MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

December 2, 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-11416

Subject: 1st 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 "1st MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3".

Enclosed are the Responses to Question 15.06.05-92, 93, 94, 97 and 99 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
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, Ltd.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. 1st MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3 (proprietary)
3. 1st 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-11416

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 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 2nd day of December, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11416
Docket No. 52-021

1st MHI's Responses to US-APWR DCD RAI No. 861-6062 Revision 3

December 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/02/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 861-6062 REVISION 3**
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.: 15.6.05-92

Follow-up to RAI 352-2369, Question 15.6.5-44:
US-APWR Tier 2 Subsection 15.6.5.3.1.3 “Post-LOCA Long Term Cooling Evaluation Model” describes the model that is used to predict the boric acid concentration in the reactor core during the LOCA long term cooling period. A more detailed description of the model is provided in Appendix B to UAP-HF-09384 “MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1” as part of the response to RAI Question 15.6.5-44. Identify the decay heat model used in the US-APWR boron precipitation analyses performed with the long term cooling evaluation model. Provide the decay heat multiplier assumed in the calculations.

The amount of liquid in the mixing volume depends on the predicted vapor volumetric fraction within this volume. The applied axial power profile can impact the volumetric vapor fraction in the core region. What axial power shape was used to determine the volumetric vapor fraction? If a bottom shape axial power distribution was not used, justify why not. Also discuss the impact of possible loop seal plugging, discussed in RAI 706-5339, Question 15.6.5-79.

ANSWER:

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, which is described in US-APWR DCD Tier 2 subsection 15.6.3.1.3.

In the post-LOCA long-term cooling, a uniform axial power profile is assumed to calculate the average core void fraction analytically and the calculation procedure of average core void fraction is described in attachment to appendix B of UAP-HF-09384 “MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1”[ref-(1)]. In addition, the calculated water volume in the core region is reduced by multiplying by a penalty factor that accounts for the void fraction dependence on the axial power distribution variation. This penalty factor is determined based on a bottom-skew shape. In the DCD evaluation, this penalty factor is set to [ ].

To demonstrate the applicability of the penalty factor, the highest possible core average void fraction is calculated from a bottom-skewed power distribution and compared to the core average void fraction calculated from a uniform power distribution.

Bottom-skewed axial power distribution

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 maximum that occurs during the fuel cycle with consideration to the axial flux difference limitation.

Core average void fraction

The core average void fraction is calculated from the bottom-skewed power distribution shown in Fig.1. The calculation procedure of the average core void fraction is described in Appendix-A of this RAI response. The calculation conditions are as follows, which are the same as in the DCD evaluation case.

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

The average void fraction calculation procedure for uniform axial power distribution is described in ref-(1). The axial void fraction distribution in the core 2 hours after the break is shown in Fig.3 for the uniform and bottom-skewed distributions. Table-1 shows the calculated core average void fraction, the ratio of the water volume between the uniform distribution, and bottom-skewed distribution cases 2 hours after the break occurs. As shown in Table-1, the core water mass differential due to the difference of axial power distribution is [ ] which is larger than the penalty factor applied in the DCD evaluation [ ]. The water volume in the mixing volume is estimated to be conservatively low by using the penalty factor, and the boric acid concentration in the mixing volume is calculated to be conservatively high.

Sensitivity to core power

The average core void fraction decreases proportionally to the decrease in the core decay heat, thus the water volume ratio between the uniform distribution case and bottom-skewed case may vary with time. Fig.4 shows the calculated time-history of the average core void fractions and core water volume ratio. The core water volume ratio slightly increases as time goes on and remains above the penalty factor [ ].

Impact of possible loop seal plugging

During the post-LOCA long-term cooling period, it can be said that the core evaporation rate depends on core heat output but does not depend on axial power distribution. Thus, the total loop flow rate does not change even if the axial power distribution changes from uniform to bottom-skewed. Consequently, the impact of possible loop seal plugging with a bottom skewed power profile is the same as described in RAI 706-5339 Question 15.6.5-79.
Regarding the RV differential head ($\Delta P_{HEAD} = \Delta P_{DC} - \Delta P_{CORE}$), the uniform distribution is conservative compared to the bottom-skewed distribution because the core void fraction in the case of a bottom-skewed distribution is larger than the uniform distribution case. Thus, the differential hydraulic head of the core is smaller, resulting in the RV differential head (ΔP_{HEAD}) being larger, and it is expected that froth level in the core and upper plenum will be higher than in the uniform distribution case.

