MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

October 29, 2010

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-10296

Subject: MHI's Responses to US-APWR DCD RAI No. 640-5051 Revision 2 (SRP 19)

References: 1) "Request for Additional Information No. 640-5051 Revision 2, SRP Section: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation," dated September 27, 2010.

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

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

As indicated in the enclosed materials, this submittal 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 this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

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

DO81
NRD

Enclosures:

1. Affidavit of Yoshiki Ogatata
2. Response to Request for Additional Information No. 640-5051, Revision 2 (Proprietary Version)
3. Response to Request for Additional Information No. 640-5051, Revision 2 (Non-Proprietary Version)

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

Contact Information
C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-10296

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 "Response to Request for Additional Information No. 640-5051, Revision 2", and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information are identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

4. The basis for holding the referenced information confidential is that it describes the unique technique of the hydrogen burning analysis results related to the US-APWR severe accident analytical models developed by MHI.

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

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

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

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

B. Loss of competitive advantage of the US-APWR created by the benefits of the 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 29th day of October 2010.

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

Enclosure 3

UAP-HF-10296
Docket No. 52-021

Response to Request for Additional Information No. 640-5051, Revision 2

October 2010
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 640-5051 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1.5.3**
DATE OF RAI ISSUE:	**09/27/2010**

QUESTION NO. : 19-466

It appears that the calculated US APWR internal flooding CDF is higher than the total internal events CDF, contributing approximately 33% to the plant total at-power CDF. Thus, please describe, to the extent practicable, why US APWR internal flooding is a significant contributor to plant risk. Did MHI consider design changes or other actions to minimize the risk due to internal flooding? If so, what did MHI consider?

ANSWER:

The risk assessment of the US-APWR internal flooding is performed under some conservative assumptions as a bounding analysis. This is because the available specific information such as piping, routing and lengths, and flooding isolation actions are limited at the design certification (DC) phase. It is expected that the internal flooding risk will be reduced with the plant-specific detailed information.

The conservative assumptions are:

- For spray events: Assume that all SSCs in the flooded areas will lose their function. The extent of impact by spray will be limited, or the spilled water flood source will be isolated or drained out in a realistic assessment.
- For flood and major flood events: Assume that the flooded water will propagate to other areas if the doors of an area are not water-tight. The flooded water will propagate to other areas through ducts or failed doors if the water level in the water-tight area exceeds approximately eight feet of depth. According to more realistic analyses, some flood scenarios will not propagate to other areas.
- For flood sequence modeling: Flood scenarios which do not cause any initiating events are also quantified by assuming a general transient event.
- No credit is taken for the isolation of a flood source by operators except for a flood source in the break room adjoining the main control room. In some scenarios it will be possible to isolate the flood sources in the case of sufficient available time available for operator actions.

- Piping lengths used for flooding frequency evaluations are set with conservative "Length + Width + Height" measurements of the flood source room. The flooding frequencies will possibly be reduced from plant-specific piping length information.

The design basis US-APWR flooding protection features are as described in the DCD Subsection 3.4.1.3. The US-APWR considered flooding protection measures to minimize risk due to internal flooding as follows:

- Separate the reactor building (R/B) into two areas, an east side and a west side. This measure assures the integrity of two trains/divisions of safety-related systems.
- Install water-tight doors for the safety-related SSC areas, safety-related electric I&C rooms, main control room and access route to the R/B from the turbine building and auxiliary building. This measure is effective to prevent flooding propagation.
- Risk-significant systems on internal flooding of current operating plants are the circulating water system and the essential service water system. Installed flood-relief panels on the T/B exterior walls are required to trip the flooded essential service water pump within 15 minutes to prevent inflow to the R/B.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on COLA.

mpact on PRA

There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 640-5051 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1.5.3**
DATE OF RAI ISSUE:	**09/27/2010**

QUESTION NO. : 19-467

In accordance with guidance provided in SRP Appendix A, Section 19.1.3.4, please explicitly describe in the US-APWR DCD the uses of the US-APWR internal flooding PRA in the design process to reduce or eliminate the weaknesses/vulnerabilities in current reactor designs, indicating the effect of new design features and operational strategies on plant risk and identifying and using the PRA-based insights and assumptions to develop design requirements to improve the US-APWR design safety profile.

