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

November 25, 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-09536

**Subject: MHI's Responses to US-APWR DCD RAI No.479-3871 Revision 1**

**References:** 1) "Request for Additional Information No. 479-3871 Revision 1, SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19.1.6," dated October 26, 2009.

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. 479-3871 Revision 1".

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

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

Sincerely,

Y. Ogata

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

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No. 479-3871 Revision 1 (Proprietary Information Included)

3. Responses to Request for Additional Information No. 479-3871 Revision 1 (Proprietary Information Excluded)

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


Contact Information

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**ENCLOSURE 1**

Docket No. 52-021
MHI Ref: UAP-HF-09536

**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. 479-3871 Revision 1" dated October 2009, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

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

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

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

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

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

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

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

Executed on this 25th day of November 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09536
Docket Number 52-021

Responses to Request for Additional Information No. 479-3871
Revision 1

November, 2009
(Proprietary Information Excluded)

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**11/25/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO.479-3871 REVISION 1**

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

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

**DATE OF RAI ISSUE:** **10/26/2009**

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**QUESTION NO. : 19-402**

(Follow-up to Question 19-397) In RAI 443, Question 19-397, the staff requested a discussion of how guidance related to temperature indication in Generic Letter (GL) 88-17 has been applied to the US-APWR. The revision provided in the October 2, 2009, response addresses only temperature measurement when the residual heat removal (RHR) heat exchanger function degrades.

a. Please discuss how the US-APWR temperature sensors provide indication representative of core exit conditions during mid-loop when the head is on the vessel.

b. Please discuss how this instrumentation continues to reflect vessel temperature following a loss of RHR flow.

c. Please discuss the controls (e.g., technical specifications (TS)) that ensure this indication is continuously available.

d. Please clarify the statement in the response to Question 19-397 that “core exit temperature instruments are located in the flow path during RHR operation.” Does this statement refer to core exit thermocouples, which would reflect in-vessel temperatures?

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**ANSWER:**

<u>Answer to item a.</u>

Temperature indicators located in the reactor coolant hot legs and at the core exit provide indications representative of core exit conditions during mid-loop when the head is on the vessel. Details of the two temperature indicators are described below.

Core exit temperature
There are 39 core exit thermocouples. Thermocouples are threaded into individual guide tubes that penetrate the reactor vessel (RV) closure head through seal assemblies and terminate at the exit flow end of the fuel assemblies. All thermocouples are arranged in two safety divisions and one non-safety division; the two safety divisions are independent.

Reactor coolant hot leg temperature
There is one resistance temperature detectors (RTDs) in each hot leg.

Answer to item b.

The core exit thermocouples are located in the RV and measure the temperature at the exit flow end of the fuel assemblies. Hence, these thermocouples continue to reflect RV temperature after losses of RHR flow.

Answer to item c.

The technical specifications regarding core exit thermocouples during shutdown is limited to MODE 4 (hot shutdown) and does not cover mid-loop operation. There are no controls that ensure the continuous availability of core exit thermocouples during mid-loop operation when the head is on the vessel.

Since the availability of core exit temperature will not increase the shutdown core damage frequency, the controls to ensure the availability of sensors will not be discussed in detail in the design certification stage. Described below is the reason MHI judge that the availability of core exit thermocouples has small impact on LPSD risk.

The core exit thermocouples will be the only temperature sensors available to measure core exit temperature when RHR flow is lost. However, the loss of RHR flow can be detected and alarmed by RHR flow rate sensors. If low flow rate in the RHR loop has been detected, the operators will be aware that decay heat removal function is lost and would initiate mitigation functions. Accordingly, the unavailability of core exit thermocouples will not directly result in undetected loss of decay heat removal event. The availability of core exit thermocouples would have only a small impact on LPSD risk. In the PRA, only the RHR pump outlet pressure sensors (flow rate detector) are taken credit for the detection of loss of RHR flow.

Answer to item d.
The statement in response to Question 19-397 refer to the hot leg temperature RTDs. Responses to Questions 19-397 and 19-351 only describes the temperature indication by the hot leg temperature sensors.

The additional statement in DCD 19.1.6.1, as described in response to 19-351, will be revised as follows.

"Indications of temperature

~~As for inaccurate hot leg temperature measurement after 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.~~ Two types of instruments are provided in US-APWR design to measure RV temperature. The first one is core exit thermocouples located inside the RV. The second is resistance temperature detectors in the reactor coolant hot leg. In the event of loss of RHR flow, the core exit thermocouples can continue to measure RV temperature."

