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

February 6, 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-09042

**Subject: MHI's Responses to US-APWR DCD RAI No. 138-1704 Revision 1**

**Reference:** 1) "Request for Additional Information No. 138-1704 Revision 1, SRP Section: 19-Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19.1.6.1," dated January 9, 2009

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

Enclosed are the responses to the RAIs contained within Reference 1. Of these RAIs, the following 12 RAIs will not be answered within this package.

19-206
19-207
19-208
19-209
19-214
19-218
19-222
19-230
19-234
19-236
19-247
19-249

MHI will need additional analyses and surveys for the responses to these RAIs. The responses to these RAIs will be submitted by 10th March.

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).

D081
NRO

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittal. His contact information is below.

Sincerely,

Y. Ogata

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

Enclosure:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No.138-1704 Revision 1 (proprietary version)

3. Responses to Request for Additional Information No.138-1704 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-09042

**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. 138-1704, Revision 1", dated February 2009, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain 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 the development of the unique design parameters.

B. Loss of competitive advantage of the US-APWR created by the benefits of the Control Rod Drive Mechanism operation.

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 6th day of February 2009.

Y. Ogata

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

Docket No. 52-021
MHI Ref: UAP-HF- 09042

Enclosure 3

UAP-HF-09042
Docket No. 52-021

Response to Request for Additional Information
No. 138-1704 Revision 1

February 2009
(Non-Proprietary)

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**2/6/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO.138-1704 REVISION 1**

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

**APPLICATION SECTION:** **19.1.6.1**

**DATE OF RAI ISSUE:** **1/9/2009**

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**QUESTION NO. : 19-210**

The US-APWR PRA assumes that multiple systems (e.g., residual heat removal (RHR), SI) do not need heating, ventilation, and air conditioning (HVAC) to function. Page 6A.14.4-2 of the PRA states that "according to ambient air temperature analysis, ambient air temperature will not exceed the design limit of the mitigation system during the 24 hours mission time regardless of the availability of HVAC." The staff needs additional information on this assumption. Specifically:

a. Provide a summary and results of the ambient air temperature analysis for each room or area that is assumed not to need HVAC, and state to which systems the analysis applies.

b. Discuss the PRA studies that indicate HVAC is essential to maintain ambient air temperature below the limit for the emergency feedwater (EFW) motor-driven pumps (see page 6A.14.4-2 of the PRA). Discuss why these studies are applicable only to the EFW motor-driven pumps.

c. The responses to Questions 9.4.5-23 and 9.4.5-24 indicate that the air handling units and chilled water system are required to support primary systems that mitigate a design basis accident or transient, so they are required as part of the OPERABILITY definition in TS whenever those primary systems must be OPERABLE. Discuss the differences in assumptions and analyses that result in the HVAC systems being required for TS but not for the PRA.

d. Since the ambient air temperature could be site-specific, discuss the mechanism (e.g., COL item) for COL applicants to verify these calculations and update the PRA if necessary.

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**ANSWER:**

Answer to the questions a. and b.

The following areas and equipments placed in each area are considered in PRA.

- Class 1E GTG area : GTG

- Engineered safety features (ESF) area : CS/RHR pumps and safety injection pumps

- Class 1E electrical area : I&C, switch gear, battery and battery charger

- Main control room (MCR)

- EFWP area : EFW motor-driven pumps and EFW turbine-driven pumps

The loss of HVAC in these areas can affect the equipments placed in them. In PRA, the loss of HVAC is modeled when the ambient temperature is estimated to exceed the allowable temperature in case of loss of HVAC from engineering judgment or ambient temperature analysis.

In the ESF area (safety injection pumps area and CS/RHR pumps area), the ambient air temperature analysis has been performed. In the analysis, change in the ambient temperature up to 24 hours after ESFs operation was evaluated on the condition that the HVAC is inoperable. The initial air temperature is assumed to be the same as in the normal operation condition. The analysis results showed that the ambient air temperature at 24 hours after ESFs operation is approximately below 120 F for each area. It has been confirmed that the ambient temperature does not exceed the allowable temperature (130 F) during mission time of 24hours after loss of HVAC according to the analysis. Therefore the loss of HVAC is not modeled for ESFs.

In other areas, the behavior of ambient temperature has been discussed through engineering judgment considering the area-specific characteristics.

- Class 1E GTG area : Because GTG is designed to draw air and to exhaust cooling air by using the own ventilation fan during an accident (see the DCD Chapter 9.5.8), the HVAC in the Class 1E GTG area is not required for GTG cooling. The ambient temperature rise is estimated to be suppressed through the engineering judgment. Therefore the loss of HVAC is not modeled for GTG.

- Class 1E electrical area : If the HVAC should stop during an accident, the ambient temperature rise is judged to be suppressed because it takes enough time for the ambient temperature to rise due to heat generation of electrical equipments and because operations such as opening the doors and installation of temporary fans by the personnel are expected. Therefore the loss of HVAC is not modeled for electrical equipments.

- MCR : The operator actions in the MCR can be also implemented with the remote shutdown consol (RSC). The HVAC in the MCR is not required for RSC cooling. Therefore the loss of HVAC is not modeled for operator actions in the MCR.

- EFWP area : EFW turbine-driven pumps are designed to operate for several hours without HVAC. In PRA, the HVAC for EFW turbine-driven pumps is not modeled because recovery of core cooling by RHR is expected during this time.
  On the other hand, EFW motor-driven pumps have low resistance to high temperature compared with turbine-driven pumps and are cooled only with HVAC instead of both CCW and HVAC unlike other safety-related pumps. Thus EFW motor-driven pumps are judged to be inoperable in case of loss of HVAC without performing the ambient temperature analysis. Therefore the loss of HVAC is modeled for EFW motor-driven pumps.

As described above, the HVAC only for EFW motor-driven pumps has been modeled in the PRA.

<u>Answer to the question c.</u>

The requirements for HVAC in TS are different from in PRA. The operation of equipments presupposes the operation of relevant supporting systems including HVAC in TS. On the other hand, in PRA, it is assumed that equipments function properly if they are in operation only in the mission time and if the ambient temperature is below the allowable temperature in case of loss of HVAC.

Answer to the question d.

The temperature of external air is determined by bounding conditions. Therefore the behavior of ambient air temperature is not assigned as COL item.

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

**2/6/2009**

**US-APWR Design Certification**

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**QUESTION NO.: 19-211**

(Follow-up to Question 19-58) The response to Question 19-58 states that "[t]he section 19.2.5 of DCD which describes the accident management will be revised reflecting this RAI. Development of accident management program is one of the COL items identified in Chapter 19, and will include a shutdown response guideline as part of the program to incorporate the discussions given in NUMARC 91-06." Provide marked-up pages of an interim revision so the staff can review the proposed changes.

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**ANSWER:**

Proposed change from the DCD revision 1 includes the following descriptions with underline.

### 19.2.5 Accident Management

Accident management includes 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 RV; (3) maintain containment integrity as long as possible; and (4) minimize offsite releases (Reference 19.2-7). Accident management extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design-basis into severe fuel damage regimes, and by making use of existing plant equipment and operator skills and creativity to terminate severe accidents and limit offsite releases. The US-APWR design incorporates accident management approaches in the severe accident regime and is articulated in the present subsection.

As discussed in Subsections 19.2.2 and 19.2.3 of the DCD, while the US-APWR has enhanced features for the prevention and mitigation of severe accidents, accident management remains an important element of defense-in-depth. Essential features of accident prevention and mitigation in the US-APWR design are basically the same as in operating reactors and have greater diversity of countermeasures. Accident management is used to relieve the operators of the need for rapid decisions based on operator skills and creativity, and permit greater reliance on support from outside sources, within a proceduralized framework.

Severe Accident Management Framework

The US-APWR applicant develops a severe accident management framework to guide the COL applicant in the development of plant-specific accident management procedure for the US-APWR design. This accident management procedure discusses the anticipated structure for the decision-making process, the goals to be accomplished in accident management, a summary of possible strategies for the US-APWR accident management, and potential adverse impacts of accident management strategies. A severe accident management framework includes:

- An approach for evaluating plant conditions and challenges to plant safety functions;
- Operational and phenomenological conditions that may influence the decision to implement a strategy, and which will need to be assessed in the context of the actual event; and
- A basis for prioritizing and selecting appropriate strategies, and approaches for evaluating the effectiveness of the selected actions.

The following countermeasures and operating actions are essentially addressed in the US-APWR severe accident management framework in accordance with the NRC guidance specified in the Reference 19.2-7.

(1) To prevent core damage

(During operations at power)

Key function of accident management to prevent core damage is to keep the core in a condition covered by coolant water. During operations at power, this includes core cooling, secondary cooling, containment cooling, isolation of containment bypass path, power supply, and component cooling. Countermeasures and operator actions for each function are described below.

- Accident management of core cooling function is to prevent core damage in case of LOCA and loss of safety injection. The CS/RHR pump has the function to inject the water from RWSP into the cold leg piping by switching over the CS/RHR pump lines to the cold leg piping (i.e. alternative core cooling operation). If all of safety injection systems are not available, operators are required to switch over the RHRS lines to the cold leg injection.
- Accident management of secondary cooling function is to prevent core damage in case of non-LOCA events. If emergency feedwater pumps cannot feed water to two intact SGs, operators are required to attempt to open the cross tie-line of emergency feedwater pump discharge line in order to feed water to two or more SGs by operable pumps. In case of loss of all feedwater and SG secondary side dried-out, operators are required to initiate the feed and bleed operation by starting the safety injection pump and opening the safety depressurization valve.
- Accident management of alternative containment cooling function is to prevent core damage in case of LOCA and loss of containment spray. This feature actually prevents containment failure before core damage, but not core damage itself. If containment fails

before core damage, containment temperature and pressure immediately decrease and coolant water, which is very likely to be in steam state under this condition, is rapidly lost, and eventually core damages. The containment fan cooler system is utilized as alternative containment cooling by promoting natural circulation in containment. If CSS is not activated when containment pressure monitor detects that the pressure reaches the design pressure, operators are required to switch the cooling water supply from the chilled water system to the CCW system.

- Accident management of the isolation of containment bypass path is to prevent core damage in case of SGTR and failure of ruptured SG isolation. In case of SGTR and if MSIV or turbine bypass valves (TBV) are failed to close, operators are required to close the valves, which are manual-handling valves installed upstream of TBV, in order to isolate the failed SG.
  If ruptured SG cannot be isolated, operators are required opening safety depressurization valves and intact SG secondary forced cooling with opening main steam depressurization valves to depressurize RCS. After that, it is required to connect RHR system to move into heat removal with RHR operation mode.
  If it is failed to move RHR operation mode, operators are required feed and bleed operation by starting the safety injection pump and opening the safety depressurization valve.

- Accident management of power supply is to prevent core damage in case of loss of offsite power and complete loss of emergency ac power. If both offsite power and emergency ac power are lost, operators are required to connect alternate ac power to the emergency bus.

- Accident management of component cooling function is to prevent core damage in case of loss of CCW. Either non-essential chilled water system cooling tower or ~~fire water service system~~ fire protection water supply system provides alternative component cooling water to charging pumps in order maintain RCP seal water injection. Operator action is required to connect non-essential chilled water system cooling tower or ~~fire water service system~~ fire protection water supply system to component cooling water line to charging pumps, and supply alternative component cooling water to charging pumps.

(During LPSD operations)

During LPSD operations, accident management functions to prevent core damage include gravitational water injection from SFP, activation of safety injection system, recovery of RCS water level by utilizing charging pumps, heat removal through the secondary system including reflux cooling, and RHR isolation.

- 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.

- SI system is forced off during LPSD operations for maintenance purposes; therefore it is highly likely that function of SI system is intact and available for core cooling. Operators are required to manually activate the SI system for emergency injection.

- 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.

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

- 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.

- 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.

