MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

September 18, 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-08196

Subject: MHI's Responses to US-APWR DCD RAI No.56-999

References: 1) "Request for Additional Information No. 56-999 Revision 0, SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: PRA," dated August 21, 2008

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Responses to Request for Additional Information No. 56-999 Revision 0".

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

As indicated in the enclosed materials, this document 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 the submittals. His contact information is below.

Sincerely,

Y. Ogata

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

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No.56-999 Revision 0 (proprietary)

3. Responses to Request for Additional Information No.56-999 Revision 0 (non-proprietary)

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

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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-08196

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 "Responses to Request for Additional Information No.56-999 Revision 0" dated September 2008, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing 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 design and methodology developed by MHI for performing the design of the US-APWR reactor.

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 development of

methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by benefits of 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 18th day of September 2008.

Y. Ogata

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

Enclosure 3

UAP-HF-08196
Docket Number 52-021

Responses to Request for Additional Information No.56-999
Revision 0

September, 2008
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/18/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.56-999 REVISION 0**

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

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **8/21/2008**

QUESTION NO. : 19-106

More detailed, specific and clear information about the transient progression (plant response) is needed in the "Event Description" sections in Chapter 3 of the PRA report (Sections 3.2.1.1 to 3.2.17.1). For example, the steam generator tube rupture (SGTR) event described in 3.2.5.1 is not clear or detailed enough for the staff to understand how the complex transient progression of a SGTR accident and its mitigation was modeled in the PRA. It is stated that "MSIV and turbine by-pass valves will be automatically closed by CV isolation signal." It is not clear whether the CV isolation signal closes automatically all MSIVs and turbine by-pass valves (TBVs) or just those associated with the faulted SG line. If the latter is true, please list the design features used in conjunction with the CV isolation signal to isolate the faulted SG. Also, it is stated that "the main steam relief valve, main steam safety valves are required to re-close to isolate failed SG." Since the re-closure of the main steam safety valves (MSSVs) depends on the RCS primary pressure and the primary to secondary leak rate, a discussion is needed to explain how this is achieved (e.g., systems used, timing of events, operator actions, success criteria and assumptions made) under the various conditions encountered in the expected accident sequences (e.g., failure of TBVs).

ANSWER:

"Event Description" sections in Chapter 3 will be revised in revision 1 of the PRA report to include more detailed, specific and clear information about the transient progression. Description of "CV isolation signal" in the SGTR event is an editorial error and this description will be amended. Example of the amendment is show in attachment A.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

The PRA technical report will be revised.

Attachment A for Question 19-106

Example of the amendment of the Technical Report, "US-APWR Probabilistic Risk Assessment", MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/18/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.56-999 REVISION 0**

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

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **8/21/2008**

QUESTION NO. : 19-107

Note (1) in Table 3.2.5.3-1 "Steam Generator Tube Rupture Event Success Criteria" of the PRA report states: "SG isolation utilizing the following valves but not all: EFW isolation valves, main steam relief valve, main steam relief valve stop valve, main steam dump valve, MSIV, turbine bypass valve, and main steam relief valve." This statement does not appear to be in complete agreement with the definition of the top event SGI (Ruptured SG Isolation). SGI is defined (page 3-21) as follows: "The failure of isolation of failed SG is caused by the main steam relief valves fail to close, the turbine by-pass valve fail to close, the main steam safety valve fail to close." Please clarify by stating which valves are utilized to isolate the ruptured steam generator, what assumptions are made and what operator actions are required.

ANSWER:

The following events or series of events are considered to result in failure of "SGI (Ruptured SG isolation)".

- Main steam line isolation fails and the actuated (opened) turbine bypass valves fail to close, and manual closure of turbine bypass isolation valves also fail.
- The main steam relief valve fails to re-close after actuation and the associated stop valve fails to isolate the main steam relief valve
- The main steam safety valve fail to re-close after actuation
- Pipe rupture in the main steam line cause by water hammer as a consequence of failure to isolate emergency feedwater.

In order to succeed SGI, event or series of event described above should not occur.

The possibility of the turbine by-pass valve, main steam relief valve, and main steam safety valves to actuate after turbine trip is conservatively assumed as follows:

- Both turbine bypass valve and main steam relief valves will open
- If either the turbine bypass valve or the main steam relief valves does not open, main steam safety valves will open.

The following are operator actions required.

