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TOKYO, JAPAN

January 9, 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-09009

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

References: 1) "Request for Additional Information No.88-1438 Revision 1, SRP Section: 19.1.6 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: PRA," dated October 29, 2008.
2) Letter MHI Ref: UAP-HF-08271 from Y. Ogata (MHI) to U.S. NRC, "MHI's Responses to US-APWR DCD RAI No.88-1438," dated November 27 , 2008.

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

Enclosed are the second responses to the RAIs contained within Reference 1. In the initial responses submitted with Reference 2, MHI committed to submit responses to RAI 19-138, 19-139, 19-140, 19-141, 19-142, 19-144, 19-145, 19-147 and 19-149 within 75 days after RAI issue date.

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

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

D081
NRO

Contact Information

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Enclosure 1

UAP-HF-09009
Docket No. 52-021

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

January 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/9/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-138

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

ANSWER:

Mitigation systems that are not credited in POS 8-1 but credited in other POSs are heat removal via steam generators (SGs) and gravity injection. Failure of SSC related to these two mitigation systems are not model as fault trees in the LPSD PRA.

Heat removal from SGs is a function credited in at-power PRA and is an important function required to mitigate most of the initiating events that occur during at-power. SSCs that highly impact the reliability of SG heat removal functions are risk significant SSCs for at-power operation. Since heat removal from SGs is not available during all POSs during shutdown, those SSCs are less risk significant during LPSD than in at-power. So detail examination of important SSCs are not discussed in the DCD.

Gravity injection is a mitigation system specific to LPSD. SSCs used for gravity injection are, spent fuel pit (SFP), drain line from the SFP to the RHR piping, and the isolation valves in the drain line (SFS-VLV-021A, SFS-VLV-021D, RHS-VLV-034A and RHS-VLV-034D in the P&ID). Components used to supply water from the recirculation refueling water storage pit (RWSP) to SFP, which are the refueling water recirculation pumps and associate valves and piping, are also necessary to provide continuous injection.

Since gravity injection function is only credited as a mitigation function for POS 4-2 and POS 8-2, but not for all POSs during shutdown, we consider that the static failure of the SSCs related to gravity injection do not have high impact on total shutdown risk. Accordingly these SSCs were excluded from important SSCs.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

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

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

APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-139

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

ANSWER:

As described in response to question 19-46, the RCS cannot be maintained below 212°F (100°C) by heat removal from one RHR, when the time after plant trip is less than 36 hours. In such cases the accident progression varies significantly depending on the existence of RCS opening. The following 2 cases can be considered.

Case 1: RCS is closed

In the case where the RCS is closed and only one RHR train is operating, the RCS pressure will increase until the RCS temperature reaches approximately 243°F (117°C), postulating decay heat level at the beginning of mid-loop operation (10 hours after plant trip). Then decay heat can be fully removed from the RCS by the one RHR train. This RCS condition is within the design specification of the RHR, and therefore, the RHR can function without degradation.

Case 2: RCS is opened

In the case where the RCS if opened, vapor generated in the reactor vessel will be released from the RCS opening, resulting decrease in RCS inventory. If the RCS opening cannot be closed, the RCS water level will decrease and the RHR will eventually be inoperable. This scenario can occur at an early stage of POS 4-1 where the decay heat is high.

When only one RHR train becomes operating at an early stage of POS 4-1, and the RCS is maintained opened, the event progression was evaluated to be as table 19-139-1. The results shown in this table is based on static evaluation conservatively assuming that the RCS opening is large enough to maintain the RCS pressure at atmospheric pressure and reflux cooling by SGs ineffective. The decay heat assumed in this calculation is based on the decay heat at 12 hours after plant trip, which is the time when POS 4-1 is entered.

Table 19-139-1 Event progressions following partial loss of RHR during POS 4-1

Event		Description of event	Relative time [hr]
1	Partial loss of RHR	Partial loss of RHR occurs and decay heat is removed by only one RHR train. RCS level is 120 mm (4.7 inches) below top of main coolant piping, which is the water level during mid-loop operation.	0
2	Loss of RHR due to low RCS water level	RCS level decreases to the level of main coolant piping center. The operating RHR is assumed to fail.	6.3
3	Core uncovered	RCS level decreases to the level of core top.	10.0

RCS temperature can be maintained below 212°F (100°C) by one operating RHR train if partial loss of RHR occurs beyond 36 hours after plant trip. In such cases bulk boiling will not occur, and the RCS level will not significantly decrease. The RHR can continuously operate and remove decay heat.

