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

April 29, 2013

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

SUBJECT: CHAPTERS 4, 15, 17, AND 19 OF THE SAFETY EVALUATION REPORT WITH OPEN ITEMS FOR CERTIFICATION OF THE US-APWR DESIGN AND SAFETY EVALUATIONS OF SUPPORTING TOPICAL REPORTS

Dear Mr. Borchardt:

During the 603rd meeting of the Advisory Committee on Reactor Safeguards, April 11-12, 2013, we met with representatives of the NRC staff and Mitsubishi Heavy Industries, Ltd. (MHI) to review the following chapters of the Safety Evaluation Report (SER) with Open Items associated with the United States Advanced Pressurized Water Reactor (US-APWR) design certification application:

- Chapter 4, "Reactor"
- Chapter 15, "Transient and Accident Analyses"
- Chapter 17, "Quality Assurance and Reliability Assurance"
- Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation"

We also reviewed the SERs for the following Topical Reports:

- MUAP-07008-P(R2), "Mitsubishi Fuel Design Criteria and Methodology"
- MUAP-07009-P(R0), "Mitsubishi Thermal Design Methodology"
- MUAP-07010-P(R4), "Non-LOCA Methodology"
- MUAP-07011-P(R3), "Large Break LOCA Code Applicability Report for US-APWR"
- MUAP-07013-P(R2), "Small Break LOCA Methodology for US-APWR"

These Topical Reports describe analysis methods that apply to the US-APWR design and may be used for other potential licensing applications.

Our US-APWR Subcommittee reviewed these chapters of the Design Control Document (DCD) and the associated Topical Reports during meetings on July 9-10, 2012; October 18-19, 2012; January 15, 2013; and February 21-22, 2013. Technical aspects of the US-APWR design, analytical methods that are described in the Topical Reports, and the open items identified in each of the SER chapters were discussed at those meetings. We also had the benefit of the documents referenced.

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CONCLUSIONS AND RECOMMENDATION

1. We have not identified any additional issues in SER Chapters 4, 15, 17, and 19 that would preclude certification of the US-APWR design.

2. The conditions and limitations in the SERs for Topical Reports MUAP-07008-P(R2), MUAP-07009-P(R0), MUAP-07010-P(R4), MUAP-07011-P(R3), and MUAP-07013-P(R2) provide appropriate constraints for use of the reference analysis methods for the US-APWR design.

3. Based on our review of these chapters, we have identified the following items that merit additional attention:

   - Revision 3 of the US-APWR probabilistic risk assessment (PRA) is adequate to meet the regulatory requirements and limited objectives for high-level risk information to support the US-APWR design certification. Substantial technical improvements to the PRA would be needed to support detailed plant-specific risk-informed applications and operational programs.

   - The staff should ensure that the MHI thermal-hydraulic models used for PRA success criteria and the initial stages of severe accident progression are benchmarked against values from an NRC-approved thermal-hydraulic code, consistent with the approach in Draft Revision 3 of Standard Review Plan Chapter 19 for passive plant designs.

4. We plan to review the staff's resolution of the open items in SER Chapters 4, 15, and 19 during future meetings. Analyses described in these chapters are affected by the design and operation of systems discussed in SER chapters that we have not yet reviewed. We will comment on safety implications of any system interactions in future interim letters and in our final report.

5. Separate from the US-APWR design certification review, we will examine the adequacy of current regulatory guidance, analysis methods, and industry practices to evaluate and manage the risk from the pellet cladding interaction (PCI) failure mechanism for PWR fuel during Anticipated Operational Occurrences (AOOs).

BACKGROUND

The US-APWR is a four-loop pressurized water reactor (PWR) with a large dry containment. The design includes a combination of active and passive safety systems, arranged in four divisions. Reactor protection, safeguards actuation, and other instrumentation and control functions are developed through integrated digital platforms. Other notable design features include advanced passive accumulators, elimination of low pressure injection pumps, a refueling water storage pit inside the containment, a core debris spreading area below the reactor vessel, and gas turbine generator emergency power supplies.

MHI submitted a DCD with its application for the US-APWR design certification on December 31, 2007. Revision 1 of the DCD was submitted on August 29, 2008; Revision 2 on October 27, 2009; and Revision 3 on March 31, 2011.

We have agreed to review the SER on a chapter-by-chapter basis to identify technical issues that may merit further consideration by the staff. This process aids the resolution of concerns and facilitates timely completion of the US-APWR design certification review. Accordingly, the staff has provided SER Chapters 4, 15, and 19 with open items for our review. SER Chapter 17 does not contain any open items. The staff's SER and our review of these chapters address DCD Revision 3.

MHI has also prepared several Topical Reports that describe analysis methods that apply for the US-APWR design and may be used for other potential licensing applications. We are reviewing each Topical Report and its SER in conjunction with the relevant DCD chapter. Our reviews have focused primarily on applicability of the reference analytical methods to the US-APWR design certification. The SERs for the Topical Reports that are addressed in this letter report are complete, and they do not contain any open items.