- Operator action to close main steam isolation valve
- Operator action to close turbine bypass isolation valve in case turbine bypass valve fails to re-close
- Operator action to close the main steam relief valve stop valve in case main steam relief valve fails to re-close

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/18/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.56-999 REVISION 0**

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

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **8/21/2008**

QUESTION NO. : 19-108

The US-APWR SGTR event tree does not explicitly model some potential plant responses and associated human actions that are usually considered in operating reactor PRAs. Examples are: (1) Lifting of main steam safety valve (MSSV) on each SG due to secondary pressure spikes following turbine trip but prior to the opening of the turbine bypass valves (TBVs); (2) Operator actions to depressurize the RCS below the lower SG relief valve setpoint before the faulted SG can be isolated (although in the USAPWR the non-safety air-operated MSRV can be isolated by valve MOV 523X, this could cause one or more MSSVs to open if the RCS pressure remains above their setpoints). It appears that the US-APWR PRA assumes (1) no MSRV or MSSV lift occurs unless at least one of the associated TBVs fails to open, and (2) the operation of the EFW system (2 pumps to 2 intact SGs) is adequate to depressurize the RCS below the lower SG relief or safety valve setpoint and keep the core covered with water (even if the high head injection fails), thus no operator action is needed before the faulted SG can be isolated. Please verify these assumptions, discuss their basis, and list any important design and operational features of the US-APWR design that improve the response to SGTR accidents as compared to operating PWR designs.

ANSWER:

(1) Valves of main steam line during turbine trip are modeled as described below.

a. Even in cases where the TBVs open, there is a possibility that the MSRV may open. The PRA therefore conservatively assumes that the MSRV will open even if TBVs succeeds to open.

b. If the TBVs and the MSRV both open, the secondary pressure will not exceed further more to cause MSSV lift. The PRA therefore assumes that the MSSV lift open only when either the TBVs or MSRV fail to open.

(2) As discussed in response to RAI#56 question 19-106, US-APWR requires operator actions to depressurize the RCS to an equivalent pressure with the secondary side of the ruptured SG in order to stop leakage from the RCS, which is also considered in operating PWRs. Since there is considerable time available for operators to stop leakage from the RCS in the sense of core damage prevention, and that there are many alternate measures that can be performed, it is assumed that the probability of core damage due to failure of these operator actions is very low. Operator action to equalize the RCS pressure with the secondary side pressure after isolation of the faulted SG is therefore not modeled in the sequences which the faulted SG is successfully isolated. Operator action to equalize the pressure is assumed to succeed in those sequences.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/18/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.56-999 REVISION 0**

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

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **8/21/2008**

QUESTION NO. : 19-109

In the US-APWR PRA it is assumed that the RCS pressure does not reach the pressurizer safety valve (PSV) lift setpoint and, therefore, no failure to re-close is modeled in most event trees (except for total or partial loss of component cooling water). The same assumption is made regarding accident sequences that could challenge the secondary side main steam safety valves (MSSVs). Please explain the basis for not modeling failure to re-close of PSVs and MSSVs in event trees other than LOCCW and PLOCW.

ANSWER:

Failure of pressurizer safety valve to re-close after a transient event accompanying RCS pressure increase is categorized as small pipe break LOCA (SLOCA) event. The initiating event frequency of "Stuck Open Safety/Relief Valve" reported in NUREG/CR-6928 is considered and added in the SLOCA event frequency of US-APWR. Additionally, no relief valves that automatically actuate upon detection of high RCS pressure are installed in the US-APWR. Consideration of "Stuck Open Safety/Relief Valve" frequency of NUREG/CR-6928 in the SLOCA frequency is therefore conservative.

Failure of pressurizer safety valve to re-close can occur during loss of support systems events such as loss of offsite power (LOOP), loss of CCW (LOCCW) and partial loss of CCW (POCCW). If pressurizer safety valve sticks open during such events and loss of coolant occurs, mitigation of the event can be challenging in comparison to SLOCA event, because the reliability of mitigation functions are lower than the case of SLOCA event. Stuck open safety valve event during loss of support system events are therefore separately modeled from the SLOCA event and respectively address in the LOOP, LOCCW and POCCW event trees.

If main steam relief valves fail to re-close, associated line of the secondary side system will be depressurized and cause reactor coolant system (RCS) cooldown. Moreover, there is a possibility that adequate heat removal by the SG associated with the faulted secondary side system line cannot be performed. RCS cooldown during this event is however considered not to have serious impact from the perspective of core damage. The frequency of scenarios that involve an event that causes main steam safety valve to lift and is followed by the failure of the safety valve to re-close is considerably lower than the frequency of main steam line break event. Failure of main steam safety valves to re-close is therefore not modeled in the event tree except for steam generator tube rupture event.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/18/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.56-999 REVISION 0**

