

December 2, 2008

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: INTERIM LETTER 5: CHAPTERS 19 AND 22 OF THE U.S. NUCLEAR
REGULATORY COMMISSION STAFF'S SAFETY EVALUATION REPORT WITH
OPEN ITEMS RELATED TO THE CERTIFICATION OF THE ECONOMIC
SIMPLIFIED BOILING-WATER REACTOR DESIGN

Dear Dr. Shack:

I am responding to your letter of October 29, 2008, regarding the Advisory Committee on Reactor Safeguards (ACRS or Committee) meeting on October 2, 2008.

During the full committee meeting, the staff discussed its safety evaluation reports (SERs) with open items (OIs) for Chapter 19 of the economic simplified boiling-water reactor design (ESBWR) certification application. These discussions included the status and technical concerns of the OIs identified in Chapter 19 and Chapter 22 of the staff's SER. The ACRS raised several specific concerns in its letter.

Enclosed are the staff's responses to the ACRS comments. The staff will continue to work with GE Hitachi Nuclear Energy to obtain satisfactory resolution to the OIs presented in the SERs.

Thank you for your comments. I appreciate the willingness of the ACRS to engage with the staff on a chapter-by-chapter SER review. I believe this process has greatly facilitated the staff's work, and we look forward to continued interactions with the Committee for the remaining chapters of the ESBWR design certification application.

Sincerely,

/RA Bruce Mallett for/

R. W. Borchardt
Executive Director
for Operations

Enclosure:
Staff Response to ACRS Comments

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
Commissioner Svinicki
SECY

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U.S. Nuclear Regulatory Commission
Staff Response to the Advisory Committee on Reactor Safeguards Interim Letter
Dated October 29, 2008
Regarding Safety Evaluation Reports with Open Items
on the Economic Simplified Boiling-Water Reactor
Design Certification Application

Following are the responses of the U.S. Nuclear Regulatory Commission (NRC) staff to comments from the Advisory Committee on Reactor Safeguards (ACRS or the Committee) on the staff's safety evaluation report (SER) with open items for Chapter 19, "PRA and Severe Accidents," and Chapter 22, "Regulatory Treatment of Non-Safety Systems," of the economic simplified boiling-water reactor (ESBWR) design certification application. The staff plans to discuss final resolution of these concerns during the ACRS committee meetings on the final SER for the ESBWR design certification application.

ACRS Comment:

"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 the ESBWR Design Certification."

Staff Response:

Documentation of the ESBWR PRA Revision 3 is not complete. To-date, the staff has received only a summary of the changes made to the PRA between Revisions 2 and 3 from GE Hitachi Nuclear Energy (GEH). GEH has indicated that it will complete the full documentation of Revision 3 in December 2008. The staff is planning to conduct an onsite audit of Revision 3 of the PRA and supporting documentation in January 2009.

ACRS Comment:

"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 staff is aware of these issues and will soon conduct an on-site audit of the PRA Revision 3 and supporting documentation."

Staff Response:

As GEH indicated at the Committee meeting on October 2, 2008, many of these errors and modeling inconsistencies were in Revision 2 of the PRA and have been corrected in Revision 3. The staff is aware of these issues and plans to conduct an onsite audit in January 2009. During the onsite audit, the staff will confirm that Revision 3 of the PRA adequately addresses all of these issues. In addition, the staff plans to examine the effectiveness of the GEH self-assessment and quality control processes during the audit to ensure that the GEH PRA meets the required quality level.

ACRS Comment:

“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.”

Staff Response:

The staff briefly discussed the bases for these assumptions with the ACRS Subcommittee for ESBWR Design Certification during the meeting with GEH on August 21, 2008. The staff plans to further review the bases for these assumptions and their cumulative impact with GEH during its onsite audit of Revision 3 of the PRA and supporting documentation scheduled for January 2009.

ACRS Comment:

“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.”

Staff Response:

The staff is aware of the current GEH efforts to address uncertainty in passive system reliability in a systematic fashion. The staff will review the results of this effort, and will discuss its findings with the ACRS.

ACRS Comment:

“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.”

Staff Response:

The staff agrees that some of the PRA data are generic and cannot yet be design specific because of the lack of experience. However, programs exist as described below to ensure that, as experience is gained further in the design process and during plant operation, the PRA data are adjusted appropriately to accurately reflect the reliability of structures, systems, and components (SSCs).

The staff considers the ESBWR Design Reliability Assurance Program (D-RAP), and its role as a foundation for operational reliability assurance over the life of the plant, to be an acceptable method of compensating for the lack of demonstrated reliability in new and novel equipment designs and the uncertainty it creates in the PRA.

