MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

August 28, 2008

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco,

Docket No. 52-021
MHI Ref: UAP-HF-08154

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

References: 1) "Request for Additional Information No. 39 Revision 0, SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19," dated July 29, 2008.

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

Enclosed are the responses to the RAIs that are contained within Reference 1.

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 responses to the RAIs (Enclosure 2) and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all information in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

Sincerely,

Y. Ogata

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

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Enclosures:

1. Affidavit of Yoshiki Ogata

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

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

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

Contact Information

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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-08154

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

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

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Responses to Request for Additional Information No.38 Revision 0" dated August, 2008, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

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

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

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

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

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

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

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

Executed on this 28th day of August 2008.

Y. Ogata

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

Enclosure 3

UAP-HF-08154
Docket Number 52-021

Responses to Request for Additional Information No.39 Revision 0

August, 2008
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19.1**

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

QUESTION NO. : 19-44

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

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ANSWER:

This question will be answered later, within 60 days after RAI issue date.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19.1**

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

QUESTION NO. : 19-45

(Follow-up to Question 19-6) Additional information is needed on reflux cooling during shutdown in the US-APWR. Specifically: (a.) Provide a description and results of design-specific analyses demonstrating the effectiveness of reflux cooling in the US-APWR at the RCS levels assumed in the plant operating states (POS) that credit the steam generators for heat removal. Include the calculated pressures and temperatures. NUREG-1410, cited in response to Question 19-6, showed different responses at different RCS levels. (b.) Discuss the impact of the time delay, temperature and pressure increase, and any associated mode changes on subsequent plant response. (c.) Provide the input parameters (including temperature, pressure, and decay heat load) used to calculate operator action timing after losses of residual heat removal (RHR) in all POS. (d.) Provide the assumed contents of the steam generator tubes during all phases of plant shutdown. Will nitrogen be injected in the steam generator tubes to speed draining? If so, how does the nitrogen content affect the steam condensing surface for reflux cooling and any subsequent repressurization?

ANSWER:

This question will be answered later, within 105 days after RAI issue date.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-46

Clarify the success criteria for RHR in all POS (both in the initiating event assessment and as a mitigating system). Provide a description and results of the calculations performed to justify these success criteria. If the number of trains required is different from the numbers used to support system analyses and/or development of technical specifications (TS), state why.

ANSWER:

TS requires at least two RHR trains to be operating, to assure sufficient capacity to transit from initial stage of RHR operation to cold shutdown. On the other hand, the PRA considers mitigation functions to be successful as long as systems can perform its minimum required function to prevent core uncover. The success criteria based on this criteria was determined by simple static evaluation and engineering judgment. Specifically, from simple static evaluation, time after plant trip where decay heat removal from RCS by one train of the RHR, keeping the hot-leg temperature below 100 °C, can be achieved was estimated to be 36 hours. Success criteria for RHR during POS 8-1 are therefore set to be "one train operating" (both in the initiating event assessment and as a mitigating system). This evaluation indicates that during other POS such as POS 3 and some portion POS 4, singular RHR train cannot remove decay heat keeping the hot-leg temperature below 100 °C. Even though boiling in RCS may occur under such circumstances, RHR will continuously remove decay heat until RHR function degrades. Decay heat will gradually decrease during this period. (It should also be noted that since boiling will not actually occur at 100 °C due to increase in RPS pressure, this evaluation is conservative.) Risk evaluation for POS other than POS 8-1 are based on simplified evaluation and detailed success criteria analysis were not performed for these POS. Success criteria for POS 8-1 were judged to be applicable to other POS.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-47

(Follow-up to Question 19-20) Provide additional information on how the sensitivity study included in the response to Question 19-20 was performed.

(a) Does the stated CDF include contributions from all POS?

(b) Were operator actions related to systems without TS requirements assumed to fail?

(c) Were operator actions depending on sensors and indication without TS requirements assumed to fail?

(d) Was automatic isolation of the low pressure letdown line on low level assumed to fail?

(e) Why does Table 19-20-1 indicate that the charging pumps, the refueling water storage pit (RWSP), and the refueling water storage auxiliary tank (RWSAT) are available? There are no TS requiring these components to be available during MODES 5 and 6. If changes to the sensitivity case are made, provide updated results for all POS after considering any PRA changes resulting from other question responses.

ANSWER:

(a) Yes, CDF stated in response to Question 19-20 includes contributions from all POSs.

(b) Operator actions related to systems without TS requirements were basically assumed to fail. The only exception is operator actions to perform, which is described in item (e) of this response.

(c) Yes, operator actions depending on sensors and indication without TS requirements were assumed to fail.

(d) No, automatic isolation of the low pressure letdown line on low level was assumed to be available in the analysis performed for response to Question 19-20.

(e) Even though there are no TS requirements for charging pump, RWSP and RWSAT, these components are assumed to be available in the analysis performed for response to Question 19-20,. The charging pump will be running for purification during normal shutdown and the RWSP is considered to have enough water to fill the refueling cavity during normal shutdown, except for cavity flooding state. The RWSAT is assumed available during shutdown since it is used to fill the cask. There is a possibility that the water volume in the RWSAT is not sufficient to supply water to the RCS via charging pumps. However, RWSAT can be used as a water source for charging injection because water can be supplied to the RWSAT from the RWSP. Therefore, we judge the assumptions applied to the sensitivity analysis performed for Question 19-20 to be reasonable.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-48

TS 3.9.5 and 3.9.6 for RHR during MODE 6 add a new note to the language of the standard technical specifications (STS) in NUREG-1431. Note 2 in TS 3.9.5 states that "[o]ne RHR pump operation is permitted, provided that decay heat is sufficiently small." Note 3 in TS 3.9.6 states that "[o]ne or two RHR loops operation is permitted, provided that decay heat is sufficiently small." The term "sufficiently small" is not defined in either TS or their bases. Define "sufficiently small." How are the operators expected to determine that this condition is met? Revise the TS to provide the operators with a clear understanding of the requirements for RHR during refueling.

