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CPSES-200701238
Log # TXX-07107

Ref. # 10CFR50.46

July 31, 2007

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: **COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)**
DOCKET NOS. 50-445 AND 50-446
SUBMITTAL OF THE CPSES UNITS 1 AND 2 LARGE AND SMALL BREAK LOCA ANALYSES

REFERENCE:
1. Letter logged TXX-07063, dated April 4, 2007 from Mike Blevins to the NRC.
2. Letter logged TXX-07047, dated Feb, 22, 2007 from Mike Blevins to the NRC.

Dear Sir or Madam:

Per Reference 1, TXU Generation Company LP, (Luminant Power) submitted proposed changes to the Comanche Peak Steam Electric Station (CPSES) Unit 1 and Unit 2 Technical Specifications to allow the use of several NRC-approved accident analysis methodologies to be used to establish core operating limits. Included in that submittal were different methods for analyzing the small break loss of coolant accident (LOCA) and the large break LOCA. As prescribed in the NRC's Safety Evaluations approving the generic use of these methods, and in compliance with 10CFR50.46, Luminant Power is herein transmitting the evaluation models and results developed in accordance with those methodologies for NRC review.

In addition, per Reference 2, Luminant Power committed to submit to the NRC, for review, a plant-specific Westinghouse NOTRUMP-based small break LOCA evaluation model for CPSES Unit 1 for application to the beginning of Cycle 14 operation in the Fall of 2008. (Commitment Number 27436). The evaluation models transmitted herein satisfy that commitment.

The evaluation models and results for the large break LOCA analyses for Comanche Peak Units 1 and 2, performed using the ASTRUM Best-Estimate LOCA methodology, as described in Reference 1, are included as Attachments 1 and 2, respectively. The Comanche Peak Units 1 and 2 evaluation models for the NOTRUMP-based small break LOCA analyses are presented in Attachment 3.

A001
NRR

If you have any questions regarding this submittal, please contact Mr. J. D. Seawright at (254) 897-0140.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC,
Its General Partner

Mike Blevins

By: Fred W. Madden
Director, Oversight & Regulatory Affairs

Attachments -
1. APPLICATION OF WESTINGHOUSE BEST-ESTIMATE LARGE BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNIT 1
2. APPLICATION OF WESTINGHOUSE BEST-ESTIMATE LARGE BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNIT 2
3. APPLICATION OF WESTINGHOUSE SMALL BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNITS 1 AND 2

c - B. S. Mallett, Region IV
B. K. Singal, NRR
Resident Inspectors, CPSES

Attachment 1 to TXX-07107

APPLICATION OF WESTINGHOUSE BEST-ESTIMATE LARGE BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNIT 1

APPLICATION OF WESTINGHOUSE BEST-ESTIMATE LARGE BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNIT 1

Background

A best-estimate loss of coolant accident analysis has been completed for Comanche Peak Unit 1. This license amendment request (LAR) for operating license NPF-87 (Comanche Peak Unit 1) requests approval to apply the Westinghouse best-estimate large break loss of coolant accident (LOCA) analysis methodology.

Westinghouse obtained generic NRC approval of its original topical report describing best-estimate large break LOCA methodology in 1996. NRC approval of the methodology is documented in the NRC safety evaluation report appended to the topical report [1]. This methodology was later extended to 2-loop Westinghouse plants with Upper Plenum Injection (UPI) in 1999 as documented in the NRC safety evaluation report appended to the UPI topical report [2].

Westinghouse recently completed a program to revise the statistical approach used to develop the Peak Cladding Temperature (PCT) and oxidation results at the 95^{th} percentile. This method is still based on the Code Qualification Document (CQD) methodology [1],[2] and follows the steps in the Code Scaling Applicability and Uncertainty (CSAU) methodology. However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The Automated Statistical Treatment of Uncertainty Method (ASTRUM) methodology replaces the response surface technique with a statistical sampling method in which the uncertainty parameters are simultaneously sampled for each case. The approved ASTRUM evaluation model is documented in WCAP-16009-P-A [3].

This report summarizes the application of the Westinghouse ASTRUM BELOCA evaluation model to Comanche Peak Unit 1 for the large break LOCA accident analysis. Table 1 lists the major plant parameter assumptions used in the BELOCA analysis for Comanche Peak Unit 1. Both Luminant Power and its analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses. These interface processes, along with vendor internal processes for assessing evaluation model changes and errors, are used to identify the need for LOCA analyses impact assessments.

Method of Thermal Analysis for Comanche Peak Unit 1

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in 10 CFR 50.46 [4], both the Nuclear Regulatory Commission (NRC) and the industry recognized that stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling System (ECCS) would be conservatively underestimated, resulting in predicted peak clad temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1. Identify, through separate effects and integral effects experiments, the degree of conservatism in those models required in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.

2. Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without a loss in safety to the public. It was also found that some plants were being restricted in operating flexibility by overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 [5]. The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models", to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best estimate codes is provided in Regulatory Guide 1.157 [6].

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology [7]. This method outlined an approach for defining and qualifying a best estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three and four loop pressurized water reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and has been approved by the NRC [1]. This methodology was later extended to 2-loop Westinghouse plants with Upper Plenum Injection (UPI) in 1999 as documented in the NRC safety evaluation report appended to the UPI topical report [2].

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical TReatment of Uncertainty Method [3]. This method is still based on the CQD methodology and follows the steps in the CSAU methodology. However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method in which the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in WCAP-16009-P-A [3]. The ASTRUM methodology remains applicable to three and four loop PWRs, as well as 2-loop Westinghouse plants with UPI. This methodology was also extended to CE design PWRs.

The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95th percentile of the Peak Clad Temperature (PCT), Local Maximum Oxidation (LMO), and Core Wide Oxidation (CWO) with 95% confidence level. These parameters are needed to satisfy the 10 CFR 50.46 criteria with regard to PCT, LMO, and CWO.

Downcomer boiling is considered as appropriate in the ASTRUM methodology. The WCOBRA/TRAC computer code determines if downcomer boiling will occur for a particular transient. If downcomer boiling is determined to occur in a transient, WCOBRA/TRAC includes the effects of downcomer boiling in the transient calculation.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A [3] as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A [3] was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report appended to this WCAP. The Best Estimate LBLOCA analysis and associated model for Comanche Peak Unit 1 is unit-specific.

Description of a Large Break LOCA Transient

Before the break occurs, the RCS (Reactor Coolant System) is assumed to be operating normally at full power in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. A large break is assumed to open instantaneously in one of the main RCS cold leg pipes.

Immediately following the cold leg break, a rapid system depressurization occurs along with a core flow reversal due to a high discharge of sub-cooled fluid into the broken cold leg and out of the break. The fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up, while the core power decreases due to voiding in the core. The hot water in the core, upper plenum, and upper head flashes to steam, and subsequently the cooler water in the lower plenum and downcomer begins to flash. Once the system has depressurized to the accumulator pressure, the accumulator begins to inject cold borated water into the intact cold legs. During the blowdown period, a portion of the injected Emergency Core Cooling System (ECCS) water is calculated to be bypassed around the downcomer and out of the break. The bypass period ends as the system pressure continues to decrease and approaches the containment pressure, resulting in reduced break flow and consequently, reduced core flow.

As the refill period begins, the core continues to heat up as the vessel begins to fill with ECCS water. This phase continues until the lower plenum is filled, the bottom of the core begins to reflood, and entrainment begins.

During the reflood period, the core flow is oscillatory as ECCS water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. The pumped upper plenum and cold leg injection ECCS water aids in the filling of the vessel and downcomer, which subsequently supplies water to maintain the core and downcomer water levels and complete the reflood period.

ASTRUM Analysis Results for Comanche Peak Unit 1

The results of the Comanche Peak Unit 1 ASTRUM analysis are summarized in Table 2. Table 3 contains a sequence of events for the limiting PCT transient.

The scatter plot presented in Figure 1 shows the effect of the effective break area on the analysis PCT. The effective break area is calculated by multiplying the discharge coefficient CD with the sample value of the break area, normalized to the cold-leg cross sectional area. Figure 1 is provided to show the break area is a significant contributor to the variation in PCT.

From the 124 calculations performed as part of the ASTRUM analysis, different cases proved to be the limiting PCT and limiting LMO transient for Comanche Peak Unit 1. Figure 2 shows the predicted clad temperature transient at the PCT limiting elevation for the limiting PCT case. Figure 3 presents the clad temperature transient predicted at the LMO elevation for the limiting LMO case. Due the low PCT results, CWO remained on the order of 0 percent for all cases.

Figure 4 through 17 illustrate the key major response parameters for the limiting PCT transient. The reference point for the lower plenum liquid level presented in Figure 11 is the bottom of the vessel (10.1 feet below the bottom of the active fuel). The reference point for the downcomer liquid level presented in Figure 12 is the point at which the outside of the core barrel, if extended downward, intersects with the vessel wall (6.1 feet below the bottom of the active fuel). The reference point for the core collapsed liquid levels presented in Figures 13 and 16 is the bottom of the active fuel.

The containment backpressure utilized for the LBLOCA analysis compared to the calculated containment backpressure is provided in Figure 18. The worst single failure for the LBLOCA analysis is the loss of one train of ECCS injection (consistent with the ASTRUM Topical);

however, all containment systems which would reduce containment pressure are modeled for the LBLOCA containment backpressure calculation.

10 CFR 50.46 Requirements

It must be demonstrated that there is a high level of probability that the following limits set forth in 10 CFR 50.46 are met:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 1492°F for Comanche Peak Unit 1, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F", is demonstrated. The result is shown in Table 2.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. Since the resulting LMO for the limiting case is 0.23 percent for Comanche Peak Unit 1, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Local Maximum Oxidation of the cladding less than 17 percent", is demonstrated. The result is shown in Table 2.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. While the limiting LMO is determined based on the single Hot Rod, the CWO value can be conservatively chosen as that calculated for the limiting Hot Assembly Rod (HAR) when there is significant margin to the regulatory limit. The limiting HAR total maximum oxidation is 0 percent for Comanche Peak Unit 1. Thus, a detailed CWO calculation is not needed because the calculations would include many lower power assemblies and the outcome would always be less than the limiting HAR total maximum oxidation. Therefore, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent", is demonstrated. The result is shown in Table 2.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The grid crush calculations currently in place for Comanche Peak Unit 1 remain unchanged with the application of the ASTRUM methodology [3], therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at Comanche Peak Unit 1 to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology [3].

Based on the ASTRUM analysis results (see Table 2), it is concluded that Comanche Peak Unit 1 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

References

[1] Bajorek, S. M., et. al., 1998, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1, and WCAP-14747 (Non-Proprietary).

[2] Dederer, S. I., et. al., 1999, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," WCAP-14449-P-A, Revision 1 and WCAP-14450, Revision 1 (Non-Proprietary).

[3] Nissley, M. E., et.al., 2005, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A and WCAP-16009-NP-A (Non-Proprietary).

[4] "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 4, 1974.

[5] Information Report from W.J. Dircks to the Commissioners, "Emergency Core Cooling System Analysis Methods," SECY-83-472, November 17, 1983.

[6] "Best Estimate Calculations of Emergency Core Cooling System Performance," Regulatory Guide 1.157, USNRC, May 1989.

[7] Boyack, B., et. al., 1989, "Quantifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident," NUREG/CR-5249.

Table 1 - Major Plant Parameter Assumptions Used in the BELOCA Analysis for Comanche Peak Unit 1

Parameter	Value
Plant Physical Description	
• SG Tube Plugging	≤ 10%
Plant Initial Operating Conditions	
• Reactor Power	≤ 3612 MWt (± 0.6% uncertainties)
• Peaking Factors	$F_Q \leq 2.5$ $F_{\Delta H} \leq 1.60$
• Axial Power Distribution	See Figure 19
Fluid Conditions	
• T_{AVG}	574.2 – 6.5 °F ≤ T_{AVG} ≤ 589.2 + 6.5 °F
• Pressurizer Pressure	2250 – 30 psia ≤ P_{RCS} ≤ 2250 + 30 psia
• Reactor Coolant Flow	≥ 95,700 gpm
• Accumulator Temperature	88 °F ≤ T_{ACC} ≤ 120 °F
• Accumulator Pressure	603 psia ≤ P_{ACC} ≤ 693 psia
• Accumulator Water Volume	6119 gal ≤ V_{ACC} ≤ 6597 gal
• Accumulator Boron Concentration	≥ 2300 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection Flow	Minimum
• Safety Injection Temperature	40 °F ≤ T_{SI} ≤ 120 °F
• Safety Injection Initiation Delay Time	≤ 17 sec (with offsite power) ≤ 27 sec (without offsite power)
• Containment Pressure	Bounded (minimum)

Table 2 – Comanche Peak Unit 1 Best Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	1492	< 2200
95/95 LMO[(1)] (%)	0.23	< 17
95/95 CWO[(2)] (%)	0.00	< 1

Notes:
1. Local Maximum Oxidation
2. Core Wide Oxidation

Table 3 – Comanche Peak Unit 1 Best Estimate Large Break Sequence of Events for the Limiting PCT Case

Event	Time (sec)
Start of Transient	0.0
Safety Injection Signal	5.8
PCT Occurs	~10
Accumulator Injection Begins	12.0
End of Blowdown	25.5
Bottom of Core Recovery	30.0
Accumulator Empty	30.5
Safety Injection Begins	32.8
End of Transient	550.0

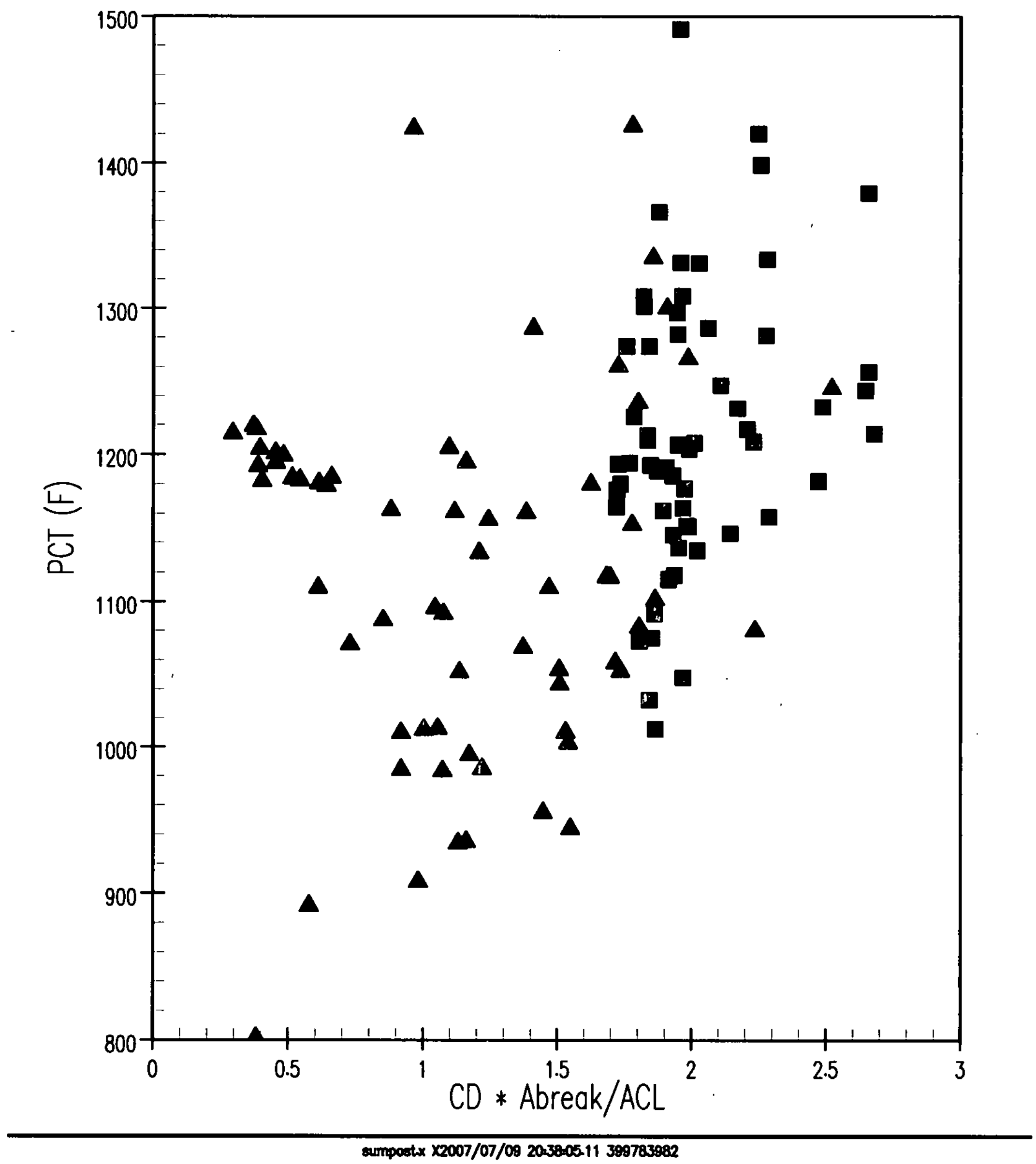

Figure 1 – Comanche Peak Unit 1 HOTSPOT PCT versus Effective Break Area Scatter Plot (CD = Discharge Coefficient, Abreak = Break Area, ACL = Cold Leg Area)

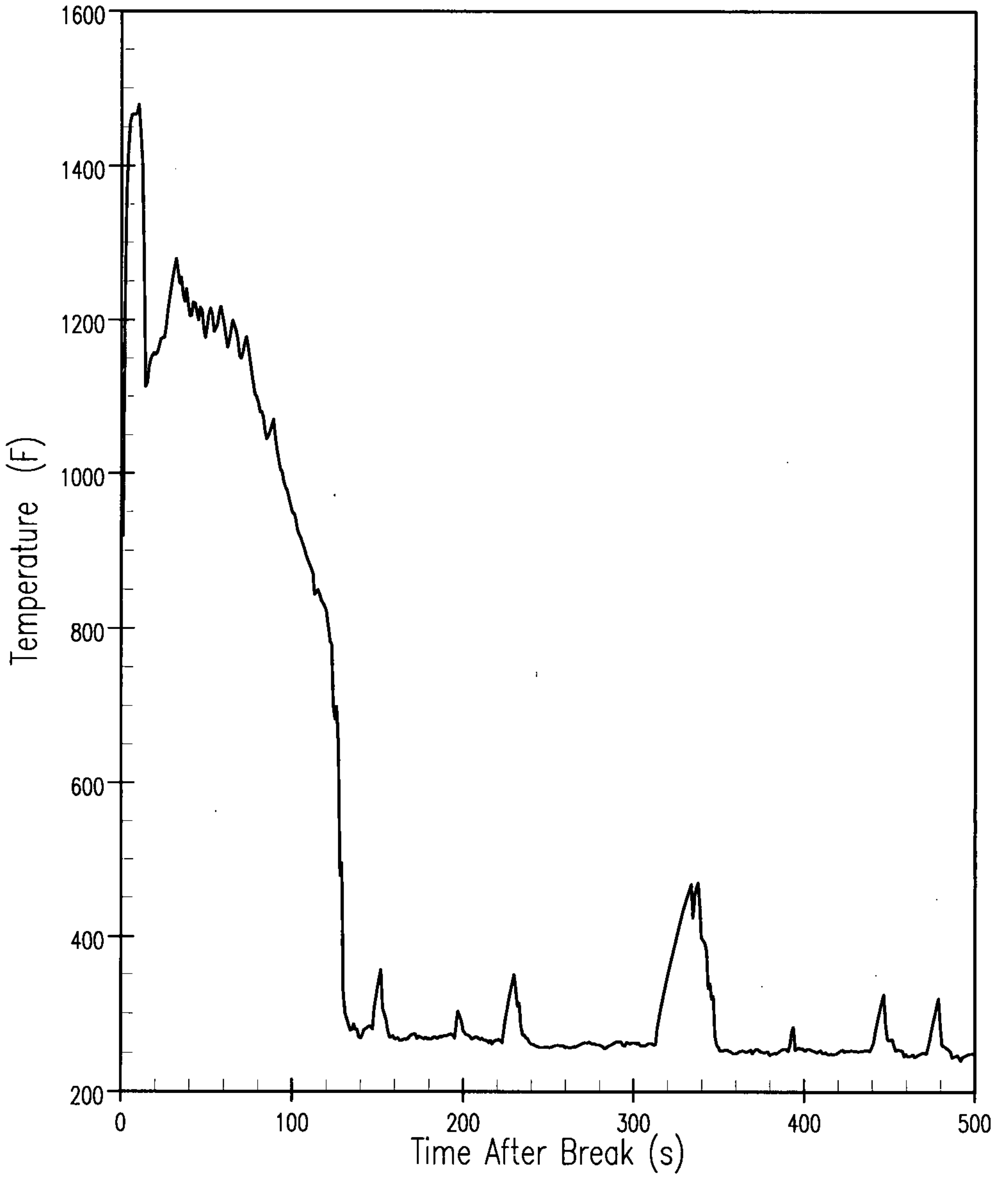

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Figure 2 – Comanche Peak Unit 1 HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case

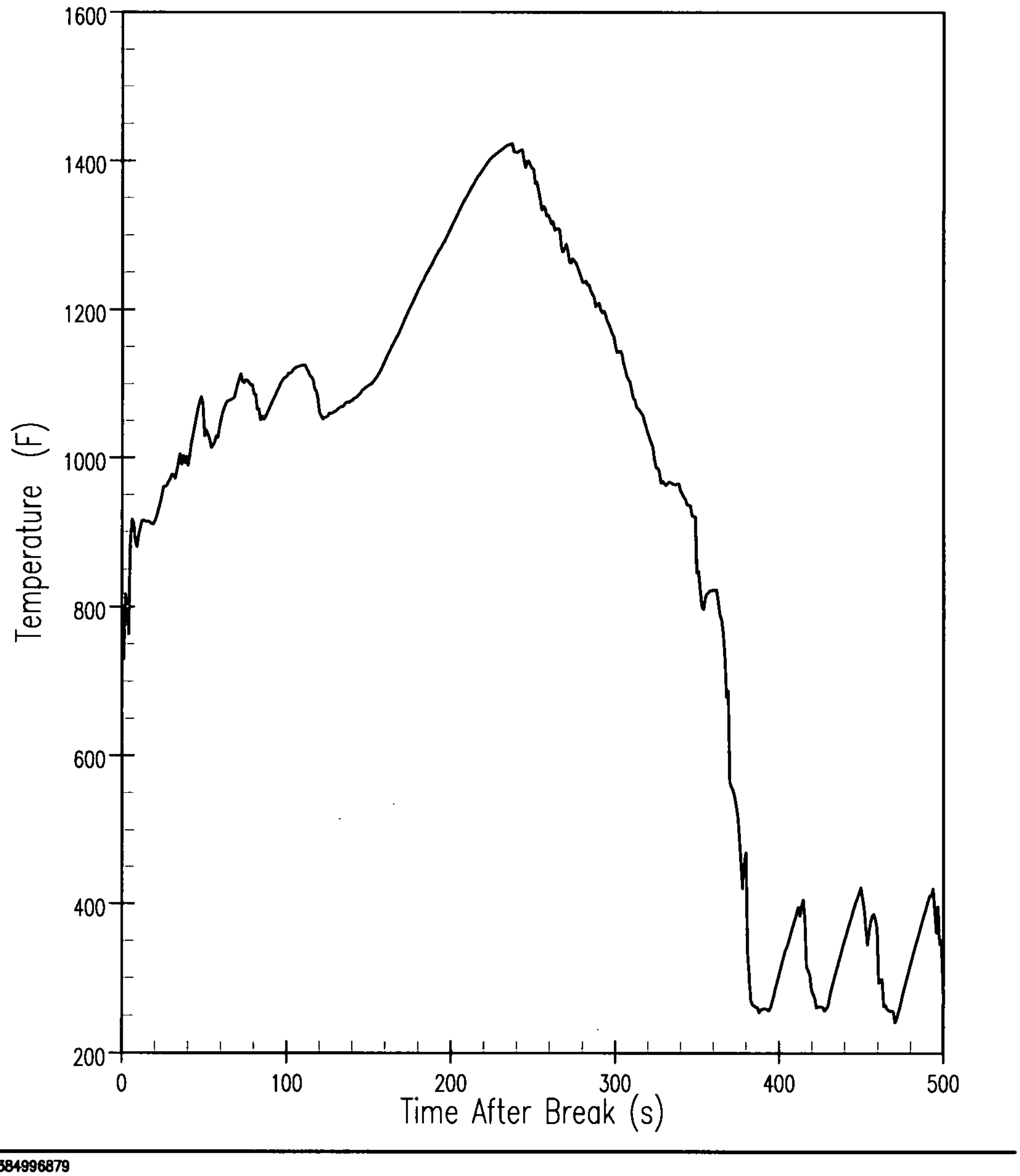

Figure 3 – Comanche Peak Unit 1 HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting LMO Case

