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

December 8, 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-08281

Subject: MHI's Responses to US-APWR DCD RAI No. 97-1551 Revision 1

Reference: 1) "Request for Additional Information No. 97-1551 Revision 1, SRP Section: 19-Probabilistic Risk Assessment and Severe Accident Evaluation," dated September 10, 2008

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

Enclosed is the responses to the RAIs contained within Reference 1.

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

Sincerely,

Y. Ogata

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

Enclosure:

1. "Responses to Request for Additional Information No. 97-1551 Revision 1"

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

D081
NRO

Docket No. 52-021
MHI Ref: UAP-HF-08281

Enclosure 1

UAP-HF-08281
Docket No. 52-021

Responses to Request for Additional Information
No.97-1551 Revision 1

December 2008

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-183

In Section 6A.5.1.4 "Test and Maintenance" of the PRA report (MUAP-07030, Rev 1) it is stated that the emergency feedwater (EFW) pumps are tested every three months. The following statement is also made in Section 6A.5.1.4: "Periodic tests for the motordriven and the turbine-driven emergency feedwater pumps are performed through the minimum flow line. In addition, pump flow tests are performed through test lines to check the pump operability." From these statements it is not clear whether both types of tests (i.e., the one through the minimum flow line and the full flow test) are performed every three months. Please explain how the frequency of each of these tests is used to calculate failure probabilities, such as failure of pumps to start or motor-operated valves to open on demand. Also, please discuss how the 24-month testing frequency of components (e.g., check valves or motor-operated valves) located downstream of each test line was taken into consideration in assessing their failure probabilities. In addition, it is stated in Section 6A.5.1.4 that "During test and maintenance, isolation valves for each EFW pump discharge tie line are all kept open." The human error to fail to close some of these valves following test or maintenance and, thus, the failure to separate the four trains does not appear to have been modeled in the PRA. Please explain, including assumptions and design features that are available, if any, to prevent such human errors.

ANSWER:

The EFW pumps are tested every 3 months utilizing the full flow line and minimum flow line. Both flow lines will be check during the EFW pump test. The motor operated valves in the EFW supply line will be tested every 3 months, and check valves will be tested every 24 months.

Failure probabilities of the valves and pumps to operate on demand are based on demand failure probability data reported in NUREG/CR-6928. The failure data used to evaluate these failure

probabilities include equipments tested under test intervals longer than 24 months, and therefore, we judge that these failure probability data are applicable to the valves modeled in the EFW fault tree.

The EFW pumps can achieve their function even when the EFW pump discharge tie line valves are left open. Human error to leave open the EFW pump discharge tie line valves will be important when a large break in the EFW piping occurs in one of the EFW trains during the mission time. The combined probability of EFW piping to break when the tie line valves have been left open is very small. The human error to leave the tie line valves after maintenance is therefore not modeled.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-184

In Section 6A.5.2.2.1 "Loss of Offsite power" of the PRA report (MUAP-07030, Rev 1) it is stated: "Two of four EFW pumps must supply feedwater to the associated SGs if the EFW pump discharge tie-line isolation valves are kept closed without power supply from the offsite power." However, the staff notices that these motor-operated valves are supplied by dc power. Please clarify.

ANSWER:

EFW pump discharge tie-line isolation valves are supplied by dc power and can be operated under station black out condition. The description in the PRA technical report which the staff has pointed out will be amended during the next update. In addition, these valves are modeled as dc power operated valves in the PRA so there is no impact on the PRA model.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
The PRA technical report documentation will be amended during the next revision.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-185

It is stated in Section 10.4.9.2.2 of the Design Control Document (DCD): "The manual valves in the suction line flow paths from the EFW pits to the M/D and T/D EFW pumps are normally closed." However, these manual valves are modeled in the PRA as "locked open" during normal operation at power. Please clarify.

ANSWER:

The manual valves in the suction line flow paths from the EFW pits to the M/D and T/D EFW pumps are normally opened. Statement in Section 10.4.9.2.2 of the DCD will be amended.

Impact on DCD
Section 10.4.9.2.2 of the DCD will be amended as follows during the next revision.

The statement
"The manual valves in the suction line flow paths from the EFW pits to the M/D and T/D EFW pumps are normally closed."
will be changed as below.
"The manual valves in the suction line flow paths from the EFW pits to the M/D and T/D EFW pumps are normally opened."

