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March 12, 2009

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09094

Subject: **Amended MHI's Responses to US-APWR DCD RAI No. 149-1744 Revision 1**

Reference:
1) "Request for Additional Information No. 149-1744 Revision 1, SRP Section: 19-Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19.1," dated January 9, 2009
2) Letter MHI Ref: UAP-HF-09046 from Y. Ogata (MHI) to U.S. NRC, "MHI's Responses to US-APWR DCD RAI No.149-1744," dated February 6, 2009
3) Letter MHI Ref: UAP-HF-09086 from Y. Ogata (MHI) to U.S. NRC, "MHI's Responses to US-APWR DCD RAI No. 149-1744 Revision 1," dated March 10, 2009

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

Enclosed are the second responses to the RAIs contained within Reference 1. In the initial responses submitted with Reference 2, MHI committed to submit responses to19-280, 19-281 and 19-283 within 60 days after RAI issue date.

This letter contains the amended version of the responses to the RAIs contained within Reference 3, which had typographic error regarding RAI question number in its enclosure. The RAI response contained in this amended version involves no technical changes from that of Reference 3.

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.

D081
NRO

Enclosure:

1. "Responses to Request for Additional Information No. 149-1744 Revision 1"

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

Contact Information

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Enclosure 1

UAP-HF-09094
Docket No. 52-021

Responses to Request for Additional Information
No.149-1744 Revision 1

March 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 149-1744 REVISION 1**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-280

The following statement is made in Section 9.3.3 "Type C Human Actions" of Revision 1 of the US-APWR PRA report: "...the time available to complete actions is not estimated at the design certification stage, but an evaluation is performed to assure that identified type C human actions are possible to perform in the time available." Please discuss the nature of the evaluation that was performed to assure that there is adequate time available for operators to complete each identified action. Also, for each identified type C action, list the estimated average time that it will take the operators to complete the action and the assumed available time window, and include a brief discussion on how these times impact the various factors used in calculating the human error probabilities.

ANSWER:

Available times and related discussions for "Type C Human Actions" are listed in the Attachment-1 and 2.

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.

Attachment-1 (At power PRA : Table for RAI 19-280)

Type C Human Action	Time window	Time necessary to complete actions	Remarks
In the case of loss of CCW event, operators will connect the non-essential chilled water system or the fire suppression system to the CCWS in order to cool the charging pump and maintain RCP seal water injection.	1 hour	About 30 minutes (Engineering Judgment)	Occur RCP seal LOCA within 1 hour after loss of RCP seal cooling
If emergency feed water pumps cannot feed water to two intact SGs, operators will attempt to open the cross tie-line of EFW pump discharge line in order to feed water to two more than SGs by one pump.	47 minutes	Few minutes (remotely open from the CCM)	In 5A.3.1 of PRA report, 1 pump is enough for core cooling. But MAAP code has uncertainty. In 5A.3.3 of PRA report, SG dry out time is 47 minutes at loss of feed water.
The CS/RHR System has the function to inject the water from RWSP into the cold leg piping by switching over the CS/RHR pump lines to the cold leg piping if all safety injection systems failed (Alternate core cooling operation). Alternate core cooling operation may be required under conditions where containment protection signal is valid. In such cases, alternate core cooling operation is prioritized over containment spray, because prevention of core damage would have higher priority than prevention of containment vessel rupture.	30 minutes	Few minutes (remotely open from the CCM)	Refer to 5A.1.5 of PRA report. Analysis for alternate core injection during LOCA.
In the case of loss of running train CCW cooling function, with running train CCW flow rate – low signal, CCW pressure – low signal, and running charging pump condition, operators start another stand-by charging pump in order to maintain RCP seal water injection.	1 hour	Few minutes (remotely open from the CCM)	Occur RCP seal LOCA within 1 hour after loss of RCP seal cooling

