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December 19, 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-08300

Subject: **MHI's Responses to US-APWR DCD RAI No. 104-1622 Revision 1**

Reference: 1) "Request for Additional Information No. 104-1622 Revision 1, SRP Section: 19-Probabilistic Risk Assessment and Severe Accident Evaluation," dated November 20, 2008

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

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. 104-1622 Revision 1"

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

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Enclosure 1

UAP-HF-08300
Docket No. 52-021

Responses to Request for Additional Information
No.104-1622 Revision 1

December 2008

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/19/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.104-1622 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/20/2008

QUESTION NO. : 19-201

The following statement is made in Section 6A.9.1.3 of Revision 1 of the PRA report: "These valves [Reactor Coolant Pump Thermal Barrier Cooling Water Outlet Valves] automatically close on a high flow rate signal at the outlet of the component cooling line in the event of in-leakage from the thermal barrier, and prevents the in-leakage from the entire CCWS." However, these valves are not shown in the simplified system diagrams included in the PRA and their failure is not listed in the fault tree basic events (Table 6A.9-7). Please explain whether and how the failure of these valves to close automatically was modeled in the PRA. Is the capability of automatic closure of these valves an important design feature that prevents or mitigates certain accidents? If this is true, please describe such accidents and discuss any assumptions that were made in evaluating their associated risks or in screening out such accidents from a detailed evaluation in the PRA.

ANSWER:

The PRA assumes operator actions to close the CV isolation valve in the CCW line (NCS-MOV-436A or NCS-MOV-438A for reactor coolant pump A and B in the P&ID) when the RCP thermal barrier cooling water outlet valves (NCS-FCV-1319A and NCS-FCV-1319B for reactor coolant pump A in the P&ID) have failed to close. Taking credit of this manual action and the automatic actuation of the RCP thermal barrier cooling water outlet valves, the in-leakage from the thermal barrier was considered to have similar impact as small LOCA or very small LOCA, but with low event frequency. For this reason this event was screened out from detailed analysis.

When in-leakage from the thermal barrier occurs, a series of RCP thermal barrier cooling water outlet valves automatically close upon detection of high flow rate at the component cooling line. If either of the valves closes, further leakage can be prevented. Even if the isolation valves both fail to close, the operator will manually close the containment vessel (CV) isolation valve in the component cooling line.

Should the CV isolation valve be closed, the in-leakage can be handled as leakage from reactor coolant system (RCS) in the CV.

The conditional probability P of preventing further in-leakage to the component cooling water (CCW) line given failure in the thermal barrier can be estimated as below.

$P = P_A \times P_B$

P_A: Failure of the two reactor coolant pump thermal barrier cooling water outlet valves to close.
This probability can is considered to be dominated by the common cause failure (CCF) probability of two motor operated valves to close on demand. Assuming the generic failure probability of motor operated valves to close 1E-3, and the beta factor 4.7E-2, the CCF probability of the valves is 4.7E-5. Taking into account the contribution from other combinations of failures, P_A is estimated to be approximately 1E-4.

P_B: Failure of CV isolation valve to close.
This probability is considered to be dominated by failure of motor operated valves to close and the human error of the operator. The generic failure probability of motor operated valves to close is 1E-3. Taking in to consideration that human error probabilities (HEP) of similar operation actions that require operation of valves from the main control room are the order of 1E-3, P_B is estimated to be approximately 1E-2.

From the discussion above, the conditional probability of failing to prevent further in-leakage to the CCW line given failure in the thermal barrier is estimated to be approximately 1E-6. Moreover, conditional probability of failing to stop loss of coolant, which can be achieved by the reactor coolant pump thermal barrier cooling water outlet valves, given failure in the thermal barrier is approximately 1E-4. These low probabilities imply that in-leakage from thermal barrier will have similar impact or less than that of leakage from RCS piping with high confidence.