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-186

It is stated in Section 10.4.9.2.1 (item D) of the DCD: "Two 50% EFW pits are provided. Both EFW pits together contain the minimum water volume required for maintaining the plant at hot standby condition for 8 hours and performing plant cooldown for 6 hours until the RHRS can start to operate." However, in the PRA a 24-hour mission time was considered for sequences with successful delivery of emergency feedwater (EFW) with no credit for operator action to supply water from the demineralized water storage tank. Furthermore, the need to remove heat using the residual heat removal system (RHRS), even before 24 hours into the accident, was not modeled. Please explain.

ANSWER:

The plant can achieve stable plant condition by maintaining hot standby condition or moving to cold shutdown. It is an usual approach for at-power PRA to consider the successful end state as hot standby condition if this condition is achievable. As it is for conventional PRAs, the US-APWR PRA does not consider transition to cold shutdown and therefore additional water supply to the EFW system is not modeled.

The actual mission time for the EFW system to perform plant cool down would be much shorter than 24 hours as state in 10.4.9.2.1 of the DCD. The US-APWR PRA conservatively considers a 24 mission time for systems that would actually be required to operate for less than 24 hours. This consideration is applied to the mission time considered for the EFW pumps and this is the reason a mission time of 24 hours is considered for the EFW system.

In addition, if the plant were to continue hot standby condition, the two EFW pits together contain enough water volume to maintain hot standby condition for 24 hours without water supply.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-187

A failure to open probability of 7E-4 per demand is assumed for the normally locked closed manual isolation valves VLV-006A and VLV-006B (PW2A and PW2B in the PRA) located in the line connecting the two EFW pits and for the manual isolation valve VLV-004 (PW3XV in the PRA) from the secondary tank (demineralized water storage tank) to the line connecting the two EFW pits. Please explain the basis of the assumed failure probability in terms of testing frequency and other relevant testing or monitoring requirements for these manual valves. Also, please explain why the operator failure to open valve VLV-004 (PW3XV in the PRA) has not been modeled in the PRA.

ANSWER:

There are no testing requirements for the tie line valves between the two EFW pits. However, valves EFS-VLV-006A, EFS-VLV-006B and EFS-VLV-004 are identified as risk significant structures systems and components (SSCs) and will be under control of the reliability assurance program (RAP).

The generic data for manual valves reported in NUREG/CR-6928 are applied to the failure probability of the manual valves. The failure probabilities of the manual valves in NUREG/CR-6928 are based on valves of annual demand rates ranging from approximately 1 to 12. The probability of manual valves that are tested less than every 12 months may have higher failure probabilities than that reported in NUREG/CR-6928.

In the EFW system analysis model, failures of the manual valves are modeled along with operator action failure to change over water sources. The human error probability to open each manual valve is evaluated as 4E-3, which is five times higher than the hardware failure probability applied to the manual valves. Even if the manual valves were to be tested with long intervals, the increase in failure probabilities of manual valves due to long test intervals is considered to be much smaller than the

human error probability to operating valves. For this reason, the use of the generic data was considered not to have impact on system reliability.

Operator action failure to open valve EFS-VLV-004 is considered in the human error basic event "EFWOO01PW2A". Operator tasks considered for "EFWOO01PW2A" is described in the PRA technical report (MUAP-07030 R1), page 9-40 of chapter 9. Action to open EFS-VLV-004 (PW3) is handled as item 4 in the human error quantification sheet.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-188

The emergency feedwater (EFW) line control throttle motor-operated valves (AWAA, AWBA, AWCA and AWDA) regulate feedwater flow to the steam generators (SGs). Since these valves are not in the EFW pump test lines, they are tested and verified every 24 months (US-APWR PRA Section 6A.5.3). It is assumed that failure to control is due only to demand stresses during their 24-hour mission time (no standby stresses during the 24 months between testing are considered). Please provide the basis for this assumption. Also, please explain why (1) the common cause failure (CCF) of the control valves to control the flow and (2) instrumentation and control (I&C) failures are not modeled in the PRA.

ANSWER:

The flow control motor operated valves are kept opened during normal plant conditions and they remain as is when they have failed during standby. Even if the function to control has failed before the occurrence of the initiating event, the valves will be in opened state and the EFW pumps can provide water to the SGs. Moreover, the water level in the SG will be maintained within the level necessary for secondary side cooling by automatic actuation of EFW isolation valves even if control valves do not operate, as discussed in question 19-190 of RAI No.97. For this reason, failures to control the valves during standby are not modeled. CCF of the control valves to control during the mission time needs to be considered and such failures will be incorporated in the PRA during the next PRA update.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
The PRA will be revised to take into account the failure to control the valves during the mission time of EFW.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-189

No common cause failure (CCF) of the emergency feedwater (EFW) pit water level sensors (other than miscalibration) is modeled in the PRA. Miscalibration is considered for the sensors of each pit separately (i.e., no miscalibration error across all sensors is considered). Please discuss.

