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United States Nuclear Regulatory Commission
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Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Additional Information Supporting the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

References:
(1) Letter from R. M. Krich (Commonwealth Edison Company) to U.S.NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000

(2) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations," dated November 27, 2000

(3) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for Exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," dated December 8, 2000

In Reference 1, we submitted the "Power Uprate Licensing Report for Byron Station and Braidwood Station." Subsequent to that submittal, a discrepancy was discovered in the Small Break Loss of Coolant Accident (SBLOCA) analysis, Section 6.1.1, "Small Break LOCA," of the Licensing Report. A non-conservative input was discovered in Table 6.1.1-1a, "Safety Injection Flows Used in the Small Break LOCA Analysis." A linear interpolation for safety injection (SI) flow was assumed in the analysis for the Reactor Coolant System (RCS) pressure range of 1200 to 1300 psia. However, specifically modeling the RCS pressure data points at 1250 and 1275 psia resulted in a slightly lower SI flow than predicted by the linear interpolation. Westinghouse Electric Company, LLC, the reactor vendor, performed an evaluation of this discrepancy. The evaluation resulted in a limiting peak cladding temperature (PCT) of 1624 °F

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and 1627 °F for the two-inch, low-RCS average temperature (i.e., T_{ave}) condition break for the Unit 1 and Unit 2 analyses, respectively. This is an increase of 22 °F for the Unit 1 limiting PCT and 13 °F for the Unit 2 limiting PCT as compared to the original limiting values of 1602 °F for Unit 1 and 1614 °F for Unit 2. The two-inch, low-T_{ave} condition break remained the limiting case for Unit 1; however, the Unit 2 limiting condition shifted from the three-inch, hi-T_{ave} condition break to the two-inch, low-T_{ave} condition break. The other break sizes of 1.5, three and four inches, were determined to be non-limiting when compared to the two-inch break. The overall results of the SBLOCA analyses remained acceptable. The revised pages of Licensing Report Section 6.1.1 are included in Attachment 1.

A typographical error was also identified on Licensing Report Table 6.5.5-4, "Results for Byron/Braidwood Unit 1 Outside Containment Cases from 102% Power with [Auxiliary Feedwater] AFW Failure." The value for Case D, peak steam temperature at or before steam line isolation was corrected to 384.0 °F vice the original value of 383.1 °F. The corrected table is included in Attachment 2.

In Reference 2, we responded to an NRC request for additional information. In our response to Question G.1, we indicated that the feedwater line break accident was analyzed using the RCS thick-metal mass heat transfer model from the LOFTRAN computer program. This specific heat transfer model has not been previously reviewed and approved by the NRC for Byron and Braidwood Stations. In Attachment 3, we are providing a copy of WCAP-7907-S1-P, Revision 1, "LOFTRAN Code Description, Supplement 1 – LOFTRAN Thick Metal Mass Heat Transfer Models," dated January 2001, for your review. WCAP-7907-S1-P, Revision 1, describes the LOFTRAN thick metal mass heat transfer model used in the analysis for the feedwater line break event supporting the Byron Station and Braidwood Station Power Uprate Program. It also includes information supporting the validation of the model. This methodology is also applicable to the Loss of Normal Feedwater event. WCAP-7907-S1-P contains information proprietary to Westinghouse Electric Company, LLC. Therefore, we are requesting that this information be withheld from public disclosure. Accordingly, an affidavit signed by an authorized representative of the Westinghouse Electric Company, the owner of the information, is provided in Attachment 3 and sets forth the basis on which the information may be withheld from public disclosure by the NRC and addressing the considerations listed in paragraph (b)(4) of 10 CFR 2.790, "Public inspections, exemptions, requests for withholding." A proprietary information notice and copyright notice are also provided in Attachment 3. A non-proprietary version of the WCAP is included in Attachment 4.

On January 31, 2001, a telephone conference call was held between members of the NRC and the Exelon Generation Company (EGC), LLC, organizations to discuss questions pertaining to meteorological data related to the Byron and Braidwood Stations Power Uprate License Amendment Request. Our responses to these questions are documented in Attachment 5.

In Reference 3, we requested an exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." Specifically, the requested exemption would allow the use of American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code, Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels,

Section XI, Division 1," Case N-640, "Alternative Requirement Fracture Toughness for Development of [pressure – temperature] P-T Limit Curves for ASME B&PV Code Section XI, Division 1," and Westinghouse Electric Company Report, WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating [pressurized water reactor] PWR and [boiling water reactor] BWR Plants," in calculating the reactor pressure vessel (RPV) P-T limits. Based on subsequent conversations with the NRC, it appears that NRC approval of the use of WCAP 15315 may not occur in time to support anticipated uprated power operations at Byron and Braidwood Stations. Therefore, the RPV P-T limit curves for Unit 1 and 2 at Byron Station and Braidwood Station have subsequently been generated using the existing NRC approved methodologies in lieu of incorporating Code Case N-588, Code Case N-640 and WCAP 15315 into the P-T limit curve methodology. The new P-T curves generated with the existing NRC approved methodology will adequately support operations at uprated power conditions for all units at Byron and Braidwood Stations. We would request that the NRC complete its review and approval of Code Case N-588, Code Case N-640, and WCAP 15315.

Should you have any questions or concerns regarding this information, please contact Mr. J. A. Bauer at (630) 663-7287.

Respectfully,

R. M. Krich
Director – Licensing
Mid-West Regional Operating Group

Attachment 1:	Power Uprate Licensing Report for Byron Station and Braidwood Station – Revised Small Break LOCA Analysis, Section 6.1.1
Attachment 2:	Power Uprate Licensing Report for Byron Station and Braidwood Station – Revised Table 6.5.5-4, "Results for Byron/Braidwood Unit 1 Outside Containment Cases from 102% Power with AFW Failure"
Attachment 3:	Affidavit and WCAP-7907-S1-P, Revision 1, "LOFTRAN Code Description, Supplement 1 – LOFTRAN Thick Metal Mass Heat Transfer Models" (proprietary)
Attachment 4:	WCAP-7907-S1-NP, Revision 1, "LOFTRAN Code Description, Supplement 1 – LOFTRAN Thick Metal Mass Heat Transfer Models" (non-proprietary)
Attachment 5:	Meteorological Data Information Supporting the Byron Station and Braidwood Station Power Uprate License Amendment Request

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
Office of Nuclear Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
EXELON GENERATION COMPANY, LLC	)	**Docket Numbers**
BYRON STATION UNITS 1 AND 2	)	**STN 50-454 AND STN 50-455**
BRAIDWOOD STATION UNITS 1 AND 2	)	**STN 50-456 AND STN 50-457**

SUBJECT: **Additional Information Supporting the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich

Director – Licensing

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 20th day of February, 2001.

Notary Public

" OFFICIAL SEAL "
Timothy A. Byam
Notary Public, State of Illinois
My Commission Expires 11/24/2001

ATTACHMENT 1

Power Uprate Licensing Report for Byron Station and Braidwood Station

Revised Small Break LOCA Analysis, Section 6.1.1

6.0 NSSS ACCIDENT ANALYSES

This section provides the results of the analyses and/or evaluations that were performed for the Nuclear Steam Supply System (NSSS) accident analyses in support of the Power Uprate Program. The accident analysis areas addressed in this section include:

- Small-Break Loss-of-Coolant Accident (LOCA), Hot Leg Switchover, and Post-LOCA Long Term Cooling
- Non-LOCA Events
- Steam Generator Tube Rupture Transient
- LOCA Containment Integrity
- Main Steamline Break Consequences
- LOCA Hydraulic Forces
- Radiological Consequences (Doses)

The Large-Break LOCA submittal, using Best Estimate Methodology, is being prepared separately from this report and will be provided later.

The detailed results and conclusions of each analysis are presented within each subsection.

6.1 Loss-of-Coolant Accident (LOCA) Transients

6.1.1 Small-Break LOCA

6.1.1.1 Introduction

This section contains information regarding the Small-Break Loss-of-Coolant Accident (SBLOCA) analysis and evaluations performed in support of the uprate project for Byron and Braidwood Units 1 and 2. The purpose of analyzing the Small-Break LOCA is to demonstrate conformance with the 10 CFR 50.46 (Reference 1) requirements for the conditions associated with the uprating. Important input assumptions, as well as analytical models and analysis

methodology for the Small-Break LOCA, 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 Small-Break LOCA would be exceeded due to the uprated power and assumed plant parameters.

6.1.1.2 Input Parameters and Assumptions

The important plant conditions and features are listed in Table 6.1.1-1. Several additional considerations that are not identified in Table 6.1.1-1 are discussed below.

Figure 6.1.1-1 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 +13%). 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 2-line segment K(Z) envelope based on the peaking factors shown in Table 6.1.1-1.

Figure 6.1.1-2 provides the SI flow versus pressure curve modeled in the Small-break LOCA analysis. The flows shown in Figure 6.1.1-2 account for a 5% flow reduction to account for future pump degradation. The flow from one Safety Injection (SI) pump and one Centrifugal Charging (CV) pump were assumed in this analysis.

6.1.1.3 Description of Analyses/Evaluations Performed

<u>Analytical Model</u>

For Small-breaks, the NOTRUMP computer code (References 2 and 3) 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. The NOTRUMP computer code is a one-dimensional general network code incorporating a number of advanced features. Among these advanced features 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. The NOTRUMP Small-break LOCA Emergency Core Cooling System (ECCS) Evaluation Model was developed to determine the RCS response to design basis Small-break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 4).

The RCS model is nodalized into volumes interconnected by flow paths. The broken loop is modeled explicitly, while the intact loops are lumped together into a second loop. 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 Small-break LOCA. 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 a version of the LOCTA-IV code (Reference 5) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. Figure 6.1.1-3 illustrates the code interface for the Small-break Model.

<u>Analysis</u>

This uprate analysis has considered 16 different break cases as indicated by the result Tables 6.1.1-7 through 6.1.1-10. A break spectrum of 1.5, 2, 3, and 4-inch breaks was considered for both Units 1 and Units 2 at Hi and Low Tavg conditions. The Low Tavg 2 inch break remained limiting for Units 1 and a shift in limiting break size and conditions to the Low Tavg 2 inch break occurred for the Units 2.

The most limiting single active failure assumed 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 assumed to occur coincident with reactor trip. This means that credit may be taken for at most one high head safety injection (HHSI) pump, one charging pump (CV/SI), and one low head, or residual heat removal (RHR) pump. In this analysis, one HHSI pump and one CV/SI pump are modeled. The RHR is not considered in Small-break LOCA analyses because the shutoff head is lower than the RCS pressure during

the portion of the transient considered here. The Small-break LOCA analysis performed for the Byron/Braidwood uprate project assumes ECCS flow is delivered to both the intact and broken loops at the RCS backpressure. The broken and intact loop SI flows are illustrated in Figure 6.1.1–2. The assumption of LOOP and the failure of a diesel generator to start as the limiting single failure for Small-break LOCA is part of the NRC approved methodology and does not change as a result of the uprated conditions. The single failure assumption is extremely limiting due to the fact that one train of ECCS, one motor driven auxiliary feedwater (AF) pump, and power to the reactor coolant pumps (RCPs) are all 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 assumed to be in a full power (102%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions assumed in the analysis are given in Table 6.1.1-1. 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. In the case of continuous heat addition to the secondary during a period of quasi-equilibrium, an increase in the secondary system pressure 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 1857 psia, is reached. LOOP is assumed 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 is delayed 40 seconds after the occurrence of the low pressure condition. This delay accounts for signal processing, diesel generator start up and emergency power bus loading consistent with the assumed loss-of-offsite power coincident with reactor trip, as well as the pump acceleration 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 Small-break LOCA analysis for the boron content of the injection water. In addition, credit is taken in the Small-break LOCA analysis for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position. A rod drop time of 2.7 seconds was assumed while also considering an additional 2 seconds for the signal processing delay time. An additional 1.3 second delay has also been modeled for added conservatism. Therefore, a total delay time of 6.0 seconds from the time of reactor trip signal to full rod insertion was used in the Small-break LOCA analysis.

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

During the earlier part of the Small-break transient (prior to the assumed 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 analysis is terminated when the 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 assumed to be isolated in 8 seconds following the generation of the low pressurizer pressure SI signal, consisting of a 2 second signal delay time and a 6 second main feedwater isolation valve stroke time. Additional makeup water is also provided to the secondary using the auxiliary feedwater (AF) system. An AF actuation signal is modeled off the low pressurizer pressure SI signal, resulting in the delivery of AF system flow 90 seconds after the generation of the SI signal. 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 (minimum), as is the case in the 3-inch and 4-inch break cases, the cold leg accumulators begin to inject borated water into the reactor coolant loops. In the case of the 1.5 and 2-inch breaks however, the transient is terminated without the aid of accumulator injection.

6.1.1.4 Acceptance Criteria for Analyses/Evaluations

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 at uprated conditions.

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 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, but are assessed in Section 6.1.3 as part of the evaluation of ECCS performance.

The criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA.

6.1.1.5 Results

In order to determine the conditions that produced the most limiting Small Break LOCA case (as determined by the highest calculated peak cladding temperature), a total of 8 break cases were examined for each of the Units 1 and Units 2. These cases included the investigation of variables, including break size and RCS average temperature, to ensure that the most severe postulated Small Break LOCA event was analyzed. The following discussions provide insight into the analyzed conditions.

Limiting Temperature Conditions

For Byron/Braidwood Units 1 and Units 2, the temperature window analyzed was based on a nominal vessel average temperature range of 565°F to 598°F, which includes ±10°F to bound uncertainties. The analysis showed that for both units, the Low T_{AVG} 2-inch case is limiting. The limiting case transient for each pair of units will be discussed below.

Byron/Braidwood Units 1 SBLOCA Results Discussion

The results of 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 are shown in Tables 6.1.1-2 and 6.1.1-3. Inherent in the limiting Small Break analysis are several input assumptions (see Section 6.1.1.2 and Table 6.1.1-1), while Tables 6.1.1-7 and 6.1.1-8 provide the key transient event times.

For the Small Break LOCA uprate analysis, the limiting case for Units 1 was the Low T_{avg} 2-inch break case. A summary of the transient response for the limiting Units 1 case is shown in Figures 6.1.1-4 through 6.1.1-14. These figures present the response of the following parameters.

- RCS Pressure
- Core Mixture Level
- Top Core Node Vapor Temperature
- Broken Loop and Intact Loop Secondary Side Pressure
- Break Vapor Flow Rate
- Break Liquid Flow Rate
- Broken Loop and Intact Loop Accumulator Flow
- Pumped Safety Injection Mass Flow Rate for the Intact and Broken Loops
- Peak Cladding Temperature
- Hot Spot Fluid Temperature
- Hot Spot Rod Surface Heat Transfer Coefficient

Upon initiation of the limiting Low T_{avg} 2-inch break for Units 1, there is an initial rapid depressurization of the RCS followed by an intermediate equilibrium at around 1250 psia (see Figure 6.1.1-4). Following the equilibrium, the RCS pressure gradually depressurizes but never reaches the accumulator injection setpoint of 600 psia (see Figure 6.1.1-10). 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 fission products decay is accomplished via a two-phase mixture level covering the core. The core mixture level and cladding temperature transient plots for the Units 1 Low T_{avg} 2-inch break calculations are illustrated in Figures 6.1.1-5 and 6.1.1-12. These figures show that the peak cladding 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 6.1.1-6).

A comparison of the flow provided by the safety injection system to the intact and broken loops can be found in Figure 6.1.1-11. The cold leg break vapor and liquid mass flow rates are provided in Figures 6.1.1-8 and 6.1.1-9 respectively. Figures 6.1.1-13 and 6.1.1-14 provide additional information on the fluid temperature at the hot spot and hot rod surface heat transfer coefficient at the hot spot, respectively. Figure 6.1.1-7 depicts the secondary side pressure for both the intact and broken loops for the Units 1 Low T_{avg} 2-inch break case.

Safety Injection Evaluation

Subsequent to completion of the break spectrum study, a non-conservative discrepancy was discovered in the safety injection flows of Table 6.1.1-1a. The data points at 1250 and 1275 psia were not included in the original cases and thus the flows modeled were non-conservatively higher in the analysis. The limiting 2 inch Low Tavg case was performed with the revised safety injection data and resulted in a slight increase in PCT. Tables 6.1.1-3 and 6.1.1-8 have been updated to reflect the results. Also, Figures 6.1.1-4 through 6.1.1-14 have been updated as well. The impact of the SI discrepancy has also been evaluated on the other break sizes and resulted in a negligible impact on those. Thus, the original results demonstrated herein remain applicable to those break sizes.

Additional Break Cases

Studies documented in Reference 6 have determined that the limiting small-break transient occurs for breaks of less than 10 inches in diameter in the cold leg. To ensure that the 2-inch diameter break was the most limiting, calculations were also performed with break equivalent diameters of 1.5, 3, and 4 inches. The results of the break spectrum cases are given in Tables 6.1.1-2 and 6.1.1-3. Figures 6.1.1-15 through 6.1.1-23 refer to the non-limiting break cases analyzed for Units 1 at the Low T_{avg} conditions. Figures 6.1.1-24 through 6.1.1-35 refer to the non-limiting break cases analyzed for Units 1 at the High T_{avg} conditions. The following plots have been included in Figures 6.1.1-15 through 6.1.1-35.

1. RCS Pressure Transient
2. Core Mixture Level
3. Peak Cladding Temperature

The PCTs for each of the breaks considered are shown in Tables 6.1.1-2 and 6.1.1-3, these PCTs are less than the limiting 2-inch Low T_{avg} break case.

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

1. The RCS pressure is gradually decaying
2. The net mass inventory is increasing
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.

Byron/Braidwood Units 2 SBLOCA Results Discussion

The Units 2 PCT results are shown in Tables 6.1.1-4 and 6.1.1-5. Inherent in the limiting Small Break analysis are several input assumptions (see Section 6.1.1.2 and Table 6.1.1-1), while Tables 6.1.1-9 and 6.1.1-10 provide the key transient event times. For the Small Break LOCA uprate analysis, the limiting case for Units 2 was the Low Tavg 2-inch break case. A summary of the transient response for the limiting Units 2 case is shown in Figures 6.1.1-36 through 6.1.1-46. These figures present the response of the following parameters.

- RCS Pressure
- Core Mixture Level
- Top Core Node Vapor Temperature
- Broken Loop and Intact Loop Secondary Side Pressure
- Break Vapor Flow Rate
- Break Liquid Flow Rate
- Broken Loop and Intact Loop Accumulator Flow
- Pumped Safety Injection Mass Flow Rate for the Intact and Broken Loops
- Peak Cladding Temperature
- Hot Spot Fluid Temperature
- Hot Spot Rod Surface Heat Transfer Coefficient

Upon initiation of the limiting Low Tavg 2-inch break for Units 2, there is an initial rapid depressurization of the RCS followed by an intermediate equilibrium at around 1250 psia (see Figure 6.1.1-36). Following the equilibrium, the RCS pressure gradually depressurizes but never reaches the accumulator injection setpoint of 600 psia (see Figure 6.1.1-42). 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 fission products decay is accomplished via a two-phase mixture level covering the core. The core mixture level and cladding temperature transient plots for the Units 2 Low Tavg 2-inch break calculations are illustrated in Figures 6.1.1-37 and 6.1.1-44. These figures show that the peak cladding 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 6.1.1-38).

A comparison of the flow provided by the safety injection system to the intact and broken loops can be found in Figure 6.1.1-43. The cold leg break vapor and liquid mass flow rates are provided in Figures 6.1.1-40 and 6.1.1-41, respectively. Figures 6.1.1-45 and 6.1.1-46 provide additional information on the fluid temperature at the hot spot and hot rod surface heat transfer coefficient at the hot spot, respectively. Figure 6.1.1-39 depicts the secondary side pressure for both the intact and broken loops for the Units 2 Low Tavg 2-inch break case.

<u>Safety Injection Evaluation</u>

Subsequent to completion of the break spectrum study, a non-conservative discrepancy was discovered in the safety injection flows of Table 6.1.1-1a. The data points at 1250 and 1275 psia were not included in the original cases and thus the flows modeled were non-conservatively higher in the analysis. The prior limiting case was the 3 inch High Tavg case. However, the effect of the safety injection discrepancy has a negligible impact on the 3 inch cases. Because changes in safety injection have a greater effect on smaller break sizes and the 2 inch Low Tavg case was only 10°F lower than the 3 inch High Tavg case, the 2 inch Low Tavg case was performed with the revised safety injection data. This resulted in a higher PCT and thus a shift in the limiting break size to the 2 inch Low Tavg case. Tables 6.1.1-5 and 6.1.1-10 have been updated to reflect the results. Also, the prior 3 inch High Tavg limiting case

Figures 6.1.1-36 through 6.1.1-46 have been updated to reflect the new 2 inch Low Tavg limiting case. Note that for the Zirc-4 cladding evaluation below, the results are based on the High Tavg 3 inch prior limiting case. Although the new Low Tavg 2 inch limiting case has not been performed with Zirc-4 cladding, the prior results are being applied to the new limiting case because the effects are expected to be similar. The impact of the SI discrepancy has also been evaluated on the other break sizes and resulted in a negligible impact on those. Thus, the original results demonstrated herein remain applicable to those break sizes.

Additional Break Cases

Studies documented in Reference 6 have determined that the limiting small-break transient occurs for breaks of less than 10 inches in diameter in the cold leg. To ensure that the 2-inch diameter break was the most limiting, calculations were also performed with break equivalent diameters of 1.5, 3, and 4 inches. The results of the break spectrum cases are given in Tables 6.1.1-4 and 6.1.1-5. Figures 6.1.1-56 through 6.1.1-58 and Figures 6.1.1-62 through 6.1.1-67 refer to the non-limiting break cases analyzed for Units 2 at the Low T_{avg} conditions. Figures 6.1.1-47 through 6.1.1-55 and Figures 6.1.1-59 through 6.1.1-61 refer to the non-limiting break cases analyzed for Units 2 at the High Tavg conditions. The following plots have been included for these figures.

1. RCS Pressure Transient

2. Core Mixture Level

3. Peak Cladding Temperature

The PCTs of each of the breaks considered are shown in Tables 6.1.1-4 and 6.1.1-5. In each case, the PCTs are less than the limiting 2-inch break case.

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

1. The RCS pressure is gradually decaying

2. The net mass inventory is increasing

3. The core mixture level is recovered

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.

ZIRLO/Zirc-4 Cladding Evaluation

Since ZIRLO and Zirc-4 fuel have different physical characteristics as modeled by the SBLOCTA code, explicit calculations for Zirc-4 fuel have been performed (See Table 6.1.1-6). The Zirc-4 fuel was found to be non-limiting at beginning of life (BOL) conditions for the Units 1 Low T_{avg} 2-inch case. Figure 6.1.1-68 illustrates the PCT plot for the Unit 1 Low T_{avg} Zirc-4 case.

The Zirc-4 fuel was found to be slightly (~1°F) limiting for the Units 2 Hi T_{avg} case. A burnup credit of 6,000 MWD/MTU was taken in order to make the Zirc-4 fuel non-limiting compared to the ZIRLO fuel. This burnup restriction will be tracked in the SPIL current limits from this point forward. The calculated PCT for Zirc-4 fuel at 6000 MWD/MTU was found to be 1601°F (see Figure 6.1.1-69), which is less limiting than the ZIRLO fuel PCT for the Units 2 Hi T_{avg} 3-inch case. Considering the revised SI Flow, the Zirc-4 fuel evaluation applies to the new limiting Unit 2 break.

At the time at which this analysis is implemented, no fresh Zirc-4 fuel is expected to be inserted into the core. All of the Zirc-4 fuel which may be used at uprated operation will have a minimum burnup of one cycle. The Zirc-4 minimum, core-wide, fuel-pin burnup is expected to be well in excess of 6000 MWD/MTU. Therefore, assuming that this is the case, the ZIRLO fuel will be considered more limiting with a PCT of 1614°F in comparison to the 1601°F PCT for the Zirc-4 fuel at 6000 MWD/MTU. This confirmation will have to be explicitly verified as part of the SPIL process when the uprated ZIRLO fuel is being implemented. If this burnup criterion can be satisfied during the reload, as is expected, then no additional PCT penalty will be needed for Zirc-4 fuel.

The fuel temperatures/pressures used in these calculations were based on NRC approved fuel performance code (PAD 3.4) which addresses all the helium release related issues. This analysis has been performed using the most limiting temperature/pressure as calculated for non-IFBA VANTAGE 5 fuel. The standard Westinghouse position is that non-IFBA fuel bounds IFBA fuel for SBLOCA analyses.

6.1.1.6 Conclusions

A break spectrum of 1.5, 2, 3, and 4 inch diameters have been considered at both high and low vessel average temperatures for all Byron and Braidwood Units. A peak cladding temperature of 1624°F was calculated to be limiting for Units 1. This limiting PCT occurred for the 2-inch low T_{avg} break case. Zirc-4 fuel is bounded by ZIRLO fuel for Units 1.

A peak cladding temperature of 1627°F was calculated to be limiting for Units 2. This limiting PCT occurred for the 2-inch Low T_{avg} break case. Beyond 6000 MWD/MTU, PCT for Zirc-4 fuel is bounded by PCT for ZIRLO fuel.

The analyses presented in this section show that the accumulator and safety injection subsystems of the Emergency Core Cooling System, together with the heat removal capability of the steam generator, provide sufficient core heat removal capability to maintain the calculated peak cladding temperatures below the required limit of 10 CFR 50.46.

6.1.1.7 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled 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. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plant," NUREG-0611, January 1980.

5. Bordelon, F. M. et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (proprietary) and WCAP-8305 (non-proprietary), June 1974.

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 6.1.1-1

Input Parameters Used in the Small Break LOCA Analysis

Input Parameter	Value	
Core Rated Thermal Power-100%	3586.6	
Fuel Type	17 X 17 V5+	
Total Core Peaking Factor, F_Q	2.6	
FDH	1.7	
FNZ	1.53	
PHA	1.514	
Maximum Axial Offset	+13%	
Initial RCS Loop Flow	92,000 gpm/loop	
Initial Vessel T_{avg}	Max: 598.0 °F Min: 565.0 °F	
Initial Pressurizer Pressure	2300 psia	
Pump Type	With RCP Weir	
Low Pressurizer Pressure Reactor Trip Signal	1857 psia	
Trip Signal Processing Time	2.0 seconds	
Reactor Trip Delay Time	6.0 seconds	
Aux. Feedwater Temp. (Maximum)	125 °F	
Number and Types of Pumps Available Following a LOOP	1 Diesel Driven	
AF Flow (Minimum)	560 gpm Total to 4 SGs (at 1284 psia or less)	
AF Delay Time (Maximum)	90 seconds	
AF Actuation Signal	LPP SI	
STEAM GENERATORS	BWI SG	D5 SG
Max AF Enthalpy Switchover Purge Volumes, ft^3	160 ft^3	60 ft^3
SGTP (Maximum)	5%	10%
Max. MFW Isolation Delay Time	2 seconds	
MFW Isolation Ramp Time	6 seconds	
MFW Isolation Signal	LPP SI	
Isolation of Steam Line	LPP RT/LOOP	

Table 6.1.1-1 (cont.) Input Parameters Used in the Small Break LOCA Analysis		
Input Parameter	**Value**	
Steam Generator Secondary Water Mass, lbm/SG	111,000	79,194
Pressure Drop from SG to Steam Header	20 psi	
Containment Spray Flowrate for 2 Pumps (Maximum)	9255 gpm	
RWST Deliverable Volume (Minimum)	180888 gallons	
SI Temp at Cold Leg Recirc.	212 °F	
ECCS Configuration	1 IHSI Pump and 1 Charging Pump spill to RCS Pressure	
ECCS Water Temp.	Max: 120 °F	
Safety Injection Signal	1715 psia	
SI Signal Delay Time	40 seconds	
ECCS Flow vs Pressure	See Table 6.1.1-1A	
Initial Accumulator Water/Gas Temperature	130 °F	
Initial Nominal Acc. Water Vol.	950 ft^3	
Min Acc. Cover Press. (With Uncertainty Consideration)	600 psia	

Table 6.1.1-1a Safety Injection Flows Used in the Small Break LOCA Analysis (Flows Account for 5% Reduction Due to Pump Degradation)		
RCS Pressure (psia)	**Intact Loop (lbm/sec)**	**Broken Loop (lbm/sec)**
15	92.8	37.1
100	90.1	36.0
200	86.6	34.6
300	83.4	33.3
400	80.1	32.0
500	76.5	30.5
600	72.7	29.0
700	68.9	27.5
800	64.6	25.8
900	59.9	23.9
1000	54.1	21.6
1100	47.2	18.8
1200	37.9	15.1
1250	30.1	12.0
1275	25.8	10.3
1300	25.5	10.1
1400	20.5	8.1
1500	19.0	7.5
1600	17.3	6.8
1700	15.4	6.1
1800	13.5	5.4
1900	11.5	4.6
2000	9.3	3.7
2100	7.0	2.8
2200	3.3	1.3
2300	0.0	0.0

Table 6.1.1-2 Units 1 Hi T_{avg} Case SBLOCTA Results				
	1.5 Inch	**2 Inch**	**3 Inch**	**4 Inch**
PCT (°F)	765	1570	1514	1428
PCT Time (s)	22680.6	3434.5	1834.7	1019.1
PCT Elevation (ft)	11.25	11.75	11.5	11.25
Burst Time (s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A
Max. Local ZrO_2 (%)	0.01	1.17	0.89	0.29
Max. Local ZrO_2 Elev (ft)	11.25	11.75	11.50	11.25
Core-Wide Avg. ZrO_2 (%)	0.00	0.17	0.14	0.05

Table 6.1.1-3 Units 1 Low T_{avg} Case SBLOCTA Results				
	1.5 Inch	2 Inch	3 Inch	4 Inch
PCT (°F)	731	1624	1457	1292
PCT Time (s)	22894.8	3455.5	2013.5	1099.7
PCT Elevation (ft)	11.50	11.50	11.5	11.25
Burst Time (s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A
Max. Local ZrO2 (%)	0.00	1.4	0.62	0.11
Max. Local ZrO2 Elev (ft)	11.50	11.50	11.50	11.25
Core-Wide Avg. ZrO2 (%)	0.00	0.22	0.10	0.02

