MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

September 11, 2008

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-08170

Subject: MHI's 2nd Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's 2nd Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology." The enclosed materials provide MHI's remaining responses to the NRC's "Request for Additional Information ("RAI") on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology," dated July 16, 2008. The first response was previously provided to the NRC by MHI letter UAP-HF-08141, dated August 22, 2008. In accordance with the agreement reached during the June 26, 2008 conference call, the remaining RAI responses (numbers 2.1-16, 3.2-2, 3.2-7, 3.2-10, and 5.5-2) are included in the enclosed document.

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

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

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

Sincerely,

Y. Ogata

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

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's 2nd Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology (proprietary)
3. MHI's 2nd Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology (non-proprietary)

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

Contact Information

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ENCLOSURE 1

Docket No. 52-021

MHI Ref: UAP-HF-08170

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 2nd Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology," dated September 2008, 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 11th day of September, 2008.

Y. Ogata

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

Enclosure 3

UAP-HF-08170
Docket No. 52-021

September 2008

MHI's 2nd Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology

(Non-Proprietary)

RAI 2.1-16

A reactor coolant pump model has been added to MARVEL during the evolution to MARVEL-M. Has this model been tested against pump vendor test data? If so, please provide the comparisons.

Response

The reactor coolant pump (RCP) characteristics incorporated into MARVEL-M are based on the RCP unit test data that MHI has performed. MHI has also compared the calculated coast down curve with the test data for a 4-loop Japanese plant in order to confirm the validity of the MARVEL-M RCP model. Figures 2.1-16.1 and 2.1-16.2 compare the measured coast down 4-loop plant data to the MARVEL-M calculation results. Figure 2.1-16.1 shows a test of a two pump coast down (Partial Loss of Forced Reactor Coolant Flow) and Figure 2.1-16.2 shows a test of a four pump coast down (Complete Loss of Forced Reactor Coolant Flow). Both the calculations and the measurements were performed at hot zero power initial conditions. As can be seen from the figures, the calculated values from MARVEL-M show excellent agreement with the measured values. Therefore, the reactor coolant pump model in MARVEL-M is valid for both uniform and non-uniform RCS flow conditions.

Figure 2.1-16.1 RCS Flow Comparison between a Typical 4-Loop Plant and MARVEL-M (Partial Loss of Forced Reactor Coolant Flow)

Figure 2.1-16.2 RCS Flow Comparison between a Typical 4-Loop Plant and MARVEL-M (Complete Loss of Forced Reactor Coolant Flow)

RAI 3.2-2

The TWINKLE code as approved for use by Westinghouse needs to be shown to provide acceptable results when used by MHI. This becomes more important when the code has also been modified. The comparisons in Section 3.2.1 show reasonable agreement with ANC for reactor conditions of interest in rod ejection accident analysis. However, no information is given to show how well TWINKLE-M models steady state power and reactivity measurements at an operating PWR, as well as how it models neutron kinetics. Please provide additional information which helps validate the code as used at MHI.

Response

MHI was introduced to the TWINKLE code by Westinghouse and has been using TWINKLE-M in domestic licensing analyses for many years, as described in MUAP-07010 Section 2.2. The additional comparisons provided in this RAI response supplement the extensive verification and validation activities already performed by MHI for the TWINKLE-M code.

Figures 3.2-2.1 and 3.2-2.2 compare the radial power distributions of two typical 3-loop operating plants to the TWINKLE-M calculations. These comparisons show good agreement and demonstrate the accuracy of the three-dimensional (3D) steady state calculation using TWINKLE-M.

As a verification of the transient capability of TWINKLE-M, the OECD rod ejection 3D transient benchmark problem was performed. This benchmark problem is designed to confirm the ability of various kinetics code solution methods to give common core fuel burnup, cross sections, rod cluster control assembly (RCCA) macroscopic cross sections, fuel properties, and several core physics parameters important to transient calculations. Figures 3.2-2.3 through 3.2-2.5 compare the results calculated using TWINKLE-M to the OECD reference problem solution for the case where one rod is ejected from the bottom of the core at hot zero power initial conditions. These comparisons show good agreement between TWINKLE-M and the reference solution, which demonstrates the validity of the neutron kinetics models used in the TWINKLE-M code.

References:

1. NEACRP-L-335(Revision 1), H. Finnemann and A. G. Galati, "NEACRP 3-D LWR Core Transient Benchmark," Final Specifications, October 1991 (January 1992).
2. NEA/NSC/DOC(93)25, H. Finnemann, H. Bauer, A. Galati, and R. Martinelli, "Results of LWR Core Transient Benchmarks," October 1993.
3. Proc. Joint Int. Conf. Mathematical Methods and Supercomputing for Nuclear Applications, Saratoga Springs, N.Y., American Nuclear Society, 1, 302-313 (1997), M. P. Knight and P. Bryce, "Derivation of a refined PANTHER solution to the NEACRP PWR rod-ejection transients."