(2) To terminate the progress of core damage if it begins and to retain the core within the reactor vessel

(During operations at power)

Core damage is identified by that both core outlet temperature and containment radiation level exceed criteria.

Essential countermeasure for termination of core damage progression and retention of core within the reactor vessel is to recover borated water injection into the reactor vessel. This is achieved by operating the safety injection system or alternative injection system. Safety injection system is provided primarily to prevent core damage however in case it fails to operate, recovery of safety injection system may be possible. Countermeasures and operator actions for each function are described below.

- Operator recovers the safety injection into RV before vessel melt through if possible.

- The alternative injection systems, such as CS/RHR pump and CVCS, are employed in case the safety injection system is down. Recovery action of the failed safety injection system is continued taken while the alternative core injection system is in operation.

- If RCS pressure is higher than the injection pump shut off head despite RCS is depressurized in case of a severe accident, additional depressurization is utilized if available.

(During LPSD operations)

During LPSD operations, accident management functions to terminate the progress of core damage are fundamentally same with the ones for operations at power.

(3) To maintain containment integrity as long as possible

(During operations at power)

Key functions of accident management to maintain containment integrity during operations at power are containment vessel isolation and decay heat removal from containment vessel. Decay heat removal is achieved in case both molten core cooling due to reactor cavity flooding and depressurization of containment vessel atmosphere are succeeded. Prevention of early containment failure due to temperature induced SGTR, hydrogen detonation and direct containment heating is also considered. Countermeasures and operator actions for each function are described below.

- Core damage is detected then operator confirms that containment vessel is properly isolated. Containment isolation may be done before core damage and hence it is required to reconfirm after core damage.

- Accident management of reactor cavity flooding is in order to cool down molten core relocated from RV breach to the reactor cavity. Decay heat is released to water and removed from containment vessel. The reactor cavity flooding is achieved utilizing the CSS and/or ~~fire water service system~~ fire protection water supply system. Molten core cooling prevents containment failure due to basemat melt through, hydrogen generation due to MCCI, etc. Operator action is initiated if the water level in the reactor cavity is lower than a criterion when core damage is detected. CSS is manually activated and water flows into the reactor cavity by gravity through the drain line. In order to utilize the ~~fire water service system~~ fire protection water supply system for the reactor cavity flooding, it is necessary to establish lineup before activating the fire water service pump.

- Accident management of containment heat removal is in order to prevent containment overpressure failure. The containment heat removal is achieved utilizing either CSS or alternative containment cooling by containment fan cooler system. CSS is one of engineered safety features and operator action is required if CSS is not automatically activated. Containment fan cooler system is a non-safety system and the fan operation is not credited during a severe accident. Cooling water is switched from chilled water system to CCW system. In order to apply the alternative containment cooling, operator pressurize CCW surge tank. This is in order to prevent boiling of CCW in the cooling unit of containment fan cooler system. ~~Fire water service system~~ Fire protection water supply system is employed in case neither CSS nor alternative containment cooling is available in order to acquire longer recovery time. ~~Fire water service system~~ Fire protection water supply system is lined up to the containment spray header and provides water as spray droplet. This operation temporarily depressurizes containment however the ~~fire water service system~~ fire protection water supply system does not contain a heat exchanger, and thus has no ability to remove heat from containment to terminate the containment pressurization.

- Accident management of prevention of early containment failure is through prevention of containment bypass, HPME and hydrogen detonation. RCS depressurization is in order for prevention of HPME and temperature-induced SGTR. When core damage is detected, severe accident dedicated depressurization valve is opened and if necessary safety depressurization valve is opened. In case water supply to SG is available, main steam depressurization valve is opened to enhance primary system cooling and depressurization if needed. Water supply to SG is recovered or controlled to avoid FP release due to temperature induced SGTR through secondary system, also to depressurize RCS. Main feedwater system or emergency feedwater system are employed for this function and operation is required when SG water level decreases below a criterion if available. Combustible gas control is in order to prevent containment failure especially due to hydrogen detonation. Although the combustible gas control is automatically achieved by hydrogen ignition system, in case CSS fails and containment vessel atmosphere is kept inerted for certain duration, CSS recovery may lead containment vessel atmosphere to combustible condition under high hydrogen concentration. In such case containment depressurization is suspended at a relatively high containment pressure. This operation is taken if combustible gas concentration is more than certain value before or when containment depressurization is in operation.

During LPSD operations

It is highly likely that containment is open during LPSD operations in order for various maintenance activities. The accident management functions to maintain containment integrity during LPSD include firstly recovery of containment isolation from the environment, and secondary heat removal from the isolated containment.

- If an accidental incident is observed, operators are required to immediately close the openings such as equipment hatch and airlock. It is also necessary to establish a method to alert personnel to evacuate containment in the event of adverse environmental conditions.

- For decay heat removal, accident management functions are fundamentally same with the ones for operations at power, i.e. reactor cavity flooding, activation of CSS or alternate containment cooling by natural circulation, or otherwise firewater injection to spray header.

(4) To minimize offsite release

(During operations at power)

Key function of accident management to minimize offsite release during operations at power is fission products removal from containment vessel atmosphere. CSS and ~~fire water service system~~ fire protection water supply system are utilized to reduce the amount of airborne FP in the containment atmosphere. Countermeasures and operator actions for each function are described below.

- Operator recovers CSS even after containment vessel failure if available.

- If CSS is not available, operator recovers ~~fire water service system~~ fire protection water supply system connected to the spray header if available.

(During LPSD operations)

It is highly likely that containment is open during LPSD operations in order for various maintenance activities. The accident management functions to minimize offsite release during LPSD include firstly recovery of containment isolation from the environment, and secondary deposition of fission products within the containment.

- If an accidental incident is observed, operators are required to immediately close the openings such as equipment hatch and airlock.

- Operators are required to activate CSS after confirming that the containment isolation is established and personnel in the containment all evacuated.

- If CSS is not available, operators are required to establish the lineup of the fire protection water supply system to the spray header.

Impact on DCD

DCD will be further revised in accordance with the description of 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

**2/6/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO.138-1704 REVISION 1**

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

**APPLICATION SECTION:** **19.1.6.1**

**DATE OF RAI ISSUE:** **1/9/2009**

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**QUESTION NO. : 19-212**

(Follow-up to Question 19-68) The response to Question 19-68 states that "MHI [Mitsubishi Heavy Industries, Ltd.] will revise the shutdown PRA to use a shutdown specific loss of offsite power (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." Discuss in greater detail the nature of these "conservative evaluations" and the new detailed assessments. Provide marked-up pages of an interim revision so the staff can review the proposed changes.

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**Answer:**

"Conservative evaluations" mean that higher failure probability of offsite power recovery and higher human error probability are applied to the top event "AC" and "RH" of the LOOP event tree.

Offsite power recovery failure probability depends on an allowable time. The allowable time is assumed as 1 hour for all POSs. However, actual allowable time of each POS is different. The actual allowable times are longer than 1 hour (the shortest time is 1.5 hours and the longest time is 24 hours). Thus, the failure probabilities of offsite power recovery will be decreased if actual allowable time is applied for all POSs.

HRA of the top event "RH" has performed the following tasks.

(1) Identify the failed running RHR pump,
(2) RHRS lineup, and
(3) Start (or Re-start) the intact RHR pumps.

In the case of LOOP, above task (1) and (3) are required to re-start RHR pumps. The task (2) is not required because status of RHRS valves is as is. If task (2) is excluded from operator actions of top event RH, human error probability of RH will decrease from the existing PRA result.

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

Impact on COLA
There is no impact on COLA.

Impact on PRA
The PRA technical report will be revised reflecting this RAI response.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**2/6/2009**

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**QUESTION NO. : 19-213**

The initiating event fault trees for the shutdown PRA presented in Attachment 20B of the PRA transfer to electrical system fault trees (e.g., EPS-STR69KA/B/C, EPS-69KEP(A/B/C)3, EPS-480EPC3, EPS-DCEPC3) that could not be located in the PRA. Are these fault trees the same as the electrical fault trees presented in Attachment 20A of the PRA, except that they are quantified with a one-hour mission time for the initiating event analysis? If the fault trees used for the initiating events are logically different from those presented in the PRA, provide them.

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**Answer:**

The electrical system fault trees modeled in the initiating event fault trees are not logically different from the electrical fault trees presented in Attachment 20A of the PRA report. The fault trees have been quantified using one-hour mission time for the initiating event analysis.

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

**2/6/2009**

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**QUESTION NO. : 19-215**

The OPSLOOP (consequential LOOP given a reactor trip) basic event appears in many locations in the shutdown PRA. Is an initiating event during shutdown expected to affect the offsite power grid, or does this event represent the probability of a random LOOP occurring during the accident? Discuss how the probability was derived.

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**Answer:**

OPSLOOP is the basic event of a random loss of offsite power during an accident at plant shutdown. The probability of this event is assigned from the probability of a consequential loss of offsite power given a reactor trip reported in NUREG/CR-6890 which used for at-power PRA. This is conservative for LPSD PRA because there is not a negative impact such as the electric disturbance due to the switchover of power sources following the generator trip at loss of off-site power.

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

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**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO.138-1704 REVISION 1**

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

**APPLICATION SECTION:** **19.1.6.1**

**DATE OF RAI ISSUE:** **1/9/2009**

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**QUESTION NO. : 19-216**

The system models in the shutdown PRA appear not to include any instrumentation and control (I&C) failures (e.g., sensors, indicators, software). (One exception is the electrical system model, which includes signal and software failures related to gas turbine starting.) Discuss why these failures, which could impede the operators' understanding of shutdown events and ability to take action, are not included in the model.

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**Answer:**

Instrumentation and control (I&C) failures (e.g., sensors, digital modules, and software) are modeled for the automatic actuation equipments, but not modeled for the manual operation equipments. This is because it is judged that I&C failure probability is much lower than the operator action one.

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. : 19-217**

The shutdown PRA fault trees include outages for several SSCs (e.g., diesel-driven and motor-driven fire suppression pumps, gas turbines, charging pumps). These out-of service probabilities are generally in the 1E-2 to 8E-3 range and appear to be in addition to the maintenance schedule stated in DCD Table 19.1-80. Discuss the rationale for including these basic events and how the probabilities were estimated. Discuss why outages for other SSCs were not included.

---

**Answer:**

In the shutdown PRA, the scheduled maintenance outages are Table 19.1-80 of the US-APWR DCD revision 1. In addition to the DCD Table 19.1-8, the maintenance outage of fire suppression pumps and alternate gas turbine generators are also considered because the SSCs which mitigate the initiating events are used in the shutdown PRA.

The out-of service probabilities are set for each POS. However, the maintenance outage events of the class 1E gas turbine generators, the charging pumps and the fire suppression pumps have been duplicated for each POS and for year averaged (out-of service probabilities are generally in the 1E-2 to 8E-3 range) unfortunately. Those year averaged maintenance outage events should be eliminated in the shutdown PRA.

These inconsistent and the over conservative basic events will be deleted from the model for the next revision of the PRA.

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

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

Impact on PRA

Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-219**

For several of the human error probabilities (HEP) evaluated in the US-APWR PRA, the human reliability analysis (HRA) in Chapter 9 of the PRA states that "frequent training has made operators very familiar with the accident sequence, and the lower bound of total HEP is assessed." Provide additional justification for this statement. How are these accident sequences and operator actions communicated to training developers to ensure that the sequences are as familiar to the operators as assumed? Revise DCD Table 19.1-113 to indicate which operator actions were given lower HEPs because of this training assumption.

---

**Answer:**

The lower bound of the total HEP is assessed for some of the operator actions which are assumed "Extremely high stress".

DCD Table 19.1-115 "Key Assumptions" in DCD Revision 1 (i.e., Table 19.1-113 in DCD Revision 0) will be revised to involve the assumption, "frequent training has made operators very familiar with the accident sequence," as shown in the attached table. These assumptions would be assured with the training programs in DCD Chapter 13.

Impact on DCD
Revise the DCD Table 19.1-115 Key Assumptions at next revision as below.