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

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US-APWR Design Certification

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Docket No.52-021

RAI NO.: NO.104-1622 REVISION 1

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/20/2008

QUESTION NO. : 19-202

The success criteria of the component cooling water (CCW) system are discussed in Section 6A.9.2.2 and Table 6A.9-2 of Revision 1 of the US-APWR PRA report. However, these success criteria are not clearly defined and appear to be inconsistent with the definition of the corresponding fault tree top events as well as with event tree top events. For example, fault tree CWS-00A (Table 6A.9-3) is described as the failure to supply adequate CCW through supply header line "A" during an accident that is not initiated by a very small LOCA (VSLOCA) or a loss of offsite power (LOOP). However, the VSLOCA and LOOP event trees indicate that CCW is needed to cool several credited mitigating equipment, such as the containment spray/residual heat removal (CS/RHR) pumps. Therefore, the fault trees CWS-00A, CWS-00B, CWS-00C and CWS-00D (appropriately modified to address event-specific conditions, if any) are also applicable to VSLOCA and LOOP accidents. Although it is not explicitly stated, the fault trees that are reported as specific to VSLOCA and LOOP accidents in Section 6A.9.4.1 (e.g., CWS-VS-00A1 and CWS-R2) are needed to model the loss of CCW to the nonsafety related charging pumps and the reactor coolant pump (RCP) seals, respectively. In addition, the definition of the fault tree top events CWS-R2 and CWS-R4 for LOOP sequences, and the associated success criteria, are not clear. For example, the fault tree top event "CWS-R4" is defined as "CCWS fails to restart given B.O sequence" and indicates that the re-start of one of four CCW pumps following a LOOP event is needed for success without explaining how this fault tree is used and why CCW flow to any one of the four supply headers is a success. Please clarify and revise as necessary.

ANSWER:

Fault tree "CWS-00A", which is the case CCW supply headers are not required to be isolated, is only applied to event heading "CHI" in the VSLOCA event tree. For other event headings in the VSLOCA event tree, such as those for high head injection and CS/RHR, fault trees CWS-00A, CWS-00B, CWS-00C and CWS-00D are applied. Current description for VSLOCA in Table 6A.9-2 only describes the success criteria of the CCW system for charging system and does not cover all the cases applied to this event tree. Table 6A.9-2 will be amended to capture other success criteria applied for VSLOCA.

One train of the CCW system has the ability to remove the heat load required to maintain RCP seal integrity. Even if RCP seal LOCA has occurred, the event can be mitigated if one safety train is operable. From the consideration that the prevention or mitigation of RCP seal LOCA can be achieved if at least one CCW train is operable, the success criteria for avoiding core damage was represented by the success of one CCW train to restart. For this reason, success criteria for the CCW system applied in "CWS-R4" and "CWS-R2" is set as one CCW train to operate.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Table 6A.9-2 in Attachment 6A of the PRA report will be amended to clearly describe the success criteria for CCW applied in the VSLOCA event.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **11/20/2008**

QUESTION NO. : 19-203

The component cooling water (CCW) supply and return header isolation valves are tested every 24 months at refueling. Please verify the applicability of operating reactor demand failure rates to equipment used in the US-APWR which has much longer testing intervals (e.g., 24 months). In addition, please address the following: (1) The presence of the same event descriptions for different events in Revision 1 of the US-APWR PRA report (e.g., CWSPCBDCWPD and CWSPCBDCWPD-R in Table 6A.9-7); (2) The reason for not considering common cause failure (CCF) of the CCW supply and return header isolation valves to control; and (3) The probability of 1.2E-8 used for the CCF of all four CCW pumps (basic event CWSCF4PCYR-FF) and all four emergency service water system (ESWS) pumps (basic event SWSCF4PMYR-FF) to continue running for their entire mission time.

ANSWER:

Unreliability data for motor operated valves used in NUREG/CR-6928 include data from various systems with demands per year ranging from 0.1 to 20. Although the 24 month test interval may be longer than conventional plants, the test interval is within the range of the collected data, and therefore we judge that the failure probability can be applied. Additionally, the risk achievement worth (RAW) of the CCW supply and return header isolation valves are less than 2.0. Uncertainties of the failure probabilities applied to these valves have small impact on the core damage frequency.

