**MITSUBISHI HEAVY INDUSTRIES, LTD.**
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**TOKYO, JAPAN**

May 8, 2009

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-09227

**Subject:** **MHI's Responses to US-APWR DCD RAI No.266-2201 Revision 1**

**References:** 1) "Request for Additional Information No. 266-2201 Revision 1, SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19.1.6," dated March 9, 2009.

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

Enclosed are the responses to the RAI contained within Reference 1. MHI committed to submit responses to the RAI 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.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No. 266-2201 Revision 1 (proprietary version)

3. Responses to Request for Additional Information No. 266-2201 Revision 1 (non-proprietary version)

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


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## ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-09227

## 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 "Responses to Request for Additional Information No. 266-2201 Revision 1" dated May 2009, 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 8th day of May 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09227
Docket Number 52-021

Responses to Request for Additional Information
No. 266-2201 Revision 1

May 2009
(Proprietary Information Excluded)

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/8/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO. 266-2201 REVISION 1**

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

**APPLICATION SECTION:** **19.1.6**

**DATE OF RAI ISSUE:** **3/9/2009**

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**QUESTION NO.: 19-306**

(Follow-up to Question 19-138) The response to Question 19-138 does not adequately justify exclusion of passive structures, systems, and components (SSC) from the steam generator (SG) and gravity injection (GI) systems from the lists of important SSCs in the design control document (DCD). Failure of these components (e.g., piping in the gravity drain line) could have the same risk impact as failure of active components already included in the reliability assurance program (RAP). Discuss the rationale for including only active components from these systems, and revise the DCD as appropriate.

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**ANSWER:**

SSCs that highly impact the reliability of SG heat removal function are risk significant SSCs for at-power operation and are inputted to the reliability assurance program. Passive components used for SG removal are same between at-power and low power and shutdown (LPSD), and therefore, important SSCs will be the same.

If gravity injection were assumed to be inoperable throughout all POSs, the LPSD core damage frequency (CDF) increases approximately 20%. This implies that SSCs specific to gravity injection will not have risk achievement worth (RAW) higher than 2. Although failure of passive components used for gravity injection could have the same risk impact as failure of active components, they will not have RAW values exceeding the RAW criteria for risk important SSCs. Passive SSCs that generally have high reliability compared to active components are excluded from important SSCs.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

# RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/8/2009

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

| | |
|---|---|
| **RAI NO.:** | **NO.266-2201 REVISION 1** |
| **SRP SECTION:** | **19 – Probabilistic Risk Assessment and Severe Accident Evaluation** |
| **APPLICATION SECTION:** | **19.1.6** |
| **DATE OF RAI ISSUE:** | **3/9/2009** |

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**QUESTION NO. : 19-307**
(Follow-up to Question 19-139) The response to Question 19-139 does not address the implications of the residual heat removal (RHR) system success criterion completely. Provide additional information on the following subjects, and revise the DCD and probabilistic risk assessment (PRA) as appropriate.
a. Case 1 states that the reactor coolant system (RCS) condition during pressurization is within the design specification of the RHR system. Case 2 indicates that RHR fails only when level drops to the center of the main coolant piping. However, while the temperature cited is lower than the design temperature stated in DCD Section 5.4.7, it is unclear that the RHR system is designed to pump saturated or boiling water. Generic Letter (GL) 2008-01 indicates that gas accumulation of less than five percent by volume can degrade or fail pumps. Provide a description and results of a calculation of the percent gas by volume over time following a loss of RHR. Justify the assumption that the RHR pumps can operate without degradation when there is boiling in the RCS, with reference to GL 2008-01. Discuss how the function of decay heat removal is met with one RHR pump if boiling occurs in the RCS.
b. The response discusses the effect of a stricter success criterion only on the loss-of- RHR (LORH) initiating event during plant operating state (POS) 4-1. Discuss how the success criteria for support systems such as component cooling water (CCW), essential service water (ESW), and electrical power were modified when the RHR success criterion was changed. State the effects of these changes on the calculated initiating event frequencies and the overall PRA results and insights.
c. Following a loss-of-coolant accident (LOCA), over-drain event (OVDR), or failure to maintain water level (FLML), the success criterion for RHR is one of one train, given that one train is unavailable and two trains were disabled by the initiating event. If two trains of RHR are needed to remove decay heat, the RHR function would fail. Discuss the impact of this scenario on the PRA results and insights. (Note that Question 19-262 addresses recovery of additional RHR trains in this scenario.)

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*Answer:*

<u>Answer to the question a.</u>
The intention of response 19-139 with regard to boiling is described as follows:

In Case 1, as RCS is closed, the RCS is pressurized by decay heat. Boiling does not occur in the RCS. So RHR pumps are not affected by boiling.

In Case 2, RCS is opened, boiling occurs in core region when only one RHR system is operating. Steam bubble from the core is coming up to upward of the RCS and vented to the opening of the RCS. In this situation, RCS flow is very slow, so the potential of steam bubble entrainment to RHR suction is very low. Therefore, boiling does not affect the RHR pump operation.

Saturated water may affect RHR pump operation due to boiling or low NPSH. US-APWR RHR system is designed to pump saturated water. RHR pump suction piping is designed with downward pitch to prevent boiling in the suction piping. RHR pump is installed the lowest floor in the reactor building to ensure sufficient NPSH available. Therefore, RHR pump can operate with saturated water condition.

In this case, since inventory of the RCS gradually decreases due to evaporation, the RCS water level also gradually decreases. The low RCS water level causes vortex and air entrainment in the RHR suction pipe connection, and then the RHRS will eventually be inoperable. Vortex does not occur at the water level of center of the main coolant pipe, but it is assumed that RHR system can not function below this water level conservatively.