**Table 19.1-115 Key Assumptions (Sheet 2 of 4)**

| Key Assumptions |
|---|
| **Operator actions during LPSD events**<br><br>a. When the RCS is under atmospheric pressure, gravity injection from SFP is effective. Operator will perform the gravity injection by opening the injection flow path from SFP to RCS cold legs, and supplying water from RWSP to SFP.<br><br>b. In the case of loss of CCW/ESW, operator will perform alternate charging pump cooling in order to maintain RCS injection by establishing the injection flow path from fire suppression tank to charging pump and from charging pump to fire suppression tank, and starting the fire suppression pump.<br><br>c. In case LOCA occurs in RHR line, operator will perform isolation of the RHR hot legs suction isolation valves.<br><br>d. In case the RCS water level decreases during mid-loop operation and the failure of automatic isolation valve occurs, operator will perform the manual isolation of low-pressure letdown line.<br><br>**Operator actions during severe accidents**<br><br>a. Operators manually initiate active severe accident mitigation systems except for the containment isolation system and combustible gas control system which start up automatically with signals.<br><br>b. In the loss of support system sequences, operators will attempt to recover CCW/ESW or ac power while suppressing containment overpressure with firewater injection into spray header.<br><br>The following operator actions are assumed to be "frequent training has made operators very familiar with the accident sequence".<br><br>**Operator actions during at power events**<br><br>a. Establish the Alternate CCWS by Non-essential Chilled Water System Cooling Tower<br><br>b. Establish the Alternate CCWS by Fire Protection Water Supply System<br><br>c. Connect the Alternate ac Power Source to Class 1E Bus<br><br>**Operator actions during LPSD events**<br><br>a. Establish the Alternate CCWS by Fire Protection Water Supply System<br><br>b. Connect the Alternate ac Power Source to Class 1E Bus<br><br>c. Start Standby Safety Injection Pump<br><br>d. Charging Injection System Establish Operation with refilling the refueling water storage auxiliary tank (RWSAT) from refueling water storage pit (RWSP)<br><br>e. Gravitational Injection Operation |

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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**QUESTION NO. : 19-220**

Page 19.1-29 of the DCD states that a basic HEP of 0.03 was selected as a conservative HEP for type A (pre-initiator) human errors. A screening value of 0.05, not 0.03, is recommended for use in the Accident Sequence Evaluation Program HRA Procedure (ASEP) for cases where no plant visit or interaction is possible, as is the case at the design certification stage. (See page 3-32 of NUREG-1842 and page 4-2 of NUREG/CR-4772 for further details.) Justify the use of 0.03 as the basic HEP, and discuss, with support from sensitivity studies as necessary, the impact on the PRA results and insights of this choice.

---

**Answer:**

As the staff has pointed out, no plant visit or interaction is possible at the design certification stage. The HEP of 0.03 was selected based on the consideration that the modeled pre-initiator actions, such as calibration of sensors and test and maintenance, are not specific to US-APWR design and the expected reliability of these activities would be similar to that of operating plants. The applied basic HEP of 0.03 is not a conservative value, and therefore, the statement in DCD page 19.1-29 will be amended.

Type A human errors contribute less than 0.5% of the LPSD CDF as a total. This implies that the pre-initiator human errors are insignificant on the CDF and the HEP increase will not have significant impact on the PRA results and insights. Total contribution of type A human errors to the at-power CDF is less than 0.1%. The sensitivity of at-power risk to the type A human error probability is also very small. Accordingly, the uncertainty of this HEP has small impact on the PRA results and risk insights.

Impact on DCD

There first paragraph in page 19.1-29 of the DCD revision 1 will be amended as follows:

A basic HEP (BHEP) of .03 was selected as a ~~conservative~~ HEP for type A human errors. The BHEP of.03 do not include any recovery factors (RF), and represents a combination of a generic HEP of .02 assessed for an error of omission (EOM) and a generic HEP of .01 assessed for an error of commission (ECOM), with the conservative assumption that an ECOM is always possible if an EOM does not occur. The estimated HEP that is used for PRA model considers the recovery factors and dependence effect on the BHEP.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

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**QUESTION NO. : 19-221**

Does the PRA assume that locked valves are locked via a "pull-to-lock" mechanism (or software equivalent) in the control room or by a physical lock (or de-energized breaker) that must be removed locally? Discuss the impact of this assumption on the HRA (e.g., RSSOO02RHR2, in which valves must be unlocked and opened).

---

**Answer:**

Locked motor-operated valves are controlled by de-energized breaker in the class 1E electrical room. The assumptions on the HRA are assumed the operation in the main control room (MCR). The class 1E electrical rooms are located in the neighbor of the MCR and it is easy to access from MCR. Therefore, the HRA assumed that the operability in the class 1E rooms is as same as the operability in the MCR.

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. : 19-223**

(Follow-up to Question 19-56) The response to Question 19-56 did not justify the exclusion of failures (e.g., spurious operation or inadvertent opening of particular valves) that could result in a loss of reactor coolant system (RCS) inventory outside containment. Such losses of inventory would not return to the refueling water storage pit (RWSP), so the impact on mitigating systems would be different from that assessed for loss-of-coolant accidents (LOCA). Identify all flow diversion pathways that would lead to loss of RCS inventory outside containment and justify exclusion of associated failures and accident sequences from the shutdown PRA for both internal and external events.

---

**Answer:**

In the shutdown PRA, the amount of water in the RWSP required for mitigation of LOCA events is determined by the amount of water required to maintain the RCS water level above fuel top when RHR function is lost. The RWSP has sufficient water to replenish the inventory lost by evaporation for more than 24 hours, even at POS 3 when decay heat generation is severe. For this reason, loss of RCS inventory outside the containment does no impact the availability of mitigation systems during LOCA, and accordingly, the accident sequences are the same with LOCA inside the containment.

Chemical and volume control system (CVCS), residual heat removal system (RHRS) and the refueling water storage system (RWS) provide potential flow diversion pathways that would lead to loss of RCS inventory outside containment. However, such flow diversion pathways can be isolated by the operator upon detection of low RCS water level or by recovery action by workers in the field. Therefore, even if flow diversion outside the containment occur, RCS inventory leak is unlikely to continue and result in serious degradation of the core cooling function.

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. : 19-224**

(Follow-up to Question 19-56) The staff needs additional information on the HEP for inadvertent opening of valves 9815A/B/C/D (RHR valves MOV-025A/B/C/D). The errors from NUREG/CR-1278 referred to in Table 9.3.2-2 of the PRA are designated as "turn rotary control in wrong direction when there is no violation of populational stereotypes" and an error of omission when a long checkoff list is used. It is not clear that either of these error types applies to inadvertent opening of the valve during the two situations identified in the response to Question 19-56 (draining the refueling cavity and full-flow test of the RHR pump). Specifically:

a. Discuss why these error types were selected.
b. Discuss why the two identified situations are the only cases in which the valves could be inadvertently operated.
c. State which valve controls are expected to be near the controls for valves 9815A/B/C/D, such that inadvertent operation could occur.
d. Discuss how the design change to lock the valves closed (also addressed in RAI 88, Question 19-141) affects the HEP. If the HEP is reduced, discuss whether pipe ruptures, other failure modes of the valves, or other flow diversion pathways that are currently screened out become significant enough that they should be modeled as LOCA initiators.

---

**Answer:**

a. The specified error type of valves 9815 A/B/C/D which causes LOCA is failure of re-closure of the opened valves 9815 A/B/C/D by operators due to operation procedures. Two error types are considered n HEP evaluation. One is "fail to close" valves and one is "omit to close" valves.

b. "Two identified situations are the only cases" means that the opening of valves by operators with the procedures is limited in the specified POSs. The LOCAs due to spurious opening has been considered for all POSs.

c. Not specified the locations for control valves in this evaluation. Also the valves 9815 A/B/C/D are modified administratively by locked close control, any other operation errors which lead to LOCA would not occur any more.

d. As mentioned above, the locked close control of valves 9815 A/B/C/D would reduce the possibility of spurious opening valves. However it does not reduce the possibility of re-closing failure given valves opened. The major causes of LOCA are human errors and the probabilities are higher than the probabilities of other failure modes as staff has indicated. Therefore, the human error is the significant initiator of LOCA.

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. : 19-225**

The staff needs additional information to understand the modeling of the LOA top event (isolation following a LOCA during shutdown). Specifically:

a. The description of the LOAOO02LC human error in Chapter 9 of the PRA does not discuss how the operators decide whether the cause of the low RCS level is a LOCA or a failure to maintain water level (FLML) event. (The staff observes that the LOAOO02OD human error for over-drain (OVDR) and FLML events includes identification of the drain valve status.) Because different actions are taken in the two scenarios, this decision is important. Discuss how the operators make this decision and the mechanism for communicating this scenario to the developers of procedures and training.
b. The fault tree presented on page 20A.8.B-1 of the PRA represents isolation of a single train of RHR. If the only indication of a LOCA is low RCS water level, it may not be clear which train of RHR has the LOCA. Discuss how the operators determine which train to isolate, or whether all running trains of RHR are expected to be isolated. If more than one train of RHR will be isolated, revise the PRA as appropriate.
c. The fault tree presented on page 20A.8.B-1 of the PRA does not include common cause failure (CCF) of the valves to close. Justify this omission.
d. The fault tree presented on page 20A.8.B-1 of the PRA does not include any support systems (e.g., electrical power) for the valves. This omission removes a potential dependency with other top events in the sequence. Justify the omission of support systems for the valves.

---

**Answer:**

Answer to a:

When the RCS level becomes low, the letdown isolation valve will be automatically closed.

If this valve fails to close, the operators will close a downstream valve of the letdown isolation valve. Even though performed the above operations, if the RCS level continues to go down, operators will manually close the valve which is located at RHR pump suction suspecting the possibility of LOCA. Operators will not decide whether the cause of the low RCS level is a LOCA or a failure to maintain water level (FLML) event. Operator will do step by step following the symptom based procedure without decision of initiating event.

Answer to b:
It may be difficult to identify which train of RHR has the LOCA by operators. All operating trains of RHR are assumed to be unavailable in this PRA LOCA model conservatively. At the plant state on page 20A.8.B-1, the number of operating trains of RHR is one. So the fault tree presented on page 20A.8.B-1 is consistent with above assumption.

Answer to c:
The CCF between the isolation valve of No.9000A and the valve of No.9001A has not been assumed. There is clearly difference in the environment of usage between these valves. No.9000A valve is under high pressure, on the other hand No.9001A valve is not under so high. In addition, even if CCF is assumed between these valves, the failure probability of CCF will be 1.0E-3×β (7.6E-2) =7.6E-5. This probability is much lower than the probability of human error for operating these valves, which is 2.6E-3. So the failure probability of CCF does not affect to the result.

Answer to d:
The failure probabilities of support systems (e.g., electrical power) for the valves have been omitted since those probabilities will be much lower than the probability of human errors for operating these valves.

<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. : 19-226**

Discuss how each of the error types presented in Table 9.3.2-1 of the PRA (quantification of RCS drain operation failure) were selected. Several of these error types (e.g., set a rotary control to an incorrect setting) appear to be inappropriate for either the action taken or the type of control room interface assumed for the US-APWR.
Additionally, justify the assumption that each of the valves operated in the fourth task is—as stated in item 1 in Table 20-13 of NUREG/CR-1278—clearly and unambiguously labeled and set apart from valves that are similar in size and shape, state, and presence of tags.

---

**Answer:**

(1) Selection of the error types on the quantification of RCS drain operation failure
Basis of the error type selection for each task is shown in Table 1. As shown in the table, a few error types are inappropriate or over-estimated for US-APWR. The effect for CDF by modifying the HEP is less than 1 %. Changes of HEP of RCS drain failure will be incorporated in next PRA update.