As discussed in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," issued March 2007, the NRC expects reactor designers seeking a design certification to develop a D-RAP. The stated purpose of the ESBWR D-RAP is to ensure that plant safety, as estimated by the PRA, is maintained as the detailed design evolves through the implementation and procurement phases. This program also ensures that the design documentation provides pertinent information for the future plant owner/operator so that equipment reliability is maintained commensurate with the PRA assumptions during the entire plant life. One key element of the D-RAP is the determination of dominant failure modes of risk-significant SSCs. The determination of dominant failure modes includes historical information, analytical models, and existing requirements. Because many boiling-water reactor (BWR) systems and components have compiled a significant historical record, an evaluation of that record is performed. For those SSCs for which there is not an adequate historical basis to identify critical failure modes, an analytical approach is used.

The ESBWR D-RAP, as described in Chapter 17 of the ESBWR DCD requires combined operating license holders to include the following in their operational reliability assurance activities:

- Evaluate and maintain the reliability of risk-significant SSCs as identified in the D-RAP.
- Establish a reliability database using historical data on equipment performance as available. The compilation and reduction of these data provide the plant with a source of component reliability information. Data used in PRA fault-tree analyses may also be a viable initial source.
- Use surveillance and testing to establish the level of performance or condition being maintained for SSCs within the scope of the D-RAP and identify declining trends between surveillances before performance or condition degrading to unacceptable levels undetected (or failure) to the extent possible.

The staff has reviewed the ESBWR D-RAP and discussed the results with the ACRS as part of the Committee's review of Chapter 17 of the staff's SER on the ESBWR design. The staff finds the ESBWR D-RAP to be acceptable, with the exception of the methodology for identifying and prioritizing risk-significant SSCs, which is still under review, and will be addressed in the final SER for Chapter 17.

ACRS Comment:

"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."

Staff Response:

GEH established the minimum success criteria for passive systems in the ESBWR PRA using thermal-hydraulic analysis of accident scenarios performed with the Modular Accident Analysis Program (MAAP), Version 4.0.6, as described in its response to Request for Additional Information (RAI) 19.1.0-1 (GEH letter MFN 07-214, dated May 16, 2007). In RAI 19.1.0-1 S01, the staff raised several questions regarding the validity of these analyses, and this issue became an open item in the staff's draft SER. GEH addressed these questions in letters MFN 07-324, Supplements 1 and 2, dated April 11, 2008, and June 26, 2008, respectively, and discussed its responses with the ACRS Subcommittee for ESBWR Design Certification at the August 21, 2008, meeting.

The staff has completed its review of the success criteria evaluation for passive systems in the ESBWR, including responses to the RAIs related to the thermal-hydraulic analyses. The staff has concluded that MAAP, Version 4.0.6, is an acceptable tool for establishing ESBWR PRA success criteria and that the minimum + 1 approach for specifying success criteria is acceptably conservative. The staff accepts this approach, in part, on the basis of sensitivity studies performed by GEH which indicate that the core damage frequency is not sensitive to changes in the values of the success criteria determined from the minimum + 1 approach.

ACRS Comment:

"The technical basis of the failure probability estimates for the digital instrumentation and controls (DI&C) systems should be provided."

"During our reviews of the Office of Nuclear Regulatory Research program on DI&C [digital instrumentation and control] 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 [following] failure probability estimates for these DI&C systems are provided in Table 5.2-3 of the ESBWR PRA, Revision 3.

- 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."

Staff Response:

The staff agrees that the technical bases for these failure probabilities in the DI&C model need to be understood. The staff plans to review the bases for these probability assumptions further with GEH during its onsite audit of Revision 3 of the PRA and supporting documentation scheduled for January 2009.

In RAI 19.1-22, the staff requested that GEH describe how it determined the failure probabilities for the basic and common-cause events in the system model for the DI&C system and discuss the contribution of various common-cause failure modes for hardware and software. In its response (GEH letter MFN 06-373 dated October 6, 2006), GEH indicated that, because of a lack of design detail, some values were assumptions to be verified later in the design process. It also indicated that common-cause hardware and software failures could not be discussed until the design was complete. The PRA model assumes a digital trip module failure probability of 6.00×10^{-4} . Common-cause failure probabilities are determined using this value and the Multiple Greek Letter calculation method.

ACRS Comment:

“Specific issues need to be clarified to ensure the functionality of the Basemat-internal Melt Arrest and Coolability [BiMAC] device as a ‘defense-in-depth’ measure for severe accident conditions.”

“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.”

“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).”

Staff Response:

As part of its ongoing review of the test report for the BiMAC device, the staff issued a number of RAIs requesting GEH to address these issues. The staff has received, and is currently reviewing, responses to all RAIs except the one related to the transient behavior, which is forthcoming. The responses address the Committee’s concerns regarding scaling, flow instability, type of sacrificial material, and metallic melt explicitly. The staff may issue followup RAIs after completing the review of the GEH responses.

ACRS Comment:

“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.”

Staff Response:

The staff agrees with this comment and is currently preparing an RAI requesting that GEH address these issues.

ACRS Comment:

“...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.”

Staff Response:

The staff agrees with this comment and is currently preparing an RAI requesting that GEH address this issue.