Figure 3.2-2.1 Radial Power Distribution Comparison between a Typical 3-loop Operating Plant (Unit-A) and TWINKLE-M

Figure 3.2-2.2 Radial Power Distribution Comparison between a Typical 3-loop Operating Plant (Unit-B) and TWINKLE-M

**Figure 3.2-2.3 Average Power Fraction
One Rod Ejection at HZP
Comparison between TWINKLE-M and OECD Reference Solution**

**Figure 3.2-2.4 Doppler Effective Fuel Temperature at Hot Spot
One Rod Ejection at HZP
Comparison between TWINKLE-M and OECD Reference Solution**

**Figure 3.2-2.5 Radial Power Distribution at Maximum Power
One Rod Ejection at HZP
Comparison between TWINKLE-M and OECD Reference Solution**

RAI 3.2-7

Finite difference codes were originally meant to have mesh sizes on the order of a transport mfp. The mesh spacing used in the comparisons to ANC is 11 cm in all directions. In order to understand why the coarse mesh 2x2 results are in agreement with the 4x4 results please address the following: Is the ejected rod worth calculated for each case (Table 3.2.2-1) or is the worth for one case fixed somehow to be in agreement (within 5 pcm) of the other case? What is the maximum fuel enthalpy (or temperature) for the two cases? It is assumed that the maximum hot channel factor is for an entire assembly. Please provide a comparison of additional hot channel factors (at a minimum for the assemblies surrounding the ejected rod). What would be the result if the axial mesh were also changed?

Response

1. Ejected Rod Worth

The ejected control rod worth (in MUAP-07010-P Table 3.2.2-1) is calculated for both the 2x2 mesh case and the 4x4 mesh case without any adjustment. These values are almost identical because the same cross-section data is used and the power distribution for each case is adjusted to agree with the power distribution of the static design code (ANC). Note that for the safety analysis in Chapter 15 of the Design Control Document, the ejected control rod worth is adjusted to the design limit determined by the ANC calculation because this parameter is critical for the rod ejection analysis and therefore should be conservative. The comparison of the nuclear power transient for the two mesh sizes shown in Figure 3.2.2-1 of MUAP-07010-P would be closer to each other if such an adjustment were performed for those cases.

2. Fuel Temperature

The Doppler effective fuel temperature at the hot spot for the 2x2 and 4x4 mesh calculations generally agree with each other, as shown in Figure 3.2-7.1, although the 2x2 mesh solution conservatively results in a slightly higher maximum effective fuel temperature.

3. Hot Channel Factors

The maximum hot channel factor is calculated for a single mesh. Figures 3.2-7.2 through 3.2-7.5 show the time-varying assembly-wise radial power distribution at the axial height of F_Q for the RCCA ejection transient for the 2x2 and 4x4 mesh calculations. This sequence of figures indicates that the power transient in assemblies around the ejected rod is in good agreement between the 2x2 and 4x4 mesh transient calculations.

4. Axial Mesh Sensitivity

A sensitivity study analysis was performed using [ ] in the axial direction in the active core region compared to the [ ] that were utilized in the cases shown in Section 3.2.2 of MUAP-07010-P. For these comparisons, the mesh division in the radial direction is 2x2 mesh per assembly, unless otherwise noted. The results of the axial mesh division sensitivity study are shown in Table 3.2-7.1 and Figure 3.2-7.6 through Figure 3.2-7.8. Figure 3.2-7.6 compares the average axial power distribution for the [ ] calculation, the [ ] calculation, and the ANC calculation for three different core conditions. The comparisons indicate that the axial power distributions of the [ ] calculations are generally closer to the ANC calculations than the [ ] calculations, but that the differences are not significant. Note that the axial power distributions shown in Figure 3.2-7.6 are calculated by TWINKLE-M without any adjustment of the diffusion coefficients in the reflector region.

The comparisons shown in Figure 3.2-7.7 and Figure 3.2-7.8 indicate that the difference

between the [ ] calculations has little impact on the nuclear power transient. Note that the diffusion coefficients in the reflector for both cases are adjusted for the transient calculation so that the [ ] axial power distribution matches the ANC power distribution, as was shown in Figure 3.2.1-5 of MUAP-07010-P. The difference in the nuclear power gradient between the [ ] cases is caused by the difference in the ejected rod worth, which is shown in Table 3.2-7.1.

In summary, the [ ] division in the axial direction is sufficient for the non-LOCA safety analyses of the US-APWR.

Table 3.2-7.1 Calculation Conditions and Results of the RCCA Ejection

	[ ]	[ ]
Initial power (fraction of nominal) Average coolant temperature (°F) RCS pressure (psia)	10^{-9} 557 2250	Same as left
Ejected worth (pcm)	600	594
Delayed neutron fraction (%)	0.44	Same as left
Neutron lifetime (microseconds)	8.0	Same as left
Maximum core power (fraction of nominal)	3.12	2.97
Maximum hot channel factor	27.5	27.2

