MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

July 28, 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-09408

Subject: MHI's Responses to US-APWR DCD RAI No.391-2974 Revision 1

Reference: 1) "Request for Additional Information No. 391-2974 Revision 1, SRP Section: 06.03 – Emergency Core Cooling System" dated June 15, 2009

With this letter, Mitsubishi Heavy Industries, LTD. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "Responses to Request for Additional Information No.391-2974 Revision 1" a document package that responds to the NRC's Requests for Additional Information dated June 15, 2009.

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

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

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

Sincerely,

Y. Ogata

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

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NRO

Enclosures:
1 - Affidavit of Yoshiki Ogata

2 - Responses to Request for Additional Information No.391-2974 Revision 1 (proprietary)

3 - Responses to Request for Additional Information No.391-2974 Revision 1 (non-proprietary)

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

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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-09408

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 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 "MHI's Response to US-APWR DCD RAI No. 391-2974 Revision 1" dated July 2009, and have determined that the document contains proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The basis for holding the referenced information confidential is that it describes the unique design of the safety analysis, developed by MHI (the "MHI Information").

4. The MHI Information is not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and detailed design for its software and hardware extending over several years. Therefore public disclosure of the materials would adversely affect MHI's competitive position.

5. The referenced information has in the past been, and will continue to be, held in confidence by MHI and is always subject to suitable measures to protect it from unauthorized use or disclosure.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information.

7. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's application for certification of its US-APWR Standard Plant Design.

8. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design and testing of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

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 28th day of July, 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09408
Docket No. 52-021

Responses to Request for Additional Information No.391-2974
Revision 1

July 2009
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 391-2974**
SRP SECTION:	**06.03 – EMERGENCY CORE COOLING SYSTEM**
APPLICATION SECTION:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-1

RAI 6.3.1.2-1

In Section 6.3.1.2, identify those components of the ECCS required for safe shutdown in the event the normal systems are unavailable, including the number of pumps required and the flow control valves. Summarize the functions of these components during the shutdown without normal systems.

ANSWER:

The description of the function, and the number of components required for the safe shutdown will be added in the Subsection 6.3.1.2.

Impact on DCD

Subsection 6.3.1.2 will be revised as follows:

The portions of the ECCS also operate in conjunction with the other systems of the cold shutdown design. The primary function of the ECCS during a safety grade cold shutdown is to ensure a means for feed and bleed for boration, and make up water for compensation of shrinkage. **For boration and make up for compensation for shrinkage, operation of two trains of high-head injection system, each of which includes one safety injection pump and one flow control valve, are required. For letdown of reactor coolant, operation of one train of emergency letdown system including one flow control valve and one stop valve is required.** Details of the safe shutdown design bases are discussed in Chapter 5, Subsection 5.4.7.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-2

RAI 6.3.1.2-2

Is the safe shutdown mission of the ECCS components to enable the operator to maintain the plant in hot standby for approximately 14 hours, until the normal systems are restored? Or to bring the plant to either hot shutdown or cold shutdown conditions?

ANSWER:

The safe shutdown mission is to bring the plant to cold shutdown condition using the safety-related systems only. The function of ECCS in safe shutdown operation is to borate the reactor coolant by "feed and bleed," and make up for compensation of shrinkage during safe shutdown operation.

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.

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DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-3

RAI 6.3.1.3-1

Revise the text to indicate that the function of the pH control is to enhance the *iodine* retention capacity in the containment *recirculation water*. On Page 6.3-9, it is stated that the dilution time of the NaTB is approximately 12 hours. Provide the diluted NaTB mass versus time and the RWST pH value versus time for a postulated large break LOCA. Demonstrate that the RWST water can reach pH 7.0 within acceptable time to suppress the iodine in the containment air space.

ANSWER:

DCD Ch. 6, Subsection 6.3.1.3 "Containment pH Control," the 2nd sentence is revised as NRC pointed out.

The information which NRC required is described in detail in the response to the QUESTION NO.06.05.02-1 in Reference 1. Please refer to Reference 1.

Reference

1. Letter MHI Ref: UAP-HF-09199 from Y. Ogata (MHI) to U.S. NRC, "MHI's Second Responses to US-APWR DCD RAI No. 234" dated April 22, 2009

Impact on DCD

Subsection 6.3.1.3 will be revised as follows:

NaTB baskets are located in the containment and are capable of maintaining the desired post-accident pH conditions in the recirculation water. The pH adjustment is capable of maintaining containment water pH at least 7.0, to enhance the **iodine** retention capacity in the containment **recirculation water** and to avoid stress corrosion cracking of the austenitic stainless steel components.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-4

RAI 6.3.2.2.5-1

In Section 6.3.2.2, it is stated that "The size of the NaTB transfer pipes and refueling cavity drain pipes are selected to minimize the head loss during a transfer of solution. The containerized NaTB solution overflows at the same flow rate as the spray water that flows into the container. Therefore, the NaTB dissolved in the container flows into the RWSP without losses from spilling over onto the containment operating floor. The dissolution time of the NaTB is approximately 12 hours."

The flows from the sprays that fall onto the NaTB containers appear to be very difficult to predict accurately. Where is this calculation performed that predicts the dilution via the sprays and has it be verified? During the aging of the plant, many characteristics of the sprays may change slightly because of maintenance activities and other effects that may change the spray distribution onto the NaTB containers to change. Has the change spray distribution after time been considered in your evaluation. How do you know that you do not lose NaTB into the containment environment? How does the amount and time of NaTB dilution impact the pH of water in the containment?

ANSWER:

DCD Section 6.3, Figure 6.3-10 shows the containment spray pattern on the floor where the NaTB baskets are installed. One spray pattern indicated in circle, regardless of its size, means that inside the circle is covered by the designed spray flow from one spray nozzle. The maximum spray flow rate that flows into the NaTB basket container was calculated by using conservatively larger number of these spray pattern circles which cover the container. Even if this conservatively estimated spray water flows into the container, the pressure loss in the transfer piping is lower than the difference of elevation between the container and RWSP, that is, the driving force to gravity injection. Therefore, the NaTB solution in the container does not overflow from the container to the outside of RWSP.

The change of spray pattern and spray distribution after time is not considered for the following reasons:

- The erosion wastage of passive components such as spray nozzles and spray piping does not occur since they are not used in the plant normal operation. In addition, the corrosion wastage does not occur since the austenitic stainless steel is used for their materials, and they are exposed in the RWSP water, which is controlled water chemistry.

- The aging degradation of CS/RHR pump head is previously considered in designing spray flow rate, that is, in designing the resistance of flow control orifice installed on the spray header.

The amount and solution rate of NaTB is used as the input conditions in the pH analysis of the recirculated water shown in the response to the QUESTION NO.06.05.02-1 of Reference 1. Please refer to the Reference 1 for details.

Reference

1. Letter MHI Ref: UAP-HF-09199 from Y. Ogata (MHI) to U.S. NRC, "MHI's Second Responses to US-APWR DCD RAI No. 234" dated April 22, 2009

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.

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APPLICATION SECTION:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-5

RAI 6.3.1.4-1

Revise DCD Table 6.3-1, "Response of US-APWR to TMI Action Plan," to show TMI Action Item II.K.3.15 is not applicable to the US-APWR design, by entering *N/A* under the US-APWR Design column.

ANSWER:

"N/A" will be added to the column US-APWR design of DCD Table 6.3-1, Response to US-APWR to TMI Action Plan, item II.K.3.15.

Impact on DCD

Table 6.3-1, item II.K.3.15 will be revised as follows:

No.	Regulatory Position	US-APWR Design
II.K.3.15	Modify break detection logic to prevent spurious isolation of high pressure core injection and reactor core isolation cooling systems (Applicable to BWR's only)	N/A

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:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-6

RAI 6.3.1.4-2

Revise DCD Table 6.3-1 TMI Action Item II.K.3.45 to include the complete description of the concern, by adding the following text after *that would reduce* "the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs". Therefore, it is not applicable to US APWR.

ANSWER:

The texts, "the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs" will be added to the column Regulatory Position in DCD Table 6.3-1, Response to US-APWR to TMI Action Plan, item II.K.3.45.

Impact on DCD

Table 6.3-1, item II.K.3-15 will be revised as follows:

No.	Regulatory Position	US-APWR Design
II.K.3.45	Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce **the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs**	N/A

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-7

RAI 6.3.1.4-3

Update DCD Table 6.3-1 to address SRP 6.3, "Emergency Core Cooling System, "Acceptance Criteria, which requests applicants address the following TMI Action Items(these items are not in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," C.I.6.3, Emergency Core Cooling System):

1. II.K.3.16 of NUREG-0737, with regard to providing an evaluation of methods to reduce challenges and failures of reactor coolant system relief valves for BWRs.
2. II.K.3.24 of NUREG-0737, with respect to the adequacy of space cooling for longterm operation of HPCI and RCIC systems for BWRs to maintain the operating environment within allowable limits.
3. II.D.3 of NUREG-0737, with respect to the requirements that reactor coolant system relief and safety valves be provided with a positive indication in the control room of flow in the discharge pipe
4. II.F.2 of NUREG-0737, with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling.

ANSWER:

The items II.K.3.16, II.K.3.24, II.D.3, and II.F.2 will be added to DCD Table 6.3-1, Response to US-APWR to TMI Action Plan.

Impact on DCD

The items II.K.3.16, II.K.3.24, II.D.3 and II.F.2 will be added in Table 6.3-1 as follows:

No.	Regulatory Position	US-APWR Design
II.K.3.16	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only).	N/A
II.K.3.24	Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (Applicable to BWR's only).	N/A
II.D.3	Provide direct indication of reactor coolant system relief and safety valve position (open or closed) in the control room.	The direct indication of reactor coolant system relief and safety valve position (open or closed) is provided in the MCR.

No.	Regulatory Position	US-APWR Design
II.F.2	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's.	Following instrumentations are provided for indication of inadequate core cooling (ICC). ・Degrees of subcooling ・Reactor vessel water level (RVWL) ・Core exit temperature The degrees of subcooling indicates the loss of subcooling, occurrence of saturation and achievement of a subcooled condition following core recovery. The RVWL provides information to the operator on the decreasing liquid inventory in the reactor. The core exit temperature sensors monitor the increasing core exit temperatures associated with ICC and the decreasing core exit temperatures associated with recovery from ICC. These instrumentations are also provided as PAM variables with an unambiguous, easy-to-interpret indication. Refer to DCD Subsection 7.5.1.1.3 for more detail.

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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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

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APPLICATION SECTION:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-8

RAI 6.3.1.4-4

Table 6.3-2, "Response of US-APWR to Unresolved Safety Issues," USI A-1, states "*the probability of water hammer in ECCS is discussed in Subsection 6.3.2.1.1,*" under the column US-APWR Design. Revise this statement to be consistent with the text DCD Section 6.3.2.1.1, "High Head Injection System," which describes design features to preclude void formation and water hammer. The text under GL 86-07 should also be similarly revised. In addition, discuss the frequency of the periodic in-service full-flow testing. Clarify whether this is a technical specification requirement.

ANSWER:

The columns US-APWR Design in Table 6.3-2, item A-1 and Table 6.3-4, item GL 86-07 will be revised to be consistent with the text in DCD Section 6.3.2.1.1.

Periodic in-service full-flow testing is conducted once every three months. Technical specification requires that the frequency of the safety injection pump testing is specified in accordance with "Inservice Testing Program." In the Testing Program Plan, the frequency of the periodic safety injection pump testing is once every three months in accordance with the ASME OM Code test requirements. Please refer to DCD Chapter 3, Section 3.9, Table 3.9-13 Pump IST (Sheet 1 of 7).

