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December 12, 2013

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

Attention: Mr. Perry Buckberg

Docket No. 52-021
MHI Ref: UAP-HF-13305

Subject: MHI's Response to US-APWR DCD RAI No. 1061-7266 (SRP 19)

Reference: 1) "Request for Additional Information No. 1061-7266, Review Section: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19," dated November 20, 2013.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "Response to Request for Additional Information No. 1061-7266."

Enclosed is the response to three RAI questions 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 submittal. His contact information is below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, LTD.

Enclosure:

1. Response to Request for Additional Information No. 1061-7266

D081
NRO

CC: P. Buckberg
J. Tapia

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Enclosure 1

UAP-HF- 13305
Docket No. 52-021

Response to Request for Additional Information No. 1061-7266

December 2013

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/12/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 1061-7266**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19**
DATE OF RAI ISSUE:	**11/20/2013**

QUESTION NO. : 19-594

The staff has reviewed the applicant's response to RAI 7081, Question 19-585. The staff understands that the purpose of isolating the low pressure letdown line during an overdraining event is to isolate the line before the level reaches the top of the core, not before the RHR pumps become inoperable. The staff also understands how the OVDR sequences were calculated. The staff also reviewed the additional indication to assist the operator given a loss of decay heat removal function and subsequent re-pressurization of the RCS in addition to the RCS narrow range indication. This indication could yield higher than actual RCS level readings if the RCS is vented via the pressurizer or another high elevation vent. The staff also understands that the RHR system indication may not reflect actual RCS conditions if the RHR pumps are not running. The staff has the following two requests for information:

(1) Please update Chapter 19 of the DCD documenting whether a single charging pump is sufficient to keep the core covered and match decay heat given a non-isolated letdown line and an isolated letdown line. This issue is relevant to all OVDR, LOCA, and LOOP scenarios.

(2) The staff requests the applicant to update the risk insights table to note that the safety related CETs are important until the reactor vessel head is removed due to potential inaccuracies in RCS level indication following an extended loss of the DHR function and subsequent boiling in the RCS.

ANSWER:

(1) The response to RAI 1056-7236 Question 01.05-10 (Ref. UAP-HF-13282, dated December 2, 2013) demonstrates that a single charging pump is capable of providing sufficient coolant to keep the core covered (at saturation temperature). The flow rate for a charging

pump does not depend on the low-pressure letdown line isolation. The letdown during refueling operation is taken from downstream of the CS/RHR heat exchangers via the low-pressure letdown line valve (RHS-AOV-024B or C). Then the water flows into the volume control tank (VCT). The charging pump takes suction from the VCT. When the low-pressure letdown isolation valve is closed, the water level of the VCT would decrease. In the case that the VCT water level reaches the low level setpoint, the automatic makeup starts. In the case that the VCT water level reaches the low-low level setpoint, the charging pump suction is automatically changed from the VCT to the RWSP (refer to DCD Subsection 9.3.4). Therefore, the isolation of the low-pressure letdown line does not affect the charging pump suction. In addition, the charging line and the low-pressure letdown lines are independent, as shown in DCD Figures 5.4.7-2 and 9.3.4-1 and the low-pressure letdown line isolation does not affect the charging line.

The design feature for the charging pump will be documented in DCD Subsection 19.1.6.1, as shown in the attached markups.

(2) DCD Table 19.1-119 will be revised, as shown in the attached DCD markup.

Impact on DCD

DCD Subsection 19.1.6.1 and Table 19.1-119 will be revised, as shown in the attached markups.

Impact on R-COLA

R-COLA Part 2 FSAR Table 19.1-119R will be revised to be consistent with this change.

Impact on PRA

There is no impact on the PRA.

Impact on Topical/Technical Report

There is no impact on Topical and Technical Reports.

