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TOKYO, JAPAN

August 25, 2011

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

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-11273

Subject: MHI's Response to US-APWR DCD RAI No. 788-5883 Revision 3 (SRP 15.1.5)

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Response to US-APWR DCD RAI No. 788-5883 Revision 3 (SRP 15.1.5)". The enclosed material provides MHI's response to the NRC's "Request for Additional Information (RAI) 788-5883 Revision 3," dated July 26, 2011.

As indicated in the enclosed materials, Enclosure 2 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 Response to US-APWR DCD RAI No. 788-5883 Revision 3 (SRP 15.1.5) (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 788-5883 Revision 3 (SRP 15.1.5) (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-11273

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 Response to US-APWR DCD RAI No. 788-5883 Revision 3 (SRP 15.1.5)", dated August 24, 2011, 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 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 25th day of August, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11273
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 788-5883 Revision 3
(SRP 15.1.5)

August 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/25/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 788-5883 REVISION 3**
SRP SECTION:	**15.01.05 - STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)**
APPLICATION SECTION:	**15.1.5**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.01.05-6
Three different cases are analyzed for the main steam line break event in DCD 15.1.5. While the first two were individual runs, Case C represents a series of runs to address variations in break size and power level. From Figure 15.1.5-26, it appears the DNBR limiting event for Case C is an approximately 0.4 ft^2 break initiated at 100% power. Because this Case C data point appears to be more limiting than either Case A or Case B, include more details of this particular Case C run in the DCD to demonstrate the acceptance criteria of SRP 15.1.5 are met. Include a sequence of events that identifies mitigating system actuation and time related variations of key parameters (as was done for Case A and Case B).

ANSWER:

DCD Section 15.1.5 is revised to provide additional figures and a time sequence of events for the limiting point associated with Case C.

Impact on DCD

DCD Section 15.1.5 is revised to expand the time sequence of events table and include several additional figures for Case C. Additionally, DCD Subsection 15.1.5.3.3 "Results" Item (3) "Case C – Spectrum of Breaks from Power with Offsite Power" is revised to include a discussion of the newly added Case C results. The mark-up of the affected portions of DCD Section 15.1.5 is provided in Attachment 1.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/25/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 788-5883 REVISION 3**
SRP SECTION:	**15.01.05 - STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)**
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QUESTION NO.: 15.01.05-7

Item 7 of the SRP 15.1.5 Acceptance Criteria for initial plant conditions states that while the minimum core flow results in the minimum DNBR margin, maximum initial core flow may cause increased reactor coolant system cooldown and be the more limiting assumption. In order to find that the input parameters were suitably conservative, please provide justification for selecting a minimum value for the initial core flow.

ANSWER:

The reviewer is correct that maximum initial core flow may cause an increased RCS cooldown. This increased cooldown could result in an increase in reactivity that could ultimately affect core power and DNBR. On the other hand, minimum DNBR is also directly dependent on core flow. As stated by the reviewer, the minimum core flow typically results in a more severe DNBR. Since these two effects are in opposite directions, MHI has performed a sensitivity study to determine the impact of assuming maximum core flow. The results of the sensitivity study are shown in Figures 15.01.05-7.1 through 15.01.05-7.8. The core average heat flux in Figure 15.01.05-7.3 shows that the sensitivity Case 1 is slightly higher than the DCD case. However, the DCD case already considers conservative assumptions regarding the Doppler temperature coefficient as described in the MHI response to Question No.15.0.0-30, RAI No. 786-5881 Revision 3. Therefore, MHI also included an additional sensitivity study case (Case 2), which assumes that the Doppler temperature coefficient is the minimum value, in the same figures. These results demonstrate that the core average heat flux of Case 2 is smaller than the DCD case. Therefore DNBR is not more severe than the DCD case. Additionally, Figures 15.01.05-7.1 and 15.01.05.7-2 show that the increase in core flow does not significantly affect core reactivity and the return to power. These results confirm MHI's position that the combination of parameters assumed in the DCD is suitably conservative to meet the intent of the SRP acceptance criteria. Therefore, MHI continues to assume the minimum core flow for the DCD Section 15.1.5 analysis.