Answer to the question b.

The electrical power system is modeled in the LORH FT. The success criterion for the electrical power system is modified at the same time the RHR success criterion is changed. The contribution of the electrical power system on the LORH frequency is negligible small even if the RHR success criterion is changed.
On the other hand, the change of the success criteria for the CCW and the ESW is not considered at the same time the RHR success criterion is changed. If the success criteria for the CCW and the ESW are changed to two trains throughout POS 4-1, the CDF for the loss of CCW/ESW (LOCS) during POS 4-1 will increase approximately 20 %, but the increase of the total LPSD CDF is less than 1 %.

Answer to the question c.

Evaluation in the response to question 19-139 has not been considered that in LOCA and OVDR events heat removal by standby CS/RHR pumps (RH) becomes unavailable when the success criterion for RHR is changed during POS 4-1. But even if RH in LOCA and OVDR events becomes unavailable during POS 4-1, the CDF for both LOCA and OVDR events during POS 4-1 will increase approximately 20 %, the increase of the total LPSD CDF is approximately 1 %. Additionally, the total increase of the CDF of LOCA, OVDR, LOCS and LORH when the RHR success criterion is changed during POS 4-1 is approximately 3 %. Accordingly, the result of changing the success criteria of RHR train to one train in POS 4-1 is negligible small impact on the PRA and insights.

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.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/8/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO. 266-2201 REVISION 1**

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

**APPLICATION SECTION:** **19.1.6**

**DATE OF RAI ISSUE:** **3/9/2009**

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**QUESTION NO.: 19-308**

(Follow-up to Question 19-140) The response to Question 19-140 states that the "maintenance rule process and ... configuration risk management program are implemented to evaluate the risk of configurations being entered during shutdown." The maintenance rule risk assessment process described in NUMARC 93-11, Section 11, presents incremental core damage probability (ICDP) and incremental large release probability (ILERP) thresholds that are compared to specific configurations and durations. Because the US-APWR PRA includes maintenance unavailability assumptions, the resulting values of core damage frequency (CDF) and large release frequency (LRF) may not be the same as the baseline values used for the maintenance rule. The staff estimated a core damage probability (CDP) of 5E-7 for POS 4-1 based on information in Tables 19.1-79 and 19.1-85. Since the US-APWR shutdown PRA assumes that CDF equals LRF, the large release probability (LRP) in this POS is also 5E-7. If this LRP represented the baseline LRP, configurations raising CDF and LRF by about a factor of three would result in an ILRP above the threshold for configurations that should not be entered voluntarily. The staff needs additional information to understand the use of the US-APWR shutdown PRA for the maintenance rule. Specifically:

a. State whether the assessment of "when the conditional risk impacts are high" will be made using the shutdown PRA or qualitative considerations.

b. If the shutdown PRA is expected to be used, discuss how the baseline CDF and LRF will be calculated during shutdown. Section 11 of NUMARC 93-11 indicates that the configuration-specific CDF may consider the zero-maintenance model, but allows use of the baseline average-maintenance model. During shutdown, certain equipment must be out of service (e.g., charging pumps disabled for low-temperature overpressure (LTOP) considerations), so "zero" maintenance may not be appropriate.

c. Provide estimates of baseline CDF and LRF in each POS, where the baseline is defined as above in part (b).

d. Discuss the impact of equating CDF and LRF on the maintenance rule configuration assessment.

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**ANSWER:**

The question correctly references NUMARC 93-01 Section 11 for the determination of action thresholds for specific configurations. The ICDP and ILRP are established in section 11.3.7.2. It is noted that if ILERP is above 1E-06 the defined actions include the need to "assess non quantifiable factors" and "establish risk management actions". These are un-quantified and the normal treatment is to enter the action threshold quantitatively but thereafter include non-quantifiable factors and actions. The expected treatment for the US-APWR would be the same as currently applied.

It is correct that the current PRA equates CDF at LPSD with LRF but it is not correct that the CDF is equated to LERF (which is the metric for the maintenance rule) or that LRF would be used in its implementation. Thus to implement the maintenance rule the determination of LERF must be made. There are potential actions which could be taken after postulated accidents at mid-loop to preclude core damage or mitigate the impact. These are not credited in the LPSD PRA at this stage. The admonition in Section 11 of NUMARC 93-01 to "establish risk management actions" in configurations with risk as determined by ICDP above 1E-06 or ILERP above 1E-07 might involve assuring that the equipment hatch is closed or can be easily closed, or adding other contingencies to reduce the risk. Thus the calculated value is only the beginning first estimate of risk and may well not represent the actual risk incurred. Actual risk is reduced by risk management actions whether quantified or not.

It is demonstrably conservative to assume that the equipment hatch is open during mid-loop operation. This is normally not the case and would be controlled by the configuration risk management program. Therefore the postulated core damage event at mid-loop would likely not fit the definition of LERF. This will be determined by the final PRA at the time of fuel load that meets the then completed standard for LPSD and Level II analyses. It will be assessed in actual plant configuration planning during the implementation of the maintenance rule.

Implementation of the maintenance rule for LPSD involves evaluating many plant states (perhaps 60) including those associated with mid-loop. Each time equipment is taken out of service or returned to service represents a new configuration that is assessed. The definition of POS in the LPSD PRA simplifies this treatment and bounds the cases expected to occur. The shutdown PRA is used to determine the core damage risk of the higher risk POS conditions. Often the implementation of the maintenance rule involves combining this knowledge with the evaluation of time to uncover the core which is a precursor to core damage or the time to boil which is an earlier precursor to core damage. Thus a combination of the LPSD PRA and the much larger set of configurations possible are evaluated to determine the risk of boiling, or core un-coverage either of which can be used as a surrogate for core damage. Such surrogates are validated against the knowledge of the LPSD PRA. Of course in every case the determination of numerical CDF and LERF using the PRA can be made if it is needed.

