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July 24, 2008

AEP-NRC-2008-11
10 CFR 50.46

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
Completion of Commitment Regarding Small Break Loss-of-Coolant Accident Analysis 8.75-inch Case (TAC No. MD5297)

REFERENCES:
1. Letter from Mark A. Peifer, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Small Break Loss-of-Coolant Accident Evaluation Model Reanalysis," AEP:NRC:7046, dated March 29, 2007 (ML071000431).
2. Letter from Peter S. Tam, NRC, to Mano K. Nazar, I&M, "D. C. Cook Nuclear Plant Unit 1 (DCCNP-1) – Request for Additional Information Regarding Reanalysis of Small-Break Loss-of-Coolant Accident (TAC MD5297)," dated August 10, 2007 (ML072050570).
3. Letter from Mark A. Peifer, I&M, to U. S. NRC Document Control Desk, "Response to Request for Additional Information Regarding the Reanalysis of Unit 1 Small Break Loss-of-Coolant Accident," AEP:NRC:8046, dated February 29, 2008 (ML080740053).

Dear Sir or Madam:

By Reference 1, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, provided the Nuclear Regulatory Commission (NRC) the reanalyzed small break loss-of-coolant accident (SBLOCA) to meet commitments made in accordance with 10 CFR 50.46(a)(3)(ii). The reanalysis used the NRC-approved Westinghouse NOTRUMP SBLOCA Emergency Core Cooling System (ECCS) Evaluation Model methodology.

By Reference 2, the NRC transmitted a request for additional information (RAI) on the CNP Unit 1 SBLOCA reanalysis. By Reference 3, I&M provided a response to the RAI and described erroneously high ECCS recirculation alignment flow rate assumptions which yielded a non-conservative reactor coolant system inventory recovery rate for the 8.75-inch break case following the five-minute Residual Heat Removal (RHR) flow interruption. By Reference 3, I&M committed to

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provide updated information on the 8.75-inch case by June 30, 2008. In a telephone conversation on June 27, 2008, the NRC Project Manager concurred that the due date could be extended to July 25, 2008. The enclosure to this letter provides a new Unit 1 SBLOCA analysis report, replacing the report transmitted by Reference 1. Revision bars in the left margin indicate the changed portions of the report.

While correcting the Reference 1 analysis due to the erroneously high ECCS recirculation alignment flow rates used in the 8.75-inch break case, it was also discovered that the Safety Injection pump performance curve was modeled with a polynomial curve fit equation that did not adequately match the desired performance curve data. Also, hydraulic resistance adjustments to the RHR system were determined to be needed to achieve conservative minimum RHR flow rates relative to an independent hydraulic model that had been correlated to plant surveillance testing data. These corrections and adjustments resulted in changes to the ECCS flow rates used in the enclosed Unit 1 SBLOCA reanalysis. Subsequent assessment determined that only non-limiting break scenarios were affected. The 3.25-inch break case continues to remain limiting with respect to peak cladding temperature and maximum local oxidation. Additional changes were made to Table 1 of the enclosure to clarify the calorimetric uncertainty and the Reactor Coolant System flow rate.

This letter contains no new regulatory commitments. Should you have any questions, please contact Mr. John A. Zwolinski, Regulatory Affairs Manager, at (269) 466-2478.

Sincerely,

Joseph N. Jensen
Vice President, Site Support Services

KAS/rdw

Enclosure: Donald C. Cook Unit 1 Small Break LOCA Analysis Report (60 pages)

c: T. A. Beltz, NRC Washington, DC
J. L. Caldwell, NRC Region III
K. D. Curry, Ft. Wayne AEP
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MDEQ – WHMD/RPS
NRC Resident Inspector

Enclosure to AEP-NRC-2008-11

Donald C. Cook Unit 1 Small Break LOCA Analysis Report

D. C. COOK UNIT 1 SMALL BREAK LOCA ANALYSIS REPORT

1 INTRODUCTION

This section contains information regarding the analysis performed for Donald C. Cook Nuclear Plant Unit 1 (D. C. Cook Unit 1) to analyze the high head safety injection cross-tie isolated case (i.e., one or both cross-tie valves closed) for a postulated Small Break Loss-of-Coolant Accident (SBLOCA). The purpose of analyzing the SBLOCA is to demonstrate conformance with the 10 CFR 50.46 (Reference 1) requirements for the conditions associated with the high head safety injection cross-tie isolated. Important input assumptions, as well as analytical models and analysis methodology for the SBLOCA are contained in subsequent sections. Analysis results are provided in the form of tables and figures, as well as a more detailed description of the limiting transient. The analysis has shown that no design or regulatory limit related to the SBLOCA transient would be exceeded due to the high head safety injection cross-tie isolated and plant operation with the associated plant parameters.

2 INPUT PARAMETERS AND ASSUMPTIONS

The important plant conditions and features for D. C. Cook Unit 1 that are supported by this analysis are listed in Table 1. Additional consideration for several parameters identified in Table 1 are discussed below.

Figure 1 depicts the hot rod axial power shape modeled in the Small Break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core (the axial offset is +20%). Such a distribution is limiting for Small Break LOCA since it minimizes coolant swell while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The chosen power shape has been conservatively scaled to a standard 2-line segment K(Z) envelope based on the peaking factors shown in Table 1.

Figures 2 and 3 provide the ECCS pumped injection flow versus pressure curves modeled in the Small Break LOCA analysis during the injection phase. Figure 2 shows the flows from one charging (CHG) pump, one high head safety injection (HHSI) pump and one residual heat removal (RHR) pump, where the broken (or faulted) loop injects to RCS pressure. Figure 3 shows flows from one CHG pump, one HHSI pump and one RHR pump, where the faulted loop CHG flow injects to RCS pressure and the faulted loop HHSI/RHR flow spills into containment because the break is postulated along the accumulator line. Note that hereafter, pumped injection subsystems of the ECCS (CHG, HHSI and RHR) are referred to collectively as safety injection (SI).

The analysis utilizes an adjusted nominal vessel average temperature (T_{avg}) of 577.4°F (with ± 4°F uncertainty specified by NOTRUMP-EM) to support the D. C. Cook Unit 1 specific T_{avg} value of 575.4°F with +5.1°F uncertainty. The analysis supports operation for a nominal full-power T_{avg} range of 553.7°F to 575.4°F with +5.1°F/-4.1°F uncertainty. Additionally, the analysis utilizes a nominal pressurizer pressure of 2250 psia (plus +67 psi uncertainty) and supports operation at nominal pressurizer pressures of 2100 psia and 2250 psia with ±67 psi uncertainty.

3 DESCRIPTION OF ANALYSES AND EVALUATIONS

Analytical Model

The requirements for an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50. For LOCAs due to Small Breaks, less than 1 square foot in area, the Westinghouse NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 2, 3, and 4) is used. The Westinghouse NOTRUMP Small Break LOCA ECCS Evaluation Model (NOTRUMP-EM) was developed to determine the RCS response to design basis Small Break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 5).

The NOTRUMP-EM consists of the NOTRUMP and LOCTA-IV computer codes. The NOTRUMP code is employed to calculate the transient depressurization of the Reactor Coolant System (RCS), as well as to describe the mass and energy release of the fluid flow through the break. Among the features of the NOTRUMP code are: calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, regime-dependent drift flux calculations in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. These features provide NOTRUMP with the capability to accurately calculate the mass and energy distribution throughout the RCS during the course of a SBLOCA.

The RCS model is nodalized into volumes interconnected by flow paths. The broken loop and each of the three intact loops are modeled explicitly, primarily to model the asymmetric safety injection flows that result from closure of one or both valves in the high head safety injection cross-tie. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multi-node capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a calculation of the behavior of the loop seal during a SBLOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Fuel cladding thermal analyses are performed with SBLOCTA, a SBLOCA version of the LOCTA-IV code (Reference 3), using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. The SBLOCTA code models the hot rod and the average hot assembly rod, assuming a conservative power distribution that is skewed to the top of the core. Figure 4 illustrates the code interface for the Small Break Model.

Analysis

The SBLOCA analysis for D. C. Cook Unit 1 considered a spectrum of eleven different break cases, including 1.5-, 2-, 2.5-, 2.75-, 3-, 3.25-, 3.5-, 3.75-, 4-, 6- and 8.75-inch breaks. The 3.25-inch break was found to be limiting for peak cladding temperature (PCT) and local oxidation. The 1.5-inch case showed no core uncovery and therefore PCT information was not calculated.

The most limiting single active failure used for a Small Break LOCA is that of an emergency power train failure which results in the loss of one complete train of ECCS components. In addition, a Loss-of-Offsite Power (LOOP) is postulated to occur coincident with reactor trip. This means that with the assumed loss of emergency power there is a loss of one CHG pump, one HHSI pump and one RHR pump. The Small Break LOCA analysis performed for D. C. Cook Unit 1 models the ECCS injection phase flow as being delivered to both the intact and broken loops at the RCS backpressure for breaks smaller than the accumulator line inner diameter (1.5-inch through 6-inch breaks). For breaks equal to or greater than the accumulator line inner diameter (8.75-inch breaks), the broken loop flow spills to containment pressure. Note that for the 8.75-inch breaks, the broken loop CHG flow is assumed to inject to the cold leg at the RCS backpressure since it is not affected by the accumulator line break (CHG injects via a separate connection to the cold leg). The ECCS pumped injection flow rates for these scenarios are illustrated in Figures 2 and 3. The LOOP and the failure of an emergency diesel generator to start as the limiting single failure for SBLOCA is part of the NRC approved methodology. The single failure assumption is extremely limiting due to the fact that one train of safety injection (SI), one motor driven auxiliary feedwater (AFW) pump, and power to the reactor coolant pumps (RCPs) are all modeled to be lost. Any other active single failure would not result in a more limiting scenario since increased SI flow would improve the overall transient results.

Prior to break initiation, the plant is in a full power (100.34%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions used in the analysis are given in Table 1. Subsequent to the break opening, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS fluid. The heat transfer between the RCS and the secondary system may be in either direction and is a function of the relative temperatures of the primary and secondary conditions. In the case of continuous heat addition to the secondary side during a period of quasi-equilibrium, an increase in the secondary system pressure (due to the assumed turbine trip following the reactor trip discussed below) results in steam relief via the steam generator safety valves.

When a Small Break LOCA occurs, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure reactor trip setpoint, conservatively modeled as 1860 psia, is reached. LOOP is postulated to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1715 psia, is reached. Safety injection flow is delayed 54 seconds after the occurrence of the low-pressure condition. This delay conservatively accounts for signal processing, diesel generator start up and emergency power bus loading consistent with the loss-of-offsite power coincident with reactor trip, as well as the pump acceleration and valve delays.

The following countermeasures limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the SBLOCA analysis for the boron content of the injection water. In addition, credit is taken in the SBLOCA analysis for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, considering the most reactive RCCA is stuck in the full out position. A rod drop time of 2.4 seconds was used while also considering an additional 2 seconds for the signal processing delay time. Therefore, a total delay time of 4.4 seconds from the time of reactor trip signal to full rod insertion was used in the SBLOCA analysis.

2. Injection of borated water provides sufficient flooding of the core to prevent excessive cladding temperatures.

During the earlier part of the Small Break transient (prior to the postulated loss-of-offsite power coincident with reactor trip), the loss of flow through the break is not sufficient to overcome the positive core flow maintained by the reactor coolant pumps. During this period, upward flow through the core is maintained. However, following the reactor coolant pump trip (due to a LOOP) and subsequent pump coastdown, a period of core uncovery occurs. Ultimately, the Small Break transient is terminated when the top of the core is recovered or the core mixing level is increasing, and ECCS flow provided to the RCS exceeds the break flow rate.

The core heat transfer mechanisms associated with the Small Break transient include the break itself, the injected ECCS water, and the heat transferred from the RCS to the steam generator secondary side. Main feedwater (MFW) is conservatively isolated in 8 seconds following the generation of the pressurizer low-pressure SI signal. Additional makeup water is also provided to the secondary using the auxiliary feedwater (AFW) system. An AFW actuation signal is derived from the pressurizer low-pressure reactor trip signal and results in the delivery of AFW flow 80 seconds after reactor trip. The heat transferred to the secondary side of the steam generator aids in the reduction of the RCS pressure.

Should the RCS depressurize to approximately 600 psia (accumulator minimum pressure), the cold leg accumulators begin to inject borated water into the reactor coolant loops as reflected in Table 6.

4 ACCEPTANCE CRITERIA AND RESULTS

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Criteria 1 through 3 are explicitly covered by the Small Break LOCA analysis modeling high head safety injection cross-ties closed.

For criterion 4, the appropriate core geometry was modeled in the analysis. The results based on this geometry satisfy the peak clad temperature (PCT) criterion of 10 CFR 50.46 and consequently, demonstrate that the core remains amenable to cooling.

For criterion 5, Long-Term Core Cooling (LTCC) considerations are not directly applicable to the Small Break LOCA transient analysis addressed herein, with the exception of predicting switchover from ECCS injection phase to ECCS recirculation phase and ensuring the SBLOCA transient remains terminated.

The acceptance criteria were established to provide a significant margin in ECCS performance following a LOCA.

In order to determine the conditions that produced the most limiting SBLOCA case (as determined by the highest calculated peak cladding temperature), eleven break cases were examined for D. C. Cook Unit 1. These cases were investigated to capture the most severe postulated Small Break LOCA event. The following discussion provide insight into the analyzed conditions.

The results of the generic study documented in Reference 6 demonstrate that the cold leg break location is limiting with respect to postulated cold leg, hot leg and pump suction leg break locations. The PCT results for D. C. Cook Unit 1 are shown in Tables 4 and 5. Inherent in the Small Break analysis are several input parameters (see Section 1.2 and Table 1), while Table 6 provides the key transient event times.

Limiting Break Case

For D. C. Cook Unit 1 the SBLOCA analysis with high head safety injection cross-tie isolated showed that the 3.25-inch break is the limiting case. A time-in-life study to determine the limiting PCT for this case considering clad burst concluded that the maximum PCT occurs at beginning-of-life (BOL). A summary of the transient response for the limiting PCT case is shown in Figures 5 through 15. These figures present the response of the following parameters:

- RCS Pressure
- Core Mixture Level
- Core Exit Vapor Temperature
- Broken and Intact Loops Secondary Pressures
- Break Vapor Flow Rate
- Break Liquid Flow Rate
- Broken and Intact Loops Accumulator Flow Rates
- Broken and Intact Loops Pumped Safety Injection Flow Rates
- Clad Temperature at PCT Elevation
- Hot Spot Fluid Temperature at PCT Elevation
- Rod Film Heat Transfer Coefficient at PCT Elevation

Upon initiation of the limiting 3.25-inch break for D. C. Cook Unit 1, there is an initial rapid depressurization of the RCS followed by an intermediate equilibrium at approximately 1150 psia (see Figure 5). The limiting 3.25-inch break depressurizes to the accumulator injection setpoint of 600 psia at approximately 1264 seconds (see Figure 11). During the initial period of the Small Break transient, the effect of the break flow rate is not sufficient to overcome the flow rate maintained by the reactor coolant pumps as they coast down. As such, normal upward flow is maintained through the core and core heat is adequately removed. Following reactor trip, the removal of the heat generated as a result of the decay of fission products is accomplished via a two-phase mixture level covering the core. The core mixture level and peak clad temperature transient plots for the limiting break calculations are illustrated in Figures 6 and 13, respectively. These figures show that the peak clad temperature occurs near the time when the core is most deeply uncovered and the top of the core is being cooled by steam. This time is characterized by the highest vapor superheating above the mixture level (refer to Figure 7). For D. C. Cook Unit 1, the limiting PCT time-in-life was determined to be BOL.

A comparison of the flow provided by the safety injection system to the intact and broken loops can be found in Figure 12. The cold leg break vapor and liquid mass flow rates are provided in Figures 9 and 10, respectively. Figures 14 and 15 provide additional information on the fluid temperature at the hot spot and hot rod surface heat transfer coefficient at the hot spot, respectively. Figure 8 depicts the secondary side pressure for both the intact and broken loops for the limiting PCT break case.

Maximum Local Oxidation

For the D. C. Cook Unit 1 SBLOCA analysis, the maximum local oxidation case was the 3.25-inch break case. Based on the time-in-life study, the maximum local transient oxidation is 3.61% at 11,500 MWD/MTU. The limiting transient oxidation occurs at the hot rod burst elevation and includes both outside oxidation and post-rupture inside oxidation in the burst region. Pre-existing (pre-transient) oxidation was also considered and the sum of the pre-transient and transient oxidation remains below 17% at all times in life, for all fuel resident in the core.

Core Wide Average Oxidation

Tables 4 and 5 indicate that for the D. C. Cook Unit 1 SBLOCA analysis, the core wide average oxidation for all cases is less than 1%. Therefore the calculated total amount of hydrogen generation is less than the 1% limit defined by 10 CFR 50.46.

Additional Break Cases

Generic studies documented in Reference 6 determined that the limiting PCT Small Break transient occurs for breaks of less than 10-inches in diameter in the cold leg. For D. C. Cook Unit 1, the limiting PCT is captured by the 1.5-, 2-, 2.5-, 2.75-, 3-, 3.25-, 3.5-, 3.75-, 4-, 6- and 8.75-inch break spectrum. The beginning-of-life (BOL) results for these break spectrum cases are given in Table 4. Figures 16 through 44 address the non-limiting cases (1.5-, 2-, 2.5-, 2.75-, 3-, 3.5-, 3.75-, 4-, 6- and 8.75-inch) analyzed for D. C. Cook Unit 1. The 1.5-inch case did not show core uncovery, therefore PCT information was not calculated. The plots for each of the additional non-limiting break cases include:

1. RCS Pressure
2. Core Mixture Level
3. Clad Temperature at PCT Elevation (Note no PCT plots provided for 1.5-inch case)

The PCTs for each of the additional breaks considered are shown in Table 4 and are less than the limiting 3.25-inch break case.

Small breaks larger than 8.75 inches are non-limiting with respect to the limiting 3.25 inch break case with a transient response similar to or better (i.e. less limiting) than the 8.75 inch break case. The improved response is due to earlier RCS depressurization to the RHR cut-in pressure as compared to the 8.75 inch case, resulting in increased delivery of the RHR flow and earlier transient termination.

Switchover from ECCS Injection Phase to ECCS Recirculation Phase

When the RWST volume of 280,000 gallons is delivered via safety injection and containment spray, NOTRUMP predicts switchover from ECCS injection phase to ECCS recirculation phase. At that time RHR flow is re-aligned to the sump and an interruption in RHR flow for up to 5 minutes may occur. For break cases that have a calculated RCS pressure at or below the RHR cut-in pressure, the 5 minute interruption in RHR flow is considered. The applicable transients were shown to satisfy the analysis termination conditions, as discussed in more detail below.

Transient Termination

The 10 CFR 50.46 criteria continue to be satisfied beyond the end of the calculated transient due to the presence of the following conditions:

1. The RCS pressure is gradually decreasing or reached equilibrium.
2. The net mass inventory is increasing or reached equilibrium.
3. The core mixture level is recovered, or recovering due to increasing mass inventory.
4. As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel cladding temperatures will continue to decline indicating that the temperature excursion is terminated.

5 CONCLUSIONS

The Small Break LOCA analysis for D. C. Cook Unit 1 considered a break spectrum of 1.5-, 2-, 2.5-, 2.75-, 3-, 3.25-, 3.5-, 3.75-, 4-, 6- and 8.75-inch diameters. The limiting peak cladding temperature of 1725°F was calculated at BOL for the 3.25-inch case and a maximum local transient oxidation of 3.61% was calculated at the limiting time-in-life of 11,500 MWD/MTU for the 3.25-inch case.

The analysis presented herein shows that the accumulator and safety injection subsystems of the Emergency Core Cooling System, together with the heat removal capability of the steam generators, provide sufficient core heat removal capability to maintain the calculated peak cladding temperature for Small Break LOCA below the required limit of 10 CFR 50.46. Furthermore, the analysis shows that the local cladding oxidation and core wide average oxidation, including consideration of pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation, are less than the 10 CFR 50.46 (Reference 1) limits. Note that the core wide average oxidation results illustrate that the total hydrogen generation is less than 1%.

Table 7 provides a results summary for the D. C. Cook Unit 1 SBLOCA analysis. Results include PCT, maximum local transient oxidation and total hydrogen generation.

6 REFERENCES

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974, as amended in Federal Register, Volume 53, September 1988.

2. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, (proprietary) and WCAP-10080-NP-A (non-proprietary), August 1985.

3. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (proprietary) and WCAP-10081-NP-A (non-proprietary), August 1985.

4. Thompson, C. D. et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Rev. 1 (proprietary), July 1997.

5. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plant," NUREG-0611, January 1980.

6. Rupprecht, S. D. et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (proprietary), October 1986.

Table 1
Input Parameters Used in the Small Break LOCA Analysis

Input Parameter	Value
Core Rated Thermal Power-100%	3304 MWt
Calorimetric Uncertainty, %	0.34
Fuel Type	15 x 15 Upgrade Fuel
Total Core Peaking Factor, F_Q	2.32
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.55
Hot Assembly Average Power Factor, P_{HA}	1.38
Maximum Axial Offset, %	20
Initial RCS Loop Flow, gpm/loop	83,200
Initial Vessel T_{avg}, °F	577.4$^{(1)}$
Initial Pressurizer Pressure (plus uncertainties), psia	2317$^{(2)}$
Reactor Coolant Pump Type	93A
Pressurizer Low-Pressure Reactor Trip Setpoint, psia	1860
Reactor Trip Signal Delay Time, seconds	2.0
Rod Drop Time, seconds	2.4
Auxiliary Feedwater Temperature (Maximum), °F	120
AFW Flow (Minimum) to all 4 Steam Generators, gpm	427.25
AFW Flow Delay Time (Maximum), seconds	80
AFW Actuation Signal	Reactor Trip/Low Pressurizer Pressure
Maximum AFW Piping Purge Volume, ft^3	78
Steam Generator Tube Plugging (Maximum), %	10
Maximum MFW Isolation, seconds	8
MFW Isolation Signal	Safety Injection Actuation
Steam Generator Secondary Water Mass, lbm/SG	101,169
Containment Spray Flowrate for 2 Pumps, gpm	7,400
RWST Deliverable Volume (Minimum), gallons	280,000

Notes:

(1) Analysis supports operation over the range of nominal full-power T_{avg} values of 553.7°F – 575.4°F.

(2) Analysis supports operation at nominal initial pressurizer pressure (without uncertainties) of 2100 psia and 2250 psia.