Table 6.1.1-4 Units 2 Hi T_{avg} Case SBLOCTA Results				
	1.5 Inch	**2 Inch**	**3 Inch**	**4 Inch**
PCT (°F)	912	1086	1614	1537
PCT Time (s)	16234.8	2804.5	1618.5	889.0
PCT Elevation (ft)	11.25	11.25	11.50	11.25
Burst Time (s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A
Max. Local ZrO2 (%)	0.03	0.06	1.48	0.65
Max. Local ZrO2 Elev (ft)	11.25	11.25	11.50	11.25
Core-Wide Avg. ZrO2 (%)	0.00	0.01	0.23	0.11

Table 6.1.1-5 Units 2 Low T_{avg} Case SBLOCTA Results				
	1.5 Inch	2 Inch	3 Inch	4 Inch
PCT (°F)	874	1627	1452	1313
PCT Time (s)	17491.8	3071.7	1805.6	992.3
PCT Elevation (ft)	11.00	11.50	11.50	11.25
Burst Time (s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A
Max. Local ZrO2 (%)	0.03	1.59	0.61	0.14
Max. Local ZrO2 Elev (ft)	11.00	11.50	11.50	11.25
Core-Wide Avg. ZrO2 (%)	0.00	0.24	0.09	0.02

Table 6.1.1-6
ZIRC-4
SBLOCTA Results

	Units 1 Low T_{AVG} 2 Inch	Units 2 Hi T_{AVG} 3 Inch BU = BOL	Units 2 Hi T_{AVG} 3 Inch BU = 6000 MWD/MTU
PCT (°F)	1601	1615	1601
PCT Time (s)	3494.5	1618.5	1624.7
PCT Elevation (ft)	11.75	11.50	11.75
Burst Time (s)	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A
Max. Local ZrO2 (%)	1.30	1.5	1.48
Max. Local ZrO2 Elev (ft)	11.75	11.5	11.50
Core-Wide Avg. ZrO2 (%)	0.19	0.23	0.45

Table 6.1.1-7 Units 1 Hi T_{avg} Case NOTRUMP Results				
Event Time (sec)	**1.5 Inch**	**2 Inch**	**3 Inch**	**4 Inch**
Break Initiation	0	0	0	0
Reactor Trip Signal	147.1	82.3	54.4	24.7
S-Signal	159.3	93.9	66.5	35.6
SI Delivered	199.3	133.9	106.5	75.6
Loop Seal Clearing*	2692	1362	586	350
Core Uncovery	15020	2112	863	637
Accumulator Injection	N/A	N/A	2002	920
RWST Switchover Time	1146.7	1143.3	1136.1	1114.8
PCT Time	22680.6	3434.5	1834.7	1019.1
Core Recovery**	>TMAX	>TMAX	2960	2200

* Loop seal clearing is defined as break vapor flow > 1 lb/s

** For the cases, where core recovery is > TMAX, basis for transient termination can be concluded based on the following arguments: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

Table 6.1.1-8 Units 1 Low T_{avg} Case NOTRUMP Results				
Event Time (sec)	**1.5 Inch**	**2 Inch**	**3 Inch**	**4 Inch**
Break Initiation	0	0	0	0
Reactor Trip Signal	79.0	41.8	17.7	10.2
S-Signal	123.3	65.2	27.1	14.1
SI Delivered	163.3	105.2	67.1	54.1
Loop Seal Clearing*	2845	1469	647	380
Core Uncovery	16040	2268	1032	731.1
Accumulator Injection	N/A	N/A	2119	991
RWST Switchover Time	1146.8	1144.6	1137.5	1116.3
PCT Time	22894.8	3455.5	2013.5	1099.7
Core Recovery**	>TMAX	>TMAX	2955	2150

* Loop seal clearing is defined as break vapor flow > 1 lb/s

** For the cases, where core recovery is > TMAX, basis for transient termination can be concluded based on the following arguments: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

Table 6.1.1-9
Units 2 Hi T_{avg} Case
NOTRUMP Results

Event Time (sec)	1.5 Inch	2 Inch	3 Inch	4 Inch
Break Initiation	0	0	0	0
Reactor Trip Signal	142.1	80.2	59.0	24.6
S-Signal	154.6	91.8	71.6	36.4
SI Delivered	194.6	131.8	111.6	76.4
Loop Seal Clearing*	2254	1114.2	485	292
Core Uncovery	9810	1809.3	771	510
Accumulator Injection	N/A	N/A	1732	990.8
RWST Switchover Time	1147.4	1144.3	1132.9	1111.3
PCT Time	16234.8	2804.5	1618.5	889.0
Core Recovery**	37098	4740	2754	2378

* Loop seal clearing is defined as break vapor flow > 1 lb/s

** For the cases, where core recovery is > TMAX, basis for transient termination can be concluded based on the following arguments: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

Table 6.1.1-10 Units 2 Low T_{avg} Case NOTRUMP Results				
Event Time (sec)	**1.5 Inch**	**2 Inch**	**3 Inch**	**4 Inch**
Break Initiation	0	0	0	0
Reactor Trip Signal	77.7	41.3	17.6	10.2
S-Signal	117.8	59.6	27.2	14.3
SI Delivered	157.8	67.2	96.6	54.3
Loop Seal Clearing*	2381	1219.0	549.3	311.9
Core Uncovery	10750	1930	717.9	614.5
Accumulator Injection	N/A	N/A	1928.1	882.6
RWST Switchover Time	1147.5	1144.9	1133.9	1112.7
PCT Time	17491.8	3071.7	1805.6	992.3
Core Recovery**	36950	5543	2826	2103

* Loop seal clearing is defined as break vapor flow > 1 lb/s

** For the cases, where core recovery is > TMAX, basis for transient termination can be concluded based on the following arguments: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

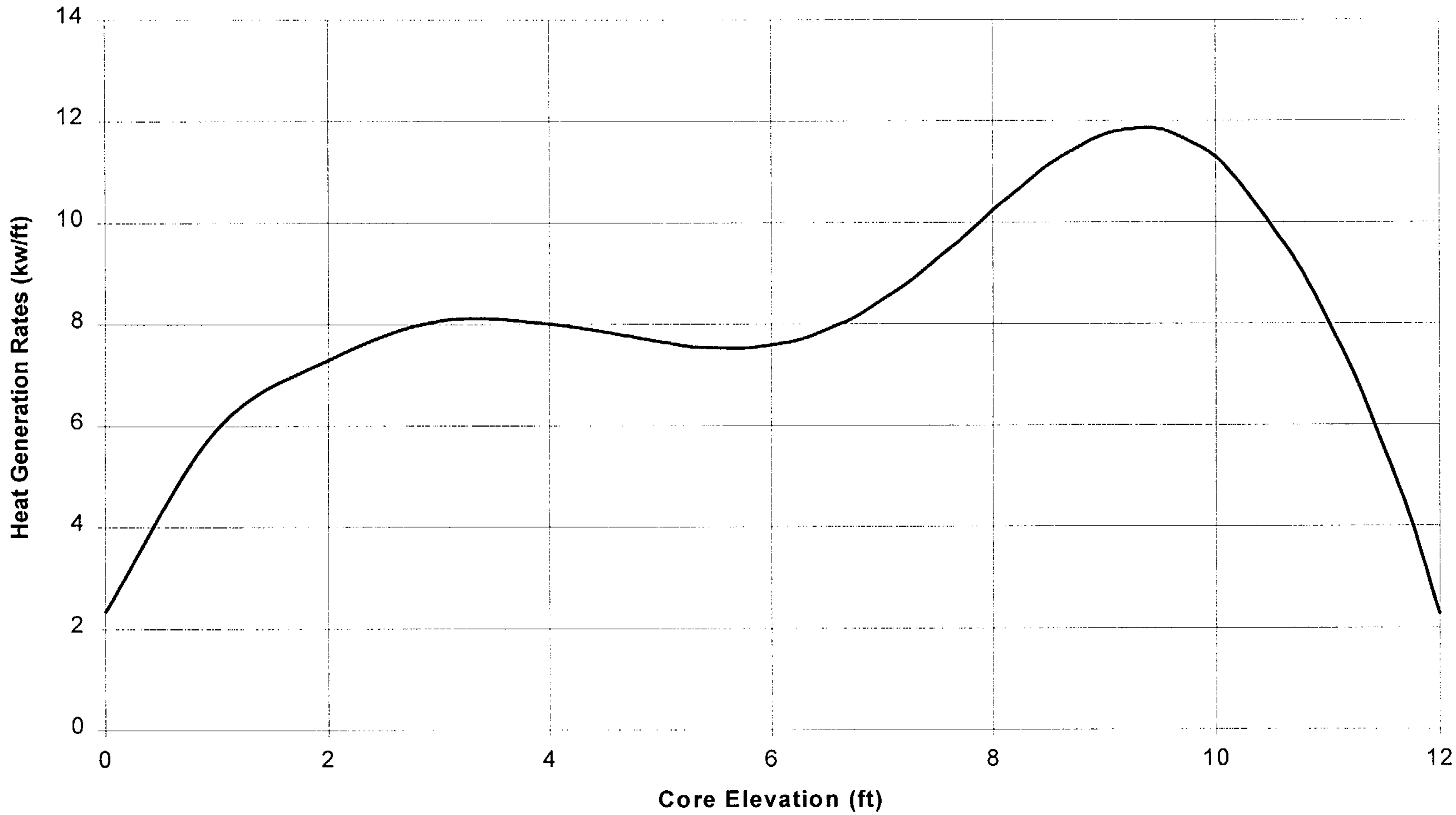

Figure 6.1.1–1
Small Break Hot Rod Power Shape

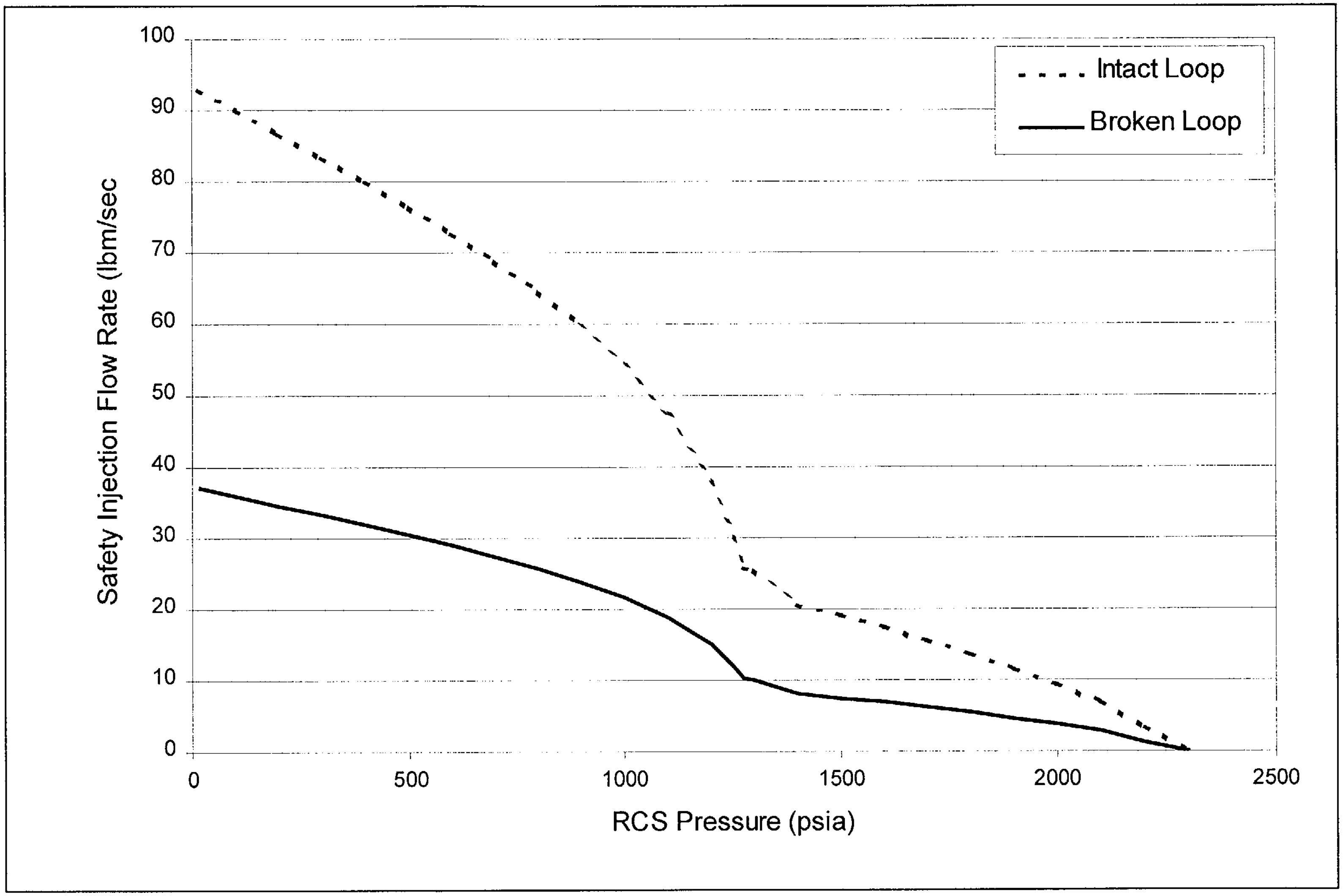

Figure 6.1.1–2

Small Break LOCA Safety Injection Flows

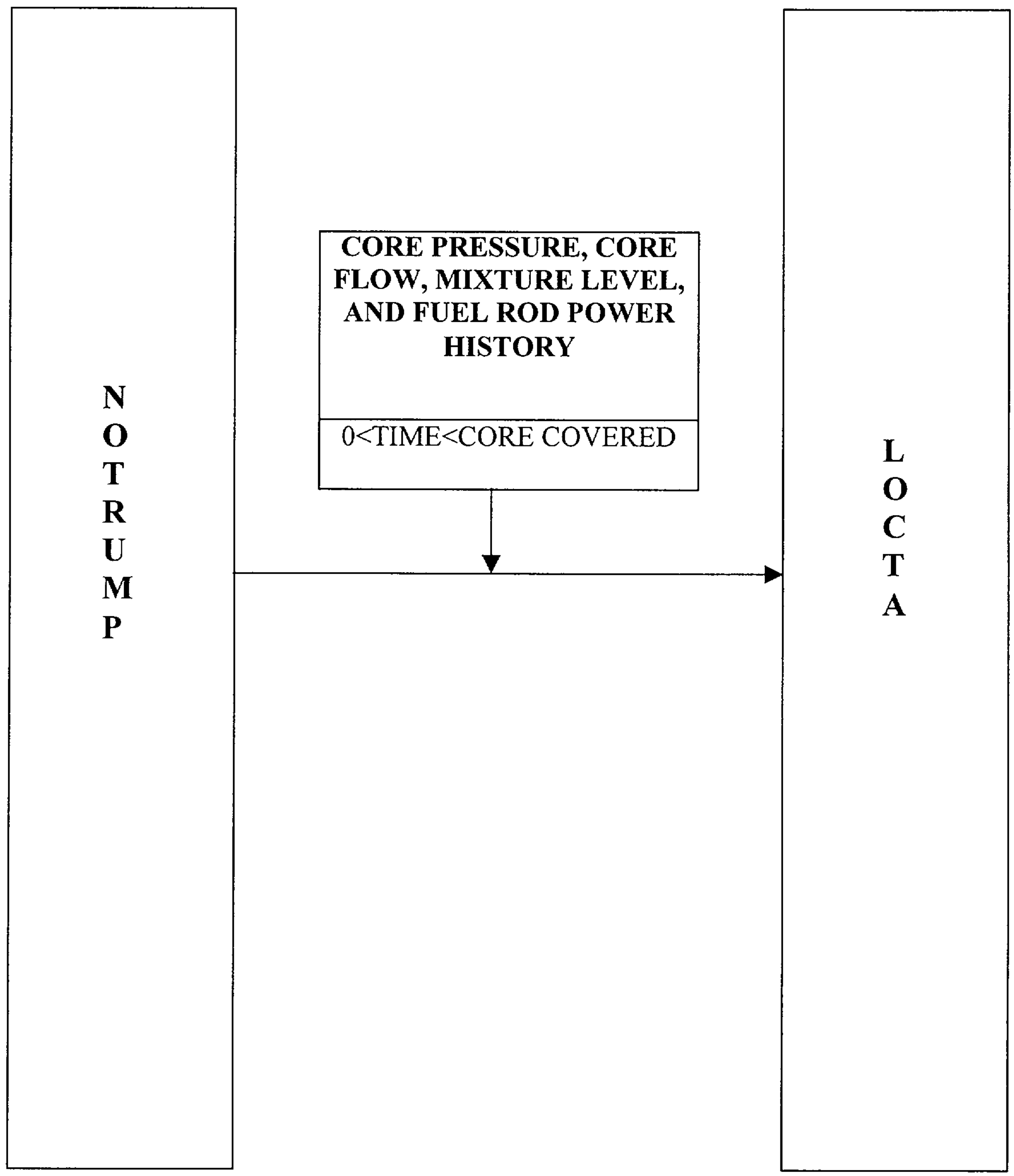

Figure 6.1.1–3
Code Interface Description
for Small Break Model

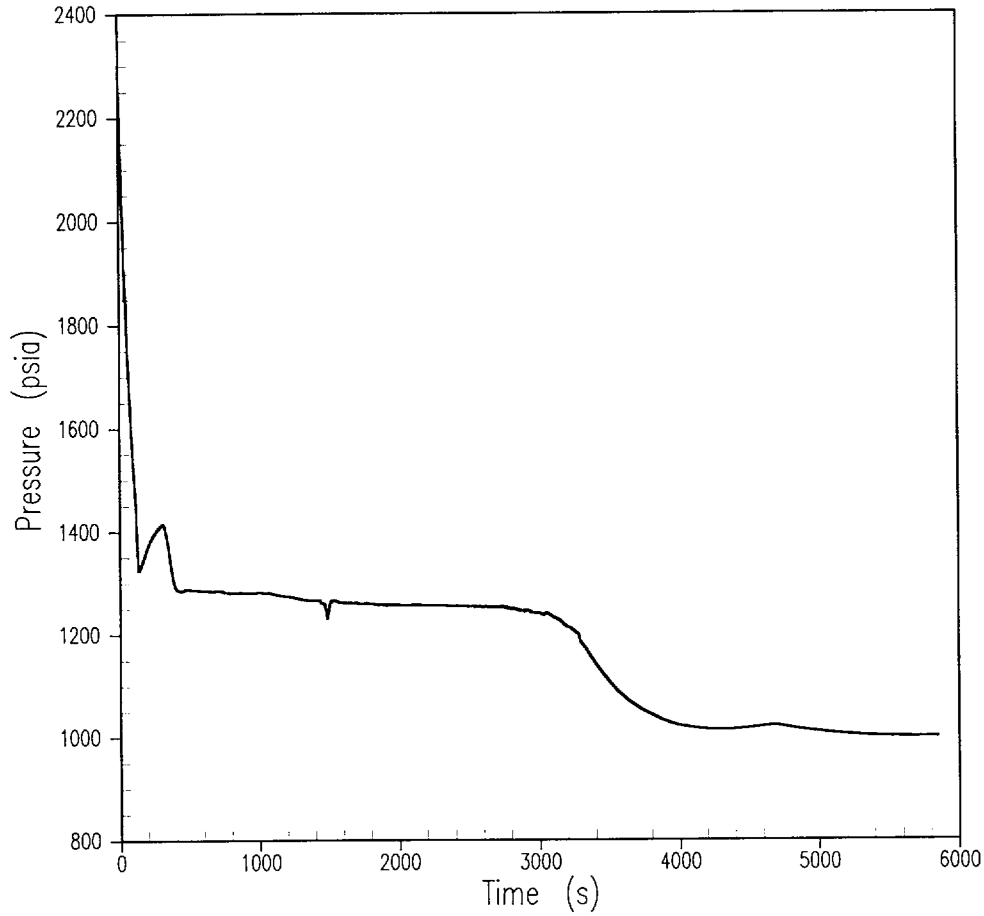

Figure 6.1.1 – 4
Units 1 Low T_{avg} 2-Inch
RCS Pressure

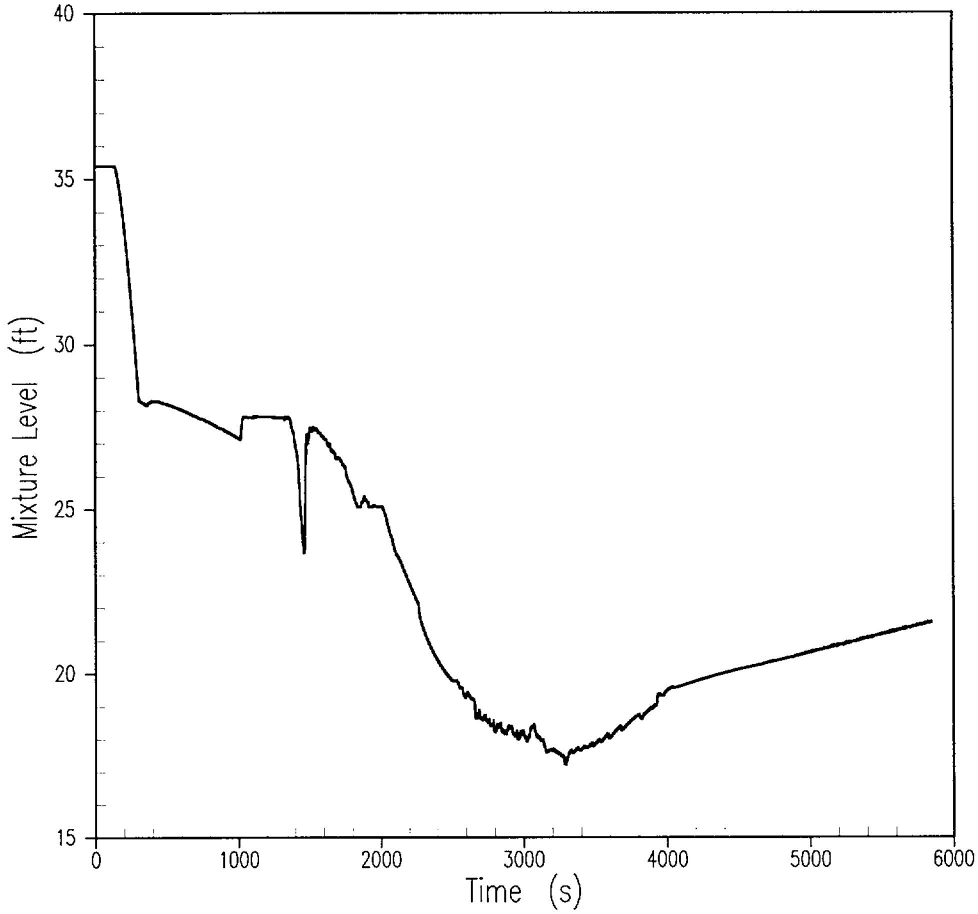

Figure 6.1.1 – 5
Units 1 Low T_{avg} 2-Inch
Core Mixture Level

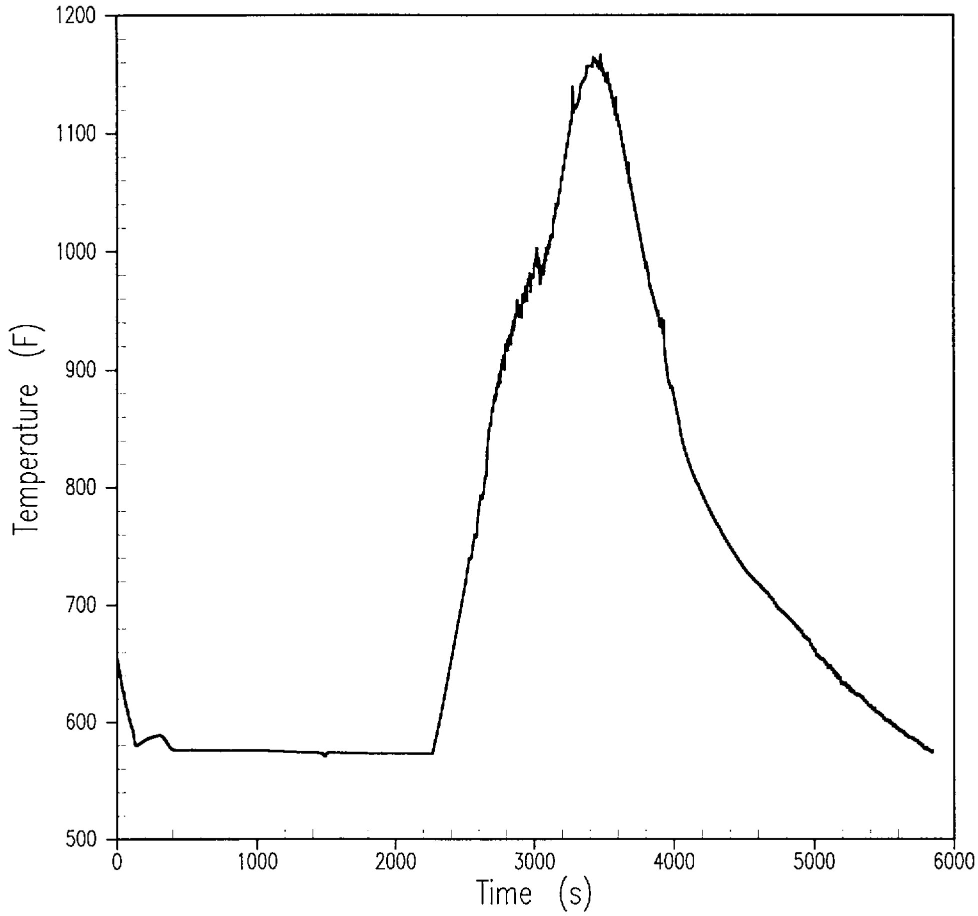

Figure 6.1.1 – 6
Units 1 Low T_{avg} 2-Inch
Core Exit Vapor Temperature

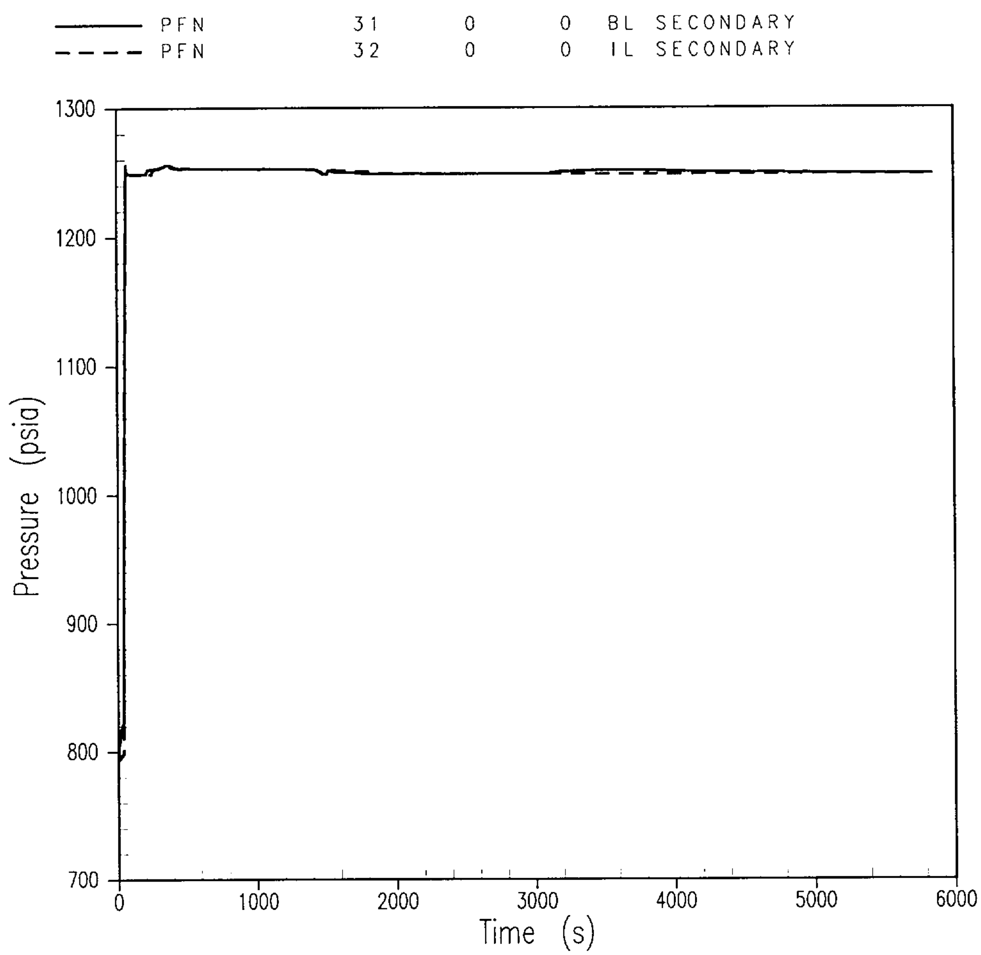

Figure 6.1.1 – 7
Units 1 Low T_{avg} 2-Inch
Broken Loop and Intact Loop Secondary Pressure

