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TOKYO, JAPAN

July 10, 2009

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09362

Subject: **MHI's 1st Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 06/11/2009**

Reference: 1) "REQUEST FOR ADDITIONAL INFORMATION ON TOPICAL REPORT MUAP-07013-P, 'SMALL BREAK LOCA METHODOLOGY FOR US-APWR'," dated June 11, 2009.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") an official document entitled 'MHI's 1st Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 06/11/2009'. In the enclosed document, MHI provides the 20 (twenty) out of 31 (thirty-one) items requested in Reference 1. The remaining responses to the RAI in Reference 1 will be transmitted to the NRC by separate correspondence, no later than August 10, 2009 (60 days after the issuance of the formal RAI), as agreed by NRC and MHI.

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), proprietary data recorded on CD (Enclosure 4), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the bases of MHI request that all materials designated as "Proprietary" in Enclosure 2 and all data recorded on CD of Enclosure 4 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.

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NRO

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata

2. MHI's 1st Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 06/11/2009 (proprietary)

3. MHI's 1st Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 06/11/2009 (non-proprietary)

4. CD: "M-RELAP5 Input Data for the MHI Advanced Accumulator Test Analysis"

The files contained in this CD are listed in Attachment 1.

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

Contact Information
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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-09362

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 "MHI's Partial Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) 'Small Break LOCA Methodology for US-APWR'" 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). Similarly, all data recorded on the enclosed CD 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 10th day of July, 2009.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-09362

MHI's 1st Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 06/11/2009

July 2009
(Non-Proprietary)

REQUEST 1-2
(Related RAI 1-1)

Scaling of the test facility relative to the US-APWR needs to be addressed. In compliance with Step 6 of the evaluation model development and assessment process (EMDAP), as identified in Regulatory Guide 1.203, provide quantitative scaling analysis to ensure that the data from separate effects tests (SET) and integral effects tests (IET), and the models based on those data, will be applicable to the analysis of the US-APWR.

a) Identify non-dimensional groups that govern the physical phenomena to be examined by the test facilities and compare the similitude between the facilities and the US-APWR.
b) Based on the US-APWR-specific scaling analysis, address Step 8 of the EMDAP (evaluate the effects of IET distortions and SET scale up capability).
c) Assess scalability of models (Step 15 of the EMDAP) - this is to demonstrate that models and correlations implemented in M-RELAP5 for the PIRT high ranking phenomena are appropriate for the SBLOCA evaluation specific to the configuration and conditions of the US-APWR.

Examples of quantitative scaling analysis are:

a. Novak Zuber, Wolfgang Wulff, Upendra S. Rohatgi and Ivan Catton, "Application of Fractional Scaling Analysis (FSA) to Loss of Coolant Accidents (LOCA), Part 1: Methodology Development", Paper: 153, 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Popes' Palace Conference Center, Avignon, France, October 2-6, 2005.
b. Wolfgang Wulff, Novak Zuber, Upendra S. Rohatgi and Ivan Catton, "Application of Fractional Scaling Analysis (FSA) to Loss of Coolant Accidents (LOCA), Part 2. System Level Scaling for System Depressurization", Paper: 111, 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Popes' Palace Conference Center, Avignon, France, October 2-6, 2005.
c. Ivan Catton, Wolfgang Wulff, Novak Zuber, and Upendra S. Rohatgi, "Application of Fractional Scaling Analysis (FSA) to Loss of Coolant Accidents (LOCA), Part 3. Component Level Scaling for Peak Clad Temperature", Paper: 055, 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Popes' Palace Conference Center, Avignon, France, October 2-6, 2005.
d. Wolfgang Wulff and Upendra S. Rohatgi, "System Scaling for the Westinghouse AP600 Pressurized Water Reactor and Related Test Facilities, Analysis and Results", NUREG/CR-5541, January 1998.
e. J. N. Reyes, Jr., Qiao Wu and John B. King, Jr. "Scaling Assessment for the Design of the OSU APEX-1000 Test Facility", OSU-APEX-03001 (Rev. 0), May 12, 2003.
f. Revision 1: R. E. Gamble, A. F. Fanning and V. Chandola, Revision 2: P. Saha, "ESBWR Scaling Report", NEDO-33082, Revision 1, December 2002, Revision 2, April 2008.
g. S. Banerjee, M. G. Ortiz, T. K. Larson and D. L. Reeder, "Top Down Scaling Analyses Methodology for AP600 Integral Tests", INEL-96/0040, May 1997.

RESPONSE

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REQUEST 4-13
(Related RAI 4-1)

The heat transfer logic at the end of blowdown must be precisely defined in the documentation so that it can be verified against the code and the analysis.

RESPONSE

The heat transfer mode selection logic to prevent the return to nucleate boiling once CHF has been predicted during the blowdown and to prevent the return to transition boiling after the cladding surface superheat exceed 300 R during the blowdown is introduced in M-RELAP5 to satisfy the Appendix K requirements. The M-RELAP5 modifications for these Appendix K requirements are as follows.

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REQUEST 4-14

(Related RAI 4-7)

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RESPONSE

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REQUEST 4-15
(Related RAI 4-10)

PIRT Tables have some rankings that are break size dependent. Is the ranking in PIRT Table (4.3.2-1, MUAP-07013-P) the highest value for all break sizes? If not, please revise the table and provide the highest ranking for each phenomenon. Also, is there a method (for example a sensitivity analyses) of confirming the ranking in the PIRT?

RESPONSE

Section 4.3-1 of MUAP-07013 states that the (*) ranking implies that the ranking is break size dependent and that it may be insignificant for certain break sizes. Therefore, the (*) ranking indicates that the table represents the highest value for all transients. It is emphasized here that MHI's ranking are primarily expert based, combining the technical know-how of internal MHI experts and international LOCA experts. This method is primarily applied to confirm the ranking in the PIRT.

Furthermore, a comprehensive quantitative scaling analysis for the US-APWR is being performed as discussed in the responses to RAIs 1-2 and 5-2. In the process of scaling analysis, the importance of each phenomenon addressed in the PIRT will be quantitatively examined.

REQUEST 5-2
(Related RAI 5-1)

The MHI response does not provide quantitative scaling analysis that supports their claim that the differences between the US-APWR and a conventional PWR in active core height and number of grid spacers is negligible.

Provide M-RELAP5 validation runs for the prediction of CHF and PCT in a 12-ft fuel assembly and the scaling analysis that demonstrates the validity of extending the M-RELAP5 capability to predict CHF and PCT in a 14-ft fuel assembly typical of the US-APWR core.

RESPONSE
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REQUEST 6-5
(Related RAI 6-2)

To satisfy TMI action plan requirements in NUREG-0737 requires assessing against Semiscale and LOFT data. MHI is requested to include Semiscale test S-UT-8 and LOFT test L3-1 in their assessment matrix.

RESPONSE
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REQUEST 6-6
Related RAI 6-2

Based on the information provided in UAP-HF-09041-P, "MHI's 2nd Part Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (RO) "Small Break LOCA Methodology for US-APWR", February 2009, in response to Request 6.2, in Table 5, "Summary of Validation Status for M-RELAP5-Specific Models," it is not clear how the assessments were performed. Provide a table that lists all Appendix K features that were used for each of the assessment cases

RESPONSE

The M-RELAP5, which is based on the RELAP5-3D, in its current state, has already contained a number of models that enable it to meet many of the Appendix K requirements with no modification. Therefore, many of the Appendix K requirements can be met by simply providing the appropriate input in the plant model. This includes the appropriate plant nodalization together with appropriate initial conditions, boundary conditions, and the proper code options and also performing sensitivity calculations, if necessary. However, some Appendix K requirements can only be achieved through the implementation of new models or the modification of existing RELAP5-3D models. A few models must be also validated by additional comparison with appropriate test data to confirm the applicability of the models to Small Break LOCA Evaluation Model calculations for the US-APWR.

Actions to conform to Appendix K requirements are divided into three categories:

Category 1: required no action (addressed by conservative input data)
Category 2: required models are missing and need to be added into M-RELAP5 with appropriate verification
Category 3: additional validation using test data needs to be performed before utilized in M-RELAP5

Table RAI-6-6.1 lists the new functions in M-RELAP5 that based on the Appendix K models utilized in the assessment cases in the US-APWR SBLOCA analyses.

Table RAI-6-6.1 Appendix K Models used for the Assessment Cases of US-APWR SBLOCA Analyses *(Only those New Functions in M-RELAP5 described here)*

No.	Appendix-K Models Implemented in M-RELAP5	Category	Assessment Approach	Test Facilities
1	Initial Stored Energy (FINE Code Gap Conductance Model)	1	Provide appropriate input. Steady state temperature distribution and stored energy in the fuel are calculated for the burn-up that yield highest PCT.	None
2	Fission Product Decay (ANS Standard 1971 Decay Heat Model)	2	ANS Standard 1971 is installed. Fission product decay heat shall be 1.2 times the values of infinite operating time in the ANS Standard 1971.	None
3	No Return to Transition Boiling/Nucleate Boiling during Blowdown Phase	2	Coding modification. Return to transition boiling or nucleate boiling is not allowed during the blowdown phase.	None
4	Metal Water Reaction Rate (Baker-Just Model)	2	Coding modification. Baker-Just equation is installed. Influence of the metal/water reaction was calculated. The reaction was assumed non-steam limited and the inside of the cladding reacts after the rupture.	None
5	Cladding Swelling and Rupture (ZIRLO™ Cladding Swelling/Rapture Model)	2	Coding modification. The gap conductance was varied in accordance with changes in gap dimensions and other applicable variables.	None
6	Discharge Model (Moody Critical Flow Model)	2	Sensitivity study was performed. Two-phase discharge rate was calculated using the Moody model with at least three (3) values of discharge coefficient.	None
7	CHF and Post-CHF Heat Transfer Models (Modification to Post-CHF Heat Transfer Logic)	3	Coding modification and assessment were performed. The conservative modifications were validated through benchmarking using the test data obtained from the THTF uncovered heat transfer and reflood tests, and from the ROSA-IV/ LSTF SBLOCA test.	ORNL-THTF ROSA-IV/LSTF

REQUEST 7-13
(Related RAI 7-1)

Does the M-RELAP5 model of the US-APWR include a bypass flow path between the upper head and the downcomer that could potentially allow steam to leak from the core region? If this bypass flow path is present in the model evaluate its effect on the progression of the SBLOCA and the potential for the bypassed steam to sweep some of the ECC water out the break.

RESPONSE

The bypass flow path between the upper head and the downcomer is considered in the US-APWR SBLOCA analyses. This flow path is modeled by [ ] as shown in **Figure RAI-7-13.1**.

As the loops are sealed at the cross-over legs during the loop seal clearance period, the bypass flow path between the upper head and the downcomer has the potential for relieving vapor generated in the core. If the vapor is relieved through this bypass flow path, the pressure above the core is reduced and then the core liquid level depression is reduced. The reduction of the liquid level depression lowers the cladding temperature when the core is uncovered. In this way, there is a possibility that the bypass flow path affects the PCT during the loop seal clearance period.

However, a large amount of liquid still remains in the upper head, and the bypass flow path is covered by the liquid during the loop seal clearance period for the break sizes in which the core is uncovered. Then, there is essentially no vapor relief through the bypass flow path during the loop seal clearance period, and the PCT is hardly affected by the bypass flow path. The collapsed liquid level transients in the upper head are shown in Figure RAI-7-1.3 of the response to RAI 7-1 (Ref. 1).

When the liquid level in the upper head is lower than the top of the spray nozzles, the vapor generated in the core flows directly into the upper downcomer and from there towards the broken cold leg. As ECC water from the high head safety injection systems is injected into the downcomer in the downward direction below the cold legs by the flow guide, the vapor flow from the spray nozzles does not impede this ECC water flow into the core. For the larger break sizes, the accumulator injects ECC water when the primary pressure decreases. The vapor from the intact steam generator sweeps part of this ECC water out the break. There remains a possibility that the vapor from the spray nozzles also sweeps part of the ECC water out the break. The vapor mass flow at the outlet of the intact reactor coolant pumps and at the spray nozzles for the 1.0ft^2 cold break is shown in **Figure RAI-7-13.2.** As the vapor flow from the spray nozzles is negligible compared with the vapor flow from the intact loop, its effect on the ECC water is considered to be small.

The bypass flow path between the upper head and the downcomer is considered in the US-APWR SBLOCA analyses, but its effect on the PCT or on the sweep of ECC water out the break is considered to be small.

References:

1. Mitsubishi Heavy Industry, Ltd., MHI's Partial Responses to NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR", UAP-HF-09002, January 16, 2009.