Table 1 (continued)
Input Parameters Used in the Small Break LOCA Analysis

Input Parameter	Value
SI Temp at Cold Leg Recirculation Time (Maximum), °F	190
ECCS Configuration	1 CHG pump, 1 HHSI pump, 1 RHR pump - faulted loop injects to RCS pressure (1.5-inch through 6-inch breaks) 1 CHG pump, 1 HHSI pump, 1 RHR pump –no RHR/HHSI in the faulted loop because the break is postulated along the accumulator line, faulted loop CHG flow injects to RCS pressure (8.75-inch)
ECCS Water Temperature (Maximum), °F	105
Pressurizer Low-Pressure Safety Injection Setpoint, psia	1715
SI Flow Delay Time, seconds	54
ECCS Flow vs. Pressure (Injection Phase)	See Tables 2 and 3
Initial Accumulator Water/Gas Temperature, °F	130
Initial Nominal Accumulator Water Volume, ft^3	946
Minimum Accumulator Pressure, psia	600

Table 2
Safety Injection Flows Used in the Small Break LOCA Analysis - Injection Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump - faulted loop injects to RCS pressure – 1.5-inch through 6-inch breaks)

RCS Pressure (psia)	Broken Loop (lbm/sec)	Intact Loops (lbm/sec)		
	Loop 1	Loop 2	Loop 3	Loop 4
14.7	188.63	158.38	175.00	159.93
114.7	127.20	102.43	117.32	103.38
214.7(1)	38.89	12.50	34.37	12.50
314.7	37.49	12.21	33.10	12.21
414.7	35.93	11.90	31.73	11.90
514.7	34.30	11.57	30.26	11.57
614.7	32.56	11.23	28.64	11.23
714.7	30.73	10.88	26.98	10.88
814.7	28.80	10.54	25.22	10.54
914.7	26.74	10.18	23.35	10.18
1014.7	24.34	9.82	21.14	9.82
1114.7	21.49	9.46	18.54	9.46
1214.7	18.00	9.10	15.32	9.10
1314.7	13.03	8.70	10.73	8.70
1414.7(2)	10.36	8.29	8.29	8.29
1514.7	9.82	7.86	7.86	7.86
1614.7	9.27	7.42	7.42	7.42
1714.7	8.72	6.98	6.98	6.98
1814.7	8.08	6.47	6.47	6.47
1914.7	7.41	5.93	5.93	5.93
2014.7	6.72	5.38	5.38	5.38
2114.7	5.94	4.76	4.76	4.76
2214.7	5.02	4.01	4.01	4.01
2314.7	0	0	0	0

Notes:

(1) RHR cut-in pressure

(2) HHSI cut-in pressure

Table 3
Safety Injection Flows Used in the Small Break LOCA Analysis – Injection Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump –no RHR/HHSI in the faulted loop because the break is postulated along the accumulator line, faulted loop CHG flow injects to RCS pressure – 8.75-inch break)

RCS Pressure (psia)	Broken Loop (lbm/sec)		Intact Loops (lbm/sec)		
	Loop 1 – CHG	Loop 1 – RHR/HHSI	Loop 2	Loop 3	Loop 4
14.7	16.34	157.03	139.36	156.01	140.26
34.7	16.27	215.73	126.84	99.71	127.64
54.7	16.20	274.56	115.46	32.18	116.17
74.7(1)	16.13	303.81	97.04	12.90	97.63
94.7	16.06	323.64	72.95	12.84	73.37
114.7	15.98	345.72	43.70	12.78	43.93
134.7(2)	15.91	365.86	12.73	12.73	12.73
154.7	15.84	365.86	12.67	12.67	12.67
214.7	15.63	365.86	12.50	12.50	12.50
314.7	15.25	365.86	12.21	12.21	12.21
414.7	14.83	365.86	11.90	11.90	11.90
514.7	14.45	365.86	11.57	11.57	11.57
614.7	14.03	365.86	11.23	11.23	11.23
714.7	13.60	365.86	10.88	10.88	10.88
814.7	13.17	365.86	10.54	10.54	10.54
914.7	12.73	365.86	10.18	10.18	10.18
1014.7	12.29	365.86	9.82	9.82	9.82
1114.7	11.83	365.86	9.46	9.46	9.46
1214.7	11.38	365.86	9.10	9.10	9.10
1314.7	10.88	365.86	8.70	8.70	8.70
1414.7	10.36	365.86	8.29	8.29	8.29
1514.7	9.82	365.86	7.86	7.86	7.86
1614.7	9.27	365.86	7.42	7.42	7.42
1714.7	8.72	365.86	6.98	6.98	6.98
1814.7	8.08	365.86	6.47	6.47	6.47
1914.7	7.41	365.86	5.93	5.93	5.93
2014.7	6.72	365.86	5.38	5.38	5.38

Table 3 **Safety Injection Flows Used in the Small Break LOCA Analysis – Injection Phase** **(1 CHG pump, 1 HHSI pump, 1 RHR pump –no RHR/HHSI in the faulted loop because the break is postulated along the accumulator line, faulted loop CHG flow injects to RCS pressure – 8.75-inch break)**					
RCS Pressure (psia)	**Broken Loop (lbm/sec)**		**Intact Loops (lbm/sec)**		
	Loop 1 – CHG	Loop 1 – RHR/HHSI	Loop 2	Loop 3	Loop 4
2114.7	5.94	365.86	4.76	4.76	4.76
2214.7	5.02	365.86	4.01	4.01	4.01
2314.7	0	365.86	0	0	0

Notes:

(1) HHSI cut-in pressure

(2) RHR cut-in pressure

Table 4 SBLOCTA BOL Results										
Break Size (in)	**2**	**2.5**	**2.75**	**3**	**3.25**	**3.5**	**3.75**	**4**	**6**	**8.75**
PCT (°F)	968.4	1433.4	1452.5	1584.0	1725.0	1705.3	1517.9	1411.2	670.1	1132.2
PCT Time (s)	2284.4	2684.0	2140.6	2000.5	1483.4	1249.3	1129.8	986.2	404.1	3277.9
PCT Elevation (ft)	11.0	11.50	11.50	11.75	11.75	11.75	11.50	11.25	11.0	11.50
Max. Local ZrO_2 (%)	0.03	0.70	0.54	1.26	2.08	1.72	0.56	0.26	0.0	0.04
Max. Local ZrO_2 Elev. (ft)	11.0	11.50	11.50	11.75	11.75	11.50	11.50	11.25	11.0	11.50
Core-Wide Avg. ZrO_2 (%)	0.0	0.09	0.07	0.17	0.30	0.26	0.08	0.04	0.0	0.01

Table 5 SBLOCTA Limiting Results from the 3.25-inch Time-in-Life Study		
Time-in-Life (MWD/MTU)	BOL	11,500
PCT (°F)	**1725.0**	1720.5
PCT Time (s)	1483.4	1480.3
PCT Elevation (ft)	11.75	11.75
Hot Rod Burst Time (s)	---	1478.4
Hot Rod Burst Elevation (ft)	---	11.75
Max. Local Transient ZrO_2 (%)	2.08	**3.61**
Max. Local Transient ZrO_2 Elev. (ft)	11.75	11.75
Core-Wide Avg. ZrO_2 (%)	0.30	0.22

Table 6
Time Sequence of Events

Event Time	1.5-inch	2-inch	2.5-inch	2.75-inch	3-inch	3.25-inch	3.5-inch	3.75-inch	4-inch	6-inch	8.75-inch
Break Initiation (s)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal (s)	89.8	45.9	27.9	22.7	19.0	16.2	14.0	12.3	10.9	6.0	4.5
S-Signal (s)	112.8	60.9	38.2	31.6	26.8	23.2	20.3	17.9	15.7	8.4	6.7
SI Flow Delivered (1), (s)	166.8	114.9	92.2	85.6	80.8	77.2	74.3	71.9	69.7	62.4	60.7
Loop Seal Clearing (2) (s)	2492	1341	857	628	516	445	386	390	302	146	31
Core Uncovery (4) (s)	N/A	1897	1027	1017	963	780	664	630	602	374	2631
Accumulator Injection (s)	N/A	N/A	3065	2129	1707	1264	1031	940	823	346	169 (3)
RWST Volume Delivered (5) (s)	2166	2158	2145	2138	2130	2121	2114	2110	2106	2043	1591
PCT Time (BOL) (s)	N/A	2284	2684	2141	2001	1483	1249	1130	986	404	3278
Core Recovery (4) (s)	N/A	6663	4032	4081	3977	3840	3973	4110	3404	423	4530

Notes:

(1) SI is assumed to begin 54.0 seconds (SI delay time) after the S-Signal.

(2) Loop seal clearing is assumed to occur when the steam flow through the broken loop, loop seal is sustained above 1 lbm/s.

(3) For the 8.75-inch case accumulator injection begins for Loops 2-4 only because the Loop 1 (broken loop) accumulator line is the location of the break and assumed to spill to containment.

(4) The latest point of sustained core uncovery/recovery is reported.

(5) The analysis assumes minimum usable RWST volume (280,000 gal) delivered via ECCS injection and containment spray before the low level RWST water level signal for switchover to cold leg recirculation is reached.

Table 7 **SBLOCA Results Summary**	
Peak Cladding Temperature (°F)	1725
Maximum Local Transient Oxidation (%)	3.61
Total Hydrogen Generation (%)	< 1%

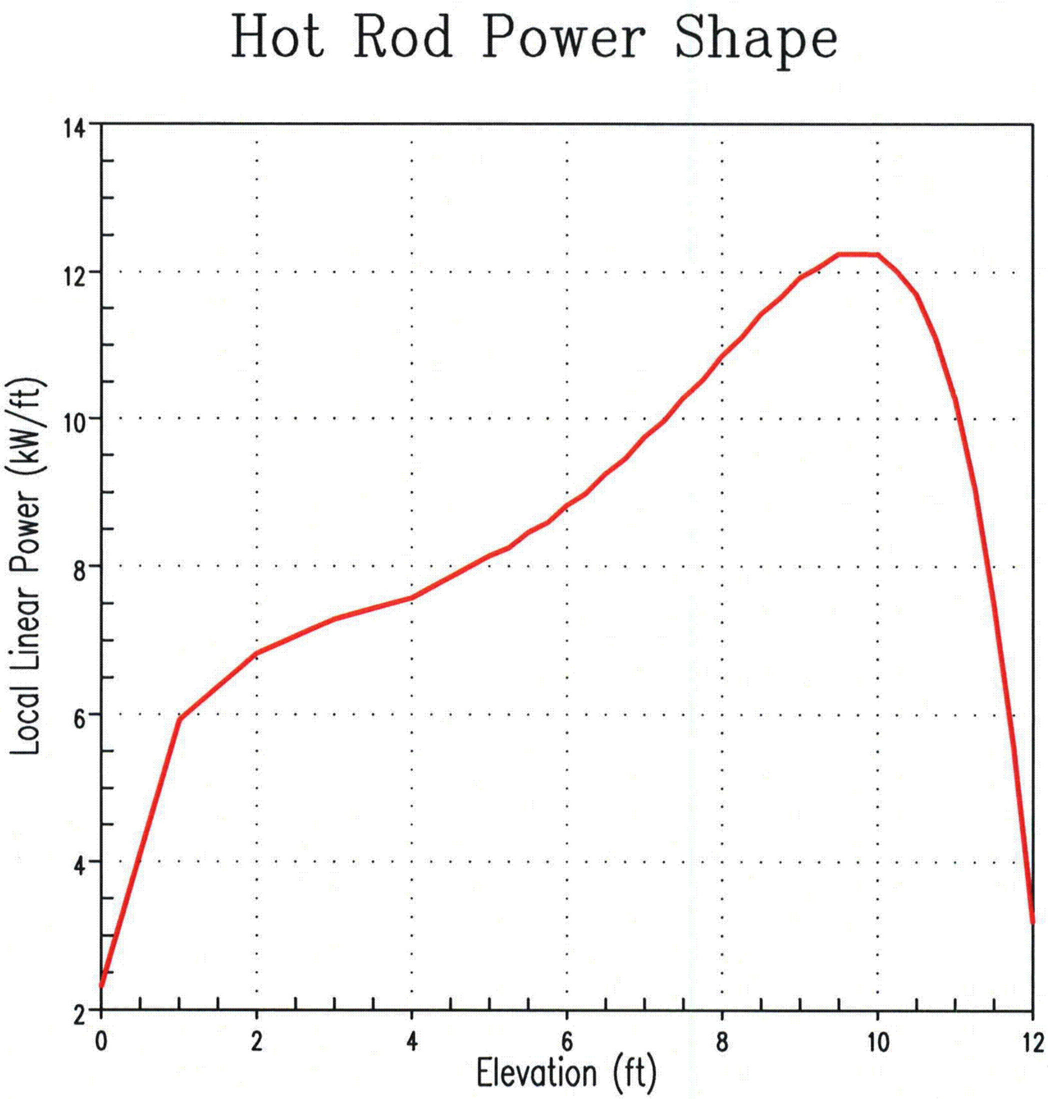

Figure 1
Small Break Hot Rod Power Shape

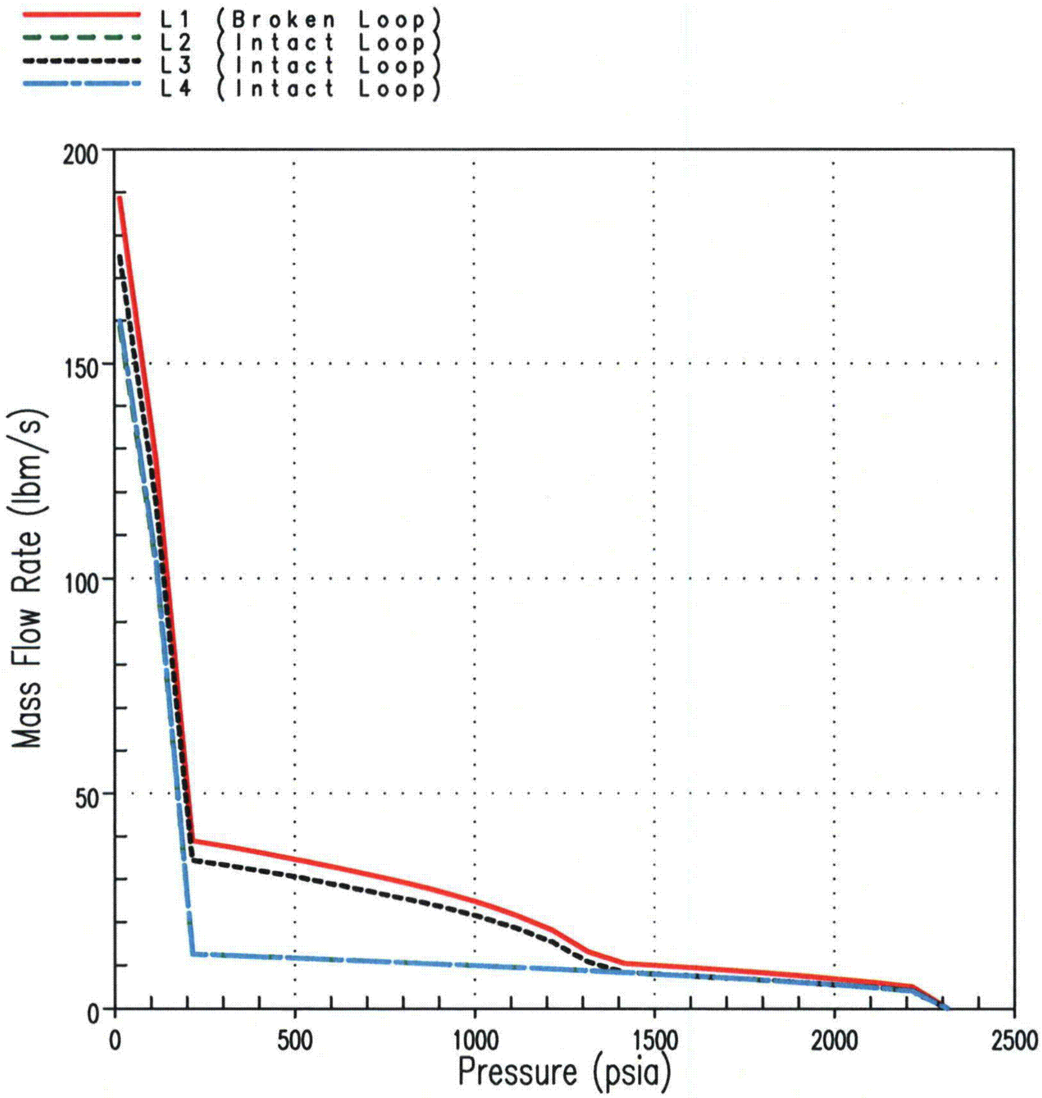

Figure 2
Small Break LOCA Safety Injection Flows – Injection Phase
1 CHG pump, 1 HHSI pump, 1 RHR pump - faulted loop injects to RCS pressure – 1.5-inch through 6-inch breaks)

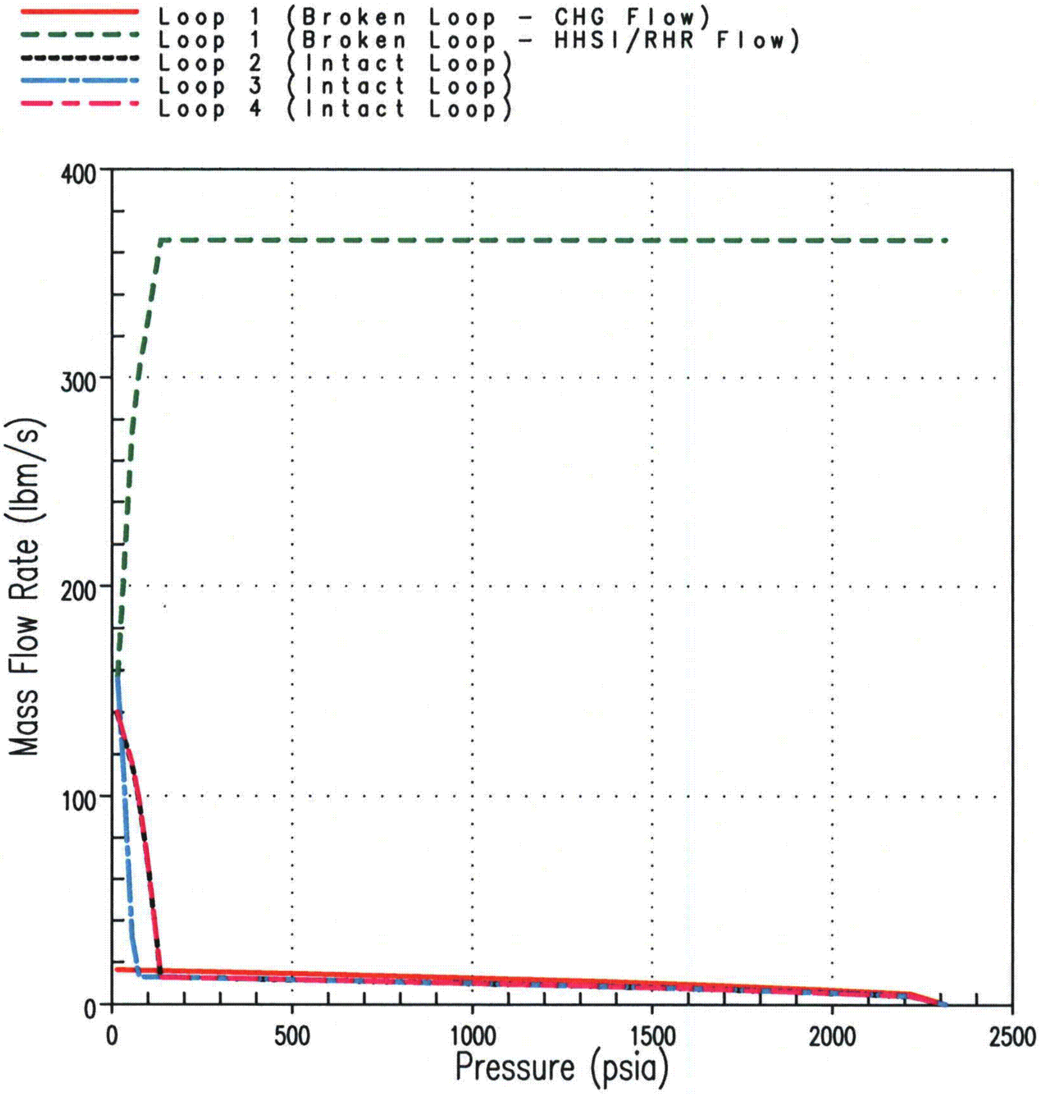

Figure 3
Small Break LOCA Safety Injection Flows – Injection Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump –no RHR/HHSI in the faulted loop because the break is postulated along the accumulator line, faulted loop CHG flow injects to RCS pressure – 8.75-inch break)

