDO

FOR SIGNATURE OF :

** GRN **

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Borchardt, EDO

DESC:

ROUTING:

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 (EDATS: OEDO-2008-0418)

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ACRS File

DATE: 05/28/08

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CONTACT:

NRO

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

May 23, 2008

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

SUBJECT: 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

Dear Mr. Borchardt:

During the 552nd meeting of the Advisory Committee on Reactor Safeguards, May 8-9, 2008, we discussed five Chapters from 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 January 16-17 and April 9, 2008, to discuss technical aspects of the ESBWR design, as well as the open items and the Combined License (COL) action items identified in each of these SER Chapters. During these meetings, we had the benefit of discussions with representatives of the NRC staff and General Electric-Hitachi Nuclear Americas, LLC (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 and on Chapters 9, 10, 13, and 16 in our March 20, 2008, letter. Our reviews have not addressed security matters and their impact on the ESBWR design.

CONCLUSIONS AND RECOMMENDATIONS

1. Based on our review of the staff's SER Chapters 4, 6, 15, 18, and 21, we have identified the following issues that merit additional attention:
 - Confirmation of coupled neutronic and thermal-hydraulic stability including interactions between the core and chimney.
 - Assurance of proper operation of the vacuum breaker system by appropriate surveillance testing, leakage monitoring, and isolation capability.
 - Demonstration of the performance of passive safety systems addressing issues such as gas binding.
 - Assurance that the proposed principles of Human Factors Engineering are appropriately integrated in the ESBWR design.

2. We plan to review the staff's resolution of the open items in SER Chapters 4, 6, 15, 18, and 21 during future meetings.
3. ESBWR systems described in these Chapters interact with other systems discussed in other SER Chapters that have not been reviewed. We will comment on potential safety implications of any system interactions in future interim letters and in our final report.

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. The ESBWR also has passive containment cooling to ensure heat transport to the ultimate heat sink for all accident scenarios. To cope with a severe reactor accident, the ESBWR design incorporates a lower drywell core retention device and allows passive drywell flooding to provide long-term debris cooling.

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 consideration, thereby aiding effective resolution of any concerns, as well as assisting timely completion of the review of the ESBWR design certification application. Accordingly, the staff has provided SER Chapters 4, 6, 15, 18, and 21 with open items and COL action items for our review.

DISCUSSION

Based on the information presented to us to date, we have the following comments on aspects of the ESBWR design addressed in these Chapters.

Chapter 4: Reactor

The ESBWR design includes the fuel system (fuel rods and fuel assemblies), the nuclear design, the associated thermal-hydraulic design, reactor materials, and the functional design of the control rod drive system.

The ESBWR core consists of 1132 fuel bundles and 269 control blades. The reference core design is based on an equilibrium cycle loading pattern, and associated fuel characteristics. This design is similar to that of current Boiling Water Reactors (BWRs) except that the recirculation pumps and associated piping are eliminated with flow through the reactor core accomplished via natural circulation. This natural circulation flow depends on the balance between the frictional pressure drop and the gravity head supplied from the difference of the downcomer coolant density and the average core/chimney density.

To reduce the frictional pressure drop, the fuel assemblies General Electric ((GE) 14E fuel) for the ESBWR design are shorter than those for currently operating BWRs by approximately 2 feet. This change affects the grid spacer positions as well as the part-length rod heights. Most importantly, such changes will affect the Critical Heat Flux (CHF) limits. To demonstrate that these CHF limits can be predicted by the GE Critical Quality-Boiling Length (GEXL) correlation with uncertainties, GEH performed full-scale prototypic tests at Stern Laboratories Inc. for GE14E fuel. Based on a comparison of the experimental data with the GEXL predictions, GEH has presented a reasonable case to confirm the applicability of the GEXL correlation.

The ESBWR natural circulation flow directly determines the void exiting the reactor core. Spatial or temporal variations in the void can have a feedback effect on the overall core recirculation ratio causing an oscillation in the core flow. This can cause an oscillation in the core power due to neutronic feedback effects. In current BWRs, the two-phase coolant flow exits the core and discharges into an open chimney common to all fuel assemblies in the core. Any spatial variations in void that would affect the two-phase average density across the core exit are minimized since the gravity head only contributes 25-30 percent of the total driving head.

The Dodewaard BWR was a small (183 MWt), natural circulation power reactor. In the core design of that reactor, sets of 4 fuel assemblies discharged two-phase coolant flow into a single chimney, thus spatial void variations were small. Empirical data from its operation indicated that naturally occurring unsteady two-phase flow perturbations were damped and did not lead to density wave oscillations or power-flow instabilities.

