MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

December 28, 2010

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-10348

Subject: **Revision to MHI's RAI Responses on US-APWR DCD Chapter 15.6.5 SBLOCA**

References: 1) Mitsubishi Heavy Industries, Ltd., "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1", UAP-HF-09384 dated July 17, 2009.
2) Mitsubishi Heavy Industries, Ltd., "MHI's Response to US-APWR DCD RAI No. 514-4040 Revision 2", UAP-HF-10039 dated February 5, 2010.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") an official document entitled 'Revision to MHI's RAI Responses on US-APWR DCD Chapter 15.6.5 SBLOCA'. In the enclosed document, MHI provides 4 (four) revised RAI responses (QUESTION No. 15.06.05-26, 30, 33, and 62). The original responses for QUESTIONS 15.06.05-26, 30, and 33 were previously submitted in Reference 1, and that for QUESTION 15.06.05-62 was in Reference 2.

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). Any proprietary information that is written inside a bracket in the proprietary-version is replaced by the designation "[ ]" without any text, in the non-proprietary-version.

This letter includes a copy of proprietary version (Enclosure 2), a copy of non-proprietary version (Enclosure 3), and the Affidavit of Atsushi Kumaki (Enclosure 1) which identifies the bases of MHI request 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 this submittal. His contact information is provided below.

Sincerely,

Atsushi Kumaki for

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

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NRO

Enclosures:

1. Affidavit of Atsushi Kumaki

2. Revision to MHI's RAI Responses on US-APWR DCD Chapter 15.6.5 SBLOCA (Proprietary)

3. Revision to MHI's RAI Responses on US-APWR DCD Chapter 15.6.5 SBLOCA (Non-Proprietary)

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

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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-10348

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Atsushi Kumaki, being duly sworn according to law, depose and state as follows:

1. I am Group Manager, Licensing Promoting Group in 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 " Revision to MHI's RAI Responses on US-APWR DCD Chapter 15.6.5 SBLOCA" and have determined that portions of the report 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 technical report 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 in the report identified as proprietary by MHI 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 codes and files developed by MHI for the fuel of the US-APWR and also contains information provided to MHI under license from the Japanese Government. These codes and files were developed at significant cost to MHI, since they required the performance of detailed calculations, analyses, and testing extending over several years. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI and the Japanese Government.

5. 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.

6. 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 of new fuel 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 28th day of December, 2010.

Atsushi Kumaki
Group Manager- Licensing Promoting Group in APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

ENCLOSURE 3

UAP-HF-10348

Revision to MHI's RAI Responses on US-APWR DCD Chapter 15.6.5 SBLOCA

December 2010
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Original Issued on 7/17/2009
Revision 1 Issued on 12/28/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 352-2369 REVISION 1**
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:	**5/4/2009**

QUESTION NO.: 15.06.05-26

Discuss how the flow distribution in the core (fraction of flows through various bypasses modeled in the core) is validated (MUAP-07025-P, section 3.2). Section 4.1.3 mentions the total core flow being 91% of the RCS flow rates. Please explain.

ANSWER:

The estimated minimum value of the effective core cooling flow is 91.0 percent of the RCS flow rate, which is the total RCS flow minus the design core bypass flow of 9% (including uncertainty).

DCD Section 4.4.4.2 describes five sources of core bypass flow that are unavailable for providing core cooling as follows (1) flow through the head-cooling spray nozzles into the RV upper head, (2) flow directly going out the outlet nozzle through the gap between the RV and the core barrel, (3) flow through the neutron reflector, (4) flow through the control guide thimbles and in-core instrumentation guide tubes, and (5) flow in the extra gap region between the peripheral fuel assemblies and the neutron reflector.

These bypass flow paths are modeled for the LOCA analyses except the gap flow between RV and core barrel which is negligibley small. The flow through the control guide thimbles and in-core instrumentation guide tubes, and the flow in the extra gap region between the peripheral fuel assemblies and the neutron reflector are represented by one flow path.