Figure RAI-7-13.1 Nodalization Scheme of Reactor Vessel

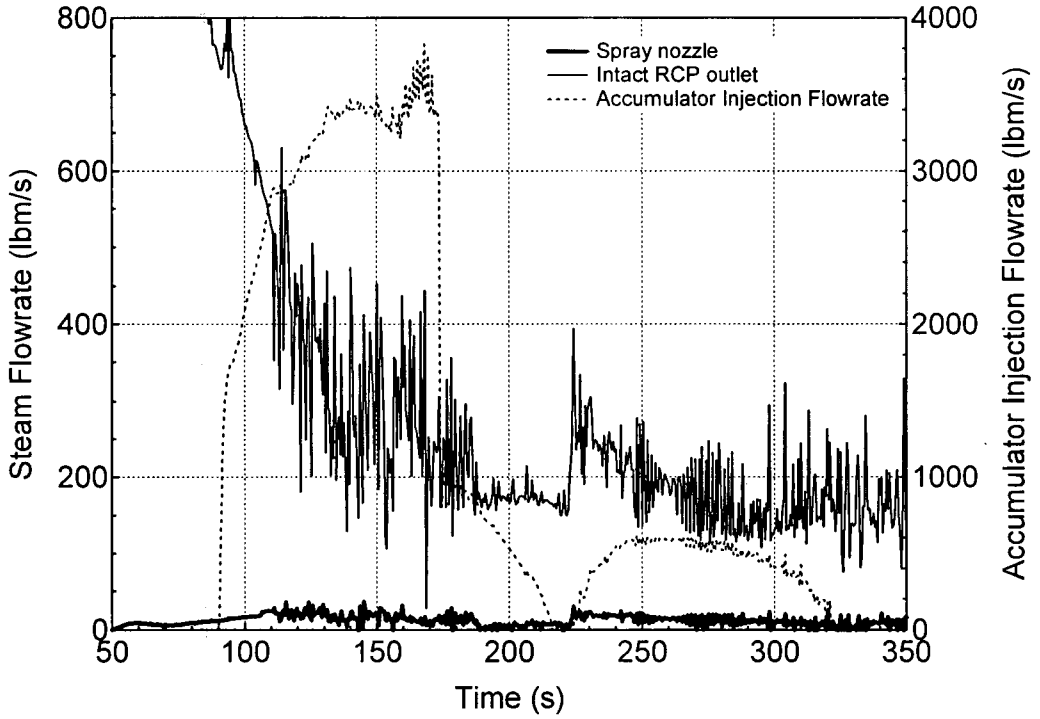

Figure RAI-7-13.2 Steam Flowrate at Intact RCP Outlets and at Spray Nozzles

REQUEST 7-14

The break size spectrum for the APWR is different than previously considered by the NRC for SBLOCA, since the limiting case is at the high end of the range (1.0 ft^2). For PWRs limiting breaks have historically been in the range of 2 to 4 inches. It is not certain that the APWR break size range can be characterized fully by SBLOCA phenomena. To assist the NRC in determining whether the governing phenomena are more like a SBLOCA or LBOCA (e.g. to resolve end of bypass, FLECHT heat transfer applicability, amount of dissolved Nitrogen in RCS from accumulator based on accumulator water level) the following information is requested:

For the limiting 1.0 ft^2 break and the 7.5 in break, provide plots of the following parameters:

a. the cold leg to downcomer mass rate for each loop,
b. the downcomer to lower plenum mass flow rate,
c. the mass flow rate into and exiting the average and hot channel (vapor and liquid components as appropriate),
d. the core bypass inlet mass flow rate,
e. the neutron reflector inlet mass flow rate,
f. the accumulator water levels as functions of time,
g. the equivalent two-phase levels in the average and hot channel as functions of time,
h. the peak cladding temperature as a function of time and identify the SBLOCA phase (i.e., blowdown, loop-seal clearing, natural circulation, etc.) time spans on the plots.
i. the heat transfer coefficient at the peak clad temperature location and the heat transfer correlation identifier and annotate the plot to indicate the time when an Appendix K heat transfer lockout occurs.

RESPONSE

The requested data are provided in **Figures RAI-7-14.1 to RAI-7-14.26**.

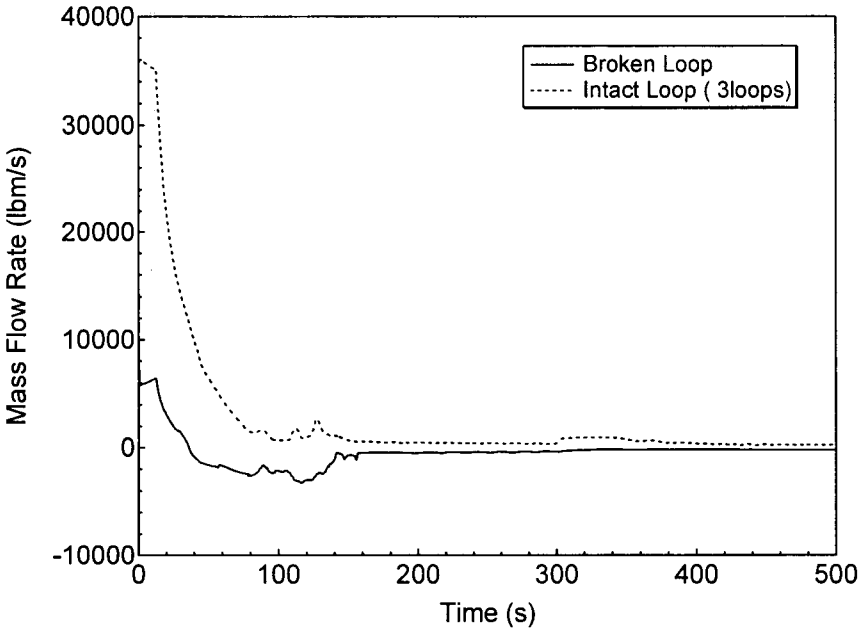

Figure RAI-7-14.1 Cold-leg to Downcomer Mass Flowrates for 7.5-inch Break

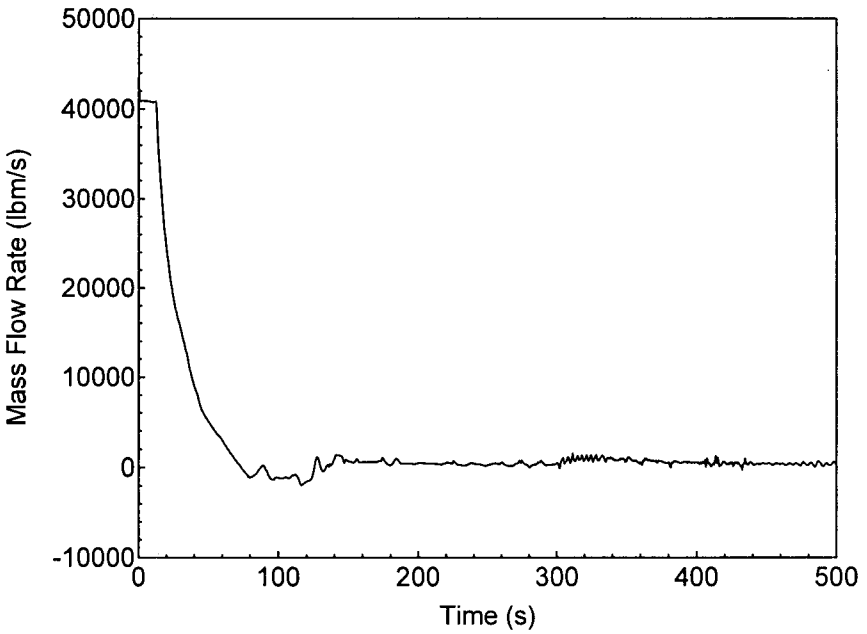

Figure RAI-7-14.2 Downcomer to Lower Plenum Mass Flowrate for 7.5-inch Break

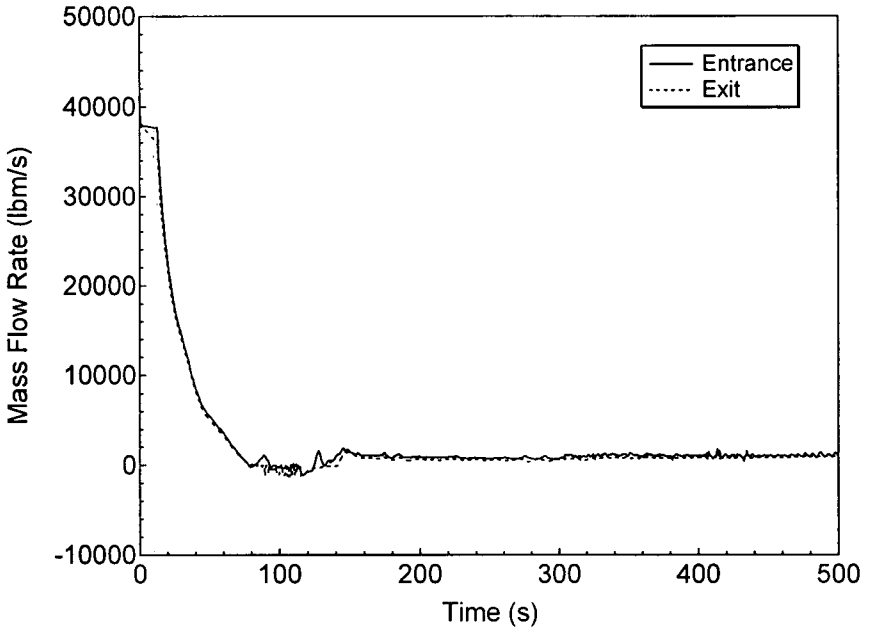

Figure RAI-7-14.3 Average Core Entrance and Exit Liquid Mass Flowrates for 7.5-inch Break

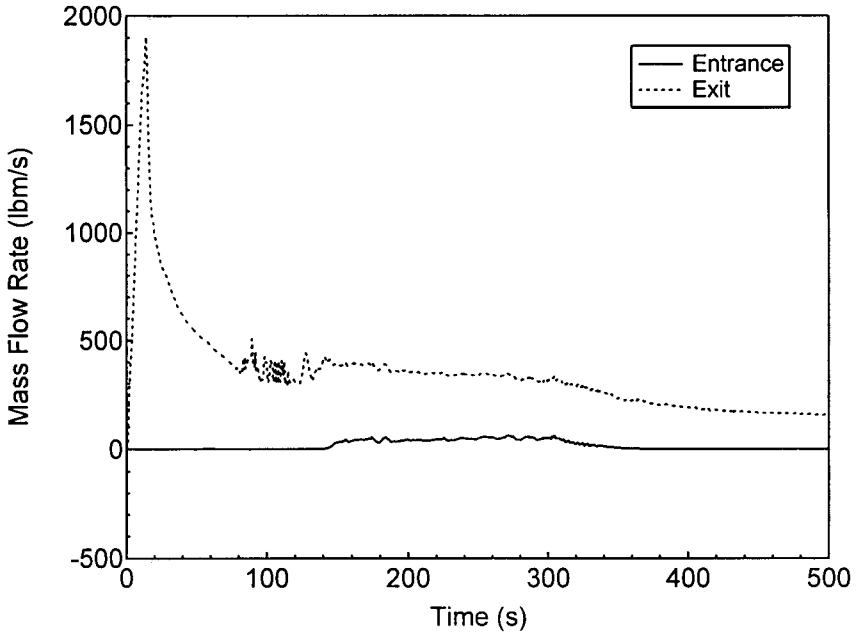

Figure RAI-7-14.4 Average Core Entrance and Exit Vapor Mass Flowrates for 7.5-inch Break

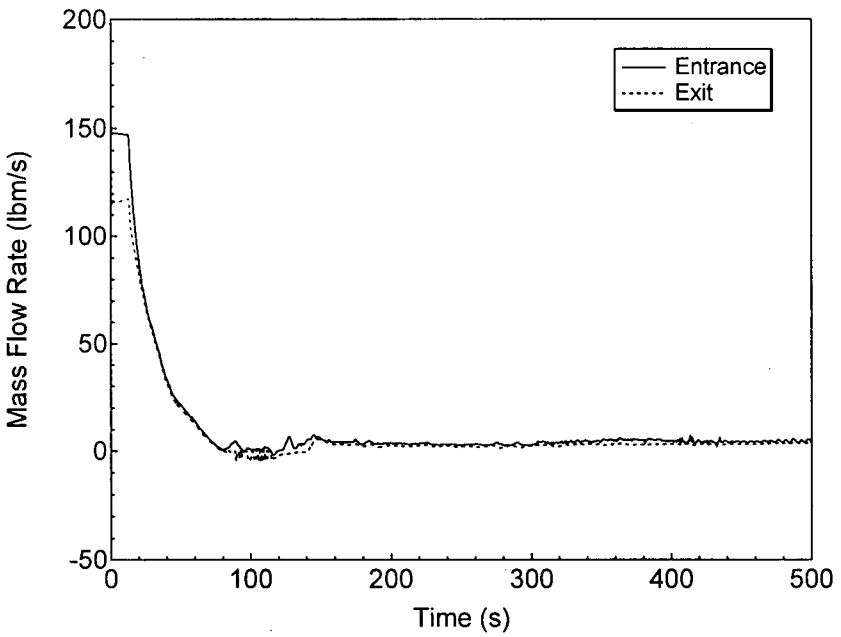

Figure RAI-7-14.5 Hot Assembly Entrance and Exit Liquid Mass Flowrates for 7.5-inch Break

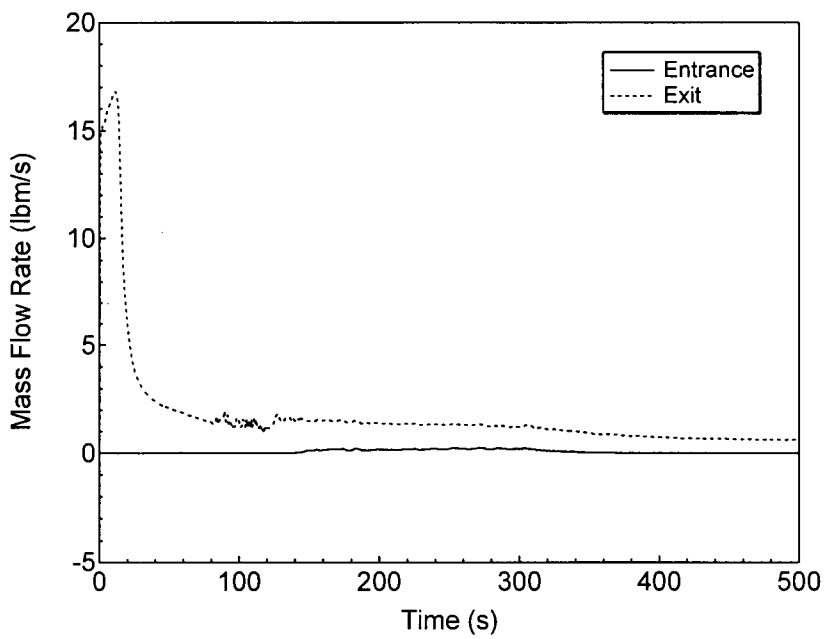

Figure RAI-7-14.6 Hot Assembly Entrance and Exit Vapor Mass Flowrates for 7.5-inch Break