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QUESTION NO. : 19-595

The staff reviewed the applicant's response to RAI 7090, Question 19-592. Question 19-592 concerns the risk associated with losses of RCS inventory when the refueling cavity is flooded, considering the temporary fuel racks located in the refueling cavity. The staff also reviewed the core damage frequency estimates for plant operational states (POSs) 5, 6, and 7 as reported in Table 19.548-1 of the response to RAI 5855, Question 19-548. Should an overdrain event occur, operator actions are needed to prevent core uncovering. The staff understands there are no automated mitigation strategies. Thus, in order for the risk of drain down events to be negligible contributors of risk, failure of the operator to terminate the drain path and/or add RCS inventory has to be very low (on the order of 1E-5). Given a bounding HEP of 1E-5 and the initiating event frequencies reported in the Question 19-548 response, the core damage frequency from draindown events during modes 5, 6, and 7 totals 6.6E-9/yr which represents more than ten percent of the shutdown CDF.

The staff also notes that there is one RCS wide range level instrument which was added to DCD Table 19-119 as risk significant based on RAI 6281, Question 19-565. However, this instrument does not appear to be safety related, backed up by AAC GTGs, or part of PCMS. Also, the applicant's response to Question 19-592 assumes that TS 3.9.5 and TS 3.9.7 will be applicable during "No Mode" with all fuel out of the reactor and possible fuel in the containment racks. The staff requests the following information:

1. The risk contribution from draindown events should be reported in Chapter 19 of the DCD.

2. To support a very low HEP estimate for the operator failing to terminate drain path:

(a) For conditions when the fuel will be stored in the temporary fuel racks and fuel is no longer in the reactor vessel, such that there are no Technical Specification (TS) limiting conditions of operation (LCO) that apply (no mode), please verify and document in the DCD and TS

that TS 3.9.5 and TS 3.9.7 will be applicable during "no mode" operations.

(b) Please document in the FSAR in Chapter 7 and Chapter 19 what system is used to indicate the RCS wide range level instrument and associated alarms in the control room (PCMS or PSMS).

(c) Please document in FSAR Chapter 7 and Chapter 19 how the RCS wide range level instrument and associated alarm will receive power during a loss of offsite power.

(d) Please add to the DCD risk insights Table 19-119 and Chapter 5.4.7. , that the drain lines that flow from the refueling cavity to the RWSP are administratively locked closed after flooding and prior to fuel movement. Similarly, the fill line, which is located at an elevation one foot above the reactor flange, is administratively locked closed after flooding and prior to fuel movement.

This RAI is related to RAI 7184 Question 09.01.02-53.

ANSWER:

1. The risk contribution from drain-down events is reported in DCD Rev. 4 Chapter 19. Specifically, Table 19.1-89 addresses the CDF contribution of drain-downs (OVDR), failure to maintain level (FLML), and loss of coolant accidents (LOCA) during LPSD conditions. The contribution to LPSD CDF for these events totals 6.5E-8 /yr. The risk from a drain down scenario in POSs 5, 6 and 7 is described as insignificant in DCD Subsection 19.1.6.1 due to isolation time and containment isolation capability. The RAI estimate of 6.6E-9/yr is approximately only 10% of the LPSD loss of inventory CDF contribution of 6.5E-8 /yr. However, such an event is only 4% of CDF associated with LPSD of 1.8E-7/yr and less than 0.3% of the total CDF of 2.9E-6/yr. Since the mechanisms for the particular type of event are already discussed and ranked for their contributors to LPSD CDF in the DCD, the risk contribution from drain-down events is adequately reported in DCD Rev. 4, Chapter 19.

2.

(a) The response to RAI 1055-7184 Question 09.01.02-53 (Ref. UAP-HF-13273, dated December 2, 2013) changed the applicability of TS 3.9.5 and 3.9.7 to include the time in which one or more irradiated fuel assemblies are seated in the containment racks.

(b) DCD Subsection 7.7.1.14 and Table 19.1-119 "LPSD assumptions", Item 26 will be added to include the information on the refueling cavity water High and Low level indicator and Main Control Room alarm "RCS-LIA-011-N" in Figure 5.1-2 (Sheet 3 of 3) as described in DCD Sections 3.1.6.4.1, 9.1.4.2.1.13 and 9.1.4.2.2.2.

(c) Table 19.1-119 "LPSD assumptions", Item 26 will be added to include the information on the power supply for the monitoring function of PCMS during a LOOP event from the station batteries which are backed up by the alternate ac source, as described in DCD Subsection 7.1.1.10.