The flow rate bypassing the core is validated as follows:

Each bypass flow rate is determined based on the balance of the pressure drops through the core and the bypass flow path, Then, it can be obtained by solving the following equation;

$$dP = K \times \rho \times (Q/A)^2 / 2g_c$$

In the above equation, following nomenclatures apply.

dP: Corresponding pressure loss in the main flow through the core (psi)
K: Pressure loss coefficient at typical cross sectional area of the bypass flow path (-)
ρ : Coolant density in the flow path (lb_m/in^3)
Q: Bypass flow rate (in^3/s)
A: Typical cross sectional area of the bypass flow path (in^2)
g_c: Conversion factor ($(lb_m in/s^2)/lb_f$)

The pressure loss, dP, for the US-APWR core is based on the experimental data from the flow test of MHI 12-ft fuel assembly, which shares the same components such as spacer grids with the US-APWR fuel. The differences on the number of grids and fuel rod length are corrected for the US-APWR 14-ft core, based on the pressure loss data of each component. Although the evaluation must be valid and treated conservatively, ~~a~~ the hydraulic test for the US-APWR fuel assembly ~~planned~~ performed in 2010 (Ref.1) ~~to complete by 2010 will~~ provided the confirmatory data lower than expected, that causes the bypass flow rate to be smaller than expected. The reduced bypass flow rate tends to mitigate consequences of small break LOCAs.

Pressure loss coefficient for each path is derived based on the conventional experimental data and/or widely used correlations described in (Ref.~~1~~2), which have been experienced in the MHI designed PWRs. The best estimated value of the bypass flow rate was calculated as 7.5% of RCS flow rate.

The uncertainty regarding the core bypass flow is estimated by considering the manufacturing tolerances of the bypass flow paths and uncertainties in the pressure drop through the core and bypass flow paths. The total uncertainty of the bypass flow rate was conservatively estimated as 1.5% of the RCS flow rate (20% of the best estimate bypass flow). Thus, the maximum bypass flow rate was determined as 9% of the RCS flow rate (91% core flow rate), which includes above mentioned uncertainty, and is used as a conservative value for the LOCA analysis.

Reference

1. Okamoto, et. al, "Test Result Report of Pressure Loss and Lift Force of US-APWR Fuel Assembly (UA-ST3170) ", Nuclear Development Corporation, 2010 (in Japanese).
~~1.~~2. Idelchik, I.E., "Handbook of hydraulic resistance" 3rd edition, CRC Press, 1994

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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US-APWR Design Certification

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Docket No. 52-021

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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:	**5/4/2009**

QUESTION NO.: 15.06.05-30

Discuss the uncertainty of the pressure measurement at the pressurizer and its impact on the SCRAM and ECCS timing (MUAP-07025-P, Section 4.1.9).

ANSWER:

Increase in the pressurizer pressure ~~rises RCS temperature at the initial steady state, which~~ results in a conservative estimate for the transient evolution, particularly for the PCT due to delay in SCRAM and ECCS timing. For US-APWR small break LOCA analyses, the initial pressure is conservatively assumed to be the nominal design value plus (+) 30 psia, which accounts for the measurement errors in terms of the pressurizer pressure. The uncertainty in the pressurizer measurement, which affects the SCRAM and ECCS timing, is conservatively taken into account for the US-APWR SBLOCA safety analyses.

~~If any uncertainty is included in the pressure measurement device and system, for example if the actual pressurizer pressure is higher than the measured/detected pressure, there is no impact on the reactor trip (SCRAM) and ECCS actuation timing, because these reactor actions are dependent on the pressure decrease from the initial level which is determined based only on the measurement/detected pressure.~~

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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APPLICATION SECTION:	**15.6.5**
DATE OF RAI ISSUE:	**5/4/2009**

QUESTION NO.: 15.06.05-33

Section 4.2.4 (MUAP-07025-P) indicates that the maximum time step sizes used for 7.5- inch and 1-ft^2 break analysis are larger than the time step size for the DVI break analysis. This appears to be contradictory to expectations, since larger break transients may require smaller time steps due to more rapid changes. Discuss how these maximum time steps were chosen. What are the maximum time step sizes for other break sizes and locations?