Impact on DCD

Table 6.3-2, item A-1 is revised will be revised as follows:

No.	Regulatory Position	US-APWR Design
A-1	**WATER HAMMER** A number of water hammers have been experienced in several systems (e.g., SG feed water ring/piping, ECCS, RHRS, Containment Spray System, Sea Water System, Main Feed Water System, Main Steam System) but most of them were relatively small damage of piping support. Although they did not result in radioactive release to outside of plant, establishing a systematic review procedure is necessary addressing continuous occurrence of the event and potential to plant safety.	~~The probability of water hammer in ECCS is discussed in Subsection 6.3.2.1.1.~~ **The safety injection piping is normally filled with water by filling and venting prior to operation, and drain-down to RWSP is prevented by four check valves provided in series between the safety injection pump and the direct vessel injection (DVI) nozzle on the reactor vessel. In addition, water column separation could not be formed by the difference of elevations between the RWSP water level during normal operation and the highest point in the safety injection piping. Therefore, the void causing water hammer could not be formed in the safety injection piping.** **In addition, ECCS has the pump full-flow testing line branched off the safety injection line at the highest point in the containment and is led to RWSP. If the void would be formed in the system due to insufficient venting, the void in piping could be dynamically vented to RWSP through the periodic safety injection pump testing using this full- flow line, and the system is maintained in the water solid condition.**

Table 6.3-4, item GL 86-07 will be revised as follows:

No.	Regulatory Position	US-APWR Design
GL 86-07	**TRANSMITTAL OF NUREG-1190 REGARDING THE SAN ONOFRE UNIT 1 LOSS OF POWER AND WATER HAMMER EVENT** On November 21, 1985, San Onofre Unit 1 Nuclear Power Plant experienced a loss of ac electrical power and failure of multiple check valves followed by a severe water hammer in the secondary system which caused a steam leak and damaged plant equipment (e.g., main feedwater pump trip, main feedwater pump suction pipe break). The NRC investigated and documented the factual information and their findings and conclusions associated with the event in NUREG-1190, "Loss of Power and Water Hammer Event at San Onofre Unit 1, on, November 21, 1985." The NRC requested all reactor licensees and applicants to review the information in NUREG-1190. The NRC requested the utility to reply relating to the validity of check valves and report the status of implementation of provision for USI A-1, "Water Hammer."	~~The probability of water hammer in ECCS is discussed in Subsection 6.3.2.1.1.~~ **The safety injection piping is normally filled with water by filling and venting prior to operation, and drain-down to RWSP is prevented by four check valves provided in series between the safety injection pump and the direct vessel injection (DVI) nozzle on the reactor vessel. In addition, water column separation could not be formed by the difference of elevations between the RWSP water level during normal operation and the highest point in the safety injection piping. Therefore, the void causing water hammer could not be formed in the safety injection piping.** **In addition, ECCS has the pump full-flow testing line branched off the safety injection line at the highest point in the containment and is led to RWSP. If the void would be formed in the system due to insufficient venting, the void in piping could be dynamically vented to RWSP through the periodic safety injection pump testing using this full- flow line, and the system is maintained in the water solid condition.**

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

7/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

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QUESTION NO.: 06.03-9

RAI 6.3.1.4-5

Table 6.3-2, USI A-2, "Asymmetric Blowdown Loads On Reactor Primary Coolant Systems," states consideration the hypothetical break is not necessary because the USAPWR uses the leak before break (LBB) concept. SRP 3.6.3, "Leak-Before-Break Evaluation Procedures," Acceptance Criteria 3, states "*LBB cannot be applied to individual welded joints or other discrete locations.*" A break at a reactor coolant system (RCS) nozzle inside the reactor cavity area around the reactor pressure vessel (RPV) would result in asymmetric loads on the RPV internals (large differential pressure across the core barrel) that should be considered in the design of the RPV supports. Failure of the supports could affect core coolablity. Expand the explanation to justify why not taking into consideration this loading function and specifically explain how the LBB concept prevents asymmetric blowdown loads.

ANSWER:

SRP 3.6.3, "Leak-Before-Break Evaluation Procedures," Acceptance Criteria 3, states "Approval of the elimination of dynamic effects from postulated pipe ruptures is obtained individually for particular piping systems at specific nuclear power units. LBB is applicable only to an entire piping system or analyzable portion thereof. LBB cannot be applied to individual welded joints or other discrete locations. Analyzable portions are typically segments located between piping anchor points. When LBB technology is applied, all potential pipe rupture locations are examined. The examination is not limited to those postulated pipe rupture locations determined from SRP Section 3.6.2."

The implication of SRP 3.6.3, Acceptance Criteria 3 is that LBB concept is applicable not only to individual welded joints or those postulated pipe rupture locations determined from SRP Section 3.6.2 but also an entire piping system or analyzable portion thereof.

LBB evaluation results of reactor coolant loop (RCL) piping are described in technical report MUAP-09010, rev.1 "Summary of Stress Analysis Results for Reactor Coolant Loop Piping", and LBB concept is successfully applied to RCL piping.

Therefore, asymmetric blowdown loads are not taken into consideration on the RPV internals.

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.

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QUESTION NO.: 06.03-10

RAI 6.3.1.4-6

Table 6.3-3, "Responses of US-APWR to Generic Safety Issues," GI-105 states the design pressure of the residual heat removal (RHR) system is *900 lb*. Revise this number to the appropriate unit of psia or psig, as appropriate.

ANSWER:

The texts, "design rating of 900 lb" in Table 6.3-3, item 105 will be revised to "design pressure of 900 psig," and also, "The RHR 900 lb design rated system" will be revised to "The RHR 900 psig design pressure system."

Impact on DCD

Table 6.3-3 item 105 will be revised as follows:

No.	Regulatory Position	US-APWR Design
105	**INTERFACING SYSTEM LOCA AT LWRS** The low pressure systems are connected to RCPB using check valves. The leak of check valves could result in the failure of low pressure system. In BWR plants, leak testing for pressure isolation valve in the low pressure system which connects to the RCS is specified to be performed every 18 months in the Tech. Spec. However, 30 failures of RCPB function have occurred in 200 BWR years of operating experience. Among 30 failures, 20 cases are inadvertent remained-open check valves after maintenance by human-error, and 10 cases are stuck-open check valves.	In the US-APWR, the discharge of boric acid water from the accumulators, below the standpipe, replaces the low head safety injection function in typical US PWR plants. As such, there are no "low head" systems outside the containment associated with ECCS. The Residual Heat Removal (RHR) System is a low pressure system that is connected to the RCS and located outside the containment. The RHR system is designed to prevent an interfacing system LOCA by having a ~~rating~~ **design pressure** of 900 ~~lb~~ **psig.** The RHR 900 ~~lb.~~ **psig** ~~rated~~ **design** pressure system can withstand the full RCS pressure. Two motor operated valves in series on the RHR suction line with power lockout capability during normal power operation minimize the probability of RCS pressure entering the RHR system. Even if both these valves are opened during normal power operation, the RHR system is designed to discharge the RCS inventory to the in-containment RWSP.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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Docket No. 52-021

RAI NO.:	**NO. 391-2974**
SRP SECTION:	**06.03 – EMERGENCY CORE COOLING SYSTEM**
APPLICATION SECTION:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-11

RAI 6.3.1.4-7

Table 6.3-3 GI-105 states the RHR is designed to discharge the RCS inventory to the incontainment RWSP if both motor operated valve should open during normal operations. Provide a reference point to the RHR design description and the relief system design description. Since 900 pounds (assuming psi is meant) is significantly less than the RCS system pressure (2250 psi), explain how the RHRS can withstand the full RCS pressure with only one MOV without check valves. Figure 6.2.4-1 indicates that there is a single MOV inside containment for system isolation. A single failure may result in a LOCA outside the containment. How will this be prevented?

ANSWER:

The description of preventing intersystem LOCA is described in Subsection 5.4.7.1 and 19.2.2.5. The wall thickness of 900 psi design is sufficient to prevent pipe rupture in case of RCS pressure of 2250 psi.

For example, typically schedule 80 is selected in 900 psi design piping of the US-APWR. In case of 10-inch and schedule 80 piping, this piping failure pressure is above 5,000 psi, shown in Table 2-18 of NUREG/CR-5603 "Pressure-Dependent Fragilities for Piping Component". Therefore, this design is effective against intersystem LOCA.

Two Motor operated isolation valves (RHS-MOV-001A and 002A, RHS-MOV-001B and 002B, RHS-MOV-001C and 002C, and RHS-MOV-001D and 002D) are installed in series between RCS and RHRS on CS/RHR pump suction side on each train. Not only one MOV. Please see Figure 5.4.7-2. Figure 6.2.4-1 is for only containment barrier, so detail of this configuration is shown in Figure 5.4.7-2. Therefore, single failure does not result in a LOCA outside the containment. In this view point, detail description is shown in Subsection 7.6.1.1.

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.

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QUESTION NO.: 06.03-12

RAI 6.3.1.4-8

Table 6.3-3, GSI-191, refers to Subsection 6.2.2.3 where it is stated the "...selection, purchase, and installation of specific insulation products are controlled by...the COL applicant." The RWSP is also the containment sump. What debris loads are used for the LOCA analysis? In addition, these debris loads must be assumed to be present at the onset of the ECCS operation. Discuss the effects of the fiber, particulates, and chemical debris on in-vessel cooling (downstream effects) throughout the LOCA event. Clarify what would be included in the DCD if some of these materials will be provided by COL applicants.

ANSWER:

The design basis of LOCA generated non-chemical debris that potentially impacts on ECCS operation was discussed in the DCD Reference 6.2-34 "MUAP-08001(R1) US-APWR Sump Strainer Performance" This report was then revised as "Revision 2" and submitted to the NRC in December, 2008 (Ref.1), incorporating plant specific chemical effects tests results (Ref. 2 and 3) to determine the characteristics of chemical debris generated during post-LOCA long term recirculation. Following evaluations, including downstream effects (Ref. 4), were performed in accordance with R.G 1.82 (R3) to assure the ECC operation safely:

1. Break selection (Ref.1)
2. Debris generation (Ref.1)
3. Debris characteristics (Ref.1)
4. Debris head loss (Ref.1)
5. Net positive suction head of ECC/CS pumps (Ref.1)
6. Downstream effects (in-vessel and ex-vessel) (Ref.4)
7. Upstream effects (Ref.1)
8. Chemical effects (Ref.1, 2, 3, and 4)
9. Structural Analysis of the strainer (Ref.5)

Refer the associated technical reports listed in the bottom for the additional details for discussions of debris impacts on ECC operation.

The DCD subsection 6.2.2.3 states that "Selection, purchase, and installation of specific insulation products are controlled by administrative programs developed by the COL Applicant". This statement and associated COL Item 6.2(9) are subject to be revised as per MHI letter, UAP-HF-08259 "Transmittal of COL Information Update for US-APWR Design Control Document Revision 1" (Ref. 6), dated November 7, 2008. Followings were the proposed changes transcribed from the letter:

- The statement in subsection 6.2.2.3 will be revised to read:

  ~~Selection, purchase, and installation of specific insulation products are controlled by administrative programs developed by the COL Applicant.~~ **<u>Insulation is a purchased product and its use is controlled to meet the parameters provided in the US-APWR Sump Strainer Performance.</u>**

- The DCD subsection 6.2.8 COL Item 6.2(9) will be deleted.

