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November 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-08263

Subject: MHI's Responses to US-APWR DCD RAI No. 86-1426 Revision 0

References: 1) "Request for Additional Information No. 86-1426 Revision 0, SRP Section: 19-Probabilistic Risk Assessment and Severe Accident Evaluation," dated October 20, 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.

Enclosures:

1. "Responses to Request for Additional Information No. 81-1253 Revision 0"

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

Contact Information
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Enclosure 1

UAP-HF-08263
Docket No. 52-021

Responses to Request for Additional Information No.86-1426 Revision 0

November 2008

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/19/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.86-1426 REVISION 0

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 10/20/2008

QUESTION NO. : 19-128

A failure probability of 1.2E-5 per demand is used for the failure to open of the accumulator injection line check valves as well as for the failure to open of the check valves at the direct vessel injection (DVI) lines and the check valves at the RHR/CSS cold leg injection lines. This failure probability was taken from NUREG/CR-6928 "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." This failure to open probability estimate, which is significantly lower than most of the other failure probabilities reported in the literature and documented in Table 7.1-1 of the PRA report, raises questions about the applicability of the NUREG/CR-6928 data to the US-APWR injection line (accumulator, DVI and RHR/CSS) check valves. The staff notes that the failure to open probability of 1.2E-5 per demand for check valves reported in NUREG/CR-6928 is based on 38,550 demands with zero failures of 729 components over a seven year period (1997 to 2004). These components were from various systems and the check valves in many of these systems operate in different conditions (e.g., in terms of the pressures that keep them closed, the presence of borated water and the length of time between flow testing) than the US-APWR injection line check valves. This argument is also supported by the average demand per year (6.6 demands per component) for the check valve population considered in the NUREG/CR-6928 study. This average demand suggests that the average testing interval for the considered population is much smaller than the two-year flow testing interval (at shutdown) of the US-APWR injection line check valves. Furthermore, the NUREG/CR-6928 study includes a caution statement (Section A.2.12.5) regarding the use of the study results in case of limited system and failure mode data sets. It should, also, be noted that the probability recommended in the Advanced Light Water Reactor (ALWR) Utility Requirements Document for the failure of a check valve to open is 1E-4 per demand (which is about a magnitude higher that the probability used in the US-APWR PRA) and that this probability is based on seven actual failures that occurred before 1990. This issue should also be addressed in conjunction with uncertainties associated with common cause failure (CCF) to open probabilities of injection line check valves (considering CCF of valves both within a system and among systems using same or similar valves). Please respond with a discussion which addresses the staff concerns regarding the modeling of the failure to open of injection line check valves in the PRA.

ANSWER:

Equipments of new built plants are considered to have enhanced reliabilities compared to existing plants. The data of NUREG/CR-6928 are based on recent plant experience and are judged to better represent the reliability of equipments of new built plants rather than failure rates based obtained on experience before 1990. In addition, data analyzed in NUREG/CR-6928 for check valves to open on demand involves 38550 demands, which is more than twice the plant specific data collected in the ALWR Utility Requirements Document.

Unreliability data for check valves used in NUREG/CR-6928 include data from various systems with demands per year ranging from 0.1 to 20. NUREG/CR-6928 is aware of the variation of failure probabilities depending on the demand per year chosen (Section A.1.2), but chooses this range of data to estimate the unreliability of equipments. 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. The 95% upper bound of the failure probability to fail to open, which is 5E-5 /d, is also considered applicable to the check valves since the 24 month test interval of the equipments is within the range of unreliability data used. This value is less than the standby failure probability estimated from the ALWR Utility Requirements Document failure rate. We consider that the ALWR Utility Requirements Document data are conservative for recently operating plants new built plants.

Caution statement in NUREG/CR-6928 (Section A.2.12.5) discusses the use of unreliability derived for specific systems from limited number of data. The generic reliability data the US-APWR applies are supported by a considerable amount of data.

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.

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APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **10/20/2008**

QUESTION NO. : 19-129

The success criteria for the accumulators during a medium LOCA is shown as 2 of 3 accumulators in Section 6A.2.2.2 of the PRA report but as 1 of 3 in Table 6A.2-6 "Fault Tree Common Cause Events." Please clarify and verify the success criteria used in the common cause failure analysis of accumulators for medium LOCAs.

ANSWER:

The success criterion for the accumulators during medium break LOCA is two of three accumulators. The description in Table 6A.2-6 of the PRA technical report (MUAP-0730 R1) is an editorial error and will be fixed in the next revision. The fault tree used for the failure of accumulators during medium break LOCA is "ACC-0LL" as shown in Table 6A.2-8 of the PRA technical report revision1. This fault tree is the same as the fault tree used for large break LOCA (with the same boundary condition used) that considers two of three accumulators as the success criteria for common cause failure.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Table 6A.2-6 "Fault Tree Common Cause Events." In the PRA technical report will be amended in the next PRA technical report update.

