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November 27, 2008

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-08271

Subject: MHI's Responses to US-APWR DCD RAI No. 88-1438

Reference: 1) "Request for Additional Information No.88-1438 Revision 1, SRP Section: 19.1.6 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: PRA," dated October 29, 2008.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document as listed in Enclosure.

Enclosed are the responses to 15 RAIs contained within Reference 1. Of these RAIs, 9 will not be answered within this package. They are; 19-138, 19-139, 19-140, 19-141, 19-142, 19-144, 19-145, 19-147 and 19-149. These RAIs require additional time for internal discussions and will be answered by 11th of January, 2009.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, 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,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure:

1. "Responses to Request for Additional Information No. 88-1438 Revision 1"

CC: J. A. Ciocco
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D081
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Contact Information

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Docket No. 52-021
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Enclosure 1

UAP-HF-08271
Docket No. 52-021

Responses to Request for Additional Information No.88-1438 Revision 1

November 2008

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/27/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.88-1438 REVISION 1**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-136

Clarify the criterion for declaring that shutdown probabilistic risk assessment (PRA) sequences lead to core damage (e.g., water level, boiling, peak clad temperature). Describe the thermal-hydraulic analyses that were performed for each scenario to determine whether core damage occurs.

ANSWER:

The criterion of core damage for the shutdown PRA is collapsed liquid level below top of active fuel. Thermal-hydraulic analysis were performed for reflux cooling during mid-loop operation. Detail of this analysis is described in response to question 19-45 of RAI #39. For other scenarios in which water injection to the reactor coolant system is credited as a mitigation function, determination of core damage is based on whether or not the injected water rate exceeds the rate of evaporation due to decay heat. Available times for operator actions were evaluated as described in response to question 19-69 of RAI #39.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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RAI NO.: **NO.88-1438 REVISION 1**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-137

Revise the design control document (DCD) to include additional information on the reduction factors used in POS other than 8-1 based on the number of mitigation systems and human error dependency, similar to that provided on page 20-22 of the PRA technical report (MUAP-07030). In addition, provide the following information:

a. How the "conservative value of human error" was chosen, as well as a justification that it is conservative

b. Why exceptions were taken to use higher values for certain combinations of mitigating systems or functions

c. How the number of operation tasks for each scenario were determined, given that procedures are not yet complete.

ANSWER:

The discussion provided in page 20-22 of the PRA technical report will be incorporated in the next revision of the DCD. Additional information requested by the staff are written below.

a. The unconditional human error probabilities of actions credited in POS 8-1 are shown in sheet 4 and 5 of table 9.2.3-1 in chapter 9 of the PRA technical report. Operator actions required to operate mitigation systems, which are classified as type "Cp" human actions*[1)], have failure probabilities ranging from 3.1E-2 to 2.6E-3 with an average value of 1.3E-2. The representative unconditional human error probability to calculate reductions factors was chosen as 3E-2 considering these human error probabilities assessed for POS 8-1.

b. Some of the mitigation functions require operator actions that have the same tasks that are required for other mitigation functions. Such operator tasks are considered to have very high dependencies. For combinations of mitigation functions that require same operator actions, an higher value was used to reflect the high dependencies between the operator actions.

c. A series of operator tasks required for each mitigation function was counted as one operator action. For the In the low power shutdown PRA, all mitigation functions, except for emergency power supply and the restart of component cooling water pumps and essential water pumps after loss of offsite power, require an operator action. The number of type Cp operator actions in a accident sequence is equal to the number of mitigation systems in the sequence that require operator action.

*1) Type Cp human actions

Human actions taken after the occurrence of an initiating event is classified into two types, type Cp and type Cr.

Type Cp actions are actions that are essential to operate the mitigation function. Human error to perform type Cp actions are modeled directly under the top gate of the mitigation function fault tree linked to each event headings. Examples of type Cp actions in LPSD PRA are actions to connect alternate ac power source to the safety bus, actions to start safety injection, actions to start secondary side cooling and others.

Type Cr actions are actions taken to recover a mitigation function when it has failed. Human error to perform type Cr actions are modeled under a "and" gate, coupled with the hardware failure of the mitigation system. Examples of type Cr actions in LPSD PRA are actions to isolate a leak that has occurred in the ECCS piping, actions to start a standby pump when a running pump fails, and others.

