MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

April 24, 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-09137

Subject: Supplemental Information to US-APWR DCD RAI No. 92-1237 Revision 0

References:
1) "Request for Additional Information No. 92-1237 Revision 0, SRP Section: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation," dated November 5, 2008
2) Letter MHI Ref: UAP-HF-08275 from Y. Ogata (MHI) to the U.S. NRC, "MHI's Responses to US-APWR DCD RAI No. 92-1237," dated December 5, 2008.
3) "Summary of February 17-19, 2009, Public Meeting with Mitsubishi Heavy Industries, On Design Control Document, Chapter 19 - Probabilistic Risk Assessment (Level 2 and Low Power/Shutdown) And Severe Accident Evaluation," dated March 16, 2009 (ML090570848)

The U.S. Nuclear Regulatory Commission ("NRC") and representatives of Mitsubishi Heavy Industries, Ltd. ("MHI") held a Category 1 public meeting on 17 to 19 February 2009. One of the topics discussed in this meeting was on the technical contents corresponded in References 1 and 2. As the conclusion of the meeting, MHI proposed to the NRC to provide supplemental information for the action items summarized in Reference 3, especially items #1 to 10 except for #5.

With this letter, MHI transmits to the NRC a document as listed in Enclosures.

Enclosed includes the first responses on five action items #4, 6, 8, 9 and 10. Rest of the action items require additional time for internal discussions and computations, and will be answered by 28th 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

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information is below.

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Supplemental Information to US-APWR DCD RAI No. 92-1237 Revision 0 (proprietary version)

3. Supplemental Information to US-APWR DCD RAI No. 92-1237 Revision 0 (non-proprietary version)

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

<u>Contact Information</u>

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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-09137

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 "Supplemental Information to US-APWR DCD RAI No. 92-1237 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 24th day of April 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09137
Docket Number 52-021

Supplemental Information to US-APWR DCD RAI No. 92-1237
Revision 0

April 2009
(Non-proprietary)

Supplemental Information to Question 19-156 of RAI#92-1237

The NRC staff requested MHI to further consider on the issue of hydrogen control in the following three aspects.

(1) The assumption of perfect mixing may be optimistic for some specific breaks. There is a need to bound the analysis with a few cases that cover the spectrum of scenarios. MHI needs to provide more information on structures and the containment configuration for specific breaks as it may impact release.

(2) MHI needs to address the hydrogen release rate for various cases in order to confirm that the chosen Large LOCA sequence is a bounding case.

(3) LOCA analysis is used as a substitution for high pressure scenarios to some degree in the evaluation of hydrogen control. MHI needs to explain why this is adequate.

Answer to Question (1)

MHI qualitatively analyzed the influence of the break locations for atmosphere mixing by reviewing the latest containment arrangement drawings presented in Figures 1.2-14 to 25 of the DCD Chapter 1.

- RCS break near SG, RCP or pressurizer surge line

  As seen in these drawings, the SG compartments where SGs and RCPs are arranged are relatively large and freely open each other, in which the pressurizer surge line is also arranged. Hence, if RCS breaks near SG, RCP or pressurizer surge line, released gas can be expected well mixed and the assumption of perfect mixing is acceptable.

- RCS break near RV

  If RCS breaks near RV, it may be possible that released gas flows into the reactor cavity, which is a compartment not very widely open to other compartments compared to the SG compartment discussed above. However, significant accumulation of gas can be prevented from the design perspective. The US-APWR is designed that the reactor cavity is filled with water before core damage by utilizing containment spray system or other water supplying functions. Hence, there would be no or very limited space in the reactor cavity where combustible gas flows into and accumulates after core damage. In case the function to supply water to the reactor cavity is completely lost, it is considered that whole containment atmosphere is inerted because of very high steam concentration. This prevents hydrogen burn in nature.

It is concluded through the above examination that no compartments are identified in which hydrogen accumulates with high concentration due to release from a specific RCS break and causes violent combustion. The assumption of perfect gas mixing in each compartment is therefore acceptable in the analysis of hydrogen control.

Answer to Question (2)

Proprietary

Proprietary

Proprietary

Answer to Question (3)

The description answered in Question 19-156, *"the hydrogen release behavior from hot leg creep rupture for high pressure scenario is same as in LOCA sequence"*, does not intend to substitute high pressure scenarios with LOCA in analyzing the hydrogen control.

As presented in Subsection 15.3.3 of the PRA technical report [1] as well as discussed in the Answer to Question (2) above, two representative accident cases were selected for the US-APWR hydrogen control analysis, i.e. LOCA and transient scenarios. This clearly shows the position of MHI that LOCA is not used as the substitution for transient events. What MHI intended to express in the answer to Question 19-156 is simply that the release location of hydrogen is from hot leg break regardless of LOCA or transient with hot leg creep rupture, and hydrogen is mixed and controlled in the SG compartment.

Therefore, in order to address the difference in hydrogen release behavior, as answered in the previous question (2), MHI additionally analyzed the hydrogen release for transient with hot leg creep rupture scenario. As the result, it is identified that more challenging hydrogen release for transient with hot leg creep rupture scenario is not from hot leg rupture but from pressurizer relief tank rupture disk, same as the transient with RCS depressurization scenario presented in PRA technical report. Hydrogen release from hot leg creep rupture is not very significant as shown in the answer to question (2), and it can be considered that the release from hot leg rupture is properly controlled.