ANSWER:

As pointed out by the staff, these notes are not clear for operators and are difficult to define the term "sufficiently small". These notes in TS 3.9.5 and 3.9.6 will be deleted. This amendment will be incorporated in chapter 16 of DCD rev.1.

Impact on DCD

TS 3.9.5 in Chapter 16 will be amended as follows in revision 1 of the DCD.

---NOTE--

~~2. One RHR pump operation is permitted, provided that decay heat is sufficiently small.~~

Also, this revision impacts revision 1 of the DCD in TS 3.9.6 with the following corrections:

---NOTE--

~~3. One RHR pump operation is permitted, provided that decay heat is sufficiently small.~~

Impact on COLA

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

Impact on PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-49

(Follow-up to Question 19-7) Discuss how the initiating events, mitigating systems, and operator actions that are not modeled in POS 8-1, but are modeled in other POS, have been considered in the development of PRA-based insights and input to other programs such as the reliability assurance program (RAP), TS, and human factors engineering (HFE).

ANSWER:

"Loss of RHR Caused by Failing to Maintain Water Level (FLML)" is an initiating event not model in POS 8-1 but modeled in other POS. Frequency of this initiating event is calculated considering the plant configuration during the POS, and sequences of this initiating event is similar to "loss of RHR due to over-drain" modeled in POS 8-1. Therefore, we judge that insights such as importance of this initiating event can be derived by the simplified evaluation applied to other POS.

There are two mitigation systems that are not modeled in POS 8-1, but modeled in other POS. The first one is a "SG: Decay heat removed from the RCS via SGs", and the second one is a "GI: Gravitational injection". Components related to these mitigation systems are treated in the RAP as risk important components. (Refer to Table 17.4-1 of DCD rev.1)

Operator actions necessary to perform the above two mitigation systems were considered as important human action. (Refer to the response of RAI 19-2 and 19-7) These insights have been provided to the HFE.

The results of other POS were input to chapter 16. There is no particular sections in the TS that have been changed base on insights from the risk evaluation.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-50

(Follow-up to Question 19-10) Provide a more detailed response to Question 19-10. The statement "[t]he effects of the flood barriers are considered only separation barriers between the east side and the west side of the reactor building" is unclear. Page 19.1-87 of the Design Control Document (DCD) states that the east and west side of the reactor building are physically separated by barriers such as watertight doors and that propagation between the two sides is not considered. Are these barriers maintained during shutdown? If so, how are they controlled? Clarify what other flood barriers credited in the at-power analysis could be removed during shutdown and how these barriers will be controlled. If barriers are not controlled during shutdown, revise the shutdown flooding PRA to reflect the propagation paths.

ANSWER:

The flood barriers that separate the reactor building between east side and west side are important to safety for the operation of the facility. These doors are monitored and controlled during plant operation and maintenance.

Administrative procedures and maintenance procedures will be in place to assure that the doors are maintained closed during plant operation shutdown. When work is on-going and the doors are opened, a watch will be posted to assure that they are closed if needed.

Administrative procedures and maintenance procedures are a COL issues described in COL 13.5(1) and COL 13.5(7) in the DCD Chapter 13 subsection 13.5.

COL 13.5(1) The COL Applicant is to develop administrative procedures describing administrative controls over activities that are important to safety for the operation of a facility.

COL 13.5(7) The COL Applicant is to describe the classifications of maintenance and other operating procedures, the operating organization group or groups responsible for following each class of procedure, and the general objectives and character of each class and subclass.

Control of water tight doors during shutdown by procedures will be address as key assumptions in the DCD Chapter 19.

Impact on DCD

DCD will be revised to address the information discussed for this RAI.

Impact on COLA

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

Impact on PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-51

(Follow-up to Question 19-18) The response to Question 19-18 states that safety-related trains are separated by fire-resistant walls or doors. Are these barriers maintained during shutdown? If so, how are they controlled? Clarify what other fire barriers credited in the at-power analysis could be removed during shutdown and how these barriers will be controlled. If barriers are not controlled during shutdown, revise the shutdown fire PRA to reflect the propagation paths.

ANSWER:

RG1.189 1.5 specifies that the Licensee may implement compensatory measures for degraded or nonconforming conditions of fire protection features as follows.

"Temporary changes to specific fire protection features that may be necessary to accomplish maintenance or modifications are acceptable, provided interim compensatory measures, such as fire watches, temporary fire barriers, or backup suppression capability, are implemented. For common types of deficiencies, the technical specifications or the NRC-approved FPP generally note the specific compensatory measures. For unique situations or for measures that the approved FPP does not include, the licensee may determine appropriate compensatory measures (............) A licensee may opt to implement an alternative compensatory measure stated in its FPP, or combination of measures."

In US-APWR, every fire protection feature will be controlled, conforming to Fire Protection Program (FPP) which will be established based on the requirements of the regulatory guide in COLA-stage. Consequently, the probability of fire protection features function to be damaged will be very low during shutdown condition as well as operating condition.

In shutdown fire PRA of US-APWR, evaluation of fire propagation path has been performed based on conservative assumptions. That is, fire barriers such as fire doors, fire dampers and so forth installed in the fire compartment that may be under maintenance outage is assumed to be open during maintenance work. Result of the shutdown fire PRA is therefore applicable.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-52

Revise DCD Sections 19.1.6.3.2, internal fire at low power and shutdown (LPSD), and 19.1.6.3.3, internal flood at LPSD, to include the information discussed on page 19.0-15 of Standard Review Plan (SRP) Section 19.0. For each POS, provide:

- Mean core damage frequency (CDF) and large release frequency (LRF)
- Significant core damage and large release sequences
- Significant initiating events
- Significant functions; structures, systems, and components (SSC), and operator actions
- PRA assumptions and PRA-based insights
- Results and insights from importance, sensitivity, and uncertainty analyses

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ANSWER:

The required additional information will be involved in the next revision of DCD.

