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February 12, 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-09040

**Subject: MHI's 4<sup>th</sup> Response to NRC's Requests for Additional Information on US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P**

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's 4<sup>th</sup> Response to NRC's Requests for Additional Information on US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P". The enclosed materials provide MHI's responses to the NRC's "Requests for Additional Information (RAIs) US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P," dated February 3, 2009. MHI has previously responded to RAIs on the Non-LOCA Topical Report in MHI letters UAP-HF-08141 dated August 22, 2008, UAP-HF-08170 dated September 12, 2008, and UAP-HF-08245 dated November 19, 2008.

As indicated in the enclosed materials, these documents contain information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is replaced by the designation "[ ]".

This letter includes a copy of the proprietary version of the RAI response (Enclosure 2), a copy of the non-proprietary version of the RAI response (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all material designated as "Proprietary" in Enclosure 2 be withheld from disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc., if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

Sincerely,



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

DOBI  
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's 4<sup>th</sup> Response to NRC's Requests for Additional Information on US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P (proprietary)
3. MHI's 4<sup>th</sup> Response to NRC's Requests for Additional Information on US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P (non-proprietary)

CC: J. A. Ciocco  
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## ENCLOSURE 1

Docket No. 52-021  
MHI Ref: UAP-HF-09040

### 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 "MHI's 4<sup>th</sup> Response to NRC's Requests for Additional Information on US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P" dated February 12, 2009, 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 12<sup>th</sup> day of February, 2009.

A handwritten signature in black ink, appearing to read "Y. Ogata". The signature is written in a cursive style with a horizontal line extending from the end of the name.

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Yoshiaki Ogata

**Enclosure 3**

**UAP-HF-09040  
Docket No. 52-021**

**February 2009**

**MHI's 4<sup>th</sup> Response to NRC's Requests for Additional Information on  
US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P**

**(Non-Proprietary)**

**RAI 2.1-2-1**

Please confirm the following understanding of the DNBR lookup table methodology: The MARVEL-M lookup table is a database of DNBR, core inlet temperatures, system pressures, ... and core heat flux data generated from VIPRE-01M steady state calculations. Using system analyses to determine the core inlet temperature and system pressure during a transient, an interpolative scheme is used to determine the normalized heat flux and thus the DNBR at each time step through the analysis. This simplified DNBR lookup table methodology is only used for non-LOCA events that have constant core flow rate and "are bounded by the applicable power distribution." In the event that the parameters of the calculation exceed the limitations of the lookup table, the analyst is flagged to use VIPRE-01M to directly calculate DNBR based on a DNB correlation in VIPRE-01M.

**Response**

The reviewer's understanding of the DNBR lookup table methodology is basically correct. There is no MARVEL-M output message that alerts the user that they have exceeded the limitations of the lookup table. The user must know the limitations of the methodology and apply it appropriately. The MARVEL-M input variable PMAXDNB allows the user to input an upper limit on the RCS pressure associated with the DNBR lookup tables. If the RCS pressure exceeds the user input upper limit, the pressure is assumed to be equal to the upper limit value when determining DNBR.



**Figure 2.1-2-1.1**      **DNBR versus Time**  
**One or More Dropped RCCAs within a Group or Bank**

**RAI 2.1-2-2**

How is the number of DNBRs evaluated in the generating the DNBR lookup tables (N) chosen? What degree of interpolation is used when using the table? .....

Verify that the DNBR lookup tables cover the full operating space (pressures, temperatures, flow rates) that the methodology is used for. Explain what is meant by "applicable power distribution" with regard to the use of the simplified DNBR tables.

**Response**

The MARVEL-M simplified DNBR lookup table array is defined by the MARVEL-M variables MDNB for a thimble cell and MDNBS for a typical cell. The array is defined for [ ] of DNBR (MARVEL-M variable WMIRC) corresponding to the DNBR limit, the DNBR at nominal conditions and [ ] additional DNBR points between these [ ]

Additional details of these input variables are described in Section 2.4 \$DNBRD of Part II of the MARVEL-M Manual (GEN0-LP-480). Revision 5 of the MARVEL-M Manual was submitted to the NRC by MHI letter UAP-HF-08245 on November 11, 2008.

The DNBR is interpolated using a third order Lagrange interpolation formula at the normalized heat flux calculated by MARVEL-M.

The power distribution assumed in the lookup table is the same as that used in the thermal hydraulic design, which is described in Section 4.4 of the US-APWR DCD, and covers normal operating conditions. For events where the power distribution exceeds the design power distribution, VIPRE-01M is used to calculate DNBR rather than the MARVEL-M DNBR lookup table, except for the RCCA drop as was described in the response to RAI 2.1-2-1.

The DNBR lookup tables cover the operating range of pressure and temperature that are protected by the over temperature  $\Delta T$ , over power  $\Delta T$ , low pressurizer pressure and high pressurizer pressure reactor trips. The DNBR lookup tables are not applied for decreasing core flow rate conditions.



**RAI 2.1-8-1**

The discussion provided in response to RAI 2.1-8 by MHI contained an error and was inadequate for the reviewer to evaluate the modeling of the four major thermal resistances in the calculation of the overall heat transfer coefficient for the steam generators. Provide clarification on the calculation of the initial values of the four thermal resistances,  $R_{pf}^0$ ,  $R_{tube}^0$ ,  $R_{bo}^0$ , and  $R_{foul}^0$ , and define all parameters and variables. Provide clarification on their calculation during the transient, and their combination to yield the overall heat transfer coefficient during the transient. Provide a (hypothetical) numerical example to clarify for the reviewer the overall procedure as modeled in the code from initial conditions through a transient. Discuss the overall heat balance and the fouling resistance. Provide the bases for the steam generator thermal resistances at nominal conditions? Provide comparisons to data if possible.

**Response**

The previous response to RAI 2.1-8 did contain errors in the formulas for the thermal resistances. The last several formulas in the response for the MARVEL-M input variables *RRPF*, *RRTM*, *RRBO* and *RRFOUL* should be written as follows:

$$RRPF = \frac{R_{pf}^0}{R_{tot}^0}$$

$$RRTM = \frac{R_{tube}^0}{R_{tot}^0}$$

$$RRBO = \frac{R_{bo}^0}{R_{tot}^0}$$

$$RRFOUL = 1 - RRPF - RRTM - RRBO$$

Note: *RRFOUL* is not an actual MARVEL-M input variable, but is calculated internally using the above equation.  
The superscript zeroes denote nominal conditions in these equations.

These equations, as written above, show the relationship between the mathematical models and the input variables. Descriptions of the equations and input are also found in Section 1.5.1 of the MARVEL-M Manual (GEN0-LP-480). As noted by the NRC, the values of *RRPF*, *RRTM*, *RRBO* and *RRFOUL* are input as constants into MARVEL-M. The MARVEL-M input parameters of *RRPF*, *RRTM*, *RRBO* and *RRFOUL*, are evaluated at nominal conditions.

$R_{tot}^0$  is internally scaled at nominal conditions as follows:

At nominal conditions, the SG overall heat transfer coefficient  $(UA)_{SG}$  can be calculated internally from the nominal SG power ( $q_{SG}$ ), the nominal SG temperature ( $T_s$ ) and the nominal primary side temperature ( $T_{avg}^0$ ) as shown in the equation below.

$$q_{SG} = (UA)_{SG} (T_{avg}^0 - T_s)$$

$U_0$  can be calculated from the overall heat transfer coefficient and the total resistance at nominal conditions is calculated with the equation below.

$$R_{tot}^0 = 1/U_0$$

Next, the overall heat transfer coefficient in the steam generators as a fraction of the nominal value can be obtained by combining the above equations as shown below:

$$\frac{U}{U^0} = \frac{R_{tot}^0}{(R_{pf} + R_{tube} + R_{foul} + R_{bo})}$$

$$\frac{U}{U^0} = \frac{1}{\left(\frac{R_{pf}}{R_{tot}^0} + \frac{R_{tube}}{R_{tot}^0} + \frac{R_{foul}}{R_{tot}^0} + \frac{R_{bo}}{R_{tot}^0}\right)}$$

where

$$\frac{R_{pf}}{R_{tot}^0} = \left(\frac{R_{pf}^0}{R_{tot}^0}\right) \cdot \left(\frac{1 + 10^{-2}T_0 - 10^{-5}T_0^2}{1 + 10^{-2}T - 10^{-5}T^2}\right) \left(\frac{Q}{Q_0}\right)^{-0.8}$$

$$\frac{R_{tube}}{R_{tot}^0} = \left(\frac{R_{tube}^0}{R_{tot}^0}\right) \frac{8.0 + 0.0051 \cdot T_t^0}{8.0 + 0.0051 \cdot T_t}$$

$$\frac{R_{foul}}{R_{tot}^0} = \frac{R_{foul}^0}{R_{tot}^0}$$

$$\frac{R_{bo}}{R_{tot}^0} = \left(\frac{R_{bo}^0}{R_{tot}^0}\right) \left(\frac{q_{sg} / A_{sg}}{q_{sg}^0 / A_{sg}^0}\right)^{-0.75} \cdot \exp\left(\frac{P_s^0 - P_s}{900}\right)$$

As described, nominal  $(UA)_{SG}$  is internally calculated and its transient value is modified based on transient conditions. Likewise, the thermal resistances are also modified based on transient conditions.