(2) The manual valve management
PRA assumes that the manual valves are managed as presented in Table 9.3.2-1 of the PRA report because the manual valves are used to the RCS water level indicator and the valve operation failure leads directly to over-drain event.
Considering that the operators can easily detect the valve operation failure when perform the calibration of water level indicators, this assumption is not key assumption because the assumed scenario (the valve operation failure leads directly to over-drain event) is unlikely to occur.

**Table 1 Basis of the error type selection on the HEP Quantification of RCS Drain Operation Failure (Causing Over-drain Event)**

| Critical Tasks | | THERP Table No. | Item | HEP (Median) | THERP Description | Basis of the error type selection |
|---|---|---|---|---|---|---|
| 3 | **Stop draining with monitoring pressurizer water level** | – | – | – | – | – |
| | Omission | 20-7 | 2 | 3.0E-3 | When procedures with checkoff provisions are correctly used: Long list, > 10 items | Assume the Item (2). |
| | Fail to operate | 20-12 | 5 | 1.0E-4 | Turn rotary control in wrong direction (for two-position switches , see item 8): when there is no violation of populational stereotypes<br>(Item (8) Turn a two-position switch in wrong direction or leave it in the wrong setting: Divide HEPs for rotary controls by 5.) | Select the Item (5) as a valve operation failure, and additionally, select the Item (8) considering the US-APWR digital control panel design.<br>("Item (8)" will be added in the left column "Item" in next PRA technical report revision.) |
| 4 | **Service the reactor cavity water level indicator and RCS water level indicator** | – | – | – | – | – |
| | Omission | 20-7 | 2 | 3.0E-3 | Refer the Task No.3 | Same as Task No.3 |
| | Failure to close the RCS water drain valve on the water level indicator line | 20-13 | 1 | 1.0E-3 | Making an error of selection in changing or restoring a locally operated valve when the valve to be manipulated is: Clearly and unambiguously labeled, set apart from valves that are similar in all of the following: size and shape, state, and presence of tags. | Assume the Item (1). |
| | Failure to open the RCS water level indicator line isolation valve 01 | 20-13 | 1 | 1.0E-3 | Same as above | Same as above |
| | Failure to open the RCS water level indicator line isolation valve 02 | 20-13 | 1 | 1.0E-3 | Same as above | Same as above |
| 5 | **Calibrate the reactor cavity water level indicator and RCS water level indicator** | – | – | – | – | – |
| | Omission | 20-7 | 2 | 3.0E-3 | Refer the Task No.3 | Same as Item No.3 |
| | Miscalibration | 20-10 | 1 | 3.0E-3 | Analog meter | The left side column assumption is conservative because a digital I&C equipment is employed in US-APWR. The estimated HEP is 1.0E-3 if the Item (2) "Digital meter" is selected.<br>The HEP will be changed to 1.0E-3 in next PRA update. |
| 10c | **Operate the drain valve** | – | – | – | – | – |

| | | | | | | |
|---|---|---|---|---|---|---|
| | Fail to operate | 20-12 | 9 | 3.0E-3 | Set a rotary control to an incorrect setting (for two-position switches, see item 8)<br>(Item (8) Turn a two-position switch in wrong direction or leave it in the wrong setting: Divide HEPs for rotary controls by 5.) | Select the Item (5) as a valve operation failure, and additionally, select the Item (8) considering the US-APWR digital control panel design.<br>("Item (8)" will be added in the left column "Item" in next PRA technical report revision.)<br>In addition, the HEP shown in the left column (3.0E-3) is higher than the probability presented in THERP (2.0E-4). The HEP will be changed to 2.0E-4 in next RRA update. |

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

Impact on COLA
There is no impact on COLA.

Impact on PRA
Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-227**

(Follow-up to Question 19-73) The RCS draining procedure outlined on page 9-7 of the PRA states that RCS water level is monitored by a temporary indicator (steps 7 and 10a). However, the response to Question 19-73 states that the three water level instruments are permanent equipment. Clarify whether water level during shutdown is monitored using temporary or permanently installed indication. Revise the DCD and/or PRA to correct the discrepancy.

---

**Answer:**

All RCS water level indicators that monitor RCS draining are permanent equipments. The description of "temporary" in PRA report is needed to revise. This description will be revised at next PRA technical report revision.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Next revision of the PRA report will be reflected this response.

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**QUESTION NO.: 19-228**

(Follow-up to Question 19-73) The response to Question 19-73 states that "[i]f errors occur in the measurement [of RCS level] due to differential pressure caused by RCS inventory swept into the pressurizer, it can be considered that all RHR pumps are inoperable. In such a situation, water level in the core can be obtained by measuring the reactor vessel water level." How will the operators be informed of this possibility and be directed to observe vessel level instead of RCS level? Discuss how this insight is communicated to the developers of procedures and training.

---

**ANSWER:**

The erroneous measurement of the RCS water level under the situations that NRC is concerning is negligible because the US-APWR RCS is designed to prevent water seal in the surge line. The answer to the Question 19-73 is one of the examples that can be considered as a possible procedure. Diagnosis of the water level in hot leg can be supplemented by the reactor vessel water level; however this operators' diagnosis is not credited in the PRA, moreover the concerned situation is unrealistic for the US-APWR design. MHI does not intend to develop detailed procedures to deal with this adverse nonetheless impractical situation for the 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. : 19-229**

The HEPs for the RCS draining procedure are developed assuming that the operators are skilled and the stress level is optimum. Given that draining to mid-loop is a high-risk evolution that is likely to be performed only during infrequent refueling outages, provide additional justification for the skill and stress assumptions. What assumptions about shutdown procedures and training underlie the assumption about the operators' skill and stress levels?

---

**Answer:**

Draining is part of the normal procedures during shutdown. Operators perform draining procedure in a similar fashion to other normal procedures.
Additionally, US-APWR adopts automatic extraction isolation valves as a counter measure against over-drain.
Considering that the backup function is provided in case draining procedure has failed, the stress level during draining procedure is considered "optimum" just as other normal procedures.

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. : 19-231**

Page 20B.3-3 of the PRA states that a one-hour mission time was applied when developing the loss of CCW (LOCS) initiating event frequency. However, Table 20.B.3-5 shows mission times of 24 hours for piping leaks and 0.1 hours for heat exchanger plugging. Discuss why these failures were treated differently, and describe the impact on the shutdown PRA results and insights.

---

**Answer:**

Mission time of 24 hours for piping leak that is described in the table was incorrect. Correct value is 1 hour. The next version of the PRA report will be revised to modify incorrect description of page 20B.3-3.

The reason why the mission time of 0.1 hour for heat exchanger plugging which is described in the table is as below.

In the PRA model, the failure rate data of shell and tube type is applied for CCW heat exchanger failure. The heat exchangers of US-APWR CCW are plate type exchangers, and the failure rate of the plugging are applied as 1/10 of that for shell and tube type exchangers. Chapter 7.2 page 7-2 and 7-3 of the PRA technical report describes why the failure rate of the shell and tube type heat exchangers plugging is 1/10 of that for plate type exchanger. The value 0.1 shown in the column for exchanger plugging in Table 20.B.3-5 is the result of the true mission time (1 hour) multiplied by the reducing factor 0.1.

In the next version of the PRA report, clarify this calculation such as using the failure rate of the plate type exchanger plugging (data base ID:RHPF) and one hour mission time.

These revisions do not impact on the PRA results.

Impact on DCD
There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

The description in the Table 20.B.3-5 of the PRA report (MUAP-07030(R1)) will be revised.

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**Docket No.52-021**

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**APPLICATION SECTION:** **19.1.6.1**

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**QUESTION NO. : 19-232**

The CCF probability of the CCW heat exchangers presented in Table 20B.3-13 of the PRA is lower than the value expected based on the information in Tables 7.1-1 and 8.5-3 of the PRA. Discuss how the CCW heat exchanger CCF probability used in the shutdown LOCS model was developed.

---

**Answer:**

The CCF probability of the CCW heat exchangers is calculated from following formula.

CCF probability (7.5E-10/hr) =
Failure probability of plate-type heat exchanger (6.0E-8/hr)
$\times \beta(0.05)$
$\times \gamma(0.5)$
$\times \delta(0.5)$
($\beta$ : beta factor $\gamma$ : gamma factor $\delta$ : delta factor)

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. : 19-233**

The approach to developing the LOCS and loss of RHR (LORH) initiating event frequencies in the shutdown PRA depends on evaluating a system failure probability over a one-hour mission time and multiplying that probability by the appropriate POS duration and number of shutdown events per year. Discuss how failures on demand in both the main and support systems are handled differently from time-based failure rates in this assessment.

---

**Answer:**

In the fault tree analysis for initiating events, it is assumed that an initiating event occurs when a time based incipient event, which is a precursor of an initiating event, is followed by a series (or single) of events that result in failure to mitigate the incipient event. Events that are credit to mitigate the impact of the incipient event are either demand type or events with a 24 hour mission time.

In the LPSD PRA, incipient events are given a one hour mission time. This "one hour mission time" is not an actual "hour" but a value applied to make the basic event a time based event.

In the fault trees for initiating events, the cutsets are only the combination of time-based events under operation, or the combination of time-based events under operation and the demand-based events under stand-by. Therefore, the sum of the cutsets for initiating events multiplying that probability by the appropriate POS duration and number of shutdown events per year results in the initiating event frequency.

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

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

Impact on PRA
There is no impact on PRA.

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**QUESTION NO. : 19-235**

Table 20B.1-1 of the PRA indicates that the FLML initiating event is caused by either failure of the letdown line or failure of the chemical and volume control system (CVCS), combined with failure of letdown line isolation. These three failures are treated as independent and combined to determine an overall initiating event frequency for each applicable POS. Justify the assumption that these three failures are independent—that is, that no single failure or CCF (e.g., software, valves) could result in a FLML initiating event.

---

**Answer:**

The RCS level is manually controlled (to adjust balance of flow of the charging and the letdown by watching the RCS water level indicators). The automatic letdown line isolation system is designed to operate under the set point of the water level. The level indicators are calibrated before start the drain and they are the redundant and reliable system. The possible common cause failure will be the failures of passive equipment during the operation. Those common cause failures of the passive equipment will be negligible small.

For the letdown line isolation, the failure mode of the letdown line isolation valves are the demand type failures such as the failure of close valves and failure of the signals. These failure modes for the letdown line and the CVCS are different. Therefore, there are not CCFs between both systems.

Consequently, FLML initiating event does not occurred due to the any single failure or CCFs within each system because they do not cause the failure of letdown line isolation valves, failure of letdown line and the CVCS simultaneously.

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. : 19-237**

The footnote to Table 20.7-1 in the PRA indicates that the RCS is opened in POS 4-2, 4-3, 8-1, and 8-2. The body of the table indicates that the reactor vessel upper plenum is open in POS 8-1 and the pressurizer safety valve is removed in POS 4-2, 4-3, 8-1, and 8-2. However, DCD Tables 19.1-76 and 19.1-77 state that the RCS is closed in POS 4-3 and 8-1 and open in POS 4-2 and 8-2. Additional clarification is needed:

a. Clarify what the footnote means by "RCS is opened" (i.e., which penetrations are assumed open, and their sizes). The treatment of POS 4-1 and 8-3 in Table 20.7-1 indicates that "vented" and "opened" have two different meanings.
b. State whether the RCS is open or closed in POS 4-3 and 8-1.
c. Discuss any impact of this discrepancy on the shutdown PRA.
d. Revise the DCD and/or PRA so that the designation is consistent.

---

**Answer:**

Question a.
The footnote "(*)" of Table 20.7-1 means that the status of "RHR Relief valve" and "Pressurizer Spray valve Line" will have no effect on this PRA.
However, MHI would amend Table 20.7-1 to clarify the definitions as follows.

> RHR Relief Valves are "Enable" through POS 4-2 to POS 8-2, and
> Pressurizer Spray Vent Lines are "Open" through POS 4-2 to POS 8-2.