Figure 15.01.05-7.1 Core Reactivity versus Time Initial Core Flow Sensitivity Study Steam System Piping Failure Event

Figure 15.01.05-7.2 Reactor Power versus Time Initial Core Flow Sensitivity Study Steam System Piping Failure Event

Figure 15.01.05-7.3 Core Heat Flux versus Time
Initial Core Flow Sensitivity Study
Steam System Piping Failure Event

Figure 15.01.05-7.4 RCS Pressure versus Time
Initial Core Flow Sensitivity Study
Steam System Piping Failure Event

Figure 15.01.05-7.5 Pressurizer Water Volume versus Time
Initial Core Flow Sensitivity Study
Steam System Piping Failure Event

Figure 15.01.05-7.6 Core Average Temperature versus Time
Initial Core Flow Sensitivity Study
Steam System Piping Failure Event

Figure 15.01.05-7.7 Steam Generator Pressure versus Time
Initial Core Flow Sensitivity Study
Steam System Piping Failure Event

Figure 15.01.05-7.8 Steam Flow versus Time
Initial Core Flow Sensitivity Study
Steam System Piping Failure Event

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

This completes MHI's response to the NRC's question.

The lower RCS flow generally causes the reactor coolant system cooldown for Case B to be slower and the associated reactivity transient is less severe, resulting in a later return to criticality and lower peak power. Most of the parameters behave in a similar manner as in Case A. As with Case A, long term decay heat can be removed by controlled steam relief from the intact steam generators and later, by the residual heat removal system.

The minimum DNBR in Case B is less limiting than the minimum DNBR in Case A because of the reduced core cooling and lower heat flux. However, the minimum DNBR remains well above the 95/95 limit for the W-3 correlation and the fuel cladding temperature would not increase significantly during this transient. This accident does not challenge the design pressures for either the reactor coolant pressure boundary or the main steam system.

(3) Case C – Spectrum of Breaks from Power with Offsite Power

A spectrum of break sizes from at-power conditions was analyzed to demonstrate that the period of the transient before post-trip shutdown does not result in DNBRs below the 95/95 limit.

At rated power, the increased reactivity causes an increase in core power. For small breaks, the response is similar to the steam flow increase event in that the power may not reach a reactor trip setpoint. For intermediate size breaks, the power increase results in an over power ΔT reactor trip. For large breaks, up to and including the double-ended rupture of a steam pipe, the reactor is tripped on low steam line pressure, which also causes ESF actuation (including safety injection, main feedwater isolation, and emergency feedwater isolation).

Figure 15.1.5-26 provides a summary of the key results of this analysis of the at-power breaks in the form of plots of initial steam flow, peak power, and minimum DNBR as a function of break area (per steam generator). A line is included on each plot for 100% and 75% initial power levels. As expected, the initial break flow is only a function of initial steam generator pressure and break area, so initial break flow decreases with decreasing break area, and some break areas are small enough that the feedwater control system may be able to keep up with the steam flow, resulting in a new steady state power below the overpower reactor trips. The peak power and minimum DNBR curves from 100% power, however, show three distinct regions: no trip for small breaks, over power ΔT trips for intermediate break sizes, and low main steam line pressure trips (and main steam line isolation) for the larger breaks. The low main steam line pressure signal occurs so rapidly for the larger breaks that the reactivity feedback has not caused power to increase before the trip and steam line isolation occur (the peak power and minimum DNBR are approximately equal to the initial full power value).

Figure 15.1.5-26 also shows that breaks initiating from lower power levels are less limiting than for full power.

As shown in Figure 15.1.5-26, the limiting Case C DNB event is an intermediate break of approximately 0.4 ft^2 initiated at 100% power. Figures 15.1.5-27 through 15.1.5-32 provide plots of system parameters versus time for the core response analysis for this limiting at power break. The corresponding sequence of events is provided in Table 15.1.5-1. The increase in reactivity results in the power increase shown in Figure 15.1.5- DCD_15.01.05-6

27. The reactor trips on over power ΔT, effectively limiting the decrease in DNB. Additionally, Figure 15.1.5-29 shows that the RCS pressure decreases from its initial value such that the maximum reactor coolant system pressure remains below 110% of design. DCD_15.01.05-6

The Case C analysis shows that the minimum DNBR remains above the 95/95 limit for the WRB-2 correlation for all break sizes. Thus the fuel cladding temperature would not increase significantly during this transient. Additionally, ~~T~~these cases do not challenge the design limits for the reactor coolant pressure boundary or the main steam system.

