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

December 24, 2013

Mr. Mark A. Satorius
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: CHAPTERS 6 AND 7 OF THE SAFETY EVALUATION REPORT WITH OPEN ITEMS FOR CERTIFICATION OF THE US-APWR DESIGN AND RELATED LONG-TERM CORE COOLING ISSUES

Dear Mr. Satorius:

During the 610th meeting of the Advisory Committee on Reactor Safeguards, December 4-7, 2013, we met with representatives of the NRC staff and Mitsubishi Heavy Industries, Ltd. (MHI or the applicant) 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 6, "Engineered Safety Features"
- Chapter 7, "Instrumentation and Controls"

We also reviewed elements of the US-APWR design and supporting tests and analyses which address long-term core cooling and resolution of Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance." This letter report contains our interim observations and recommendations on these issues.

Our US-APWR Subcommittee reviewed these matters during meetings on November 30, 2011; September 20, 2012; April 25, 2013; September 17, 2013; and October 1, 2013. Technical aspects of the US-APWR design as well as the open items identified in each of these SER chapters were discussed at those meetings. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The staff should re-examine the technical justification for not installing hydrogen igniters at the apex of the containment dome.

2. The staff should confirm that the US-APWR Emergency Operating Procedures contain unambiguous guidance to ensure that containment pressure is controlled, refueling water storage pit (RWSP) cooling is established, and the full inventory of buffering agent is delivered to the RWSP during a design basis accident.

3. The staff should ensure that sufficient design information is available to provide assurance that watchdog timers will produce the desired reactor protection and engineered safety features actuation failure state signals independently from the Mitsubishi Electric Total Advanced Controller (MELTAC) platform software.

4. Elements of the digital instrumentation and control system design affect the human factors engineering evaluations which are the subject of SER Chapter 18. We will comment on any safety implications from those interfaces in our review of that chapter.

5. Best estimate analyses with explicit consideration of uncertainties should be performed to determine the available net positive suction head (NPSH) for the containment spray / residual heat removal pumps and the high head injection pumps during design basis loss of coolant accident (LOCA) scenarios.

6. The RWSP strainer head loss performance evaluations should explicitly account for uncertainties that are based on experimental data.

7. The core blockage head loss performance evaluations should explicitly account for uncertainties that are based on experimental data.

BACKGROUND

The US-APWR is a four-loop pressurized water reactor 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 Design Control Document (DCD) with its application for the US-APWR design certification on December 31, 2007. Revision 3 of the DCD was submitted on March 31, 2011 and Revision 4 on September 10, 2013.

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 Chapters 6 and 7 of the SER with Open Items for our review. The staff's SER and our review of these chapters address DCD Revision 3 and supplemental material that has since been included in DCD Revision 4.

On May 8, 2008, the Commission issued a Staff Requirements Memorandum stating:

"The ACRS should advise the staff and Commission on the adequacy of the design basis long-term core cooling approach for each new reactor design based, as appropriate, on either its review of the design certification or the first license application referencing the reactor design."

This issue is addressed in Chapter 6 of the SER. As part of our review, we examined elements of the US-APWR design and supporting tests and analyses which address long-term core cooling and resolution of GSI-191. This letter report contains our interim observations and recommendations on these issues.

DISCUSSION

For this interim report, we note the following observations and recommendations on selected elements of the design and analyses that are addressed in SER Chapters 6 and 7.

<u>Chapter 6: Engineered Safety Features</u>

Location of Hydrogen Igniters

The US-APWR contains 20 hydrogen igniters that are distributed among open areas of the containment and in selected subcompartments where hydrogen may be released or collected. The igniters are energized automatically by the emergency core cooling system actuation signal. Nine igniters are AC-powered, and eleven igniters are powered from a dedicated DC supply with a rated battery life of 24 hours.

Igniters are located inside upper areas of each steam generator compartment and the pressurizer compartment. However, no igniters are located at the apex of the containment dome. It is not apparent that the current distribution of igniters will ensure that hydrogen cannot migrate through open areas in the containment and collect in the dome region. The staff should re-examine the technical justification for not installing hydrogen igniters in the containment dome.

