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TOKYO, JAPAN

April 18, 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-11106

Subject: 1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0 (15.06.05)

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0". The enclosed materials provide MHI's response to Questions 15.06.05-88 and 15.06.05-90 of the NRC's "Request for Additional Information (RAI) 719-5352 Revision 0," dated March 17, 2011.

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 response (Enclosure 2), a copy of the non-proprietary version of the RAI response (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
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, Ltd.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. 1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0 (proprietary)
3. 1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0 (non-proprietary)

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-11106

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and 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 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 "1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0," 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 18th day of April, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11106
Docket No. 52-021

1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0

April 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

4/18/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 719-5352 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:	**3/17/2011**

QUESTION NO.: 15.6.05-88

In the US-APWR design, a switchover from direct vessel ECCS injection mode to a simultaneous injection mode involving direct vessel and hot leg ECCS injection is used to prevent boric acid precipitation and to ensure core cooling following a LOCA. During the simultaneous injection mode, the steam flow through the reactor hot legs can cause liquid entrainment and thus impede delivery of ECCS flow into the upper plenum. In addition, liquid holdup in the hot leg horizontal and inclined sections as well as in the connected steam generator regions can increase the loop resistance. In turn, this will cause a corresponding increase of the upper plenum pressure thus limiting the growth of the control mixing volume. US-APWR FSAR Section 15.6.5.3.3.3 "Post-LOCA Long Term Cooling Evaluation Results" only refers to entrainment threshold calculations as an evaluation basis for concluding that sufficient reactor core cooling is provided following the switchover to simultaneous ECCS injection after a LOCA.

Describe the entrainment model and provide the results from entrainment calculations performed for the US-APWR to demonstrate that hot leg injection is capable of preventing effectively boric acid precipitation for this reactor design. Discuss the applicability of the selected correlations under US-APWR specific conditions. List all assumptions made in the calculations including assumptions related to the decay heat model and core decay rate calculations as well as ECCS performance. Provide an assessment for the earliest point in time after which the liquid delivery into the upper plenum is sufficient enough to compensate for the core boil-off rate and flush the core.

Address possible impacts of assumptions and uncertainties associated with key parameters on the critical time point obtained. Present plots showing the time variation of quantities such as pressure, temperature, injected ECCS flow rate, steam flow rate, liquid flow rate, and entrainment rate as used and obtained in the analysis..

ANSWER:

In the US-APWR, the operator switches the operating DVI lines to the hot leg injection line (simultaneous RV and hot leg injection mode) to prevent the boric acid precipitation in case of LOCA. Two of four injection lines are switched for the simultaneous injection mode in practice. In the LOCA safety analysis it is assumed that only one injection line is used for hot leg injection.

The earliest hot leg switch over time is determined from the following three criteria.

(1) The time when the hot leg steam velocity drops below the entrainment threshold
(2) The time when necessary injection water flow can flow to the reactor vessel against counter current flow in hot leg
(3) The time when hot leg injected flow exceeds core boil-off flow and can dilute the boric acid concentration in the core.

<u>(1) Liquid entrainment threshold in hot leg</u>

The liquid entrainment threshold in the hot leg can be estimated from the Ishii-Grolmes liquid entrainment onset criterion (ref-[1]). This entrainment correlation is valid for flow conditions where the liquid phase does not take up a significant volume of the pipe such as in the hot legs in post-LOCA and viscous effects in the liquid are not dominant, i.e. the liquid phase is in the turbulent regime.

The liquid entrainment onset correlation (ref-[1]) can be expressed as follows;

$$j_g = N_\mu^{0.8}\left(\frac{\rho_f}{\rho_g}\right)^{0.5}\left(\frac{\sigma}{\mu_f}\right) \quad \text{for } N_\mu < \frac{1}{15} \qquad \text{(eq-1)}$$

where N_μ is the viscosity number and j_g is the superficial velocity of vapor phase.
N_μ is expressed as follows

$$N_\mu = \frac{\mu_f}{\left\{\rho_f\sigma\left(\frac{\sigma}{g\left[\rho_f-\rho_g\right]}\right)^{0.5}\right\}^{0.5}} \qquad \text{(eq-2)}$$

The following properties of saturated liquid and vapor phase at atmospheric conditions (14.7 psia) are used conservatively.