Impact on DCD

The DCD will revised to include more information of the reduction factors used in POSs other thanPOS 8-1.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

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SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-138

Page 19.1-128 states that "important SSCs [structures, systems, and components] and operator actions of other POS are qualitatively extracted based on the mitigation system which is available for each POS...[p]assive components are excluded from important SSCs because generally the failure rate of passive mode is lower than active mode." State which passive components were excluded based on this assumption. Provide additional justification for this assumption, given that components with an assumed low failure rate may have extremely high risk achievement worth (RAW) values, and attention to these components is important to ensure that plant risk is as low as was assumed during design certification.

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-139

(Follow-up to Question 19-46) The response to Question 19-46 indicates that during POS 3 and portions of POS 4, a single residual heat removal (RHR) train does not remove enough decay heat to prevent boiling, but that "RHR will continuously remove decay heat until RHR function degrades." Provide a justification for assuming that the RHR pump will continue to operate when there is boiling in the reactor coolant system (RCS). How long will the single RHR pump continue to operate before "function degrades" and a loss of RHR occurs? When will the core be uncovered in this scenario? If the RHR pump cannot function for its entire mission time in POS 3 and portions of POS 4, losses of RHR and subsequent mitigation strategies should be quantitatively evaluated in these POS using two trains of RHR as the success criterion..

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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RAI NO.: NO.88-1438 REVISION 1

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: 19.1.6

DATE OF RAI ISSUE: 10/29/2008

QUESTION NO. : 19-140

(Follow-up to Question 19-47) Question 19-20(c) requested a sensitivity study for the shutdown PRA that credits only the systems required to be operable according to technical specifications (TS), since voluntary measures that are not required by current regulations could be withdrawn by licensees without NRC approval. The response to Question 19-47 clarified that the sensitivity study credited automatic isolation of the low pressure letdown line on low level, the charging pumps, the refueling water storage pit (RWSP), and the refueling water storage auxiliary tank (RWSAT) despite the fact that there are no TS requiring these components to be available during MODES 5 and 6. Standard Review Plan (SRP) Section 19.0 states that the design-phase PRA is used to demonstrate whether the plant design, including the impact of site-specific characteristics, represents a reduction in risk compared to existing operating plants. The PRA is also used to identify and support the development of specifications such as inspections, tests, analyses, and acceptance criteria (ITAAC); reliability assurance program (RAP); TS; and combined license (COL) items. So that the staff can make these conclusions in its final safety evaluation report (FSER):

a. Provide the results of a sensitivity study that specifies guaranteed failure for all operator actions, equipment, and sensors related to systems that are not required to be operable during shutdown, including those listed above.

b. Since the response to Question 19-20(a) states that shutdown risk will be ensured through the configuration risk management program (part of the maintenance rule), provide the incremental core damage probability (ICDP) and incremental large release probability (ILRP) for each POS (where incremental is defined between the zeromaintenance shutdown PRA and the sensitivity study performed for part (a) above). Describe how the values were calculated, including how demand-based events such as overdrain were treated separately. Compare the values to the guidelines in NUMARC 93-01, Section 11, and discuss what risk management actions, if any, would be needed if minimum TS compliance were planned..

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-141

(Follow-up to Question 19-56) Provide additional information on flow diversions during shutdown. Specifically:

a. Revise page 19.1-102 of the DCD to clarify which valves can cause flow diversion to the RWSP if they are inadvertently opened. Include both the valve numbers from the PRA (9815A/B/C/D) and the valve numbers from the design piping and instrumentation diagrams (P&ID) in the rest of the FSAR, so that input to other programs such as human factors is accurate. (Note that valves 9815A/B/C/D in Figure 19.1-2, Sheet 4, appear to be the same as valves MOV-025A/B/C/D in Figure 5.4.7-2, but that the P&ID-to-PRA reference table submitted to the NRC on March 3, 2008, indicates that valves 9815A/B/C/D correspond to MOV-021A/B/C/D.)

b. Tier 1, Table 2.4.5-2 indicates that the active safety function of MOV-025A/B/C/D is to transfer open. Describe the scenarios for which MOV-025A/B/C/D are expected to transfer open to perform a safety function, and discuss the impact on plant operations of the design change to lock these valves closed. Discuss whether the valves can still be operated remotely, as stated in Table 2.4.5-2, if they are locked closed. Confirm that the impact of this design change has been addressed in analyses or documentation related to section 5.4.7 of the DCD. Revise the DCD as needed to reflect the response.

c. Revise Tier 1, Figure 2.4.5-1, to indicate that valves MOV-025A/B/C/D are now locked closed.

d. Revise Figure 19.1-2, Sheet 4, to indicate that valves 9815A/B/C/D are now locked closed.