PRA report revision 1 would be issued due to the changes of plant layout. Internal fire PRA and internal flood PRA at LPSD has been performed using the internal events PRA model at LPSD. These evaluations have been performed focusing on the representative POS that is POS 8-1. Also simplified evaluations have been performed for other POSs.

The following additional information for Internal fire PRA and internal flood PRA at LPSD would be involved in the PRA report revision 1 referring "Chapter 20 Internal Events Low-Power and Shutdown Risk Assessment" of the PRA report.

- PRA assumptions

- Mean core damage frequencies (CDF). (Large release frequency (LRF) is assumed as same as CDF) for all POSs
- Initiating events frequencies for each POS
- Insights from the PRA at LPSD
- Dominant scenarios (POS8-1)
- Dominant cutsets (POS8-1)
- Results of importance analysis (Fussell-Vesely Importance (FV) and risk achievement worth (RAW)) (POS8-1)
- Results of uncertainty analysis (POS8-1)
- Results of sensitivity analysis (POS8-1)

Followings are parts of additional information in the PRA report revision 1 for internal fire PRA and internal flood PRA at LPSD.

1. Internal Fire at LPSD

The total CDF of internal fire at LPSD is 4.8E-08/RY and CDF of each initiating event and each POS is shown in Table 1-1. Internal fire risk is not significant for US-APWR.

Table 1-1 Core damage frequency of each initiating event and each POS of internal fire at LPSD

The CDF of POS 8-1 of internal fire at LPSD is 1.9E-08/RY. Also the results of uncertainty analysis of POS 8-1 are

95%th percentile	6.3E-08/RY
Mean	1.8E-08/RY
Median	6.7E-09/RY
5%th percentile	1.5E-09/RY

Attached Table 1-2 to Table 1-5 are examples of the dominant scenarios, dominant cutsets and importance of internal fire at LPSD.

Table 1-2 Dominant scenarios of internal fire at LPSD (POS 8-1)

Table 1-3 Examples of dominant cutsets of internal fire at LPSD (POS 8-1)

Table 1-4 Examples of all basic events (hardware and human error) FV Importance of internal fire at LPSD (POS 8-1)

Table 1-5 Examples of all basic events (hardware and human error) RAW Importance of internal fire at LPSD (POS 8-1)

2. Internal Flood at LPSD

The total CDF of internal flood at LPSD is 5.7E-08/RY and CDF of each initiating event and each POS is shown in Table 2-1. Internal flood risk is not significant for US-APWR.

Table 2-1 Core damage frequency of each initiating event and each POS of internal flood at LPSD

The CDF of POS 8-1 of internal flood at LPSD is 1.8E-08/RY. Also the results of uncertainty analysis of POS 8-1 are

95%th percentile	6.8E-08/RY
Mean	1.8E-08/RY
Median	7.6E-09/RY
5%th percentile	4.2E-10/RY

Attached Table 2-2 to Table 2-5 are examples of the dominant scenarios, dominant cutsets and importance of internal flood at LPSD.

Table 2-2 Dominant scenarios of Internal flood at LPSD (POS 8-1)

Table 2-3 Examples of dominant cutsets of internal flood at LPSD (POS 8-1)

Table 2-4 Examples of all basic events (hardware and human error) FV Importance of internal flood at LPSD (POS 8-1)

Table 2-5 Examples of all basic events (hardware and human error) RAW Importance of internal flood at LPSD (POS 8-1)

Above additional information and more detail descriptions will be involved in "Chapter 23 Internal Fire Risk Assessment" and in "Chapter 22 Internal Flood Risk Assessment" of PRA Report Revision 1 (MUAP-07030(R1).

Impact on DCD

The required additional information will be involved in the next revision of DCD.

Impact on COLA

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

Impact on PRA

PRA technical report will be revised.

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-53

(Follow-up to Question 19-11) The response to Question 19-11 states that the failure rate of the suction strainers is unchanged from the at-power model. State the additional assumptions (e.g., related to containment cleanliness and foreign materials exclusion programs) that support this key assumption. Revise the table of risk insights in Chapter 19 of the DCD to include the assumptions and their dispositions.

ANSWER:

Regarding the assumption in the response to Question 19-11, a foreign materials exclusion program is stated in DCD chapter 6.2.2.3 as shown below;

"Preparation of a foreign materials exclusion program is the responsibility of the COL Applicant. This program addresses other debris sources such as latent debris inside containment. This program minimizes foreign materials in the containment."

The table of risk insights in Chapter 19 of the DCD will be revised to include the assumptions and their dispositions.

Impact on DCD

The DCD will be revised reflecting this response to this RAI

Impact on COLA

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

Impact on PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19**

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

QUESTION NO. : 19-54

Page 19.1-29 of the DCD indicates that the human error probabilities (HEP) for type B human failure events (i.e., over-drain and human-induced loss-of-coolant accidents (LOCA) during shutdown) are taken directly from NUREG/CR-1278 and that performance shaping factors (PSF) are not considered. NUREG-1842, the comparison of human reliability analysis (HRA) methods to good practices, states that involvement of HRA specialists is a prerequisite to a valid THERP analysis, and that treatment of everything as "nominal" without justification may indicate inadequate HRA and human factors considerations in making the THERP judgments. Provide additional information on how the THERP analysis of the type B human failures was performed. Discuss the involvement of HRA practitioners and human factors specialists in the development of the US-APWR PRA.

ANSWER:

MHI has more than 20 years of experience performing PRA including HRA for PWR plants in Japan. MHI PRA engineers are very well experienced and have sufficient knowledge of plant behavior to perform HRA.

MHI agrees that the original description in the DCD is misleading as pointed by the NRC staff. The human error probabilities (HEP) are evaluated considering the performance shaping factors (PSF) in accordance with the discussion in NUREG/CR-1278, including evaluation of HEP causing an initiating event in LPSD PRA.