The CDP for POS 4-1 is 5E-8. The CDF for POS 4-1 shown in table 19.1-85 is based on evaluation assuming 0.5 shutdowns per year and 39.2 hr duration for POS 4-1. Accordingly the CDF for POS 4-1 is 1.3E-9 /hr or 1.2E-5 /Y. CDP for POS 4-1 after 39.2 hr duration is 5E-8.

In response to question 19-308(a) the determination will be made with a combination of methods and reflect the knowledge of the LPSD PRA, detailed knowledge of the actual specific configuration planned, and evaluations of risk management actions taken.

In response to 19-308(b) the determination of risk will be made with a model and considerations that reflect the actual plant conditions at the time of the configuration. The reference to the use of the "zero maintenance model" refers to a standard tactic to achieve this calculation. That is, the probability that equipment is out of service an average time in the base PRA is replaced with an actual set of knowledge regarding whether or not the equipment is in service or out of service, the latter including cases in which the equipment will not or cannot function. In actual implementation the maintenance rule for shutdown is always applied in this manner as the plant conditions are well known and controlled. The use of the "zero maintenance" model is appropriate as a starting point since all equipment which is actually out of service is given a probability of 1.0 of being out of service, and equipment in service is given a zero probability of being in maintenance.

In response to 19-308(c) the zero-maintenance CDF has been estimated for the DCD rev.0 PRA results based on the tactic described above. The zero-maintenance configuration was set as the configuration with outage components, shown in table 19.1-80 of the DCD rev.1, set to be standby. Mitigation functions that are not applicable during specific POSs were set as guaranteed failure in the event tree for those POSs and were not taken credit. In this analysis, initiating event frequencies for each POSs and associated conditional core damage frequencies were separately estimated and then combined to calculate the zero maintenance CDF. The initiating event frequency of LOCW (loss of component cooling water) and LORH (loss of RHR caused by other failures such as pump failure) decreased due to increase in redundancy while frequencies of other initiating events are not impacted by configuration change. The conditional core damage probabilities given initiating events for the zero maintenance configuration are approximately the same with those of the base case since operator error are the dominant contributors of the core damage scenarios, as can seen in the dominant cut sets for POS 8-1. Therefore, conditional core damage probabilities where assumed to be the same with the DCD rev.0 PRA results. The estimated zero maintenance CDF for each POSs are shown in table 19-266. Since operator action error is the dominant contributor of US-APWR LPSD PRA, the difference in core damage frequency of zero maintenance and base case is small. The LPSD risk assessment at DCD stage does not take credit for the containment vessel (CV) condition and LRF is equated to CDF. The LRF values will be less if the CV has been taken credit.

Table 19-266 Zero maintenance CDF for LPSD

| POS | 3 | 4-1 | 4-2 | 4-3 | 8-1 | 8-2 | 8-3 | 9 | 11 | Total |
|---|---|---|---|---|---|---|---|---|---|---|
| CDF (/Y) | 1.3E-8 | 2.1E-8 | 1.4E-8 | 3.0E-8 | 4.8E-8 | 1.4E-8 | 1.5E-8 | 1.5E-8 | 2.6E-8 | 2.0E-7 |

In response to 19-308(d) the assumption that CDF equals LRF is quite conservative and inadequate for maintenance rule configuration assessment. It is only used at the DCD stage to evaluate the risk level and will be revised to better reflect the actual LERF during the completion of the PRA to meet the standards which do not define LRF but rely on LERF. The treatment of LRF at this stage will have no impact on the maintenance rule as the model considering the CV condition will be developed for maintenance rule configuration assessment.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/8/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO. 266-2201 REVISION 1**

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

**APPLICATION SECTION:** **19.1.6**

**DATE OF RAI ISSUE:** **3/9/2009**

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**QUESTION NO.: 19-309**

(Follow-up to Question 19-210) The staff needs clarification on several aspects of the response to Question 19-210.
a. The response states that "[t]he analysis results showed that the ambient air temperature at 24 hours after ESFs [engineered safety features] operation is approximately below 120 F for each area." Describe the analyses performed, including the initial temperature and any assumed operator actions (e.g., opening doors or installing fans). Define "approximately below 120 F."
b. Operator actions such as opening doors and installing temporary fans are described to justify not modeling loss of heating, ventilation, and air conditioning (HVAC) for the class 1E electrical area. Discuss how long it will take the operators to detect a loss of HVAC and carry out these actions, and compare this duration to the time before the equipment overheats. Compare the combined failure probability of HVAC and the operator recovery action with the dominant failure probabilities of the related electrical equipment. If the failures are of similar likelihood, the HVAC failure and operator actions should be modeled.
c. Describe the assumed "bounding conditions" for external air temperature. Discuss the effect that higher ambient temperatures would have on the operator actions addressed above.

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**ANSWER:**

Item a

The initial temperature is set to 105 F, which is the maximum design temperature during plant operation for the safeguard component room. The analysis assumes that at time 0, HVAC is lost and the safeguard components start to run. No operator action, such as opening of doors is taken into credit. Heat source in the room is the operating pump and the heat sink is the concrete wall.

Room temperature analysis was performed for two rooms, containment spray pump room and safety injection pump room. The containment spray pump room temperature increases up to 115

F at 24 hours after loss of HVAC. The safety injection pump room increases up to 117 F at 24 hours after loss of HVAC.

