MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

April 10, 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-09163

Subject: MHI's Second Responses to US-APWR DCD RAI No.197-1800 Revision 0

References:
1) "Request for Additional Information No. 197-1800 Revision 0, SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19," dated February 9, 2009.
2) Letter MHI Ref: UAP-HF-09085 from Y. Ogata (MHI) to the U.S. NRC, "MHI's Responses to US-APWR DCD RAI No. 197-1800," dated March 11, 2009.

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

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 to #19-304 by 10th of April 2009.

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. Second Responses to Request for Additional Information No. 197-1800 Revision 0 (proprietary version)

3. Second Responses to Request for Additional Information No. 197-1800 Revision 0 (non-proprietary version)

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

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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-09163

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 "Second Responses to Request for Additional Information No. 197-1800 Revision 0" dated April 2009, 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 10th day of April 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09163
Docket Number 52-021

Second Responses to Request for Additional Information No. 197-1800 Revision 0

April 2009
(Proprietary Information Excluded)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

4/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.197-1800 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **2/9/2009**

QUESTION NO. : 19-304

Please describe and justify the criteria that would be used to manually depressurize the reactor coolant system during a high-pressure severe accident.

For sequences AM001 and AM002 on the effectiveness of RCS depressurization for a small-break LOCA and a main steam line break outside containment, respectively, leading to a severe accident, RCS depressurization was enabled 10 minutes after core melt. What are your definitions of core damage and core melt? At the times of core damage and core melt, what are the core outlet temperatures, the amounts of hydrogen generated, and the damage fractions for the hot leg and steam generator tubes for each sequence? For each sequence, please provide plots, from the start of the sequence until the time of vessel failure, of the core-to-upper plenum natural circulation, the natural circulation between the upper plenum and the steam generators, and the countercurrent natural circulation flow rates in the hot legs and in the steam generators.

Please verify that the study of the effectiveness of RCS depressurization features in Section 15.6.2 of the PRA is based on these two sequences. Note that, in Section 15.6.2, it is stated that the depressurization valves are assumed to be manually opened 10 minutes after core damage.

Since the large release frequency (LRF) in existing reactors is dominated by high pressure sequences where the secondary side has been depressurized, please provide analyses of variations of the two cases described above, in which one or more steam generators have been depressurized prior to the onset of zircaloy oxidation. Provide the same results requested above for comparison, and report when the hot leg and/or the steam generator tubes would fail from creep rupture.

ANSWER:

For actual plant operation of the US-APWR, the core outlet temperature and the containment dose rate are utilized as the criterion to implement manual depressurization of the reactor coolant system. If the core outlet temperature exceeds a certain degree (to be evaluated) as well as the containment dose rate exceeds a certain level (to be evaluated), it should be recognized as core is damaged and necessary to take countermeasures immediately. In general it is considered that these two physical properties tend to rise very rapidly in the event of core damage.

On the other hand, for the US-APWR severe accident analysis, "Core damage" is used as a generic term for the process of core disruption, but does not have a clear definition. To the contrary, "core melt" is clearly defined as a point of time in the accident progression analyses when the maximum core temperature exceeds 2500K (4041°F). In the US-APWR accident progression analyses, "cladding rupture" is also defined as a point of time when the maximum cladding temperature exceeds 1000K (1341°F) at which fission product release from the core starts. In the DCD and PRA report (MUAP-07030(R1)), "core damage" represents "core melt" when it is used as an analysis condition, and also it is used as a generic term when used as a phase of accident progression. Subsection 15.6.2.1 in the PRA report states that "depressurization valve open 10 minutes after core damage." This is the same meaning of "depressurization valve open 10 minutes after core melt." In the accident progression analyses, it is assumed that core damage is detected at the time of "core melt."

RCS depressurization by manual opening of the depressurization valve (DV) is assumed implemented after core damage is detected under a high RCS pressure sequence. In the following discussions, the criteria for RCS depressurization are justified by showing the analysis results for accident sequences AM001 and AM002 which correspond to Case 1 and Case 2 in subsection 15.6.2 of PRA report, respectively.