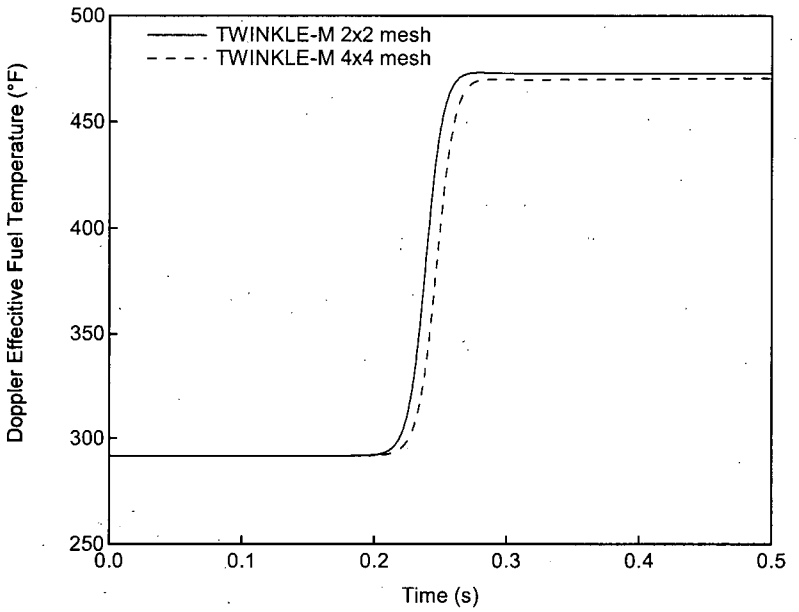


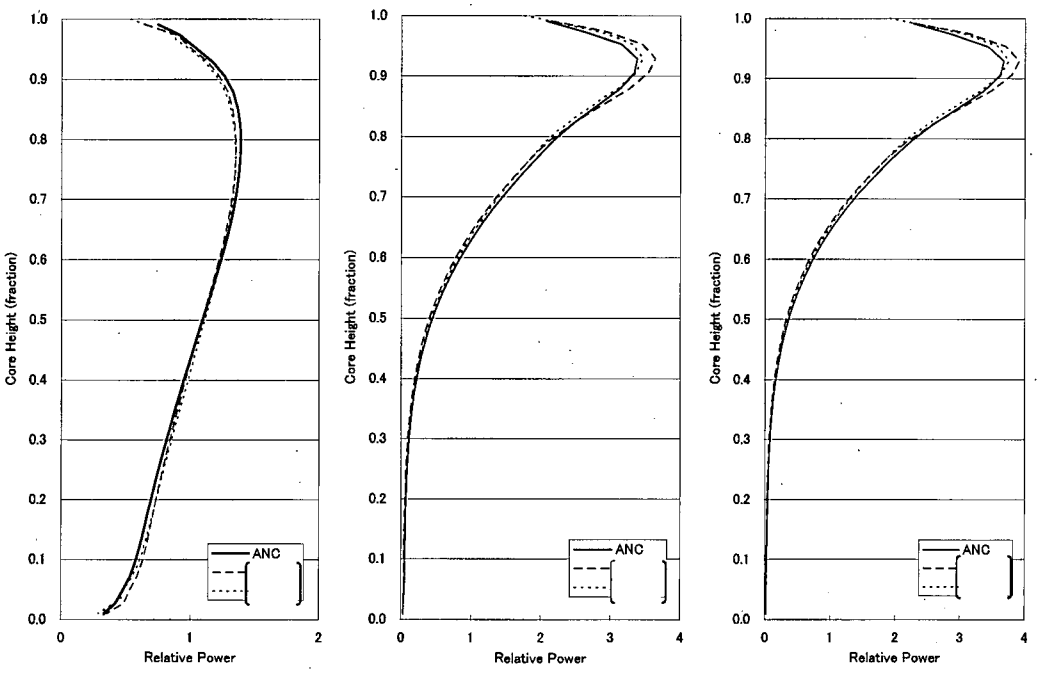
Figure 3.2-7.1 Doppler Effective Fuel Temperature at Hot Spot RCCA Ejection at EOC HZP Comparison between 2x2 and 4x4 Mesh in TWINKLE-M

Figure 3.2-7.2 Assembly-wise Radial Power Distribution at Axial Height of F_Q, RCCA Ejection at 0.0 second, EOC HZP in TWINKLE-M

Figure 3.2-7.3 Assembly-wise Radial Power Distribution at Axial Height of F_Q, RCCA Ejection at 0.1 second, EOC HZP in TWINKLE-M

Figure 3.2-7.4 Assembly-wise Radial Power Distribution at Axial Height of F_Q, RCCA Ejection at 0.2 second, EOC HZP in TWINKLE-M

Figure 3.2-7.5 Assembly-wise Radial Power Distribution at Axial Height of F_Q, RCCA Ejection at 0.3 second, EOC HZP in TWINKLE-M



Case 1
BOC HFP
All RCCAs Out

Case 2
EOC HZP
RCCA at Insertion limit

Case 3
EOC HZP
One RCCA Ejected

Figure 3.2-7.6 Average Axial Power Distribution Comparison between ANC and Axial Mesh Sensitivity Cases (No adjustment of the diffusion coefficient of the reflector region)