Type C Human Action	Time window	Time necessary to complete actions	Remarks
When station blackout occurs, with emergency bus voltage – low signal after connecting EGTs, operators connect the alternative ac power with alternate Gas turbines to class 1E bus in order to recovery emergency ac power.	1 hour	About 30 minutes (Engineering Judgment)	Occur RCP seal LOCA within 1 hour after loss of RCP seal cooling
In the case of fail to isolate failed SG, but success to depress enough RCS by secondary side cooling and Safety Depressurization Valve in SGTR event, operators do RCS pressure control in order to prepare to early RHR cooling in order to ensure long term heat removal. (RCS pressure control means stopping SI safety injection and starting charging pump. RCS pressure under SI injection keeps higher for connecting RHR system. Charging pump is back up for failure RHR cooling after stopping SI injection)	Several hours (Engineering Judgment)	Few minutes	In section 2.1.3 of the technical report for Level 3, it takes about 27 hours to uncover the core due to leakage from failed SG in the case of isolated failed SG.
In the case of above, if operators fail to move RHR cooling after SI injection control, operators start to bleed and feed operation. Operators open safety depressurization valve and start the safety injection pump in order to ensure long term heat removal.	Same as above	Same as above	Same as above
When the main steam isolation valve fail to close in SGTR event, with status signal of this valve, operators try to close this valve in order to stop leakage of RCS coolant from the failed SG.	Same as above	Same as above	Same as above
When the main steam isolation valve fail to close in SGTR event, with SG pressure indication after above operation, operators close turbine bypass stop valves in order to stop leakage of RCS coolant from the failed SG.	Same as above	About 30 minutes (Engineering Judgment)	Same as above

Type C Human Action	Time window	Time necessary to complete actions	Remarks
In the case of loss of failed SG isolation function in SGTR event, with SG pressure indication after above operation, operators open main steam depressurization valve of intact SG loop in order to promote SG heat removal and to depressurize RCS and move to cool down and recirculation operation.	Same as above	Few minutes	Same as above
In the case of loss of secondary side cooling function by emergency feedwater system in transient events including turbine trip, load loss event etc., with emergency feedwater pump flow rate, operators start to recover main feedwater system in order to maintain secondary side cooling.	47 minutes	Few minutes (remotely open from the CCM)	In 5A.3.1 of PRA report, 1 pump is enough for core cooling. But MAAP code has uncertainty. In 5A.3.3 of PRA report, SG dry out time is 47 minutes at loss of feed water.
In the case of loss of contain spray system function, with contain spray pump flow rate and CV pressure-high signal, operators provide preparation for CV natural recirculation cooling operation in order to remove heat from CV. This preparation contains CCW pressurization with N2 gas, disconnection heat load of non-safety chiller and CRDM etc. and connection to containment fan cooler units.	1 hour	About 30 minutes (Engineering Judgment)	Occur RCP seal LOCA within 1 hour after loss of RCP seal cooling
When any two EFW pumps that commonly utilize at EFW pit have failed, operators supply water to operating EFW pumps from alternate EFW pit or demineralized water storage pit in order to ensure the water source.	4 hours	About one hour (Engineering Judgment)	The total inventory of two EFW pits is required for 8 hours cooling for hot shutdown in design.
When the CV isolation signal fail to automatically actuate, with CV pressure abnormally high signal, operators manually actuate the CV isolation signal in order to remove heat from the containment vessel.	Several hours (Engineering Judgment)	Few seconds	There are generally several hours between CV pressure abnormally high signal point and CV yield pressure point.

Type C Human Action	Time window	Time necessary to complete actions	Remarks
When the ESF actuation signal fail to automatically actuate, with pressurizer pressure-abnormally low signal or CV pressure abnormally high signal or ECCS actuation failure alarm, operators manually actuate the ESF actuation signal in order to recover the ESF actuation signal.	2 hours	Few seconds	Table 14-3 tells there are 2.26hours before core uncovered even in Medium LOCA with following condition. No ECCS except accumulator, but emergency feed water system is actuated.
When the CCW header tie-line isolation valves fail to automatically close with specific signals which contain SI signal plus UV signal, P signal, and surge tank level low signal, operators manually close these valves in order to separate CCW header.	Several hours	Few seconds	A CCW pump will not cause cavitations behavior under the condition of one pump with two ways.