The ESBWR is a much larger reactor where sets of 16 fuel assemblies discharge two-phase coolant into a single chimney. There is a greater likelihood of spatial void variations regionally and at the assembly exits, along with density wave oscillations and associated power-flow instabilities. However, all thermal-hydraulic analyses by GEH and independent calculations by the staff indicate that these instabilities are damped and the reactor remains stable. GEH is completing more detailed thermal-hydraulic studies of the design to address the staff's open item related to power-flow stability. We note that full-scale confirmation of power-flow stability will be ultimately determined during start-up testing.

Chapter 6: Engineered Safety Features

The engineered safety features of the ESBWR design are those systems provided to mitigate the consequences of postulated accidents, which include containment cooling systems, fission product containment, emergency core cooling systems, and control room habitability systems. When the reactor vessel is at high pressure, passive natural circulation decay heat removal is accomplished by the isolation condenser system. The isolation condenser system includes heat exchangers which transfer heat to water pools on top of the containment and vent directly to the environment. The automatic depressurization system depressurizes the reactor vessel in the event of an accident or when the isolation condenser system is unavailable. Once depressurized, the gravity driven cooling system provides coolant to refill the vessel. Given the actuation logic of these systems, analyses show that the core does not uncover during any design basis accident.

The passive containment cooling system (PCCS) removes decay heat from the drywell. Similar to the isolation condenser system, the PCCS heat exchangers sit in pools of water on top of the containment that vent to the environment. After steam condensation, the PCCS heat exchangers vent noncondensables to the suppression pool. This venting occurs whenever buildup of noncondensables degrades heat exchanger performance to the point the drywell pressure increases above wetwell pressure until the differential pressure is equal to the submergence of the PCCS vent line in the suppression pool.

GEH TRACG analyses indicate that the main steamline break is the limiting accident that challenges the PCCS and maximizes the containment pressures and temperatures. Using MELCOR, with initial and boundary conditions for the reactor vessel derived from GEH TRACG analyses, the staff and its contractors performed audit analyses to examine the containment response for this accident. These simulations yielded predictions of containment pressures and temperatures that are quite similar to those performed by GEH using TRACG. Both TRACG and MELCOR analyses include assumptions that need careful examination. First, the dominant factor that affects containment pressurization is bypass leakage from the drywell to the wetwell through the vacuum breakers. The vacuum breakers allow gas and steam flow from the wetwell to the drywell to equalize pressure, when the pressure in the drywell decreases below the wetwell pressure.

The vacuum breakers must perform with high reliability to allow the PCCS to function properly and maintain the pressure well below the containment design pressure. GEH tests indicate good performance of this new vacuum breaker design under a range of normal as well as adverse operational conditions. Both GEH and the staff have confirmed that design basis analyses indicate that if the vacuum breaker leakage is more than double its maximum allowable limit, containment pressurization for the main steamline break can exceed containment design pressures. The ESBWR design includes a leak detection system and isolation valves that close in case of vacuum breaker failure. The staff identified the reliability of the vacuum breaker system as a significant open item until GEH develops a protocol for acceptable periodic testing, leakage detection, and isolation capability. We concur with the staff that assurance of proper operation of the vacuum breaker system is important.

In its safety analyses for containment pressurization, GEH used simplifying assumptions regarding the treatment of noncondensable gases. The code used for these analyses, TRACG, does not have the capability to realistically model spatial mixing and stratification of noncondensable gases and steam. Thus, homogeneous mixing is assumed for any compartment or computational volume in the analyses. With this assumption, noncondensable gases are transported more readily to the wetwell during the accident, thereby overestimating their contribution to the containment pressure. GEH has, therefore, stated that TRACG analysis results are conservative. Audit calculations performed by the staff and its contractors using MELCOR confirm that the results are conservative.

Although these calculations suggest that the containment pressurization results are conservative, the final factor that affects containment pressurization is the need to ensure proper gravity drainage of water from locations such as the gravity driven cooling system to the vessel during the accident progression and from the PCCS to the gravity driven cooling system tanks. The details of the piping geometry and arrangements in these systems are critical in assuring that noncondensable gases are not initially present or do not become trapped during operation of the gravity driven cooling system and the PCCS. The staff is aware of these concerns and we urge that detailed design of the piping arrangements be reviewed to ensure gases will not be trapped in these piping systems. GEH needs to demonstrate the performance of these passive safety systems given the potential for gas pockets. GEH also needs to develop a protocol for periodic testing and venting of passive safety systems to minimize the presence of noncondensable gases.