Variables relevant to the effectiveness of RCS depressurization for the accident sequences AM001 and AM002 are shown in the following figures.

- Accident Sequence AM001 (RCP Seal LOCA & RCS Depressurization)
  : Figures 19-304-1(1) through 19-304-1(6)

- Accident Sequence AM002 (Main Steam Line Break (MSLB) & RCS Depressurization)
  : Figures 19-304-2(1) through 19-304-2(6).

In these figures, the timings evaluated as "cladding rupture" and "core melt" are plotted. The timing of "vessel failure" is also plotted in some figures of them. In these figures, upper plenum gas temperature is plotted as the substitution of core outlet temperature. For each sequence, the natural circulation flow rate of gas in the RCS is increased and then the upper plenum gas temperature goes up rapidly at the point of "cladding rupture" or "core melt." Hydrogen is also generated especially after "core melt," resulting in further RCS gas temperature rise. As stated above, the upper plenum gas temperature tends to rise and dose rate in the containment is considered to be increased significantly because fission products have been released to the containment at the point of "core damage." It is therefore concluded reasonable that core damage can be detected at the point of "core melt" defined in the accident progression analysis.

In the accident sequence AM001 which is classified as a small pipe break LOCA, RCS pressure varies from medium to low because of manual opening of DV 10 minutes after "core melt." The creep damage fraction for AM001 remains small. On the other hand, in the accident sequence AM002, hot leg creep rupture occurs approximately simultaneously with RCS depressurization. Regarding the SG tubes, the creep damage fractions are evaluated to remain small, even if the secondary side pressure is low because of MSLB initiated accident sequence. The operator actions for RCS depressurization can be completed immediately after detection of core damage because manual opening of DV is implemented in the main control room. The operation time of 10 minutes assumed for these analyses is therefore considered rather conservative. In these analyses, only the timing of hot leg creep rupture or SG tube creep rupture is evaluated and the subsequent physical phenomena such as release from the rupture are ignored to set the area of rupture as zero.

The behaviors of RCS pressure for these sequences are shown in Figures 14-140 and 14-148 of the

PRA report. Since the RCS pressure at the timing of vessel failure is sufficiently low in either sequence, there is little possibility of high pressure melt ejection if RCS depressurization is implemented.

The discussions on the criteria and effectiveness for RCS depressurization described in subsection 15.6.2 of the PRA report are therefore considered appropriate.

In the PRA report, the MSLB initiated accident sequence without RCS depressurization (AP201) is also evaluated. In the analysis, only the timing of hot leg creep rupture or SG tube creep rupture is evaluated in a same manner with the LOCA sequence discussed above.

Variables for accident sequence AP201 are shown in the following figures.

- Accident Sequence AP201 (MSLB)
  : Figures 19-304-3(1) through 19-304-3(6)

The accident progression from the timing of "cladding rupture" to "core melt" is similar to that of the accident sequence AM002. In this accident sequence, natural circulation of gas in the RCS continues after "core melt" different from the sequence AM002, because RCS depressurization fails. In this MAAP analysis, the creep damage fraction for hot legs reaches one, faster than that for SG tubes. It is therefore considered that hot leg creep rupture possibly occurs even in an accident sequence where the secondary side pressure is low.

Since there are inherently high uncertainties in the models of induced SGTR, it is also possible that the creep damage fraction for SG tubes reaches one faster than that for hot legs. In the evaluation, the creep damage fraction remains nearly equal to zero at the point of "core melt", and it is also expected that reliable detection of core damage and prompt RCS depressurization are implemented as stated above. However it is still necessary to carefully address the possibility of the induced SGTR in the evaluation of the level 2 PRA in considering the inherently high uncertainties as well as the state-of-the-art study results on this topic. Additional discussion on the evaluation of induced SGTR will be provided in the answer for RAI#19-303 by 28 April 2009.

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Impact on DCD
There is no impact on DCD from this RAI.

Impact on COLA
There is no impact on COLA from this RAI.

Impact on PRA
There is no impact on PRA from this RAI.