CONCLUSIONS AND RECOMMENDATIONS

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

   The U.S. Nuclear Regulatory Commission (NRC) staff acknowledges the importance of the locations of the 20 hydrogen igniters located throughout the containment in controlling the hydrogen concentration below the regulatory limit, thereby protecting containment integrity following a beyond design-basis accident (DBA).

   The igniters have been located near the likely hydrogen release points, such as the pressurizer relief tank rupture disk, in the dome above each steam generator, in the dome above the pressurizer room, in the dome above the exhaust piping for the severe accident depressurization valve, and throughout the containment.

   The United States-Advanced Pressurized Water Reactor (US-APWR) applicant, Mitsubishi Heavy Industries, Ltd. (MHI), has evaluated the atmospheric mixing and the potential for deflagration to detonation transition (DDT) of hydrogen in the containment. Its analysis paid particular attention to the dome volume, modeling it with 150 nodes. The NRC staff found that the applicant's analysis shows, for all significant beyond DBAs, hydrogen concentrations throughout the containment are maintained below 10 percent by volume, below the DDT level which could lead to a loss of containment integrity.

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.*

   As described in Design Control Document (DCD) Section 6.3.2.2.5, "NaTB Baskets and NaTB Basket Containers," crystalline sodium tetraborate decahydrate (NaTB) is stored in containment and is used to raise the refueling water storage pit (RWSP) pH from 4.3 to at least 7.0 following a loss-of-coolant accident (LOCA). Achieving a pH of 7.0 or greater is important to avoid stress-corrosion cracking and reduce hydrogen generation. DCD Section 6.3.2.2.5, states the dissolution time for the NaTB is approximately 12 hours, implying that containment sprays should run for 12 hours to achieve a pH of at least 7.0. In DCD Section 6.5.2.1, "Design Bases," the following statement is made: "When the containment pressure is reduced sufficiently and the operator determines that containment spray is no longer required, the operator terminates containment spray." At the July 9, 2013, Advisory Committee on Reactor Safeguards (ACRS) Chapter 15, "Transient and Accident Analysis," Phase 3, subcommittee meeting, members raised a concern that operators could terminate containment spray based solely on containment pressure and hence terminate containment sprays prior to 12 hours.

Furthermore, in DCD Section 6.2.2.2, "System Design," a similar statement is made: "Following a DBA, the containment pressure approaches atmospheric pressure. When the containment pressure is reduced sufficiently and the operator determines that containment spray is no longer required, the operator terminates containment spray. The operator closes the containment spray header isolation valves and aligns system flow through the CS/RHR heat exchanger back to the RWSP through the full flow test line. The pit water is then recirculated and cooled."

The NRC staff agrees with the ACRS's concern that the Emergency Operating Procedures guidance should be clear and include both containment pressure and an elapsed time of 12 hours as conditions for operator actions terminating containment spray and realigning the containment spray system for RWSP cooling. As such, the staff plans to issue a request for additional information (RAI) against Chapter 6, "Engineered Safety Features," requesting that the applicant clarify conditions for containment spray termination and RWSP cooling. This is expected to result in a modification to the DCD and a revision to the staff's Chapter 6 safety evaluation.

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.*

The NRC staff is working with the applicant to resolve this recommendation concerning the watchdog timers (WDT). During its review of Chapter 7, "Instrumentation and Controls," of the DCD and referenced material, audits of undocketed information, and interactions with the applicant, the staff gained a clear understanding of the WDT operation. The staff determined that the WDTs will provide the desired reactor protection and engineered safety features actuation failure state signals independently from the Mitsubishi Electric Total Advanced Controller platform software. However, upon receipt of your letter, the staff found that additional clarification can be made in the docketed material to better reflect this aspect of the WDT operation. The staff has initiated discussion with the applicant to address this issue and will keep ACRS informed of its efforts.

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.*

The NRC staff acknowledges that comments regarding the safety implications of instrumentation and controls system design on human factors engineering (HFE) evaluations will be withheld until the Chapter 18, "HFE," safety evaluation is reviewed.

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.*

The NRC staff acknowledges that best estimate analyses could be performed but finds that available net positive suction head (NPSH) is adequately addressed, which is consistent with the staff's guidance and the Commission direction. This is discussed in more detail below.

The NRC staff and ACRS agree that the applicant's NPSH evaluation uses a portion of the positive pressure in containment, that is, the portion which corresponds to the sump's saturation vapor pressure, to calculate the available NPSH for spray and injection pumps. The staff and ACRS also agree that some NSPH evaluations should use best estimate analyses to determine the amount of containment accident pressure that is needed to maintain adequate NPSH for spray and injection pumps.

On January 31, 2011, the NRC staff issued Commission Paper SECY-11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," to address the issues related to the use of containment accident pressure (CAP). On March 15, 2011, the Commission selected Option 1 of SECY-11-0014 and endorsed the staff's recommended approach on the use of CAP.