<u>Item b</u>

Loss of HVAC caused by failure in fan will initiate an alarm upon detection of fan trip or low flow. An alarm will be initiated upon detection of high temperature in class1E electrical room when the room temperature reaches 95 F, which is the design temperature.

Class 1E electrical room is located close to the main control room in the reactor building and can easily access after detection of abnormal condition of the class 1E electrical room HVAC system. Operator action to recover loss of HVAC in the class 1E electrical area is expected to be performed within 30 minutes. Loss of HVAC can be detected when the room temperature is approximately 95 F. On the other hand equipments in the class 1E electrical area can operate under 122 F and there is margin before the room temperature exceeds this value after detection of loss of HVAC. It should also be noted that exceeding this room temperature will not immediately result in equipment failure. Even if the breakers in the electrical rooms fail due to high temperature, they are likely to fail "as is" and do not impact the availability of frontline systems, which are usually started at sometime close to the initial event.

The class 1E electrical room HVAC system consists of four redundant trains, each is sized to satisfy 100% of the cooling and heating demand of two trains, i.e., train A or B can provide cooling and heating for both trains A &B, and train C or D can provide cooling and heating for both trains C & D.

Loss of class 1E electrical room HVAC is caused by failure two trains of the HVAC system, failure of the running train followed by the failure of the standby train. In such redundant systems, the failure probability is dominated by the common cause failure (CCF) during the 24 hour mission time. Applying the failure rate reported in NUREG/CR-6928 and the MGL parameters listed in the PRA report, total of CCF probabilities of two fans and two chillers, which have comparatively large failure rates, is 1.2E-4. It should be noted that the two fans and chillers both have diversity in initial conditions, that is, one train is running and the other is standby, so the likelihood of CCF probabilities are actually expected to be lower than 1.2E-4. Considering the existence of other failure modes that can result in loss of HVAC, the probability of loss of HVAC of a two Class 1E train rooms is estimated to be 2E-4.

The human error probabilities for operator recovery action given loss of HVAC is expected to be lower than 0.1. This is judged taking into account the timely detection enabled by alarms and the short time expected for the operator to access to the class 1E electrical room.

Combined failure probability of HVAC and the operator recovery action, discussed above, is expected to be lower than 2E-5. Even if room temperature occurs and equipments fail, equipments such as breakers are likely to fail "as is", so the rise in room temperature will not result in functional failure of all systems that have dependency with the equipments within the room.

This probability 2E-5 is lower than the conditional core damage probability of the dominant cutsets shown in Table 19.1-87 of the DCD. This implies that even if the class 1E electrical room HVAC system has been modeled, the impact to the results is small.

The operator recovery action for loss of class 1E electrical room HVAC is an important action to the PRA, and therefore, this action will be incorporated in the list of important assumptions.

Item c

Ambient design air temperature (0% annual exceedance maximum) is 115 F, as described in chapter 2 of the DCD. Class 1E electrical area is connected to the hall in the reactor building which is maintained below 105 F during normal operation. Taking into consideration that there are no significant heat sources in the hall and that the surrounding concrete has considerable heat capacity, the temperature in the reactor building hall is not expected to impact ability of operator action.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**5/8/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO. 266-2201 REVISION 1**

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

**APPLICATION SECTION:** **19.1.6**

**DATE OF RAI ISSUE:** **3/9/2009**

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**QUESTION NO.: 19-310**

(Follow-up to Question 19-211) The staff needs additional information to understand several aspects of the shutdown accident management framework provided in response to Question 19-211.

a. Should the framework clarify when the GI and SG mitigating functions can be used?

b. If the framework will be used to develop detailed guidance and procedures, discuss whether the mitigating functions should be listed in the order they will be used (e.g., GI is not the first means of recovering RCS inventory).

c. Clarify the statement that the safety injection (SI) "system is forced off during [shutdown] operations for maintenance purposes; therefore it is highly likely that function of SI system is intact and available for core cooling." DCD Table 19.1-81 indicates that at least two SI trains are in standby (not in maintenance) in all POS. Pumps that are out of service for maintenance are likely not to be intact and available for core cooling.

d. The statement that "operators are required to manually open the safety depressurization valves" (SDV) to avoid LTOP conditions conflicts with the response to Question 19-150, which states that the RHR relief valves are used to avoid LTOP and the SDVs provide backup if the relief valves fail. Clarify the actions taken when charging and SI are used during shutdown.

e. How do the operators determine that "an accidental incident is observed," requiring immediate containment closure? How long will it take to close containment, with and without electrical power? Discuss how procedures and training, including the accident management framework, will ensure that containment is closed before steaming from the RCS makes operator action unachievable.

f. The response to Question 19-26 indicated that the shutdown response guideline will be developed ensuring that NUMARC 91-06 is satisfied. Discuss how the accident management framework addresses NUMARC 91-06 and the more detailed guidance in GL 88-17.

g. The response to Question 19-73 stated that the accident management program will ensure that indication of temperature, pressure, and level is available during shutdown, but the framework does not address this subject. Discuss how availability of these sensors and indicators will be ensured.

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**ANSWER:**

(a) Please find the response to question b. that includes description of GI and SG. The specific timing for operation (e.g. RCS water level is below a certain value) should be viewed in the actual procedure to be developed.

(b) As pointed by the staff, MHI would like to propose changing the order of the mitigation functions in accordance with the appearance in the event trees.

- If loss of coolant water through RHRS is identified, operators are required to manually isolated the failed RHR train and stop leakage of coolant water.

- Malfunction of RHR pumps may be because of decrease of RCS water level. If the water level in the RCS is insufficient for RHR pump suction, RHR pumps are forced stopped in order to avoid failure due to cavitations. Operators are required to control the CVCS charging pumps to provide water to recover the RCS water level, accordingly the RHR function recovers. This charging injection is also expected for the decay heat removal. In parallel, operators are required to establish the lineup between RWSP and RWSAT to continuously provide source for CVCS. Water supply from RWSP to RWSAT is achieved by a motor-driven pump.