<u>Reference</u>

1. MUAP-08001 (R2) "US-APWR Sump Strainer Performance" December, 2008
2. MUAP-08006 (R1) "US-APWR Sump Debris Chemical Effects Tests Plan", November, 2008
3. MUAP-08011 (R0) "US-APWR Sump Debris Chemical Effects Test Results", November, 2008
4. MUAP-08013 (R0) "US-APWR Sump Strainer Downstream Effects", December, 2008.
5. MUAP-08012 (R0) "US-APWR Sump Strainer Stress Report", December, 2008
6. Letter from Yoshiki Ogata, MHI, to NRC dated November 7, 2008; Docket No. 52-021 MHI Ref: UAP-HF-08259; Subject: Transmittal of COL Information Update for US-APWR Design Control Document Revision 1.

Impact on DCD

No changes are proposed beyond the changes described above MHI letter UAP-HF-08259. (Ref. 6)

Impact on COLA

No changes are proposed beyond the changes described above MHI letter UAP-HF-08259. (Ref. 6)

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-13

RAI 6.3.1.4-9

Table 6.3-4, "Responses of US-APWR to Generic Letters and Bulletins," BL 01-01 and BL 02-01 states these bulletins are not applicable as the US-APWR does not have penetrations in the RPV head for safety injection. These bulletins address the potential for boric acid leakage from the control rod drive mechanism nozzles or incore instrumentation system nozzles and subsequent corrosion of the RPV upper head. Identify the in-service inspection (ISI) plan which monitors the RPV upper head for boric acid accumulation and describe the design features that facilitate the inspection.

ANSWER:

The ISI plans to be implemented for the Reactor Vessel Closure Head relating to the boric acid leakage will be described in US-APWR DCD Section 5.2.4, as part of the response to RAI No. 254-2075 Rev.0 (MHI Ref. UAP-HF-09178, dated April 17, 2009).
Design features that facilitate ISI of the RV Closure Head are described in US-APWR DCD Section 5.3.3.7.

Impact on DCD

No changes are proposed beyond those described in MHI letter UAP-HF-09178.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-14

RAI 6.3.1.4-10

Revise Table 6.3.4, GL 89-10, "Item b" to include the complete description by adding the following text after review and revise ", as necessary, the methods for selecting and setting all switches."

ANSWER:

Table 6.3-4 GL 89-10, item b will be revised to add the following text after review and revise, "as necessary, the methods for selecting and setting all switches."

. **Impact on DCD**

DCD Table 6.3-4, item GL 89-10 will be revised as follows:

No.	Regulatory Position	US-APWR Design
GL 89-10	**SAFETY-RELATED MOTOR-OPERATED VALVE TESTING AND SURVEILLANCE** NRC requires the following actions to ensure that valve motor-operator switch settings (torque, torque bypass, position limit, overload) for motor-operated valves (MOVs) in several specified systems are selected, set, and maintained so that the MOVs will operate under design-basis conditions for the life of the plant: a. Review and document the design basis for the operation of each MOV. b. Using the results from item a., establish the correct switch settings, a program to review and revise**, as necessary, the methods for selecting and setting all switches**. c. Individual MOV switch settings should be changed, as appropriate, to those established in response to item b. The MOV should be demonstrated to be operable by testing. d. Prepare or revise procedures to ensure that correct switch settings are determined and maintained throughout the life of plant. Each MOV failure and corrective action taken, including repair, alteration, analysis, test, and surveillance, should be analyzed or justified and documented.	The Testing and Surveillance of MOVs is discussed in DCD Chapter 3, Subsection 3.9.6. Environmental Qualification is discussed in DCD Chapter 3, Section 3.11.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-15

RAI 6.3.1.4-11

Update Table 6.3-4 to include the following and address them in the relevant DCD sections properly:

1. NRC Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors.
2. NRC Generic Letter 2008-01: Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.
3. NRC Bulletin 2003-01: Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors.
4. NRC Bulletin 2003-02: Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity.

ANSWER:

Table 6.3-4 will be updated with addition of GL 2004-02, GL 2008-01, and BL 2003-01.

For the US-APWR, NRC Bulletin 2003-02 is not applicable as the US-APWR Reactor Vessel does not have any lower head penetrations.

Impact on DCD

The items GL 2004-02, GL 2008-01, BL 2003-01, and BL 2003-02 will be added in Table 6.3-4 as follows:

No.	Regulatory Position	US-APWR Design
GL2004-02	POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS NRC requested all PWR licensee to perform a mechanistic evaluation of the potential for the adverse effects of post-accident debris blockage and operation with debris-laden fluids to impede or prevent the recirculation functions of the ECCS and CSS following all postulated accidents for which the recirculation of these systems is required, using an NRC-approved methodology. Individual addressees may also use alternative methodologies to those already approved by the NRC; however, additional staff review may be required to assess the adequacy of such approaches. Implement any plant modifications that the above evaluation identifies as being necessary to ensure system functionality.	This issue is discussed in Subsection 6.2.2.2.6, 6.2.2.3, and following technical reports: MUAP-08001 "US-APWR Sump Strainer Performance" MUAP-08013 "US-APWR Sump Strainer Downstream Effects"

No.	Regulatory Position	US-APWR Design
GL 2008-01	MANAGING GAS ACCUMULATION IN EMERGENCY CORE COOLING, DECAY HEAT REMOVAL, AND CONTAINMENT SPRAY SYSTEM The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter (GL) to address the issue of gas accumulation in the emergency core cooling, decay heat removal (DHR), and containment spray systems for following purposes: (1) to request addressees to submit information to demonstrate that the subject systems are in compliance with the current licensing and design bases and applicable regulatory requirements, and that suitable design, operational, and testing control measures are in place for maintaining this compliance (2) to collect the requested information to determine if additional regulatory action is required	In the US-APWR, the following design provisions are provided in order to prevent void forming in the system: - To reduce gas intrusion into the safety-related pump system, fully submerged strainers is installed to function as vortex suppressor. - To mitigate any possible gas buildup in the RCS, a temperature instrument is installed on the line from the Engineered Safety Feature to the RCS for detection in the MCR. - To prevent boric acid water containing dissolved nitrogen from flowing back from the accumulator tank to RHRS, RHRS return line and accumulator injection line are segregated. - Pump test line is provided in order to allow the dynamic venting of the system through the periodic pump full-flow testing.

No.	Regulatory Position	US-APWR Design
BL2003-01	POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY SUMP RECIRCULATION AT PRESSURIZED-WATER REACTORS NRC requested all PWR licensee to provide a response to state that the ECCS and CSS recirculation functions have been analyzed with respect to the potentially adverse post-accident debris blockage effects identified in this bulletin, taking into account the recent research findings described in the Discussion section, and are in compliance with all existing applicable regulatory requirements. Applicable Regulatory Guidance was Draft	Compliance with R.G 1.82 Rev.3 is discussed in Table 6.2.2-2, and following technical reports: MUAP-08001 "US-APWR Sump Strainer Performance" MUAP-08013 "US-APWR Sump Strainer Downstream Effects"

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-16

RAI 6.3.2.1-1

The text in DCD Section 6.3.2.1.1, "High Head Injection System," uses decimal representations to describe elevations (for example 43.8 ft), while the referenced Figure 6.3-3, "SIS Elevation Diagram," provides the same information in the more traditional format of feet and inches (for example 43 ft-9 in). Revise the text to feet and inches to be consistent with the figure.

ANSWER:

The elevations in the text of Subsection 6.3.2.1.1 will be revised to be consistent with the Figure 6.3-3.

Impact on DCD

Subsection 6.3.2.1.1, the 2nd paragraph, the 1st and the 4th sentences will be revised as follows:

Figure 6.3-3 presents an elevation drawing of the SIS. System piping would normally be filled and vented from the RWSP to the reactor vessel injection nozzles at elevation ~~39.3 ft~~ **39 ft-3 in** prior to startup. Thus, the injection piping is completely filled with water. A series of four check valves are installed between each SI pump and the direct vessel injection (DVI) nozzles at the reactor vessel. This series of check valves provides a "keep full" function, while preventing a drain-down to the RWSP. As shown, ~~24.2 ft.~~ **24ft-2in** are available between the 100% RWSP level at elevation ~~19.5 ft.~~ **19 ft-6 in**, and the highest SI piping at elevation ~~43.7 ft~~ **43 ft-8 in**.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-17

RAI 6.3.2.1-2

Has the hydrodynamic loads evaluation for the emergency letdown system spargers in the RWSP been completed? If the analysis has been completed, provide the reference to the evaluation.

ANSWER:

The hydrodynamic loads evaluation for the emergency letdown system spargers in the RWSP has not been completed yet. However, the hydrodynamic load may not be the problem because the reactor coolant discharged from the sparger during the emergency letdown is as small as 265 gpm.

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.

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QUESTION NO.: 06.03-18

RAI 6.3.2.1-3

Provide test data which demonstrates that Figure 6.3-15, the High Head Safety Injection Flow Characteristic Curve (Minimum Safeguards), is conservative relative to actual pump performance. Provide details on how the test data was generated. Describe the testing conditions and their relevance to the actual system conditions during normal operation and postulated accidents.

ANSWER:

Figure 6.3-15, The High Head Safety Injection Flow Characteristic Curve (Minimum Safeguards) used the minimum head curve in the Figure 6.3-4, Safety Injection Pump Performance Flow Requirement. Figure 6.3-4 is developed based on the experience and not on the actual pump testing data. The actual pump has not been manufactured, and Table 6.3-4 will be the requirement for the manufacturer.

The testing is performed during the normal operation or the plant shutdown. The pressure difference (psi) measured by pressure instrumentation installed at the inlet and outlet of the pump is converted to the pump head (ft) and is confirmed to meet the requirement. The pump head confirmed by the test is secured even in the accident since the pump head is not affected by the fluid temperature. However, the pump discharge pressure is affected by the fluid temperature (i.e., fluid density becomes smaller as the fluid temperature becomes higher, and the discharge pressure (psig) becomes lower even if the pump head (ft) is the same), therefore, the fluid density at the temperature in the accident is used in developing Figure 6.3-15.

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.

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QUESTION NO.: 06.03-19

RAI 6.3.2.2-3

Modify the text in DCD Section 6.3.2.2.2 to include the evaluation used to develop the required capacity (724 ft^3) for the small injection flow rate or provide a reference.

ANSWER:

The basis of the required capacity (724 ft^3) at the small flow rare injection is described in Section 3.1 in Reference 1, and the reference will be provided.

Reference

1. The Advanced Accumulator, MUAP-07001-P, Rev. 2. , September 2008.

Impact on DCD

Subsection 6.3.2.2.2, the 3rd paragraph will be revised as follows:

To maintain downcomer water level and establish post-LOCA core re-flood conditions, large accumulator injection flow is followed by an assumed 180 seconds of accumulator injection flow at a small flow rate (followed by the injection flow from the SI pumps). The required capacity of each accumulator at the small injection flow rate is approximately 724 ft^3, which is increased to approximately 784 ft^3 **(Ref. 6.3-3)**.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-20

RAI 6.3.2.2.1-1

In Section 6.3.2.2.1 Safety Injection Pumps , second sentence states that "This SI pump flow rate is based on two SI pumps operating (active failure of one SI pump and one SI pump out of service), with each SI pump delivering 1,057 gpm against near atmospheric pressure." The design flow given on Table 6.3-5 is 1,540 gpm. Explain why the design flow of 1,540 gpm is greater than the run out flow of 1,057 gpm?

ANSWER:

The SI pump design flow of 1540 gpm is greater than the safety injection flow rate of 1057 gpm by including the margin and also including the minimum flow rate.

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.

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QUESTION NO.: 06.03-21

RAI 6.3.2.2.4-1

In Section 6.3.2.2.4 ECC/CS Strainers, the fourth sentence states that "The strainer sizing accommodates the estimated amount of debris potentially generated in containment." Provide a relevant reference and explain how this is accomplished?