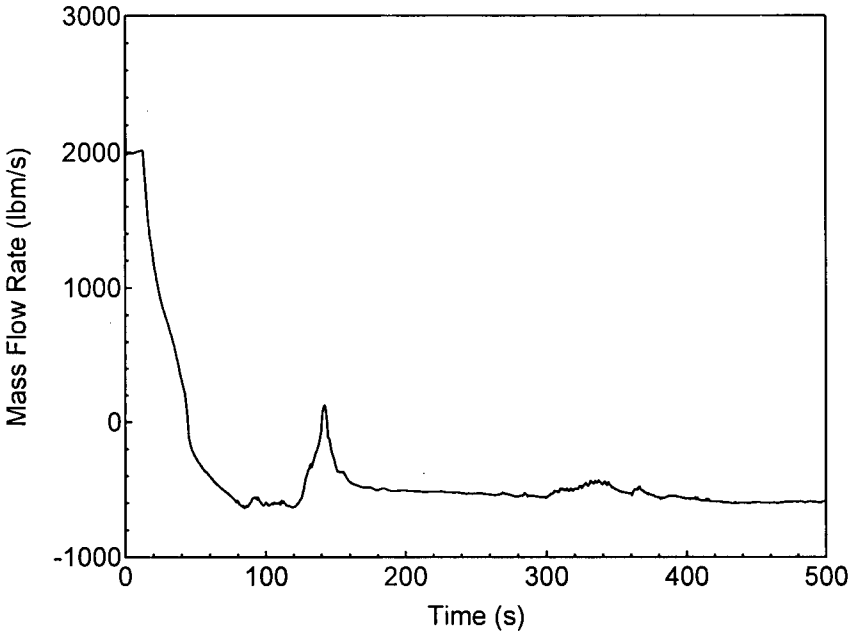

Figure RAI-7-14.7 Core Bypass Entrance Mass Flowrate for 7.5-inch Break

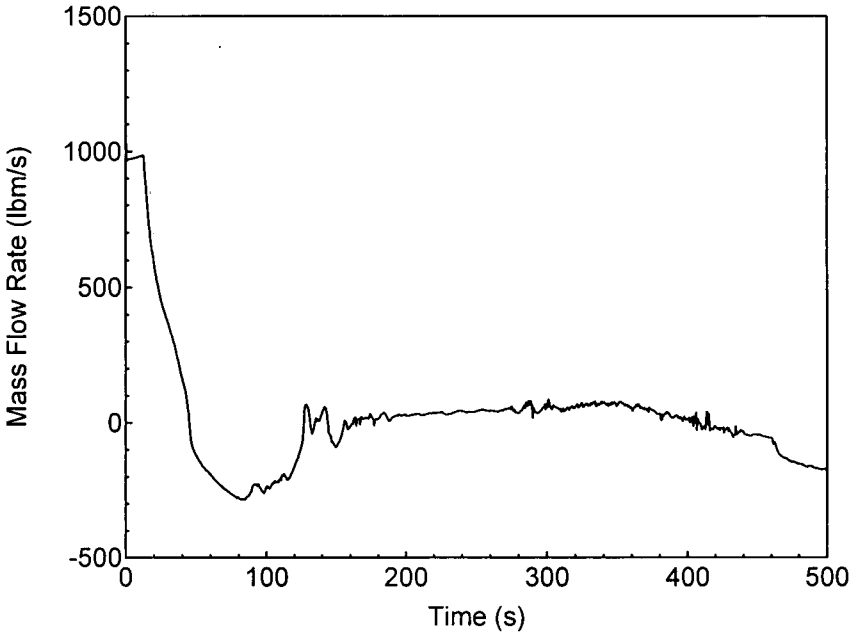

Figure RAI-7-14.8 Neutron Reflector Entrance Mass Flowrate for 7.5-inch Break

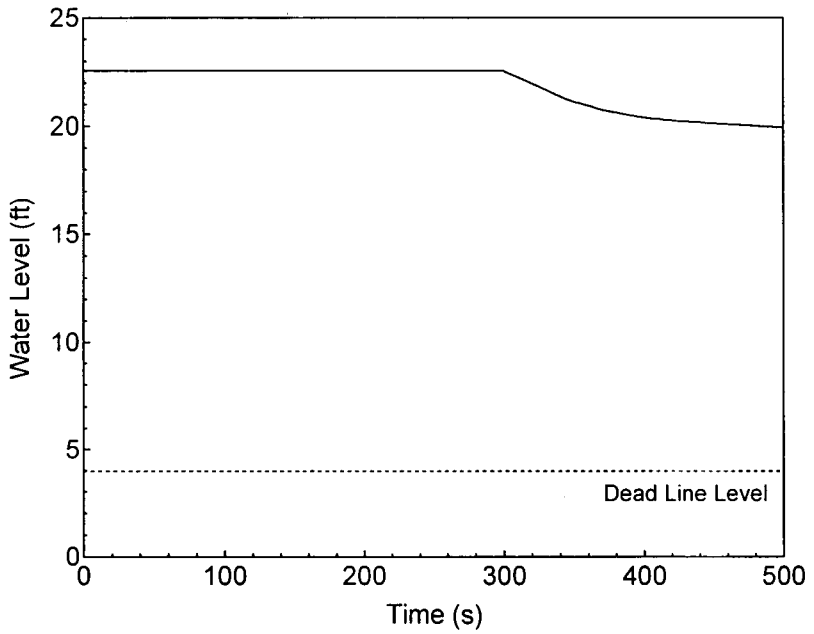

Figure RAI-7-14.9 Accumulator Water Level for 7.5-inch Break

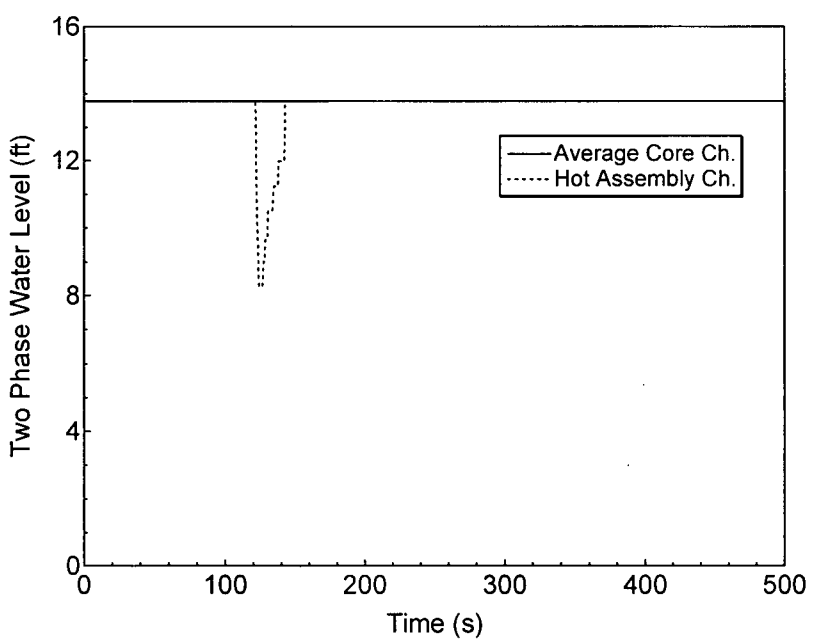

Figure RAI-7-14.10 Core Two-Phase Level for 7.5-inch Break

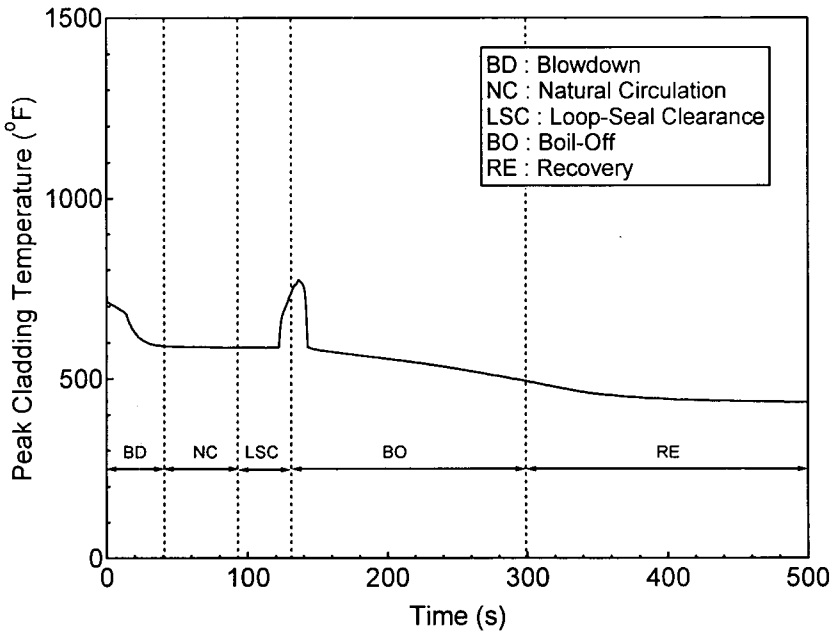

Figure RAI-7-14.11 PCT at ALL Elevations for Hot Rod in Hot Assembly for 7.5-inch Break