ANSWER:

Internal flooding events involve the rupture of water lines that result in a release of water that can directly cause the failure of required mitigating systems and/or cause system failure because of submersing or spraying required components. The effects of internal flooding are highly plant-specific depending on the layout of equipment within the plant and the relative isolation of rooms.

The typical plant vulnerabilities by internal flooding in the current reactors are identified by individual plant examinations summarized in NUREG-1560.

The major contributors to flooding events involve water system breaks of large quantity water systems such as a service water system and circulating water system that cause the failure of multiple mitigating systems. The causes of major contributors are breaks of large quantity water systems. The plants with the largest flood contributions are typically dominated by floods that affect support systems that have plant-specific designs such as service water system.

Various countermeasures for reducing internal flooding risk in current reactors are also proposed, such as:

- modifying doors separating the turbine building basement from areas containing safeguards equipment to prevent flood propagation into these areas,
- installing water-tight doors,
- increasing protection of components from flood effects,
- installing isolation valves and development of procedures for isolation,
- revising procedures for inspecting the floor drains and flood barriers, and

- conducting periodic inspections of cooling water piping and components.

The US-APWR is expected to satisfy the NRC safety goal and to reduce or eliminate known weaknesses of existing operating plants that are applicable to the new design by introducing appropriate features and requirements. The US-APWR has safety-related SSCs in the reactor building (R/B) and power source buildings (PS/B). Therefore, the US-APWR introduced the following design requirements to protect the R/B and PS/B against internal flooding:

- Prevent the flood propagation to multiple mitigation systems (more than two out of four trains of safety systems in the R/B and PS/B) by:
    - Separation of the R/B into two areas of an east side and a west side.
    - Installation of water-tight doors for the safety-related SSC areas, safety-related electric I&C rooms, and main control room.
    - Requirement of the isolation of a flooded essential service water pump within 15 minutes to prevent inflow into the R/B.

- Prevent inflow into the R/B from adjoining buildings, such as the T/B and A/B.
    - Install water-tight doors between the R/B and adjoining buildings.
    - Install flood relief panels on the T/B exterior walls to drain the flooded water from the circulating water system to the yard.

The separation between the east side and west side in the R/B is the most important design feature for mitigation of internal flooding consequences. The most important operating strategy is maintaining the closed state of the water-tight doors between the east side and the west side. The doors are controlled by administrative controls, such as COL item 13.5(7), which is described in DCD Table 19.1-119, "Key Insights and Assumptions".

<u>Impact on DCD</u>

The following description will be added to Subsection 19.1.5.3 of DCD Chapter 19.

(Insert the paragraph before the first paragraph of Subsection 19.1.5.3)

The US-APWR design features for reducing internal flooding risk and use of the internal flooding PRA in the design process are as follows.

The US-APWR is expected to satisfy the NRC safety goal and to reduce or eliminate known weaknesses of existing operating plants that are applicable to the new design by introducing appropriate features and requirements. The US-APWR has safety-related SSCs in the reactor building (R/B) and power source buildings (PS/B). Therefore, the US-APWR introduced the following design requirements to protect the R/B and PS/B against internal flooding:

- *Prevent the flood propagation to multiple mitigation systems (more than two out of four trains of safety systems in the R/B and PS/B) by:*
    - *Separation of the R/B into two areas of an east side and a west side.*
    - *Installation of water-tight doors for the safety-related SSC areas, safety-related electric I&C rooms, and main control room.*
    - *Requirement of the isolation of a flooded essential service water pump within 15 minutes to prevent inflow into the R/B.*

- *Prevent inflow into the R/B from adjoining buildings, such as the T/B and A/B.*
    - *Install water-tight doors between the R/B and adjoining buildings.*

- *Install flood relief panels on the T/B exterior walls to drain the flooded water from the circulating water system to the yard.*

<u>Impact on COLA</u>

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 640-5051 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1.5.3**
DATE OF RAI ISSUE:	**09/27/2010**

QUESTION NO. : 19-468

In the US-APWR DCD Section 19.1.5.3.1 (for example), which discusses the internal flooding PRA models, the floor drain system is credited to drain fluid and minimize the impact of flooding. Is the potential drain system blockage considered in the PRA model? If yes, please describe, otherwise, justify the exclusion.