"vented" means that RCS has small opening area but mitigation with gravitational injection is not effective.
"opened" means that RCS has large enough opening area to mitigate with gravitational injection.
So that the treatment of POS 4-1 and 8-3 in Table 20.7-1 is "vented".

Question b.
As the response to RAI 19-44, RCS is "opened" in POS 4-3 and 8-1.

Question c.
Current PRA model does not take a credit on the gravitational injection for POS 4-3 and 8-1. However, above mentioned, MHI would amend the PRA model to take the credit on the gravitational injection for POS 4-3 and 8-1. In this case, the base case CDF is expected to reduce about 10% from the base case CDF.

Question d.
MHI will amend the PRA report and the DCD to involve the results of the case which take credit on the gravitational injection for POS 4-3 and 8-1.

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

Impact on COLA
There is no impact on COLA.

Impact on PRA
The PRA technical report will be revised reflecting the responses to this RAI.
Table 20.7-1 of the PRA report is revised as below.

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**QUESTION NO. : 19-238**

(Follow-up to Questions 19-7 and 19-45) The response to Question 19-45 indicates that the operator must open the main steam depressurization valve (MSDV) to initiate steam generator cooling following a loss of RHR. Page 19.1-103 of the DCD states that the secondary cooling function fails if the main steam relief valves (MSRV) fail to open manually. Clarify whether the operator is expected to open the air-operated MSRV, the motor-operated MSDV, or both to enable secondary cooling during shutdown. In addition, DCD Tables 19.1-107 to 19.1-114 appear not to include any valves from the main steam system as important SSCs for POS other than 8-1. Justify the exclusion of these valves and any required support systems from the tables of important SSCs and from any other programs that use the tables as input (e.g., RAP).

---

**Answer:**

The operator is expected to open the motor-operated MSDV to enable secondary cooling during shutdown and is not expected to open the air-operated MSRV.
The motor-operated MSDV should be included in DCD Tables 19.1-107 to 19.1-108 and Tables 19.1-112 to 19.1-114 as important SSCs for POS other than 8-1. The next version of the DCD will be revised to add the descriptions about the motor-operated MSDV in these Tables. However, DCD Tables 19.1-109 to 19.1-111 corresponding to POS4-2, POS4-3, POS8-1 and POS8-1 in which the secondary cooling function is unavailable are not necessary to be revised.

Additionally, MSDV has been already included as risk significant SSC for at-power PRA because secondary cooling is significant function in also at-power PRA. Thus, there is no impact on PAP.

Impact on DCD

DCD Tables 19.1-107 to 19.1-108 and Tables 19.1-112 to 19.1-114 will be revised as follows:

Table 19.1-19.1-107 Differences of Important SSCs between POS 3 and POS 8-1

| No | System | Description | Remarks |
|---|---|---|---|
| 1 | LOW PRESSURE LETDOWN LINE | N/A | This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation.<br>Therefore, this system is not modeled in POS 3. |
| 2 | RESIDUAL HEATA REMOVAL SYSTEM | Main active components of RHRS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 3: B,C,D trains. | RHR A-train is outage. |
| 3 | ~~EMERGENCY FEED WATER SYSTEM~~<br>HEAT REMOVAL via SGs | MOTOR DRIVEN EFW PUMP C<br>MAIN STEAM DEPRESSURIZATION VALVE C,D | This system is unavailable in POS 8-1 because SG is isolated from the RCS. But it is available in POS 3.<br>Motor driven EFW pump B and MSDV A,B are ~~is~~ outage. |
| 4 | HIGH HEAD INJECTION SYSTEM | Main active components of HHIS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B trains, POS 3: C,D trains. | SI pump A,B are outage. |
| 5 | CHEMICAL VOLUME CONTROL SYSTEM | Main active components of CVCS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B charging pumps, POS 3: B charging pump. | Charging pump A is outage. |
| 6 | EMERGENCY ELECTRIC POWER SUPPLY SYSTEM | Main active components of EPS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 3: A,B,C,D trains. | |
| 7 | COMPONENT COOLING WATER SYSTEM | Main active components of CCWS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 3: B,C,D trains. | CCW A-train is outage. |

Component identifiers used in this table are specific to PRA.

Corresponding components for the identifiers can be identified in Figure 19.1-2.

Table 19.1-19.1-108 Differences of Important SSCs between POS 4-1 and POS 8-1

| No | System | Description | Remarks |
|---|---|---|---|
| 1 | RESIDUAL HEATA REMOVAL SYSTEM | Main active components of RHRS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 4-1: B,C,D trains. | RHR A-train is outage. |
| 2 | ~~EMERGENCY FEED WATER SYSTEM~~<br>HEAT REMOVAL via SGs | MOTOR DRIVEN EFW PUMP C<br>MAIN STEAM DEPRESSURIZATION VALVE C,D | This system is unavailable in POS 8-1 because SG is isolated from the RCS.But it is available in POS 3.<br>Motor driven EFW pump B is outage. |
| 3 | HIGH HEAD INJECTION SYSTEM | Main active components of HHIS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B trains, POS 4-1: C,D trains. | SI pump A,B are outage. |
| 4 | CHEMICAL VOLUME CONTROL SYSTEM | Main active components of CVCS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B charging pumps, POS 4-1: B charging pump. | Charging pump A is outage. |
| 5 | EMERGENCY ELECTRIC POWER SUPPLY SYSTEM | Main active components of EPS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 4-1: B,C,D trains. | GTG A-train is outage. |
| 6 | COMPONENT COOLING WATER SYSTEM | Main active components of CCWS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 4-1: B,C,D trains. | CCW A-train is outage. |

Component identifiers used in this table are specific to PRA.

Corresponding components for the identifiers can be identified in Figure 19.1-2.

## Table 19.1-19.1-112 Differences of Important SSCs between POS 8-3 and POS 8-1

| No | System | Description | Remarks |
|---|---|---|---|
| 1 | RESIDUAL HEATA REMOVAL SYSTEM | Main active components of RHRS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 8-3: A,B,C,D trains. | |
| 2 | ~~EMERGENCY FEED WATER SYSTEM~~<br>HEAT REMOVAL via SGs | MOTOR DRIVEN EFW PUMP B<br>MAIN STEAM DEPRESSURIZATION VALVE A,B,C,D | This system is unavailable in POS 8-1 because SG is isolated from the RCS.But it is available in POS 8-3.<br>Motor driven EFW pump C is outage. |
| 3 | COMPONENT COOLING WATER SYSTEM | Main active components of CCWS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 8-3:A,B,C,D trains. | |

Component identifiers used in this table are specific to PRA.

Corresponding components for the identifiers can be identified in Figure 19.1-2.

## Table 19.1-19.1-113 Differences of Important SSCs between POS 9 and POS 8-1

| No | System | Description | Remarks |
|---|---|---|---|
| 1 | LOW PRESSURE LETDOWN LINE | N/A | This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation.<br>Therefore, this system is not modeled in POS 9. |
| 2 | RESIDUAL HEATA REMOVAL SYSTEM | Main active components of RHRS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 9: A,B,C,D trains. | |
| 3 | ~~EMERGENCY FEED WATER SYSTEM~~<br>HEAT REMOVAL via SGs | MOTOR DRIVEN EFW PUMP B<br>MAIN STEAM DEPRESSURIZATION VALVE A,B,C,D | This system is unavailable in POS 8-1 because SG is isolated from the RCS.But it is available in POS 9.<br>Motor driven EFW pump C is outage. |
| 4 | HIGH HEAD INJECTION SYSTEM | Main active components of HHIS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B trains, POS 9: A,B,C,D trains. | |
| 5 | COMPONENT COOLING WATER SYSTEM | Main active components of CCWS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 9:A,B,C,D trains. | |

Component identifiers used in this table are specific to PRA.

Corresponding components for the identifiers can be identified in Figure 19.1-2.

## Table 19.1-19.1-114 Differences of Important SSCs between POS 11 and POS 8-1

| No | System | Description | Remarks |
|---|---|---|---|
| 1 | LOW PRESSURE LETDOWN LINE | N/A | This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation.<br>Therefore, this system is not modeled in POS 11. |
| 2 | RESIDUAL HEATA REMOVAL SYSTEM | Main active components of RHRS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 11: A,B,C,D trains. | |
| 3 | ~~EMERGENCY FEED WATER SYSTEM~~<br>HEAT REMOVAL via SGs | MOTOR DRIVEN EFW PUMP B<br>MAIN STEAM DEPRESSURIZATION VALVE A,B,C,D | This system is unavailable in POS 8-1 because SG is isolated from the RCS.But it is available in POS 11.<br>Motor driven EFW pump C is outage. |
| 4 | HIGH HEAD INJECTION SYSTEM | Main active components of HHIS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B trains, POS 11: A,B,C,D trains. | |
| 5 | EMERGENCY ELECTRIC POWER SUPPLY SYSTEM | Main active components of EPS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 11: A,B,C,D trains. | |
| 6 | COMPONENT COOLING WATER SYSTEM | Main active components of CCWS are the same as POS 8-1.<br>However, trains which are available differ from POS 8-1 as follows;.<br>POS 8-1: A,B,C trains, POS 11:A,B,C,D trains. | |

Component identifiers used in this table are specific to PRA.

Corresponding components for the identifiers can be identified in Figure 19.1-2.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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**QUESTION NO. : 19-239**

Section 20A.1.A of the PRA lists cutsets for the HPI2-LOSP and HPI2-LON fault trees, but these fault trees are not presented in Section 20A.1.B where the HPI2 tree is provided or described in the text of Section 20A.1. Provide the HPI2-LOSP and HPI2-LON fault trees, and revise the PRA accordingly.

---

**Answer:**

The cutsets for the HPI2-LOSP and HPI2-LON have been calculated using the same fault trees in the section 20A.1.B and the boundary conditions respectively as below.

HPI2-LOSP: given the logical value "True" for the basic event OPSLOOP
HPI2-LON: deleted the power supply failure from HPI2 fault tree.

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. : 19-240**

Section 20A.1 of the PRA indicates that failure of limit switches associated with motor operated valves (MOV) 8820A/B, 8805A/B, and 8807A/B (piping and instrumentation diagram (P&ID) designators MOV-001A/B, MOV-009A/B, and MOV-011A/B, respectively) are risk-significant, because spurious closure of these valves would disable the safety injection line. The staff needs additional information to understand these important failures. Specifically:

a. Discuss why the limit switches are modeled separately from other valve failures. The data source for MOV failures, NUREG/CR-6928, indicates that the MOV component boundary includes the valve, the valve operator, local circuit breaker, and local instrumentation and control circuitry.
b. Discuss why limit switches are only modeled for these three sets of valves in the safety injection system and not for any other valves or systems.
c. Justify the exclusion of CCF of the MOV limit switches from the shutdown PRA.
d. Discuss why the limit switches for the valves are not explicitly included in the RAP, although the valves themselves are.

---

**Answer:**

Limit switches associated with motor operated valves detect the valve position (open or closed) and notice the states to the operators in the control room. Also the failures of the limit switches do not affect the valve function directly. The function of valve will be failed only when the simultaneous failure with valve spurious operation and associated limit switches. Therefore, it is not necessary to model a limit switch failure because the simultaneous failure is unlikely to occur. Unfortunately, some limit switch failures for valves (8820A/B, 8805A/B, and 8807A/B) have been remained in the model. Revise these unnecessary failure modes by the next update of PRA model. The elimination of those failure modes does not affect the result of PRA.

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

<u>Impact on COLA</u>

There is no impact on COLA.

Impact on PRA

Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-241**

For the HPIOO02S operator action, page 9-52 of the PRA states that the operators must start the SI pump, while page 20A.1-8 of the PRA states that the operators manually initiate the emergency core cooling system (ECCS) actuation signal. Clarify whether the operators are expected to start the pump directly or via the ECCS actuation signal, and revise the PRA accordingly.