Containment Spray Operating Time

The design basis accident analyses are performed with the assumption that one train of emergency core cooling system equipment is out of service for maintenance, and a second train is disabled by failure of its emergency power supply. These conditions leave two trains of the containment spray system (CSS) available for event mitigation. Two CSS trains provide sufficient spray flow to maintain containment pressure below its design value of 68 psig during the most limiting design basis accident.

The US-APWR design provides a buffering agent to neutralize the high concentration of boric acid in the RWSP and thereby reduce the potential for corrosion during long-term cooling scenarios. The sodium tetraborate decahydrate (NaTB) buffering agent is stored in baskets that are located around the periphery of the containment. A portion of the containment spray flow is directed through the baskets, dissolving the NaTB and delivering it to the RWSP. The design basis accident analyses indicate that approximately 12 hours of spray flow is required to ensure that the NaTB is completely dissolved and that RWSP water reaches the desired post-accident pH conditions.

The applicant indicated that the draft US-APWR Emergency Response Guidelines instruct the operators to stop spray flow and to realign the CSS pumps for RWSP cooling when containment pressure is reduced below the spray actuation setpoint. Operation of two CSS trains will reduce containment pressure below the spray actuation setpoint several hours before the 12-hour time to fully dissolve the NaTB additive. Therefore, it is not apparent how the operators will be instructed to control CSS operation in these scenarios to ensure that containment pressure is controlled, RWSP cooling is established, and the full inventory of NaTB additive is delivered to the RWSP. The staff should confirm that the Emergency Operating Procedures contain adequate guidance to ensure that these functions are accomplished without ambiguity.

Chapter 7: Instrumentation and Controls

Deterministic Generation of Protection System Failure State Signals

The applicant has proposed changes to the DCD and Technical Report MUAP-07005, "Safety System Digital Platform - MELTAC," which clarify the configuration and design functions of watchdog timers in several modules of the integrated protection and safety monitoring system (PSMS). The watchdog timers provide hardware-based detection of an interruption in the cyclical execution of each module's signal processing or data communications functions. Their intended purpose is to ensure that an appropriate Failure state safety signal is generated, independently of malfunctions in the processing system hardware or the platform software. The reactor protection system (RPS) Failure state is a reactor trip signal. To avoid an undesired spurious safeguards actuation, the engineered safety features actuation system (ESFAS) Failure state is "as-is."

For example, the applicant states that timeout of a Central Processing Unit (CPU) Module watchdog timer will generate Failure state output signals from that module. It will also stop updates of the module's Alive Counter status signals. Termination of the Alive Counter updates will cause the other PSMS divisions to treat the affected CPU Module as being in its Failure state. These actions are processed by the CPU basic diagnostic software, which may not be operational if the CPU has locked up. Thus, watchdog timer actuation of the RPS and ESFAS Failure states on timeout appears to depend on the MELTAC platform software.

The summary descriptions note that the Failure state signals are activated by a "hardware mechanism." However, the available information does not describe how the watchdog timers achieve the desired signal states independently from the MELTAC platform software. For example, the Failure state signals should be generated, even if corrupt data cause the affected CPU Module software to lock up. The staff should ensure that sufficient design details (e.g., descriptions and simplified diagrams) are available to provide assurance that the watchdog timers will produce the desired RPS and ESFAS Failure state signals independently from the MELTAC platform software.

Interface with Human Factors Engineering Evaluations

Elements of the digital instrumentation and control system design affect the human factors engineering evaluations which are the subject of SER Chapter 18. We will comment on any safety implications from those interfaces in our review of that chapter.

<u>Issues Related to Long-Term Core Cooling and Resolution of GSI-191</u>

Net Positive Suction Head for Emergency Core Cooling System Pumps

The US-APWR design relies on positive pressure in the containment to maintain adequate NPSH for operation of the containment spray / residual heat removal (CS/RHR) pumps and the high head injection system (HHIS) pumps that take suction from the RWSP. The CS/RHR pumps have a design required NPSH of approximately 8.5 psi. The design required NPSH for the HHIS pumps is slightly lower at approximately 8.2 psi. Accounting for losses through the RWSP strainers and the pump suction piping, the static head of water at minimum level in the RWSP provides an available NPSH of approximately 9.1 psi, or a margin of approximately 0.6 psi for the CS/RHR pumps.

