MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

December 22, 2011

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-11448

Subject: 2nd MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3 (SRP 15.06.05)

Reference: 1) Request for Additional Information No. 861-6062 REVISION 3 - Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary", dated 10/31/2011.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "2nd MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3".

Enclosed are the Responses to Question 15.06.05-95, 96 and 100 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 in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is replaced by the designation "**[ ]**".

This letter includes a copy of the proprietary version of the RAI Responses (Enclosure 2), a copy of the non-proprietary version of the RAI Responses (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all material designated as "Proprietary" in Enclosure 2 be withheld from 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 this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

Yoshiki Ogata
Director- APWR Promoting Department
Mitsubishi Heavy Industries, Ltd.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. 2nd MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3 (proprietary)
3. 2nd MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3 (non-proprietary)

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

Contact Information

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ENCLOSURE 1

Docket No. 52-021

MHI Ref: UAP-HF-11448

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and state as follows:

1. I am Director, 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 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 "2nd MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3," and have determined that the document contains 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 basis for holding the referenced information confidential is that it describes the unique design of the safety analysis, developed by MHI (the "MHI Information").

4. The MHI Information is not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and detailed design for its software and hardware extending over several years. Therefore public disclosure of the materials would adversely affect MHI's competitive position.

5. The referenced information has in the past been, and will continue to be, held in confidence by MHI and is always subject to suitable measures to protect it from unauthorized use or disclosure.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information.

7. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's application for certification of its US-APWR Standard Plant Design.

8. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design and testing of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

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 22nd day of December, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11448
Docket No. 52-021

2nd MHI's Responses to US-APWR DCD RAI No. 861-6062
Revision 3

December 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/22/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 861-6062 REVISION 0**
SRP SECTION:	**15.06.05 – LOSS OF COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY**
APPLICATION SECTION:	**15.6.5**
DATE OF RAI ISSUE:	**10/31/2011**

QUESTION NO.: 15.6.05-95

Follow-up to RAI 719-5339, Question 15.6.5-79:

1. In the response to RAI 706-5339, Question 15.6.5-79, MHI stated that the axial power shape affects the core average void fraction. MHI then performed a sensitivity calculation where the core average void fraction was reduced to 80% of the base case value. Provide a basis for the 80% value.
2. Using a uniform axial power shape and reducing the core average void fraction may not be conservative relative to using different axial power shapes. Provide the results of a thermal-hydraulic analysis quantifying the two-phase mixture level in the US-APWR reactor during the long term cooling phase assuming the most limiting break size, break location, and ECCS performance conditions. The analysis should include loop seal piping becoming plugged as well as the most limiting axial power shape.
3. In the axial power sensitivity, the reactor vessel pressure differential falls to a value just below the loop pressure differential for a short time. This implies that the core mixture level is below the bottom of the hot leg elevation. At the same time, the conclusion is that the two-phase mixture level is always maintained above the bottom of the hot leg elevation. How can the conclusion statement be made when the condition that is observed does not meet the requirement?

ANSWER:

The core average void fraction was reduced to 80% of the uniform power distribution case for sensitivity study in the previous response to RAI 706-5339 Question 15.06.05-79[(ref-1)]. This assumption of 80% was not derived from the actual power distribution, but it is so conservative that this ratio covers actual axial power distribution variation. To demonstrate the conservativeness of the ratio of 80%, the lowest possible core average void fraction was calculated from a top-skewed power distribution and compared to the core average void fraction calculated from a uniform power distribution.

<u>Top-skewed axial power distribution</u>

Fig.1 shows assumed axial power distribution, and integral of axial power from the bottom of the

core is shown in Fig.2. The axial power distribution is determined such that the axial power integral at the lower portion of the core is the minimum that occurs during the fuel cycle within the limits of the axial flux difference variation.

Core average void fraction

The core average void fraction is calculated from the top-skewed power distribution shown in Fig.1. The calculation procedure of the core average void fraction is described in Appendix A of the ref-(2). The calculation conditions are as follows, which are the same as in the previous RAI response, Question 15.06.05-79 of ref-(1).

- Atmospheric pressure is assumed for system pressure
- Core inlet flow temperature is set to saturation temperature at atmospheric pressure

A cold leg break was assumed since hot leg break does not causes boric acid precipitation in the core during post-LOCA long term condition as explained in section 15.6.5.2.3 of DCD. The maximum size break was assumed to maintain low system pressure. The minimum HHIS flow rate was assumed but the flow rate assumption does not actually matter since most of the injected flow through the DVI line spills without going to the core.

