April 24, 2007

LICENSEE: MITSUBISHI HEAVY INDUSTRIES (MHI)

FACILITY: US-APWR STANDARD DESIGN PRE-APPLICATION REVIEW

SUBJECT: SUMMARY OF MARCH 1, 2007, PUBLIC MEETING ON RADIATION DOSE ASSESSMENT, SEVERE ACCIDENT MITIGATION, AND PROBABILISTIC RISK ASSESSMENT (PRA) FOR THE US-APWR.

On March 1, 2007, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of MHI at NRC Headquarters, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss PRA, severe accident mitigation, and radiation dose assessment methodologies for the US-APWR design certification. MHI announced its intention to submit a design certification application for the US-APWR in December 2007. A list of attendees is provided as Enclosure 1. MHI presented handouts that are shown in Enclosures 2, 3, and 4 and can be accessed through the Agencywide Documents Access and Management System (ADAMS) accession numbers ML070670194, ML070670192, and ML070670188.

Dose Assessment

MHI opened the meeting by discussing the radiation dose analysis that will be used in the US-APWR design certification application. MHI presented the dose analysis for normal operation and accident conditions, and discussed the methods for radiation shielding protection. MHI plans to use the latest revisions to NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," and Draft Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants," for the dose analysis and calculation codes that have already been approved by the NRC. The PWR-GALE code will be used to determine the atmospheric dispersion factors that will bound the atmospheric conditions at the plant sites. MHI stated that at the Combined Operating License (COL) stage, the COL applicants will validate the design certification analysis and compare the analysis to the site specific atmospheric dispersion factors. MHI presented the radioactive gaseous and liquid effluent pathways that will be analyzed using the atmospheric dispersion model.

When discussing the radiation dose analysis for accident conditions, MHI informed the NRC staff that it plans to use the RADTRAD and MicroShield codes to analyze several accidents. These accidents consist of the loss-of-coolant accident, steam generator tube rupture, main steam line break, fuel handling accident, rod ejection accident, and spent fuel cask drop accident. Next, MHI discussed the safety features available to mitigate these types of accidents and the fission product pathways that resulted from these particular accidents. MHI also discussed the applicable NRC regulations and regulatory guidance that would be used, including 10 CFR 50.67. The NRC staff informed MHI that 10 CFR 100.21 should be used instead of 10 CFR 50.67, since 10 CFR 50.67 only applies to current operating plants.

MHI then presented its methods and acceptance criteria for radiation shielding protection. Discussions were held on the acceptance criteria, radiation sources, and shielding design. During the acceptance criteria discussion, MHI stated that it would follow the requirements of 10 CFR 50.67 for personnel exposure; however, the NRC staff noted that 10 CFR 50.34 was the correct regulation that should be used instead of 10 CFR 50.67. MHI presented several codes that will be used for radiation shielding design. The ORIGEN2 Code will be used for source term calculations for core inventory and spent fuel, the MicroShield code will be used for almost all of the contained sources besides neutron source, G3 Code will be used for scattered gamma-ray dose calculation, MCNP Code will be used for complicated geometries, and the DORT and ANSIN codes will be used for the reactor pressure vessel and reactor coolant system (RCS) equipment.

Severe Accident Treatment and Mitigation

MHI discussed the severe accident analysis and its plans for submitting this analysis as part of the design certification application in December 2007. MHI plans to comply with the Three Mile Island requirements contained in 10 CFR 50.34(f), develop an appropriate PRA model, and resolve medium and high priority generic safety issues.

MHI identified eight issues associated with severe accidents. They are hydrogen mixing and combustion, core debris coolability, steam explosion, high pressure melt ejection and direct containment heating, temperature induced steam generator tube rupture, molten core concrete interaction, equipment survivability, and early and late containment overpressure failure. MHI also presented plans for mitigating these types of severe accidents. The hydrogen mixing and combustion accident will be mitigated by using a large dry containment that provides adequate strength to contain most hydrogen burns. Combustible gas will be controlled by the use of hydrogen monitors and gas igniters. Core debris coolability will be addressed by providing sufficient reactor cavity floor area for debris spreading. The reactor cavity floor will be made of basalt concrete. This feature will serve to mitigate any molten core concrete interaction.

The dedicated severe accident depressurization valves will be used to reduce RCS pressure to avoid high pressure melt ejection and direct containment heating. These dedicated severe accident valves can also be used to reduce the likelihood of temperature induced steam generator tube rupture. The firewater injection system and a drain line to the reactor cavity can also be used to reduce RCS pressure. The large dry containment, the decay heat removal system, and the backup firewater pump will serve to mitigate the effects of containment overpressure.

PRA methodology

The last presentation given by MHI pertained to the PRA analysis for the US-APWR design certification and PRA model. The PRA model will be based upon available design information. For the areas where specific information will be provided at the COL stage, the PRA model will determine the limiting boundary conditions and operating assumptions.

The NRC staff and MHI discussed the technical elements that comprise the Level 1, 2, and 3 analysis. The US-APWR Level 1 PRA discussion on technical elements focused on internal events at full and low power operation and shutdown, internal flooding, internal fire, seismic events, and limited external events. Likewise, the discussion on level 2 PRA technical elements involved internal floods and fires, and low power and shutdown events. The NRC staff questioned MHI on the PRA analysis and on the application of Regulatory Guide 1.200. The NRC staff stated that the PRA analysis for the US-APWR needs to meet the Commission's safety goals. The NRC staff informed MHI that DG-1145 will be revised soon and this revision would impact the US-APWR PRA analysis. MHI stated that it plans to submit the PRA Level 1 and 2 Technical Elements as part of the design certification application in December 2007, and submit the Level 3 PRA analysis in March 2008.

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

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Public Meeting on U.S.-APWR
March 1, 2007
Attendance Lists

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Stephen R. Monarque	NRC
Alan Levin	AREVA
Lynn Mrowca	NRC
Nick Saltos	NRC
Michelle Hart	NRC
Mark Beaumont	Washington Group International
Ed Fuller	NRC
David Bessette	NRC/Research
Hien Le	NRC
Hiroyuki Naito	Nuclear and Industrial Safety Agency
Yumi Kawanago	Mitsubishi Heavy Industries (MHI) - MNES
Yoshinori Takechi	MHI
Hiromasa Nishino	MHI
Dennis Buschbaum	TXU Power
Takahiro Imamura	MNES
Hiroshi Goda	MNES
Masayuki Kambara	MHI
Kiyoshi Yamauchi	MHI
Toshisada Kato	MHI
C. Keith Paulson	MHI
Alfred Torri	RMA
Makoto Toyama	MHI
Katsuya Kuroiwai	MHI
Futoshi Tanaka	MHI

Enclosure 1

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Todd Ansehmi	Anion Science and Technology
Ryuji Zwasaki	Toshiba
John H. Bickel	Talisman International, LLC
Josh McGuire	NRC

Enclosure 2

US-APWR Pre-Application Review Dose Evaluation

(ML070670194)

Enclosure 3

US-APWR Pre-Application Review Severe Accident Treatment and Mitigation

(ML070670192)

Enclosure 4

US-APWR, Pre-Application Review Overview PRA Methodology

(ML070670188)

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