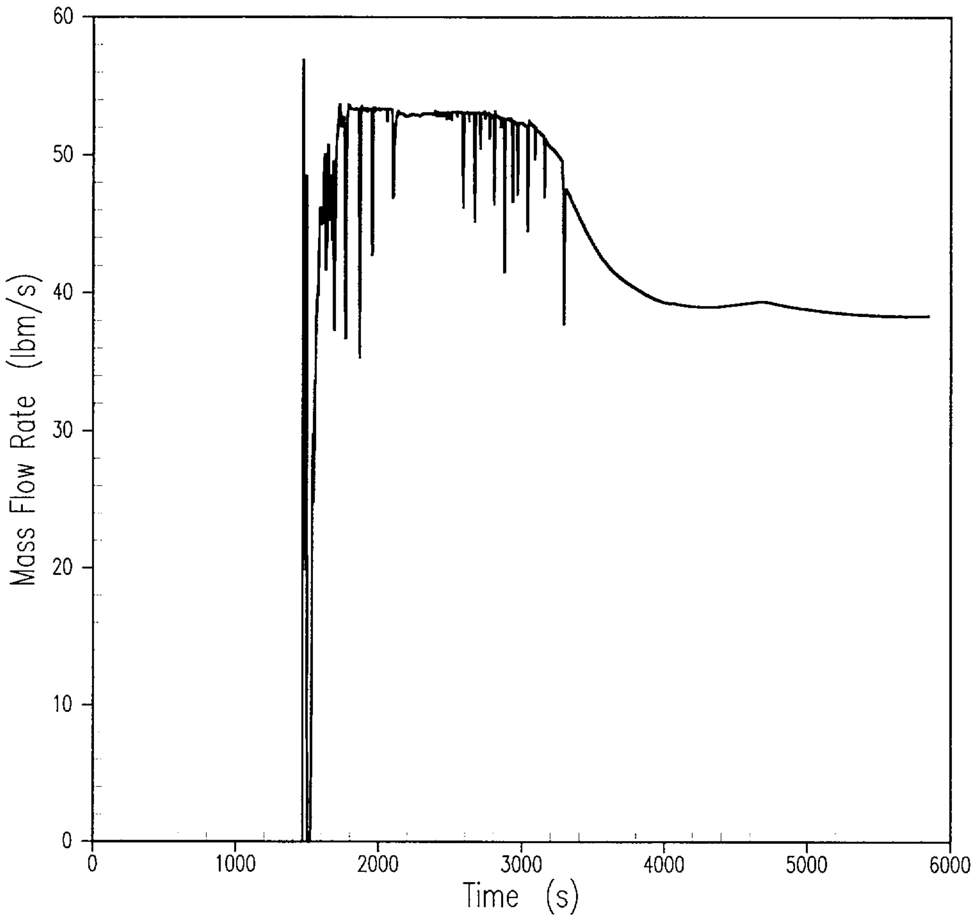

Figure 6.1.1 – 8
Units 1 Low T_{avg} 2-Inch
Break Vapor Flow Rate

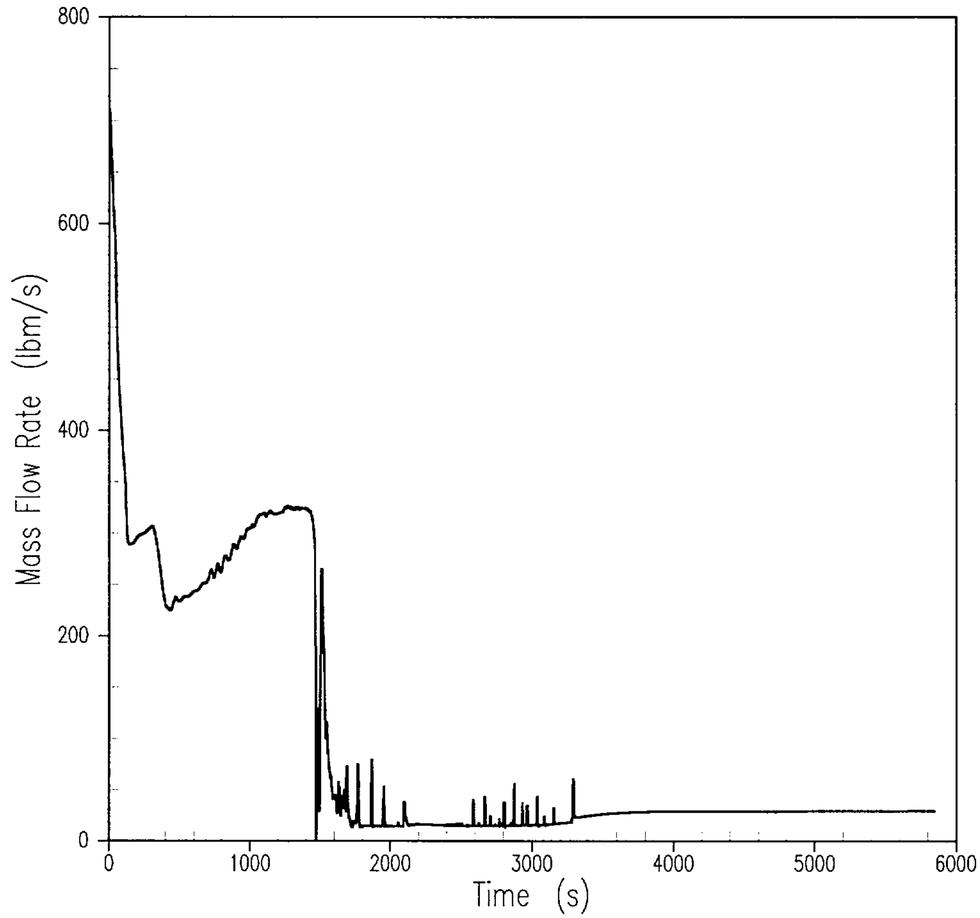

Figure 6.1.1 – 9
Units 1 Low T_{avg} 2-Inch
Break Liquid Flow Rate

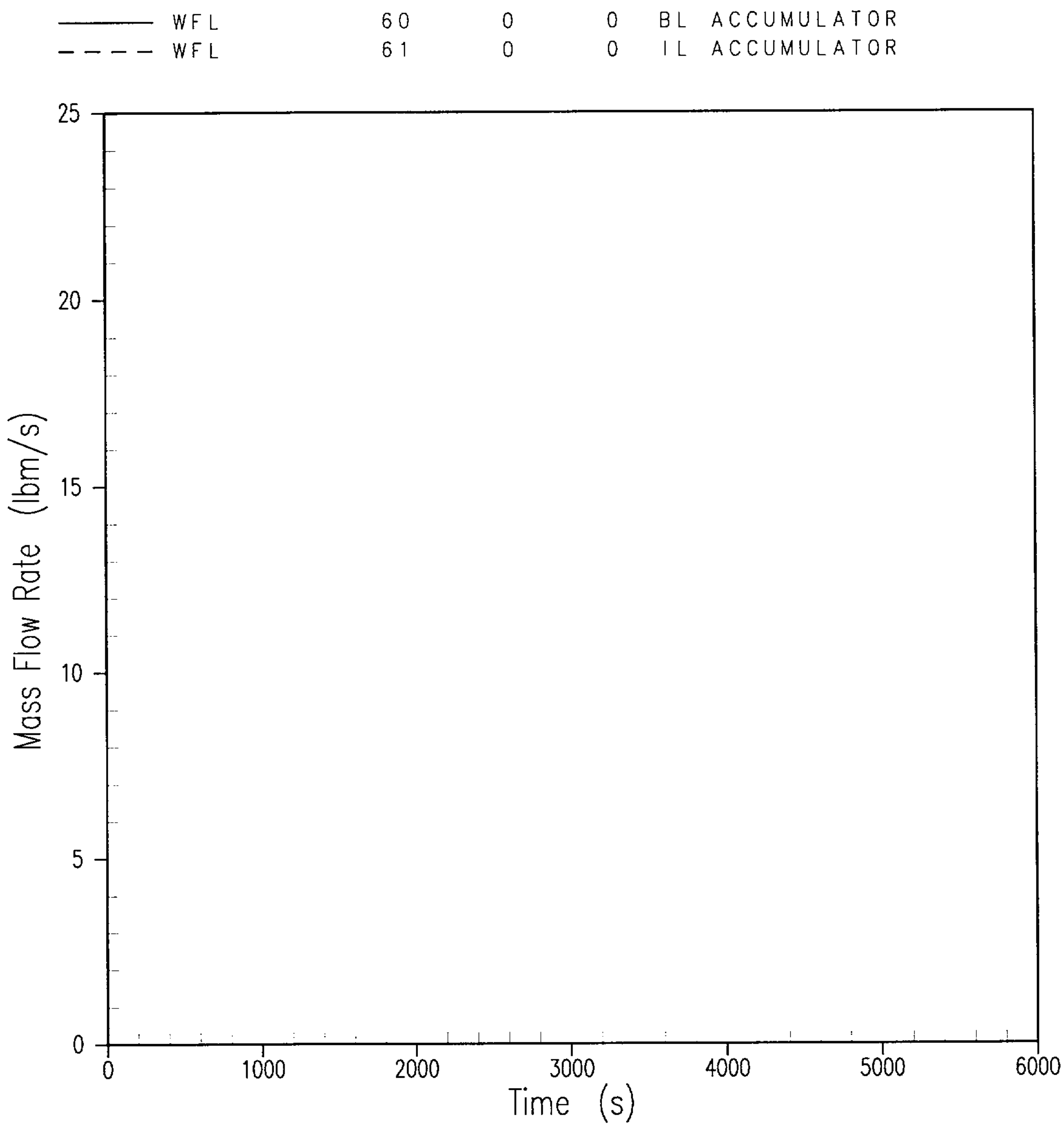

Figure 6.1.1 – 10
Units 1 Low T_{avg} 2-Inch
Broken Loop and Intact Loop Accumulator Flow Rate

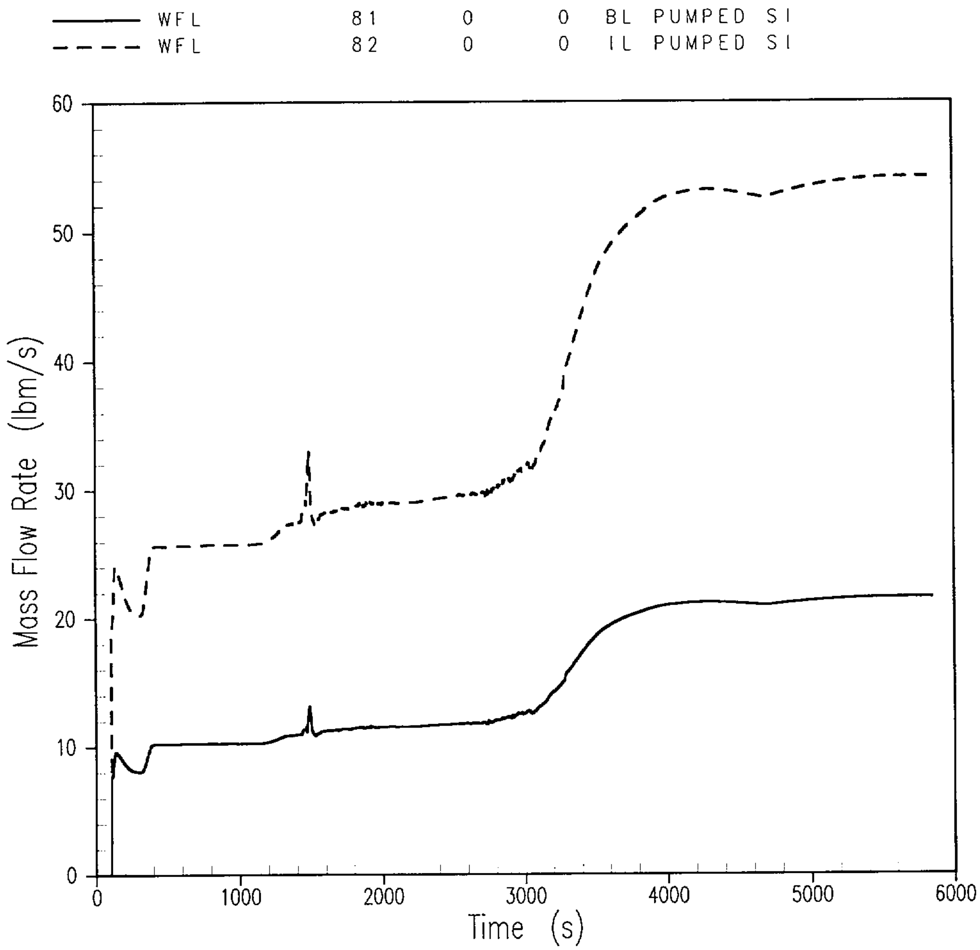

Figure 6.1.1 – 11
Units 1 Low T_{avg} 2-Inch
Broken Loop and Intact Loop Pumped Safety Injection Flow

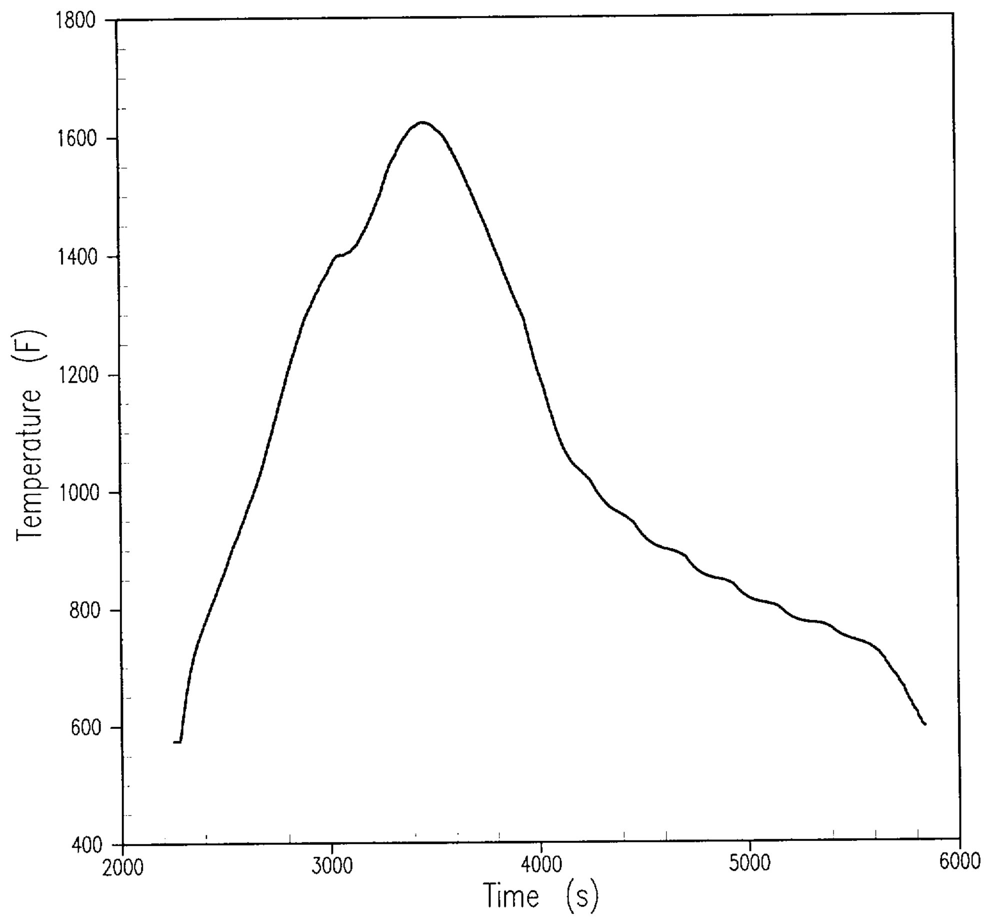

Figure 6.1.1 – 12
Units 1 Low T_{avg} 2-Inch
Peak Clad Temperature at 11.50 ft.

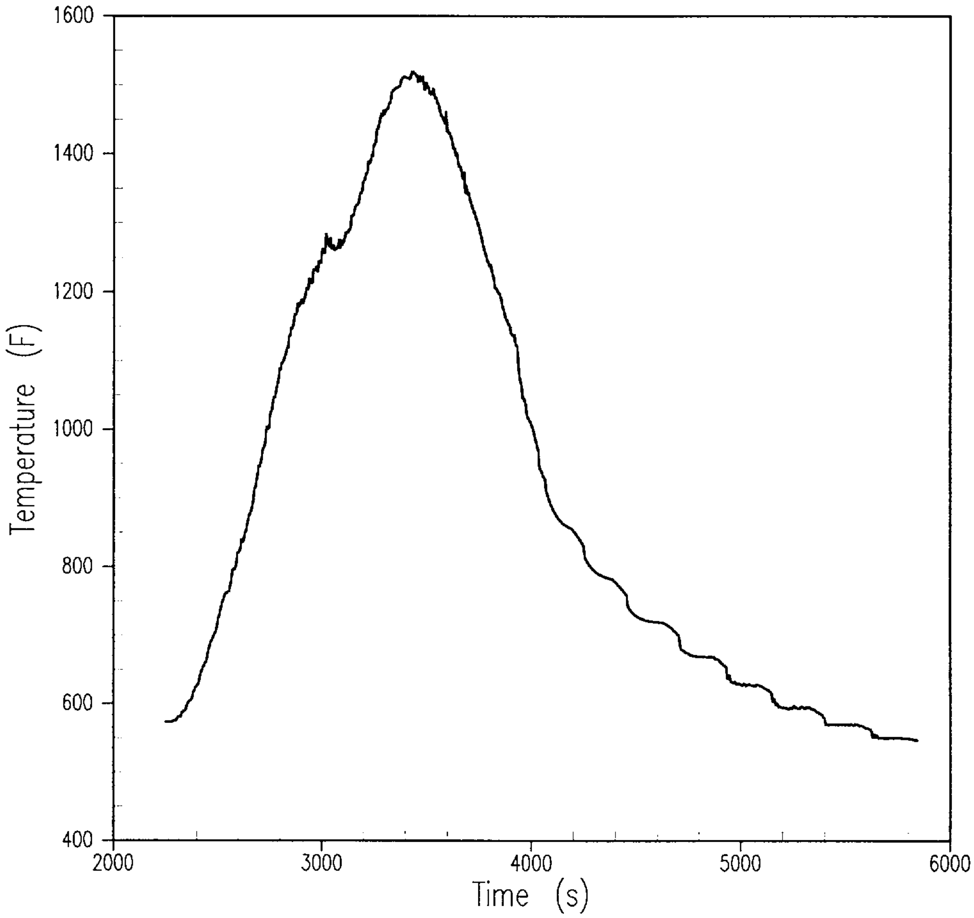

Figure 6.1.1 – 13
Units 1 Low T_{avg} 2-Inch
Hot Spot Fluid Temperature

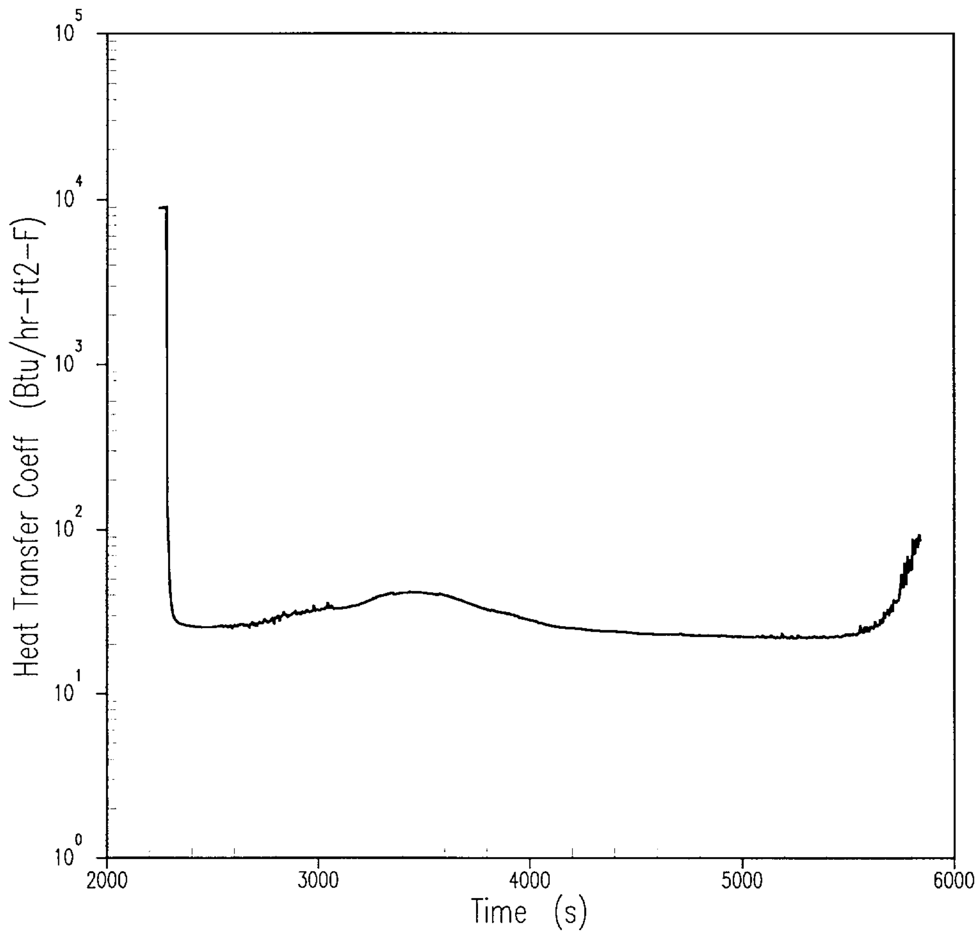

Figure 6.1.1 – 14
Units 1 Low T_{avg} 2-Inch
Rod Film Heat Transfer Coefficient at 11.50 ft.

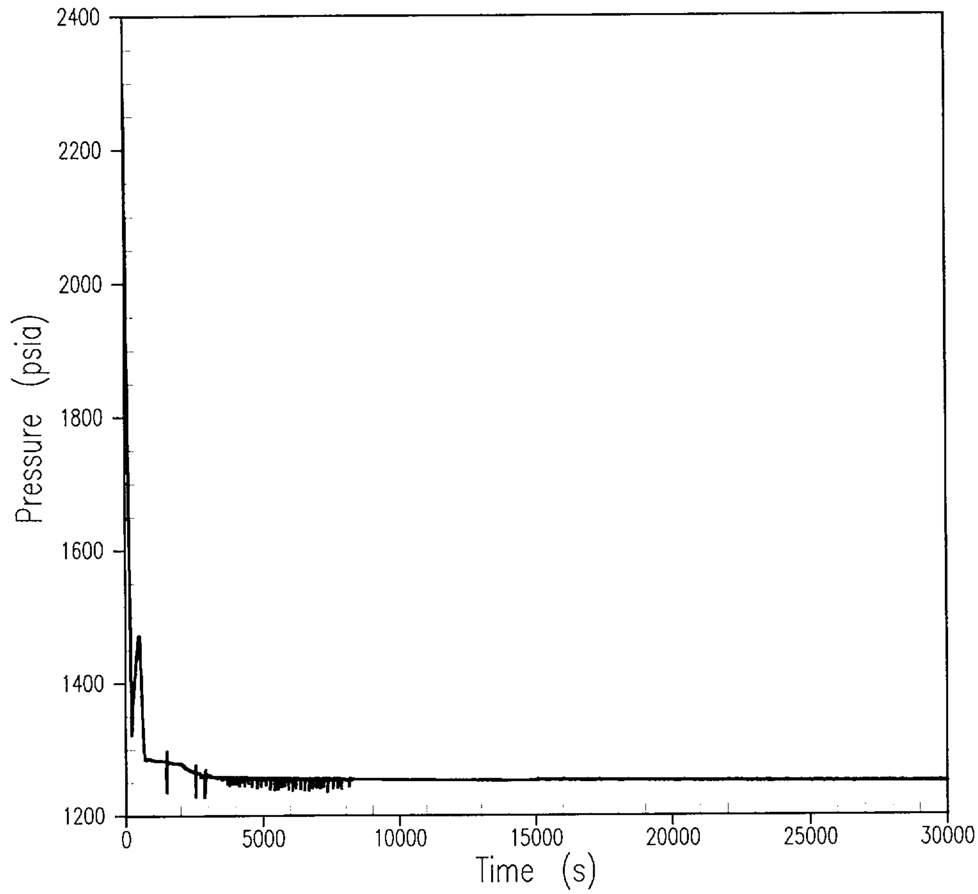

Figure 6.1.1-15

Units 1 Low T_{avg} 1.5-Inch

RCS Pressure

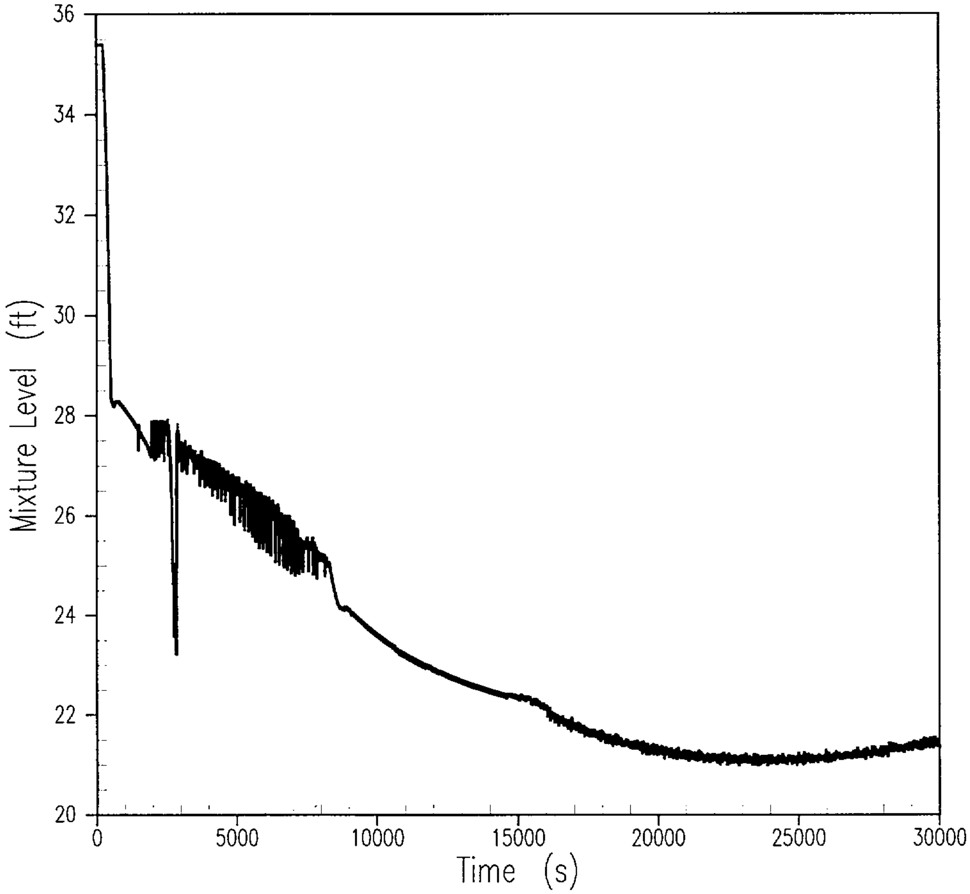

Figure 6.1.1-16

Units 1 Low T_{avg} 1.5-Inch

Core Mixture Level

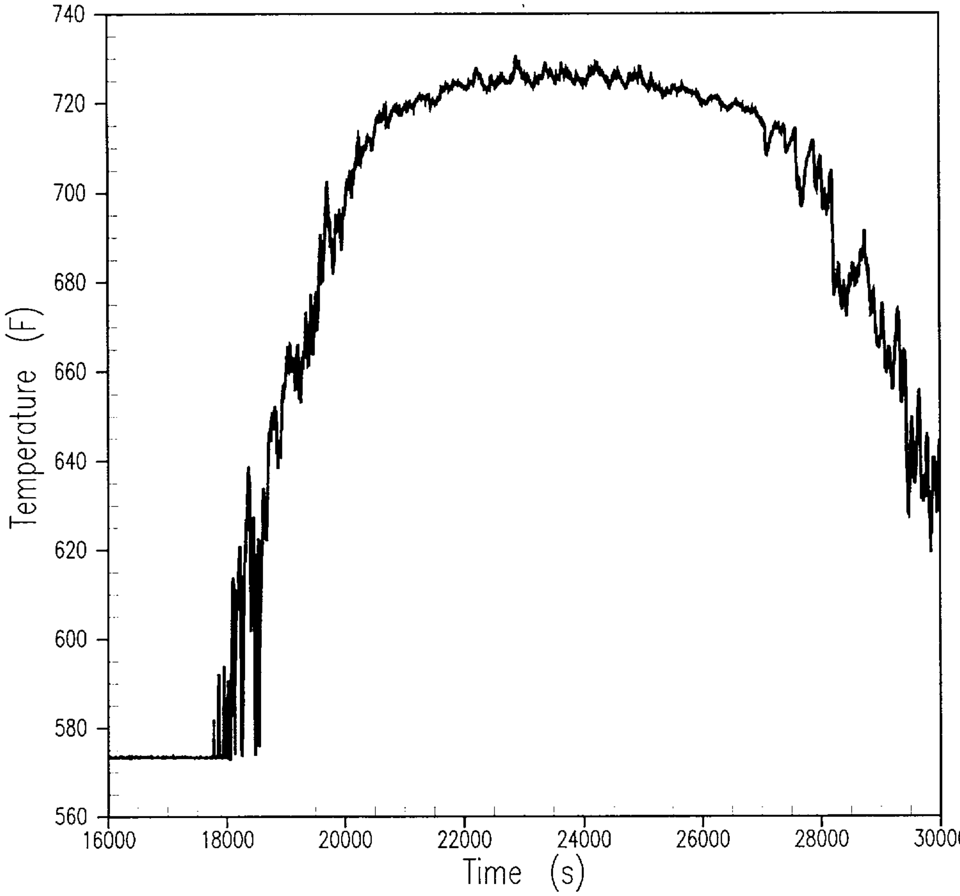

Figure 6.1.1-17

Units 1 Low T_{avg} 1.5-Inch

Peak Clad Temperature at 11.5 ft.