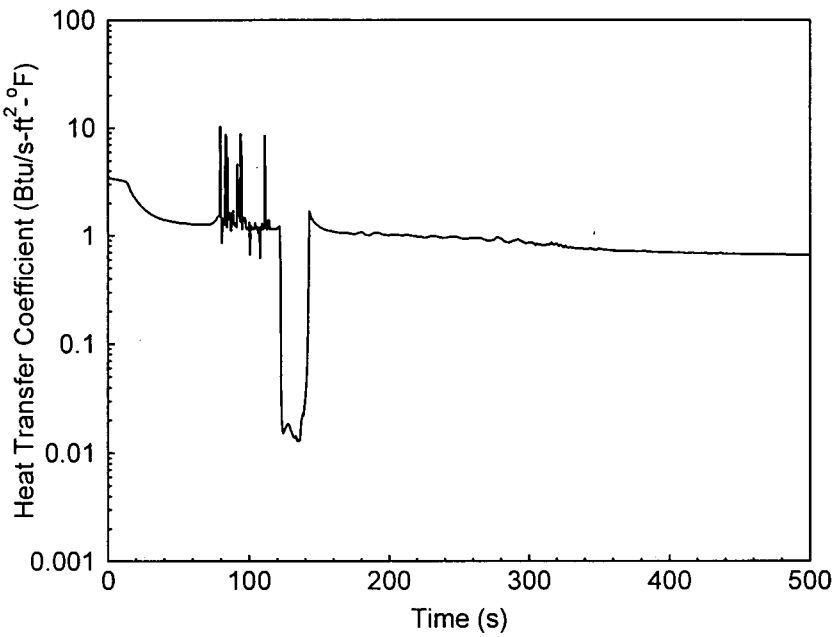

Figure RAI-7-14.12 Heat Transfer Coefficient at PCT Location for 7.5-inch Break

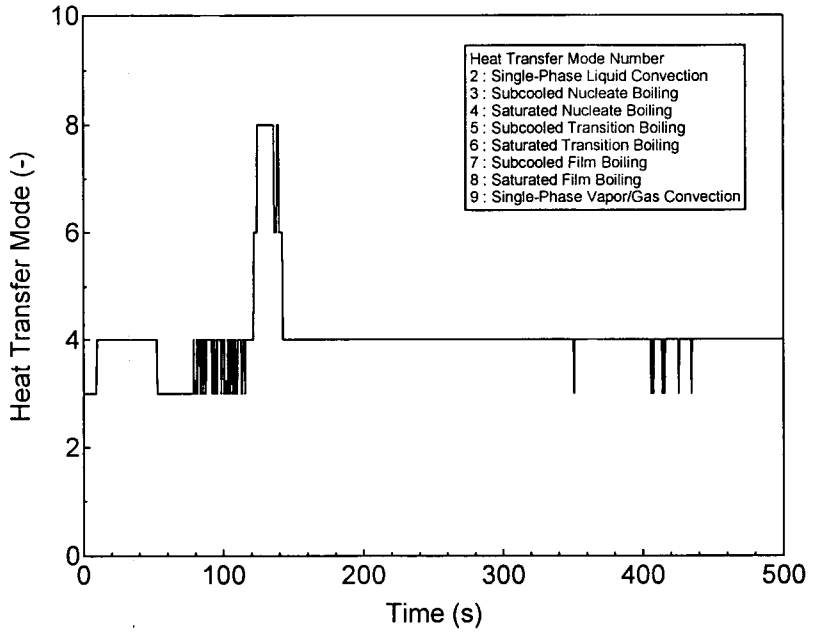

Figure RAI-7-14.13 Heat Transfer Mode at PCT Location for 7.5-inch Break

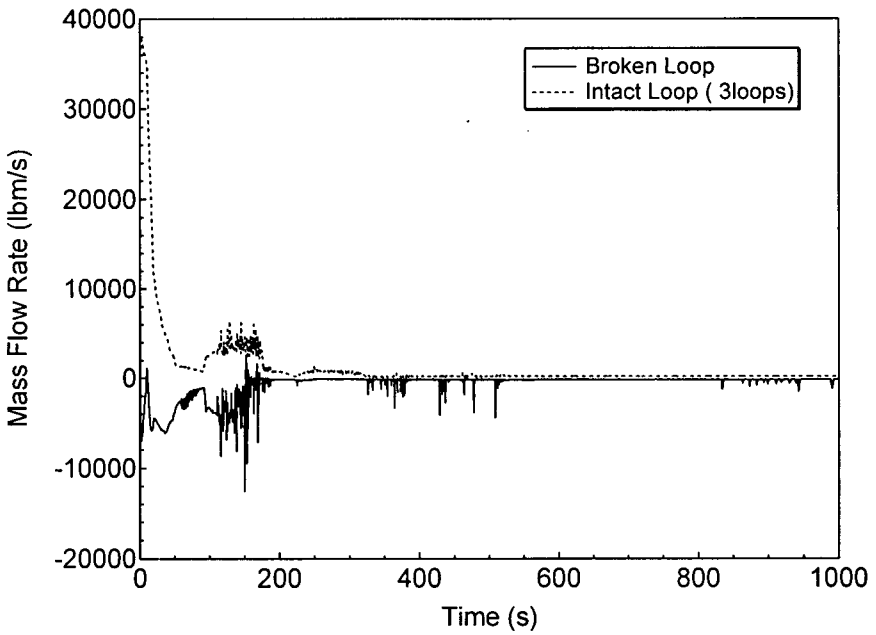

Figure RAI-7-14.14 Cold-leg to Downcomer Mass Flowrates for 1-ft^2 Break

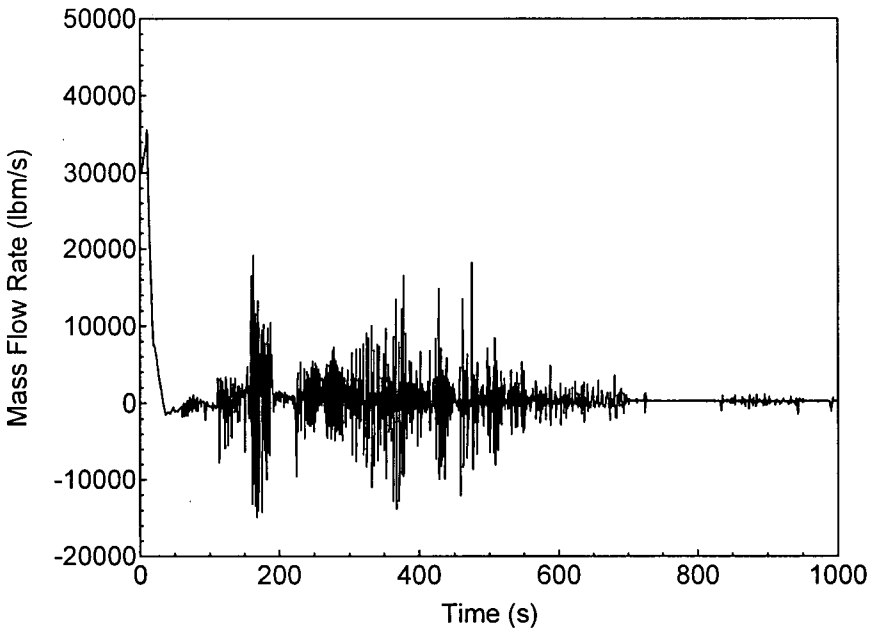

Figure RAI-7-14.15 Downcomer to Lower Plenum Mass Flowrate for 1-ft^2 Break

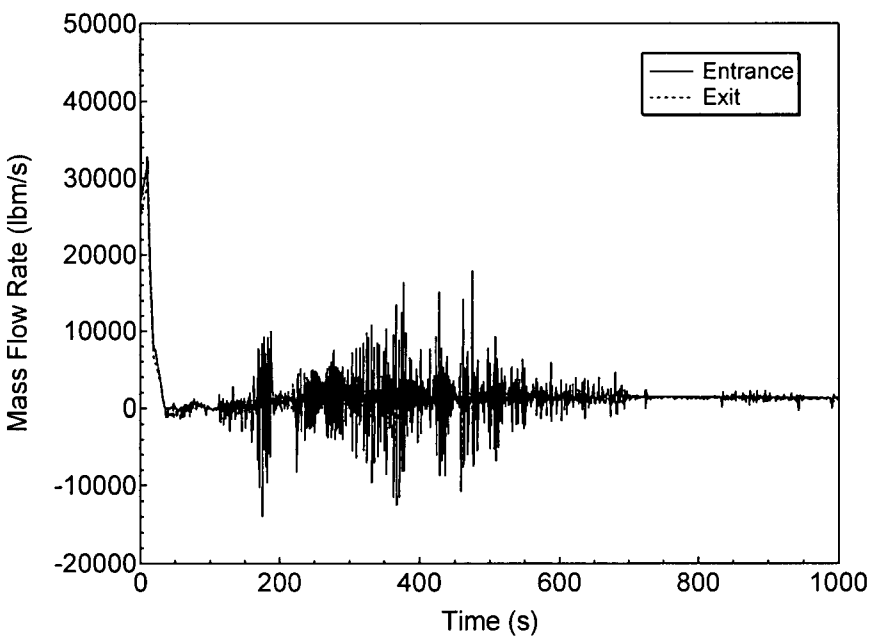

Figure RAI-7-14.16 Average Core Entrance and Exit Liquid Mass Flowrates for 1-ft^2 Break

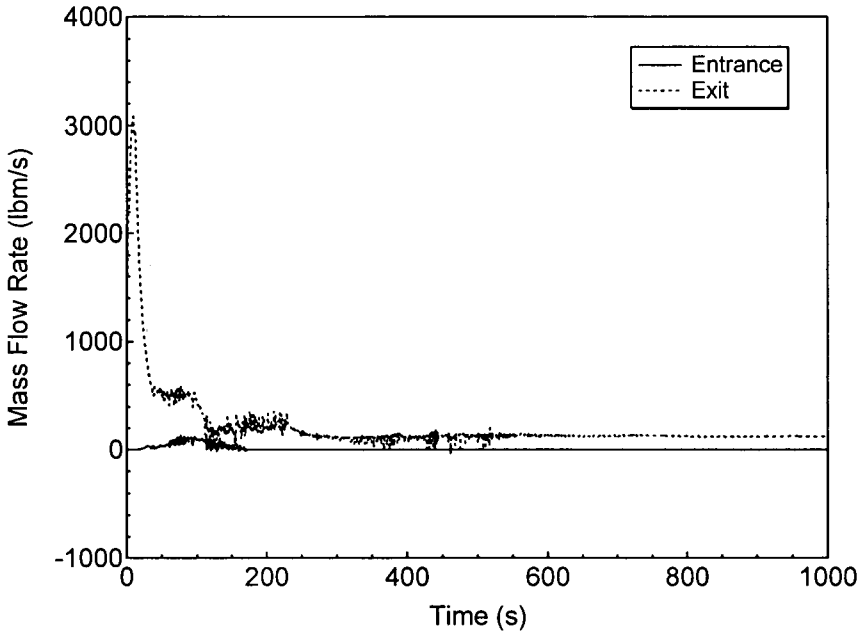

Figure RAI-7-14.17 Average Core Entrance and Exit Vapor Mass Flowrates for 1-ft^2 Break

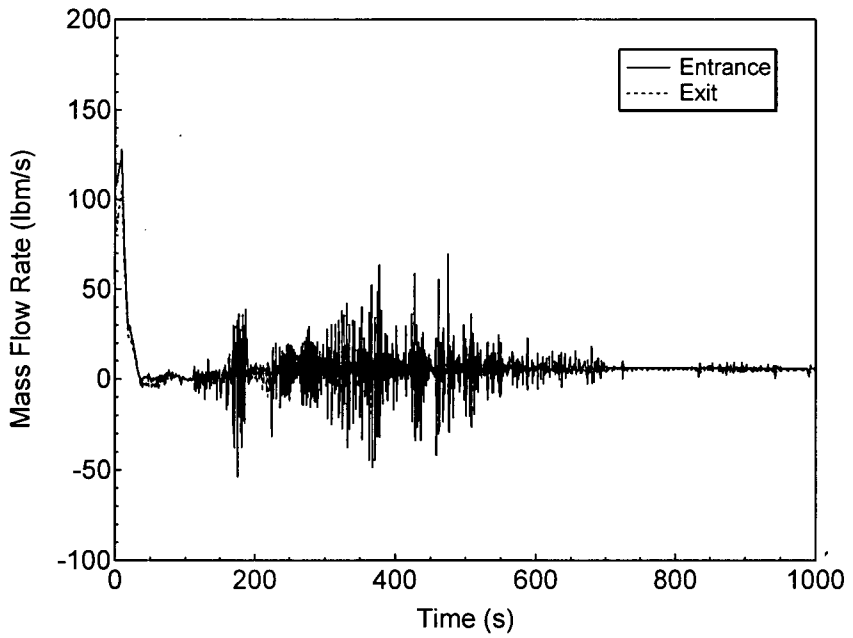

Figure RAI-7-14.18 Hot Assembly Entrance and Exit Liquid Mass Flowrates for 1-ft^2 Break

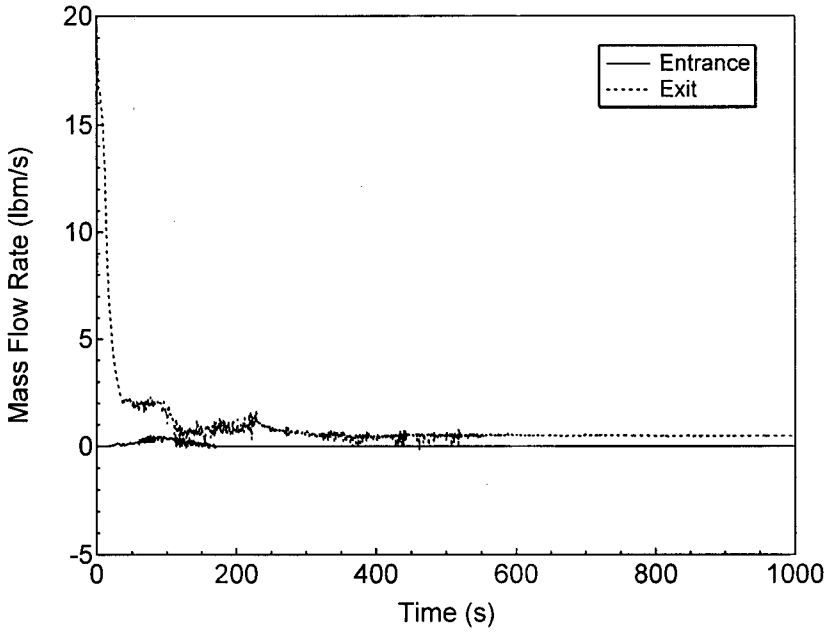

Figure RAI-7-14.19 Hot Assembly Entrance and Exit Vapor Mass Flowrates for 1-ft^2 Break

