MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

September 25, 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-08200

Subject: **MHI's Second Responses to US-APWR DCD RAI No.39**

References: 1) "Request for Additional Information No. 39 Revision 0, SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19," dated July 29, 2008.
2) Letter MHI Ref: UAP-HF-08154 from Y. Ogata (MHI) to U.S. NRC, "MHI's Responses to US-APWR DCD RAI No.39," dated August 28, 2008.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Second Responses to Request for Additional Information No.39 Revision 0".

Enclosed are the second responses to the RAIs contained within Reference 1. In the initial responses submitted with Reference 2, MHI committed to submit responses to RAI 19-44, 55, 64, 65, 67, 68, 69, 71, 76, 77, and 78 within 60 days after RAI issue date.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

D081
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Second Responses to Request for Additional Information No.39 Revision 0 (proprietary)

3. Second Responses to Request for Additional Information No.39 Revision 0 (non-proprietary)

CC: J. A. Ciocco
C. K. Paulson

<u>Contact Information</u>

C. Keith Paulson, Senior Technical Manager
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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-08200

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Second Responses to Request for Additional Information No.39 Revision 0, September, 2008", and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design and methodology developed by MHI for performing the design of the US-APWR reactor.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with development of methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by benefits of modeling information.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 25th day of September 2008.

Y. Ogata

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

Docket No. 52-021
MHI Ref: UAP-HF-08200

Enclosure 3

UAP-HF-08200
Docket Number 52-021

Second Responses to Request for Additional Information No.39
Revision 0

September, 2008
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.39-548 REVISION 0

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-44

Gravity injection to the reactor coolant system (RCS) from the spent fuel pool (SFP) during shutdown is a mitigation strategy that is not typically seen for reactors in the United States. Overdraining the SFP could result in damage to the stored fuel. Remove discussion of this mitigation strategy and all credit in the shutdown probabilistic risk assessment (PRA), or provide the following information for the staff's evaluation: (a.) Detailed elevation drawings of the RCS and SFP, with the elevation of the high point vent (e.g., pressurizer manway) and expected equilibrium level clearly indicated (b.) Design features and associated inspections, test, analyses, and acceptance criteria (ITAAC) to ensure that the SFP cannot be drained to a level that would endanger the spent fuel (c.) Analysis results showing the gravity injection flow rate required to prevent boiling in the RCS (d.) Graphs of the driving head and gravity injection flow rate expected at various SFP levels (e.) Analysis of the consequences of overdraining the SFP (e.g., zirconium fire) (f.) Detailed procedural guidance for the evolution, including precautions and limitations provided to the operators (g.) Discussion of the controls to ensure that gravity injection does not occur inadvertently

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ANSWER:

MHI has been considering the gravity injection from SFP as an effective mitigation strategy. Therefore, MHI provides following information for NRC staff.

(a) **Elevation of the RCS and SFP**: Relationship of the elevation of reactor vessel (R/V), steam generator (S/G), pressurizer (Pzr) and SFP water level is shown in Figure 1. SFP water level is sufficiently higher than the elevation of main coolant pipe (MCP) and SG manway to inject water to RCS by gravity as shown in Figure 1. Suction nozzle of SFP is located in the upper part of SFP to prevent the water below the suction nozzle from being drained.