σ = 0.12988 [lbm/sec^2]	surface tension of liquid
μ_f = 190.3x10^{-6} [lbm/(ft-s)]	viscosity of liquid
μ_g = 8.24x10^{-6} [lbm/(ft-s)]	viscosity of vapor
ρ_f = 59.83 [lbm/ft^3]	density of liquid
ρ_g = 0.0373 [lbm/ft^3]	density of vapor

From (eq-2), viscosity number (N_μ) can be calculated.

$$N_\mu = \frac{1.903\times10^{-4}}{\left\{59.83\cdot0.12988\left(\frac{0.12988}{32.2\left[59.83-0.037308\right]}\right)^{0.5}\right\}^{0.5}} = 7.53\times10^{-4}$$

From (eq-1), the following results are obtained for the liquid entrainment threshold in terms of superficial steam velocity in the hot leg.

$$j_g = (7.53\times10^{-4})^{0.8}\left(\frac{59.83}{0.037308}\right)^{0.5}\left(\frac{0.12988}{1.903\times10^{-4}}\right) = 86.7\ \text{[ft/s]}$$

Applying this vapor velocity (j_g) in the hot leg, the following total core steam mass flow rate (W_g) at the entrainment threshold becomes:

$$W_g = N_{LOOP} \times A_{HL} \times \rho_g \times j_g = 67.81 \text{ [lbm/s]}$$

where

N_{LOOP} = 4 [#] number of loop

A_{HL} = 5.241 [ft^2] single hot leg flow area

The decay heat (P) which results in the core steam flow rate is obtained.

$$P = W_g(h_g - h_{SI}) = 65787.2 \text{ [Btu/s]}$$

where

h_g = 1150.3 [Btu/lbm] specific vapor enthalpy (saturated)

h_{SI} = 180.13 [Btu/lbm] specific liquid enthalpy of injection water (conservatively assumed as saturation)

US-APWR licensed power including uncertainty is 4303108 [Btu/s] (4451x1.02 [MWt]). The decay heat fraction (P/P_0) is obtained.

P/P_0 =[ ]

The decay heat fraction corresponds to about <u>5400 [s]</u> after reactor trip for Appendix K decay heat. Steam flow in the hot legs should drop below the entrainment threshold after this time.

<u>(2) Counter Current Flow Limitation in hot leg</u>

After the hot leg switch over, hot leg injected water flows from the hot leg to the reactor vessel whereas core vapor flows from the reactor vessel to the steam generators. This creates a counter current flow condition in hot leg. In this situation, it is necessary that sufficient injection flow reaches the reactor vessel so that the injection flow can flush the core.
The counter current flow limitation in hot leg is described as follows (ref-[2]).

[ ] (eq-3)

where

$$Ku_g = \frac{j_g \rho_g^{1/2}}{\left[\sigma g(\rho_f - \rho_g)\right]^{1/4}}$$ Kutateladze number (vapor) (eq-4)

$$Ku_f = \frac{j_f \rho_f^{1/2}}{\left[\sigma g(\rho_f - \rho_g)\right]^{1/4}}$$ Kutateladze number (liquid) (eq-5)

$$j_g = \frac{W_{g-HL}}{A_{HL}\rho_g}$$ Superficial vapor velocity (eq-6)

$$j_f = \frac{W_{f-HL}}{A_{HL}\rho_f}$$ Superficial liquid velocity (eq-7)

$W_{g\text{-}HL}$: Vapor mass flow rate in one hot leg

$W_{f\text{-}HL}$: Liquid mass flow rate in one hot leg

Atmospheric pressure is conservatively assumed as a system pressure. Physical properties are the same as described above.