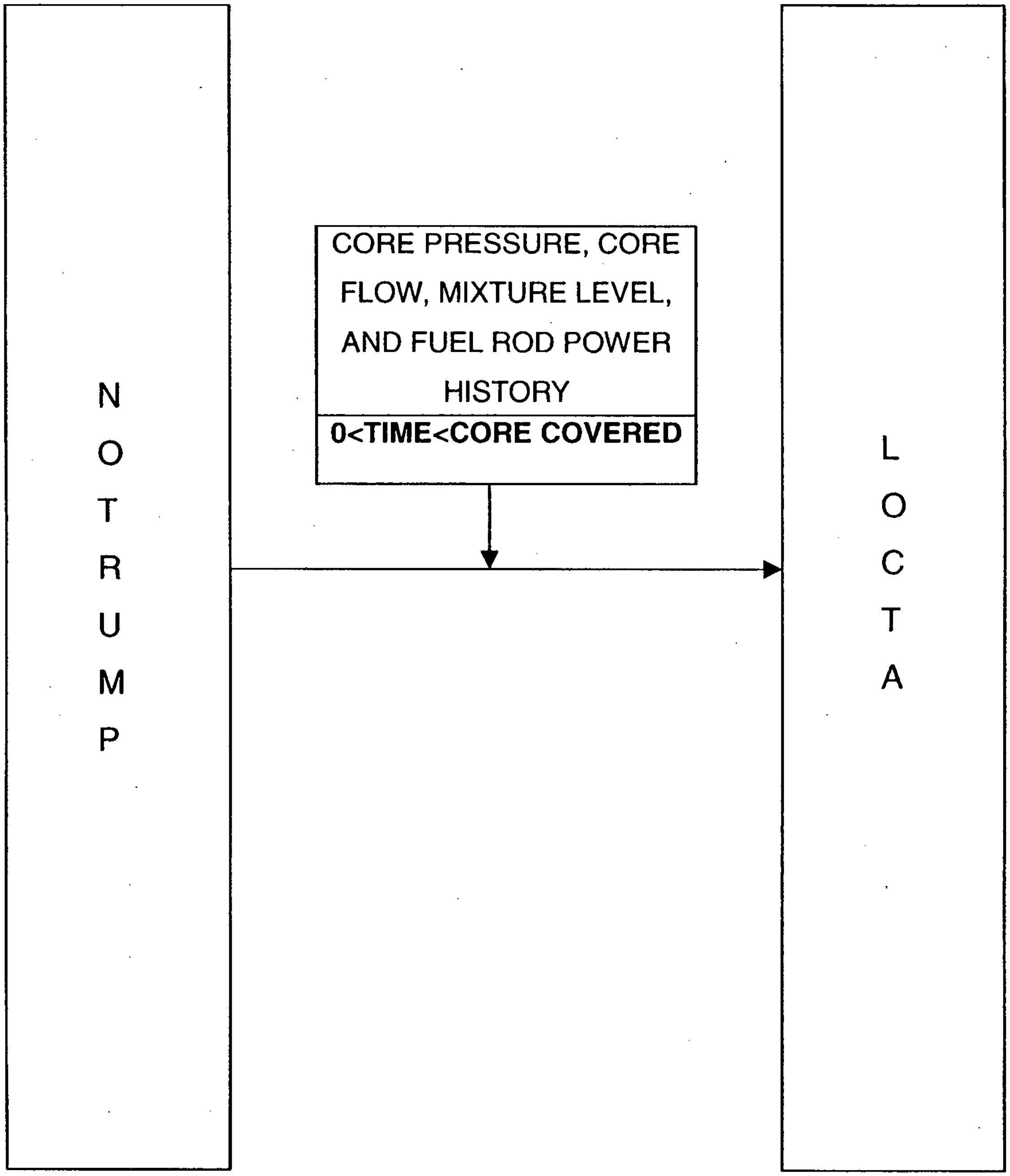

Figure 4
Code Interface Description
for Small Break Model

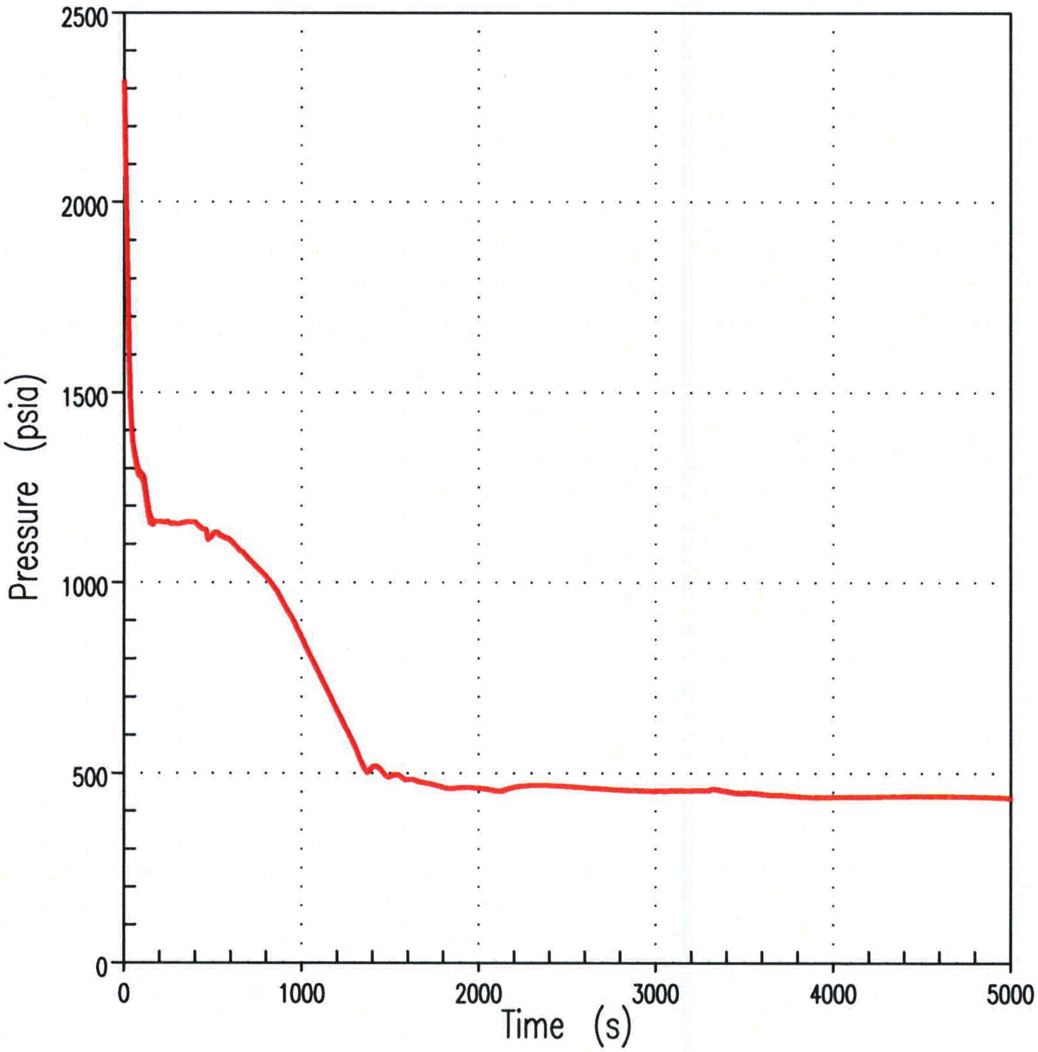

Figure 5
3.25-inch Break
RCS Pressure

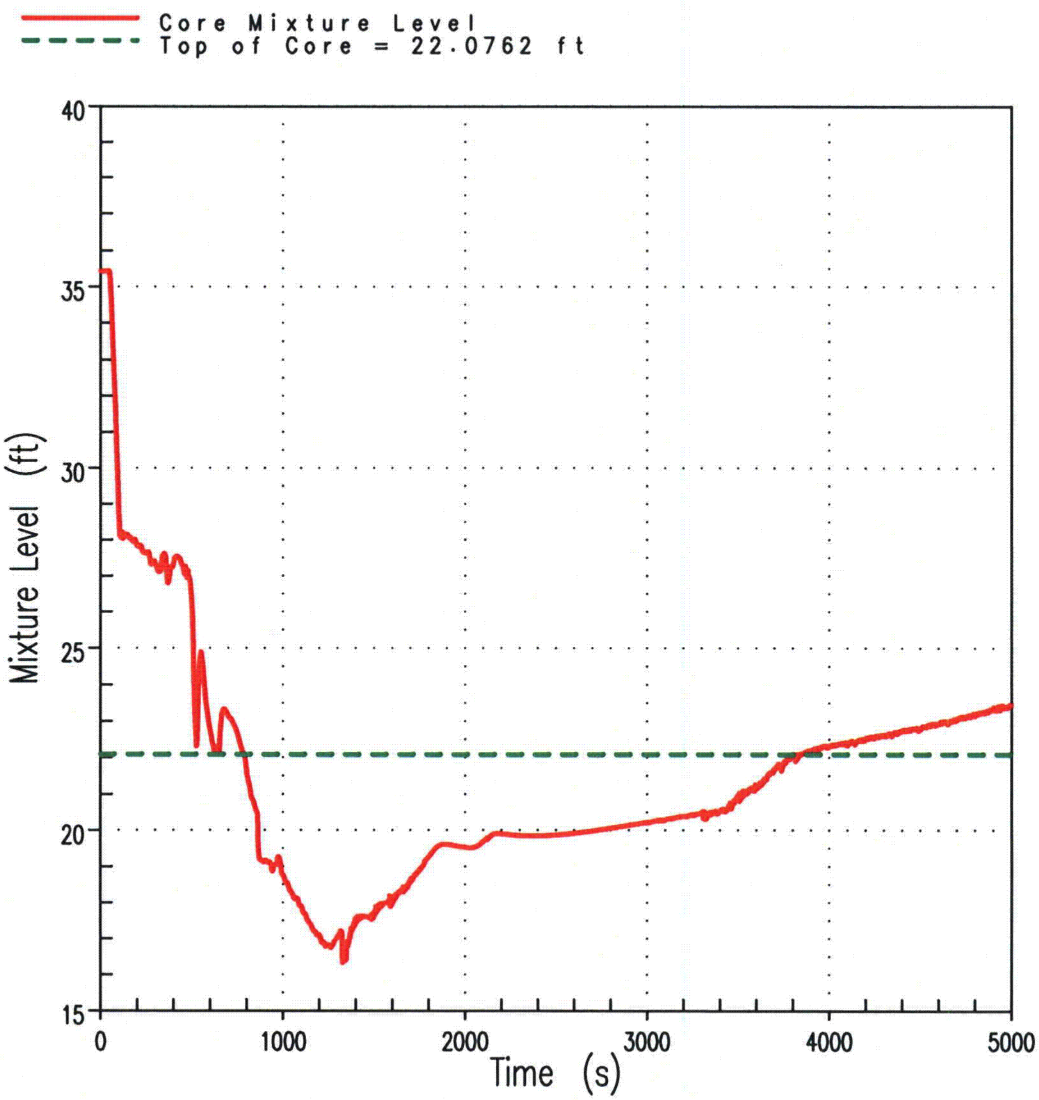

Figure 6
3.25-inch Break
Core Mixture Level

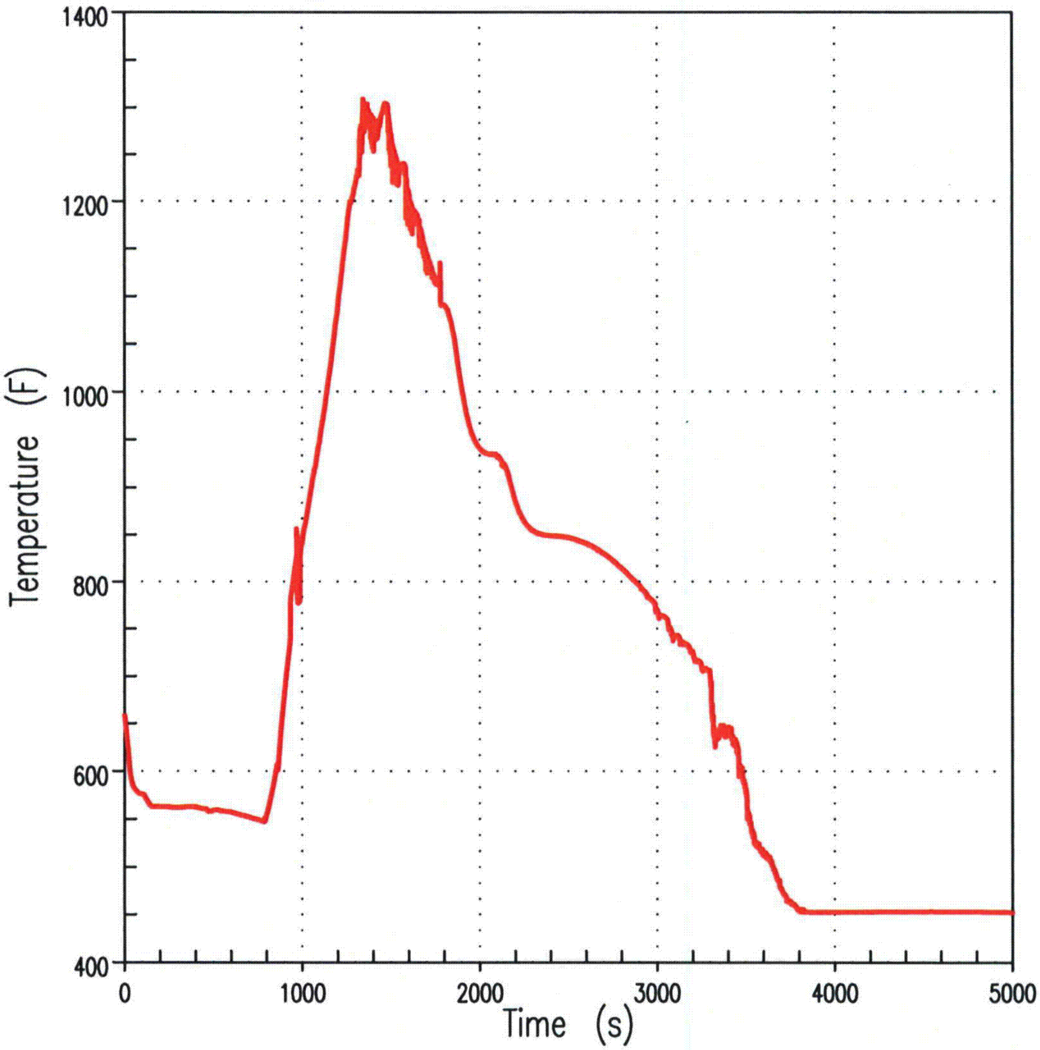

Figure 7
3.25-inch Break
Core Exit Vapor Temperature

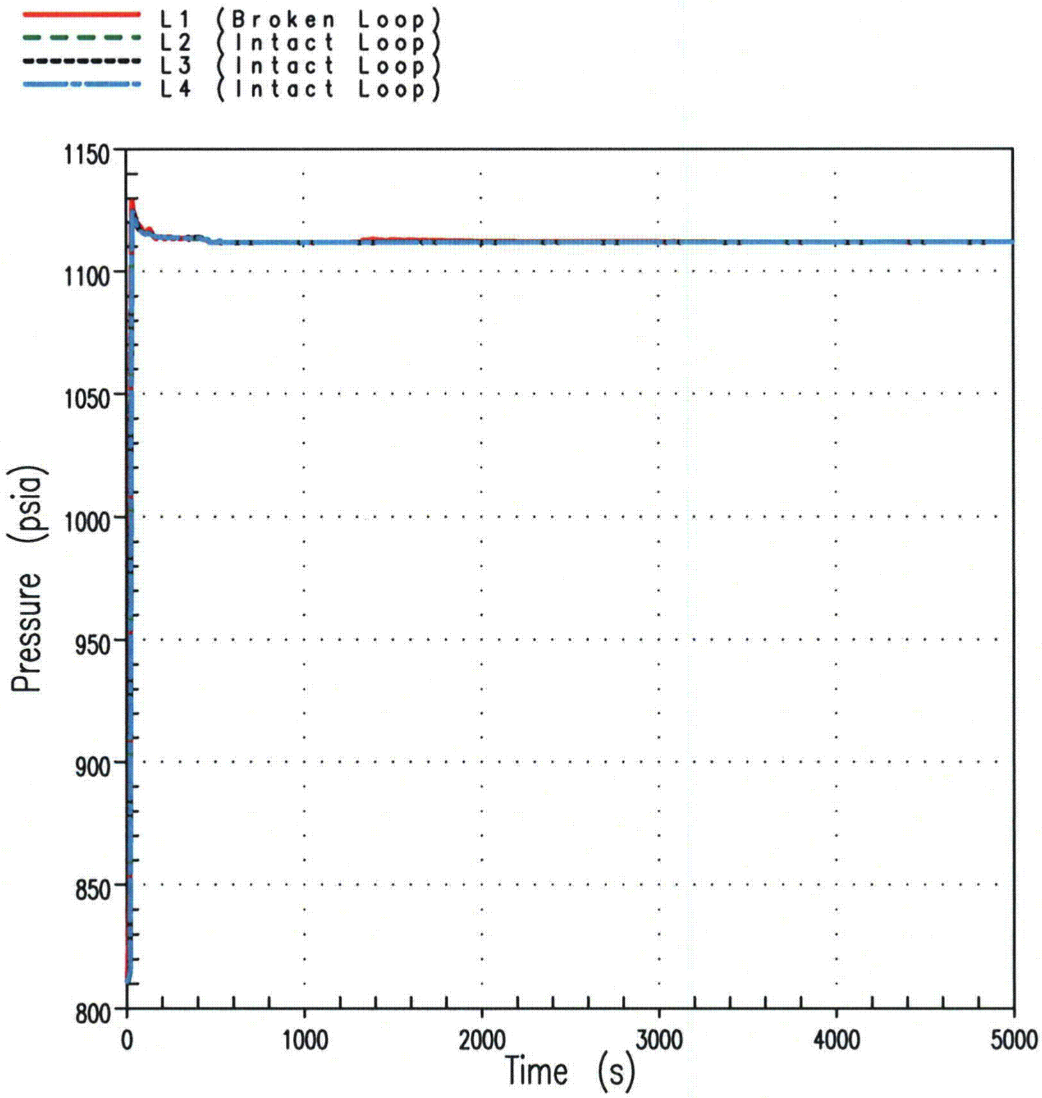

Figure 8
3.25-inch Break
Broken and Intact Loops Secondary Pressures

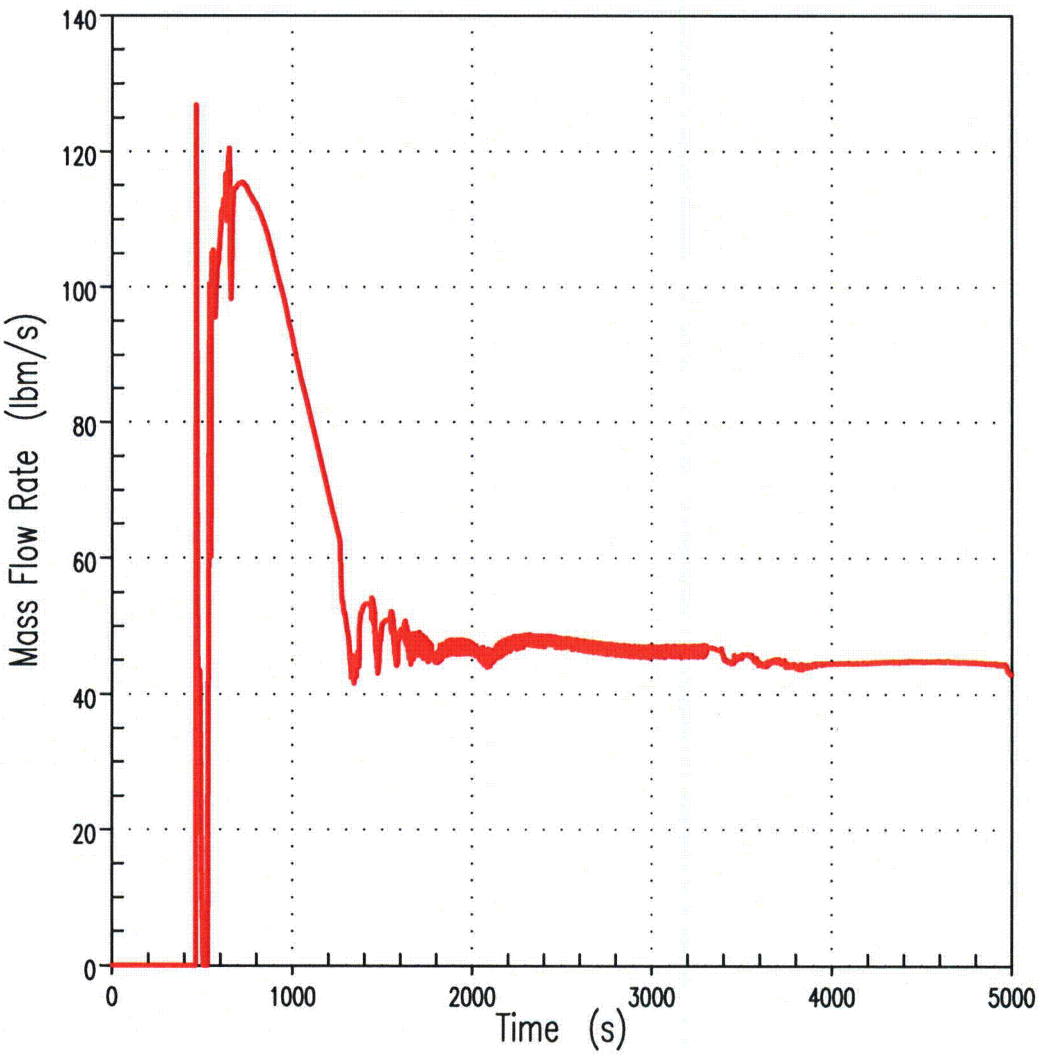

Figure 9
3.25-inch Break
Break Vapor Flow Rate

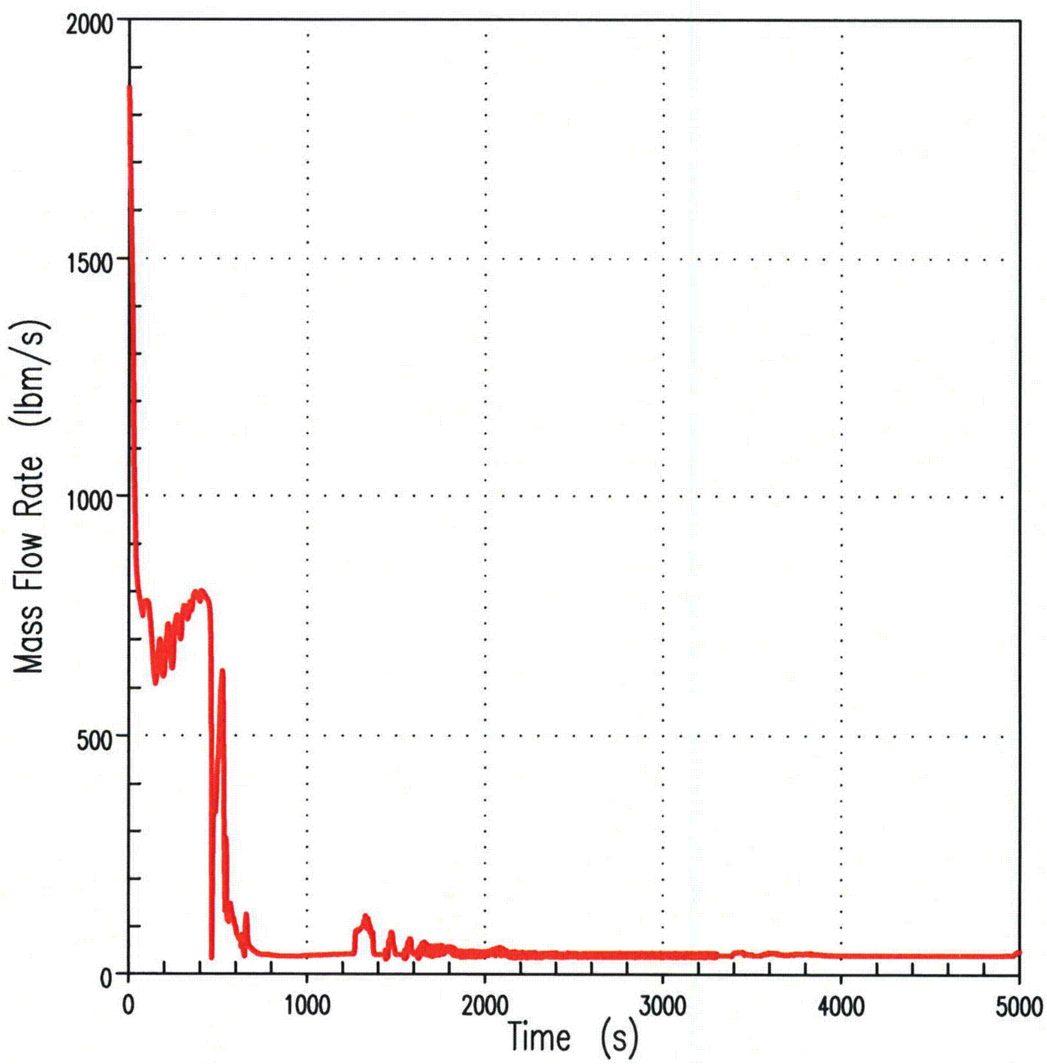

Figure 10
3.25-inch Break