Impact on DCD

Section 19.6 will be revised. Statement regarding indication of temperature will be revised as shown in item d of this response.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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**Docket No.52-021**

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

**DATE OF RAI ISSUE:** **10/26/2009**

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**QUESTION NO. : 19-403**

(Follow-up to Question 19-395) The response to RAI 443, Question 19-395 lists three specifications that a temporary water level sensor (installed to provide accurate level indication when the reactor coolant system (RCS) is vented at a high elevation) will meet. Please document these assumed specifications in the Design Control Document (DCD) where the temporary sensor is suggested.

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**ANSWER:**

The assumed specifications of the temporary water level sensors will be documented in the DCD 19.1.6.1. The following statement will be incorporated.

In addition to the narrow range and middle range mid-loop water level sensors, a temporary water level sensor that refer pressure at the bottom of cross over leg and reactor vessel top vent is provided when the reactor coolant system (RCS) is vented at a high elevation. This sensor will satisfy the following specifications.

- Water level can be read outside the containment vessel (CV) in order to be effective during events which involve harsh environment in the CV.
- Tygon tubing monometer will not be used.
- Piping diameter will be sufficient enough to prevent delay in response

Impact on DCD

The additional statement to DCD 19.1.6.1 described in response to 19-351 will be revised as follows.

"

- Indications of water

  Three types of permanent instruments are provided in US-APWR design to measure RCS water level for shutdown. The first one is narrow range water level instrument, the second one is mid range water level and the third one is wide range water level. Narrow range and mid range water level instruments that refer pressure at the bottom of cross over leg and pressurizer gas phase are provided to measure RCS water level during midloop operation.

  In addition to the narrow range and middle range mid-loop water level sensors, a temporary water level sensor that refer pressure at the bottom of cross over leg and reactor vessel top vent is provided when the reactor coolant system (RCS) is vented at a high elevation. This sensor will satisfy the following specifications.
  - Water level can be read outside the containment vessel (CV) in order to be effective during events which involve harsh environment in the CV
  - Tygon tubing monometer will not be used
  - Instrumentation piping diameter will be sufficient enough to prevent delay in response "

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**11/25/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO.479-3871 REVISION 1**

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

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

**DATE OF RAI ISSUE:** **10/26/2009**

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**QUESTION NO. : 19-404**

(Follow-up to Question 19-392) The response to RAI 443, Question 19-392, describes several operational assumptions (e.g., frequent operability checks of instrumentation and control (I&C) systems during shutdown) that support the applicant's conclusion that I&C hardware failures are unlikely during shutdown.

a. Because these assumptions are important to achieving low shutdown risk, please revise the DCD to include them.

b. The response suggests that the risk achievement worth (RAW) values for I&C hardware common-cause failures (CCF) would be similar to those for software. However, software and sensor failures are still being incorporated in the model, as stated in the response to RAI 369, Question 19-342, so these values are not available to the staff. Please discuss whether any I&C hardware not currently included in the RAP would be expected to be important during shutdown. Please discuss how the list of risk-important equipment was modified to account for any such omissions.

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**ANSWER:**

Answer to item a.
Assumptions applied to I&C systems in the LPSD PRA will be documented in the DCD. Key assumption regarding I&C system during shutdown are as follows:

- During plant shutdown, the operability of I&C systems used for mitigation functions such as residual heat removal (RHR), charging injection, refueling water storage auxiliary tank

(RWSAT) replenishment by refueling water recirculation pump (RWRP) are frequently checked through maintenance activities and evolution of plant operating states. Local I&C equipments of these components as well as the safety logic system (SLS) can be checked and the I&C hardware are considered to be reliable during plant shutdown. Local I&C equipments of the safety injection pumps, which is a mitigation function during plant shutdown, may not be operated or tested during plant shutdown. However, the diverse actuation system (DAS) can be used to initiated safety injection when the I&C systems have failed, and therefore, signals to actuate safety injection pumps are also reliable. Manual operation of the safety injection pumps through the DAS is available during plant shutdown.

- Restoration of I&C equipments can be performed within a short period of time by exchanging the faulted card.

Answer to item b.
Failures of sensors essential for manual actions and automatic signals have been modeled in the LPSD PRA model for DCD revision 2. The sensors that are additionally modeled cover those necessary for automatic and manual actions. Failures of sensors are assumed to result in failures in cognition or detection of the event and lead to failures in operator actions and automatic signals.
Sensors modeled in the LPSD PRA that have been identified as risk important SSCs are the followings:

- Component cooling water (CCW) supply header pressure sensors
  This instrument is necessary to detect loss of CCW. Failure of this instrument leads to failure of operator actions to initiate alternate CCW that utilize the fire protection system or non-essential chilled water system.

- Residual heat removal (RHR) pump outlet flow meter (pressure sensors)
  This instrument is necessary to detect losses of RHR flow. Failure of this instrument leads to failure of operator action to promptly start the standby RHR pumps to continue RHR.