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

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **8/21/2008**

QUESTION NO. : 19-110

Please explain and address the following information in the PRA documentation in Chapter 3 (event trees) and Chapter 6 (system analysis):
(a) MLOCA and SLOCA event tree top event HIB (High Head Injection System) success criteria: 1 of 3 in Chapter 3 and 2 of 3 in Chapter 6.
(b) The top event BLA stands for "bleed" for most event trees (SLOCA, VSLOCA, SGTR, SLBO, SLBI, FWLB, TRANS, PLOCW and LOOP) and Tables 3.2.1.2-1 to 3.2.17.2-1 indicate that event BLA has as input event (fault tree top event) the event FAB. However, in Chapter 6 (Section 6A.1.4.1) the fault tree HPI-FAB is defined for "feed and bleed" and FAB is not defined (a fault tree labeled FAB is shown in Appendix 6A.1.B just for "bleed"). Also, for event trees where high head injection is not possible without "bleed" should not the event BLA precede the event for high head injection? In addition, for SGTR, event BLA has an additional input event named FAB-SG-DP2 which is not defined.
(c) The top event CHI (Charging Injection System) in the VSLOCA event tree has input event CHI-VS (Table 3.2.4.2-1). However, Section 6A.4.4.1 states that the fault tree of the charging injection system is CHI and Table 6A.4-2 indicates that the "System Identifier" is HIA.
(d) Several fault tree top events for the Alternate Containment Cooling System (e.g. NCC-VS-DP2, NCC-VS-DP3 and NCC-VS-DP4) are listed in Tables of Chapter 3 without any definitions in either Chapter 3 or Chapter 6.
(e) Table 3.2.5.2-1 indicates that top event SRB in the SGTR event tree has MSPSGDP1 as the input event (fault tree). However, Table 6A.6-1 defines fault tree MSP-SG but not MSP-SG-DP1. Also, it is not clear what the success criteria for SRB are: 2 of 3 EFW pumps according to Table 3.2.5.3-1; 1 of 4 EFW pumps according to Note (2) of Table 3.2.5.3-1; 2 out of 4 EFW pumps to 3 intact SGs according to Table 6A.5-2; 2 of 3 EFW pumps according to Table 6A.5-5; and 1 of 3 EFW pumps according to the MSPSG fault tree (page 6A.6.B-42).
(f) Table 3.2.5.2-1 indicates that top event HT in the SGTR event tree has HIT-SG-DP1 as the input event (fault tree). However, fault tree HIT-SG-DP1 is not defined anywhere in Chapter 3 and 6. It appears that the fault tree HIT-SG-DP1 was renamed to HIT (included in Attachment 6A.14.6.B) without reference or link.
(g) The success criterion for Alternate Containment Cooling event FNA8 (SGTR event tree) is stated in Chapter 3 (Table 3.2.5.3-1) as supplying water to the containment vessel recirculation unit by 1 of 4 CCW pumps. However, in Chapter 6 of the PRA (Section 6A.14.1.2.2) it is stated

that 2 of 4 CCW pumps are required.
(h) Table 3.2.7.2-1 indicates that top event EFB in the SLBO event tree has EFW-SLBO as the input event (fault tree). However, fault tree EFW-SLBO is not defined anywhere in Chapter 3 and 6. It appears that the fault tree EFW-SLBO was renamed to EFW-SB (included in Section 6A.5.4.1) without reference or link. Also, it is not clear what the success criteria for top events EFB (applied to the SLBO event tree) and EFD (applied to the SLBI event tree) are: 1 of 4 EFW pumps to 2 of 3 intact SGs according to Note (1) of Tables 3.2.7.3-1 and 3.2.8.3-1 but 2 of 4 EFW pumps are required according to Table 6A.5-2.
(i) The top event designator EFD for "Emergency Feedwater system" appears in event trees SLBI and FWLB, including Tables 3.2.8.2-1 and 3.2.9.2-1. However, in Section 3.2.8.2 (page 3-31) the designator for the same event is EFB. Also, the success criterion for EFD (or EFB?) in event tree FWLB is not clear. Note (1) of Table 3.2.9.3-1 indicates that supply of water by 1 out of 3 EFW pumps to 2 out of 3 intact SGs is required while Table 6A.5-2 indicates that supply by 2 out of 4 EFW pumps to 2 out of 3 intact SGs is required.
(j) The top event MSI (Main steam line isolation) in the FWLB event tree has input event MSR-I-00. However, in Chapter 6 (Section 6A.8.4.1), fault tree MSR-I-00 is not given as been applicable to FWLB. Also, please explain why fault tree MSR-I-00 (main steam line isolation failure) is applicable to both SLBI and FWLB accident sequences.
(k) The top event FBA1 (Bleed and Feed) in the TRANS event tree has input event HPIFAB-TR-DP1 (Table 3.2.10.2-1). However, event HPI-FAB-TR- DP1 is not defined and Section 6A.1.4.1 states that the fault tree for "bleed and feed" is HPI-FAB.
(l) The top event CRB2 (Alternate core injection) in the PLOCW event tree has three input events (RSS-RHR-SL, RSS-RHR-SLPL-DP2, and RSS-RHR-SLPL-DP3 in Table 3.2.13.2-1). However, the last two input events are not defined.
(m) Top events ROD (reactor trip control rod) and SCF (reactor trip digital system) in the ATWS event tree have input events that are not discussed in Chapter 6.

ANSWER:

Information of event tree headings and associated fault trees and success criteria are provided in the attachment of response to RAI#40, question 19-100. These information will be incorporated in revision 1 of the PRA technical report.

Other information requested is shown below:

Response to question (j)

In the event of FWLB, unlimited steam release can occur depending on the location of the break and result in degradation of heat removal function of SGs. It is therefore conservatively assumed that main steam isolation is necessary in the event of FWLB as it is in the event of SLBI.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

The PRA technical report will be revised.