Break Liquid Flow Rate

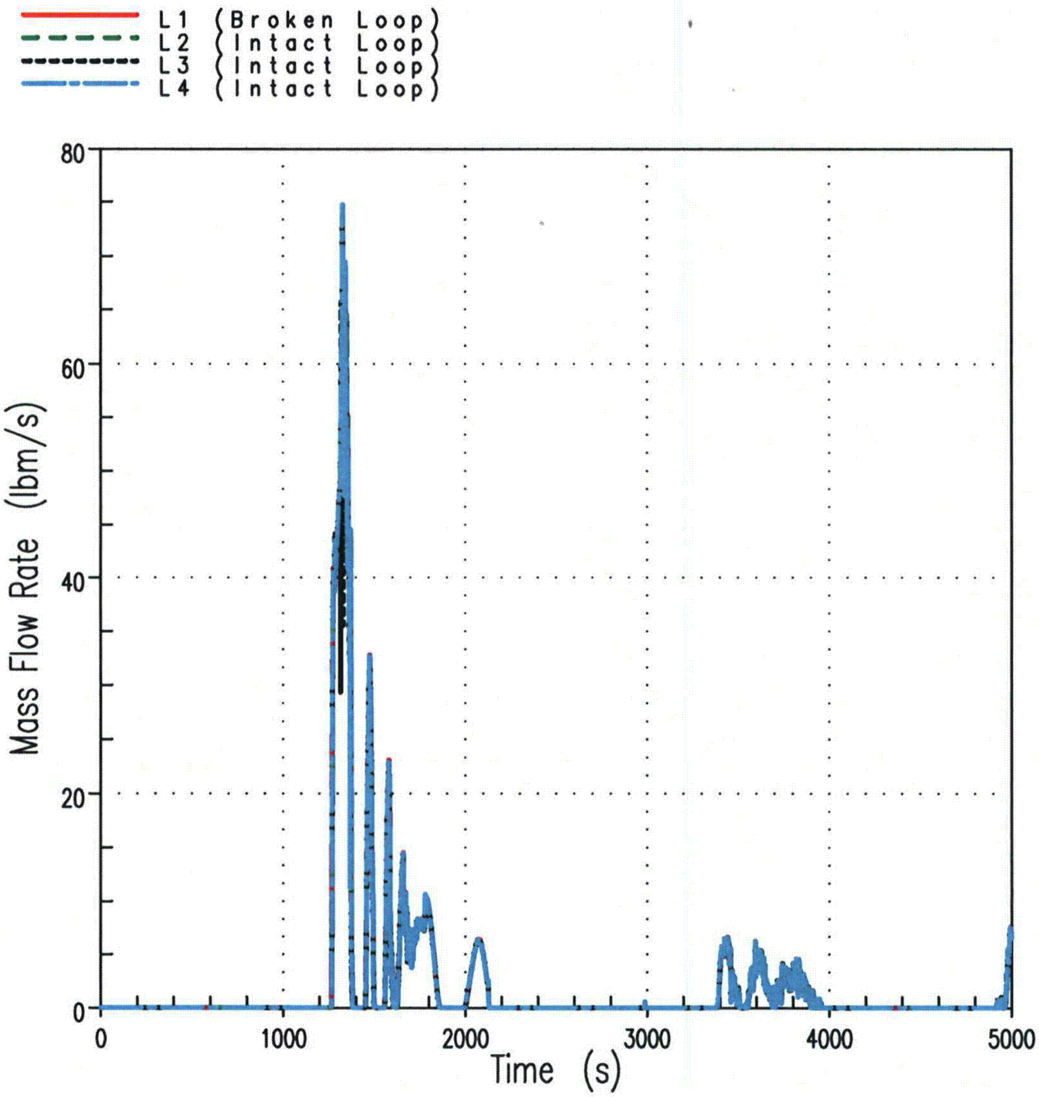

Figure 11
3.25-inch Break
Broken and Intact Loops Accumulator Flow Rates

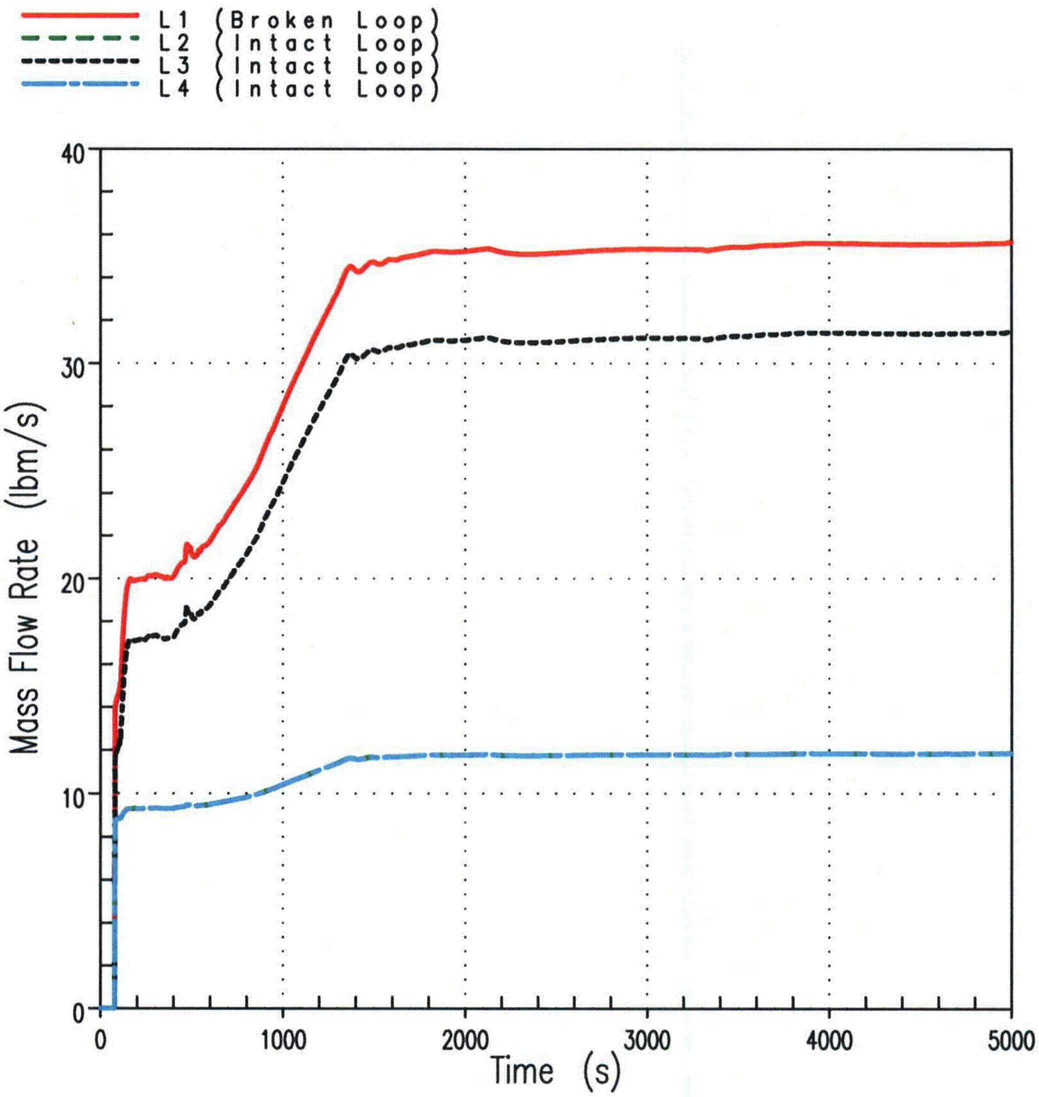

Figure 12
3.25-inch Break
Broken and Intact Loops Pumped Safety Injection Flow Rates

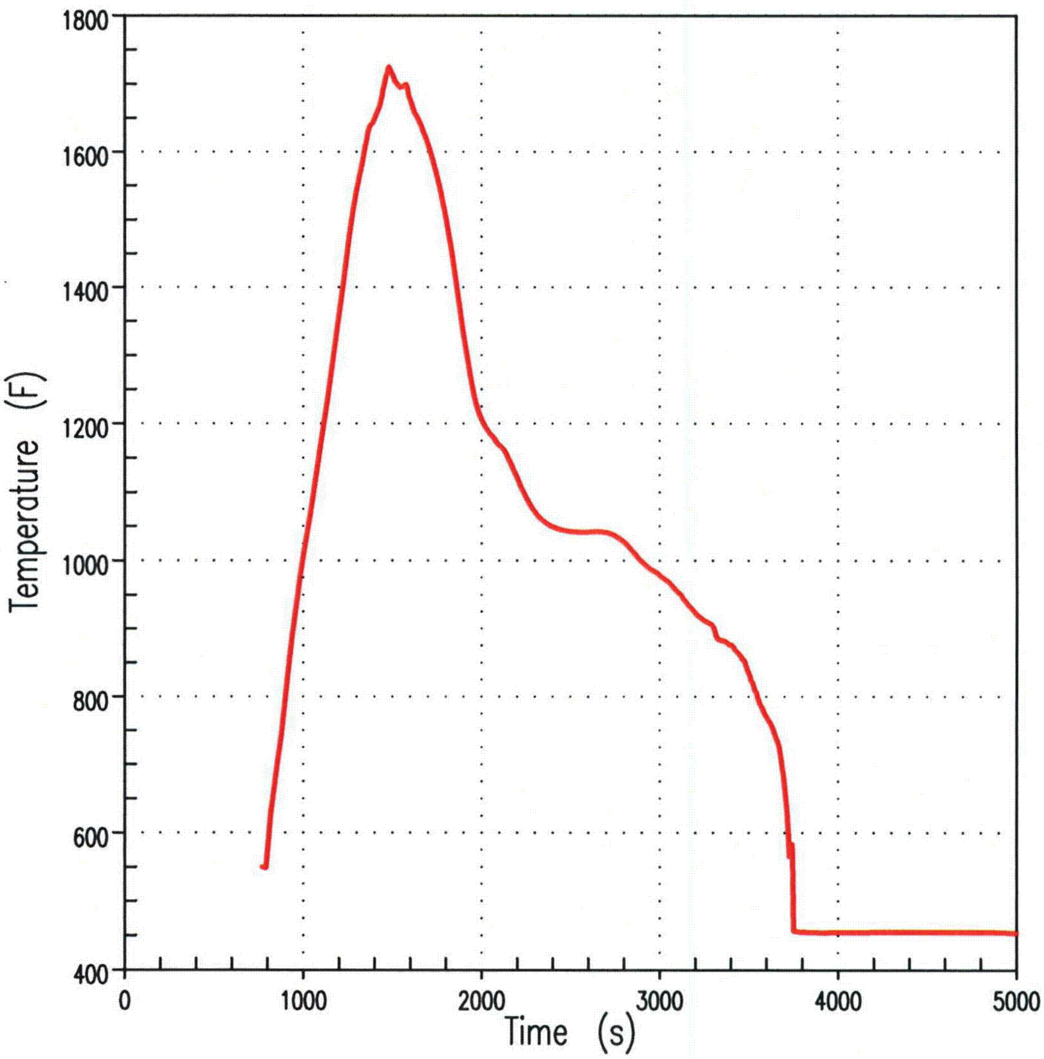

Figure 13
3.25-inch Break
Clad Temperature at PCT Elevation (11.75 ft)

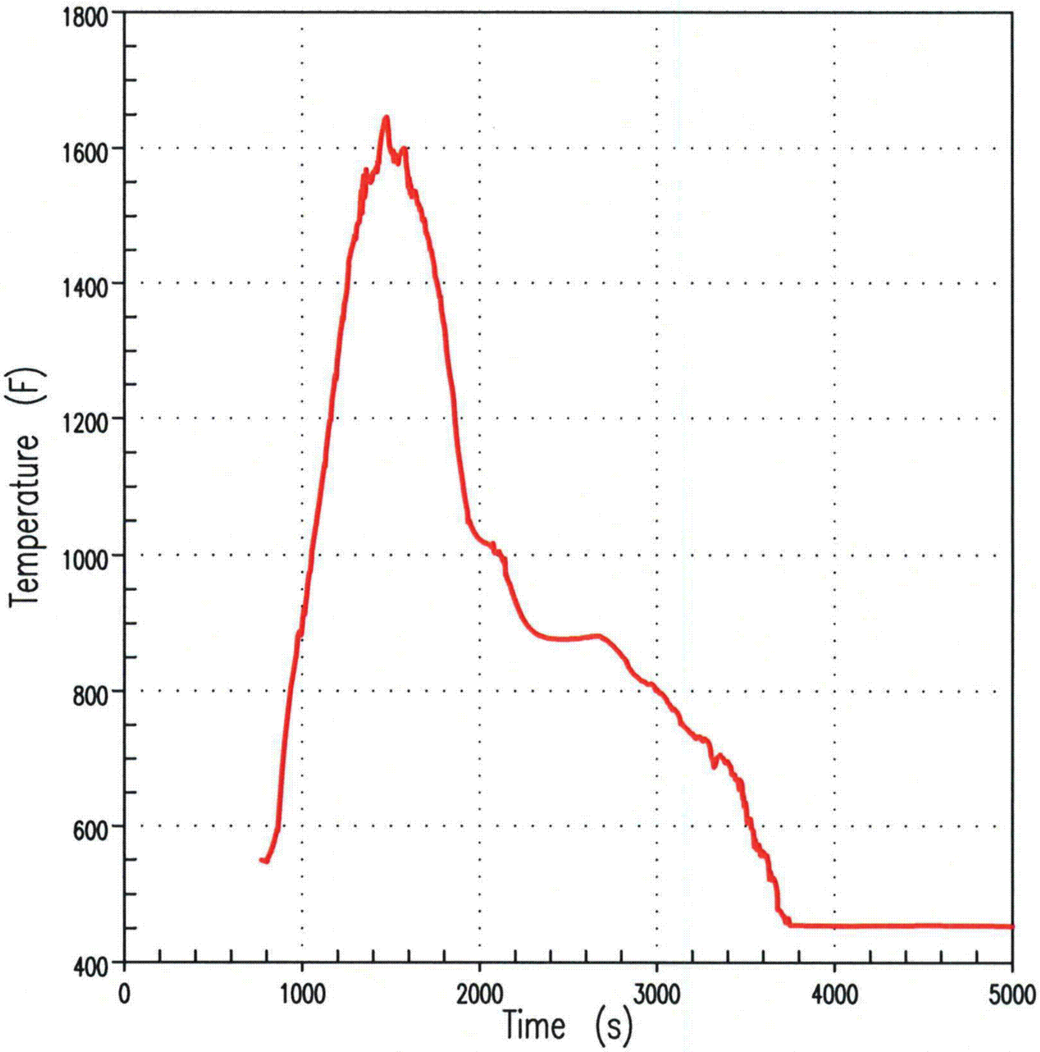

Figure 14
3.25-inch Break
Hot Spot Fluid Temperature at PCT Elevation (11.75 ft)

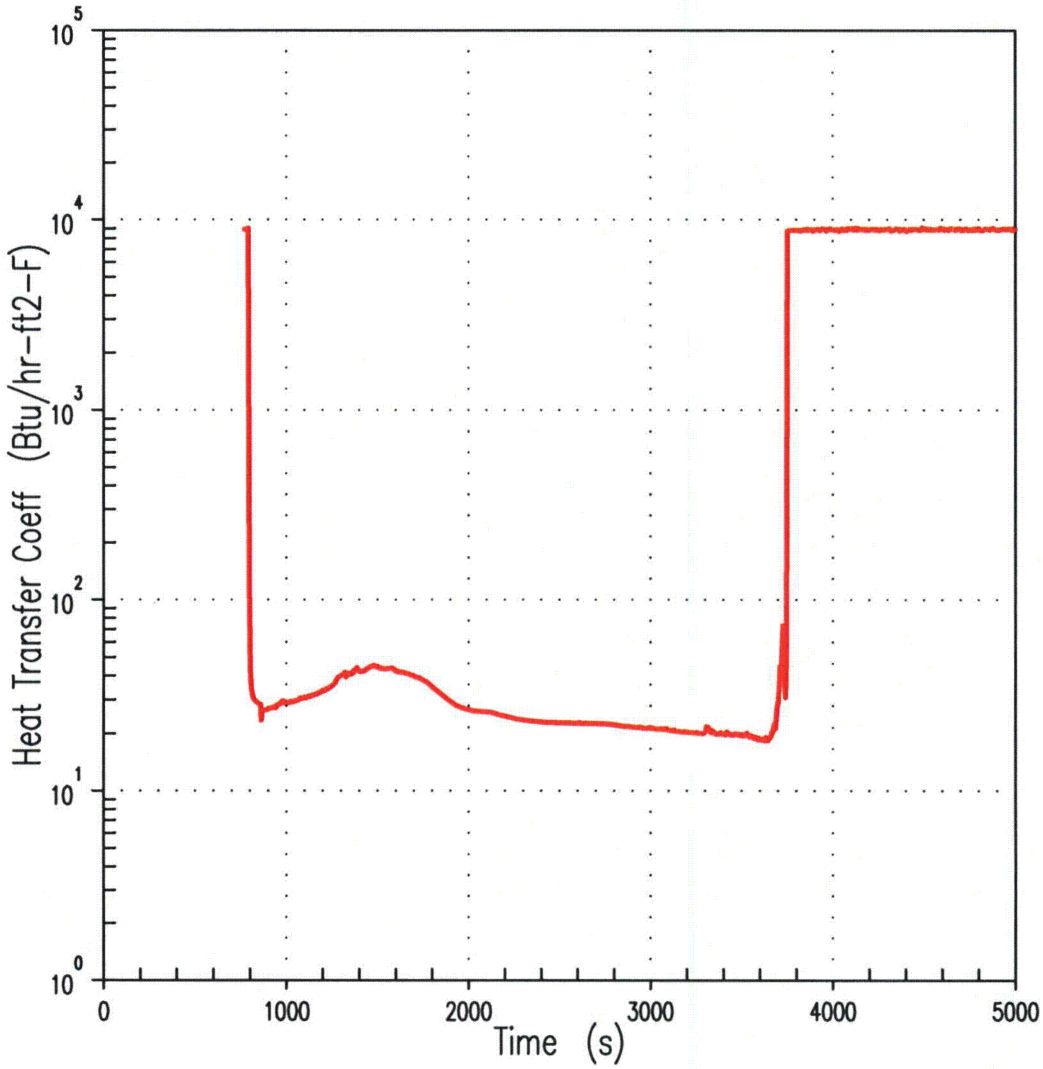

Figure 15
3.25-inch Break
Rod Film Heat Transfer Coefficient at PCT Elevation (11.75 ft)