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-142

(Follow-up to Question 19-61) The statement about mid-loop water level in the response to Question 19-61 is still unclear. The proposed paragraph would read:

"The level in the primary system is lowered to near the mid-line of the hot and cold legs. The RCS water level should be higher than 0.33 feet above the mid-loop and the RHR flow of 1,550 to 2,650 gpm should be supplied. At this water level, the air/water interface is at close proximity to the RHR suction nozzles located on the hot legs, but the higher RCS water level applied for the US-APWR design reduces the possibility of air entrainment into the RHR pump suction."

Even as revised, it appears that the antecedent of "this water level" is "0.33 feet above the mid-loop" and the statement continues to contradict itself. If the intention, as stated in the response to Question 19-61, is that "this water level" refers to the mid-loop water level used in conventional plants, the discussion in section 5.4.7.2.3.6 should be revised to state this intention.

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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APPLICATION SECTION: 19.1.6

DATE OF RAI ISSUE: 10/29/2008

QUESTION NO. : 19-143

(Follow-up to Questions 19-65) The response to Question 19-65 states that a vent valve in the pressurizer spray system is opened during draining of the RCS. Clarify by both P&ID and PRA number which valve is opened, and provide the effective area of this vent path.

ANSWER:

The pressuizer spray line vent valve opened during RCS draining is RCS-VLV-153 in the P&ID. There in no PRA number assigned to this valve since the mechanical failure of this valve it is not modeled in the PRA as a basic event.

A flow restrictor with a 0.15 inch inner diameter is installed at the connection of the vent line. Flow from the vent path will be restricted by this flow restrictor. The effective area of this vent path is therefore approximately 0.018 square inches.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-144

(Follow-up to Questions 19-65 and 19-75) The response to Question 19-75 states that "[d]uring POS 4-3 and 8-1, there is a period of time where the reactor vessel upper plenum is closed and the RCS vent paths are opened. If loss of RHR occurs during this condition, RCS pressure may exceed the design pressure of nozzle dams after initiation of bulk boiling in the RCS." The list of expeditious actions in Generic Letter (GL) 88-17 includes a direction to "[i]mplement procedures and administrative controls that reasonably assure that all hot legs are not blocked simultaneously by nozzle dams unless a vent path is provided that is large enough to prevent pressurization of the upper plenum of the [reactor vessel]."

a. Provide a description and results of calculations of RCS pressure following loss of RHR in POS 4-3 and 8-1.
b. Provide a description and results of the calculation to determine adequate vent size to meet GL 88-17.
c. Discuss how the direction in GL 88-17 to prevent pressurization is met if RCS pressure can exceed the design pressure of the nozzle dams.
d. Discuss the impact on plant operations of loss of reactor coolant via the steam generator manways following overpressure of the nozzle dams. How has this scenario has been addressed in the PRA? Are the success criteria and timing for safety injection in this scenario different from the injection functions already postulated in the loss of RHR event tree?

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-145

(Follow-up to Question 19-44) Provide the volume of water in the spent fuel pool (SFP) above the level of the gravity drain suction nozzle. State how long it would take for this water volume to be exhausted at the expected flow rate. If exhaustion would occur before the mission time of the shutdown PRA, discuss what strategy would be necessary following exhaustion (e.g., makeup from another water source) and how this strategy is addressed both in the PRA and in programs that receive input from the PRA (e.g., RAP, ITAAC, TS, human factors).

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-146

(Follow-up to Question 19-44) Clarify which valves must be opened to begin a gravity drain from the SFP. The response to Question 19-2 indicates that valves RHR-VLV-031A/D and RHR-VLV-032A/D must be opened, but Figures 5.4.7-2 and 9.1.3-1 (transfer points 2949 and 3806) indicate that the two cross-tied drain lines from the SFP connect to the RHR system via locked-closed valves VLV-033A/D and VLV-034A/D. Confirm that the correct valve numbers have been used in programs that receive input from the PRA (e.g., RAP, ITAAC, TS, human factors).

ANSWER:

Valves of the residual heat removal system that must be opened to begin gravity injection from the spent fuel pit are valves RHS-VLV-033A/D and VLV-034A/D. Valves numbers indicated in the response to question 19-2 were editorial errors.