Supplemental Information to Question 19-169 of RAI#92-1237 (modeling of alternate containment cooling for MAAP code)

The NRC staff requested MHI to evaluate to determine the ability of fan cooler unit during severe accident whether it functions adequately with postulated hot gas or non-condensable build up which might impact effectiveness.

It should be noted that containment cooling employing fan cooler unit during severe accident (i.e. alternate containment cooling) is a backup function of the containment spray system as explained in Subsection 11.2.2.7 of the PRA technical report [1]. It is proposed that this function is only employed when none of the containment spray system trains are functional. The containment atmosphere is therefore pressurized over the design condition by steam when the alternate containment cooling is applied. It may possible that postulated hot gas or non-condensable gas accumulates in containment during severe accident; however the total amount of these non-steam gases is considerably smaller than that of steam.

This can be confirmed through examining the severe accident progression analysis result obtained by employing MAAP code. It is presented in Chapter 14 of the PRA technical report [1], in which the efficiency of the alternate containment cooling is also evaluated.

The Sequence-AM003 and Sequence-AM004 are the cases discussed about the efficiency of the alternate containment cooling, and graphs of hydrogen and steam concentrations are depicted in Figures 14-159 and 14-160 for AM003, and Figures 14-167 and 14-168 for AM004, respectively. AM003 is a sequence that containment fails before core damage, hence there is no accumulation of hydrogen, and on the other hand, steam concentration reaches as high as approximately 80%. AM004 is a sequence of small break LOCA without containment spray, and accumulation of hydrogen is approximately 6% at highest, and on the other hand, steam concentration reaches approximately 75%. For both sequences, the highest steam concentration is observed immediately before starting the alternate containment cooling.

The experiment to determine the effectiveness of alternate containment cooling was performed for several gas conditions [2]. The gas conditions were determined to model the postulated severe accident containment atmosphere, which had the steam concentration approximately 40 to 70% corresponding to the gas pressures. Thermal properties of fan cooler unit coil such as heat exchange capacity were measured for these gas conditions with varying the natural convection flow rate as a parameter. The tested gas conditions were very similar to the containment atmosphere during severe accident calculated for the US-APWR in light of the gas composition.

Considering above discussions, it is concluded that the effect of the hot gas or non-condensable build up has been incorporated with sufficient degree in determining the effectiveness of alternate containment cooling. It can be therefore considered that the accumulation of non-condensable gas during postulated severe accident for the US-APWR has very little adverse impact to the effectiveness of the alternate containment cooling and can be negligible.

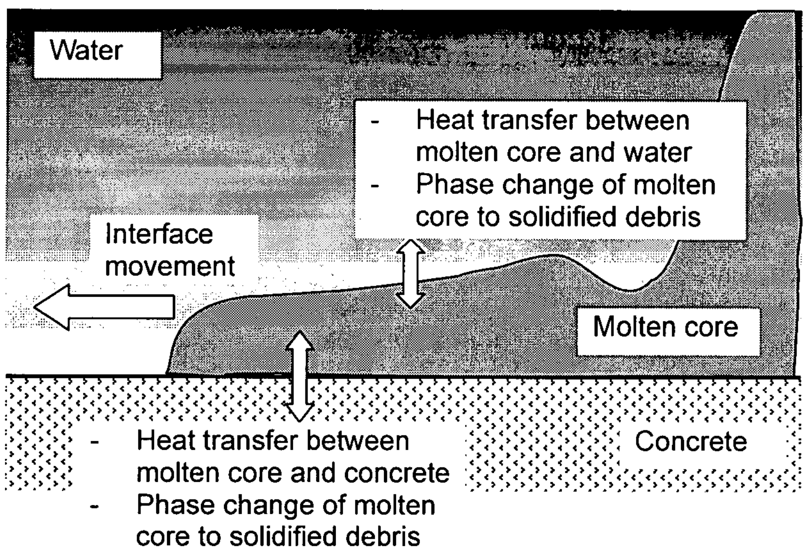
Supplemental Information to Question 19-161 of RAI#92-1237

The NRC staff requested MHI to demonstrate the basis for use of FLOW-3D by reference of test, etc.

In the answer to Question 19-161 of RAI#19-1237, three requirements to model debris spreading behavior were presented, i.e.

- Ability to evaluate interface movement of molten core
- Ability to evaluate solidification process of molten core
- Ability to evaluate temperature profile of molten core

Fundamental physical characteristics to be solved during the spreading process of molten core are all included in these three abilities as depicted in the schematic drawing below.



In order to validate the capability of FLOW-3D code to solve the problem of molten core spreading behavior, MHI referenced two types of studies, one is casting experiment and analysis (three casting studies are referenced in "Answer to additional item (a)", page 19-161-3 to 5), and the other is thermal transportation experiment and analysis (i.e. MINI-ACOPO experiment). The casting study covers areas of interface movement and solidification process of molten core; and the thermal transportation study covers an area of temperature profile of molten core.

From these two types of studies, it has been confirmed that FLOW-3D has adequate capability to evaluate the molten core spreading process for the US-APWR design.

References:

[1] US-APWR Probabilistic Risk Assessment, MUAP-07030 Rev.1, Mitsubishi Heavy Industries, September 2008

[2] Evaluation of the Cooling Performance of Non-safety Grade Air Recirculation System Cooling Coils, Presentation material prepared by MHI for Workshop on Severe Accident Research in Japan (SARJ-95), December 1995