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QUESTION NO. : 19-130

An important design feature of the Accumulators is the installation of two pressure indicators and two level indicators in each accumulator with high and low alarms in the main control room. These indicators are tested every two years (at refueling), according to Table 6A.2-5 of the PRA report, and their single component failure probability is relatively high (1.2E-2). The staff could not find any common cause failures (CCF) modeled in the PRA for the accumulator pressure indicators and level indicators. Please explain.

ANSWER:

CCF of sensors of the accumulators will be incorporated in the model during the next PRA update. CCF of sensors across loops will be modeled as well. The impact of this model change on the results is small since leakage from accumulator tanks are independent events with relatively low probabilities. Failures of sensors impact the availability of the accumulator only when it is combined with the failure of the accumulator tank boundaries.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
CCF of sensors of the accumulators will be incorporated in the model during the next PRA update. The impact on the PRA results is small.

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DATE OF RAI ISSUE: **10/20/2008**

QUESTION NO. : 19-131

According to information provided in Section 6A.3.1.4 of the PRA report, the periodic testing of CSS/RHR components is assumed to occur once every 24 months at refueling, except for the full flow line test of the pumps which is assumed to be performed every 48 months. These testing intervals are significantly longer than the ones used in operating nuclear power plants. Therefore, the applicability of operating reactor data to US-APWR must be investigated and justified. Studies, such as the one reported in NUREG/CR-5823 "Analysis of Standby and Demand Failures Modes," October 1992, indicate that testing policies can have a significant impact on component performance since components fail from both "standby" stresses (e.g., corrosion, oxidation and boron precipitation) and "demand" stresses (e.g., vibration and stressing of shafts). For example, the NUREG/CR-5823 study found that motor-operated valves (MOVs) exhibit mostly standby stress failure modes while standby and demand stresses are equally important for emergency diesel generators (EDGs). The longer testing intervals of the US-APWR design as compared to operating reactors, raises questions about the applicability of the demand failure rate data that are obtained from operating reactor experience to the US-APWR design PRA. The staff believes that further investigation of the applicability of the operating experience data to US-APWR is necessary. The impact of any significant design and operational differences to the data used in the PRA should be addressed for all affected systems and components. Please discuss.

ANSWER:

The full flow test will be performed once every 3 months and CS/RHR pump and the valves in the full flow line will be tested. The description in the PRA technical report (MUAP-0730 R1) regarding the test interval of full flow line is an editorial error and will be amended in the next revision.

Motor operated valves of the residual heat removal system and containment spray system that are tested once every 24 months are RHS-MOV-001A(B,C,D), RHS MOV-002A(B,C,D) and RHS- MOV-026A(B,C,D). Other motor operated valves are tested quarterly.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Section 6A.3.1.4 of the PRA technical report will be amended.

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QUESTION NO. : 19-132

It is stated in Section 6A.3.3 “Boundary Conditions and Assumptions” of the PRA report that error of omission to close several valves following testing of the CSS/RHR system has been considered. However, no errors of omission are listed in Table 6A.3-11(basic events) or in Table 6A.3-13 (human errors) and no such basic events appear in the related fault trees. Please explain.

ANSWER:

Valve positions will be check from the main control room after outages or testing. Valves that have been aligned in the wrong position will be detected and fixed to the correct position within a short period of time. The probability of valves to be left in the incorrect position after testing is therefore considered to be very low. Accordingly, human error of omission to close valves are not modeled in the fault trees.

Section 6A.3.3 of the PRA technical report will be amended incorporating this discussion.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
PRA technical report will be amended reflecting the response of this RAI.

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The fault tree analysis of the charging injection (CHI) system (fault tree CHI-VS) indicates that the reliability of the system is dominated by the common cause failure (CCF) of two pairs of motor-operated valves (MOVs) which are tested every 18 months. One pair is the two Volume Control Tank (VCT) stop valves which are open during normal power operation and at least one of them must close for success. The other pair is the charging pump Refueling Water Storage Auxiliary Tank (RWSAT) suction isolation valves which are closed during normal power operation and both must open for success. A failure probability of 1.0E-3 per demand is used for the failure to open as well as for the failure to close of these MOVs. This failure probability was taken from NUREG/CR-6928 "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants" and it is significantly lower than most of the other failure probabilities reported in the literature and documented in Table 7.1-1 of the USAPWR PRA report. The staff notes that the failure to open or to close probability of 1.0E-3 per demand for MOVs reported in NUREG/CR-6928 is mostly based on quarterly testing intervals as opposed to the 18-month testing intervals proposed by the USAPWR design CHI system MOVs. It should, also, be noted that the failure rate recommended in the Advanced Light Water Reactor (ALWR) Utility Requirements Document for the failure of an MOV to open or to close is 1E-5 per hour. If this failure rate were used, the failure probability of an MOV to open or to close would be 6.5E-2 per demand (instead of 1E-3 used in the US-APWR PRA) and the CCF probability would be 1.1E-3 (instead of 4.7E-5 used in the US-APWR PRA). Please provide justification for the failure data used for MOVs in the US-APWR PRA.