Table 2.1-8-1.1 gives an example of the required parameters in the equations above and the calculation results for the resistances in the Complete Loss of Forced Coolant Flow analysis discussed in Section 6.2 of MUAP-07010. Figure 2.1-8-1.1 shows the transient of these resistances calculated by MARVEL-M. These results indicate that the primary convection film resistance ( $R_{pf}$ ) increases associated with the decrease in reactor coolant flow and the secondary side boiling heat transfer resistance ( $R_{bo}$ ) increases associated with the decrease in feedwater flow caused by a reactor trip, while the tube metal resistance ( $R_{tube}$ ) and the fouling resistance ( $R_{foul}$ ) stay constant.

**Table 2.1-8-1.1 Calculation of Thermal Resistances during a Transient  
 - Complete Loss of Forced Reactor Coolant Flow**

Parameter	0.0 seconds	5.0 seconds	10.0 seconds
$T$ (°F)			
$T_t$ (°F)			
$Q$ (fraction)			
$q_{sg}$ (fraction)			
$A_{sg}$ (fraction)			
$P_s$ (psia)			
$R_{pf} / R_{tot}^0$			
$R_{tube} / R_{tot}^0$			
$R_{foul} / R_{tot}^0$			
$R_{bo} / R_{tot}^0$			
$R_{tot} / R_{tot}^0$			



**Figure 2.1-8-1.1 Thermal Resistances versus Time  
 - Complete Loss of Forced Reactor Coolant Flow**

**RAI 2.1-10-1**

There may be physical effects in the coolant flow not captured by modeling safety injection into the cold leg instead of the Direct Vessel Injection featured in the US-APWR. Justify why cold leg injection is more conservative.

**Response**

The MARVEL-M pressure model is not dependent on the location of injection. Therefore, the only effect of this modeling difference is in the time for the safety injection to reach the core. The previous response to RAI 2.1-10 describes that modeling safety injection into the cold leg instead of the direct vessel injection, as is physically done in the plant, adds additional purge volume which conservatively delays the effect of the boron reaching the core. Therefore, modeling the safety injection by cold leg injection rather than direct vessel injection is more conservative.

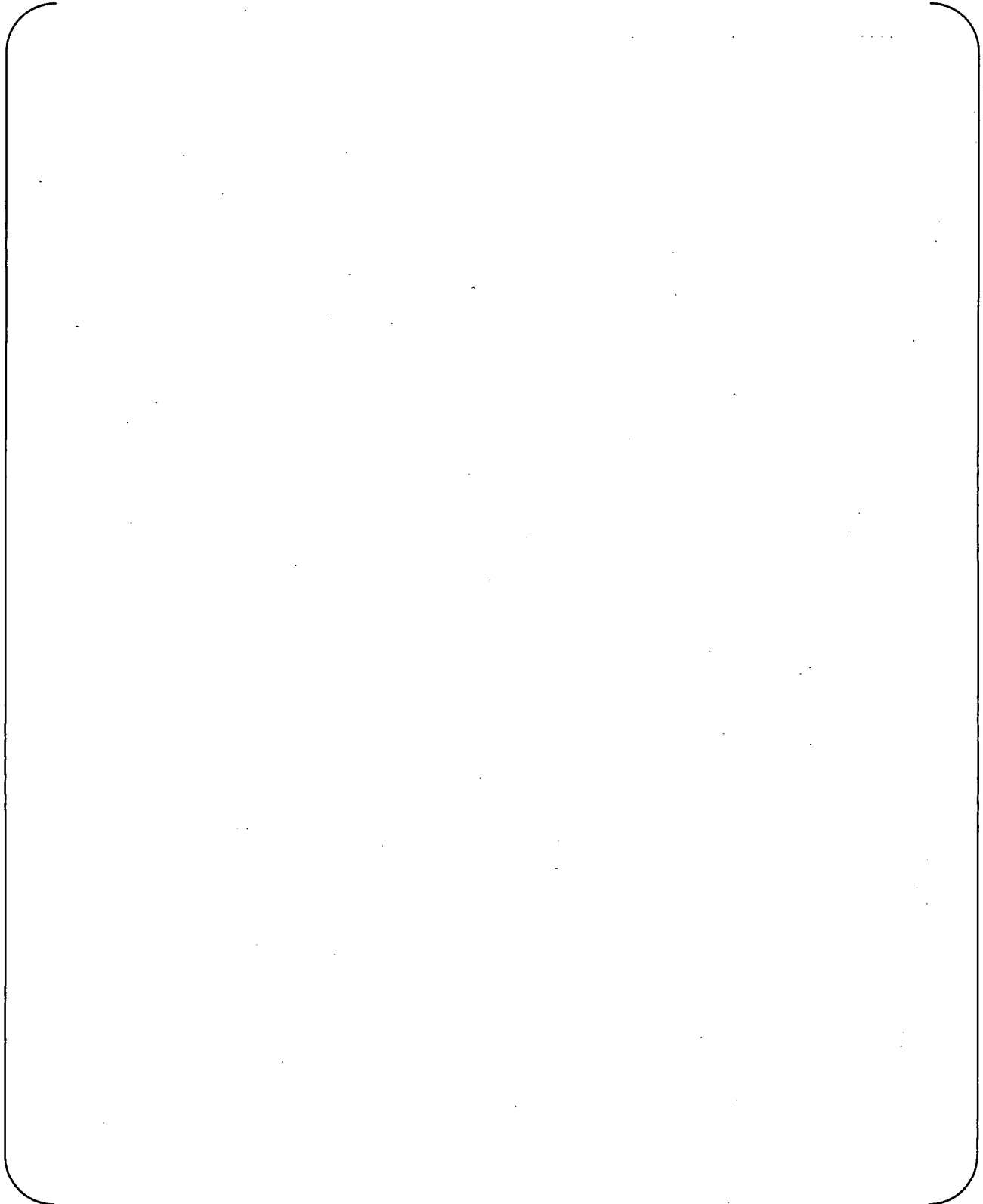
**RAI 2.1-13-1**

In the MHI response to RAI Appx.E-1 (in UAP-HF-08141, Docket No. 52-021, August 22, 2008), it is stated, "It should also be noted that uniformity in vessel inlet mixing can best be approximated by perfect mixing for very low flows such as during natural circulation conditions. Perfect mixing is assumed for the Steam Generator Tube Rupture event due to natural circulation conditions that exist during most of the event." This statement appears to contradict the MHI response to RAI 2.1-13. MHI is requested to clarify and comment on the discussion above.

**Response**

The original RAI response stated that, "The design value of  $f_{mi}$  is utilized for all DCD Chapter 15 non-LOCA events for the US-APWR, except for the steam line break, which uses the conservative value of  $f_{mi}$ ." This statement was incorrect because there are other Chapter 15 events that do not use the design or conservative values. The purpose of this response is to clarify the MARVEL-M mixing values that are used for the DCD Chapter 15 non-LOCA events that require this input. Table 2.1-13-1.1 shows the mixing factor in each of the non-LOCA events that are analyzed using the MARVEL-M code. In addition, the basis for the values used in the mixing model for each event is also shown in the notes below the table.

**Table 2.1-13-1.1 Mixing Factors in Non-LOCA Events Analyzed Using MARVEL-M**





**RAI 2.1-16-1**

Describe or demonstrate that the MARVEL-M's pump model is adequate for the full range of Chapter 15 events.

**Response**

For the Chapter 15 events where the reactor coolant pumps (RCPs) are running, constant reactor coolant system (RCS) flow is used, so there is no need for the RCP model. For other events, the RCP parameters are verified from the RCP scale data used to generate the homologous curves used in the Chapter 15 analysis. Three of the Chapter 15 events, partial loss of flow, complete loss of flow, and locked rotor (shaft seizure) have already been verified by comparisons to LOFTRAN, shown in Sections 3.1.2 through 3.1.4 of topical report MUAP-07010. Additionally, the partial loss of flow and complete loss of flow were verified by comparison to actual test data as shown in the response to RAI 2.1-16 provided in UAP-HF-08170-P.