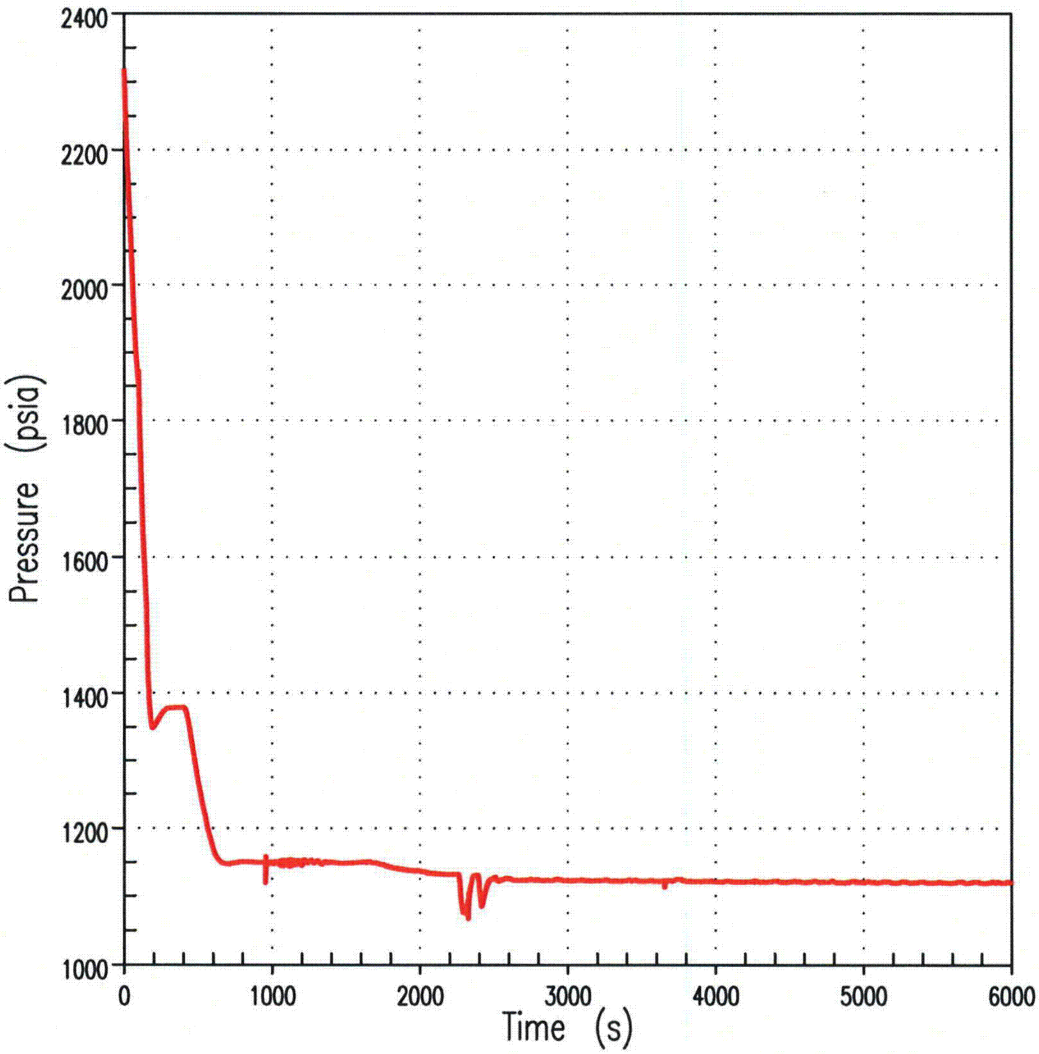

Figure 16
1.5-inch Break
RCS Pressure

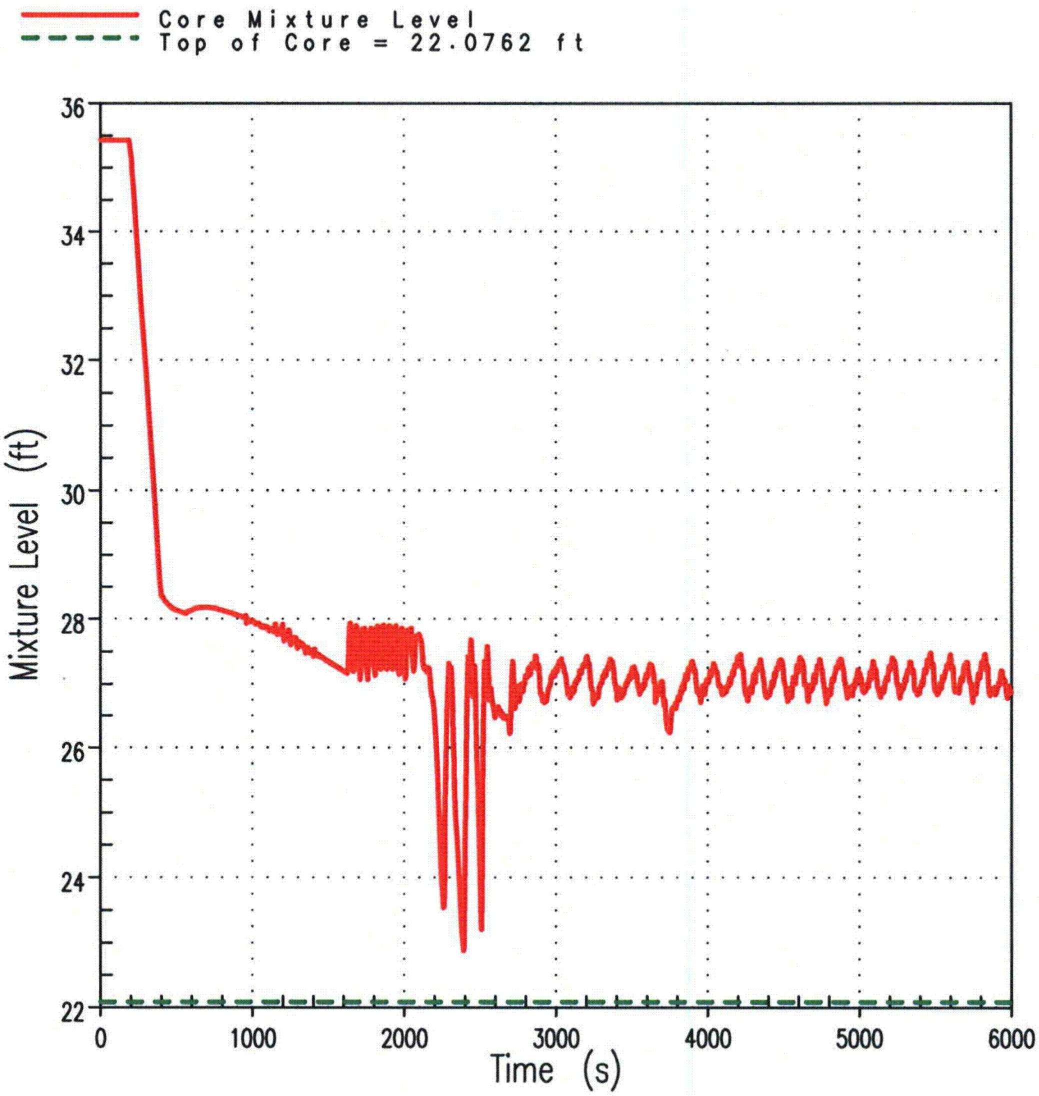

Figure 17
1.5-inch Break
Core Mixture Level

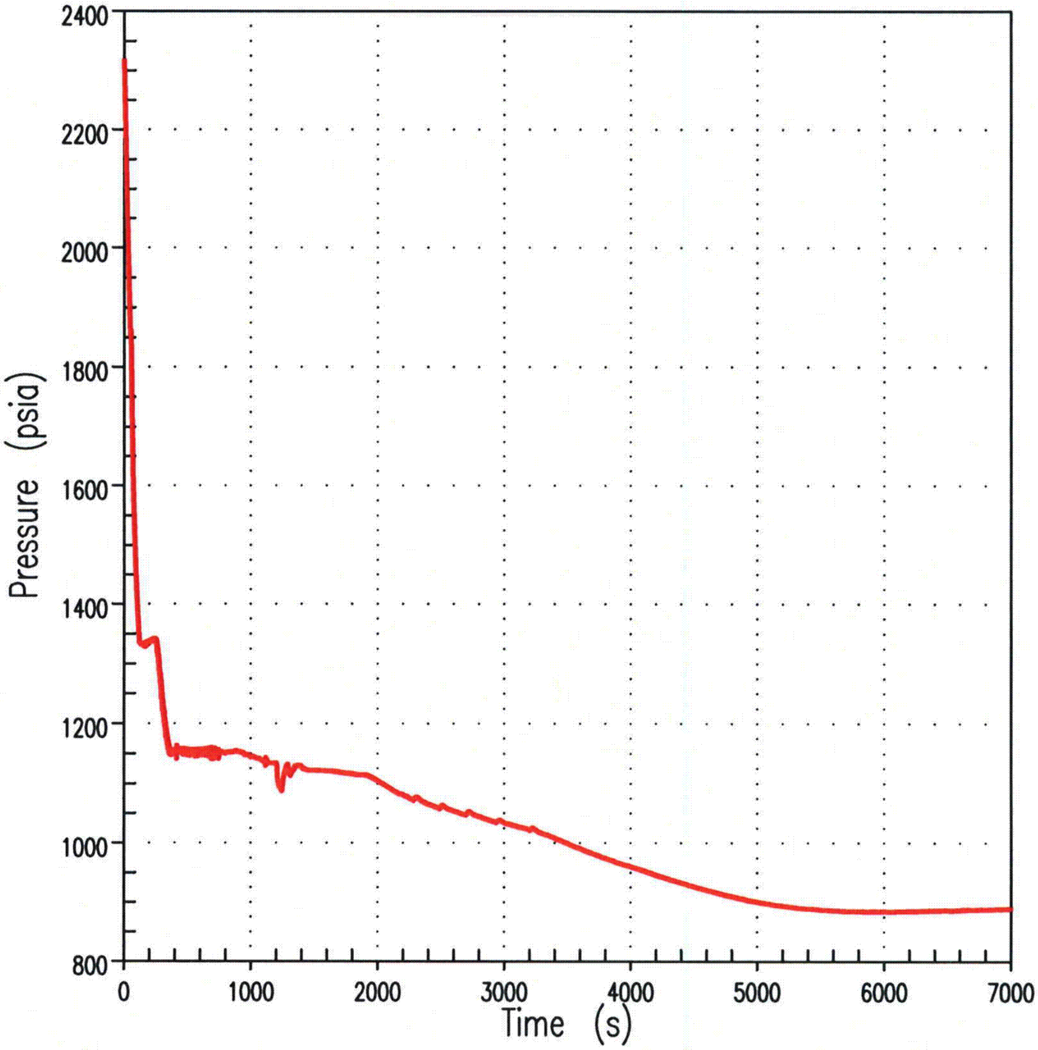

Figure 18
2-inch Break
RCS Pressure

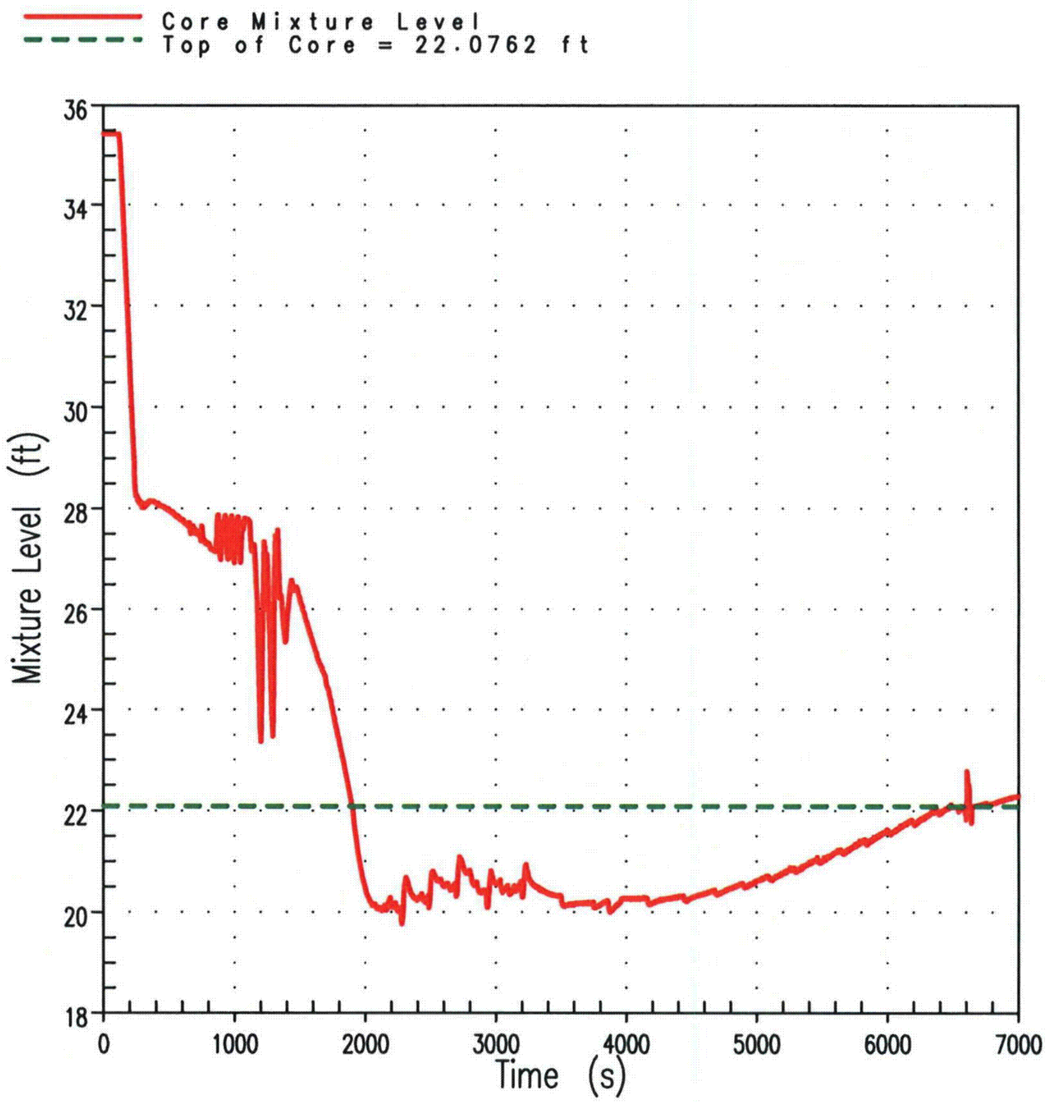

Figure 19
2-inch Break
Core Mixture Level

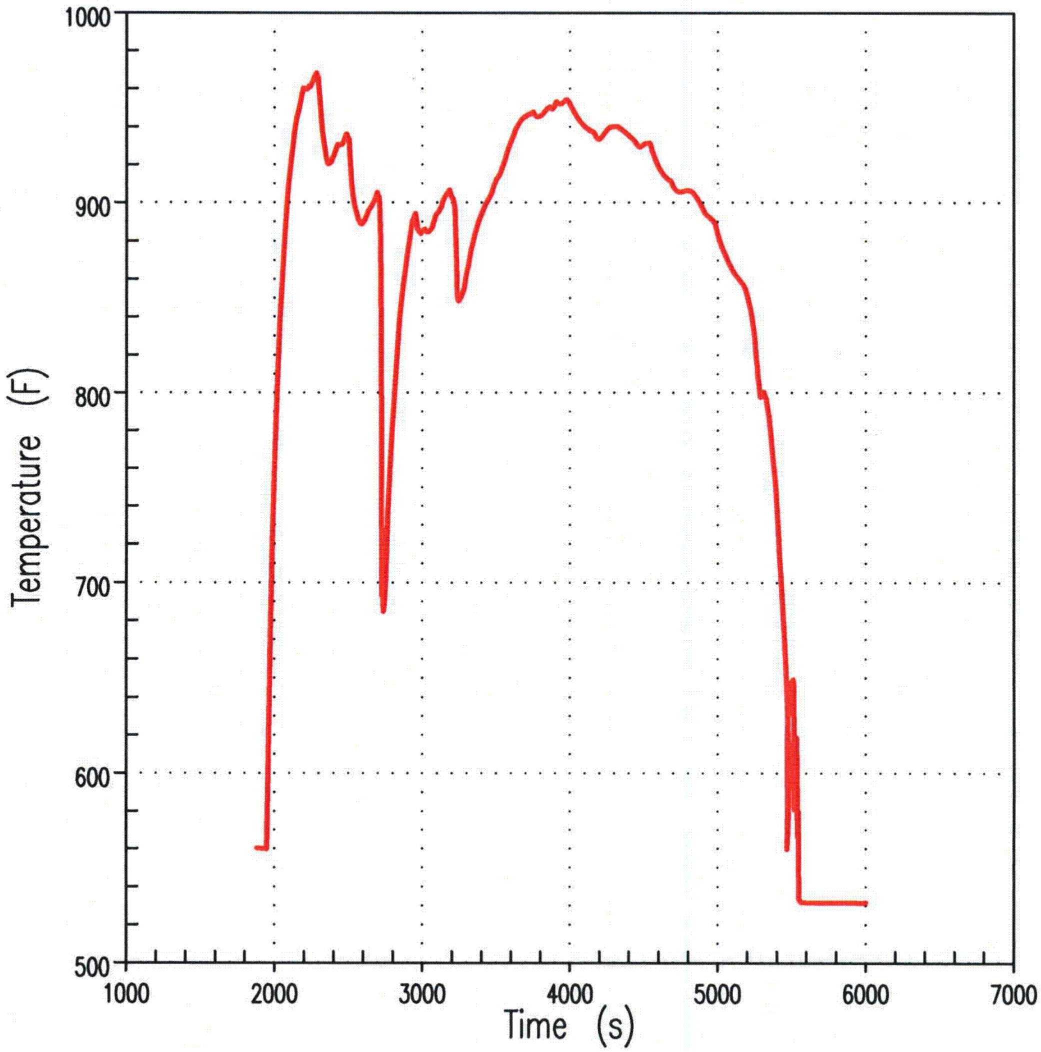

Figure 20
2-inch Break
Clad Temperature at PCT Elevation (11.0 ft)

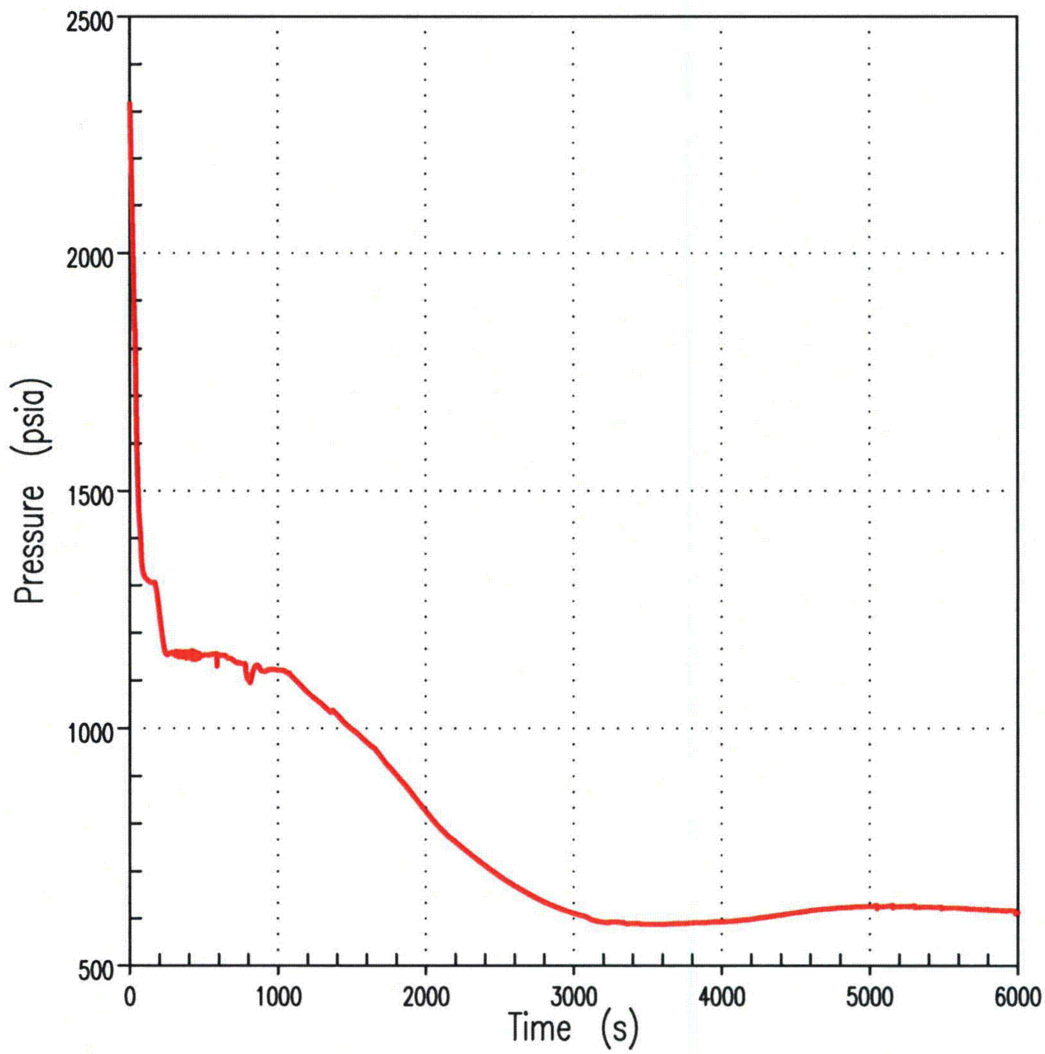

Figure 21
2.5-inch Break
RCS Pressure

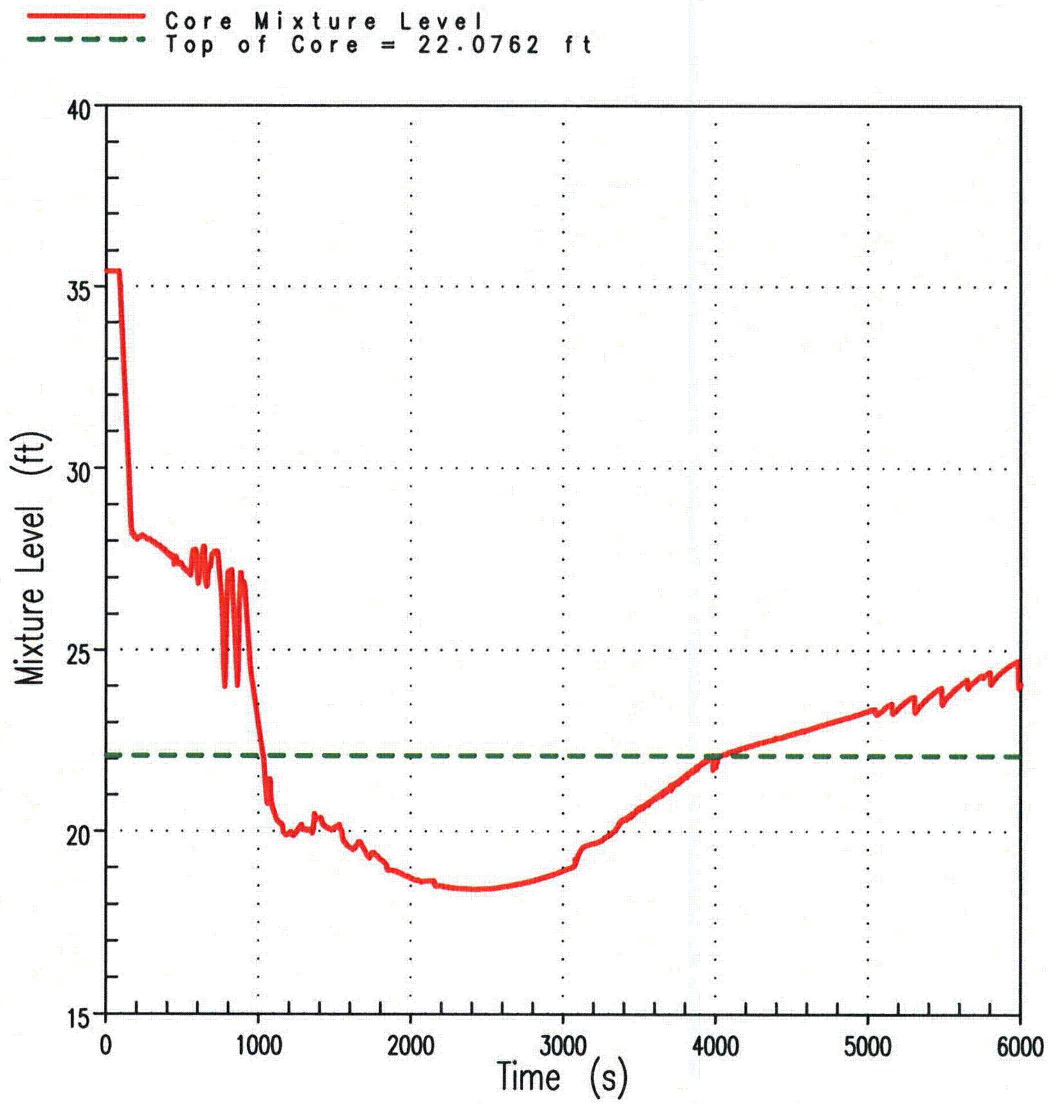

Figure 22
2.5-inch Break
Core Mixture Level

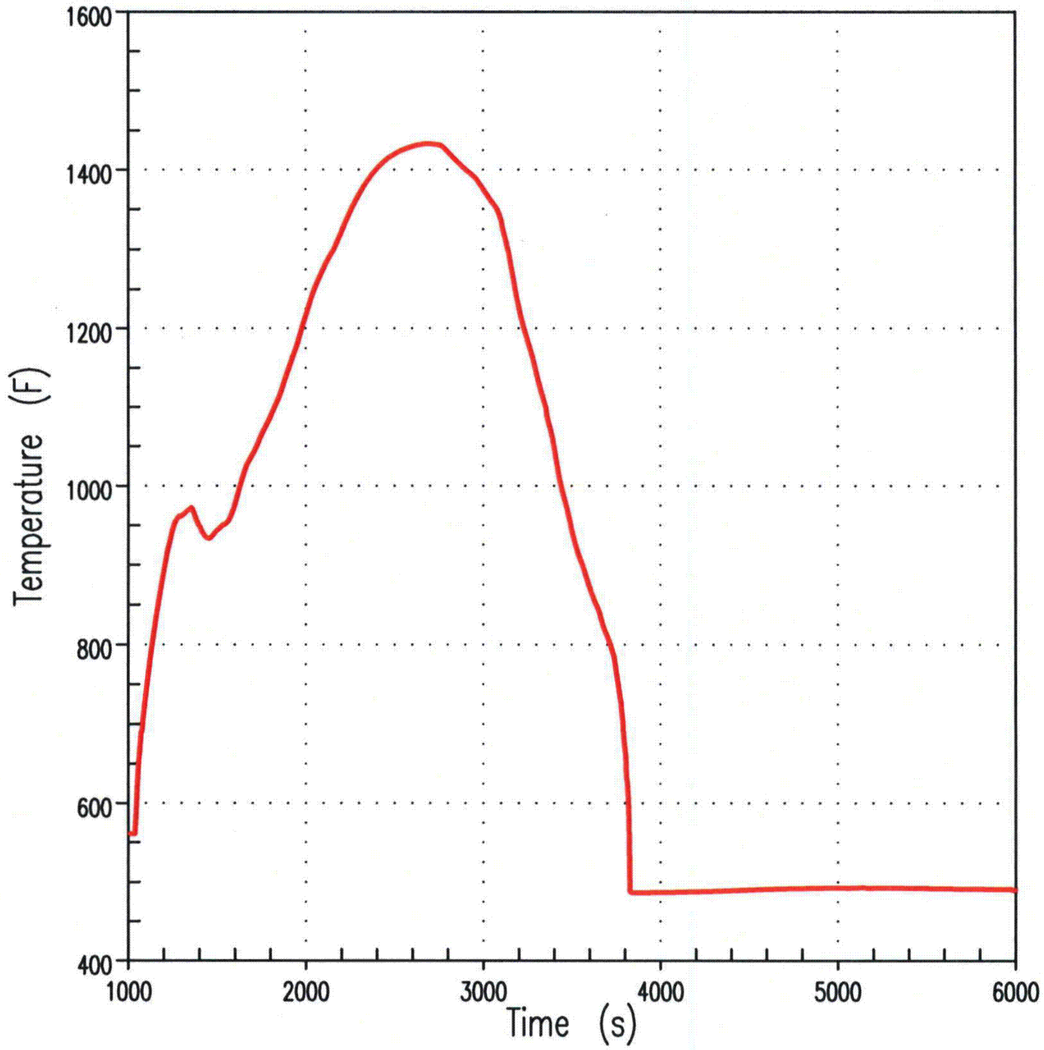

Figure 23
2.5-inch Break
Clad Temperature at PCT Elevation (11.5 ft)