Table-1 Average Core Void Fraction Two Hours after Break

Type	Uniform Distribution	Bottom-skewed Distribution
Average Core Void Fraction	[ ]	[ ]
Core Water Volume 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 average core void fraction (α_{uni}, α_{btm}) and water volume ratio

Reference:

1. UAP-HF-09384 “MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1” (July 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

12/02/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 861-6062 REVISION 3**
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.: 15.6.05-93

Follow-up to RAI 352-2369, Question 15.6.5-45:
According to Appendix B, "Post-LOCA Long Term Cooling Evaluation Model for USAPWR," to the RAI 352-2369, Question 15.6.5-44, response, the boric acid precipitation calculation is initiated at the beginning of the reflood phase to determine the timing of boric acid precipitation. However, the response to RAI 352-2369, Question 15.6.5-45, refers to the point in time when the operators recognize a LOCA to characterize the timing of manual switchover to hot leg injection and states that the related operator action procedures will be provided in future Emergency Procedure Guidelines (EPGs).
Explain how it will be ensured in the EPGs that the timing of such manual switchover to hot leg injection will be defined so that it occurs prior to the point of boric acid precipitation as predicted by post-LOCA long term cooling evaluation model.

ANSWER:

In the US-APWR Emergency Response Guidelines (ERGs), a High Energy Line Break (HELB) is initially identified in the E-0 guideline (Reactor Trip or Safety Injection) by abnormal containment conditions and following that the E-1 guideline (Loss of Reactor or Secondary Coolant) is implemented. If the HELB is a secondary side break inside containment, the operator would terminate SI using the EOS-1.1 guideline (Safety Injection Termination). Boron precipitation is not a concern for these secondary side breaks. If the event is a small break LOCA, the operator will attempt to stop SI pumps sequentially. If the operator is successful at stopping all SI pumps, boron precipitation is not a concern. However, if all SI pumps cannot be stopped due to the size of the LOCA, boron precipitation is a concern. To prevent boron precipitation, the E-1 guideline contains a step to manually switchover to hot leg injection. The E-1 step instructs the operator to close the DVI Isolation valve on an injecting line and open the corresponding hot leg injection isolation valve at the designated time based on the DCD analysis (4 hours). These actions are simple and can be completed quickly. In the E-1 guideline, the time to perform this action is measured from event initiation (i.e. reactor trip and SI). It is expected that the operator will reach this step within the E-1 guideline well before the required action time. Therefore, the structure of these ERGs ensures that manual switchover to hot leg injection will occur before boric acid precipitation.

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

12/02/2011

US-APWR Design Certification

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APPLICATION SECTION:	**15.6.5**
DATE OF RAI ISSUE:	**10/31/2011**

QUESTION NO.: 15.6.05-94

Follow-up to RAI 352-2369, Question 15.6.5-48:
In response to RAI 352-2369, Question 15.6.5-48, it was stated that the mixing volume makeup flow rate is the sum of core evaporation rate and the change rate of the mixing volume water mass. Compare the calculated makeup flow rate using this method with the flow rate that would be obtained from the pressure difference between the downcomer and core. Also compare this flow rate to the total HHIS flow rate. It was also stated in the original RAI response that "In the post-LOCA long-term cooling period, downcomer water level is maintained more than cold leg bottom elevation by HHIS."
Explain how the downcomer water level can be above the cold leg bottom for a large cold leg break (double ended guillotine break).

ANSWER:

In the post-LOCA long-term cooling, downcomer hydraulic head exceeds the core side hydraulic head and a positive flow is established through the core. The core enters a pool boiling condition, under which only steam flows out of the core. To maintain this situation, it is required that the HHIS flow rate to exceed the core make up flow rate, or the core side mixture level decreases and below the top of the fuel.

The core makeup flow is just slightly greater than the core vapor generation rate. The difference is the increase in liquid in the mixing volume with the core and upper plenum. The pressure at the top of the downcomer is equal to the pressure at the top of the upper plenum minus the pressure drop due to steam flow from the upper plenum through the hot legs, steam generators, crossover legs, pumps, and cold legs to the break. The pressure at the top of the upper plenum is equal to the pressure at the top of the downcomer plus the static head of liquid in the downcomer minus the flow pressure drop in the downcomer minus the static head of two phase mixture in the core and upper plenum minus the flow pressure drop through the core and upper plenum. The flow resistance pressure drop in the downcomer is practically negligible since the down-flow velocity through the downcomer becomes very slow, less than 0.5 in/sec, within 30 min. As compared with the downcomer side, the flow resistance through the core to the upper plenum can become much larger because of the two-phase flow effects. The loop flow resistance is evaluated as described in the response to RAI 352-2369, Question 15.6.5-48.