ANSWER:

The DCD subsection 6.3.2.2.4 last sentence in first paragraph discusses relevant reference (i.e., subsection 6.2.2) with respect to compliance with RG.1.82. The subsection 6.2.2.2.6 first paragraph states that the strainer large surface to account for potential debris blockage and maintain safety performance, corrosion resistance, and a strainer hole size to minimize downstream effects. Additional design attributes are referred in the US-APWR Sump Strainer Performance document (Ref 6.2-34).

Impact on DCD

The DCD subsection 6.3.2.2.4, first paragraph will be revised to provide a relevant reference more precisely as follows:

> Four independent sets of strainers are provided inside the RWSP as part of the ECCS and CSS. ECC/CS strainers are provided for preventing debris from entering the safety systems, which are required to maintain the post-LOCA long-term cooling performance. ECC/CS strainers are designed to comply with RG 1.82. Strainer compliance with RG 1.82 is discussed in Subsection 6.2.2**.2.6**.

The DCD subsection 6.3.2.2.4, last sentence in forth paragraph will be revised to provide a relevant reference more precisely as follows:

> The strainer sizing accommodates the estimated amount of debris potentially generated in containment. **(Subsection 6.2.2.2.6)**

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-22

RAI 6.3.2.2.4-2

The fifth sentence of Section 6.3.2.2.4 ECC/CS Strainers states that "The RWSP water chemistry is controlled so as to minimize the chemical effects between the sump water and potentially corrosive materials in containment is considered. Clarify how the chemistry is controlled to minimize the chemical effects or point to the right reference.

ANSWER:

The cited statement in the DCD subsection 6.3.2.2.4 fifth paragraph is not appropriate. The RWSP water chemistry is NOT controlled so as to minimize the chemical effects between the sump water and potentially corrosive materials in containment. The cited statement will be revised properly.

Subsequent statement to the above, in the DCD subsection 6.3.2.2.4 sixth paragraph, "The COL Applicant is responsible for developing a program to maintain RWSP water chemistry including surveillance test procedures", is also subject to be deleted. Refer the MHI letter, UAP-HF-08259 "Transmittal of COL Information Update for US-APWR Design Control Document Revision 1", dated November 7, 2008. Accordingly, COL Item 6.3(4) will be deleted.

Impact on DCD

The DCD subsection 6.3.2.2.4, fifth paragraph will be deleted as follows:

> ~~The RWSP water chemistry is controlled so as to minimize the chemical effects between the sump water and potentially corrosive materials in containment is considered.~~

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-23

RAI 6.3.2.2.5-2

In Section, it is stated that "NaTB in baskets is dissolved in spray water in the containers. The solution containing NaTB is discharged from each container to the RWSP through 4-inch diameter NaTB solution transfer pipes." Are there any postulated LOCA breaks that could cause debris to block the four inch lines? If yes, provide the evaluation results and the relevant references.

ANSWER:

There are no credible postulated LOCA breaks that could cause debris to block the four inch lines of NaTB baskets. Followings are the clarifications for that:

- Postulated pipe break possibly considered for debris blockage the four inch lines of NaTB baskets is primary side pipe break. There is no need to consider secondary side (i.e., main steam pipe or feed water pipe) break for the impact on the basket's function.
- The location of postulated primary side pipe breaks are lower portions inside secondary shield walls where is far from the NaTB baskets location.
- Since layers of gratings are provided inside secondary shield walls, only small/fine debris that could pass though grating is possibly blown up to containment atmosphere over the NaTB baskets.
- Therefore, there is no "large" LOCA debris which potentially blocks the four inch lines of NaTB basket.

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.

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QUESTION NO.: 06.03-24

RAI 6.3.2.2.3-1

It is stated in Section 6.3.2.2.3 that "The RWSP capacity includes an allowance for instrument uncertainty and the amount of holdup volume loss within the containment." Where are the instrument uncertainty and holdup volume loss assessments documented? Provide the references.

ANSWER:

The assessment of holdup volumes associated with RWSP water volume calculation is provided in the following references:

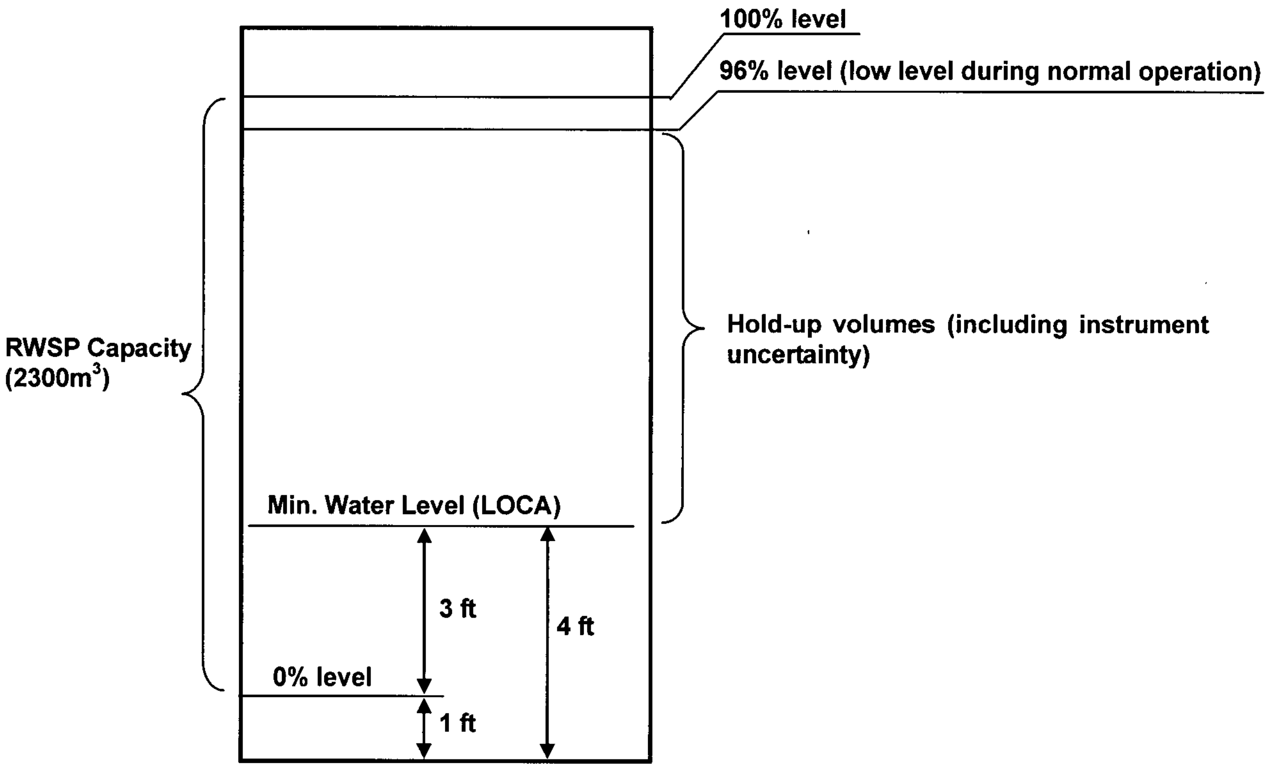
- The DCD subsection 6.2.2.2.5 discusses the capacity of the RWSP and hold up volumes as follows:

  The RWSP is located on the lowest floor inside the containment, with a minimum 81,230 ft^3 capability available, it is designed with sufficient capacity to meet long-term post-LOCA coolant needs, including holdup volume losses. Potential holdup areas within the containment are depicted in Figure 6.2.1-9.

- The technical report MUAP-08001 "Sump Strainer Performance" subsection 3.7 ""Upstream effect" discusses detail of hold up volume (i.e., 1644 m^3) calculation. Table 3-10 in MUAP-08001 provides summary of each hold up volumes in the containment.

The discussion of instrument uncertainty associated with RWSP water volume calculation is not provided in the DCD. However, it was considered for determination of RWSP capacity as follows:

The nominal RWSP capacity (i.e., 81,230 ft^3 = 2300 m^3) was defined as water volume between 0% to 100% water level of the RWSP. During normal operation, the RWSP is controlled to have minimum 96% capacity by makeup operation. During LOCA, the RWSP is replenished with water which has been released to the containment sufficient to maintain adequate net positive suction head to the safety injection and containment spray/ residual heat removal pumps throughout the event. In the event, hold up volumes (i.e., 1644 m^3) is considered not contribute to the replenishment, and subtracted from initial RWSP water volume with considering 1.7% of instrument uncertainty. Following figure is a schematic of water volume calculation of the RWSP.



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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 391-2974**

SRP SECTION: **06.03 – EMERGENCY CORE COOLING SYSTEM**

APPLICATION SECTION: **6.3**

DATE OF RAI ISSUE: **6/15/2009**

QUESTION NO.: 06.03-25

RAI 6.3.2.2.3-2

"Figure 6.2.1-1 through Figure 6.2.1-4 are plots of containment internal pressure and temperature versus time for the most severe primary and secondary system piping failures." What is the most severe pipe break with respect to containment pressure and temperature? What is the worst case pipe break with respect to creating debris that could enter the RWSP and potentially plug the recirculation?

ANSWER:

Maximum containment pressure, 57.5psig is predicted on the assumption of double ended break at the pump suction of the primary system. Figure 6.2.1-1 indicates the corresponding pressure transient. On the other hand, maximum containment temperature, 355 ^{0}F is predicted on the assumption of double ended break of the secondary system during full power operation and under offsite power available. Figure 6.2.1-40 indicates the corresponding temperature transient.
Various assumptions of accident condition and results for containment analysis are summarized in Table 6.2.1-6 through 6.2.1-8 of DCD. Above two cases are also included in these tables.

The worst case pipe break with respect to creating debris that could enter the RWSP and potentially plug the recirculation is the main coolant pipe break as discussed in the section 3.1 of technical report MUAP-08001. Refer the response to Question 06.02.02-20 in the RAI-354-2585 (UAP-HF-09365, "MHI's Responses to US-APWR DCD RAI No. 354-2585 Revision 0" dated July 7, 2009) for further information discussing the worst case pipe break impacts on ECC operation.

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.

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QUESTION NO.: 06.03-26

RAI 6.3.2.2.3-3

In Section 6.3.2.2.3, the auxiliary RWSP storage tank is designed to ensure the required volume for refueling operations. Provide sufficient information regarding this tank and the relevant piping/pump system and describe whether this tank is intended to be used during accident conditions (i.e., LOCA and AOOs). If it could be activated during these events, describe the necessary operator actions and the equipment availability during these events.

ANSWER:

The refueling water storage auxiliary tank (RWSAT) is used as the water source performing the following functions:

- Supply boric acid water to the refueling canal and the fuel inspection pit during the refueling operation.
- Supply boric acid water to the refueling canal and the cask pit during the carrying out operation for the spent fuel.
- Back up water source for the charging pumps.

The boric acid water in the RWSAT is supplied to refueling canal, fuel inspection pit, and cask pit using the Refueling Water Recirculation Pumps shown in Figure 6.3-7, DCD Section 6.3.

The RWSAT has not any safety-related function and is not used during accident condition.