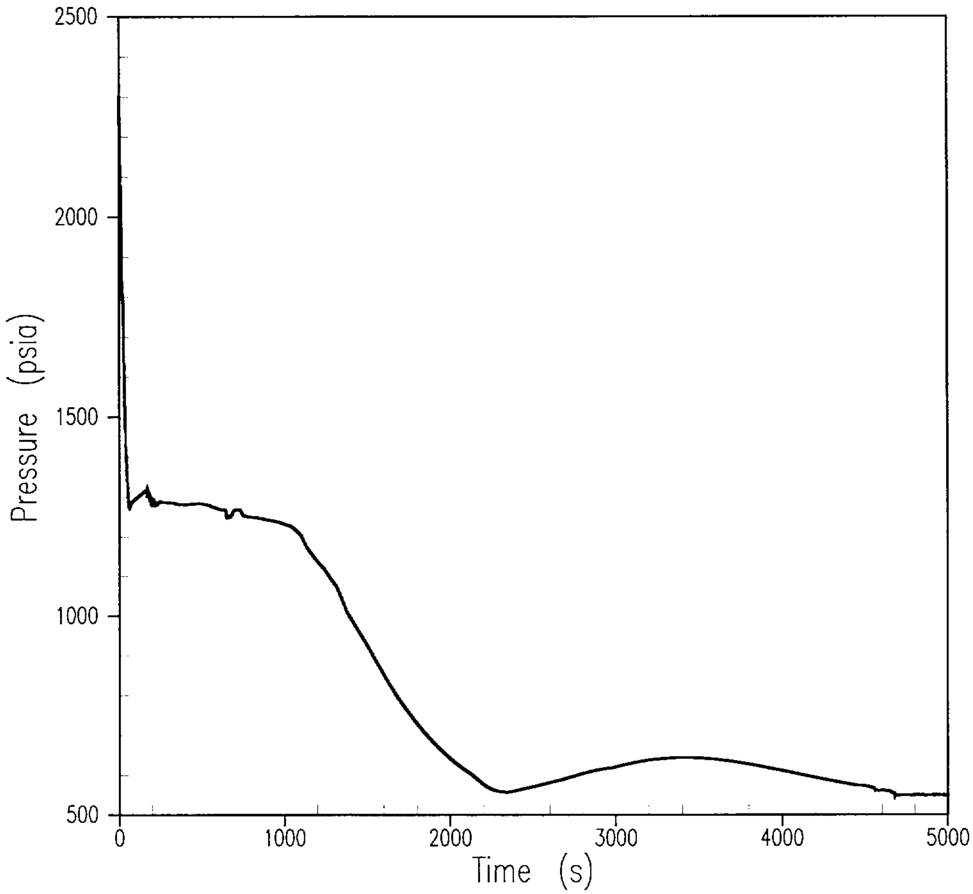

Figure 6.1.1-18

Units 1 Low T_{avg} 3-Inch

RCS Pressure

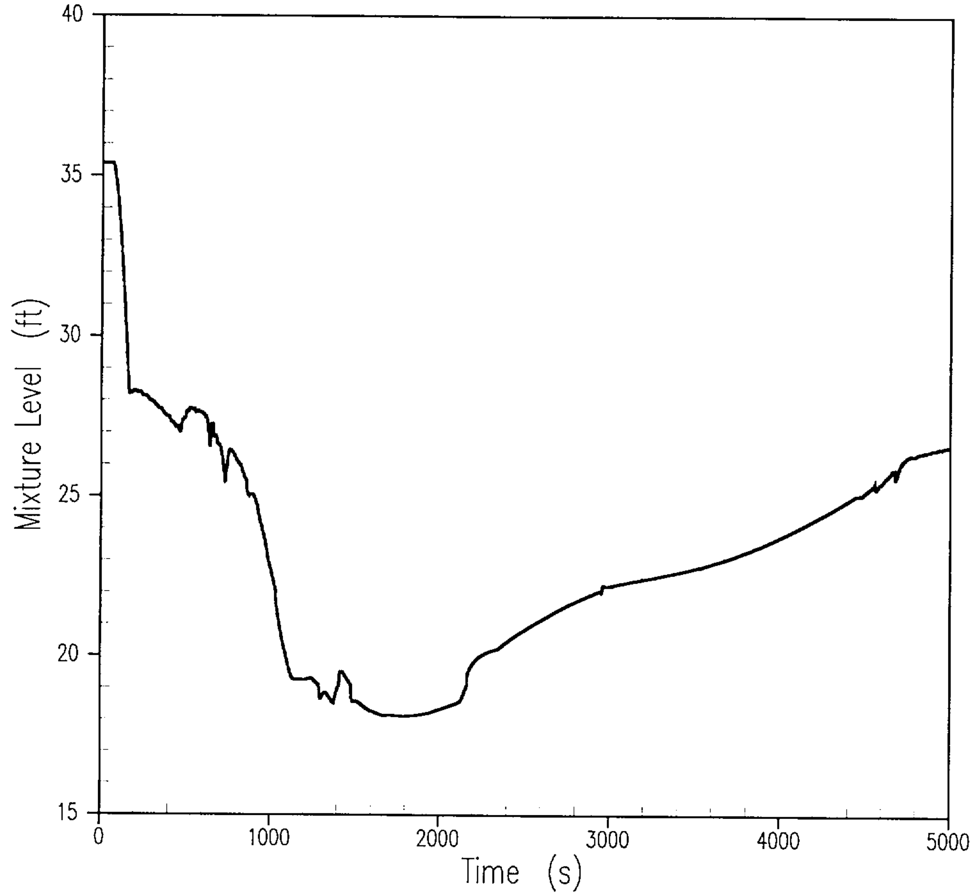

Figure 6.1.1-19

Units 1 Low T_{avg} 3-Inch

Core Mixture Level

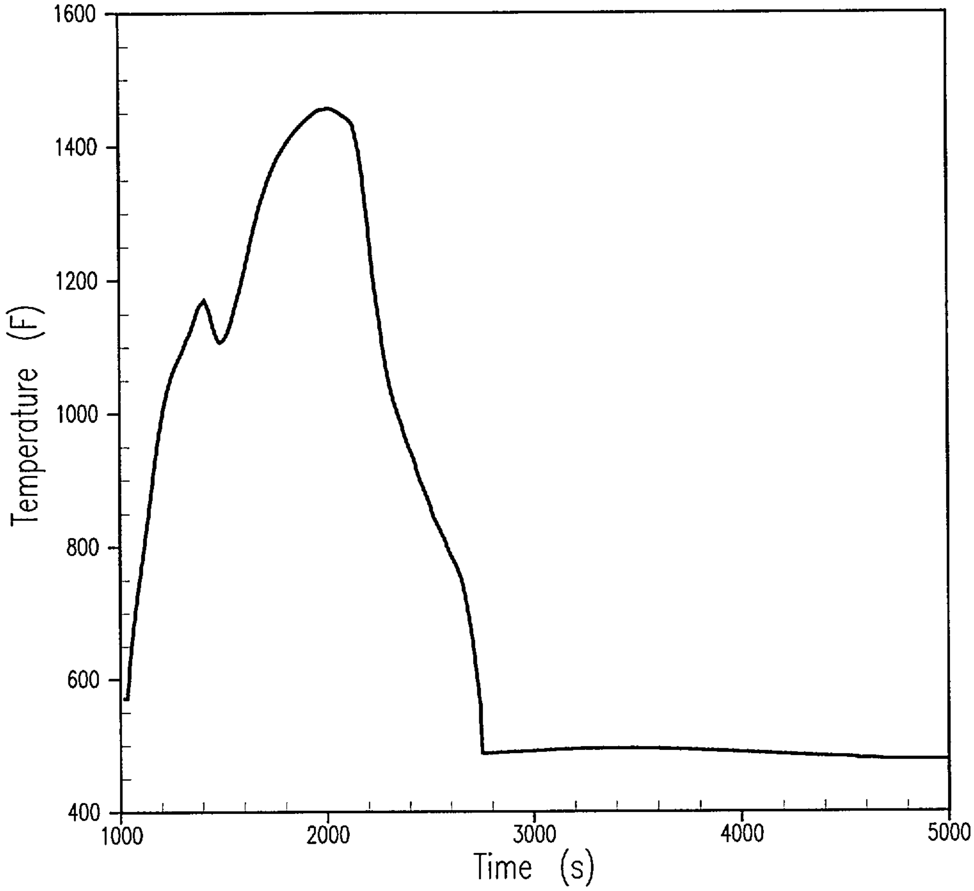

Figure 6.1.1-20

Units 1 Low T_{avg} 3-Inch

Peak Clad Temperature at 11.5 ft.

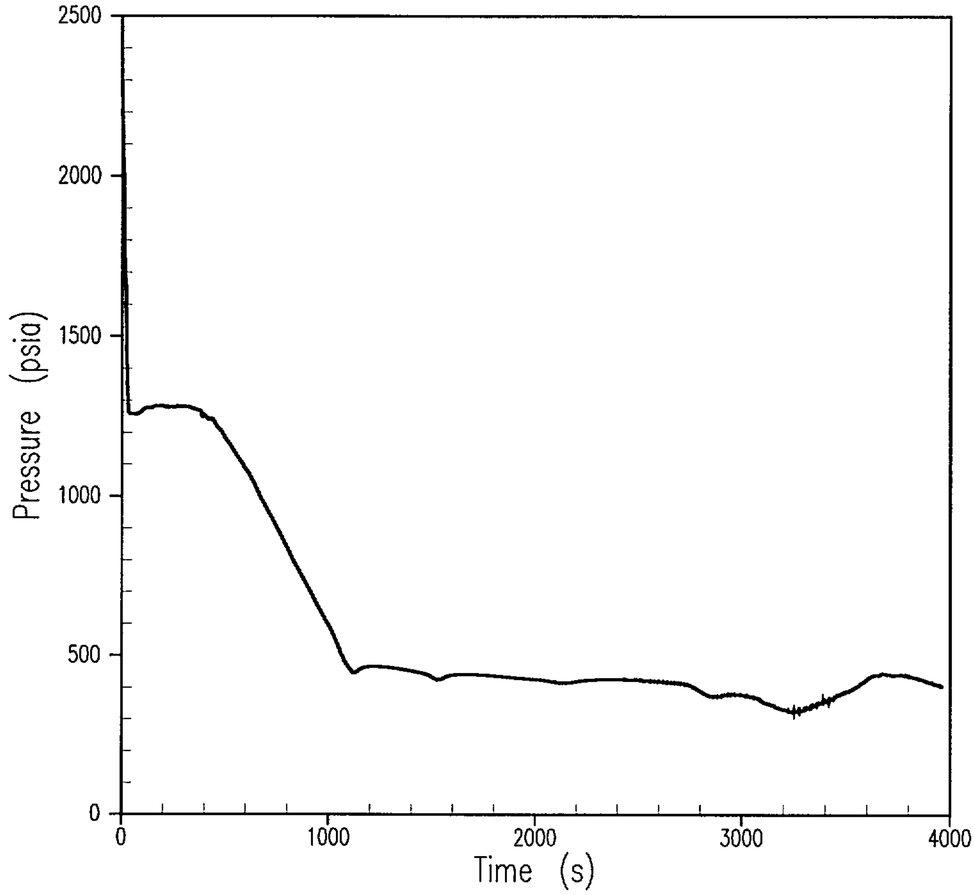

Figure 6.1.1-21

Units 1 Low T_{avg} 4-Inch

RCS Pressure

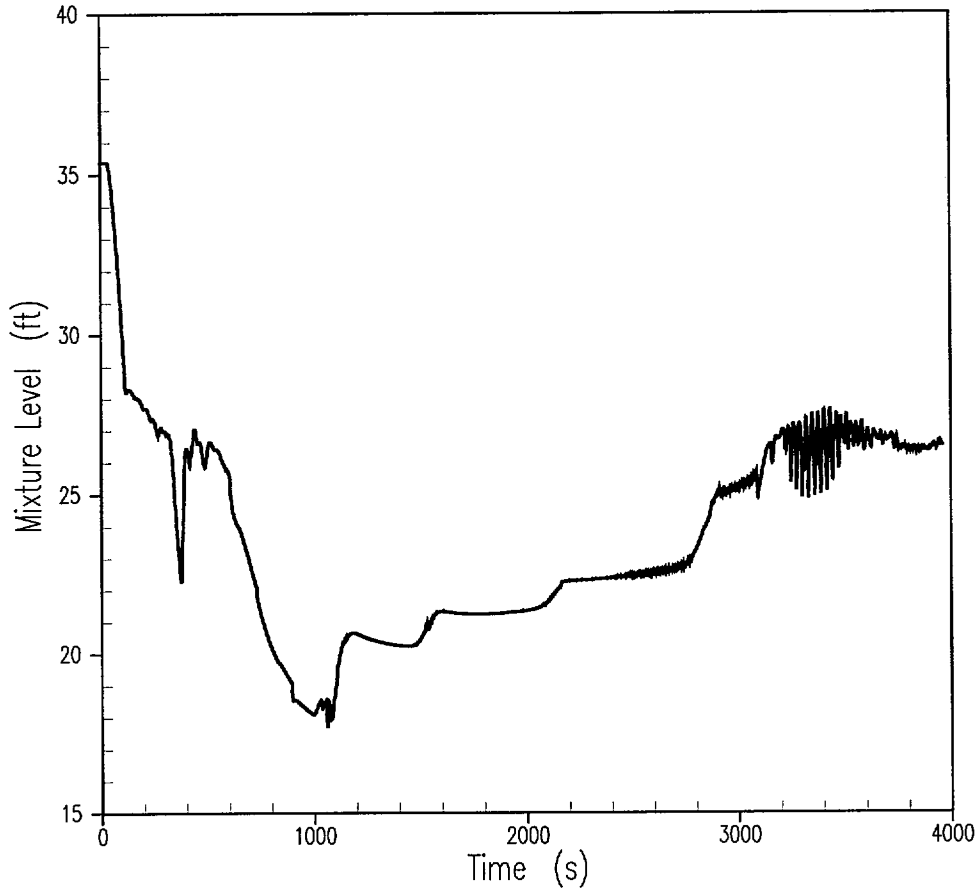

Figure 6.1.1-22

Units 1 Low T_{avg} 4-Inch

Core Mixture Level

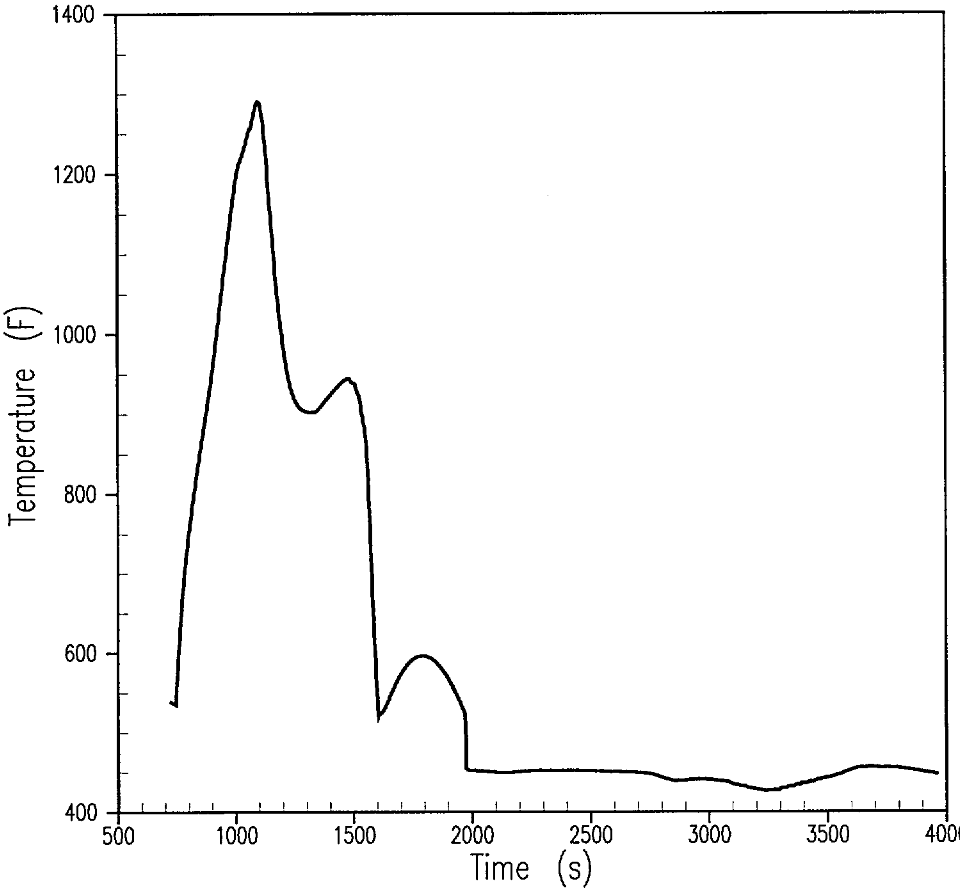

Figure 6.1.1-23

Units 1 Low T_{avg} 4-Inch

Peak Clad Temperature at 11.25 ft

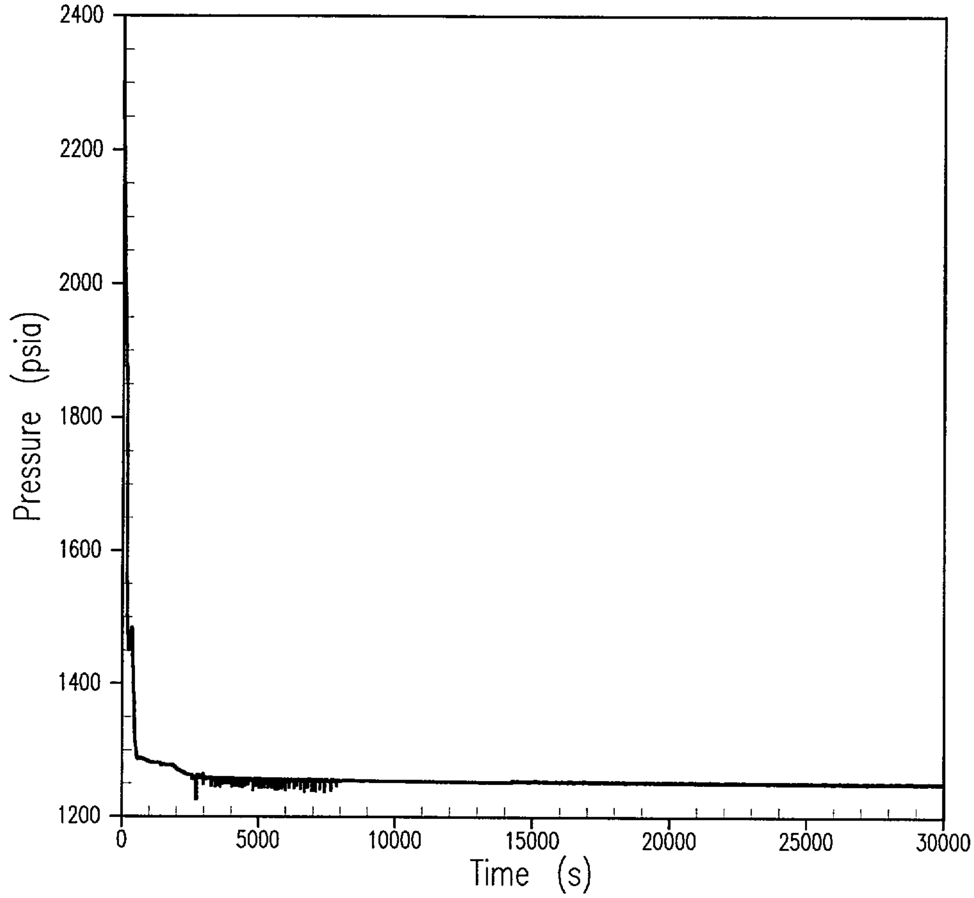

Figure 6.1.1-24

Units 1 High T_{avg} 1.5-Inch

RCS Pressure

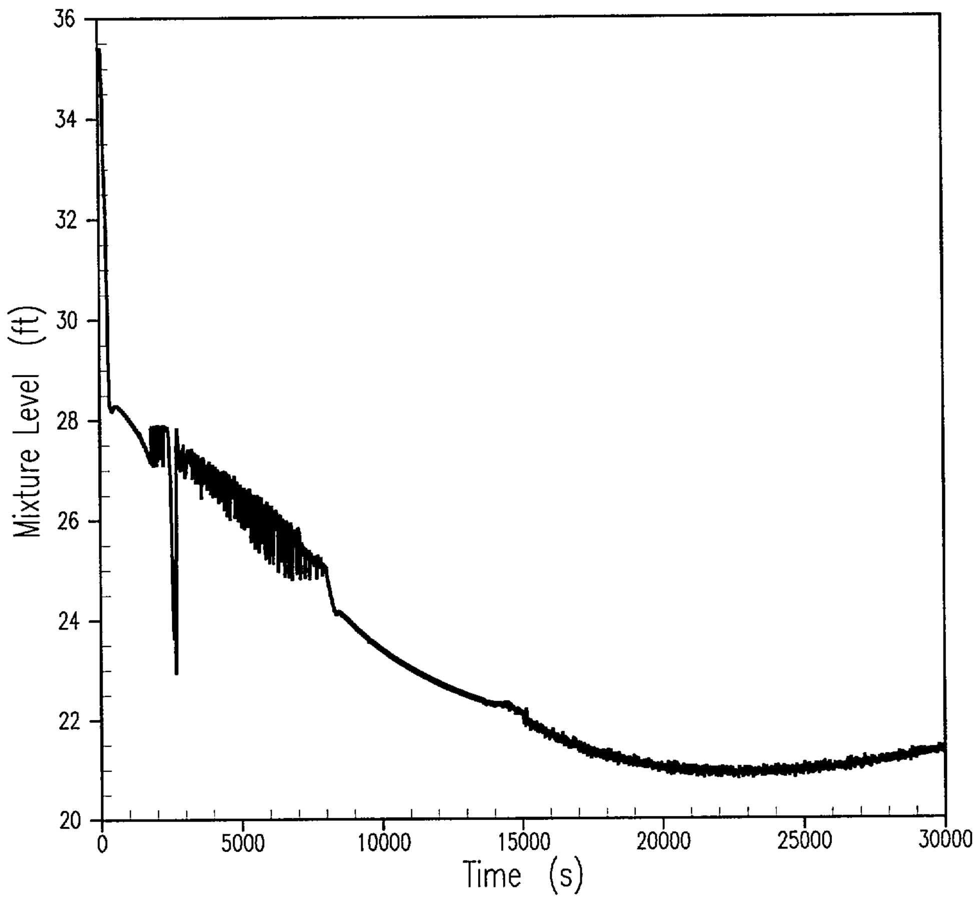

Figure 6.1.1-25

Units 1 High T_{avg} 1.5-Inch

Core Mixture Level

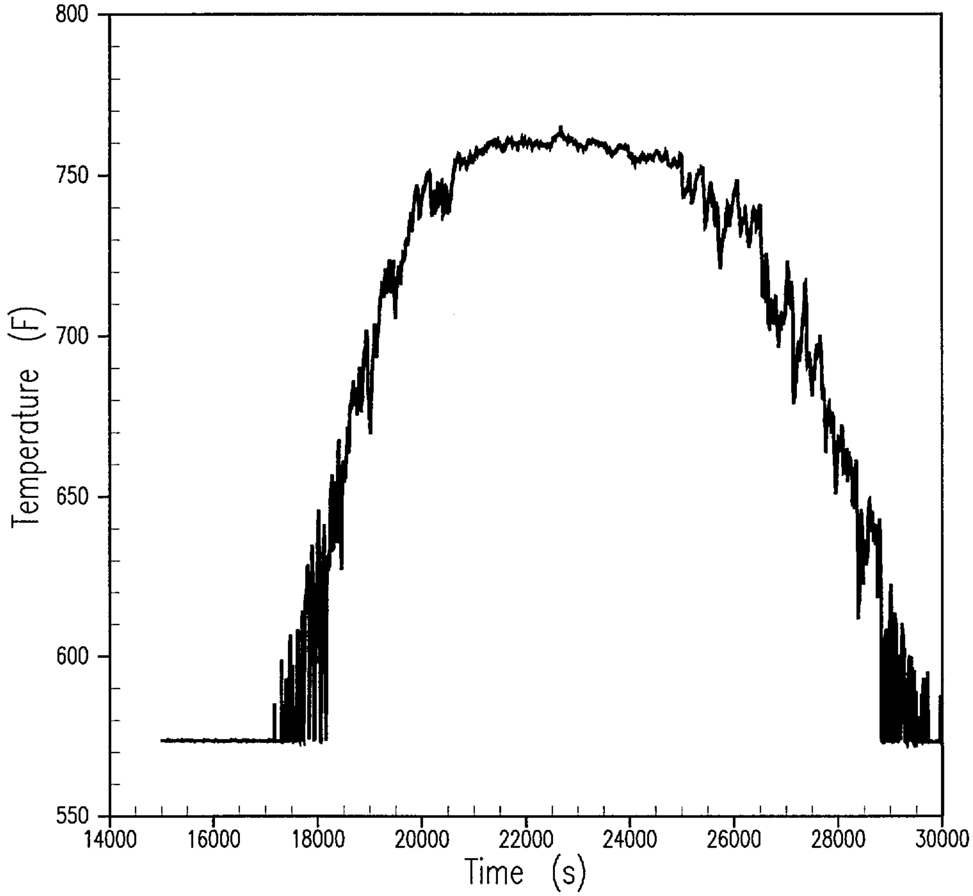

Figure 6.1.1-26

Units 1 High T_{avg} 1.5-Inch

Peak Clad Temperature at 11.25 ft.

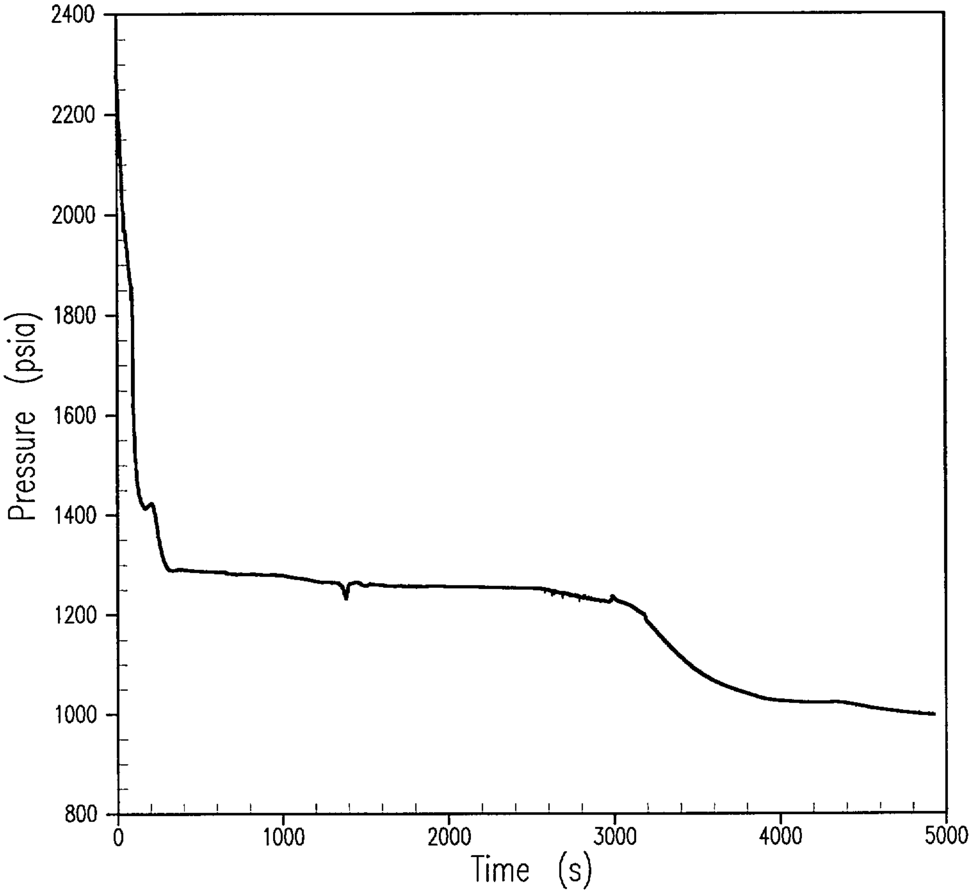

Figure 6.1.1-27

Units 1 High T_{avg} 2-Inch

RCS Pressure

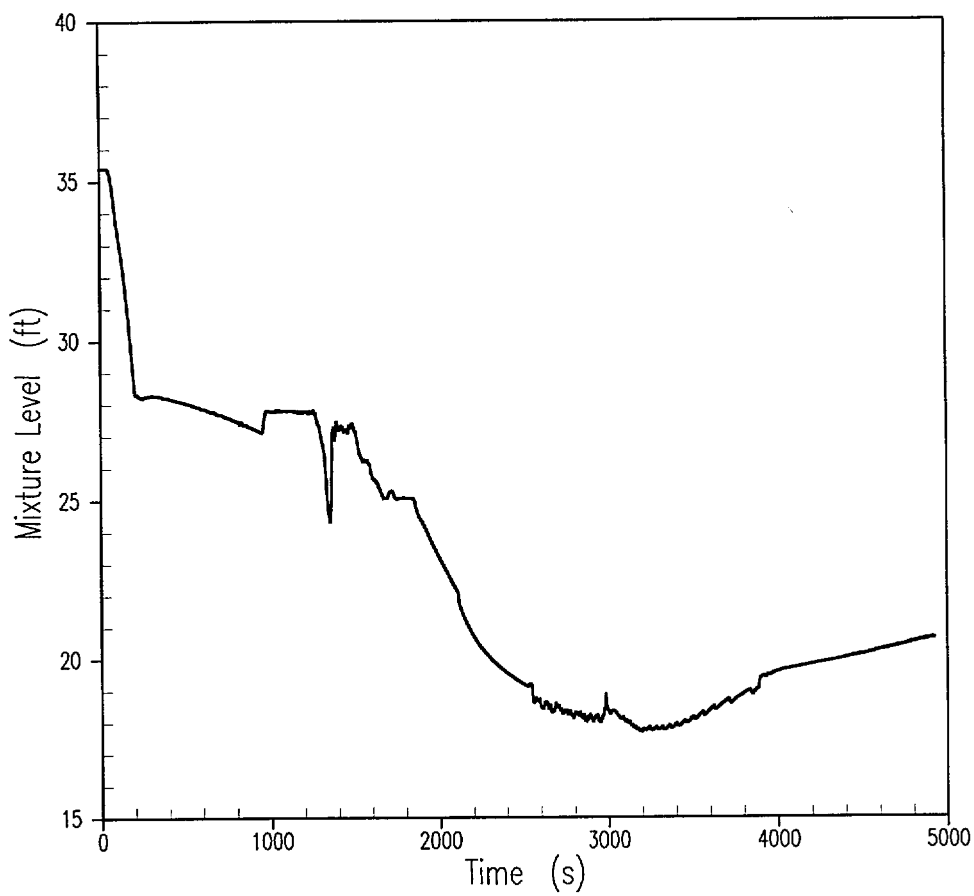

Figure 6.1.1-28

Units 1 High T_{avg} 2-Inch

Core Mixture Level

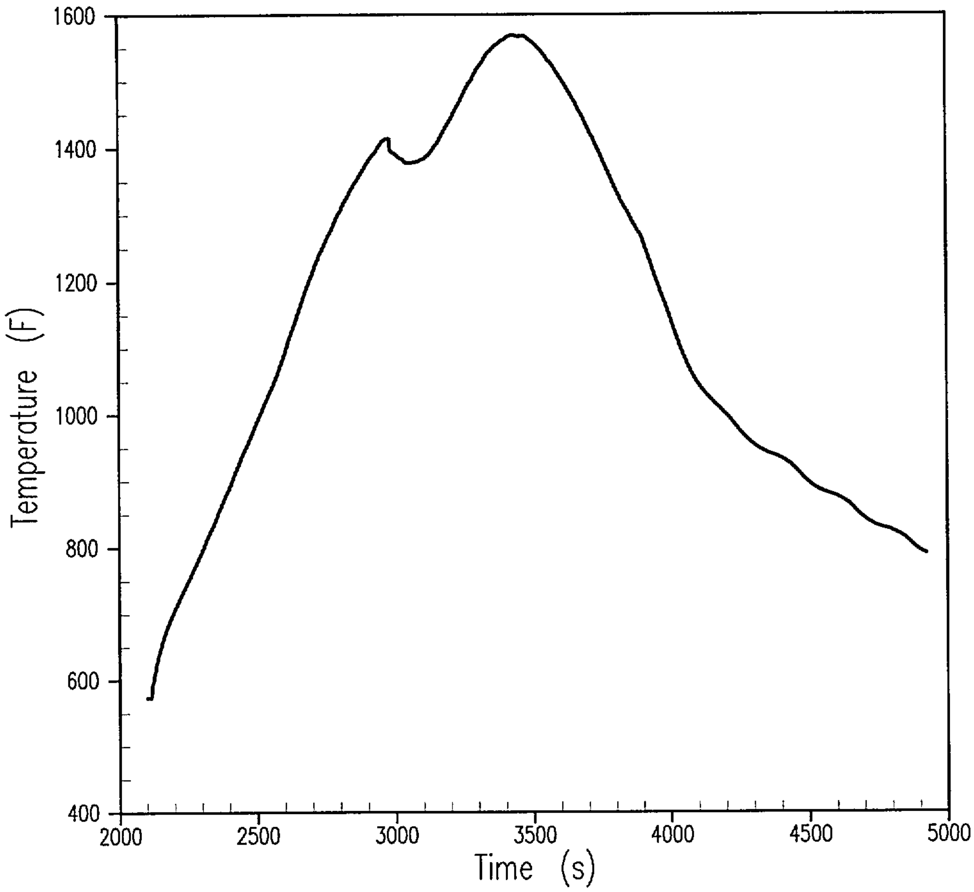

Figure 6.1.1-29

Units 1 High T_{avg} 2-Inch

Peak Clad Temperature at 11.75 ft.