---

**Answer:**

The direct operation of the SI pump as described in the Chapter 9 page 9-52 of the PRA report is correct. The next version of the PRA report will revise the description of page 20A.1-8 correctly.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Revise the description of page 20A.1-8 of the PRA report (MUAP-07030(R1)).

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**QUESTION NO. : 19-242**

In the shutdown PRA, the HPI2 fault tree transfers to the RWS fault tree to model RWSP failures. However, the RWS fault tree is not presented in Attachment 20A of the PRA, only in Section 6A.14.3.B. Discuss whether the at-power model of the RWSP is directly applicable to the shutdown PRA. If modifications to the RWS fault tree were made for the shutdown assessment, revise Attachment 20A of the PRA to include the fault tree.

---

**Answer:**

The shutdown PRA has used the fault tree of RWSP used by the at-power PRA in the section 6A.14.3.B.

The states (opening / closing) of the valves of RWSP are same as at-power and at shutdown. Also the RWSP contains enough water during the evaluated POSs. Therefore the same fault trees can be used for the at-power PRA and the shutdown PRA.

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. : 19-243**

The HPI-401A fault tree in the shutdown appears to model the parallel test line paths in the SI system: a 2-inch line with locked-open valve 8825A (P&ID designator VLV-023A) and 4-inch line with locked-closed valve 8813A (P&ID designator MOV-024A). The top event indicates that the failure represents "insufficient flow [in the SI system] due to test line A failure." However, the fault tree models leaks and plugging of the components in the locked-open line and leaks in the locked-closed line. It is unclear how these failure mechanisms would result in insufficient flow in the SI system. Discuss the failures HPI-401A and HPI-401B, which is similar, are intended to represent and how these failures affect SI system operation.

---

**Answer:**

The HPI-401A fault tree and HPI-401B fault tree in the shutdown model are assumed that the plugging of 8825A (8825B) valves are causes of failure of HHIS pumps starting-up and the external leaks of 8825A (B), 8813A (B) valves are causes of insufficient flow conservatively. However, HHIS pumps are able to start-up even though the plugging of valve 8825A (B) at the shutdown states. Also the leak from small line valves 8825A (B) or 8813A (B) located in the containment vessel are not the causes of RCS inventory make-up function. Those assumptions will be over estimated.

Therefore, these conservative failure modes would be eliminated at the next revision of PRA report.

<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>
PRA model and PRA report (MUAP-07030(R1)) would be revised at the next revision to eliminate the

following failure modes from the fault trees HPI-401A and HPI-401B.

- Plug of valves 8825A(B) (P&ID designator VLV-023A(B))
- Leak of valves 8825A(B) (P&ID designator VLV-023A(B))
- Leak of valves 8813A(B) (P&ID designator MOV-024A(B))

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**QUESTION NO. : 19-244**

The shutdown PRA appears not to model CCFs of equivalent valves in trains A and B of the SI system (e.g., CCF of valves 8820A and B to close). Justify the exclusion of these failures from the model.

---

**Answer:**

The CCFs modeled for the motor-operated valves are 'failure to open' and 'failure to close' as described in the Chapter 8 Table 8.5-3 of the PRA report. The motor-operated valves 8820A and B are normally open, and also remaining open during the accident. This is the reason why the CCFs of these valves are not modeled.

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. : 19-245**

The cutsets for the HPI2 fault tree in the shutdown PRA do not include CCFs of two trains of any support systems (e.g., electrical, CCW) that could disable the two SI trains. Because only the top 17 cutsets are presented, it is unclear whether support systems are modeled appropriately. Discuss, with support from additional cutsets as needed, how CCFs of SI support system components are modeled in the shutdown PRA. If crossconnects are available that allow more than two trains of support systems to support the two SI trains (and thereby requiring failures of larger groups of components to disable SI), describe these cross-connects and the operator actions needed to enable them.

---

**Answer:**

The support systems for HPI2 fault trees are modeled appropriately. The cutsets of the support systems for HPI2 fault trees are lower than the top 17 cutsets. The most higher contribution of the support system cutset of HPI2 fault trees is the external leak from CCW B header piping (basic event identifier : CWSPNELCCWB). The contribution of the cutset is 0.02 percent and the 35th in the cutsets of HPI2.

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. : 19-246**

Page 20A.3-1 of the PRA indicates that only a small amount of water is available in the refueling water storage auxiliary tank (RWSAT) in a certain POS because the RWSAT water is used for refueling. Clarify to which POS this statement refers, given that the PRA does not model the POS in which the refueling cavity is flooded. If the RWSAT is full during the POS modeled in the PRA, discuss why the PRA assumes that RWSAT makeup is needed. Discuss the impact of the use of RWSP water for refueling on the assumption that it is available for RWSAT makeup.

---

**Answer:**

The POSs that the PRA does not model the RWSAT in which the refueling cavity is flooded are POS5, POS6 (no fuels in the core) and POS7.

The PRA model assumed that the RWSAT is needed RWSAT makeup conservatively. This is because that the inventory of RWSAT is dependent on operating process and it may be possible in a certain state that the inventory of the RWSAT is not sufficient.

The POSs in which the RWSP water is possible to be used for refueling are POS5, POS6 and POS7. Those POSs are out of scope of the shutdown PRA model because of the sufficient water in the cavity for residual heat removal.

<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. : 19-248**

According to the DCD, the charging pump and high head SI pump flow rates (275 gallons per minute (gpm) and 1540 gpm, respectively) are lower than that of the RHR pumps. The success criteria for the CV and SI top events require only one of either type of pump. Given that a single RHR train does not remove enough decay heat to prevent boiling in certain POS (see Questions 19-46 and 19-139), justify the success criteria for charging and SI for all modeled POS. As needed, provide a description and results of supporting thermal-hydraulic calculations.

---

**Answer:**

The loss of inventory due to the decay heat is the largest in POS3 (348gpm) among the modeled POSs. Rate of inventory loss at POS3 exceeds the normal charging pump rated flow (275gpm). However, the maximum flow rate of the charging pump is about 440gpm, and the pump head at this flow rate is sufficiently higher than the RCS pressure during POS 3 (398 psi - 469 psi). Therefore, injection by one charging pump can recover the loss of inventory due to decay heat. Accordingly, the success criteria for charging pump that require only one pump is adequate in all POSs.
In addition, flow rate of SI pump is sufficiently larger than that of charging pump, so the success criteria for SI pump requiring only one pump is also adequate in all POSs.

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. : 19-250**

Clarify how the CCF probabilities for the CHIMVCD121BC-ALL and CHIMVOD121DEALL (failures of VCT valves to close and RWSAT valves to open, respectively) were derived. The assumed probability of MOVs failing to open or close on demand is 1.0E-3 (PRA Table 7.1-1), and the assumed beta factor for two charging MOVs is 0.14 (PRA Table 8.5-3). In contrast, the CCF probability cited on Page 20A.5.A-21 of the PRA is 4.7E-5.

---

**Answer:**

The beta factor value (0.047) from OTHER2 CCF data in "Generic Components data" based on NUREG/CR-5485 was applied to the both valves.

However, the factor value (0.14) for motor operated valves will be more applicable to the MOVs of the charging system. The effect of this data revision will be negligible.

In the next revision of the PRA report, beta factor value (0.14) will be used for the two MOVs.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-251**

Clarify why the probability of a charging pump failing to start (CHIPMBDCHPA/B) is presented as 2E-3 in PRA Table 20A.3-5, as 1.796E-3 in the cutsets for the CHI and CHI21 fault trees, and as 1.8E-3 in PRA Table 18.2-1. Discuss the impact of this difference on the shutdown PRA results and insights.

---

**Answer:**

The probability (1.8E-3) of a charging pump failing to start, which is presented in the cutsets for the CHI and CHI21 fault trees and in the PRA report Table 18.2-1 (the result of the importance analysis), is the single failure probability of the charging pump failing to start described Chapter 7 Data Analysis of the PRA report. On the other hand, the probability (2.0E-3) presented in Table20A.3-5 (the basic event list) is the sum of the single failure probability (1.8E-3) and the CCF probability (2.0E-3×$\beta$(0.1)=2.0E-4).

US-APWR PRA used the CCF group function of the RiskSpectrum software. With this function, the CCF events in the cutsets are generated automatically by specifying the grouping of the CCF events and the MGL parameters. In the PRA report, the basic event lists describe the probabilities including the CCF values. The quantified results (the cutsets and the importance analysis) describe the values without CCF values.

<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. : 19-252**

The CHI11 fault tree in the shutdown PRA includes the RWSSTRWSP basic event for RWSP failure, rather than transferring to the RWS event tree, as in the HPI2 fault tree. The failure probability of RWSSTRWSP in the CHI11 fault tree is approximately ten orders of magnitude lower than the RWS failure probability indicated in the HPI2 fault tree. Discuss why RWSP failures are modeled differently for the charging system (where the RWSP provides makeup to the RWSAT) than for the SI system.

---

**Answer:**

The basic event (RWSSTRWSP) for RWSP failure in the fault tree (CH11) of the charging system is necessary to link the fault tree (RWS) of RWSP as well as the fault tree (HPI2) of high head injection system. Although the dependency of the RWSP between the charging injection system and the high head injection system is appropriately incorporated with the linking, the increase of CDF will not significant because the human error probabilities dominate the CDF of the shutdown PRA.

The revised PRA model will be incorporated in the next PRA technical report revision.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-253**

The CHI-13A fault tree, which models the A train of RWR for the shutdown PRA, models external leaks from valves 005 and 007 in the B train and from valve 006 in the A train. The CHI-13B fault tree models the B train similarly. Valves 005, 006, and 007 are in series in each train. Clarify why each fault tree includes equipment from the other train.

---

**Answer:**

In fault tree analysis of RWR pump line failure, it is assumed that the function of both trains could be lost by the lack of injection flow during the external leak event. That is to say, any credit for train separation is not taken in this case.

Therefore, failure of different train components is also modeled together.

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. : 19-254**

The shutdown PRA appears not to model CCFs of RWR pumps A and B. Justify the exclusion of these failures from the model.

---

**Answer:**

CCFs of RWR pumps have not been modeled because the unavailability of this system is dominated by human errors and the CCFs of RWR have not been specified.
However, MHI would like to reconsider the CCFs of RWR pumps A and B and to update the model by the next PRA model update.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-255**

The CHI fault tree, which models the CV top event in the shutdown PRA, includes the CHIOO02CV21 and CHIOO02CV2 operator actions. CHIOO02CV21 models operator failure to start the charging pump remotely to recover RCS level when SI fails following a LOCA or OVDR event. CHIOO02CV2 appears to model initiation of makeup to the RWSAT from the RWSP so that charging injection can continue long-term. However, CHIOO02CV2, as described on page 9-48 of the PRA, also includes operator failure to start the charging pump. Clarify why starting the charging pump is required for both operator actions. Does the charging pump trip on low RWSAT level and need to be restarted?

---

**Answer:**

The operator failure to start the charging pump does not need to be modeled in the CHIOO02CV2. The next version of the PRA will be revised to eliminate the unnecessary HEP for operator failure to start the charging pump from CHIOO02CV2, but this revision has negligible impact on the PRA results.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-256**

The CHIOO02CV212 basic event appears in many cutsets in Chapter 20 of the PRA, but could not be found in any of the fault trees in Attachment 20A. Chapter 9 indicates that the HEP for this basic event is the sum of the HEPs for CHIOO02CV21 and CHIOO02CV2. As discussed in the previous question, these operator actions appear to be dependent (i.e., both require starting the charging pump). Provide the fault trees in which the CHIOO02CV212 basic event is modeled, and justify the addition of the two HEPs.

---

**Answer:**

The failure of the charging injection pump start-up in a fault tree of the top event CV is modeled by dividing into two basic events (CHIOO02CV21 and CHIOO02CV2) considering the dependency of CHIOO02CV21 between the top event MC and CV.

On the other hand, the dependencies of the human error do not need to be consider for a sequence that bypasses top event MC and for an event tree that does not have top event MC. The basic event CHIOO02CV212 which put together the CHIOO02CV21 and CHIOO02CV2, is used for the sequences and event trees.