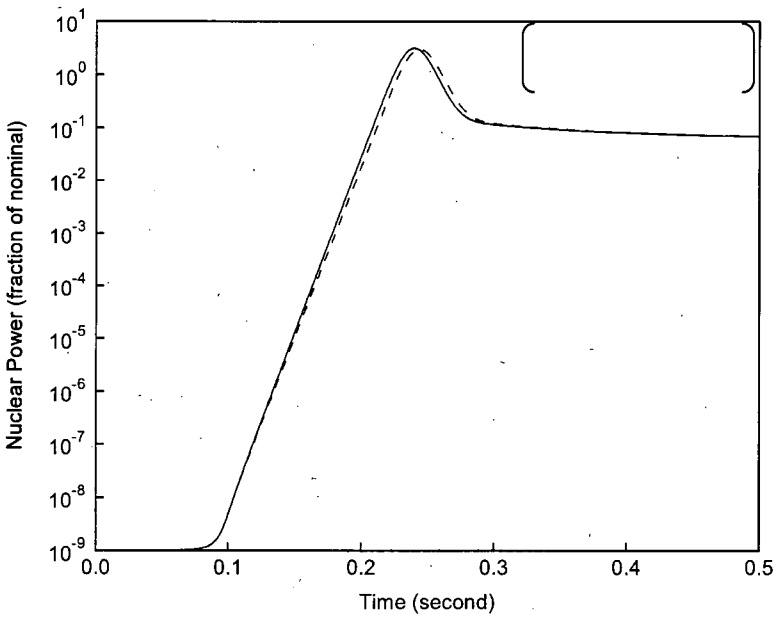


Figure 3.2-7.7 Nuclear Power, RCCA Ejection at EOC HZP Comparison between ANC and Axial Mesh Sensitivity Cases

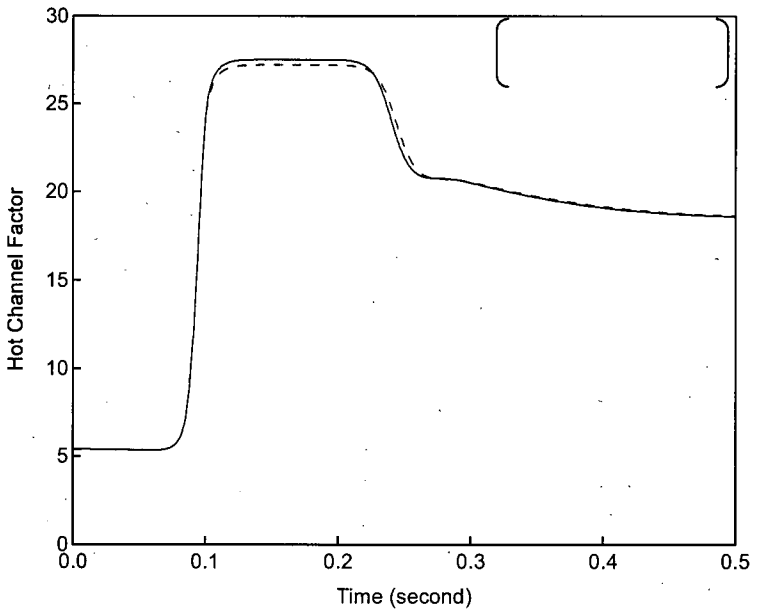


Figure 3.2-7.8 Hot Channel Factor, RCCA Ejection at EOC HZP Comparison between ANC and Axial Mesh Sensitivity Cases

RAI 3.2-10
Please provide discussion of the effect of axial mesh size on the axial power distribution.

Response
See the response to RAI 3.2-7.

RAI 5.5-2

Are the calculated results for the feedwater system pipe break sensitive to the assumed mixing factors, and have any sensitivity calculations been performed? If so, please provide the comparisons.

Response

The extent of reactor plenum inlet mixing has an effect on events characterized by non-uniform loop behavior, in particular, events resulting in the heatup or cooldown of a single reactor coolant loop. Among the events that result in a non-uniform loop heatup, the feedwater system pipe break event is the most extreme. A sensitivity study was performed to confirm the effect of the assumed mixing factors for this event. Two different cases were analyzed: the first case assumes a vessel inlet mixing factor that is 10% less than the base case (Case 1) and the second case assumes a vessel inlet mixing factor that is 10% more than the base case (Case 2). The results of the sensitivity study of the vessel inlet mixing factor for the feedwater system pipe break are shown in Figure 5.5-2.1 through Figure 5.5-2.12. The two cases with modified mixing factors are compared to the base case for various parameters. For these parameters, the effect of the change in the inlet mixing factor is negligibly small. These results demonstrate that the effects of changes to the vessel inlet mixing factor are small for the feedwater system pipe break event.