These discussions indicate that in the initial stage of POS-1, when decay heat generation is high, decay cannot be continuously removed by single RHR train if the RCS opening (pressurizer spray line vent) is kept open. When the success criteria for RHR function is two trains to be operating, the initiating event frequency of losses of RHR function will increase approximately five times from the case where single train is the success criterion. The core damage frequency of losses of RHR event (LORH) during POS 4-1 is 3.3E-10 /yr in the DCD PRA. If the success criterion of RHR train is two trains throughout

POS 4-1, which is a conservative assumption, the resulting CDF for LORH during POS 4-1 is approximately 1.7E-9 /yr. This increase of the CDF is less than 1% of the total LPSD CDF. Accordingly, setting the success criteria of RHR train as one train in POS 4-1 is considered to have negligible impact on the PRA and its insights.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

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

APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-140

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

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

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

ANSWER:

It is standard practice in the United States for plants in shutdown to evaluate the risk of configurations being entered as a result of the implementation of the maintenance rule process and application of standard configuration control practices. These practices assure that removing a number of related systems from service at the same time is carefully considered and virtually never done when the conditional risk impacts are high.

The RAI question states a request to consider performing a risk calculation removing any credit for, "systems (not) required to be operable according to technical specifications (TS), since voluntary measures that are not required by current regulations could be withdrawn by licensees without NRC approval". It also states that, "The response to Question 19-47 clarified that the sensitivity study credited automatic isolation of the low pressure letdown line on low level, the charging pumps, the refueling water storage pit (RWSP), and the refueling water storage auxiliary tank (RWSAT) despite the fact that there are no TS requiring these components to be available during MODES 5 and 6."

It is not accepted PRA practice to perform a PRA while only taking credit for certain SSCs. PRA practice is to model all the SSCs in the plant that could perform a safety function and then to realistically consider the probability that each SSC so-modeled would be unavailable or fail to perform that function.

The acceptance limits for judging CDF and LERF results for design certification PRAs are based on annual average risk metrics and not for time varying risk metrics that are dependent on the specific time dependent configurations of the plant. Annual average CDF estimates must consider the probability of the plant configurations that can impact the conditional risk. Before any meaningful configuration dependent sensitivity study can be performed it is important to have agreement on the decision criteria to be used to evaluate the results. MHI is not aware that there exist any decision criteria for time dependent or configuration dependent CDF values. These criteria must address the probability of the conditions being imposed on the calculation.

The "risk" calculated in response to the question would be artificially high and incorrect if calculated as a sensitivity since this configuration cannot occur and has not occurred. In fact the calculation of CDF at LPSD takes into consideration the event or configuration duration. Since this configuration would have zero duration it would have zero risk.

The following key assumption will be added in Table 19.1-115.
"Maintenance rule process and application of configuration risk management program are implemented to evaluate the risk of configurations being entered during shutdown. These practices assure that removing a number of related systems from service at the same time is carefully considered and virtually never done when the conditional risk impacts are high. "

Impact on DCD

Table 19.1-115 (Sheet 2 of 4) of chapter 19,will be revised as follows:

Key assumptions
Operator actions during LPSD events 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. 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. c. In case LOCA occurs in RHR line, operator will perform isolation of the RHR hot legs suction isolation valves. 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. **Operator actions during severe accidents** 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. 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. **Maintenance activities** a. Maintenance rule process and application of configuration risk management program are implemented to evaluate the risk of configurations being entered during shutdown. These practices assure that removing a number of related systems from service at the same time is carefully considered and virtually never done when the conditional risk impacts are high.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/9/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-141

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

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

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

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

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

ANSWER:

a.

The last two sentences of the first paragraph that describes "Loss of coolant accident (LOCA)" in page 19.1-102 of DCD revision 1 will be revised as follows:

In this evaluation, inadvertent transfer to the RWSP from the RHR is assumed. This diversion can happen if ~~a~~ the containment spray/residual heat removal pump full-flow test line stop valves (RHS-MOV-025A/B/C/D) ~~motor-driven valve~~ is opened. This event is defined as a loss of all RHR trains.

b.
The CS/RHR pump full-flow test line stop valves are manually opened by the operator from the main control room (MCR) and align the CS/RHR pumps to remove heat from the containment for an extended period of time by continuous circulation of water from the refueling water storage pit (RWSP) once the containment spray is no longer required. Since these valves are locked close by having electrical power removed, the valves must be unlocked from the control center by the operator, prior to opening the valve from the MCR.