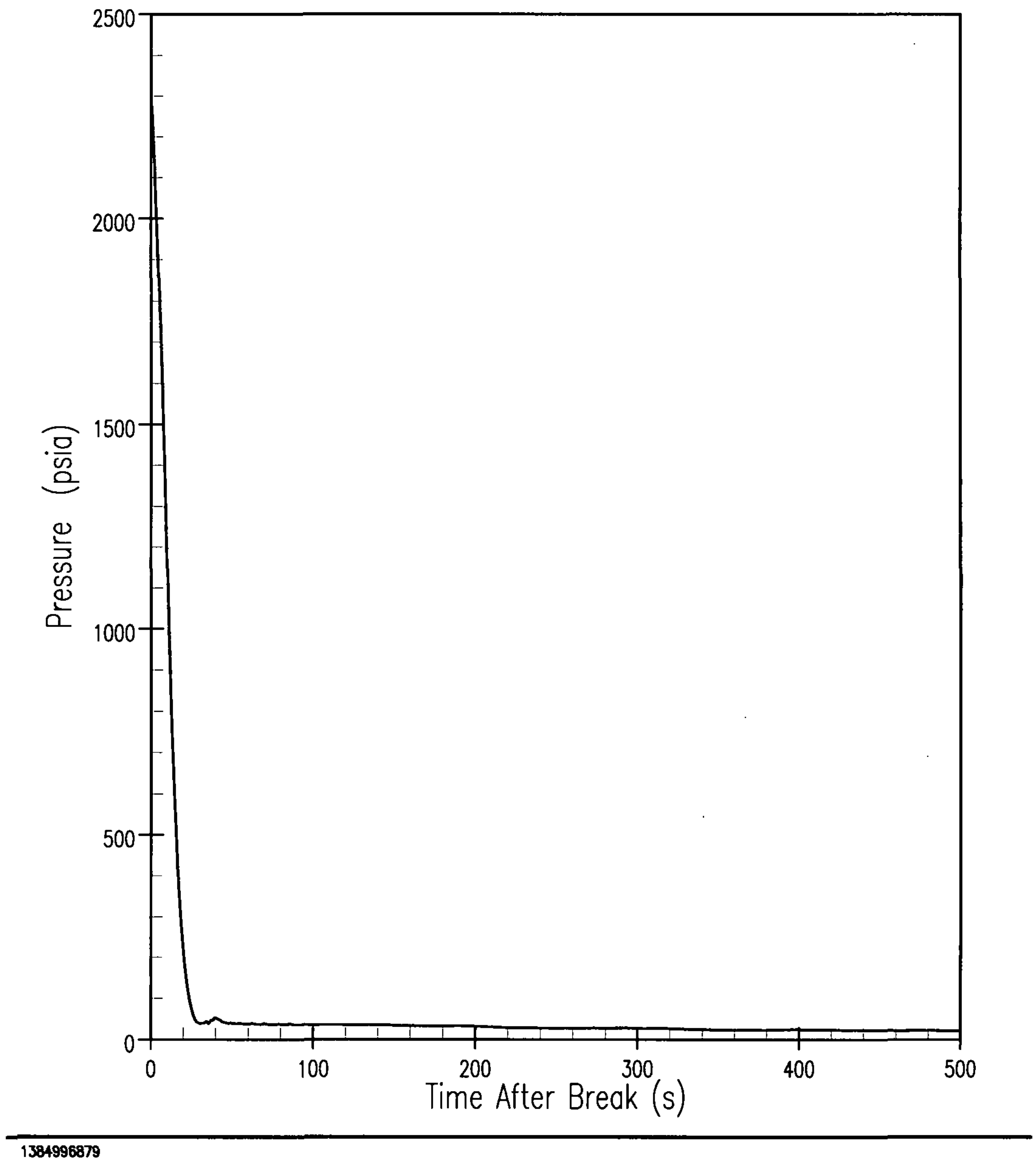

Figure 4 – Comanche Peak Unit 1 Pressurizer Pressure for the Limiting PCT Case

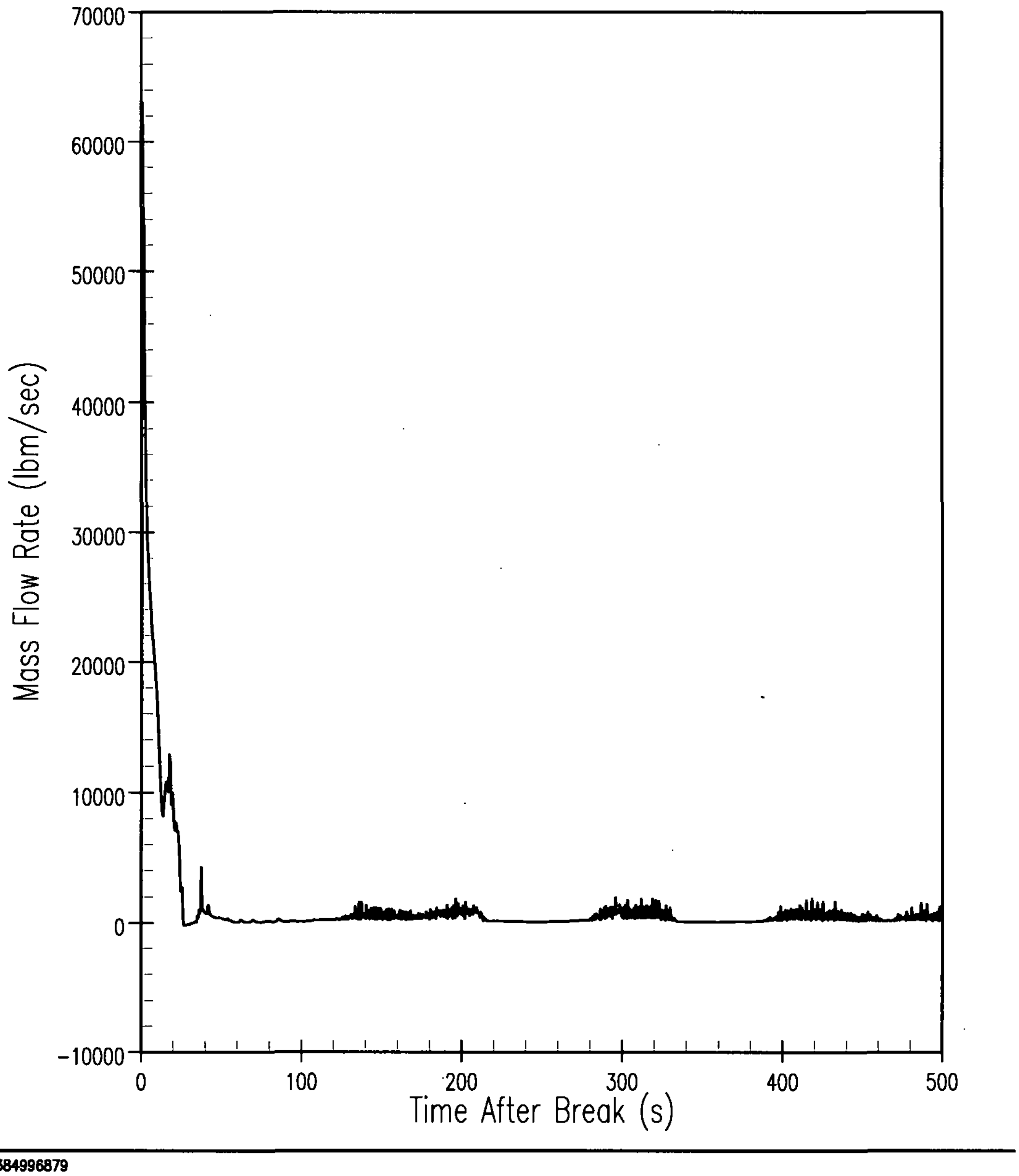

Figure 5 – Comanche Peak Unit 1 Vessel Side Break Flow for the Limiting PCT Case

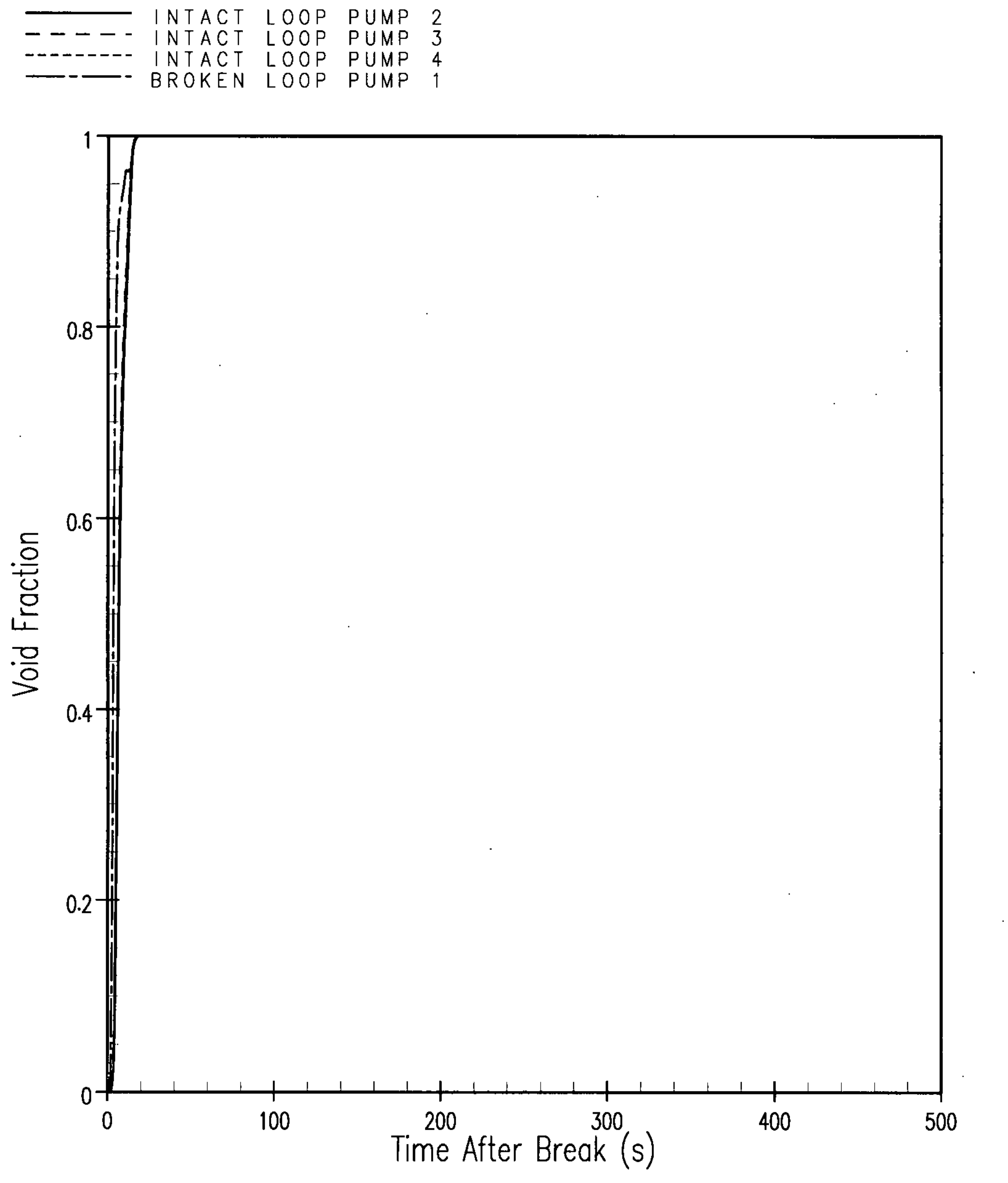

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Figure 6 – Comanche Peak Unit 1 Void Fraction in Pumps for the Limiting PCT Case

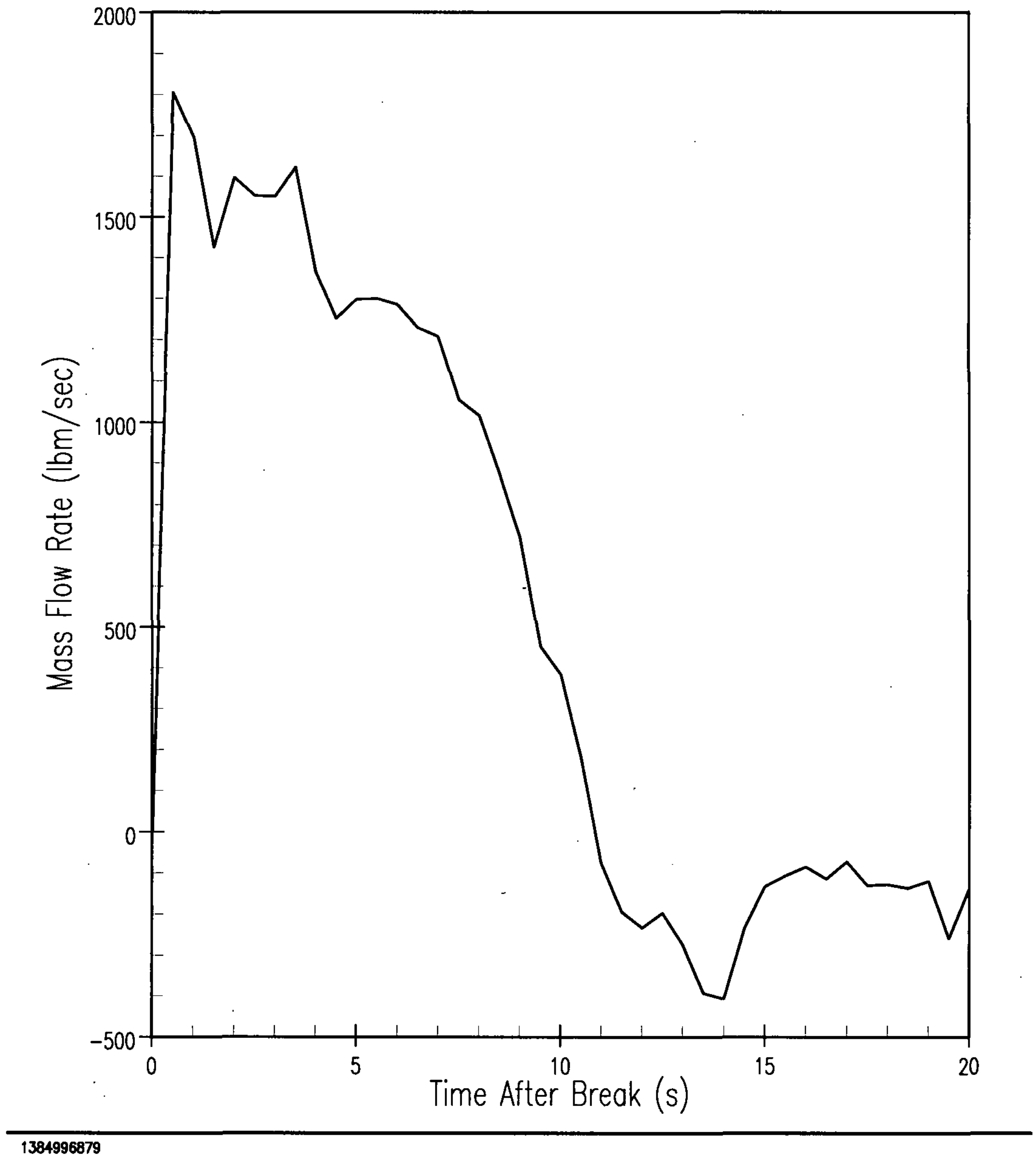

Figure 7 – Comanche Peak Unit 1 Vapor Flow at Top of Core for the Limiting PCT Case

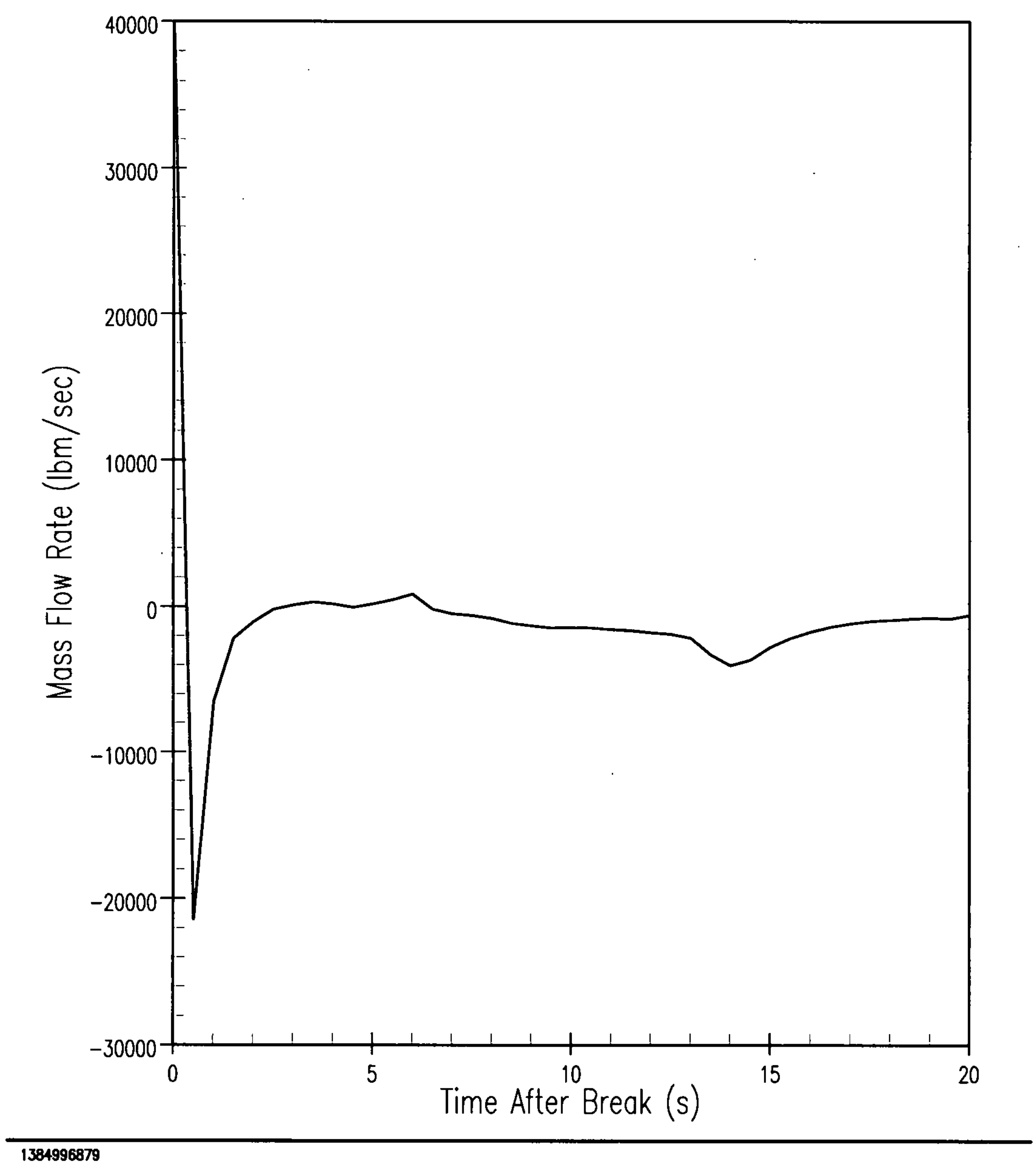

Figure 8 – Comanche Peak Unit 1 Total Flow at Bottom of Core for the Limiting PCT Case

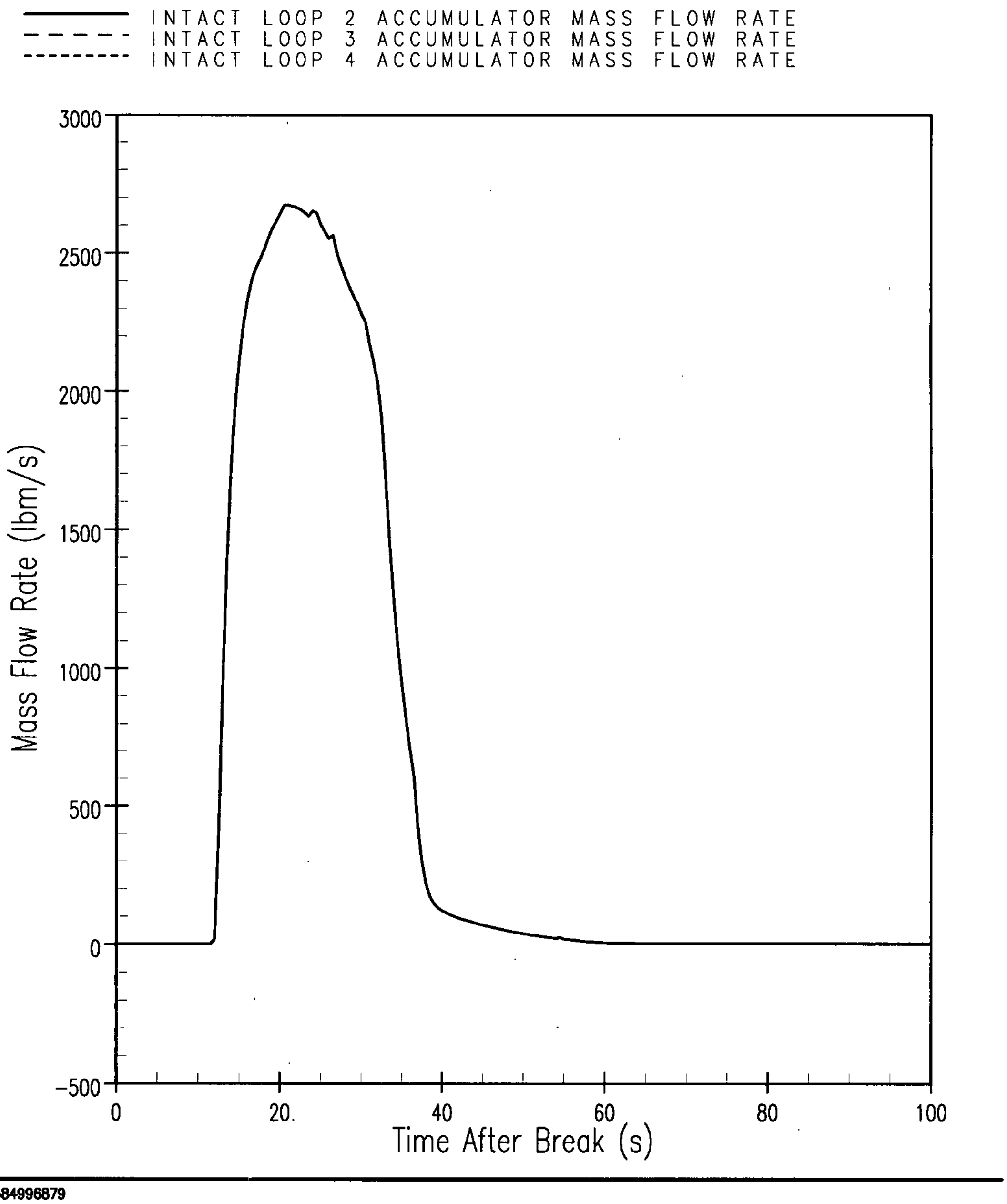

Figure 9 – Comanche Peak Unit 1 Accumulator Injection Flow for the Limiting PCT Case

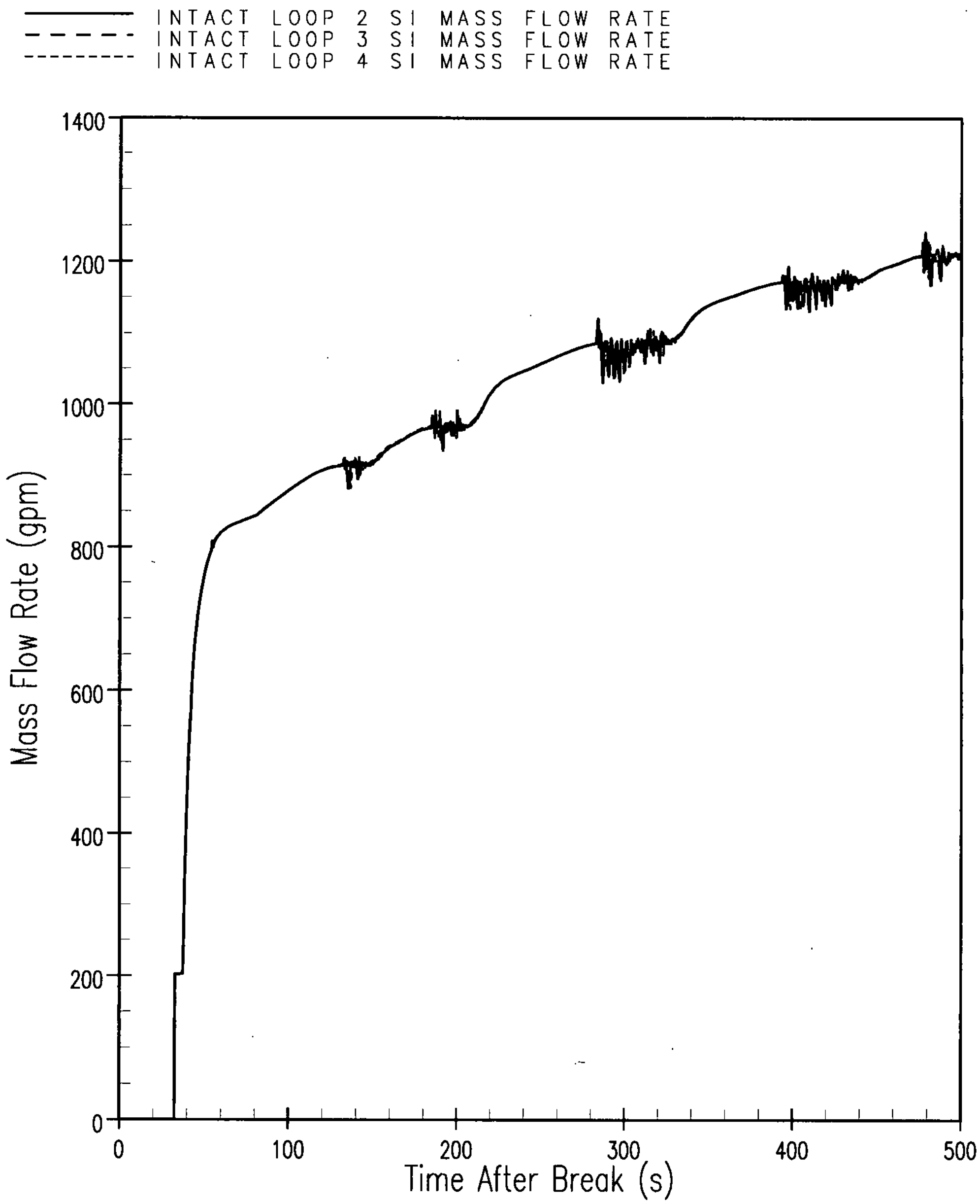

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Figure 10 – Comanche Peak Unit 1 Safety Injection Flow for the Limiting PCT Case

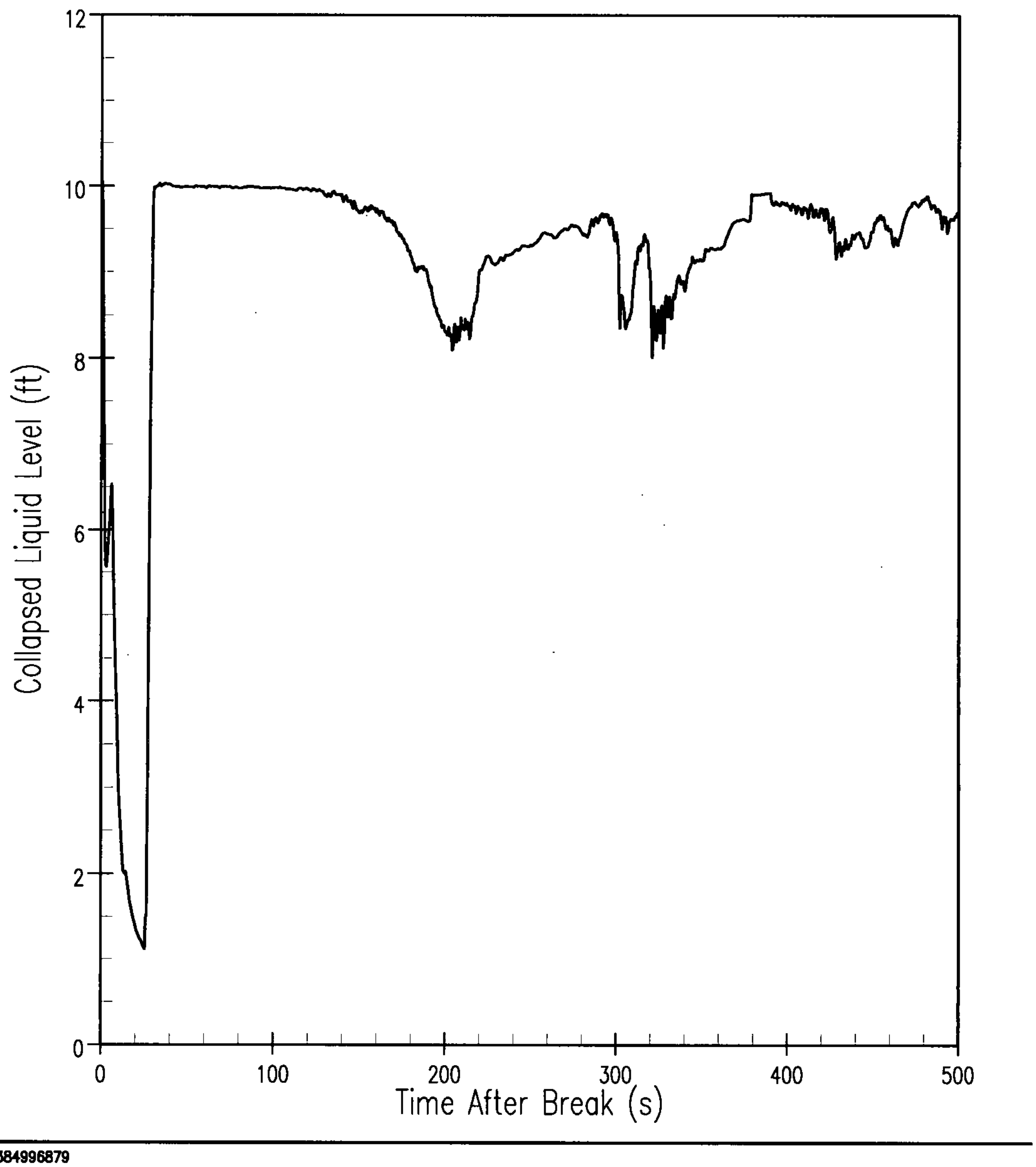

Figure 11 – Comanche Peak Unit 1 Lower Plenum Collapsed Liquid Level for the Limiting PCT Case

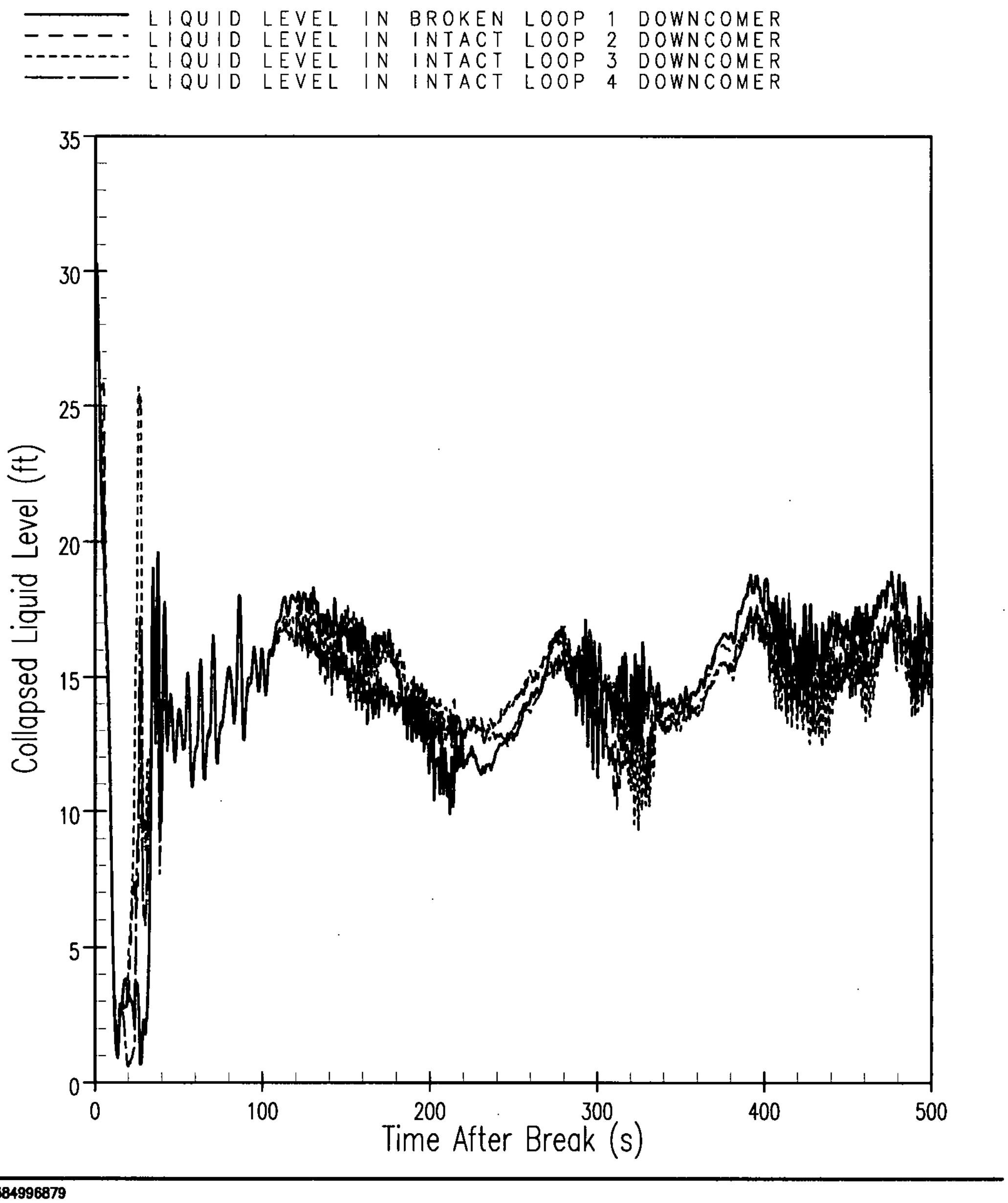

Figure 12 – Comanche Peak Unit 1 Downcomer Collapsed Liquid Levels for the Limiting PCT Case