By the way, since pump names in Figure 6.3-7 are not consistent with that in other section, pump names will be revised as follows:

Refueling Water **Recirculation** ~~Recirculating~~ Pump - A
Refueling Water **Recirculation** ~~Recirculating~~ Pump - B

Impact on DCD

Pump names in Figure 6.3-7 will be revised as follows:

Refueling Water **Recirculation** ~~Recirculating~~ Pump - A
Refueling Water **Recirculation** ~~Recirculating~~ Pump - B

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-27

RAI 6.3.2.2-1

On Page 6.3-4, it is stated that void formation due to water column separation in the SI piping is precluded because the available head (24 ft-2 in) is less than the head (30 ft) which would result in column separation and no delay is assumed between the system initiation and the injection flow into the vessel downcomer. The liquid vapor pressure is generally accepted as the cavitation inception pressure in standard column separation numerical models. Is this model being used? Provide the details of the column separation analysis.

ANSWER:

The liquid vapor pressure is used as the inception pressure of the water column separation. The water temperature in the piping is assumed to be the same as the maximum temperature in the building where the piping is installed during normal operation. This temperature is 120 °F in the containment, and 105 °F in the reactor building. In this case, the higher value, 120 °F was conservatively used. The liquid vapor pressure of water at 120 °F is 1.695 psia. This value is multiplied by the specific volume of saturated water at 120 °F, 0.01620 ft^3/lb, and the water head is:

$1.695 \times 12^2 \times 0.01620 = 3.954$ (ft)

Standard atmospheric pressure of 14.70 psia is converted to the water head at 120 °F:

$14.70 \times 12^2 \times 0.01620 = 34.29$ (ft)

Therefore, the minus water head to begin the water column separation is:

$34.29 - 3.954 = 30.34$ (ft) $\cong$ 30 (ft)

The minus head of 24 ft - 2 in developed in the SI piping is blow the 30 ft and consequently, the water column separation does not occur.

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.

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QUESTION NO.: 06.03-28

RAI 6.3.2.2-2

It is not clear that void formation in the SI piping can be precluded during a postulated LOCA. During the blowdown phase, quick depressurization would occur through the entire primary system. Flashing could occur in the un-isolated sections of SI piping (between the RPV DVI nozzle and the closest check valve (SIS-VLV-013A, B, C, and D). Justify why there is no void formation. If there is, demonstrate that the water hammer would not occur in these sections during a LOCA, or would not result in damage to the piping sufficient to degrade the SI injection.

ANSWER:

As shown in Figure 6.3.28-1, the SI piping is horizontal at the inlet of the reactor vessel, and this horizontal piping may be filled with vapor due to the rapid depressurization of the RCS.

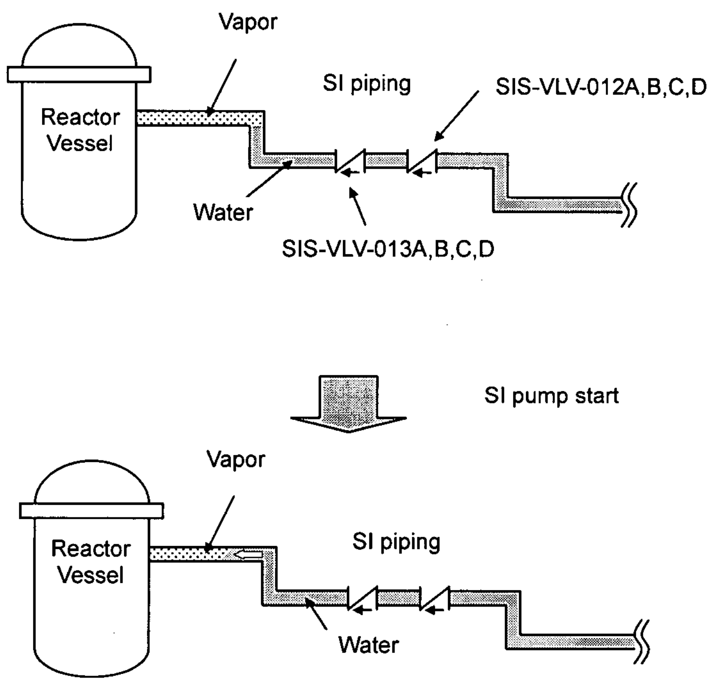

Figure 6.3.28-1 Conceptual figures of aspect in safety injection piping

Figure 6.2.28-2 shows the steam hammer mechanism due to the collapse of steam void being quickly condensed by the cold water.

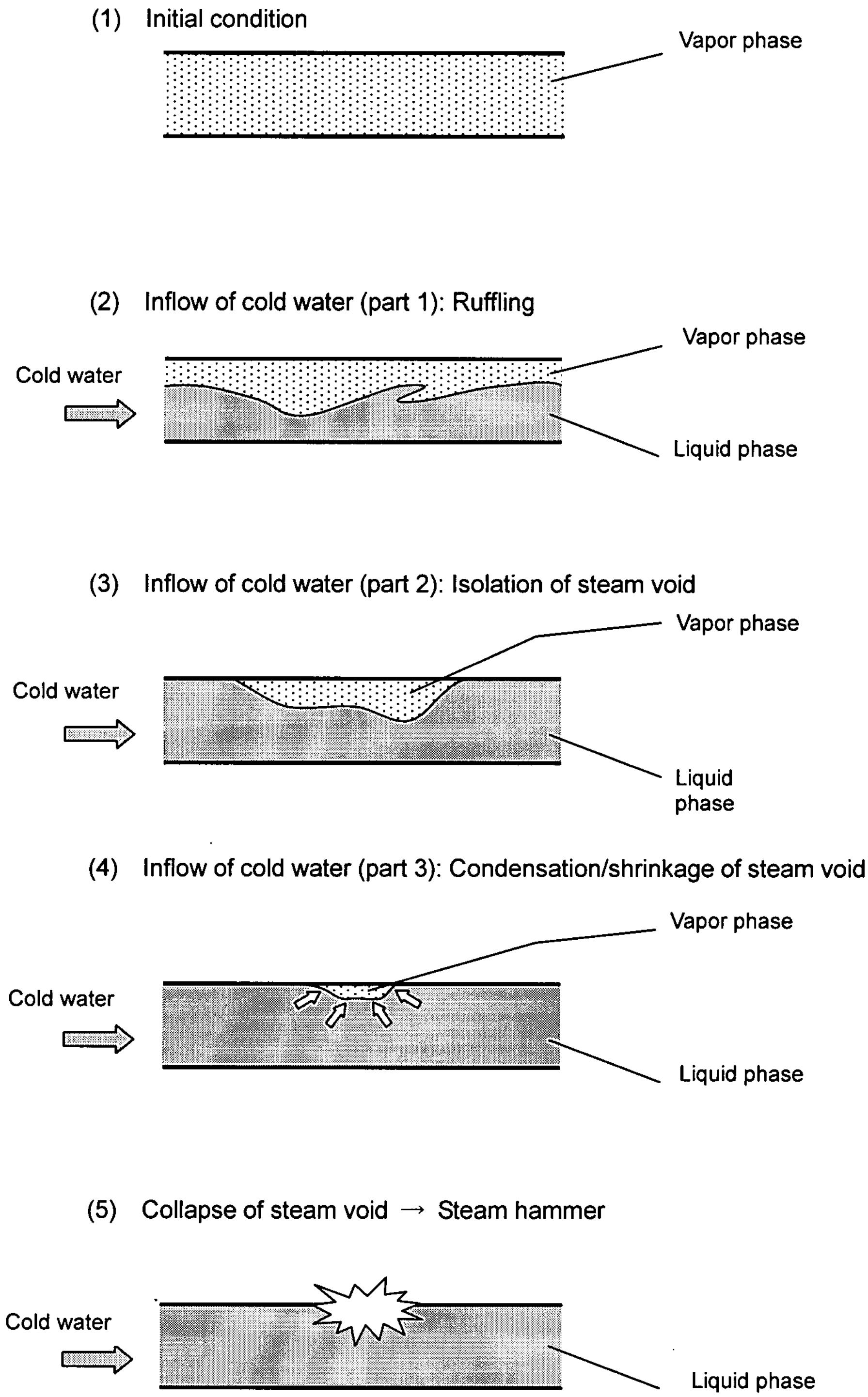

Figure 6.3.28-2 Steam Hammer Generating Mechanism

From the mechanism as shown in Figure 6.3.28-2, the following two conditions must be met for steam hammer generation:

Condition 1: Two-phase environment where vapor and liquid exist together
Condition 2: Steam void is surrounded by cold water

For the SI piping of the US-APWR, even if the condition 1 is realized, the condition 2 is not realized since the inner diameter of SI piping is as small as 3.44 inch and flow velocity is too high to form the separated flow of vapor and liquid as shown in Figure 6.3.28-2 (2). Consequently, steam void is not entrained into liquid phase, and is pushed out without the steam hammer as shown in Figure 6.3.28-1 (2).

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.

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QUESTION NO.: 06.03-29

RAI 6.3.2.2-3

The SI pumps are horizontal, multi-stage centrifugal type pumps. Typically these pumps require cooling (from non-safety related and safety-related cooling systems) to protect the motors and seals to ensure they can provide the required flow for the duration of the accident. Describe the cooling system(s) and include the cooling system(s) in the failure modes and effects analysis.

ANSWER:

SI pump is supplied cooling water from the Component Cooling Water System (CCWS). During an accident, CCWS is divided into four independent trains, and the failure of one CCWS train does not result to the simultaneous loss of function of more than two SI pumps. For detailed description of the CCWS, please refer to the DCD Chapter 9, subsection 9.2.2, "Component Cooling Water System." Section 6.3, Table 6.3-6 Failure Mode and Effect Analysis - Safety Injection System will be revised with addition of "Component Cooling Water" as item 11.

Impact on DCD

The item 11, "Component Cooling Water" will be added in Table 6.3-6 as follows:

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
11. Component Cooling Water	Failure to deliver fluid due to loss of Component Cooling Water	LOCA, Non-LOCA Safe Shutdown	Same as Item 1.	Same as Item 1.	
	Failure to deliver fluid due to loss of Component Cooling Water with one SI train out of service.				

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-30

RAI 6.3.2.2-4

The building area which houses the SI pumps and instrumentation should be provided with HVAC to protect the pumps and instrumentation from excessive temperatures. Describe the HVAC system and include the HVAC system in the failure modes and effects analysis.

ANSWER:

During an accident, the Safeguard Component Area in the reactor building, where SIS components are installed, is maintained in the adequate environmental condition by the Safeguard Component Area HVAC System. The Safeguard Component Area HVAC System consists of four trains of completely independent subsystems; therefore, the failure of one train does not result to simultaneous loss of SIS function of more than two trains. For detailed description of the Safeguard Component Area HVAC System, please refer to the DCD Chapter 9, Subsection 9.4.5, "Engineered Safety Feature Ventilation System." Section 6.3, Table 6.3-6 Failure Mode and Effect Analysis - Safety Injection System will be revised with addition of "Safeguard Component Area HVAC" as item 12.

Impact on DCD

The item 12, "Safeguard Component Area HVAC" is added in Table 6.3-6 as follows:

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
12. Safeguard Component Area HVAC	Failure to deliver fluid due to loss of Safeguard Component Area HVAC	LOCA, Non-LOCA Safe Shutdown	Same as Item 1.	Same as Item 1.	
	Failure to deliver fluid due to loss of Safeguard Component Area HVAC with one SI train out of service				

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-31

RAI 6.3.2.2-5

Summarize or reference, the NPSH analyses performed to ensure there is an adequate head available for all accident conditions.

ANSWER:

The NPSH analysis is summarized in DCD section 6.2, subsection 6.2.2.3, the 8th and the 9th paragraphs. The detailed analysis is provided in Reference 1, and is referred in subsection 6.2.2.3, the 6th paragraph.

Reference

1. US-APWR Sump Strainer Performance, MUAP-08001-P, Rev. 2, (Proprietary), and MUAP-08001-NP, Rev. 2, (Non-Proprietary), December 2008.

