May 1, 2007

LICENSEE: MITSUBISHI HEAVY INDUSTRIES (MHI)

FACILITY: US-APWR STANDARD DESIGN PRE-APPLICATION REVIEW

SUBJECT: SUMMARY OF THE MARCH 22, 2007 PUBLIC MEETING ON ENGINEERED SAFETY FEATURES (ESF), SMALL-BREAK LOSS-OF-COOLANT ACCIDENT (SBLOCA), AND CONTAINMENT RESPONSE ANALYSIS METHODOLOGIES FOR THE US-APWR

On March 22, 2007, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of Mitsubishi Heavy Industries (MHI) at NRC Headquarters in Rockville, Maryland. The purpose of the meeting was to discuss the engineered safety features (ESF), small-break loss-of-coolant accident (SBLOCA), and containment response analysis methodologies for the US-APWR [Advanced Pressurized Water Reactor] design certification application. MHI announced its intention to submit a design certification application for the US-APWR in December 2007. A list of meeting attendees is provided as Enclosure 1. MHI presented handouts that are shown in Enclosures 2, 3, and 4 and can be assessed through the Agencywide Documents Access and Management System (ADAMS) accession numbers ML070820518, ML070820439, and ML070820447, respectively.

ESF Analysis

MHI opened the meeting by discussing the five systems that are part of the ESF: the containment system, the emergency core cooling system (ECCS), the habitablility system, the fission product removal and control system, and the emergency feedwater (EFW) system.

MHI stated that the US-APWR containment functional design includes a containment vessel that is designed to minimize leakage and meets NRC requirements for radioactive releases. The design pressure and temperature of the containment vessel are defined by the loss-of-coolant (LOCA) and the main steam line break (MSLB). The containment vessel consists of a pre-stressed, post-tensioned concrete structure and a reinforced concrete foundation slab. The inside surface of the containment vessel is lined with carbon steel. MHI noted that this type of containment vessel is already adopted by many NRC licensed pressurized-water reactor (PWR) plants.

The US-APWR containment heat removal system consists of four independent trains with four containment spray (CS) and residual heat removal (RHR) pumps, and four CS and RHR heat exchangers. The CS contains a spray ring header that is composed of four concentric, interconnected rings. The purpose of the CS is to reduce pressure and temperature in containment to acceptable levels and provide long term containment cooling following a LOCA. MHI then discussed the various operating scenarios for these systems during power and shutdown operations.

MHI stated that the US-APWR containment isolation system provides appropriate isolation of lines penetrating the containment to prevent the release of radioactive products to the environment following a LOCA. This system is designed to provide an automatic isolation of lines penetrating the containment on a containment isolation signal, such as low reactor pressure. There is no guard pipe for the containment sump isolation valve and containment sump discharge line. MHI stated that a containment sump isolation valve guard pipe design is not required because the design of this line can meet the requirements set forth in NUREG-0800, "Standard Review Plan (SRP) for Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 6.2.4. MHI notified the NRC staff that current U.S. Pressurized Water Reactors (PWRs), such as McGuire 1 and 2, do not have a containment sump isolation valve guard pipe design.

The second US-APWR ESF that MHI discussed was the ECCS. The purpose of this system includes providing core cooling during a LOCA, performing emergency boration to mitigate accidents, and performing emergency boration and letdown and emergency makeup to assure safe shutdown. This system consists of four independent ECCS trains that take suction from the refueling water storage pit (RWSP). Each train has a safety injection pump (SIP) that is capable of injecting water both directly into the reactor vessel and into the hot leg, and an advanced accumulator that injects water into the cold leg. The RWSP is located inside containment at the lowest part of the containment vessel and provides a continuous suction source for both the SIPs and the CS and RHR pumps. The advanced accumulators integrate the function of low head safety injection into their design, and the long accumulator injection time allows more time for the SIPs to start. MHI informed the NRC staff that the ECCS design also implements several design features from Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance," such as, locating the RWSP over the four containment sumps, with passive sump screens installed over the sump areas. This increases the net positive suction head and provides sufficient surface area for the strainer. Additional design features related to GSI-191 include having the appropriate insulation material and pH control buffer.