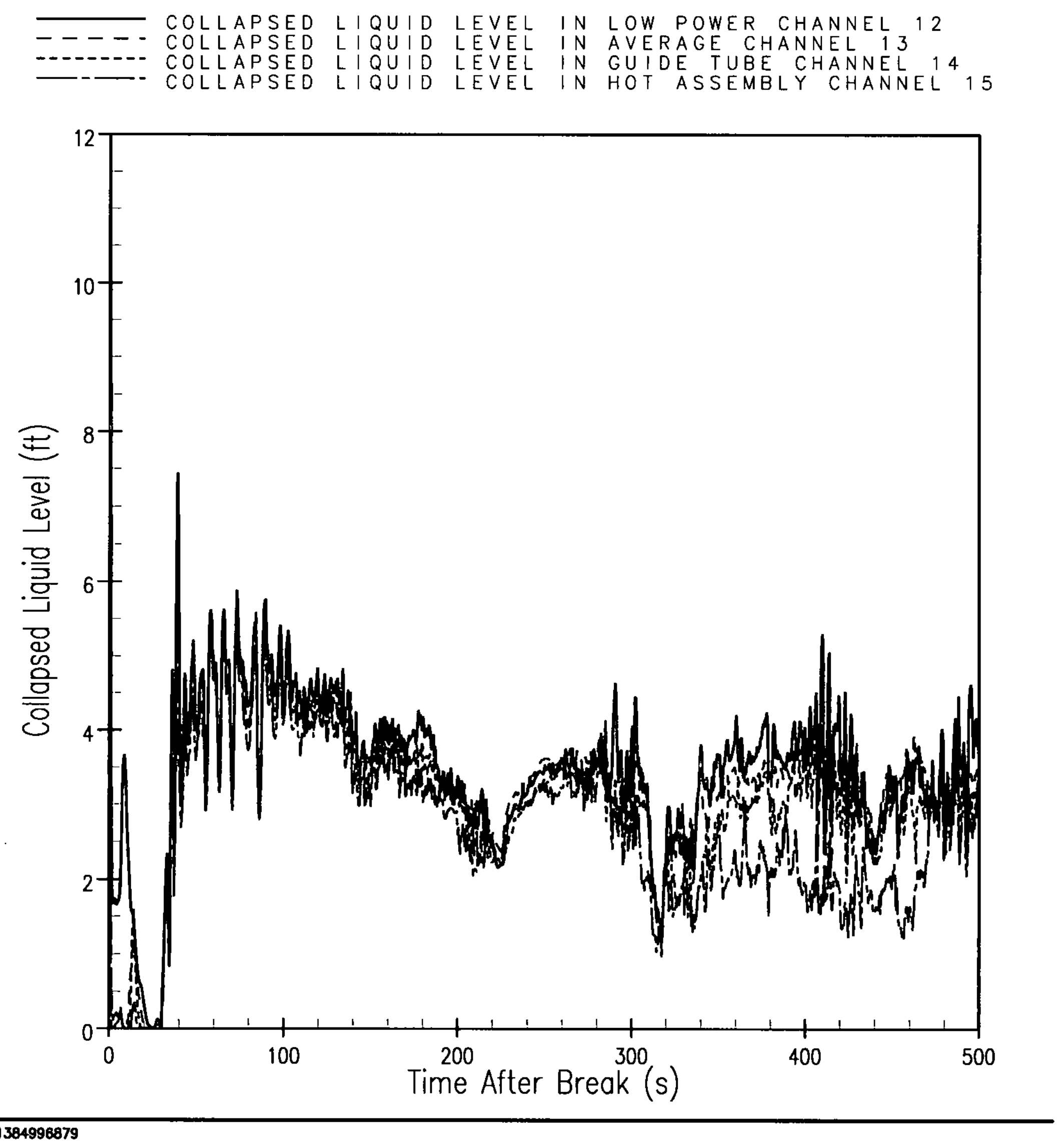

Figure 13 – Comanche Peak Unit 1 Core Collapsed Liquid Levels for the Limiting PCT Case

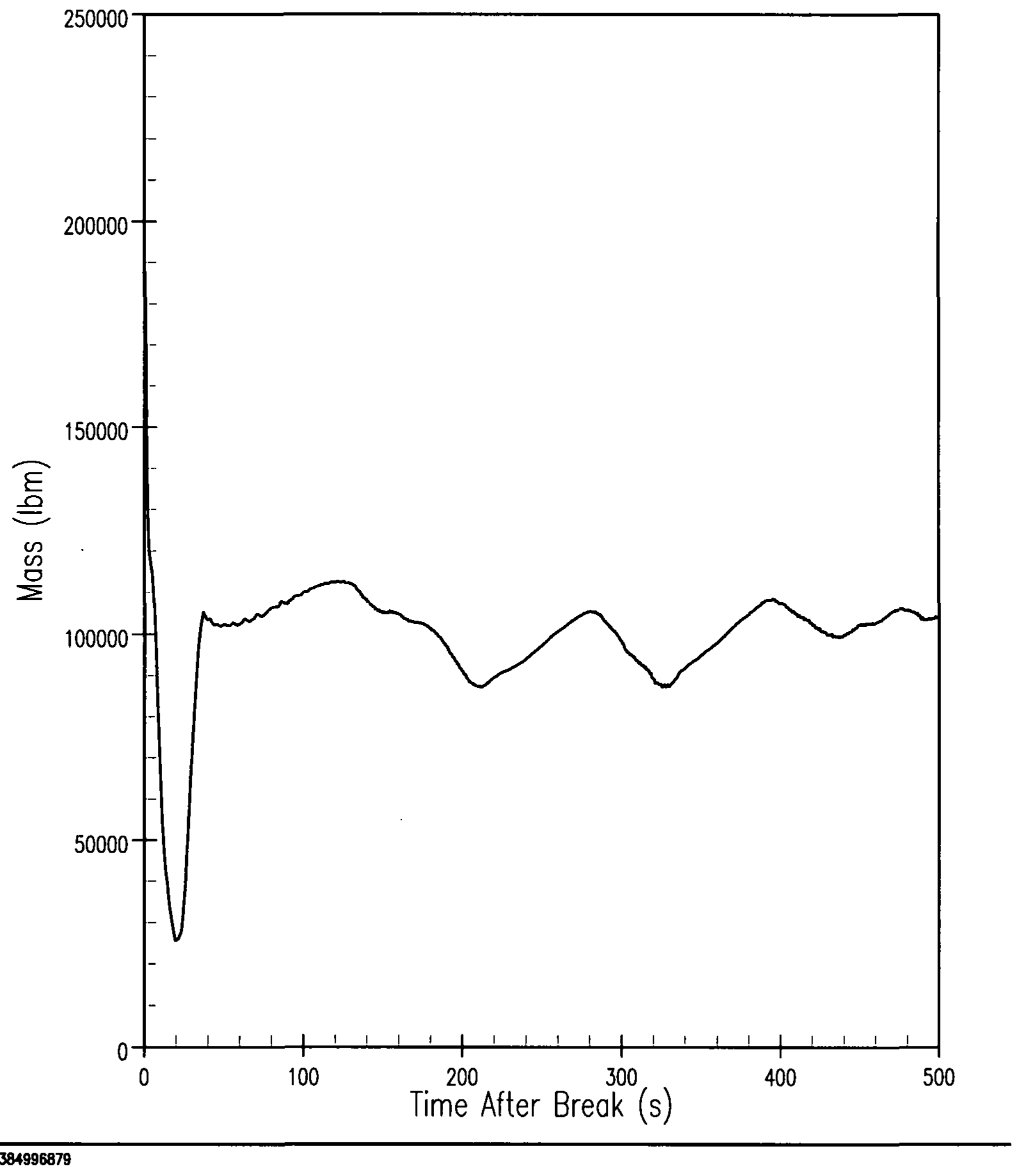

Figure 14 – Comanche Peak Unit 1 Vessel Fluid Mass for the Limiting PCT Case

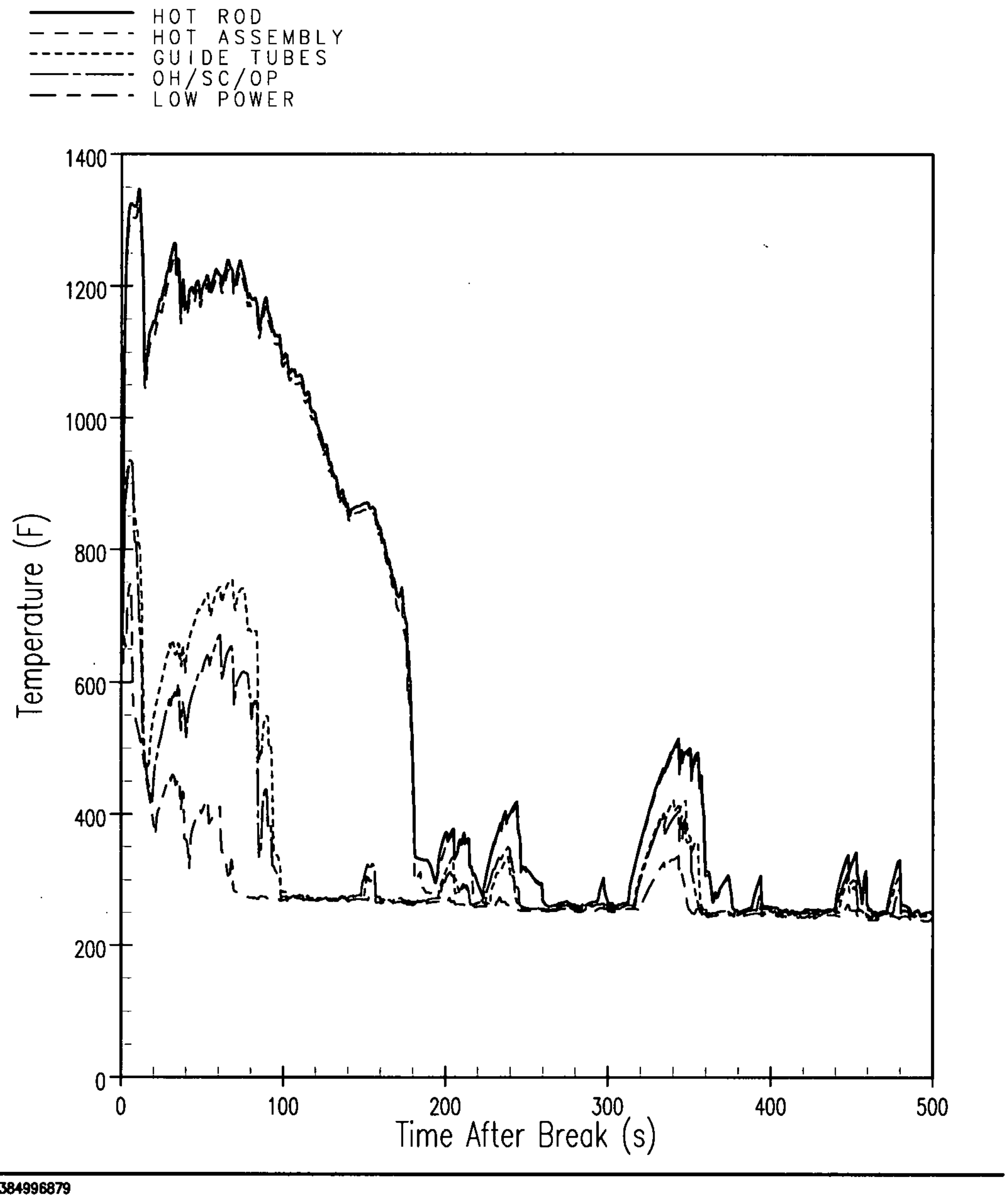

Figure 15 – Comanche Peak Unit 1 WCOBRA/TRAC Peak Clad Temperature for all 5 Rod Groups for the Limiting PCT Case

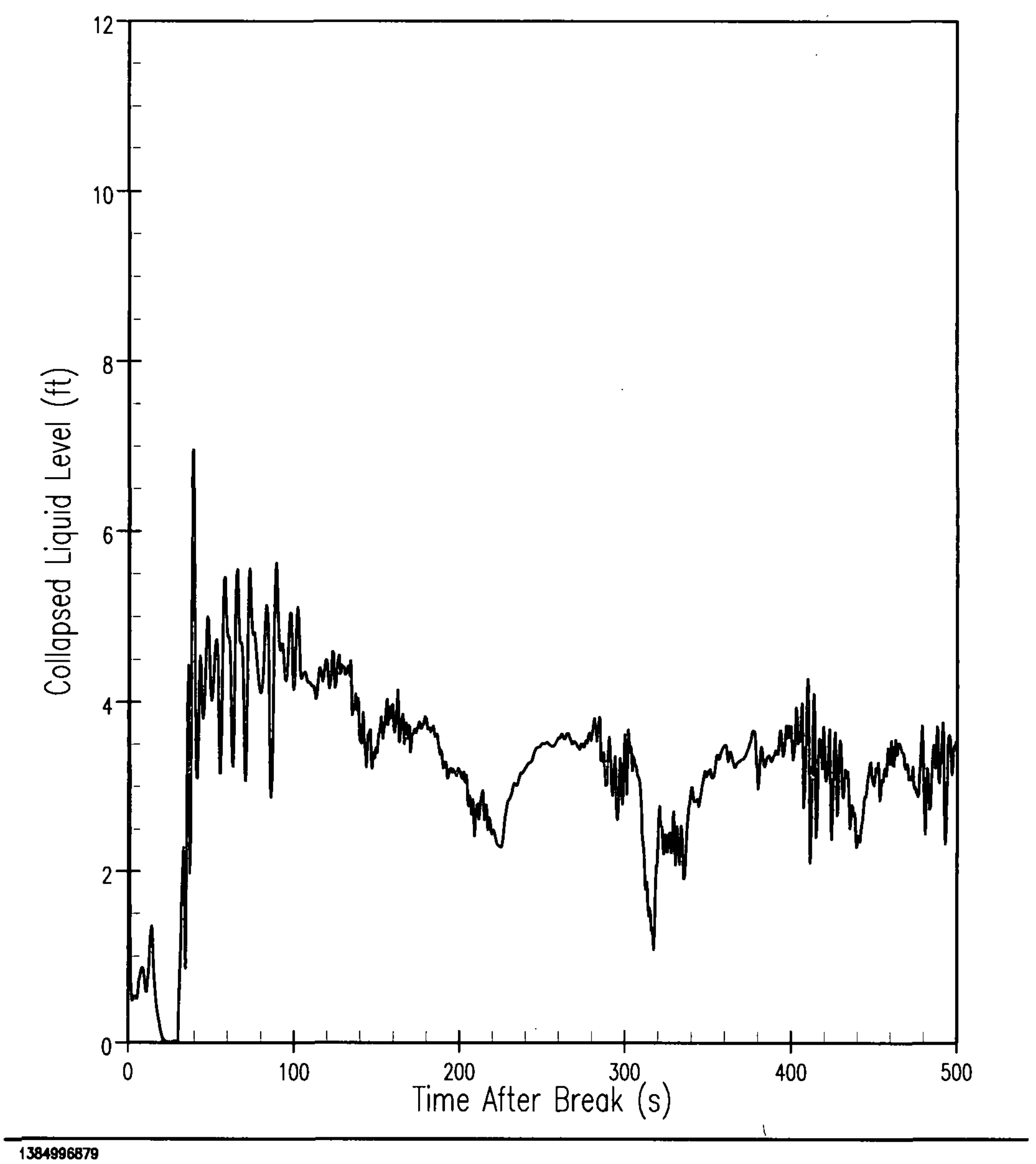

Figure 16 – Comanche Peak Unit 1 Average Core Collapsed Liquid Level per Assembly for the Limiting PCT Case

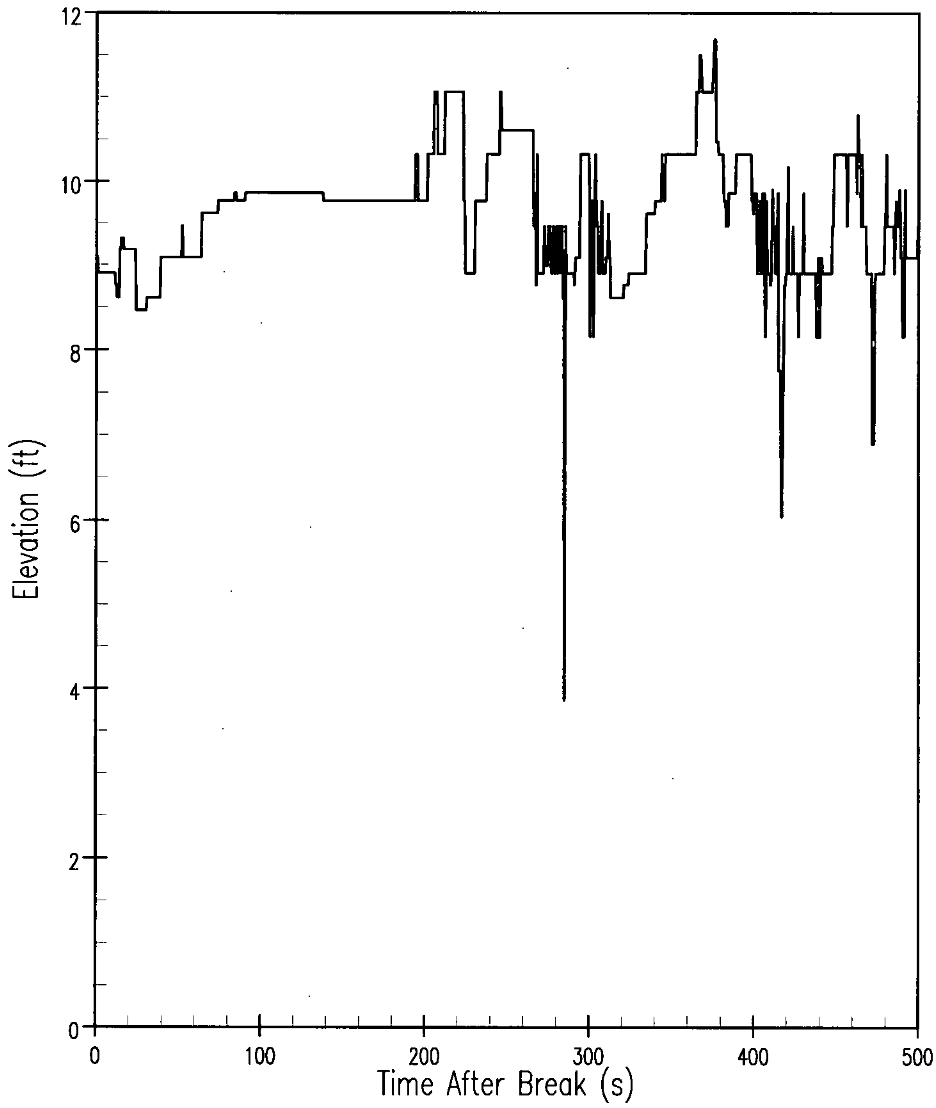

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Figure 17 – Comanche Peak Unit 1 Peak Clad Temperature Elevation for the Hot Rod for the Limiting PCT Case

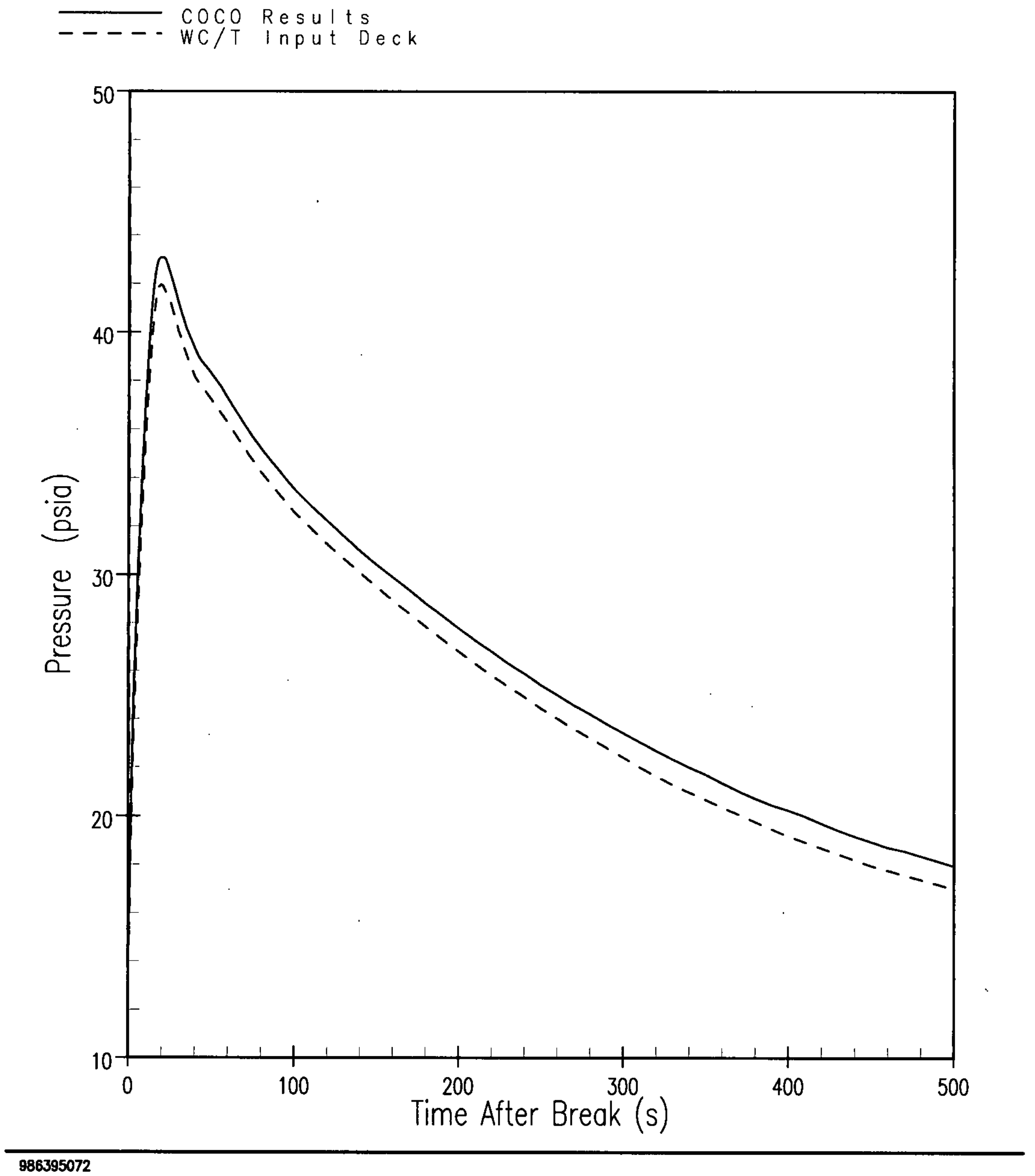

Figure 18 – Comanche Peak Unit 1 Analysis Versus Calculated Containment Backpressure

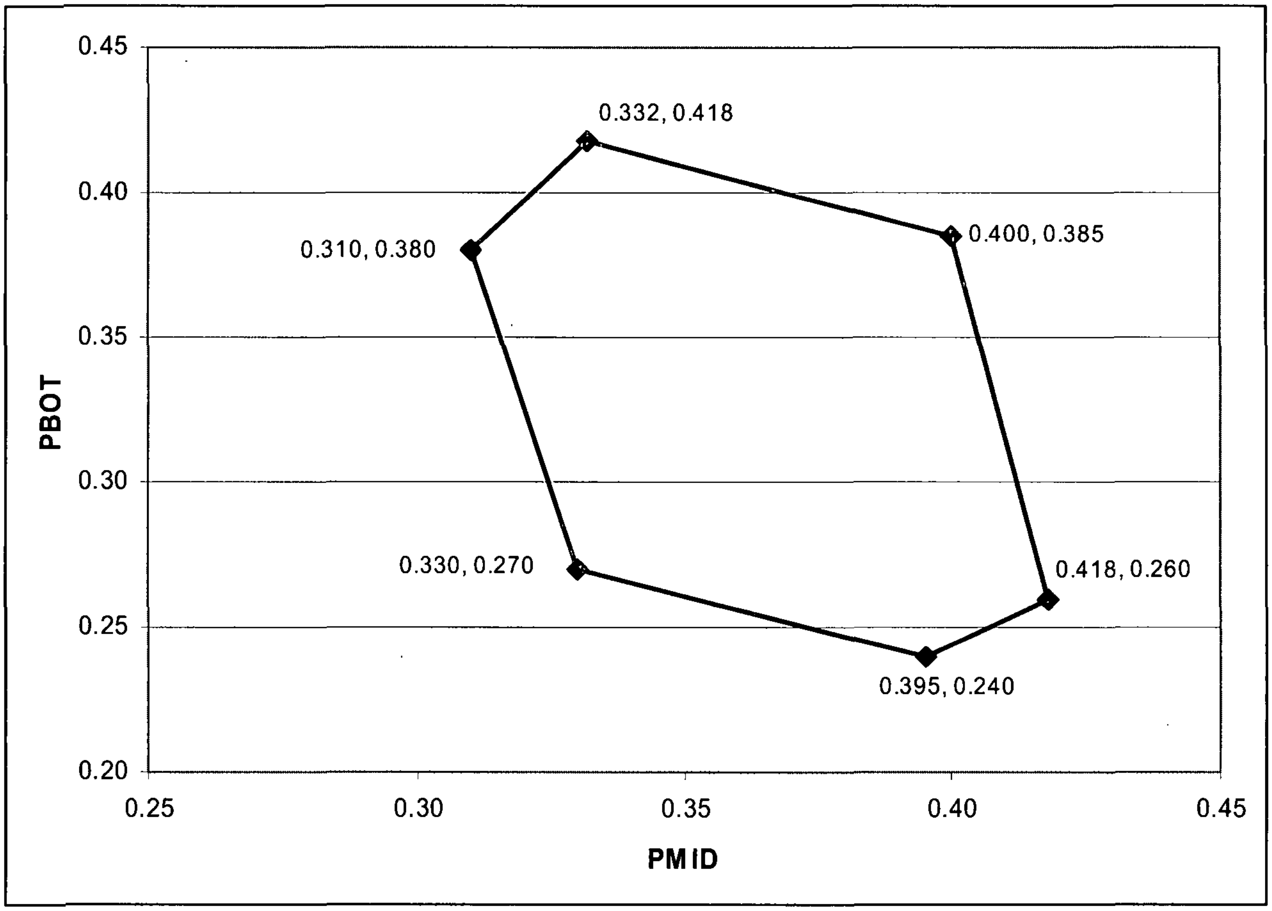

Figure 19 – Comanche Peak Unit 1 BELOCA Analysis Axial Power Shape Operating Space Envelope

PBOT = integrated power fraction in the bottom third of the core
PMID = integrated power fraction in the middle third of the core

Attachment 2 to TXX-07107

APPLICATION OF WESTINGHOUSE BEST-ESTIMATE LARGE BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNIT 2

APPLICATION OF WESTINGHOUSE BEST-ESTIMATE LARGE BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNIT 2

Background

A best-estimate loss of coolant accident analysis has been completed for Comanche Peak Unit 2. This license amendment request (LAR) for operating license NPF-89 (Comanche Peak Unit 2) requests approval to apply the Westinghouse best-estimate large break loss of coolant accident (LOCA) analysis methodology.

Westinghouse obtained generic NRC approval of its original topical report describing best-estimate large break LOCA methodology in 1996. NRC approval of the methodology is documented in the NRC safety evaluation report appended to the topical report [1]. This methodology was later extended to 2-loop Westinghouse plants with Upper Plenum Injection (UPI) in 1999 as documented in the NRC safety evaluation report appended to the UPI topical report [2].

Westinghouse recently completed a program to revise the statistical approach used to develop the Peak Cladding Temperature (PCT) and oxidation results at the 95^{th} percentile. This method is still based on the Code Qualification Document (CQD) methodology [1],[2] and follows the steps in the Code Scaling Applicability and Uncertainty (CSAU) methodology. However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The Automated Statistical Treatment of Uncertainty Method (ASTRUM) methodology replaces the response surface technique with a statistical sampling method in which the uncertainty parameters are simultaneously sampled for each case. The approved ASTRUM evaluation model is documented in WCAP-16009-P-A [3].

This report summarizes the application of the Westinghouse ASTRUM BELOCA evaluation model to Comanche Peak Unit 2 for the large break LOCA accident analysis. Table 1 lists the major plant parameter assumptions used in the BELOCA analysis for Comanche Peak Unit 2. Both Luminant Power and its analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses. These interface processes, along with vendor internal processes for assessing evaluation model changes and errors, are used to identify the need for LOCA analyses impact assessments.

Method of Thermal Analysis for Comanche Peak Unit 2

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in 10 CFR 50.46 [4], both the Nuclear Regulatory Commission (NRC) and the industry recognized that stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling System (ECCS) would be conservatively underestimated, resulting in predicted peak clad temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1. Identify, through separate effects and integral effects experiments, the degree of conservatism in those models required in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.

2. Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without a loss in safety to the public. It was also found that some plants were being restricted in operating flexibility by overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 [5]. The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models", to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best estimate codes is provided in Regulatory Guide 1.157 [6].

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology [7]. This method outlined an approach for defining and qualifying a best estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three and four loop pressurized water reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and has been approved by the NRC [1]. This methodology was later extended to 2-loop Westinghouse plants with Upper Plenum Injection (UPI) in 1999 as documented in the NRC safety evaluation report appended to the UPI topical report [2].

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical TReatment of Uncertainty Method [3]. This method is still based on the CQD methodology and follows the steps in the CSAU methodology. However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method in which the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in WCAP-16009-P-A [3]. The ASTRUM methodology remains applicable to three and four loop PWRs, as well as 2-loop Westinghouse plants with UPI. This methodology was also extended to CE design PWRs.

The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95th percentile of the Peak Clad Temperature (PCT), Local Maximum Oxidation (LMO), and Core Wide Oxidation (CWO) with 95% confidence level. These parameters are needed to satisfy the 10 CFR 50.46 criteria with regard to PCT, LMO, and CWO.

Downcomer boiling is considered as appropriate in the ASTRUM methodology. The WCOBRA/TRAC computer code determines if downcomer boiling will occur for a particular transient. If downcomer boiling is determined to occur in a transient, WCOBRA/TRAC includes the effects of downcomer boiling in the transient calculation.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A [3] as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A [3] was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report appended to this WCAP. The Best Estimate LBLOCA analysis and associated model for Comanche Peak Unit 2 is unit-specific.

Description of a Large Break LOCA Transient

Before the break occurs, the RCS (Reactor Coolant System) is assumed to be operating normally at full power in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. A large break is assumed to open instantaneously in one of the main RCS cold leg pipes.