The average void fraction calculation procedure for uniform axial power distribution is described in ref-(3). The axial void fraction distribution in the core 2 hours after the break is shown in Fig.3 for the uniform and top-skewed distributions. Table-1 shows the calculated core average void fraction for each axial power distribution case and the ratio of the void fraction between the uniform distribution and the top-skewed distribution cases 2 hours after the break occurs. As shown in Table-1, the core average void fraction ratio due to the difference of axial power distribution is **[ ]**, which is larger than the assumed value of 80% in the sensitivity study. As a result, the void fraction value assumed in the sensitivity study was conservatively low.

Sensitivity to core power

The core average void fraction decreases proportionally to the decrease in the core decay heat, thus the void fraction ratio between the uniform power distribution case and the top-skewed power distribution case may vary with time. Fig.4 shows the calculated time-history of the core average void fractions. The core void fraction ratio does not change and remains above the 80% (0.8) which was assumed in the sensitivity study of the previous response to RAI 706-5339 Question 15.06.05-79[(ref-1)].

Revised sensitivity study of axial power shape

As discussed above, the lowest possible value of the void fraction which accounts for the axial power distribution variation is **[ ]** times of the void fraction calculated from a uniform power distribution. Based on this result, the sensitivity study concerning axial power distribution described in the previous response to RAI Question 15.06.05-79 of ref-(1), is revised as follows.

Fig.5 shows the comparison of calculated time-histories of the RV differential head (ΔP_{HEAD}) and the external loop flow resistance (ΔP_{LOOP}) in the case where the core average void fraction is decreased to **[ ]** of the uniform power distribution case.
The external loop flow resistance does not change from the base case of ref-(1) since core boiling rate is same as base case of ref-(1) and loop flow rate does not change. The RV differential head becomes smaller than the base case due to decreasing the core average void fraction, however, it still remains above the external loop flow resistance. That is, the two-phase mixture level is maintained above the hot leg bottom elevation up to the hot-leg switch-over time

(4 hour).

Conclusion

The lowest possible core average void fraction is calculated based on the axial power distribution, which occurs in actual plant. The calculated lowest core average void fraction is compared to the value of uniform distribution case, and the void fraction ratio is confirmed to exceed the 80% value, which is assumed for the sensitivity study case in the previous response to RAI Question 15.06.05-79 of ref-(1).
Next, the RV differential head is re-evaluated using the new void fraction ratio and re-compared to the external loop flow resistance, which is calculated in the response to RAI Question 15.06.05-79 of ref-(1). As a result, it is confirmed that RV head differential remains above the external loop flow resistance pressure drop at all time prior to the hot leg switch over, which means the two-phase core mixture will not fall below the bottom of the hot leg elevation.

Furthermore, it is assumed that no loop seal plugging occurs before the hot leg switch over occurs if the counter current flow phenomena at the RCP suction uphill side are taken into consideration. This assumption further reduces the loop flow resistance pressure drop relative to the RV head differential, which will increase two-phase mixture level in the core and upper plenum. The loop seal phenomena are described in the response to RAI 15.06.05-97 of ref-(2) in detail.

Table-1 Core Average Void Fraction Two Hours after Break

Type	Uniform Distribution	Top-skewed Distribution
Core Average Void Fraction	[ ]	[ ]
Core Average Void Fraction Ratio	[ ]	

[ ]

Fig.1 Axial power distribution

[ ]

Fig.2 Integral of axial power from bottom of the core

Fig.3 Core void fraction axial distribution at 2 hours after LOCA

Fig.4 Time history of core average void fraction (α_{uni}, α_{top}) and void fraction ratio

[ ]

Fig.5 Differential head and loop flow resistance during post-LOCA long term period
([ ] of core average void fraction case compared to base case)

References:

1. UAP-HF-11130 "MHI's Response to US-APWR DCD RAI No. 706-5339 Revision 0 (15.06.05)" (April 28,2011)
2. UAP-HF-11416 "1st MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3 (SRP 15.06.05)" (December 2, 2011)
3. UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1" (July 17, 2009) RAI Question 15.6.5-44 Appendix B Attachment.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on a Technical/Topical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/22/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 861-6062 REVISION 0**
SRP SECTION:	**15.06.05 – LOSS OF COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY**
APPLICATION SECTION:	**15.6.5**
DATE OF RAI ISSUE:	**10/31/2011**

QUESTION NO.: 15.6.05-96

Follow-up to RAI 706-5339, Question 15.6.5-80:
Further information is needed on the Yeh correlation. The response to RAI 706-5339, Question 15.6.5-80 includes Figure 1, which compares predicted versus measured void fractions including test data at low pressure (20 psia [0.14 MPa]). However, it is not clear how many tests were actually run at this low pressure. Provide a figure showing clearly the comparison against low pressure test data and include a table that lists the test flow conditions and measured void for each data point used in assessing the correlation at low pressure.