- Heat removal through the secondary system is expected during LPSD operations, including natural circulation during the operations that RCS is fully filled with water and reflux cooling during mid-loop operations. Operators are required to handle the related devices to achieve the secondary system cooling.

- SI system is forced off during LPSD operations for maintenance purposes; however at least two SI trains are in standby (not in maintenance) in all POSs. Therefore it is highly likely that function of SI system is maintained available for core cooling. Operators are required to manually activate the SI system for emergency injection.

- During safety injection and charging injection, conditions of low temperature and over pressure may occur if RHR relief valves are inoperable. In order to avoid the subsequent adverse event, operators are required to manually open the safety depressurization valves.

- If water in the spent fuel pit is available, operators are required to manually control several valves installed between SFP and RCS and gravitationally provide adequate amount of water to the RCS. In parallel, operators are required to establish the lineup between RWSP and SFP to continuously provide coolant water. Water supply from RWSP to SFP is achieved by a motor-driven pump.

(c) The word of “intact” in the answer was not appropriate, as pointed by the staff that some pumps should be out of service for maintenance and not available for core cooling. The intention of MHI using the word of “intact” was that SI function of decay heat removal still satisfied the success criterion even if some trains were in maintenance and out of services. MHI would therefore like to propose changing the description in the answer to RAI#19-211 as following:

"SI system is forced off during LPSD operations for maintenance purposes; however at least two SI trains are in standby (not in maintenance) in all POSs. Therefore it is highly likely that function of SI system is ~~intact and~~ maintained available for core cooling. Operators are required to manually activate the SI system for emergency injection."

(d) The description answered to RAI#19-211 was inaccurate as the appropriate accident management. It is true that SDV should be opened in order to avoid LTOP, although as pointed by the staff that operation is a backup function when RHR relief valves are not operable. Correct description should be as following to add the phrase with underline:

"During safety injection and charging injection, conditions of low temperature and over pressure may occur if RHR relief valves are inoperable. In order to avoid the subsequent adverse event, operators are required to manually open the safety depressurization valves."

(e) In the LPSD Level 2 PRA, the release probability under the condition that core damage occurs is assumed to be 1.0 conservatively considering the probability of containment kept opened during LPSD accident. Even under such assumption, the risk associated with the US-APWR design has been determined to be lower than the NRC goals of less than 1E-6/year for large release frequency (LRF) as described in the US-APWR DCD subsection 19.1.8.

Also framework of the accident management required by the R.G. 1.206 to be described has been specified in the US-APWR DCD subsection 19.2.5 including all those actions taken during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases.

Closure of containment during LPSD accident enhances safety associated with the US-APWR design which already satisfies the NRC goal, so procedure of containment closure will be included in a severe accident management framework to guide the COL applicant in the development of plant-specific accident management procedure. The procedure will be a symptom based one and the framework will be such that If loss of decay heat removal function as described in GL88-17 is observed, operators are required to immediately close the openings such as equipment hatch and airlock.

(f) At this stage, the accident management framework for LPSD described in Subsection 19.2.5 is basically to show countermeasures to be taken in order to prevent or mitigate important shutdown safety issues that are identified in NUMARC 91-06. More detail information in NUMARC 91-06 and GL 88-17 can be applied to develop the shutdown response guideline.

(g) Availability of components necessary for accident management will be controlled by the configuration risk management program as part of the maintenance rule. Availabilities of sensors and indicators credited in the accident management for low power and shutdown will be adequately controlled as well.

Impact on DCD

DCD will be revised in accordance with this RAI answer.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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**QUESTION NO. : 19-311**

(Follow-up to Question 19-212) Justify the different offsite power recovery failure probabilities used for each POS. Discuss how the probabilities were adjusted given the simplified modeling of POS other than 8-1.

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*Answer:*

"Probability of exceedance versus duration" during shutdown operation, which is shown in Table 4-1 of NUREG/CR-6890, Vol. 1, is applied to the failure probability of offsite power recovery for each POS. The duration time of each POS corresponds to core uncovered time in Table 20.7-1.
The CDF of LOOP of POSs other than 8-1 are adjusted based on results calculated in the Table 20.10-22. Specifically, AC probability adjusting is performed in the manner that the CDFs of "AC" failure scenario are divided by 0.53, which number is the failure probability of offsite power recovery during critical operation within one hour, and multiplied by offsite power recovery failure probabilities corresponding to the duration time for each POS.

<u>Impact on DCD</u>
There is no impact on DCD.

<u>Impact on COLA</u>
There is no impact on COLA.

<u>Impact on PRA</u>
There is no impact on PRA.

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**QUESTION NO. : 19-312**

(Follow-up to Question 19-216) The response to Question 19-216 states that instrumentation and control (I&C) components (e.g., sensors, indicators) are not modeled for manual actions. Discuss the consequences of failure of the sensors listed in the response to Question 19-73. If these failures would significantly increase the likelihood of an initiating event (e.g., OVDR) or impede use of a mitigating system, justify their exclusion from the shutdown PRA and the RAP.

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*Answer:*

(1) The consequence of failure of the sensors listed in the response to Question 19-73 is shown in Table 19-312.1.

(2) It is judged that instrumentation and control (I&C) components (e.g., sensors, indicators) are not necessary for DC PRA to be modeled because I&C failure probability is much lower than the operator action error probability.
But it seems to be necessary to take I&C components such as the RCS temperature sensors and RCS level sensors in Table 19-312.1into account as risk significant SSC for an expert panel of RAP.