The PRA does not model the change over from containment spray mode to RWSP circulation mode since there is considerable time for recovery actions that can be taken and the failure of this action has negligible impact on risk.

This design feature has been addressed in section 5.4.7 of DCD chapter 5.

c.
Tier 1, Figure 2.4.5-1 is not intended to describe detail design information such as valve position and locked conditions. Accordingly the figure will be unchanged. The locked closed condition of the containment spray/residual heat removal pump full-flow test line stop valves (RHS-MOV-025A/B/C/D) will be stated as key assumptions in Table 19.1-115 of chapter 19.

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

Impact on DCD

Page 19.1-102 of the DCD rev.1 will be revised to clarify which valve has the potential to cause inadvertent transfer to the RWSP from the RHR. The last two sentences of the first paragraph that describes "Loss of coolant accident (LOCA)" will be revised as follows:

In this evaluation, inadvertent transfer to the RWSP from the RHR is assumed. This diversion can happen if ~~a~~ the containment spray/residual heat removal pump full-flow test line stop valves (RHS-MOV-025A/B/C/D) ~~motor-driven valve~~ is opened. This event is defined as a loss of all RHR trains.

The following statement will be added in Table 19.1-115 of chapter 19, as a key assumption of design features:

The containment spray/residual heat removal pump full-flow test line stop valves (RHS-MOV-025A/B/C/D) are locked closed.

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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US-APWR Design Certification

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APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-142

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

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

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

ANSWER:

MHI will revise the DCD to be easy understood as shown below;

5.4.7.2.3.6 Mid-loop and Drain Down Operations

[the last paragraph in Subsection 5.4.7.2.3.6]

The level in the primary system is lowered below the upper end ~~to near the mid-line~~ of the hot and cold legs. The RCS water level should be maintained higher than 0.33 feet above the loop center ~~mid-loop~~ and the RHR flow of 1,550 to 2,650 gpm should be supplied. During mid-loop operation~~At this water level~~, the air/water interface is at close proximity to the RHR suction nozzles located on the hot legs. ~~but~~

~~the h~~Higher RCS water level during mid-loop operation applied for the US-APWR design than that for conventional plants reduces the possibility of air entrainment into the RHR pump suction.

Impact on DCD

The DCD will be revised to rev.2 reflecting this response to this RAI.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-144

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

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

ANSWER:

For POS 4-3 and POS 8-1, where the SG nozzle dams are installed and the reactor vessel upper plenum is closed, pressurizer manhole will be kept opened. The pressurizer manhole will provide a sufficient vent path large enough to prevent pressurization of the upper plenum under loss of RHR events. Opening of pressurizer manhole during POS 4-3 and POS 8-1will be stated as key assumptions in the DCD.

Administrative and operating procedures will be in place to assure that the pressurizer manhole will be kept opened during the POSs discussed above. Administrative and operating procedures are COL issues described in COL 13.5(1) and COL 13.5(7) in the DCD Chapter 13 subsection 13.5.

Impact on DCD

Table 19.1-115 (Sheet 2 of 4) of chapter 19,will be revised as follows:

Key assumptions
Operator actions during LPSD events 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. 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. c. In case LOCA occurs in RHR line, operator will perform isolation of the RHR hot legs suction isolation valves. 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. **Operator actions during severe accidents** 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. 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. Maintenance activities a. To ensure that prevent pressurization of the upper plenum under loss of RHR events, pressurizer manhole will be opened when the reactor vessel upper plenum is closed and all hot legs are blocked simultaneously by nozzle dams.

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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APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-145

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

ANSWER:

The SFP is expected to contain water volume above the level of the gravity drain suction nozzle to perform gravity injection for a certain amount of time without water supply to the SFP. However, the PRA assumes that water supply from the refueling water storage pit (RWSP) to the SFP is essential for gravity injection function. The refueling water recirculation pump utilized to supply water to the SFP can provide flow rate exceeding that required for gravity injection through the gravity drain line. Therefore, exhaustion will not occur.