For Cases A and B, the normalized core average heat flux transient is virtually identical to the normalized maximum heat flux, and is considered representative of both parameters for this event. A plot for reactor vessel inlet temperature (showing all loops) is provided in place of inlet coolant temperature to illustrate the non-uniform inlet temperatures prior to mixing in the reactor vessel inlet. Because there is significant core subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions, peak cladding temperature, and fuel centerline temperature are not provided; these are not key parameters for this event. A plot of steam generator pressure is provided in place of steam line pressure to show the non-uniform and independent response of the steam generators during this event. Steam generator water mass is presented instead of steam generator water volume. Additionally, steam line break flow rate is labeled as steam flow rate for these cases. Pressurizer safety valve flow is not reported for this event because RCS pressure remains below the pressurizer safety valve set pressure and there are no releases from the RCS inside containment. Containment parameters are not presented for these core response analyses. Containment vessel response to steam system piping failures inside the containment vessel is described and analyzed in Section 6.2.

For the Case C at-power breaks, plots of the core and RCS parameter transient response up to the time of reactor trip are ~~similar to those for the RCCA Bank withdrawal at Power transients presented in Section 15.4.2~~provided for the limiting break size.~~, and the DNBRs are calculated using the same approach. Therefore, only a plot summarizing the results (peak power, initial break flow, and minimum DNBR versus break area at two power levels) is provided.~~ DCD_15.01.05-6

15.1.5.4 Barrier Performance

Information for the bounding transient documented in Section 15.1.5.3 indicates the maximum reactor coolant system and main steam pressures remain well below 110% of their design pressures.

In response to GDC 31, this event (including operation of the ECCS under low-temperature conditions) has been considered in the design of the reactor coolant pressure boundary to assure that the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture has been minimized. Fracture toughness of the reactor coolant pressure boundary and reactor vessel is described in Sections 5.2.3 and 5.3.1.

As discussed in Section 15.0.0.9, the integrity of the reactor coolant pumps is maintained such that loss of ac power and containment isolation will not result in pump seal damage.

Table 15.1.5-1
Time Sequence of Events for the Steam System Piping Failure

Event Description	Case A Time [sec]	Case B Time [sec]
Steam pipe rupture occurs	0.0	0.0
Low steamline pressure analytical limit reached	1.5	1.5
RCP coastdown begins	N/A	4.5
MSIVs closed	10.0	10.0
Automatic isolation of EFW to faulted SG (Case B)	N/A	50.2
Safety injection pumps start	21.5	121.5
Boron reaches core	44.9	141.4
Automatic isolation of EFW to faulted SG (Case A)	51.7	N/A
Peak core heat flux occurs	89.8	152.8
Faulted SG water mass depleted	330	1420

DCD_15.01.05-6

Event Description	Case C[*1] Time [sec]
Steam pipe rupture occurs	0.0
Over power ΔT analytical limit reached	8.0
Reactor trip initiated (rod motion begins)	14.0
Minimum DNBR occurs	14.7