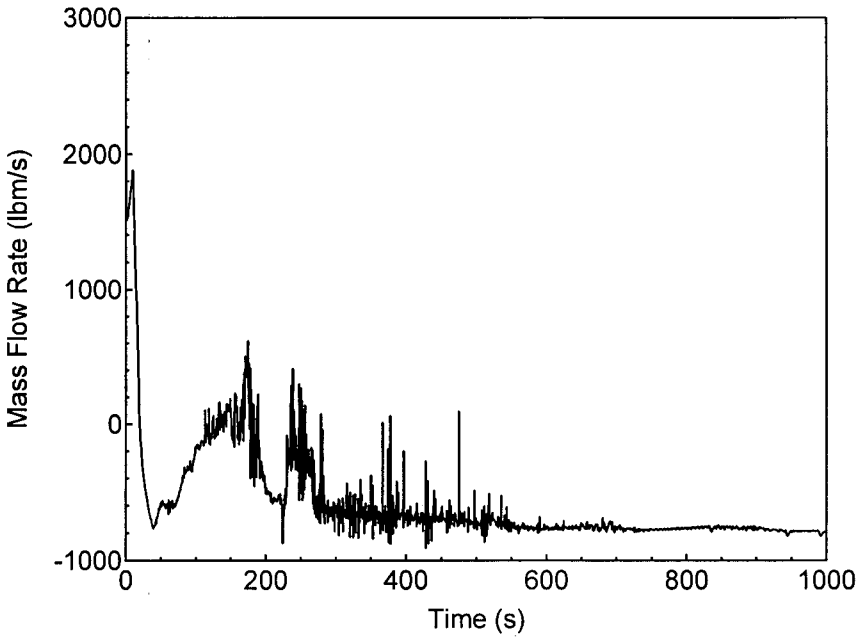

Figure RAI-7-14.20 Core Bypass Entrance Mass Flowrate for 1-ft^2 Break

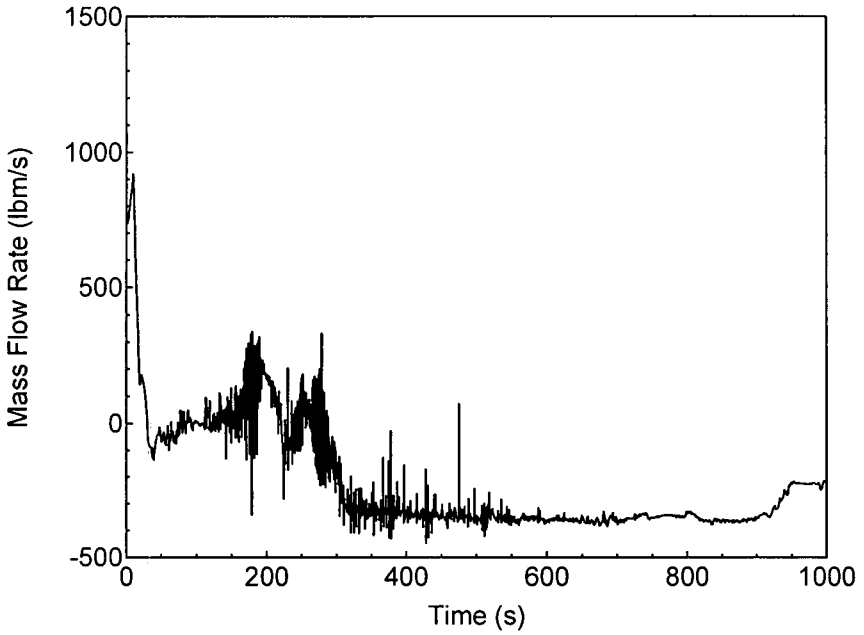

Figure RAI-7-14.21 Neutron Reflector Entrance Mass Flowrate for 1-ft^2 Break

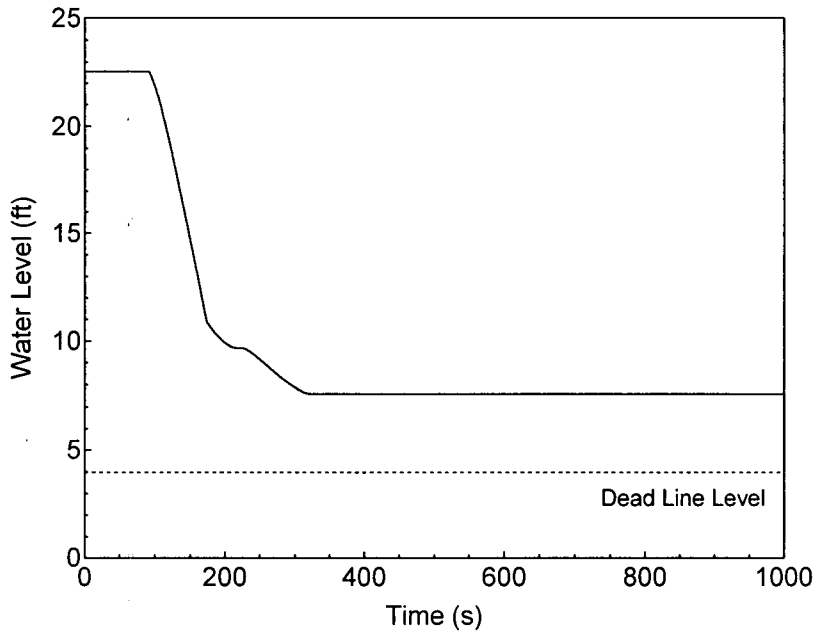

Figure RAI-7-14.22 Accumulator Water Level for 1-ft^2 Break

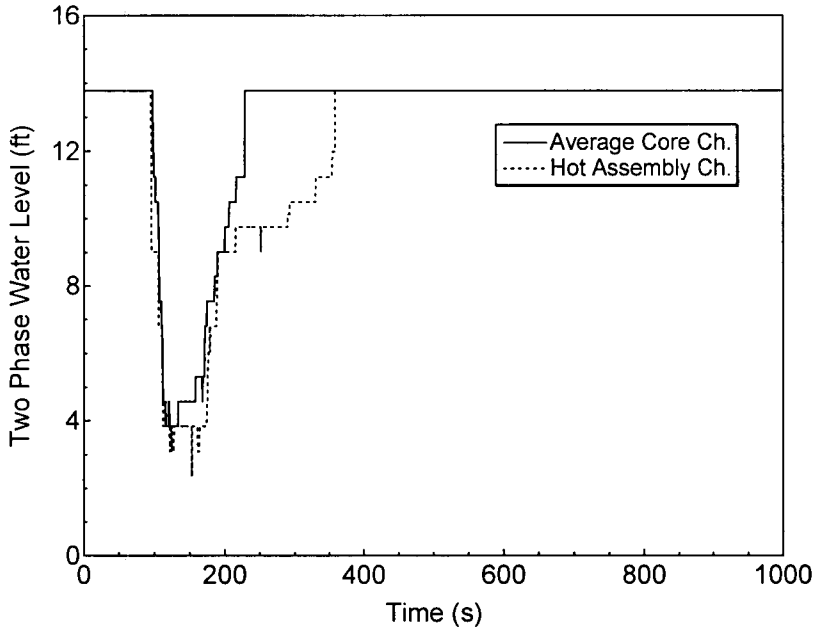

Figure RAI-7-14.23 Core Two-Phase Level for 1-ft^2 Break

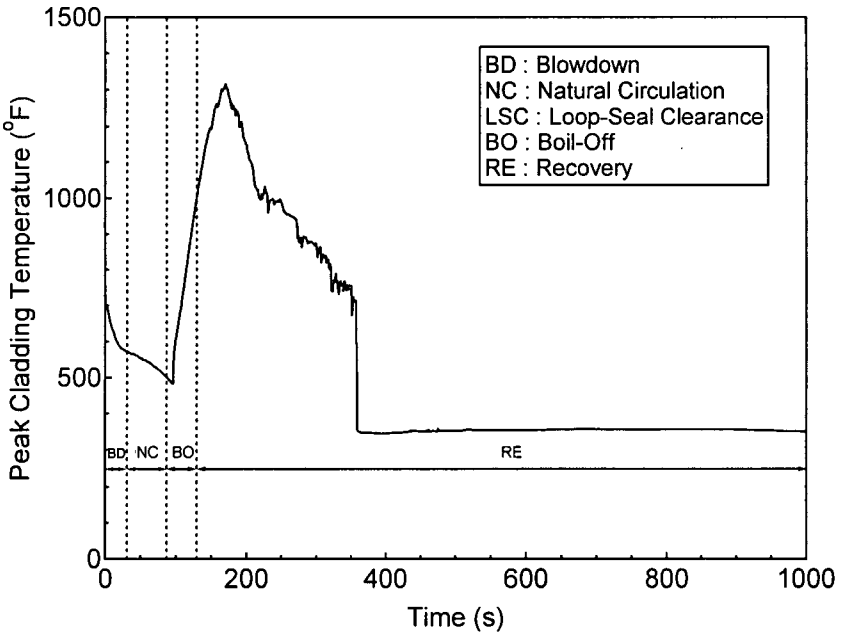

Figure RAI-7-14.24 PCT at ALL Elevations for Hot Rod in Hot Assembly for 1-ft^2 Break

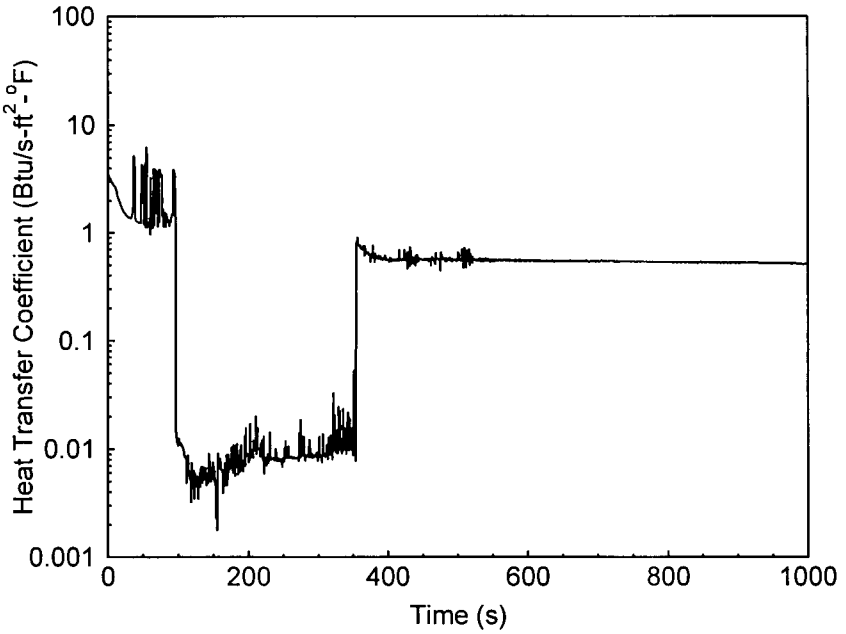

Figure RAI-7-14.25 Heat Transfer Coefficient at PCT Location for 1-ft^2 Break

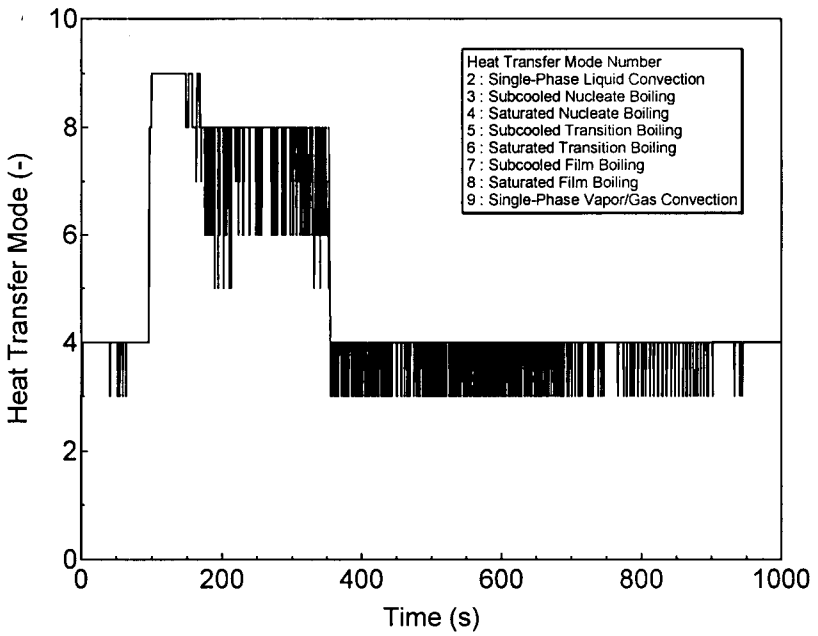

Figure RAI-7-14.26 Heat Transfer Mode at PCT Location for 1-ft^2 Break

REQUEST 7-15
(Related RAI 7-2)

Based on Section 4.4, "Reflood Model," in Volume IV of the RELAP-3D code description, "Paul Scherrer Institute (PSI) in Switzerland developed updates to improve the quench front behavior during the reactor core reflood process. A modified version of these updates was incorporated into RELAP5-3D along with a new quench front plotting capability." A literature search suggests a multiplier of 0.6 should be applied to the heat transfer coefficient for flooding rate greater than 1 inch/sec to meet Appendix K requirements, based on an assessment against FLECHT-SEASET tests.

Is the 0.6 multiplier used for SBLOCA reflood rates greater than 1 inch/second? If not, why? Are any other modifications used in M-RELAP5, such as a multiplier, to meet the Appendix K requirements?

RESPONSE

Liang et al. intend to apply the RELAP5-3D to the LOCA analysis in accordance with the Appendix K of 10 CFR 50. They performed the assessment calculations against the five FLECHT-SEASET tests to use the PSI reflood model for the reflood heat transfer calculation for a flooding rate greater than 1 inch/sec. They concluded based on this assessment that the heat transfer coefficient calculated by the original PSI model should be reduced by a factor of 0.6 to ensure reasonable conservatism (Ref. 1).

The applicability of the PSI reflood model to the high pressure condition of the SBLOCA transients has not been validated. On the other hand, it was confirmed that the default heat transfer model in the M-RELAP5 can be conservatively applied to SBLOCA conditions by the comparison with the ORNL/THTF high-pressure reflood tests. Thus, the default heat transfer model is used in the US-APWR SBLOCA analyses with M-RELAP5. As the PSI reflood model is not used in the US-APWR SBLOCA analyses, the multiplier of 0.6 is not applicable. Multipliers to the heat transfer coefficient are not introduced to meet the Appendix K requirements.

The Dougall-Rohsenow correlation is applied to the two-phase vapor convection heat transfer calculation in M-RELAP5. As described in the topical report of SBLOCA methodology for the US-APWR, the vapor physical properties of thermal conductivity, viscosity and specific heat capacity are evaluated at the film temperature and the vapor temperature is used as the sink temperature. The original correlation used the saturation temperature to evaluate the vapor properties and as the sink temperature. MHI believes that M-RELAP5 meets the Appendix K requirement about the Dougall-Rohsenow correlation, because of the lower heat transfer coefficient and the higher sink temperature.

References:
1. Thomas K. S. Liang, et al., Development of LOCA licensing calculation capability with

RELAP5-3D in accordance with Appendix K of 10 CFR50.46, Nucl. Eng. Des. Vol. 211, 2002

REQUEST 7-17
(Related RAI 7-2)

During refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam. The M-RELAP reflood model uses the Forsland-Rohsenow correlation which includes heat transfer to droplets and does not comply with the Appendix K requirement for steam cooling only.