- Volume control tank (VCT) water level sensor
  This instrument is necessary to detect low water level in the VCT. Failure of this instrument leads to failure of automatic changeover of the charging pump's water source from VCT to the refueling water storage auxiliary tank (RWSAT).

- Refueling water storage auxiliary tank water level sensor
  This instrument is necessary to detect low water level in the RWSAT. Failure of this instrument leads to failure of operator actions to replenish the RWSAT by pumping the water from the refueling water storage pit (RWSP).

- Narrow range reactor coolant system (RCS) water level sensors
  This instrument is necessary to detect low water level in the RCS. Failures of these instruments lead to failure of automatic isolation of low pressure letdown line. When failures of these sensors are associated with the failure of the middle range RCS water level sensor, the operator fails to recognize low RCS water level.

- Middle range RCS water level sensor
  This instrument is used to detect low water level in the RCS. Failure of this instrument will lead to failure to recognize low water level, when associated with failures of the narrow range RCS water level sensors.

These instruments have been incorporated in the list of risk-significant SSCs for the reliability assurance program described in DCD rev.2 Chapter 17.

Impact on DCD

Last paragraph of DCD rev.2 section 19.6.1 will be revised as follows.

"

h. Various equipments will be possible temporary in the containment during LPSD operation for maintenance. However, there are few possibilities that these materials fall into the sump because the debris interceptor is installed on the sump of US-APWR. (see Chapter 6, Subsection 6.2.2) Therefore, potential plugging of the suction strainers due to debris is excluded from the PRA modeling.

i. During plant shutdown, the operability of I&C systems used for mitigation functions such as RHR, charging injection, RWSAT replenishment by refueling water recirculation pump are frequently checked through maintenance activities and evolution of plant operating states. Local I&C equipments of these components as well as the safety logic system can be checked and the I&C hardware are considered to be reliable during plant shutdown. Local I&C equipments of the safety injection pumps, which is a mitigation function during plant shutdown, may not be operated or tested during plant shutdown. However, the DAS can be used to initiated safety injection when the I&C systems have failed, and therefore, signals to actuate safety injection pumps are also reliable. Manual operation of the safety injection pumps through the DAS is available during plant shutdown.

j. Restoration of I&C equipments can be performed within a short period of time by exchanging the faulted card.

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Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**11/25/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO.479-3871 REVISION 1**

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

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

**DATE OF RAI ISSUE:** **10/26/2009**

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**QUESTION NO. : 19-405**

(Follow-up to Question 19-394) The response to RAI 443, Question 19-394, addresses the ability of operators to recover RHR after a loss of offsite power (LOOP).

a. RHR recovery may not be achievable before boiling occurs following a station blackout. A core damage frequency (CDF) increase of less than one percent is estimated "[i]f the model has been changed," but the response does not clearly state whether the model will be revised. Please discuss whether the model will be updated to reflect the insufficient time available to perform this action, and update the DCD (including the event tree in Figure 19.1-20) accordingly.

b. In addition, the description of this action in the table attached to the response states that "[s]aturated boiling will not occur within 10 minutes even if the event has occurred at the beginning of POS [plant operating state] 4-1." No reference to calculations (e.g., those in the probabilistic risk assessment (PRA) report) is provided. Please provide the results of calculations that support the ability in all POS to recover RHR after a LOOP before boiling disables the function.

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**ANSWER:**

Answer to item a.
The LOOP event tree model will be revised according to the response to question 19-394. The changes will be incorporated in the DCD revision after revision 2.

Answer to item b.

Time available before saturated boiling can occur and disable recovery of RHR is summarized the bottom row of table 1. The analysis results provided in Table 1 are based on the MAAP code analyses same with those provided in response to RAI 19-69.

Impact on DCD

LOOP event tree model will be revised according to the response to question 19-394.

Impact on COLA

There is no impact on COLA.

Impact on PRA

LOOP event tree model will be revised according to the response to question 19-394.

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**11/25/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO.479-3871 REVISION 1**

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

**DATE OF RAI ISSUE:** **10/26/2009**

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**QUESTION NO. : 19-406**

(Follow-up to Question 19-391) The response to RAI 443, Question 19-391, states that automatic safety injection (SI) during shutdown can reduce shutdown risk, but that spurious actuation of the pumps would be a threat to workers in containment. Worker safety is clearly a primary concern during shutdown; however, the staff requests further justification for manual-only actuation of SI.

a. Please discuss the likelihood of spurious actuation of SI, with reference to the likelihood that an operator error causes SI to actuate.

b. Please discuss whether measures could be incorporated in the design to reduce the likelihood of spurious actuation or bypass the function when it would create a hazard for workers.

c. Please describe the process for evaluating potential design changes based on their risk benefit and how it was applied in this scenario.