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.

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QUESTION NO.: 06.03-32

RAI 6.3.2.2-6

Describe the "assumed large break LOCA" used to determine the SI and accumulator delivery requirements. Include any assumptions made concerning the initial conditions in the RCS, ECCS availability, and assumed failures.

ANSWER:

For the basis of the Accumulator and SI pump delivery requirements, please refer to the descriptions in the Reference 1, section 2.3, "Performance Requirements for ACC."

Reference

1. The Advanced Accumulator, MUAP-07001-P, Rev 2, September 2008.

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.

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QUESTION NO.: 06.03-33

RAI 6.3.2.2-7

In DCD Section 6.3.2.2.2, "Accumulators," it is stated "*Although four accumulators are provided, accumulator sizing is based on three accumulators to account for unavailability of flow from the accumulator installed on the broken loop during a LOCA whose contents are assumed to spill to the containment so that it does not contribute to the core injection. One third of the remaining accumulator volume is also assumed to be lost to the spill through the postulated pipe break. Two thirds of the remaining accumulator volume is available for injection.*" Is the additional loss used in the LOCA analyses or is it only used to size the accumulators by providing additional margin for the available fluid?

ANSWER:

The losses described in Subsection 6.3.2.2.2 are only used to size the accumulator and are not used in the LOCA analysis.

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.

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QUESTION NO.: 06.03-34

RAI 6.3.2.2-9

Explain why there is no upper value limit for the large injection flow rate capacity?

ANSWER:

The required function for the Advanced Accumulator in the PCT evaluation is to fill-up the lower plenum promptly during the refill period, then simultaneously raise the water level in the downcomer. Therefore, it is conservative treatment qualitatively to shorten the duration of the large flow rate mode of the Advanced Accumulator. The lower limit of the large flow capacity is the important value in design to make the large flow duration conservative in PCT evaluation. The upper limit of the large flow capacity exists as the bounding of the safety analysis condition, but it is not the important value in designing, therefore, only the lower limit is provided in Table 6.3-5, Safety Injection System Design Parameters (Sheet 2 of 3). The minimum and the maximum administrative values of the accumulator inventory are provided in the technical specification.

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.

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QUESTION NO.: 06.03-35

RAI 6.3.2.2-11

There is no ITAAC to verify the small injection flow rate required capacity. Explain why this value does not need to be verified.

ANSWER:

In the PCT evaluation, the accumulator water capacity and the large flow injection capacity are given as analysis conditions, and the small flow injection capacity recognized as the difference of the accumulator water capacity and the large flow injection capacity. Therefore, these two parameters, the accumulator water capacity and the large flow injection capacity are also verified in ITAAC.

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.

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QUESTION NO.: 06.03-36

RAI 6.3.2.2-12

The Direct Vessel Safety Injection Line Isolation Valves (SIS-MOV-011A, B, C, and D) have throttling capability for safe shutdown operations. The open or closed valve position is indicated in the MCR and RSC. TS 3.5.2, "Safety Injection System (SIS) –Operating," SR 3.5.2.2 only verifies valves to be in the correct position. For a valve has throttling capabilities shouldn't the valve position be part of the display? What process variable is monitored to determine if the correct throttled position is obtained? Please identify the process used to ensure that the valve is not left in a partially open configuration prior to entering an operating mode.

ANSWER:

SIS-MOV-011A, B, C, and D are not used for flow control except in the safe shutdown. During an accident or a transient, these valves are used with fully open position; therefore, the correct position of these valves during normal operation must be fully open. The text "in the correct position" in SR 3.5.2.2 of TS 3.5.2 means "in fully open" for these valves. Therefore, the display of throttled position of these valves need not be provided in MCR or RSC for verifying the correct position, and it is ensured that these valves do not throttled open during an event by verifying their fully open position. During a safe shutdown, the safety injection flow is controlled by the operator with monitoring pressurizer water level and safety injection flow rate.

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.

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QUESTION NO.: 06.03-37

RAI 6.3.2.2-13

Please describe the need for a throttling capability for the Safety Injection Pump Full-flow Test Line Stop Valves (SIS-MOV-024A, B, C, and D). The open or closed valve position is indicated in the MCR and RSC. For a valve has throttling capabilities shouldn't the valve position be part of the display? What process variable is monitored to determine if the correct throttled position is obtained? Are there any negative consequences of the valve being in a partially open configuration when performing the SI full-flow test?

ANSWER:

Since the full-flow test line is not completely filled with water during normal operation, flushing large flow of water could cause the water hammer. Consequently, in the full-flow testing, it is the recommended procedure that the safety injection pumps are started with safety injection pump full-flow test line stop valves (SIS-MOV-024A, B, C, and D) in fully closed position, and then, the testing flow rate could reach the full-flow with opening these valves gradually. This is the reason for these valves having throttling capability.

During an accident or a transient, these valves must be fully closed in order to ensure sufficient safety injection flow rate, therefore, their correct position in normal operation must be fully closed. During the normal operation, these valves are fully closed with electrical power removed, therefore, the display of throttled position of these valves need not be provided in MCR or RSC for verifying the correct position. The full-flow testing flow rate is controlled by the operator with monitoring safety injection pump discharge flow rate.

These valves must not be allowed to operate in throttled position except in the full-flow testing.

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.

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QUESTION NO.: 06.03-38

RAI 6.3.2.2-14

Provide the evaluation performed to size the Accumulator Safety Valves (SIS-VLV-126A, B, C, and D) relief capacity. Address the concern that the valve may need to discharge both nitrogen gas and water, if the SI failure fills the accumulator (going water solid).

ANSWER:

Each of the accumulator safety valves (SIS-VLV-126A, B, C, D) is required to release the maximum accumulator makeup flow rate of 100 gpm (13.4 ft^3/min) by the SI pump. On the other hand, the release set pressure of the safety valve is 700 psig (715 psia). The required release capacity of 13.4 ft^3/min at 700 psig is converted at the standard condition (15 psia), and obtained:

13.4 ft^3/min X (715 psia / 15 psia) = 639 ft^3/min

The design release capacity of SIS-VLV-126A, B, C, D is set to 1500 ft^3/min, which is the same flow rate as the Accumulator N_2 Supply Line Safety Valve (SIS-VLV-116), and this capacity exceeds the required release capacity of 639 ft^3/min.
The release capacity of SIS-VLV-116 is set to 1500 ft^3/min in order to release the maximum N_2 supply flow rate of 1423 ft^3/min.

The most limiting failure of excessive makeup to accumulator is the stuck open of Accumulator Makeup Flow Control Valve (SIS-HCV-989). A flow restraint orifice is provided on the accumulator makeup line to prevent the accumulator makeup water flow rate from exceeding 100 gpm even if SIS-HCV-989 fully opens. The excessive makeup due to the failure is announced to operator by the accumulator high water level alarm. The operator can terminate the excessive makeup by stopping the SI pump within sufficient period since it takes about 45 minutes for an accumulator to be filled completely with water from the high water level is alerted. Therefore, it is quite unlikely that water is released from the safety valve due to the failure of Safety Injection System.

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.

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QUESTION NO.: 06.03-39

RAI 6.3.2.2-15

The Emergency Letdown Line Isolation Valves (SIS-MOV-032B, D) have throttling capability, based on a review of DCD Figure 6.3-2, "ECCS Piping and Instrumentation Diagram." The open or closed valve position is indicated in the MCR and RSC. Please describe the need for a throttling capability and include this capability in the DCD description. For a valve has throttling capabilities shouldn't the valve position be part of the display? What process variable is monitored to determine if the correct throttled position is obtained? Are there any negative consequences of the valve being in a partially open configuration when starting the feed and bleed procedure?

ANSWER:

The function of the Emergency Letdown System is to divert the letdown to RWSP as a substitute for CVCS during the safe shutdown. SIS-MOV-032B and D are provided with throttling capability to enable the control of letdown flow rate. The text of DCD Chapter 6, Subsection 6.3.2.2.6.17, "Emergency Letdown Isolation Valve" will be revised to add the description of throttling capability of SIS-MOV-032B, D.

SIS-MOV-032B and D are not used for flow control except in the safe shutdown. During an accident or a transient, these valves are used with fully closed position; therefore, the correct position of these valves during normal operation must be fully closed. The text "in the correct position" in SR 3.5.2.2 of TS 3.5.2 means "in fully closed" for these valves. Consequently, the display of throttled position of these valves need not be provided in MCR or RSC for verifying the correct position, and it is ensured that these valves do not throttled open during an event by verifying their fully closed position. During a safe shutdown, the emergency letdown flow rate is controlled by the operator with monitoring pressurizer water level and safety injection flow rate.

Impact on DCD

Subsection 6.3.2.2.6.17 will be revised with addition of the following sentence to the end of the 1st. paragraph:

"2nd. Emergency Letdown Line Isolation Valves (SIS-MOV-032B and D) have the throttling capability to enable the control of letdown flow rate."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-40

RAI 6.3.2.2-16

On Figure 6.3-2, "ECCS Piping and Instrumentation Diagram (Sheet 3 of 4)," Note 4 states: "to be provided with overpressurization protection," on the nitrogen supply line segment outside containment. Explain the need for this protection and does this statement mean there is a COL action item to provide this feature in the design?

ANSWER:

Note 4 is applicable to the pressure gage PI-916. The measuring range of this pressure gage is 0 to 700 psig, however, the design pressure of the nitrogen supply line outside containment on which the pressure gage is installed is 2485 psig. The Note 4 requires that the pressure gage is designed to withstand the design pressure of the line, and is not the COL action.

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.

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QUESTION NO.: 06.03-41

RAI 6.3.2.2-17

Figure 6.3-2 refers to equipment/system "CVDT." It also appears on Figure 6.2.4-1, "Containment Isolation Configurations (Sheet 3 of 50)." This is not identified in the Acronyms and Abbreviation table. Define the equipment/system and add the definition to the table.

ANSWER:

The CVDT is the tank in Liquid Waste Management System and is the acronym of "containment vessel reactor coolant drain tank." CVDT is defined in the DCD Chapter 11, Section 11.2, Liquid Waste Management System. CVDT will be added in "ACRONYMS AND ABBREVIATIONS" in DCD Chapter 6.

Impact on DCD

The followings are added in "ACRONYMS AND ABBREVIATIONS" in DCD Chapter 6.

CVDT containment vessel reactor coolant drain tank

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-42

RAI 6.3.2.2-18

The Accumulator Nitrogen Supply Containment Isolation Valve (SIS-AOV-114), does not appear to have position indication in the MCR and RSC. As a containment isolation valve, the open or closed valve position should be indicated in the MCR and RSC. Is the valve position indicated in the MCR and RSC? If not, why is position indication not required?

ANSWER:

The open or closed position of the Accumulator Nitrogen Supply Containment Isolation Valve (SIS-AOV-114) is displayed in the MCR and RSC. The description will be added in Subsection 6.3.2.2.6.20 that the open or closed position is displayed in the MCR and RSC.

Impact on DCD

Subsection 6.3.2.2.6.20 will be revised as follows:

One normal closed air operated globe valve is aligned in the accumulator nitrogen supply line as a containment isolation valve. The valve is closed automatically on receipt of a containment phase "A" isolation signal. **The open or closed valve position is indicated in the MCR and RSC.** The accumulator nitrogen supply containment isolation valve (SIS-AOV-114) is Equipment Class 2, seismic category I.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-43

RAI 6.3.2.3-1

Summarize the Equipment Class for the ECCS components and piping inside containment to ensure that they meet, at a minimum, Equipment Class 2 and Seismic Category I. For Equipment Class 3 and Class 4 components and piping identify the seismic design category associated with each class.