The range of the hydraulic head and flow resistance of the core side is defined to be from the core inlet elevation to the bottom of the hot leg elevation in the upper plenum, corresponding to the upper range of the assumed mixing volume, described in DCD 15.6.5.3.1.3.

The balance of the flow resistance and the hydraulic differential the inside reactor vessel is based on the methodology developed by M. ishii and I. Kataoka for the two-phase natural circulation system. The mass flow rate through the reactor vessel is mathematically expressed as follows:

$$\frac{1}{2}\rho_{m,0}^{2}v_{m,0}^{2}\sum_{i}\left\{\left(K_i+f\frac{L_i}{D_i}\right)\left(\frac{A_0}{A_i}\right)^2\frac{1}{\rho_{m,i}}\right\}=\rho_{SI}\cdot g\cdot\Delta H_{DC}-\sum_{i}\left(\rho_{m,i}\cdot g\cdot\Delta H_i\right) \qquad \text{(eq-1)}$$

where;

ρ_m:	Average fluid density	(lbm/ft^3)
v_m:	Average flow velocity	(ft/sec)
K_i:	Form loss factor	(-)
f:	Friction factor	(-)
L_i:	Length of i	(ft)
D_i:	Hydraulic diameter of node i	(ft)
A_i:	Flow area of node i	(ft^2)
ρ_{SI}:	HHIS Liquid density	(lbm/ft^3)
g:	Gravity acceleration	(ft/sec^2)
ΔH_{DC}:	Vertical length of downcomer	(ft)
ΔH_i:	Vertical length of i (core side)	(ft)

$\sum_{i}\Delta H_i=\Delta H_{DC}$

= (bottom of the cold leg elevation) – (bottom of the core inlet elevation)

The subscripts 0 and i represent the reference component and i-th node, respectively. The reference density ($\rho_{m,0}$) and velocity ($v_{m,0}$) correspond to the core inlet elevation conditions in this evaluation. In deriving the above equation, the dynamic effect on the momentum balance is ignored since the mass flow rate gradually decreases during the post-LOCA long-term period. In this case, saturated water is assumed for HHIS water.

In evaluating the two-phase flow system, the integrate
d momentum equation (eq-1) is represented by a combination of single-phase and two-phase components. Ishii and Kataoka provided the following to account for the two-phase multiplication effect on the frictional and local pressure drops[ref-(1)].

$$\frac{1}{2}\rho_{m,0}^{2}v_{m,0}^{2}\sum_{i}\left(N_{f,i}+N_{o,i}\right)\frac{1}{\rho_{m,i}}=\rho_{SI}\cdot g\cdot\Delta H_{DC}-\sum_{i}\left(\rho_{m,i}\cdot g\cdot\Delta H_i\right) \qquad \text{(eq-2)}$$

The $N_{f,i}$ and $N_{o,i}$ are the friction number and orifice number, respectively. The non-dimensional parameters accounting for the two-phase effect in the friction and local losses are as follows :

$$N_{f,i}=\left(f\frac{L_i}{D_i}\right)\left\{\frac{1+\chi\left(\Delta\rho/\rho_g\right)}{\left\{1+\chi\left(\Delta\mu/\mu_g\right)\right\}^{0.25}}\right\}_i\left(\frac{A_0}{A_i}\right)^2 \qquad \text{(eq-3)}$$

$$N_{o,i}=K_i\left\{1+\chi\left(\Delta\rho/\rho_g\right)\right\}_i\left(\frac{A_0}{A_i}\right)^2 \qquad \text{(eq-4)}$$

where;

$\Delta\rho$:	difference of saturated water and vapor density	(lbm/ft^3)
ρ_g:	saturated vapor density	(lbm/ft^3)
x:	flow quality	(-)
$\Delta\mu$:	difference of liquid and vapor viscosity	(lbm/ft/s)
μ_g:	vapor viscosity	(lbm/ft/s)

To accurately calculate pressure loss in the core side, the core is divided into [ ] nodes and the upper plenum is divided into [ ] nodes vertically. The void fraction (α_i) in each node is calculated as described in appendix-A of the response to RAI 861-6062. A uniform axial power distribution is assumed consistent with the evaluation described in DCD.

From the equations (eq-2),
(eq-3) and (eq-4), the core inlet flow velocity ($v_{m,0}$) is found by iterative convergence calculation. After which, the mass flow rate is calculated from the core inlet velocity.