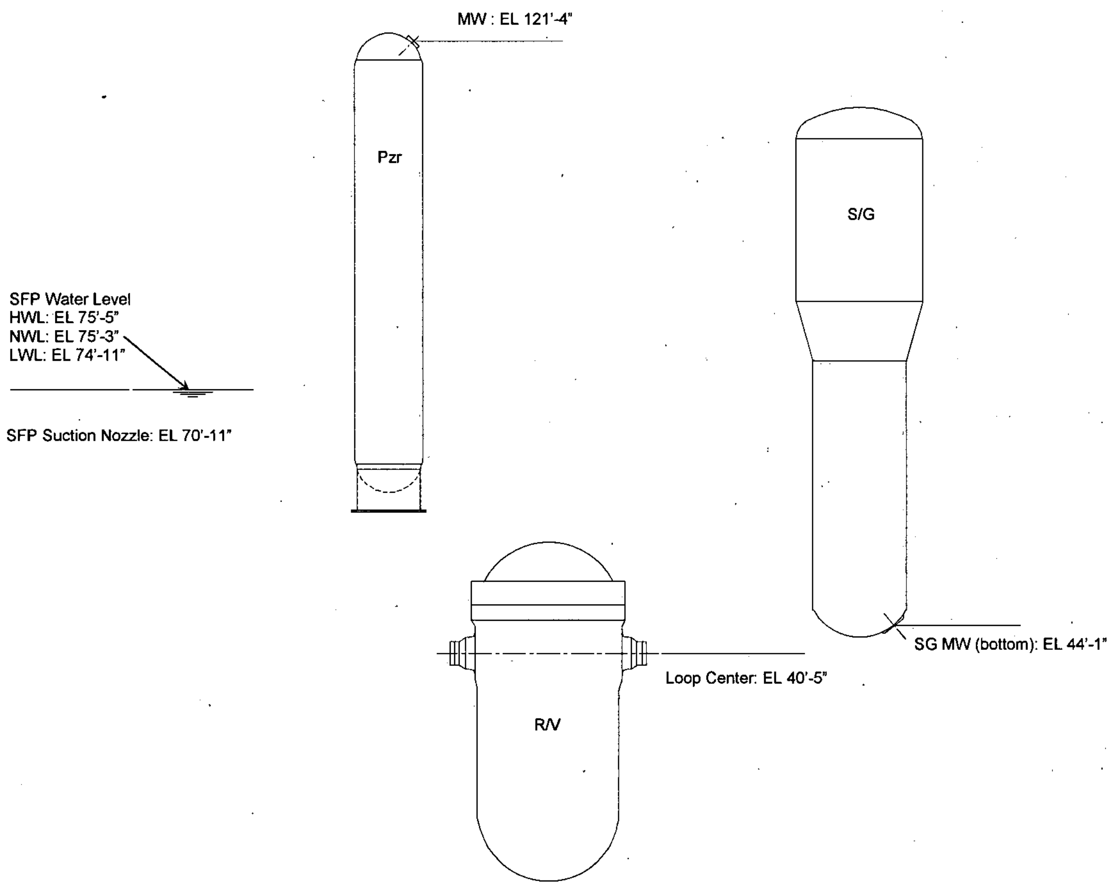

Figure 1 Relationship of the elevations

(b) **Design feature and ITAAC to ensure that the SFP cannot be drained**:

SFP is designed not to connect with the equipment drain system to preclude unanticipated drainage. In addition, suction nozzle of SFP is located on the upper part of SFP and siphon breakers are installed in cooling water return piping. These design features can prevent SFP water from unanticipated drainage.

As described in Subsection 2.7.6.2.1 of Tier 1, one of the key design features of the spent fuel storage facilities is that "To preclude unanticipated drainage, the spent fuel pit is not connected to the equipment drain system." In Table 2.7.6.2-1, item 2 indicates ITAAC for verification of functional arrangement. This ITAAC ensures that the SFP cannot be drained to a level that would endanger the spent fuel.

(c) **Required gravity injection flow rate**:

According to the assumed maintenance schedule, the time after shutdown for opening the SG manhole during the mid-loop operation is 51hr (beginning of POS 4-2). The timing of 51hr after shutdown is the most severe condition for the gravity injection because RCS is closed until that time. However, MHI conservatively applied 40hr as the elapsed time to the requirement of gravity injection. The required flow rate for gravity injection was estimated in a manner that decay heat under the condition described above can be removed by evaporation according to the latent heat under the atmospheric pressure condition. Therefore, the required flow rate for gravity injection is 173gpm or more.

(d) **Graphs of the driving head and gravity injection flow rate**:

Gravity injection flow rate by various SFP water levels is shown in Figure 2. Normal operating range of SFP water level is from EL 74'-11" to EL 75'-5". When SFP water level is within this range, the flow rate for gravity injection is around 195 gpm, which exceeds the required flow rate 173 gpm. Therefore, the gravity injection system has sufficient capacity for accident mitigation.