Responses to other questions are written below.

(1)
Basic events "CWSPCBDCWPD" and "CWSPCBDCWPD-R" both represent the failure of the same pumps to start. The letters "-R" are used to show that this failure has occurred after the restart of CCW system and ESW system following loss of offsite power (LOOP). Since all trains of the CCW system and ESW system restart after recovering from LOOP, CCF of four trains to start on demand may occur,

while this combination of CCF do not occur in other initiating events. To address such CCF events, different basic event name were given for failure that occur after restart.

(2)
The CCW supply and return header isolation valves are gate valves with no function to regulate flow. The basic event "fail to control" are used for valves that are required to regulate flow. This failure mode does not apply to the CCW supply and return header isolation valves, and therefore, single failure basic events representing "failure to control" will be removed from the model.

(3)
For CCFs that occur among components with asymmetric initial conditions, a modified set of MGL parameters were applied. Detailed discussion including the MGL parameters applied is documented in the PRA technical report (MUAP-07030 R1), section 8.7 of chapter 8.

Since the modified set of MGL parameters can only be applied to CCF with group size of four, the CCF group function of the RiskSpectrum software could not be applied. For this reason, no CCF group identifiers, which is listed in table 6A.9-8 of the PRA technical report, is assigned to these CCF events.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
The fault trees in the PRA model will be revised to eliminate the "fail to control" basic events applied to CCW supply and return header isolation valves.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 11/20/2008

QUESTION NO. : 19-204

The CS/RHR heat exchanger cooling water outlet motor-operated valves (labeled 114A, B, C and D in the PRA) are discussed in Attachment 6A.9 (Component Cooling Water/ Essential Service Water System) of Revision 1 of the US-APWR PRA report but they are modeled in Attachment 6A.3 (Containment Spray/Residual Heat Removal System). A clear definition of a system's boundary in the fault tree analysis is necessary to avoid missing important failures in the PRA model. Please include a statement in Attachment 6A.9 referring to the modeling of these valves in Attachment 6A.3. The CS/RHR heat exchanger cooling water outlet motor-operated valves are tested every 24 months at refueling. Please verify the applicability of operating reactor demand failure rates to equipment used in the US-APWR which has much longer testing intervals (e.g., 24 months).

ANSWER:

The failure in CCW that only impact the CS/RHR heat exchanger function is modeled in the CS/RHR system model. A statement clarifying that the heat exchanger out let valves are model in Attachment 6A.3 will be included in Attachment 6A.9.

The CS/RHR heat exchanger cooling water outlet motor-operated valves are tested every three months. Statement in section 6A.9.1.4 of Attachment 6A, that states periodic tests are performed every 24 months will be amended. A thorough review of test intervals applied to components will be performed 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
A statement will be included in Attachment 6A.9 referring to the modeling of CS/RHR heat exchanger cooling water outlet motor-operated valves in Attachment 6A.3. Test intervals applied to each components will be reviewed.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **11/20/2008**

QUESTION NO. : 19-205

Please address the following questions related to "alternate component cooling" (i.e., cooling of the charging pumps when component cooling water is lost by using the fire protection water supply system or the non-essential chilled water system) discussed in Attachment 6A.10 of Revision 1 of the PRA report:

(a) Motor-operated valves ACWCH6A and B are shown as normally open valves which must close for "alternate component cooling." However, Tables 6A.10-4 (Basic Events) and 6A.10-5 (Common Cause Failure Events) do not include a failure to close event but a failure to open event. This is also the case in the associated fault tree ACW (gates ACW-CH06A and B) where the failure of these valves to open, instead of the failure to close, is shown.