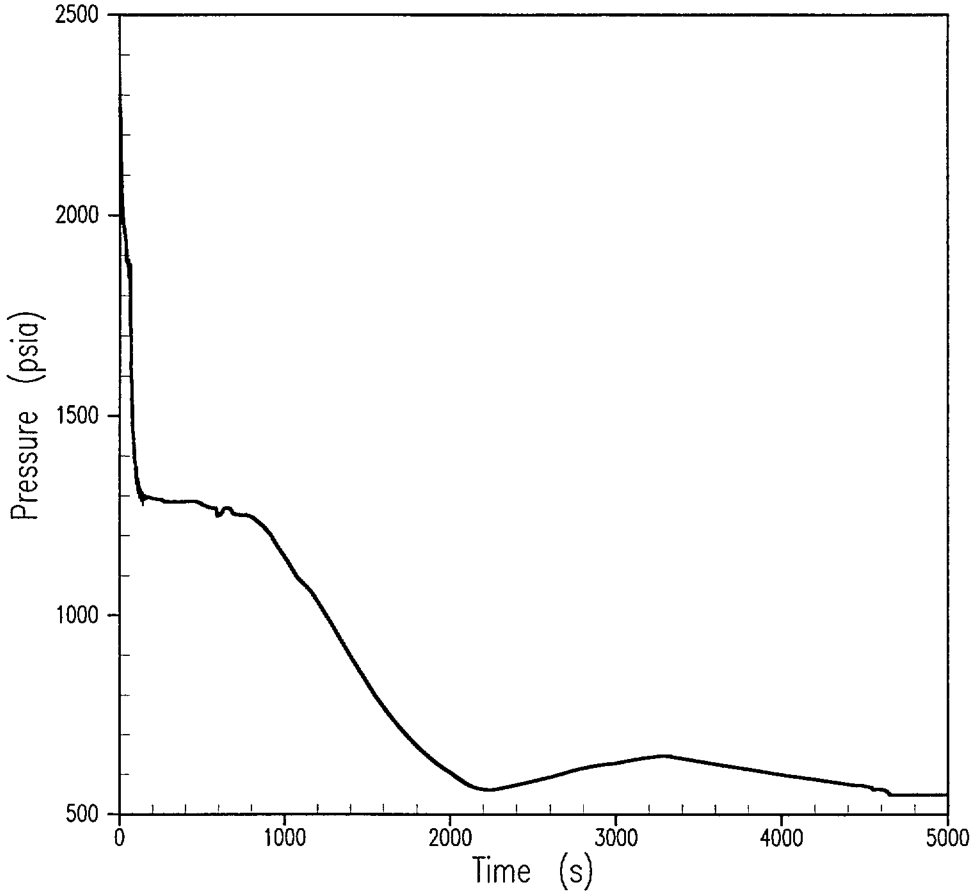

Figure 6.1.1-30

Units 1 High T_{avg} 3-Inch

RCS Pressure

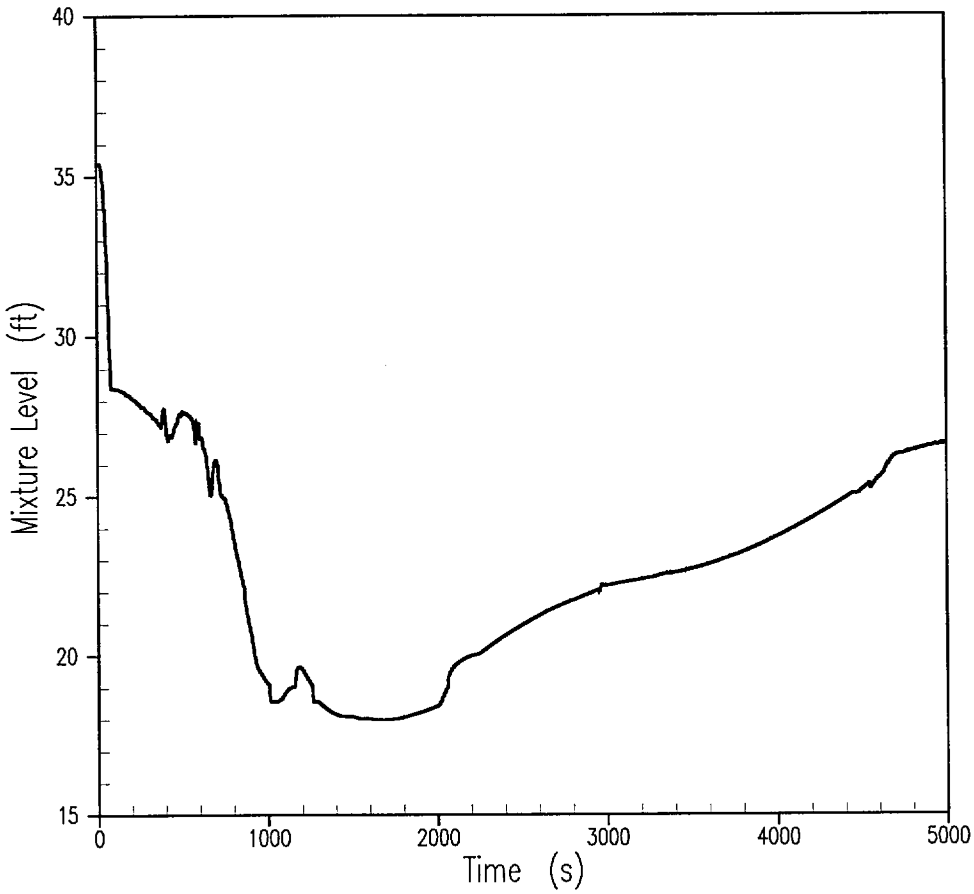

Figure 6.1.1-31

Units 1 High T_{avg} 3-Inch

Core Mixture Level

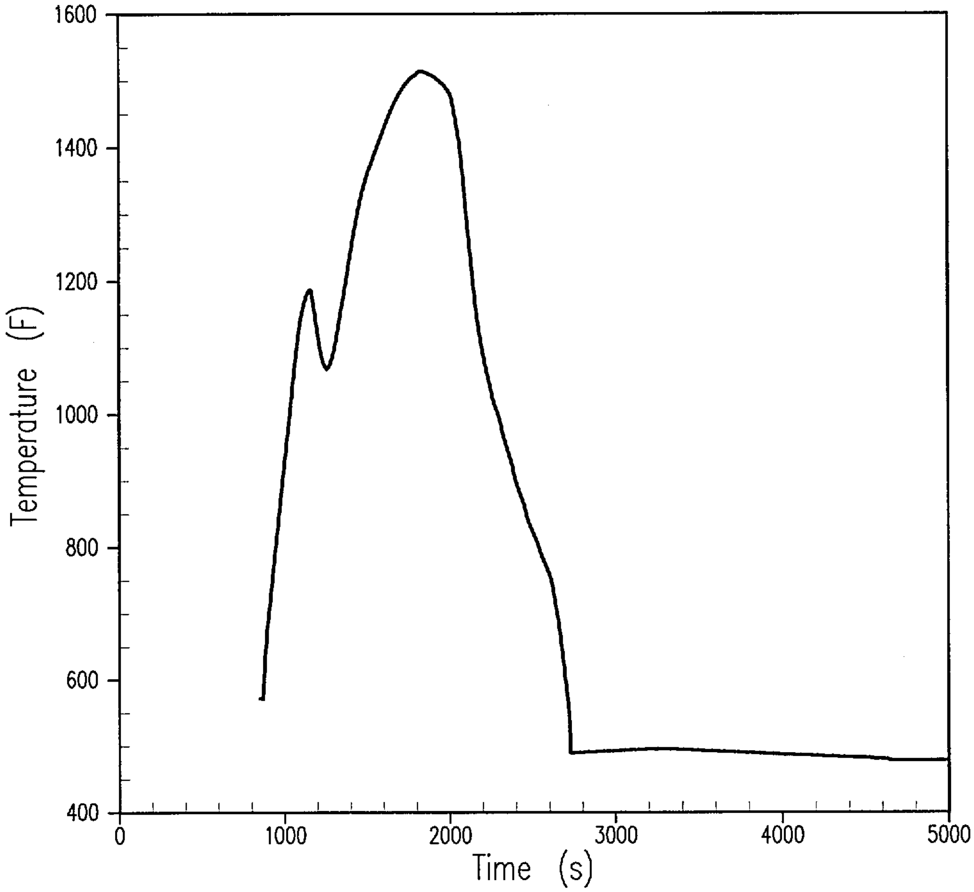

Figure 6.1.1-32

Units 1 High T_{avg} 3-Inch

Peak Clad Temperature at 11.5 ft.

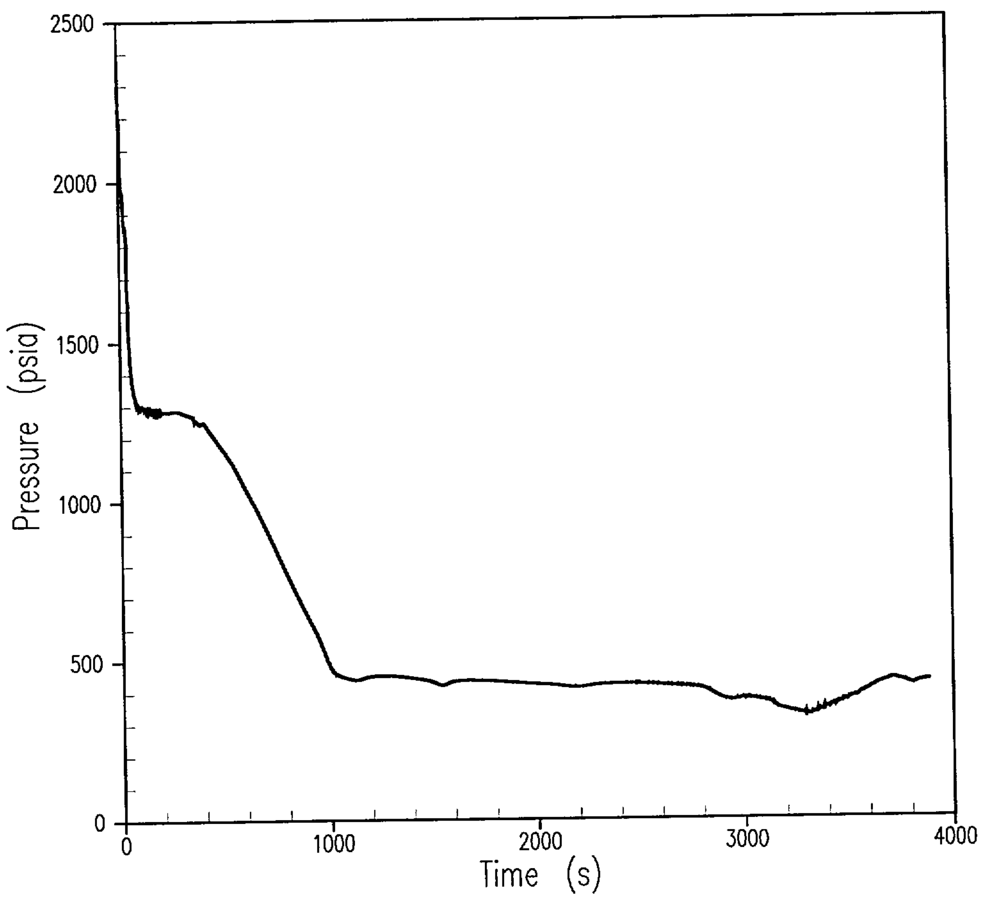

Figure 6.1.1-33

Units 1 High T_{avg} 4-Inch

RCS Pressure

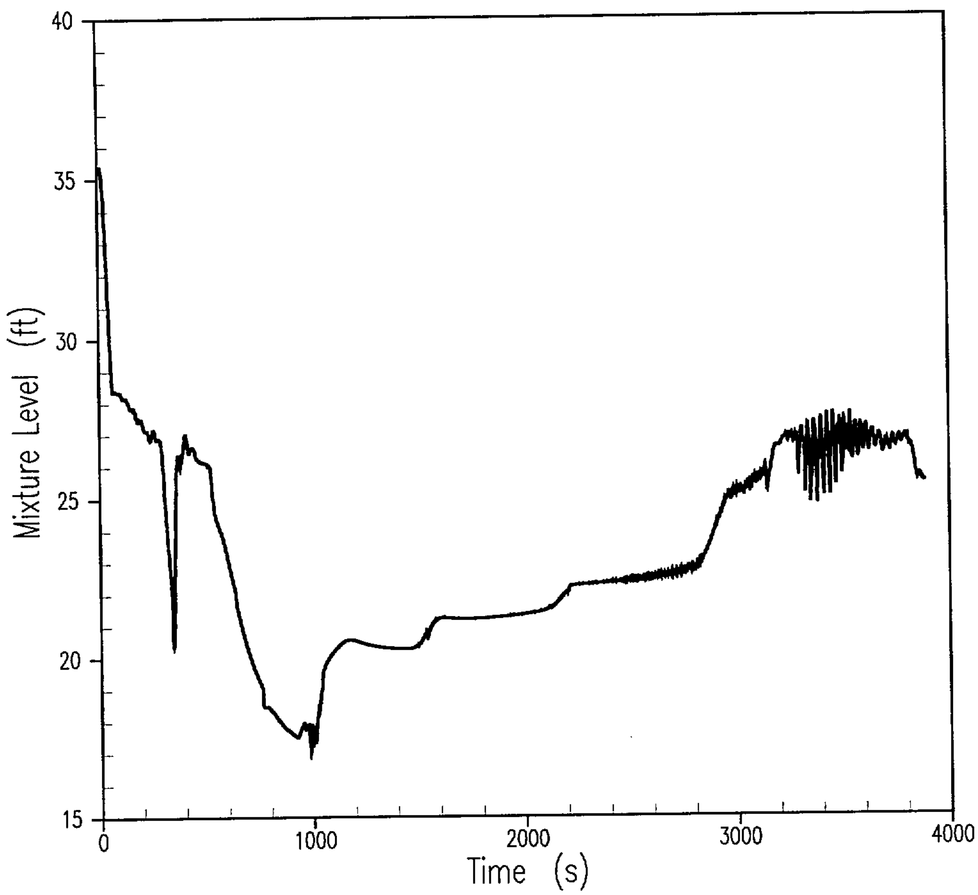

Figure 6.1.1-34

Units 1 High T_{avg} 4-Inch

Core Mixture Level

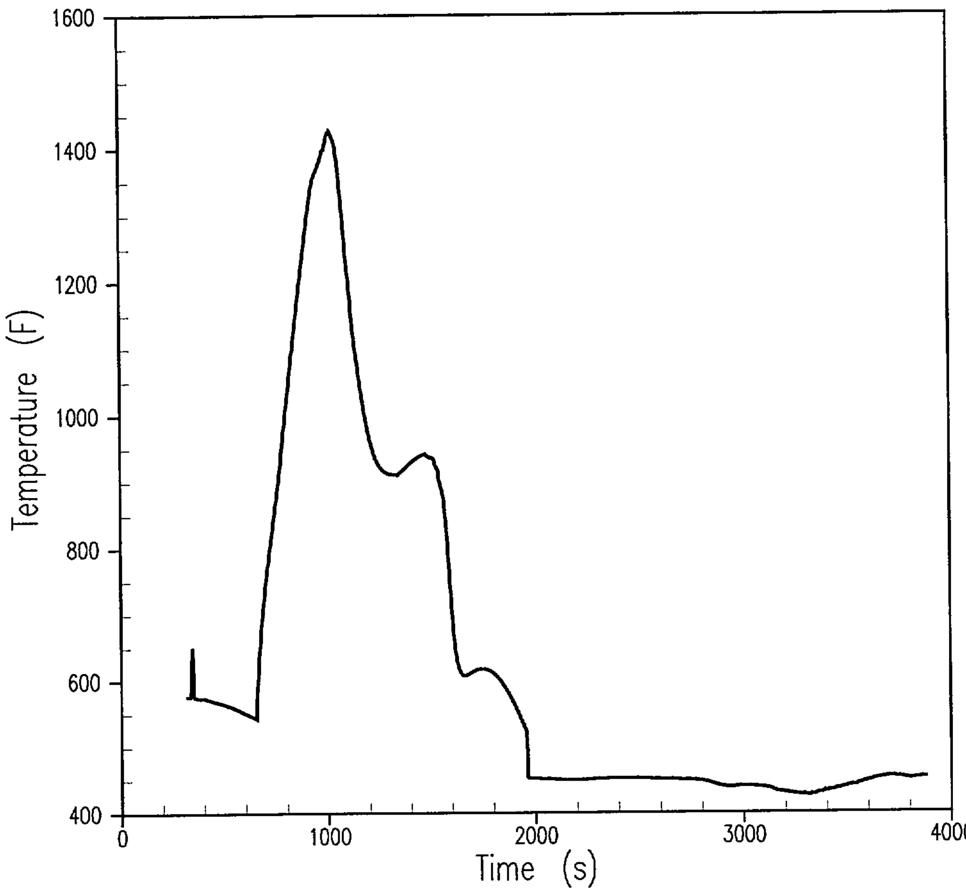

Figure 6.1.1-35

Units 1 High T_{avg} 4-Inch

Peak Clad Temperature at 11.25 ft.

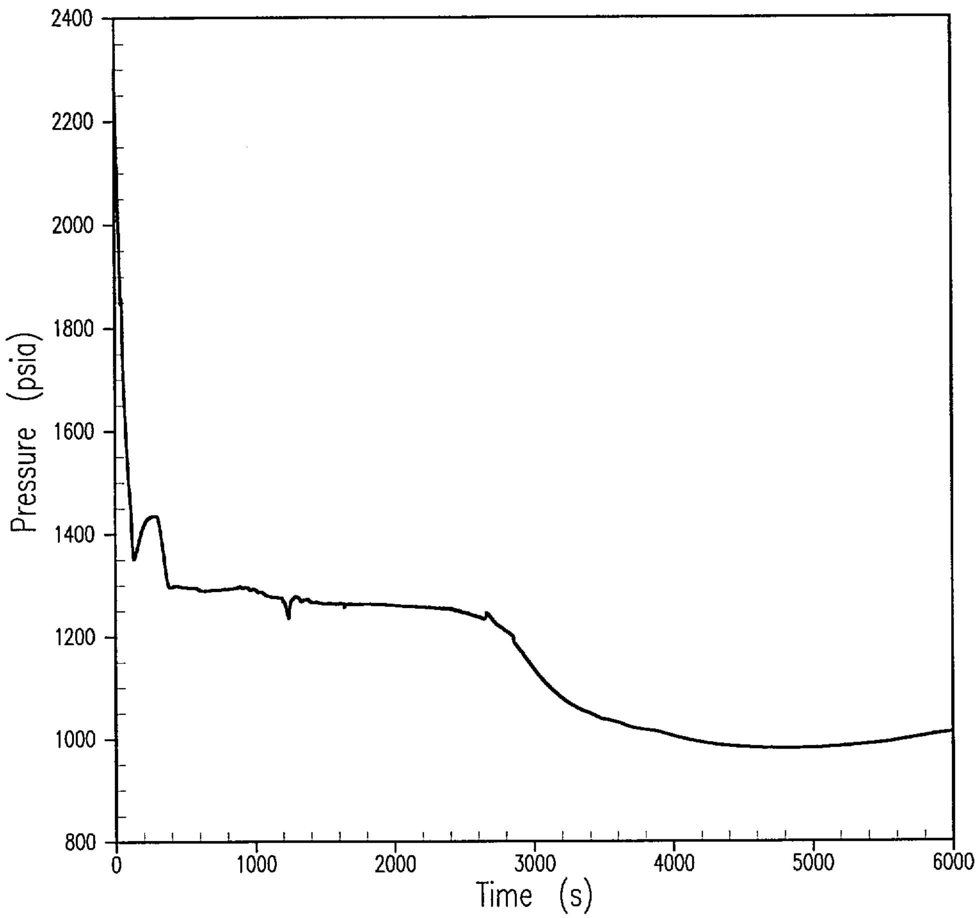

Figure 6.1.1 – 36
Units 2 Low T_{avg} 2-Inch
RCS Pressure

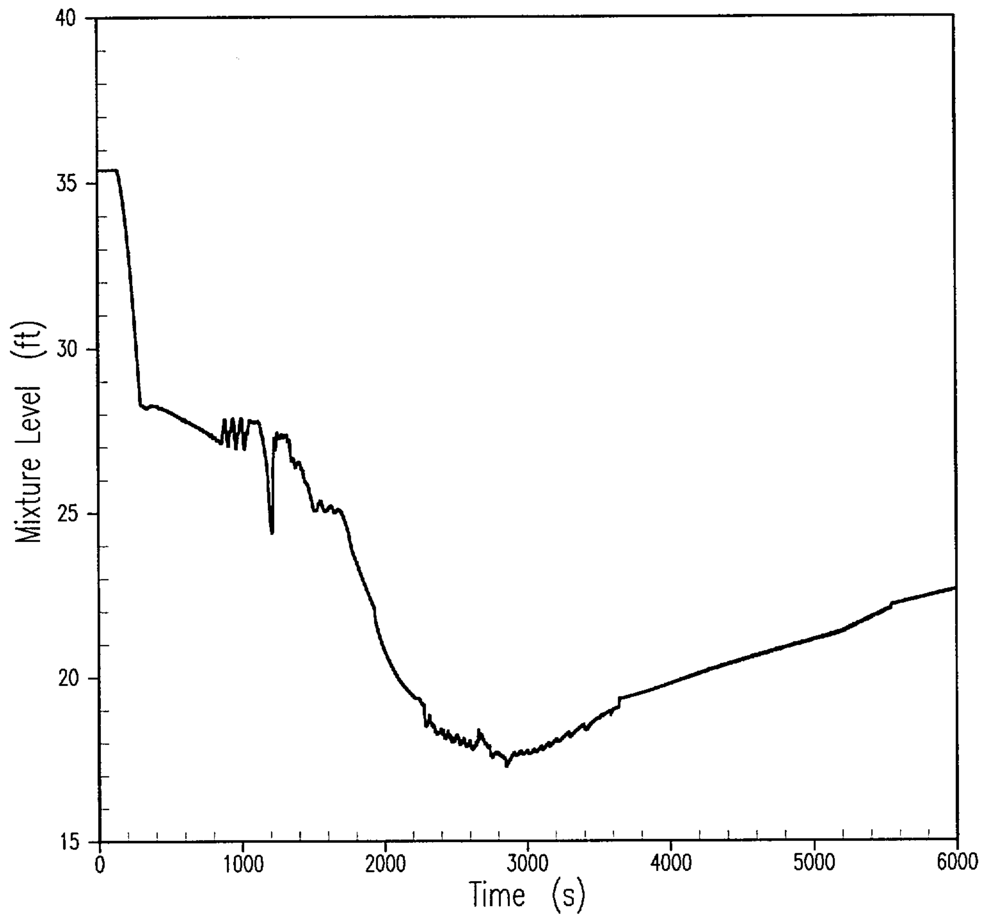

Figure 6.1.1 – 37
Units 2 Low T_{avg} 2-Inch
Core Mixture Level

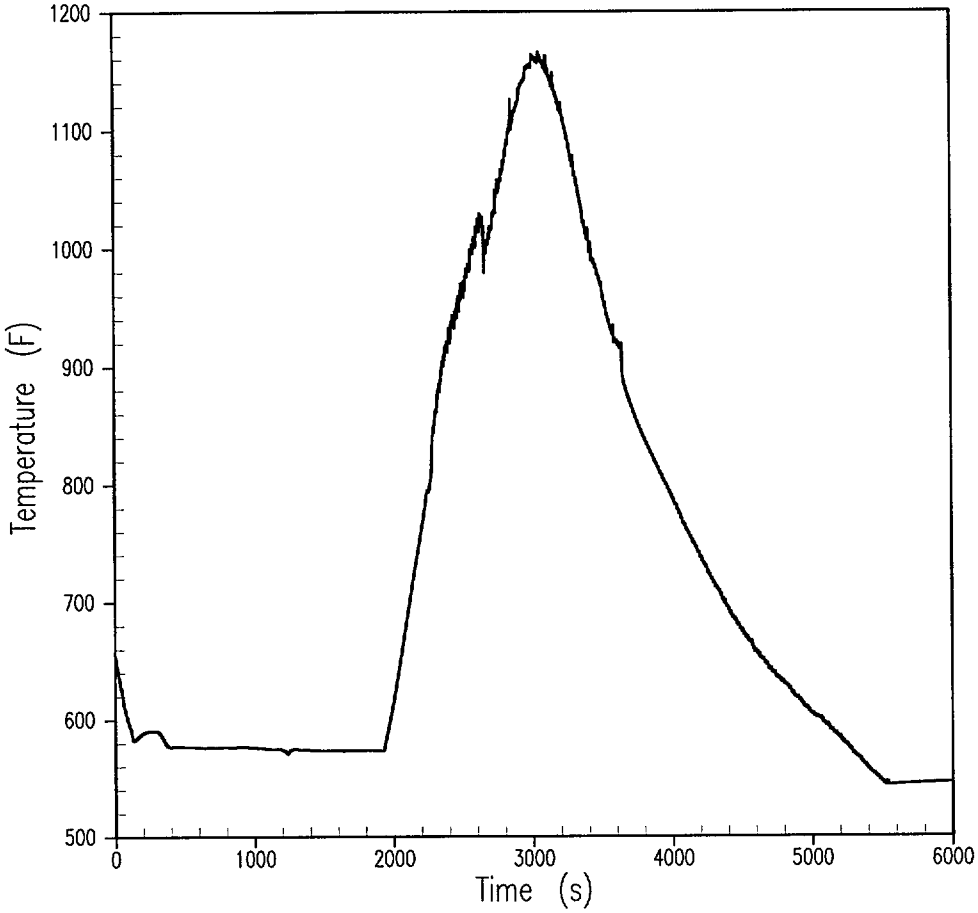

Figure 6.1.1 – 38
Units 2 Low T_{avg} 2-Inch
Core Exit Vapor Temperature

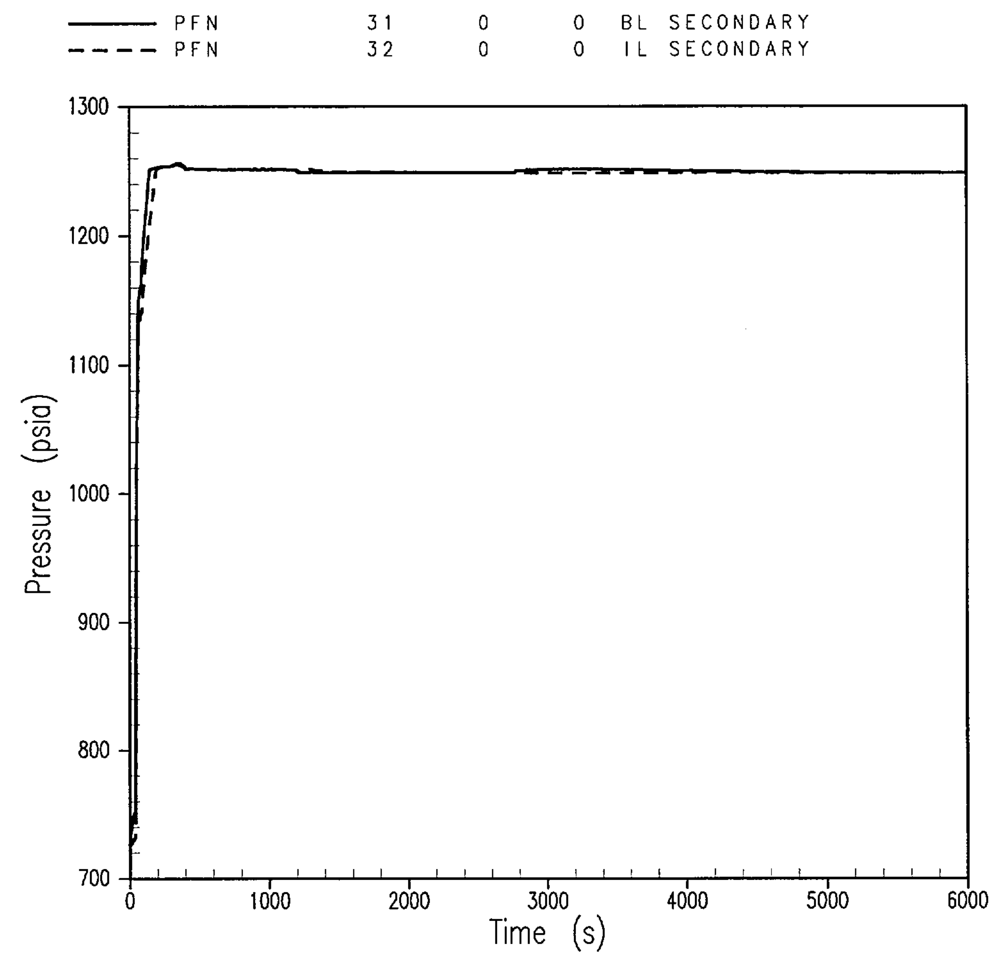

Figure 6.1.1 – 39
Units 2 Low T_{avg} 2-Inch
Broken Loop and Intact Loop Secondary Pressure