Fig.1 shows the estimated hydraulic head differential between the downcomer side and the core side. As mentioned above, the flow resistance pressure drop is included in this hydraulic head differential. Fig.2 shows the estimated flow rate driven by the RV pressure differential including loop flow resistance (W_{dP}). Fig.2 shows the calculated core makeup flow rate (W_{makeup}) and minimum HHIS flow rate (W_{SI}). Most of core make up flow is balancing the core evaporation rate, which gradually decreases proportionally to the decrease in core decay heat. In contrast to this, the estimated flow rate driven by the RV pressure differential increases as time passes. This is because the decreasing rate of the pressure loss in the core due to the decrease of steam generation in the core exceeds the decreasing rate of the RV hydraulic head difference.

The HHIS is injected into the upper downcomer through the DVI lines, which are located at the cold leg elevation. The HHIS flow rate exceeds the core makeup flow rate as shown in Fig.2, thus a part of the injected water travels into the core through the downcomer and the lower plenum, and the remaining water flows to the cold leg through the upper downcomer and spills out of the break point which is located at the cold leg. There must be a gradient of liquid depth increasing from the break back toward the downcomer to maintain the flow of the excess HHIS liquid. Therefore, the downcomer water level is maintained above the bottom of the cold leg by the HHIS surplus flow.

Fig.1 Estimated hydraulic head differential between the downcomer side and the core side

Fig.2 Comparison of flow rate

References:

1. M.Ishii and I. Kataoka, 'Similarity Analysis and Scaling Criteria for LWRs under Single-Phase and Two-Phase Natural Circulation', NULEG/CR-3267, March 1983.

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.

12/02/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 861-6062 REVISION 3**
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APPLICATION SECTION:	**15.6.5**
DATE OF RAI ISSUE:	**10/31/2011**

QUESTION NO.: 15.6.05-97

Follow-up to RAI 719-5352, Question 15.6.5-88:
Demonstrate the applicability of the Ishii-Grolmes liquid entrainment onset correlation with regard to the conditions under which it was applied in the analysis provided in the response to RAI 719-5352, Question 15.6.5-88. Consider the impact of possible loop seal plugging, discussed in RAI 706-5339, Question 15.6.5-79, and associated variation in the hot leg steam flow rate on the assessment results for of the earliest time when the operator can switch over to hot leg injection.

ANSWER:

This entrainment correlation is valid for flow conditions where the liquid phase does not take up a significant volume fraction of the pipe and the viscous effects in the liquid are not dominant, meaning that it is valid for the liquid phase in the turbulent regime. In this sense, this correlation applies if there is almost no water in the hot leg(s) before the hot leg switch-over (HLSO). In a case where a significant amount of water level is formed in the hot leg, and this situation is more likely to occur in practice, the counter current flow (CCF) between the hot leg and the steam generator inlet plenum occurs before the HLSO. In that case, it is more suitable to apply the evaluation based on the CCFL correlation described in item (2) of RAI 719-5352 Question No. 15.6.5-88 rather than the evaluation based on Ishii-Grolmes liquid entrainment onset correlation, in terms assessing the validity the HLSO time.

Impact of possible loop seal plugging

During the post-LOCA long term period, there is a possibility that some loops will experience plugging due to the back-flow of ECCS water. In the response to RAI 706-5339 Question 15.6.5-79, it is explained that the number of loops that form a loop seal in the RCP suction leg is determined by the balance between the additional differential pressure due to the loop seal and the flow resistance of the unsealed loop. It is also shown that one loop may become sealed by ECC back flow approximately 30 minutes later, while another loop may become sealed approximately 2 hours later.

This evaluation is conservative but does not consider counter current flow phenomena in the cold leg and RCP suction leg. In practice, it is assumed that loop seal plugging does not occur for any significant period of time by considering counter current flow limitation (CCFL) in the RCP suction uphill side. Considering no loop seal plugging occurs and the core generated steam flows through all four loops, the evaluation of the liquid entrainment threshold time in hot leg described in RAI 706-5339, Question 15.6.5-88 (1), is valid.

Here, the CCFL correlation applied to RCP suction uphill side for US-APWR is Tien's correlation[ref-(1)], which is also adopted in the M-RELAP5 code for US-APWR[ref-(2)]. The correlation is described as follows.