Please explain how using the heat transfer coefficient model, which includes the Forsland-Rohsenow correlation, satisfies the Appendix K steam cooling only requirement.

In M-RELAP5, radiation to droplets is then added to the final film boiling coefficient. Please explain how this term satisfies the Appendix K steam only criterion.

RESPONSE

The FLECHT tests which simulate the flow blockage were performed before Appendix K to 10 CFR 50 was issued. With reflood rate of one inch per second or higher, improvement of heat transfer was found around the blockage. The improved heat transfer was shown to be caused by break-up of the entrained droplets and increased turbulence by the blockage. On the contrary, heat transfer was degraded for the blockage test at a reflood rate of 0.6 inch per second. The degradation was presumed to be the result of a lack of the entrained droplet effect and the steam flow reduction around the blockage. Based on the test results, heat transfer coefficient based on unblocked FLECHT tests is approved when the reflood rate is one inch per second or higher even if the flow blockage occurs. And the assumption that cooling is only by steam, and the flow blockage might affect both local steam flow and heat transfer is required when the reflood rate is less than one inch per second. The actual purpose of Appendix K requirement for the steam cooling is considered to neglect the entrained droplet effect by the blockage and to account the heat transfer degradation due to the flow reduction around the blockage for low flood rate condition.

The Forsland-Rohsenow correlation is used in PSI reflood model. As the PSI reflood model is not used in US-APWR SBLOCA analyses, the Forsland-Rohsenow correlation is not applied.

The general heat transfer model incorporated in M-RELAP5 considers not only convective cooling to steam but also film boiling cooling to liquid through vapor film and radiation to water droplets is applied to the reflood heat transfer calculations in the US-APWR SBLOCA analyses. The conservatism of this heat transfer model is confirmed by the comparison with ORNL/THTF high-pressure reflood test data. MHI intends to evaluate the FLECHT-SEASET data to confirm the conservatism of the heat transfer model under the

lower pressure condition which is typical to larger break size in the response to RAI 7-16. It is expected that the heat transfer model of M-RELAP5 is conservative and is adequately applied to the reflood heat transfer calculation for no blockage cases.

When fuel cladding deformation due to cladding swelling or rupture occurs, the resulting flow blockage affects hydrodynamic behavior. Flow area change and additional pressure loss due to the flow blockage are considered in M-RELAP5 regardless of the reflood rate. The flow blockage will reduce the coolant flow around the blockage and will degrade the heat transfer. In addition, the entrained droplet effect caused by the blockage is conservatively neglected in M-RELAP5. Thus, the flow blockage effect on the heat transfer is conservatively considered in M-RELAP5 regardless the reflood rate. The flow blockage does not occur in the US-APWR SBLOCA analyses.

A reflood rate for 1.0ft^2 of cold leg break is shown in **Figure RAI-7-17.1**. As the initial fuel rod stored energy is already released before the reflood starts, and decay heat in the reflood period is small in the SBLOCA, steam generation in the core is small compared with the LBLOCA. So, the steam binding effect is negligible in SBLOCAs. This causes a higher reflood rate in SBLOCA than in LBLOCA. The estimated reflood rate is greater than 1.0 inch per second as shown in Figure RAI-7-17.1. Therefore, the Appendix K requirement for steam cooling does not apply to the SBLOCA analyses of the larger break sizes.

It can be concluded that the reflood heat transfer model in M-RELAP5 satisfies the Appendix K requirement.

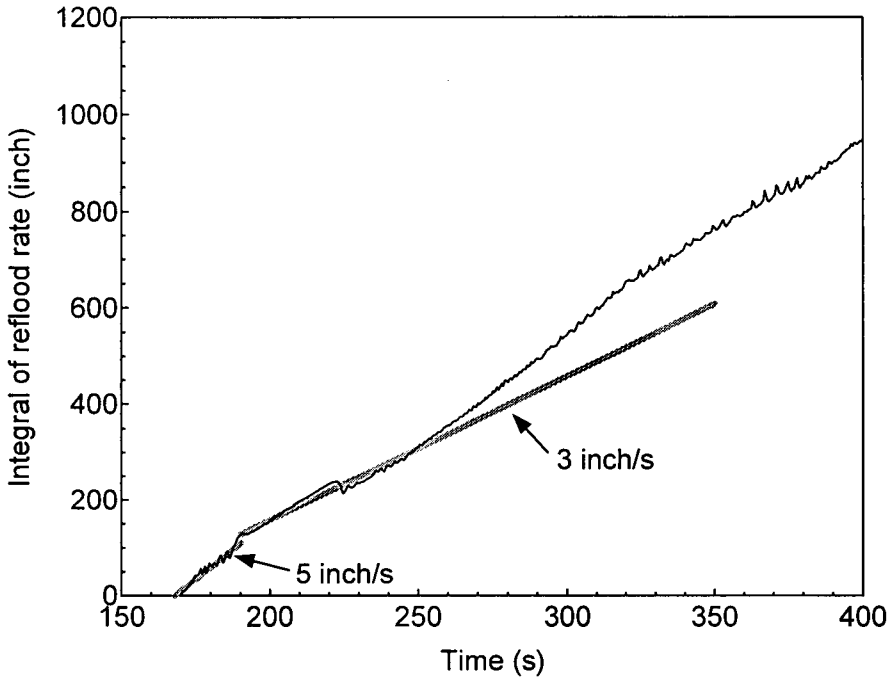

Figure RAI-7-17.1 Estimated Reflood Rate for 1.0ft^2 of Cold-leg Break

REQUEST 7-18
(Related RAI 7-2, 7-10)
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RESPONSE
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REQUEST 7-19
(Related RAI 7-8)
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RESPONSE
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REQUEST 7-21
(Related RAI 7-8)
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RESPONSE
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REQUEST 8-3

The 1/2 scale test will be simulated with RELAP5/MOD3.3 as part of the confirmatory calculations. The purpose is to benchmark the advanced accumulator modeling approach used in RELAP5/MOD3.3 (uses control systems) against experimental data.

Please provide the M-RELAP5 input file that was used for the 1/2 scale advanced accumulator tests.

RESPONSE

The M-RELAP5 input files are supplied in ENCLOSURE 4 of this document, and the M-RELAP5 input data for the 1/2-scaled advanced accumulator test analysis are recorded on the attached CD-R. The detail of these inputs is described in Reference 1.

References:

1. Mitsubishi Heavy Industry, Ltd., Transmittal of the Calculation Book and Input Deck of Full Height 1/2 Scale Test Analysis on Advanced Accumulator for WCOBRA/TRAC and M-RELAP5., UAP-HF-08030, January 17, 2008.

REQUEST 8.1.2-13
(Related RAI 8.1.2-5)

The mass fluxes for Test K and N in Table RAI-8.1.2-5.1 are not the same as those in Table 8.1.2-1. This should be clarified. Also, note there is substantial energy imbalance, especially for tests J, BB and CC (Some of them are more than 15%. They will be even larger when divided by (d)-(c) rather than (d).)

Discuss the impact of these imbalance on the conclusion that the code prediction is reasonable and validates the code.

RESPONSE

First, the boundary condition in terms of the mass flux used for M-RELAP5 THTF test analysis is described as follows. In the experiment, the volumetric flow rate was measured both at the channel inlet and outlet sections, and one low-flow orifice meter and one turbine meter were equipped at each of the inlet and outlet sections. The fluid density was also measured by using the densitometer equipped in the outlet spool piece to determine the mass flux from the volumetric flow rate. However, the experimental test report states that the vapor density computed from the measured pressure and steam temperature was used in determining the mass flux at the channel outlet. As discussed later, there observed significant uncertainties in the measured steam temperature; therefore, the mass flux at the inlet channel section is used throughout the M-RELAP5 calculation with the exception of the ultimately high flow tests (Tests I and L) where the inlet mass fluxes were not correctly measured. It is noted that the Tests I and L are excluded from our code assessment database because of the higher power density in comparison with that expected in the actual plant accident.

Next is our consideration in confirming the heat balance from the test data. In the experiment, the vapor temperature distribution was determined from the energy conservation based upon the steam temperature measured by the subchannel thermocouple rake at the end of heated length. Although the vapor temperature distribution was directly measured by using the thermocouple array rods, the direct measurement indicated several unphysical behaviors, especially, for the ultimately low mass flux tests (Tests K and N). The details are described in Section 5.3.2.1 and Appendix A.5 of Reference 1.

In determining the experimental vapor temperature distribution for the upper portion of the test channel (steam cooling region), the measured outlet mass flux was used along with the energy conservation. Because a large deviation is found between the inlet and outlet mass fluxes only for the lower flow tests (Tests K and N), we decided to apply the mass fluxes measured at the outlet in derivation of the heat balance only for these two tests, while the measured inlet mass flux is used for the other tests as M-RELAP5 THTF test analysis. Test FF was also performed under the low flow rate condition. However, the measured two-phase mixture level was higher than Tests K and N and approached beneath the end of the heated length in Test FF. Under this condition, assumption of the steam only flow at the channel outlet was uncertain and the temperature distribution shall not be evaluated based on the energy conservation with the measured outlet mass flux. Therefore, the measured inlet mass flux was used in the heat balance calculation for Test FF.

The above is the reason why the mass fluxes for Tests K and N in Table RAI-8.1.2-5.1 (Ref.2) are not the same as those in Table 8.1.2-1 of MUAP-07013-P (R0). The mass fluxes used for each of the M-RELAP5 test analysis and heat balance calculation prepared for the NRC's request are compared with the measured inlet and outlet mass fluxes in **Table RAI-8.1.2-13.1**.

Hereafter, the imbalance observed in derivation of the heat balance calculation is discussed. As reviewer pointed out there remains a significant imbalance in the derived heat balance table as shown in **Table RAI-8.1.2-13.2**. To clarify possible sources of this imbalance, first, errors contained in the measured parameters used for the heat balance calculation are summarized in **Table RAI-8.1.2-13.3**, which are referred to appendices of the experimental test report (Ref.1). The table shows that significant measurement uncertainties (above 30%) were included in the measured fluid temperatures, while errors of the measured bundle power are below about 6 % and those of the mass flux are below 10% except for Test FF. Therefore, the uncertainties included in the measured parameters can be one of the possible sources.

MHI would like to give a comment regarding the significantly large uncertainty in the measured temperature. First, Section 5.3.2 of the experimental test report (Ref.1) shows that uncertainties of the bundle cross-section average vapor temperature were about ±50 °F (±28 K) near the end of the heated length for Test K, while Appendix A of the same report indicates uncertainties of the outlet temperature reached ±260 K. The latter uncertainties are shown in Table RAI-8.1.2-13.3, although the experimental test report mentions nothing about the difference between the two series of the uncertainties. One possibility is that the temperature shown in Section 5.3.2 may be the data obtained by the thermocouple array rods and the uncertainty is primarily corresponding to the equipment uncertainty and to the deviation caused by averaging several temperatures measured at the same axial elevation. On the other hand, the temperature shown in Appendix A may be obtained by the energy balance along with the vapor temperature measured by the subchannel thermocouple rake. For the latter, Section A.6 of the test report states "*Use of an energy balance to calculate vapor temperature requires computation of bundle heat losses*", and the heat loss was determined from axial integral of the measured temperature gradient across the test channel shroud box. This indicates a possibility that the temperature uncertainty shown in Appendix A involves the uncertainty due to the heat loss measurement and calculation in addition to the uncertainty of the thermocouple rake. In other words, the imbalance in the heat balance calculations may be due to the uncertainty of the heat loss in addition to the measurement uncertainties.

Second, in the mixture level swell tests, Tests AA to FF, where the mixture level was high approached beneath the end of the heated length, there possibly remains the liquid film and entrainment even at the channel outlet. In calculating the heat balance (Ref.2), on the other hand, we approximately assumed the steam only flow even for the mixture level swell tests. This may induce the additional uncertainty for Tests AA to FF.

The above are the possible sources of the imbalance appearing in the heat balance calculations, and MHI supposes that the primary sources come from the measurement uncertainties. The uncertainties are taken into account also in the measured uncovery heat transfer coefficient for Tests I to N, and in the mixture level swell for all the subject tests, independently from the heat balance calculations MHI performed.

Figures RAI-8.1.2-13.1 to RAI-8.1.2-13.6 compare the M-RELAP5 calculation results with

the measurements in terms of the uncovery heat transfer and the collapsed and mixture level with the errors. Taking account the measurement uncertainties, M-RELAP5 is capable of predicting these key parameters with best-estimate or conservative accuracy. (Note that the axial variation of the predicted and measured uncovery heat transfer coefficient is addressed in MHI response to REQUEST 8.1.2-10 in Reference 2.) In addition, 2% increase in the core thermal power assumed for the US-APWR DCD Small Break LOCA analyses assures the sufficient conservatism required in the safety analyses.

References:

1. T. M. Anklam et al., Experimental Investigations of Uncovered-Bundle Heat Transfer and Two-Phase Mixture-Level Swell under High-Pressure Low-Heat Conditions, NUREG/CR-2456, ORNL-5848, 1982.
2. Mitsubishi Heavy Industry, Ltd., MHI's Partial Responses to NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR", UAP-HF-09002, January 16, 2009.