In this evaluation, the time when hot leg injection flow rate will not be impeded by the countercurrent steam flow is calculated. It is assumed one hot leg injection is active and core boiloff steam flows through all four loops equally. The following equation is satisfied when the hot leg injection flow rate balances core boiloff flow.

$$W_{f-HL} = 4 \times W_{g-HL} \qquad \text{(eq-8)}$$

Consequently the maximum liquid mass flow rate ($W_{f\text{-}HL}$) can be obtained from the following relation.

[ ] (eq-9)

From (eq-4) through (eq-7) and (eq-8), the following equation is obtained.

$$Ku_g = \gamma \cdot Ku_f, \quad \text{with } \gamma = \frac{1}{4}\left(\frac{\rho_f}{\rho_g}\right)^{0.5} \qquad \text{(eq-10)}$$

By substituting (eq-10) into (eq-9), the maximum Kutateladze number for liquid (Ku_f) is,

[ ]

Finally, the maximum superficial liquid velocity and mass flow rate are obtained from (eq-5) and (eq-7).

$$j_f = Ku_f \frac{\left[\sigma g\left(\rho_f - \rho_g\right)\right]^{1/4}}{\rho_f^{1/2}} = 0.227 \text{ [ft/s]}$$

$$W_{f-HL} = j_f \times A_{HL} \times \rho_f = 71.18 \text{ [lbm/s]}$$

The decay heat (P) which results in the core steam flow rate is obtained.

$$P = W_{f-HL}\left(h_g - h_{SI}\right) = 69056.7 \text{ [Btu/s]}$$

where

h_g = 1150.3 [Btu/lbm] specific vapor enthalpy (saturated)

h_{SI} = 180.13 [Btu/lbm] specific liquid enthalpy of injection water (conservatively assumed as saturation)

US-APWR licensed power including uncertainty is 4303108 [Btu/s] (4451x1.02 [MWt]). The decay heat fraction (P/P_0) is obtained.

P/P_0 = [ ]

The decay heat fraction corresponds to about 4560 [s] after reactor trip for Appendix K decay heat. Injection flow can reach the RV without being impeded by CCF in hot leg after this time.

(3) Core Boiloff Compensation

The purpose of the hot leg switch-over is preventing the boric acid precipitation. To dilute the boron concentration in the core, the hot leg injection rate should exceed the core boiloff rate. As mentioned above, one injection line is used for hot leg injection in safety analysis assumption. Hot leg injection flow rate decreases as system pressure increases, thus the highest plausible system pressure for hot leg injection should be assumed conservatively. In this sense, system pressure is assumed to be at the main steam safety valve setting pressure since the hot leg switch-over time is supposed to after one-hour and system pressure is thought to decrease to below the setting pressure by that time.

In this evaluation, the time when hot leg injection flow rate exceeds core boiloff rate is calculated. The temperature of injection water is assumed to be [ ] which envelopes the maximum RWSP water transient temperature in case of LOCA. The following physical properties are used in this evaluation.

P_{sys} = 1200 [psia]	System pressure (main steam safety valve setting pressure)
T_{HL} = [ ]	Hot leg injection water temperature
h_{HL} = [ ]	Specific enthalpy at P=1200 [psia] and T = [ ]
h_g = 1184.2 [Btu/lbm]	Specific enthalpy of saturated vapor at P=1200 [psia]

The minimum hot leg injection flow rate at P=1200 [psia] is 73.99 [lbm/sec]. Core bypass flow rate fraction except upper head bypass is [ ] %, which makes the effective core inlet flow rate (W_{in}),

W_{in} = [ ]

The decay heat (P) which results in the core steam flow rate is obtained.

P = [ ]

US-APWR licensed power including uncertainty is 4303108 [Btu/s] (4451x1.02 [MWt]). The decay heat fraction (P/P_0) is obtained.