ANSWER:

The floor drainage systems collect liquid waste from equipment and floor drains during all modes of operation. The waste is then separated from the contaminated effluents and transferred to the processing and disposal systems. The floor drainage systems are designed in accordance with 10CFR50, Appendix A, General Design Criteria (GDC) 2, 4 and 60 as described in DCD Subsection 9.3.3.

In general, small leaks will drain to disposal systems, and the extent of impacts by the leaked water will be limited. However, the internal flooding risk assessment of DCD Subsection 19.1.5.3 makes the conservative assumption that spray scenarios affect all SSCs in the area, including small leaks and ignoring the drainage capability. The probability of drain system blockage will be very low, and small leaks might possibly be isolated by operators because of the time they have to respond.

For example, in the B-EFW pump room (FA2-103-01), a small flood area (49,480 in^2) takes more than three hours (approximately 75,000 [l] / 380 [l/min.] = 197 [min.]) for the water level in the room to rise to eight feet at 100gpm (typical floor drain capability).

<u>Impact on DCD</u>

There is no impact on the DCD.

<u>Impact on COLA</u>

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 640-5051 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1.5.3**
DATE OF RAI ISSUE:	**09/27/2010**

QUESTION NO. : 19-469

The internal flooding insights provided in Table 19.1-119 of US-APWR DCD are limited to the design insights but not risk insights. The phrase "risk insights" refers to the results and findings that come from the internal flooding PRA. Thus, please update and enhance this table to include the internal flooding risk insights, such as:

- Which rooms are more significant to internal flood risk and why
- Which systems are more significant to internal flood risk and why
- Which systems are more significant to internal flood frequency and why
- Flood isolation/mitigation insights
- Propagation insight
- Operator action insights
- Major contributors to the uncertainty associated with the risk estimates
- Etc.

ANSWER:

The following risk insights identified from the internal flooding PRA will be added to Table 19.1-119, "Key Insights and Assumptions" of the US-APWR DCD.

(1) Rooms significant to internal flood risk

The following four areas are significant to internal flood risk and contribute about 80% of the total CDF.

Flood Source Area	Room Name	Location	CDF(/ry)
FA2-102-01	A-Emergency Feedwater Pump (T/D) Room	R/B B1F East Side	3.6E-07
FA2-108-01	D-Emergency Feedwater Pump (T/D) Room	R/B B1F West Side	3.3E-07
FA2-414-01	Main Steam and Feedwater Piping Room	R/B 3F East Side	2.2E-07
FA2-415-01	Main Steam and Feedwater Piping Room	R/B 3F West Side	1.7E-07

The most significant rooms to internal flood risk are the turbine driven (T/D) emergency feedwater pump rooms which are installed in water tight areas. If a large internal flood occurs in a room, the flood water is assumed to propagate to other areas on the R/B east or west side and cause failure of two safety-related systems. In the other rooms, T/D pumps are available because the rooms are protected by water-tight doors.

Main steam and feedwater piping rooms are also significant rooms to internal flood risk. The internal flooding frequencies of these rooms are higher than other rooms because of the many water sources in the piping rooms.

(2) Systems significant to internal flood risk

The most significant system to internal flood risk is the emergency feedwater (EFW) system. The risk-significant cutsets of internal flooding are human errors to switch-over the EFW water sources. This is because the risk-significant flood scenarios might possibly affect the SSCs on either side of the R/B, and the failure of two EFW pumps of the affected side is assumed. In this scenario, switch-over of the EFW pit or the realignment of the EFW source to the intact side of the EFW lines is required. When detailed plant-specific information is available, there is determined to be sufficient time to perform these actions.

(3) Systems significant to internal flood frequency

Significant systems to internal flooding frequencies are the following four systems, the emergency feedwater system (EFWS), main feedwater system (MFWS), main steam system (MSS) and circulating water system (CWS). These systems contain longer runs of piping in the R/B.