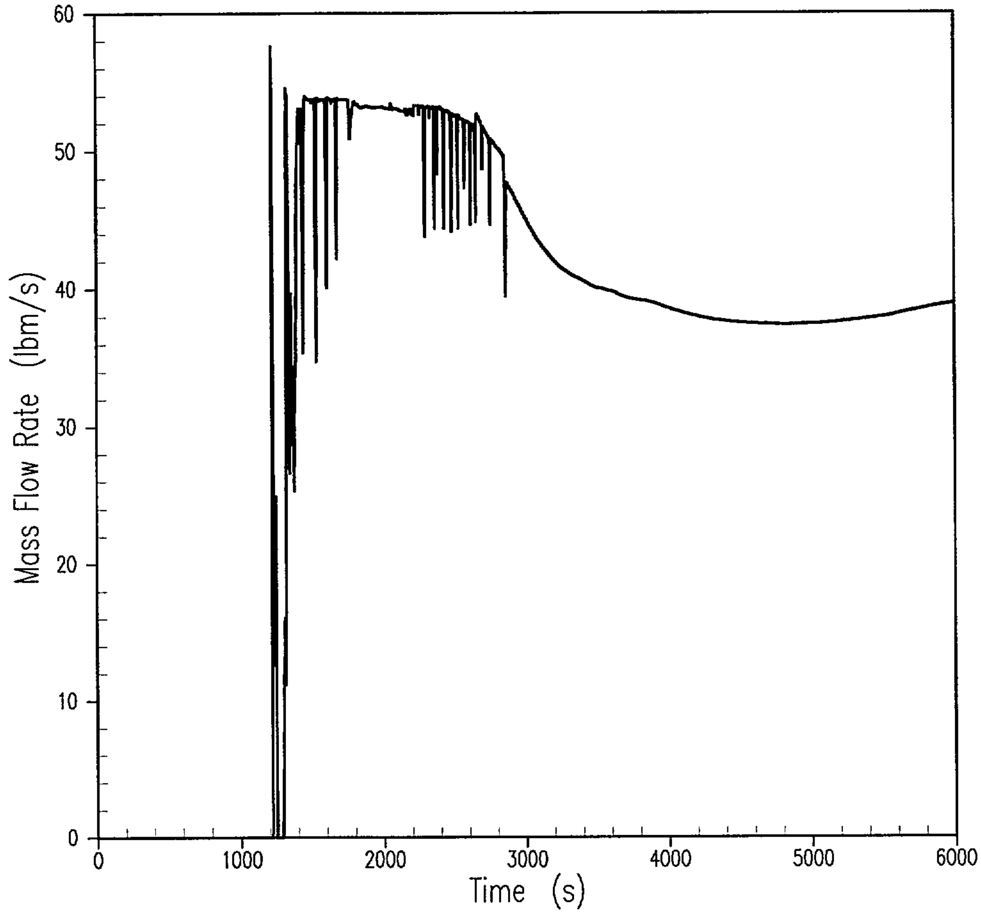

Figure 6.1.1 – 40
Units 2 Low T_{avg} 2-Inch
Break Vapor Flow Rate

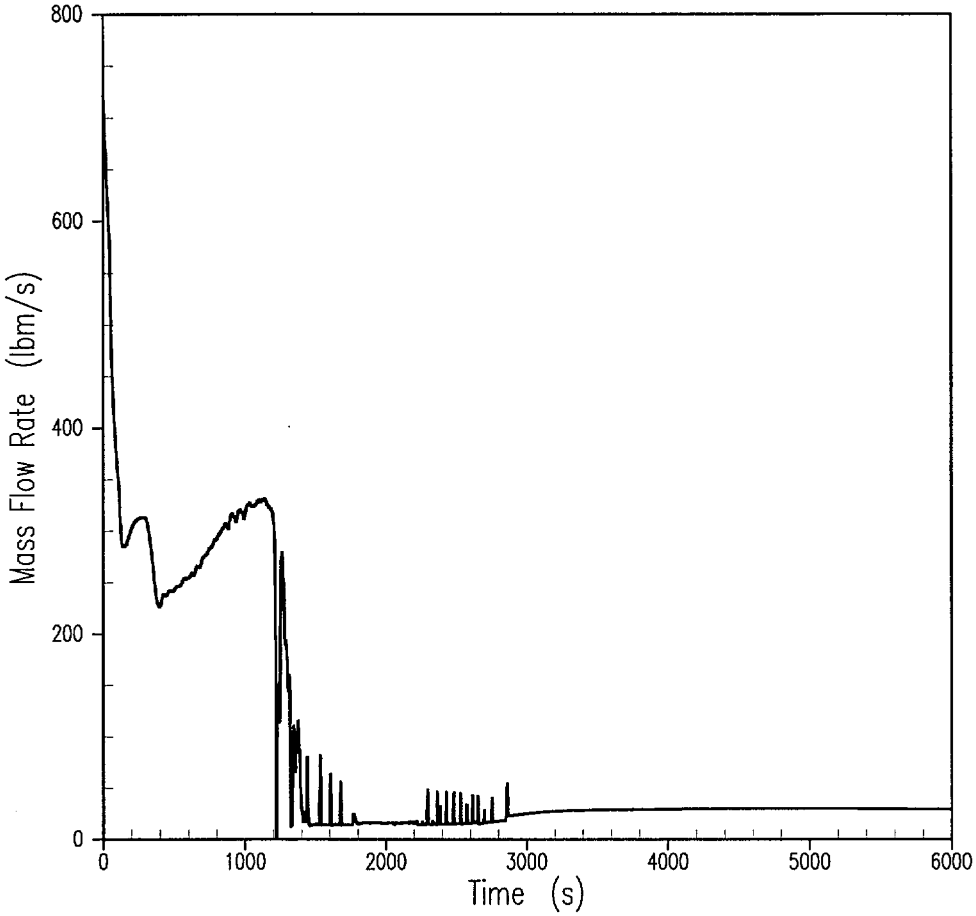

Figure 6.1.1 – 41
Units 2 Low T_{avg} 2-Inch
Break Liquid Flow Rate

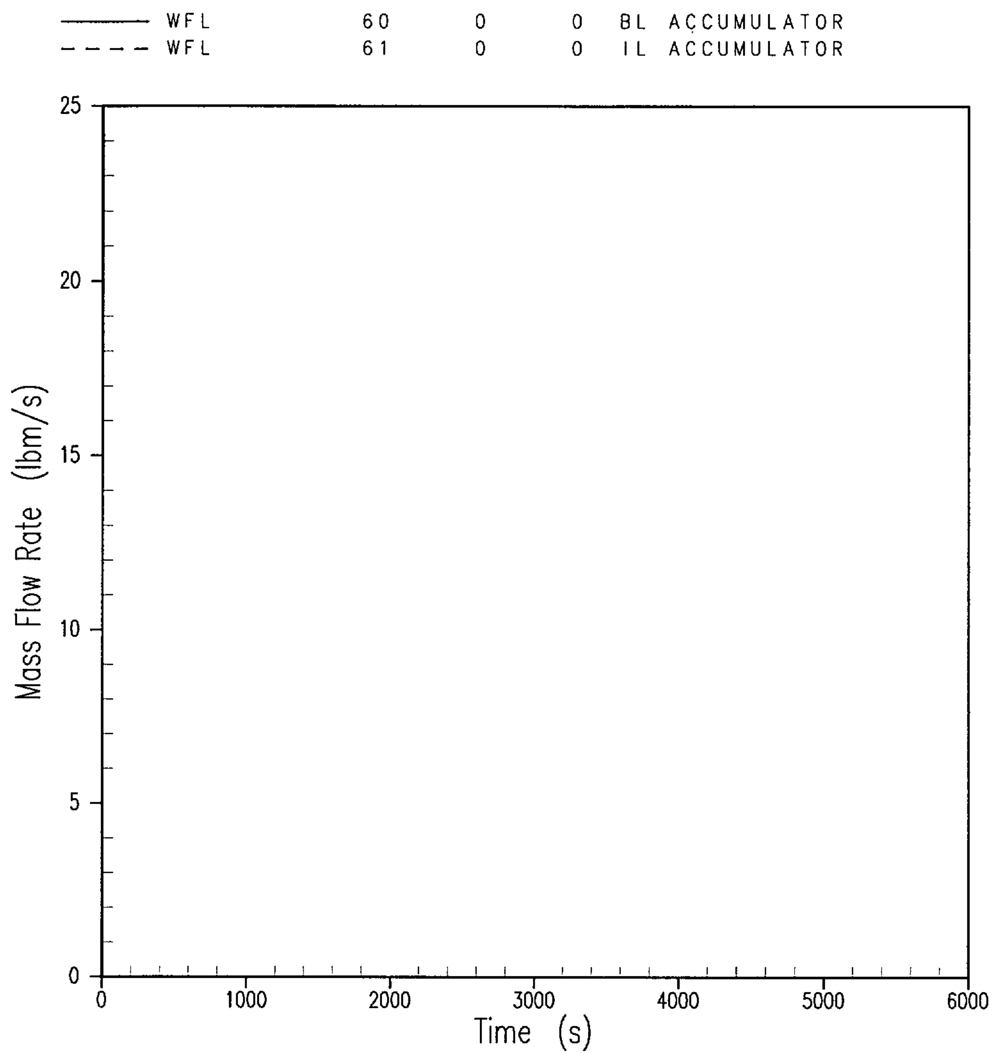

Figure 6.1.1 – 42
Units 2 Low T_{avg} 2-Inch
Broken Loop and Intact Loop Accumulator Flow Rate

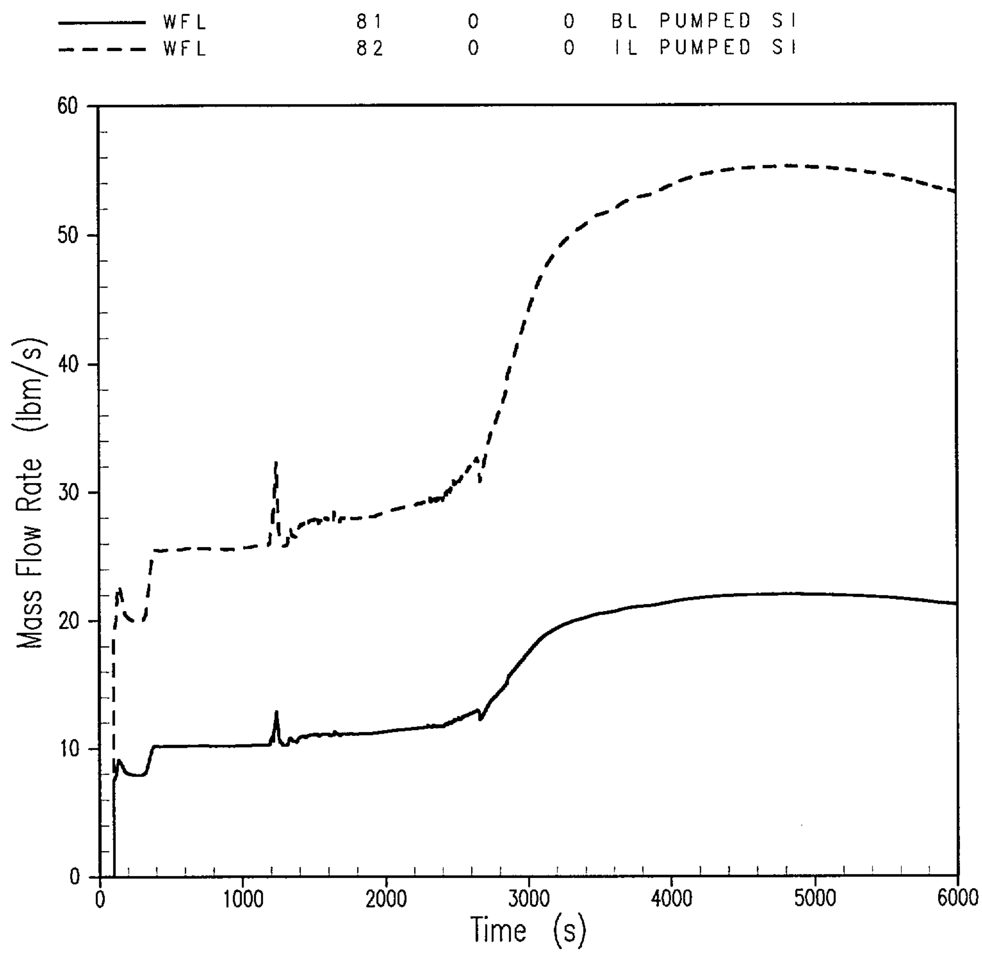

Figure 6.1.1 – 43
Units 2 Low T_{avg} 2-Inch
Broken Loop and Intact Loop Pumped Safety Injection Flow Rate

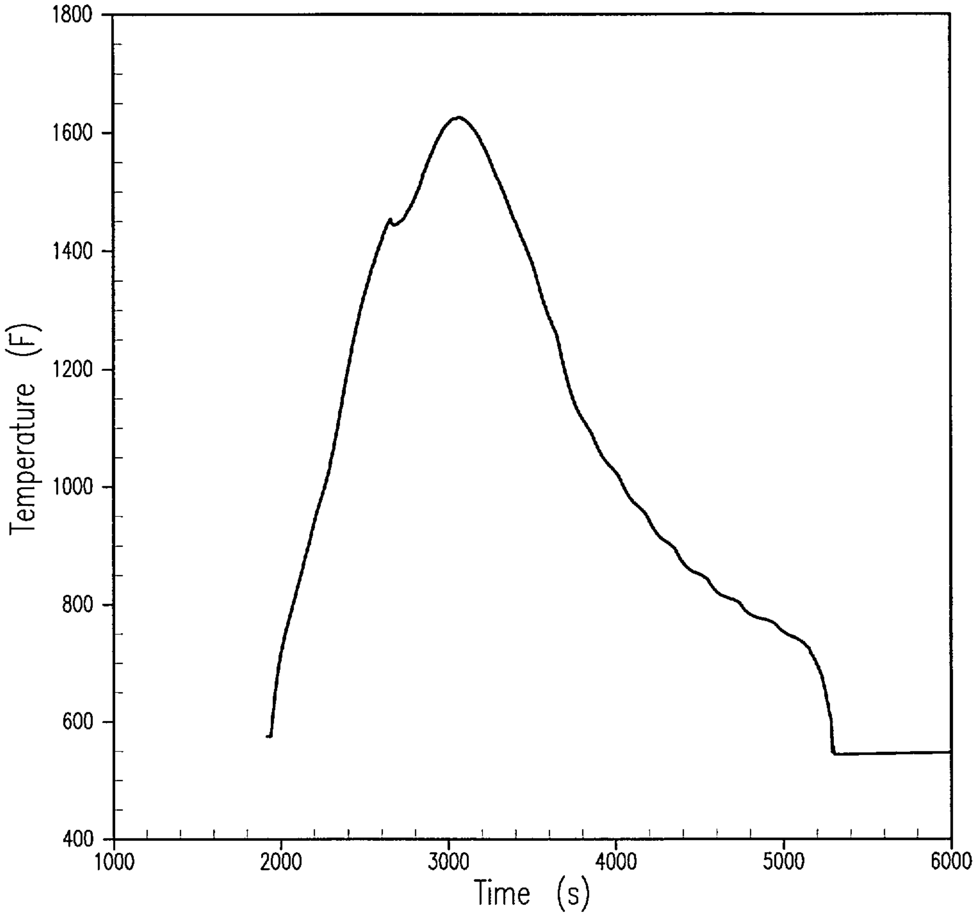

Figure 6.1.1 – 44
Units 2 Low T_{avg} 2-Inch
Peak Clad Temperature at 11.5 ft.

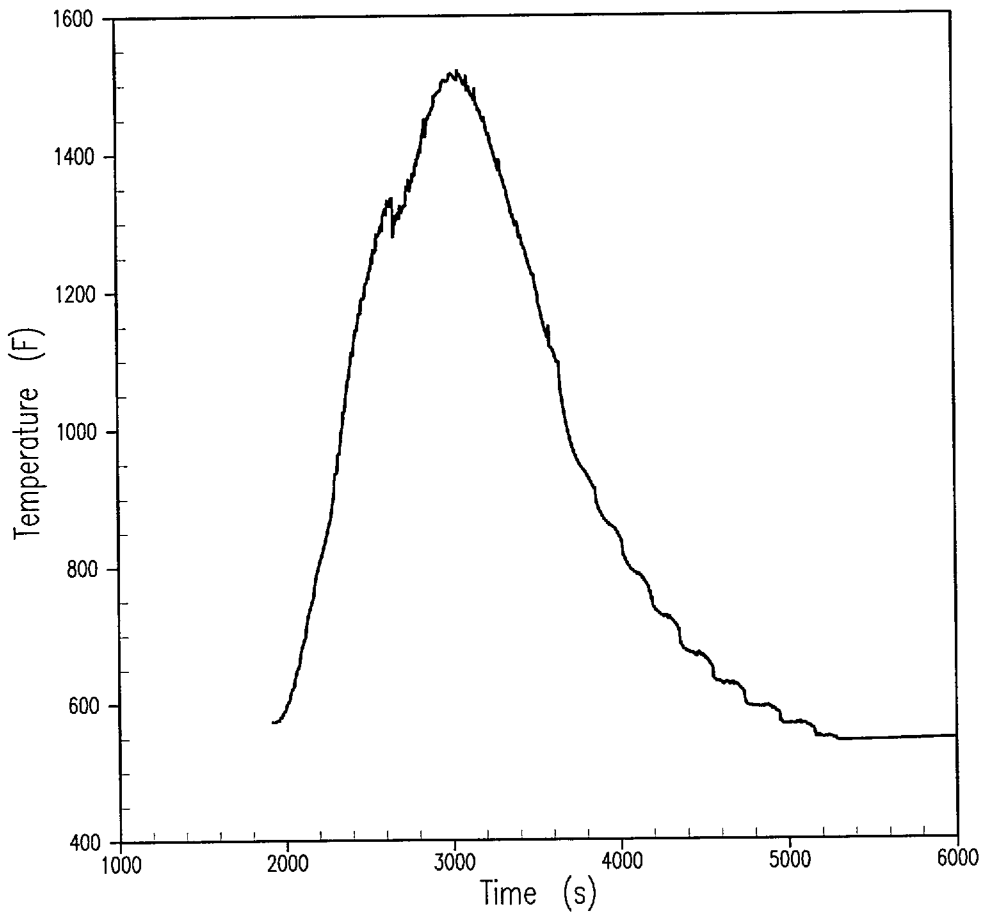

Figure 6.1.1 – 45
Units 2 Low T_{avg} 2-Inch
Hot Spot Fluid Temperature

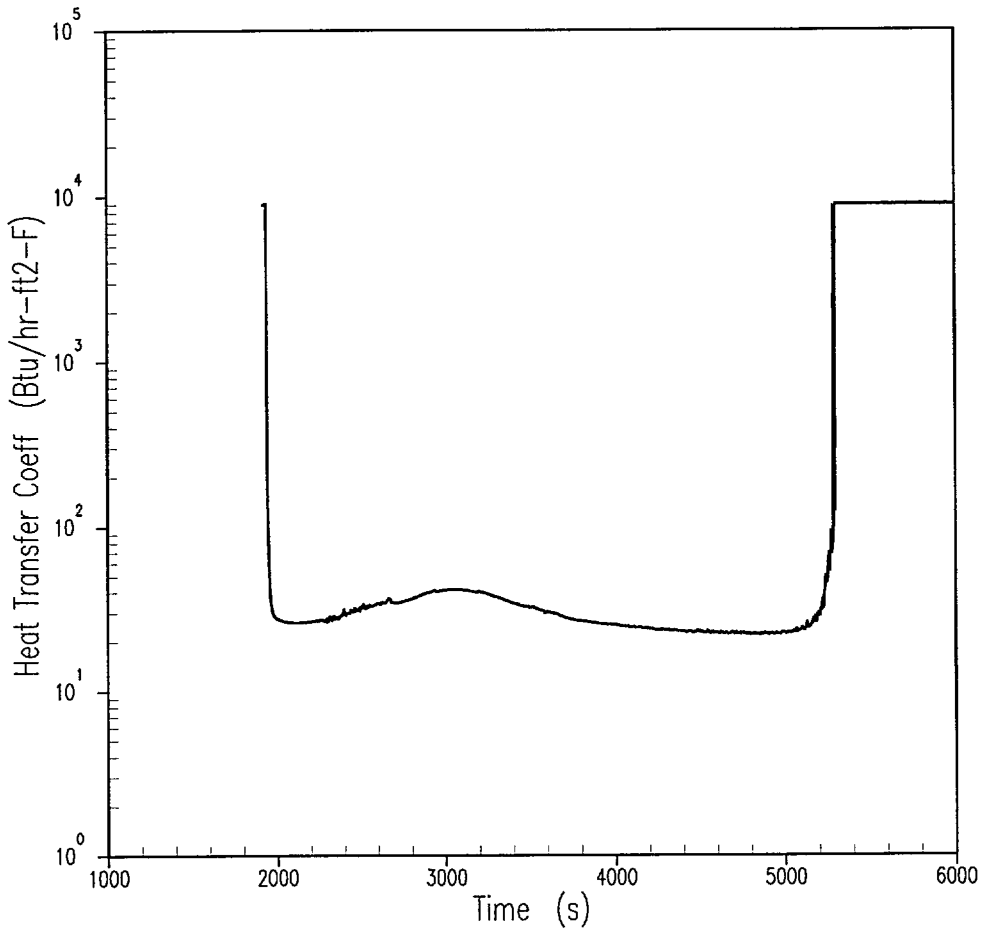

Figure 6.1.1 – 46
Units 2 Low T_{avg} 2-Inch
Rod Film Heat Transfer Coefficient at 11.5 ft.

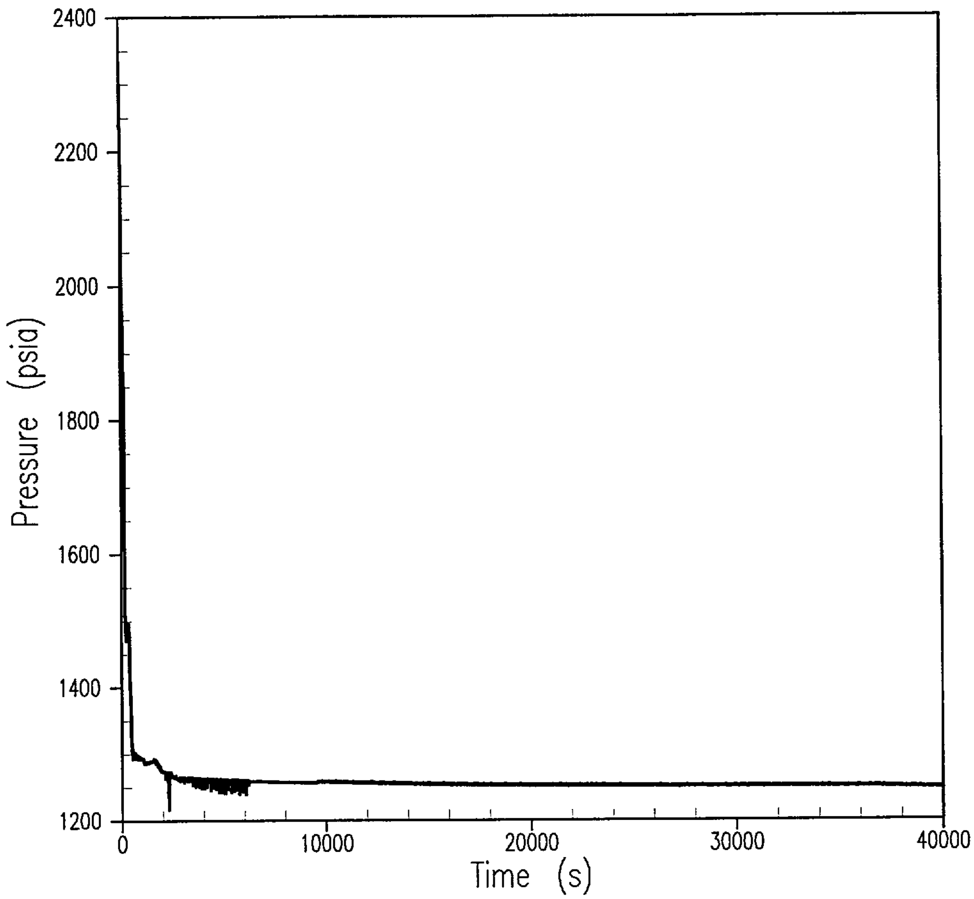

Figure 6.1.1-47

Units 2 High T_{avg} 1.5-Inch

RCS Pressure

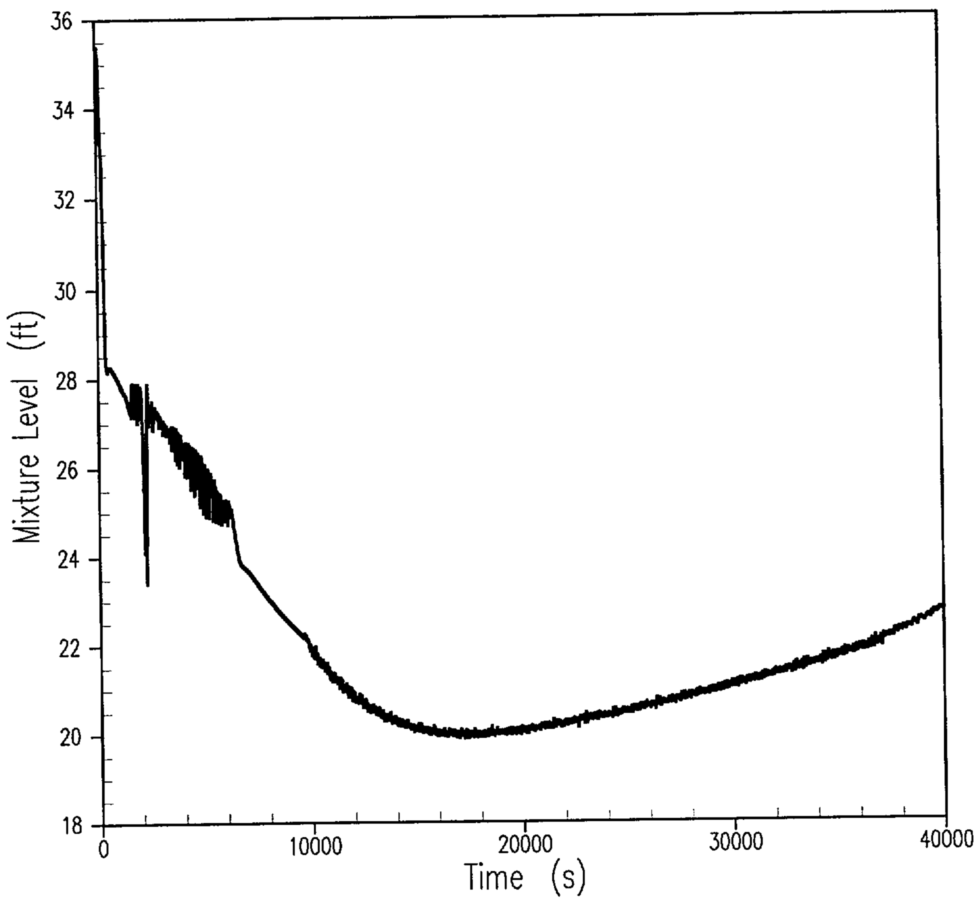

Figure 6.1.1-48

Units 2 High T_{avg} 1.5-Inch

Core Mixture Level

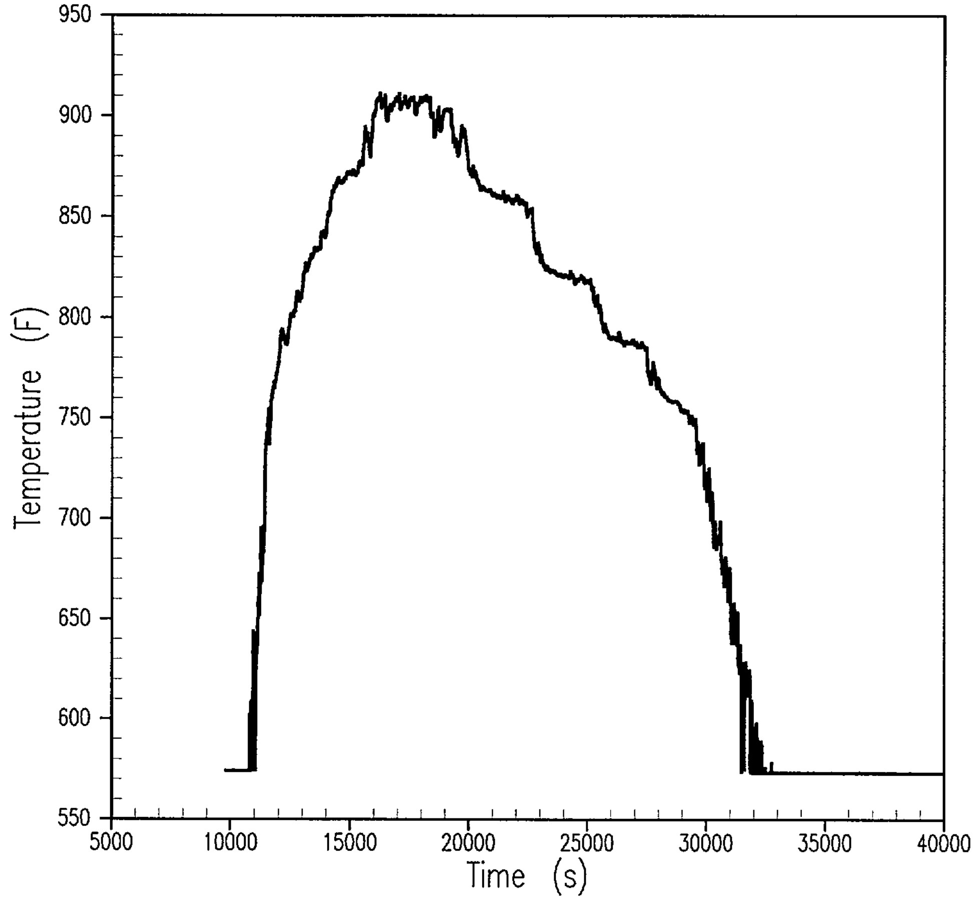

Figure 6.1.1-49

Units 2 High T_{avg} 1.5-Inch

Peak Clad Temperature at 11.25 ft.

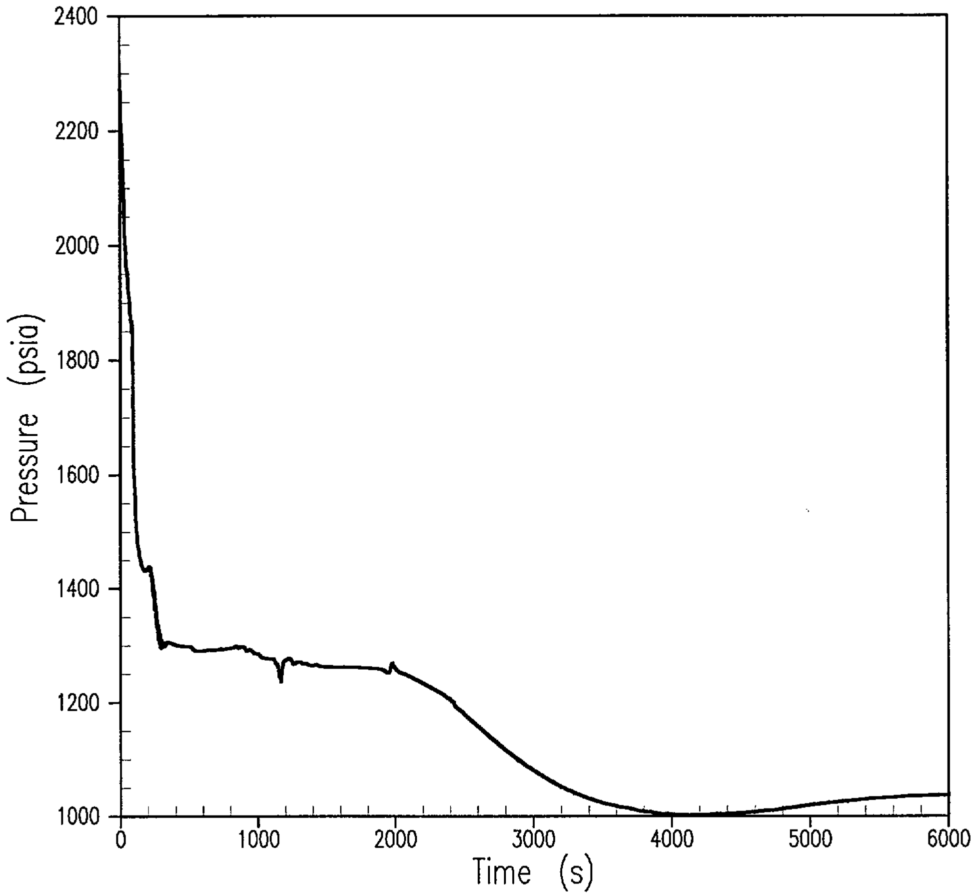

Figure 6.1.1-50

Units 2 High T_{avg} 2-Inch

RCS Pressure

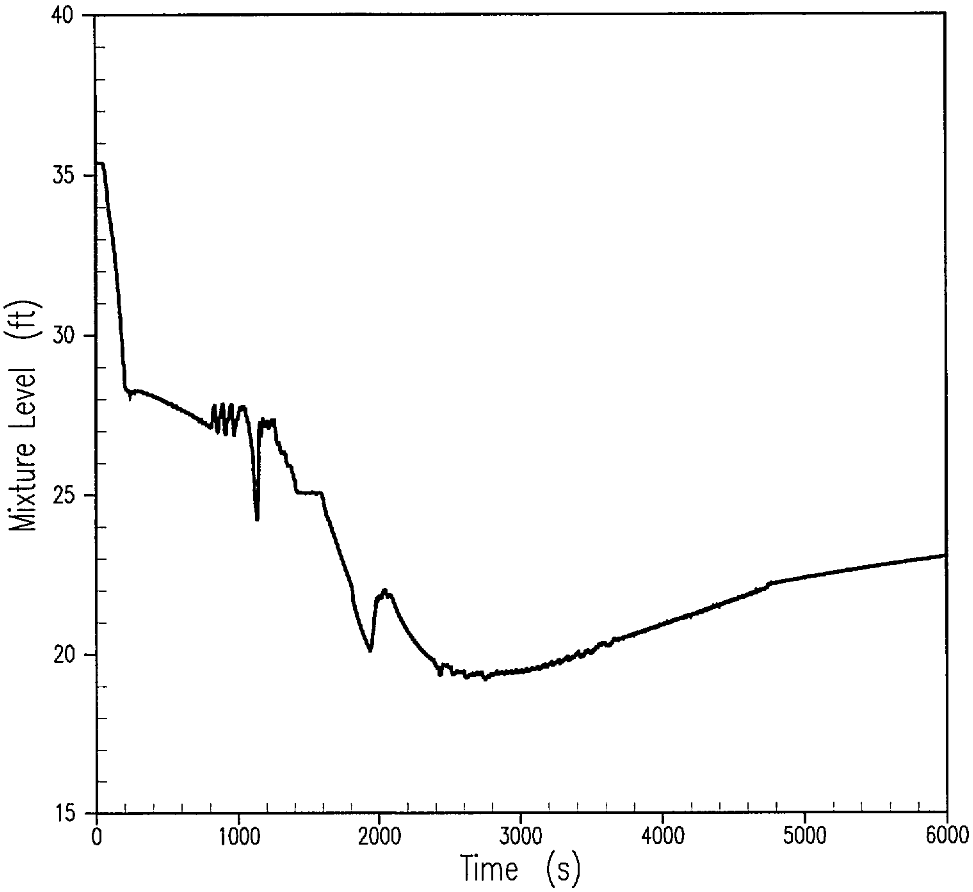

Figure 6.1.1-51

Units 2 High T_{avg} 2-Inch

Core Mixture Level

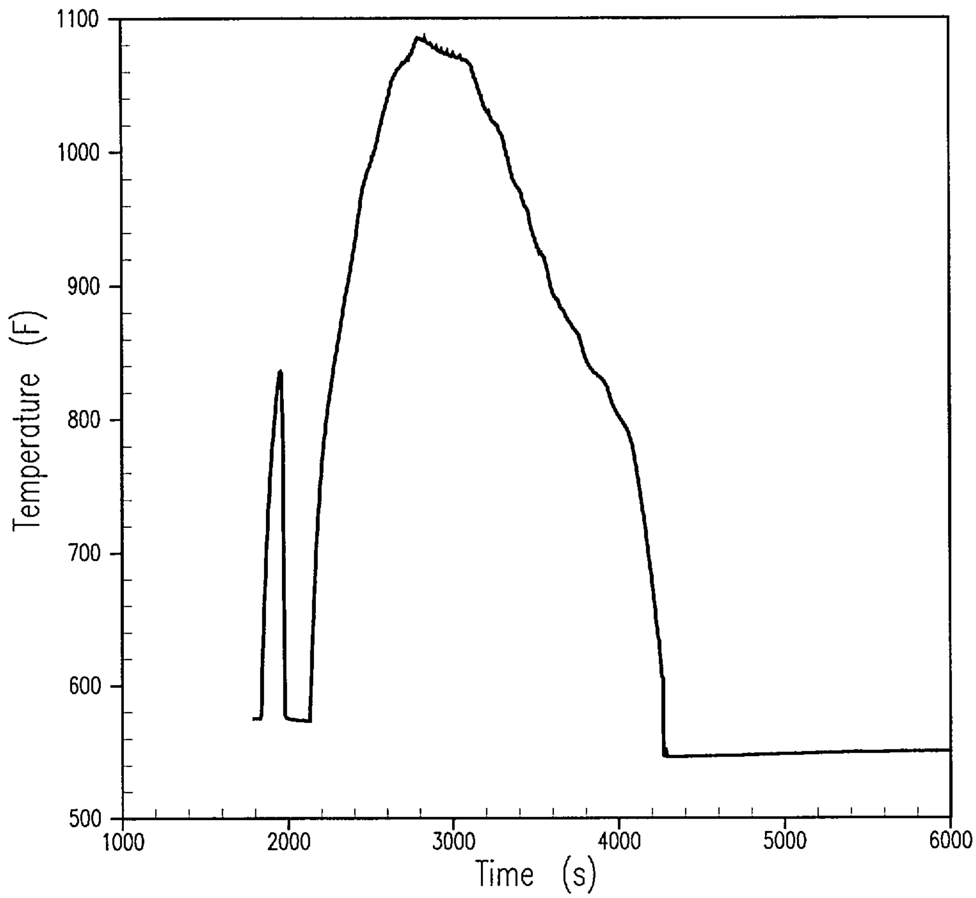

Figure 6.1.1-52

Units 2 High T_{avg} 2-Inch

Peak Clad Temperature at 11.25 ft.

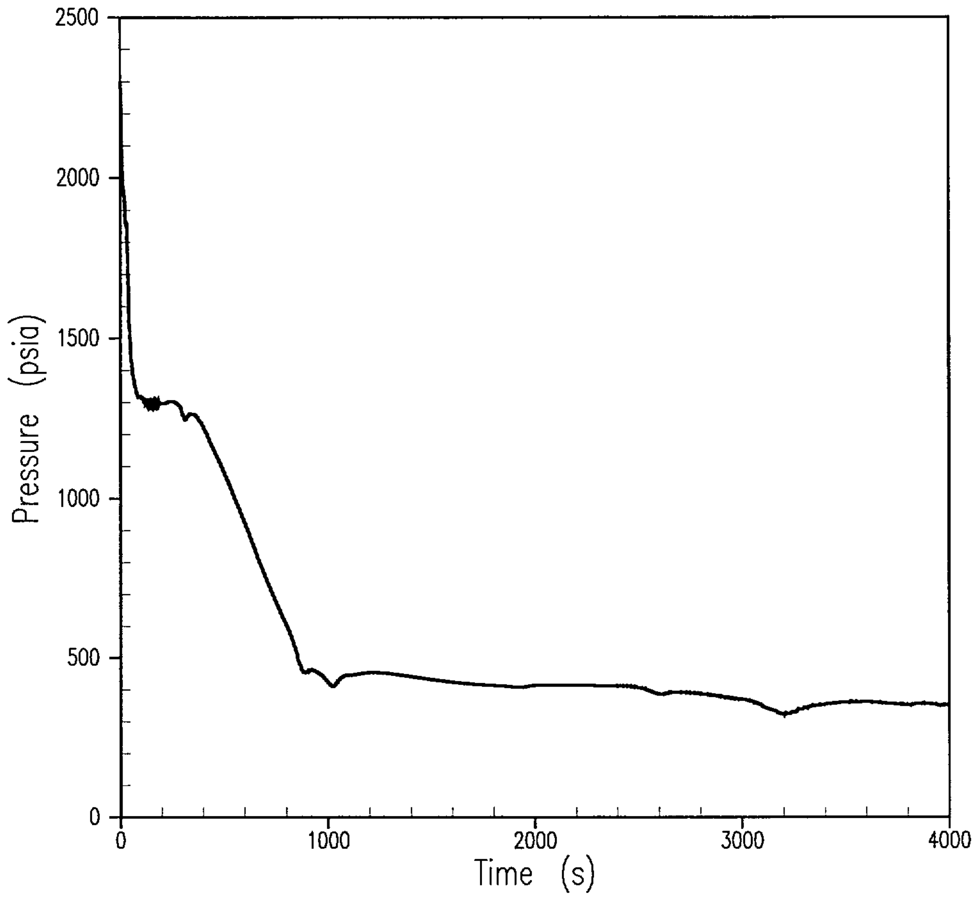

Figure 6.1.1-53

Units 2 High T_{avg} 4-Inch

RCS Pressure

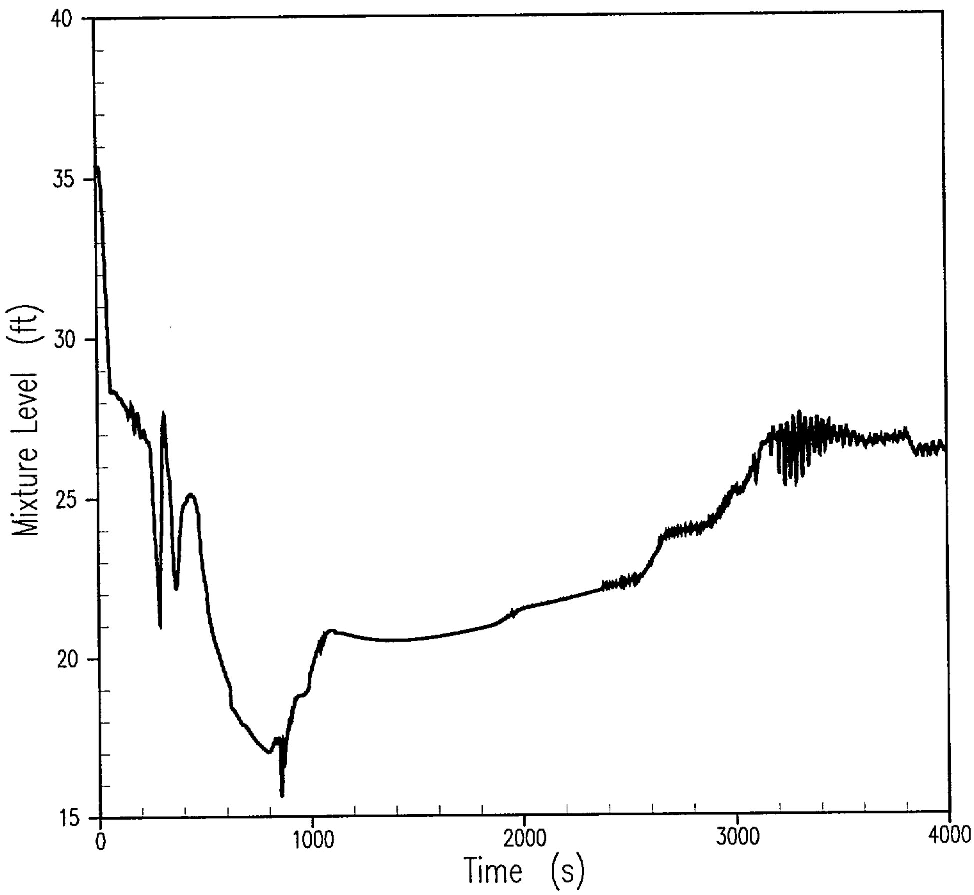

Figure 6.1.1-54

Units 2 High T_{avg} 4-Inch

Core Mixture Level

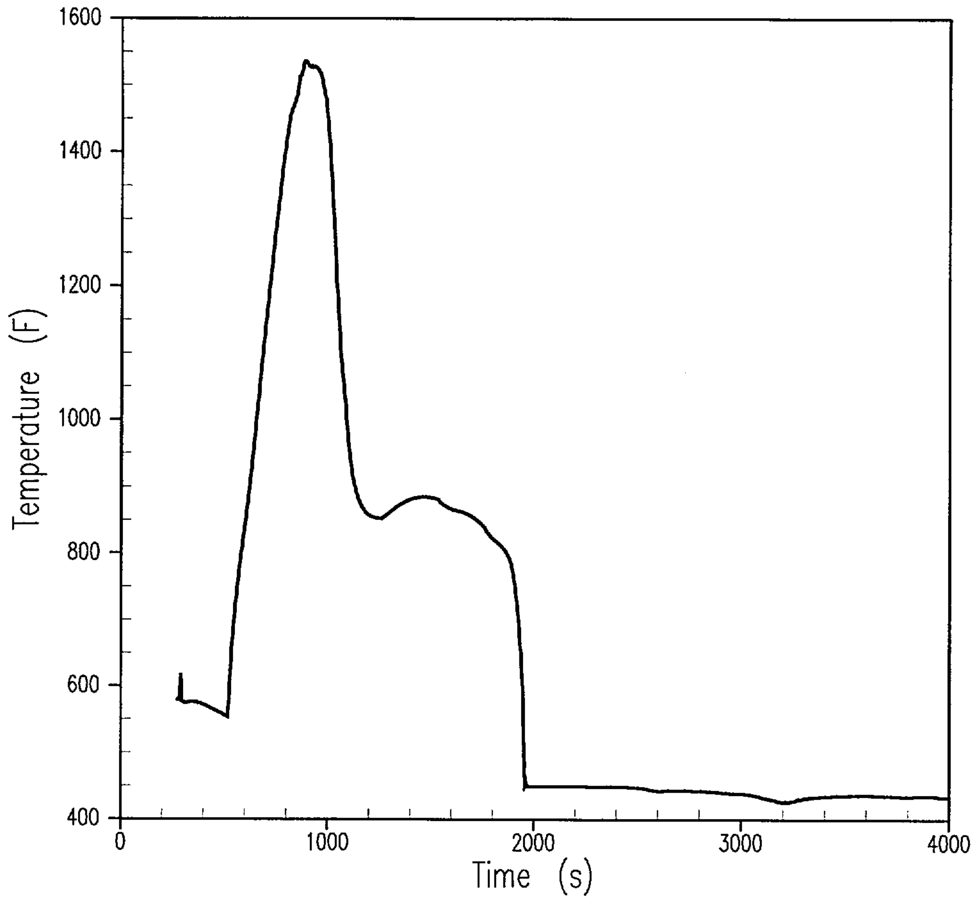

Figure 6.1.1-55

Units 2 High T_{avg} 4-Inch

Peak Clad Temperature at 11.25 ft.

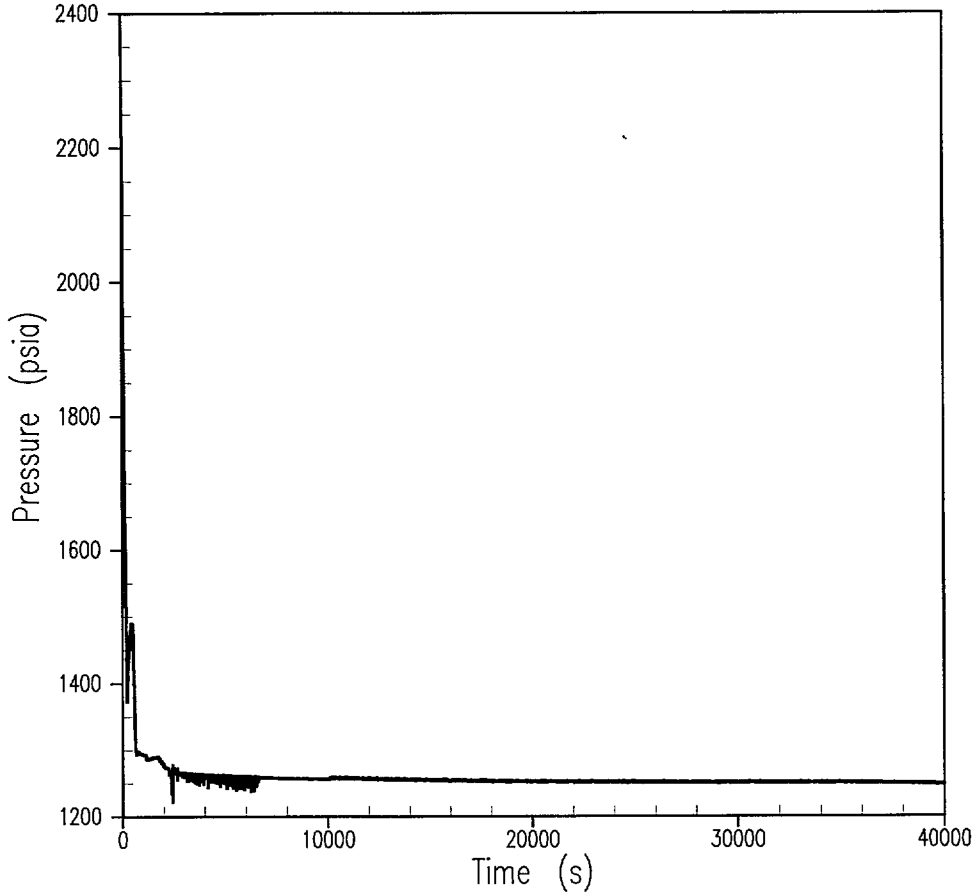

Figure 6.1.1-56

Units 2 Low T_{avg} 1.5-Inch

RCS Pressure

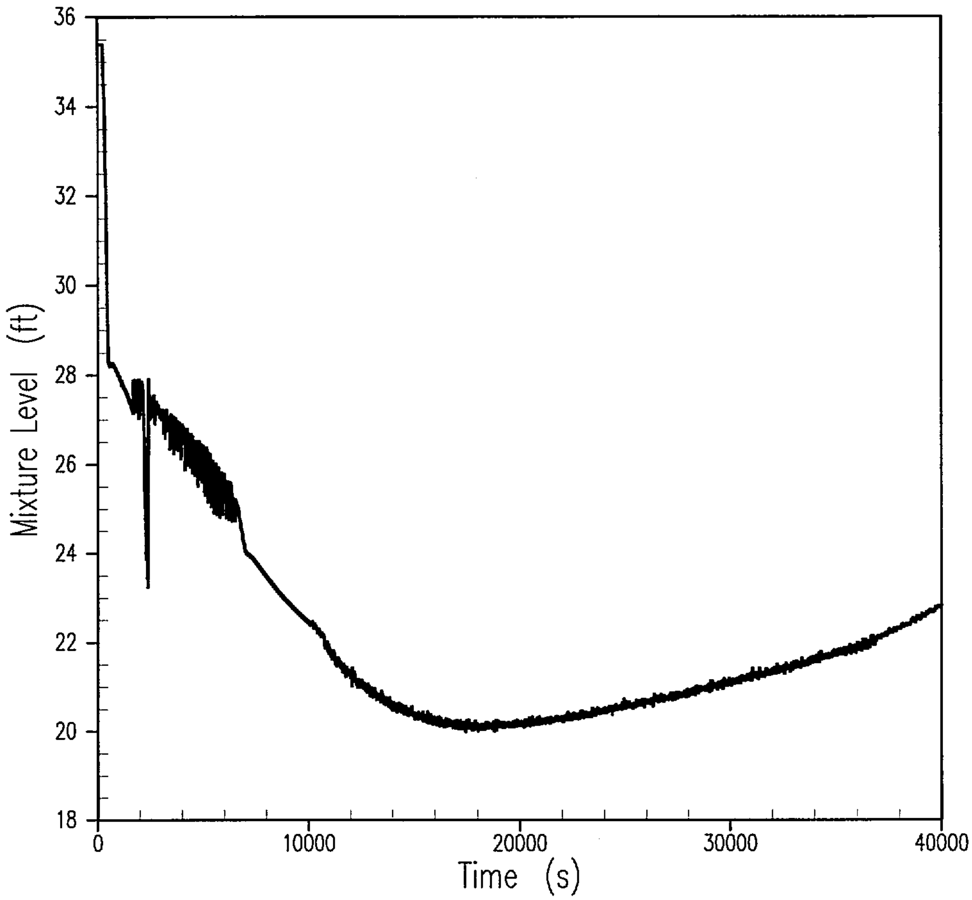

Figure 6.1.1-57

Units 2 Low T_{avg} 1.5-Inch

Core Mixture Level

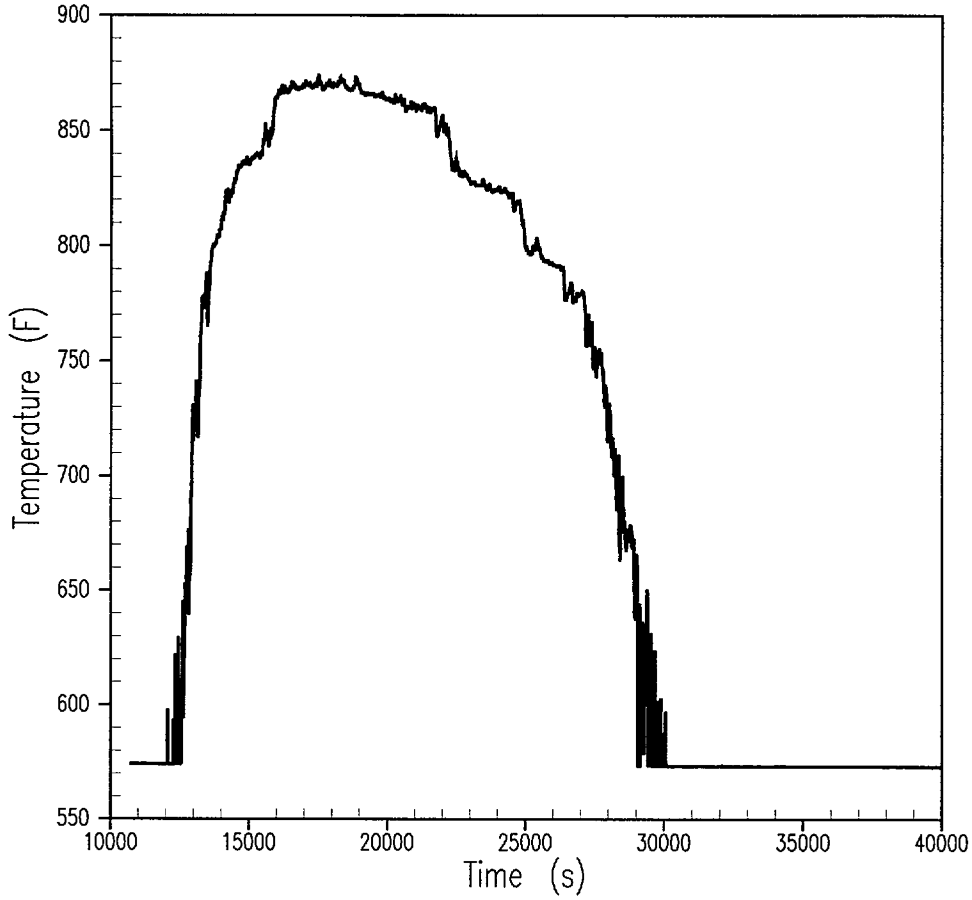

Figure 6.1.1-58

Units 2 Low T_{avg} 1.5-Inch

Peak Clad Temperature at 11.00 ft.

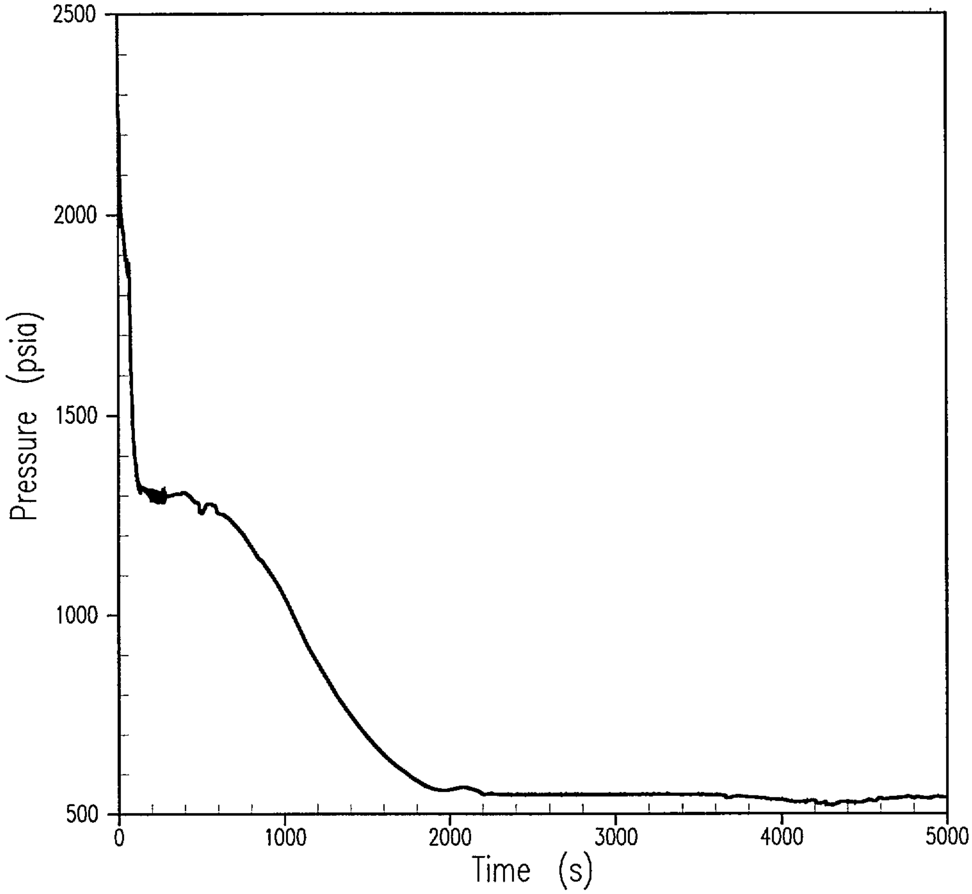

Figure 6.1.1-59

Units 2 High T_{avg} 3-Inch

RCS Pressure

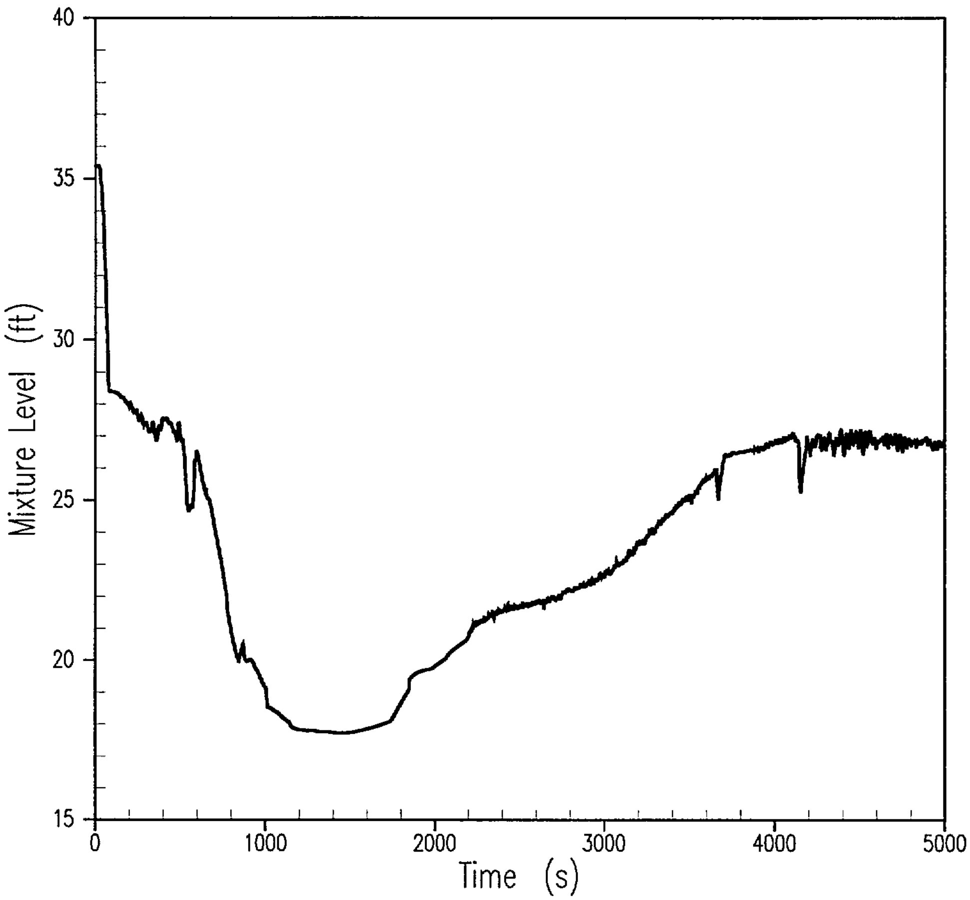

Figure 6.1.1-60

Units 2 High T_{avg} 3-Inch

Core Mixture Level

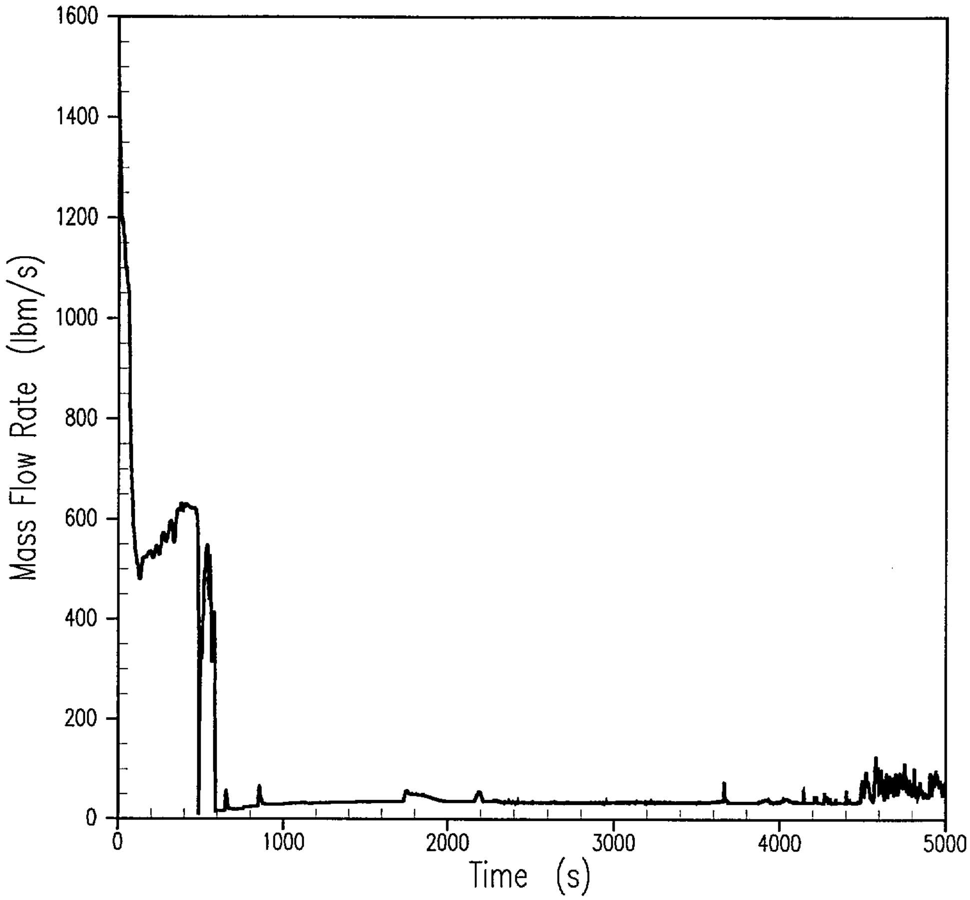

Figure 6.1.1-61

Units 2 High T_{avg} 3-Inch

Break Liquid Flow Rate

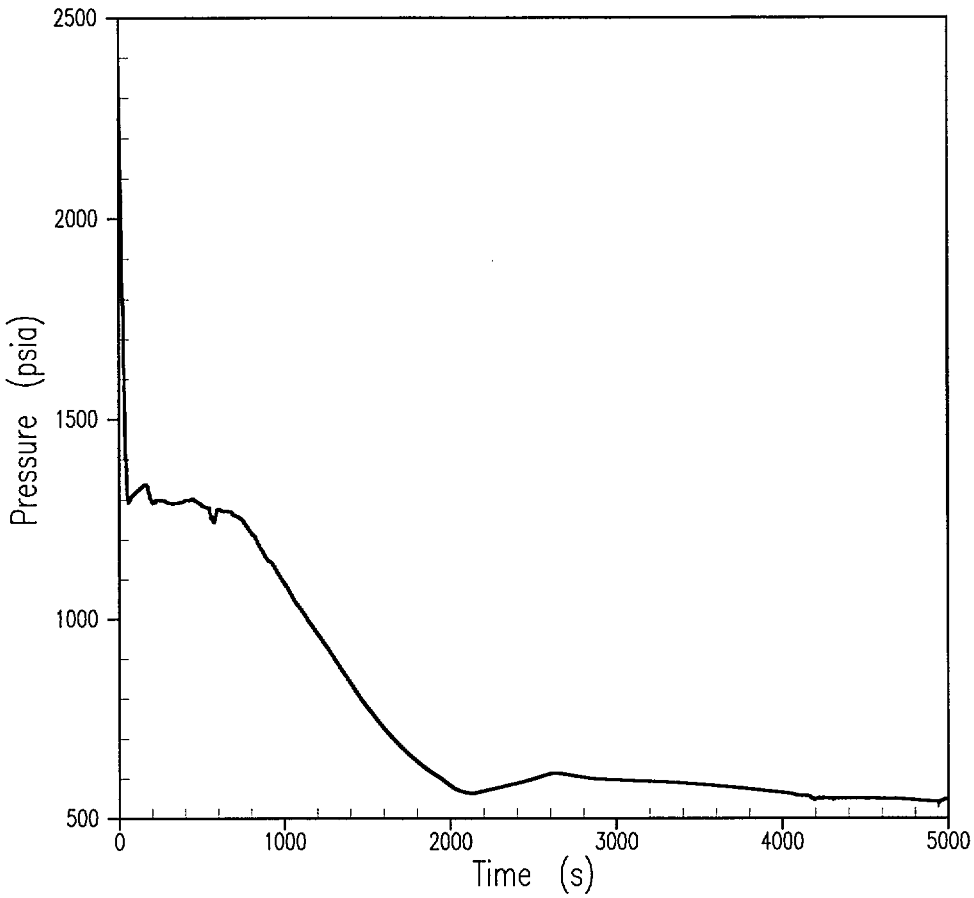

Figure 6.1.1-62

Units 2 Low T_{avg} 3-Inch

RCS Pressure

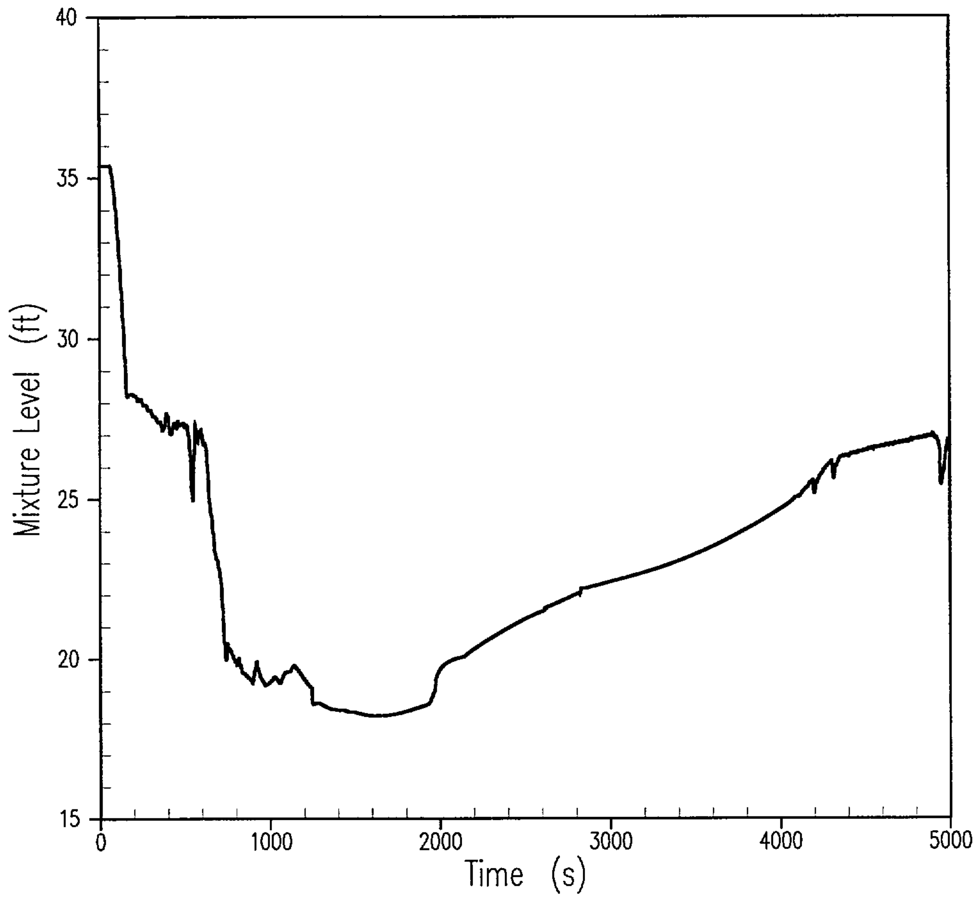

Figure 6.1.1-63

Units 2 Low T_{avg} 3-Inch

Core Mixture Level

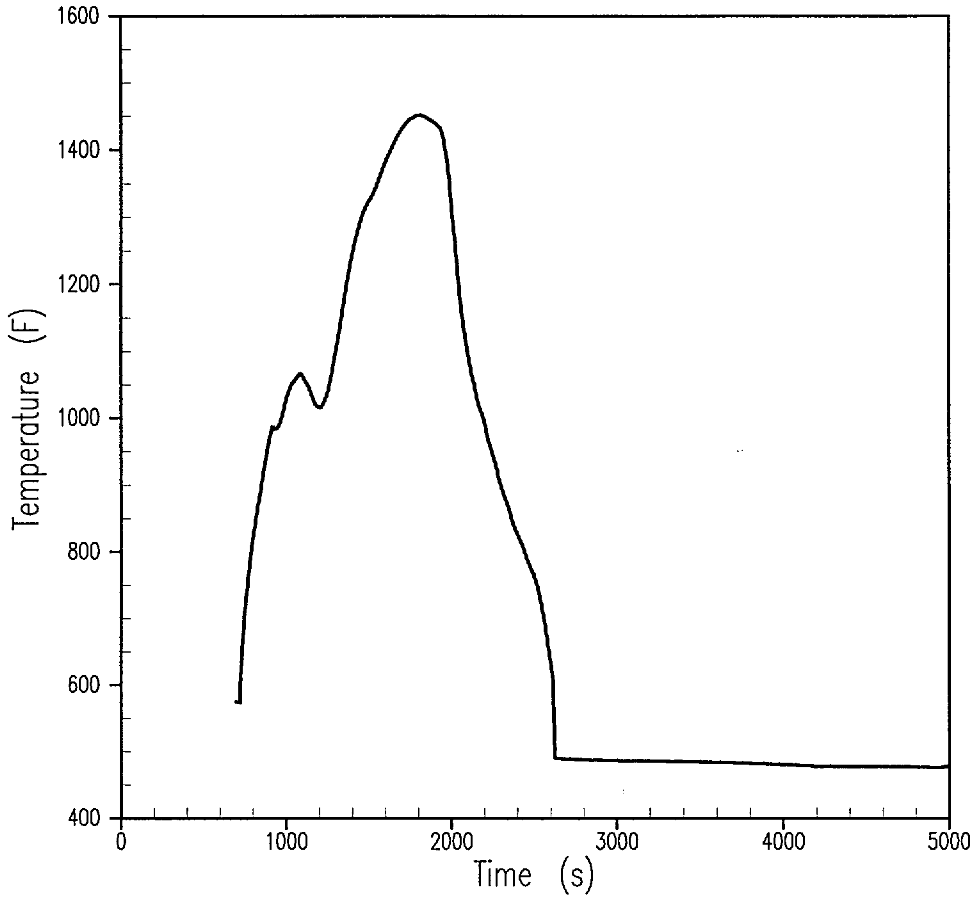

Figure 6.1.1-64

Units 2 Low T_{avg} 3-Inch

Peak Clad Temperature at 11.5 ft.

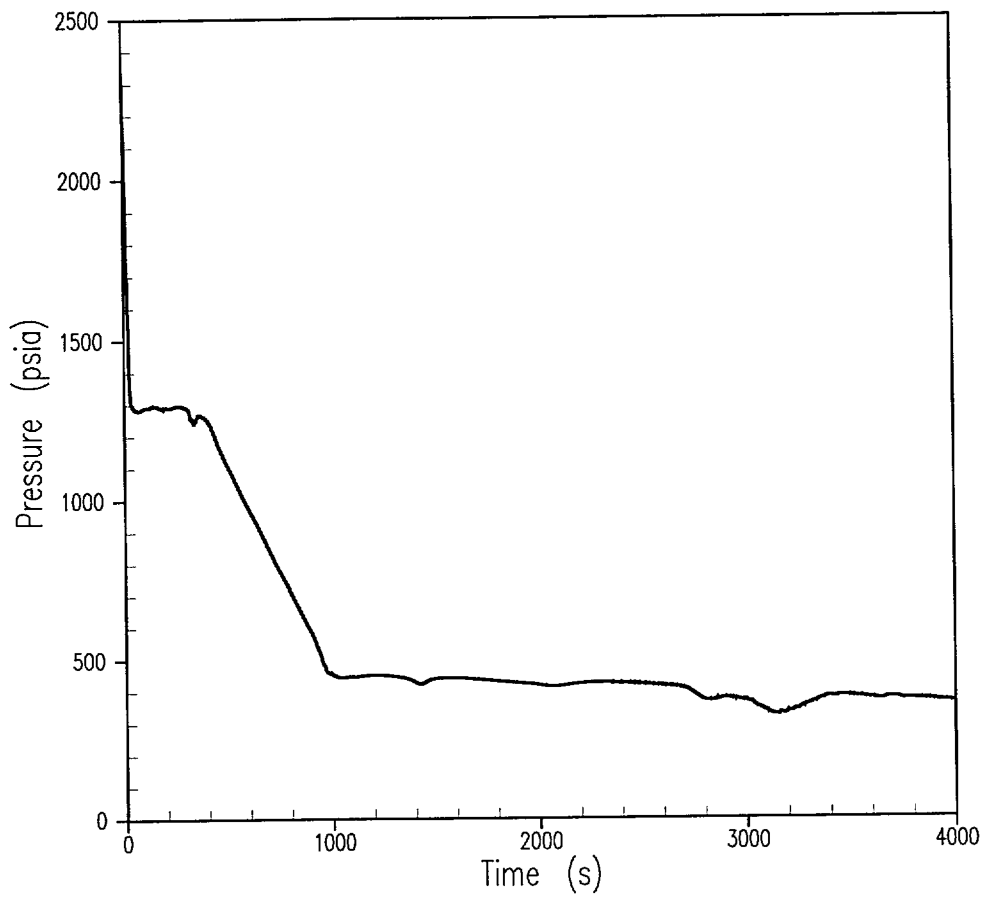

Figure 6.1.1-65

Units 2 Low T_{avg} 4-Inch

RCS Pressure

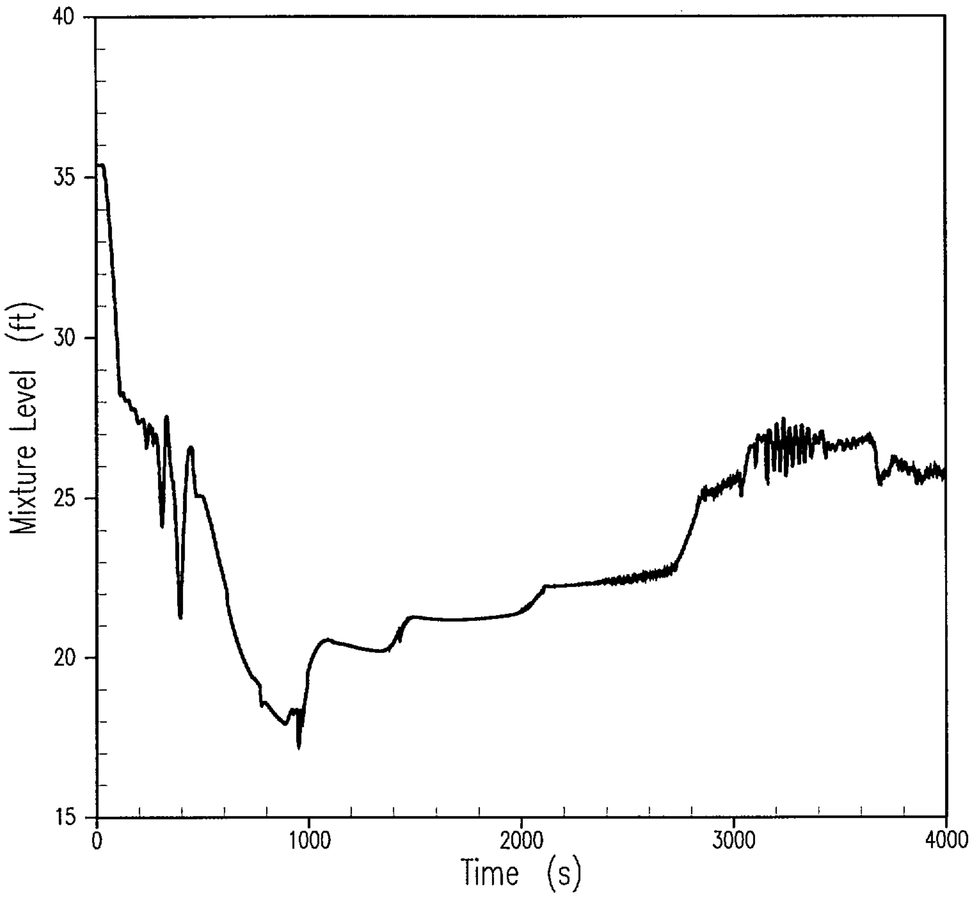

Figure 6.1.1-66

Units 2 Low T_{avg} 4-Inch

Core Mixture Level

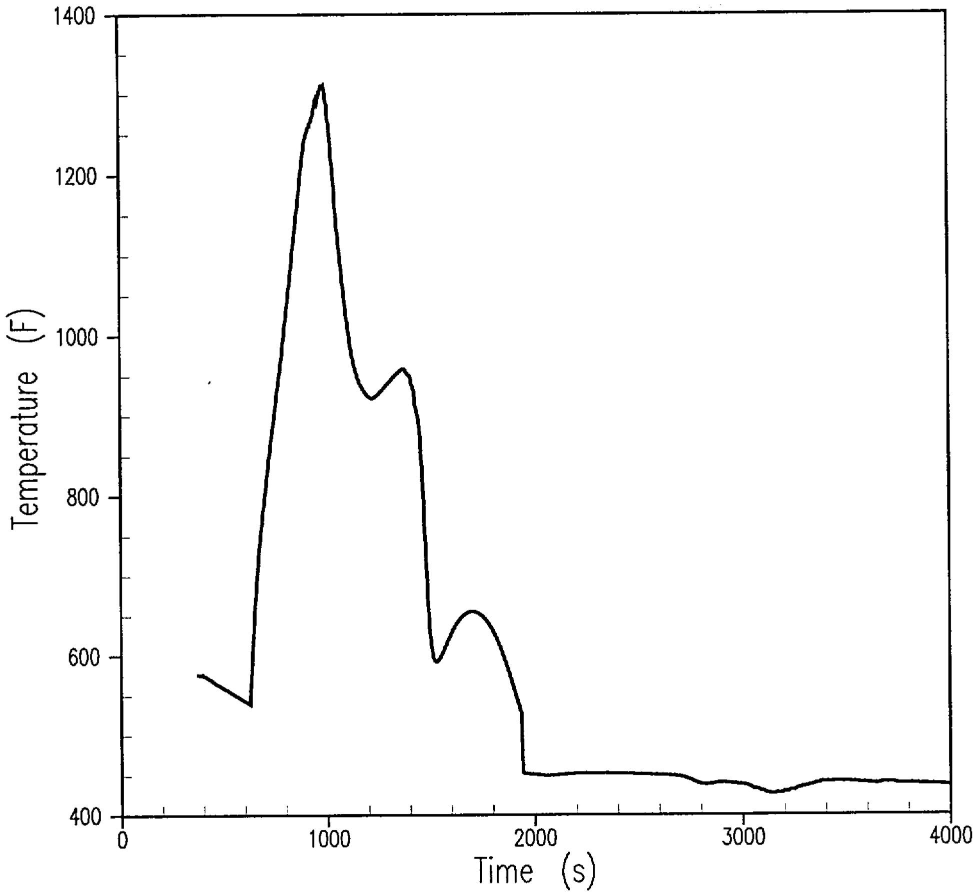

Figure 6.1.1-67

Units 2 Low T_{avg} 4-Inch

Peak Clad Temperature at 11.25 ft.

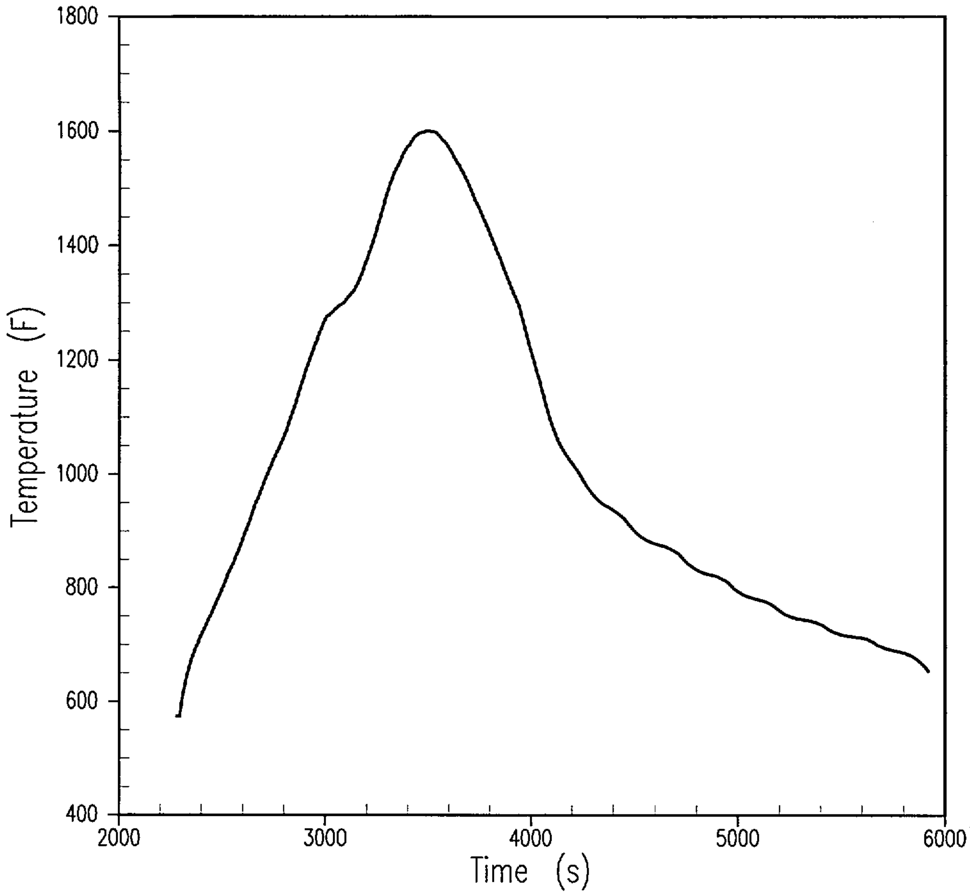

Figure 6.1.1-68

Units 1 Low T_{avg} 2-Inch Zirc-4

Peak Clad Temperature at 11.75 ft.

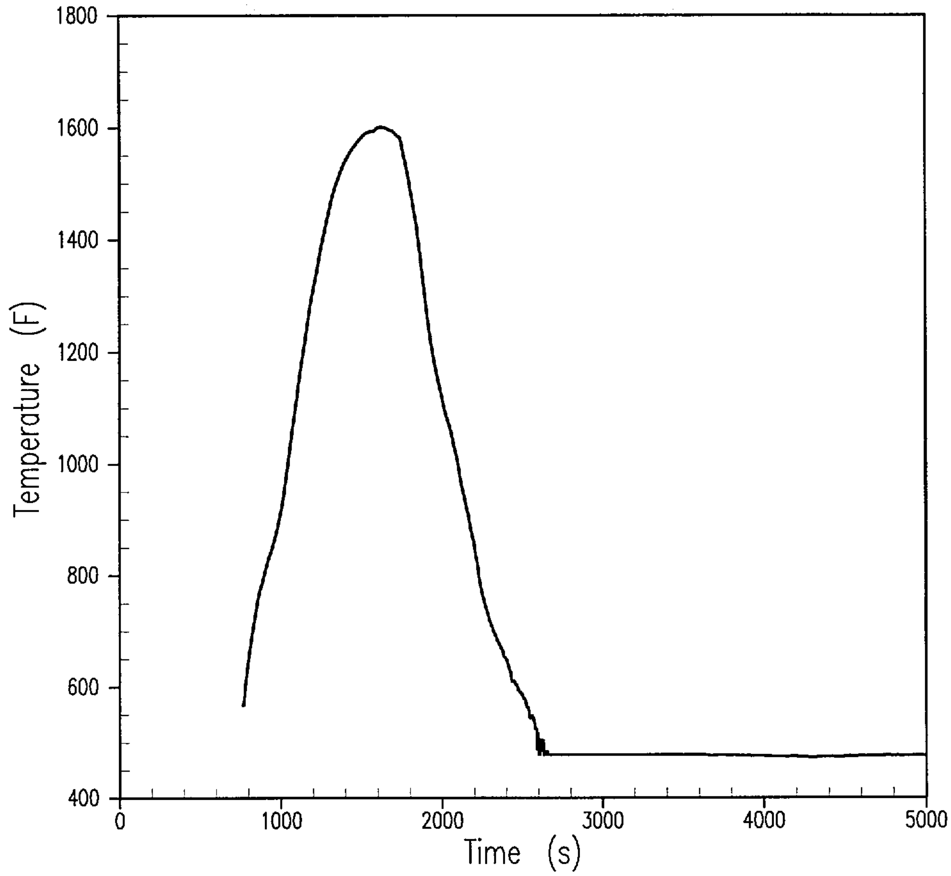

Figure 6.1.1-69

Units 2 High T_{avg} 3-Inch Zirc-4, BU = 6K

Peak Clad Temperature at 11.75 ft.

ATTACHMENT 2

Power Uprate Licensing Report for Byron Station and Braidwood Station

Revised Table 6.5.5-4
"Results for Byron/Braidwood Unit 1 Outside Containment Cases from 102% Power with AFW Failure"

20.

Table 6.5.5-4
Results for Byron/Braidwood Unit 1 Outside Containment Cases from 102% Power with AFW Failure

Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	AFW	102	0.1	1800.0	328.7	330.7	1800.0
B	AFW	102	0.2	1230.0	391.9	395.4	1230.0
C	AFW	102	0.3	806.1	387.1	390.4	807.1
D	AFW	102	0.4	681.5	384.0	387.5	682.5
E	AFW	102	0.5	543.4	383.4	386.4	543.5
F	AFW	102	0.6	450.0	382.8	386.3	450.8
G	AFW	102	0.7	387.6	382.5	385.9	388.9
H	AFW	102	0.8	337.9	381.6	384.7	338.5
I	AFW	102	0.9	301.4	380.3	383.1	301.7
J	AFW	102	1.0	271.7	379.4	383.3	272.7
K	AFW	102	1.1	246.8	379.5	382.6	247.5
L	AFW	102	1.2	16.8	303.3	305.3	10.4
M	AFW	102	1.4	12.8	301.2	303.2	6.9
N	AFW	102	2.0	10.6	301.8	303.8	4.7
O	AFW	102	4.4	9.1	309.3	312.1	9.6