ANSWER:

The equipment class, location, and seismic category of ECCS components and piping is summarized in DCD Chapter 3, Section 3.2, "Classification of Structures, Systems, and Components," Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," item 4, "Safety Injection System" (Sheet 13 of 53, 14 of 53 and 15 of 53).

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.

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QUESTION NO.: 06.03-44

RAI 6.3.2.4-1

Describe the material specification characteristics for ECCS valves, both for the seating surfaces and stems, to prevent failures and to reduce wear.

ANSWER:

DCD Chapter 6, Subsection 6.3.2.4, "Material Specifications and Compatibility" will be revised with adding the description of the material specifications of ECCS valves (seat surface and stem).

Impact on DCD

Subsection 6.3.2.4, 1st paragraph will be revised as follows:

All surfaces of the SIS in contact with borated reactor coolant, or a mixture of borated reactor coolant and NaTB, are austenitic stainless steel. The nitrogen supply piping is carbon steel. The accumulator vessels are stainless clad carbon steel. **The surfaces of SIS valve seating are hard-faced to prevent failure and to reduce wear. In addition, valve stem materials are selected considering corrosion resistance, high-tensile properties and resistance to surface scoring by packing.** The complete material specifications are presented in Section 6.1. System and component purchasing and procurement activities are performed within the guidelines provided by Chapter 17, "Quality Assurance."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-45

RAI 6.3.2.5-1

The failure modes and effects analysis (FMEA) presented in Table 6.3-6, "Failure Modes and Effects Analysis - Safety Injection System," appears to be a very high level summary. For example, the SI failure mode is described as "Failure to deliver flow." This could result from, for example, failure to start (mechanical, electrical, or I&C failure), failure to run (mechanical or electrical failure), or failure to delivery required flow (mechanical failure). The flow could be excessive which could result in pump run-out, or could be inadequate and not provided adequate make-up. In addition, failure of a closed valve to open or of an open valve to close would result in no flow. Describe the level of detail used to develop Table 6.3-6, or provide a reference to the detailed FMEA evaluation.

ANSWER:

MHI does not consider that the FMEA in Table 6.3-6 is a very high level summary. This FMEA is developed based on failures describer in the 1st. column. The item 1 is the mechanical failure of SI pump. The electrical failure is described in the item 9, and the I&C failure is described in the item 8. The failure to deliver flow due to the valve failure is addressed in the item 2 and 3. The pump run-out is addressed in the item 13, which will be added in accordance with response of this RAI, Question 06.03-49.

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.

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QUESTION NO.: 06.03-46

RAI 6.3.2.5-2

Table 6.3-6 indicates the "Failure Detection Method" is based on information provided in the MCR. Is the same information available from the RSC? If not, explain how the operator monitors the status of safety systems when the MCR in uninhabitable.

ANSWER:

The information provided in the MCR described in Table 6.3-6 "Failure Detection Method" is applicable to the RSC.

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.

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QUESTION NO.: 06.03-47

RAI 6.3.2.5-3

Item 4. Accumulator discharge valve, in Table 6.3-6, Column "Effect on System Operation," should be clarified as "No effect on plant safety because the accumulator nitrogen gas volume can be vented to the containment atmosphere by opening the accumulator nitrogen discharge valve SIS-MOV-121A or B."

ANSWER:

The description in the column "Effect on System Operation" will be more clarified.

Impact on DCD

Table 6.3-6, item 4 will be revised as follows:

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
4. Accumulator discharge valve SIS-MOV-101A (SIS-MOV-101B, C and D analogous)	Failure to close on demand	Safe shutdown; isolate accumulator A from the RCS prior to depressurization to prevent introducing nitrogen into RCS	No effect on plant safety because the accumulator nitrogen gas volume can be vented **to the containment atmosphere** by opening the accumulator nitrogen discharge valve, and atmospheric vent SIS-MOV-121A or B.	Valve position indication in MCR.	Valves SIS-MOV-121A and B are parallel vents to the atmosphere and are powered from different Class 1E supplies.
	Failure to close on demand with a Class 1E supply out of service		No effect on plant safety because the accumulator nitrogen gas volume can be vented **to the containment atmosphere** by opening the accumulator nitrogen discharge valve, and atmospheric vent SIS-MOV-121A or B.		

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 06.03-48

RAI 6.3.2.5 -4

The long term cooling limiting failure is based on leakage from a valve or pump seal. Leakage is detected and alarmed in the MCR. Is the same information available from the RSC? If not, explain how the operator monitors the status of safety systems when the MCR in uninhabitable.

ANSWER:

The same information provided in the MCR is available from the RSC.

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.

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QUESTION NO.: 06.03-49

RAI 6.3.2.5-5

The failure modes and effects analysis needs to consider the inadvertent (I&C failure) opening of a hot leg injection isolation valve (SIS-MOV-014A, B, C, or D) prior to the operator switch over to hot leg injection. State whether the MHI analysis considers this and if not, provide justification why it does not.

ANSWER:

The inadvertent (I&C failure) opening of a hot leg injection isolation valve (SIS-MOV-014A, B, C, and D) is considered in the design although it is not described in the FMEA. DCD Subsection 6.3, Table 6.3-6, "Failure Modes and Effects Analysis - Safety Injection System" will be revised with addition of this failure as the item 13.

Impact on DCD

The item 13, I&C for hot leg injection isolation valve control will be added to Table 6.3-1 as follows:

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
13. I &C for hot leg injection isolation valve SIS-MOV-014A Control (B, C and D analogous)	Failure to deliver fluid due to one SI pump run out caused by Inadvertent open off demand	Small-break LOCA (not DVI LOCA)	No effect on plant safety because three, 50% SI pumps remain and only two are required.	Valve position indication in MCR and RSC. SI pump operating information in the MCR and RSC includes flow, suction and discharge pressure, pump motor current, and RUN indication for each pump.	
		Small-break LOCA (DVI LOCA)	No effect on plant safety because three SI pumps remain and only One SI pump spills and only one is required.		
		Large-break LOCA	No effect on plant safety because three, 50% SI pumps remain and only two are required.		
		Non-LOCA	No effect on plant safety because three, 50% SI pumps remain and only two pumps are required.		
		Safe shutdown; provide emergency boration and preserve RCS inventory	No effect on plant safety because three, 50% SI pumps remain and only two are required.		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
13. I &C for hot leg injection isolation valve SIS-MOV-014A Control (B, C and D analogous) (cont.)	Failure to deliver fluid due to one SI pump run out cased by inadvertent open off demand with one SI train out of service	Small-break LOCA (not DVI LOCA)	No effect on plant safety because two, 50% SI pumps remain and only two are required.	Valve position indication in MCR and RSC. SI pump operating information in the MCR and RSC includes flow, suction and discharge pressure, pump motor current, and RUN indication for each pump.	
		Small-break LOCA (DVI LOCA)	No effect on safety because two SI pumps remain. One SI pump spills and only one is required.		
		Large-break LOCA	No effect on safety because two, 50% SI pumps remain and two pumps are required.		
		Non-LOCA	No effect on safety because two, 50% SI pumps remain and two are required.		
		Safe shutdown; provide emergency boration and preserve RCS inventory	No effect on safety because two, 50% SI pumps remain and two are required.		

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI 6.3.2.6-1

Does the SIS design consider protection from fires? If so, identify the appropriate DCD section(s) which address SIS fire protection. If not, how does the design assure adequate SI availability should a fire occur?

ANSWER:

The SIS design considers protection from fires in accordance with the fire protection requirements. The fire protection is described in the DCD Chapter 9, Section 9.5 "Other Auxiliary Systems," Subsection 9.5.1 "Fire Protection Program" and Appendix 9A "Fire Hazard Analysis."

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.

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QUESTION NO.: 06.03-51

RAI 6.3.2.8-1

Hot leg injection and emergency letdown operations are stated to be taken from the MCR. Can these operations be performed from the RSC? If so, then Table 6.3-6 should be revised accordingly. If not, explain how these operations can be taken when the MCR is uninhabitable.

ANSWER:

Hot leg injection and emergency letdown operations can also be performed from the RSC. Table 6.3-6 "Failure Detection method" will be revised. In addition, other items (All pumps and valves) in Table 6.3-6 will be also revised.

Impact on DCD

Table 6.3-6 will be revised as follows :

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
1. SI Pump A (B, C, and D analogous)	Failure to deliver flow	Small-break LOCA (not DVI LOCA)	No effect on plant safety because three, 50% SI pumps remain and only two are required.	SI pump operating information in the MCR **and RSC** includes flow, suction and discharge pressure, pump motor current, and RUN indication for each pump.	
		Small-break LOCA (DVI LOCA)	No effect on plant safety because three SI pumps remain. and only One SI pump spills and only one is required.		
		Large-break LOCA	No effect on plant safety because three, 50% SI pumps remain and only two are required.		
		Non-LOCA	No effect on plant safety because three, 50% SI pumps remain and only two pumps are required.		
		Safe shutdown; provide emergency boration and preserve RCS inventory	No effect on plant safety because three, 50% SI pumps remain and only two are required.		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
1. SI Pump A (B, C, and D analogous) (cont.)	Failure to deliver flow with one SI train out of service	Small-break LOCA (not DVI LOCA)	No effect on plant safety because two, 50% SI pumps remain and only two are required.	SI pump operating information in the MCR **and RSC** includes flow, suction and discharge pressure, pump motor current, and RUN indication for each pump.	
		Small-break LOCA (DVI LOCA	No effect on safety because two SI pumps remain. One SI pump spills and only one is required.		
		Large-break LOCA	No effect on safety because two, 50% SI pumps remain and two pumps are required.		
		Non-LOCA	No effect on safety because two, 50% SI pumps remain and two are required.		
		Safe shutdown; provide emergency boration and preserve RCS inventory	No effect on safety because two, 50% SI pumps remain and two are required.		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
2. Direct vessel safety injection line isolation valve SIS-MOV-011A (SIS-MOV-011B, C and D analogous)	Failure to throttle on demand	Safe shutdown	No effect on plant safety because associated SI pump A can be stopped. Three SI trains remain and only two are required.	Valve position indication in MCR **and RSC.**	
	Failure to close on demand	LOCA; re-align two SI pumps to hot leg injection	No effect on plant safety because remaining two SI trains can realign and only one is required.		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
2. Direct vessel safety injection line isolation valve SIS-MOV-011A (SIS-MOV-011B, C and D analogous) (cont.)	Failure to throttle on demand with one SI train out of service	Safe shutdown	No effect on plant safety because SI pump A can be stopped. Two SI trains remain and two are required.	Valve position indication in MCR **and RSC**.	
	Failure to close on demand with one SI train out of service	LOCA; re-align one SI pump to hot leg injection	No effect on plant safety because remaining one SI train can realign and only one is required.		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
3. Hot leg injection isolation valve SIS-MOV-014A (B, C and D analogous)	Failure to open on demand	LOCA; re-align two SI trains to hot leg injection	Failure prevents use of SI train A for hot leg injection. No effect on plant safety because the remaining two SI trains can realign and only one is required (two normally used).	Valve position indication in MCR **and RSC**.	
	Failure to open on demand while one SI train is out of service	LOCA; realign one SI train to hot leg injection	Failure prevents use of SI train A for hot leg injection. No effect on plant safety because one SI train can realign and only one is required.		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
4. Accumulator discharge valve SIS-MOV-101A (SIS-MOV-101B, C and D analogous)	Failure to close on demand	Safe shutdown; isolate accumulator A from the RCS prior to depressurization to prevent introducing nitrogen into RCS	No effect on plant safety because the accumulator nitrogen gas volume can be vented by opening the accumulator nitrogen discharge valve, and atmospheric vent SIS-MOV-121A or B.	Valve position indication in MCR **<u>and RSC</u>**.	Valves SIS-MOV-121A and B are parallel vents to the atmosphere and are powered from different Class 1E supplies.
	Failure to close on demand with a Class 1E supply out of service		No effect on plant safety because the accumulator nitrogen gas volume can be vented by opening the accumulator nitrogen discharge valve, and atmospheric vent SIS-MOV-121A or B.		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
5. Accumulator nitrogen discharge valve SIS-MOV-121A (SIS-MOV-121B analogous)	Failure to open on demand	Safe shutdown; vent accumulator A, B, C, or D of nitrogen prior to RCS depressurization	No effect on plant safety because the common nitrogen vents to atmosphere valves SIS-MOV-121A and B are connected in parallel; only one valve is needed to vent the nitrogen from accumulators.	Valve position indication in MCR **and** **RSC**.	Valve SIS-MOV-101A and B can be on both electrical train A and B. Valve SIS-MOV-101C and D can be on electrical train C and D. Therefore, if one electrical train is out of service, Valve SIS-MOV-101A can be closed.
	Failure to open on demand with one Class-1E electrical supply out of service		No effect on plant safety because valve SIS-MOV-101A can be closed (power from alternate Class-1E supply).		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
6. Accumulator nitrogen supply line isolation valve SIS-MOV-125A (SIS-MOV-125B, C and D analogous)	Failure to open on demand	Safe shutdown; vent accumulator nitrogen prior to RCS depressurization	No effect on plant safety because the accumulator nitrogen is not normally vented in safe shutdown. (Accumulator discharge valve SIS-MOV-101A is closed on RCS depressurization to prevent introducing nitrogen into the RCS on shutdown. See Item 4 above).	Valve position indication in MCR **and RSC**.	Valve SIS-MOV-101A and B, and valve SIS-MOV-125C and D can be on both electrical train A and B. Valve SIS-MOV-101C and D, and valve SIS-MOV-125A and B can be on both electrical train C and D.
	Failure to open on demand with one electrical supply out of service		No effect on plant safety because the accumulator nitrogen is not normally vented in safe shutdown. (Accumulator discharge valve SIS-MOV-101A is closed on RCS depressurization to prevent introducing nitrogen into the RCS on shutdown. See Item 4 above).		