ANSWER:

Test interval of the two pair of motor operated valves are one per 24 months. The PRA technical report will be amended to fix this editorial error in the next revision.

Equipments of new built plants are considered to have enhanced reliabilities compared to existing plants. The data of NUREG/CR-6928 are based on recent plant experience and are judged to better represent the reliability of equipments of new built plants rather than other data bases which are based on operating experiences before 1990. The failure probability of valves to operate in NUREG/CR-6928 is supported by a considerable number of operating experiences of over 230000 demands with 244 failures. Failure probability reported in NUREG/CR-6928 was therefore considered to be applicable.

Unreliability data for motor operated valves used in NUREG/CR-6928 includes data from various systems with different values of demands per year ranging from 0.1 to 20. NUREG/CR-6928 is aware of the variation of failure probabilities depending on the demand per year chosen (Section A.1.2), but chooses this range of data to derive the unreliability of equipments. 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 judged that the failure probability can be applied. The 95% upper bound of the failure probability to fail to open is 3E-3 /d is also applicable to the check valves since the 24 month test interval of the equipments is within the range of unreliability data used.

Sensitivity analysis has been performed assuming the failure probability of all motor operated valves to be 3E-3 /d, the 95% upper bound failure probability of motor operated valves to operate. This failure probability was also considered for common cause failure probabilities. The resulting core damage frequency (CDF) was 1.3 E-6 /ry, which is approximately 10% higher than the base case CDF. We consider that uncertainity of the reliability of motor operaed valve has approximately 10% impact on CDF.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
The PRA report will be amended to fix the editorial error regarding the test interval charging injection system valves. The discussion on the basis for choosing the NUREG/CR-6928 will be incorporated in the PRA technical report during the next revision.

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In Table 6A.4-3 of Chapter 6A.4 "Charging Injection System" of the US-APWR PRA document it is stated: "The water of the refueling water auxiliary tank is injected into the cold leg piping through the charging line by the charging pump, provided that the water of the refueling water storage pit is supplied to the tank." In the same Table, the operator action to "Supply water from the RWSP to RWSAT" is stated. However, this operator action is not listed in Table 6A.4-6 "Fault Tree Human Error Events for Charging Injection System" or anywhere else in Chapter 6A.4-3. Please clarify how the operator action needed to supply water from the RWSP to RWSAT, including the failures of any associated hardware, were modeled in the PRA.

ANSWER:

Human error of operators to supply water from RWSP to RWSAT will be incorporated in the "charging injection system" fault tree.

If the water in the refueling water storage auxiliary tank (RWSAT) becomes low, the operator needs to identify the low water level in the RWSAT and initiate RWSAT refill. The operator must perform the following actions.

- Close air operated valve RWS-AOV-022
- Open manual valve RWS-VLV-021
- Open manual valve RWS-VLV-052
- Start either of the refueling water recirculation pump

Human errors of operators to perform these actions will be modeled in the PRA during the next update. In order to assess the impact of this human error, a sensitivity analysis was performed by assuming a conservative human error probability of 0.5 as the failure probability to refill the RWSAT. Resulting increase in CDF is approximately 3%. Taking into account that the actual human error probability for this operator action is expected to be less than 0.5, the impact of incorporating this human error in the PRA model is considered to have small impact on the PRA results.

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 and the PRA model will be revised in the next update. The impact on the PRA results is small.

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APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **10/20/2008**

QUESTION NO. : 19-135

The tag numbers used in Figure 19.1-2 (line diagram of the Charging Injection System) for the components modeled in the PRA are different from those used in the US-APWR Design Control Document (DCD). This major inconsistency, which is also present in several other systems (e.g., the Emergency Feedwater System), complicates the review of the PRA and its proposed applications, such as the implementation of the riskmanaged technical specifications (RMTS) program, and makes the interpretation and communication of the PRA results and insights cumbersome. The staff expects the applicant to remove these inconsistencies throughout the PRA document before the staff's review of the design certification PRA is completed.

ANSWER:

The PRA technical report will be revised to remove inconsistencies of tag numbers with other chapters of DCD. Tables listing the basic events used in the PRA along with tag numbers used in other chapters of the DCD is provided in chapter 6 of the PRA technical report revision 1. These tag numbers will be eventually incorporated in the basic event names and description used in the PRA model as well.

Impact on DCD
Tag numbers consistent with other chapters of the DCD will be incorporated in the documentation.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Tag numbers consistent with other chapters of the DCD will be incorporated in the PRA technical report and the model as a prioritized item for DCD and application review.