DISCUSSION

We have not identified any additional issues in SER Chapters 4, 15, 17, and 19 that would preclude certification of the US-APWR design. We plan to review the resolution of the open items identified in these SER chapters during future meetings.

The staff's evaluations of Topical Reports MUAP-07008-P(R2), MUAP-07009-P(R0), MUAP-07010-P(R4), MUAP-07011-P(R3), and MUAP-07013-P(R2) conclude that the described methods, models, and analysis codes are acceptable for use to support the US-APWR design certification, within the conditions and limitations that are specified in each SER. We concur with the staff's conclusions and agree that the conditions and limitations provide appropriate constraints for use of the reference analysis methods for the US-APWR design.

For this interim report, we note the following observations and recommendations on selected elements of the design and analyses that are addressed in these chapters.

Chapter 4: Reactor

The US-APWR is required to operate without a significant number of fuel failures in the event of AOOs. Many of these events can affect the entire core. Peak linear heat generation rates of the fuel rods may increase rapidly to levels that are significantly higher than their licensed limits for normal operation. These transients can persist for seconds to minutes until automatic systems or the control room operators detect and terminate the event.

When certain AOOs occur, the fuel temperature increases rapidly. The hotter fuel releases fission products to the gap, thermally expands, and creates significant tensile strains on the inner surface of the zirconium alloy cladding. If the duration of the event is on the order of a few minutes, the fuel cladding may fail by the pellet cladding interaction (PCI) stress corrosion mechanism.

MHI and the staff have evaluated the proposed US-APWR fuel in accordance with current regulatory guidance and methods that have been used for other PWR fuel designs. The analyses show that the fuel is in compliance with the pellet cladding mechanical interaction (PCMI) regulatory limit of less than 1% cladding strain. There are no criteria or explicit guidance in the Standard Review Plan to address PCI failures due to stress corrosion cracking.

There is no indication that the US-APWR fuel is more susceptible to PCI failures than fuel for other new reactor designs or currently operating PWRs. Therefore, PCI failures are not a specific safety concern for the US-APWR.

The PCMI cladding strain limit does not provide a suitable metric for PCI vulnerability. Extensive operational experience, as well as numerous power ramp test programs, have demonstrated that PCI failures can occur at strains on the order of 0.1%, or one-tenth the regulatory PCMI limit. This experience and test data have led us to initiate a separate effort to examine the adequacy of current regulatory guidance, analysis methods, and industry practices to evaluate and manage the risk from the PCI failure mechanism for PWR fuel during AOOs.

Chapter 15: Transient and Accident Analyses

The US-APWR includes several design features to mitigate design basis accidents. For example, the increased fuel length (14 feet) results in an average linear heat generation rate that is 20% lower than that in typical PWRs. The reactor coolant system volume, steam generator heat transfer area, and pressurizer volume per unit of heat generation are also larger than typical PWRs. Emergency core cooling is provided by four advanced accumulators and a four-train high head injection system (HHIS) with a shutoff head of approximately 1,960 psig and a peak flow rate that is nearly three times greater than typical PWR high pressure injection systems. During a large loss of coolant accident (LOCA), the advanced accumulators initially deliver high flow to compensate for coolant loss and then rely on passive flow dampers to deliver water at a smaller flow rate for longer-term makeup.

Calculations performed by MHI and confirmatory calculations by the staff demonstrate that the safety margins for the US-APWR are considerably larger than predicted for typical operating PWRs. Small Break LOCA (SBLOCA), Large Break LOCA (LBLOCA), and non-LOCA accident peak cladding temperatures (PCTs) are estimated to be several hundreds of degrees below the 2200 °F regulatory limit.

In the case of SBLOCAs, the predicted US-APWR response differs from that of typical PWRs. Calculations performed by MHI and the staff indicate that the US-APWR limiting break size is 1 ft^2 (13.5-inch diameter). This break size is considerably larger than the 2- to 4-inch diameter limiting break sizes that are predicted for most PWRs. In addition, calculations performed by MHI indicate that the PCTs occur during the post-blowdown phase of the accident. MHI completed the SBLOCA calculations using their own version of the RELAP5-3D code, M-RELAP5, that has been modified to include model and input assumptions required for Appendix K analyses. MHI validated M-RELAP5 using numerous code-to-data comparisons, which were reviewed and accepted by the staff. Staff audit calculations using the NRC-developed version of RELAP5/MOD3.3 confirmed the limiting SBLOCA size and the magnitude and timing of PCTs. In addition, staff calculations indicated that the larger diameter limiting break size is due to the much larger initial HHIS flow rate.

We concur with these findings and commend the staff for their thorough review of these analyses.