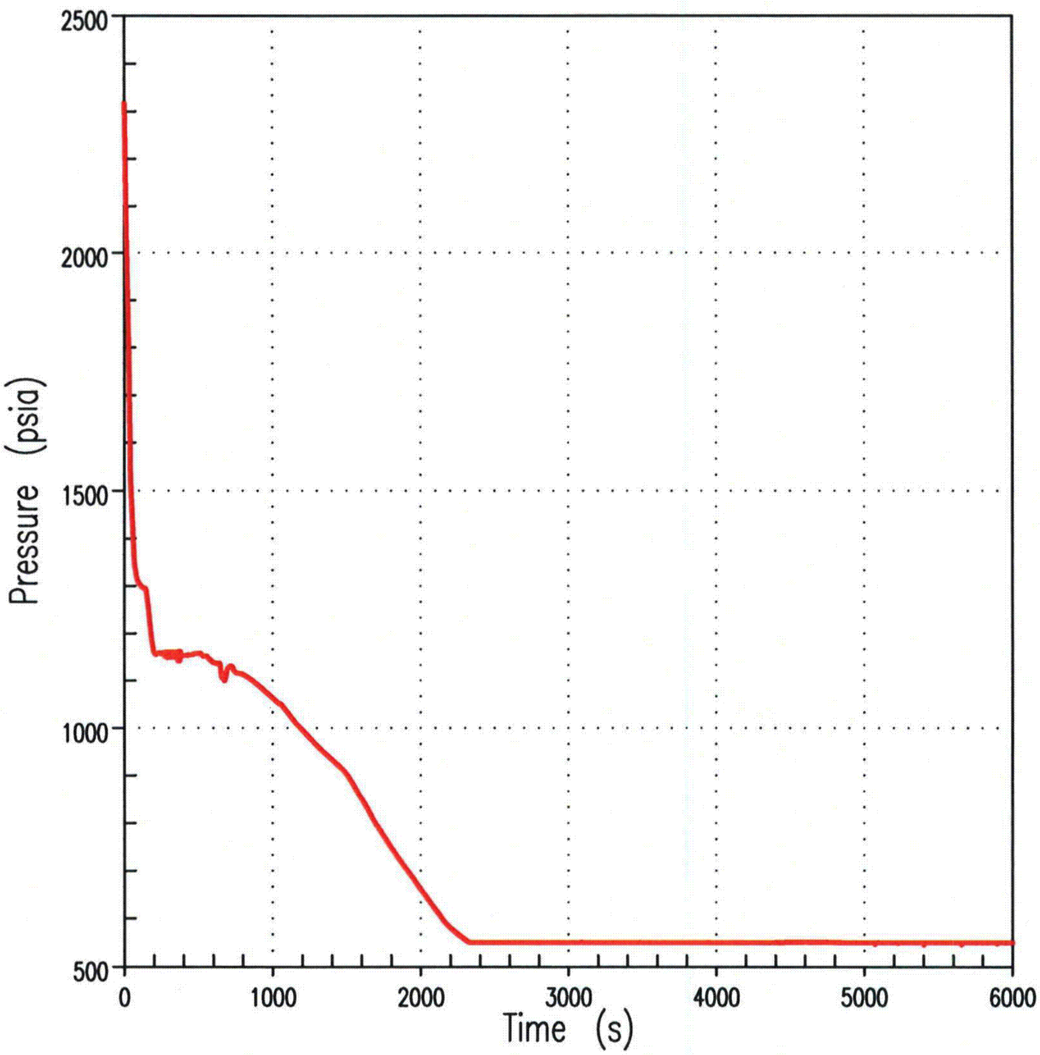

Figure 24
2.75-inch Break
RCS Pressure

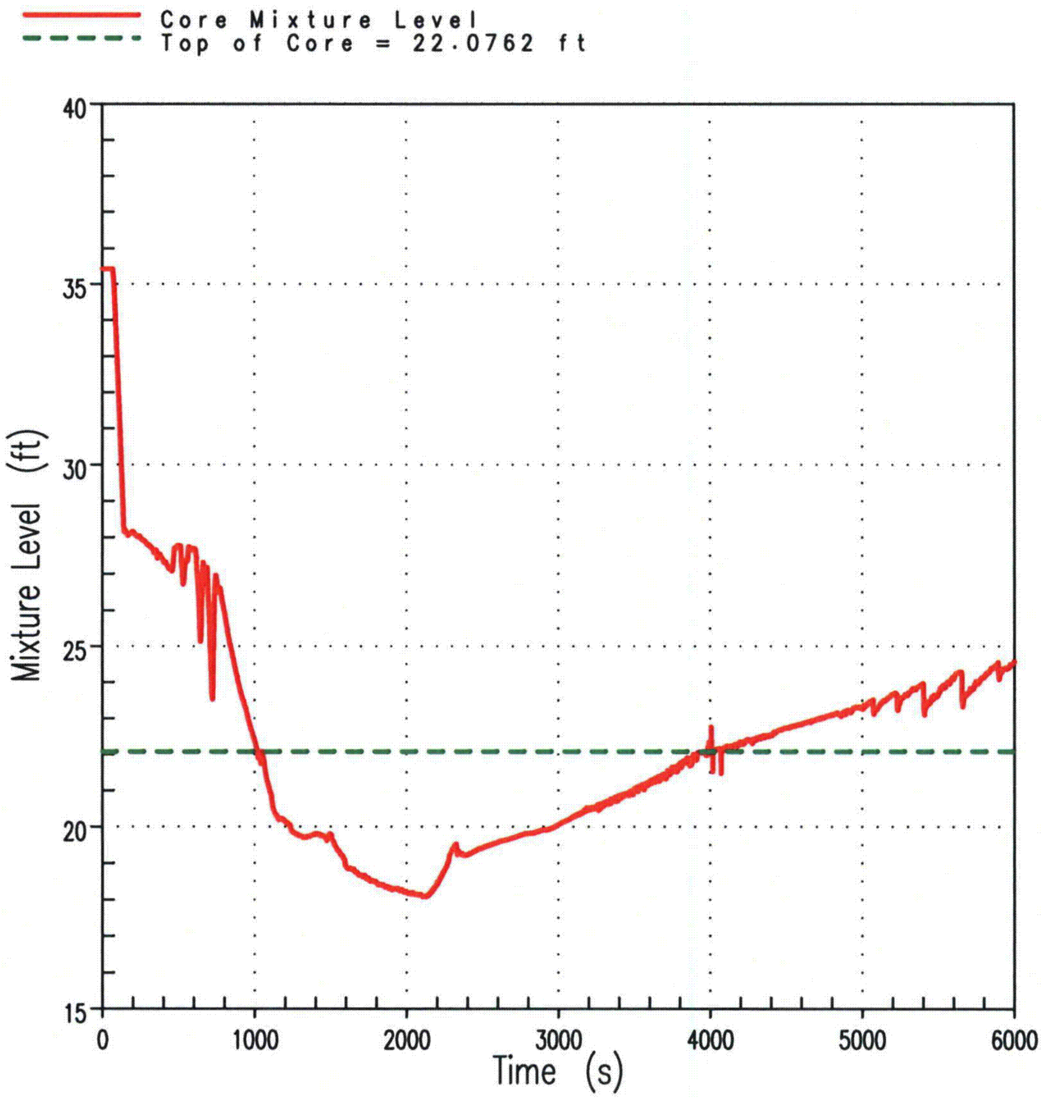

Figure 25
2.75-inch Break
Core Mixture Level

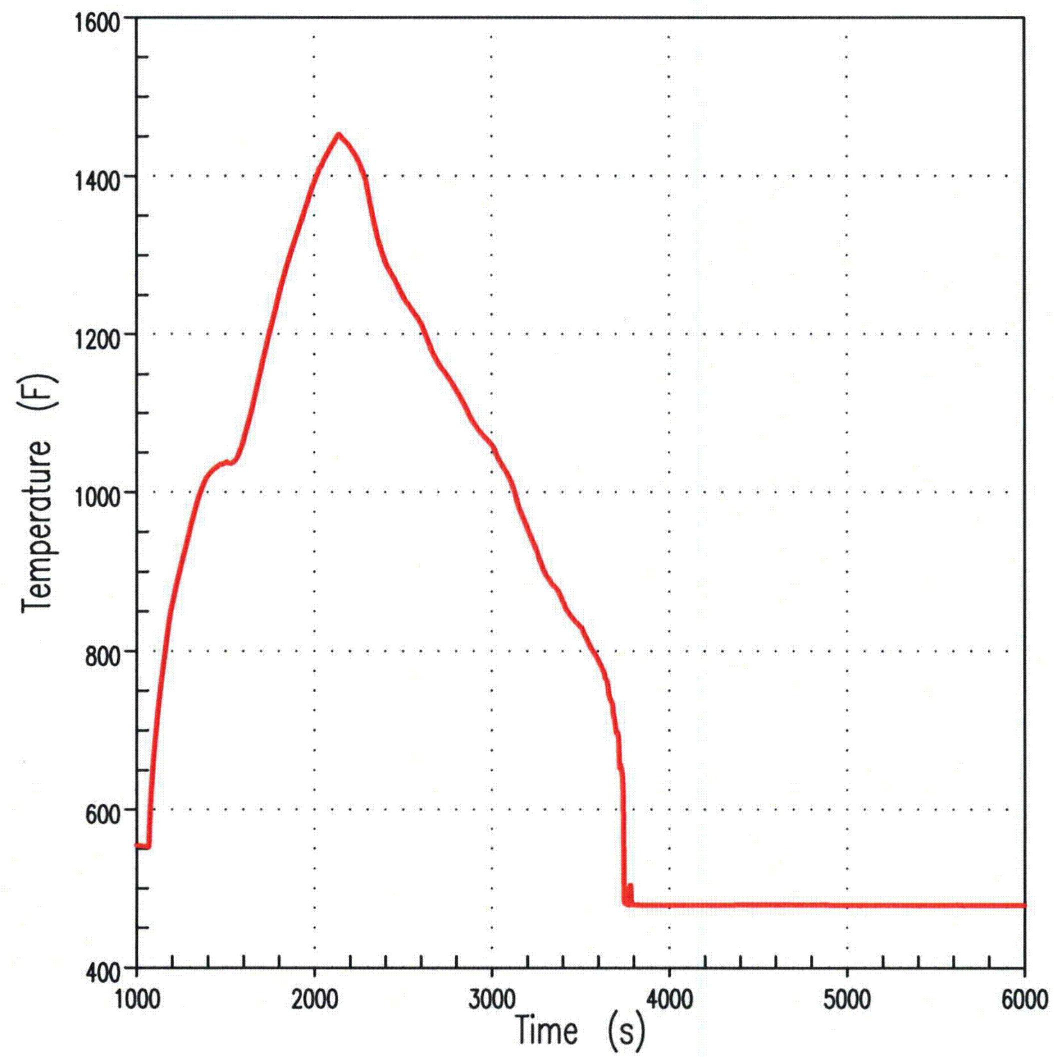

Figure 26
2.75-inch Break
Clad Temperature at PCT Elevation (11.5 ft)

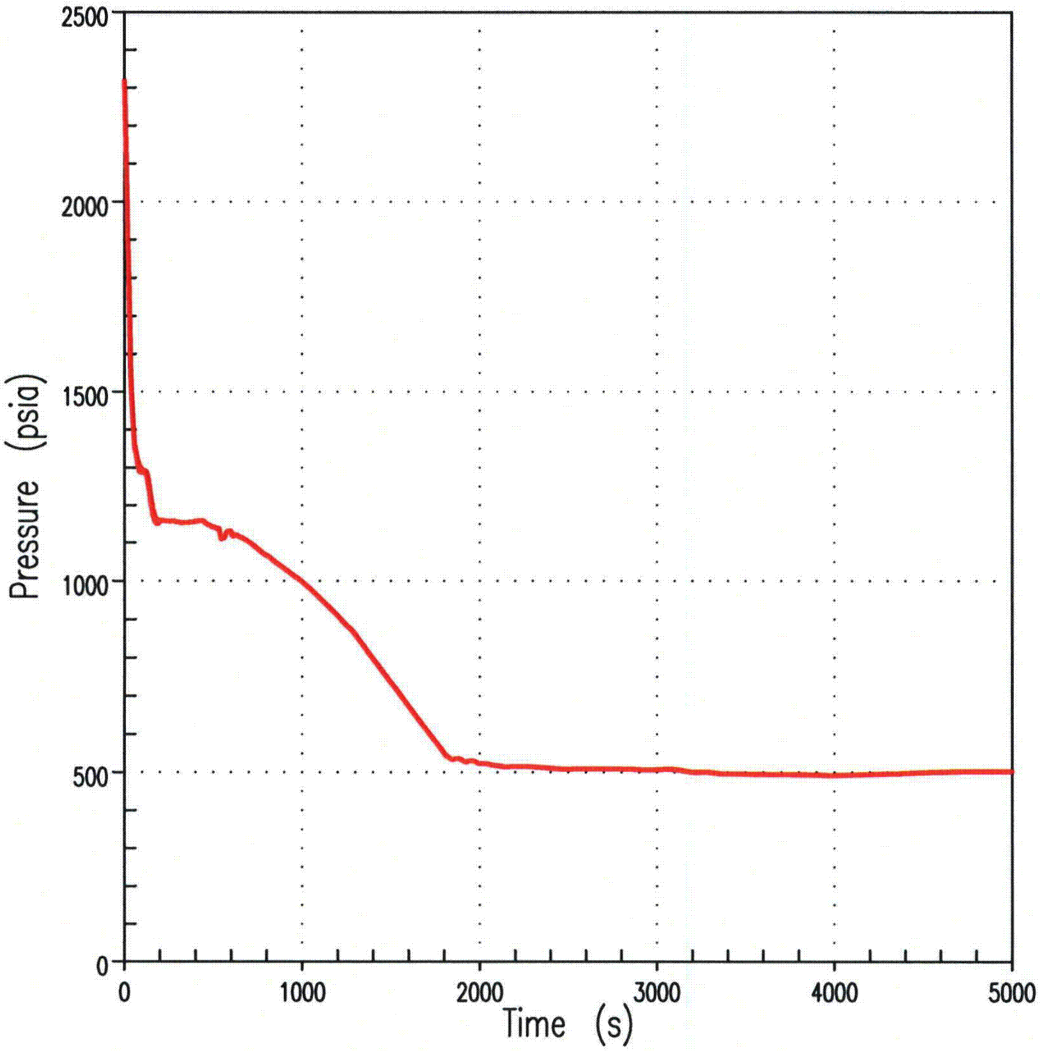

Figure 27
3-inch Break
RCS Pressure

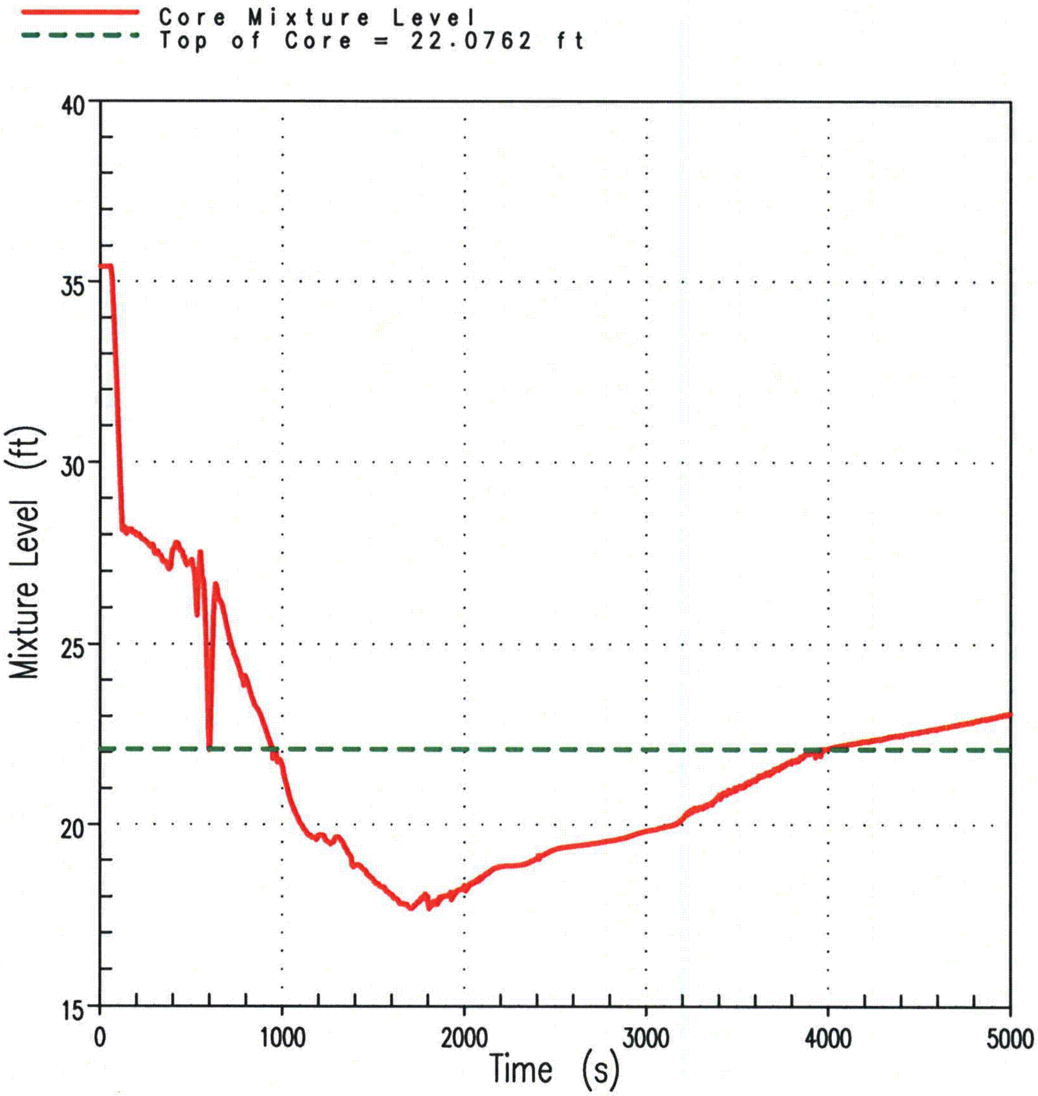

Figure 28
3-inch Break
Core Mixture Level

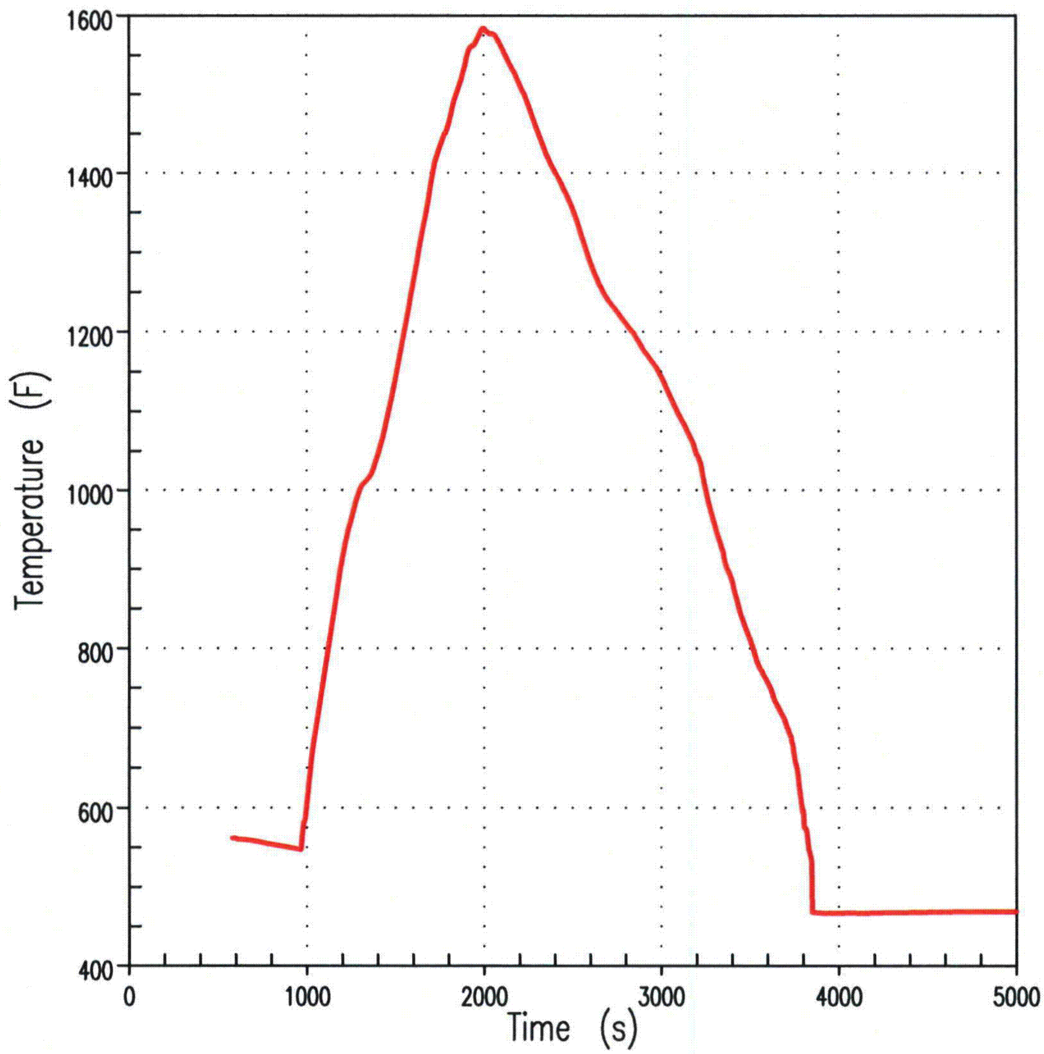

Figure 29
3-inch Break
Clad Temperature at PCT Elevation (11.75 ft)

