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February 08, 2012

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-12023

Subject: Revised Response to US-APWR DCD RAI No.669-5219 Revision 2 (SRP 19.0)

References: 1) Letter MHI Ref: UAP-HF-10345 from Y. Ogata to U.S. NRC "MHI's Responses to US-APWR DCD RAI No. 669-5219 Revision 2 (SRP 19.0)" dated December 27, 2010

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "Revised Response to Request for Additional Information No.669-5219 Revision 2 ".

Enclosed is the revised response to RAI contained within Reference 1.

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
Director - APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure:

1. Revised Response to Request for Additional Information No. 669-5219 Revision 2

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NRO

CC: J. A. Ciocco
J. Tapia

Contact Information

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Enclosure 1

UAP-HF-12023
Docket Number 52-021

Revised Response to Request for Additional Information No.669-5219 Revision 2

February, 2012

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

02/08/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 669-5219 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19**
DATE OF RAI ISSUE:	**11/29/2010**

QUESTION NO. : 19-494

The staff has reviewed MHI's response to RAI 19-442. Based on the US-APWR shutdown risk results on page 19.1-146 of the DCD, the shutdown CDF equals the shutdown LRF frequency. No credit was given for containment closure in the risk assessment. In their response to RAI 19-442, MHI reported that the USAPWR shutdown CDF removing all equipment not required by TS to be 2.1E-5 per reactor year. This result means that the LRF removing all equipment not required by Technical Specifications (TS) to be 2.1E-5 per reactor year which exceeds the Commission's safety goals for new reactors. The staff concludes that voluntary initiatives must be implemented by the COL applicant for the USAPWR design to meet the Commission's safety goals. The staff is requesting MHI to consider adding shutdown TS in accordance with Criterion 4 of 10CFR50.36 (c)(2)(ii) so that this design meets the Commission's safety goals for new reactors without voluntary initiatives or justify in the DCD why these actions are not necessary.

ANSWER:

The RAI states that "...the LRF removing all equipment not required by Technical Specifications (TS) to be 2.1E-5 per reactor year which exceeds the Commission's safety goals for new reactors" and requests that MHI "...consider adding shutdown TS in accordance with Criterion 4 of 10CFR50.36 (c)(2)(ii) so that this design meets the Commission's safety goals for new reactors,,," MHI has not identified any NRC policy, regulation, or guidance document that specifies that the NRC safety goals shall be met following the assumption that selected plant safety equipment is not available for consideration in the PRA, or that only equipment covered by Technical Specifications can be credited in the PRA.

The NRC safety goal policy was approved under the Staff Requirements Memo (SRM) for SECY 90-016. This policy requires new plants to demonstrate how the risk associated with the design compares against the Commission's goals of less than 1E-4/year for core damage frequency and less than 1E-6/year for large release frequency. These goals are implemented in NRC guidance in SRP (NUREG-0800) Chapter 19. In accordance with SRP Chapter 19, Regulatory Guide (RG) 1.200 provides the guidance for how the PRA is performed. RG 1.200, Table 2, Summary of Technical Characteristics and Attributes of a PRA, states that accident sequence development analysis includes necessary and sufficient equipment (safety and non-safety) reasonably expected to be used to mitigate initiators. There is no guidance in the SRP or RG 1.200 that the DCD demonstrate compliance with the NRC safety goal policy by performing a PRA using only equipment required to be in service by Technical Specifications. The US-APWR DCD, Section 19.1 clearly provides PRA documentation, per RG 1.200, that the US-APWR meets the NRC safety goals.