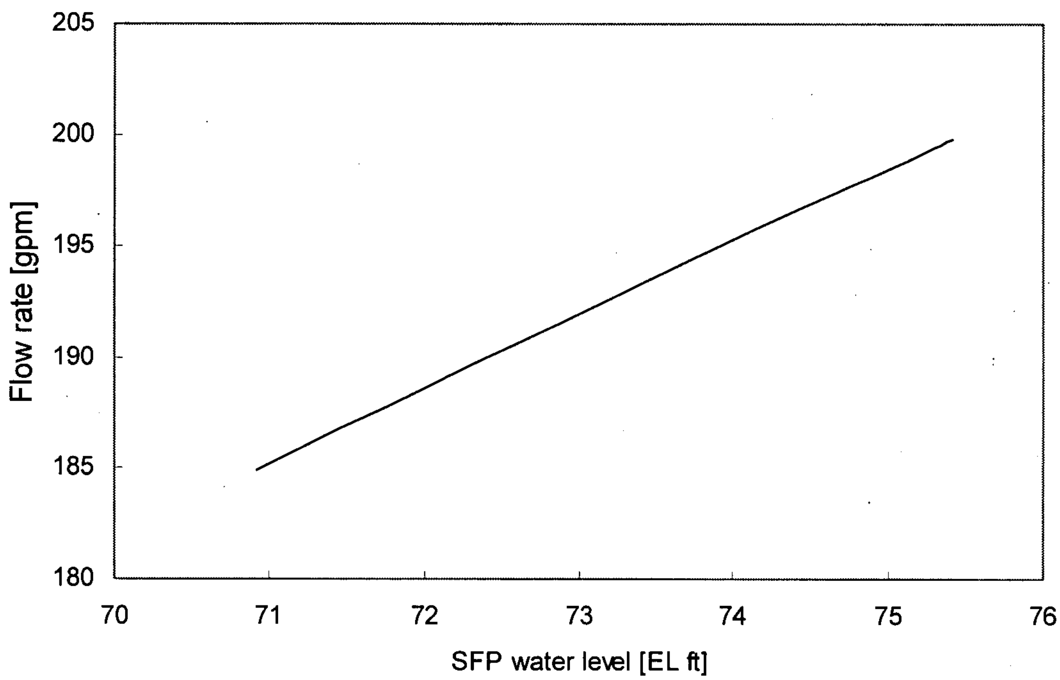

Figure 2 Gravity injection flow rate

(e) **Analysis of the consequences of overdraining the SFP**:

A overdraining from the SFP does not occur because of the design feature of US-APWR as described in the response for (b). Therefore, it is not necessary to analyze the consequences of overdraining the SFP.

(f) **Detailed procedural guidance**:

Detailed procedure for gravity injection from SFP is described in response to question No. 19-2 of RAI No.1. Please see the response.

In this operation, the following parameters should be monitored. (i) RCS water level to confirm that the RCS water level is recovered; (ii) SFP water level to keep SFP water level between normal SFP water levels; (iii) RWSP water level to confirm there is sufficient water volume to inject water to RCS.

(g) **Discussion of the controls to ensure that gravity injection does not occur inadvertently**:

Two isolation valves per one line are installed between RHRS and SFPCS (RHS-VLV-033A, -034A, -033D and -034D). These valves are usually locked closed and under administrative control to prevent inadvertent gravity injection. In addition, if inadvertent gravity injection occurs, SFP water level low water level alarm in the MCR alert the operators.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI as the response contains only additional information

<u>Impact on COLA</u>

There is no impact on COLA from this RAI as the response contains only additional information.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-55

(Follow-up to Question 19-14) The response to Question 19-14 states that if HEPs are set to high values, the conditional core damage probability (CCDP) will increase and will depend on the error assumed. The staff needs more information to understand the importance of human reliability assumptions to the shutdown CDF. Provide the results of a sensitivity study with all HEPs related to both initiating events and mitigating systems set to a high value, such as 0.5 or the 95th percentile value.