(b) The probability of operator failure (human error) to establish "alternate component cooling" by using the fire protection water supply system (ACWOO02FS) or the non-essential chilled water system (ACWOO02CT) is assumed to be 2E-2. This probability is the "lower bound" of the human error probability (HEP) estimated in Chapter 9 of the US-APWR PRA report for events ACWOO02FS and ACWOO02CT. It is argued that the lower bound of the HEP is a good estimate because "the frequent training has made operators very familiar with the accident sequence" and this reduces the "extremely high" stress level used in the calculation of the HEP. However, there is no COL action requirement proposed to ensure that such "frequent training" will take place.

(c) Explain how conditional HEPs were estimated. For example, the failure probability to establish "alternate component cooling" by using the non-essential chilled water system (event ACWOO02CT) given failure to establish "alternate component cooling" by using the fire protection water supply system (event ACWOO02FS probability set to 1) is assumed to be 5.1E-1 without any discussion or explanation.

(d) There is no reference to test and maintenance requirements for the motor-driven or diesel-driven pumps of the fire protection water supply system and the nonessential chilled water system or for the motor-operated isolation valves that are required to change status to establish "alternate component cooling." Please verify the applicability of operating reactor demand failure rates to equipment used in the US-APWR which has much longer testing intervals (e.g., 24 months) and different testing and maintenance requirements.

(e) The failure to establish "alternate component cooling" by using the fire protection water supply system, due to common cause failure (CCF) of motor-operated valves (MOVs) in both the supply and return lines, was modeled. However, it appears that not all CCF combinations of MOVs were considered, such as the CCF of MOVs ACWCH1A and ACWCH2B. Please discuss.

ANSWER:

(a)
The motor-operated valves ACWCH6A and B are required to close when alternative component cooling is established, as the staff has pointed out. The descriptions of the basic events and common cause failure events considered for these valves will be amended. Since the same failure probabilities are applied to both failures to open and to close, this change will not impact the quantification.

(b)
Frequent training of operations to establish alternate component cooling water will be added in the list of key assumptions. This key assumption will be incorporated in the training program defined as a COL item in DCD Section 13.2 (i.e. COL 13.2(1)).

Moreover, this operator action is identified as risk important operator action and will be inputted to the Human Factors Engineering Program as PRA insight.

(c)
Dependency of the two human errors ACWOO02CT and ACWOO02FS are considered to be "high". The dependency level evaluation for these operator actions is described in Table 9.4.3-1 (sheet 4 of 11) of chapter 9 in the PRA technical report. Dependency of the actions is considered to be high since the actions are assumed to be performed by the same crew and the actions taken in a close location.

Conditional human error probabilities are calculated by the approach described in section 9.4.3 of the PRA technical report, chapter 9. For operator actions that have high dependency, the conditional human error probability given that the proceeding action has failed is calculated as (1+N)/2, where N is the unconditional human error probability.
The unconditional human error probability N of ACWOO02CT is 2.0E-2. The conditional human error probability given that operation to supply fire protection water to the CCW line, is therefore 5.1E-1.

(d)
Maintenance of pumps of the fire protection water supply system will be controlled by the Fire Protection Program (DCD section 9.5.1).
The non-essential chilled water system consists of four pumps each with 33+1/3 % capacity required during normal operation. Two or three of the four pumps will be normally running, applying a rotating shift. The standby state of each non-essential chilled water system pump will not exceed 6 months.

(e)
Common cause failure between valves used to supply chilled water and fire protection water to the component cooling water line are not modeled. Common cause failures that involve failure of two of these motor operated valves may result in failure of alternate component cooling.

Each of such common cause failure are estimated to occur with a probability of 4.7E-5, based on estimation considering the failure probability (1E-3) and the beta factor (4.7E-2) of motor operated valves. This probability is less than 0.5% of the probability to fail to establish alternate component cooling by a series of human errors. This implies that the common cause failure of the valves that are not modeled have negligible impact on the CDF result.

Impact on DCD
Frequent training of operations to establish alternate component cooling water will be added in the list of key assumptions.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.