*1 Limiting case at power break

DCD_15.01.05-6

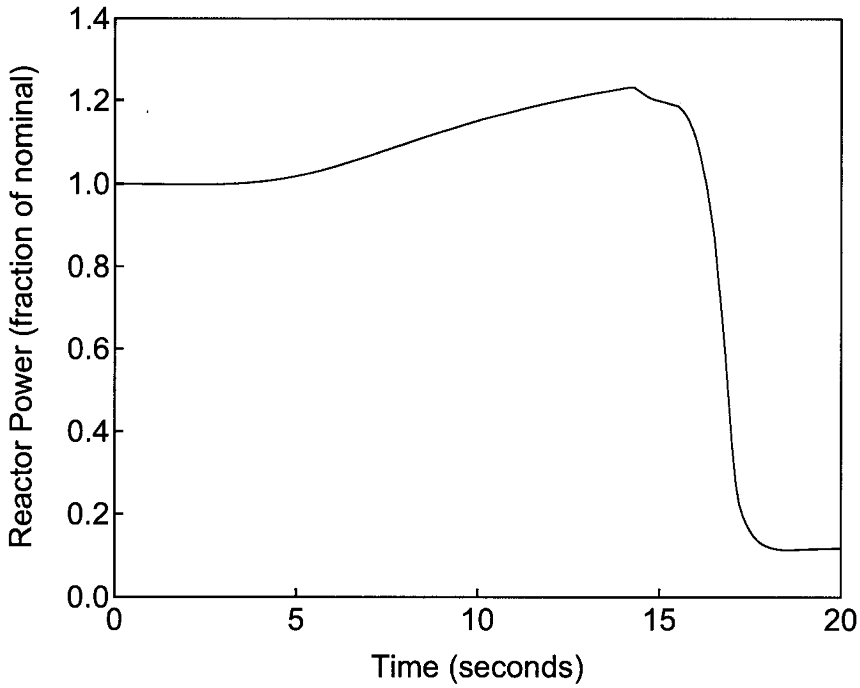

Figure 15.1.5-27 <u>**Reactor Power versus Time**</u>

<u>**Steam System Piping Failure**</u>
<u>**- Case C: Limiting Case for Spectrum of Breaks at 100% Power**</u>

DCD_15.01.05-6

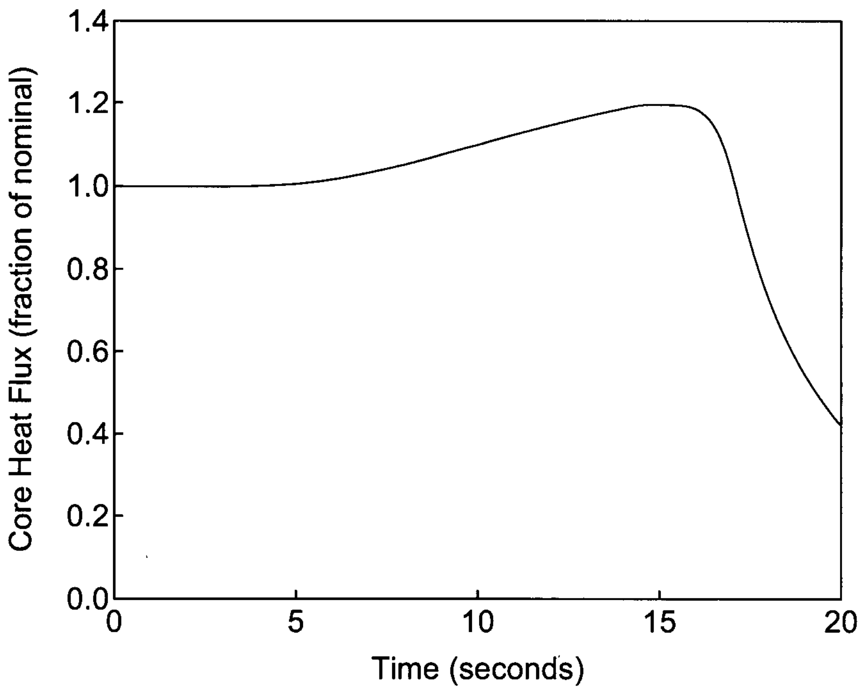

Figure 15.1.5-28 **Core Heat Flux versus Time**

Steam System Piping Failure
- Case C: Limiting Case for Spectrum of Breaks at 100% Power