[ ] (eq-1)

where

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

$$Ku_f = \frac{j_f \rho_f^{1/2}}{\{\sigma g(\rho_f - \rho_g)\}^{1/4}}$$ Kutateladze number (liquid) (eq-3)

$$j_g = \frac{W_{g-XL}}{A_{XL}\rho_g}$$ Superficial vapor velocity (eq-4)

$$j_f = \frac{W_{f-XL}}{A_{XL}\rho_f}$$ Superficial liquid velocity (eq-5)

$W_{g\text{-}XL}$: Vapor mass flow rate in one pump suction leg uphill side
$W_{f\text{-}XL}$: Liquid mass flow rate in one pump suction leg uphill side
A_{XL} = [ ] (ft^2) single RCP suction leg flow area

System pressure is assumed to be at atmospheric pressure. The following properties of saturated liquid and vapor phases at atmospheric conditions (14.7 psia) are used.

σ = 0.12988 (lbm/sec^2)	surface tension of liquid
ρ_f = 59.83 (lbm/ft^3)	density of liquid
ρ_g = 0.0373 (lbm/ft^3)	density of vapor

It is assumed that core steam flows into four loops for the initial period of post-LOCA long term cooling, and ECCS back-flow is impeded completely at the RCP suction uphill side due to the CCFL. While the following relation is satisfied, ECCS back-flow cannot enter into the RCP suction leg.

[ ] (eq-6)

By substituting (eq-2) into (eq-6), the following equation is obtained.

$j_g >$ [ ] (eq-7)

Finally, the minimum superficial vapor velocity (j_{g-min}) to maintain the complete CCFL condition can be calculated.

$j_{g\text{-}min} = \left[\quad \right] = 65.96$ (ft/sec)

From (eq-4), the minimum vapor mass flow rate can be obtained.

$W_{g\text{-}XL} = j_{g\text{-}min} \times A_{XL} \times \rho_g = [\quad]$ (lbm/sec)

where

$A_{XL} = [\quad]$ (ft^2) flow area of RCP suction leg

The minimum core steaming flow rate ($W_{boil\text{-}min}$) to keep all four loops in complete CCFL condition is:

$W_{boil\text{-}min} = 4$ (loop) x $W_{g\text{-}XL} = [\quad]$ (lbm/sec)

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

$$P = W_{boil-\min}\left(h_g - h_{SI}\right) = [\quad] \text{ (Btu/s)}$$

where

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

$h_{SI} = 180.13$ (Btu/lbm) specific liquid enthalpy of injection water (saturated)

US-APWR licensed power including uncertainty is 4,303,100 (Btu/s) (4,451x1.02 (MWt)). The decay heat fraction (P/P_0) is obtained.

$P/P_0 = [\quad]$

The decay heat fraction corresponds to about 16,500 (sec) after reactor trip for Appendix K decay heat.

As a result, the complete CCFL at the RCP suction uphill side is maintained for over 4 hours (14,400 sec), no loop plugging occurs, and the core generated steam flows through all four loops prior to the hot-leg switch-over. Therefore, the evaluation of liquid entrainment threshold time in the hot leg described in RAI 706-5339, Question 15.6.5-88 item (1), is still valid.

References:

1. C. L. Tien, K. S. Chung, and C. P. Liu, "Flooding in Two-phase countercurrent flows", EPRI NP-1283, December 1979.
2. Small Break LOCA Methodology for US-APWR Section 8.2, 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.

Impact on Technical/Topical Report

There is no impact on a Technical/Topical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/02/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 861-6062 REVISION 3**

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.: 15.6.05-99

US APWR DCD Tier 2 Section 15.6.5.3.1.3, "Post-LOCA Long Term Cooling Evaluation Model," states that the post-LOCA long term cooling evaluation model is similar to the model described in several references, including the following: "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, "Post LOCA Long Term Cooling Model" due to Discovery of Non-Conservative Modeling Assumptions During Calculations Audit, NRC letter dated November 23, 2005, D.S. Collins to G.C. Bischoff." Provide an explanation of how the subject document relates to the US APWR post-LOCA long term cooling evaluation model and how any non-conservatisms are treated

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ANSWER:

The intention of including the reference, "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, "Post LOCA Long Term Cooling Model" due to Discovery of Non-Conservative Modeling Assumptions During Calculations Audit, NRC letter dated November 23, 2005, D.S. Collins to G.C. Bischoff.", is to show that MHI's evaluation model complies with the NRC's request for the post-LOCA long term cooling evaluation.

The reference explains that the NRC staff no longer approves the use of Westinghouse topical report (TR) CENPD-254-P, "Post-LOCA Long-term Cooling Model". In addition, the 11th line of page 2 states "the following four items will also needed to be addressed by licensees on a plant-specific basis in any future submittals regarding post-LOCA LTC."

The summary of the four items are:

(1) The mixing volume must be justified, and the void fraction must be taken into consideration.