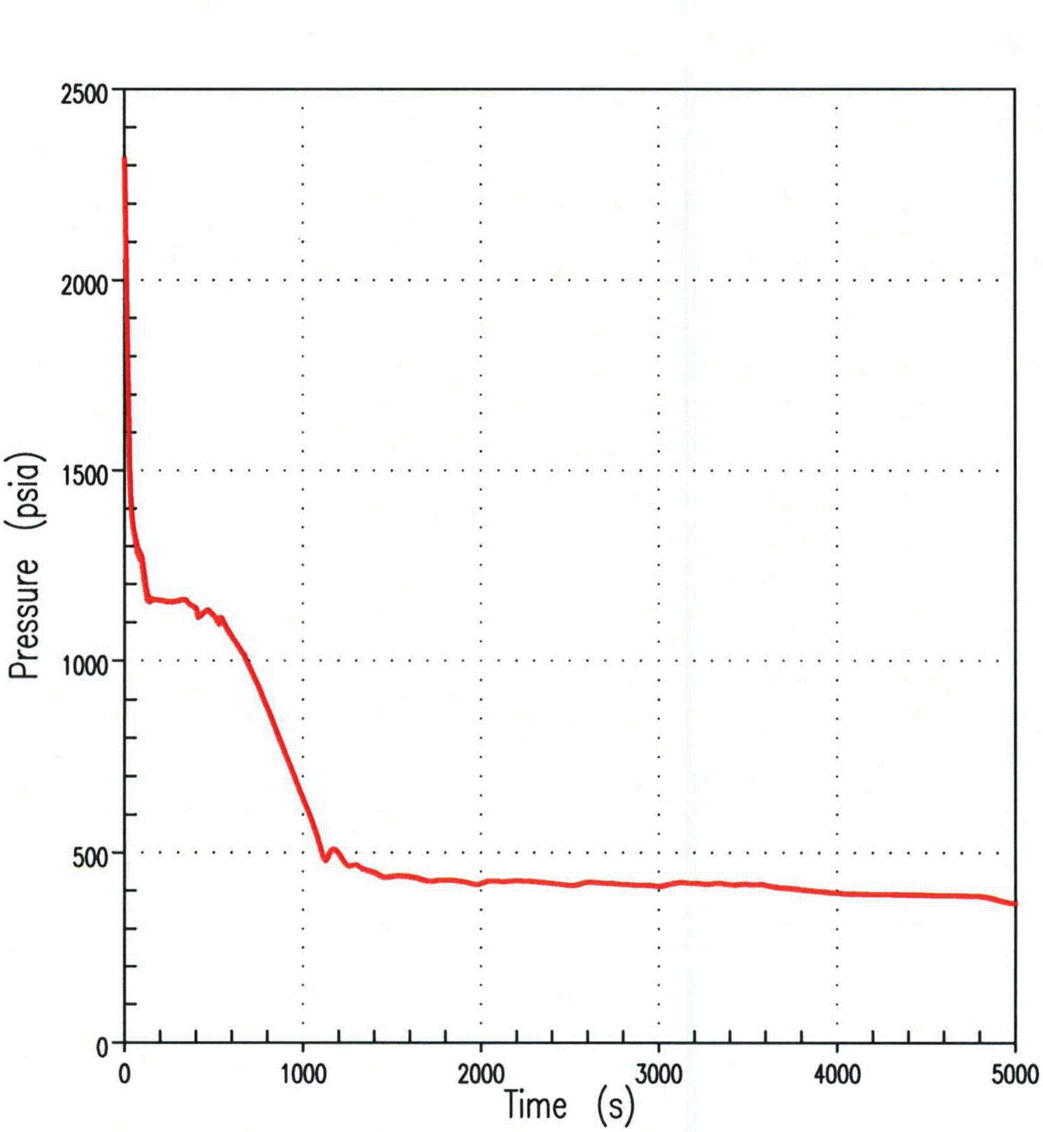

Figure 30
3.5-inch Break
RCS Pressure

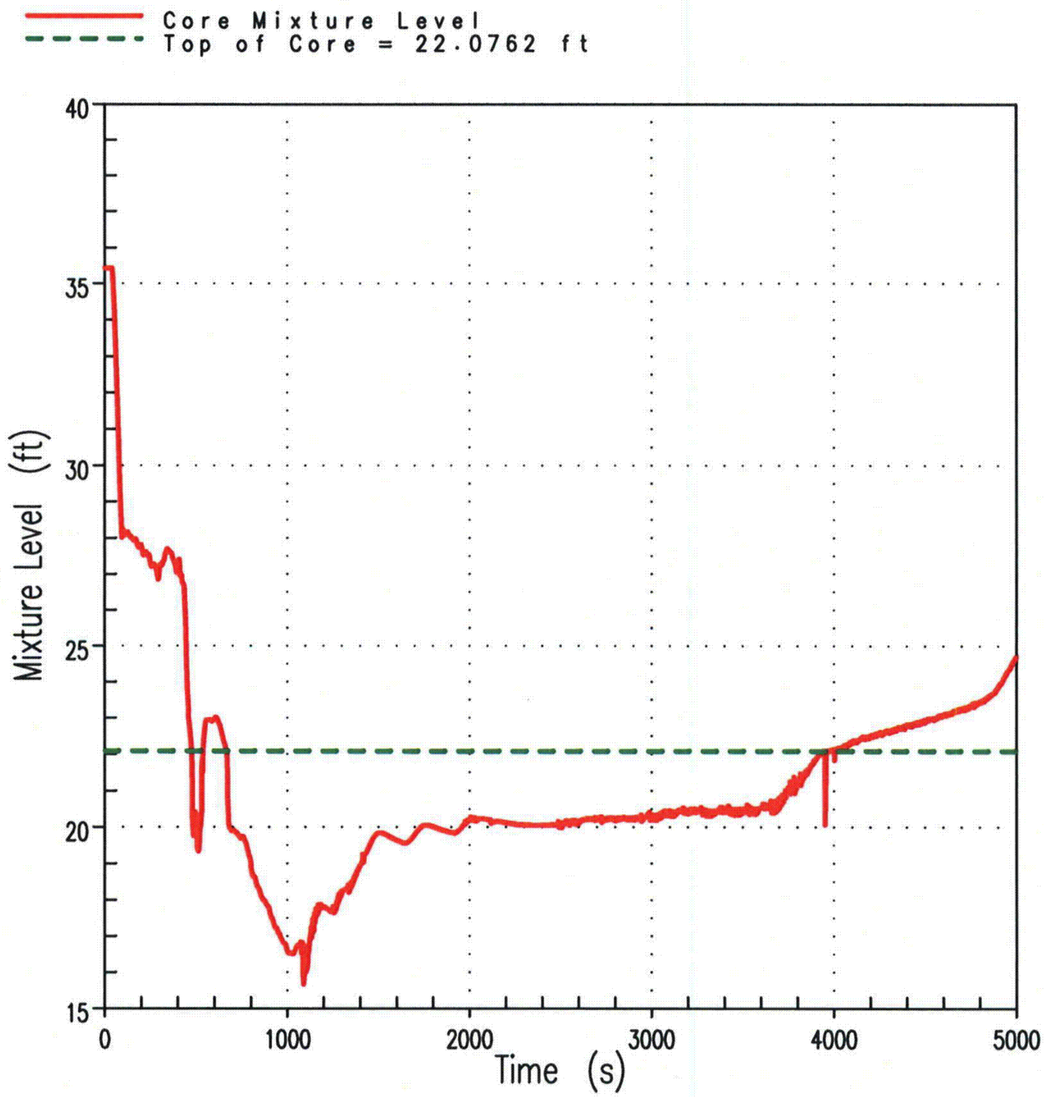

Figure 31
3.5-inch Break
Core Mixture Level

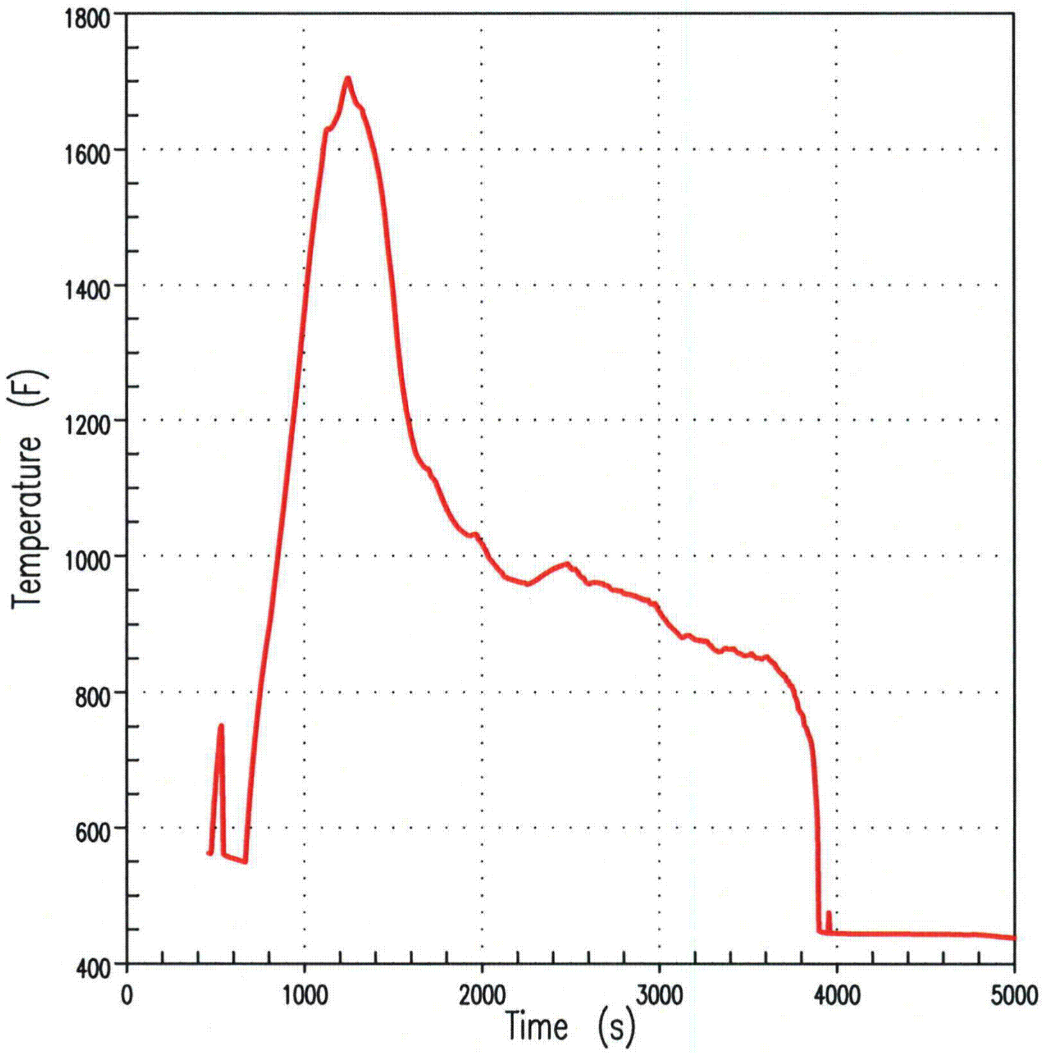

Figure 32
3.5-inch Break
Clad Temperature at PCT Elevation (11.75 ft)

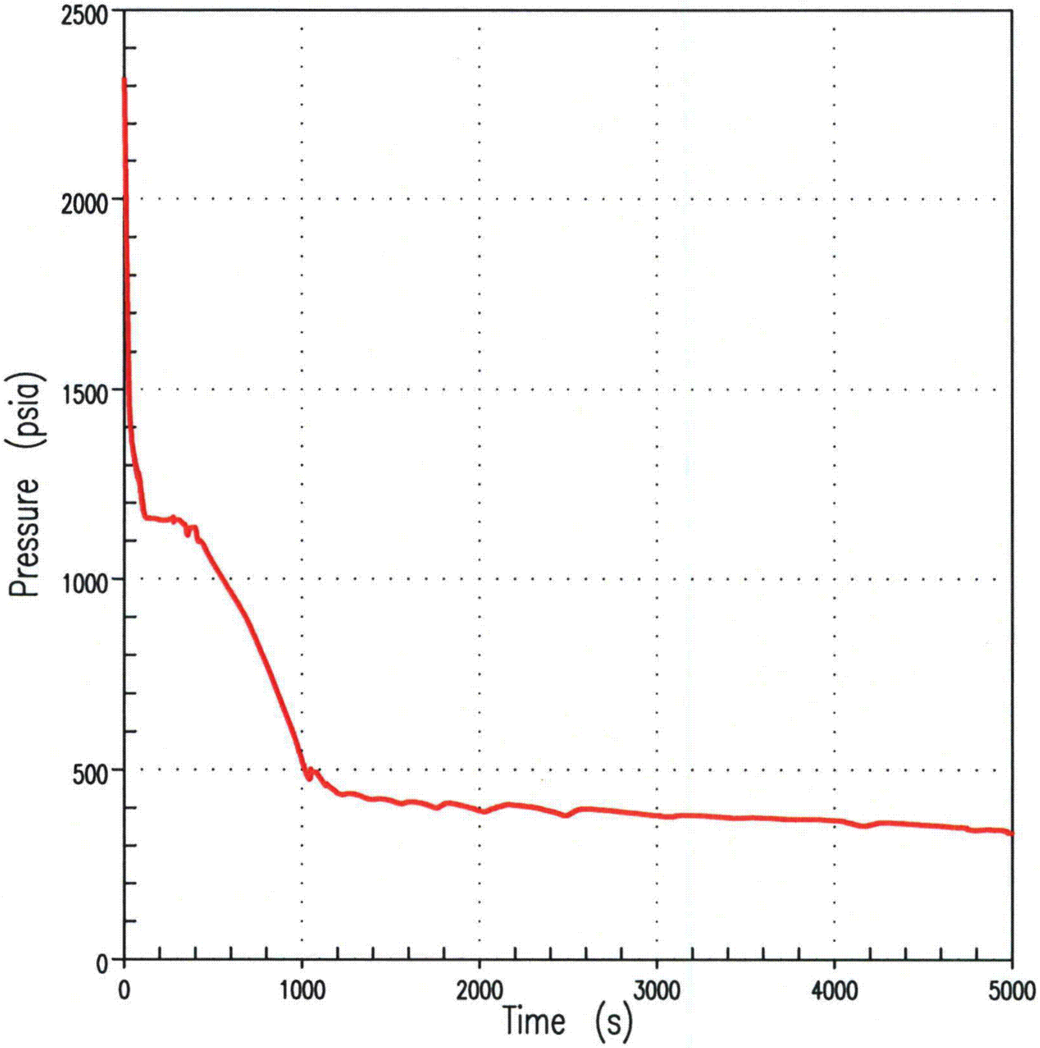

Figure 33
3.75-inch Break
RCS Pressure

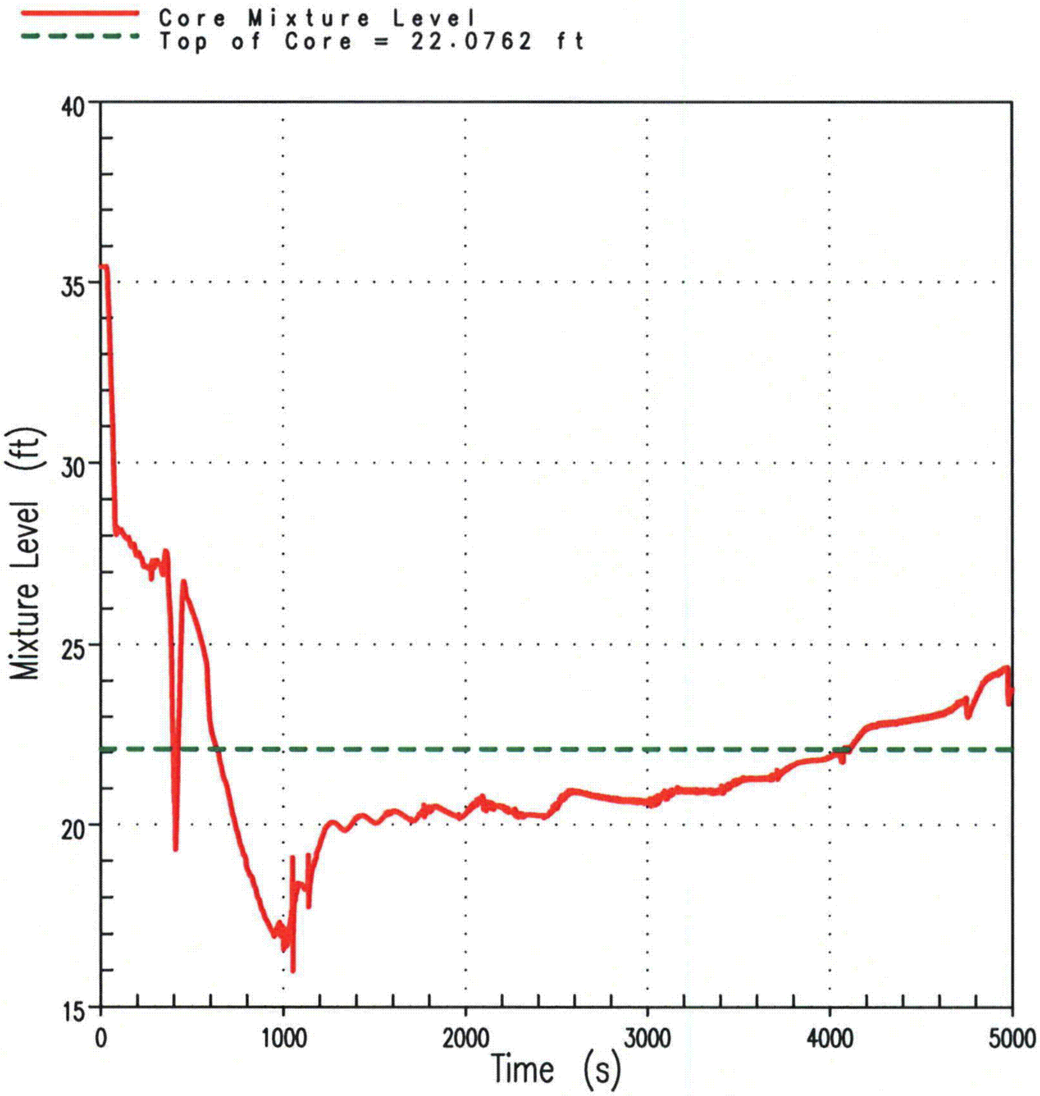

Figure 34
3.75-inch Break
Core Mixture Level

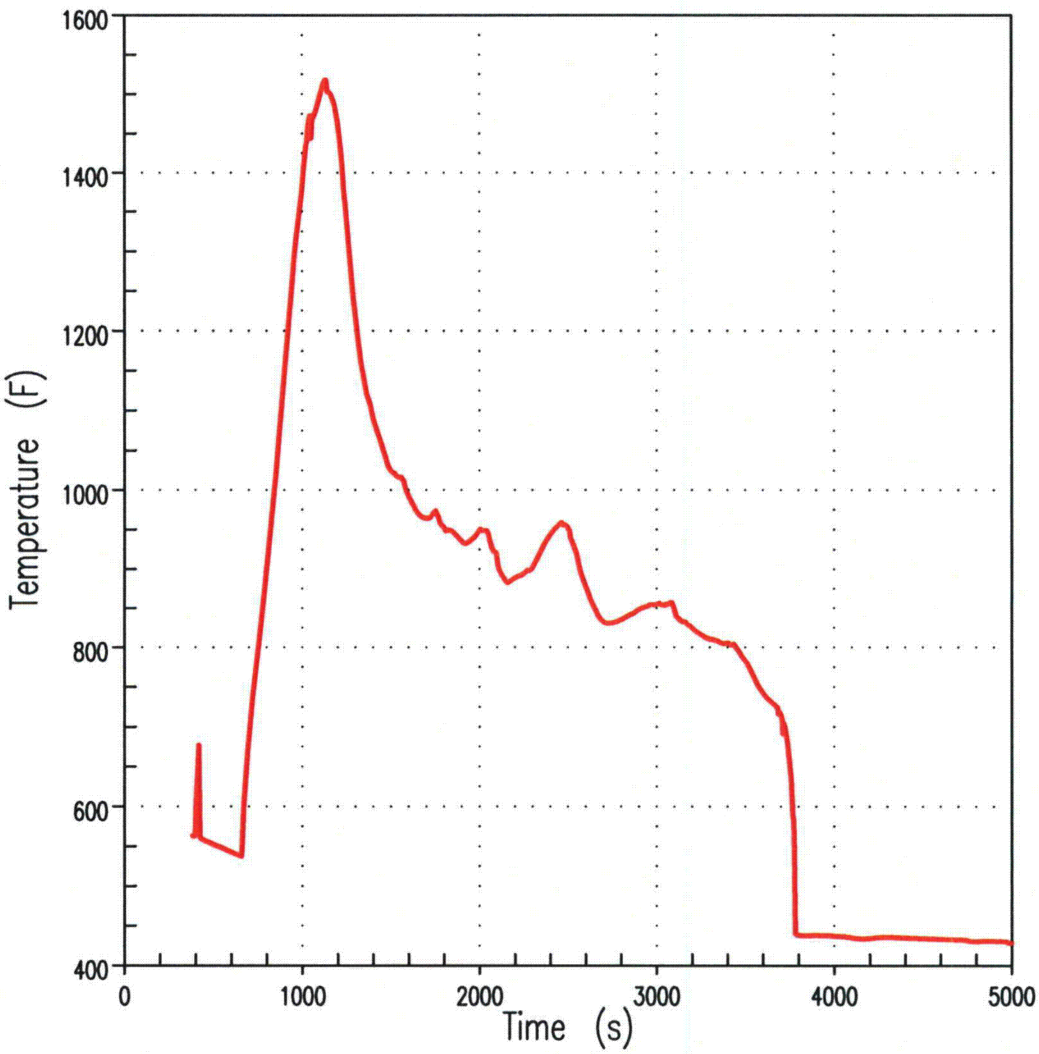

Figure 35
3.75-inch Break
Clad Temperature at PCT Elevation (11.5 ft)

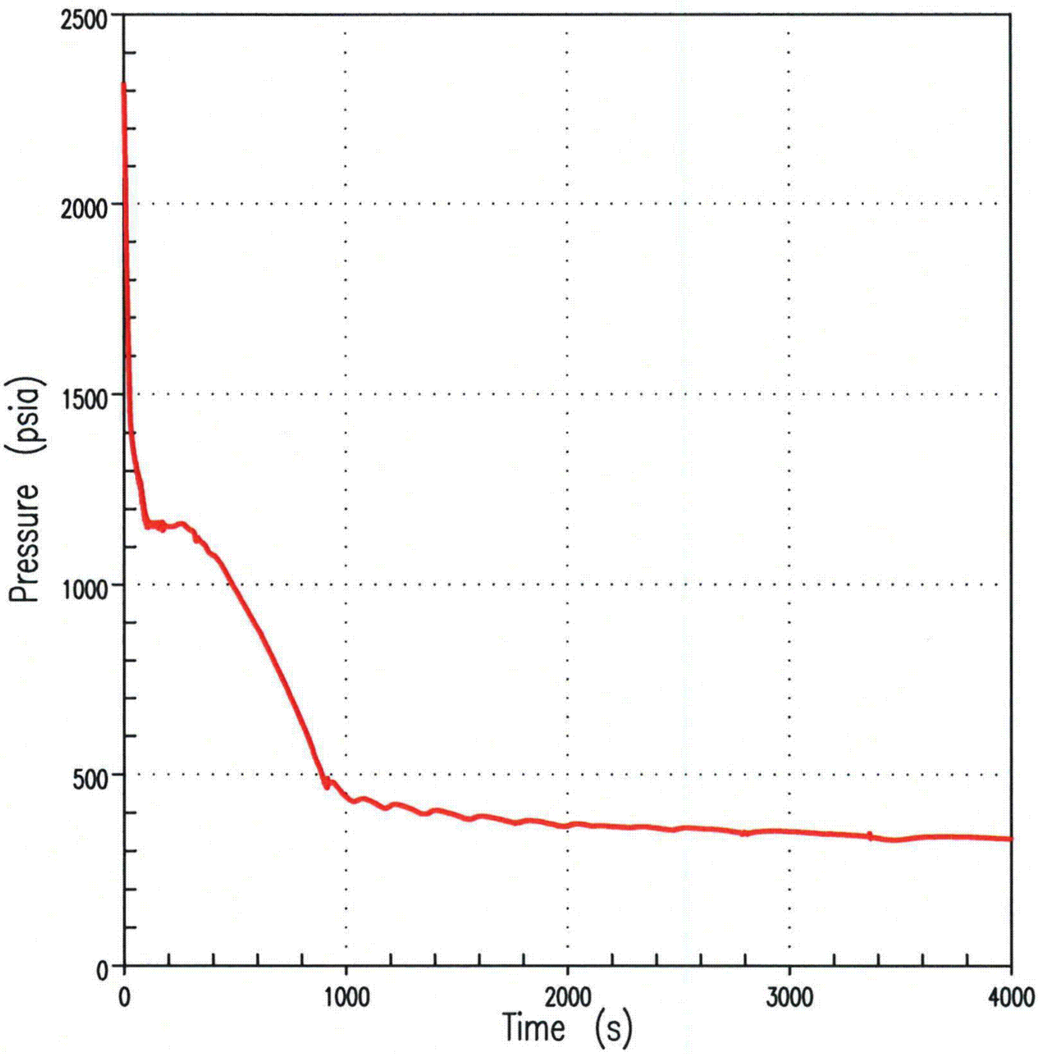

Figure 36
4-inch Break
RCS Pressure

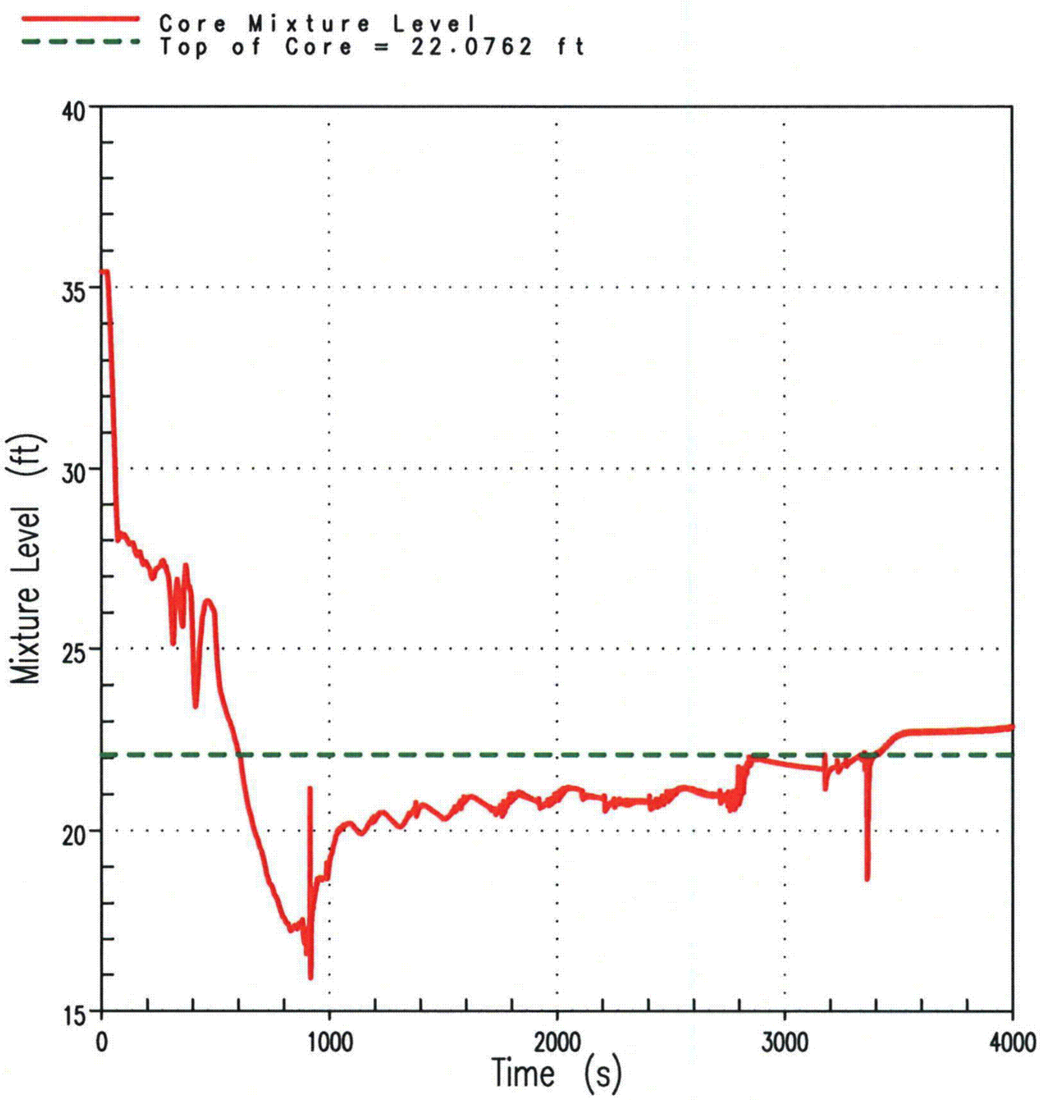

Figure 37
4-inch Break
Core Mixture Level

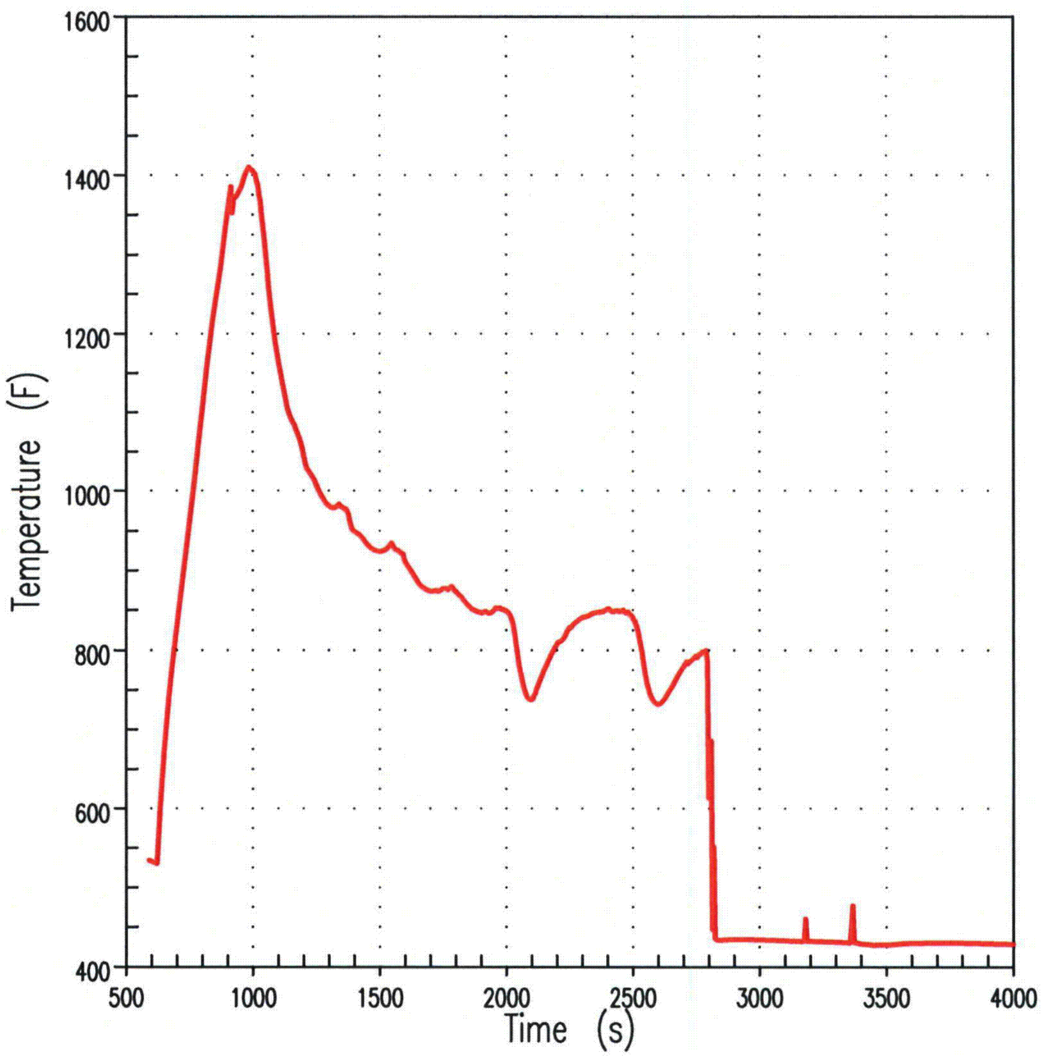

Figure 38
4-inch Break
Clad Temperature at PCT Elevation (11.25 ft)