ANSWER:

In generally, a transient with larger break size requires smaller time step size due to the Courant limitation for the semi-implict numerical scheme which is applied to M-RELAP5. In all calculations by M-RELAP5, including the both plant and experimental test calculations, the automatic time step algorithm is used, in which the applied time step size is determined not to violate the hydraulic Courant number in all the calculation nodes. In addition, adequacy of the maximum allowable time step size is verified in the process of the plant calculation as is described in the technical report MUAP-07025-P (R2) (Ref.1). In the report, the maximum allowable time step size used for the DVI break is smaller than that for the 7.5-in break, and is same as that for the 1-ft^2 break (Table RAI-15.6.5-33.1). This seems to be contradictory to the numerical theory described above, since the maximum coolant velocity is expected to be smaller in the DVI break than in the 1-ft^2 break.

MHI has performed an additional time step sensitivity calculation for the DVI injection line break so as to re-evaluate the applicable time step size. Specifically, the maximum allowalble time step of [ ], which is same as the time step size for the 7.5-in cold leg break, is applied and its sensitivity on the cladding temperature is confirmed. The result of the cladding temperature is shown in Figure RAI-15.6.5-33.1, and is compared with the result obtained using the reference time step size ([ ]) in Reference 1. The figure shows [ ] is small enough for the DVI line break calculation, as well as for the 7.5-in break calculation. It is noted that a coarser time step size ([ ]) is sufficiently applicable for cold leg breaks of less than 6-in since no heat-up occur during the transients. In conclusion, the relation of applicable time step sizes for all the break cases, including the DVI break, is consistent with the expectation from the numerical theory.

~~However, the time-step size required to simulate the transient accurately differs from the case where the different break location is assumed even if the break sizes are mutually identical in the two cases. This is because the phenomenon of interest and the location providing the Courant limitation might be different in the two calculations. MHI has individually determined the maximum time step sizes used for 7.5 inch (and 1 ft2) and DVI break cases, based on the sensitivity analysis in terms of the time step.~~ ~~**Table RAI 15.6.5 33.1**~~ ~~shows the relation between the break position, location and the maximum allowable time step determined for the US APWR SBLOCA sensitivity calculations described in MUAP 07025 P (R0).~~

Reference:

1. Mitsubishi Heavy Industries, Ltd., Small Break LOCA Sensitivity Analyses for US-APWR, MUAP-07025-P (R2), October, 2010.

Table RAI-15.6.5-33.1 Maximum Allowable Time-Step Sizes for US-APWR SBLOCA Sensitivity Calculations

Break Location	Break Size or Break Area	Break Orientation	Maximum time step (s)
Cold leg	1-inch	bottom	
	2-inch	bottom	
	3-inch	bottom	
	4-inch	bottom	
	5-inch	bottom	
	6 inch	bottom, top, side	
	6.5-inch	bottom, top, side	
	7-inch	bottom, top, side	
	7.5-inch	bottom, top, side	
	8-inch	bottom, top, side	
	9-inch	bottom	
	10-inch	bottom	
	11-inch	bottom	
	12-inch	bottom, top, side	
	13-inch	bottom, top, side	
	1-ft^2 (=13.5-inch)	bottom, top, side	
Hot leg	2-inch	bottom	
	1-ft^2 (=13.5-inch)	bottom	
Crossover leg	2-inch	bottom	
	1-ft^2 (=13.5-inch)	bottom	
DVI injection line	3.4-inch	N/A	
Pressurizer steam phase	about 6.7-inch	N/A	