P/P_0 = [ ]

The decay heat fraction corresponds to about <u>5600 [s]</u> after reactor trip for Appendix K decay heat. Hot leg injection flow can flush the core after this time.

<u>Conclusion</u>

There are three criteria to determine the earliest hot-leg injection time

From the viewpoint of (1) Liquid entrainment threshold, the earliest time is <u>5400 [s].</u>
From the viewpoint of (2) CCFL in hot leg, the earliest time is <u>4560 [s]</u>.
From the viewpoint of (3) Core boiloff compensation, the earliest time is <u>5600 [s]</u>.

As a result, MHI evaluates the earliest time when the operator can switch-over the injection lines is 100 [min] (6,000 [sec]).

After the hot leg switch-over, core boron concentration begins to decrease, thus the size of control volume is not a problem at that time.

References:
1. Ishii, M.; Grolmes, M. A., Inception Criteria for Droplet Entrainment in Two-Phase Concurrent Film Flow, AIChE Journal, Vol. 21, No. 2, pp. 308-319, 1975.
2. Small Break LOCA Methodology for US-APWR Section 8.1.5, MUAP-07013-P Rev.2 (Proprietary) and MUAP-07013-NP (Non-Proprietary) Rev.2, October 2010.

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.

Table 15.06.05-88-01 Decay Heat Fraction

The decay heat of 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards: "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors", October 1971) is used in accordance with 10CFR50 Appendix K requirements.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

4/15/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 719-5352 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:	**15.06.05**
DATE OF RAI ISSUE:	**3/17/2011**

QUESTION NO.: 15.06.05-90

In the response to RAI Question 15.6.5-56 provided in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1" (July 2009), the issue of inherent boron dilution during small break LOCAs in the US-APWR is discussed. Referring to an evaluation by the applicant, the RAI response cites a minimum core boron concentration required to maintain the reactor subcritical. This value is used as a criterion for assessing the available margin to recriticality following the restart of natural circulation and associated transport of diluted condensate towards the core inlet. It is explained that this value is based on the assumptions stated in the above referenced RAI response. It is also stated that the uncertainty associated with the core criticality evaluation is taken into account.

Provide a full list of reactor core conditions that have been assumed in the criticality calculation for determining the minimum core boron concentration required to maintain the reactor subcritical. In particular, specify the reactor core temperature, reactor coolant pressure, and core life cycle point in time. In addition, quantify any conservative margins included in the calculated minimum core boron concentration such as available shutdown margin and additions to the criticality result for conservatism.

ANSWER:

The reactor core conditions used in the recriticality analysis for the potential boron dilution scenario following SBLOCAs are listed in Table RAI-15.06.05-90.1. The table includes geometric data, core state data, and boron concentration data. In addition, the methodology used to calculate the boron concentration for the recriticality evaluation is described.

As discussed in Table RAI-15.06.05-90.1, various conservatisms are taken into consideration for this safety analysis. The primary conservative assumptions are as follows:

[

]

[

]

The recriticality results are summarized in Table RAI-15.06.05-90.2, where the subcriticality at the beginning of cycle (BOC) and the end of cycle (EOC) of the 24 months core. The boron concentrations used for the recriticality analysis at each core state were derived as described in Table RAI-15.06.05-90.1 using a three-dimensional core simulator[1].

Table RAI-15.06.05-90.2 shows that [

]

Table RAI-15.06.05-90.2 also shows that [

]

Reference:

1. Mitsubishi Heavy Industries, Ltd., Qualification of Nuclear Design Methodology using PARAGON/ANC, MUAP-07019-P (R0), December 2007.

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.

Table RAI-15.06.05-90.1 List of Reactor Core Conditions for Recriticality Evaluation

Table RAI-15.06.05-90.1 List of Reactor Core Conditions for Recriticality Evaluation

Table RAI-15.06.05-90.1 List of Reactor Core Conditions for Recriticality Evaluation

Table RAI-15.06.05-90.2 Recriticality Results for Various Core States