Immediately following the cold leg break, a rapid system depressurization occurs along with a core flow reversal due to a high discharge of sub-cooled fluid into the broken cold leg and out of the break. The fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up, while the core power decreases due to voiding in the core. The hot water in the core, upper plenum, and upper head flashes to steam, and subsequently the cooler water in the lower plenum and downcomer begins to flash. Once the system has depressurized to the accumulator pressure, the accumulator begins to inject cold borated water into the intact cold legs. During the blowdown period, a portion of the injected Emergency Core Cooling System (ECCS) water is calculated to be bypassed around the downcomer and out of the break. The bypass period ends as the system pressure continues to decrease and approaches the containment pressure, resulting in reduced break flow and consequently, reduced core flow.

As the refill period begins, the core continues to heat up as the vessel begins to fill with ECCS water. This phase continues until the lower plenum is filled, the bottom of the core begins to reflood, and entrainment begins.

During the reflood period, the core flow is oscillatory as ECCS water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. The pumped upper plenum and cold leg injection ECCS water aids in the filling of the vessel and downcomer, which subsequently supplies water to maintain the core and downcomer water levels and complete the reflood period.

ASTRUM Analysis Results for Comanche Peak Unit 2

The results of the Comanche Peak Unit 2 ASTRUM analysis are summarized in Table 2. Table 3 contains a sequence of events for the limiting PCT transient.

The scatter plot presented in Figure 1 shows the effect of the effective break area on the analysis PCT. The effective break area is calculated by multiplying the discharge coefficient C_D with the sample value of the break area, normalized to the cold-leg cross sectional area. Figure 1 is provided to show the break area is a significant contributor to the variation in PCT.

From the 124 calculations performed as part of the ASTRUM analysis, different cases proved to be the limiting PCT and limiting LMO transient for Comanche Peak Unit 2. Figure 2 shows the predicted clad temperature transient at the PCT limiting elevation for the limiting PCT case. Figure 3 presents the clad temperature transient predicted at the LMO elevation for the limiting LMO case. Due the low PCT results, CWO remained on the order of 0 percent for all cases.

Figure 4 through 17 illustrate the key major response parameters for the limiting PCT transient. The reference point for the lower plenum liquid level presented in Figure 11 is the bottom of the vessel (10.1 feet below the bottom of the active fuel). The reference point for the downcomer liquid level presented in Figure 12 is the point at which the outside of the core barrel, if extended downward, intersects with the vessel wall (6.1 feet below the bottom of the active fuel). The reference point for the core collapsed liquid levels presented in Figures 13 and 16 is the bottom of the active fuel.

The containment backpressure utilized for the LBLOCA analysis compared to the calculated containment backpressure is provided in Figure 18. The worst single failure for the LBLOCA analysis is the loss of one train of ECCS injection (consistent with the ASTRUM Topical); however, all containment systems which would reduce containment pressure are modeled for the LBLOCA containment backpressure calculation.

10 CFR 50.46 Requirements

It must be demonstrated that there is a high level of probability that the following limits set forth in 10 CFR 50.46 are met:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 1632°F for Comanche Peak Unit 2, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F," is demonstrated. The result is shown in Table 2.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. Since the resulting LMO for the limiting case is 0.71 percent for Comanche Peak Unit 2, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Local Maximum Oxidation of the cladding less than 17 percent," is demonstrated. The result is shown in Table 2.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. While the limiting LMO is determined based on the single Hot Rod, the CWO value can be conservatively chosen as that calculated for the limiting Hot Assembly Rod (HAR) when there is significant margin to the regulatory limit. The limiting HAR total maximum oxidation is 0 percent for Comanche Peak Unit 2. Thus, a detailed CWO calculation is not needed because the calculations would include many lower power assemblies and the outcome would always be less than the limiting HAR total maximum oxidation. Therefore, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent," is demonstrated. The result is shown in Table 2.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The grid crush calculations currently in place for Comanche Peak Unit 2 remain unchanged with the application of the ASTRUM methodology [3], therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at Comanche Peak Unit 2 to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology [3].

Based on the ASTRUM analysis results (see Table 2), it is concluded that Comanche Peak Unit 2 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

References

[1] Bajorek, S. M., et. al., 1998, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1, and WCAP-14747 (Non-Proprietary).

[2] Dederer, S. I., et. al., 1999, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," WCAP-14449-P-A, Revision 1 and WCAP-14450, Revision 1 (Non-Proprietary).

[3] Nissley, M. E., et.al., 2005, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A and WCAP-16009-NP-A (Non-Proprietary).

[4] "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 4, 1974.

[5] Information Report from W.J. Dircks to the Commissioners, "Emergency Core Cooling System Analysis Methods", SECY-83-472, November 17, 1983.

[6] "Best Estimate Calculations of Emergency Core Cooling System Performance," Regulatory Guide 1.157, USNRC, May 1989.

[7] Boyack, B., et. al., 1989, "Quantifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident," NUREG/CR-5249.

Table 1 - Major Plant Parameter Assumptions Used in the BELOCA Analysis for Comanche Peak Unit 2

Parameter	Value
Plant Physical Description	
• SG Tube Plugging	≤ 10%
Plant Initial Operating Conditions	
• Reactor Power	≤ 3612 MWt (± 0.6% uncertainties)
• Peaking Factors	$F_Q \leq 2.5$ $F_{\Delta H} \leq 1.60$
• Axial Power Distribution	See Figure 19
Fluid Conditions	
• T_{AVG}	$574.2 - 6.5\ °F \leq T_{AVG} \leq 589.2 + 6.5\ °F$
• Pressurizer Pressure	$2250 - 30$ psia $\leq P_{RCS} \leq 2250 + 30$ psia
• Reactor Coolant Flow	≥ 95,700 gpm
• Accumulator Temperature	$88\ °F \leq T_{ACC} \leq 120\ °F$
• Accumulator Pressure	603 psia $\leq P_{ACC} \leq$ 693 psia
• Accumulator Water Volume	6119 gal $\leq V_{ACC} \leq$ 6597 gal
• Accumulator Boron Concentration	≥ 2300 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection Flow	Minimum
• Safety Injection Temperature	$40\ °F \leq T_{SI} \leq 120\ °F$
• Safety Injection Initiation Delay Time	≤ 17 sec (with offsite power) ≤ 27 sec (without offsite power)
• Containment Pressure	Bounded (minimum)

Table 2 – Comanche Peak Unit 2 Best Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	1632	< 2200
95/95 LMO[1] (%)	0.71	< 17
95/95 CWO[2] (%)	0.00	< 1

Notes:
1. Local Maximum Oxidation
2. Core Wide Oxidation

Table 3 – Comanche Peak Unit 2 Best Estimate Large Break Sequence of Events for the Limiting PCT Case

Event	Time (sec)
Event	0.0
Start of Transient	5.3
Safety Injection Signal	10.5
Accumulator Injection Begins	24.0
End of Blowdown	31.0
Bottom of Core Recovery	32.3
Safety Injection Begins	33.1
Accumulator Empty	~85
PCT Occurs	550.0
End of Transient	

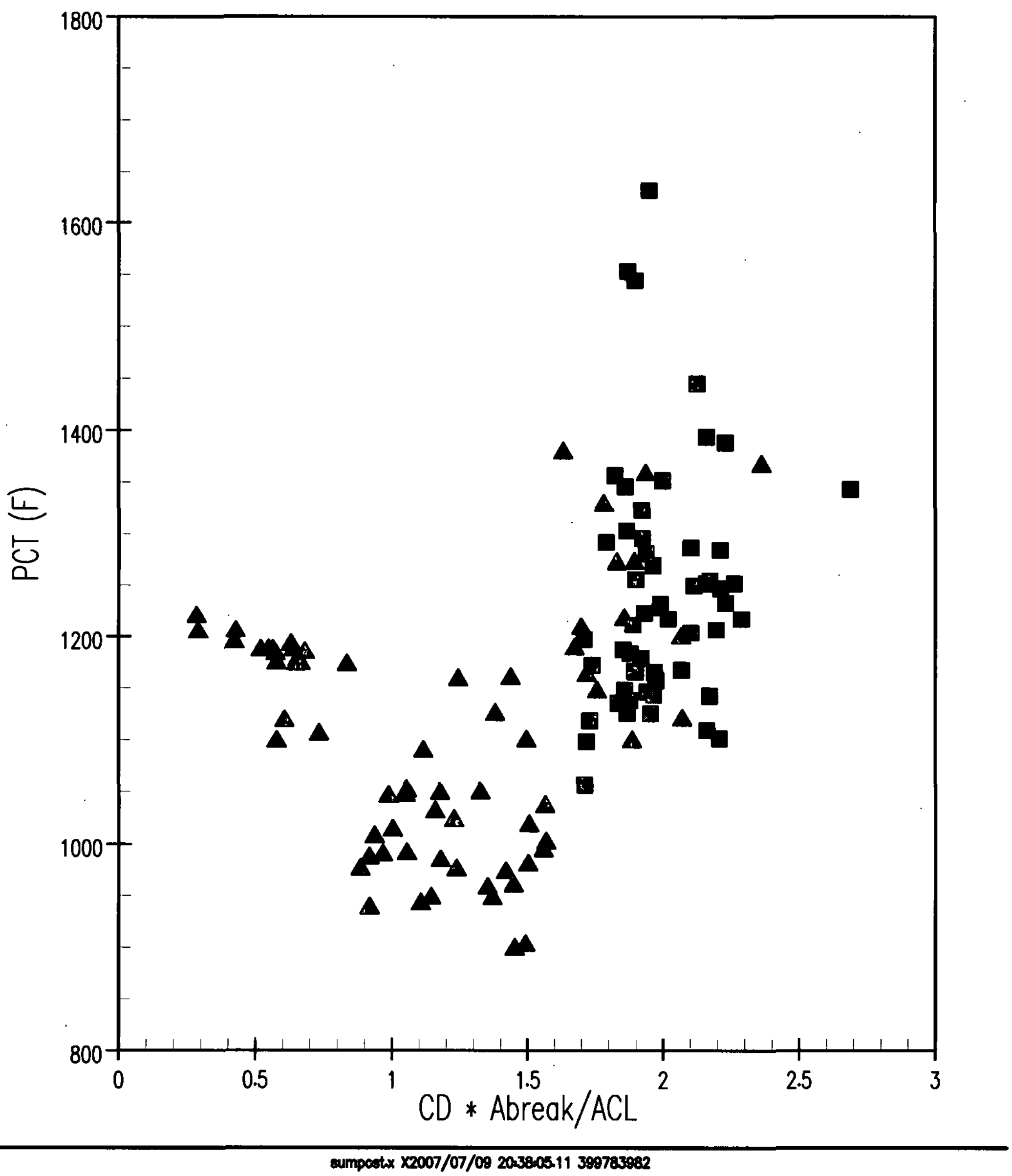

Figure 1 – Comanche Peak Unit 2 HOTSPOT PCT versus Effective Break Area Scatter Plot (CD = Discharge Coefficient, Abreak = Break Area, ACL = Cold Leg Area)

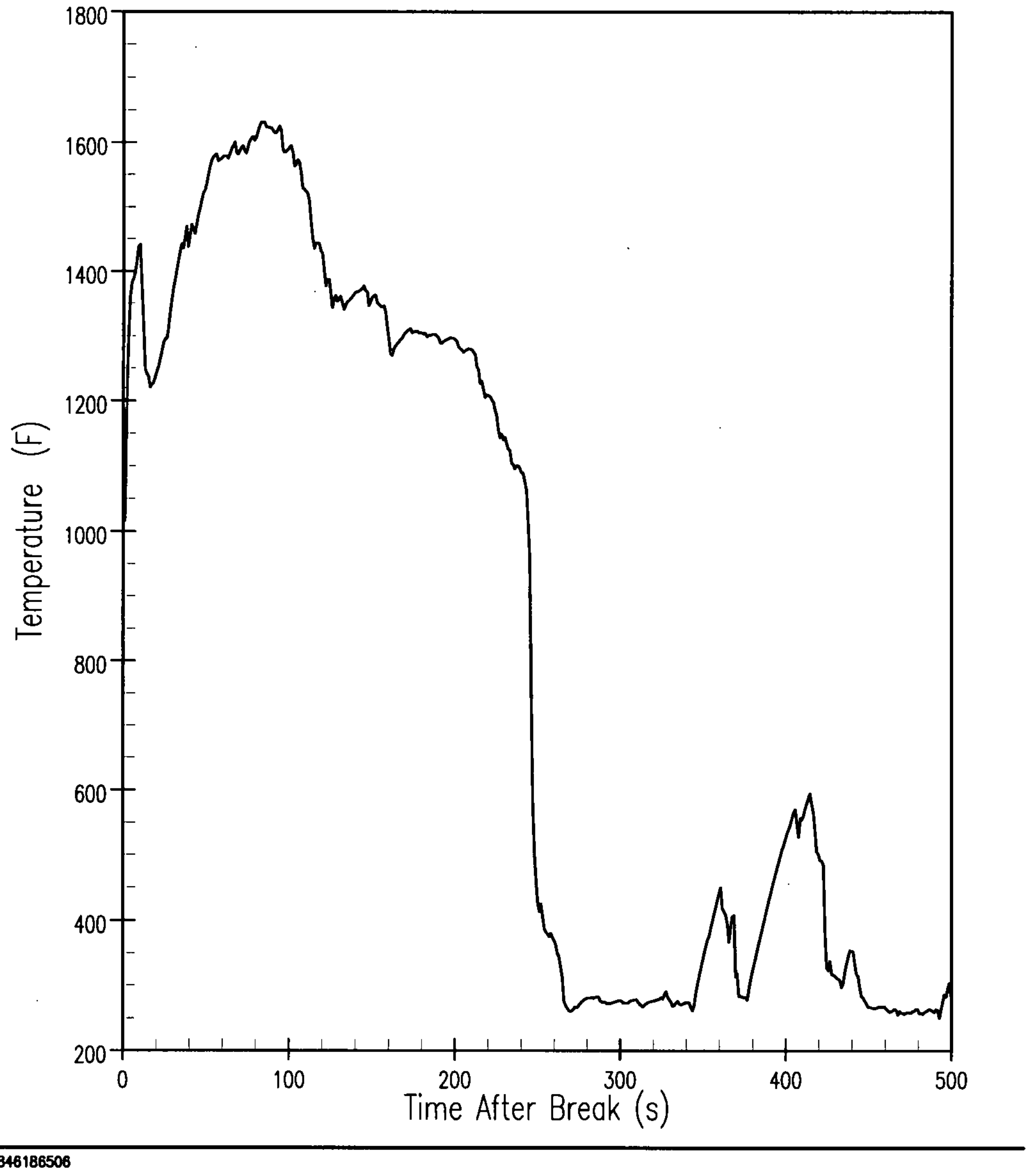

Figure 2 – Comanche Peak Unit 2 HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case

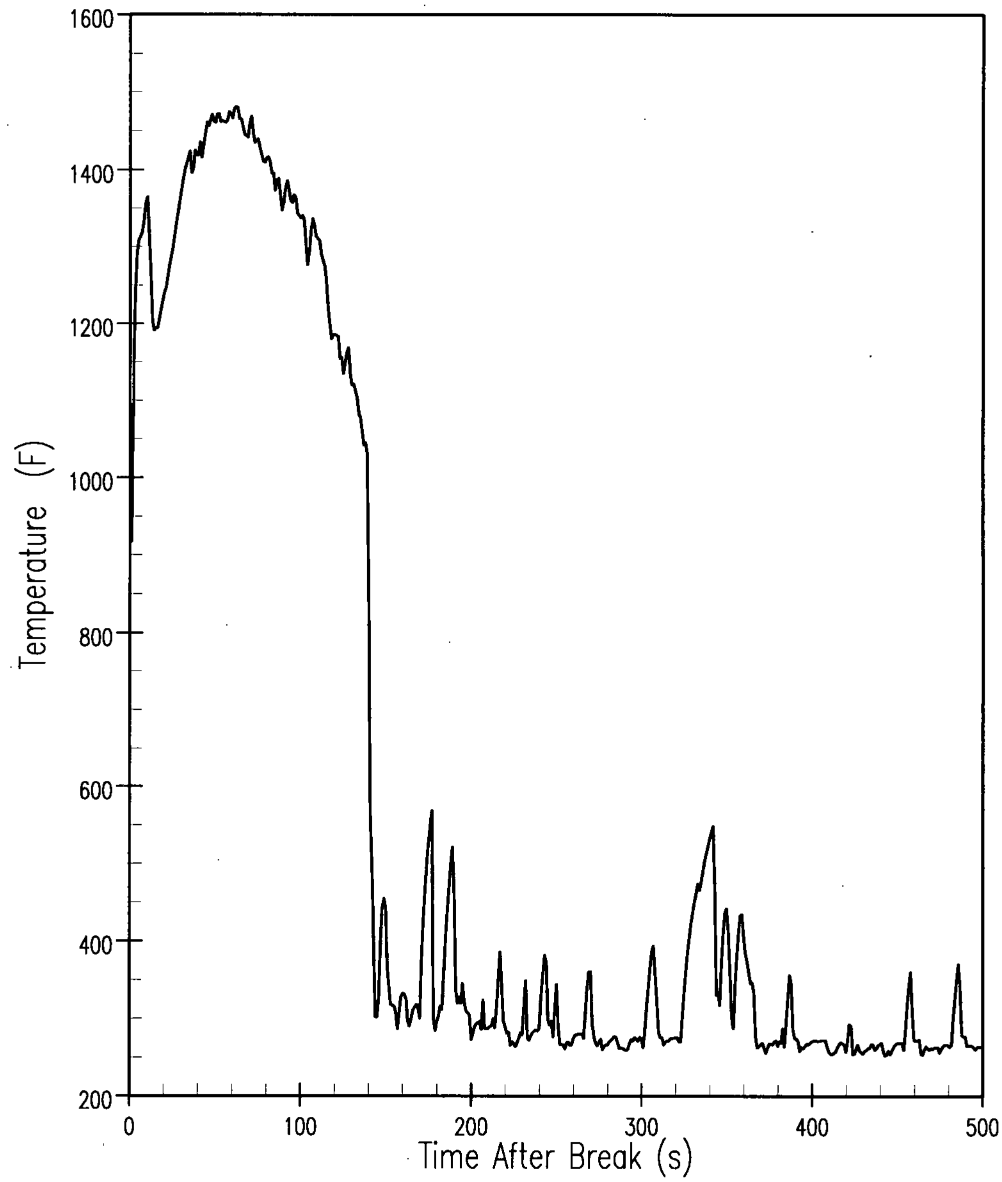

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Figure 3 – Comanche Peak Unit 2 HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting LMO Case

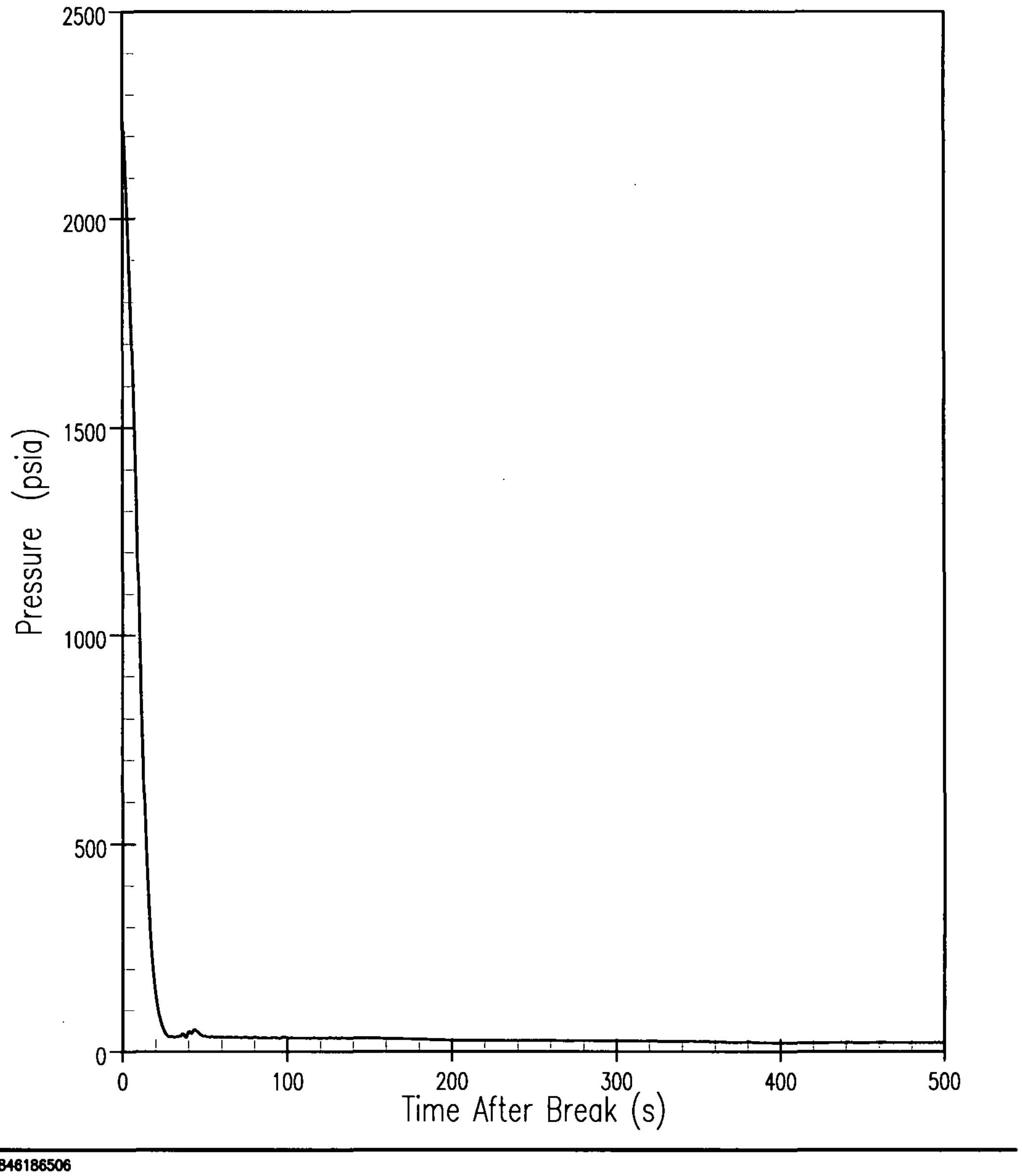

Figure 4 – Comanche Peak Unit 2 Pressurizer Pressure for the Limiting PCT Case

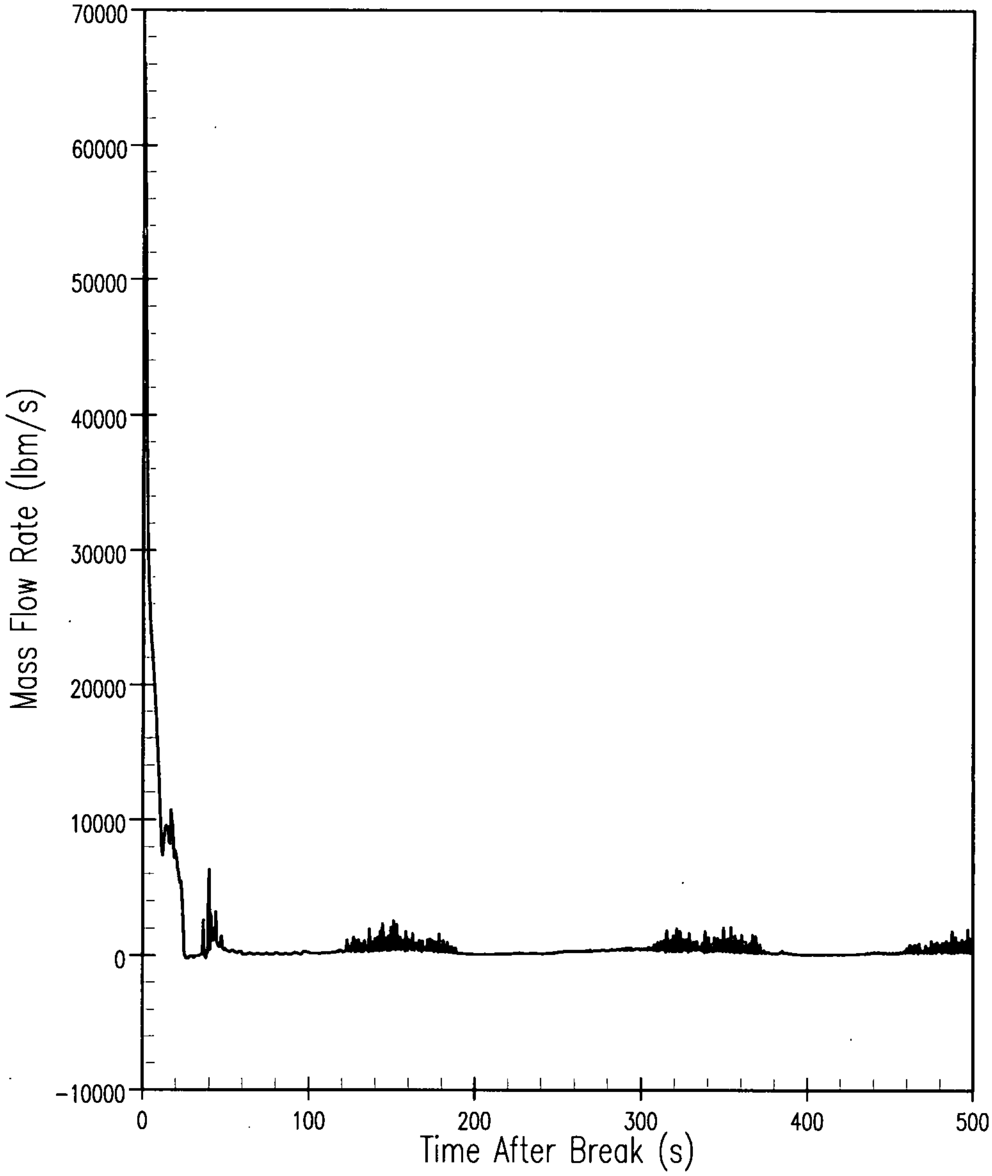

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Figure 5 – Comanche Peak Unit 2 Vessel Side Break Flow for the Limiting PCT Case