The US-APWR design basis accident analyses conclude that the RWSP reaches a maximum temperature of 256 °F during the limiting large break LOCA event. Temperature remains above 212 °F for approximately 10 hours during this scenario, based on the assumed analysis conditions with only two trains of operating CS/RHR and HHIS pumps. At the maximum RWSP temperature of 256 °F, the NPSH deficit for the CS/RHR pumps is approximately 18 psi, without credit for the containment accident pressure.

The analyses account only for the available NPSH that is afforded by the RWSP saturation vapor pressure. That pressure is less than the calculated containment pressure throughout the progression of the analyzed accident scenario. Therefore, the applicant concludes that additional containment pressure margin is always available to prevent pump cavitation. The analyses are performed according to design basis assumptions that maximize energy input into the containment and the RWSP, and account for only limited leakage from the containment. The applicant has stated that no design changes are justified to reduce the required credit for containment pressure to maintain adequate pump NPSH.

We have advised against reliance on containment accident pressure credit to ensure that adequate NPSH is maintained for emergency core cooling pumps, noting that this practice jeopardizes a fundamental principle of defense in depth. Conditions such as unexpected containment leakage or actuation of the containment fan coolers could cause loss of NPSH for the HHIS pumps and the CS/RHR pumps, with consequential loss of cooling for the reactor fuel coupled with the inability to control containment pressure if core damage occurs.

For currently operating reactors, we have noted that best estimate analyses should be performed to determine the amount of containment accident pressure that is needed to maintain adequate NPSH and the duration for which that pressure is needed. We have also noted that plant-specific probabilistic risk assessment (PRA) models may be used to provide risk-informed assurance that the frequency of scenarios which require containment accident pressure is acceptably low, and the uncertainties in those scenarios are understood. The current US-APWR design-level PRA does not satisfy the scope, level of detail, or technical quality attributes to directly support this type of risk-informed evaluation.

Best estimate analyses should be performed for the limiting large break LOCA event to determine the amount of containment accident pressure that is needed to maintain adequate NPSH and the duration for which that pressure is needed. The analyses should account for conditions with two, three, and all four trains of the CS/RHR pumps and HHIS pumps operating to provide better information about the ranges of margin that are available during these scenarios. In all cases, the comparison of available NPSH with required NPSH should include explicit consideration of the uncertainties in both parameters.

RWSP Strainer Blockage

To substantially reduce the amount of LOCA-generated debris, the US-APWR uses reflective metal insulation (RMI) as the primary insulation material inside a 'clean' containment, in which latent debris is limited to small amounts by administrative controls. Jets that emanate from large pipe breaks would impinge on RMI surfaces which do not fragment into fine debris. Fibrous insulation is excluded from the zone of influence of LOCA jets, and only qualified coatings are used. These measures, the NaTB buffering agent, and restrictions on the aluminum surface area exposed to post-LOCA conditions contribute to substantial reductions in the amounts of fibrous material, particulate debris, and chemicals which increase head losses in strainers and downstream components. The design is based on a total LOCA debris loading of 106 ft^3 of RMI and 3 ft^3 of coating materials that are dislodged by jet impingement, 30 lbm of latent fibrous material, 170 lbm of latent particulates, 300 lbm of aluminum hydroxide and 330 lbm of sodium aluminum silicate from chemical reactions, and other miscellaneous debris in quantities sufficient to block 200 ft^2 of the surface area of each strainer. The strainer testing program also accounted for approximately 2 ft^3 of additional fibrous debris and 200 lbm of additional coating debris to demonstrate extra operational margin.

The RWSP strainers contain stacks of perforated disks with holes that are 0.066 inch in diameter. The total surface area of each strainer is approximately 2750 ft^2. Design basis emergency core cooling and containment spray functions can be accomplished by a minimum of two pump trains. The applicant's analyses allocate approximately 85% of the total containment debris to one RWSP strainer, based on an assumed flow distribution with only two operating pumps.

The large strainer area and 0.066-inch hole size are designed to accommodate the relatively low NPSH available for the CS/RHR pumps and HHIS pumps. As a consequence, the head losses appear to be quite low, as measured in a limited number of prototypical tests. However, such large area, low resistance strainers allow fine fibers and particulates to pass through, with the amount of such "bypass" being approximately 30-50% for fibers.