**RAI 2.1-16-2**

Explain why the measured and predicted (using MARVEL-M) pump coast down curves are so similar. Have these measured data of pump coast down used to validate MARVEL-M been compared to the LOFTRAN code?

**Response**

For the comparison of MARVEL-M to actual plant test data, best estimate values are used in MARVEL-M rather than the more conservative values used in the DCD Chapter 15 safety analyses. In particular, the best estimate values for pump inertia and loop pressure drops lead to the excellent agreement between MARVEL-M and plant data shown in Figures 2.1-16.1 and 2.1-16.2. In the response to RAI 2.1-16, the measured plant data was compared to MARVEL-M, not LOFTRAN. However, MARVEL-M and LOFTRAN were compared for these same events, partial and complete loss of flow, in Sections 3.1.2 and 3.1.3 of the Non-LOCA Methodology Topical Report MUAP-07010. The comparisons show that MARVEL-M agrees very well with actual plant data and with LOFTRAN results for the pump coast down.

**RAI 3.1-2-1**

Did the Uncontrolled RCCA Bank Withdrawal at Power event have the largest differences between LOFTRAN and MARVEL-M?

**Response**

MHI has performed numerous comparisons between MARVEL-M and LOFTRAN. Only cases which resulted in significant differences between the results were investigated further to determine the reason for the differences. Two primary differences between the code models have been identified and previously described.

- (1) Topical report MUAP-07010 identifies differences in the core heat flux model that results in differences for peak pressure for the loss of flow event.
- (2) The UAP-HF-08141-P response to RAI 3.1-2 discusses differences in the pressurizer spray models that result in differences in results for the uncontrolled RCCA bank withdrawal at power event.

Overall, MARVEL-M compares extremely well with LOFTRAN and is suitable for Chapter 15 analyses.

**RAI 3.2-3-1**

Please confirm that the approach used in TWINKLE-M modifies the two group diffusion coefficients for both the radial reflector region and two axial reflector regions.

**Response**

The first (fast) group diffusion coefficient is modified for both the axial and radial reflector regions. The second (thermal) group diffusion coefficient is not modified because the desired effect on the TWINKLE-M power distributions is achieved by modifying the first group diffusion coefficient for these reflector regions.

**RAI 3.2-3-2**

Detail the process by which the diffusion coefficient in the reflector is modified before it is input into TWINKLE-M.

**Response**

[Empty response box]

**RAI 3.2-4-1**

MHI asserts that the differences between the two codes in the assemblies with control rods inserted for the HZP case is due to the different modeling of the spatial dependence and the fact that there is a strong spatial gradient near the control rods. This is a reasonable explanation; however, it is not consistent with their response to 3.2-5. The explanation of the differences at the location of a control rod for the HFP case is claimed to be because the burnup is different in the controlled and surrounding assemblies. However, since the reactor is expected to be operated with control rods withdrawn, the burnup differential between controlled assemblies and surrounding assemblies should be no different than between any other set of adjacent assemblies. Please provide further clarification of the responses to RAI 3.2-4 and 3.2-5.

**Response**

The topical report provided comparisons of the radial power distribution between TWINKLE-M and ANC for several cases. The BOC, HFP, all RCCAs out case was shown in Figure 3.2.1-2 and the EOC, HZP, RCCAs at the insertion limit case was shown in Figure 3.2.1-3. There were several RAIs about the differences in these comparisons. RAI 3.2-4 discussed the HZP case (Figure 3.2.1-3) and RAI 3.2-5 discussed the HFP case (Figure 3.2.1-2). In the response to RAI 3.2-4, it was stated that the largest differences between TWINKLE-M and ANC for the HZP case were in assemblies with the control rods inserted. This could be verified by Figure 3.2.1-1 in the topical report that showed the locations of the RCCAs. In the response to RAI 3.2-5, it was stated that the largest differences between TWINKLE-M and ANC for the HZP case were in assemblies near the periphery of the core where adjacent fuel assemblies had large differences in burnup. However, no burnup information was included in that response.

The example that was provided by MHI in the responses to RAIs 3.2-4 and 3.2-5 included adjacent assemblies with significant differences in burnup, as shown in Figure 3.2-4-1.1. The figure also indicates the assemblies with the largest differences in power between ANC and TWINKLE-M for the HFP case (based on Figure 3.2.1-2 of the topical report). The figure confirms the explanation in the response to RAI 3.2-5 that the largest differences in power for the HFP case occur in assemblies with fresh fuel that are adjacent to high-burnup assemblies near the periphery of the core. As a result, the differences between TWINKLE-M and ANC can be attributed to the large differences in burnup.



**Figure 3.2-4-1.1 Burnup Distribution at BOC (0.15 GWD/MTU)**

**RAI 3.2-6-1**

Comment on the positive Doppler temperature coefficients presented in Table 3.2-6.1.

**Response**

The positive values in Table 3.2-6.1 are typographical mistakes. The Doppler temperature coefficient values for both TWINKLE-M and ANC should be  $-1.9 \text{ pcm}/^{\circ}\text{F}$ .

**RAI 3.2-7-1**

MHI states that the ejected rod worth is the same with both mesh sizes in part because the steady state power distributions are the same. It is true that the power distributions should be similar as they are both tuned to the same ANC-generated power distributions. The result shown for the Doppler fuel temperature is similar for the two meshes with the 2x2 mesh giving a higher value. MHI also supplies radial power distributions at different times during a rod ejection accident to show that the effect of mesh size is small. Several comparisons of the effect of axial mesh are also shown to be insensitive to mesh size. The conclusions implicitly assume that the 4x4 mesh yields a converged solution. What changes would occur if a smaller mesh (6x6 or 8x8 radially or >76 mesh points axially) were used?

**Response**

As discussed in previous RAI responses, the nodal expansion method used in ANC can more accurately calculate steep flux gradients than the finite difference method used in TWINKLE-M. In general, using a finer mesh size in a finite difference code will allow for a more accurate calculation in regions of steep flux gradients. Therefore, it is expected that using a finer mesh size in TWINKLE-M would result in power distributions that more closely match those generated by ANC. However, the reactivity insertion and peaking factor from TWINKLE-M are already adjusted to match the safety limit value which is determined based on the ANC calculations. The hot spot temperature response is highly dependent on the reactivity insertion and peaking factor and is not affected by the small differences in power distribution between TWINKLE-M and ANC. Therefore, it is not necessary to use a finer mesh size, since the final result will not be affected.



**RAI 3.2-7-2**

What is the time step size used in the TWINKLE-M simulation of the rod ejection accident?  
How is this time step size determined?

**Response**

Section 4.6 of the TWINKLE-M Input Manual, GEN0-LP-517(R0), provides a standard set of time steps, reproduced in the table below. This standard set of time step values were used in the TWINKLE-M analysis of the rod ejection accident. In addition to this standard set of time steps, a sensitivity analysis is performed to judge that the time steps used in the analysis are appropriate. If necessary, the time steps are adjusted appropriately. The TWINKLE-M Input Manual was submitted to the NRC by letter UAP-HF-08138 on August 1, 2008.



**RAI 3.2-7-3**

In Figure 3.2-7.6, why is no adjustment made of the diffusion coefficient in the reflector region?

**Response**

Figure 3.2-7.6 showed the average axial power distribution comparison between ANC and the results from TWINKLE-M. No adjustment is made to the diffusion coefficient in order to better see the difference between the case. If the diffusion coefficient in TWINKLE-M was adjusted, then the cases would agree very well and would be difficult to distinguish from each other in the figure.

**RAI 3.2-8-1**

Were the adjustments made to neutron lifetime and delayed neutron fraction adjusted to minimize the difference when changing mesh size? Were these only made for the comparison of mesh sizes or is this part of the procedure for calculating transients?

**Response**

The neutron lifetime and delayed neutron fraction are key parameters that affect the transient calculation. These parameters are adjusted in order to match the safety limit. The parameters were adjusted separately for the two mesh sizes, but in each case the adjustment is only made in order to match the safety limit. This adjustment is part of the procedure for calculating transients; it occurs uniquely for every analysis.

**RAI 3.2-9-1**

Explain how the adjustments made in the responses to RAIs 3.2-7 through 3.2-9 relate to the way in which analysis is generally done with TWINKLE-M.

**Response**

As described in the preceding response to RAI 3.2-8-1, adjustments to key parameters are made as part of each analysis. The parameters are adjusted to match the values from the safety analysis limit.