(d) Table 19.1-119 "LPSD assumptions", Item 27 will be added and DCD Subsection 9.1.4.2.1.13 will be revised to include the risk insights on locked closed valves.

Impact on DCD

DCD Subsection 7.7.1.14 will be added; Subsections 9.1.4.2.1.13 and Table 19.1-119 will be revised, as shown in the attached markups.

Impact on R-COLA

R-COLA Part 2 FSAR Table 19.1-119R will be revised to be consistent with this change.

Impact on PRA

There is no impact on the PRA.

Impact on Topical/Technical Report

There is no impact on Topical and Technical Reports.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/12/2013

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QUESTION NO. : 19-596

The staff evaluated the applicant's response to RAI 5651, Question 19-506 and RAI 6953, Question 19-578. In response to Question 19-506, MHI responded that SG nozzle dams can withstand RCS pressure up to 32.0 psig. From the MAAP analysis results reported in Table 19.506-1 of the Question 19-506 response, the applicant concluded that to maintain the RCS pressure below the design pressure of the SG nozzle dam, it is necessary to remove at least three pressurizer safety valves or the pressurizer manway. The m-RELAP5 code analysis also indicates that removal of at least three pressurizer safety valves or the pressurizer manway are sufficient for over pressure protection. The staff then evaluated the design of the US-APWR pressurizer. The US-APWR pressurizer height from the surge line to the pressurizer spray nozzle is approximately 74 feet. The staff understands the pressurizer safety valves are installed on separate relief lines at the top of the pressurizer. The pressurizer manway is located in the dome of the pressurizer. The staff is concerned that given surge line flooding following a loss of DHR, subsequent RCS boiling and initiating RCS injection via a Safety Injection pump could cause the pressurizer to fill. The staff is concerned that the SG nozzle dams cannot withstand the pressure given by the water head in a flooded pressurizer. The staff is requesting that the SG nozzle dam design pressure be increased to handle a potentially flooded pressurizer given (1) an open manway and (2) removed pressurizer safety valves, or justify why this design enhancement is not needed to meet the staff recommendations in GL 88-17

ANSWER:

The height from the SG nozzle dam to the top of the pressurizer is approximately 80 feet. As defined by this RAI, the pressure when the pressurizer is solid, i.e. filled by Safety Injection, slightly exceeds the design pressure of the SG nozzle dam; however, such a condition is unlikely to result in core damage. That is, if the vent path for Safety Injection changes from the pressurizer to a nozzle dam location, the scenario would correspond to that considered in

POSs 4-2 and 8-2, where SG manways are open and no SG nozzle dam is installed.

From the perspective of GL 88-17 (Section 2.7 of Enclosure 2 to GL 88-17, "Guidance for Meeting Generic Letter 88-17"):

> A part of the concern is that nozzle dams may not have sufficient strength to withstand the pressure that may result under accident conditions. Loss of a nozzle dam while pressurized under loss of DHR conditions could cause rapid RV voiding.

MHI concludes that "the pressure that may result under accident conditions" is the water head in the flooded pressurizer due to the surge line flooding following a loss of RHR and subsequent RCS boiling. (I.e., the flooded pressurizer caused by RCS injection via SI pump differs from the accident condition discussed in GL 88-17. This is because RCS injection using the SI pump removes decay heat from the reactor core and keeps the core covered.) The M-RELAP5 code used in the response to RAI 749-5651 Question 19-506 (Ref. UAP-HF-12092, dated April 25, 2012) modeled the pressurizer and the surge line piping and calculated the effect of the surge line flooding following a loss of RHR and subsequent RCS boiling; the primary system pressure in Table 19.506-1 reflects this condition. In the maximum pressure case, in which the three pressurizer safety valves are removed, the maximum pressure was evaluated to be 20.5 psig. This calculated pressure is less than the 32.0 psig SG nozzle dam design pressure and there is a design margin of approximately 50%. Therefore, the design pressure meets the recommendation in GL 88-17 for the loss of decay heat removal.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical/Technical Report

There is no impact on Topical and Technical Reports.

7.7.1.11 Radiation Monitoring System

The radiation monitoring system (RMS) section of the PCMS provides non-safety area and process radiation monitoring to generate displays and alarms. Refer to Chapters 11 and 12 for additional related details.