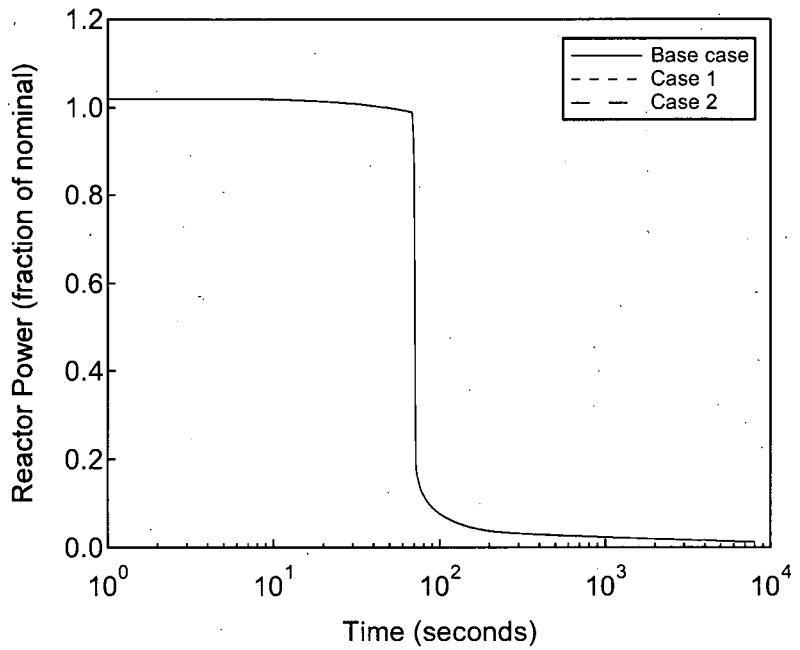


Figure 5.5-2.1 Reactor Power versus Time Feedwater System Pipe Break

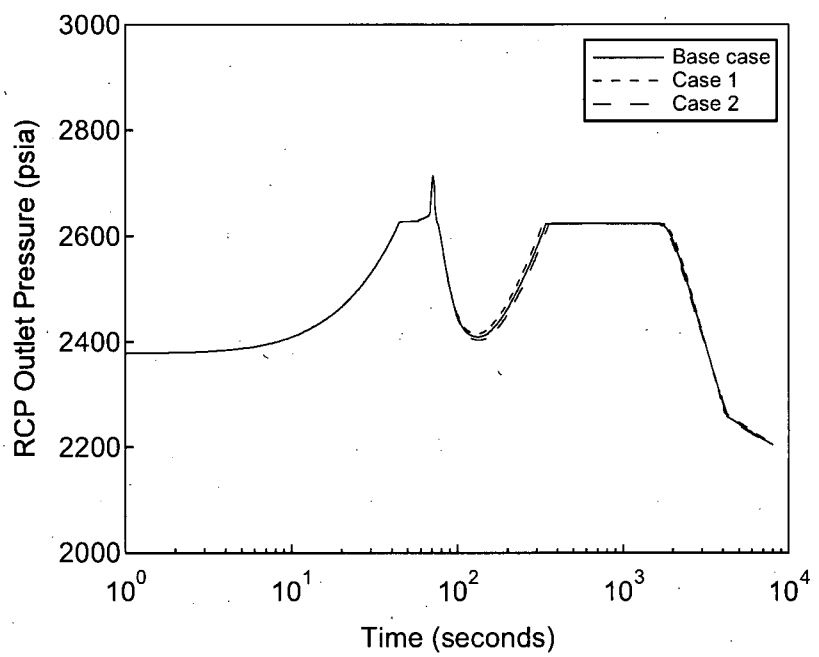


Figure 5.5-2.2 RCP Outlet Pressure versus Time Feedwater System Pipe Break

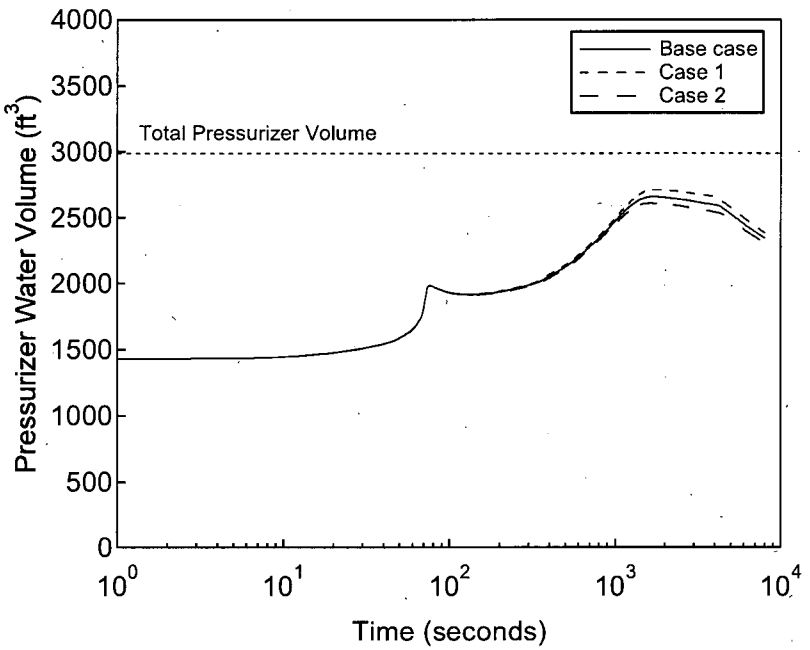


Figure 5.5-2.3 Pressurizer Water Volume versus Time Feedwater System Pipe Break

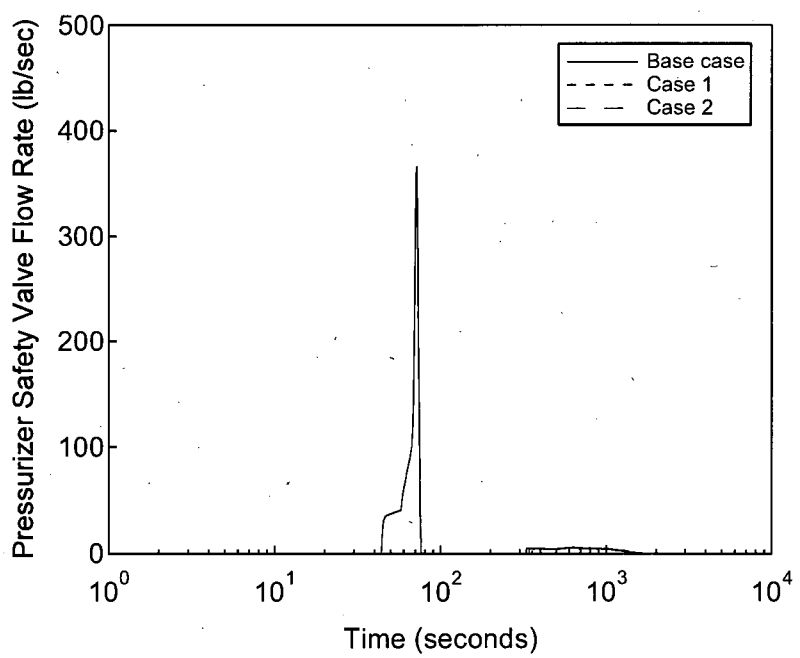


Figure 5.5-2.4 Pressurizer Safety Valve Flow Rate versus Time Feedwater System Pipe Break