The third US-APWR ESF that MHI discussed was the habitability system. This system is designed to protect the operators from airborne radioactivity, smoke, and toxic gas by maintaining suitable conditions in the main control room (MCR) envelope. The MCR envelope consists of the MCR, the technical support center (TSC), kitchen, toilet, etc. The habitability system has a MCR and TSC HVAC system that consists of four MCR and TSC air handling units, two MCR and TSC emergency supply filter units and fans, and airtight isolation dampers. The habitability system automatically transfers to emergency mode on a safety injection signal or a high radiation signal, and can also be manually transferred to recirculation mode when smoke is detected in the air intake line.

The fourth US-APWR ESF that MHI discussed was the fission product removal and control system, which consists of pH control and annulus air cleanup system. For pH control, an appropriate buffer agent is added to provide the proper sump water pH adjustment following a LOCA. The annulus air cleanup system prevents uncontrolled radioactive release from the containment penetrations and safeguard components to the environment. This system has two annulus air cleanup filter units, which maintain the annulus area and safeguard component area at a negative pressure, during accident conditions. This system is also used for containment depressurization during normal operation. The annulus air cleanup system automatically initiates on a safety injection signal.

The fifth US-APWR ESF that MHI discussed was the EFW. The purpose of the EFW is to maintain the capability of the steam generators to remove reactor coolant system heat. The EFW can supply power to its pumps via diverse power supplies, and is automatically initiated during transients. Whenever a MSLB, feedline break, or steam generator tube rupture occurs, the EFW is automatically isolated from the faulty steam generator.

SBLOCA Methodology

MHI discussed the SBLOCA methodology, and its plans for submitting a topical report by the end of July 2007. MHI stated that it will use one-dimensional modeling with the M-RELAP5 code, a modified version of RELAP5-3D, for the SBLOCA analysis. This code has multi-dimensional thermal hydraulic and kinetic modeling capability and is equivalent to RELAP5/MOD3.2. It will be applied to SBLOCA analysis with a break size of less than 1.0 ft^2 and will incorporate 10 CFR 50.46 and 10 CFR Part 50, Appendix K requirements.

MHI also discussed phenomena identification and ranking tables. (These tables helped to identify each phenomena that needs to be taken into consideration.) MHI discussed how it accounted for each phenomena within the M-RELAP5 code. MHI will incorporate empirical correlations into the M-RELAP5 code in order to correctly model the advanced accumulator characteristics. Condensation phenomena in the downcomer can be affected due to direct vessel injection. MHI noted that the M-RELAP5 code is capable of adequately modeling the direct vessel injection region with a “Branch Component” feature. The NRC staff informed MHI that the safety injection water temperature will rise following a break due to the rise in RWSP water temperature. However, MHI noted that this temperature increase would be small and M-RELAP5 is capable of modeling safety injection water temperature as the boundary condition. Metal heat release and bypass flow within the neutron reflector can affect the mixture level in the core. MHI will address this phenomena by modeling the neutron reflector as a separate channel with heat structure in the M-RELAP5 code.

The NRC staff questioned MHI on the volume of the US-APWR reactor coolant system. MHI responded that the volume was about 12,000 cubic feet. The NRC staff notified MHI that it had performed a SBLOCA analysis, using the code, scaling, applicability, and uncertainty (CSUA) methodology and RELAP5 code. The NRC staff referred MHI to NUREG-CR-5818, “Uncertainty Analysis of Minimum Vessel Liquid Inventory During a SBLOCA in a Babcock and Wilcox Plant - an Application of the CSUA methodology using the RELAP5/MOD3 Computer Code,” dated December 31, 1992, ADAMS Accession No. 9302230168. In addition, the NRC staff informed MHI of a letter that was sent to Westinghouse regarding an error that was found in a post-LOCA long term cooling model. During an audit of this post-LOCA long term cooling model, the NRC staff identified an error regarding the build-up of boric acid following a large-break LOCA. After the meeting, MHI was provided a copy of the NRC staff’s letter to Westinghouse, dated August 1, 2005, ADAMS accession number ML051920310.