(2) The mixing volume should be treated as a time-dependent quantity, and pressure drop should be considered. The Boric-acid concentration should be maintained below the precipitation limit.

(3) The solubility limit must be justified.

(4) If using 10 CFR Part 50 Appendix K model, the decay heat multiplier must be 1.2 for the

entire time period.

These items are addressed in the current evaluation model, and can be found in Section 15.6.5.3.1.3.

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.

Appendix A

Average Core Void Fraction Calculation Procedure

MITSUBISHI HEAVY INDUSTRIES, LTD.

NOMENCLUTURE

SIMBOL	UNIT	DESCRIPTION
A_{core}	(ft^2)	Core flow area
g	(ft/sec^2)	Gravitational constant of 32.174
$h(i)$	(Btu/lbm)	Fluid average enthalpy at the center of the i^{th} node
h_g	(Btu/lbm)	Saturated vapor density
h_{SI}	(Btu/lbm)	Enthalpy of SI water
$h_{TOP}(i)$	(Btu/lbm)	Fluid average enthalpy at the upper bound of the i^{th} node
$j_g(i)$	(ft/sec)	Superficial rising velocity of steam for i^{th} node.
$j_f(i)$	(ft/sec)	Superficial rising velocity of water for i^{th} node.
K	(-)	Coefficient
$L(i)$	(ft)	Node length for i^{th} node.
m	(-)	Coefficient
$p(i)$	(1/ft)	Relative linear power density for i^{th} node. (As a basis of average core linear power density)
$\Delta P(i)$	(-)	Relative power for i^{th} node.
$\Delta P_N(i)$	(-)	Normalized node power for i^{th} node
$P_N(i)$	(-)	Integrated normalized core power for i^{th} node
Q_{core}	(Btu/sec)	Core power
S_f	(lbm/ft)	Surface Tension
$u_{rb}(i)$	(ft/sec)	Critical rising velocity of void for i^{th} node.
$W_g(i)$	(lbm/sec)	Vapor mass flow for i^{th} node.
$W_f(i)$	(lbm/sec)	Liquid mass flow for i^{th} node.
W_{sys}	(lbm/sec)	Core mass flow
Z_{core}	(ft)	Core height
$z(i)$	(ft)	Elevation from core bottom for i^{th} node.
$\alpha(i)$	(-)	Void fraction for i^{th} node.
α_{AVG}	(-)	Core average void fraction
$\rho_g(i)$	(lbm/ft^3)	Density of saturated steam for i^{th} node
$\rho_f(i)$	(lbm/ft^3)	Density of saturated water for i^{th} node
$\chi(i)$	(-)	Flow path quality for i^{th} node.

1. Introduction

This appendix describes a procedure for calculating a core average void fraction from an arbitrary axial core power profile.

2. Assumption

The following two items are assumed to calculate the core average void fraction during the post-LOCA long term cooling phase.

Assumption (1): Pressure in a core region is assumed to be constant.
Assumption (2): Core flow rate at the inlet of the core region is assumed to be equal to the core coolant evaporation rate.
Assumption (3): There is no super heat in the core.

3. Calculation Procedure

In the post-reflood long term cooling phase, the lower plenum is fully filled with water injected from the Safety Injection (SI) and ECCS systems. The temperature of the water injected into the core is considered to be approximately equal to the SI water. Therefore, the core mass flow at the inlet of the core can be estimated by the following equation according to Assumption (2) described above.

$$W_{SYS} = \frac{Q_{CORE}}{h_g - h_{SI}} \tag{1}$$

Where

Q_{core}	: Core power
h_g	: Saturated vapor density
h_{SI}	: Enthalpy of SI water
W_{sys}	: Core mass flow

As shown in Figure 1, this axial power profile consists of a data set of relative linear power densities at many representative points along the axial core height, each of which corresponds to an averaged value of the relative linear power density for each node, representing the core nodalization. Based on the axial core power profile mentioned above, the relative power for each node can be calculated by the following equations (2) and (3).