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
7. Emergency letdown line isolation valves SIS-MOV-031B and SIS-MOV-032B (SIS-MOV-031D and SIS-MOV-032D analogous)	Failure to open on demand	Safe shutdown; emergency letdown (RWSP feed and bleed)	No effect on plant safety because redundant emergency letdown from the RCS loop D is available and adequate for safe shutdown.	Open/close position indication MCR **and RSC.**	Four emergency letdown isolation valves are on different dc power electrical trains. On line mainte-nanc e of dc power electrical train is prohibited.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI 6.3.2.8 -2

Table 6.3-6 identifies manual operation of the accumulator nitrogen discharge valve for the safe shutdown plant condition should the accumulator discharge valve fail. This action should be included in DCD Section 6.3.2.8, "Manual Actions."

ANSWER:

The manual operation of the accumulator nitrogen discharge valve for the safe shutdown plant condition in case of the accumulator discharge valve fail will be added to DCD Subsection 6.3.2.8, "Manual Actions."

Impact on DCD

Subsection 6.3.2.8 will be revised with addition of the following paragraph under 2nd paragraph:

During safe shutdown, operator closes remotely the accumulator discharge valves by the operator's manual action before the RCS pressure decreases to the accumulator operating pressure in order to prevent the discharge of nitrogen from accumulators to the RCS. If the accumulator discharge valve could not be closed due to a single failure, operator opens remotely the accumulator nitrogen supply line isolation valve and the accumulator nitrogen discharge valve by the operator's manual action, and discharges the nitrogen in the accumulator to containment atmosphere and depressurizes the accumulator.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI 6.3.3.3-1

The DCD states: "The ECCS is designed with redundancy so that the specified safety functions are performed assuming a single failure of an active component for a shortterm following an accident, and assuming either a single failure of an active component or a single failure of a passive component for a long-term following an accident."
However in DCD Section 6.3.2.5, "System Reliability," it is stated "During long term cooling, the most limiting active failure, or a single passive failure, equal to the leakage that would occur from a valve or pump seal failure, may occur." Does the description in Section 6.3.2.5 mean the two failure considerations are either (1) a limiting active failure with up to total loss of ECCS fluid or (2) a limited passive failure resulting in limited loss (leakage) of ECCS fluid?

ANSWER:

The description in Section 6.3.2.5 means the two failure considerations are either (1) a limiting active failure with up to loss of one train of ECCS function to inject water to the core, or (2) a limiting passive failure with ECCS leakage resulting in the loss of one train of the ECCS function to inject water to the core (The total loss of ECCS fluid is prevented by isolating the leaked train from RWSP. The isolated ECCS train loses the function to inject water to the core).

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.

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RAI 6.3.3.1-1

Address the following discrepancies between the information provided in Table 15.6.5-1, "US-APWR Major Plant Parameter Inputs Used in the Best-Estimate Large break LOCA Analysis," and TS 3.5.1, "Accumulators":

1. There is no TS SR for the accumulator water temperature range used in the LBLOCA analysis (70°F ≤ T_{ACC} ≤ 120 °F)
2. The accumulator pressure range used in the LBLOCA analysis is 600psia ≤ P_{ACC} ≤ 710 psia, while the TS SR 3.5.1.3 range is ≥ 586 psig and ≤ 695 psig.
3. The accumulator water volume range used in the LBLOCA analysis is 2126ft^3 ≤ V_{ACC} ≤ 2179 ft^3, while the TS SR 3.5.1.2 range is ≥ 19,300 gallons and ≤ 19,700 gallons.

ANSWER:

It is assumed that the accumulator water temperature range (70°F ≤ T_{ACC} ≤ 120 °F) used in the LBLOCA analysis is the same as the containment temperature range, because the accumulators are located inside the containment vessel. In addition, there is no TS SR for the accumulator water temperature range, which is the same as that described in NUREG-1431, Rev. 3.1, and no equipment is needed to control the accumulator water temperature.

The accumulator pressure range used in the LBLOCA analysis (600psia ≤ P_{ACC} ≤ 710 psia) is the range whose values are converted into absolute pressure, and then rounded for the TS SR 3.5.1.3 range (586psig ≤ P_{ACC} ≤ 695 psig). Therefore, the accumulator pressure range used in the LBLOCA analysis corresponds to the TS SR 3.5.1.3 range.

Regarding accumulator water volume, this question was already answered in the response to RAI 135-1818, 16-49 (Ref. 1).

Reference

1. UAP-HF-09031, "MHI's Responses to US-APWR DCD RAI No. 135-1818 Revision 0", February 4, 2009

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 391-2974**
SRP SECTION:	**06.03 – EMERGENCY CORE COOLING SYSTEM**
APPLICATION SECTION:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-55

RAI 6.3.3.1-2

Address the following discrepancy between the information provided in Table 15.6.5-1, "US-APWR Major Plant Parameter Inputs Used in the Best-Estimate Large break LOCA Analysis," and TS 3.5.4, "Refueling Water Storage Pit (RWSP)": The safety injection temperature range used in the LBLOCA analysis is 45 °F ≤ T_{SI} ≤ 120 °F, while the TS SR 3.5.4.1 range is ≥ 32 °F and ≤ 120 °F.

ANSWER:

The applicable range of the thermodynamic properties for WCOBRA/TRAC (M1.0) used in the LBLOCA analysis is ≥ 280 K (Ref. 1). Therefore, the LBLOCA analysis assumes that the range of the safety injection water temperature is 45 °F ≤ T_{SI} ≤ 120 °F.

Reference

1 T. Suzuta, *et al.*, "Large Break LOCA Code Applicability Report for US-APWR," MUAP-07011-P (R0), July 2007.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 391-2974**
SRP SECTION:	**06.03 – EMERGENCY CORE COOLING SYSTEM**
APPLICATION SECTION:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-56

RAI 6.3.3.1-3

Justify using a core power less than 100% of rated power (98% ≤ Pcore ≤ 102% of 4451MWt, Table 15.6.5-1).

ANSWER:

In the LBLOCA analyses for the US-APWR, the assumed core power is 100% of rated power with measurement uncertainty. As the statistical methodology has been applied in the LBLOCA analysis, the rated power is one of the statistical parameters and sampled randomly from the uncertainty range. [

]

The core power of less than 100% is used as the initial condition in a part of the statistical calculations. However, a 95% probability level in PCT, LMO and CWO values is achievable, which is conservative. This is because the probability of core powers higher that 100% of rated power is calculated conservatively by [ ]

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 391-2974**
SRP SECTION:	**06.03 – EMERGENCY CORE COOLING SYSTEM**
APPLICATION SECTION:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-57

RAI 6.3.4.1-1

Identify, by number, those tests in DCD Section 14.2, "Initial Plant Test Program," that specifically address ECCS performance, for example, "14.2.12.1.57 - Safety Injection Accumulator Test."

ANSWER:

DCD Chapter 6, Subsection 6.3.4.1, "ECCS Performance Tests," the 3rd. paragraph will be revised with addition of the tests and numbers in DCD Section 14.2, "Initial Plant Test Program."

Impact on DCD

Subsection 6.4.2.1, the 3rd. paragraph will be revised as follows:

Pre-operational tests first provide assurance that individual components are properly installed and connected, and then demonstrate that system design specifications are satisfied. Pre-operational testing demonstrates that limited interface requirements for support systems are satisfied. Formal review and approval of pre-operational test results (the "pre-operational plateau") are performed prior to the initial fuel loading and criticality. The pre-operational test program for the ECCS is described in **following Subsections,** Chapter 14 ~~Subsection 14.2.12.1~~ **:**

- **14.2.12.1.54 Safety Injection System (SIS) Preoperational Test**
- **14.2.12.1.55 ECCS Actuation and Containment Isolation Logic Preoperational Test**
- **14.2.12.1.56 Safety Injection Check Valve Preoperational Test**
- **14.2.12.1.57 Safety Injection Accumulator Test**

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

7/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 391-2974**
SRP SECTION:	**06.03 – EMERGENCY CORE COOLING SYSTEM**
APPLICATION SECTION:	**6.3**
DATE OF RAI ISSUE:	**6/15/2009**

QUESTION NO.: 06.03-58

RAI 6.3.5.3-1

Operator actions may be required to protect the SI pumps. The MCR and RSC should have alarms that indicate unacceptable parameters such as high bearing oil, motor winding, or motor air temperatures. Are such alarms available? If not, explain how it can be assured that two SI pumps trains remain available (when one pump is out of service) if a support system degrades or fails, as described in DCD Section 6.3.3.3, "Single Failure Considerations," which states (in part) "assuming a single failure of an active component for a short-term following an accident."

ANSWER:

The following alarms which indicate unacceptable parameters of the SI pump and motor are provided in the MCR and RSC:

- Pump bearing temperature- High
- Pump bearing oil pressure- Low
- Motor stator temperature- High
- Motor cooling air temperature- High

DCD Chapter 6, Subsection 6.3.5.3, "Safety Injection Pumps" will be revised with addition of the description of these alarms.

Impact on DCD

DCD Subsection 6.3.5.3 will be revised with the addition of the following as the last paragraph :

The following alarms which indicate unacceptable parameters of the SI pump and motor are provided in the MCR and RSC:

- **Pump bearing temperature- High**
- **Pump bearing oil pressure- Low**
- **Motor stator temperature- High**
- **Motor cooling air temperature- High**

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.