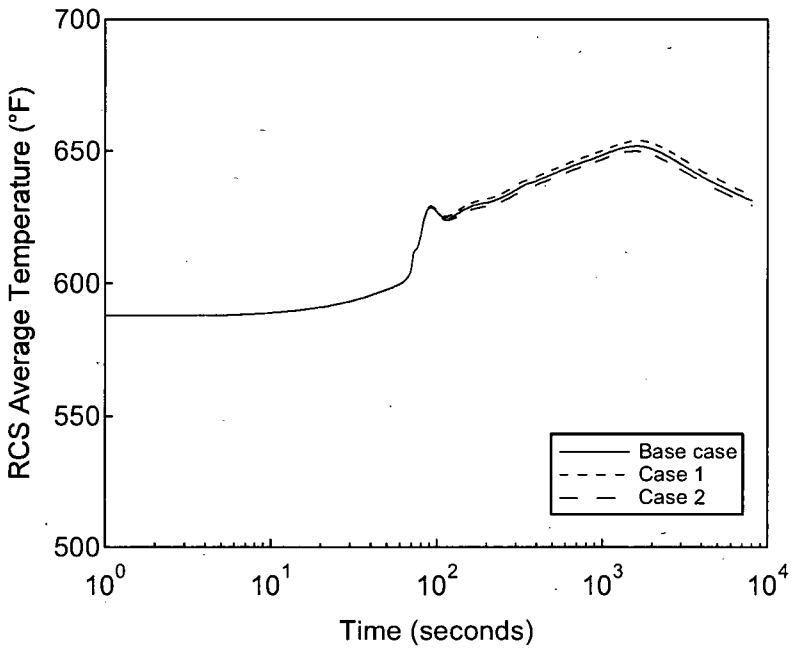


Figure 5.5-2.5 RCS Average Temperature versus Time Feedwater System Pipe Break

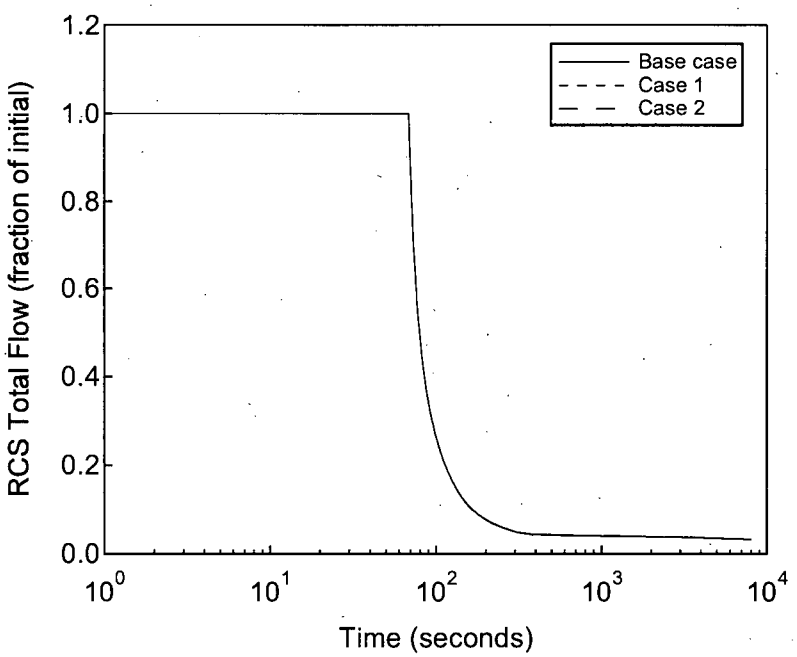


Figure 5.5-2.6 RCS Total Flow versus Time Feedwater System Pipe Break

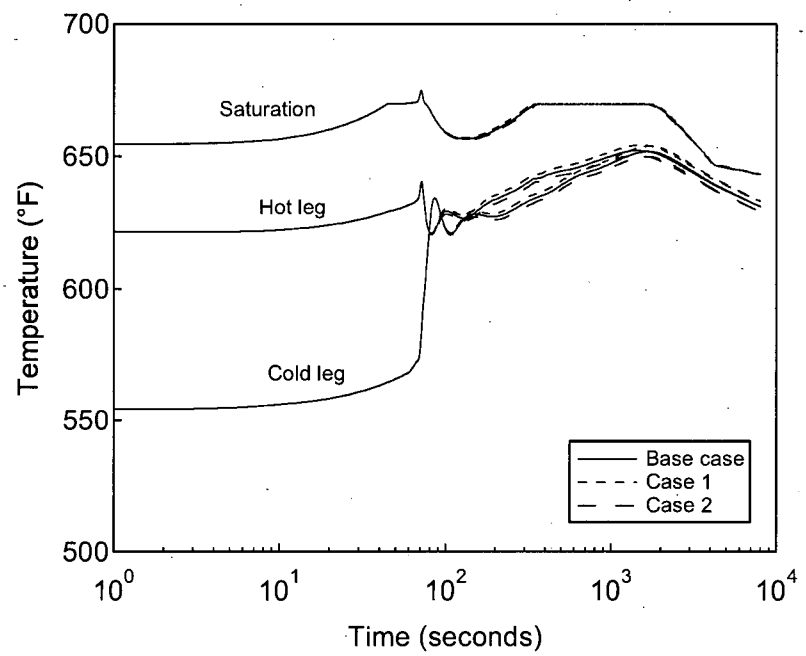