The NRC staff's approach and guidance regarding the use of CAP indicates that the use of CAP beyond the saturation vapor pressure region (up to the total pressure available) should include best estimate analyses, whereas use of CAP within the saturation vapor pressure region (limited portion of the pressure available) is considered conservative and therefore, is not in need of best estimate analyses to assess adequate margin.

The NRC staff's safety evaluation finds that the US-APWR design satisfies the staff's CAP guidance. This conclusion is based upon the applicant limiting the use of CAP to within the vapor pressure region, maximizing pump flow, satisfying the zone of maximum erosion criteria (NPSH available between 1.2 and 1.6), and accounting for uncertainties associated with NPSH required (e.g., pump speed, suction piping variation, gas evolution).

Therefore, given that the US-APWR NPSH evaluation is consistent with the NRC staff's guidance and the Commission direction, the staff, for the reasons set forth in the preceding paragraphs, concludes that the US-APWR NPSH evaluation satisfies regulatory requirements for long-term cooling.

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

The NRC staff acknowledges the uncertainties but finds that they are adequately addressed in the performance evaluations. This is discussed in more detail below.

The NRC staff and ACRS agree that the US-APWR design substantially reduces the types and amounts of LOCA-generated debris and is considered a 'clean' containment, in which latent debris is limited to small amounts by administrative controls. The US-APWR assumes a relatively large amount of latent debris compared to what is likely to be in containment, based upon current operating experience, and provides for strict control of the debris quantities. Further, changes to these limits require prior NRC review and approval.

The NRC staff and ACRS also agree that the applicant's strainer head loss tests were performed according to staff guidance. The ACRS notes that the test results indicate acceptable strainer head losses provided that the latent debris in containment meets the specifications. The staff finds the strainer to have sufficient head loss margin between the test results and the design basis.

Although the ACRS and the NRC staff find that the applicant assessed strainer performance consistent with staff guidance and the test results indicate acceptable strainer head losses, the ACRS recommends deferring the decision about acceptability of the strainer head loss performance until uncertainty estimates are provided, preferably on the basis of experiments. The areas of uncertainty identified by the ACRS are associated with debris characteristics (e.g., latent fiber and coatings) and test protocols.

The NRC staff agrees with the ACRS that uncertainty does exist with strainer performance evaluations. However, the staff concludes that satisfying the current evaluation criteria for strainer testing results in a strainer head loss that is the maximum that could occur given the design-specific conditions and balances, known conservatisms against potential non-conservatisms and uncertainties, including those associated with debris characteristics and test protocols.

The NRC staff's safety evaluation concludes that the US-APWR design satisfies the strainer evaluation criteria and, therefore, balances known conservatisms against potential non-conservatisms and uncertainties. This conclusion is based upon the large size of the installed strainer screens, limited fiber inventory (latent only), no problematic insulation installed in the zone of influence, 100 percent debris transport to the strainer, calculated filtering bed thicknesses that are less than $1/16^{th}$ of an inch, completion of design specific head loss testing consistent with staff guidance, and the requirement for establishing a robust containment cleanliness program.

The NRC staff also concludes that the US-APWR strainer evaluation readily satisfies the Option 1 closure path that was approved by the Commission in Staff Requirements Memorandum (SRM) - SECY-12-0093, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated December 14, 2012.

Therefore, given that the US-APWR strainer performance evaluation is consistent with NRC staff guidance and a Commission approved strainer performance resolution path, the staff, for the reasons set forth in the preceding paragraphs, concludes that the US-APWR strainer performance satisfies regulatory requirements for long-term cooling.

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

The NRC staff agrees with the ACRS that the strainer bypass testing performed by the applicant allows for up to [ ] bypass of the fibrous debris and that this value is particularly of note since it determines the fibrous debris inputs used in subsequent in-vessel downstream effects tests. This value was determined by fiber-only strainer bypass testing performed by the applicant using shredded NUKON fiber as the surrogate fiber source. The use of the shredded fiber source term helps to maximize conservatively the bypass since the larger fibers were shown during the applicant's strainer testing to be caught more readily by the strainer. While the selection of fiber lengths can have an impact on the overall results, the staff found that the debris generation and distribution was conservative and similar to that used in other vendor testing.

Additionally, the NRC staff found that the overall test plan, "Sump Strainer Test Plan for Fiber Only Bypass Test and Head Loss Test," (Agencywide Documents Access and Management System (ADAMS) Accession Number ML11160A107) for the fiber-only bypass testing contains several conservatisms. Three notable conservatisms used by the applicant to maximize the fiber bypass fraction were: (1) the absence of particulate and chemical debris that would normally be present, thereby reducing the fiber bypass by developing a more restrictive debris bed, (2) introduction of the fiber debris in smaller batches to increase fiber debris bypass, and (3) the assumption of no settling during transport to the sump strainer.

The largest measured bypass, [ ], was obtained during the first batch loading which contained only 12.5 percent of the debris load for one sump strainer. This batch size represented a loading [ ]. Subsequent batch loadings represent debris allocations with either [ ] and resulted in [ ].