Attachment-2 (LPSD PRA : Table for RAI 19-280)

Type C Human Action	Time window	Time necessary to complete actions	Remarks
In the case of loss of CCW cooling function, with CCW flow rate – low or CCW pressure – low, operators connect the fire suppression system to the CCWS and start the fire suppression pump in order to cool the charging pump and maintain injection to RCS.	1 hour	About 40 minutes (Engineering Judgment)	Time window is taken from previous PRA studies and experience with mid-loop operation and applied to all POSs conservatively, supported by allowable time until uncover of the reactor core calculated by MAAP.
In the case of loss of decay heat removal functions by RHRS and SGs, and loss of injection by SI pump, with RCS temperature – high or RCS water level – low, and RWSAT water level – low, operators open the injection path from RWSP to RWSAT and start the refueling water recirculation pump in order to maintain RCS water level.	1 hour	About 40 minutes (Engineering Judgment)	same as above
When RCS makeup is required during charging pump being standby, with RCS water level – low, operators start the charging pump in order to recover water level in the RCS.	1 hour	About 10 minutes (Engineering Judgment)	same as above
When station blackout occurs, with emergency bus voltage – low after connecting EGTs, operators connect the alternative ac power with alternate gas turbines to class 1E bus in order to recover emergency ac power.	1 hour	About 30 minutes (Engineering Judgment)	same as above
In the case of loss of decay heat removal functions by RHRS and SGs, with RCS temperature – high or RCS water level – low, operators start the safety injection pump in order to maintain RCS water level.	1 hour	About 30 minutes (Engineering Judgment)	same as above

Type C Human Action	Time window	Time necessary to complete actions	Remarks
When LOCA occurs, with RCS water level – low, operators close the RHR hot legs suction isolation valves in order to stop leakage of RCS coolant from RHRS where LOCA occurs.	1 hour	About 10 minutes (Engineering Judgment)	same as above
When over-draining occurs and the automatic isolation valve fails, with RCS water level – low, operators close the valve on the letdown line in order to stop draining.	1 hour	About 10 minutes (Engineering Judgment)	same as above
In the case of failure of running RHRS, with RHR flow rate – low, operators open the valves on the standby RHR suction line and discharge line and start the standby RHR pump in order to maintain RHR operating.	1 hour	About 10 minutes (Engineering Judgment)	same as above
In the case of leakage of the RWSP water from HHIS piping, CSS/RHR piping or refueling water storage system piping, with drain sump water level – abnormally high, operators close the RWSP suction isolation valves respectively in order to prevent leakage of RWSP water from failed piping.	1 hour	About 10 minutes (Engineering Judgment)	same as above
In the case of failure of running CCWS, with CCW flow rate – low, operators start the standby CCW pump in order to maintain CCWS operating.	1 hour	About 10 minutes (Engineering Judgment)	same as above
In the case of failure of running ESWS, with CCW flow rate – low, operators start the standby ESW pump in order to maintain ESWS operating.	1 hour	About 10 minutes (Engineering Judgment)	same as above

Type C Human Action	Time window	Time necessary to complete actions	Remarks
When ESW strainer plugs up, with ESW pump pressure – normal, ESW flow rate – low and differential pressure – significant, operators switch from plugged strainer to standby strainer in order to maintain ESWS operating.	1 hour	About 40 minutes (Engineering Judgment)	same as above
In the case of loss of decay heat removal functions from RHRS and SGs, and loss of injection by SI pump and charging pump, with RCS temperature – high or RCS water level – low, and SFP water level – low, operators open flow path from RWSP to SFP and the gravity injection path from SFP to RCS cold leg, then start the refueling water recirculation pump and supply water to RCS in order to maintain RCS water level.	1 hour	About 40 minutes (Engineering Judgment)	same as above
In the case of loss of decay heat removal functions from RHR, with RCS temperature – high or RCS water level – low, operators feed water to SGs by motor-driven EFW pump and open safety depressurization valve in order to remove decay heat from RCS.	1 hour	About 30 minutes (Engineering Judgment)	same as above
In the case of failure of feed or steam line associated with available motor-driven EFW pump during secondary side cooling, operators open the EFW tie-line valves in order to feed water to multiple SGs.	1 hour	About 30 minutes (Engineering Judgment)	same as above