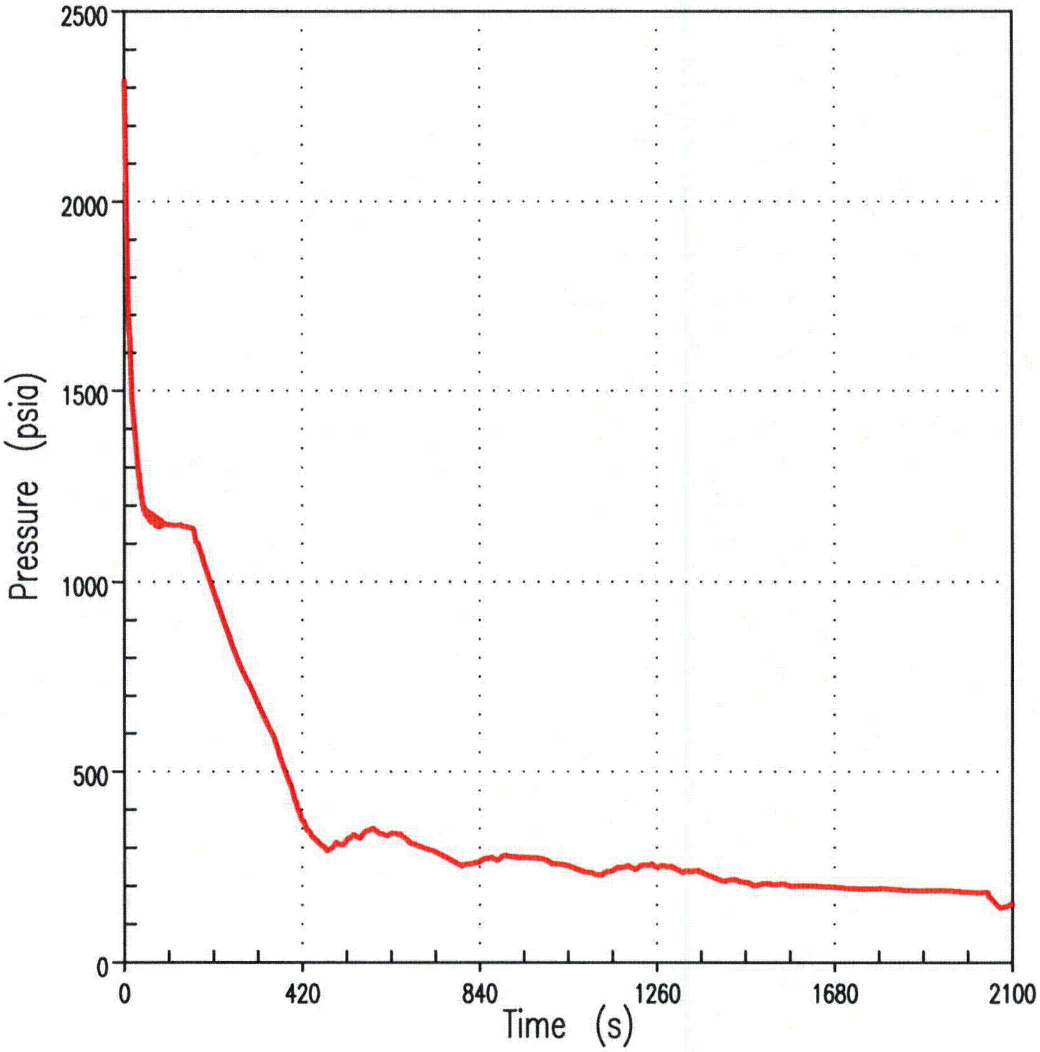

Figure 39
6-inch Break
RCS Pressure

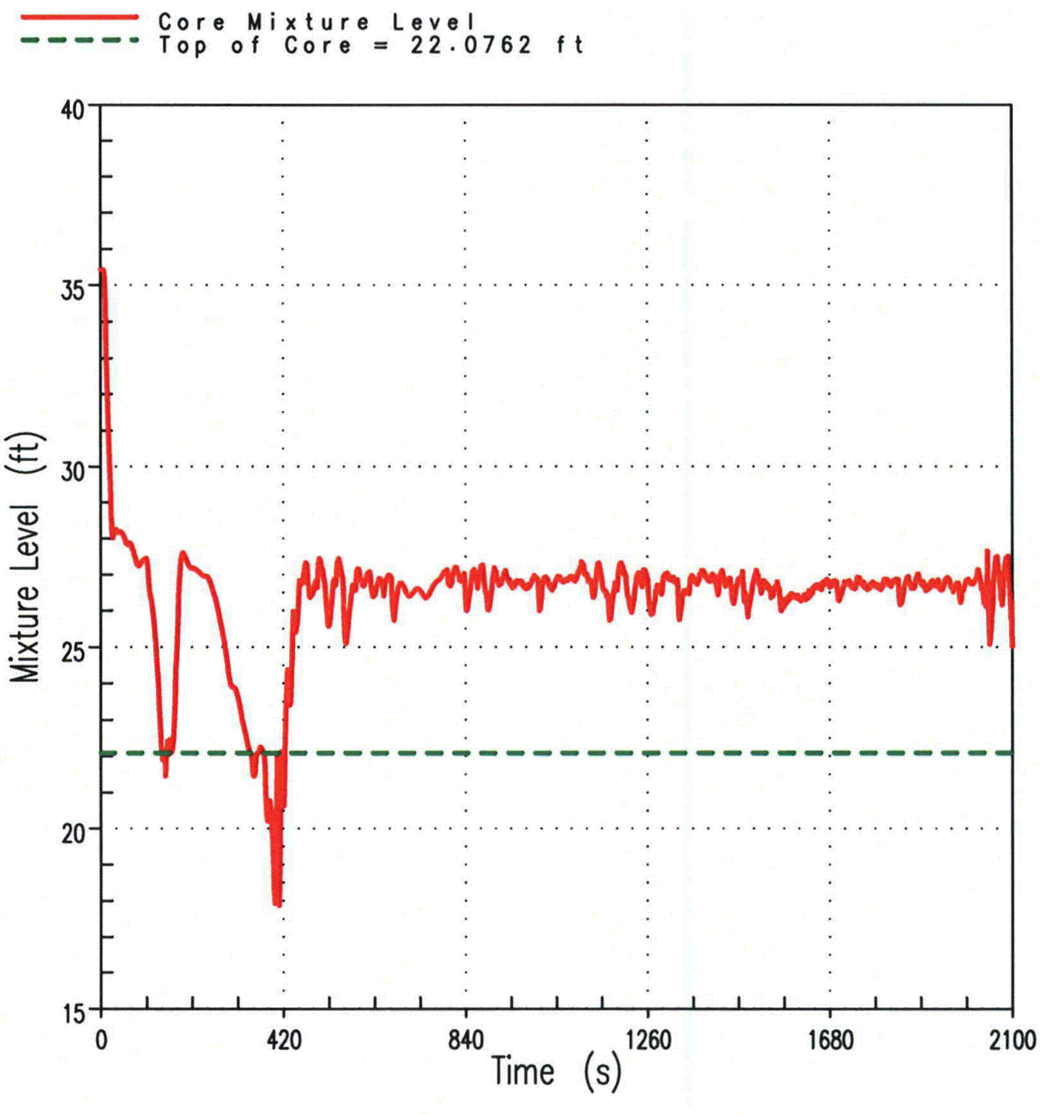

Figure 40
6-inch Break
Core Mixture Level

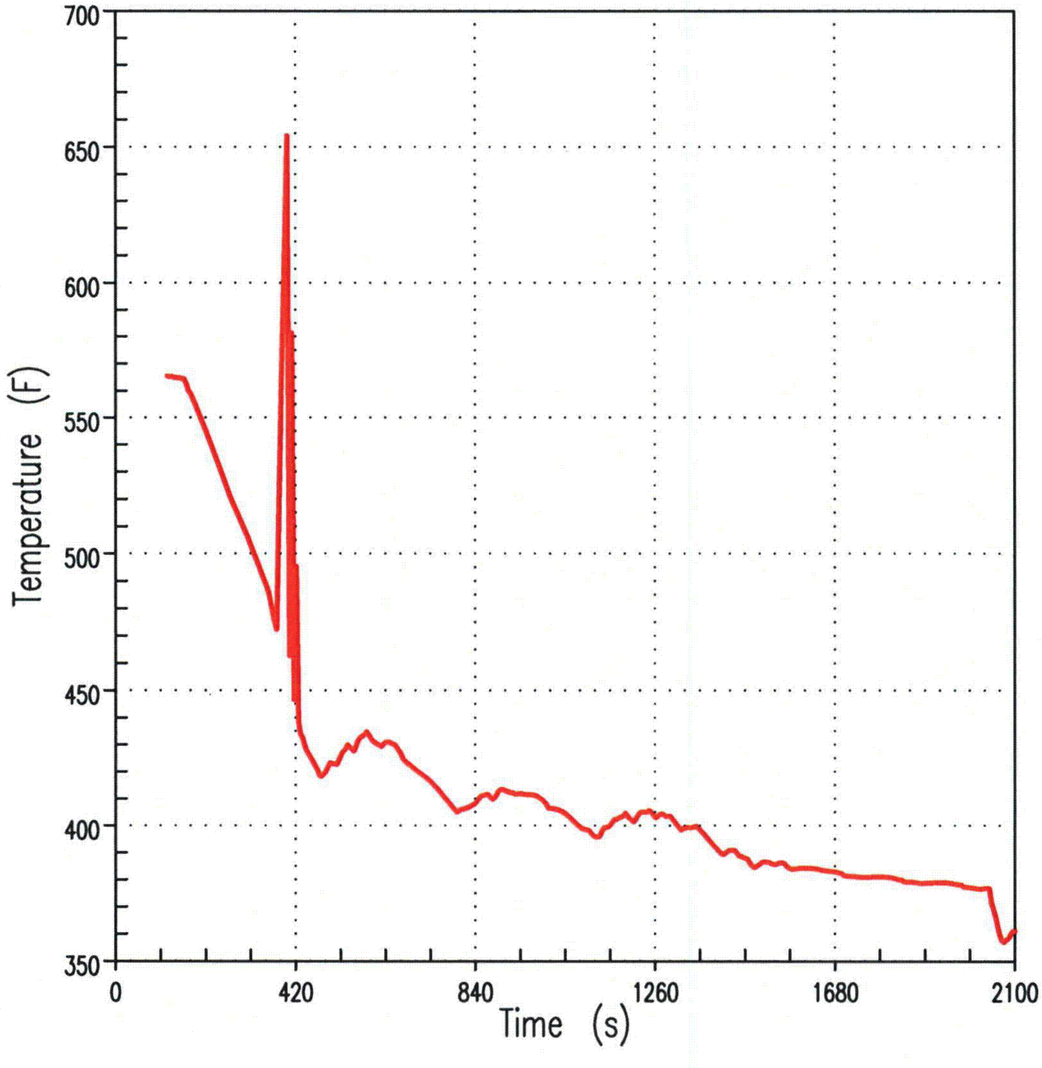

Figure 41
6-inch Break
Clad Temperature at PCT Elevation (11.0 ft)

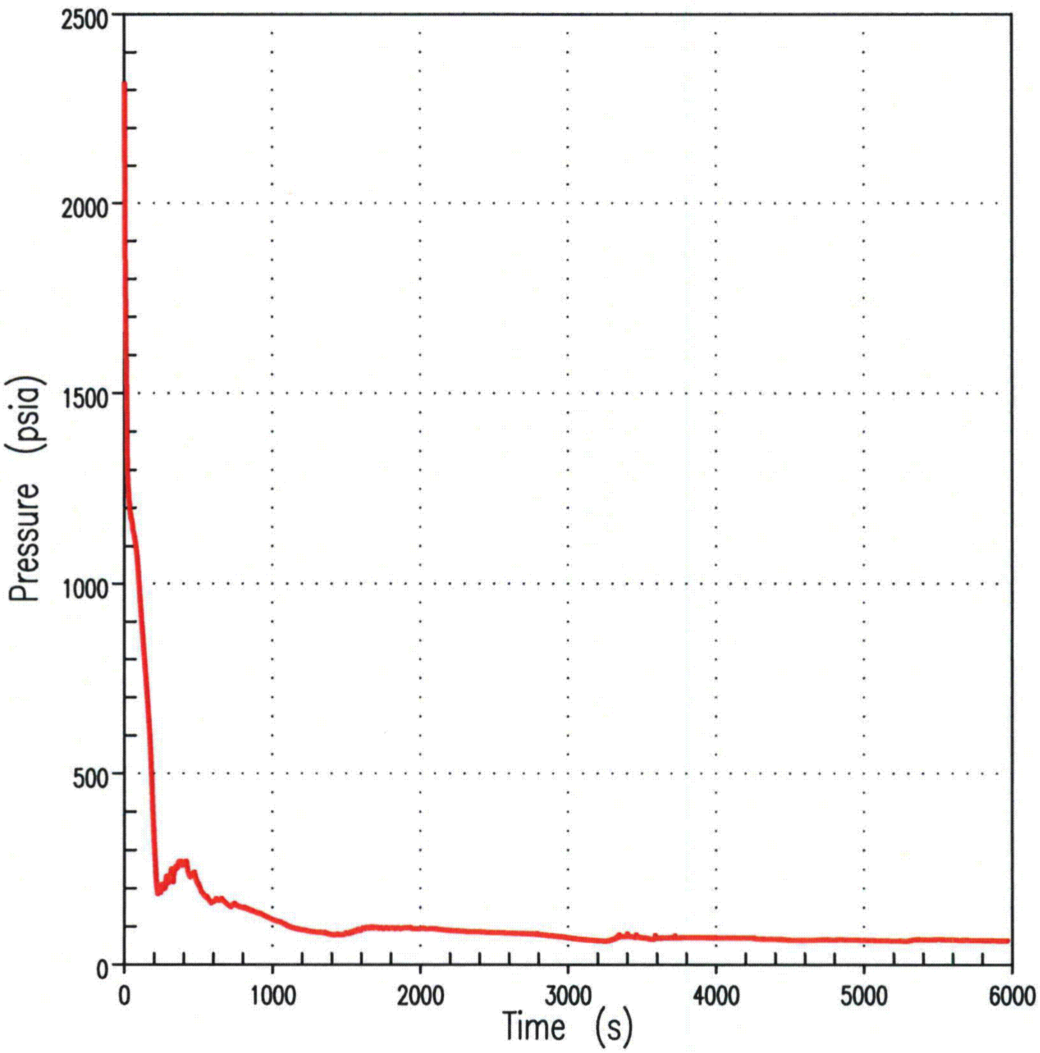

Figure 42
8.75-inch Break
RCS Pressure

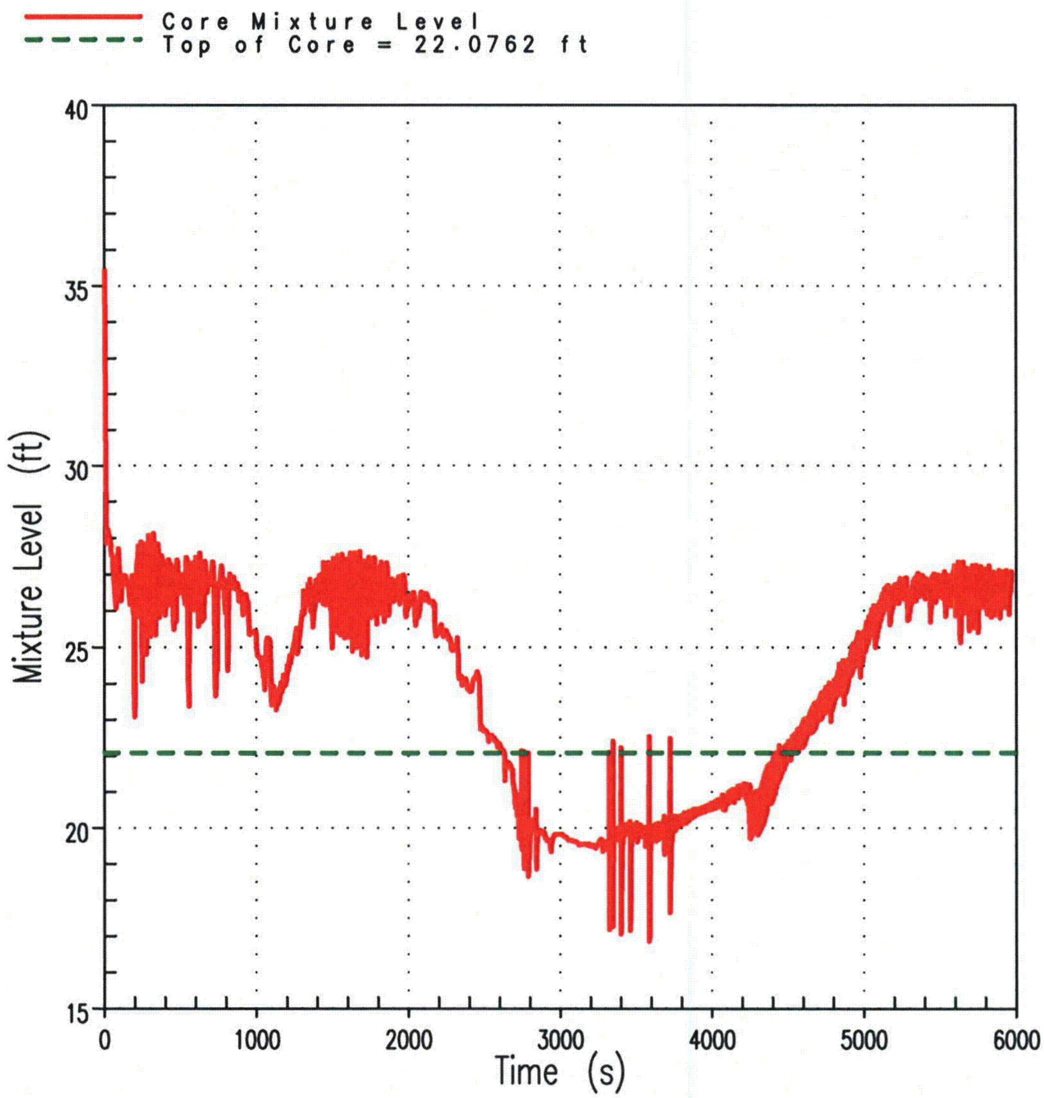

Figure 43
8.75-inch Break
Core Mixture Level

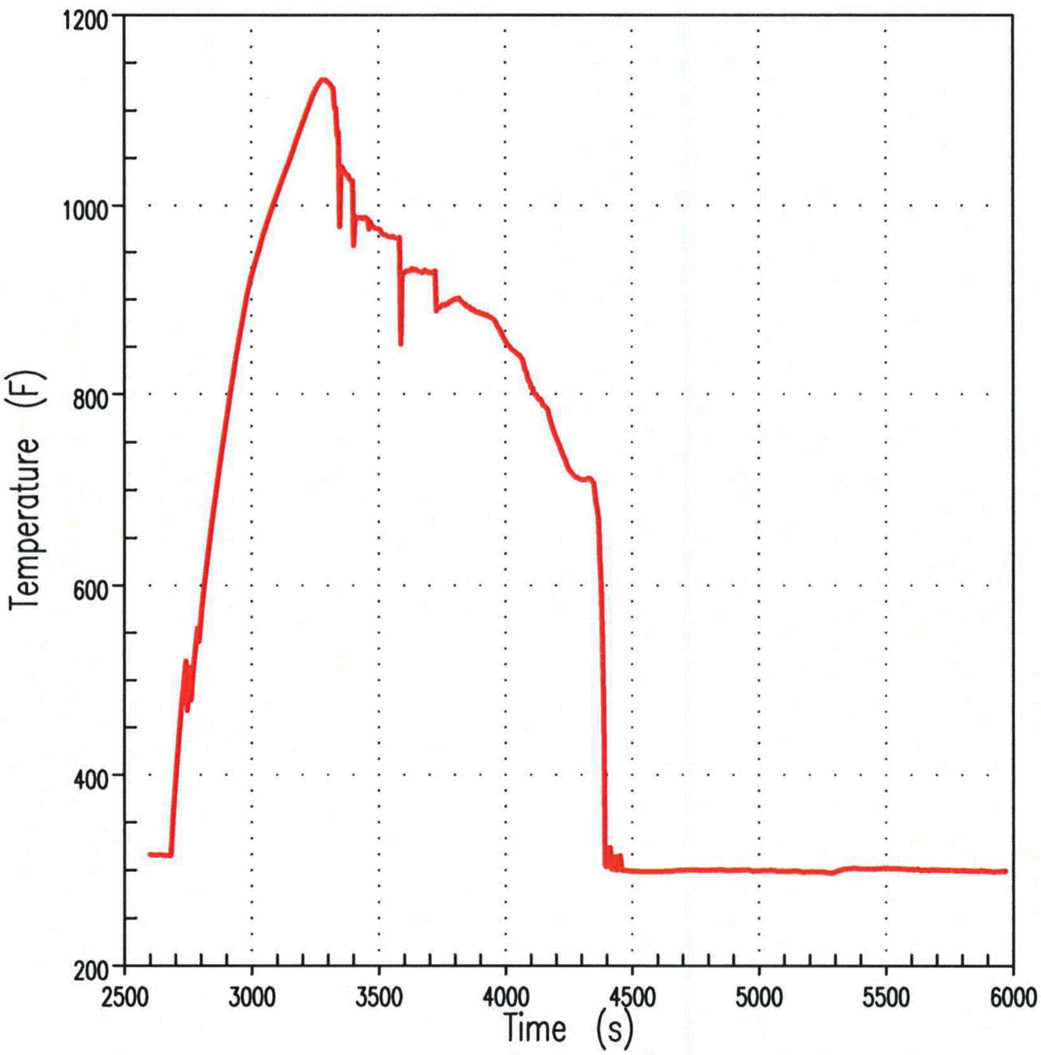

Figure 44
8.75-inch Break
Clad Temperature at PCT Elevation (11.50 ft)