Containment Response Analysis

The last presentation given by MHI was pertaining to the containment response analysis for the US-APWR. MHI stated that the GOTHIC 7.2a code will be used for this analysis. MHI also noted that this code has been previously approved by the NRC staff for U.S. licensing submittals. MHI then discussed some features of the GOTHIC 7.2a code and also summarized the US-APWR plant parameters related to the containment response analysis.

The containment design function evaluation (CDFE) will be performed for peak pressure, peak temperature, and pressure at 24 hours, assuming LOCA and MSLB accident scenarios. These evaluations will be based on SRP Chapters 6.2.1, 6.2.1.1A, and NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.” The containment geometric modeling of the CDFE consists of either a single or two volume system depending on results from a sensitivity study, and assumes minimum free volume and minimum heat sinks. Mass and energy release calculations of the CDFE will be performed using the SATAN code and the WREFLOOD code for the LOCA scenario, and MARVEL for the MSLB accident scenario. For the heat and mass transfer between the containment atmosphere and heat sinks of the CDFE, MHI will use the Diffusion Layer Model (DLM), which has already been used in previous NRC approved licensing submittals. MHI also noted that the DLM will be validated against separate effects test data. For the heat sink modeling of the CDFE, MHI presented a list of structures that would be modeled as containment internal heat sinks, such as the containment shell (liner and concrete), uninsulated pipes and supports, and internal concrete. MHI stated that the containment design evaluation methodology and analysis results will be provided in the design control document.

Members of the public were in attendance, but Public Meeting Feedback forms were not received. Please direct any inquiries to me at 301-415-1544, or srm2@nrc.gov.

/RA/

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Project No. 751

Enclosures:
1. List of Attendees
2. Mitsubishi Handout - US-APWR
   6th Pre-Application Review Meeting -
   Engineered Safety Features
   (ML070820518)
3. Mitsubishi Handout - US-APWR
   6th Pre-Application Review Meeting -
   Small Break LOCA Methodology
   (ML070820439)
4. Mitsubishi Handout - US-APWR
   6th Pre-Application Review Meeting -
   Containment Response Analysis Methodology
   (ML070820447)

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3. Mitsubishi Handout - US-APWR 6th Pre-Application Review Meeting - Small Break LOCA Methodology (ML070820439)
4. Mitsubishi Handout - US-APWR 6th Pre-Application Review Meeting - Containment Response Analysis Methodology (ML070820447)

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ADAMS Accession No.: ML070930676 PKG: ML070930687

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Attendees

Public Meeting to Discuss

Engineered Safety Features, SBLOCA, and Containment

Response Analysis Methodology Topical Reports

March 22, 2007

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Enclosure 2

Mitsubishi Handout - US-APWR
6th Pre-Application Review Meeting-
Engineered Safety Features

(ML070820518)

Enclosure 3

Mitsubishi Handout - US-APWR
6th Pre-Application Review Meeting -
Small Break LOCA Methodology

(ML070820439)

Enclosure 4

Mitsubishi Handout - US-APWR
6th Pre-Application Review Meeting-
Containment Response Analysis Methodology

(ML070820447)

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SUBJECT: SUMMARY OF THE MARCH 22, 2007 MEETING TO DISCUSS MHI'S PROPOSED TOPICAL REPORTS ON ENGINEERED SAFETY FEATURES, SMALL BREAK LOSS-OF-COOLANT ACCIDENT (SBLOCA) METHODOLOGIES, AND CONTAINMENT RESPONSE ANALYSIS METHODOLOGIES

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