Chapter 15: Accident Analysis and Chapter 21: Computer Code Validation

As noted previously, the automatic depressurization system depressurizes the reactor vessel in the event of any design basis accident or if the isolation condenser system is unavailable. Once the vessel is depressurized, the gravity driven cooling system injects coolant to refill the vessel. Given the actuation logic of these systems, analyses show that the core does not uncover during any design basis accident. Thus, the fuel integrity is not challenged by the typical design basis accidents.

Events that are not design basis accidents but are evaluated in BWR safety analysis reports are termed, "special events." GEH has retained this classification for the ESBWR. The Anticipated Transient Without Scram (ATWS) is among the special events analyzed by GEH and reviewed by the staff. Several ATWS events were analyzed by GEH to provide assurance that unacceptable plant conditions will not occur. Results of analyses performed using TRACG indicate that the ESBWR design meets appropriate acceptance criteria. ATWS is also estimated to be a low frequency event. The NRC staff is currently reviewing the applicability of TRACG for the ESBWR ATWS analysis as presented in NEDE 33083-P, Supplement 2. Approval of this analysis is contingent on the GEH response to the staff's open items regarding NEDE-33083P, Supplement 2. We concur.

Chapter 18: Human Factors Engineering

Human Factors Engineering (HFE) includes 12 areas of review:

- HFE Program Management
- Operating Experience Review
- Functional Requirements Analysis and Function Allocation
- Task Analysis
- Staffing and Qualifications
- Human Reliability Analysis
- Human-System Interface Design
- Procedure Development
- Training Program Development
- Human Factors Verification and Validation
- Design Implementation
- Human Performance Monitoring

Criteria for review of these areas are specified in the Standard Review Plan. In addition, the staff used review criteria contained in "Human Factors Engineering Program Review Model" (NUREG-0711, Rev. 2). For nuclear power plants licensed under 10 CFR Part 52, HFE program elements are to be assessed during the COL application review and all HFE Design Acceptance Criteria are to be addressed before plant startup. However, during the design certification process, Inspections, Tests, Analyses, and Acceptance Criteria (ITAACs) are formulated for HFE.

The staff plans to perform on-site audits of the HFE design process. During this audit, the staff should ensure that the HFE principles are appropriately integrated into the ESBWR design. As the staff performs these audits and reviews the proposed ITAACs, we would like to be kept informed.

We plan to review the resolution of the open items identified in the SER Chapters 4, 6, 15, 18, and 21 during future meetings. Many of the ESBWR systems described in these Chapters may interact with systems discussed in other Chapters of the SER that have not been reviewed. We also plan to comment on the potential safety implications of any system interactions in future interim letters and in our final report.

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 and Materials (ACRS/ACNW&M), dated November 1, 2007, transmitting SER with open items for Chapter 9, "Auxiliary Systems" (ML072900345 and ML072900355).
2. Memorandum from David B. Matthews, Director, DNRL, NRO, to Frank P. Gillespie, Executive Director, ACRS/ACNW&M, dated September 20, 2007, transmitting SER with open items for Chapter 10, "Steam and Power Conversion System" (ML072350529 and ML070720143).
3. Memorandum from David B. Matthews, Director, DNRL, NRO, to Frank P. Gillespie, Executive Director, ACRS/ACNW&M, dated November 1, 2007, transmitting SER with open items for Chapter 13, "Conduct of Operations" (ML072290020 and ML072290010).
4. Memorandum from David B. Matthews, Director, DNRL, NRO, to Frank P. Gillespie, Executive Director, ACRS/ACNW&M, dated November 1, 2007, transmitting SER with open items for Chapter 16, "Technical Specifications" (ML072910513 and ML072910702).
5. Letter from James C. Kinsey, Project Manager, ESBWR Licensing, GEH, to NRC, dated February 22, 2007, transmitting ESBWR Design Control Document, Revision 3 (ML070660561).
6. 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).
7. 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).

8. "TRACG Application for ESBWR Anticipated Transient Without Scram Analyses," M. D. Alamgir, Wayne Marquino, Antonio J. Barrett, Shivakumar Sitaraman, Bharat S. Shiralkar, NEDE-33083P, Supplement 2, Revision 1, February 2008 (ML080520104).
9. U. S. Nuclear Regulatory Commission, NUREG-0711, Rev. 2, "Human Factors Engineering Program Review Model," John O'Hara, James Higgins, J. Persensky, Paul Lewis, James Bongarra (ML033290024).
10. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," March 2007.
11. Code of Federal Regulations, Title 10, Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."