System		Frequency (/ry)			
		Spray	Flood	Major Flood	Total
EFWS	Emergency Feed Water System	8.1E-02	9.1E-03	2.3E-02	1.1E-01
MFWS	Main Feed Water System	2.8E-02	0.0E+00	9.5E-03	3.7E-02
CWS	Circulatiing Water System (Piping)	2.7E-02	0.0E+00	9.6E-03	3.7E-02
MSS	Main Steam System	5.2E-03	3.9E-04	4.8E-03	1.0E-02

For EFWS and MFWS, flood frequencies per piping lengths from EPRI 1013141 are relatively higher (on the order of 1E-6/yr-ft) than other systems. For CWS and MSS, the numbers of pipes and lengths of piping are relatively higher than for other systems in the R/B, though the flood frequency (on the order of 1E-7/yr-ft) is lower than that for the EFWS and MFWS.

(4) Flood isolation/mitigation and propagation insights

Except for the flood source in the break room adjoining the main control room, the isolation of flood sources by operators is not considered in this assessment.

All floors in the R/B are divided into two areas, east and west, by concrete walls and/or water-tight doors. This design mitigates the impact of flooding from one area to safety-related systems in other areas, impacting no more than two of the four trains.

(5) Significant Operator Actions

Except for the flood source in the break room adjoining the main control room, flood source isolation actions by operators is not expected in this assessment.

The most significant operator action for internal flooding is switch-over of the EFW pit or the realignment of the EFW source to the intact side of the EFW lines. This case occurs when major flooding due to failure of two trains of the EFW system propagates into the R/B east side or west side.

(6) Major contributors to the uncertainty associated with the risk estimates

The major contributors to the uncertainty associated with risk estimates are that available specific information--such as pipe routing, pipe lengths, and flooding isolation actions--are limited at the design certification (DC) phase. The risk assessment of US-APWR internal flooding is performed under some conservative assumptions as a bounding analysis. It is expected that the internal flooding risk will be reduced with the plant-specific detailed information.

In addition to incorporation of the above risk insights into DCD Table 19.1-119, the "internal flood design insights" in DCD Table 19.1-119 will be revised as shown in Table 1 to clarify the design insights.

Impact on DCD

Table 19.1-119 of DCD Chapter 19 will be revised to clarify the design insights and the risk insights as discussed above.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

Table 1 Revision of internal flood insights in DCD Table 19.1-119 (Sheet 22)

Key insights and assumptions	Dispositions
Internal flood design insights [Replace item 1, item 2 and item 3]	
1. Four redundant safety systems are located in the reactor building (R/B). Each safety system is separated into four divisions by physical barriers to assure that the functions of the systems are maintained in the event of a postulated incident.	1.2.1.7.2.1
2. The R/B consists of a radiological controlled area (RCA) and a non-radiological controlled area (NRCA) separated physically by concrete barrier walls. These concrete barrier walls are designed to preclude flooding between the RCA and the NRCA. Piping, instrumentation, HVAC duct, conduit, and cable trays are installed through the flood barrier wall above the maximum flood level or are provided with water-tight seals.	3.4.1.3
3. All floors in the RCA of the R/B are divided into two areas, east and west, by concrete walls with water-tight doors. The equipment in the east area of the RCA is the A and B train SI pumps and the A and B train CS/RHR pumps with heat exchangers. The equipment in the west area are the C and D train SI pumps and the C and D train CS/RHR pumps with heat exchangers. The concrete walls and the water-tight doors prevent flood water migration from one safety train to another. The floor drains of the east area are connected and drain into the A-R/B sump tank, and the floor drains of the west area are connected and drain into the B-R/B sump tank. There is no cross-connection between the east area drains and the west area drains.	3.4.1.5.2.1
[Add item 4 and 5]	
4. All floors in the NRCA of the R/B are divided into the two areas, east and west, by concrete walls with water-tight doors. The equipment on the east side includes two trains (A and B) of the CCW (HX and pump rooms), two trains (A and B) of the EFW (pump rooms), and the A and B train Class 1E electrical panels. The equipment on the west side includes two trains (C and D) of the CCW (HX and pump room), two trains (C and D) of the EFW (pump rooms) and the C and D train Class 1E electrical panels. The Class 1E electrical panel rooms are isolated from the corridor by concrete walls and water-tight doors. The floor drains of the east areas are connected and drain into the A-R/B non-radioactive sump, and the floor drains of the west areas are connected and drain into the B-R/B non-radioactive sump. There is no cross-connection between the east area drains and west area drains.	3.4.1.5.2.2
5. The T/B adjoins the NRCA of the R/B. The T/B is independent of the R/B to prevent internal hazards in the T/B from propagating to the R/B. Water-tight doors are installed in the doorways at ground level between the T/B and the R/B. In addition, a flood relief panel system is built into the T/B exterior walls. Actuation of the flood relief panels allows the flood water to drain out to the yard area to prevent it from affecting R/B equipment.	1.2.1.7.2.1 3.4.1.3