Elevation from the core bottom : $z(i)$ $i = 1 \sim n$
Relative linear power density : $p(i)$ $i = 1 \sim n$
(As a basis of average core linear power density)
Core height : Z_{core}

Figure 2 shows the schematic diagram for the relationship between the node size and the node elevation of primary interest. The length for ith node is given as follows.

$$L(i) = \frac{z(i+1) - z(i-1)}{2} \tag{2}$$

Where

If i=1, $L(1) = \dfrac{z(2) + z(1)}{2}$

If i=n, $$L(i)= Z_{CORE} - \frac{z(i-1)+z(i)}{2}$$

The relative power for each node can be calculated as a product of the node length and its relative linear power density, which is described in the following equation.

$$\Delta P(i) = L(i) \times p(i) \tag{3}$$

For the sake of convenience, two non-dimensional parameters, a normalized node power $\Delta P_N(i)$ and an integrated normalized core power $P_N(i)$ are introduced to calculate a fluid average enthalpy for each node. The normalized node power for the ith node can be obtained by dividing the relative power for the ith node by a total relative power integrated from the 1st to last nodes, whereas the integrated normalized core power for the ith node can be obtained by integrating the normalized node powers from the 1st to ith nodes. These non-dimensional parameters are defined in the following equations.

$$\Delta P_N(i) = \frac{\Delta P(i)}{\sum_{j=1}^{n}\{\Delta P(j)\}} \tag{4}$$

$$P_N(i) = \sum_{j=1}^{i}\{\Delta P_N(j)\} \tag{5}$$

Using equation (5), the fluid average enthalpy at the top of the ith node can be calculated in equation (6). The one at the center of the same node is defined as the averaged value of the upper and the lower bound of the node, which is described in equation (7).

$$h_{TOP}(i) = h_{SI} + (h_g - h_{SI})P_N(i) \tag{6}$$

$$h(i) = \frac{h_{TOP}(i-1) + h_{TOP}(i)}{2} \tag{7}$$

A flow path quality for each node is defined as a thermal equilibrium quality, which can be calculated from an average fluid enthalpy for each node as shown in the following equation. Since the value of the thermal equilibrium quality can be negative under a condition of a subcooled fluid state, zero is applied to the thermal equilibrium quality in case of the negative quality.

$$\chi(i) = MAX\left\{\frac{h(i) - h_f}{h_g - h_f}, 0.0\right\} \tag{8}$$

Using the core mass flow and the flow path quality described in equations (1) and (8), a vapor mass flow and a superficial vapor velocity for the ith node can be obtained as equations (9) and (10). Similarly, a liquid mass flow and a superficial liquid velocity for the ith node can be obtained as equations (11) and (12).

$$W_g(i) = W_{SYS} \times \chi(i) \tag{9}$$

$$j_g(i) = \frac{W_g(i)}{A_{CORE} \times \rho_g} \tag{10}$$

$$W_f(i) = W_{SYS} \times \{1.0 - \chi(i)\} \tag{11}$$

$$j_f(i) = \frac{W_f(i)}{A_{CORE} \times \rho_f} \tag{12}$$

Substituting the superficial vapor / liquid velocity calculated from equations (10) and (12) into the modified Yeh correlation equation which is defined in equation (13), the void fraction for the ith node can be obtained as follows.

$$\alpha(i) = K \times \left(\frac{\rho_g(i)}{\rho_f(i)}\right)^{0.239} \times \left(\frac{j_g(i)}{u_{rb}(i)}\right)^{m} \times \left(\frac{j_g(i)}{j_g(i) + j_f(i)}\right)^{0.6} \tag{13}$$

$$u_{rb} = 1.53 \times \left(\frac{S_f \times (\rho_f(i) - \rho_g(i)) \times g}{\rho_f(i)^2}\right)^{0.25} \tag{14}$$

K=0.925, m=0.67, if $\left(\frac{j_g(i)}{u_{rb}(i)}\right) < 1$

K=0.925, m=0.47, if $1 < \left(\frac{j_g(i)}{u_{rb}(i)}\right) < 4.31$

K=1.035, m=0.393, if $4.31 < \left(\frac{j_g(i)}{u_{rb}(i)}\right)$

Where

$\alpha(i)$	: Void fraction for ith node
K	: Coefficient
$\rho_g(i)$	: Density of saturated steam for ith node
$\rho_f(i)$	: Density of saturated water for ith node
$j_g(i)$	: Superficial rising velocity of steam for ith node
$j_f(i)$	: Superficial rising velocity of water for ith node
$u_{rb}(i)$	: Critical rising velocity of void for ith node
m	: Coefficient
g	: gravitational constant
S_f	: Surface Tension

This void fraction represents an averaged value of void fraction for each node. The length of each node is different. Hence, the core average void fraction can be evaluated as a weighted-averaged value of the node length, which can be described by the following equation.

$$\alpha_{AVG} = \frac{\sum_{i=1}^{n} \{\alpha(i) \times L(i)\}}{Z_{CORE}} \tag{15}$$