The evaluation conditions of PSF about stated HEP in page 19.1-29 of the DCD are explained below;

(a) Experience Level

The operator's skill level is set to "Skilled" as described in Table 20-16 of NUREG/CR-1278 assuming that the operators are sufficiently experienced.

(b) Stress Level

The operator's stress level is set to “Optimum” as described in Table 20-16 of NUREG/CR-1278 because these tasks are routine work during shutdown.

(c) Task Level

Task level is set to “Step-by-step Task” as described in Table 20-16 of NUREG/CR-1278 because these tasks consist of independent steps established in the operating procedures.

(d) Tagging Level

Tagging level is at least set to “Level 2” as described in Table 20-15 of NUREG/CR-1278 assuming each equipments properly managed to be able to easily identify by attaching an ID tag.

(e) Others;

"Human-system Interface" quality is assumed good having the following advanced design characteristics of the Main Control Board (MCB) of US-APWR;

a. Touch Operation

- Integration of monitoring and operating function
- Compact console (Full-time sit-down operation)
- Automation of high workload monitoring task

b. Improved alarm system

- Easy to distinguish important alarm by dynamic prioritization

c. Adoption of Large Display Panel

- Plant information is clearly visible to all operators and supervisors

According to Table 20-15 and Table 20-16 in NUREG/CR-1278, and assumptions (a) through (e) related to PSF, modifiers for the nominal HEPs of operator actions that cause initiating events for LPSD were estimated to be "1". Accordingly, the nominal HEP were applied to each HEP without any modifications.

MHI is aware of the limitations of THERP, and applies THERP to estimate HEP during stages where detail information of operation procedures is not available.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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Docket No.52-021

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

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

APPLICATION SECTION: **19.1**

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

QUESTION NO. : 19-55

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

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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APPLICATION SECTION: **19**

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

QUESTION NO. : 19-56

(Follow-up to Question 19-9) Provide additional information on flow diversions during shutdown. Specifically: (a.) How does the frequency of a flow diversion from the RHR system to the RWSP via motor-operated valves (MOV) 9815A/B/C/D account for the four possible valves that could cause it? (b.) How is the contribution of both spurious operation and inadvertent opening considered in the evaluation of this flow diversion? (c.) Justify the exclusion of all other failures (e.g., spurious operation or inadvertent opening of particular valves) that could result in a loss of RCS inventory inside or outside containment. For each, state how long the operator would have to respond to the flow diversion in each POS, including the flooded-cavity POS 5 and 7. (d.) Discuss how inadvertent opening of these valves has been considered in the design of the control room and in administrative controls.

ANSWER:

In the design certification phase, the maintenance schedule during shutdown has not been established for the evaluation of LOCA initiating event. Therefore LOCA due to flow diversion is conservatively assumed to occur in each POS from the RHR system to the RWSP via MOV (9815A/B/C/D). The frequency of LOCA is assumed to be 1.0E-04/ry. This frequency is estimated considering other possible scenarios including all four trains and valves as pointed by the NRC staff, although this assumption does not limit the operational administrations. It can be concluded that the risk for the US-APWR is evaluated insignificant even if the evaluation is based on conservative assumptions.

(a)

There is a possibility that a flow diversion LOCA may occur from the RHR system (running train) to the RWSP due to careless operation of MOV (9815A/B/C/D). Practically this valve operation can be expected at only two situations, i.e. draining water from the refueling cavity to the RWSP and operation of full flow examination of the RHR pump. According to this assumption, it is not necessary to consider this event in each POS. However, a single spurious operation of this MOV may result in the flow diversion LOCA, and the frequency of this spurious failure is considered to be higher than that of other scenarios. This event is therefore conservatively considered in each POS.

(b)

The contribution of the spurious operation is considered in the evaluation of this flow diversion although it is estimated very small compared to the value because of the human error. The failure probability of a spurious operation of MOV is 4.0E-8/h according to NUREG/CR-6928, and the mission time is as the POS duration time (the maximum is 55.5 hours for POS 8-1). The frequency of LOCA due to the spurious operation in POS 8-1 is evaluated as 1.1E-06/ry, considering this failure probability and this mission time. On the other hand, the value of human errors is 1.0E-04/RY. Therefore, it is concluded that the contribution of this failure mode can be disregarded in the flow diversion LOCA as it is significantly small comparing to the one due to human errors. And, regarding the inadvertent opening, two human errors are taken into account. The first assumption is a commission error, which is a spurious operation of this valve by an operator. This evaluation referred to the Table 20-12 Item 5 (turn rotary control in wrong direction) of NUREG/CR-1278. The second assumption is an omission error, which an operator forgets to close this valve. This evaluation is performed in accordance with the Table 20-7 Item 2 of NUREG/CR-1278.

(c)

As mentioned above, regarding the flow diversion LOCA, in the design certification phase, the maintenance schedule during shutdown has not been established for the evaluation of LOCA initiating event. Therefore LOCA due to flow diversion is conservatively assumed to occur in each POS from the RHR system to the RWSP via MOV (9815A/B/C/D). It is assumed that the frequency of LOCA because of this scenario is the highest among all the scenarios. And regarding the failure mode, it is explained in the response to this RAI (b). In PRA of US-APWR at the design certification phase, the cognition model of human error is described in the PRA technical report Chapter 9 section 9.3.3.2.b. The diagnosis HEP as shown in ASEP and THERP handbook for cognition failure is not used in this evaluation. The time from the occurrence of an initiating event to core uncovery is estimated more than one hour from a simplified calculation with conservative assumptions. It is therefore considered that the time is sufficient for operators to properly complete these actions.