The strainer head loss tests were performed according to staff guidance. The experiments attempted to produce conservatively high head losses. For example, they included more chemical precipitates than required by NRC accepted methods. No chemical precipitates were assumed to form above 150 °F, which is consistent with calculations performed by the applicant and the staff. The test results indicate acceptable strainer head losses, provided that the latent debris in containment meets the design specifications. However, few tests were performed and uncertainties were not estimated. Such uncertainties can arise due to several factors, notably the effects of fiber length and its distributions, size and characteristics of materials used as surrogates for coating debris and latent particulates, and the test protocols, for example the order and timing of various debris and chemical additions. Further, the effect of approach velocity, which can also be uncertain, was not evaluated.

In view of this, we recommend that a decision about acceptability of the RWSP strainer head loss performance be deferred until uncertainty estimates are provided, preferably on the basis of experiments, as no model for strainer head loss has been accepted.

Downstream Effects

With regard to debris effects downstream of the strainers, a significant consideration is that the strainer design allows 30-50% of the fiber to pass through. This "bypass" will be affected by fiber length and its distribution, but no estimate of such effects has been presented. Nonetheless, if 50% bypass is used as an estimate, the resulting fibrous debris loading is well above the staff-approved 15 g per fuel assembly (FA) value for hot leg breaks evaluated by the PWR Owners' Group (PWROG).

With regard to in-vessel effects, the primary concerns are blockage of flow through the FAs due to debris caught at the inlet and in the intermediate grids, and effects on the fuel due to precipitation of chemicals and boron. To address the core blockage issue, the applicant performed a limited number of tests in a facility in Takasago, Japan, where a one-third length FA could be accommodated.

For the hot leg break tests, a forward flow of 22.8 gpm per FA was maintained with 50% of the containment fiber load per FA, consistent with the maximum 50% bypass observed for the strainers. For the cold leg break tests, the forward flow was 3.5 gpm per FA, which corresponds to the estimated boiloff rate at 850 seconds into the event. The fiber load in the cold leg tests was 30% of the containment fiber load per FA, assuming that 60% of the injection flow reaches the core inlet. Finally, tests were done for cold leg breaks after switchover to hot leg injection, with a reverse flow of 11.4 gpm and the same fiber load as for the hot leg tests. For the hot leg break experiments, which were found to be limiting in terms of head loss, a single fiber load was tested, based on 50% bypass. This load was well above the staff approved value of 15 g per FA. Three particle-to-fiber ratios, ranging up to the maximum estimated particulate loading, were tested for this fiber load. For each of the three test scenarios (i.e., hot leg, cold leg, and switchover to hot leg injection), the pressure losses were measured to be below the available pressure head needed to maintain acceptable core cooling flow. While the tests included three particle-to-fiber ratios, the sensitivity of the results to no other parameter was examined. As a consequence, uncertainty in the results could not be determined. It is known from previous experiments that results are affected markedly by the test loop configuration, nature and characteristics of the fibers used, the amount and timing of chemical precipitates added, fluid velocity and temperature, and the source of water used.

These factors introduce uncertainties that have led the staff to limit fiber loading to 15 g per FA for currently operating PWROG reactors. This value is substantially below the values used in the US-APWR application. With regard to new reactors, a value slightly higher than the staff limit, but still substantially below that in the US-APWR application, was permitted for one applicant, but only after extensive testing to evaluate uncertainties due to flow rates, fiber characteristics, fiber loading, chemical loading, and testing protocol.

In summary, the US-APWR fiber loading per FA is substantially higher than the current staff-permitted limit for operating PWRs and is also well above the fiber loading permitted for another new reactor applicant. In view of this, we recommend that a decision about acceptability of the core blockage head loss performance be deferred until uncertainty estimates based on experimental data are provided.

With regard to the core flow rate required for cold leg breaks, the tests and analyses should account for boron precipitation downstream from the blockage. A detailed analysis of cooling performance under reduced flow conditions has been performed by another new reactor applicant. In general, the core flow rates must be well above the boiloff limits to ensure that boron does not precipitate on the fuel. In the US-APWR in-vessel effects experiments, the cold leg break flow appears to have been set on the basis of boiloff. The test flow may need to be increased above this value to ensure that boron does not precipitate, adding to the need to understand how uncertainties in the flow affect core head losses related to debris blockage.

Sincerely,

/RA/

J. Sam Armijo
Chairman

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