ANSWER:

MHI performed the sensitivity study of HEPs related to both initiating events and mitigating systems. In this study, all HEPs are set to the 95th percentile values and the result is compared with that for the baseline shutdown core damage quantification.

This sensitivity produces a CDF of 2.5E-06/ry, which is approximately ten times of the baseline CDF. This indicates that the assumption involving all HEPs may have a large impact on the results of PRA during shutdown and that the operators play an important role in maintaining a very low CDF.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-64

(Follow-up to Question 19-3) The response to Question 19-3 states that the RCS is opened by opening of the steam generator manhole lids, and that other openings such as the pressurizer manway or pressurizer safety valve (PSV) vent are opened at approximately the same time or later. Clarify whether an RCS vent is open during draining to mid-loop to prevent drawing a vacuum in the RCS.

ANSWER:

The vent valves in pressurizer spray system are opened during draining from RCS full to the mid-loop level to prevent drawing a vacuum in the RCS.

When the RCS level reaches to the mid-loop level, the steam generator chambers are completely drained. Then the steam generator manhole lids are opened, and the steam generator nozzle dams are set.

Effectiveness of the reflux cooling in conjunction with the efficiency of the vent valve in pressurizer spray system is to be discussed in RAI#39 19-45.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-65

Tables 19.1-76 and 19.1-77 of the DCD indicate that the RCS is closed and the steam generators are isolated in POS 4-3 and 8-1. Clarify the vent status of the RCS in these POS. 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]." Discuss how this condition is met during shutdown in the US-APWR.

ANSWER:

During draining from RCS full to the mid-loop level, a vent path is ensured to open the vent valve in pressurizer spray system.

All hot legs are blocked by nozzle dams after the RCS level reaches to the mid-loop level and a vent path has been securely established. The direction specified in the GL 88-17 is therefore met during this shutdown condition.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-67

TS 3.9.5 and 3.9.6, related to RHR during MODE 6, require containment closure within four hours whenever no RHR loops are available. The bases for this TS state that "[t]he Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time." Provide descriptions and results of time-to-boil calculations from the shutdown PRA that support this statement.

ANSWER:

MHI calculated the allowable time until the uncover of the reactor core including time-to-boil by MAAP. The analysis conditions and results are shown in Table 19.67-1.

In the result of the most severe case during the mode 6 (Low water level) when all RHRS is not available, the time to boiling is 19 minutes. In addition, the time until the uncover of the reactor core is 108 minutes. However, alternate mitigation systems such as safety injection, charging injection and gravity injection can prevent the uncover of the reactor core even in such a severe case. The frequency of the sequence in which all mitigation systems fail is estimated to be extremely low. In the case of success of alternate mitigation system, the completion time of 4 hours is assumed to be enough from point of view of operation time.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-68

(Follow-up to Question 19-21) The response to Question 19-21 states that availability of offsite power is assumed the same as at power and that a sensitivity analysis increasing the LOOP frequency 3 times resulted in a CDF increase of 40 percent. Generic data in NUREG/CR-6890 indicates a shutdown LOOP frequency of 0.196 per reactor shutdown year (/rsy), nearly five times higher than the value assumed in the shutdown PRA. Revise the shutdown PRA to use a shutdown-specific LOOP frequency; alternatively, provide a list of assumptions and associated requirements and controls that justify the use of an at-power LOOP frequency during shutdown. Clarify why Table 19.1-80 of the DCD indicates that the offsite power transformers are in standby status during shutdown.

ANSWER:

MHI will revise the shutdown PRA to use a shutdown-specific LOOP frequency. There are certain conservative evaluations for LOOP event in the current PRA model, such as (1) the allowable time to recovery and (2) the human error of re-start RHR pump. In the revision of the shutdown PRA, to prevent the excessive conservative evaluation, MHI will also reflect the detailed evaluation related to these points. The results of the MAAP analysis in the response to Question 19-69 will be considered in determination of the allowable time to recover. Therefore, the impact of setting a shutdown-specific LOOP frequency is estimated to be small.