(d)

The designed status of these valves (9815A/B/C/D) is changed to "Locked close: LC" valve in the DCD rev.1. Accordingly the possibility of a mechanical failure for spurious open is significantly reduced in the current design. In addition, practically this valve operation can be expected at only two situations, i.e. draining water from the refueling cavity to the RWSP and operation of full flow examination of the RHR pump. The LPSD PRA is performed considering such limited possibility for spurious operations

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-57

(Follow-up to Question 19-5) Page 19.1-98 of the DCD states that LOCAs caused by pipe rupture are unlikely to occur during shutdown. Provide further justification for the exclusion of LOCAs both inside and outside containment from the shutdown PRA. The response to Question 19-5 appears to indicate that pipe breaks are assumed to result in a loss of RHR and steam generator function, but pipe breaks also could cause a loss of inventory requiring mitigation.

ANSWER:

MHI agrees as pointed by the NRC staff that pipe rupture can cause loss of inventory, which may involve mitigative operations. MHI has already considered this event as one of the causes of LOCA events, and the frequency of a LOCA event due to pipe rupture in the RHRS is evaluated as 3.0E-06/ry, which was calculated considering the component failure probability and the pipe length of RHRS. Detailed result of this calculation is provided in table 20.5-1 of PRA technical report. The frequency of LOCA events considered in the PRA for operations at power is summarized in Table 19-57-1. The frequency of LOCA events for LPSD is estimated based on these frequencies for operations at power, as summarized in Table 19-57-2.

These values are apparently much smaller than 1.0E-04/ry that is the frequency of LOCA caused by human error. It is therefore considered reasonable to assume that the frequency of LOCA can be represented by human error that can result in loss of inventory.

Table 19-57-1 Summary of Frequency of Various LOCA Events for Operations at Power

	Frequency(/ry)	Frequency(/hr)
LLOCA	1.2E-06	1.4E-10
MLOCA	5.0E-04	5.7E-08
SLOCA	3.6E-04	4.1E-08

Table 19-57-2 Summary of Frequency of Various LOCA Events for POSs

POS	Duration time(hr)	LLOCA	MLOCA	SLOCA
3	2.3	3.2E-10	1.3E-07	9.5E-08
4-1	39.2	5.4E-09	2.2E-06	1.6E-06
4-2	12	1.6E-09	6.8E-07	4.9E-07
4-3	6	8.2E-10	3.4E-07	2.5E-07
8-1	55.5	7.6E-09	3.2E-06	2.3E-06
8-2	12	1.6E-09	6.8E-07	4.9E-07
8-3	11	1.5E-09	6.3E-07	4.5E-07
9	10	1.4E-09	5.7E-07	4.1E-07
11	43.5	6.0E-09	2.5E-06	1.8E-06

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-58

(Follow-up to Question 19-26) Revise the DCD to include a combined license (COL) information item or similar commitment that ensures the COL applicant will develop shutdown response guidelines that satisfy NUMARC 91-06, as stated in response to Question 19-26.

ANSWER:

The section 19.2.5 of DCD which describes the accident management will be revised reflecting this RAI. Development of accident management program is one of the COL items identified in Chapter 19, and will include a shutdown response guideline as part of the program to incorporate the discussions given in NUMARC 91-06.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-59

(Follow-up to Question 19-27) Footnote 8 in RG 1.206, Section C.I.19, Appendix A, states that: "'PRA-based insights' are those insights identified during the DC [design certification] process that ensure that assumptions made in the PRA will remain valid in the as-to-be-built, as-to-be-operated plant and include assumptions regarding SSC and operator performance and reliability, ITAAC, interface requirements, plant features, design and operational programs, and others. The usage of the phrase is intended to be consistent with its use in Table 19.59-29 of the AP600 design control document [DCD]." In the AP600 DCD, each insight receives a disposition such as a reference to another portion of the DCD, an ITAAC, or a COL information item. Question 19-27 requested that such a disposition be added to Table 19.1-113 for each shutdown entry, as well as the inclusion of additional features that reduce shutdown risk. Amend DCD Table 19.1-113 to add the requested dispositions for all entries, and include the assumptions and insights provided in response to Question 19-27.

ANSWER:

The DCD Table 19.1-113 will be revised adding the requested dispositions for all entries, and include the assumptions and insights provided in response to Question 19-27. The section 19.2.5 of DCD, which describes the accident management, will be revised reflecting this RAI. Development of accident management program is one of the COL items identified in Chapter 19, and will include a shutdown response guideline

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-60

Provide the assumed water volume in the RWSP during shutdown and the assumed water volume that is transferred to the refueling cavity. Would safety injection (SI) draw from the RWSP following a LOCA in POS 5 or 7 when the cavity is flooded? If so, clarify, with supporting drawings as needed, whether the suction point from the RWSP remains covered with water in POS 5 and 7.

ANSWER:

The water volume in the RWSP during shutdown is from 77,980 ft^3 through 81,230 ft^3, which is 96% to 100% of the total RWSP volume. The water volume that is transferred to the refueling cavity is 70,630 ft^3.

In the LPSD PRA of US-APWR, POS 5 and POS 7 are excluded from the PRA modeling because of the following reason.

The POS 5 and 7 are in a state that refueling cavity is filled with water. Since there is large inventory water in the cavity, even in case of LOCA there would be sufficient time until core exposure and accordingly operator action can be considered more reliable. Accordingly, CDF during these POSs is considered negligible.

Therefore, it is not necessary to consider the water volume in RWSP where SI can draw following a LOCA in POS 5 or 7.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-61

Page 5.4-42 of the DCD states that "[a]t this water level [0.33 feet above mid-loop], the air/water interface is at close proximity to the RHR suction nozzles located on the hot legs, and thus, reduces the possibility of air entrainment into the RHR pump suction." This statement appears to contradict itself. Clarify how an air/water interface close to the RHR suction nozzles reduces the possibility of air entrainment, or revise the DCD to correct the statement. In addition, discuss any design improvements made to the RCS and RHR system to reduce shutdown risk, such as self-venting suction lines, suction nozzle modifications, or vertically offset hot and cold legs.