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 149-1744 REVISION 1**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-281

The following statement is made in Section 9.3.3 "Type C Human Actions" of Revision 1 of the US-APWR PRA report: "..... human error probabilities can be updated as more specific US-APWR design and updated thermal-hydraulic analyses become available." Per RG 1.206, the detailed design will have to be consistent with all important assumptions, regarding design and operational features and related characteristics, made in the design certification PRA. These assumptions need to be identified at the design certification phase and be verified through appropriate requirements, such as ITAAC and COL action items, to ensure that they will remain valid for the as-to-be-built, as-to-be-operated plant. The staff expects MHI to perform a systematic search to identify a more complete list of important assumptions made in the human reliability analysis (HRA) as well as in all chapters of the US-APWR PRA and, for each of such assumptions, indicate how it will be ensured that they will remain valid for the as-to-be built, as-to-be-operated plant. This information should be documented in Table 19.1-115 (Section 19.1.7) of the design control document (DCD).

ANSWER:

Responses to this RAI will be involved in the responses to the RAI 19-207. Information for "Type C Human Actions" is documented in the DCD Table 19.1-115.

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

3/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 149-1744 REVISION 1**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-283

Please address the following questions regarding the sensitivity analysis discussed inSection 18.3 of the US-APWR PRA report:

(a) It was estimated that in case one safety train is out of service throughout the year (Sensitivity Case 1-1), the core damage frequency (CDF) from internal events at power operation would increase about four times (from 1.2E-6/yr to about 5E-6/yr). The risk insight from this sensitivity case appears to be included in the following statement: "This is small in consideration that trains are not normally out of severe [service]." In this statement it is implicitly recognized that a four times increase in CDF is significant but it is assumed that safety trains, normally, are not going to be out of service for extended periods, even though the technical specifications (TS) allow it. However, it is not stated how this PRA assumption about a future plant's operation will remain valid for the as-tobe-built, as-to-be-operated plant (see RG 1.206, Section C.I.19.2 item C on results and insights, and Section 19.1.7.1 of Appendix A on "PRA input to design programs and processes.") Would all safety trains be in the D-RAP program and each safety train be expected to meet appropriate availability goals, as required by the Maintenance Rule? This is an example of how the integrated results and insights from the importance, sensitivity and uncertainty analysis can be used together with important "assumptions" made in the PRA about design and operational features, in order to identify specific design certification requirements to ensure that these assumptions will remain valid for the as-to-be-built, as-to-be-operated plant. A systematic search is needed to identify such important "assumptions" and discuss how these assumptions will remain valid for the as-to-be-built, as-to-be-operated plant. This discussion often will include a reference to already proposed design material and certification requirements (e.g., crossreferences to certified material in other DCD chapters, ITAAC, TS, D-RAP and COL action items) or will identify new design certification requirements, if necessary. This kind of information should be included in Section 19.1.7 (Table 19.1-115) of the DCD.

(b) For Sensitivity Cases 1-2, 1-3 and 1-4, the incremental conditional core damage probability (ICCDP) was assessed, starting with the plant configuration of Case 1-1 (one safety train out throughout the year), and assuming that a second safety train of a single safety system becomes unavailable for a certain time interval. A more detailed discussion is needed about the objectives of these sensitivity cases and what is the basis of the time intervals assumed in assessing the various ICCDP values. Could such sensitivity cases be interpreted, and expanded if needed, to gain insights about available margins in the implementation of risk managed technical specifications (RMTS) initiatives, such as initiative 4b (risk managed completion times)? In your discussion please include the reason for not considering the "zero maintenance" plant configuration in estimating the ICCDP values.

(c) The results of Sensitivity Cases 2-1 and 2-2 indicate that the operators are assumed to be very effective in mitigating accidents (the CDF without operator actions would increase 1400 times while if the operators never failed the CDF would decrease only 2.5 times). This risk insight underlines the importance of a systematic search to identify assumptions made in the human reliability analysis (HRA) and, for each of these assumptions, indicate how it will be ensured that they will remain valid for the as-to-be built, as-to-be-operated plant. This information should be documented in Table 19.1-115 (Section 19.1.7) of the design control document (DCD).