Impact on DCD

The DCD will be revised reflecting this RAI

Impact on COLA

This RAI and its response will impact the COLA, which refers the DCD

Impact on PRA

There is an impact on PRA from this RAI, but it is estimated to be small.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-69

(Follow-up to Question 19-1) As stated in the response to Question 19-1, page 19.1-103 of the DCD indicates that the "allowable" LOOP recovery time is one hour. Provide justification for this assumption. Do any LOOP-initiated loss-of-RHR scenarios result in boiling in the RCS in less than an hour? If so, describe the scenario and provide a description and results of the time-to-boil calculation. Describe procedures and training related to closure of the equipment hatch and other containment penetrations without offsite power. State how long containment closure is expected to take both with and without offsite power.

ANSWER:

For the LOOP recovery, as described in the DCD, one hour is assumed to be the allowable time until the uncover of the reactor core. This is taken from previous PRA studies and experience with mid-loop operation, and is conservatively applied to all POSs.

As a response of this RAI, MHI calculated the allowable time until the uncover of the reactor core for each POS by MAAP. The analysis conditions and results are shown in Table 19.69-1.

Procedures and training related to containment closure are addressed in the COL item on an accident management program which will be developed in accordance with NUMARC91-06.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-71

(Follow-up to Question 19-25) The response to Question 19-25 appears to assess the impact of Type A and B outages only on LOCA-initiated accident sequences. Amend the response to include all initiating events modeled in the shutdown PRA. If the impact is significant, the baseline PRA results should be revised to reflect realistic plant outages rather than treating the exclusion of certain outage types with a sensitivity study.

ANSWER:

MHI amended the response to Question 19-25 to include all initiating events modeled in the shutdown PRA. The revised response is shown below:

In this sensitivity study, the impact of setting the yearly frequency is evaluated considering all outage types (Type A, Type B and Type C). The yearly frequency of each POS is shown in Table 19.71-1 as described in the response to Question 19-25. All initiating events for each POSs are conservatively assumed to occur because of the same factor as base case even if all outage type is considered in this sensitivity analysis. That is, only the frequency of initiating event changes according to Table 19.71-1.

This sensitivity produces a CDF of 2.7E-07/ry, which is an increase of 17 percent over the base case CDF frequency. This indicates that the impact of the yearly frequency which considered various outage types is small during plant shutdown conditions.

Note: The PRA model in response to Question 19-8 is used as a base case model in this evaluation, reflecting the requirement of RAI 19-72. The base case CDF is 2.3E-07/ry.

Table 19.71-1 Yearly frequency of each POS for LPSD PRA

POS	Type			Yearly frequency
	A	B	C	
POS 3	×	×	×	0.84
POS 4-1	N/A	×	×	0.55
POS 4-2	N/A	×	×	0.55
POS 4-3	N/A	N/A	×	0.50
POS 8-1	N/A	N/A	×	0.50
POS 8-2	N/A	N/A	×	0.50
POS 8-3	N/A	N/A	×	0.50
POS 9	N/A	×	×	0.55
POS 11	×	×	×	0.84

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-76

So that the staff can understand the US-APWR shutdown strategy, describe the expected shutdown sequence of events from entry into MODE 5 until the reactor cavity is flooded for refueling and during startup from the time when reactor cavity draining begins until entry into MODE 4. Describe the approach taken (e.g., tasks performed, systems and equipment used) for each step, including but not limited to:

(a.) Depressurization before draining the RCS

(b.) Reduction of RCS level to mid-loop

(c.) Draining the steam generator tubes

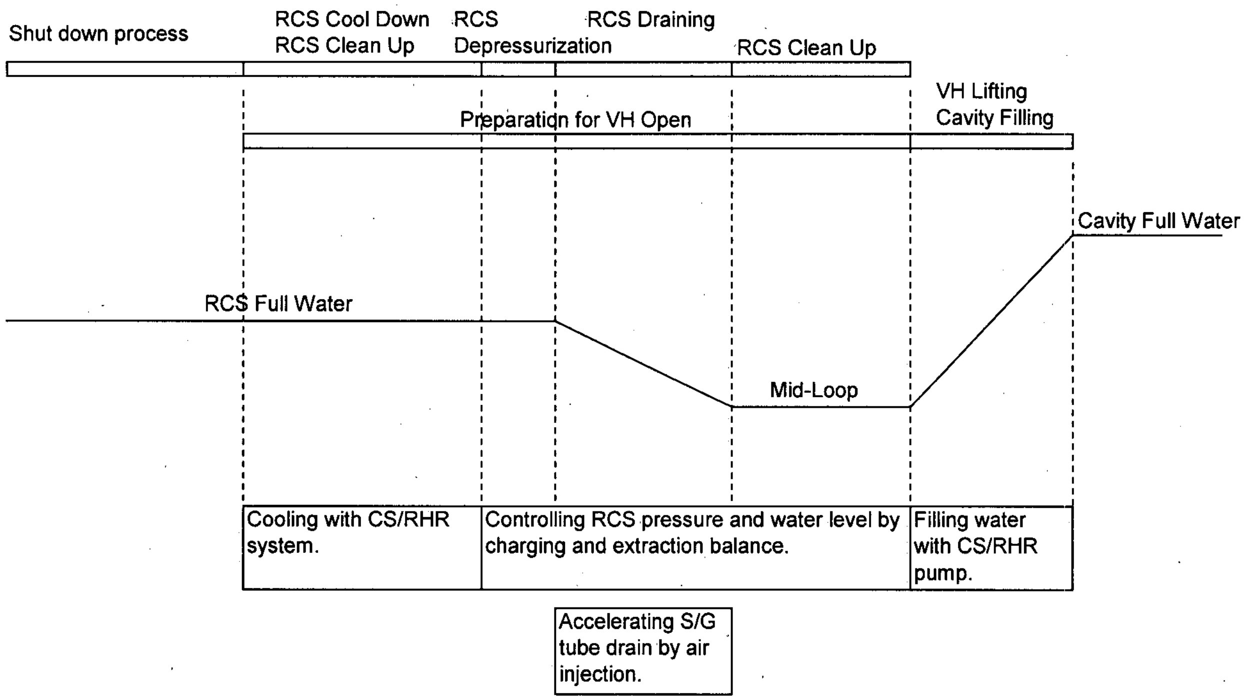
(d.) Level control during mid-loop

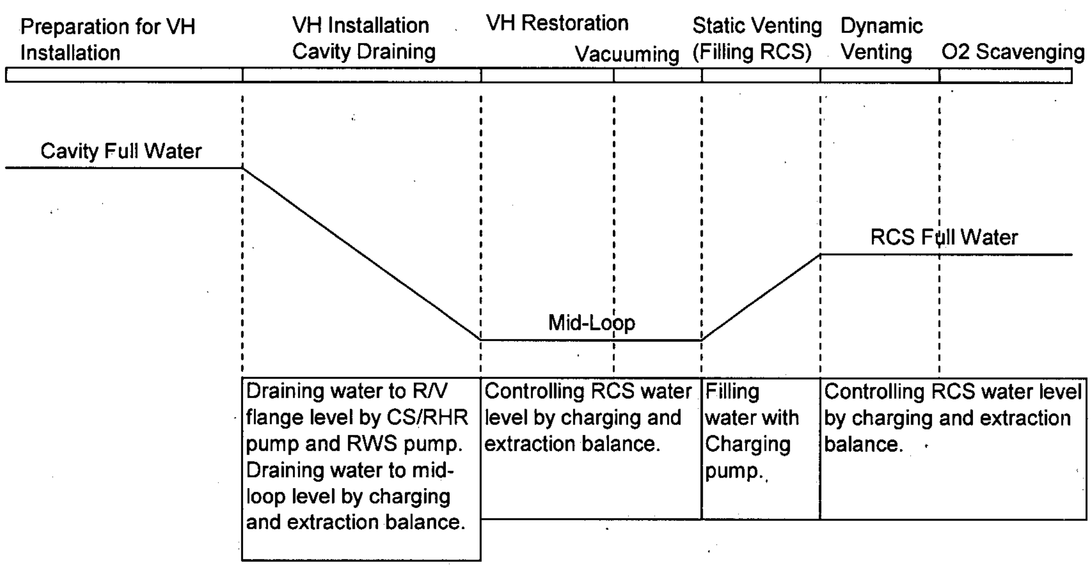
(e.) Draining the refueling cavity after refueling

(f.) Vacuum fill of the RCS

ANSWER:

Please see the following figures.