This operator action is identified as risk important operator action and is inputted to the human factors engineering program. Components used for the water supply to the SFP is identified as risk important structure system and components (SSCs) and is inputted to the RAP.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

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

APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-147

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

a. Provide the elevation of the top of the spent fuel in the SFP.
b. Confirm that 23 feet of water, as required by TS 4.3.2, can be maintained above the spent fuel even in the scenario of gravity injection to the RCS.
c. Revise Figure 9.1.3-1, which depicts the drain lines near the top of the spent fuel, as needed.
d. Revise the description of and ITAAC for the SFP in Tier 1, section 2.7.6.2, as needed, to ensure that the SFP is constructed and inspected with a gravity injection nozzle at an elevation that will not allow the spent fuel to be endangered during draining.

ANSWER:

a. The elevation of the top of the spent fuel in the SFP is at 47 feet.

b. The elevation of the suction nozzle in the latest design is at 71'-3". The bottom of suction nozzle is EL 70'-7". Therefore, even if gravity injection or some piping failure occurs, the height from spent fuel top to water level can be maintained above 23ft.

c. The P&ID differs from a detailed drawing and is intended mainly to show system configuration excluding dimensional specifications. Therefore, MHI think there is no need to revise the P&ID.

d. MHI will revise the description for the SFP in Tier 1, Section 2.7.6.2.

Impact on DCD

In Subsection 2.7.6.2.1 of Tier 1, key design feature will be revised as follows:

To preclude unanticipated drainage, the spent fuel pit is not connected to the equipment drain system and the nozzles or piping connected to the SFP are installed at an elevation that will not allow the spent fuel to be endangered during draining.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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APPLICATION SECTION: **19.1.6**

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

QUESTION NO. : 19-149

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

ANSWER:

The outline of the US-APWR shutdown sequences are described as followings. Schematic diagrams of refueling outage are shown in Figure 19-149-1.

1. RCS Cooldown Phase (MODE 5)
In this phase, residual heat is continuously removed by RHR system. CS/RHR pumps and CS/RHR heat exchangers are operating. Also, Chemical and Volume Control System (CVCS) are continuously operating for RCS clean up.
This phase is categorized as part of POS 3.

2. RCS Depressurization Phase (MODE 5)
After the pressurizer is fulfilled, RCS pressure is controlled by CVCS. In this phase, pressure control valve (CVS-PCV-104) installed in the letdown line is used to depressurize the RCS.
This phase is categorized as part of POS 3.

3. RCS Draining Phase (MODE 5)
In plant cooldown phase with RHRS operation, RCS inventory is controlled by balance of letdown flow rate and charging flow rate. These flow rates are controlled by control valves in CVCS. RCS drain is performed by changing this balance.
This phase is categorized as part of POS 3.

4. Mid-loop Operation Phase (shutdown) (MODE 5)
In this phase, RCS water level is lowered below the top of the main coolant pipe. When the RCS water level has been decreased to a level where the gas phase approaches the SG plenum, SG tube water will be drained. To promote rapid SG tube drain, air is injected to SG from the station service air system (SSAS). Water level during mid-loop operation is controlled by balance of letdown flow rate and charging flow rate in CVCS.

After SG tube drain has completed, SG nozzle dams can be installed. Inspection of SG tubes can start after these operations.

Following the installation of SG nozzle dams, RCS water level is increased above the main coolant pipe to reduce the possibility of ingestion of air to CS/RHR pumps.

Then, hydrogen peroxide is added into the RCS water for RCS water purification.

This phase is categorized as POS 4.

5. Refueling Cavity Filling Phase (MODE 6)
Concurrently with removal operation of Reactor Vessel Head, the water in the refueling water storage pit (RWSP) is transferred to refueling cavity by CS/RHR pumps.
This phase is categorized as POS 5 for the period the fuel is in the reactor vessel (RV), and categorized as POS 6 for the period the fuel has been relocated from the RV.

6. Refueling Cavity Drain Phase (MODE 6)
After core loading, the water in the refueling cavity is transferred to RWSP by CS/RHR pumps. During this operation, Reactor Vessel Head is moved to RV. The water level for mid-loop is controlled by balance of letdown flow rate and charging flow rate in CVCS.
This phase is categorized as POS 7.

7. Mid-loop operation (Startup) (MODE 5)
After Reactor Vessel Head restoration, gas phase in RCS is vacuumed by Vacuum Venting system for reducing dynamic venting period. In this phase, the water level for mid-loop is also controlled by balance of letdown flow rate and charging flow rate in CVCS.
This phase is categorized as POS 8.

8. Static Venting Phase (MODE 5)
RCS is filled by increasing the charging flow rate by CVCS.
This phase is categorized as part of POS 9.