Figure 5.5-2.7 Temperature of Faulted Loop versus Time Feedwater System Pipe Break

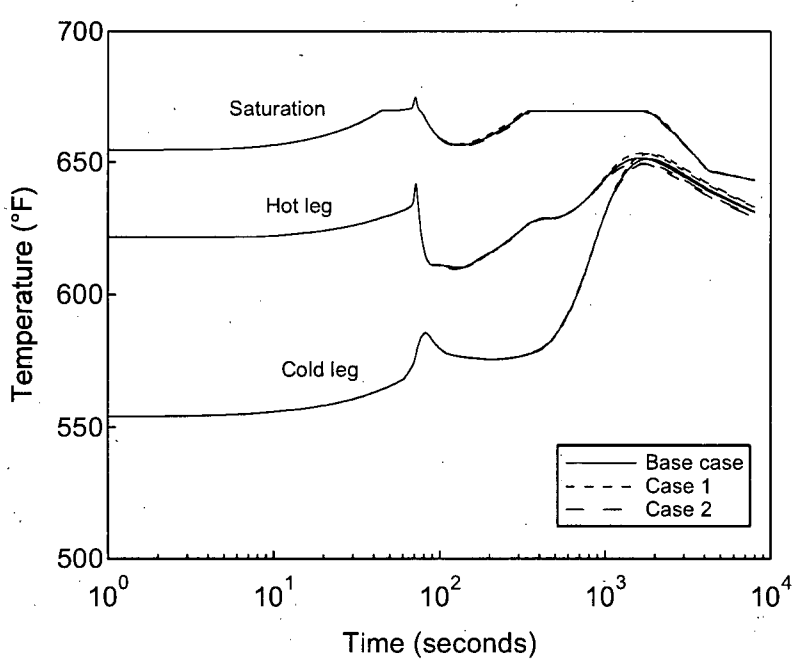


Figure 5.5-2.8 Temperature of Intact Loop without EFW versus Time Feedwater System Pipe Break

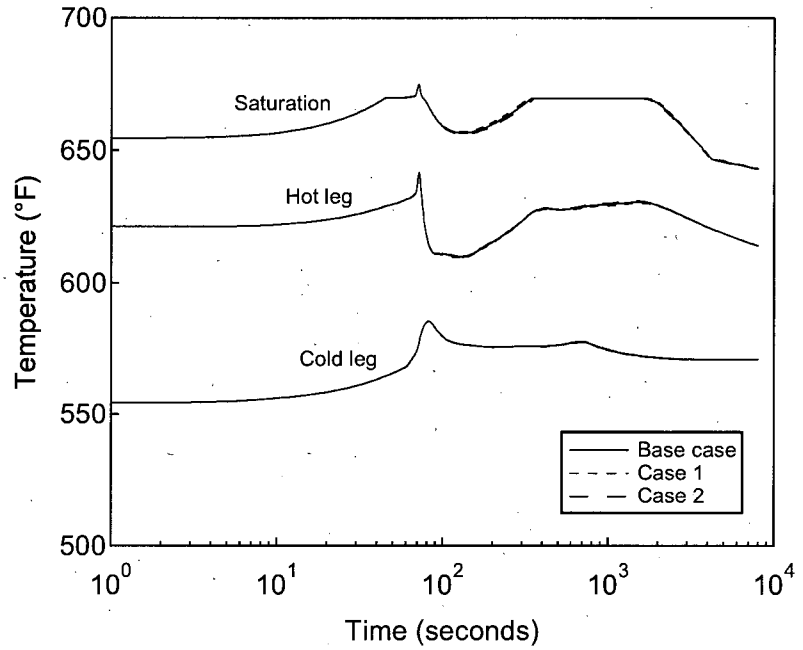


Figure 5.5-2.9 Temperature of Intact Loop with EFW versus Time Feedwater System Pipe Break