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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**QUESTION NO. : 19-313**

(Follow-up to Question 19-217) The response to Question 19-217 states that maintenance outages of fire suppression pumps and alternate gas turbine generators are considered in the PRA. These components are not listed in DCD Table 19.1-80. Amend this table to include all systems credited in the shutdown PRA for which maintenance outages are expected.

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*Answer:*

As described in the second paragraph of the response "Those years averaged maintenance outage events should (have been) eliminated in the shutdown PRA.", the response to Question 19-217 means that maintenance outage of fire suppression pumps and alternate gas turbine generators had not needed to be modeled.
And the sentence "the maintenance outage of fire suppression pumps and alternate gas turbine generators are also considered... in the shutdown PRA" refers to the idea of maintenance outage treatment in the LPSD PRA. Actually these components are assumed to be available.
Additionally, both RAW of fire suppression pumps and alternate gas turbine generators are approximately 1. Accordingly, the impact on LPSD PRA is negligible small if either fire suppression pump or alternate gas turbine generator is under maintenance outage.

<u>Impact on DCD</u>
There is no impact on DCD.

<u>Impact on COLA</u>
There is no impact on COLA.

<u>Impact on PRA</u>
There is no impact on PRA.

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**QUESTION NO.: 19-314**

(Follow-up to Question 19-221) The response to Question 19-221 states that locked motor-operated valves (MOV) are controlled by breakers in the class 1E electrical room. This room is outside the control room on a different floor. However, the human reliability analysis (HRA) considers unlocking and manipulating such valves as a single subtask performed by a reactor operator (RO) with two senior reactor operators (SRO) checking the action. Discuss who will perform the breaker manipulation and who will check the action. Justify the current treatment or revise the HRA as needed.

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**ANSWER:**

One of the two RO will move to the class 1E electrical room to "unlock" the motor operated valves by manipulating the breakers. Status of these valves are monitored in the control room and the remaining RO and the SRO in the control room can check the "unlock" action performed in the class 1E electrical room.

The description of recovery actions performed by RO/SRO will be revised. This change does not impact the HRA results.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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**QUESTION NO. : 19-315**

(Follow-up to Question 19-224) The original description of the LOCA initiating event stated that a LOCA occurs if valves 9815A/B/C/D (RHR valves MOV-025A/B/C/D) are opened. The design change to lock these valves closed has made inadvertent opening of the valves unlikely. The response to Question 19-224 suggests that the LOCA event represents failure to close the valves after the two situations identified in response to Question 19-56 (draining the refueling cavity and full-flow test of the RHR pump), spurious operation of the valves followed by operator failure to close them, or both. Revise the DCD to clarify the situation represented by the LOCA initiating event. State the POS in which the two operations are expected to occur. Given that the design change makes spurious operation unlikely, discuss whether the conservative inclusion of LOCA in every POS masks the importance of other initiating events (and their mitigating systems) to overall shutdown risk.

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*Answer:*

(1) The description in page 19.1-102 of DCD will be amended as follows to clarify the situation represented by the LOCA initiating event.

Loss of coolant accident (LOCA)
During shutdown, the RCS is under low or atmospheric pressure. LOCA caused by pipe rupture are unlikely to occur. Only LOCA events that occur by operator error are considered in the PRA of LPSD - an event that would result from the inadvertent transfer of reactor coolant out of the RCS. In this evaluation, inadvertent transfer to the RWSP from the RHR, which is caused by operator failure to close the isolation valve (9815A/B/C/D) after draining the refueling cavity and full-flow test of the RHR pump, is assumed. This diversion can happen if a motor-driven valve is opened. This event is defined as a loss of all RHR trains.

(2) "Draining the refueling cavity" is done in POS8-1.
On the other hand, "full-flow test of the RHR pump" has a possibility to be done in every POS.
POS 3 seems to have a high probability of the failure to close the isolation valve (9815A/B/C/D) after the full-flow test of the RHR pump at power. But other POSs also have a probability of above.

For this reason, it is assumed that LOCA occurs in every POS in PRA.

Impact on DCD
The description in page 19.1-102 of DCD will be amended.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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**QUESTION NO. : 19-316**

(Follow-up to Question 19-225) The response to 19-225(b) states that "[a]t the plant state on page 20A.8.B-1 [of the PRA], the number of operating trains of RHR is one." DCD Table 19.1-80 indicates that at least two trains of RHR are running in all POS. Technical specifications (TS) also require at least two trains of RHR to be in operation. Will procedures direct the operators to isolate all running RHR trains if level continues to drop after letdown isolation? Discuss how the HRA addresses isolation of multiple trains.

---

*Answer:*

(1) The description of "the number of operating trains of RHR is one" in the response to RAI 19-225 (b) is wrong. And the description of "at least two trains of RHR are running in all POS" in DCD is correct.

(2) Procedures are expected to direct the operators to isolate only the unavailable RHR train due to LOCA by identifying the unavailable train. Residual heat can be removed by the intact RHR trains after the isolation.
In PRA, the isolation failure of the unavailable RHR train due to LOCA is modeled. The isolation failure of the intact running RHR trains is not necessary to be modeled because intact RHR trains are not necessary to be isolated.
Therefore, it is not necessary to consider the isolation failure of multiple trains in HRA.

<u>Impact on DCD</u>
There is no impact on DCD.

<u>Impact on COLA</u>
There is no impact on COLA.

<u>Impact on PRA</u>
There is no impact on PRA.