The conditional PRA analyses described in MHI's response to RAI #610-4761 Question 19-442 (See UAP-HF-10246 responded on September 3, 2010) identified the safety injection system as important to risk while the plant is in shutdown mode with reduced water level in the reactor coolant system. As described below, MHI proposes to address these conditional PRA results by implementing non-Technical Specification administrative controls to ensure the availability of one train of safety injection system during shutdown, low water level conditions. The NRC has historically allowed the use of non-Technical Specification administrative controls for general risk management and shutdown risk management. In the SRM for SECY-97-168, "Issuance for Public Comment of Proposed Rulemaking for Shutdown and Fuel Storage Pool Operation," the Commission rejected the staff's proposal to implement new regulatory requirements for shutdown Technical Specifications, deciding instead to continue reliance on the industry's voluntary risk management programs, particularly NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." Within the body of the SECY the staff acknowledged these voluntary measures, stating:

> Sensitivity analysis showed little qualitative value when comparing the voluntary case (based on the assumption that current voluntary practices remain in effect) to the rule case. This is because of the substantial measures generally adopted by industry in response to generic communications. These measures include NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.

Additionally, the Standard Technical Specifications (STS) for Westinghouse PWRs, NUREG-1431, do not require an LCO for safety injection system during Modes 5 and 6. The STS are described in SRP Chapter 16 as being adequate to address the requirements in 10 CFR 50.36, including Criterion 4, excluding unique plant specific requirements. The risks in Modes 5 and 6 with mid-loop operation for the US-APWR are not significantly different than current operating PWRs and current operating PWRs have not been required to include an LCO for safety injection in Modes 5 and 6 in their adoption of STS. Hence, NRC has not historically interpreted the need for safety injection system during Modes 5 and 6 as rising to the level prescribed in 50.36, Criterion 4, i.e., "...which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." The design and PRA results for the US-APWR do not support a change in this position.

The STS for PWRs also acknowledges the use of voluntary initiatives for shutdown risk management in the following statement from the TS LCO Bases:

> The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of

electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

As an additional defense-in-depth measure, COL applicants are required to implement the Maintenance Rule, 10CFR50.65, section a(4) which defines requirements for management risk during the removal of equipment from service. It states:

> Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety.

Maintenance Rule paragraph 50.65(a)(4) requires a licensee to assess and manage the increase in risk that may result from proposed maintenance activities before performing these maintenance activities (such as the removal of safety injection system from service during shutdown conditions). DCD Section 17.6 contains a COL item [17.6(1)] that requires the COL applicant to provide a description of the Maintenance Rule program that meets 10 CFR 50.65.

NEI 07-02, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52," provides processes for assessing and managing potential increases in risk that might result from proposed maintenance activities to meet the requirements of 10 CFR 50.65. NEI 07-02 is endorsed by the NRC in a Safety Evaluation as an acceptable generic program description for use in meeting 10 CFR 50.65. The Maintenance Rule program in NEI 07-02 follows the guidance in Nuclear Management and Resources Council, Inc.(NUMARC), "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01, as endorsed and modified by Regulatory Guide (RG) 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," without any exceptions that could materially and negatively impact the effectiveness of the program.

NUMARC 93-01 Sections 11.3.5, "Scope of Assessment for Shutdown Conditions," 11.3.6, "Assessment Methods for Shutdown Conditions," and 11.3.7, "Managing Risk," address the performance of safety assessments for shutdown conditions, including mid-loop operations. The approaches discussed involve the use of both quantitative and qualitative insights from the plant safety assessment of maintenance activities. The assessment provides insights regarding the risk-significance of maintenance activities. The process for managing risk involves using the result of the assessment in plant decision-making to control the overall risk impact. NUMARC 93-01 addresses, in part, the establishment of thresholds for risk management actions quantitatively by considering the magnitude of increase of the core damage frequency (and/or large early release frequency) for the maintenance configuration. RG 1.182 states that NUMARC 93-01 provides methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.65(a)(4).

Given the existence of COL item 17.6(1), the adoption of NEI 07-02 by both APWR COL applicants, and the endorsements by the NRC staff of both NEI 07-02 and NUMARC 93-01, through a safety evaluation and in regulatory guides as acceptable program descriptions for managing risk while shutdown, there is no need for additional Technical Specifications regarding the management of shutdown risk.