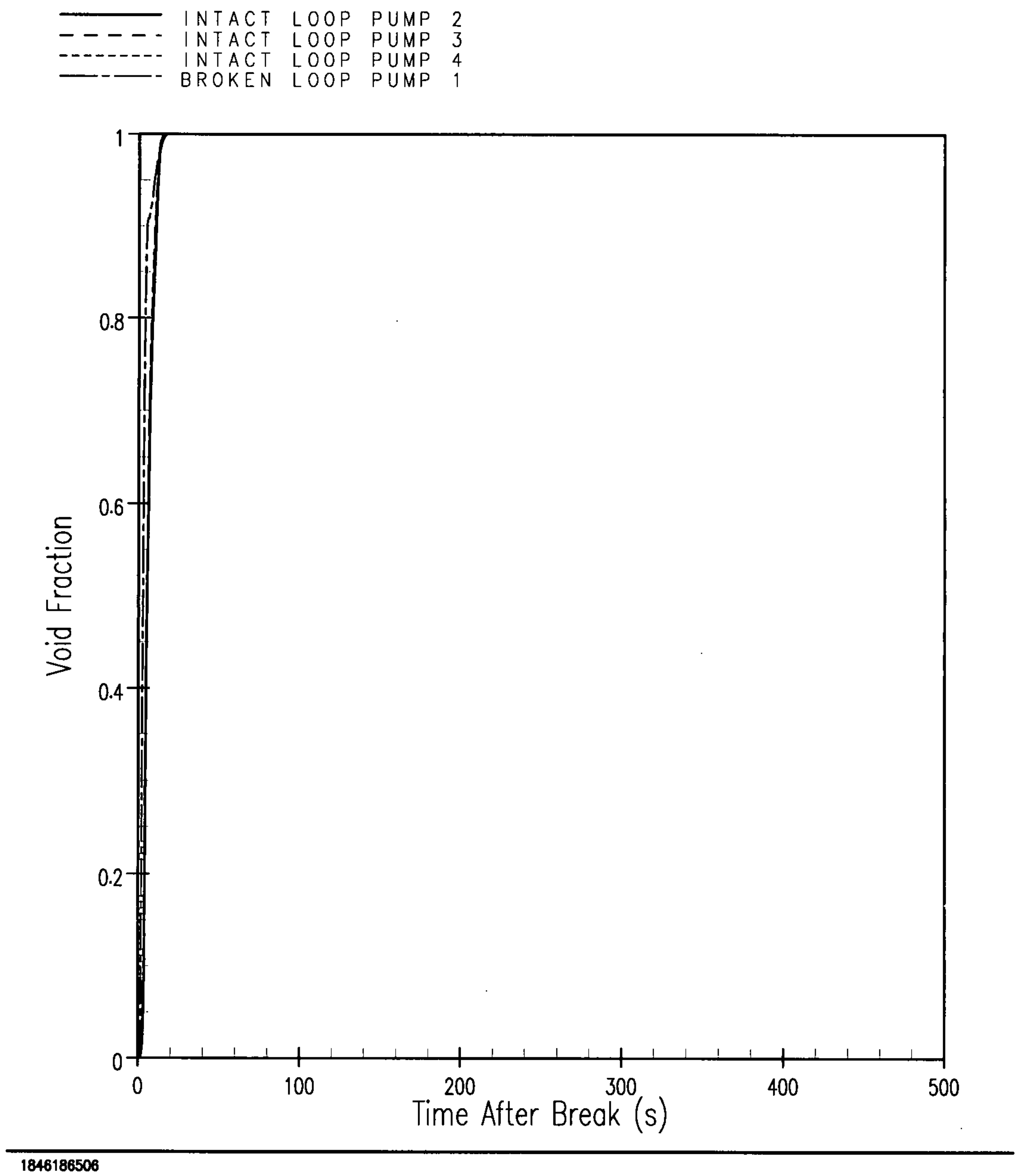

Figure 6 – Comanche Peak Unit 2 Void Fraction in Pumps for the Limiting PCT Case

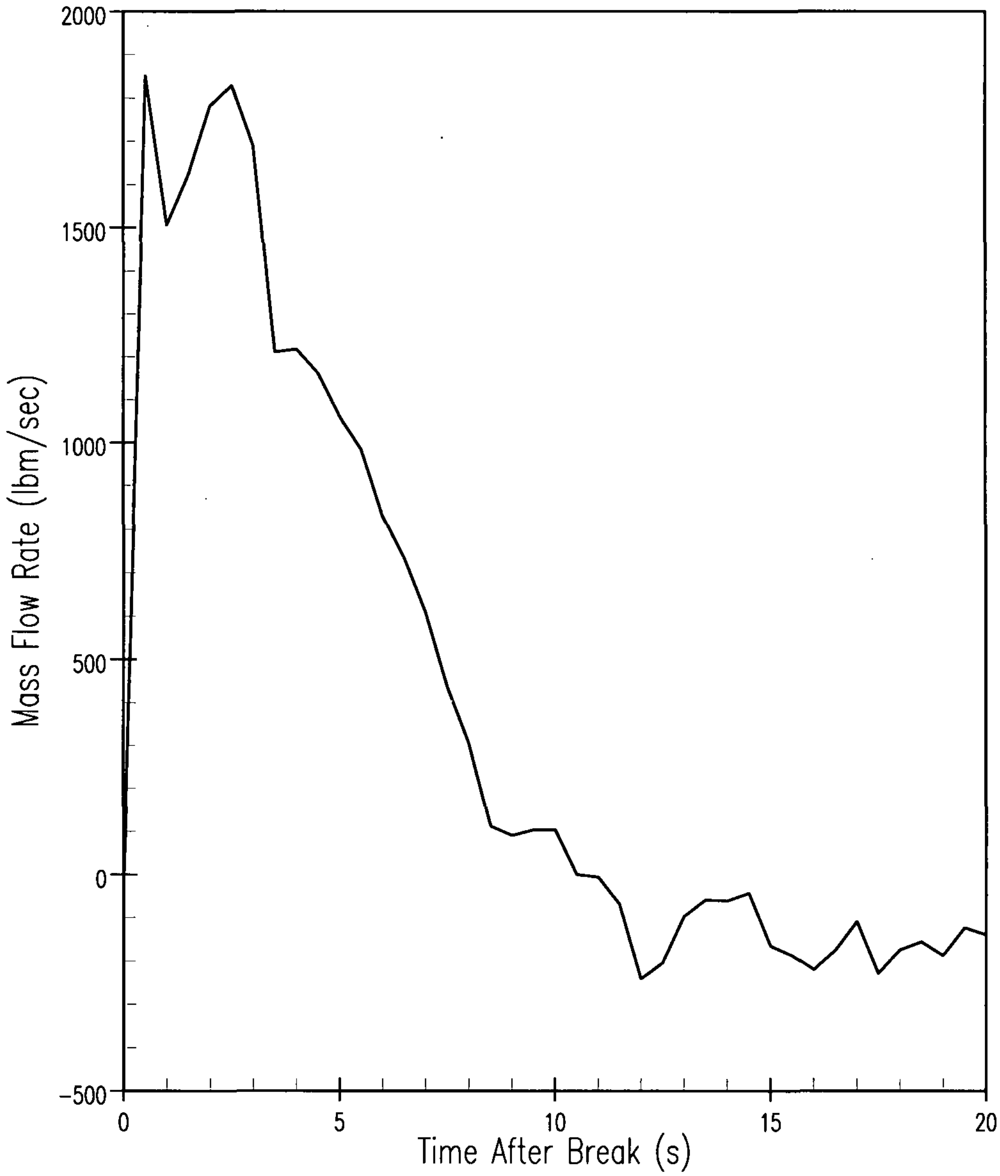

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Figure 7 – Comanche Peak Unit 2 Vapor Flow at Top of Core for the Limiting PCT Case

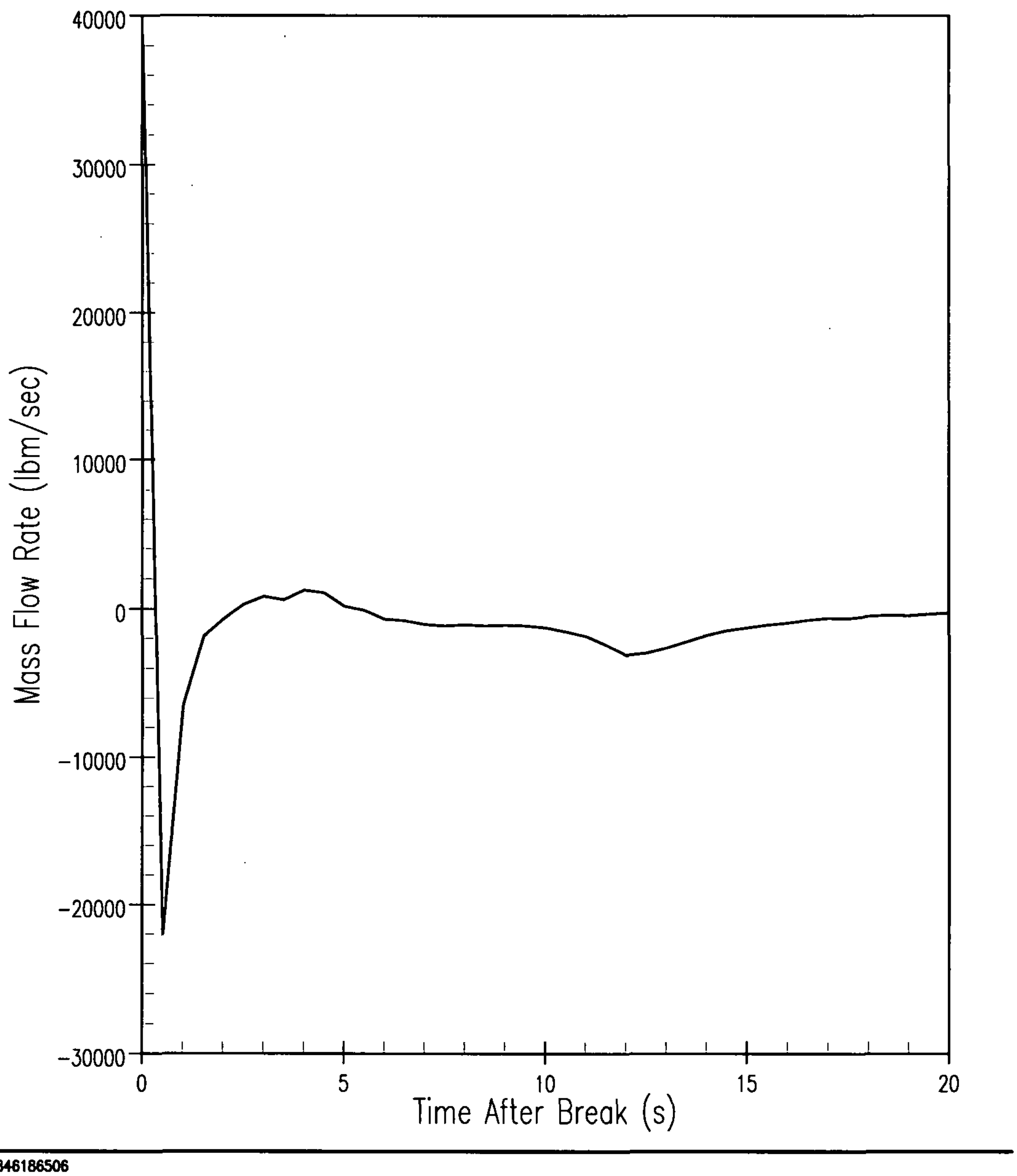

Figure 8 – Comanche Peak Unit 2 Total Flow at Bottom of Core for the Limiting PCT Case

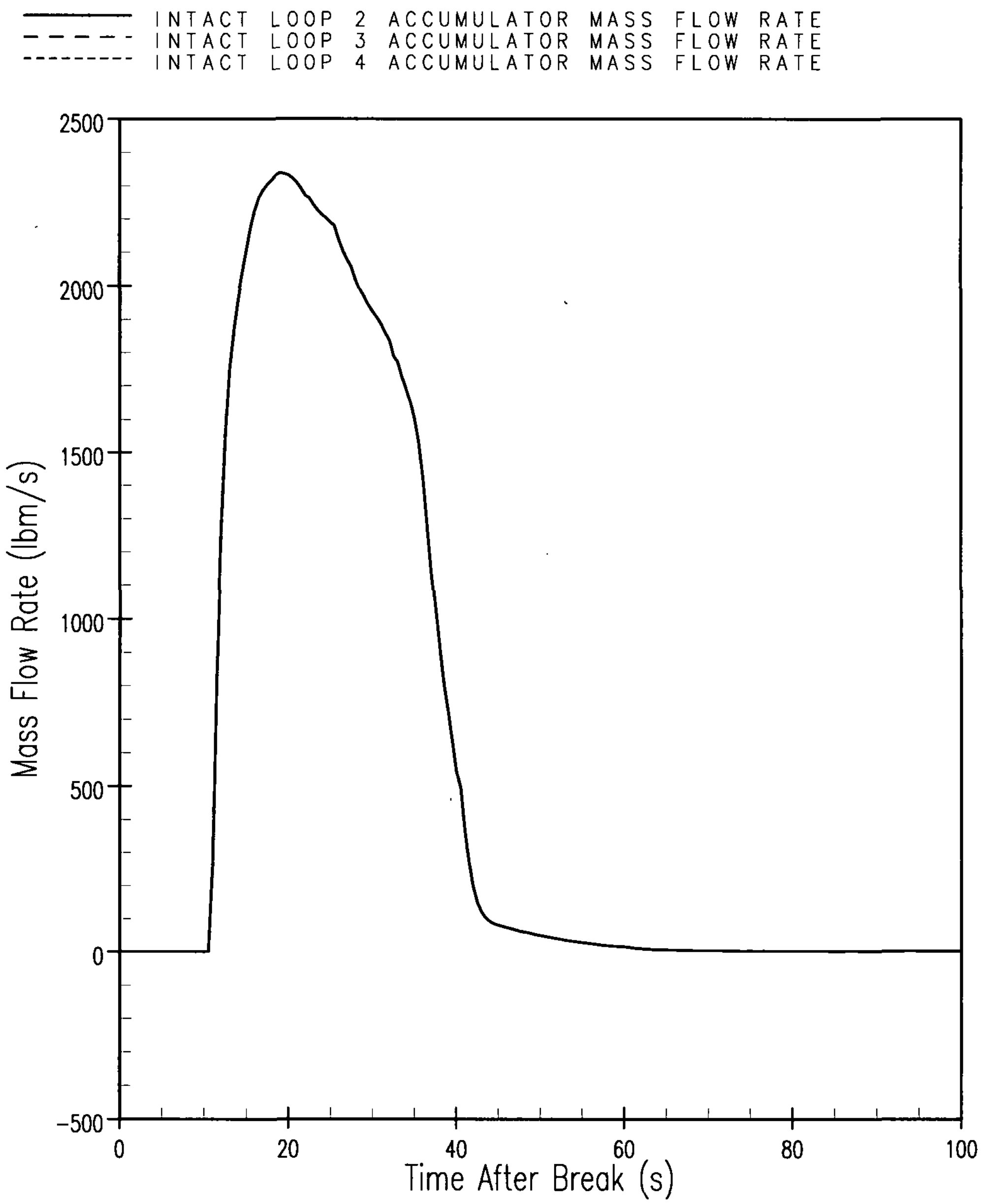

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Figure 9 – Comanche Peak Unit 2 Accumulator Injection Flow for the Limiting PCT Case

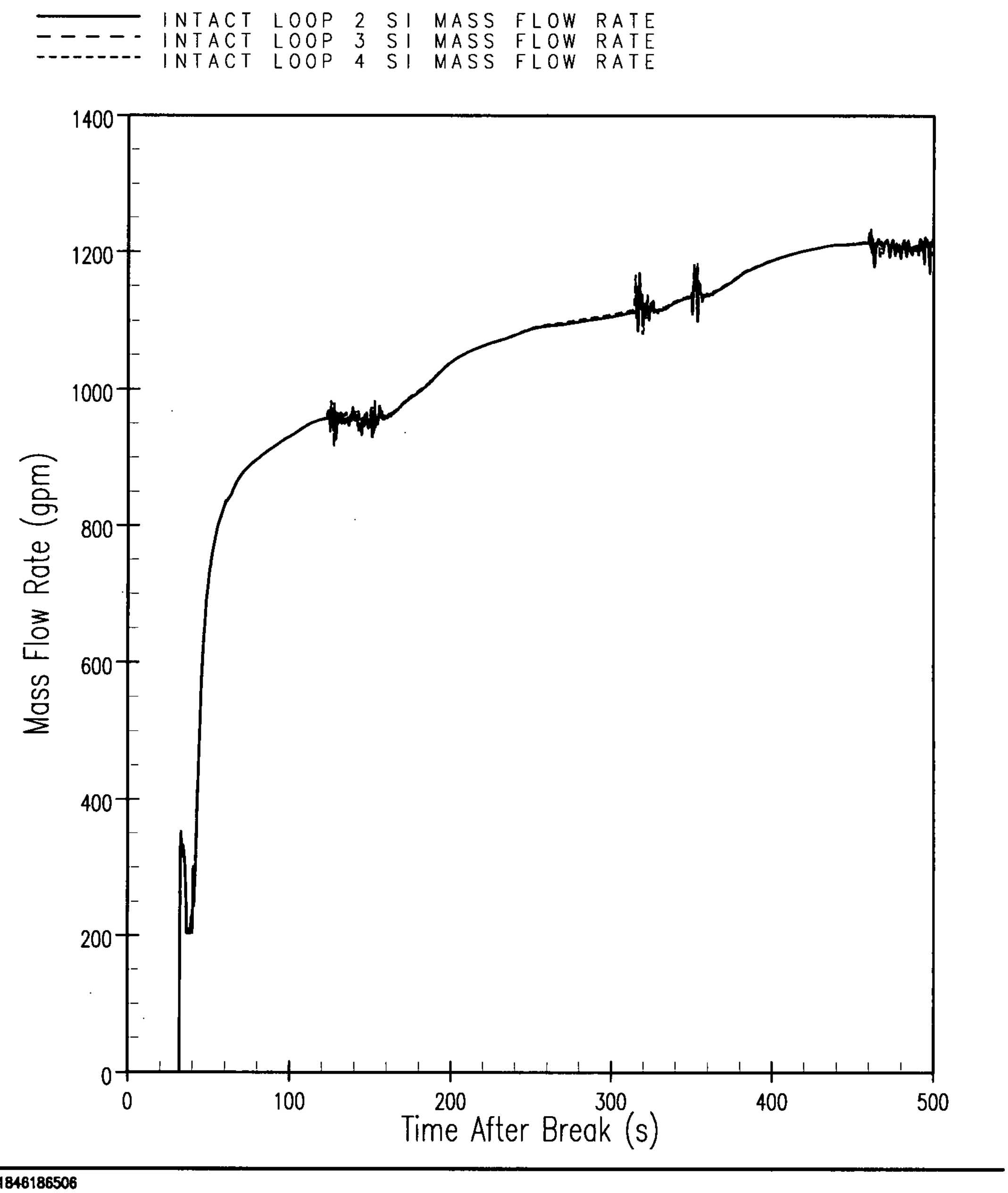

Figure 10 – Comanche Peak Unit 2 Safety Injection Flow for the Limiting PCT Case

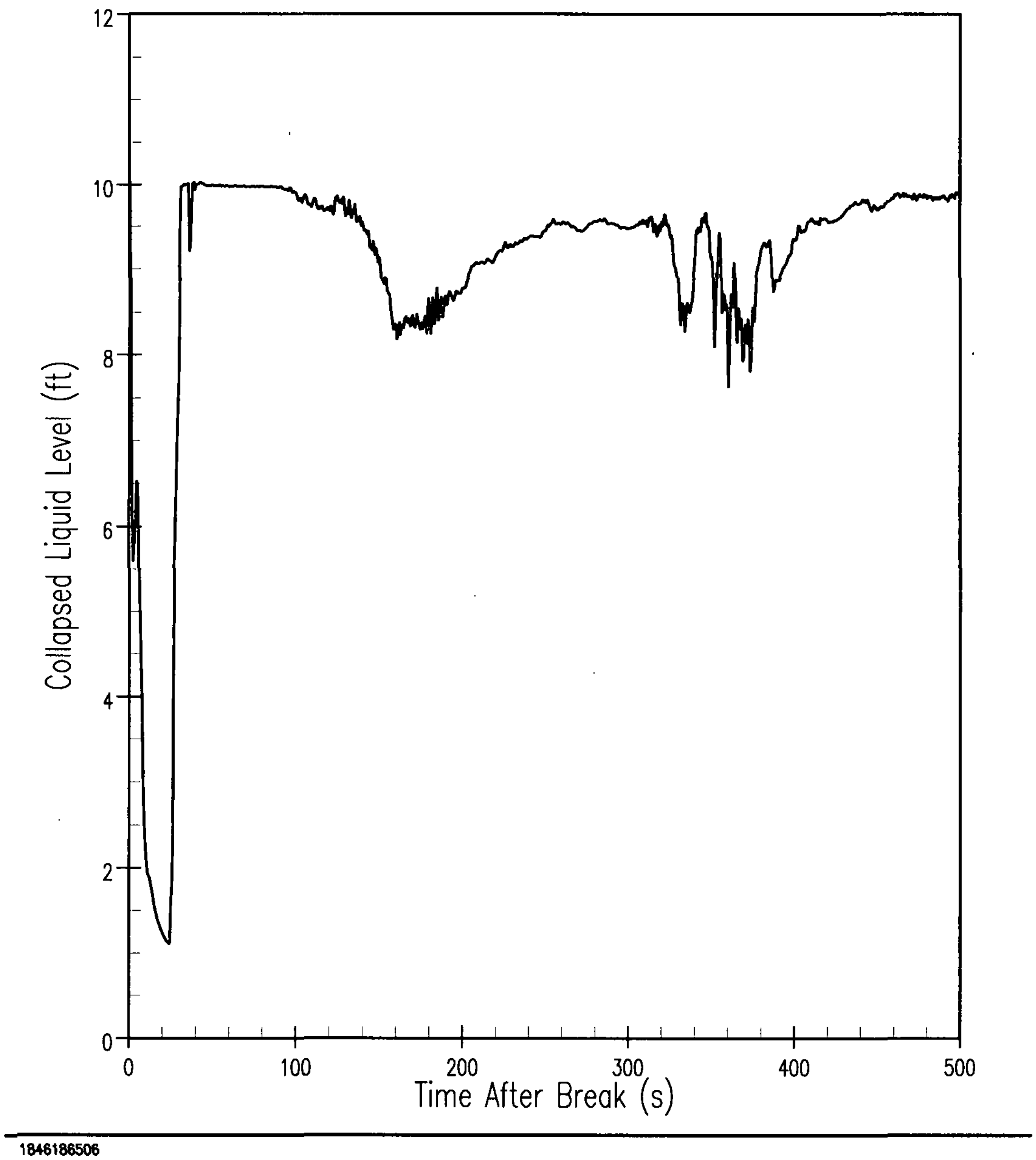

Figure 11 – Comanche Peak Unit 2 Lower Plenum Collapsed Liquid Level for the Limiting PCT Case

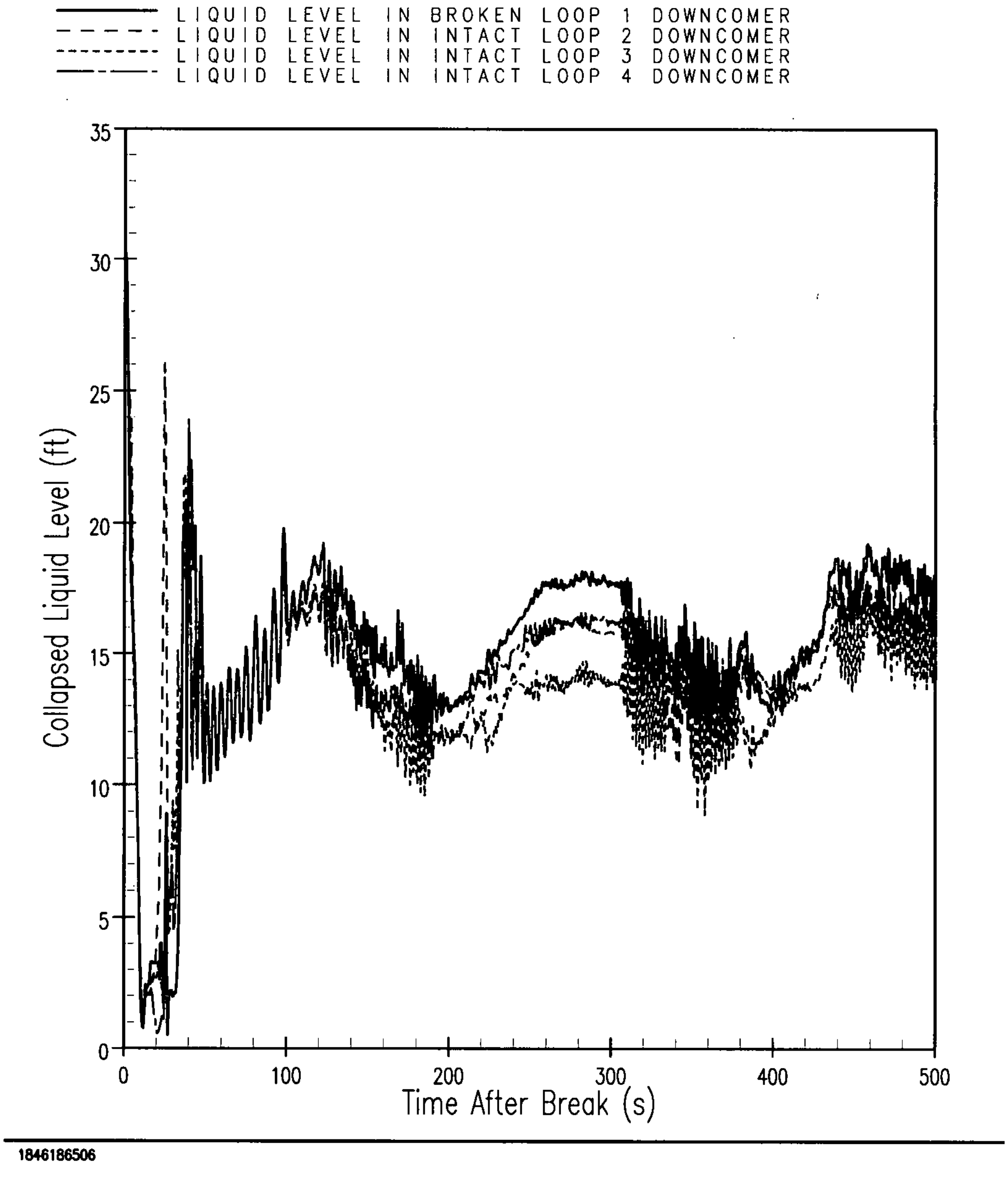

Figure 12 – Comanche Peak Unit 2 Downcomer Collapsed Liquid Levels for the Limiting PCT Case

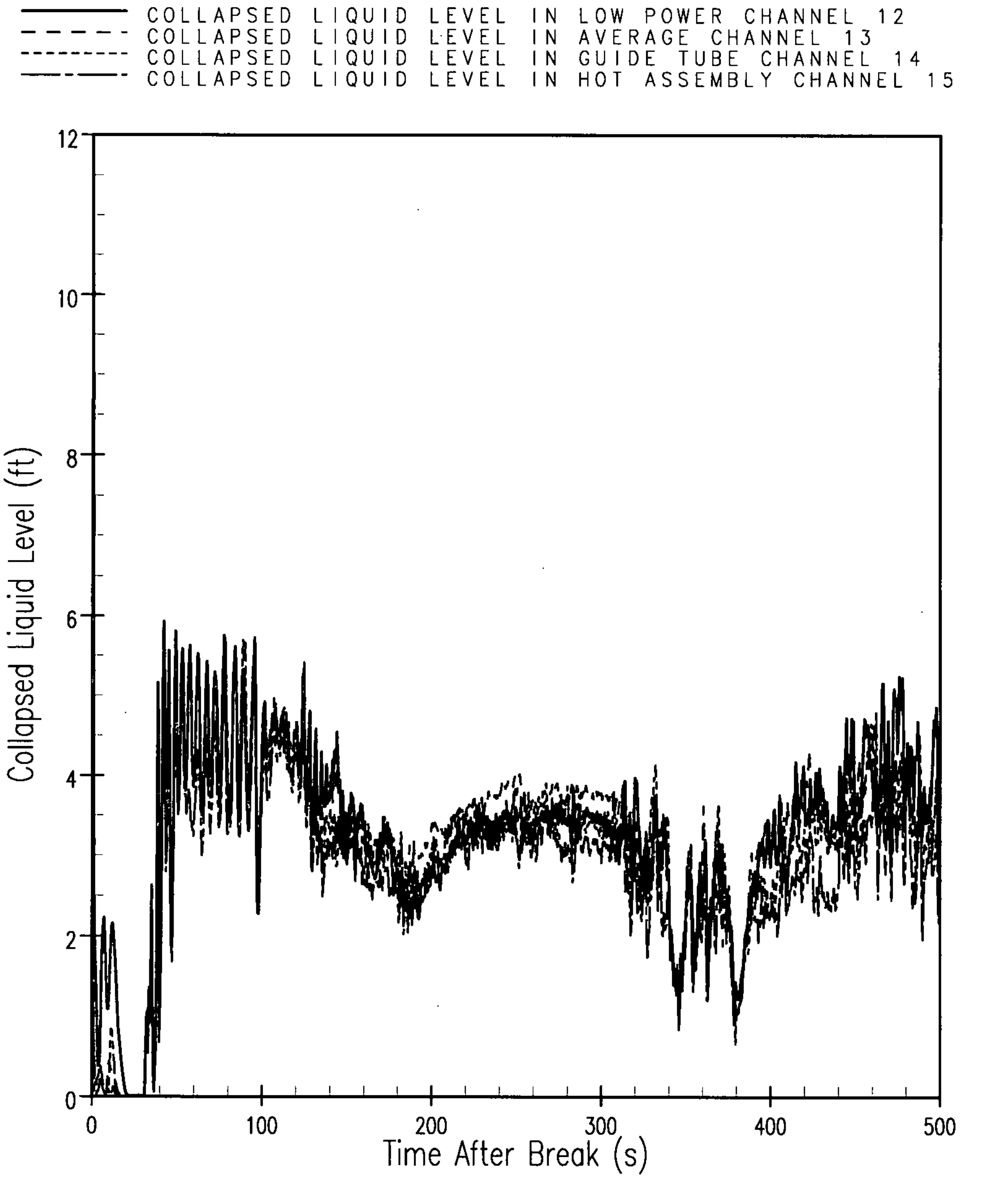

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Figure 13 – Comanche Peak Unit 2 Core Collapsed Liquid Levels for the Limiting PCT Case