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**QUESTION NO. : 19-317**

(Follow-up to Question 19-237) Information on the status of the pressurizer manhole, SG manhole, spray line vent, pressurizer safety valve, vessel head, SG nozzles, and other RCS penetrations is provided in the responses to Questions 19-4, 19-69, 19-144, 19-148, and 19-237, as well as DCD Tables 19.1-98 to 19.1-105. Add a table to the DCD that clarifies the status of all relevant RCS penetrations in all modeled POS, as well as the ability to use GI and SG as mitigating functions in each POS.

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*Answer:*

MHI will revise the DCD to add a table that clarifies the status of all relevant RCS penetrations in all modeled POS, as shown in Table 19-317-1.

Impact on DCD
The DCD will be revised reflecting the response to this RAI.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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**QUESTION NO.: 19-318**

(Follow-up to Question 19-237) The response to Question 19-237 indicates that GI will now be credited as a mitigating function in POS 4-3 and 8-1. This function was previously not modeled in detail, since it was credited only in other POS. Describe the approach to developing a detailed model for the GI function, similar to other functions credited in POS 8-1. In addition, the open pressurizer manway in POS 4-3 and 8-1 means that surge line flooding, as described in Question 19-148, may impede the injection function (see the related notification from Westinghouse, Agencywide Documents Access and Management System (ADAMS) Accession No. ML013380174). Discuss how this issue has been addressed..

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**ANSWER:**

GI will be modeled in detail same as other mitigation functions modeled in POS 8-1. Mechanical failure in the injection line, spent fuel pit (SFP) and refilling line will be modeled as well as operator error to establish gravity injection and SFP refill from the RWSP. Regarding discussion on surge line flooding during POS 4-3 and 8-1, please refer response to question 19-319. As discussed in response to question 19-319, surge line flooding is prevented by US-APWR design.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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**QUESTION NO.: 19-319**

(Follow-up to Question 19-228) The response to Question 19-228 indicates that erroneous RCS level measurement is negligible because the "RCS is designed to prevent water seal in the surge line." Does this statement refer to surge line flooding? If the pressurizer manway is opened during POS 4-3 and 8-1 to provide an adequate vent path, as stated in the response to Question 19-144, discuss how correct indication of RCS level is ensured.

---

**ANSWER:**

Regarding the NRC's concerned surge line flooding issue, MHI has reviewed the related document, NUREG-1410 as indicated in the letter from Westinghouse to TVA (ADAMS Accession No. ML013380174). And again, MHI has been convinced that the statement in the answer to the Question 19-228 "RCS is designed to prevent water seal in the surge line" is appropriate for the US-APWR.

As drawn in Figure 8.14 and Figure 8.15 of NUREG-1410 for level instrumentation indicating erroneous level caused by water holdup in pressurizer, the nozzle for surge line is connected at the middle of hot leg in the horizontal direction. On the other hand, as for the US-APWR RCS design, the surge line nozzle is connected at the top of hot leg in the vertically upward direction, as shown in Figure 5.1-4 of the DCD Chapter 5. Because of this difference of the RCS design, MHI considers, as answered to the Question 19-228, that the concerned situation induced by surge line flooding is unrealistic for the US-APWR so that possibility of erroneous measurement of the RCS water level is negligible.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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**QUESTION NO.: 19-320**

(Follow-up to Question 19-262) The response to Question 19-262 states that, following a loss of inventory (i.e., LOCA, OVDR, or FLML), recognizing the event occurrence and tripping the RHR pumps will be difficult because detection measures are not specified. The staff agrees that not crediting the pump trip and subsequent recovery of RHR is conservative. However, it is unclear why "detection measures are not specified" when operators are generally trained to monitor water level during shutdown and have procedures for recovering a loss of inventory. This response contrasts with the responses to Question 19-223, which states that flow diversion pathways can be isolated by the operator upon detection of low RCS water level, and 19-225, which states that operators manually isolate letdown if automatic isolation fails and isolate RHR if level continues to drop. The automatic isolation of letdown on low level should provide a clear cue to the operators that level has reduced abnormally, even if they did not detect the level drop previously. The staff needs additional information to understand this scenario.
a. Compare these water levels: (1) the level at which operators, by procedure or training, would likely detect a loss of inventory, (2) the level at which letdown is automatically isolated, and (3) the level at which RHR pumps begin to cavitate. For the expected flow rates during the LOCA, OVDR, and FLML events, discuss how much time is available before each level is reached.
b. Discuss the expected training and procedures related to losses of inventory during shutdown. What would the operators be expected to do at each of the three levels described above?
c. Discuss the estimated risk benefit if an RHR recovery strategy (e.g., tripping the pumps before cavitation and restarting after level is restored) were included in training and procedures. Would this risk benefit increase if the pump trip were automated?.

---

**ANSWER:**

In response to question a., the water levels for each event is listed below.

(1) The level at which operators, by procedure or training, would likely detect a loss of inventory. Since the RCS water level is manually controlled by the operator, uncontrolled water

decrease in the RCS water level will be instantly detected by the operator regardless of water level. The RCS water below-normal alarm is set 60 mm above top end of main coolant piping (MCP). Therefore, during shutdown operations when the RCS water level is maintained above the MCP, operators will detect loss of inventory at this point, as a minimum.

During SG primary side water drainage and vacuum venting operation, which is operation taken during the mid-loop POSs, the RCS water level drained and kept below the top of MCP. Since the RCS water level is manually controlled by the operator, uncontrolled water decrease in the RCS water level will be detected by the operator at least when RCS water level decreases close to the RCS loop low-level signal setpoint, which is below the normal RCS water level for midloop operation. RCS loop low-level alarm will annunciate when water level decreases to approximately 5+1/2 inches above loop center, but the operators is expected to be aware of loss of inventory before the alarm initiates since RCS water level is continuously monitored.