Chapter 19: Probabilistic Risk Assessment and Severe Accident Evaluation

PRA Technical Quality and Use for Risk-Informed Applications

Revision 3 of the US-APWR PRA has not been subjected to a formal independent peer review against the technical attributes in Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and the ASME / ANS Standard for PRA. The current PRA is adequate to meet the regulatory requirements and limited objectives for high-level risk information to support the US-APWR design certification. The PRA confirms that the four-train safety system design, reduced reliance on active equipment, alternative heat removal options, and improved spatial separation and compartmentalization provide substantial risk benefits.

We performed a limited review of selected elements of the PRA models, supporting analyses, and data. That review raised a number of questions about the completeness and level of detail in the PRA models. A more technically refined PRA would provide an improved understanding of the US-APWR risk profile and the relative importance of specific initiating events, structures, systems, components, or personnel actions to the frequency of core damage and large releases.

The staff's SER with open items for DCD Chapter 19 has concluded that the current PRA is not technically adequate to support certain proposed risk-informed applications during plant operation. We concur with that conclusion. According to combined license (COL) Action Item 19.3(1), the design certification PRA will need to be updated (e.g., to address site-specific information) and upgraded to industry standards to achieve the technical adequacy that is required to support those risk-informed applications. The PRA upgrades will be performed by the licensee, after the COL is issued and before initial fuel loading.

Severe Accident Analyses

MHI primarily evaluated US-APWR severe accident progression with the MAAP 4.0.6 code. As needed, additional methods were applied to consider specific severe accident phenomena, such as hydrogen combustion, molten debris spreading, steam explosions, temperature-induced steam generator tube rupture, and direct containment heating. MHI concluded that severe accident phenomena do not challenge the integrity of reactor coolant system piping or the containment.

The staff performed audit calculations primarily using the NRC-developed MELCOR 1.8.6 code. In many cases, the accident progression results predicted by MAAP were in agreement or more conservative than those predicted by the MELCOR calculations. Given the large uncertainties in our understanding of severe accident phenomena, this is acceptable. However, the documentation does not provide an adequate technical explanation for substantial differences between MAAP and MELCOR predictions for the timing of significant severe accident phenomena, such as vessel failure, basemat melt-through, and containment failure.

The analyses performed by the staff identified a potential for hydrogen to collect in the refueling water storage pit (RWSP) compartment. MHI has addressed this by a design change that provides battery backup power for approximately half of the containment hydrogen igniters. Pending final resolution of several outstanding requests for additional information (RAIs), the staff has concluded that containment integrity is likely to be maintained for more than 24 hours after the onset of core damage.

Draft Revision 3 of Standard Review Plan Chapter 19 for passive plant designs indicates that the staff should examine comparisons of MAAP model predictions with values from an NRC-approved thermal-hydraulic code. Such comparisons are also appropriate for non-passive plants, especially for sequences that involve medium or large break LOCAs. For example, in an 8-inch LOCA case, the staff's MELCOR analysis predicts core uncovery in 0.3 hours, while MHI's MAAP analysis predicts core uncovery in 2.26 hours. Since no comparisons with predictions from codes such as RELAP or TRACE were made, it is difficult to have confidence in the values predicted by either the MAAP or MELCOR analyses. Thermal-hydraulic predictions by codes like MAAP or MELCOR are important. These codes provide a basis for the applied success criteria and timing of operator actions in the PRA, and they are the starting point for predicting subsequent severe accident phenomena. We understand that the staff will issue an RAI to ensure that such confirmatory code comparisons are performed for the US-APWR.

Analyses that are described in Chapters 4, 15, and 19 are affected by the design and operation of systems which are discussed in SER chapters that we have not yet reviewed. We will comment on safety implications of any system interactions in future interim letters and in our final report.

Sincerely,

/RA/

J. Sam Armijo
Chairman

REFERENCES

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3. Mitsubishi Heavy Industries, Ltd., MUAP-DC017, Revision 3, Design Control Document for the US-APWR, Chapter 17, "Quality Assurance and Reliability Assurance," (ML110980226), dated March 31, 2011.

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6. NRC Memorandum, Subject: United States – Advanced Pressurized Water Reactor Design Certification Application – Safety Evaluation with Open Items for Chapter 15, "Transient and Accident Analyses," (ML121350457), dated May 21, 2012.

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16. NRC Memorandum, Subject: United States – Advanced Pressurized Water Reactor Advanced Topical Report Safety Evaluation for MUAP-07010, Revision 4, "Non-LOCA Methodology," (ML12248A358), dated September 6, 2012.

17. NRC Memorandum, Subject: United States – Advanced Pressurized Water Reactor Advanced Topical Report Safety Evaluation for MUAP-07013, Revision 2, "Small Break LOCA Methodology for US-APWR," (ML113190428), dated December 9, 2011.

18. NRC Memorandum, Subject: United States – Advanced Pressurized Water Reactor Advanced Topical Report Safety Evaluation for MUAP-07011, Revision 3, "Large Break LOCA Code Applicability Report for US-APWR," (ML120870637), dated April 5, 2012.

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