7.7.1.12 Auxiliary Equipment Control System

The auxiliary equipment control system section of the PCMS controls and monitors auxiliary systems (e.g., radioactive waste disposal system, CVCS water treatment).

HSI for the auxiliary equipment control system is located in the auxiliary equipment control room, which is located in the auxiliary building. This control room is manned periodically for auxiliary equipment operation (i.e., radioactive waste management). Key alarms are displayed on the alarm VDU, LDP and the operational VDU and key indications are provided on operational VDUs. Refer to Chapter 11 related details.

7.7.1.13 Automatic Control of Safety-Related System

There are several automatic control signals transmitted from the PCMS to the PSMS via hardwired connections, which actuate safety-related components to provide diverse preceding actuations on the safety-related signals in order to enhance the plant reliability. These automatic controls enhance the plant reliability, but are not credited in the safety analysis of DCD Chapter 15. The priority logic within the SLS ensures that a safety-related signal, either manually or automatically, generated from the PSMS has higher priority than those non-safety automatic control signals from the PCMS. These signals are transmitted from the PCMS to the PSMS via the hardwired connections with the isolation devices in the PSMS.

7.7.1.14 Monitoring and Alarm for Control Systems

DCD_19-595

The PCMS provides direct monitoring and alarm of non-safety plant systems, including control systems, for the operators via operational VDUs, alarm VDUs and LDP. This monitoring function of the PCMS is available during a LOOP event with power from the station batteries and backed up by the alternate ac power source, as described in Subsection 7.1.1.10.

These automatic controls from the PCMS to the PSMS are the same as those used in conventional standard PWR plants.

7.7.2 Design Basis Information

The control systems include the necessary features for manual and automatic control of process variables within the prescribed normal operating limits.

The PCMS design is based on the following design considerations.

7.7.2.1 Safety Classification

The PCMS is a non-safety system. The plant accident analysis of Chapter 15 does not rely on the operability of any PCMS control functions to assure safety. Safe shutdown can be achieved without reliance on any PCMS control functions.

9.1.4.2.1.13 Permanent Cavity Seal

The Permanent Cavity Seal (PCS) has a function to maintain water level in the refueling cavity during refueling operation by sealing an annular gap between the reactor vessel flange and the refueling cavity floor.

The seal is made of a stainless steel structure and permanently attached to the vessel and the floor with bolts and welds. The stress limits of ASME Code Section III, Subsection ND, are used for design of the PCS. In addition, material selection, fabrication, and examination of the PCS are in accordance with requirements of ASME Code Section II, Section IX, and Section V. ASME Code certification is not required.

Should a load, such as a fuel assembly, suspended from the polar crane or refueling machine, which are designed as single failure proof, be dropped on the seal, damage to the seal is prevented by a stainless steel guard plate (curing lid) which is installed over the PCS. Moreover. since the PCS and the guard plate are washed thoroughly with demineralized water after the draining of the refueling cavity water to remove extraneous materials such as sludge, these structures do not degrade over time.

Leakage detection systems are utilized for the PCS. Two leakage detection pipes are installed under the ring-shaped PCS directly opposite each other across the reactor vessel. Wherever around the seal leakage from the PCS should occur, the leakage water flows and accumulates into an annular space between a vertical cylindrical plate, which is attached to support ring, and the PCS, and eventually flows into one or both of the detection pipes. Once water flows into the leakage detector via the leakage detection pipe, the leak detection system provides an alarm signal to alert operators in the MCR and in the vicinity of the fuel handling system that an abnormal water level condition exists in the refueling cavity.