ANSWER:

Water level is checked every 12 hours and failure of a sensor can be detected by recognizing inconsistent output signals (water levels) from the two sensors, and are likely to be repaired within a short period of time if failure has been detected. Common cause failure of two sensors of an EFW pit that result in same incorrect output signal may not be detected and left faulted until the initiating event occurs, but such kind of failure is considered to be rare. For this reason failure of water level sensors that occur prior to the initiating event is not modeled.

Failure of sensors during the 24 hour mission time is modeled in the PRA. When the two sensors associated to an EFW pit both fail, the operator cannot detect the timing to change over the water source and the EFW pumps that are supplied water from the pit may fail. CCF may occur among the sensors and therefore the PRA will be revised to take into account of the CCF of sensors, during the next update. The probability of CCF of sensors during the mission time is much lower than the human error probability of operators to change over from the low level EFW pit, which is the order of 1E-2. Therefore, the change to the PRA model will not impact the PRA result.

The EFW pits are located in opposite sides of the reactor building. The operators need to move between the pits when calibrating the water level sensors and a certain amount of time will required to perform calibration of the both pairs of sensors. For this reason the operator action failure to calibrate the sensors in the two pits are considered to have very low dependencies and miscalibration error across all sensors are not modeled.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
CCF of water level sensors that occur during the mission time will be in incorporated in the model during the next update.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1
SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation
APPLICATION SECTION: 19.1
DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-190

The following statement is made in Section 10.4.9.2.2 of Revision 1 of the Design Control Document (DCD): "Upon LOOP, the main feedwater pumps trip and the water level of the SGs initially lowers and then recovers gradually upon initiation of the EFW flow. To maintain the adequate range of water level in SGs, the EFW flow rate is manually controlled by the operator from the MCR." Please discuss how the operator failure to manually control the EFW flow rate was addressed in the PRA.

ANSWER:

EFW flow rate is manually controlled by operator from the MCR to maintain adequate water level in the steam generators (SGs). However, water level in the SGs needed for secondary cooling can be maintained by the actuation of interlocks implemented on the EFW control valves (EFS-MOV-017A,B,C,D in the P&ID) and the EFW isolation valves (EFS-MOV-019A,B,C,D).
When the water level in the SGs are low, an automatic signal to fully open the EFW control valves and EFW isolation valves will be actuated upon detection of low SG water level. When the water level in the SGs are high, an automatic signal to close the EFW control valves and EFW isolation valves will be actuated upon detection of high SG water level. Thus the SG water level will be within the range for effective secondary side cooling regardless of operator action.
Failure of operator action to manually control the EFW flow rate is therefore not modeled in the PRA. This design characteristic will be documented in the PRA technical report during the next revision.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/8/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.97-1551 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/10/2008

QUESTION NO. : 19-191

The assumed unavailability of the EFW pumps due to maintenance outage (4E-3/demand for the M-D pumps and 5E-3/demand for the T-D pumps) is based on operating reactor experience and technical specifications (TS). In the US-APWR design, the outage of a single EFW system train is not a limiting condition for operation and, therefore, the average outage time of an EFW pump can be much higher than it is for operating reactors. Please address this issue for the EFW system (and for any other systems that are impacted by this issue).

ANSWER:

Although the technical specifications does not require all four trains to be applicable as the limiting condition for operation (LCO), the EFW pumps will not unnecessary be taken out of service. It is more likely that the actual out of service time will be determined by failure type and repair time with an expectation that the newer design pumps will experience higher reliability. The times assumed are also expected to be impacted by regulations such as MSPI and derivative requirements that impact unavailability monitoring. For this reason, the US-APWR PRA considered the generic unavailability data to be applicable to evaluate the baseline CDF.

To investigate the impact of the unavailability of EFW pumps a sensitivity analysis has been performed assuming a yearly 7 days out of service for the EFW pumps. If the each of EFW pumps were taken out of service for 7 days per year, the resulting CDF for internal events at power is 1.3E-6 /RY, which is approximately 9% increase from the base case.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.