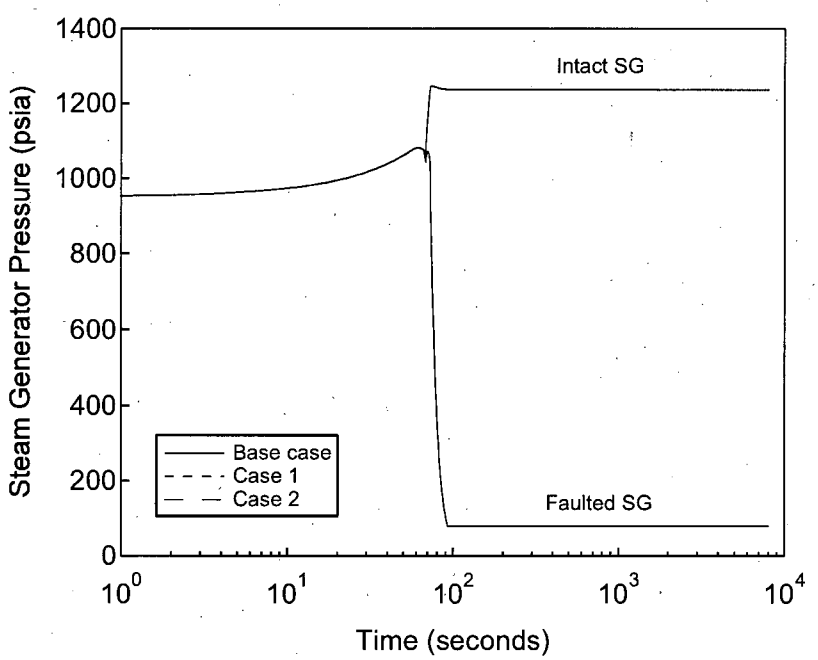


Figure 5.5-2.10 Steam Generator Pressure versus Time Feedwater System Pipe Break

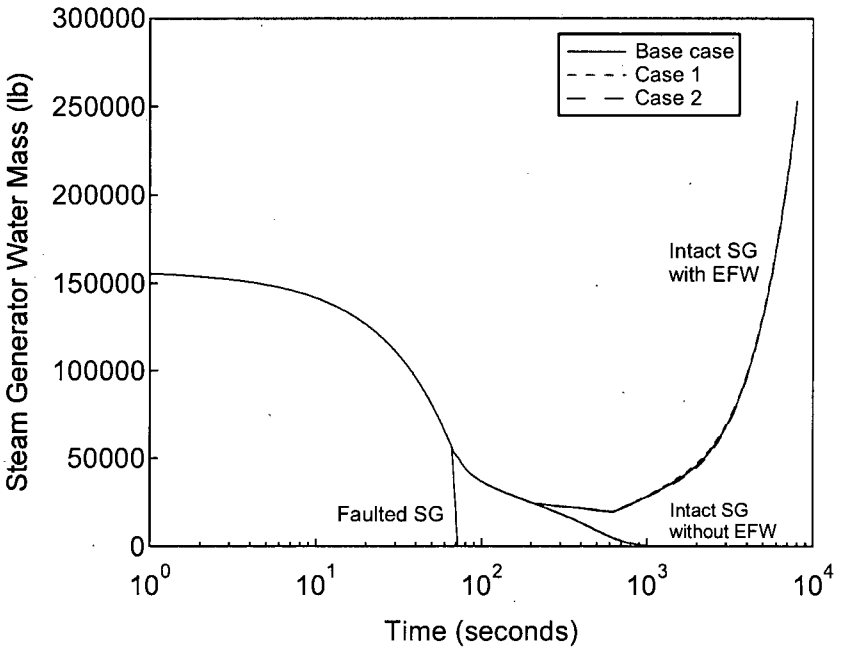


Figure 5.5-2.11 Steam Generator Water Mass versus Time Feedwater System Pipe Break

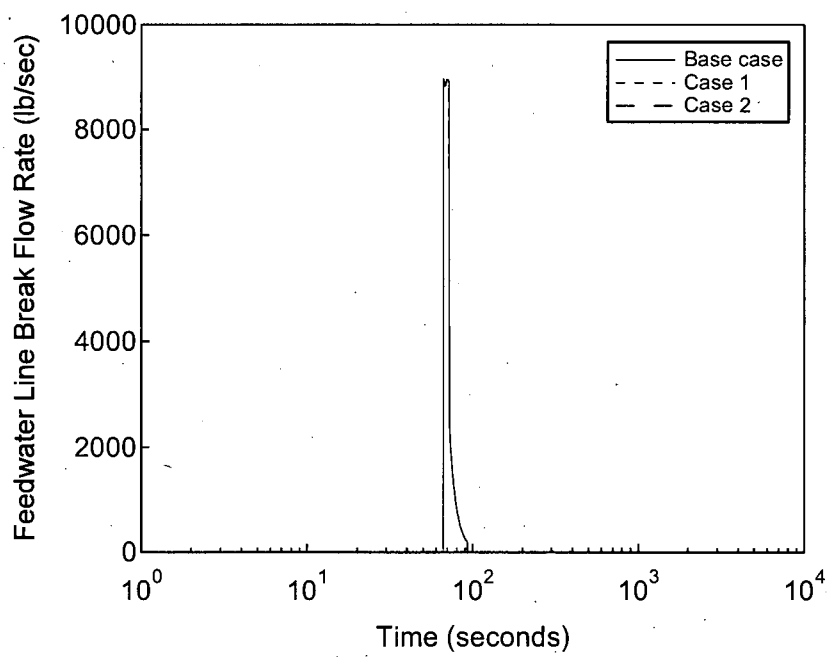


Figure 5.5-2.12 Feedwater Line Break Flow Rate versus Time Feedwater System Pipe Break