DCD_15.01.05-6

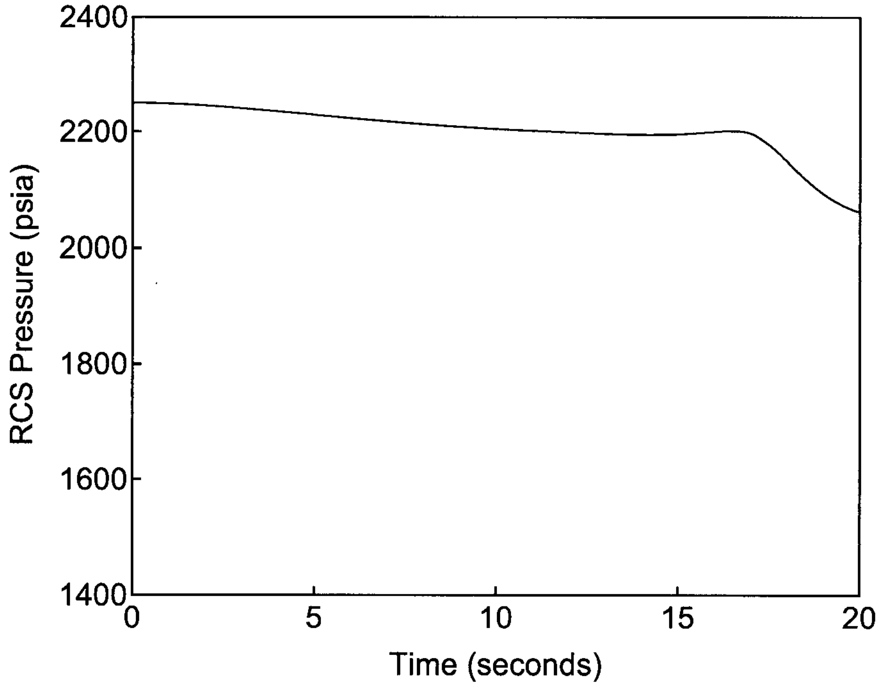

Figure 15.1.5-29 RCS Pressure versus Time

Steam System Piping Failure
- Case C: Limiting Case for Spectrum of Breaks at 100% Power

DCD_15.01.05-6

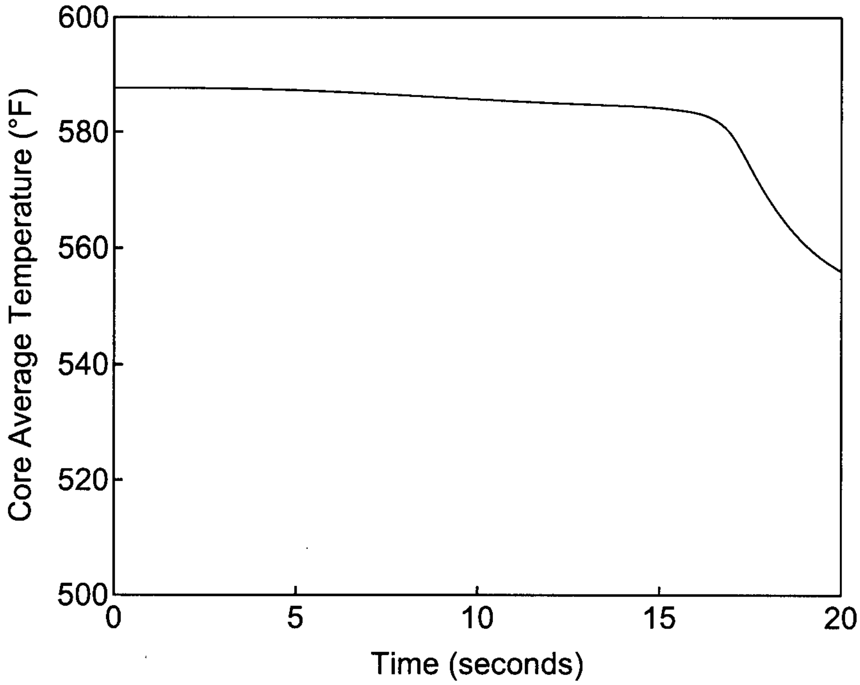

Figure 15.1.5-30 Core Average Temperature versus Time

Steam System Piping Failure
- Case C: Limiting Case for Spectrum of Breaks at 100% Power