MHI will revise DCD Table 19.1-119 "Key Insights and Assumptions" to include an administrative control to ensure the availability of a train of the SIS and associated water source (i.e., RWSP) as a RCS make up function during cold shutdown and during refueling with water level <23 ft above the top of reactor vessel flange. This new table entry will identify COL item 13.5(5) to ensure that the controls will be implemented by the COL applicants, through operating procedures including NUMARC 91-06 and NEI 07-02, which provides processes for assessing and managing potential increases in risk that might result from proposed maintenance activities to meet the requirements of 10 CFR 50.65.

MHI believes that its position is well established in regulatory history and that the NRC staff position taken in the RAI:

1. Is not supported by the NRC Safety Goal policy statement nor any NRC guidance related to the policy statement, such as SRP Chap 19
2. Is contrary to the direction provided by the Commission in SRM for SECY 97-168 which directed the staff to allow continue reliance on industry voluntary efforts to manage shutdown risk, and
3. Is not supported by NRC approved STS which does not include an LCO for the safety injection system in Modes 5 and 6, and provides Bases statements supporting MHI's position to use voluntary administrative controls for shutdown risk management.

The quantitative risk values for the US-APWR will not significantly change if the proposed administrative controls are converted into Technical Specifications, as explained in SECY-97-168. Such a change would only convert voluntary programs into NRC controlled programs which the Commission rejected in the SRM to SECY-97-168.

Impact on DCD

MHI will revise DCD Table 19.1-119 "Key Insights and Assumptions" to include an administrative control to ensure the availability of a train of the SIS and associated water source (i.e., the RWSP) as an RCS make up function during cold shutdown and during refueling with water level <23 ft above the top of reactor vessel flange. (See Attachment-1)

Impact on R-COLA

R-COLA Part 2 FSAR Table 19.1-119R will be revised, consistent with DCD Table 19.1-119.

Impact on S-COLA

S-COLA Part 2 FSAR Table 19.1-119R will be revised, consistent with DCD Table 19.1-119.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Reports

There is no impact on the Technical / Topical Reports.

Table 19.1-119 Key Insights and Assumptions (Sheet 34 of 48)

Key Insights and Assumptions	Dispositions
19. Surge line flooding may occur if decay heat removal function is lost during plant operating states where the pressurizer manway is the only vapor release pass from the RCS. Water held up in the pressurizer can erroneous readings of water level indicators measured with reference to the pressurizer. This phenomenon can also prevent gravity injection from the SFP. Measures to prevent accident evolution caused by surge line flooding are important. Adoption of both measures listed below can reduce risk from surge line flooding event. - Installation of an temporary RCP water level sensor that measure the MCP water level with reference to pressure at the reactor vessel head vent line and cross over leg when the RCS is vented at a high elevation. - Operational procedures to perform continuous RCS injections when loss of RHR occurs under conditions where the pressurizer manway is the only vapor release pass from the RCS. The temporary water level will satisfy the following specifications. - Water level can be read outside the containment vessel (CV) in order to be effective during events which involve harsh environment in the CV - Tygon tubing monometer will not be used - Instrumentation piping diameter will be sufficient enough to prevent delay in response	5.4.7.2.3.6 19.2.5 COL 19.3(6) COL 13.5(7)
20. Two types of instruments are provided in US-APWR design to measure the temperature representative of the core exit whenever the reactor vessel head is located on top of the reactor vessel. The first one is core exit thermocouples located inside the RV. The second is resistance temperature detectors in the reactor coolant hot leg. These two independent instruments will be available whenever the RCS is in a mid-loop condition and the reactor vessel head is located on top of the reactor vessel.	5.4.7.2.3.6
21. Administrative controls to ensure the availability of a train of the SIS and associated water source (i.e.RWSP) as a RCS make up function during cold shutdown and during refueling with water level <23 ft above the top of reactor vessel flange.	COL 13.5(5)

DCD_16-117
DCD_19-494