9. Dynamic Venting Phase (MODE 5)
After static venting, gas in the SG tubes is dynamically vented by RCP jog operation. The gas is collected to the RV head, and vented through RV head vent line. During this operation, the water level is controlled by balance of letdown flow rate and charging flow rate.
This phase is categorized as part of POS 9.

10. O_2 Scavenging Phase (MODE 5)
For further deoxidation, hydrazine is added into RCS water from CVCS. In this phase, RCS is being fulfilled.
This phase is categorized as part of POS 9.

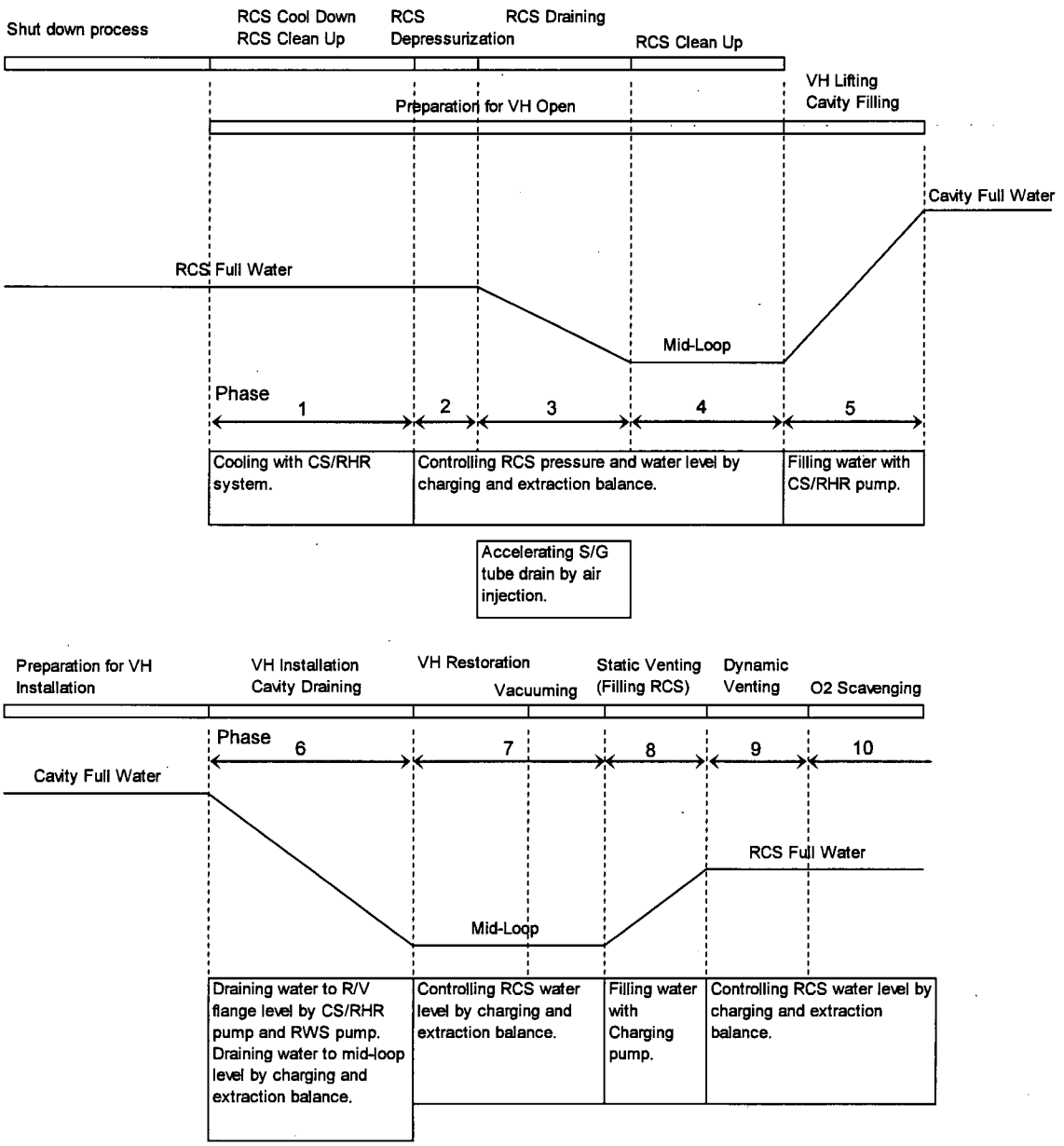

Figure 19-149-1 Schematic diagram of refueling outage

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA