INDIANA MICHIGAN POWER®
A unit of American Electric Power

Indiana Michigan Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
AEP.com

August 11, 2011

AEP-NRC-2011-52
10 CFR 50.46

Docket No.: 50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Donald C. Cook Nuclear Plant Unit 2
REVISED SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS

References:
1) Letter from R. A. Hruby, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Nuclear Plant Unit 2, Small Break Loss-Of-Coolant Accident Evaluation Model Reanalysis," AEP-NRC-2009-25, dated March 30, 2009 (ADAMS Accession Number ML091100153).
2) Letter from T. A. Beltz, NRC, to J. N. Jensen, I&M, "Donald C. Cook Nuclear Plant, Unit 2 - Request for Additional Information Regarding the Small-Break Loss-of-Coolant Accident Evaluation Model Reanalysis (TAC No. ME1147)," dated September 22, 2009 (ADAMS Accession Number ML092610017).
3) Letter from R. A. Hruby, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Unit 2, Schedule for Submittal of Revised Unit 1 and Unit 2 Small Break Loss of Coolant Accident Analyses Addressing Residual Heat Removal System Spray Issue," AEP-NRC-2010-30, dated March 29, 2010 (ADAMS Accession Number ML100060923).
4) Letter from P. S. Tam, NRC, to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant, Unit 2 (CNP-2) – Issuance of Amendment to Adopt a New Large-Break Loss-of-Coolant Accident Analysis (TAC No. ME1017)," dated March 31, 2011 (ADAMS Accession No. ML110730783).

This letter provides a revised small break loss of coolant accident (SBLOCA) analysis which addresses a Residual Heat Removal (RHR) system spray diversion issue for Donald C. Cook Nuclear Plant (CNP) Unit 2.

By Reference 1, Indiana Michigan Power Company (I&M), the licensee for CNP Unit 2, provided a report of a Unit 2 SBLOCA emergency core cooling system analysis performed using the Westinghouse NOTRUMP evaluation model. Reference 2 transmitted an NRC request for additional information (RAI) regarding the analysis. In Reference 3, I&M described an assumption in the new analysis that had been determined to be inconsistent with CNP Emergency Operating

A002
NRR

Procedure (EOP) provisions to realign the RHR pump discharge to provide containment spray under certain conditions, thus diverting RHR system flow from the Reactor Coolant System (RCS) during the cold leg recirculation phase of the accident. In Reference 3, I&M committed to revise the Unit 2 SBLOCA analysis to address the RHR flow diversion issue and transmit the revised analysis to the NRC no later than August 12, 2011. The RHR flow diversion issue has been addressed by changing the applicable EOPs such that the train cross-tie valves in Safety Injection (SI) System discharge are maintained open during the cold leg recirculation phase. Since each SI pump discharge is piped to two RCS loops if a cross-tie valve is closed, maintaining the cross tie valves open results in increased SI system flow to the RCS if only one SI pump is operating due to a postulated single failure. The increased SI system flow to the RCS compensates for the RHR flow being diverted from the RCS to containment spray. The Unit 2 SBLOCA analysis has been revised accordingly.

Additionally, the revised analysis does not include a case in which the SI system train cross-tie valves are assumed to be closed at the start of the event. Although this case was included in previous analyses, operation of the unit with the cross-tie valves closed for an unlimited period of time is no longer permitted by the Technical Specifications in accordance with Reference 4.

The enclosure to this letter provides a report of the revised analysis. As described in the enclosed report, the results of the revised SBLOCA analysis conform to the emergency core cooling system acceptance criteria of 10 CFR 50.46. The calculated peak cladding temperature of 1274°F is well below the 10 CFR 50.46 limit of 2200°F.

In Reference 3, I&M also committed to coordinate with the NRC staff to establish which of the RAI questions transmitted by Reference 2 remain applicable following submittal of the revised SBLOCA analysis, and establish a schedule for submittal of responses to the applicable questions. This commitment remains unchanged.

There are no new commitments in this submittal. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

JRW/jen

Enclosure

Donald C. Cook Nuclear Plant Unit 2, Revised Small Break Loss-Of-Coolant Accident Analysis Report

c: J. T. King – MPSC w/o enclosure
S. M. Krawec – AEP Ft. Wayne w/o enclosure
MDEQ – WHMD/RPS w/o enclosure
NRC Resident Inspector
M. A. Satorius, NRC R-III
P. S. Tam, NRC Washington, DC

Enclosure to AEP-NRC-2011-52

Donald C. Cook Nuclear Plant Unit 2

REVISED SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS REPORT

D. C. COOK UNIT 2 SMALL BREAK LOCA ANALYSIS SUMMARY REPORT

1 INTRODUCTION

The purpose of this report is to document the small break loss-of-coolant accident (SBLOCA) analysis performed for Donald C. Cook Nuclear Plant Unit 2 (D. C. Cook Unit 2) at a core power level of 3600 MWt (3612 MWt, including 0.34% uncertainty) with the emergency core cooling system (ECCS) cross-tie valves configurations as follows: 1) During the injection phase of the transient, both High Head Safety Injection (HHSI) and Residual Heat Removal (RHR) cross-tie valves are assumed to be open and 2) During the recirculation phase, the HHSI cross-tie valves are assumed to be open and the RHR cross-tie valves are assumed to be closed to isolate the trains. (It is noteworthy that HHSI refers to the fluid system designated as the Safety Injection (SI) system in the D. C. Cook Unit 2 Technical Specifications and Bases, and in previous model change and error reports submitted pursuant to 10 CFR 50.46.) The purpose of analyzing the SBLOCA is to demonstrate compliance with the 10 CFR 50.46 requirements (Reference 1). Important input assumptions, as well as analytical models and methodology for the SBLOCA analysis are contained in subsequent sections. Results are provided in the form of tables and figures, as well as more detailed descriptions of the limiting transient. The analysis has shown that no design or regulatory limit related to the SBLOCA transient would be exceeded due to plant operation with the ECCS cross-ties configurations, power level and the associated plant parameters outlined in this report.

2 INPUT PARAMETERS AND ASSUMPTIONS

The important plant conditions and features for D. C. Cook Unit 2 that are supported by this analysis are listed in Table 1. Additional considerations for several parameters identified in Table 1 are discussed below.

Figure 1 depicts the hot rod axial power shape modeled in the SBLOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core (the axial offset is +13%). Such a distribution is limiting for SBLOCA 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 through 5 provide the ECCS flows modeled in the SBLOCA analysis. Tables 2 through 5 provide the flows used to generate Figures 2 through 5, respectively. Figures 2 and 3 show the ECCS pumped injection flow versus pressure curves utilized during the injection phase. Figures 4 and 5 show the ECCS pumped injection flow versus pressure curves utilized during the cold leg recirculation phase (with HHSI available and RHR isolated, assuming active RHR spray). Note that the 1.5-, 3- and 4-inch break cases modeled more conservative ECCS pumped injection flows during cold leg recirculation than the available flow documented herein. Figures 2 and 4 show the flows from one charging (CHG) pump, one HHSI pump and one RHR pump, where the broken (or faulted) loop flow spills to reactor coolant system (RCS) pressure. Figures 3 and 5 show flows from one CHG pump, one HHSI pump and one RHR pump for the scenario where the break is postulated along the accumulator line. In this scenario, the

faulted loop CHG flow spills to RCS pressure and the faulted loop HHSI/RHR flow spills to 0 psig containment pressure. Note that hereafter, pumped injection subsystems of the ECCS (CHG, HHSI and RHR) are referred to collectively as safety injection (SI).

The analysis utilized an adjusted nominal vessel average temperature (T_{avg}) of 578.2°F (with ± 4°F uncertainty specified by NOTRUMP-EM) to support the D. C. Cook Unit 2 specific T_{avg} value of 578.1°F with +4.1°F uncertainty. The analysis supports operation for a nominal full-power T_{avg} range of 547.6°F to 578.1°F with +4.1°F/-5.6°F uncertainty. Additionally, the analysis utilizes a nominal pressurizer pressure of 2250 psia (plus +62.6 psi uncertainty) and supports operation at nominal pressurizer pressures of 2100 psia and 2250 psia with ±62.6 psi uncertainty.

3 DESCRIPTION OF ANALYSIS

3.1 ANALYTICAL MODEL

The requirements for an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Reference 1). For LOCAs due to Small Breaks, less than 1 square foot in area, the Westinghouse NOTRUMP SBLOCA ECCS Evaluation Model (References 2 and 3) with the improved condensation model (COSI) (Reference 4) is used. The Westinghouse NOTRUMP SBLOCA ECCS Evaluation Model (NOTRUMP-EM) was developed to determine the RCS response to design basis SBLOCAs, and to address Nuclear Regulatory Commission (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 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 SI flows that result from closure of one or both valves in the HHSI 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 the SBLOCA version of the LOCTA-IV code (Reference 3), using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow, core inlet enthalpy and mixture heights as boundary conditions. The small break version of the LOCTA-IV 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 6 illustrates the code interface for the Small Break Model.

3.2 ANALYSIS

The SBLOCA analysis for D. C. Cook Unit 2 considered a spectrum of cold leg breaks, including 1.5-, 2-, 3-, 4-, 6- and 8.75-inch breaks. 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. Results of the analysis are discussed in Section 5.

The most limiting single active failure used for a SBLOCA 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 SBLOCA analysis performed for D. C. Cook Unit 2 model the ECCS injection phase and cold leg recirculation phase flows as being delivered to both the intact and faulted loops at the RCS backpressure for breaks smaller than the accumulator line inner diameter (1.5-inch through 6-inch breaks) and at containment pressure for breaks equal to or greater than the accumulator line inner diameter (8.75-inch break). Note that for the 8.75-inch break, the CHG flow is assumed to inject into the broken loop 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 flows for these scenarios are illustrated in Figures 2 through 5. 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 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 modeled to be in a full power (100.34%) equilibrium condition (i.e., the heat generated in the core is being removed via the secondary system) for the nominal power level assumed. 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 SBLOCA 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. The SI signal is generated when the pressurizer low-pressure SI setpoint, conservatively modeled as 1715 psia, is reached. SI 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 LOOP 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 such that the core is rendered sub-critical. A rod drop time of 2.7 seconds was used while also considering an additional 2 seconds for the signal processing delay time. Therefore, a total delay time of 4.7 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 LOOP coincident with reactor trip), the loss of flow through the break is not sufficient to overcome the positive core flow maintained by the RCPs. During this period, upward flow through the core is maintained. However, following the RCP 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 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 7.

4 ACCEPTANCE CRITERIA

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) and are summarized 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 SBLOCA analysis.

For criterion 4, the appropriate core geometry was modeled in the SBLOCA analysis. The results based on this geometry satisfy the peak cladding temperature (PCT) criterion of 10 CFR 50.46 (Reference 1) 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 SBLOCA 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.

5 RESULTS

In order to determine the most limiting SBLOCA case(s) (as determined by the highest calculated PCT and the maximum local transient oxidation), six break cases were examined for D. C. Cook Unit 2. These cases were investigated to capture the most severe postulated SBLOCA event. The following discussion provides insight into the analyzed conditions.

The beginning of life (BOL) fuel rod heatup results for D. C. Cook Unit 2 are shown in Table 6 and Table 7 provides the key transient event times.

5.1 LIMITING BREAK CASE

The SBLOCA analysis for D. C. Cook Unit 2 showed that the 4-inch break is the limiting case for PCT and is the limiting case for maximum local transient oxidation along with the 3-inch break.

Peak Cladding Temperature

The limiting PCT is 1274°F for the 4-inch break at BOL. A summary of the transient response for the limiting PCT case is shown in Figures 7 through 17. 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 4-inch break, there is an initial rapid depressurization of the RCS followed by a very brief period of slower depressurization at approximately 1150 psia; slightly above the main steam safety valve (MSSV) lift pressure (Figures 7 and 10). A reactor trip signal is generated at 11.02 seconds followed by a SI signal at 17.71 seconds. The rate of depressurization begins to increase once the loop seal clears in the faulted loop at approximately 310 seconds, creating a vapor vent path between the top of the core and the break in the cold leg. During this initial part of the transient, the break flow is entirely liquid, which along with the low SI flow rates associated with the high system pressure results in a net reduction in the primary system mass. The accumulator injection setpoint is reached at approximately 864 seconds (Figure 13). Core uncovery begins during the injection phase at approximately 602 seconds (Figure 8), leading to the start of cladding heatup (Figure 15). The peak top core vapor temperature (Figure 9) and the corresponding PCT (Figure 15) occur at approximately 972 seconds. The PCT 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 9).

A comparison of the flow provided by the SI system to each loop is shown in Figure 14. The cold leg break vapor and liquid mass flow rates are provided in Figures 11 and 12, respectively. Figures 16 and 17 provide additional information on the hot spot fluid temperature at the PCT elevation and hot rod surface heat transfer coefficient at the PCT elevation, respectively.

Maximum Local Oxidation

The maximum local transient oxidation was calculated for the 3- and 4-inch breaks. The maximum local transient oxidation is 0.11% at BOL. The transient oxidation remains below 17% at all times in life, for all fuel resident in the core.

Core Wide Average Oxidation

Table 6 indicates that 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 (Reference 1).

5.2 NON-LIMITING BREAK CASES

Summaries of the transient responses for the non-limiting break cases (1.5-, 2-, 3-, 6- and 8.75-inch) are

shown in Figures 18 through 29. The 1.5- and 8.75-inch breaks showed no core uncovery and the 6-inch case showed minimal core uncovery (as can be seen in Figure 27), therefore PCT information was not calculated for these three cases. The plots for each of the additional non-limiting break cases include:

- RCS Pressure
- Core Mixture Level
- Clad Temperature at PCT Elevation (For 2- and 3-inch cases only)

The fuel rod heatup results for each of the additional breaks considered are shown in Table 6 and are less than the limiting 4-inch break case (Note: the 3-inch break has equivalent transient oxidation as the 4-inch break as discussed previously).

5.3 ADDITIONAL ANALYSIS DETAILS

5.3.1 Switchover from ECCS Injection Phase to ECCS Recirculation Phase

When the refueling water storage tank (RWST) volume of 280,000 gallons is delivered via SI 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.

5.3.2 Transient Termination

The 10 CFR 50.46 criteria (Reference 1) 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.

6 CONCLUSIONS

The SBLOCA analysis for D. C. Cook Unit 2 considered a break spectrum of 1.5-, 2-, 3-, 4-, 6- and 8.75-inch diameters. The analysis resulted in the limiting PCT of 1274°F for the 4-inch break and a maximum local transient oxidation of 0.11% calculated at BOL for the 3- and 4-inch breaks. The analysis is applicable to core power up to and including 3612 MWt (3600 MWt plus 0.34% uncertainty) with both the HHSI and RHR cross-tie valve(s) open during the injection phase and with HHSI cross-tie valves open and RHR cross-tie valves closed during the recirculation phase.

The analysis presented herein shows that the accumulator and SI subsystems of the ECCS, together with

the heat removal capability of the steam generators, provide sufficient core heat removal capability to maintain the calculated PCT for SBLOCA below the required limit of 10 CFR 50.46 (Reference 1). Furthermore, the analysis shows that the local cladding oxidation and core wide average oxidation, including consideration of pre-existing and post-LOCA oxidation, are less than the 10 CFR 50.46 (Reference 1) limits at all times in life for all fuel resident in the core. Note that the core wide average oxidation results illustrate that the total hydrogen generation is less than 1%.

Table 8 provides the summary of the results for the D. C. Cook Unit 2 SBLOCA analysis including PCT, maximum local transient oxidation and total hydrogen generation.

7 REFERENCES

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," 10 CFR 50.46, August 2007 and "ECCS Evaluation Models," Appendix K of 10 CFR 50, June 2000.

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

4. Thompson, C. M. 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 SBLOCA Analysis

Input Parameter	
Core Rated Thermal Power-100%	3600 MWt
Calorimetric Uncertainty, %	0.34
Fuel Type	17x17 Vantage 5 Fuel
Total Core Peaking Factor, F_Q	2.32
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.62
Hot Assembly Average Power Factor, P_{HA}	1.46
Maximum Axial Offset, %	13
Hot Rod Axial Power Shape	Figure 1
Initial RCS Loop Flow, gpm/loop	88,500
Initial Vessel T_{avg}, °F	578.2[(1)]
Initial Pressurizer Pressure (plus uncertainties), psia	2312.6[(2)]
Reactor Coolant Pump Type	93A
Pressurizer Low-Pressure Reactor Trip Setpoint, psia	1860
Reactor Trip Signal Processing Time, seconds	2.0
Rod Drop Time, seconds	2.7
Auxiliary Feedwater Temperature (Maximum), °F	120
AFW Flow (Minimum) to all 4 Steam Generators, gpm	750
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
MFW Isolation, seconds	8
MFW Isolation Signal	Safety Injection Actuation
Steam Generator Secondary Water Mass, lbm/SG	98,352
Containment Spray Flowrate for 2 Pumps, gpm	7,400
RWST Deliverable Volume (Minimum), gallons	280,000
SI Temp at Cold Leg Recirculation Time (Maximum), °F	190

Table 1 (continued)
Input Parameters Used in the SBLOCA Analysis

Input Parameter	
ECCS Configuration	1 CHG pump, 1 HHSI pump and 1 RHR pump 1.5- through 6-inch breaks -faulted loop injects to RCS pressure 8.75-inch (accumulator line break) -faulted loop CHG flow injects to RCS pressure -faulted loop HHSI/RHR flow spills to containment (0 psia)
Cross Tie Valve Position(s) – Injection Phase	Both HHSI and RHR Open
Cross Tie Valve Position(s) – Recirculation Phase	HHSI Open and RHR Closed
ECCS Water Temperature (Maximum), °F	120
Pressurizer Low-Pressure Safety Injection Setpoint, psia	1715
SI Flow Delay Time, seconds	54
ECCS Flow vs. Pressure (Injection Phase) ECCS Flow vs. Pressure (Recirculation Phase)(3)	Tables 2 and 3 Tables 4 and 5
Initial Accumulator Water/Gas Temperature, °F	130
Initial Nominal Accumulator Water Volume, ft^3	946
Minimum Accumulator Pressure, psia	600

Notes:

(1) Analysis supports operation over the range of nominal full-power T_{avg} values of 547.6°F – 578.1°F with T_{avg} uncertainty range of +4.1°F/-5.6°F.

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

(3) The 1.5-, 3- and 4-inch breaks modeled more conservative flows; therefore, the flows referenced here are bounding.

Table 2
Safety Injection Flows Used in the SBLOCA 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	177.8	155.5	157.6	156.5
34.7	169.2	148.1	150.0	149.0
54.7	159.3	139.5	141.2	140.4
74.7	148.1	129.8	131.2	130.6
94.7	135.8	119.1	120.3	119.8
114.7	121.5	106.7	107.7	107.4
134.7	103.3	91.0	91.5	91.5
154.7	77.0	68.1	68.2	68.4
174.7	31.3	28.5	27.6	28.4
194.7	31.1	28.3	27.4	28.2
214.7	30.9	28.1	27.2	28.0
234.7	30.6	27.9	26.9	27.8
254.7	30.4	27.6	26.7	27.6
274.7	30.1	27.4	26.5	27.4
294.7	29.9	27.2	26.3	27.1
314.7	29.7	27.0	26.1	26.9
414.7	28.4	25.8	25.0	25.8
514.7	27.2	24.7	23.9	24.6
614.7	25.8	23.4	22.7	23.4
714.7	24.4	22.1	21.5	22.1
814.7	23.0	20.8	20.2	20.8
914.7	21.4	19.3	18.8	19.3
1014.7	19.5	17.5	17.1	17.5
1114.7	17.4	15.6	15.2	15.5
1214.7	14.4	12.9	12.6	12.8
1314.7	10.1	8.8	8.8	8.7
1414.7	9.0	7.9	7.9	7.8

Table 2 – (continued)
Safety Injection Flows Used in the SBLOCA 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
1514.7	8.6	7.5	7.5	7.4
1614.7	8.1	7.1	7.1	7.0
1714.7	7.6	6.6	6.6	6.5
1814.7	7.1	6.2	6.1	6.1
1914.7	5.3	4.6	4.6	4.6
2014.7	4.7	4.1	4.1	4.0
2114.7	3.9	3.5	3.4	3.4
2214.7	3.1	2.7	2.7	2.6
2314.7	0.0	0.0	0.0	0.0
2414.7	0.0	0.0	0.0	0.0

Table 3
Safety Injection Flows Used in the SBLOCA Analysis – Injection Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump – faulted loop CHG flow injects to RCS pressure and faulted loop HHSI/RHR flow spills to containment (0 psia) – 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	14.1	163.7	155.5	157.6	156.5
34.7	14.1	230.7	142.6	92.1	143.5
54.7	14.0	300.3	132.0	12.2	132.9
74.7	13.9	319.2	110.1	12.2	110.7
94.7	13.9	339.5	85.1	12.1	85.5
114.7	13.8	362.1	54.4	12.0	54.6
134.7	13.7	379.3	28.8	12.0	28.8
154.7	13.7	379.5	28.6	11.9	28.5
174.7	13.6	379.6	28.3	11.9	28.2
194.7	13.6	379.6	28.0	11.8	28.0
214.7	13.5	379.7	27.7	11.8	27.7
234.7	13.4	379.8	27.4	11.7	27.4
254.7	13.4	379.8	27.2	11.7	27.1
274.7	13.3	379.9	26.9	11.6	26.8
294.7	13.2	379.9	26.6	11.5	26.5
314.7	13.2	380.0	26.3	11.5	26.2
414.7	12.8	380.3	24.7	11.2	24.7
514.7	12.5	380.6	23.1	10.9	23.1
614.7	12.2	386.8	20.8	10.6	20.7
714.7	11.8	387.8	17.6	10.3	17.6
814.7	11.5	388.9	13.9	10.0	13.8
914.7	11.1	390.1	9.7	9.7	9.5
1014.7	10.7	390.1	9.3	9.3	9.2
1114.7	10.3	390.2	9.0	9.0	8.9
1214.7	9.9	390.2	8.6	8.6	8.5
1314.7	9.5	390.2	8.3	8.3	8.1

Table 3 – (continued)
Safety Injection Flows Used in the SBLOCA Analysis – Injection Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump – faulted loop CHG flow injects to RCS pressure and faulted loop HHSI/RHR flow spills to containment (0 psia) – 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
1414.7	9.0	390.2	7.9	7.9	7.8
1514.7	8.6	390.2	7.5	7.5	7.4
1614.7	8.1	390.2	7.1	7.1	7.0
1714.7	7.6	390.2	6.6	6.6	6.5
1814.7	7.1	390.2	6.2	6.1	6.1
1914.7	5.3	390.2	4.6	4.6	4.6
2014.7	4.7	390.2	4.1	4.1	4.0
2114.7	3.9	390.2	3.5	3.4	3.4
2214.7	3.1	390.2	2.7	2.7	2.6
2314.7	0.0	390.2	0.0	0.0	0.0
2414.7	0.0	390.2	0.0	0.0	0.0

Table 4
Safety Injection Flows Used in the SBLOCA Analysis - Recirculation Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump - faulted loop injects to RCS pressure – RHR Spray active – 1.5- 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	36.3	32.1	31.0	32.2
26.7	36.3	31.9	30.8	32.2
34.7	36.2	31.8	30.7	31.9
54.7	35.9	31.6	30.5	31.8
74.7	35.8	31.5	30.4	31.6
94.7	35.5	31.2	30.1	31.4
114.7	35.4	31.1	30.0	31.2
134.7	35.1	30.8	29.7	31.0
154.7	34.8	30.7	29.6	30.8
174.7	34.5	30.5	29.4	30.7
194.7	34.4	30.1	29.2	30.4
214.7	34.1	30.0	29.0	30.1
234.7	34.0	29.7	28.8	29.9
254.7	33.6	29.6	28.6	29.7
274.7	33.4	29.4	28.5	29.6
294.7	33.2	29.2	28.2	29.3
314.7	33.0	28.9	27.9	29.2
414.7	31.6	27.9	27.0	28.1
514.7	30.4	26.6	25.7	26.7
614.7	29.2	25.5	24.6	25.6
714.7	27.8	24.2	23.4	24.4
814.7	26.3	22.8	22.2	23.0
914.7	24.8	21.6	20.9	21.6
1014.7	23.0	20.0	19.4	20.1
1114.7	21.1	18.2	17.6	18.2
1214.7	18.7	16.1	15.7	16.1
1314.7	15.6	13.2	12.8	13.2

Table 4 – (continued)
Safety Injection Flows Used in the SBLOCA Analysis - Recirculation Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump - faulted loop injects to RCS pressure – RHR Spray active – 1.5- through 6-inch breaks)

RCS Pressure (psia)	Broken Loop (lbm/sec)	Intact Loops (lbm/sec)		
	Loop 1	Loop 2	Loop 3	Loop 4
1414.7	11.7	9.5	9.4	9.5
1514.7	9.6	7.7	7.7	7.7
1614.7	9.2	7.3	7.3	7.3
1714.7	8.7	6.9	6.9	6.9
1814.7	8.1	6.5	6.5	6.5
1914.7	7.4	5.9	5.9	5.9
2014.7	6.7	5.5	5.5	5.5
2114.7	6.0	4.9	4.9	4.9
2214.7	5.2	4.3	4.3	4.3
2314.7	3.3	2.6	2.6	2.6
2414.7	0.0	0.0	0.0	0.0

Table 5
Safety Injection Flows Used in the SBLOCA Analysis – Recirculation Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump – faulted loop CHG flow injects to RCS pressure and faulted loop HHSI/RHR flow spills to containment (0 psia) – RHR Spray active – 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	15.7	20.6	32.1	31.0	32.2
26.7	15.7	39.5	31.9	12.6	32.1
34.7	15.7	39.5	31.8	12.5	31.9
54.7	15.5	39.6	31.5	12.5	31.6
74.7	15.5	39.8	31.4	12.5	31.5
94.7	15.4	39.9	31.0	12.4	31.1
114.7	15.4	40.0	30.7	12.4	31.0
134.7	15.3	40.2	30.4	12.2	30.5
154.7	15.3	40.3	30.1	12.2	30.4
174.7	15.1	40.4	30.0	12.2	30.1
194.7	15.1	40.6	29.6	12.1	29.7
214.7	15.0	40.7	29.4	12.1	29.6
234.7	15.0	40.9	29.0	12.0	29.2
254.7	14.8	41.0	28.8	12.0	28.9
274.7	14.8	41.1	28.6	12.0	28.8
294.7	14.7	41.1	28.2	11.8	28.3
314.7	14.7	41.3	27.9	11.8	28.1
414.7	14.3	42.0	26.4	11.5	26.6
514.7	13.9	42.6	24.8	11.1	24.9
614.7	13.6	56.0	22.8	10.9	23.0
714.7	13.2	57.9	20.0	10.6	20.1
814.7	12.8	59.8	16.5	10.2	16.7
914.7	12.4	62.0	12.5	9.9	12.5
1014.7	12.0	64.1	9.6	9.6	9.6
1114.7	11.5	64.1	9.2	9.2	9.2
1214.7	11.1	64.1	8.9	8.9	8.9
1314.7	10.6	-	8.5	8.5	8.5

Table 5 – (continued)
Safety Injection Flows Used in the SBLOCA Analysis – Recirculation Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump – faulted loop CHG flow injects to RCS pressure and faulted loop HHSI/RHR flow spills to containment (0 psia) – RHR Spray active – 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
1414.7	10.2	-	8.1	8.1	8.1
1514.7	9.6	-	7.7	7.7	7.7
1614.7	9.2	-	7.3	7.3	7.3

Table 6
SBLOCTA BOL Results

Break Size (in)	**2.0**	**3.0**	**4.0**
PCT (°F)	977.9	1176.5	**1273.7**
PCT Time (s)	1989.1	1603.1	971.5
PCT Elevation (ft)	11.00	11.25	11.25
Max. Local ZrO_2 (%)	0.01	**0.11**	**0.11**
Max. Local ZrO_2 Elev. (ft)	11.00	11.50	11.25
Hot Rod Axial Average ZrO_2 (%)(1)	0.00	0.02	0.02

Notes:

(1) The hot rod axial average ZrO_2 conservatively represents the core wide average oxidation, since the core wide average ZrO_2 thickness will always be less than the corresponding hot rod axial average ZrO_2 thickness.

Table 7
Time Sequence of Events

Event Time	1.5-inch	2-inch	3-inch	4-inch	6-inch	8.75-inch
Break Initiation (s)	0.00	0.00	0.00	0.00	0.00	0.00
Reactor Trip Signal (s)	89.72	46.48	19.17	11.02	6.09	4.83
S-Signal (s)	108.11	59.03	27.57	17.71	9.65	7.66
SI Flow Delivered (1), (s)	162.11	113.03	81.57	71.71	63.65	61.66
Loop Seal Clearing (2) (s)	2630	1422	551	310	146	26
Core Uncovery (4) (s)	N/A	1726	877	602	N/A	N/A
Accumulator Injection (s)	N/A	N/A	1800	864	352	168(3)
RWST Volume Delivered (5) (s)	2161.88	2151.71	2115.48	2088.59	2052.49	1565.20
PCT Time (BOL) (s)	N/A	1989.1	1603.1	971.5	N/A	N/A
Core Recovery (4) (s)	N/A	5070	N/A(6)	2830	N/A	N/A

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 8.75-inch break, accumulator injection begins for Loops 2-4 only; 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.

(6) The run was successfully terminated per the NOTRUMP transient termination criteria described in Section 5.3.2 of the report text.

Table 8
SBLOCA Results Summary

Peak Cladding Temperature (°F)	1273.7
Maximum Local Transient Oxidation (%)	0.11
Total Hydrogen Generation (%)	<1

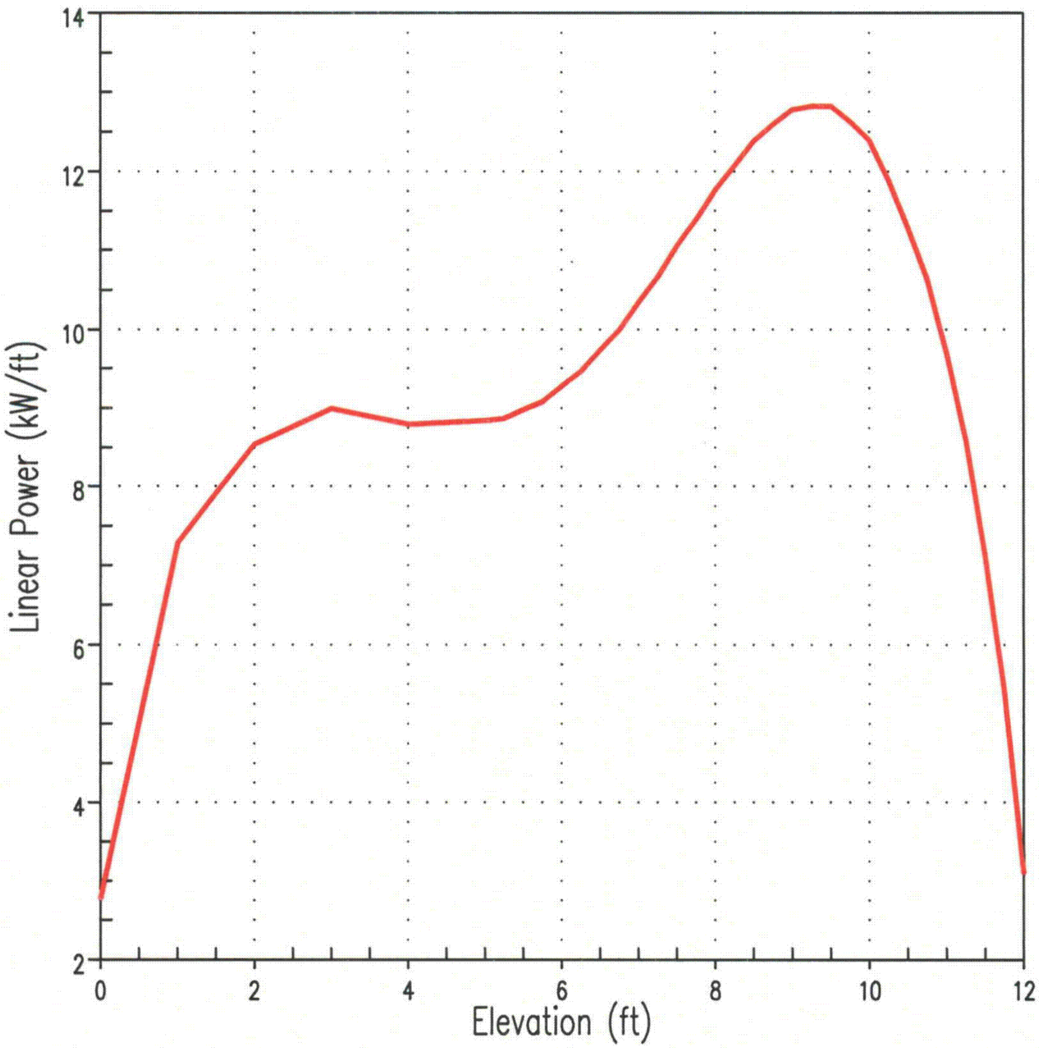

Figure 1
Small Break Hot Rod Power Shape

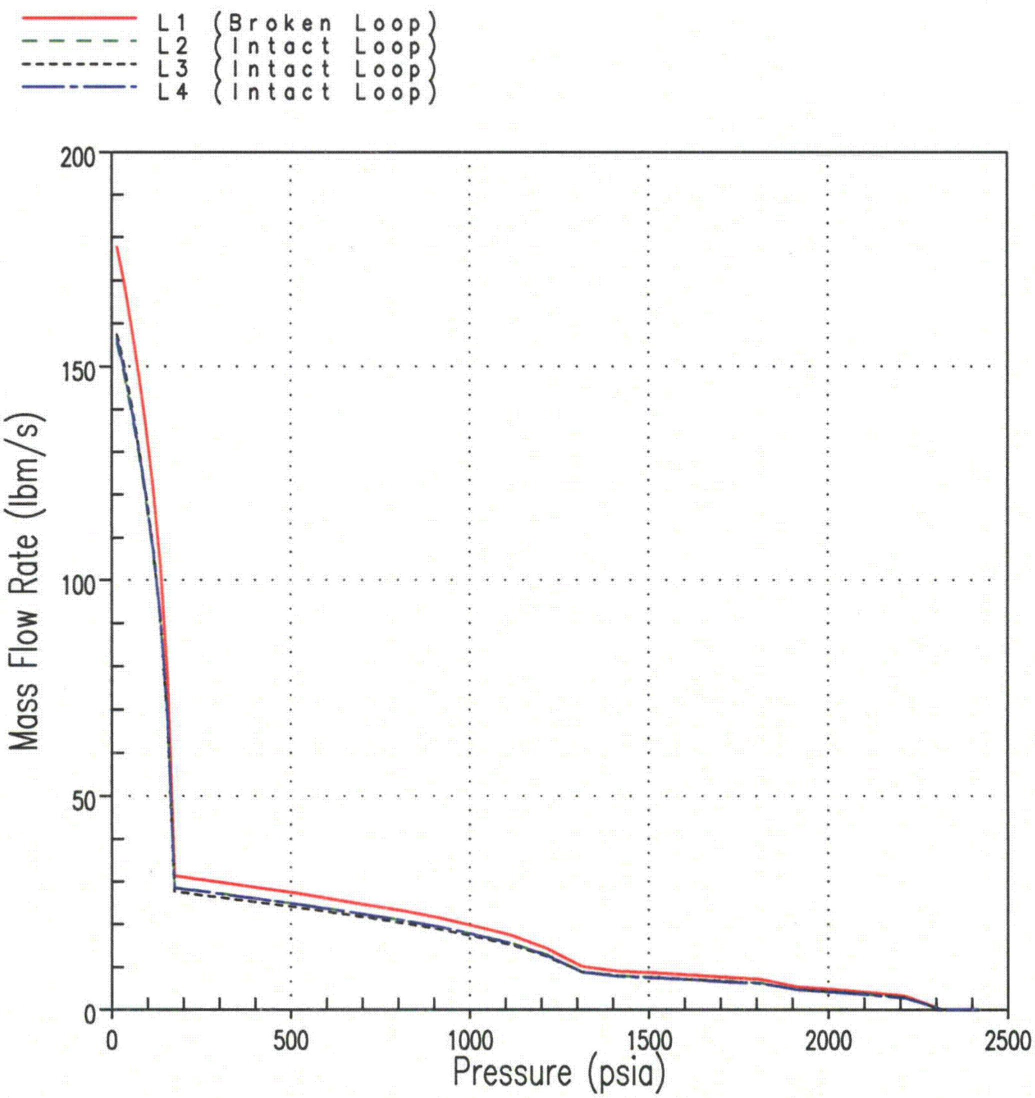

Figure 2
SBLOCA Safety Injection Flows – Injection Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump - faulted loop injects to RCS pressure – 1.5-inch through 6-inch breaks)

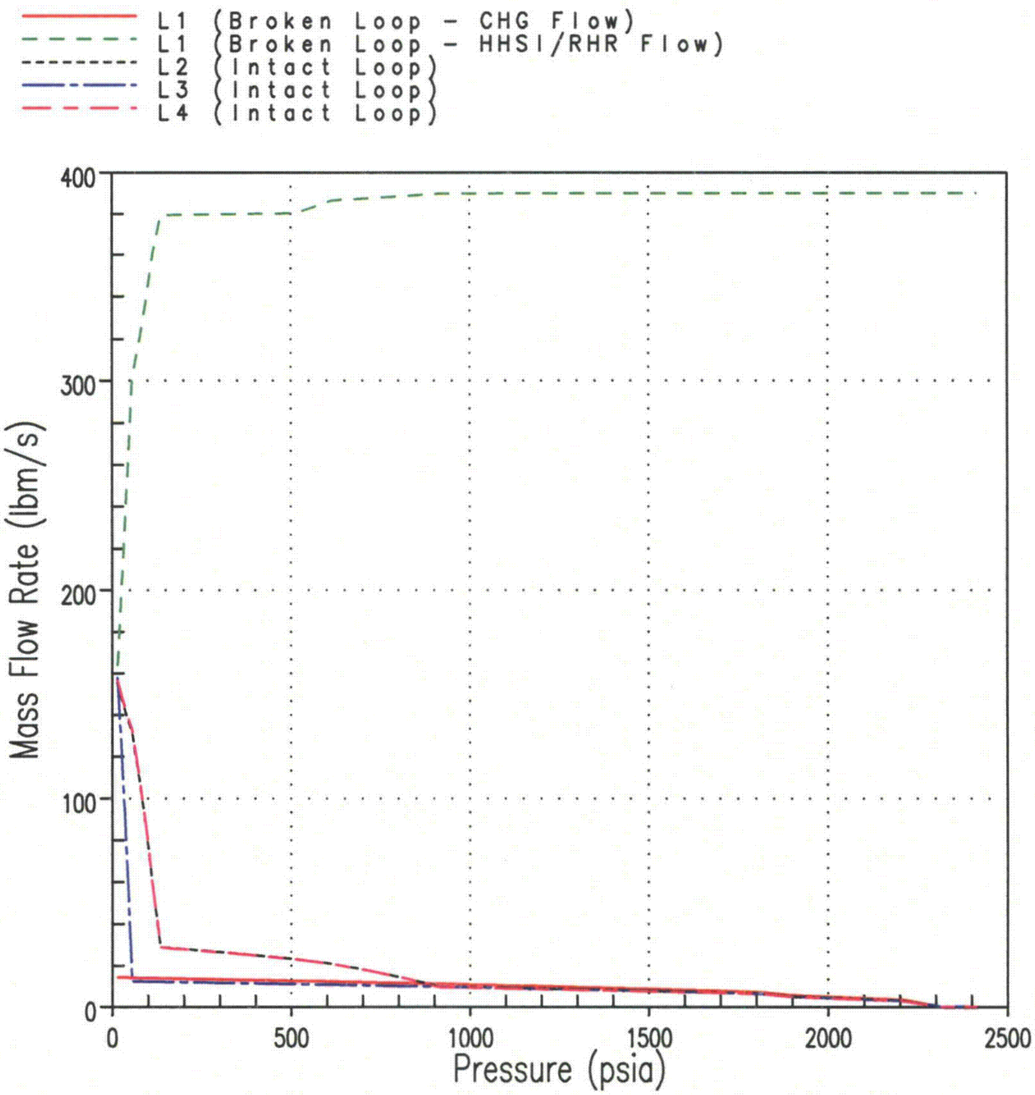

Figure 3
SBLOCA Safety Injection Flows – Injection Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump – faulted loop CHG flow injects to RCS pressure and faulted loop HHSI/RHR flow spills to containment (0 psia) – 8.75-inch break)

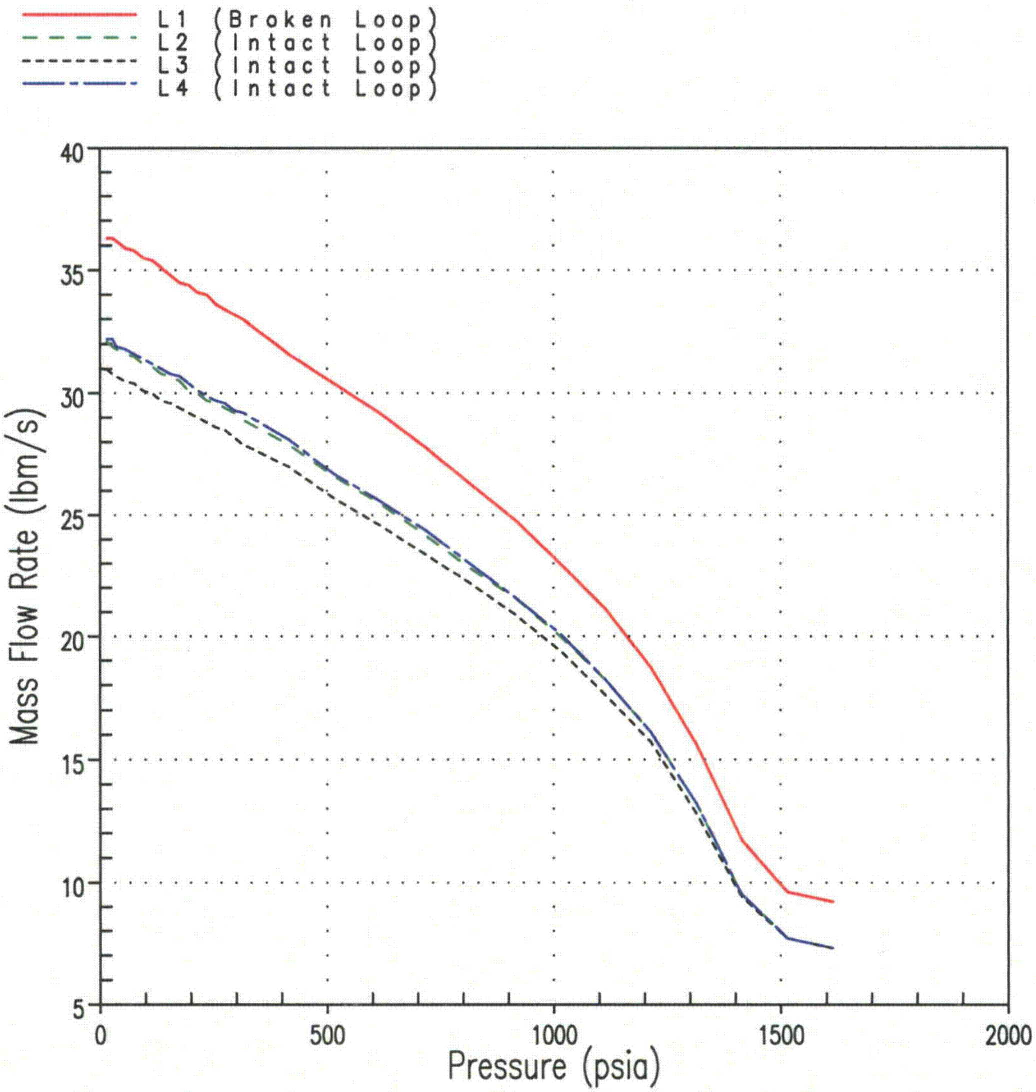

Figure 4
SBLOCA Safety Injection Flows – Recirculation Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump - faulted loop injects to RCS pressure – Spray active – 1.5-inch through 6-inch breaks)

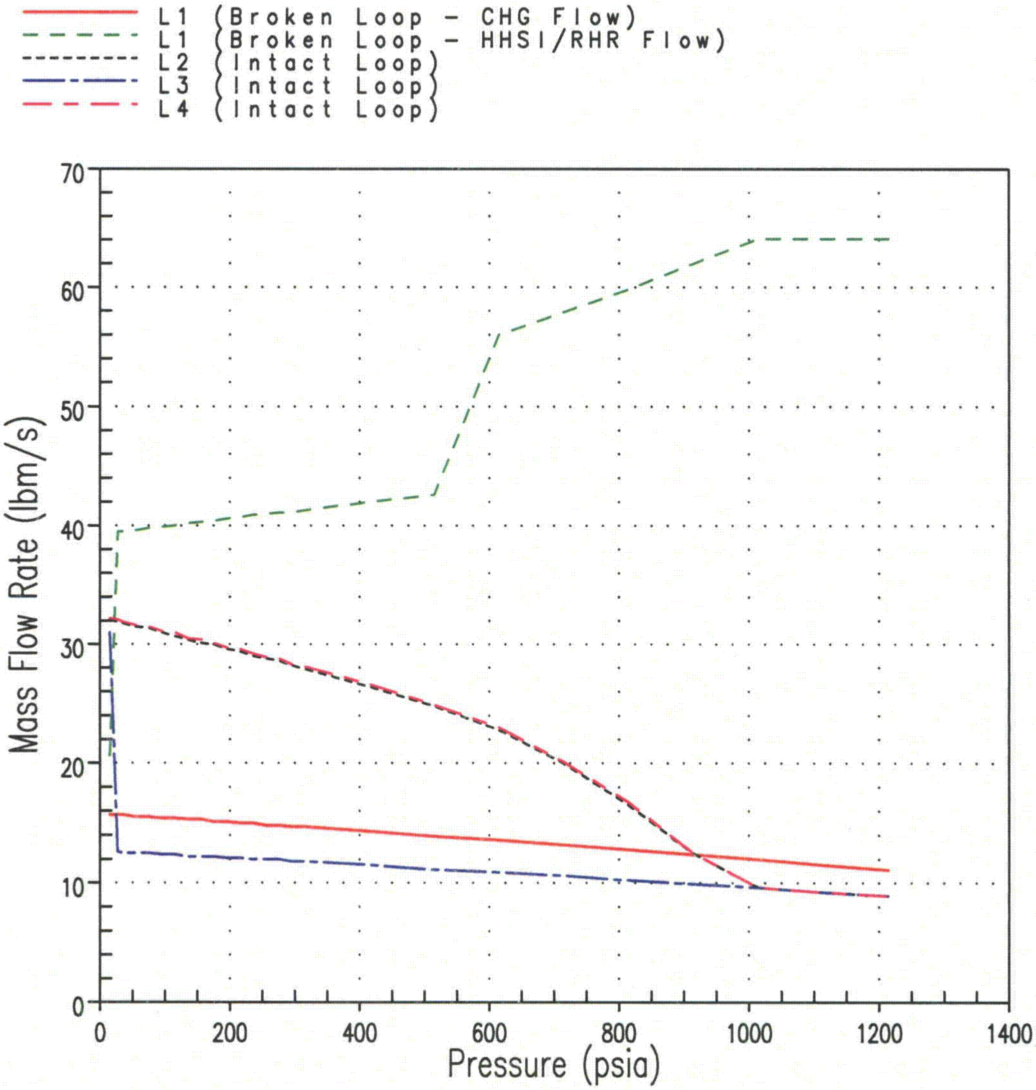

Figure 5
SBLOCA Safety Injection Flows – Recirculation Phase
(1 CHG pump, 1 HHSI pump, 1 RHR pump – faulted loop CHG flow injects to RCS pressure and faulted loop HHSI/RHR flow spills to containment (0 psia) – Spray active – 8.75-inch break)

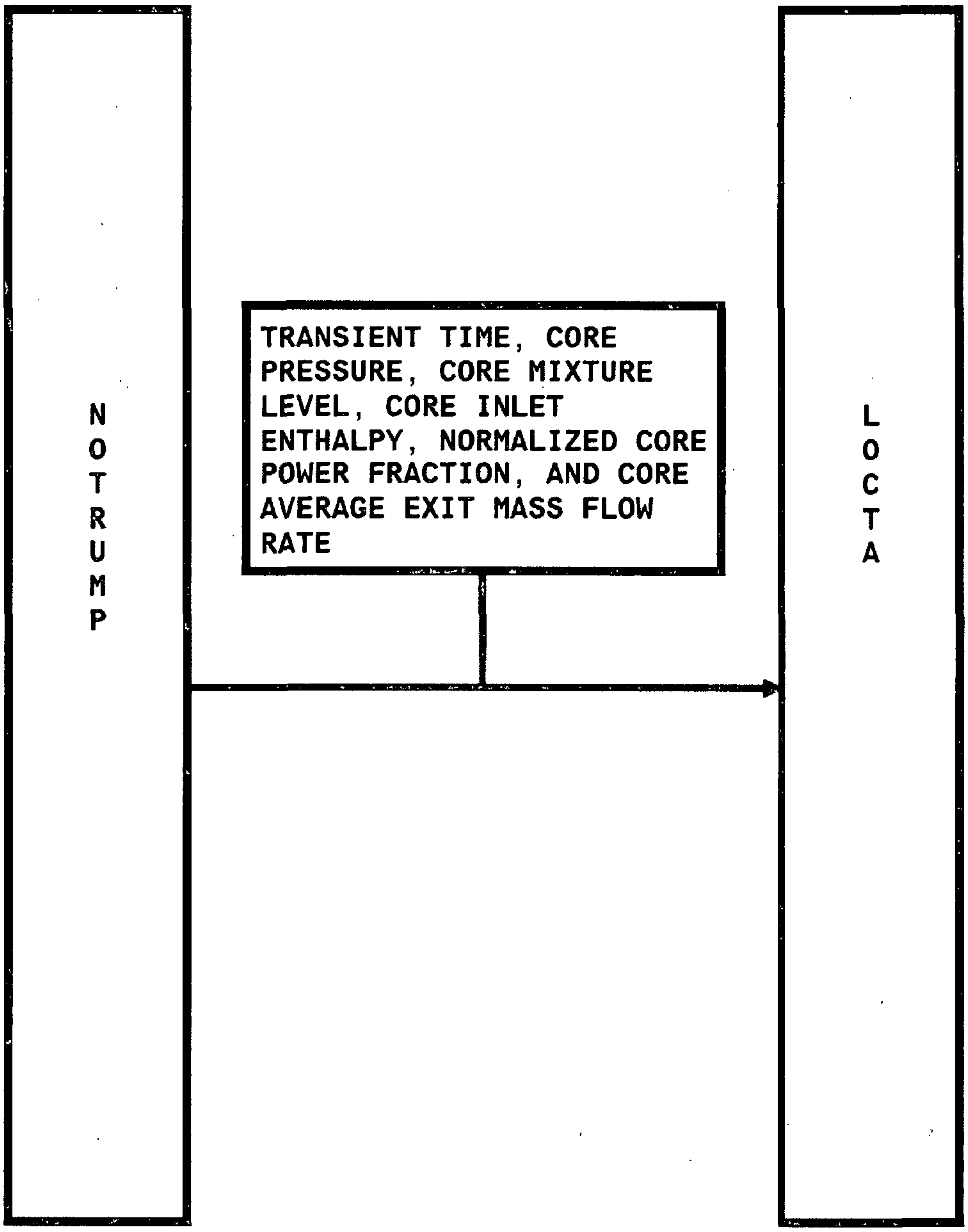

Figure 6
Code Interface Description
for Small Break Model

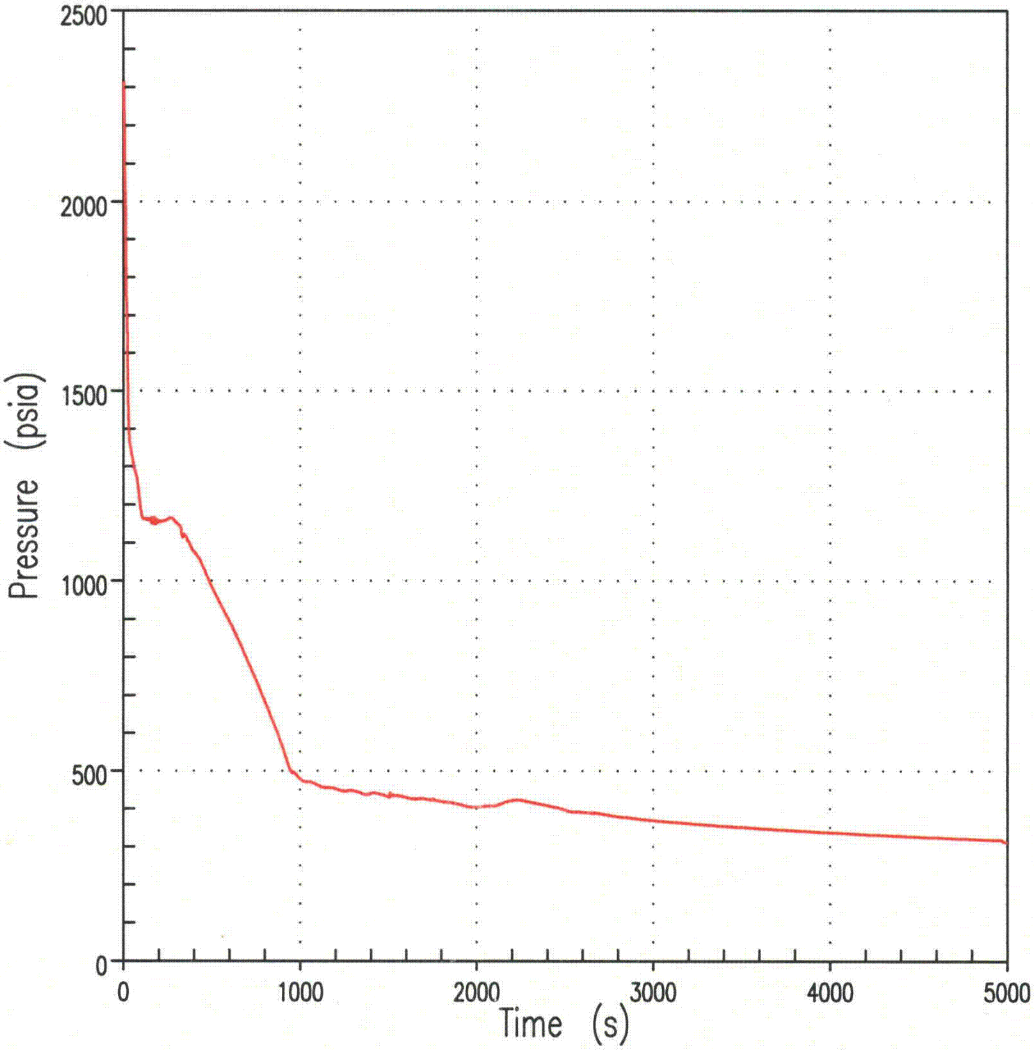

Figure 7
4-inch Break
RCS Pressure

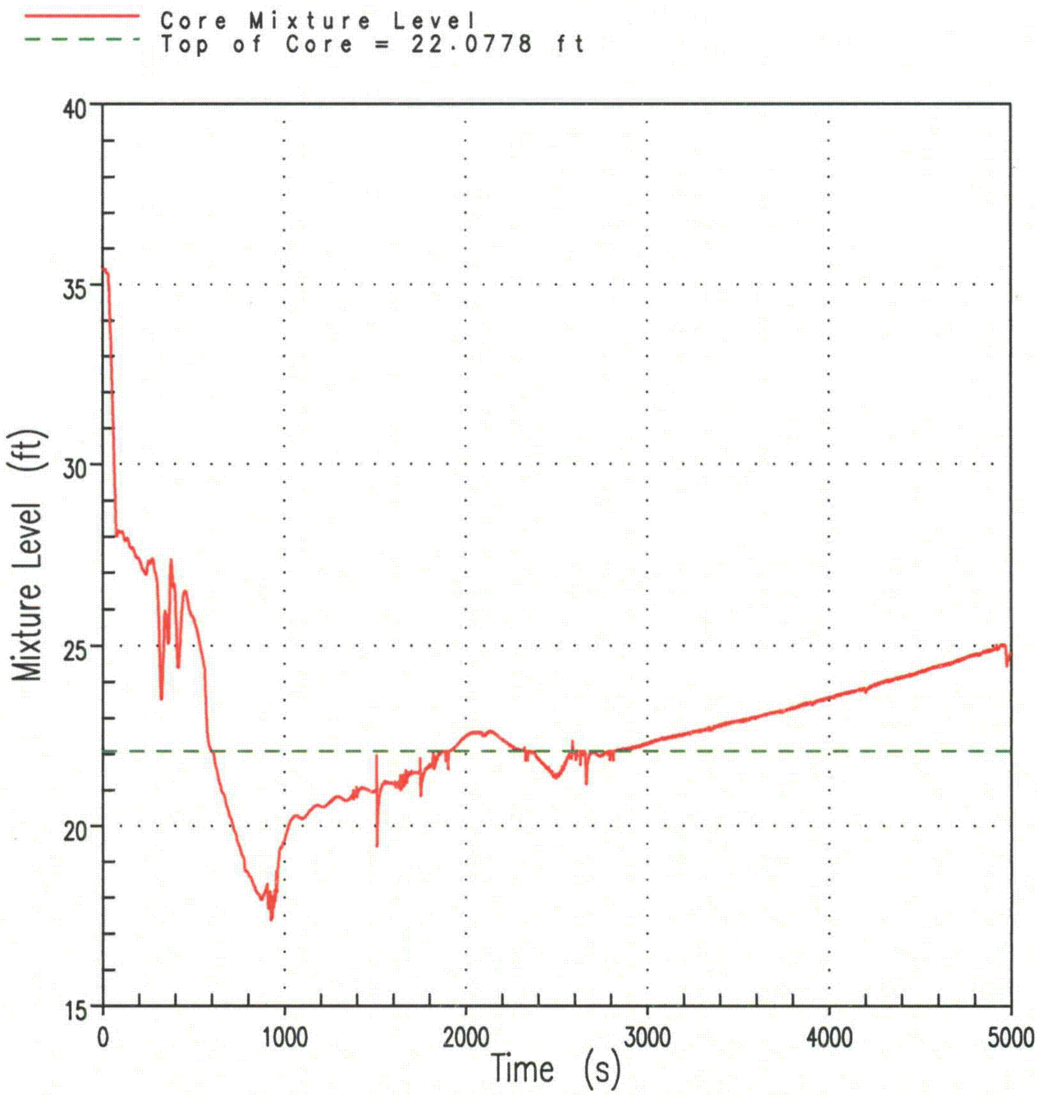

Figure 8
4-inch Break
Core Mixture Level

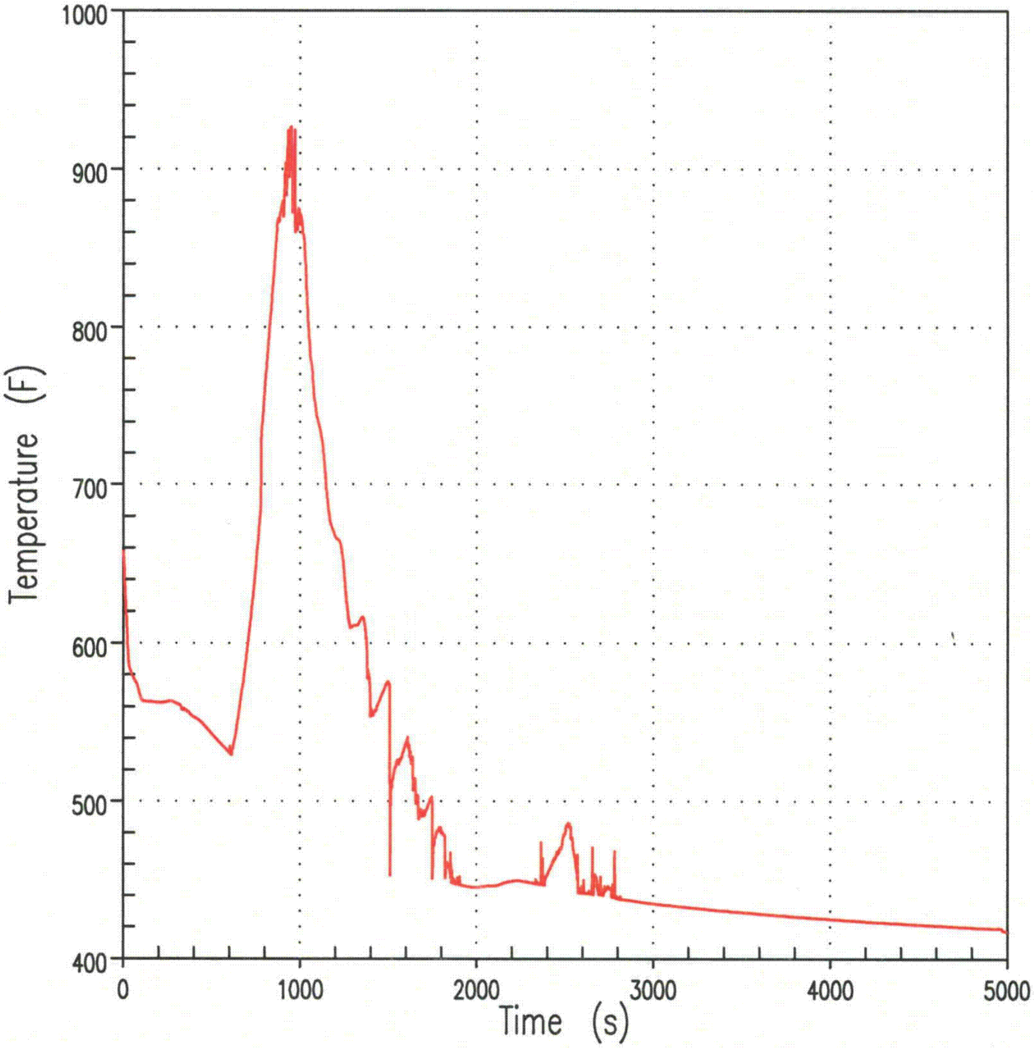

Figure 9
4-inch Break
Core Exit Vapor Temperature

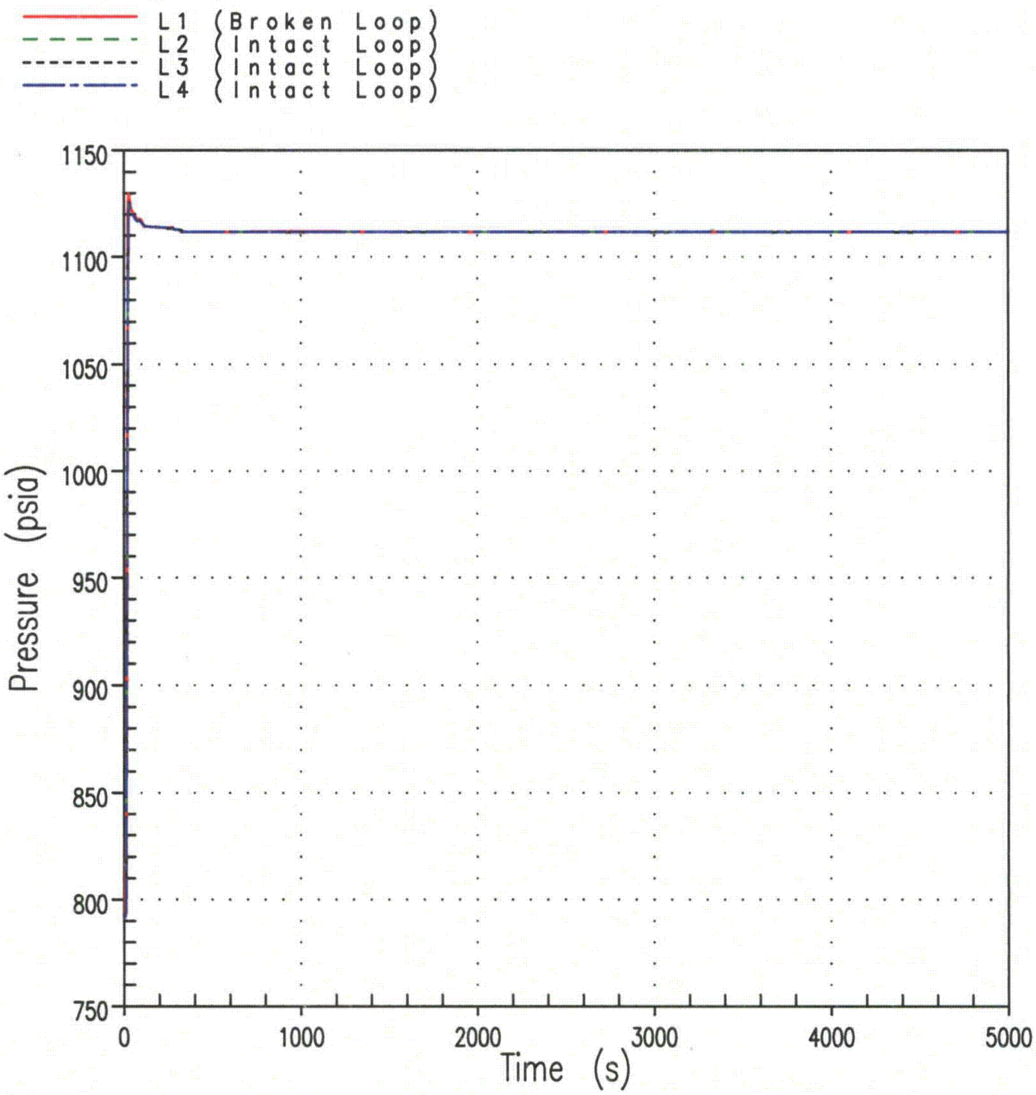

Figure 10
4-inch Break
Broken and Intact Loops Secondary Pressures

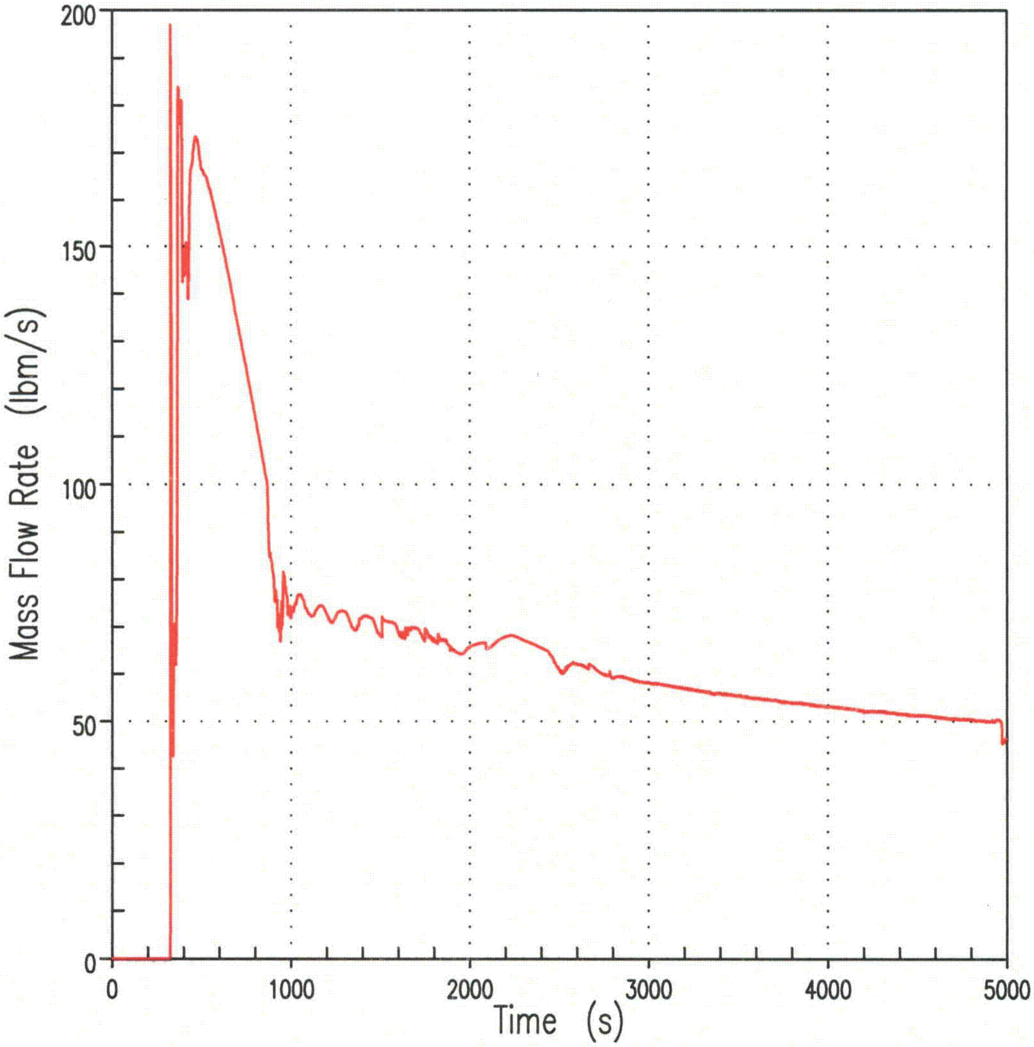

Figure 11
4-inch Break
Break Vapor Flow Rate

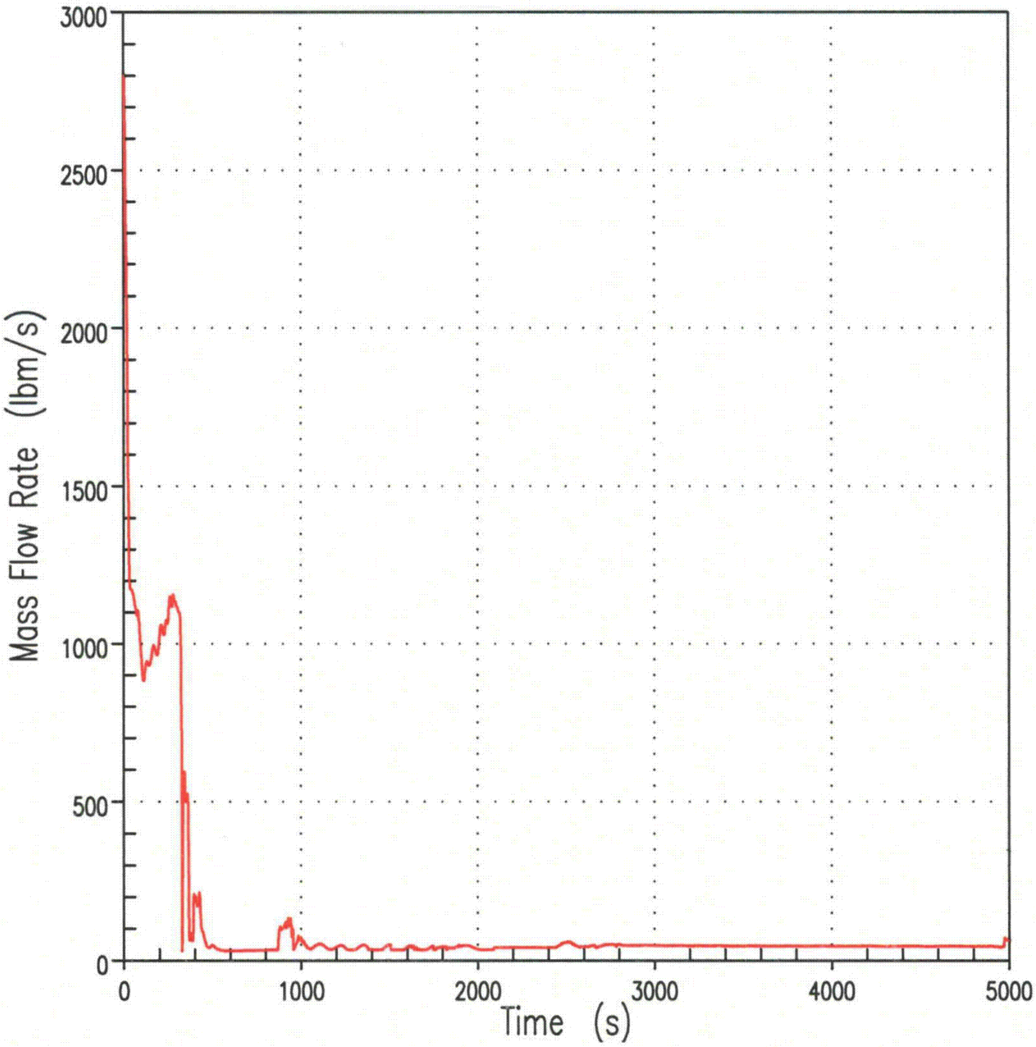

Figure 12
4-inch Break
Break Liquid Flow Rate

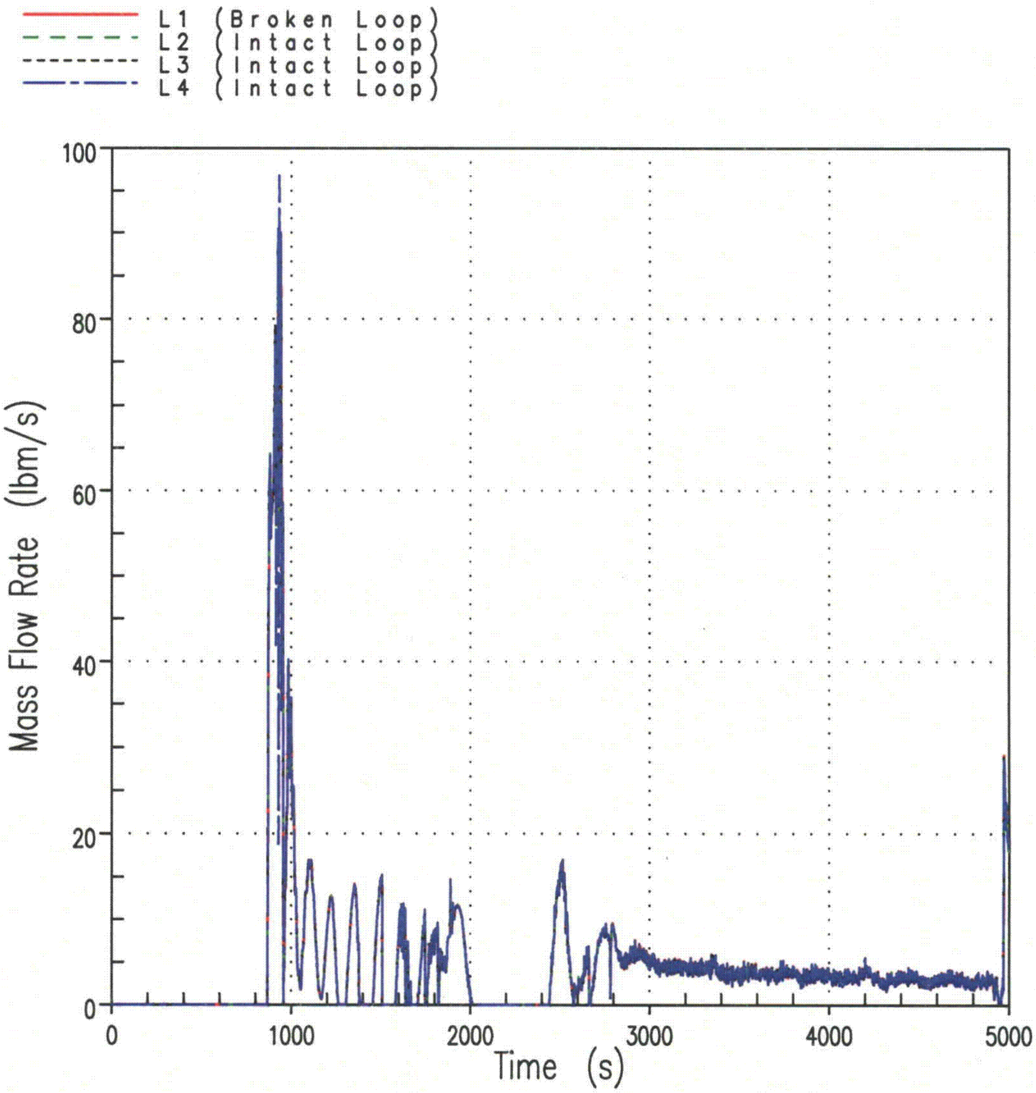

Figure13
4-inch Break
Broken and Intact Loops Accumulator Flow Rates

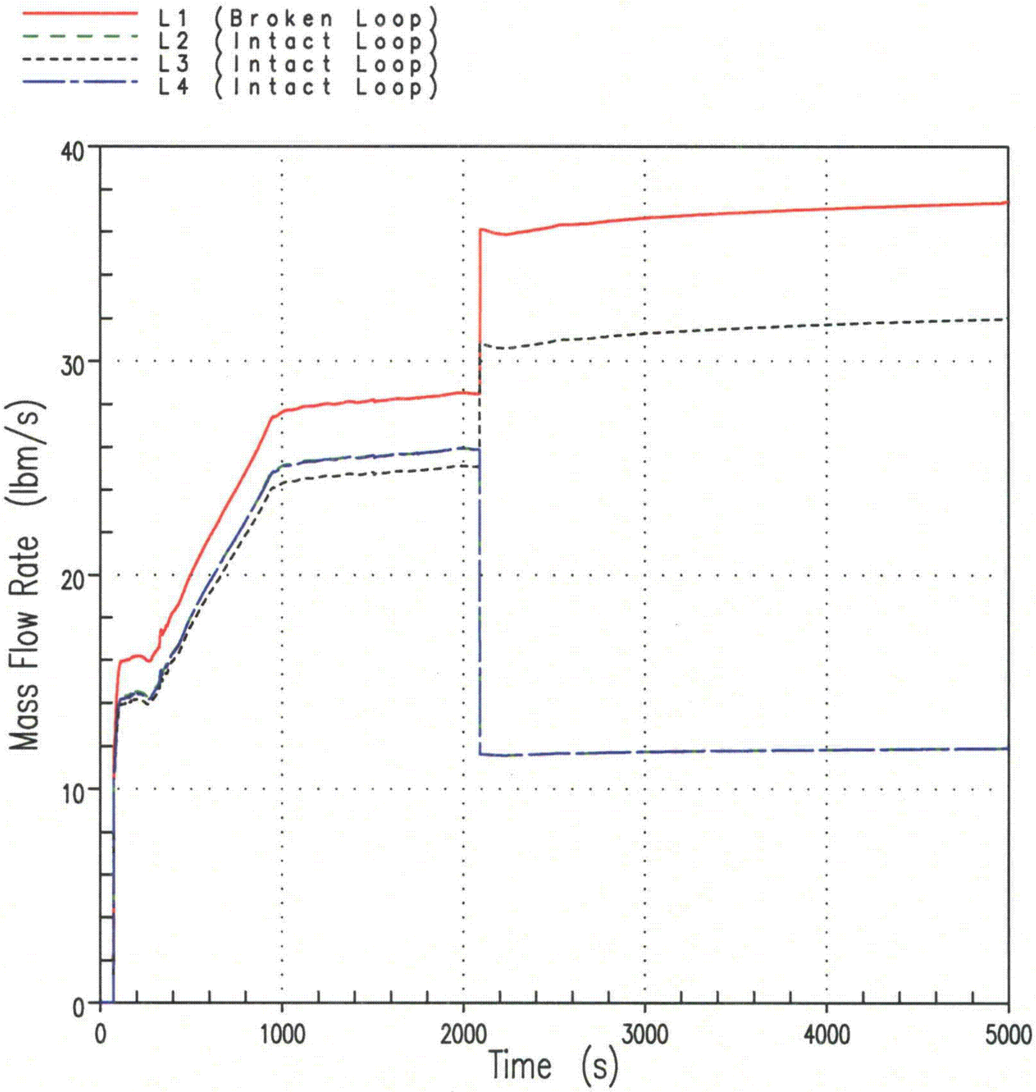

Figure 14
4-inch Break
Broken and Intact Loops Pumped Safety Injection Flow Rates

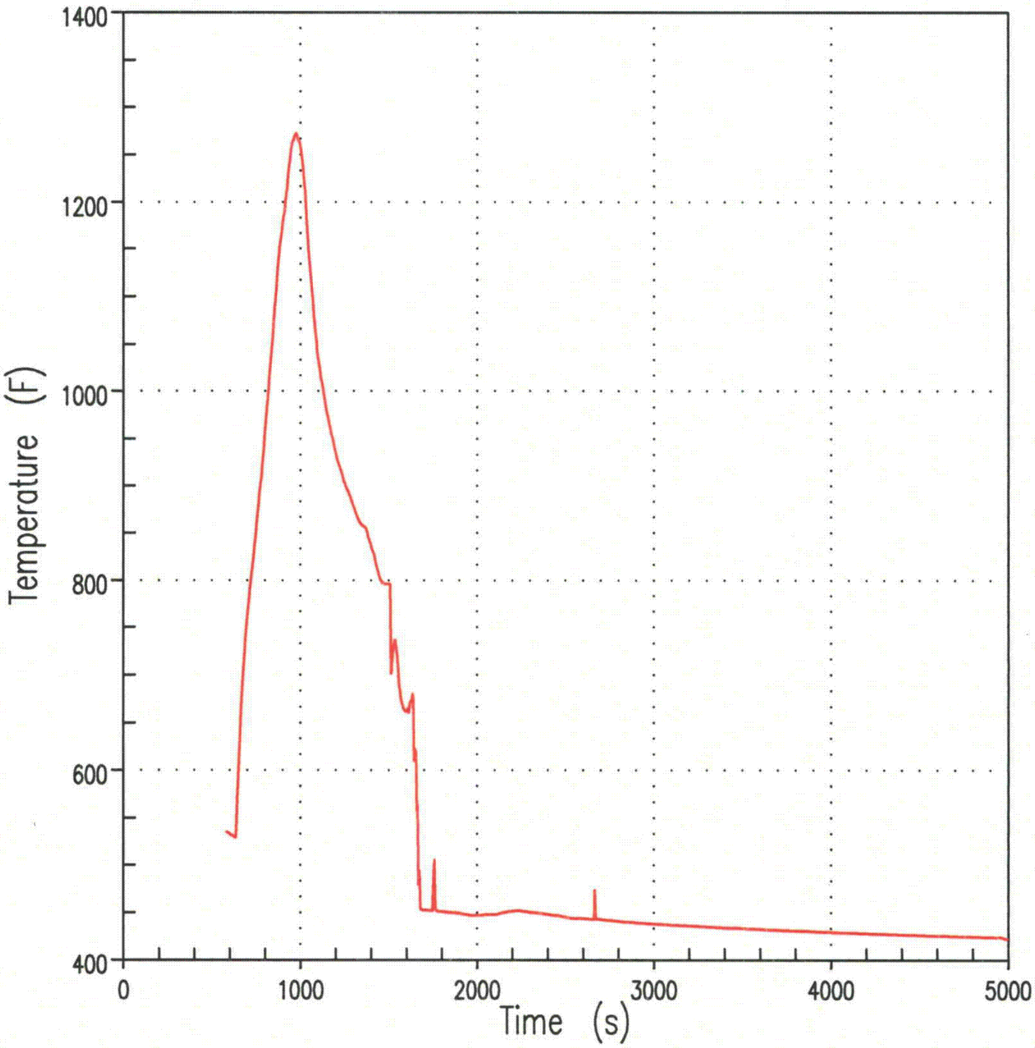

Figure 15
4-inch Break
Clad Temperature at PCT Elevation (11.25 ft)

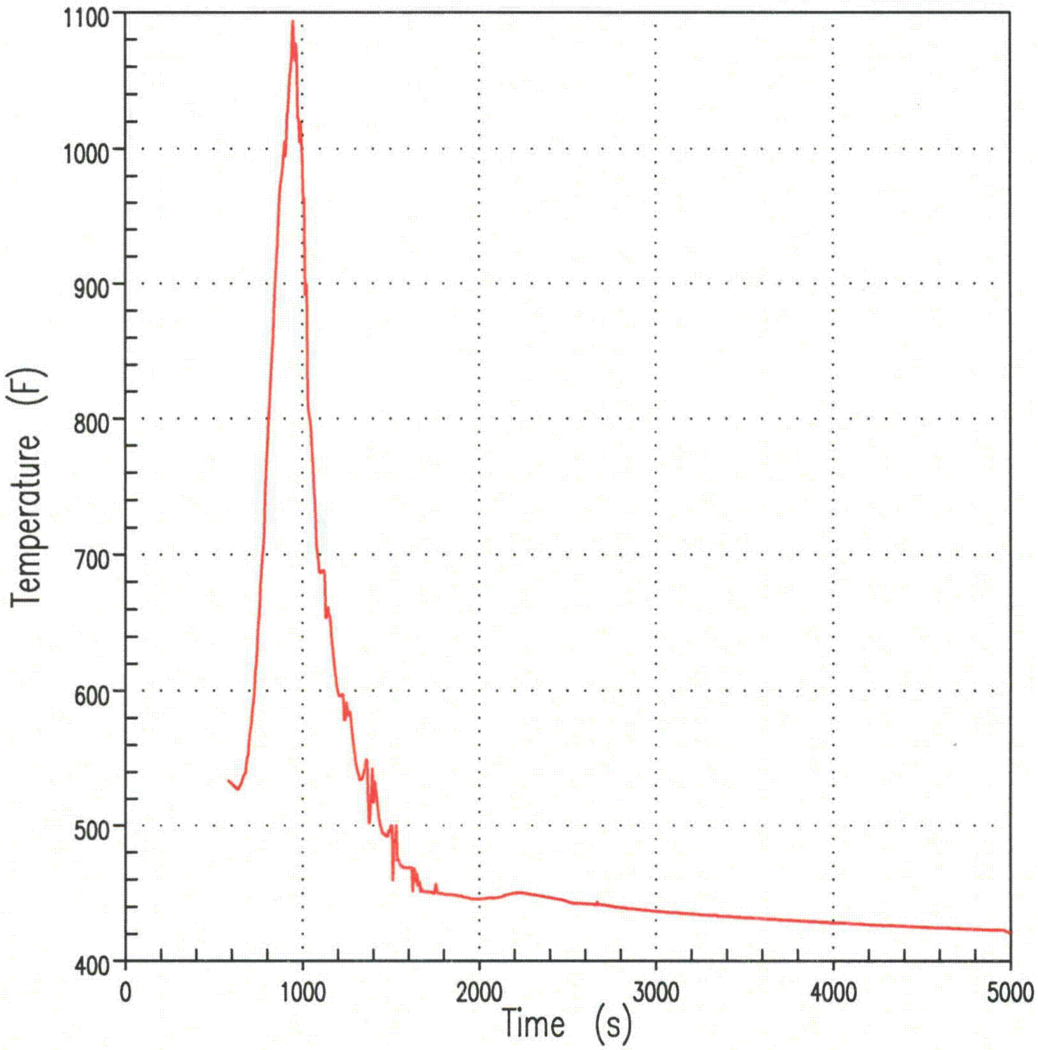

Figure 16
4-inch Break
Hot Spot Fluid Temperature at PCT Elevation (11.25 ft)

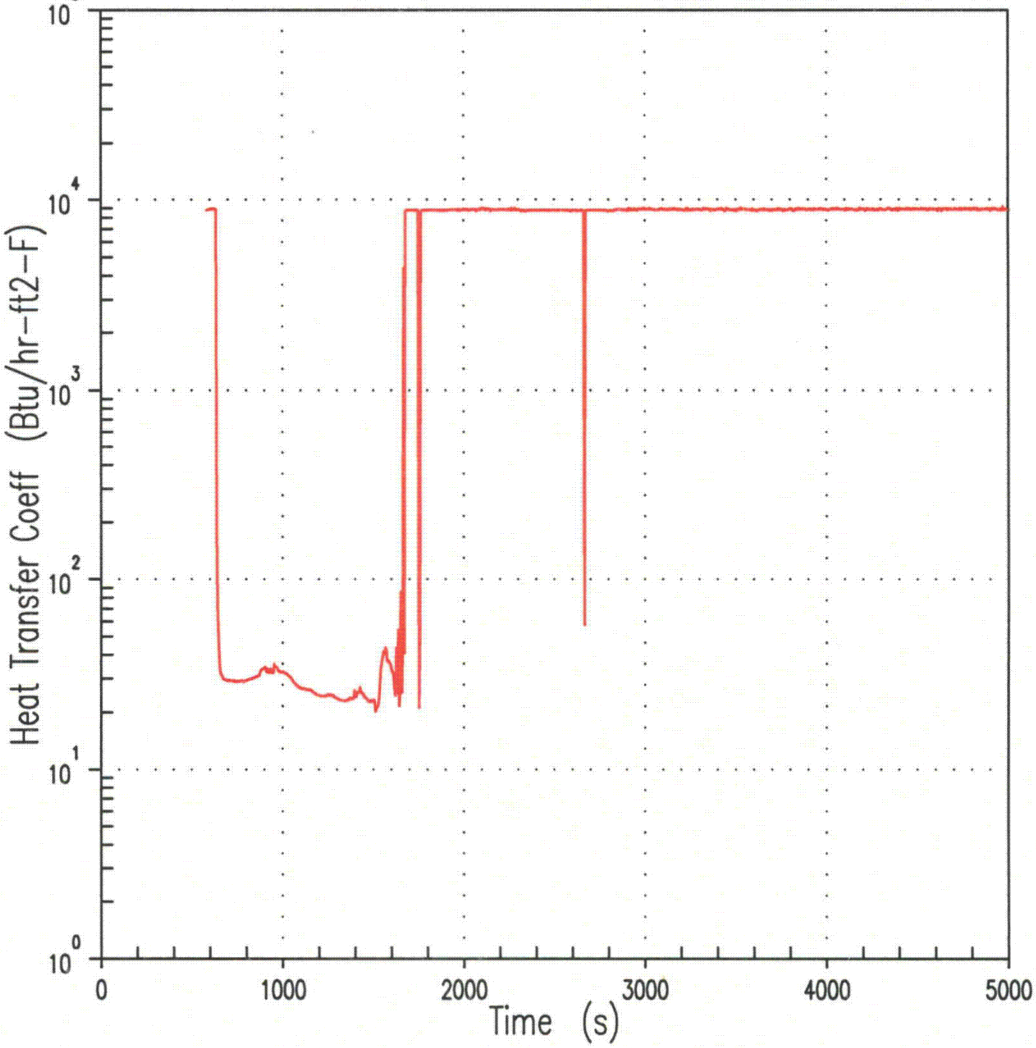

Figure 17
4-inch Break
Rod Film Heat Transfer Coefficient at PCT Elevation (11.25 ft)

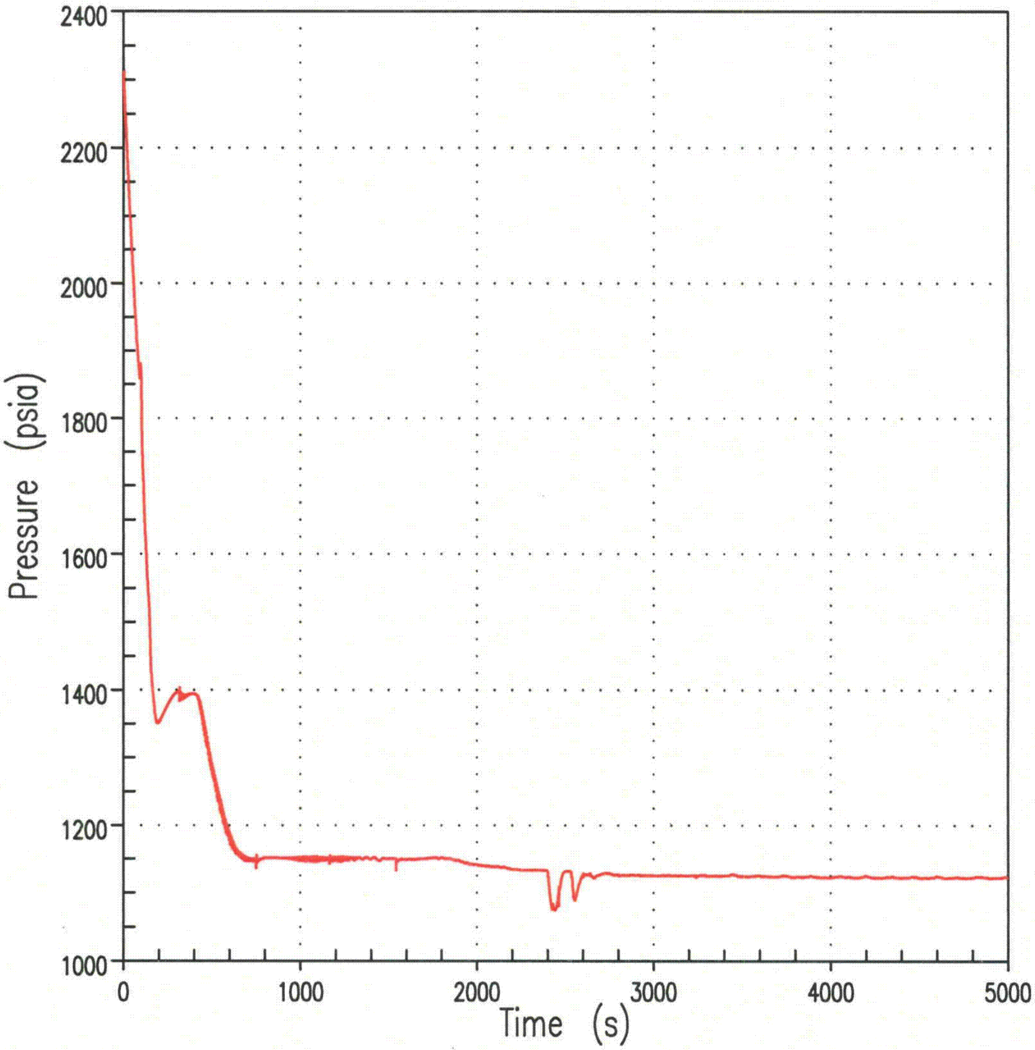

Figure 18
1.5-inch Break
RCS Pressure

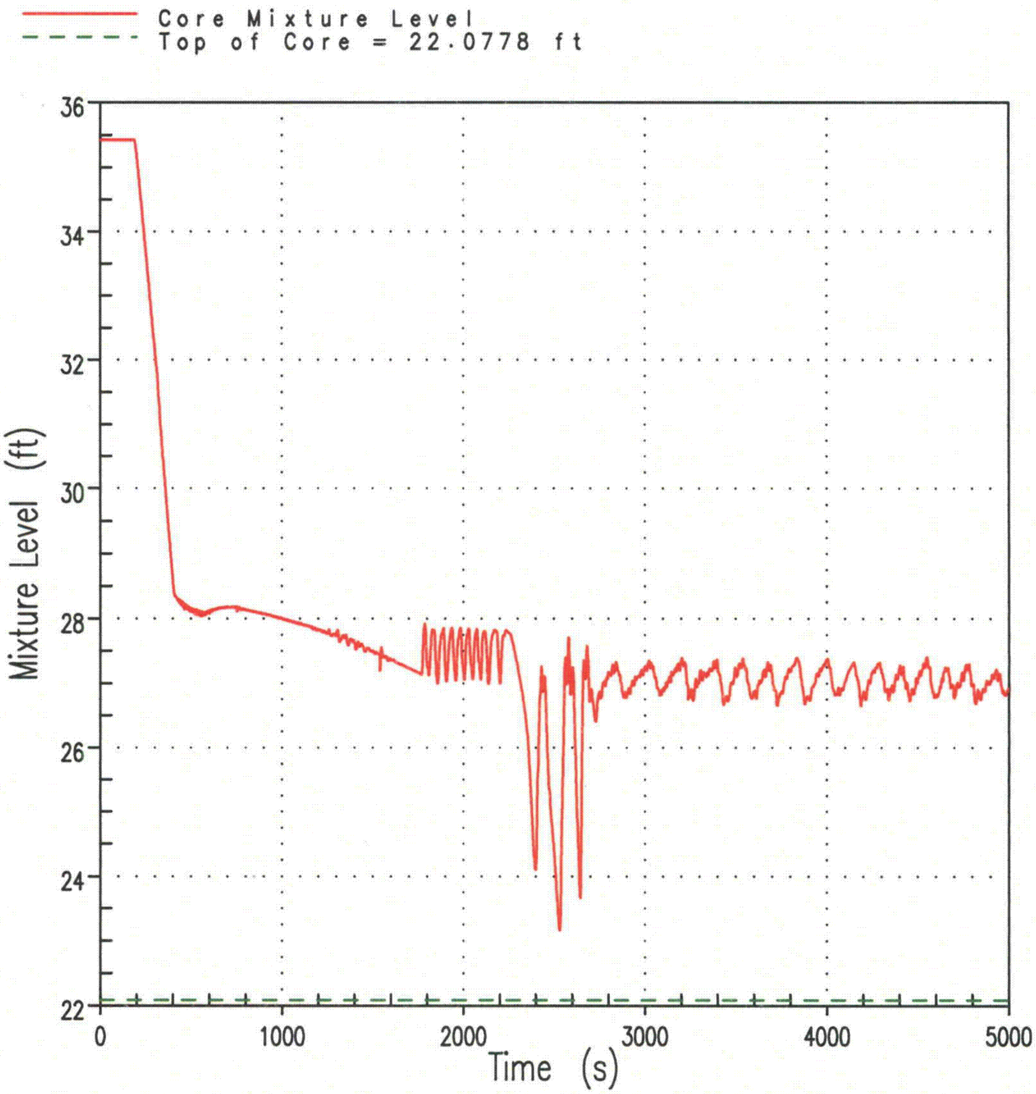

Figure 19
1.5-inch Break
Core Mixture Level

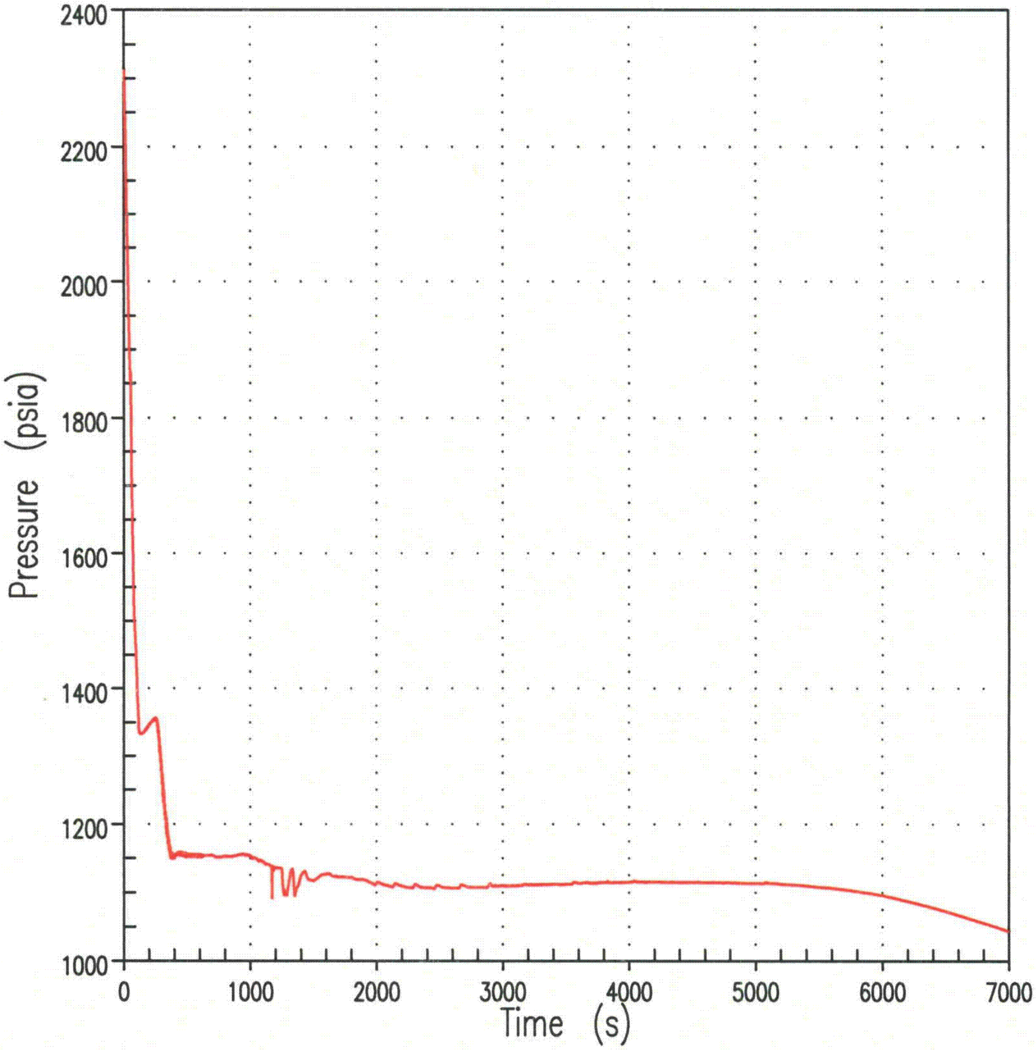

Figure 20
2-inch Break
RCS Pressure

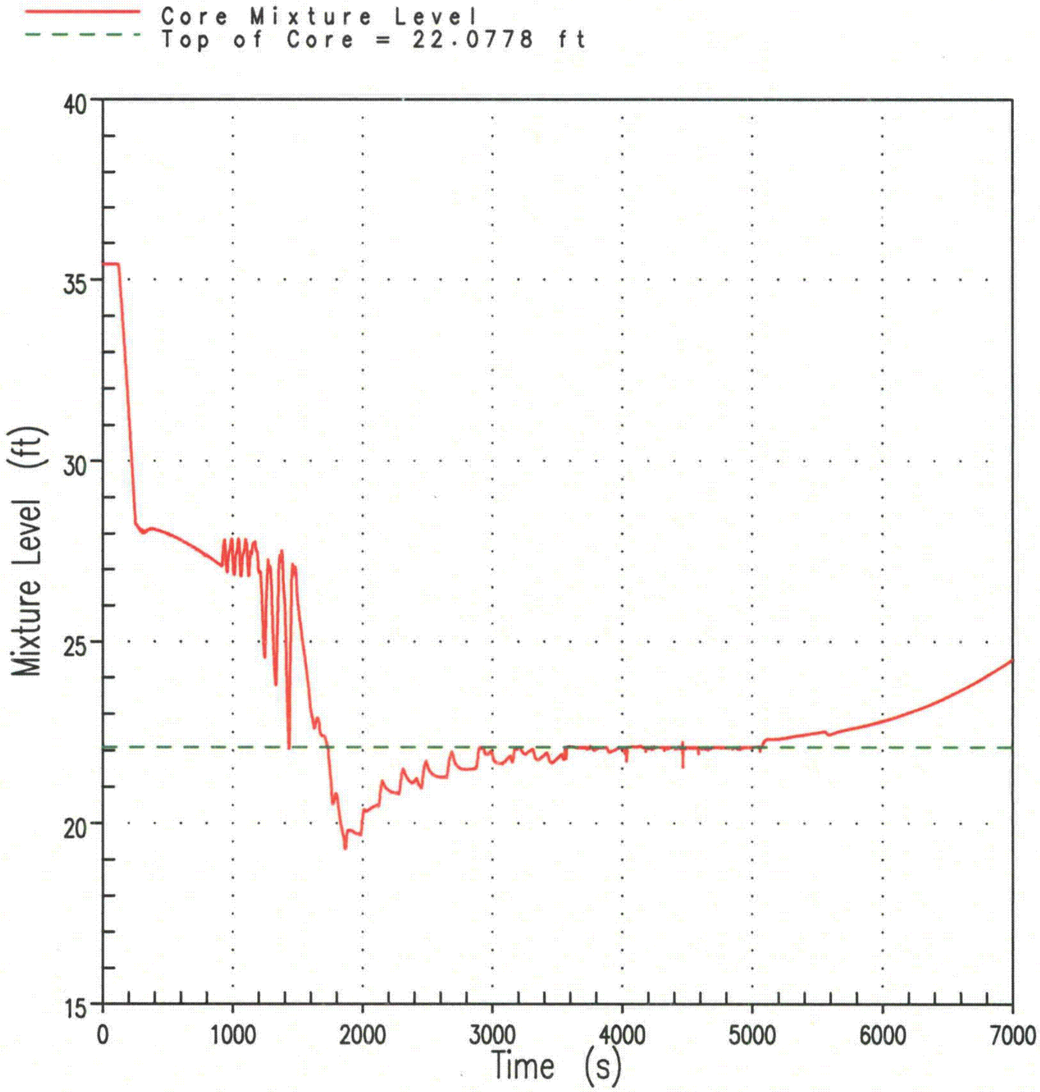

Figure 21
2-inch Break
Core Mixture Level

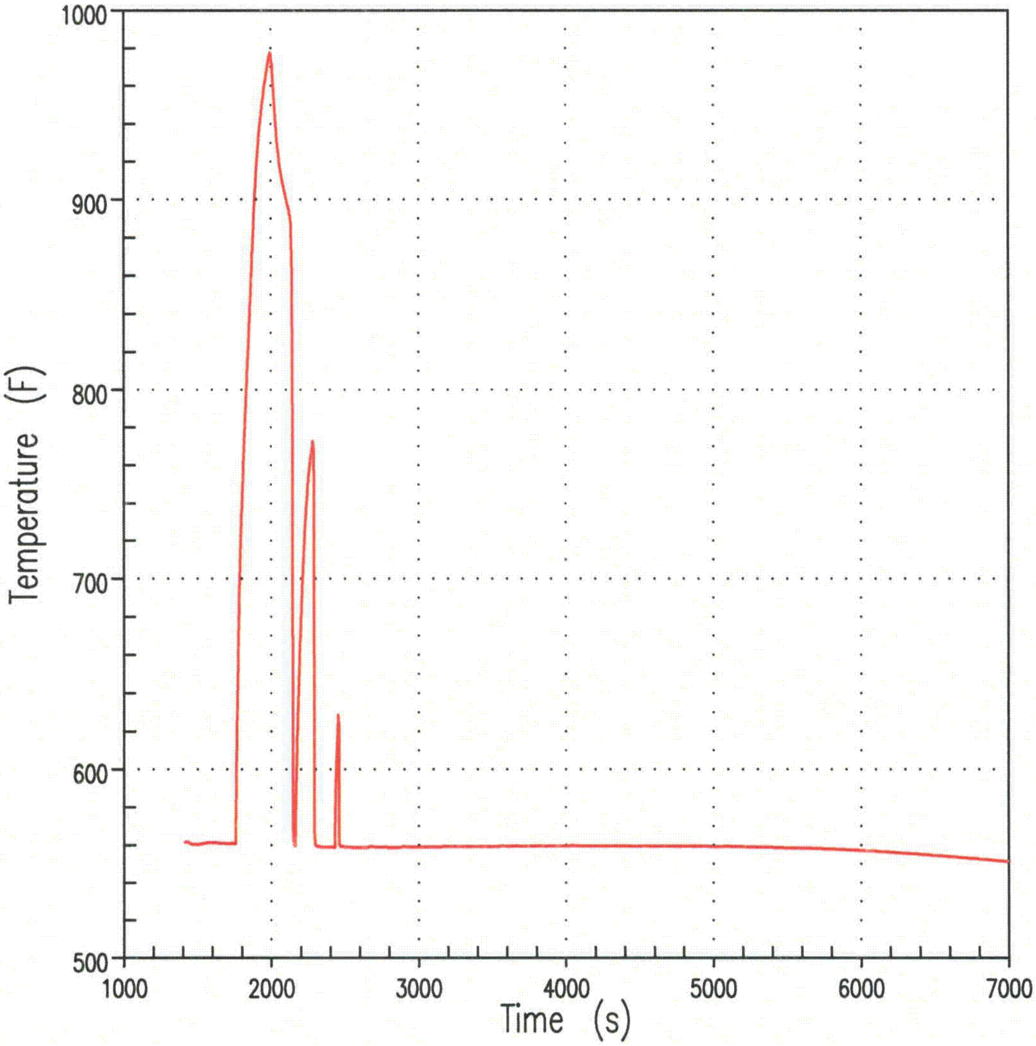

Figure 22
2-inch Break
Clad Temperature at PCT Elevation (11.00 ft)

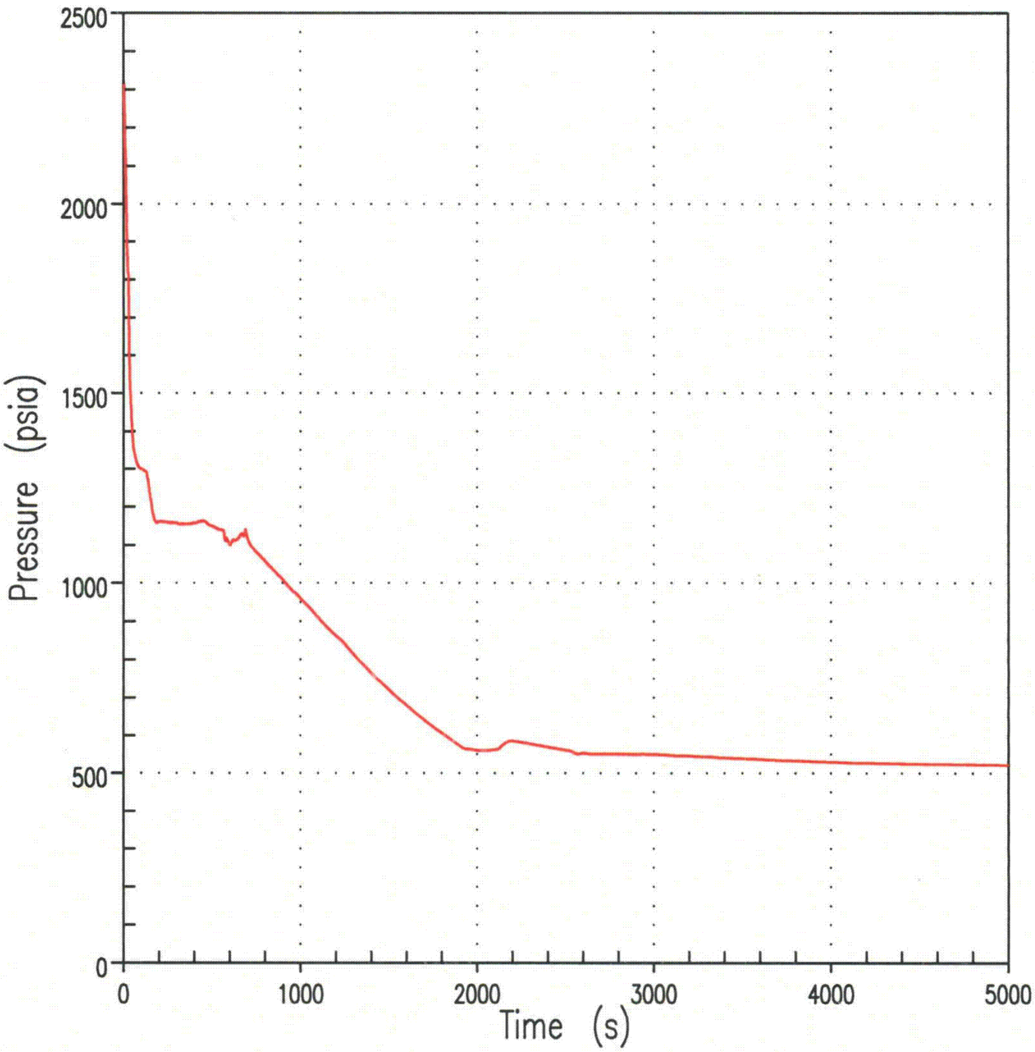

Figure 23
3-inch Break
RCS Pressure

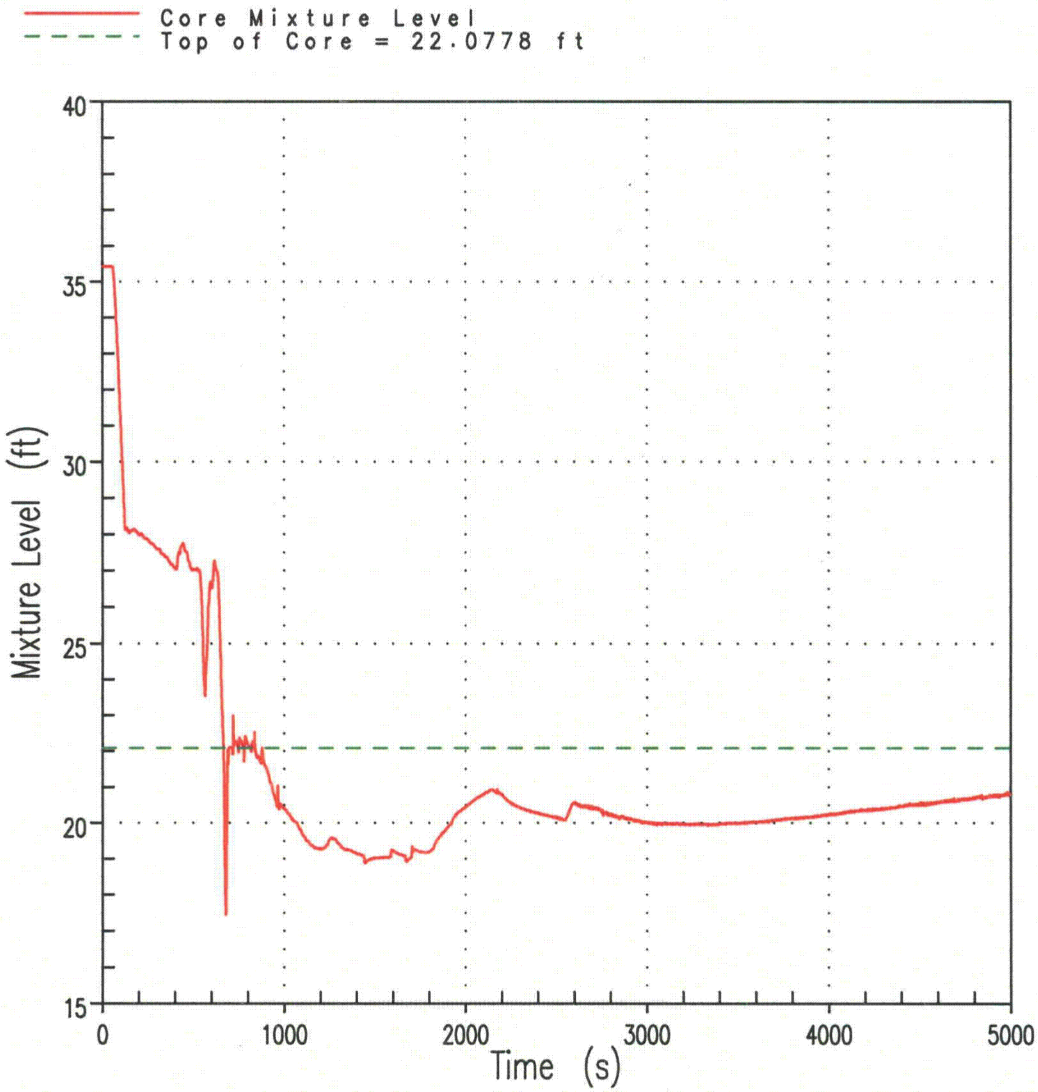

Figure 24
3-inch Break
Core Mixture Level

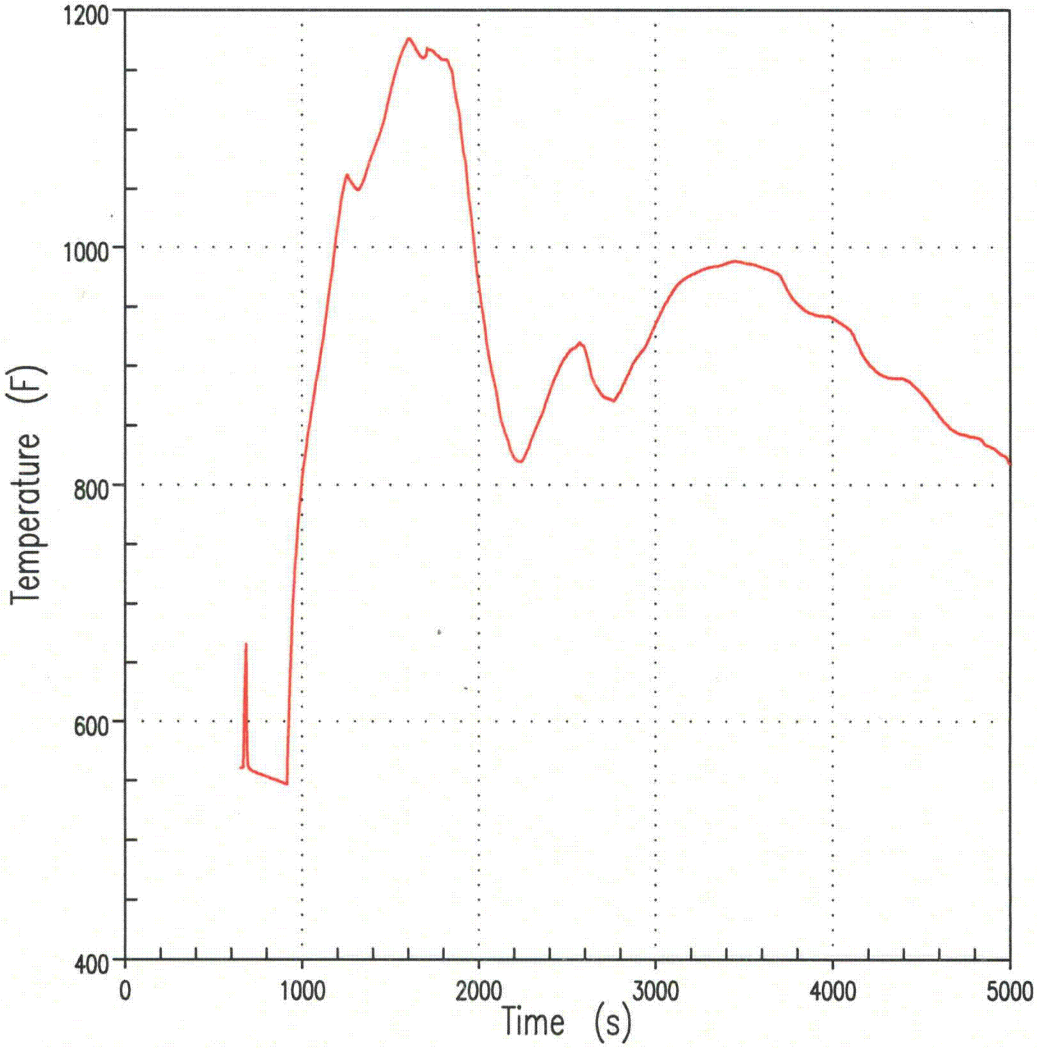

Figure 25
3-inch Break
Clad Temperature at PCT Elevation (11.25 ft)

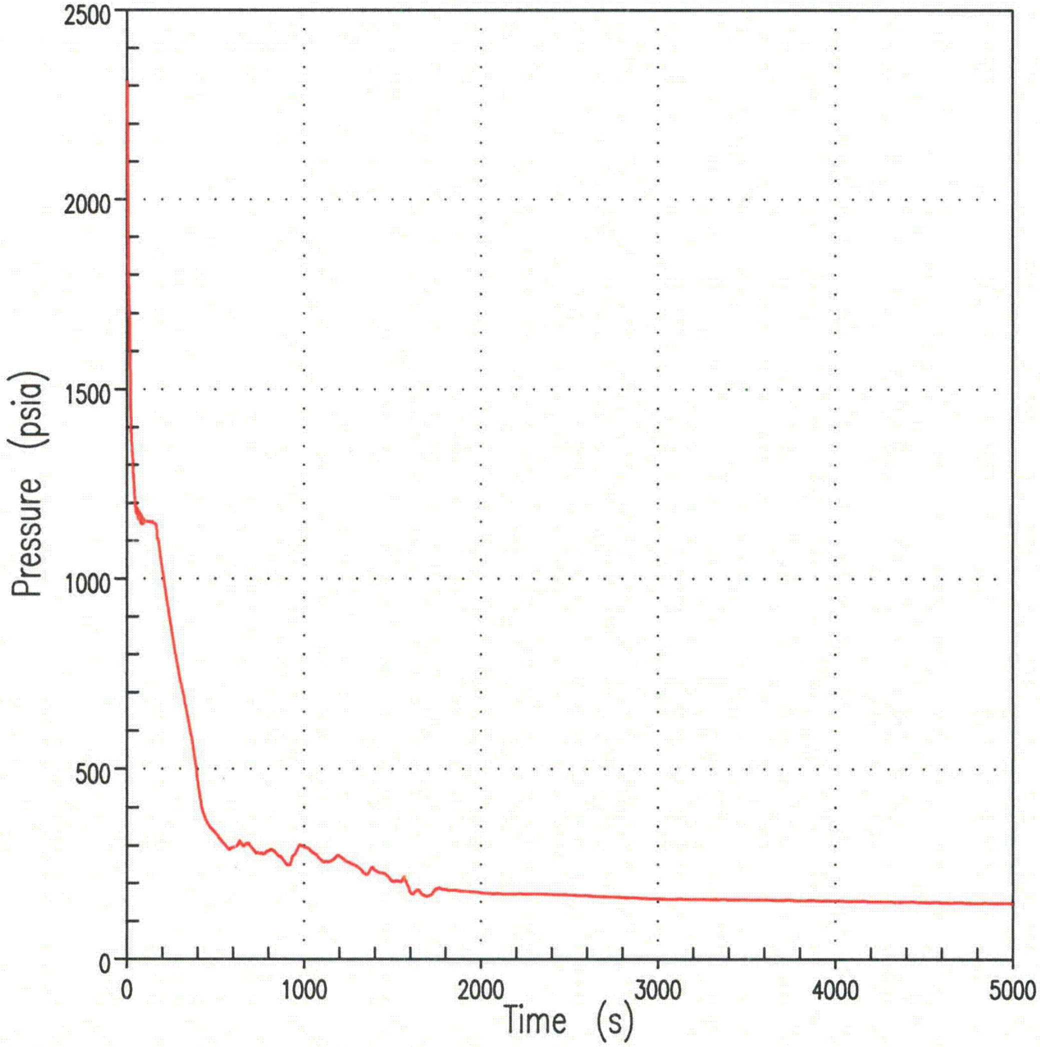

Figure 26
6-inch Break
RCS Pressure

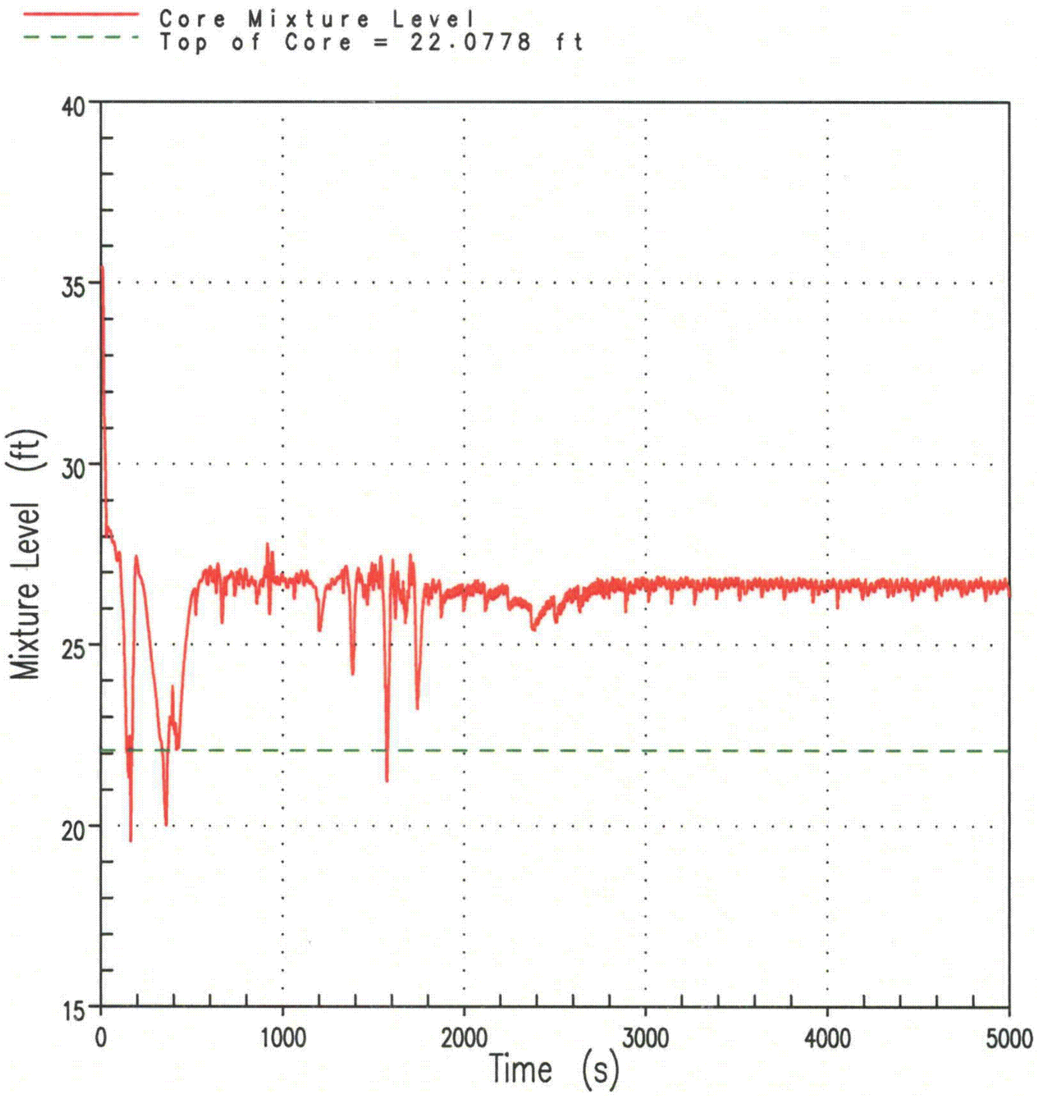

Figure 27
6-inch Break
Core Mixture Level

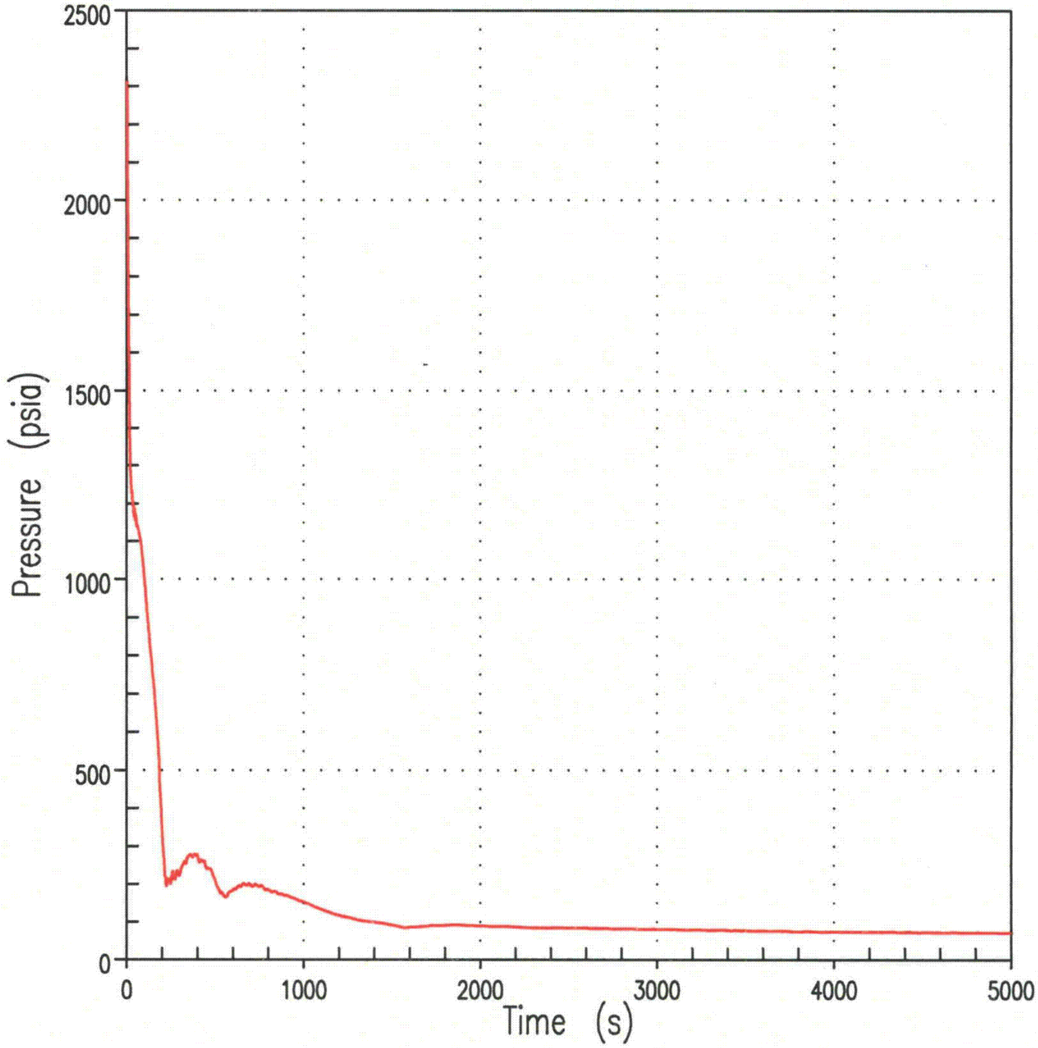

Figure 28
8.75-inch Break
RCS Pressure

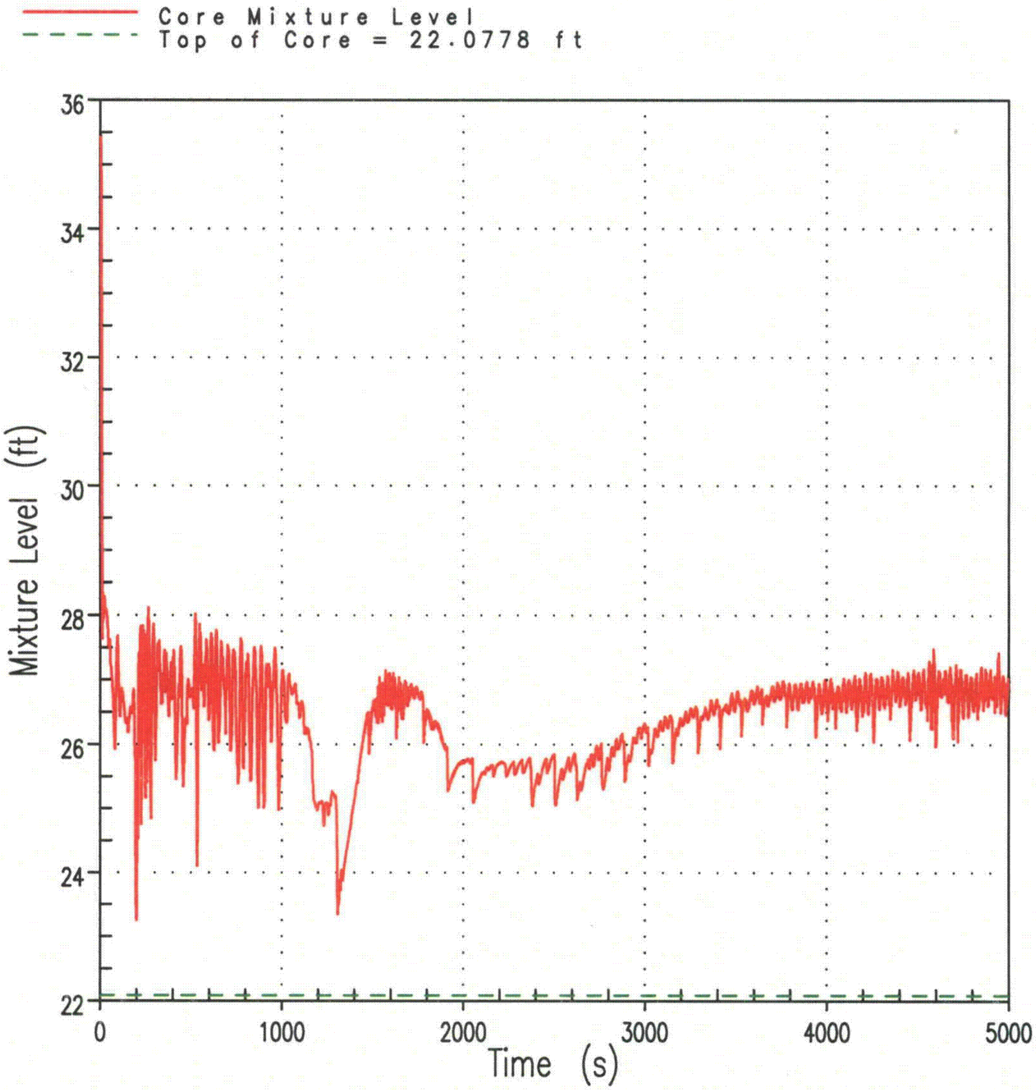

Figure 29
8.75-inch Break
Core Mixture Level