Another conservatism used in the US-APWR Generic Safety Issue (GSI)-191 methodology is the assumption that all fibrous debris stays suspended with no settling occurring. This is a conservative assumption for the US-APWR design because of the rather complicated flow path for coolant as compared with other designs. The coolant must flow over berms, through multiple large cavities, and through overflow pipes before reaching the sump strainer (see Figure E-1, "Schematic of Recirculation Water Return Path," of MUAP-08013, "US-APWR Sump Strainer Downstream Effects," Revision 5, ADAMS Accession Number ML13228A301). Although potential settling is not quantified or credited, the NRC staff recognizes this as a conservatism.

Based on the fiber preparation methodology being similar to past precedents and the conservative assumptions used in the fiber-only bypass testing, the NRC staff concluded that the applicant justified the [ ] fiber bypass as a conservative source term for fuel assembly testing.

Results of 25 fuel assembly tests were submitted in support of the US-APWR GSI-191 in-vessel downstream effects analysis, representing three separate test programs (MUAP-10021-P, "US-APWR Core Inlet Blockage Test,“ MUAP-11022-P, "US-APWR Additional Core Inlet Blockage Test," and MUAP-12004-P, "US-APWR Core Inlet Blockage Test for Test Conditions with Design Changes in Recirculation Water Flow Path to Refueling Water Storage Pit"). The test matrices included a total of three repeatability tests. The NRC staff audited the first two test programs in Takasago, Japan, and conducted a quality assurance inspection of the second test program. Although the third test program best represented the final system design in regard to coolant flow path and source terms, the staff was able to use the entire docketed test input to draw conclusions regarding the adequacy of the test procedure, the test equipment, and to some extent the repeatability of the tests. During the second test program, two different repeatability tests were performed and the results were within three percent differential pressure of each other. The staff observed that the applicant had a robust system that relied on computer controls and frequent feedback of the flow rates (measurements and adjustments each second). Additionally, the third test program hot leg break tests were performed with the same conditions as the second test program, except for the chemical debris source term. Therefore, the staff was able to compare the pressure measurements before the addition of the chemical source term and confirmed that there was good agreement between the two test programs. The staff concluded that the test facility and program were capable of reproducing consistent results. While the repeatability tests do not provide a detailed uncertainty analysis, they do provide an indication for the uncertainty magnitude when compared with available margin. The most limiting result (hot leg break scenario) presented by the applicant in MUAP-12004-P (ADAMS Accession Number ML13228A303), is [ ] of the acceptance criterion. Based on the staff's review of the robust, computer controlled test facility, the repeatability of the test data, and the available margin, the staff concluded that no extensive uncertainty analysis was necessary to reach a reasonable assurance of adequate safety.

The NRC staff and ACRS agree that MHI did not perform extensive studies of particulate to fiber (p:f) ratios in its testing program, but the staff's review was informed by information available from the Pressurized Water Reactor Owner Group (PWROG) test program. The staff found that the p:f range was broad and included the more extreme scenarios (i.e., scenarios in which all and nearly none of the available particulate source term was included). The results did not indicate a trend that would lead the staff to request additional testing in specific areas.

ACRS raised questions regarding a need to account for boron precipitation. At the time of the Chapter 6 ACRS subcommittee meeting, the NRC staff's safety evaluation contained an open item related to boron precipitation. This open item was tied to a similar open item in Chapter 15. The staff has recently reviewed the analysis provided by the applicant regarding boron precipitation in response to RAI 861-6062, Question 15.6.5-100, associated with the Chapter 15 open item. The staff is in the process of revising the SER to include its review of boron precipitation, and closing out the open items for both Chapters 15 and 6.

The results of the applicant tests indicate that the hot leg test performed at a p:f ratio of [ ] was the most limiting. This limiting differential pressure value was still [ ] of the acceptance criterion. The NRC staff and the ACRS noted that this is well below the results seen by other designs. The staff concludes that a major factor in the applicant's ability to meet the acceptance criteria with fiber source terms higher than 15 grams is the large difference in flow rates between the US-APWR and other large PWR designs. The flow rates for US-APWR are roughly half of those for the PWROG. The importance of this can be seen in the flow sensitivities presented by the applicant in Figure 7.2-1, "Calculated differential pressure vs. Particle/Fiber ratio on CL Tests," of MUAP-11022-P (ADAMS Accession Number ML13228A296). The cold leg test number 4 was repeated with flow rates of [ ] of the design flow rate. The resultant differential pressure is shown to increase with increasing flow rates. The staff concludes that this relationship to flow rates helps explain why US-APWR testing successfully demonstrated that no acceptance criteria would be exceeded even with a fiber source term greater than 15 grams.

Based on multiple conservative methodology assumptions, the NRC staff's observations of the test facility and procedure quality during the audits, and the abundance of margin, the staff concludes that there is reasonable assurance of adequate protection related to the US-APWR GSI-191 in-vessel downstream effects testing and no further testing is necessary. Therefore the staff, for the reasons set forth in the preceding paragraphs, concludes that the US-APWR GSI-191 in-vessel effects performance satisfies regulatory requirements for long-term cooling.