**RAI 5.3-1-1**

Do operating procedures allow for fully or partially inserted misaligned or inoperable control rods and if so, was this taken into account in the analysis of the design limit? .....

**Response**

All transients are assumed to begin with the most severe power distributions that are consistent with operation within the Technical Specifications (TS). In Chapter 16 of the DCD, Limiting Condition for Operation (LCO) 3.1.4 states that all shutdown and control rods shall be operable and that individual indicated rod positions shall be within 12 steps of their group step counter demand position. The impact of this LCO on the safety analysis is described in the Bases section of TS 3.1.4. The operability of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. The maximum rod misalignment is another initial assumption in the safety analysis that directly affects core power distributions and assumptions of available shutdown margin.

**RAI 5.3-1-2**

The TWINKLE-M model is adjusted to give the same design limit worth as from the ANC calculation. Is the identical configuration represented in the TWINKLE-M three-dimensional model?

**Response**

TWINKLE-M uses the identical configuration (geometry and mesh) as ANC for the three-dimensional model.

**RAI 5.3-1-3**

The adjustment is done by "changing the eigenvalue" in both the three- and one dimensional models. In the former, this would mean that a multiplier is applied to the fission rate throughout the reactor rather than to a local property of the controlled fuel assembly (e.g., absorption cross section). What is the impact on the analysis of this approximation?

**Response**

For the RCCA ejection analysis, there are two methods to add reactivity to the core. One method is to change the eigenvalue, which modifies the fission rate in all regions of the core. The second method is to decrease the local absorption cross section in the assembly where the RCCA is being ejected. While these two methods seem very different, the end results for the RCCA ejection analysis are similar. To show evidence that the results are similar, MHI is providing a sensitivity study comparing the following two cases:



The sequence of events for the sensitivity study is shown in Table 5.3-1-3.1 below. The results of the sensitivity study for the nuclear power and the fuel enthalpy as a function of time are shown in Figures 5.3-1-3.1 and 5.3-1-3.2, respectively. As the figures show, the two methods give very similar results. The hot spot peak fuel enthalpy is 77.8 cal/g for the topical report analysis, 51.1 cal/g for Case 1, and 50.9 cal/g for Case 2. The general shape of the two results is also similar to the topical report results, although the absolute values are different due to the difference in total reactivity insertion (600 pcm for the sensitivities vs. 800 pcm for the topical report). In addition, Figure 5.3-1-3.3 shows the radial enthalpy distributions for both cases<sup>\*1</sup>. These results show that the two methods give similar results and it is therefore acceptable to use the method of "changing the eigenvalue" in the R/E analysis.

\*1 Note that Figure 5.3-1-3.3 is generated from TWINKLE-M. As described in Section 5.3 (2) of MUAP-07010 "Non-LOCA Methodology", a local fuel enthalpy rise is calculated in TWINKLE-M code by integration of the local power and power density in each mesh. The values in Figure 5.3-1-3.3 are adjusted to match the fuel enthalpy of the hottest assembly to the result of VIPRE-01M. The fuel enthalpy in each assembly is the maximum fuel enthalpy of the axial nodes.

**Table 5.3-1-3.1 Sequence of Events for the RCCA Ejection (EOC HZP)**

<b>Event</b>	<b>Topical Report Time (sec)</b>	<b>Case1 Time (sec)</b>	<b>Case2 Time (sec)</b>
Rod Ejection Occurs	0.0	0.0	0.0
Neutron Flux High (low setting) Analysis Limit Reached	0.15	0.20	0.22
Peak Nuclear Power Occurs	0.16	0.22	0.24
Reactor Trip Initiated (Rod Motion Begins)	1.05	1.10	1.12
Maximum Fuel Temperature Occurs	-	-	-
Maximum Fuel Enthalpy Occurs	1.60	1.66	1.68



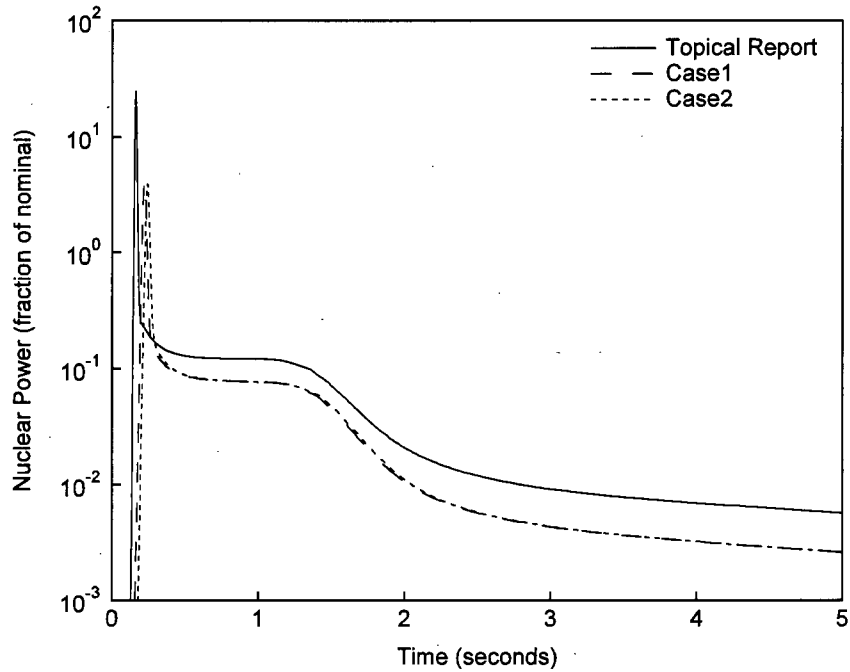


Figure 5.3-1-3.1 Nuclear Power versus Time – RCCA Ejection (EOC HZP)

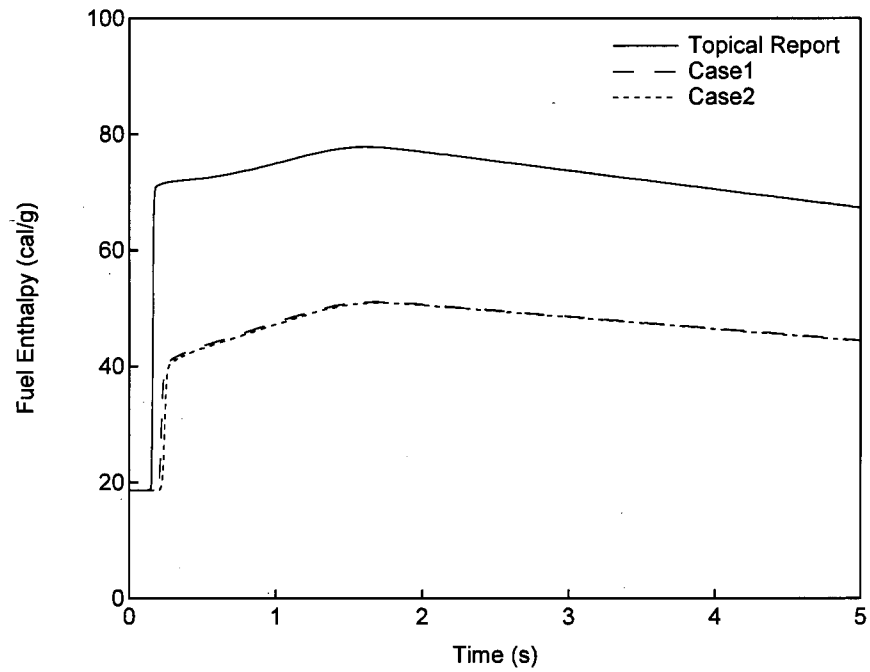
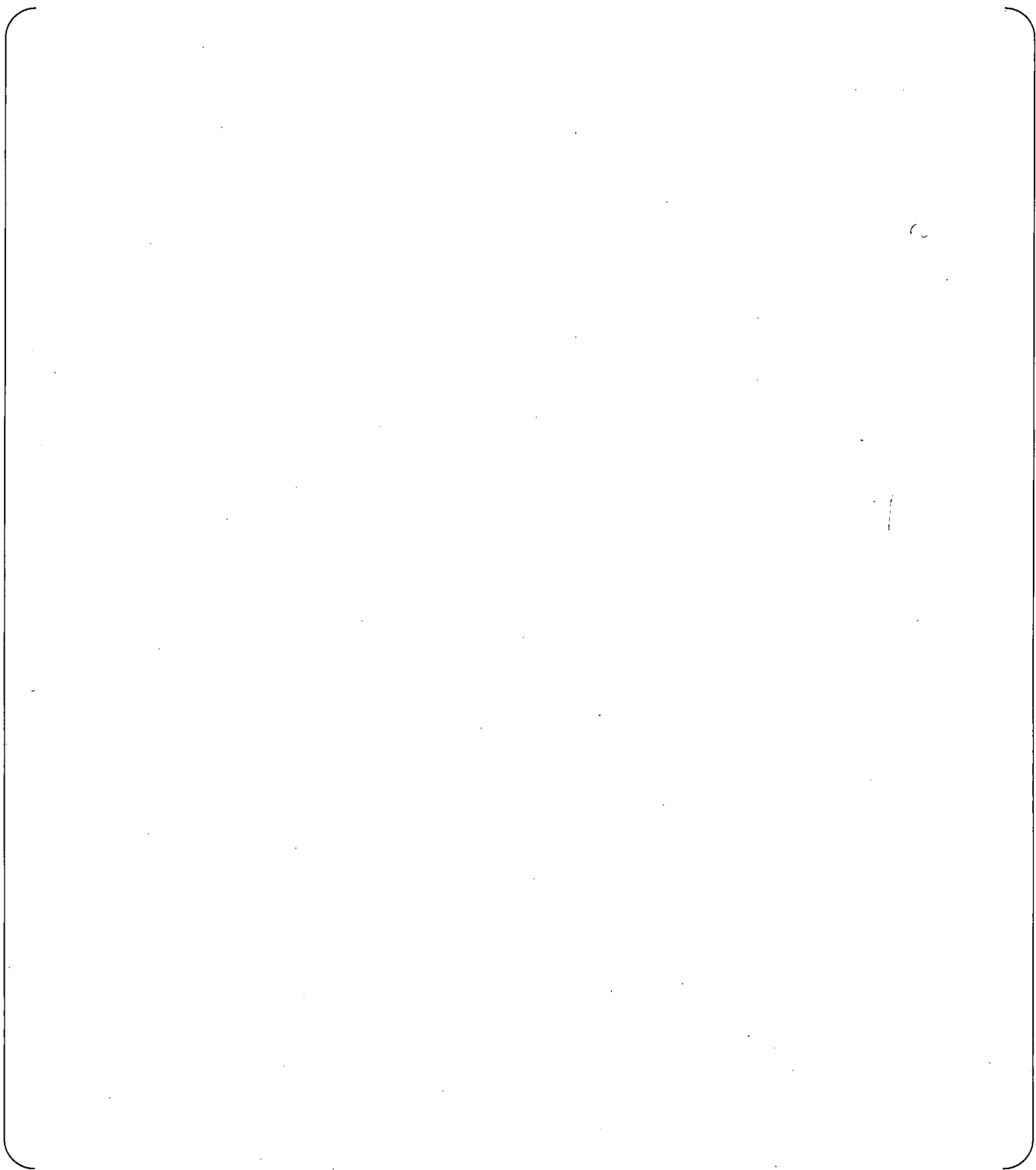