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 640-5051 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1.5.3**
DATE OF RAI ISSUE:	**09/27/2010**

QUESTION NO. : 19-470

Please revise the DCD to include a combined license (COL) action item or similar commitment that ensures the COL applicant that references the US−APWR design certification will perform as-designed and as-built information verification and will conduct walk-downs to confirm that the assumptions used in the PRA remain valid with respect to the internal flooding events.

ANSWER:

MHI agrees to incorporate the requested commitment. The DCD COL action item, COL 19.3(4) in DCD Revision 2 Section 19.3 committed that the PRA and severe accident evaluation will be updated as necessary to assess specific site information and associated site-specific external events. Further, the following commitment will be incorporated into Subsection 19.1.2.4 of DCD Chapter 19 and clarifications will be added to Subsection 19.1.2.4 to reflect the commitment to applicable PRA standards when the PRA updates are performed.

In accordance with 10 CFR 50.71(h)(1), prior to the scheduled date for initial loading of fuel, a plant-specific PRA that covers initiating events and modes for which NRC-endorsed consensus standards on PRA exist shall be developed one year prior to the scheduled date for initial loading of fuel. The plant-specific PRA will reflect the as-built plant. The plant-specific PRA model will utilize the US-APWR DCD PRA model as a baseline along with any additional modeling changes resulting from the plant-specific design. Any departures from the design used in the US-APWR DCD PRA, insights from procedure development and operator training, or other PRA modeling changes that are identified subsequent to the completion of the US-APWR DCD PRA will also be utilized. The PRA-based risk insight differences between the plant-specific PRA and the US-APWR DCD PRA will be evaluated. Plant walk-downs to confirm that the assumptions used in the PRA remain valid will also be conducted.

Impact on DCD

Subsection 19.1.2.4 of DCD Chapter 19 will be revised as follows.

19.1.2.4 PRA Maintenance And Upgrade

The objective of the PRA maintenance and upgrade program is to ensure that the PRA will be maintained and upgraded so that its representation of the as designed, as-to-be built, and as-to-be operated plant is sufficient to support the applications for which the PRA is being used. The PRA will be under configuration control and the program will contain the following key elements:

- *A process for monitoring PRA inputs and collecting new information*
- *A process that maintains and upgrades the PRA to be consistent with the as-built, as-operated plant*
- *A process that ensures the cumulative impact of pending changes is considered when applying the PRA*
- *A process that evaluates the impact of changes on previously implemented risk-informed decisions that have used the PRA*
- *A process that maintains configuration control of computer codes used to support PRA quantification*
- *Documentation of the program*

PRA maintenance involves updating the PRA models to reflect plant changes such as modifications, procedure changes, or plant performance. A PRA upgrade involves the incorporation of new methodologies or significant changes in scope or capability. These changes might include items such as new human error analysis methodology, new data update methods, new approaches to quantification or truncation; or new treatments of common cause failure (CCF).

In accordance with 10 CFR 50.71(h)(1) (Reference 19.1-15), prior to the scheduled date for initial loading of fuel, a plant-specific PRA that covers initiating events and modes for which NRC-endorsed consensus standards on PRA exist one year prior to the scheduled date for initial loading of fuel shall be developed. The plant-specific PRA will reflect the as-built plant. The plant-specific PRA model will utilize the US-APWR DCD PRA model as a baseline. Any additional modeling changes resulting from the plant-specific design, departures from the design used in the US-APWR DCD PRA, insights from procedure development and operator training, or other PRA modeling changes that are identified subsequent to the completion of the US-APWR DCD PRA will also be utilized. The PRA-based risk insight differences between the plant-specific PRA and the US-APWR DCD PRA will be evaluated. Plant walk-downs to confirm that the assumptions used in the PRA remain valid will also be conducted.

During operation, PRA will be maintained and updated in accordance with approved station procedures on a periodic basis not to exceed two refueling cycles.