DCD_15.01.05-6

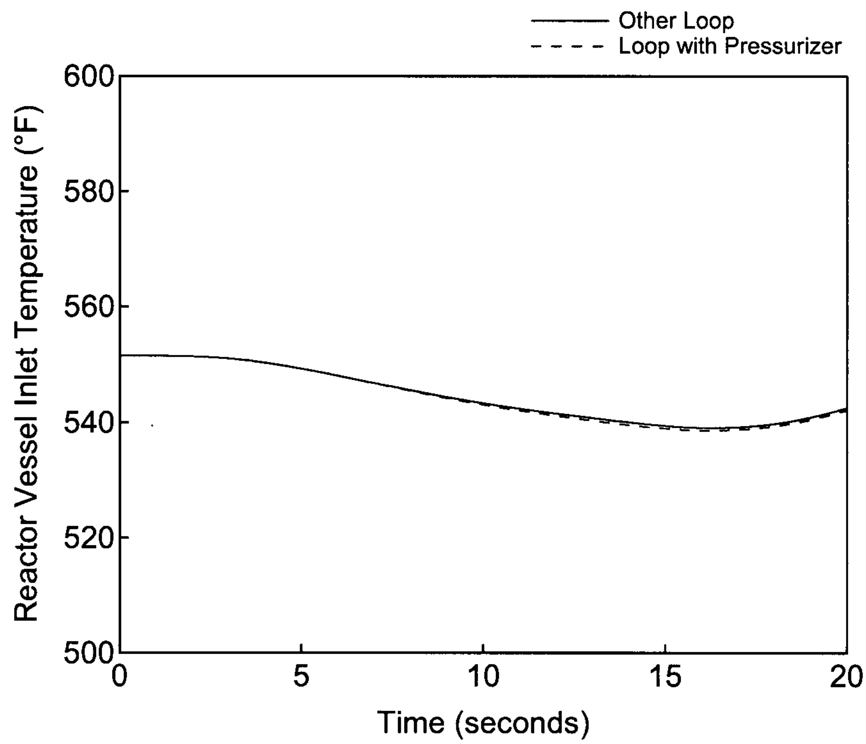

Figure 15.1.5-31 **Reactor Vessel Inlet Temperature versus Time**

Steam System Piping Failure
- Case C: Limiting Case for Spectrum of Breaks at 100% Power

DCD_15.01.05-6

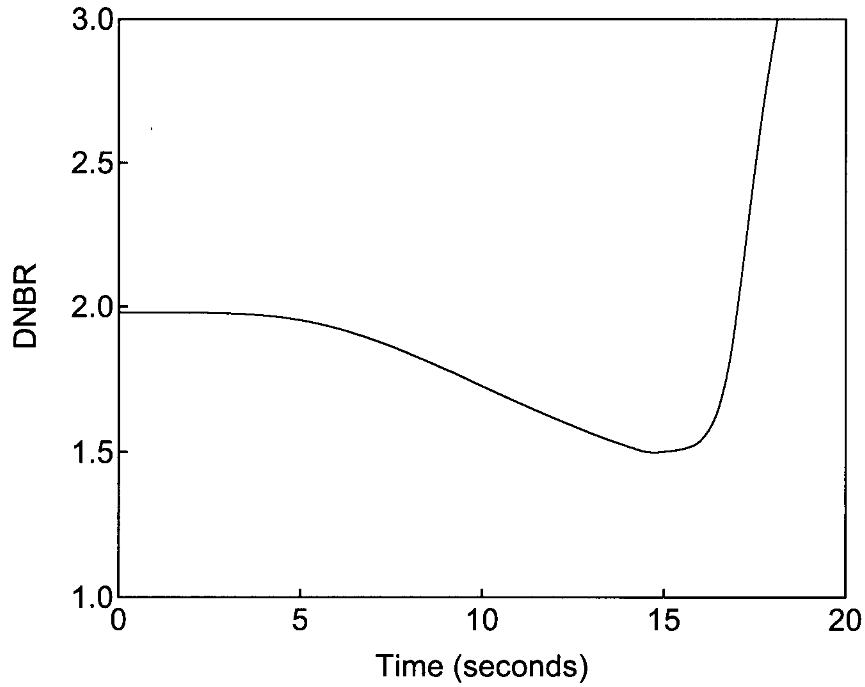

Figure 15.1.5-32 DNBR versus Time

Steam System Piping Failure
- Case C: Limiting Case for Spectrum of Breaks at 100% Power