Figure 5.3-1-3.2 Fuel Enthalpy versus Time – RCCA Ejection (EOC HZP)



**Figure 5.3-1-3.3 Radial Enthalpy Distribution at the Time of Maximum Enthalpy  
RCCA Ejection (EOC HZP)**

**RAI 5.3-2-1**

According to Figure 5.2-1, the VIPRE-M calculation needs rod power factors for 23 rods; TWINKLE-M presumably provides the "hot channel factor" for the hottest quarter (based on a 2x2 mesh) of a fuel assembly, i.e., not even for one fuel rod. Please explain what information is used by VIPRE-M to analyze the hot channel.

**Response**



**RAI 5.3-2-2**

The response to this RAI indicates that whatever the model is, it is not necessary to do a detailed 1/8 core calculation; all information is obtained by focusing on the hot channel. Does this allow you to make the assumption that the VIPRE-M model can place the ejected rod at the center of the core rather than at its realistic position?

**Response**

The reviewer's understanding of the VIPRE-01M model is correct. As discussed in the response for RAI 5.3-4 in UAP-HF-08141, the limiting parameters for the rod ejection event are fuel enthalpy, fuel temperature, and cladding temperature, which reach their respective maximum values before the transient coolant condition significantly changes and is able to affect the core. This allows for calculating the behavior of the hottest rod in the center of the VIPRE-01M 1/8 core model by applying appropriate boundary conditions.

**RAI 5.3-2-3**

The third sentence of the second paragraph in response to RAI 5.3-2 is grammatically incorrect; clarify.

**Response**

MHI would like to clarify this sentence by rewriting it as follows: The power distribution around the hot assembly is assumed to be the thermal design power distribution. This assumption has no effect on the results and, therefore, either a 1/8 core model or a single channel calculation can be used for this analysis.

The intent of the sentence is to explain that as long as the channels around the hot assembly are modeled, it does not matter what portion of the core is modeled.

The response to RAI 5.3-2, including this correction, is shown in entirety below:

The three-dimensional distribution of fuel enthalpy is calculated in TWINKLE-M using a mesh-wise average model, whereas the maximum fuel enthalpy rise is calculated in VIPRE-01M. The maximum enthalpy rise is calculated using a detailed sub-channel model in VIPRE-01M, which is, in turn, used to compensate for the difference between the mesh-wise and pin-wise enthalpy rise. The detailed procedure for how the three-dimensional distribution of enthalpy rise is adjusted to pin-wise enthalpy is provided in Section 5.3 of MUAP-07010.

To calculate the hot spot enthalpy rise in VIPRE-01M for PCMI failure, histories of core average power and hot channel peaking factor ( $F_Q$ ) calculated in TWINKLE-M are passed to VIPRE-01M. The  $F_Q$  history is scaled using a multiplier so that the maximum value is the design limit, which is applied to the hot assembly of the 1/8 core model. The power distribution around the hot assembly is assumed to be the thermal design power distribution. This assumption has no effect on the results and, therefore, either a 1/8 core model or a single channel calculation can be used for this analysis. However, MHI has selected to use the 1/8 core model for the rod ejection analysis in order to assure consistency of the base input of VIPRE-01M with the core thermal hydraulic design.

In summary, to ascertain whether or not the PCMI acceptance criteria are met, the three-dimensional distribution of enthalpy calculated by TWINKLE-M is applied, considering a peak / average ratio in the mesh using the VIPRE-01M hot spot results. Histories of core average power and hot channel factor are necessary information to calculate the maximum enthalpy rise in VIPRE-01M.

**RAI 5.3-3-1**

The response suggests that the design limit is actually just the ANC generated peak/average fuel assembly power (with some unknown adjustment for uncertainty and "safety margin") for the case with the ejected rod out and no other change in rod configuration. It is used to add a conservative factor to the analysis only because it turns out to be larger than the same ratio calculated by TWINKLE-M. Please confirm this understanding of the hot channel design limit, mathematically describe the hot channel factor, uncertainty and safety margin, and explain how the uncertainty and safety margin are applied to the design limit.

**Response**

The reviewer's understanding is basically correct. The design limit is determined based on the value calculated using the core design code ANC, considering calculation uncertainty and safety margin. The TWINKLE-M value does not include calculation uncertainty or safety margin and is therefore less than the ANC value. By using the lower (TWINKLE-M) value of the hot channel factor, the result is a higher peak power excursion. This larger peak power from TWINKLE-M is used as input to the VIPRE-01M analysis. (

)

The mathematical description of the hot channel factor is as follows: The "hot channel" is the coolant channel in which the maximum heat flux occurs and the "hot channel factor" is the maximum heat flux in the core (the heat flux in the "hot channel") divided by the average heat flux in the core.

(

**RAI 5.3-6-1**

In MUAP-07009-P and in the response to RAI 5.3-6, it is noted that mixing between assemblies is "conservatively ignored." Yet later in the response to RAI 5.3-6, it is stated that "the power of the hot assembly is assumed to be higher than the power of the surrounding assemblies, which would cause *larger flow redistribution from the hot assembly to the surrounding assemblies* and result in a more limiting coolant condition in the hot assembly." As written, these two statements are contradictory. Clarify.

**Response**

In the response to RAI 5.3-6, the word "mixing" is meant to refer to the turbulent mixing of energy and momentum, which is associated with equal mass exchange between adjacent channels due to turbulence. This is more clearly explained in MUAP-07009-P but was unclear as was written in the original response to RAI 5.3-6. The response should have used the term "turbulent mixing" rather than "mixing" for clarity.

**RAI 5.3-7-1**

The response explains how cross sections are averaged radially, but does not address other parameters such as diffusion coefficients and delayed neutron data. Please provide the missing information.

**Response**

The nuclear parameters such as the diffusion coefficients and delayed neutron data are also averaged using a neutron flux and volume weighting method. The equation for the averaging of the diffusion coefficients and the delayed neutron data is the same as was shown for the macroscopic cross section, which was given in the UAP-HF-08141-P response to RAI 5.3-7.