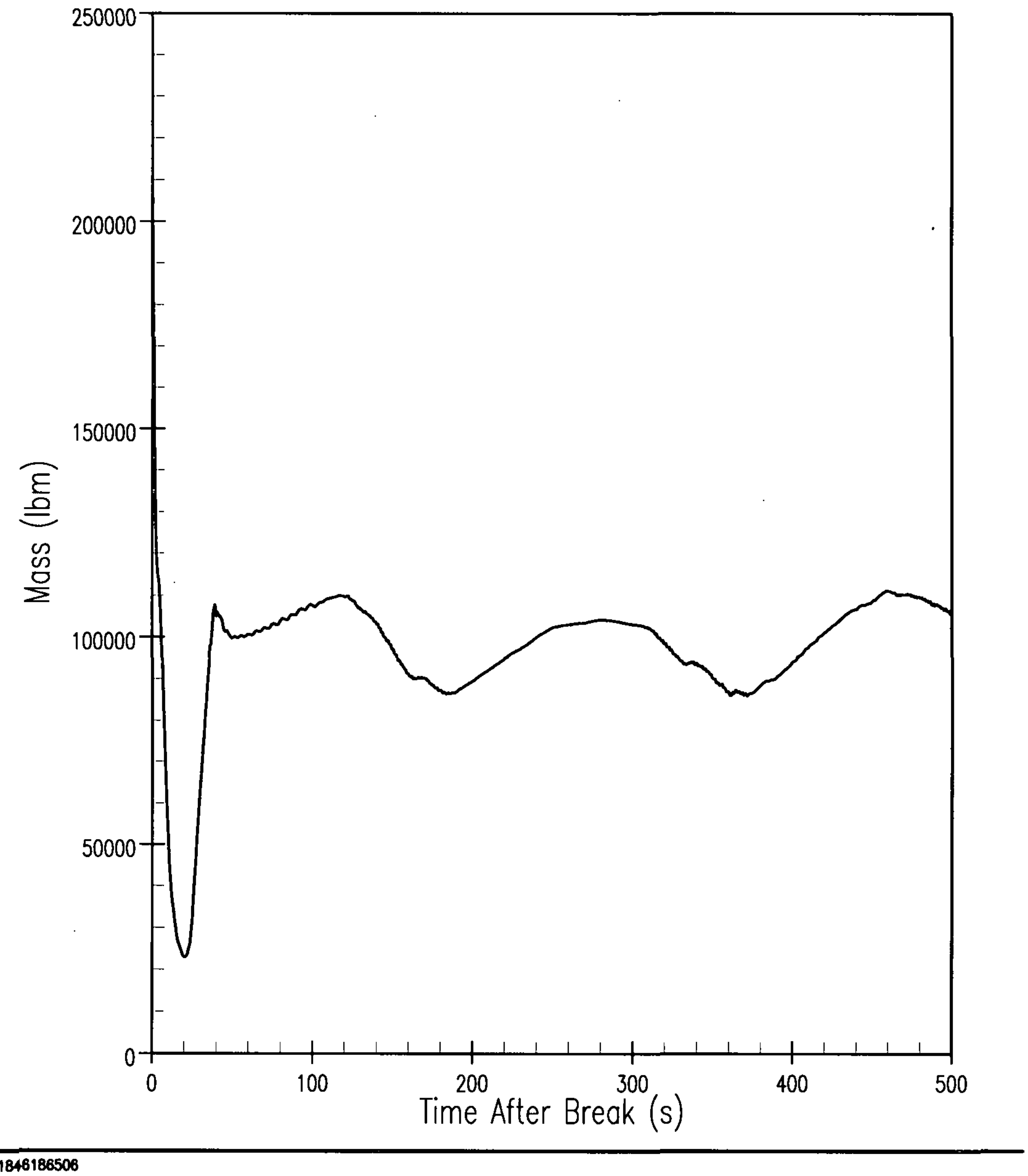

Figure 14 – Comanche Peak Unit 2 Vessel Fluid Mass for the Limiting PCT Case

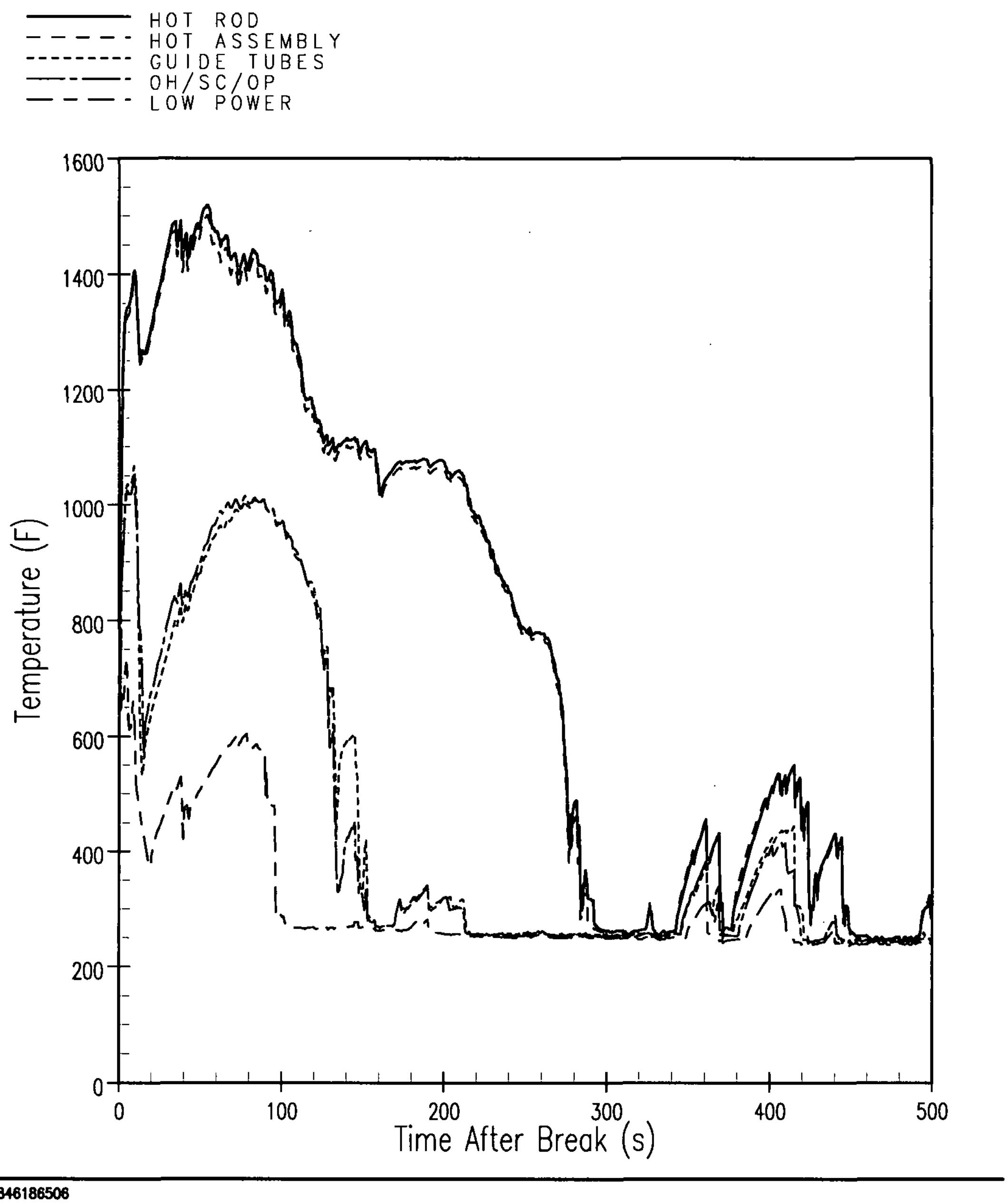

Figure 15 – Comanche Peak Unit 2 WCOBRA/TRAC Peak Clad Temperature for all 5 Rod Groups for the Limiting PCT Case

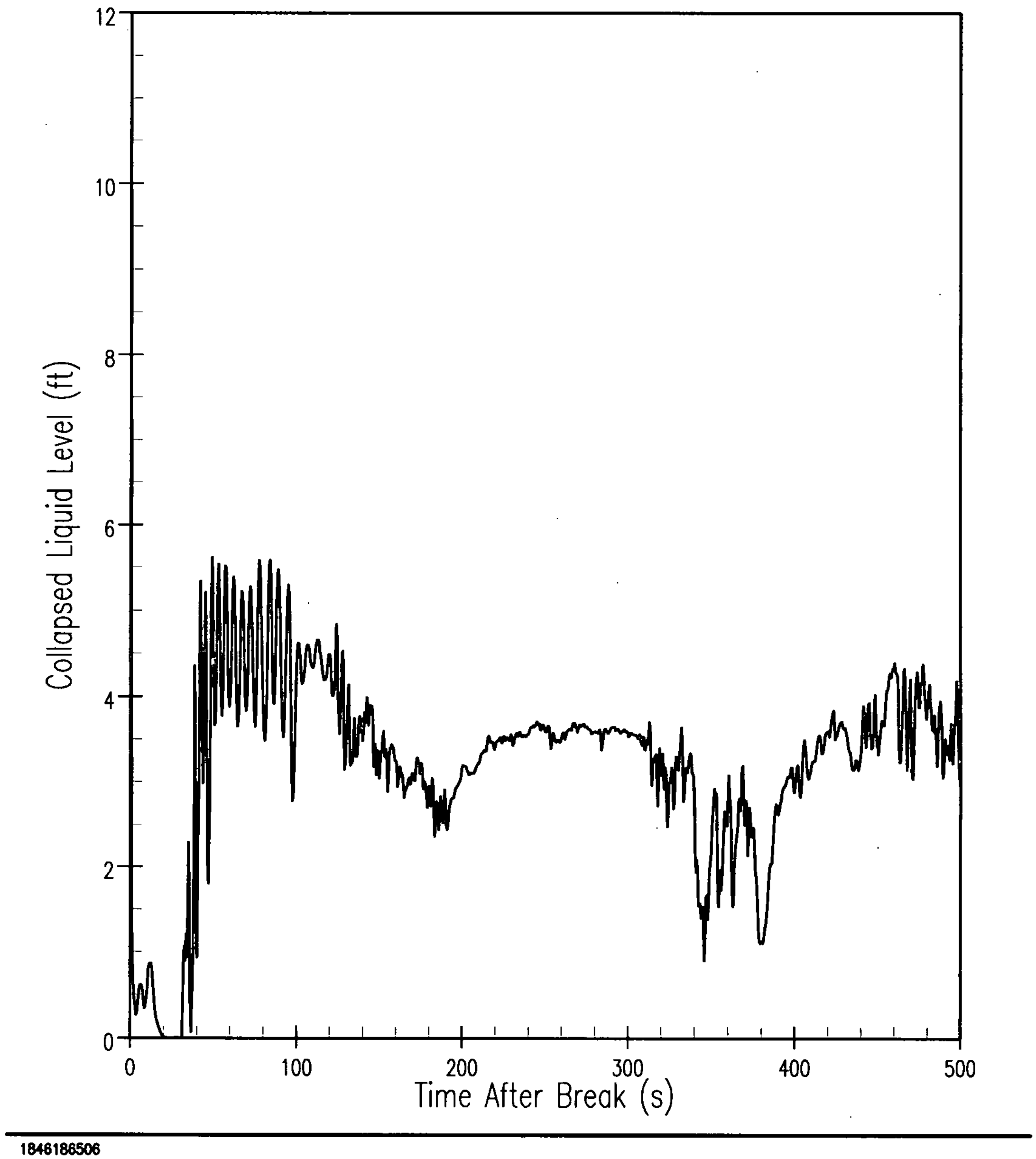

Figure 16 – Comanche Peak Unit 2 Average Core Collapsed Liquid Level per Assembly for the Limiting PCT Case

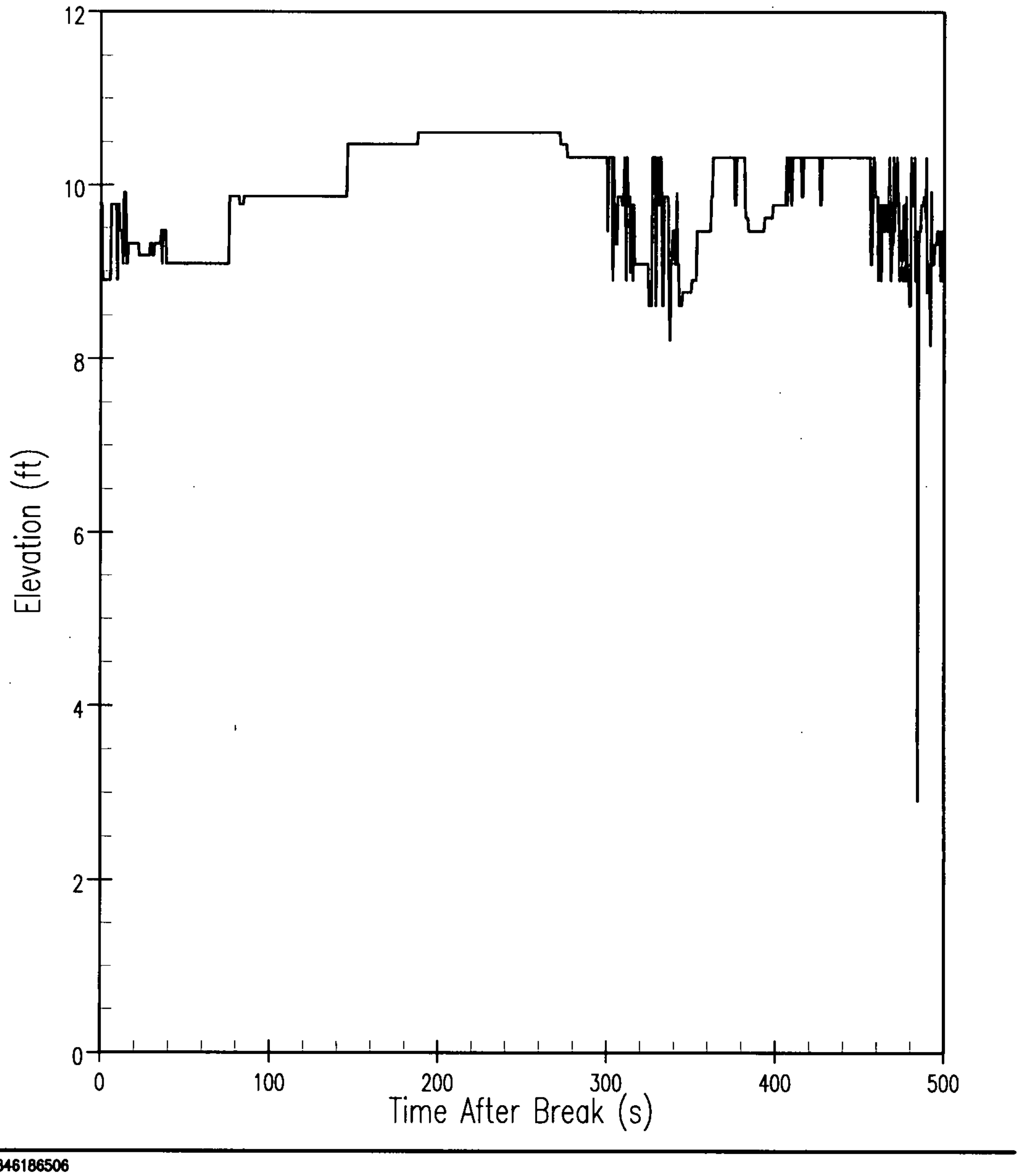

Figure 17 – Comanche Peak Unit 2 Peak Clad Temperature Elevation for the Hot Rod for the Limiting PCT Case

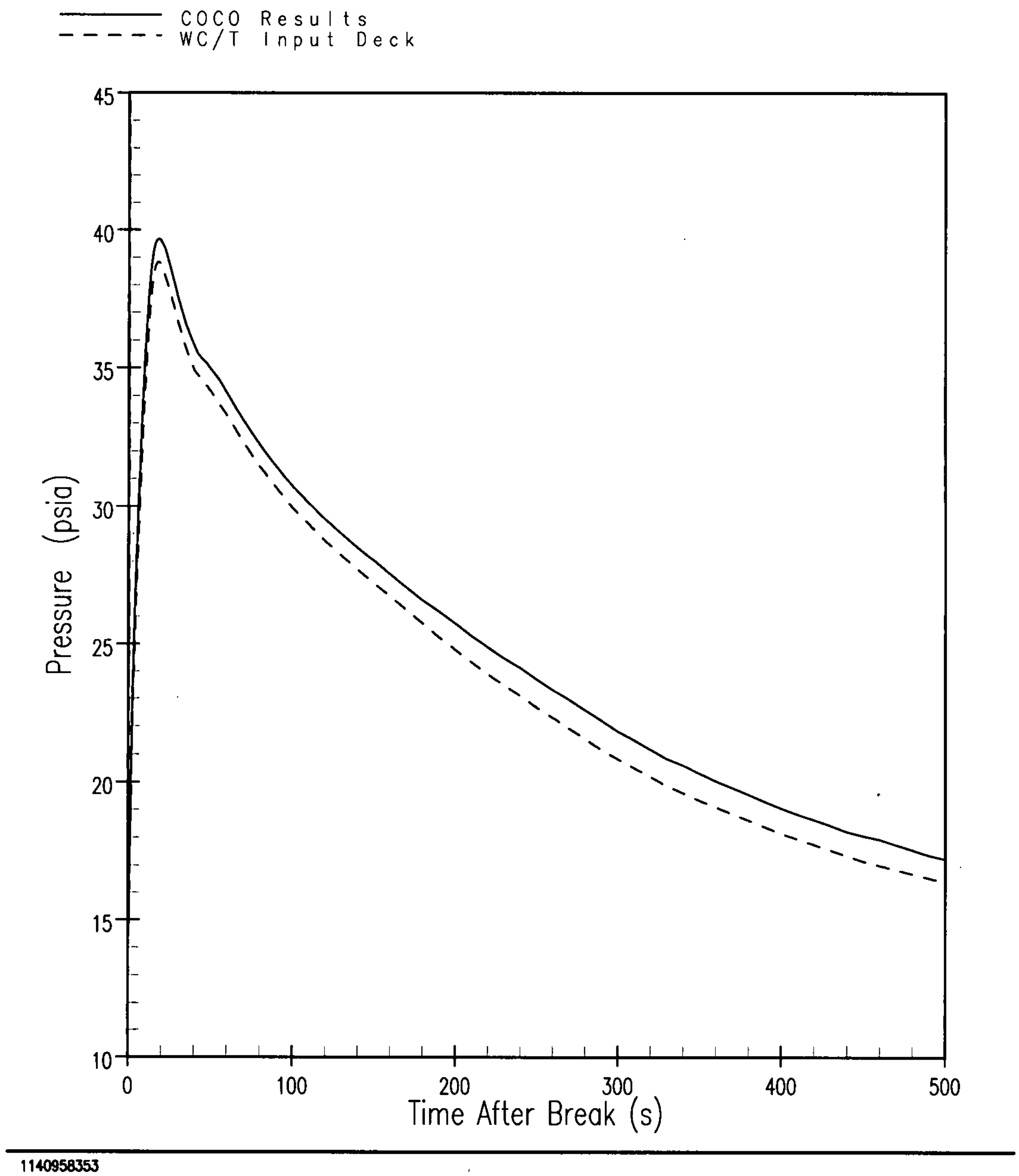

Figure 18 – Comanche Peak Unit 2 Analysis Versus Calculated Containment Backpressure

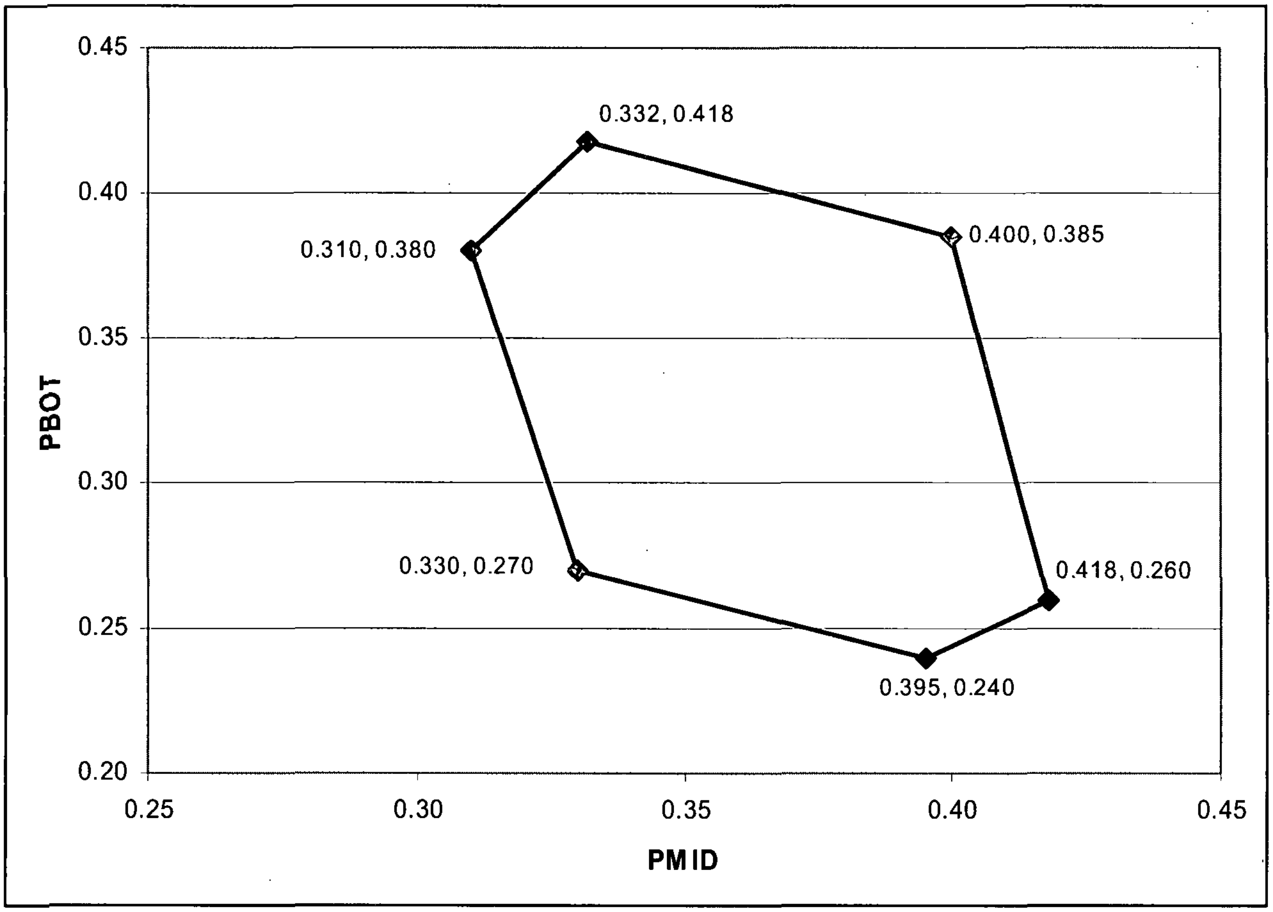

Figure 19 – Comanche Peak Unit 2 BELOCA Analysis Axial Power Shape Operating Space Envelope

PBOT = integrated power fraction in the bottom third of the core
PMID = integrated power fraction in the middle third of the core

Attachment 3 to TXX-07107

APPLICATION OF WESTINGHOUSE SMALL BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNITS 1 AND 2

APPLICATION OF THE WESTINGHOUSE SMALL BREAK LOCA METHODOLOGY TO COMANCHE PEAK UNITS 1 AND 2

Introduction

The small break loss-of-coolant-accident (SBLOCA) analyses for Comanche Peak Units 1 and 2 were completed using the 1985 Westinghouse SBLOCA Evaluation Model with NOTRUMP (NOTRUMP-EM) (References 1, 2 and 3) as a part of the Comanche Peak "Transition Project." The NOTRUMP-EM SBLOCA analyses were performed at 100.6% of the up-rated core power of 3612 MWt (NSSS power of 3628 MWt) for both units. The Unit 1 NOTRUMP model included the Δ-76 Replacement Steam Generator (RSG). The purpose of this analysis is to demonstrate conformance with the 10 CFR 50.46 (Reference 4) requirements at the up-rated conditions with NOTRUMP-EM. 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. The analysis has shown that no design or regulatory limit related to the small break LOCA would be exceeded at the conditions analyzed.

Input Parameters and Assumptions

The SBLOCA methodology using the NOTRUMP-EM was developed in accordance with the requirements of 10 CFR 50 Appendix K. This regulation was designed to produce a conservative prediction of the analysis results and includes various conservative modeling requirements such as the decay heat model (1971 ANS Infinite + 20%), the zirconium-water reaction model (Baker-Just) and the most limiting single failure criterion. For the SBLOCA analysis, loss of offsite power (LOOP) is assumed, which results in the limiting single failure assumption of the loss of one Emergency Diesel Generator (EDG) and a subsequent loss of one train of pumped Emergency Core Cooling System (ECCS). The SBLOCA analysis assumes that reactor trip occurs coincident with the LOOP, which results in the following: (a) Reactor Coolant Pump (RCP) trip and coastdown and (b) Steam Dump System being inoperable. Additional input assumptions and initial conditions for the SBLOCA analysis are found in Tables 1 – 4 for Units 1 and 2.

Description of Analyses and Calculations

Analytical Model

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

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

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 2) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. 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 1 illustrates the code interface for the NOTRUMP-EM.

Analysis Method

The small break LOCA analyses considered five different break cases each for Comanche Peak Unit 1 and Unit 2 as indicated by the results in Tables 5, 6 and 7, 8 respectively. A spectrum of cold leg breaks of equivalent diameters 2, 3, 4 and 6 inches and an accumulator line break of equivalent diameter 8.75 inch were considered. For both Units 1 and 2, the 4-inch break was found to be limiting for peak clad temperature (PCT). For both units the 2, 6 and 8.75 inch breaks resulted in minimal or no core uncovery and therefore PCT information was not calculated for these breaks. Note that intermediate (non-integer) breaks were not considered here since the limiting PCTs for the 4 inch break have considerable margin to the 2200°F limit set forth by the acceptance criteria for SBLOCA analysis.

The most limiting single active failure used for a small break LOCA is that of an emergency power train failure which results in the loss of one complete train of ECCS components. In addition, a Loss-of-Offsite Power (LOOP) is postulated to occur coincident with reactor trip. This means that credit may be taken for at most one centrifugal charging (CCP) pump, one safety injection (SIP) pump and one residual heat removal (RHR) pump. In the analyses for Comanche Peak Units 1 and 2, one CCP, one SIP and one RHR pump are modeled. The small break LOCA analyses performed for both units model the ECCS flow as being delivered to both the intact and broken loops at the RCS backpressure for breaks smaller than the accumulator line diameter (2 inch – 6 inch breaks) and at 0 psig containment pressure for the accumulator line diameter (8.75-inch breaks). For the accumulator line break, no ECCS flow is assumed in the faulted loop and the delivered flow to the intact loop is conservatively calculated based on spilling assumption to 0 psig containment back pressure. SI flows are provided in Tables 3 and 4 for each scenario.

Prior to break initiation, the plant is in a full power (100.6%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions used in the analysis are given in Table 1. Subsequent to the break opening, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS fluid. The heat transfer between the RCS and the secondary system may be in either direction and is a function of the relative temperatures of the primary and secondary conditions. In the case of continuous heat addition to the secondary 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 1860 psia, is reached. LOOP is postulated to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1715 psia is reached. Safety injection flow is delayed 22 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 loss-of-offsite power coincident with reactor trip, as well as the pump acceleration and valve delays.

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

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

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

During the earlier part of the small break transient (prior to the postulated loss-of-offsite power coincident with reactor trip), the loss of flow through the break is not sufficient to overcome the positive core flow maintained by the reactor coolant pumps. During this period, upward flow through the core is maintained. However, following the reactor coolant pump trip (due to a LOOP) and subsequent pump coastdown, the core mixture level decreases until it reaches the top of hot leg elevation (Figures 2B, 3B, 4B, 5B, 6B, 7B, 8B, 9B, 10B and 11B). 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 7 seconds (consisting of a 2 second signal delay time and a 5 second main feedwater isolation valve stroke time) following the generation of the pressurizer low-pressure SI signal. Additional makeup water is also provided to the secondary using the auxiliary feedwater (AFW) system. An AFW actuation signal is derived from the pressurizer low-pressure SI signal, resulting in the delivery of AFW system flow 60 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.

The steam generators and the break provide the principal heat removal mechanism until the steam generation in the core is sufficient to establish a flow path through the low elevation loop seal (loop seal clearing) and out of the break. This results in two-phase and ultimately all steam flow through the break which then becomes the principal heat removal mechanism. The rate of core level draining is then slowed as vapor is now allowed to enter the hot legs due to the loop seal clearing. Consistent with the NOTRUMP methodology described in Reference 2, only the faulted loop seal is allowed to clear for the cold leg breaks less than 6 inch equivalent diameter (2 – 4 inches) and both the faulted and the intact loop seals were allowed to clear for breaks ≥ 6 inch equivalent diameter (6 and 8.75 inches). When the core mixture level drops below the bottom of the hot legs, the rate of uncovery once again establishes itself. The RCS continues to depressurize and the core level continues to decrease.

For the 3 and 4 inch breaks the top of the core uncovers (Figures 2B, 3B, 5B and 9B) leading to an increase in the core exit vapor temperature (Figures 2F and 3F) and the start of clad heat up (Figures 2G, 3G, 5C and 9C). The peak cladding temperature (PCT) occurs near the time when the core is most deeply uncovered and the top of the core is being cooled by steam only. Minimal or no core uncovery occurs for the 2, 6 and 8.75 inch breaks (Figures 4B, 6B, 7B, 8B, 10B and 11B) and therefore no clad heat up calculation using the SBLOCTA code is performed for these cases. The safety injection flow rate continues to increase as the RCS pressure decreases. The accumulators begin to inject borated water into the reactor coolant loops (Figures 2E and 3E), when the RCS pressure reaches the accumulator setpoint (including uncertainties) of 603 psia. The safety injection replenishes the core level, which results in a reversal in the clad heat up transient (3 and 4 inch breaks) and a steady increase in the core mixture level. Ultimately, the small break transient analysis is terminated when the top of the core is recovered or the core mixture level is increasing and ECCS flow provided to the RCS exceeds the break flow rate.

Acceptance Criteria

The acceptance criteria for the SBLOCA analysis are specified in 10 CFR 50.46, 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. (Note that this criterion is not addressed as part of the short-term SBLOCA analysis.)

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

Results

Tables 5 and 7 provide the NOTRUMP transient timing results for Comanche Peak Units 1 and 2 respectively and Tables 6 and 8 provide the SBLOCTA fuel cladding results for Comanche Peak Units 1 and 2 respectively. The peak cladding temperature is 1013°F for Unit 1 (4 inch break) and 1209°F for Unit 2 (4 inch break), and the maximum local transient oxidation is 0.02 percent for Unit 1 and 0.05 percent for Unit 2. The sum of the pre-transient and transient oxidation remains below 17 percent at all times in life and the average oxidation is negligible for both units.

Limiting Break Case

A summary of the transient response for the limiting PCT cases for Unit 1 and Unit 2 are shown in Figures 2A – 2G and Figures 3A – 3G respectively. These figures present the response of the following parameters.

A. RCS Pressure
B. Core Mixture Level
C. Broken Loop and Intact Loop Pumped Safety Injection Flow Rate

D. Break Flow Rate vs. Total ECCS Flow Rate
E. Broken Loop and Intact Loop Accumulator Flow Rate
F. Core Exit Vapor Temperature
G. Hot Rod Clad Temperature and ZrO_2 Thickness at PCT Elevation

Additional Break Cases

Studies documented in Reference 6 have determined that the limiting PCT small break transient occurs for breaks of less than 10-inches in diameter in the cold leg. For Comanche Peak Units 1 and 2, the limiting PCT is captured by the 2, 3, 4, 6 and 8.75 inch break spectrum. Figures 4 (A and B) through 7 (A and B) provide the RCS pressure and Core mixture level plots for the 2, 3, 6 and 8.75 inch breaks for Unit 1. In addition Figure 5C provides the Hot Rod Clad Temperature and ZrO_2 Thickness at PCT Elevation for the 3 inch break for Unit 1. Figures 8 (A and B) through 11 (A and B) provide the RCS pressure and Core mixture level plots for the 2, 3, 6 and 8.75 inch breaks for Unit 2. In addition Figure 9C provides the Hot Rod Clad Temperature and ZrO_2 Thickness at PCT Elevation for the 3 inch break for Unit 2. The 2, 6 and 8.75 inch cases resulted in minimal or no core uncovery and therefore PCT information was not calculated.

To summarize, the plots for each of the additional non-limiting break cases include:

A. RCS Pressure
B. Core Mixture Level
C. Hot Rod Clad Temperature and ZrO_2 Thickness at PCT Elevation (3 inch break only)

Transient Termination

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

1. The RCS pressure is gradually decreasing.

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.