Table RAI-8.1.2-13.1 Mass Fluxes used for M-RELAP5 Test Analysis and Heat Balance Calculation

Test	Inlet Mass Flux ($kg/s/m^2$)	Outlet Mass Flux ($kg/s/m^2$)	Mass Flux[1)] ($kg/s/m^2$)	Mass Flux[2)] ($kg/s/m^2$)
3.09.10I	-	29.76	29.76	29.76
3.09.10J	12.93	12.66	12.93	12.93
3.09.10K	2.22	3.13	2.22	3.13
3.09.10L	-	29.11	29.11	29.11
3.09.10M	13.38	12.63	13.38	13.38
3.09.10N	4.33	4.60	4.33	4.60
3.09.10AA	21.15	20.26	21.15	21.15
3.09.10BB	9.44	9.62	9.44	9.44
3.09.10CC	7.22	5.03	7.22	7.22
3.09.10DD	19.82	19.48	19.82	19.82
3.09.10EE	11.00	9.62	11.00	11.00
3.09.10FF	4.83	3.85	4.83	4.83

1) Mass flux used for M-RELAP5 calculations.
Based on the inlet mass flux except for the high flow tests (Tests I and L).
2) Mass flux used for the heat balance calculations.
Based on the inlet mass flux except for the low and high flow tests (Tests, I, K, L, N)

Table RAI-8.1.2-13.2 Heat Balance Calculation Results

Test	(a)FRS Thermal Power (kW)	(b) Heat Loss (kW)	(b) Heat Loss [fraction]	(c) Inlet Flow Energy (kW)	(d) Outlet Flow Energy (kW)	{(a)-(b) +(c)-(d)}/(d)	{(a)-(b) +(c)-(d)}/{(d)-(c)}
3.09.10I	487.19	8.61	[0.018]	156.24	629.70	0.81%	1.08%
3.09.10J	234.82	12.13	[0.052]	70.49	266.53	10.00%	13.59%
3.09.10K	70.23	12.33	[0.176]	15.87	73.41	0.48%	0.61%
3.09.10L	476.22	8.13	[0.017]	143.76	582.54	5.03%	6.68%
3.09.10M	223.85	9.46	[0.042]	70.85	274.69	3.84%	5.17%
3.09.10N	103.14	16.73	[0.162]	24.21	108.10	2.33%	3.00%
3.09.10AA	278.71	5.58	[0.020]	98.27	373.50	-0.56%	-0.76%
3.09.10BB	140.45	4.83	[0.034]	45.73	166.98	8.61%	11.85%
3.09.10CC	72.42	2.53	[0.035]	36.81	126.91	-15.92%	-22.43%
3.09.10DD	283.10	8.46	[0.030]	93.67	350.88	4.97%	6.77%
3.09.10EE	140.45	5.52	[0.039]	52.71	192.47	-2.51%	-3.46%
3.09.10FF	70.23	6.46	[0.092]	22.56	83.45	3.46%	4.74%

Table RAI-8.1.2-13.3 Measurement Errors relevant to Heat Balance Calculation

Test	Bundle Power		Inlet Temperature		Outlet Temperature		Mass Flux		Heat Loss	Mixture Level
	(kW)	Error	(K)	Error	(K)	Error	(kg/m^2/s)	Error	(-)	(m)
3.09.10I	487.19	5.3%	473.0	54.8%	774.1	33.5%	29.76	6.7%	0.02	2.62
3.09.10J	234.82	5.3%	480.3	53.9%	728.4	35.6%	12.93	4.8%	0.05	2.47
3.09.10K	70.23	6.3%	466.5	55.5%	935.0	31.3%	3.13	8.3%	0.18	2.13
3.09.10L	476.22	5.3%	461.3	56.2%	715.6	36.2%	29.11	5.6%	0.02	2.75
3.09.10M	223.85	5.4%	474.4	54.6%	746.5	34.7%	13.38	4.6%	0.04	2.62
3.09.10N	103.14	5.4%	473.1	54.8%	947.9	28.8%	4.60	5.9%	0.16	2.13
3.09.10AA	278.71	5.3%	450.9	57.5%	547.0	47.4%	21.15	3.0%	0.02	3.42
3.09.10BB	140.45	5.4%	458.2	56.5%	540.8	47.9%	9.44	6.7%	0.03	3.31
3.09.10CC	72.42	5.4%	467.6	55.4%	531.5	48.8%	7.22	8.7%	0.04	3.60
3.09.10DD	283.10	5.3%	453.4	57.2%	595.4	43.5%	19.82	3.2%	0.03	3.23
3.09.10EE	140.45	5.4%	455.9	56.8%	581.0	44.6%	11.00	5.8%	0.04	3.47
3.09.10FF	70.23	5.5%	451.4	57.4%	565.8	45.8%	4.83	13.2%	0.09	3.23

All the errors refer to the appendices of the experimental test report (NUREG/CR-2456).
Measurement uncertainty of heat loss from the channel is not available.

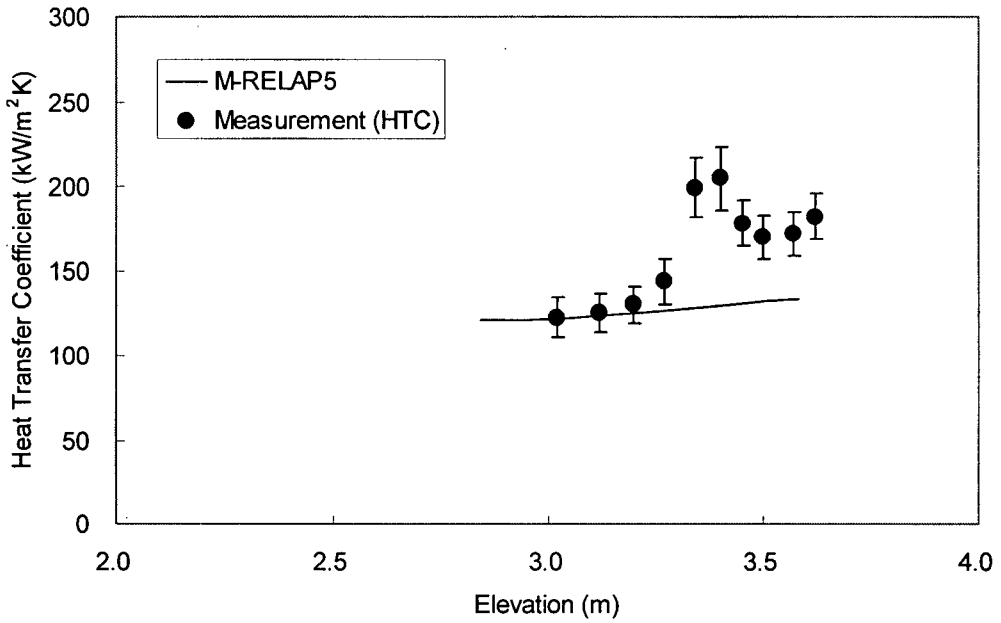

Figure RAI-8.1.2-13.1 Comparison Measured and Predicted Uncovery Heat Transfer Coefficients (Test J)

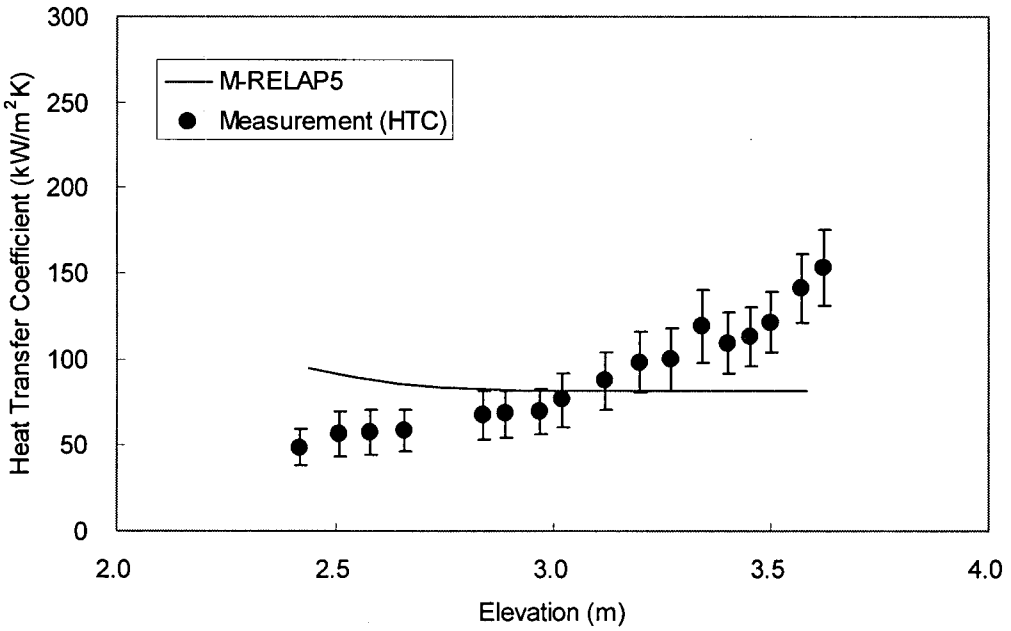

Figure RAI-8.1.2-13.2 Comparison Measured and Predicted Uncovery Heat Transfer Coefficients (Test K)

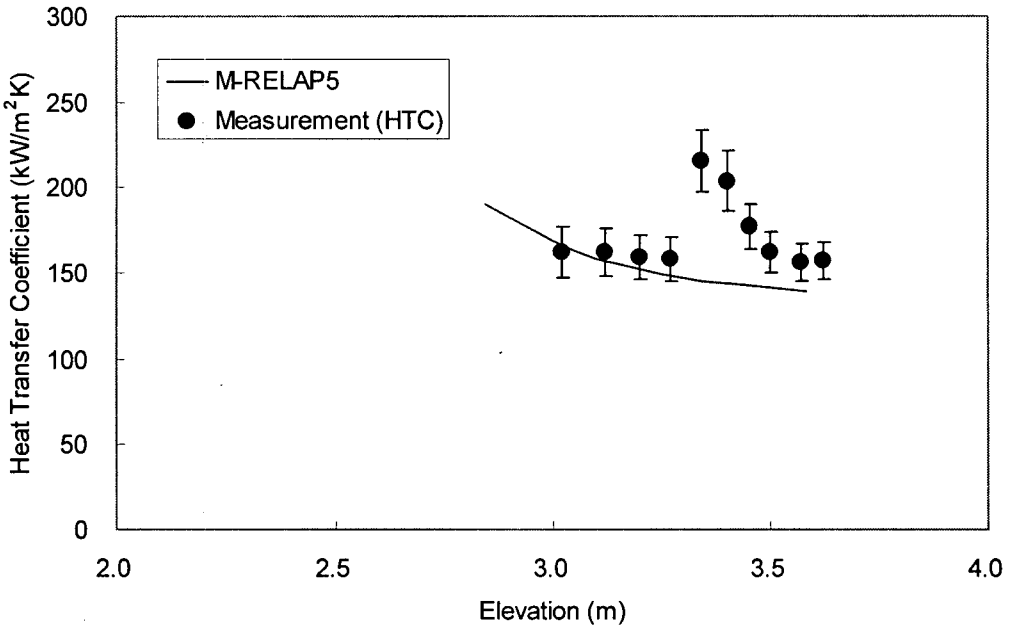

Figure RAI-8.1.2-13.3 Comparison Measured and Predicted Uncovery Heat Transfer Coefficients (Test M)

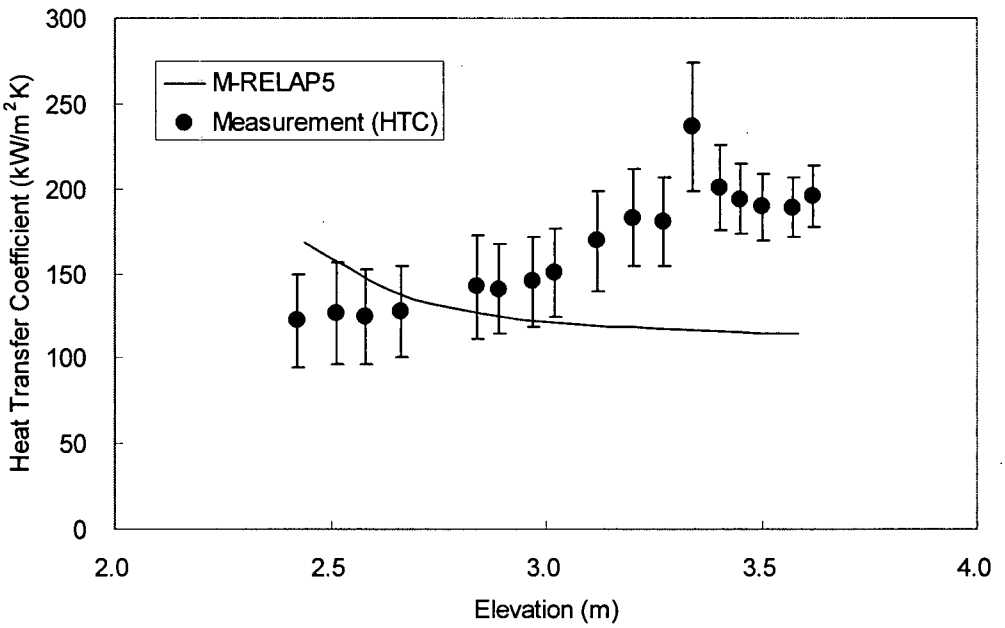

Figure RAI-8.1.2-13.4 Comparison Measured and Predicted Uncovery Heat Transfer Coefficients (Test N)