**RAI 5.3-7-2**

In response to RAI 5.3-7 of UAP-HF-08141-P MHI provided a comparison between the TWINKLE-M 3-D and 1-D core average axial power in Figure 5.3-7.1. Is the difference given in Figure 5.3-7.1 a representative or maximum expected difference? If representative, please explain if a larger difference would have a significant impact on the number of rods in DNBR.

**Response**

The comparison provided in Figure 5.3-7.1 is a representative example of the axial power distribution for the 1-D and 3-D TWINKLE-M analyses for the beginning of cycle. The difference in axial power distribution has only a very slight impact on the number of rods in DNBR, even if the differences were larger than those shown in the representative example. Additionally, the rod ejection analysis using 1-D methodology assumes a large amount of conservatism in the inserted reactivity, the hot channel factor and the Doppler weighting factor. These other conservative assumptions are more than large enough to bound the impact of the relatively small differences in axial power distribution.

**RAI 5.3-10-1**

According to the response, the control rods are modeled explicitly. However, the report says that the trip reactivity is the design limit, i.e., not what is calculated by TWINKLE-M. Please explain further.

**Response**

The rod absorption cross section is adjusted so that the total worth of the tripped rods is equal to the design limit.

**RAI App.F-1-1**

Although the final form of the "modified Zaloudek" correlation is attractive due to its simple dependence on  $(P - P_{sat})$ , comparison to experimental data is necessary to show that the correlation is accurate under the conditions for which it was developed. Please provide additional validation for the "modified Zaloudek" correlation.

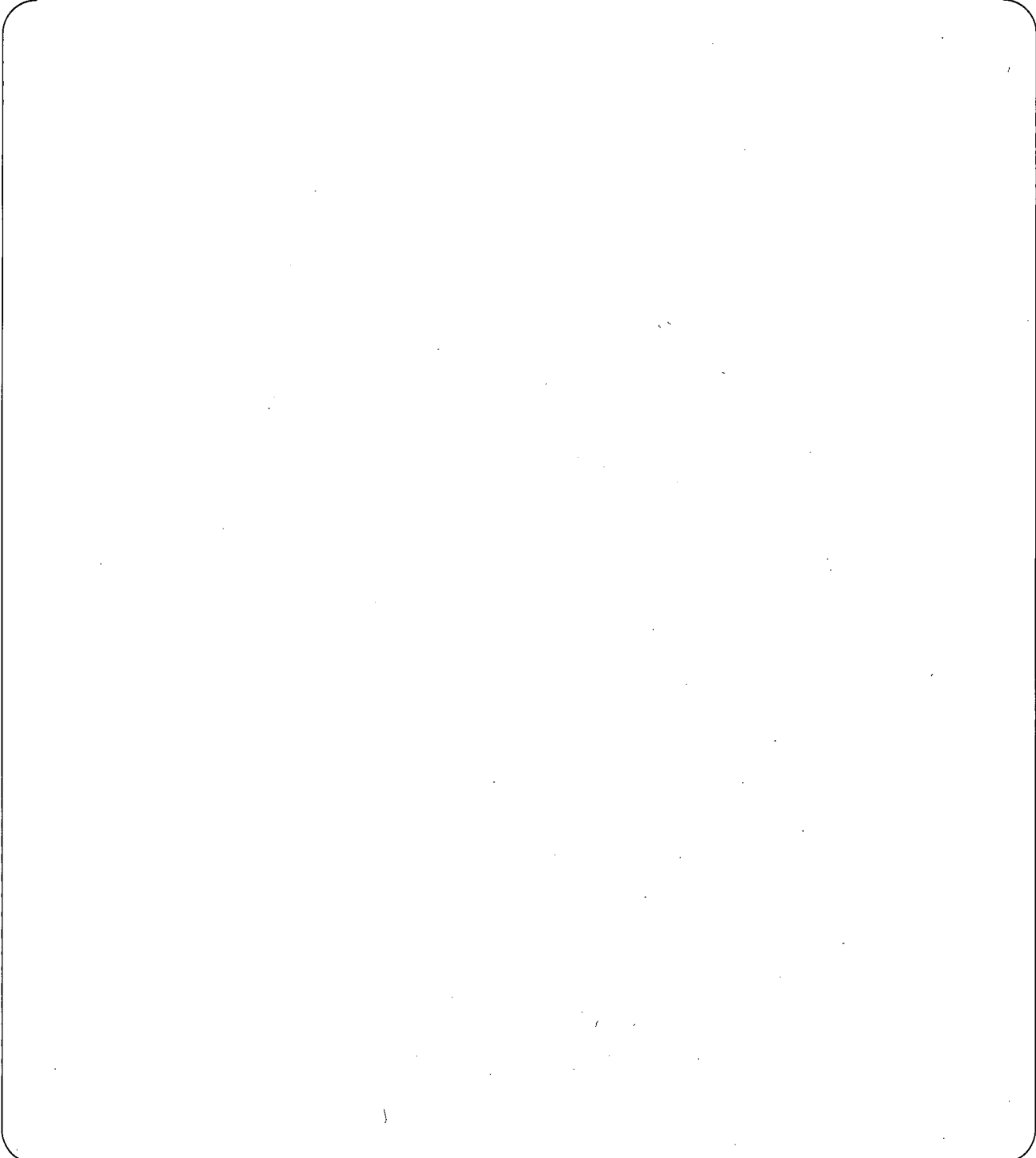
**Response**

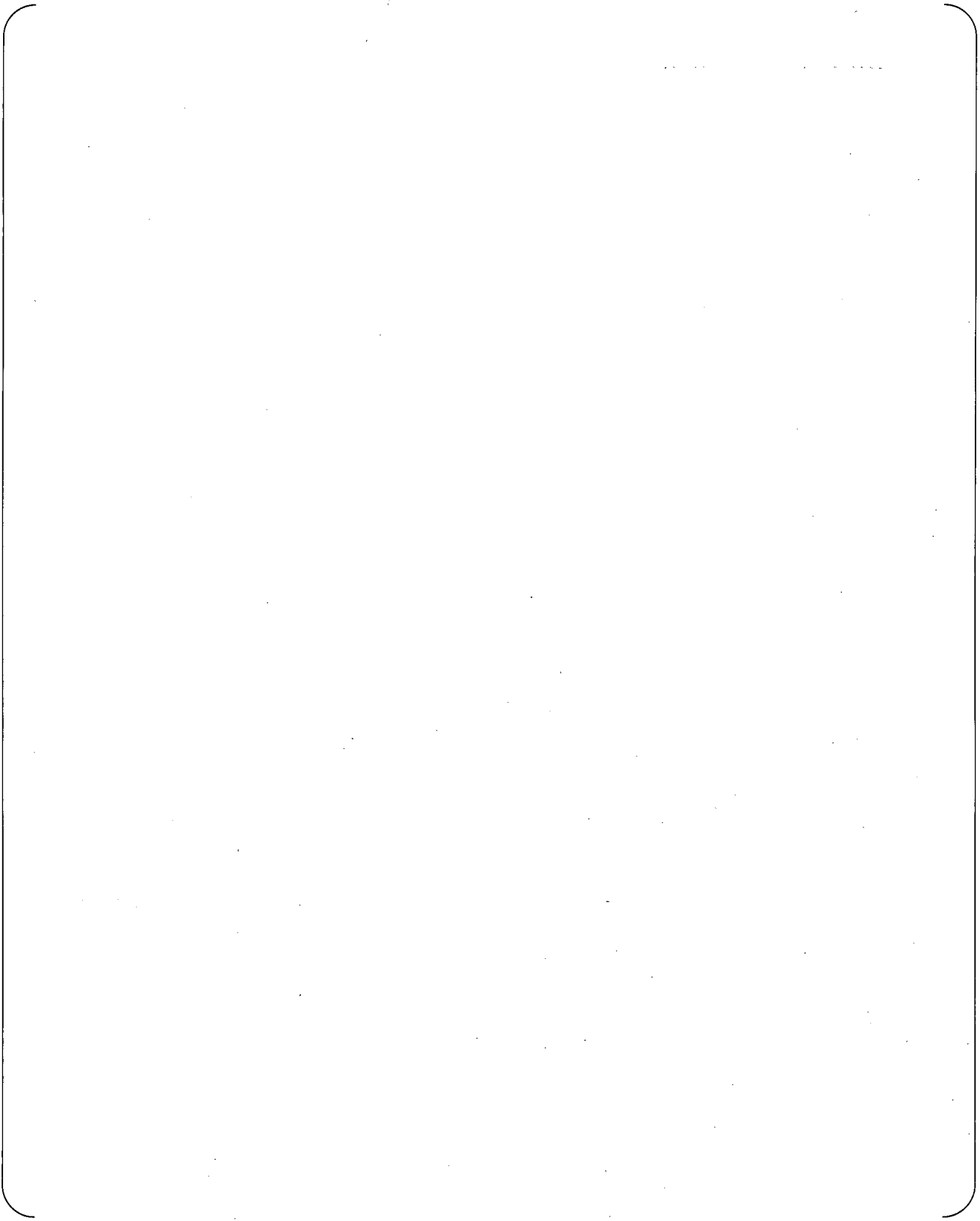
The modified Zaloudek correlation is part of the conservative break flow model used in MARVEL-M for the steam generator tube rupture (SGTR) analysis. The SGTR analysis in MARVEL-M was validated by a comparison to an actual SGTR event at the [ ]  
The MARVEL-M validation used a realistic break flow model with a discharge coefficient ( $C_D$ ) of [ ] and is described in detail in the response to RAI 3.1-6. In Appendix F of the "Non-LOCA Methodology Topical Report" (MUAP-07010), the calculation results for the US-APWR with a conservative break flow model (in which the initial break flow rate is calculated by the modified Zaloudek correlation) are compared to those of the realistic break flow model (using  $C_D=1.0$ ). Figure F-2 in the topical report shows that the conservative break flow model is conservative with respect to the realistic break flow model. Since the realistic break flow model was shown to give accurate results when compared to data from an actual SGTR event, as described in the response to RAI 3.1-6, the conservative break flow model will likewise give conservative results for an actual SGTR event. Therefore, the use of the modified Zaloudek correlation is both reasonable and conservative for the SGTR event.