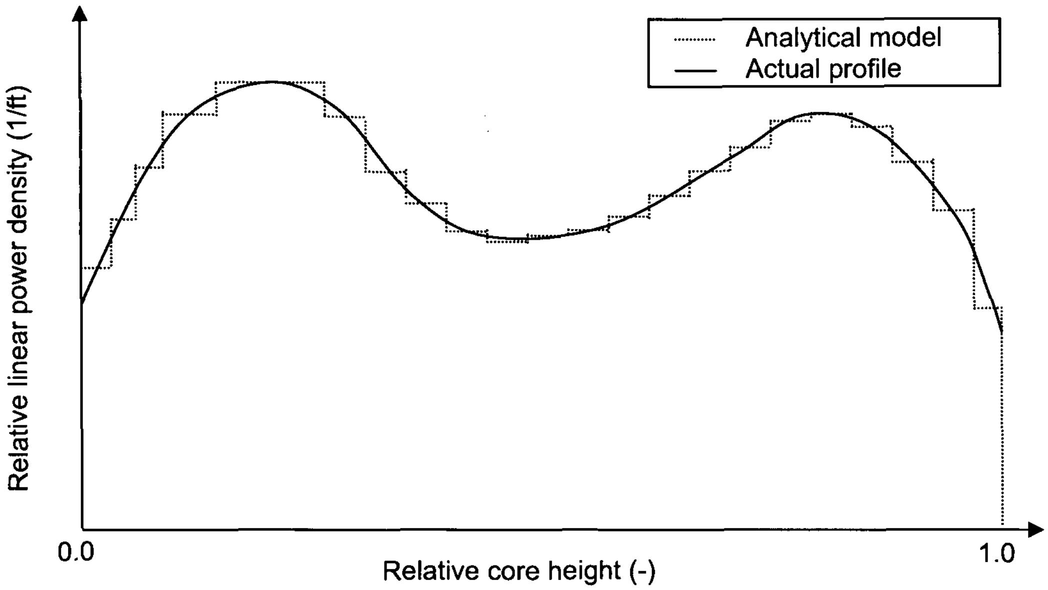

Figure 1 Axial core power profile based on a relative linear power density, and its analytical model

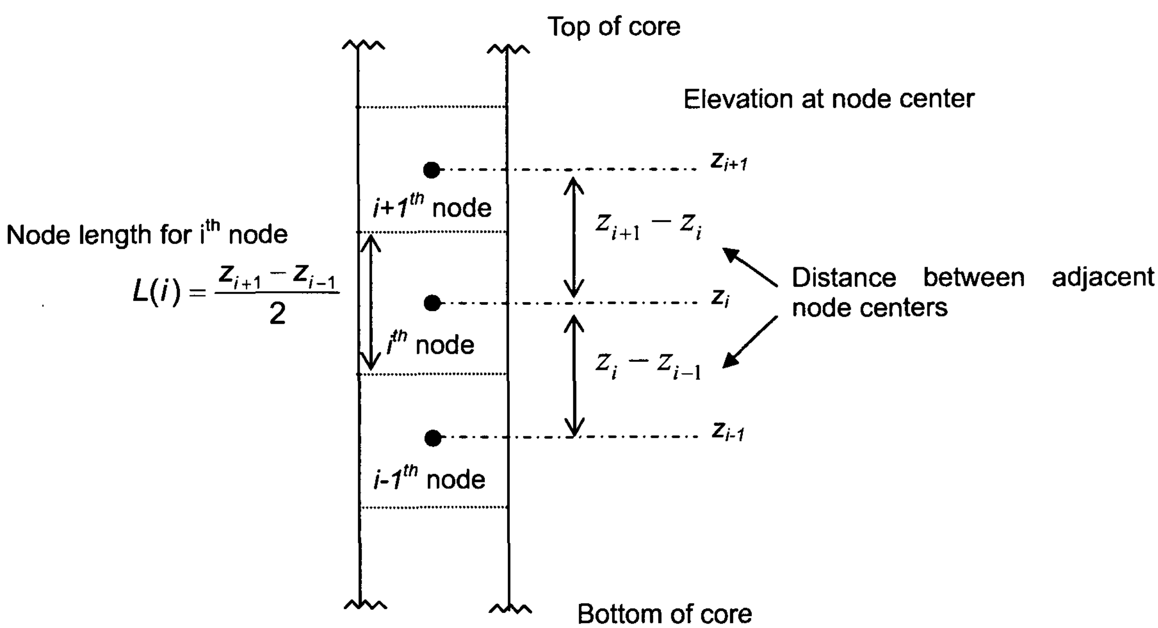

Figure 2 Schematic diagram for the relationship between node size and node elevation