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. : 19-257**

Page 20A.3-3 of the PRA states that the CHI fault tree is used for "ALL initiating events except LOOP." However, no fault tree for charging following a LOOP initiator could be located. Provide the fault tree that was used in this scenario, and revise the PRA accordingly.

---

**Answer:**

The charging injection fault trees CHI2-LOSP and CHI2-LO are used for LOOP event tree.

CHI2-LOSP: assigned "True" to the basic event OPSLLOP in the CHI fault tree.
CHI2-LO: deleted the power supply failure event from the CHI fault tree.

Add this explanation on CHI2-LOSP and CHI2-LO in the Attachment 20A Section 20A.3 of the PRA report next revision.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Add the description of this response in the next revision of the PRA report.

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**QUESTION NO. : 19-258**

For scenarios in which charging was successfully used to recover level immediately after a LOCA or OVDR (MC top event), does the PRA assume that the charging pumps would be stopped once level returns to normal? If the charging pumps continue to run, the operators would not need to start the pumps again if SI fails (CV top event). Discuss this scenario.

---

**Answer:**

Even though the water level recovers to normal level, the charging pumps are not stopped. Although the PRA models the failure mode "fail to start" of the charging pumps in the top events MC and CV, the cutsets of this failure mode are not involved in the MC success sequences due to the Boolean operation.

<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. : 19-259**

Discuss why the success criterion for the MC top event requires charging injection for 24 hours, given that the MC top event represents short-term charging injection to recover level and allow the standby RHR pump to start.

---

**Answer:**

The mission times of all fault trees are assumed as 24 hours conservatively even though the systems required the short mission times such as the MC top event.

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. : 19-260**

For the RSSOO02RHR2 operator action, page 9-55 of the PRA states that the operators must open five valves and start the standby RHR pump. The fault trees include failures of the five valves (9000C, 9001C, 9014C, 9015C, and 114C) to open and failure of the pump to start. However, page 20A.2-7 of the PRA states that the operators must manually initiate the ECCS actuation signal and containment spray (CS) actuation signal. In addition, page 20A.2-9 also states that the ECCS and CS signals must be manually actuated and does not include the discharge isolation valves (9014C and 9015C) and heat exchanger CCW inlet valve (114C) in the list of components that change status. Clarify whether the pump and valves are manipulated directly or must be actuated by manual ECCS and CS signals. If the actuation signals are used, clarify which components change state as a result of each signal. Revise the PRA as appropriate so that the various sections are consistent.

---

**Answer:**

The correct description for the RSSOOO02RHR2 is that the pump and valves are manipulated directly as described in Chapter 9 of the PRA report. The descriptions in page 20.A.2-7 and 20.A.2-9 of the PRA report will be revised to be consistent with the Chapter 9.

Change states of the discharge isolation valves (9014C and 9015C) have been shown in page 20.A.2-5 of the PRA report. Change state of the heat exchanger CCW inlet valve (114C) has been missing in the PRA component list. This valve will be added to the PRA component list in the next revision of the PRA report.

<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>

Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-261**

The CFACVELFSV6 basic event included in the RSS-02A fault tree is not defined anywhere in Chapter 20 of the PRA. Table 6-3 of the PRA indicates that the basic event corresponds to CSS-VLV-012, but this component could not be located in any P&ID. It appears likely that this basic event corresponds to an external leak in check valve 9012A (P&ID designator CSS-VLV-005A), of which there are no failures in the RHR fault trees. However, it is unclear why similar basic events are not included for the B and C trains of RHR. Clarify which component's failure is represented by CFACVELFSV6, discuss why similar failures are not included for all RHR trains, and revise the PRA as appropriate.

---

**Answer:**

The CFACVELFSV6 basic event represents the leak in check valve FSV6 (Its designator in P&ID is CSS-VLV-012). The detail description of this basic event is shown in Table 6-3 of the PRA report.

The fire protection water supply system which involves the FSV6 check valve is connected to the B train of RHR only. Therefore, it should be modeled only to the B train of RHR in this analysis.

In the next revision of the PRA report, replace the events in the B train of RHR from the current design. The modification of this model is no impact on the results of this PRA.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Next revision of the PRA report will be reflected this response.

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**QUESTION NO. : 19-262**

In the shutdown PRA, the RHR model only credits the standby train of RHR following a LOCA or OVDR event. However, if the two running RHR pumps are tripped before they cavitate and are vented after level is restored by charging injection (MC top event), it is possible that the two additional trains of RHR could be used. Discuss whether the operators would be expected (e.g., in the shutdown procedures) to recover the initially running trains of RHR and how much time would be available for this action. Describe what the impact on the shutdown PRA results and insights would be if this recovery were modeled.

---

**Answer:**

Before cavitations of the running RHR pumps occurs, recognizing of event occurrence and recovering operating (e.g. tripping RHR pump) will be difficult because the detection measures are not specified. So, the shutdown PRA model did not take a credit to the recovery of the running RHR pumps conservatively.

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. : 19-263**

(Follow-up to Question 19-52) Table 1-1 provided in the response to Question 19-52 indicates that fire-induced LOCAs are only applicable in POS 3, 9, and 11. The staff needs additional information on this analysis. Specifically:

a. Describe the fire scenarios that result in a LOCA during shutdown.

b. State which flow diversion paths were considered to result in a LOCA. The staff observes that the design change to lock closed valves 9815A/B/C/D (RHR MOVs 025A/B/C/D) may make fire-induced operation unlikely. However, other flow diversion paths, which were screened as discussed in response to Question 19-56, may open spuriously in a fire scenario. If any flow diversion paths were considered to cause OVDR or FLML initiating events rather than LOCAs, discuss this treatment.

c. Justify the exclusion of fire-induced LOCA from all POS other than 3, 9, and 11.

---

**Answer:**

a. The fire scenarios that result in LOCA

The fire scenarios that result in LOCA by fire are as follows.

- Spurious open of Safety Depressurization Valve(SDV)
  RCS conditions of POS 3 and 11 are high pressure and high temperature. Therefore, spurious open of SDV by fire result in LOCA.

- Flow diversion of reactor coolant to holdup tank (spurious open of CVS-LCV-121A)
  In POS other than 4-1 and 8-1, reactor coolant is drained from RHRS and is returned to RCS through volume control tank (VCT). CVS-LCV-121A of three-way valve on letdown line is connected to VCT side. The reactor coolant is flowed diversion to Holdup tank (HT) when CVS-LCV-121A is operated spuriously to HT side by fire. Therefore, the flow diversions of reactor coolant to HT result in LOCA.

RCP seal LOCA was excluded in the following reasons by internal fire at LPSD.

- In POS 3 and 11, seal injection or cooling by CCWS is necessary so that RCP is running. In the case of seal injection and cooling by CCWS have been damaged at the same time, it results in RCP seal LOCA. Loss of seal injection and cooling by CCWS at the same time by fire is possible to cause the RCP seal LOCA. However, there are no fire scenarios that damage these at the same time.

b. Flow diversion paths to cause LOCA, OVDR and FLML

The flow diversion paths were considered to cause not only LOCA but OVDR and FLML by internal fire at LPSD.

(1)LOCA

The fire scenarios that result in LOCA are described above.

(2)OVDR

The fire scenarios that result in OVDR by fire are as follows.

- Spurious open of CVS-PCV-104

  In POS 4-1 and 8-1, the drain flow is increased when CVS-PCV-104 of flow control valve on letdown line is opened spuriously by fire. In this case, RCS water level is decreased lower than mid-loop water level. That result in OVDR.

- Loss to VCT/HT change operation

  In POS 4-1 and 8-1, CVS-LCV-121A of three-way valve is connected to HT side. Even if RCS water level reaches mid-loop water level, reactor coolant is flowed diversionary to HT when it could not be changed to VCT side by fire. That result in OVDR.

(3)FLML

The fire scenarios that result in FLML are (a) Increase of drain flow and (b) Loss of charging flow.

(a) Increase of drain flow

- Spurious open of CVS-PCV-104

  In POS 4-1, 4-3, 8-2 and 8-3, the drain flow is increased when CVS-PCV-104 of flow control valve on letdown line is opened spuriously by fire. As the result, RCS water level is decreased lower than mid-loop water level. That result in FLML.

- Flow diversion of reactor coolant to Holdup tank(HT)

  In POS 4-1, 4-3, 8-2 and 8-3, reactor coolant is drained from RHRS and is returned to RCS through VCT. CVS-LCV-121A of three-way valve on letdown line is connected to VCT side in these POS. The reactor coolant is flowed diversion to Holdup tank (HT) when CVS-LCV-121A is operated spuriously to HT side by fire. In this case, the flow diversions of reactor coolant to HT result in LOCA.

(b) Loss of charging flow

- Spurious open of CVS-LCV-121B or CVS-LCV-121C

  In POS4-2, 4-3, 8-2 and 8-3, supply to charging pump is lost when either CVS-LCV-121B or CVS-LCV-121B of volume control tank outlet valve is closed spuriously by fire, and RCS injection from charging line is lost. In this case, RCS water level decreases lower than mid-loop water level. That result in FLML.

- Spurious stop of charging pump

  In POS4-2, 4-3, 8-2 and 8-3, injection to RCS is lost when charging pump is stopped spuriously by fire. In this case, RCS water level decreases lower than mid-loop water level. That result in FLML.

- Spurious close of MOV/AOV on charging line

In POS4-2, 4-3, 8-2 and 8-3, injection to RCS is lost when MOV or AOV on charging line is closed spuriously by fire. In this case, RCS water level decreases lower than mid-loop water level. That result in FLML.

c The reason why the fire-induced LOCA is excluded from all POS other than 3, 9, and 11

The fire-induced LOCA from all POS other than 3, 9, and 11 is not occurred because;

- Spurious open of Safety Depressurization Valve(SDV)

  In POSs other than 3 and 11, pressure is atmospheric pressure and temperature is 140°F in RCS condition. Therefore, even if SDV is opened spuriously by fire, reactor coolant does not flow out from RCS.

- Flow diversion of reactor coolant to Holdup tank(HT)

  As described in the item a, in POSs other than 3, 9 and 11, the reactor coolant is flowed diversion to HT when CVS-LCV-121A is operated spuriously by fire. Alternately, spurious operation of CVS-LCV-121A is evaluated as OVDR initiating event in POS 4-1 and 8-1 and FLML initiating event in POS 4-2, 4-3, 8-2, and 8-3.

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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**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO.138-1704 REVISION 1**

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

**APPLICATION SECTION:** **19.1.6.1**

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**QUESTION NO. : 19-264**

Table 23U-2 of the PRA appears to list conditional core damage probabilities (CCDP) for the first five entries in the column titled "CCDP" and CDFs (several orders of magnitude smaller) for the remainder of the table. Correct the entries in this table and in any other parts of the PRA where data from this table was used.

---

**Answer:**

The column titled "CCDP" of Table 23U-2 of the PRA is an editorial error. The revised Table 23U-2 of the PRA report (MUAP-7030(R1)) will be revised as the attached table. There is no impact in any other parts of the PRA.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Table 23U-2 of the PRA report (MUAP-7030(R1)) will be revised as described below.

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**QUESTION NO. : 19-265**

The shutdown flooding cutsets presented in Chapter 22 of the PRA include basic event RAM-LOCS-FM, which has a value of 1E-3 and is designated as the failure probability of one train of CCW by random failure. Describe how this failure probability was derived and why a single basic event, rather than the CCW fault trees, was used.

---

**ANSWER:**

The shutdown internal PRA has estimated the frequencies of initiating events using the fault trees as provided in Attachment 20B of US-APWR PRA report, MUAP-07030(R1). The shutdown flooding PRA has also estimated the one train random failure probabilities for the loss of RHR (LORH) and the loss of CCW (LOCS) using the same fault trees in the Attachment 20B.2 and Attachment 20B.3.