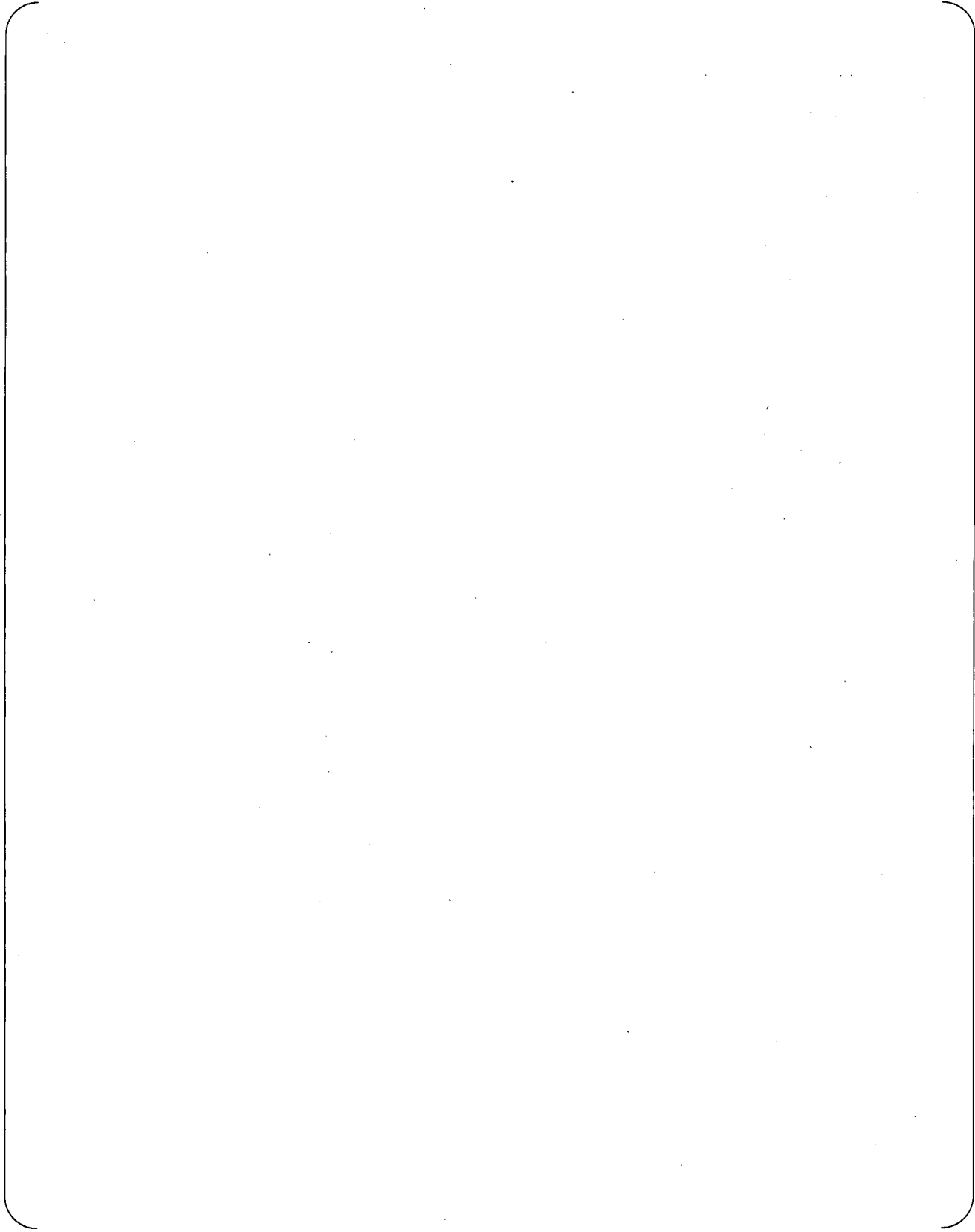
**RAI 3.1-6**

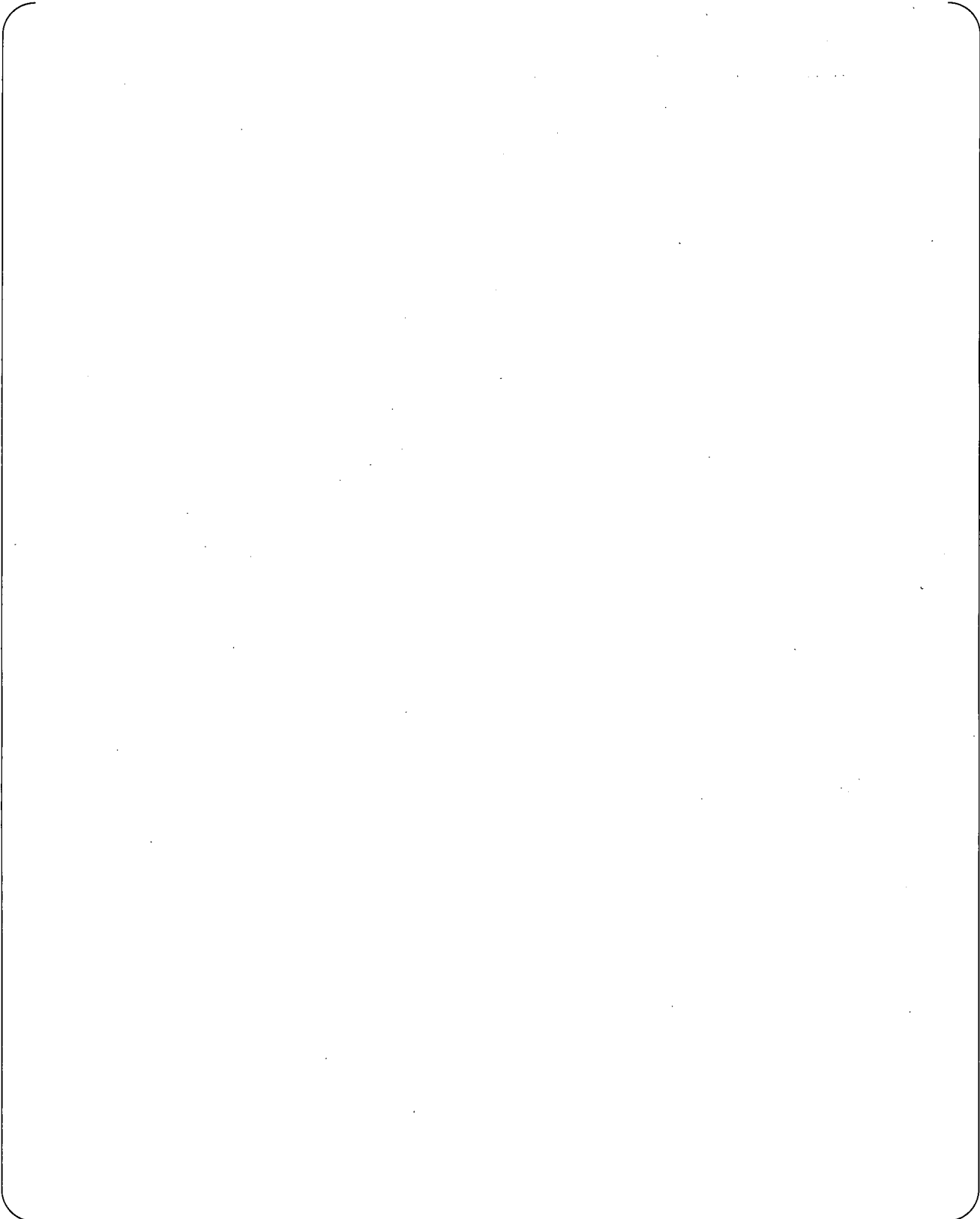
Present validation data of MARVEL-M for the steam generator tube rupture event.