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**ANSWER:**

Answer to item a.
If automatic safety injection (SI) were to be adopted for mid-loop operation, the following causes can initiate spurious actuation of the SI pumps.

- Spurious actuation of safety injection signal cause by failures in I&C system
- Spurious operation of SI pump circuit breakers
- Failure of water level sensors

Assuming that the duration of mid-loop operation is 100 hours per fuel outage shutdown, the likelihood of spurious actuation of SI is estimated to be in the order of 1E-5 per fuel outage shutdown.

It is difficult to estimate the likelihood of spurious actuation of SI caused by operator error. One situation where the operator can erroneously initiate SI during shutdown is when the operator attempts to test the SI pumps in the incorrect timing. However, this operation error will not result in water injection to the RV since the injection line to the RV is isolated and the SI's full flow line is opened during SI pump tests. If the operator performs SI pump tests at the wrong timing and also omits to open the full-flow line, operator error can result in injection to the RV, but occurrence of such multiple operator errors are considered to have low probability.

The likelihood of spurious actuation when the SI pumps are automated is estimated to be in the order of 1E-5 per fuel outage. However, some causes that result in spurious actuation of the pump, such as spurious operation of pump circuit breakers, can also occur even when SI is not automated. Therefore, the likelihood of spurious actuation of SI pumps may not be sufficiently different for cases where the pumps are automated and not automated. For either case, the likelihood of spurious actuation (either by human error or signal error) is low.

<u>Answer to item b.</u>
One of the measures to reduce the likelihood of spurious actuation is to design the signal logic to have redundancy against single failures. Such design can prevent spurious actuation of SI caused by single failures in the signal system. By strengthening the signal logic against single failures and by bypassing the automated SI when it would create hazards for workers, the risk from spurious actuation can be reduced.

However, bypassing the automated SI when it would create hazards for the workers would decrease the effectiveness of this function to reduce core damage risk. The workers are working inside the containment vessel during mid-loop operation. Since mid-loop operation is the plant operating state witch the automated SI can most effectively reduce risk, bypassing this function during mid-loop operation will minimize the risk benefit of automatic SI.

<u>Answer to item c.</u>
Processes to evaluate potential design changes depend on the objective of the design change. There are two types of design changes, (1) design changes to meet the commission's goal for core damage frequency (CDF) and large release frequency (LRF), and (2) design changes to achieve further safety. The processes to evaluate design changes are described below.

(1) Evaluating potential design changes to meet the commission's goal

Accident scenarios that contribute to high risk are analyzed and potential design and operational changes that can reduce risk to meet the commission's goal are evaluated. Each design and operation alternatives are discussed with design and operating experts to identify the impact on other design such as alignment and operability of the system. The design change to be applied is chosen based on the advantages and disadvantages of each design changes.

(2) Evaluating potential design changes to achieve further safety

The overall risk profile and dominant accident scenarios are analyzed and design and operational changes that can result in further risk reduction are proposed. Candidate design and operational changes are discussed with the design and operating experts to assess the impact on the latest design. The adoption of design and operational changes is comprehensively decided considering the impacts on other designs, cost to adopt the change, impacts on operability, and risk reduction. If the design change involves impermissible disadvantages, the design change will not be adopted.

In the case of automatic SI during low power and shutdown (LPSD), the CDF and LRF meet the commissions' goal without adoption of automatic SI. Automating the SI is not a design change to meet the commission's goal but a measure to further reduce LPSD risk. The evaluation of automatic SI during LPSD was based on the experience in Japanese PWRs.

MHI has experienced a study regarding adoption of automated SI during plant shutdown to reduce LPSD risk for Japanese PWR plant. In the study, discussions have been held between experts of plant design, low power and shutdown operation, and PRA to decide the applicability of the design change. There has been an objection from experts representing plant operation that the workers cannot perform maintenance activities under conditions threat of spurious actuation of automated SI. The adoption of automated SI during LPSD has been passed on for the following reasons.

- If the spurious actuation of SI occurs, there would be no barriers to protect the workers from high pressure water. Even though the probability of such an event can be low, such an situation that the workers are under threat should not be created.

- Even when the automated SI has been successfully initiated upon detection of low water level, the workers who are not aware of decreasing water level (if they where not aware for any reason) will be under threat of sudden actuation of SI.

MHI judged that the discussion on automated SI for the Japanese PWRs is applicable to US-APWR, and decided not to apply automatic SI during LPSD.

<u>Impact on DCD</u>

There is no impact on DCD.

<u>Impact on COLA</u>

There is no impact on COLA.

<u>Impact on PRA</u>

There is no impact on PRA.