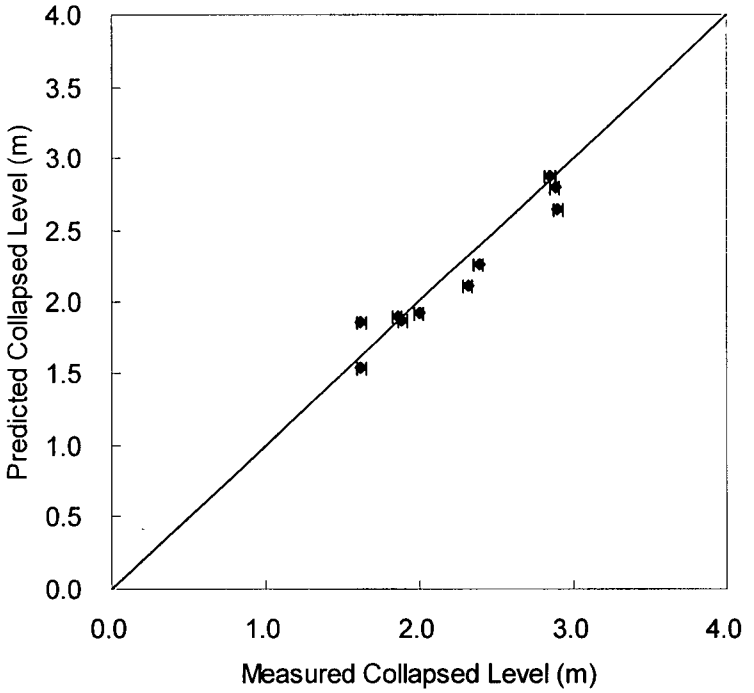

Figure RAI-8.1.2-13.5 Comparison Measured and Predicted Collapsed Levels

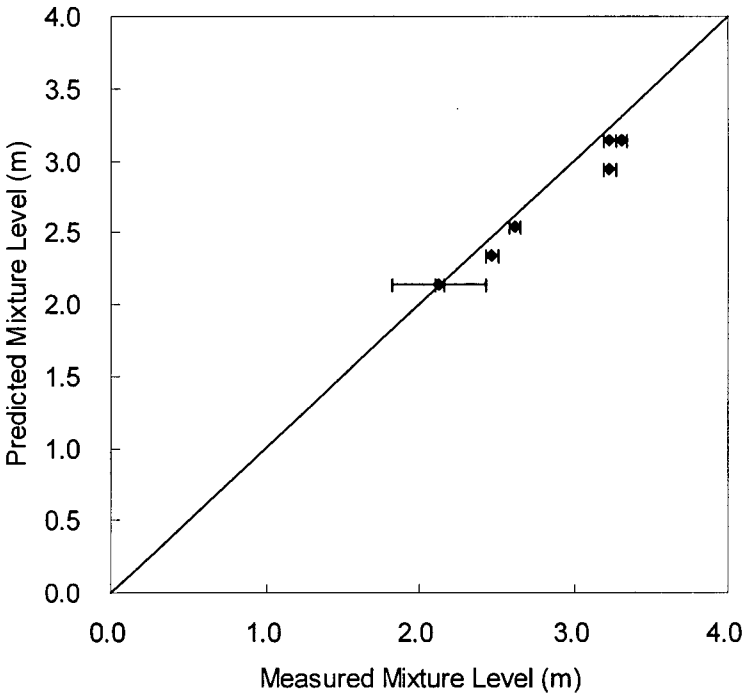

Figure RAI-8.1.2-13.6 Comparison Measured and Predicted Mixture Levels

REQUEST 8.1.4-7
(Related RAI 8.1.4-3)

The CCFL correlation has three coefficients that are provided through input. Provide locations of all the places where CCFL correlation is applied, values of CCFL coefficients, and justification.

What is the size of pipe when surface tension effect is not important? Does the set of coefficients change with a change in pressure (see Table RAI-8.1.4-3.1, January-2009 MHI response, page 93)?

RESPONSE

Table RAI-8.1.4-7.1 summarizes the location and the coefficient set of CCFL applied to US-APWR.

For the location A in the table, the hydraulic diameter is [ ] and is close to 0.75 m in the UPTF. Since those hydraulic diameter are classified as a large-scale pipe and the applicability of K*-scaling to such a large-scale pipe is considered to be high as described in the response to RAI 8.1.4-3 (Ref. 1). For the location B, the applicability was described in the response to RAI 8.1.5-1 (Ref.1). For the location C, the hydraulic diameter is also [ ] and the applicability of K*-scaling is high. K* number for 'water zero' penetration at the location C is about [ ] from the coefficient c and the value corresponds to that under a large-scale pipe in Figure RAI-8.1.4-3.1.

The effect of surface tension is relatively small where the nondimensional CCFL data are rather be correlated better with hydraulic diameter as the length-scale. The region is considered to be under D* in Figure RAI-8.1.4-3.1 (Ref.1) less than [ ].

The CCFL coefficient set in Table RAI 8.1.4-7.1 is used for the whole range of pressure without modification due to pressure.

References:
1. Mitsubishi Heavy Industry, Ltd., MHI's Partial Responses to NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR", UAP-HF-09002, January 16, 2009.
2. C. L. Tien and C. P. Liu, Survey on Vertical Two-Phase Countercurrent Flooding, EPRI NP-984 Research Project 1160-1, 1979.

Table RAI-8.1.4-7.1 Locations of CCFL correlations applied for US-APWR

REQUEST 8.1.4-9
(Related RAI 8.1.4-6)

Why does the data for 3 bar in Figure 8.1.4-2 of MUAP-07013-P (R0) show less liquid down flow when compared to the 15 bar data at the same Hg? Is this an indication of pressure dependence of the coefficients (c and m) for the CCFL correlation?

Figure 8.1.4-4 of MUAP-07013-P (R0) compares analytical results and test data at two pressures, 3 bar and 15 bar. The comparison at 3 bar shows that for a given steam mass flow, the analytical result predicts higher water mass flow than the test data. Is the set of coefficients for CCFL correlation obtained from the UPTF tests conservative for SBLOCA simulation for the US-APWR? Has the validation of the CCFL correlation been done with an independent set of data, i.e. other than the UPTF data that were used to develop the coefficients for the CCFL correlation?

Also, there are CCFL models applied at the main coolant pump suction and discharge. Please describe the basis for selecting these models including the assessments that demonstrate applicability to the US-APWR application.

RESPONSE

For vertical pipes, the CCFL correlation by Hewitt & Wallis for SG U-tubes is able to apply to the high pressure ROSA/LSTF data as described in the response to RAI 8.1.5-1 (Ref. 1). However, the effect of pressure for the hot-leg CCFL is not clear because of scarce database for the special geometry. The UPTF data is reported in detail with error band in Reference 2 and the database is shown in **Figure RAI-8.1.4-9.1**. The data scattering becomes large under a low Hg of the 3 bar case and the dependency on pressure is not clear under the wide error band.

Considering the error band in Reference 2, the upper bound of water mass flow rate under 3 bar is located at around the UPTF CCFL correlation and it does not necessarily mean that the correlation gives non-conservative results. And the range of pressure is at around 80 bar that the hot-leg CCFL is important. Since the data scattering under the low pressure 3 bar case is considered to be caused by the two-phase flow instability due to density difference between gas and liquid, the hot-leg CCFL characteristics under 15 bar are supposed to be likely case under 80 bar.
The description for the CCFL model at pump suction side is included in the response to RAI 8.1.4-7 where the model applies to the location C. Please refer to the response.

References:
1. Mitsubishi Heavy Industry, Ltd., MHI's Partial Responses to NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR", UAP-HF-09002, January 16, 2009.
2. P. A. Weiss and R. J. Hertlein, UPTF Test Results: First Three Separate Effect Tests, Nuclear Engineering and Design, 108, pp.249-263, 1988.

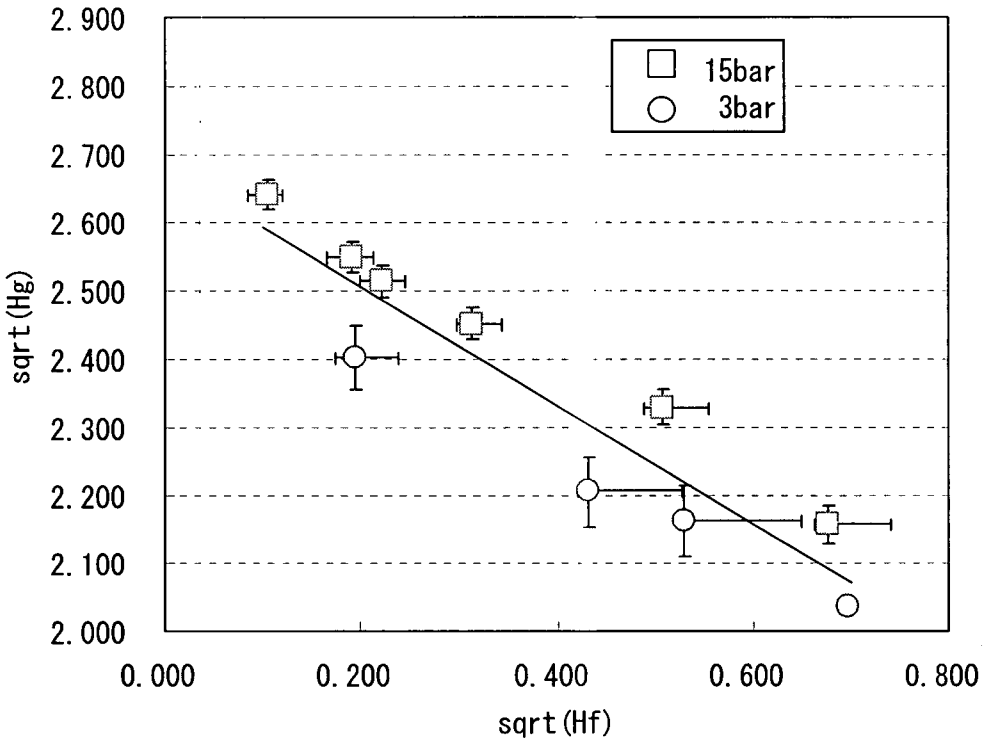

Figure RAI-8.1.4-9.1 UPTF CCFL Data with Error Band

REQUEST 8-3
(Related RAI 8-2)

Have there been error fixes in M-RELAP5 since MUAP-07013-P (R0) was issued? What impact do these error fixes have on the SBLOCA methodology as presented in MUAP-07013-P (R0) and the technical report MUAP-07025-P (R0)?

RESPONSE

Followings are differences in M-RELAP5 between the versions used in MUAP-07013-P (R0) and MUAP-07025-P (R0).
[

]
These error fixes have no impact to MUAP-07013-P (R0) experimental validation calculation.

Differences between the current M-RELAP5 and the version used in MUAP-07015-P (R0) are as follows.
[

]
The impacts of these error fixes to PCT are -1°F and +20°F for 7.5in break and 1ft^2 break cases, respectively, as reported in Reference 1.

References:
1. Mitsubishi Heavy Industry, Ltd., Reporting of Modifications and Corrections in ECCS Evaluation Models for the US-APWR LOCA Analyses, UAP-HF-08120, July 4, 2008.

REQUEST D-2
(Related RAI D-1)

Appendix D describes the model and uncertainty in the correlations. The model description is for regular accumulator and there is description of spherical accumulator that is not representative of the advanced accumulator in the US-APWR. The model does not explicitly provide expressions for calculating level in tank ("H"). Also, how does the code calculate L_{fL}? The definitions of A_g and A_f (top of page D-5) are not clear.

RESPONSE

The above request is decomposed into the following 4 (four) questions.

(1) The model description is for regular accumulator and there is description of spherical accumulator that is not representative of the advanced accumulator in the US-APWR.

(2) The model does not explicitly provide expressions for calculating level in tank ("H").

(3) Also, how does the code calculate L_{fL}?

(4) The definitions of A_g and A_f (top of page D-5) are not clear.

The corresponding answers are discussed below.

(1) The US-APWR advanced accumulator is cylindrical in the vertical direction. For the cylindrical type accumulator, the equations D-16 to D-19 and D-34 to D-35 are not used.

(2) The head of accumulator tank is calculated by the equation D-23. Therefore, the equation D-1, which calculates the cavitation factor σ_v, can be obtained as follows:

$$\sigma_v = \frac{P_D + P_{at} - P_v}{\left(P_A + \rho g H\right) - \left(P_D + \frac{\rho V_D^2}{2} + \rho g H'\right)} = \frac{P_{exit} - P_v}{P - P_{exit} + \Delta P_Z - \frac{\rho v^2}{2}}$$

P : vapor/gas dome pressure (abs)
P_v : Vapor pressure (abs)
P_{exit} : pressure at exit of surge line (abs)
ΔP_Z : elevation pressure differential between discharge line entrance and liquid surface
v : velocity at the accumulator tank outlet
ρ : Density of water

Therefore, it is not necessary to explicitly express the liquid level in tank is not necessary for calculating the cavitation factor.
[

]
(4) A_g and A_f are same as the tank flow area A_{TK}, because the advanced accumulator is

cylindrical in the vertical direction.

Appendix A

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Appendix B

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Appendix C

ATTACHMENT 1
FILES Contained in CD

CD: "M-RELAP5 Input Data for the MHI Advanced Accumulator Test Analysis"
- Proprietary information

Contents of CD

File Name	Size	Sensitivity Level
• M-RELAP5 Input for		
- case1.k0.0.i (txt format)	4KB	Proprietary
- case2.k0.0.i (txt format)	4KB	Proprietary
- case3.k0.0.i (txt format)	4KB	Proprietary
- case4.k0.0.i (txt format)	4KB	Proprietary