Changes in PRA inputs or discovery of new information will be evaluated to determine whether the new or changed information warrants a PRA maintenance or upgrade. Changes that would impact risk-informed decisions will be prioritized to ensure that the most significant changes are incorporated as soon as practical. Other changes will be incorporated during the next PRA update.

Changes to the PRA due to PRA maintenance and PRA upgrade will meet the risk assessment technical requirements of the NRC-endorsed PRA standards. Upgrades of the PRA will receive a peer review in accordance with the requirements of the NRC-endorsed PRA standards, but will be limited to aspects of the PRA that have been upgraded.

The PRA will be updated to reflect plant experience, operational experience, and PRA modeling changes, consistent with the NRC-endorsed standards. These standards are described in Section 19.1 and were in existence one year prior to the issuance of the maintenance update scheduled in compliance with 10 CFR 50.71 specified criteria and intervals.

Impact on COLA

There is no impact on the FSAR.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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DATE OF RAI ISSUE:	**09/27/2010**

QUESTION NO. : 19-471

Please explain why the number of internal flood PRA components shown in Table 22.3-2 of the US-APWR PRA is much greater than the fire PRA components provided in the Appendix 23F given that the flooding and fire zones include the same areas.

ANSWER:

Table 23F-1 "List of fire PRA components" and Table 22.3-2 "Internal Flood PRA Components and Locations" of the US-APWR PRA report, MUAP-07030(R2), summarize components in each fire area and flood area, respectively. Table 23F-1 was edited referring the cable routing list (Attachment 23E of the PRA report) and fire scenarios (Attachment 23M of the PRA report), which were used for fire scenario component quantification.

There are some differences between Table 23F-1 "List of fire PRA components" and Table 22.3-2 "Internal Flood PRA Components and Locations" of the US-APWR PRA report, MUAP-07030(R2). The locations of some air operated valves (AOVs) in Table 23F-1 listed the locations of the solenoid-operated valves for AOV valves.

Also there are some editorial errors in Table 23F-1 and Table 22.3-2. Attachment 19-471-A includes marked-up tables of Table 23F-1 and Table 22.3-2 on the PRA report.

The PRA components of both revised tables are consistent, aside from differences noted above.

<u>Impact on DCD</u>

There is no impact on the DCD.

<u>Impact on COLA</u>

There is no impact on COLA.

<u>Impact on PRA</u>

Revise Table 23F-1 and Table 22.3-2 of PRA report MUAP-07030(R2).

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 640-5051 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1.5.3**
DATE OF RAI ISSUE:	**09/27/2010**

QUESTION NO. : 19-472

The staff identifies multiple inconsistencies between the updated internal events USAPWR PRA submitted in the current DCD Revision 2 and the internal flooding PRA (i.e., flood-induced LOFF CCDP in US APWR PRA Table 22.6-2 is less than the internal events LOFF CCDP in US APWR DCD Table 19.1-23). It appears that the internal flooding PRA currently presented in the US-APWR DCD Revision 2 is quantified using the internal events PRA provided in the DCD Revision 1. Please describe the plan for updating the internal flooding PRA to be consistent with the most recent internal events PRA.

ANSWER:

The internal flooding risk evaluation is not re-quantified for DCD Revision 2. This is because it is judged that the design changes and internal events model changes will not increase the risk of internal flooding.

MHI quantified the internal flooding scenarios for the dominant 16 flooding areas (96% of total CDF) using the updated internal events models. The results of the total risks are reduced from 1.4E-06/RY to 1.0E-06/RY. This is because the key contributor "operator failure to open the valve of EFW pit discharge cross tie line" event frequency is reduced in the internal events PRA model. Therefore the impact to internal flooding will not be significant.

MHI is planning to perform the internal flooding PRA to be consistent with the latest internal events PRA model and design information and to consider a more realistic approach to reduce the internal flooding risk.

MHI is planning to update DCD section 19.1.5.3 by the end of March, 2011, in DCD Revision 3.

<u>Impact on DCD</u>

Subsection 19.1.5.3 of DCD Chapter 19 will be updated based on the revised flooding PRA.

<u>Impact on COLA</u>

There is no impact on COLA.

Impact on PRA

PRA technical report MUAP-07030(R2), Chapter 22 on internal flooding PRA, will be updated.