Conclusions

The small break LOCA analyses considered a break spectrum of 2, 3, 4, and 6 inch equivalent diameter cold leg breaks and a 8.75 inch equivalent diameter accumulator line break for Comanche Peak Units 1 and 2. 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 for small break LOCA below the 2200°F limit of 10 CFR 50.46. Furthermore, the analyses show that the maximum local oxidation (pre-transient plus transient) is less than 17 percent; the core-wide hydrogen generation is less than 1 percent; and the core geometry remains amenable to cooling.

References

1. 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.

2. 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.

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

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

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

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

Table 1 Input Parameters Used in the Small Break LOCA Analysis (Unit 1 and Unit 2)		
	Unit 1	**Unit 2**
A. Core Parameters		
100% Licensed Core Power (MWt)	3612	
Core Power Calorimetric Uncertainty, %	0.6	
Fuel Type	17x17 Vantage +	
Total Core Peaking Factor, F_Q	2.50	
Hot Rod Enthalpy Rise Peaking Factor, F_{DH}	1.60	
Hot Assembly Peaking Factor, P_{HA}	1.4245	
Axial Offset, %	13	
K(z) Limit	2 line segment	
B. Reactor Coolant System		
Thermal Design Flow, gpm/loop	95,700	
Total Core Bypass Flow, %	5.8	
Nominal Vessel Average Temperature Range, °F	Max: 589.2	
	Min: 574.2	
Vessel Average Temperature Uncertainty	± 6°	
Pressurizer Pressure (plus uncertainties), psia	2280	
Reactor Coolant Pump Type	Model 93A 7000HP with Weir	
Reactor Coolant Pump Weir Height, ft	0.4167	
C. Reactor Protection System		
Reactor Trip Setpoint, psia	1860	
Reactor Trip Signal Processing Time (includes Rod Drop Time), sec	4.4	
D. Auxiliary Feedwater System		
Maximum AFW Temperature, °F	120	
AFW Flow (Minimum) to all 4 Steam Generators,	430 (107.5gpm/SG * 4)	
Initiation Signal	Pressurizer Low-Pressure Safety	
AFW Delivery Delay Time, sec	60	
E. Steam Generators		
Steam Generator Secondary Mass, lbm/SG	101,702	92,000
Steam Generator Tube Plugging, %	10	
MFW Isolation Signal	Pressurizer Low-Pressure Safety	
MFW Isolation Delay Time, sec	2	
MFW Flow Coastdown Time, sec	5	
Feedwater Temperature, °F	450.3	
F. Safety Injection		
SI Water Temperature, °F	120	
Safety Injection Signal, psia	1715	
SI Flow Delay Time, sec	22	

Table 1 Input Parameters Used in the Small Break LOCA Analysis (Unit 1 and Unit 2) (cont.)	
G. Accumulators	
Maximum Initial Temperature, °F	120
Initial Water Volume, ft^3	850
Minimum Cover Gas Pressure (including uncertainties), psia	603
H. RWST Draindown Input	
Maximum Containment Spray Flow (2 Trains), gpm	16,000
Minimum Usable RWST Volume, gal	440,300

Table 2 Steam Generator Safety Valve Flows Per Steam Generator (Unit 1 and Unit 2)

MSSV	Set Pressure psig	Uncertainty %	Accumulation %	Rated Flow at Full Open Pressure lbm/hr
1	1185	0	3	893,160
2	1195	0	3	900,607
3	1205	0	3	908,055
4	1215	0	3	915,502
5	1235	0	3	930,397

Table 3 Safety Injection Flows for 2-inch to 6-Inch Break Sizes (Spilling to RCS Pressure) (Unit 1 and Unit 2)		
RCS Pressure (psia)	Injected Flow (lbm/s)	Spilled Flow (lbm/s)
14.7	527.37	199.93
34.7	501.77	190.22
114.7	375.32	142.33
134.7	330.65	125.43
154.7	274.99	104.37
174.7	190.59	72.51
194.7	89.88	31.43
214.7	89.88	31.84
314.7	87.24	31.27
414.7	83.89	30.06
514.7	80.43	28.81
614.7	76.81	27.49
714.7	72.89	26.09
814.7	68.79	24.60
914.7	64.45	23.04
1014.7	59.76	21.32
1114.7	54.37	19.36
1214.7	48.08	17.07
1314.7	40.89	14.45
1414.7	29.56	10.31
1514.7	23.02	7.94
1614.7	21.69	7.47
1714.7	20.26	6.99
1814.7	18.79	6.48
1914.7	17.29	5.96
2014.7	15.72	5.42
2114.7	13.70	4.72
2214.7	11.28	3.88
2314.7	7.45	2.57
2414.7	0.00	0.00

Table 4 Safety Injection Flows for 8.75-Inch Break Size (Spilling to Containment Pressure) (Unit 1 and Unit 2)		
RCS Pressure (psia)	Injected Flow (lbm/s)	Spilled Flow (lbm/s)
14.7	527.63	190.65
34.7	346.45	344.63
54.7	267.12	397.75
74.7	180.32	451.99
94.7	83.82	509.29
114.7	83.17	507.64
134.7	82.49	507.78
214.7	79.81	508.30
314.7	76.37	508.98
414.7	72.80	509.64
514.7	68.97	510.28
614.7	65.01	510.94
714.7	60.83	511.60
814.7	56.43	512.28
914.7	51.70	512.97
1014.7	46.47	513.65
1114.7	40.65	514.38
1214.7	32.94	517.74
1314.7	21.01	518.82
1414.7	19.10	519.57
1514.7	17.74	519.44
1614.7	16.34	519.73
1714.7	14.72	520.02
1814.7	12.89	520.28
1914.7	10.95	520.57
2014.7	8.54	522.85
2114.7	5.89	523.24
2214.7	2.35	523.64
2314.7	0.00	524.00
2414.7	0.00	524.00

Table 5 NOTRUMP Transient Results (Unit 1)

Event (sec)	2-inch(3)	3-inch	4-inch	6-inch(3)	8.75-inch(3)
Break Initiated	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	95.94	20.92	12.17	7.56	6.28
Safety Injection Signal	105.75	30.60	21.25	14.56	8.61
Safety Injection Begins(1)	127.75	52.60	43.25	36.56	30.61
Loop Seal Clearing Occurs(2)	1370	~535	~315	70, 135	~22
Core Uncovery	N/A	~955	~715	~440	N/A
Accumulator Injection Begins	N/A	N/A(4)	~970	~405	~200
Core Recovery	N/A	~3000	~1520	~485	N/A
RWST Low Level(5)	1608.5	1600.19	1586.95	1574.98	1474.99

Notes:

1. Safety injection begins 22.0 seconds (SI delay time) after the safety injection signal is
2. Loop seal clearing is considered to occur when the broken loop loop seal vapor flow rate is sustained above 1 lbm/s.
3. There is no core uncovery for the 2-inch and 8.75-inch cases, and only minimal core uncovery for the 6-inch case.
4. Accumulator actually injects at 3485 for TBX, but this is after the core has recovered and has no bearing on the LOCA recovery.
5. The analysis assumes minimum usable RWST volume (440,300 gal) before the low-1 RWST water level signal for switchover to cold leg recirculation is reached.

Table 6 SBLOCTA BOL Results (Unit 1)

Result	2-inch(2)	3-inch	4-inch	6-inch(2)	8.75-inch(2)
PCT, °F	N/A	949.6	1012.8	N/A	N/A
PCT Time, sec		2026.1	1058.3		
PCT Elevation, ft		10.75	11.00		
HR Burst Time(1), sec		N/A	N/A		
HR Burst Elevation(1), ft		N/A	N/A		
Max. Local Transient ZrO_2, %		0.02	0.01		
Max. Local ZrO_2 Elevation, ft		10.75	11.00		
Average ZrO_2, %		0.00	0.00		

Notes:

1. None of the hot rods nor the hot assembly rods burst during the SBLOCTA calculations.
2. The core either does not uncover or only uncovers for a very short time and therefore does not warrant SBLOCTA calculations for these break sizes.

Table 7 NOTRUMP Transient Results (Unit 2)					
Event (sec)	**2-inch[3]**	**3-inch**	**4-inch**	**6-inch[3]**	**8.75-inch[3]**
Break Initiated	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	101.81	35.60	12.52	7.79	6.19
Safety Injection Signal	111.82	44.53	21.99	14.81	8.58
Safety Injection Begins[1]	133.82	66.53	43.99	36.81	30.58
Loop Seal Clearing Occurs[2]	1140	510	255	135	~17
Core Uncovery	N/A	~670	~575	~420	N/A
Accumulator Injection Begins	N/A	N/A[4]	~850	~380	~170
Core Recovery	N/A	~2800	~1630	~470	N/A
RWST Low Level[5]	1609.1	1599.10	1585.37	1574.83	1474.55

Notes:

1. Safety injection begins 22.0 seconds (SI delay time) after the safety injection signal is
2. Loop seal clearing is considered to occur when the broken loop loop seal vapor flow rate is sustained above 1 lbm/s.
3. There is no core uncovery for the 2-inch and 8.75-inch cases, and only minimal core uncovery for the 6-inch case.
4. Accumulator actually injects at ~3330 sec for TCX, but this is after the core has recovered and has no bearing on the LOCA recovery.

Table 8 SBLOCTA BOL Results (Unit 2)					
Result	**2-inch[2]**	**3-inch**	**4-inch**	**6-inch[2]**	**8.75-inch[2]**
PCT, °F	N/A	1068.5	1209.4	N/A	N/A
PCT Time, sec		1787.9	919.7		
PCT Elevation, ft		11.0	11.0		
HR Burst Time[1], sec		N/A	N/A		
HR Burst Elevation[1], ft		N/A	N/A		
Max. Local Transient ZrO_2, %		0.04	0.06		
Max. Local ZrO_2 Elevation, ft		11.0	11.0		
Average ZrO_2, %		0.01	0.01		

Notes:

1. None of the hot rods nor the hot assembly rods burst during the SBLOCTA calculations.
2. The core either does not uncover or only uncovers for a very short time and therefore does not warrant SBLOCTA calculations for these break sizes.

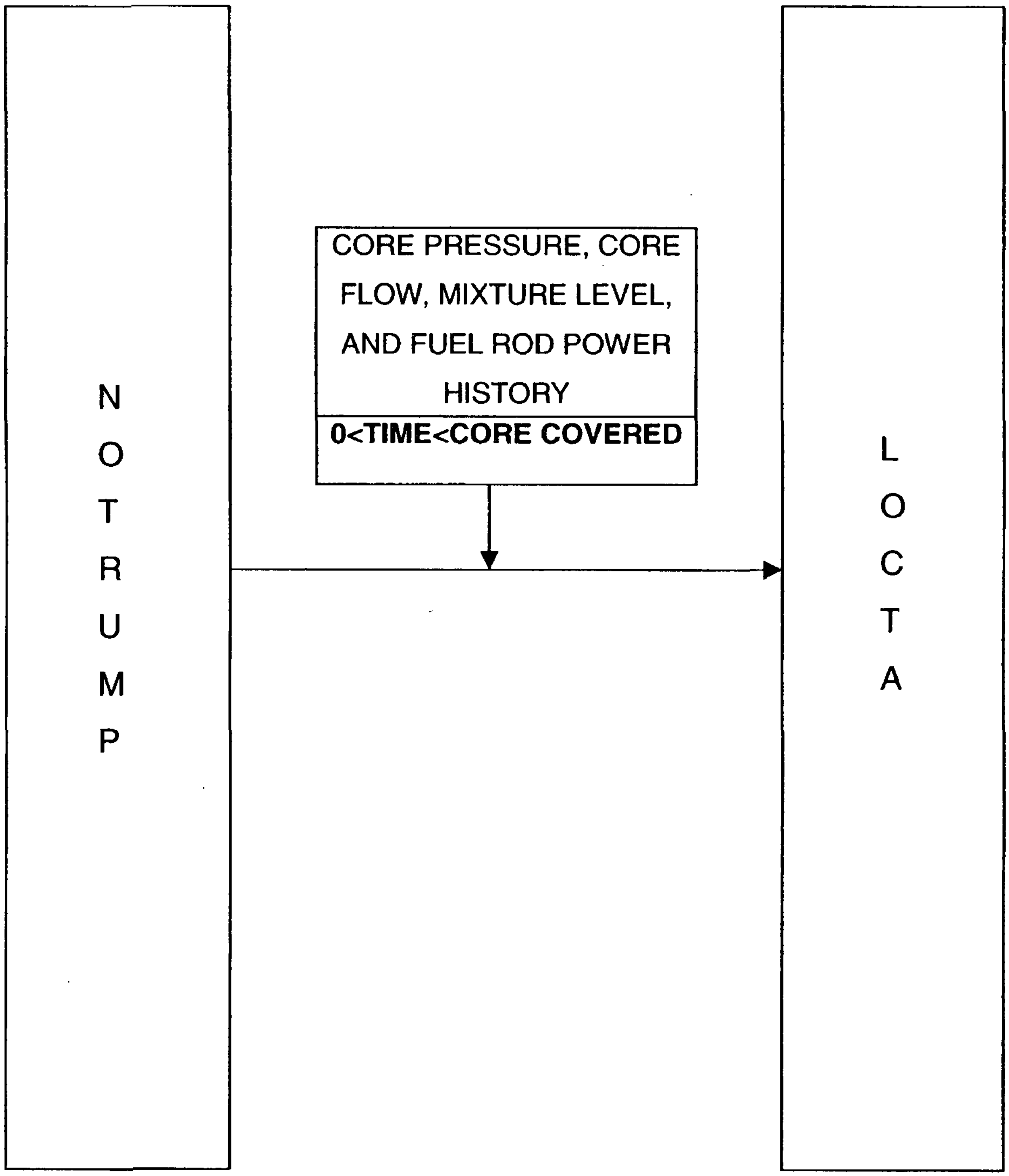

Figure 1 Code Interface Description for NOTRUMP-EM

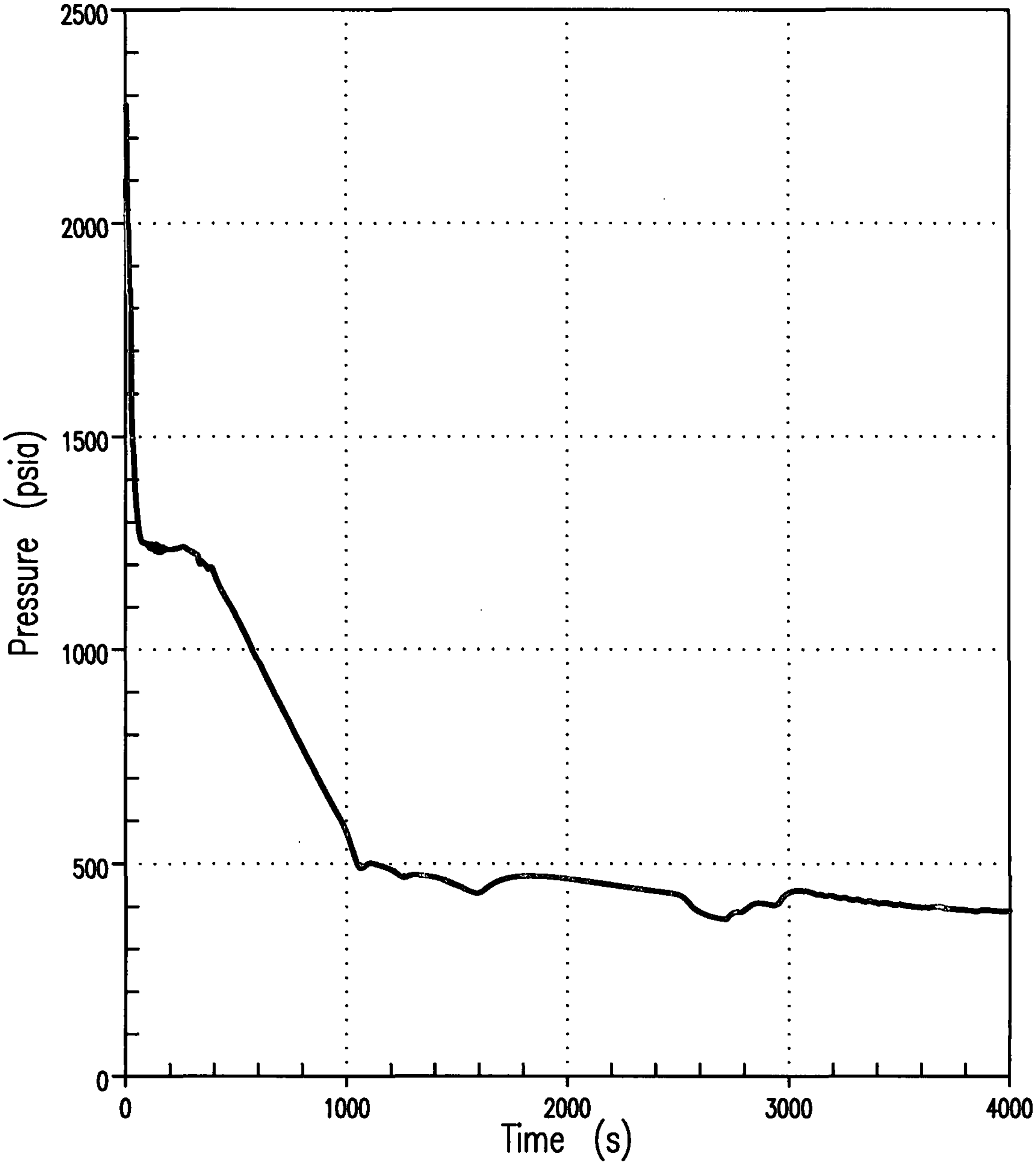

Figure 2A RCS Pressure 4-Inch Break (Unit 1)

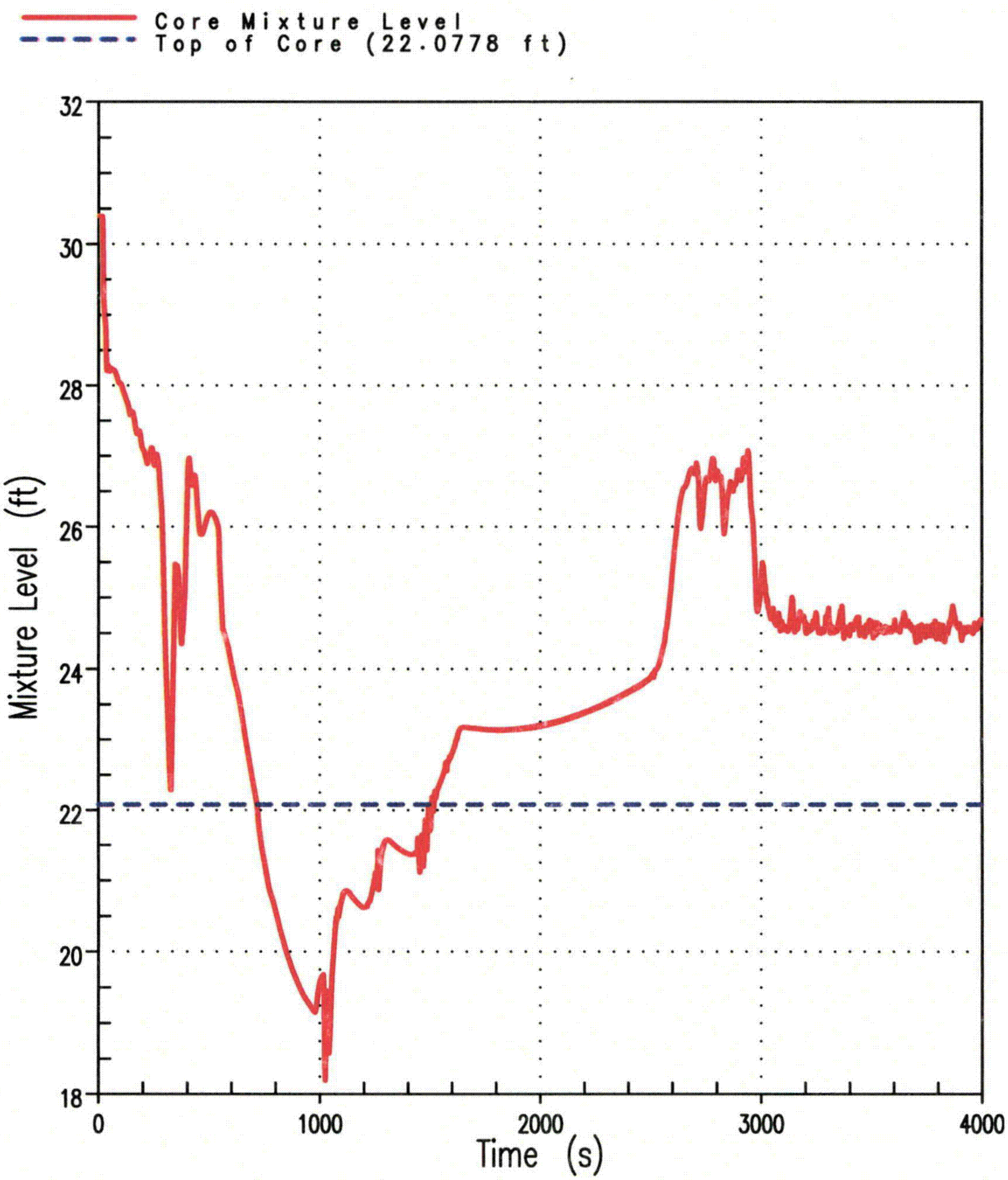

Figure 2B Core Mixture Level 4-Inch Break (Unit 1)

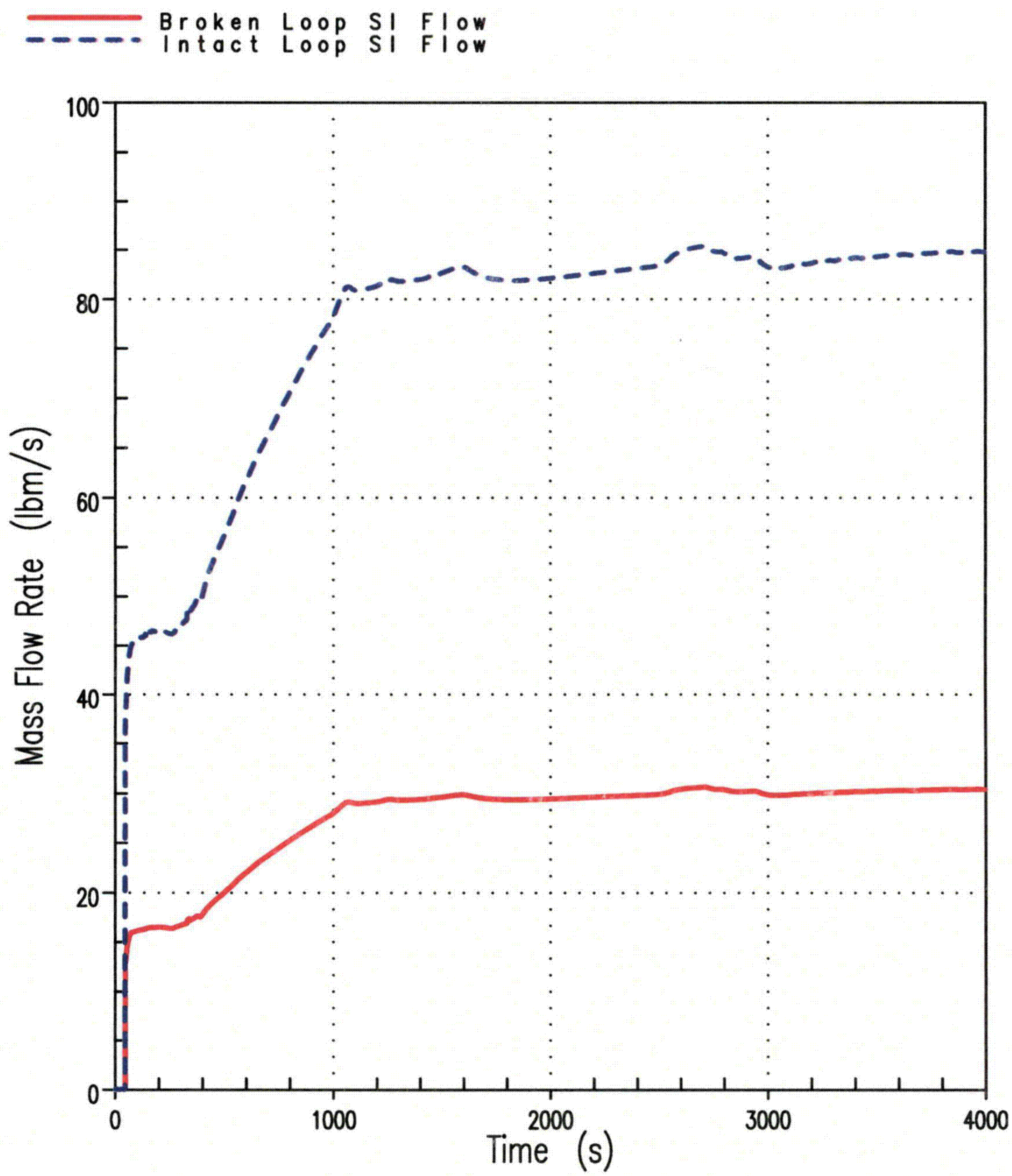

Figure 2C Broken Loop and Intact Loop Pumped Safety Injection Flow Rate 4-Inch Break (Unit 1)

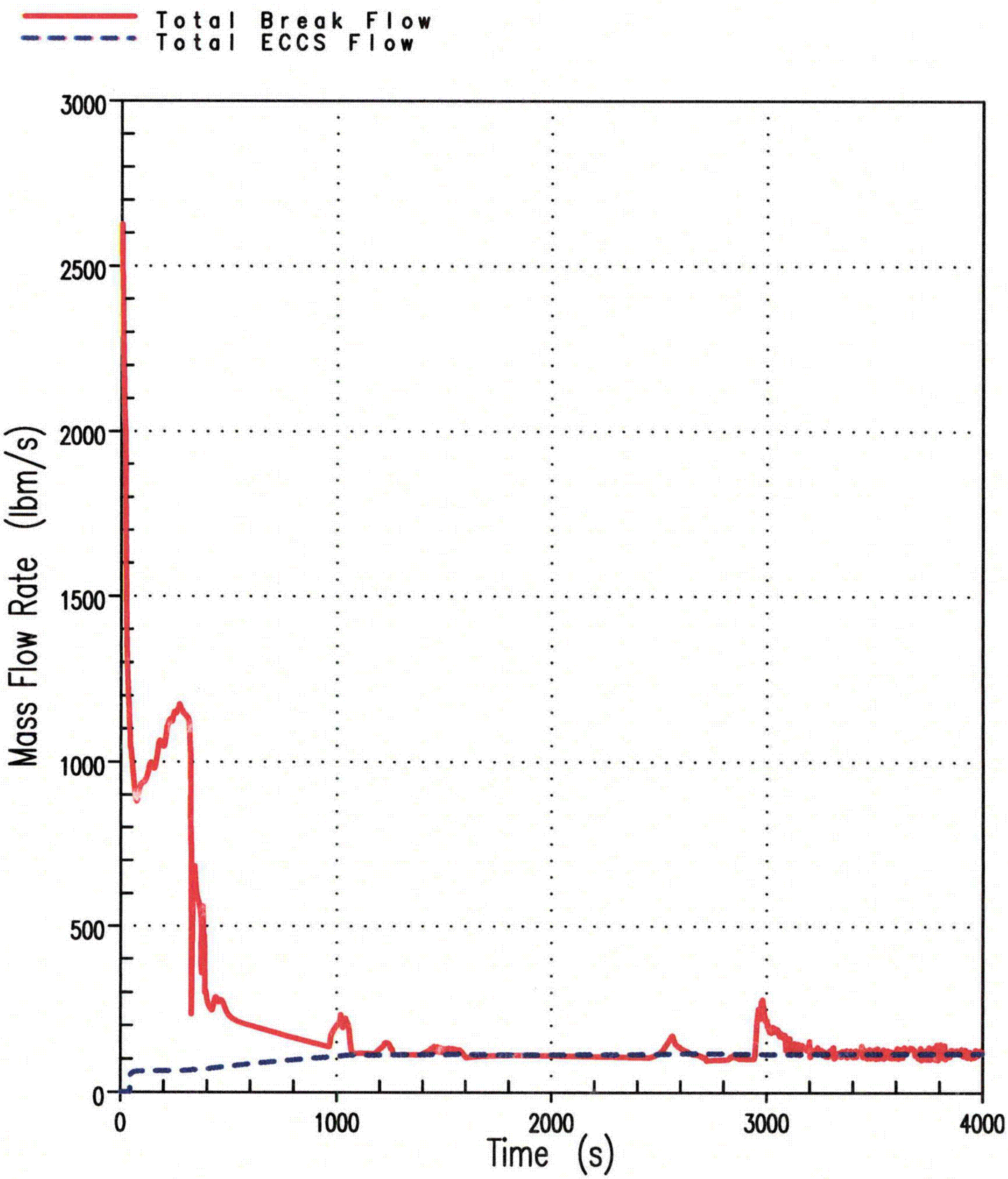

Figure 2D Break Flow Rate vs. Total ECCS Flow Rate
4-Inch Break (Unit 1)