ANSWER:

MHI agrees that the original description in the DCD is misleading. An intension of the sentence pointed by the NRC staff is that in general during mid-loop operation, the air/water interface is close to the RHR suction nozzles, but for the US-APWR design, the water level during mid-loop operation is kept higher than a conventional plant, so that this reduces the possibility of air entrainment into the RHR pump suction. MHI will revise the DCD to be easy understood as shown below;

5.4.7.2.3.6 Mid-loop and Drain Down Operations

[the third sentence in the last paragraph in Subsection 5.4.7.2.3.6]

At this water level, the air/water interface is at close proximity to the RHR suction nozzles located on the hot legs, ~~and thus,~~ but the higher RCS water level applied for the US-APWR design reduces the possibility of air entrainment into the RHR pump suction.

In US-APWR design, the RHR suction nozzles are located on the lower side and the diagonal 45 degrees of the main coolant pipe, which is same as conventional plant. Design improvements for mid-loop operation are another part such as reduction of mid-loop operation time, high reliability of water level instruments and an interlock for abnormal water level decrease, as described the DCD subsection 5.4.7.2.3.6.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-62

Justify the exclusion of low-temperature overpressure (LTOP) events from the shutdown PRA. Discuss the mass or energy input that would cause the RHR suction relief valves to open when the RCS is water-solid during shutdown. Provide the likelihood that the valves will stick open, and discuss how the shutdown PRA handles this scenario.

ANSWER:

Potential overpressurization transients to the RCS, while at relatively low temperatures, can be caused by either of two types of events to the RCS; i.e., mass input or heat input. Both types result in more rapid pressure changes when the RCS is water solid. For those low-temperature modes of operation when operation with a water solid pressurizer is possible, the CS/RHR pump suction relief valves provide low-temperature overpressure protection for the reactor coolant system. The detail of low-temperature overpressure protection is described in Subsection 5.2.2 in the DCD.

There is a TS requirement to LTOP system as shown below;

3.4.12 Low Temperature Overpressure Protection (LTOP) System

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

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

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

It is considered that a low temperature overpressure events are caused by spurious operation of these pumps and the loss of pressure relief function. The LTOP requirement is satisfied by shutting off the power supply to the charging and SI pumps or by closing isolation valves at the inlet and outlet of these pumps. Therefore the probability of the spurious operation of these pumps is assumed significantly small. In addition, venting is generally expected during the mid-loop operation. Even in some duration that venting is not available, the US-APWR provides various pressure relief capabilities including RHR suction relief valves, the safety depressurization valve and the depressurization valve. The accident scenario of the low temperature overpressure events can be therefore disregarded in the PRA because the frequency of these events is assumed to be significantly small due to these various design features.

Regarding the likelihood that the valves will stick open, it is not necessary to consider the valves stuck open when RCS venting is available because the possibility of pressure rise and the valves open is obviously low. If there is no vent from RCS, it is possible that valves will stick open following the spurious operation of pump and subsequent valves opening operation. This event is considered to be equivalent event of LOCA. However the frequency of this type of LOCA event caused by the spurious operation of pump and stuck open of valve is evaluated as significantly small compared to the frequency of the flow diversion LOCA caused by human errors. Therefore the LOCA event caused by the stuck open valves is assumed negligible in the LPSD PRA modeling.

<u>Impact on DCD</u>

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

<u>Impact on COLA</u>

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

<u>Impact on PRA</u>

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

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QUESTION NO. : 19-63

Page 19.1-97 of the DCD states that "[r]eactivity insertion event will progress phenomena very slowly by boron dilution and long grace periods so that this event has enough time to recovery." Provide further justification for the exclusion of boron dilution events from the shutdown PRA, including quantitative discussion of the "long grace periods" and associated automatic and manual mitigating actions.

ANSWER:

The reactivity insertion event due to boron dilution has been judged to be insignificant to risk because of the following factors:

- Strict administrative controls are in place to prevent boron dilution
- Boron dilution events are highly recoverable
- The consequences of re-criticality are minor unless they continue for very long.

Background to prove these factors are described in Chapter 15.4.6.2 of DCD rev1, as shown below;

"The potential for boron dilutions during refueling (Mode 6) or during shutdown operation with no reactor coolant pumps (RCPs) running (Modes 4 & 5) does not exist due to strict administrative controls.

Boron dilution is carried out by adding a predetermined batch of pure water into the reactor coolant system. When the predetermined amount of pure water has been added, the makeup water valve is automatically closed, so that boron dilution beyond the predetermined limit will not occur. When carrying out a boron dilution, the operator performs two operations: (1) changing from the automatic makeup mode to the dilution mode and (2) operating the start switch. Dilution cannot start unless both of these steps are performed. The requirement for two distinct actions reduces the likelihood of inadvertent dilution caused by operator action. For the US-APWR, planned boron dilutions are under strict administrative control.

The CVCS design inherently limits the maximum boron dilution rate so boron dilution transients proceed relatively slowly. This slow rate, together with the alarms and trips (described above) that provide operator indication ensure that sufficient time exists so that reactivity transients can be terminated by manual action to prevent criticality or a return to criticality. The alarms and trips described above are in addition to other parameters continuously available for monitoring in the control room, such as neutron flux and RCCA bank position.

If a fault in the chemical and volume control system causes the borated water or pure water flow rate to deviate from the setpoint flow rate, the operator is warned by a flow rate high deviation alarm and the water makeup isolation valve is automatically closed to stop makeup to the reactor coolant system (Section 7.6.1.3)."

The description about boron dilution event of DCD will be replaced mentioned above.

Impact on DCD

The DCD will be revised reflecting this response to this RAI

Impact on COLA

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

Impact on PRA

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

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APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-64

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

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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APPLICATION SECTION: **19.1**

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

QUESTION NO. : 19-65

Tables 19.1-76 and 19.1-77 of the DCD indicate that the RCS is closed and the steam generators are isolated in POS 4-3 and 8-1. Clarify the vent status of the RCS in these POS. The list of expeditious actions in Generic Letter (GL) 88-17 includes a direction to "[i]mplement procedures and administrative controls that reasonably assure that all hot legs are not blocked simultaneously by nozzle dams unless a vent path is provided that is large enough to prevent pressurization of the upper plenum of the [reactor vessel]." Discuss how this condition is met during shutdown in the US-APWR.