LORH fault trees: Attachment 20B.2 "Loss of RHR caused by other failures (LORH) initiating events frequencies quantification (for LPSD PRA)"
LOCS fault trees: Attachment 20B.3 "Loss of CCWS/ESWS (LOCS) initiating events frequencies quantification (for LPSD PRA)"

For example, the initiating events frequency of LOCS is estimated with the following process.

Initiating event frequency of LOCS due to the flooding
= Flooding frequency (frequency of pipe failure) X Conditional random failure probability of CCWS

Conditional random failure probability of CCWS is estimated using the fault trees of the Attachment 20B.3 considering the boundary conditions and mission time of the POS. The boundary conditions are assigned in the fault trees for the following trains/basic events.

- The trains/basic events of CCWS affected by flooding (the components in the flood area and the flood propagated areas are assumed to lose the function) are attributed guaranteed failures (input "true") and,
- The components of CCWS that are maintenance outage during the POS are attributed guaranteed failures (input "true").

The conditional random failure probability of CCWS for the flood/major flood in the POS 8-1 is calculated as 1.3E-03 (Basic event name: RAM－LOCS－FM) .

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. : 19-266**

It appears that the shutdown flooding analysis estimates the frequency of a flood at a CCW line, combined with random failure of the other CCW train, to obtain a flooding induced LOCS frequency. However, because no LORH cutsets are presented in PRA Table 22.8-6, it is unclear whether the same treatment (i.e., flood disabling one train, combined with random failures of other trains) was used for LORH. Describe the approach used to develop initiating event frequencies in the shutdown flooding analysis.

---

**ANSWER:**

The shutdown flooding analysis estimated the flooding induced LOCA, LOCS (loss of CCW) and LORH (loss of RHR). Also the flooding analysis estimated for the three flooding types (i.e., spray, flood and /major flood).

For spray, the affected area by flooding is limited in a local area. The possible flood induced initiating events are considered such as below.

- LOCA: spray from the CVCS extract line
- LOCS: spray from the CCW line
- LORH: spray from the RHR line

For flood/major flood, the flooding is possible to propagate to other areas and to cause various events. The flooded water will be accumulated in the lower level areas. CCW pumps are located on the lower level floor in the non-restricted area of R/B. Also RHR pumps are located on the lower level floor in the restricted area of R/B. So that the possible flood induces initiating events due to flood/major flood are assumed as below.

- LOCS: in the non-restricted areas of R/B
- LORH: in the restricted area of R/B

Also, flooding does not cause LOCS and LORH directly because it affects less than two trains of the safety systems from the train separation design. Initiating events LOCS and LORH will occur when other intact trains of CCWS or RHRS failed by random failure in conjunction with the flood. (Initiating events frequencies of LOCS and LORH by the flooding are estimated as noted in the reply to the previous question No.: RAI 19-265.)

The CDF of LOCS in POS8-1 is 1.6E-08/ry (90% of internal flood CDF at LPSD (POS8-1)) and the CDF of LORH is 1.8E-09/ry (9.8% of internal flood CDF at LPSD (POS8-1)) as provided in Table 22.8-15 of the PRA report MUAP-07030(R1). And so the cutsets of LORH are not presented in the PRA report Table 22.8-6 which provided the only top 50 cutsets. However, the cutsets of LORH (and LOCA) appear in the additional information Table 1 which provides from top 51 to 100 cutsets.

<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. : 19-267**

The response to Question 19-18 indicates that LORH and LOCS were excluded from the fire-induced initiating event analysis during shutdown because of train separation. However, it appears (see previous question) that LORH and LOCS were evaluated in the shutdown flooding analysis by combining flood-induced and random failures. Discuss why the approaches taken for fire and flood initiating events assessment were different

---

**Answer:**

For the shutdown internal flooding, the initiating event LORH and LOCS are considered by combining flood-induced and random failures. This is because that some major flood scenarios will be possible to affect the two trains of a system due to the flood propagation within the reactor building. If one of the remained two trains is out of service for maintenance in some POS, it is possible to cause the LORH or LOCS due to the random failure of remained one train.

For the shutdown internal fire, initiating event LORH or LOCS is also possible if the fire-induced and random failures of a system occurred simultaneously. However, fire scenarios are affected only one train by fire because each train is separated. The potential cause of LORH and LOCS by fire is the case that one train damage by fire and the random failures of two or three trains simultaneously. These initiating event frequencies are enough smaller than those of internal events.

As an example, the initiating event frequencies of LORH and LOCS due to fire are calculated under the following conservative conditions.

- The fire ignition frequency is assumed as 1.7E-04/ry of the reactor building (R/B) in the POS 8-1.
- It is assumed that two trains is operation, one train is standby and one train is outage.
- It is assumed the one operating train is damaged by the fire.

The conditional initiating event probability of LORH is 3.1E-05 and the initiating event frequency of LORH due to the fire is 5.3E-09/ry. This result is smaller three orders than 6.0E-06/ry of the internal event LORH frequency.

The conditional initiating event probability of LOCS is 1.8E-06 and the initiating event frequency of LOCS due to the fire is 3.1E-10/ry. This result is smaller three orders than 2.6E-06/ry of the internal LOCS frequency.

Therefore, LORH and LOCS are excluded from initiating event in the shutdown fire PRA because the initiating event frequencies of LORH and LOCS by internal fire are enough smaller than the frequencies of internal events.

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. : 19-268**

(Follow-up to Question 19-17) The response to Question 19-17 provided additional information on the FLML initiating event as a proprietary excerpt from the PRA. However, this information should be in the DCD so that all shutdown initiating events can be understood without reference to the PRA. Revise the DCD (e.g., page 19.1-104 where the text states that FLML does not apply to POS 8-1) to describe the failures that cause a FLML initiating event and how the initiating event frequency was derived.

---

**Answer:**

The next revision of the DCD will include the information that was attached as "Attachment A" in the reply to the question 19-17.

Impact on DCD

The following information will be added between the sentence "This sequence does not apply to POS 8-1" and the sentence "Loss of RHR caused by other failures (LORH)" in the Page 19.1-104 of the DCD revision 1.

(Additional information)

This category is loss of RHR operation during mid-loop operation caused by loss of coolant inventory. Two sub-categories are considered. One is over-drain (OVDR) and another is failure to maintain water level (FLML).

If the charging injection system or the letdown line system fail and the low-pressure letdown isolation valve fail to close after RCS water level has decreased to the level of the RV nozzle center, FLML is assumed to occur. Since POS 4-1 and POS 8-1 is the beginning of mid-loop operation, and RCS water level is decreasing and is not kept constant, it is assumed that this FLML event is not applicable. On the other hand, in POS 4-2, POS 4-3, POS 8-2 and POS 8-3, FLML is considered as an initiating event.

The frequency of IE (POS 4-2, POS 4-3) = (The frequency of failure of charging injection system + the frequency of failure of letdown line) × The probability of automatic isolation failure of low-pressure letdown line

a. The loss of charging injection system is evaluated by the fault tree. The failure frequency obtained from quantifying this FT is 3.6E-04/ry.

   ・Charging pump A : running

   ・Charging pump B : out of service

   ・Supply to a charging pump is expected only from VCT.

b. As failure of the letdown line, the external leakage or spurious operation of components on the line from letdown line to VCT was assumed. The failure frequency obtained from quantification under this assumption is 7.5E-06/ry.

c. Although the duration time of POS 4-2 is expected to be 12 hours, the evaluation is conservatively based upon 24 hours duration.

d. The failure probability of automatic isolation of low-pressure letdown line is evaluated by the fault tree. (Success Criteria is two out of two air-operated valves.) The failure probability obtained from quantifying this FT is 2.5E-03.

e. Therefore, the frequency of IE becomes the following.
   = (3.6E-04/ry + 7.5E-06/ry) × 2.5E-03 = 9.2E-07/ry

Evaluation of the frequency of IE (POS 8-2, POS 8-3) is shown below.

a. The loss of charging injection system is evaluated by the fault tree. The failure frequency obtained from quantifying this FT is 1.1E-04/ry.

   ・Charging pump A : running

   ・Charging pump B : standby

   ・Supply to a charging pump is expected only from VCT.

b. As failure of the letdown line, the external leakage or spurious operation of components on the line from letdown line to VCT is assumed. The failure frequency obtained from quantification under this assumption is 7.5E-06/ry.

c. Although the duration time of POS 8-2 is expected to be 12 hours, the evaluation is conservatively based upon 24 hour duration.

d. The failure probability of automatic isolation of low-pressure letdown line is evaluated by the fault tree. (Success Criteria is two out of two air-operated valves.) The failure probability obtained from quantifying this FT is 2.5E-03.

e. Therefore, the frequency of IE becomes the following.
   = (1.1E-04/ry + 7.5E-06/ry) × 2.5E-03 = 2.9E-07/ry

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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**QUESTION NO. : 19-269**

Table 20A.4-5 of the PRA indicates that the PRS fault tree is used to evaluate CCW and ESW pump restart following a LOOP, but this fault tree could not be located in the PRA. Provide the fault tree used in this scenario, and revise the PRA accordingly.

---

**Answer:**

Attached the PRS fault trees. The next version of the PRA report involves the PRS fault trees in Chapter 20A.4 of Attachment 20A.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Add the additional fault trees in the next revision of the PRA report.

19-269-6

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**QUESTION NO. : 19-270**

(Follow-up to Question 19-137) The response to Question 19-137 indicates that a higher value of the "correction factor" was used for sequences that include combinations of mitigating functions that require the same operator actions. This approach appears to be applied inconsistently. In the PRA, it is only clear that the higher value was used for the SDLORH-0005 and SDLOOP-0027 sequences in POS 4-2 and 8-2. For these sequences, three operator tasks are required (two from POS 8-1 plus GI) and the correction factor is increased from 0.2 to 0.5. However, the following sequences also include combinations of CV and GI or GI and SC (both of which require the RWR pumps) but do not appear to have increased correction factors: SDLOCA-0006, SDLOCA-0011, SDLOCA-0015, SDLOCS-0003, SDLOOP-0006, SDLOOP-0009, SDLOOP-0015, SDLOOP-0018, and SDLOOP-0024. Discuss this discrepancy, and revise the PRA as needed.

---

**Answer:**

The response to Question 19-137 intends to use "one rank higher correction factor", but not to use "High dependency" for the sequences that include combinations of mitigating functions that require the same operator actions. In particular, "Moderately dependency" is used as a correction factor when "Low dependency" is assessed, "High dependency" is used when "Moderately dependency" is assessed. Note that, the dependency level does not increase when "High dependency" is assessed, because it is too conservative to assess "Complete dependence (correction factor; 1.0)" between the operator actions. Also, the dependency level can not increase when "Complete dependency" is assessed.

The sequences which have increased correction factors from "Low dependency (0.1)" to "Moderately dependency (0.2)" are as follows;

- SDLOCS-0003 (POS 4-2, 8-2)
- SDLOOP-0009 (POS 4-2, 8-2), PR is done by automatic action.
- SDLOOP-0018 (POS 4-2, 8-2), GT and PR are done by automatic action.

The sequences which have increased correction factors from "Moderately dependency (0.2)" to "High dependency (0.5)" are as follows;

- SDLORH-0005 (POS 4-2, 8-2)
- SDLOOP-0027 (POS 4-2, 8-2)

The sequences which use “High dependency” and have not increased correction factors are as follows;

- SDLOCA-0006 (POS 4-2, 8-2)
- SDLOCA-0011 (POS 4-2, 8-2)
- SDLOCA-0015 (POS 4-2, 8-2)
- SDLOOP-0006 (POS 4-2, 8-2)
- SDLOOP-0015 (POS 4-2, 8-2), GT is done by automatic action.

The sequences which use “Complete dependency (1.0)” and have not increased correction factors are as follows;

- SDLOOP-0024 (POS 4-2, 8-2), GT is done by automatic action.

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.