The refueling cavity water High and Low level is monitored by a refueling cavity water level indicator and an alarm which are shown as "RCS-LIA-011-N" in Figure 5.1-2 (Sheet 3 of 3). This water level channel is ~~operable before the fuel assembly is moved from or to the R/V~~available while the refueling cavity is flooded. The low level alarm setpoint is determined using a water shielding depth necessary to keep personnel radiation dose ALARA in the fuel handling area and refueling cavity. In the event that the refueling cavity water level alarm RCS-LIA-011-N becomes inoperable, the spent fuel pit water level alarm SFS-LICA-010-S and SFS-LICA-020-S will be utilized while the fuel transfer gate valve and the gate between the refueling canal and SFP are open. The low water level alarm setpoint is also determined using a water shielding depth necessary to keep personnel radiation dose ALARA in the fuel handling area and refueling cavity. The water shielding depth requirement for the refueling cavity, and the resulting radiation dose limit in the Fuel Handling Area are described in Subsection 12.3.2.2.4. DCD_19-592 DCD_09.01.02-58 DCD_09.01.02-55

To preclude draindown events, the drain lines from the refueling cavity to the RWSP and the fill lines are administratively locked closed after flooding and prior to fuel movement. Although a rapid cavity drain-down event is unlikely, if such an event should occur, upon alarm the workers immediately place any fuel in transit in the nearest suitable safe storage location. Since the seal is visually inspected before filling the cavity, the possibility of a rapid cavity drain-down event at a flow rate of more than 1 gpm, resulting from a large crack, which would be detected through visual inspection, is excluded. Therefore, DCD_19-595

- Tygon tubing monometer will not be used
- Instrumentation piping diameter will be sufficient enough to prevent delay in response

Freeze plug may not be used for US-APWR because the isolation valves are installed considering maintenance and CCWS has been separated individual trains. Therefore, the freeze plug failure is excluded from the potential initiator.

The methods for data analysis and common cause analysis are the same as for Level 1 internal events PRA at power. The details of data analysis and CCF analysis are given in Subsection 19.1.4.1.1.

Mitigating functions during LPSD can be categorized into two groups: decay heat removal function and RCS inventory make up function. Systems that provide these functions are listed below. It is postulated that if these systems fail following an initiating event, bulk boiling and core damage will occur.

- Decay heat removal functions
  - RHR system

    If CS/RHR pumps are available, the RCS is cooled by the RHR system through RHR suction line.

  - SG and secondary side system

    When the RHR cooling is unavailable, decay heat is removed from the RCS via the SGs.

- RCS inventory make-up Functions
  - CVCS

    If the RHRS and the heat removal via SGs are unavailable, coolant to the RCS is injected by the CVCS in order to prevent bulk boiling and to maintain the RCS inventory. A single charging pump is capable of supplying the coolant to keep the core covered, regardless of isolation of low-pressure letdown line. If the operable charging pump~~s~~ fails, pump~~s~~ that ~~were~~was locked out for low temperature overpressure (LTOP) compliance can be used if available. DCD_19-594

  - High head injection system

    If the CVCS fails to operate, safety injection pumps are utilized to inject coolant to the RCS in order to maintain coolant inventory. If the operable safety injection pumps fail, pumps that were locked out for low temperature overpressure (LTOP) compliance can be used if available.

  - Gravity injection system

Table 19.1-119 Key Insights and Assumptions (Sheet 36 of 51)

Key Insights and Assumptions	Dispositions	
23. ~~Administrative~~Technical Specification controls to ensure that the availability of equipment necessary to achieve containment isolation, specifically, the equipment hatch hoist, lifting rig, and AACs while the containment remains open.	~~COL 13.5(7)~~ TS 3.4.8 TS 3.6.7 TS 3.9.6	DCD_19-494 S03
24. All containment penetrations are closed ~~immediately~~prior to the onset of steaming into containment after a loss of all RHR trains.	~~COL 13.5(7)~~ TS 3.4.8 TS 3.6.7 TS 3.9.6	DCD_19-494 S03
25. Safety-related core exit thermocouples are important until the reactor vessel head is removed due to potential inaccuracies in RCS level indication following an extended loss of the decay heat removal function and subsequent RCS boiling.	19.1.6	DCD_19-594
26. The refueling cavity water High and Low level indicator and alarm "RCS-LIA-011-N" are displayed in the Main Control Room using PCMS which is powered from the station batteries and are backed up by the alternate AC source.	3.1.6.4.1 Figure 5.1-2 7.1.1.10 7.7.1.14 9.1.4.2.2.2	DCD_19-595
27. To preclude drain-down events, the drain lines from the refueling cavity to the RWSP and the fill lines are administratively locked closed after flooding and prior to fuel movement.	9.1.4.2.1.13	DCD_19-595