(d) Sensitivity Case 3-3 involves the common cause failure of all sump screens due to clogging. It is stated that "...the probability of all four sump screens to clog at large LOCA has been assumed to be 0.0625 ..." Please explain the basis for this probability.

(e) Sensitivity Case 3-4 assumes that "all application software for digital systems, excluding that of the alternate ac power (AAC) system, is dependent and has no diversity." This statement appears to imply that it was also assumed that the application software of the diverse actuation system (DAS) is not diverse. Please verify and state the important features and characteristics of the AAC power application software that make it diverse compared to all other application software. The "assumption" in the PRA about these "features and characteristics" that make the AAC application software diverse should remain valid for the as-to-be-built, as-to-be-operated plant. Therefore, this information should be included in Section 19.1.7 (Table 19.1-115) of the DCD.

(f) MHI should consider, if necessary, the performance of additional studies to investigate the sensitivity of PRA results and insights to some of the issues raised by the staff through the RAI process (e.g., failure rates and human error probabilities).

ANSWER:

Answer to (a)

Sensitivity case 1-1 show that CDF will increase approximately four times when one safety train is out of service throughout the year. Insight from this sensitivity analysis are that availabilities of all trains of each safety systems need to be adequately controlled by configuration risk management program as required by the maintenance even though the LCO for the safety systems is three trains.

Answer to (b)

Sensitivity cases 1-2, 1-3, and 1-4 were not assuming the CT calculation of initiative 4b but the CDF results obtained from the combination of component outages can be use as reference information. The method used to calculate ICCDP does not strictly meet that of R.G.1.177 and the analysis was intended to seek the rough estimation of the sensitivity of component outages.
The time intervals assumed in assessing the various ICCDP values are based on the completion time in the generic technical specification of the US-APWR.

Sensitivity analysis to gain insights related to risk informed technical specification initiative 4b is planned to be submitted in response to RAI #151 question 19-288.

Answer to (c)
Results of sensitivity case 2-1 and 2-2 indicate that operator actions have large impact on risk. Risk important operator actions will be included in table 19.1-115 of section 19.1.7. Operator actions included in table 19.1-115 are shown in response to RAI #138 question 19-207.

Answer to (d)
Sump screens are designed to prevent clogging and the probability of common cause failure of sump screen clogging is considered to be very low. In sensitivity case 3-3, the probability of all four sump screen to clog was estimated assuming that probability of each sump screen to clog is 0.5, which is a value postulating no available information on the probability of the failure mode. Based on this value, the probability of all sump screens to clog during large LOCA was estimated to be 0.0625, which is simply 0.5 to the forth power.

Answer to (e)
DAS is hard wired and is diverse from postulated software failures of the safety system. The alternate ac power (AAC) sources are initiated by the non-safety digital I&C system and they are independent from the digital I&C system of the safety systems. Therefore, application software used to control the AAC is independent from the application software used to control the Class 1E gas turbine generators and common cause failure between these application software is unlikely to occur.

Answer to (f)
Sensitivity studies of issues raised by the staff through the RAI process will be performed and included in the DCD and/or PRA report as necessary. PRA Insights will be included in Section 19.1.7 of DCD as well.
Example of sensitivity studies additionally included in the PRA report is the sensitivity analysis of software failure probabilities raise by RAI#25. Additional sensitivity study on reliability of advanced accumulators has been performed based on the issue raised in question 19-285 of RAI#151.

Impact on DCD
There is no impact on DCD.
The following assumptions and insights will be included in Section 19.1.7 (Table 19.1-115) of the DCD.

- All trains of each safety systems will be adequately controlled by configuration risk management program.
- DAS is hard wired and is diverse from software failures of the safety Digital I&C system.
- AAC is initiated by non-safety digital I&C system and is independent from the application software used to initiate the Class 1E gas turbine generators.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.