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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APPLICATION SECTION: 19

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-66

Confirm that the reactor vessel bottom head has no penetrations that could lead to inadvertent draining of the RCS during shutdown. Discuss this design improvement in the context of shutdown risk and add it to the list of risk insights in Table 19.1-113 of the DCD.

ANSWER:

In US-APWR, the instrumentation piping are installed at upside of the RV. Therefore, the reactor vessel bottom head has no penetrations that could lead to inadvertent draining of the RCS during shutdown. This design feature will be described in Table 19.1-113 in DCD.

Impact on DCD

The DCD will be revised reflecting this response to this RAI

Impact on COLA

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

Impact on PRA

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

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APPLICATION SECTION: **19.1**

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

QUESTION NO. : 19-67

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

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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APPLICATION SECTION: **19.1**

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

QUESTION NO. : 19-68

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

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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Mitsubishi Heavy Industries

Docket No.52-021

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

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

APPLICATION SECTION: **19.1**

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

QUESTION NO. : 19-69

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

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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QUESTION NO. : 19-70

Discuss whether any gravity-driven sources of borated water (other than the SFP discussed above) are available for injection following a loss of inventory during shutdown. At operating plants, the ability to inject from the refueling water storage tank (RWST) is an important mitigation strategy during shutdown, but the RWSP in the USAPWR is below the RCS elevation. Discuss how this design feature, which enhances safety by eliminating the need for recirculation switchover following a LOCA, affects shutdown risk.

ANSWER:

For the US-APWR, only the skim of SFP water is available as the source of the gravity injection during shutdown.

As pointed by the NRC staff, in-containment RWSP, which is located below the elevation of RCS, involves both advantages and disadvantages. MHI understands there is a disadvantage of this design that the gravity injection from the RWSP is unavailable during shutdown. Therefore, as in the response to Question 19-44, the US-APWR is designed that the gravity injection of the skim of SFP water is available during shutdown. This design feature can compensate for the RWST mitigation strategy addressed in the operating plants; and overall, it is evaluated that the risk during shutdown is improved to operating plants.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-71

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

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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QUESTION NO. : 19-72

(Follow-up to Question 19-8) The response to Question 19-8 discusses the impact of modeling the charging and SI systems differently from TS requirements. Will the next update of the US-APWR PRA modify the success criteria for these systems so they match outages required by TS for LTOP? Although the impact is not large, the discrepancy is not merely a modeling assumption to be justified with a sensitivity study, but rather a condition not allowed by TS during shutdown.

ANSWER:

As commented by the NRC staff, MHI will modify the success criteria for the charging and SI systems in the next update of the US-APWR PRA so they match outages required by TS for LTOP.

Impact on DCD

The DCD will be revised reflecting this response to this RAI as this additional information is requested in order to consist with TS requirement.

Impact on COLA

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

Impact on PRA

There is an impact on PRA from this RAI, but it is very small as shown in the response to the Question 19-8.

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QUESTION NO. : 19-73

What indication of temperature, pressure, and level is available to the operators during shutdown? For each, state the type of sensor, its location in the RCS, any associated alarms and trips, and the controls that ensure the indication is available during shutdown. Discuss whether these sensors are susceptible to errors identified at current plants (e.g., errors in differential pressure caused by RCS inventory swept into the pressurizer, failures of Tygon tubing, and inaccurate hot leg temperature measurement after a loss of flow).

ANSWER:

Typical indications of temperature, pressure and level during shutdown are shown in Table 19-73.1. This table summarizes the type of sensor, location in the RCS and associated alarms.

As for RCS water level for shutdown, three types of instruments are provided in US-APWR design. The first one is narrow range water level instrument (2ch), the second one is mid range water level instrument, and the third one is wide range water level. These instruments lines are permanent equipments, so monitoring RCS water level during shutdown (e.g., mid-loop operation) does not basically use Tygon tubing.

Narrow range and mid range water level instruments measure the level between the bottom of cross over leg and pressurizer gas phase, so these instruments indicate correct water level. If errors occur in the measurement due to differential pressure caused by RCS inventory swept into the pressurizer, it can be considered that all RHR pumps are inoperable. In such a situation, water level in the core can be obtained by measuring the reactor vessel water level.

As for inaccurate hot leg temperature measurement after a loss of flow, reactor coolant hot leg temperature instruments are located in the flow path during RHR operation, so this parameter can be accurately indicated.

The section 19.2.5 of DCD which describes the accident management will be revised reflecting this RAI. Development of accident management program is one of the COL items identified in Chapter 19, and will include a shutdown response guideline as part of the program to incorporate the discussions given in NUMARC 91-06. Moreover this COL Item will include the controls that ensure the indication is available during shutdown.

Table 19-73.1 Typical indications of temperature, pressure, and level during shutdown

	Type of sensor	Location	Alarms	Remark
Reactor coolant hot leg temperature (Wide range)	Resistance Temperature Detector (RTD)	Hot leg	-	-
Reactor coolant cold leg temperature (Wide range)	RTD	Cold leg	-	-
Pressurizer pressure	Transmitter	Pressurizer	High and Low	-
Reactor coolant pressure	Transmitter	CS/RHR pump suction piping	-	-
Pressurizer water level	Transmitter	Pressurizer	High and Low	-
RCS water level (narrow)	Transmitter	Crossover leg	High, Below normal, Low and Low-low	For mid-loop operation
RCS water level (mid)	Transmitter	Crossover leg	High	For mid-loop operation
RCS water level (wide)	Transmitter	Crossover leg	High and Low	For mid-loop operation

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-74

Define "mid-loop" for the US-APWR. To what elevation will the RCS be drained to allow steam generator maintenance and nozzle dam installation? Provide the location of the RHR hot leg suction nozzle.