(2) The level at which letdown is automatically isolated
Low-pressure letdown line isolation valve are automatically closed and the CVCS is isolated from the RHRS after receiving the RCS loop low-level signal, which will initiate when water level decreases to approximately 5+1/2 inches above loop center.

(3) The level at which RHR pumps begin to cavitate
The RHR pump cavitation is expected to be prevented while RCP water level is 100 mm (approximately 4 inches) above MCP center.

For FLML and OVDR, the operator can detect abnormal RCS water level decrease before water level reaches the RCS loop low-level signal set point and be ready to manually isolate the low pressure letdown line as soon as the RCS loop low-level alarm annunciates. In case of OVDR event or FLML event when the RCP loop water level is below MCP top, the operator may only have order of minutes as a minimum before RCS water level decrease to the level at which RHR pumps begin to cavitate. However, during such POSs, the operator is continuously monitoring the RCS water level before the event initiates and therefore expected to take immediate action.

For LOCA events accompanying small flow rates comparable with low pressure letdown flow rate, operators will take the same actions expected during FLML events described above. However, since LOCA has occurred, water level will keep decreasing the operator will be aware of LOCA event. If the initial RCS water level is high, such as above MCP top, the operator will have more then ten minutes before RCS water drops to the level where RHR pumps potentially begin to cavitate. If the initial RCS water level is low, it may reach the level where RHR pumps potentially begin to cavitate within few minutes depending on the leak rate. For LOCA events accompanying large flow rates, the operator may not be available to trip the RHR pump before the RCS water level where RHR pumps potentially begin to cavitate.

In response to question b., when the operators experience an abnormal decrease in RCS level and detect a loss of inventory event, the operator will manually isolate low pressure letdown line to prevent further loss of inventory below the RCS low-level. If RCS level further decreases, regardless of manual letdown line isolation and reaches the RCS low-low-level (approximately 4 inches above MCP center), alarm will annunciate and the operator will trip the RHR pump and isolate the RHR line.

In response to question c., the risk benefit of RHR recovery strategy is expected to be small for plant configuration assumed in the LPSD PRA. This is because the reliability of RHR after losses of inventory will be dominated by operator error, which will not be impacted by the number of operable pumps. As can be seen in the dominant cutsets for the core damage scenario for LOCA that involve failure of "RHR of standby pumps", show in table 20.10-8 of the PRA report, human error to start the RHR pumps will dominate the sequence which the RHR recovery strategy will impact.

In POSs where the numbers of standby pumps are not sufficient for RHR, such as POS 4-1, RHR recovery strategy will be affective to reduce the risk during that particular POS. The RHR recovery strategy can be risk beneficial during plant configurations where there are no standby RHR pumps. Manual trip of RHR pump will be included in the procedure.

Risk benefit is not expected increase risk benefit if the pump trip were automated. Automated pump trip may increase the reliability to trip the pump before it fails by cavitation, but since the RHR pump restart will need operator action, which can fail by human error, the frequency of core damage scenario involving RHR cooling will not drastically decrease. Moreover, automatic pump trip signal will introduce a cause for spurious RHR pump trip, which will cancel out the benefit of increase the reliability to protect pump.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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**QUESTION NO. : 19-321**

(Follow-up to Question 19-263) The fire-induced flow diversions listed in the response to Question 19-263 are in different locations than those modeled for internal events. Describe the procedures that will direct operators to isolate these flow diversions, and discuss how the HRA accounts for the different response to these events.

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*Answer:*

LOCA due to the spurious opening of RHS-MOV-025A (B,C,D) has been modeled in Internal Event PRA, and it is because human error which might cause the unintentional opening of such valve during shutdown condition has been postulated in their assessment. This initiating event, however, has not been modeled in Fire PRA because of the following reasons.

- It is expected that spurious opening of RHS-MOV-025 A (B,C,D) will not occur even if the control circuit of these valves has been damaged by fire, because the mal-opening of these valves will be prevented administratively by the key-lock control at MCC(Motor Control Center).
- Inter-cable hot shorts of the power cables, which run from MCC to the location of RHS-MOV-025A (B,C,D), will be prevented by routing each cable in individual conduit. Therefore, spurious opening of RHS-MOV-025A (B,C,D) by the inter-cable hot short of three-phase power supply will not occur.

It has been modeled that the scenarios of "Spurious open of SDV (Safety Depressurization Valve)" and "Spurious operation of CVS-LCV-121A (Letdown Line Volume Control Tank Inlet Change Valve)" have resulted in LOCA in Fire PRA, but these scenarios have not been modeled in Internal Event PRA. The reasons are as follows.

a. Spurious open of SDV

Spurious opening of SDV due to fire adverse effect may result in LOCA, but it is possible to prevent the occurrence of LOCA by the operator action of closing SDV isolation valve. In Fire PRA, it has been assumed that LOCA will be caused by the spurious opening of SDV due to fire effect and the operator might concurrently fail to close SDV isolation valve by human error.

Incidentally, these scenarios have been excluded from Internal Event PRA, because the frequency of human error is lower than that of fire induced spurious opening of SDV.

b. Spurious operation of CVS-LCV-121A
Spurious operation of CVS-LCV-121A due to fire adverse effect may result in LOCA, but it is possible to prevent the occurrence of LOCA by closing RHS-AOV-024 (Low-Pressure Letdown Line Isolation Valve) automatically. In Fire PRA, it has been assumed conservatively that LOCA will be caused by the spurious opening of CVS-LCV-121A due to fire effect and automatic closing of RHS-AOV-024 might fail concurrently.
Incidentally, in Internal Event PRA this scenario has been excluded because the frequency of spurious opening of CVS-LCV-121A is low.

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.