PRA input to risk informed programs have been checked and the use of correct valve numbers has been confirmed.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: 19.1.6

DATE OF RAI ISSUE: 10/29/2008

QUESTION NO. : 19-147

(Follow-up to Question 19-44) Additional information is needed on the elevation of the gravity drain line suction nozzles in the SFP. Specifically:

a. Provide the elevation of the top of the spent fuel in the SFP.

b. Confirm that 23 feet of water, as required by TS 4.3.2, can be maintained above the spent fuel even in the scenario of gravity injection to the RCS.

c. Revise Figure 9.1.3-1, which depicts the drain lines near the top of the spent fuel, as needed.

d. Revise the description of and ITAAC for the SFP in Tier 1, section 2.7.6.2, as needed, to ensure that the SFP is constructed and inspected with a gravity injection nozzle at an elevation that will not allow the spent fuel to be endangered during draining.

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-148

(Follow-up to Question 19-44) Clarify in which POS gravity injection is credited as a mitigation strategy, and whether the reactor vessel head is removed in these POS. If the RCS is vented with a high elevation vent rather than with the head removed, surge line flooding could impede gravity injection (see the related notification from Westinghouse, Agencywide Documents Access and Management System (ADAMS) Accession No. ML013380174). Discuss how the surge line flooding issue has been addressed, if applicable.

ANSWER:

Gravity injection is credited in POS 4-2 and POS 8-2 when the reactor vessel is removed but the steam generator (SG) manhole is open. The reactor vessel head is removed during POS 4-3 and POS 8-1. However, gravity injection is not credited in these POSs since the period of time the reactor vessel is removed is short. Surge line flooding issue is not applicable since SG man hole is credited as the RCS vent for the gravity injection.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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APPLICATION SECTION: 19.1.6

DATE OF RAI ISSUE: 10/29/2008

QUESTION NO. : 19-149

(Follow-up to Question 19-76) The figures provided in response to Question 19-76 do not include enough detail for the staff to understand the US-APWR shutdown strategy. Supplement the figures with a text description, including systems and equipment in use (e.g., for depressurization, air injection to steam generator tubes, and vacuuming), necessary operator actions, and indication of when MODE and POS changes occur.

ANSWER:

This question will be answered later, within 75 days after RAI issue date.

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APPLICATION SECTION: **19.1.6**

DATE OF RAI ISSUE: **10/29/2008**

QUESTION NO. : 19-150

(Follow-up to Question 19-77) Additional information is needed on the use of RHR relief valves as the bleed path for feed-and-bleed operation during shutdown. Specifically:

a. Why are the RHR relief valves used instead of the pressurizer safety depressurization valves, which are used for feed and bleed at power?

b. Can the RHR relief valves be opened manually from the control room? If so, describe the cues and procedural guidance that would direct the operators to open the valves. If not, discuss how long it would take for pressure to increase to the RHR relief valve setpoint, the effect of the pressure increase on temporary pressure boundaries (e.g., nozzle dams), and how the PRA addresses this scenario.

c. The RHR relief valves appear not to be included in the RAP equipment list in section 17.4 of the DCD. Given that these valves are used both for LTOP protection and a primary bleed path, discuss the quantitative and qualitative justification for their exclusion from RAP. If the valves were deliberately excluded from other programs that receive input from the PRA, discuss these exclusions as well.

ANSWER:

a. RHR relief valves are spring-loaded relief valves that lift by system pressure directly acting on the valve. Safety depressurization valve will be manually opened when the RHR relief valve fail to open upon pressure increase. The purpose of depressurization is to prevent overpressure in the RCS and therefore operator does not need to voluntary open the pressure boundary before certain the pressure increases to the RHR relief valve set point.

b. RHR relief valves are spring-loaded relief valves and cannot be manually opened. Opening of the safety depressurization valve will be performed as a back up of the RHR relief valve in case they fail to open. After initiating RCS injection by the safety injection pump or the charging pump, the RCS pressure will increase up to the RHR relief valve set point within a short period of time if the pressure boundary is closed. There will be no effect on the pressure boundary. When there is considerable opening in the RCS (e.g. RCS upper plenum or man ways), the RCS pressure will be maintained atmospheric pressure regardless of injection flow. For POSs which nozzle dams are installed, steam generator man way is opened and therefore the pressure will not increase. No specific treatment of the effect of pressure increase on pressure boundaries is modeled in the PRA.

c. The contribution of RHR relief valve failure to the core damage frequency is expected to be low as discussed in response to question 19-77 of RAI No.39. However, these valves have important functions as pointed out by the staff. The RHR relief valves will be raised as an item for discussion in the expert panel for RAP.

Impact on DCD
The RHR will be discussed as a candidate for important SSCs in the RAP.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.