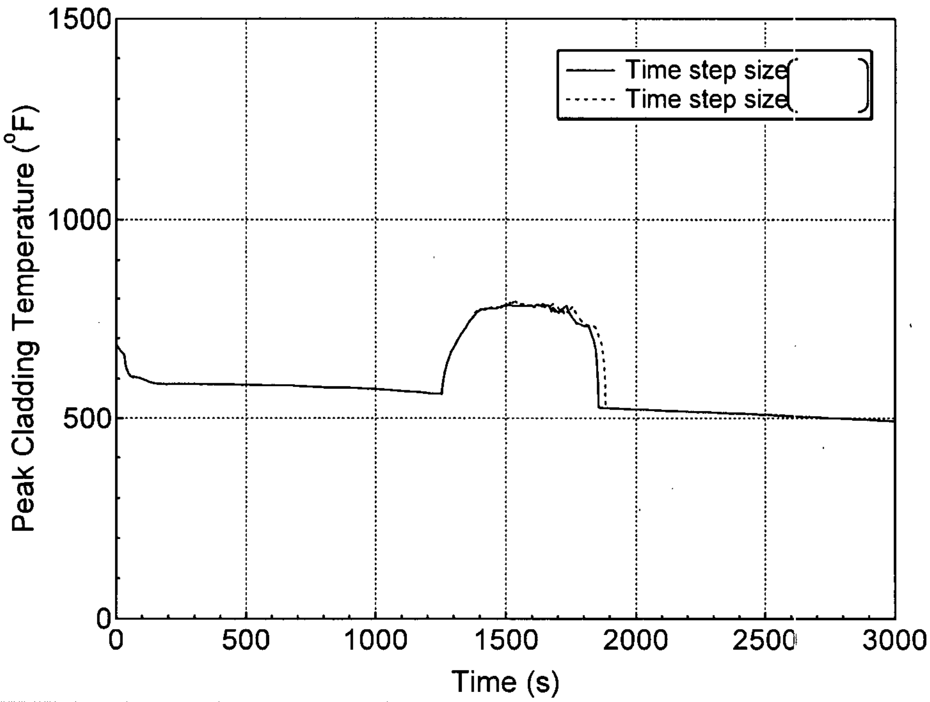

Figure RAI-15.6.5-33.1 PCT Sensitivity of Maximum Time Step Size for DVI Line Break

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 514-4040 REVISION 2**
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:	**12/17/2009**

QUESTION NO.: 15.06.05-62

PIRT Phenomena 37 Water Holdup in SG Inlet Plenum and 38 Water Hold-up in U-Tube Uphill Side highlight the importance of countercurrent flow limitation (CCFL) characteristics in the SG tubes, SG inlet plenum, and hot leg piping during loop seal clearance phase of SBLOCA. MUAP-07013-P provides comparisons of M-RELAP5 to UPTF hot leg tests and the Dukler air-water flooding tests and concludes that the M-RELAP5 model results are acceptable.

Considering the importance of CCFL relative to core cooling during the loop seal clearance phase of SBLOCA, evaluate the variability of PCT with CCFL model coefficients (both for the hot leg and the SG tubes) and justify the values used in the SBLOCA evaluation model. Responses to earlier RAIs may cover this topic.

ANSWER:

Sensitivity calculations in terms of the CCFL at the SG inlet plenum and in the SG U-tubes are ~~given in MHI's response to RAI CA-1 on the M-RELAP5 topical report MUAP-07013-P (Ref. 1)~~performed with M-RELAP5 M1.6.

[

]

~~The MHI response to RAI CA-1 concluded that the loop seal PCT is well suppressed by the enhanced capacity of the high head injection system (HHIS) in the US-APWR design.~~

Reference:
~~1. Mitsubishi Heavy Industries, Ltd., MHI's 2nd Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 09/08/2009, UAP-HF-09512, November 2009.~~
1. Mitsubishi Heavy Industries, Ltd., Small Break LOCA Sensitivity Analyses for US-APWR, MUAP-07025-P (R2), October, 2010.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.