ANSWER:

MHI recognizes that NRC's requests are: 1) to provide a figure clearly showing the comparison against low pressure test data described in Fig-1 of ref-(1), and ref-(2) to provide the experimental test conditions, such as measured flow rate and void fraction against these experiments. However, as for the request 1), it is difficult to identify each measured void fraction from the figure since the experimental data plotted on the figure are not clearly labeled, and MHI can not identify the set of measured data of primary interest.

Instead, MHI identified the test flow condition, re-calculate void fractions derived from modified Yeh's correlation, and compare the calculated void fractions with measured void fractions. There are many data included in the Fig-1 of the previous response to RAI of ref-(1), but only the FLECHT–SKEWED[ref-(3)] and FLECHT-SEASET [ref-(4)] tests are identified as low-pressure tests in that figure. Then, MHI specifically compared the calculated and measured void fraction in the same manner as ref-(2).

There are many experimental runs in the FLECHT–SKEWED[ref-(3)] and FLECHT-SEASET[ref-(4)] data sets, and MHI selected the 17 low pressure run cases. Table-1 and Table-2 summarize the selected run cases, the experimental conditions, the measured and the calculated void fractions for the FLECHT-SKEWED[ref-(3)] and FLECHT-SEASET[ref-(4)] tests. The calculated void fraction at the outlet of the core was obtained by using the modified Yeh's correlation, whereas the measured void fraction at the same point was obtained from the original test data.

Fig.1 shows the comparison between the calculated and the measured void fraction against low pressure test data for these tests. It can be seen that the calculated result indicated by a solid line shows good agreement with the test data, which are as shown in the Fig-1 of the previous response to RAI of ref-(1). This result indicates that the modified Yeh's correlation equation can properly predict the void fraction, including at low-pressure conditions.

Table-1 FLECHT-SKEWED Experimental Conditions

Table-2 FLECHT-SEASET Experimental Conditions

Fig.1 Comparison of the measured and the calculated void fraction at low pressure in the same graphical form as Figure 3 of ref-(2)

References:
1. UAP-HF-11083 "MHI's Response to US-APWR DCD RAI No. 706-5339 Revision 0" (March 2011) RAI Question 15.6.5-80
2. H. C. Yeh, "Modification of Void Fraction Calculation," Proceedings of the Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Volume 1, Taipei, Taiwan, April 5-9, 1994.
3. E.R. Rosal, C.E. Conway, and M.C. Krepinevich, "FLECHT Low Flooding Rate Skewed Test Series Data Report," NRC-2 May 1997.
4. M.J. Loftus, L. E. Hochreiter, C.E. Conway, C.E. Dodge, A. Tong, E.R. Rosal, M. M. Valkovic and S. Wong, "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Report," NUREG/CR-1532 NRC / EPRI / Westinghouse, 1980.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on a Technical/Topical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/22/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

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RAI NO.:	**NO. 861-6062 REVISION 0**
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APPLICATION SECTION:	**15.6.5**
DATE OF RAI ISSUE:	**10/31/2011**

QUESTION NO.: 15.6.05-100

Follow-up to RAI 5352, Question 15.6.5-89:
Provide an updated response to RAI 719-5352, Question 15.6.5-89 that takes into consideration relevant and conforming findings related to US-APWR core debris blockage that also accounts for any experimental test results to assess the US-APWR core blockage. Currently, such additional information is planned to be included in Revision 2 of MUAP-080013-P, "US-APWR Sump Strainer Downstream Effects," scheduled for release by MHI on August 31, 2011.

ANSWER:

After the quenching of the core at the end of reflood phase, continued operation of the ECCS supplies borated water from the RWSP to remove decay heat and to keep the core subcritical, and borated water from the RWSP is initially injected through DVI lines. In such a situation, sump strainer bypass debris is also carried by the ECCS water, which has the potential to adversely affect the boric acid concentration behavior in the reactor vessel.

The technical report MUAP-08013-P Revision 2, "US-APWR Sump Strainer Downstream Effects" (ref-(1)), describes in-vessel downstream effects of bypass debris in chapter 4. Chapter 4 of this report describes the impact of bypass debris on long-term core cooling and the boric acid mixing volume. In this response to the RAI, important effects which can impact post-LOCA long-term cooling evaluation are discussed.