**Response**















**Figure 3.1-6.1 RCS Pressure versus Time**



**Figure 3.1-6.2 Ruptured Steam Generator Pressure versus Time**



**Figure 3.1-6.3 Intact Steam Generator Pressure versus Time**



**Figure 3.1-6.4 Ruptured Steam Generator Water Level versus Time**



**Figure 3.1-6.5 Intact Steam Generator Water Level versus Time**



**Figure 3.1-6.6 Intact Steam Generator Water Level versus Time  
Sensitivity Analysis to Steam Generator Model**



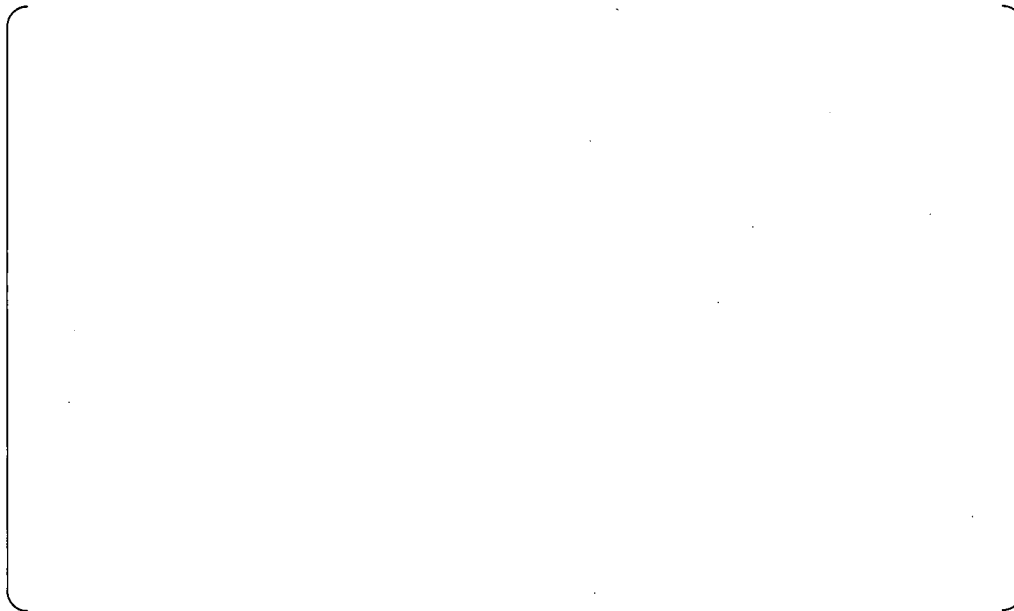
**Figure 3.1-6.7 Pressurizer Water Level versus Time**



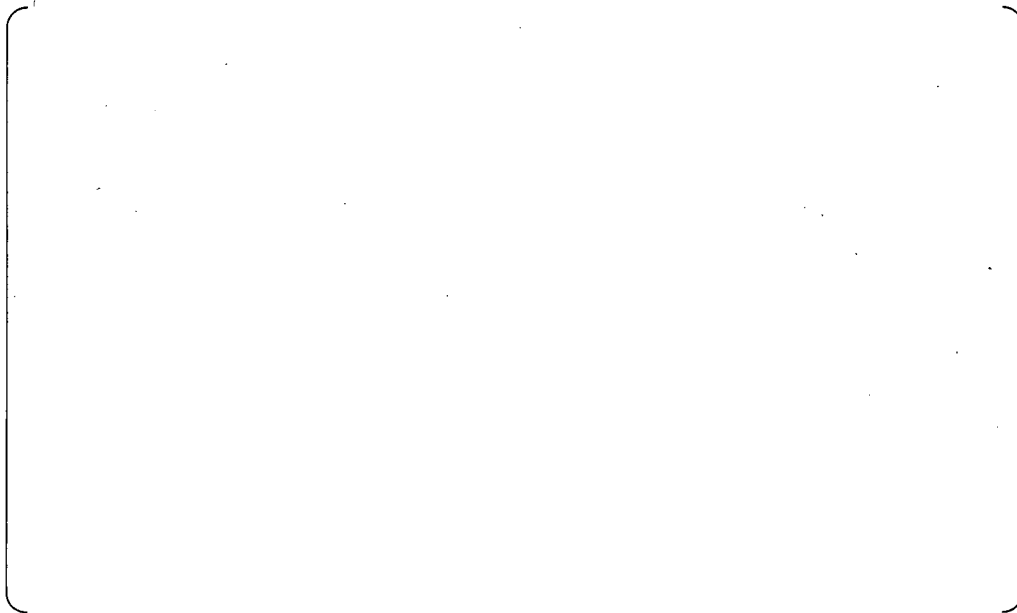
**Figure 3.1-6.8 Ruptured Loop Hot Leg Temperature versus Time**



**Figure 3.1-6.9 Intact Loop Hot Leg Temperature versus Time**



**Figure 3.1-6.10 Safety Injection Flow Rate to Deluge Line versus Time**



**Figure 3.1-6.11 Safety Injection Flow Rate to Cold Leg versus Time**

**RAI 3.1-7**

Give the history of the version of LOFTRAN used in MARVEL-M validation. Is the version of LOFTRAN employed by MHI for validation the same as that which was approved by the NRC? Detail any changes MHI has made to LOFTRAN. This is necessary because MHI is using LOFTRAN to validate the results from MARVEL-M, and it must be clear that the version of LOFTRAN employed has received licensing approval in the United States.

**Response**

[Empty response box]

**RAI 3.1-8**

MHI uses LOFTRAN as one of the means to demonstrate that the MARVEL-M code behaves as expected. For the comparison to be meaningful, the algorithms, numerical methods and, if used, correlations should be sufficiently different. Does MARVEL-M share any significant algorithms, numerical methods or correlations with the version of LOFTRAN used for the comparison? If yes, please describe the similarities. If no, provide examples where there are fundamental differences.

**Response**

MARVEL and LOFTRAN are completely different codes with respect to the FORTRAN programming. The two codes were developed independently by different individuals, and the internal subroutine structure and variable names are different. MARVEL was used for selected FSAR Chapter 15 analyses for events with non-uniform behavior (prior to the availability of 4-loop LOFTRAN in the late 1970s) such as:

- Inadvertent opening of a steam generator relief or safety valve
- Startup of an inactive loop at an incorrect temperature
- Steam system piping failure

The NRC approved the use of MARVEL for those FSAR Chapter 15 analyses. MARVEL and 4-loop LOFTRAN were both approved in 1984 by the NRC. The NRC continues to approve the use of 4-loop LOFTRAN in FSAR Chapter 15 analyses of recent US PWRs. However, MARVEL has not been used in recent US PWR plant FSAR Chapter 15 analyses since the 4-loop LOFTRAN code was approved by the NRC.

As described in detail in the response to RAI 2.1-21 in MHI letter UAP-HF-08245, MHI made some improvements and refinements to certain original MARVEL models. As part of the validation of MARVEL-M, comparisons of MARVEL-M and 4-loop LOFTRAN have been made and presented to NRC in the Non-LOCA Topical Report (MUAP-07010) and the associated RAI responses. The MARVEL-M and 4-loop LOFTRAN calculation results agree very well and in some cases are almost identical. Therefore, MARVEL-M has the same capabilities as 4-loop LOFTRAN, which continues to be approved by the NRC. MHI believes that the comparison of MARVEL-M to an NRC approved code, 4-loop LOFTRAN, is a reasonable method of code of code certification and validation.



**RAI App E-2**

Show the scale on the vertical axis of Figure E-1.

**Response**

The revised version of Figure E-1 that includes the axes scales is shown in Figure E-2.1 below.



**Figure E-2.1 DNBR versus Reactor Inlet Mixing for the Steam System Piping Failure**