ANSWER:

Typical RCS water level transition during mid-loop operation is shown in Figure 19-74.1.

Mid-loop operation for the US-APWR is implemented for oxidation operation for purification of RCS water and SG drain for ECT operation. This operation involves RCS drain, SG drain, SG nozzle dam installation and hydrogen peroxide supplement. At first stage, during RCS drain operation, the RCS water level is decreased, but kept higher than the center of main coolant pipe. The second stage, the water level is kept below the upper level of main coolant pipe to communicate between air space and SG for SG drain. In this stage, if water level decreases lower than the certain threshold levels, low water level alarm or low-low water level alarm alerts the operators. Additionally, low pressure letdown line is automatically isolated by issuance of the low water level signal to prevent water level from decreasing. The third stage, after SG drain, the water level is increased and kept between upper level of main coolant pipe and lower level of SG nozzle. This water level allows installation of SG nozzle dam. In this stage, air/water interface is higher than the upper level of main coolant pipe and main coolant pipe is fully filled with water, so this operation reduces the possibility of air entrainment into the RHR pump suction. In this stage, high water level alarm or below normal water level alarm alerts the operators that water level is in abnormal condition.

The location of the RHR hot leg suction nozzle is located on the lower side of the main coolant pipe with the diagonal 45 degrees.

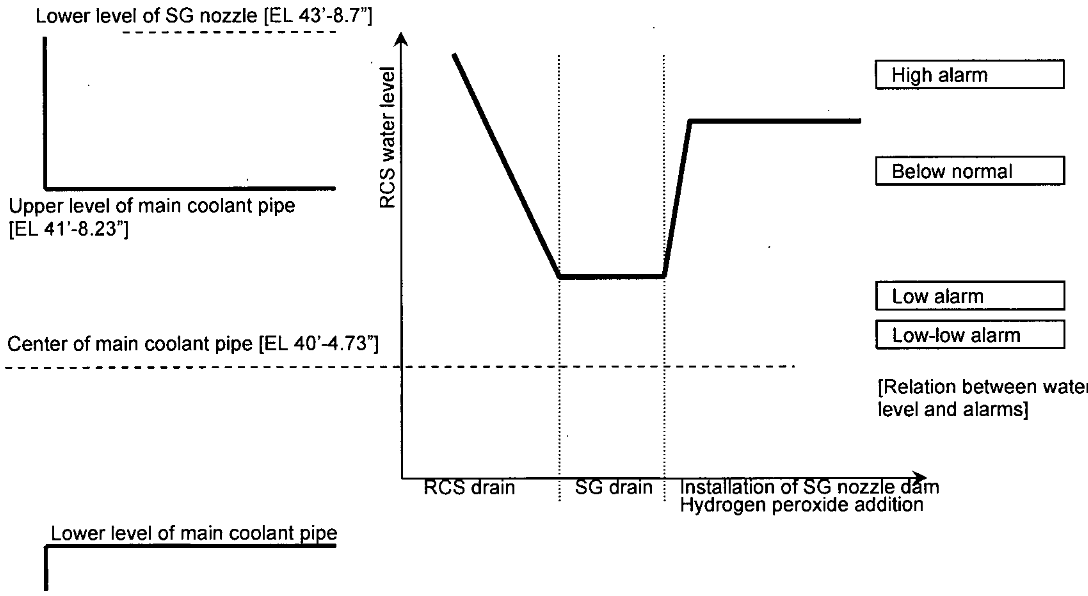

Figure 19-74.1 RCS water level transition

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-75

What is the design pressure of the nozzle dams to be used during shutdown of the USAPWR? Discuss the analysis performed to calculate this design pressure. Compare the design pressure of the nozzle dams to the expected pressure following a loss of RHR in all POS.

ANSWER:

The design pressure of the nozzle dams is 1.5 kg/cm^2 and it is designed to withstand the static head pressure postulated refueling cavity fully filled with water.

Nozzle dams will be used during POS 4-3 through 8-1. During POS 5 through 7 where the reactor vessel upper plenum is opened, RCS pressure will not increase following loss of RHR because increased coolant volume or generated steam is released outside the RCS. Accordingly, RCS pressure will not exceed the design pressure of the nozzle dams. During POS 4-3 and 8-1, there is a period of time where the reactor vessel upper plenum is closed and the RCS vent paths are opened. If loss of RHR occurs during this condition, RCS pressure may exceed the design pressure of nozzle dams after initiation of bulk boiling in the RCS. In addition, the PRA does not credit any mitigation functions that require the nozzle dams to be closed during POS 4-3 and 8-1.

Impact on DCD

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

Impact on COLA

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

Impact on PRA

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

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QUESTION NO. : 19-76

So that the staff can understand the US-APWR shutdown strategy, describe the expected shutdown sequence of events from entry into MODE 5 until the reactor cavity is flooded for refueling and during startup from the time when reactor cavity draining begins until entry into MODE 4. Describe the approach taken (e.g., tasks performed, systems and equipment used) for each step, including but not limited to: (a.) Depressurization before draining the RCS (b.) Reduction of RCS level to mid-loop (c.) Draining the steam generator tubes (d.) Level control during mid-loop (e.) Draining the refueling cavity after refueling (f.) Vacuum fill of the RCS

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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QUESTION NO. : 19-77

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

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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QUESTION NO. : 19-78

(Follow-up to Question 19-13) Table 19.1-79 in the DCD indicates that the RCS leakage test occurs between POS 9 (cold shutdown) and POS 11 (cold and hot shutdown). However, current plants' RCS leakage tests are generally performed at operating pressure and temperature, which would appear to place the plant in MODE 3. Provide further clarification to the staff on this state, specifically: (a.) Describe the general procedure for the test. (b.) Provide the temperature, pressure, and TS MODE achieved during the RCS leakage test state (POS 10). (c.) Confirm at what point in the outage the test is performed

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ANSWER:

This question will be answered later, within 60 days after RAI issue date.