<u>Trapping debris on grid spacer or cladding surface</u>

In the case where bypass debris adheres to grid spacer or cladding surface physically or chemically, the debris acts as an insulator and impedes heat transfer from fuel rod to coolant in the core. It results in a small decrease in heat transfer, but the rate of heat transfer recovers in a short time since fuel rod and cladding temperature increase slightly and the temperature differential between the cladding and the coolant increases, which compensates for the degradation of heat transfer coefficient. The small increase in cladding and fuel rod temperature does not directly affect the accumulation of boric acid in the mixing volume. The accumulation of boric acid in the mixing volume

is controlled by the core decay heat level, which is not affected by debris in the ECC. Debris trapping or plate out should not significantly affect the post-LOCA long term cooling evaluation.

Effect of suspended and settled sump debris on mixing volume

The suspended debris and settled debris ingested into the vessel may have some impact on the mixing volume. The bypass debris settled or suspended in the coolant replaces water volume in the mixing volume. The amount of bypass debris that may exist in the mixing volume is described and estimated in the ref-(1) Appendix–D. The lower plenum in the US-APWR has a large volume of more than one thousand cubic feet, thus the liquid volume replaced by debris would be a small fraction of the mixing volume which is assumed in the post-LOCA long term cooling evaluation. Because the fraction of the mixing volume displaced by debris is so small the effect on the mixing process and the boric acid contribution will be negligible.

Effect of core inlet blockage on mixing volume

The possibility of the core inlet blockage due to debris is in discussed in the ref-(1), section 4.4.1 (2). In the ref-(1) section 4.4.1 (2), it is stated that significant core inlet blockage will not take place.
As shown in the previous response to RAI 15.6.5-89(ref-2), the required mixing flow rate and velocity to maintain the boric acid concentration in the lower plenum is quite low. In this sense, the effect of a change in flow resistance between the core and the lower plenum on the mixing flow is not important. As stated in ref-(1) section 4.4.1 (2), any increase in flow resistance due to debris will not impede flow between the lower plenum and core region.

From the comprehensive point of view discussed above, the impact of accumulated bypass debris on mixing flow behavior between the core and the lower plenum is very limited and that the core region mixing volume used in the US-APWR long term cooling analyses is still valid.

Effect of two-phase core mixture level

In the post-LOCA long-term cooling evaluation, it is assumed that the core two-phase mixture level will not fall below the bottom of the hot leg before the hot leg switch-over and that the mixing volume includes the upper plenum, below the hot leg. When the core inlet blockage by the bypass debris is considered, the two-phase mixture level is not expected to fall below the bottom of the hot leg elevation for the following reasons:

- As discussed above, the potential flow blockage would occur mainly at the fuel assembly bottom nozzle. The region below the fuel inlet nozzle elevation is filled with water at or below saturation temperature and core inlet flow velocity drops to a very low level early in the post-LOCA long-term cooling period.
- The evaluation of core flow driven by RV pressure differential is discussed in DCD RAI response, Q15.06.05-94 of Ref-(3). In the RAI response, it is shown that the estimated core flow rate driven by the RV pressure differential is much larger than the core makeup flow rate, which means that there is much margin regarding flow resistance inside RV.
- In the response to RAI 861-6062, Q15.6.05-95 it was shown that the void distribution and mixture level in the core and upper plenum is determined primarily by the core power level and axial power profile. Since these do not change if debris is present the mixture level should not be changed.

Effect of blockage on alternate core coolant flow path

Sump debris may accumulate sufficiently to block some of the core bypass flow paths that are expected to dilute the boric acid concentration in the mixing volume. The following flow paths are considered as core bypass.

- Upper head spray nozzle through control guide tube
- Control-rod guide tubes, core thimble tubes
- Neutron reflector
- Hot leg nozzle gaps

These bypass flow paths or volumes are not modeled or credited in the evaluation of post-LOCA long-term cooling evaluation. Therefore, there is no impact of sump debris in these core bypass paths on the post-LOCA long term cooling evaluation.

Conclusion

The above discussion shows that the sump bypass debris may affect post-LOCA long term cooling evaluation only in terms of core inlet clogging, but its impact is very limited, hence the core region mixing volume used in the US-APWR long term cooling analyses is still valid. In conclusion, there is no need to change the hot leg switch over time (4 hour).

References:

1. MUAP-08013-P(R2), "US-APWR Sump Strainer Downstream Effects", August 2011
2. UAP-HF-11139 "MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0 (15.06.05)" (May 18,2011)
3. UAP-HF-11416 "1st MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3 (SRP 15.06.05)" (December 2, 2011)

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on a Technical/Topical Report.