Shut down process
RCS Cool Down
RCS Clean Up
RCS
Depressurization
RCS Draining
RCS Clean Up
Preparation for VH Open
VH Lifting
Cavity Filling
Cavity Full Water
RCS Full Water
Mid-Loop
Cooling with CS/RHR system.
Controlling RCS pressure and water level by charging and extraction balance.
Filling water with CS/RHR pump.
Accelerating S/G tube drain by air injection.


Preparation for VH Installation
VH Installation
Cavity Draining
VH Restoration
Vacuuming
Static Venting
(Filling RCS)
Dynamic
Venting
O2 Scavenging
Cavity Full Water
RCS Full Water
Mid-Loop
Draining water to R/V flange level by CS/RHR pump and RWS pump. Draining water to mid-loop level by charging and extraction balance.
Controlling RCS water level by charging and extraction balance.
Filling water with Charging pump.
Controlling RCS water level by charging and extraction balance.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-77

The shutdown PRA appears to credit injection from SI and charging, but the staff could not find discussion of a primary bleed path as in the at-power model. Revise the DCD (and PRA, if necessary) to include a discussion of feed and bleed during shutdown, including how equipment and operator failures of the primary bleed path are modeled in the PRA, calculations supporting operator action timing, and the success criteria for both injection and bleed capacity in all POS.

ANSWER:

As pointed out by NRC staff, there was no description about "a primary bleed path" in the DCD and also in the PRA technical report. MHI will revise the DCD and PRA technical report reflecting the response to this RAI as shown below:

Regarding the bleed path during shutdown, there is a TS requirement for LTOP system as shown below:

"3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 — An LTOP System shall be OPERABLE with a maximum of two Safety Injection (SI) pumps and one charging pump capable of injecting into the RCS and the accumulators isolated and one of the following pressure relief capabilities:

a. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 456 psig and ≤ 484 psig, or

b. The RCS depressurized and an RCS vent of ≥ 2.6 square inches."

This requirement indicates that for most POSs evaluated in the LPSD PRA, there is a primary bleed path for which the operator action is not necessary. When the plant configuration meets the requirement of "b", the failure of the primary bleed path is not considered in the LPSD PRA. In addition, when the plant configuration meets the requirement of "a", the PRA has to consider the mechanical failure of the RHR relief valve, which can potentially "fail to open". However, the probability of the mechanical failure of the RHR relief valve is smaller than that of the failure of the feed system (High head injection system or charging injection system). Additionally, the US-APWR provides various pressure relief capabilities including the safety depressurization valve and the depressurization valve even if the RHR relief valve fail. The accident scenario for the loss of bleed path events can be therefore disregarded in the PRA because the frequency of these events is considered to be negligible due to these various design features.

Impact on DCD

The DCD will be revised reflecting this response to this RAI

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/25/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.39-548 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-78

(Follow-up to Question 19-13) Table 19.1-79 in the DCD indicates that the RCS leakage test occurs between POS 9 (cold shutdown) and POS 11 (cold and hot shutdown). However, current plants' RCS leakage tests are generally performed at operating pressure and temperature, which would appear to place the plant in MODE 3. Provide further clarification to the staff on this state, specifically:

(a) Describe the general procedure for the test.

(b) Provide the temperature, pressure, and TS MODE achieved during the RCS leakage test state (POS 10).

(c) Confirm at what point in the outage the test is performed

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ANSWER:

As for the general Japanese procedure to perform the RCS leakage test, the RCS pressure and temperature are temporarily raised up to the operating condition, and after completing the RCS leakage test, the pressure and temperature are decreased to normal to perform other inspections. So RCS leakage test is performed in MODE 3.

The detailed outage time schedule including inspection items of US-APWR is to be established. So these start-up procedures are based on the Japanese typical time schedule.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.