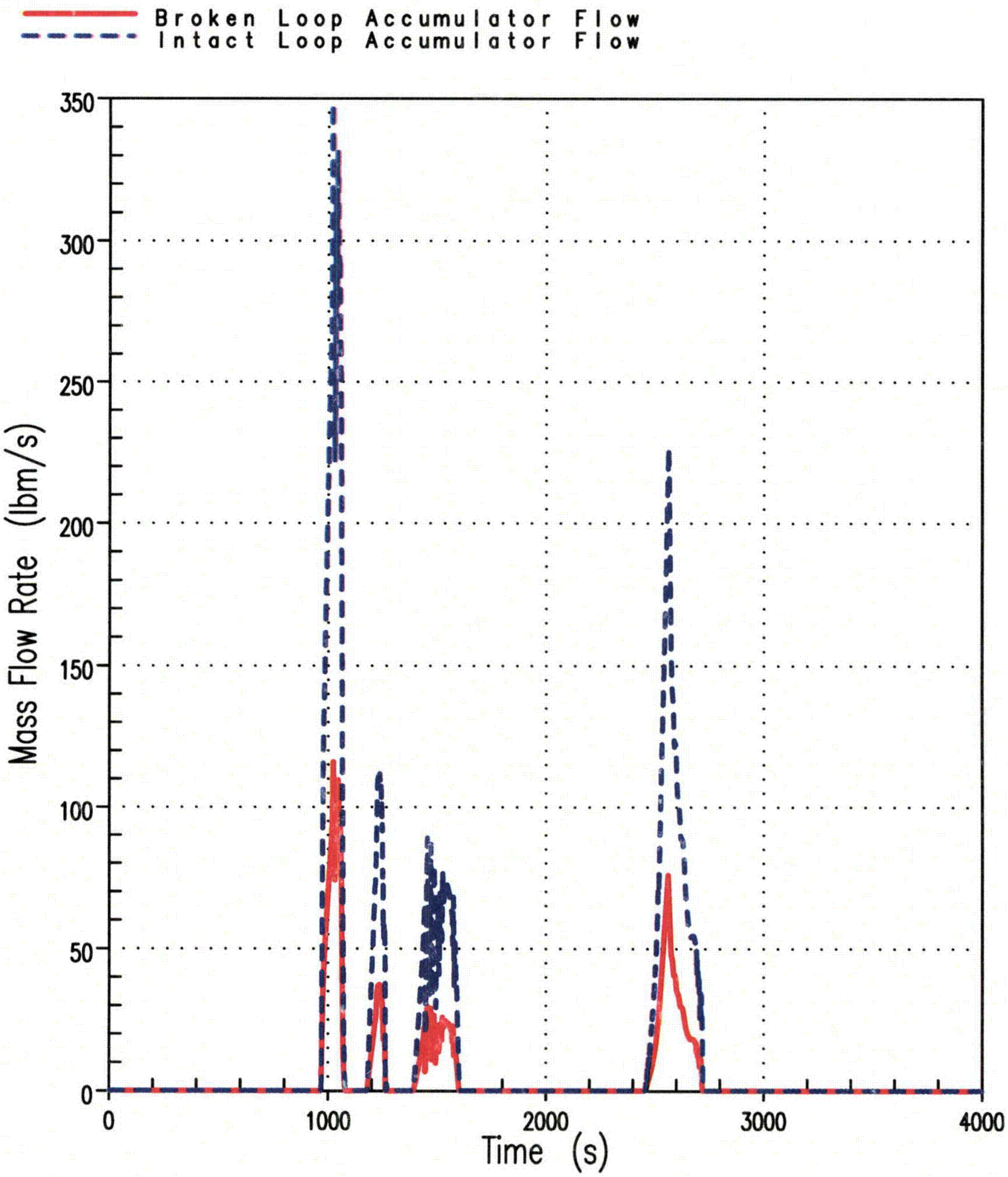

Figure 2E Broken Loop and Intact Loop Accumulator Flow Rate
4-Inch Break (Unit 1)

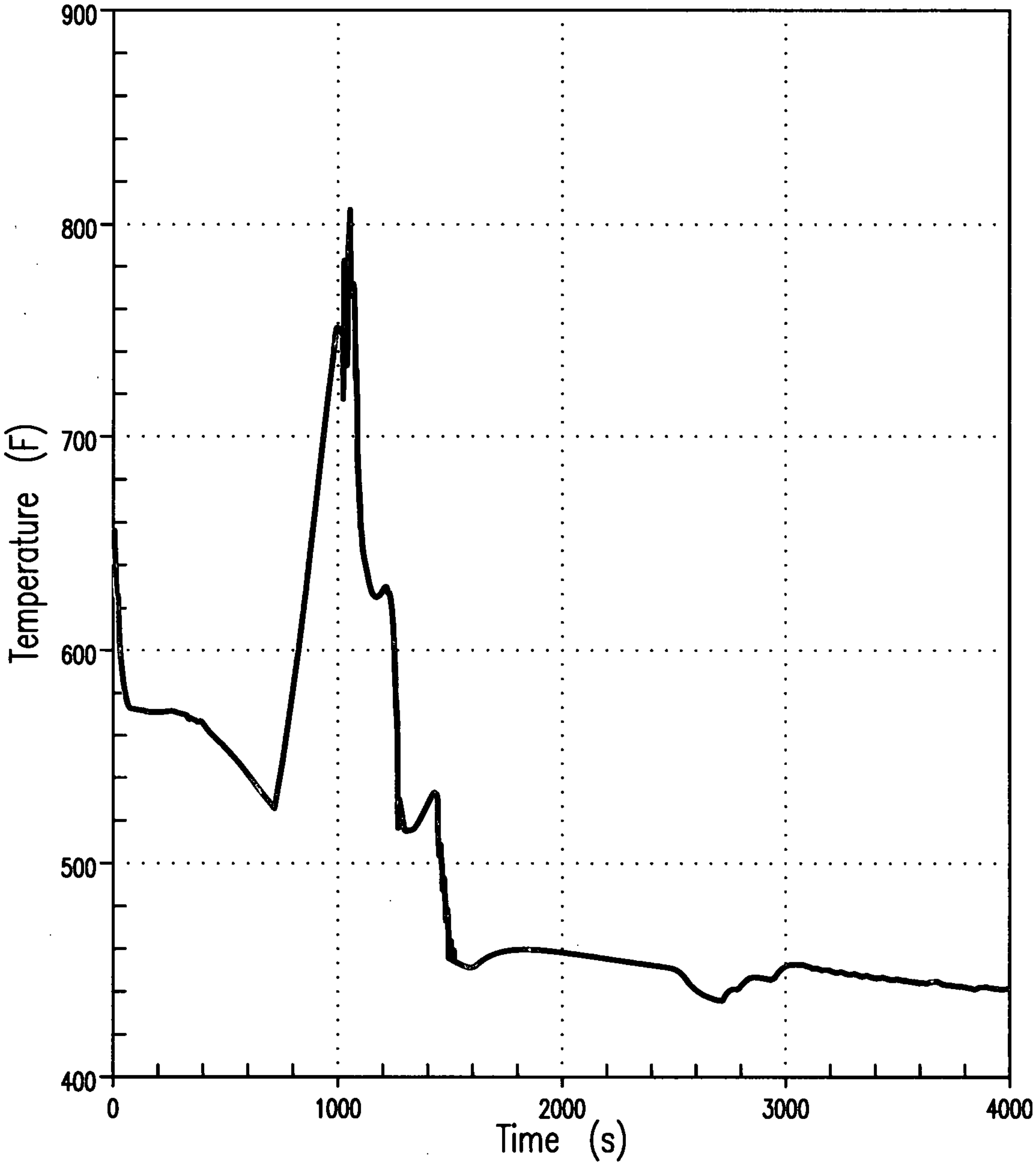

Figure 2F Core Exit Vapor Temperature 4-Inch Break (Unit 1)

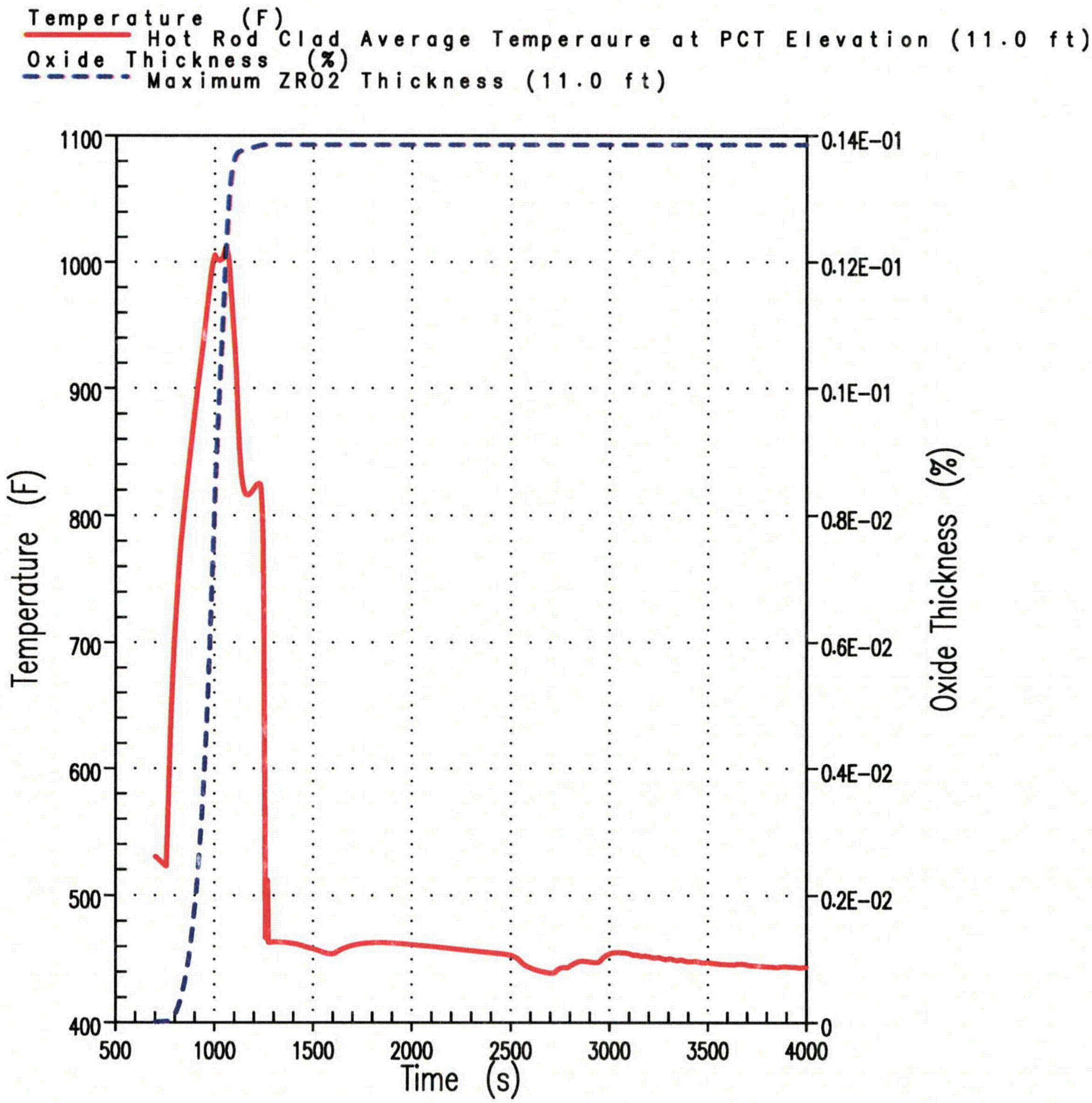

Figure 2G Hot Rod Clad Temperature and ZRO2 Thickness at PCT Elevation 4-Inch Break (Unit 1)

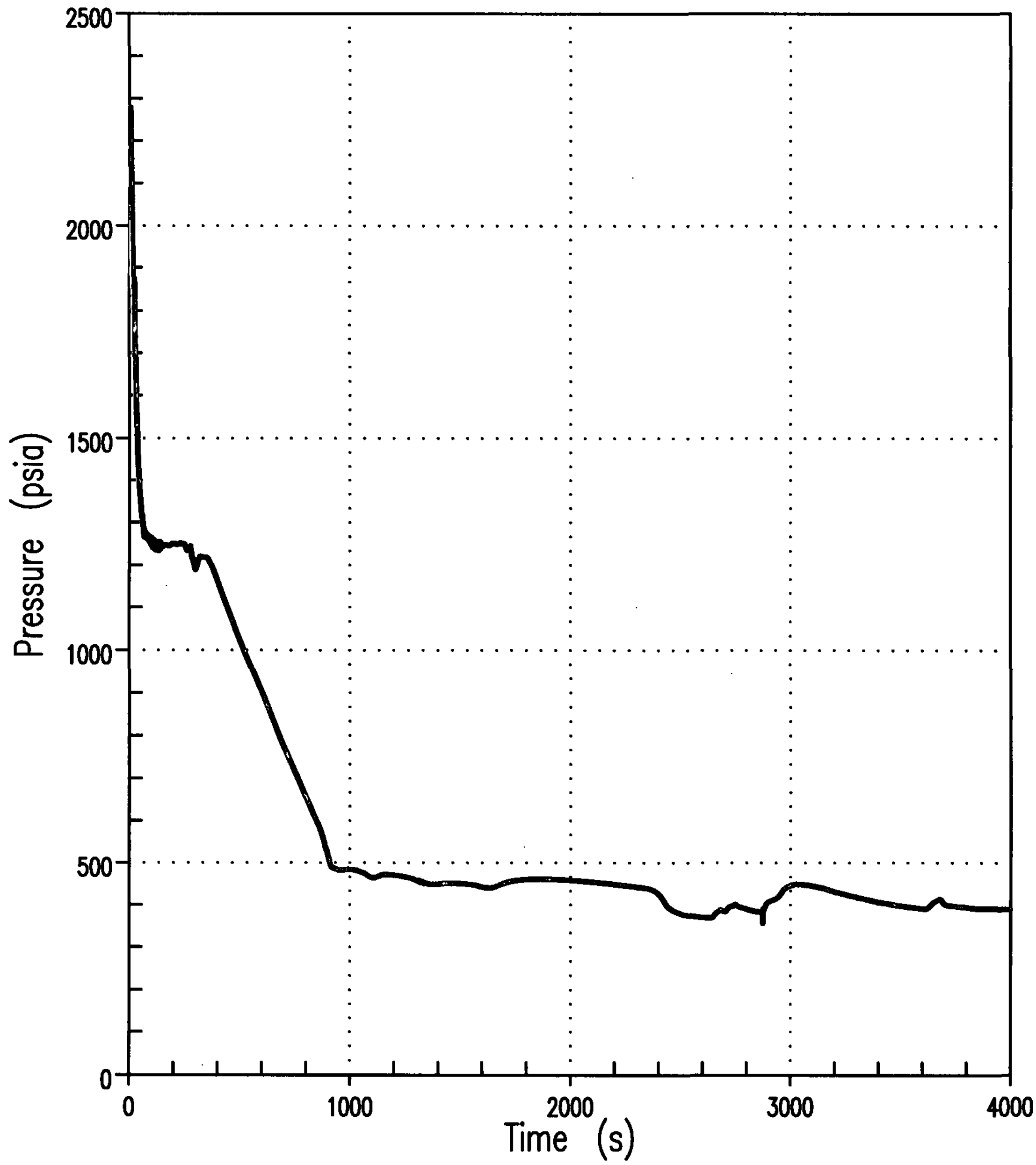

Figure 3A RCS Pressure 4-Inch Break (Unit 2)

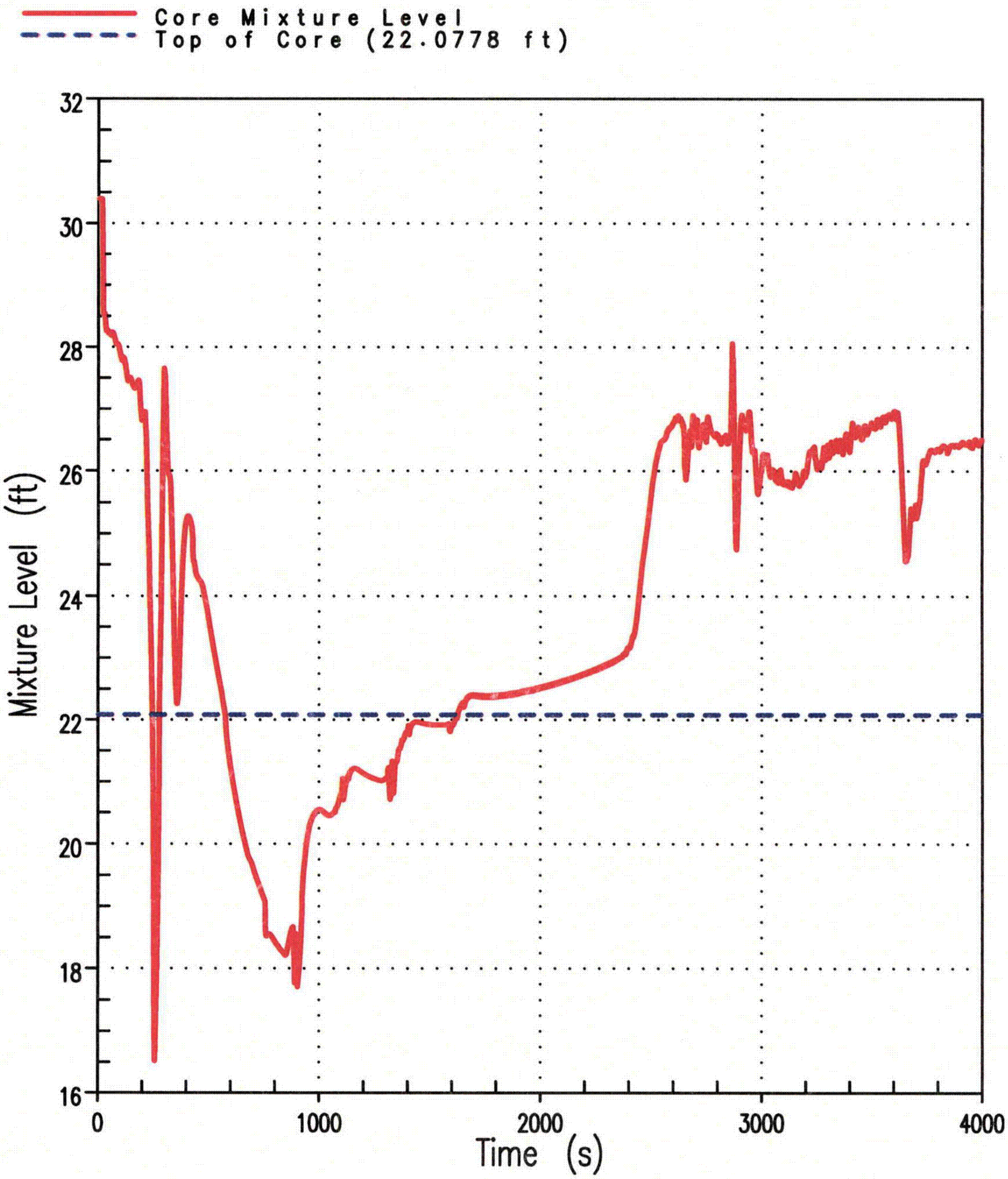

Figure 3B Core Mixture Level 4-Inch Break (Unit 2)

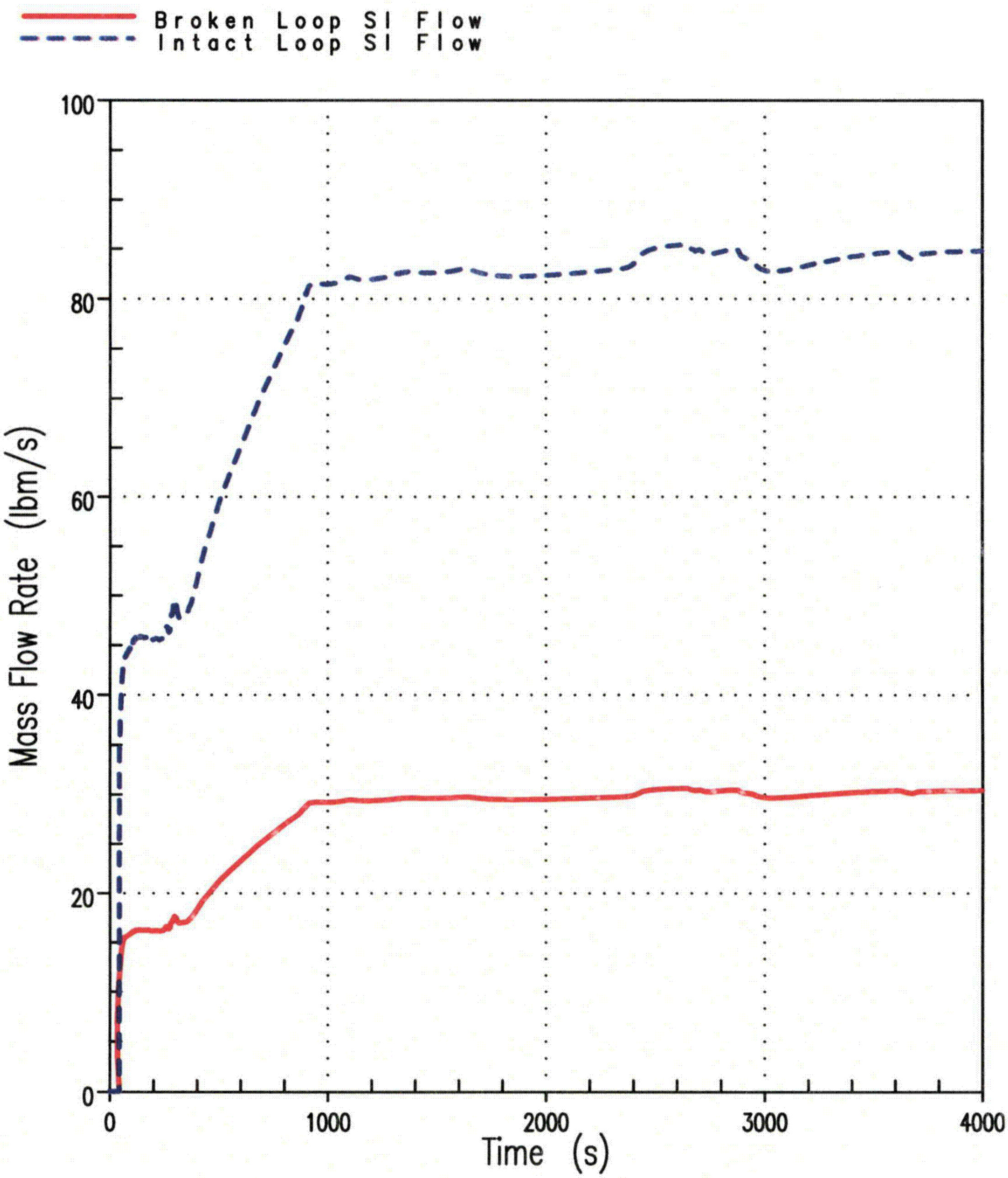

Figure 3C Broken Loop and Intact Loop Pumped Safety Injection Flow Rate 4-Inch Break (Unit 2)

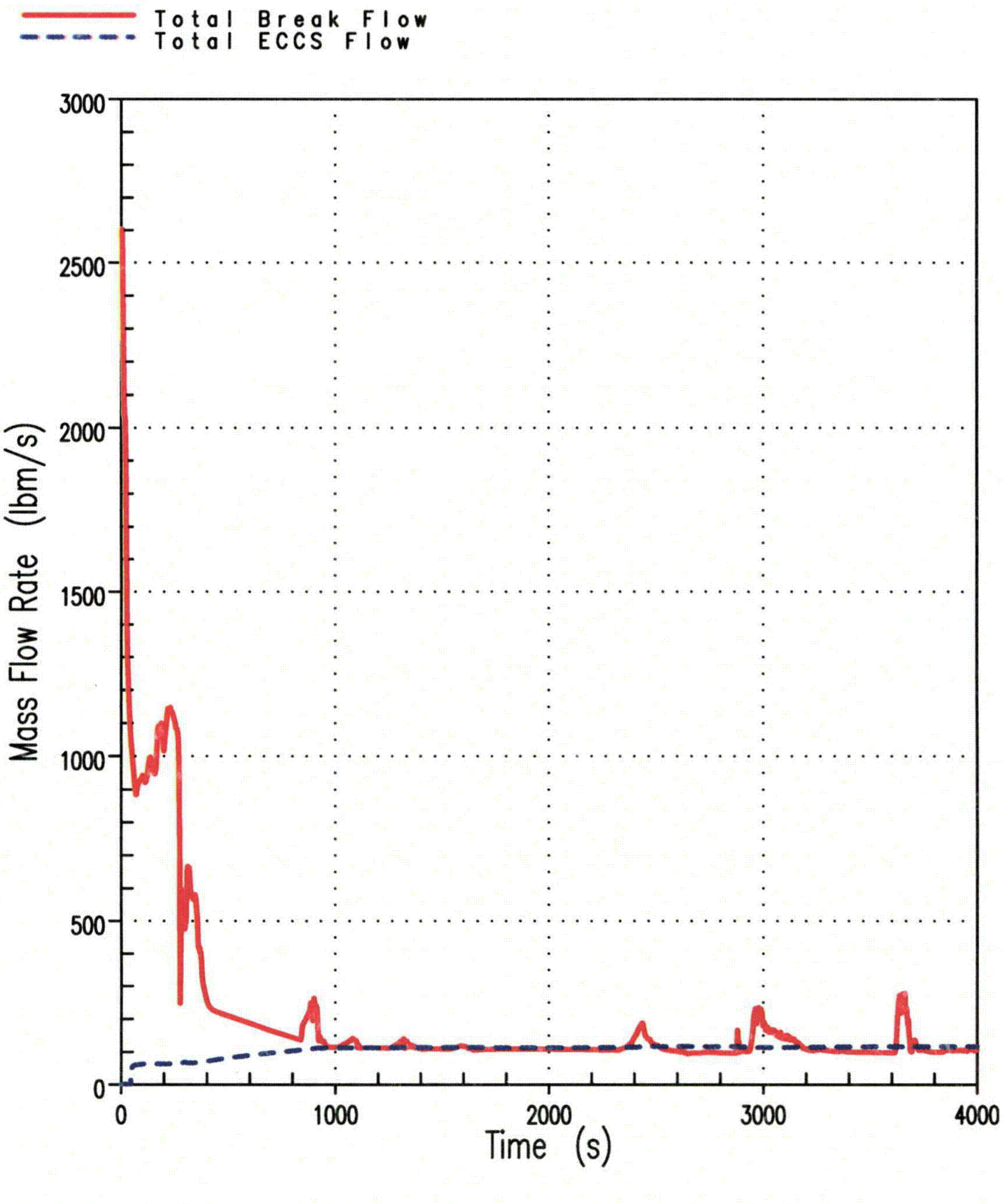

Figure 3D Break Flow Rate vs. Total ECCS Flow Rate
4-Inch Break (Unit 2)

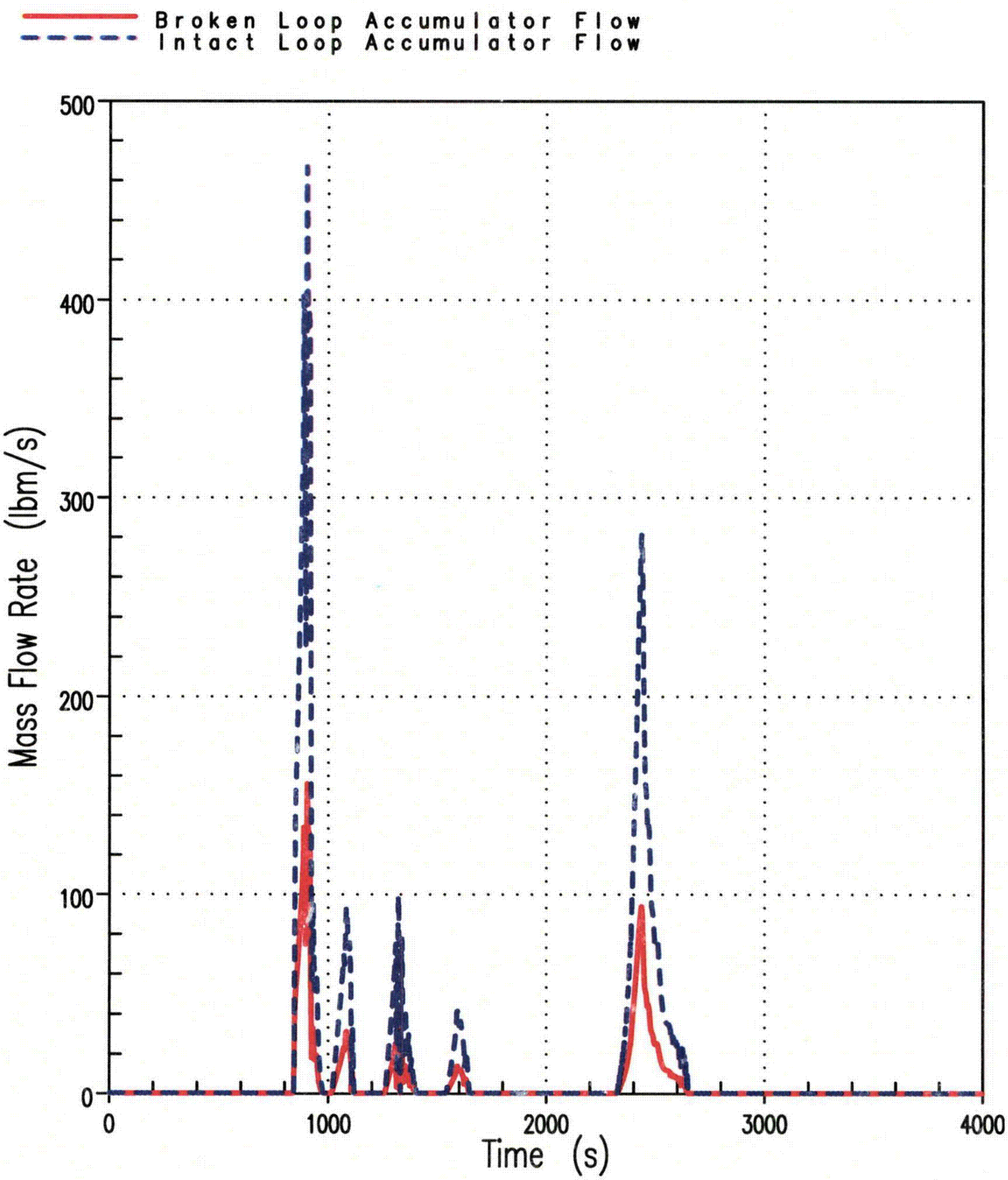

**Figure 3E Broken Loop and Intact Loop Accumulator Flow Rate
4-Inch Break (Unit 2)**

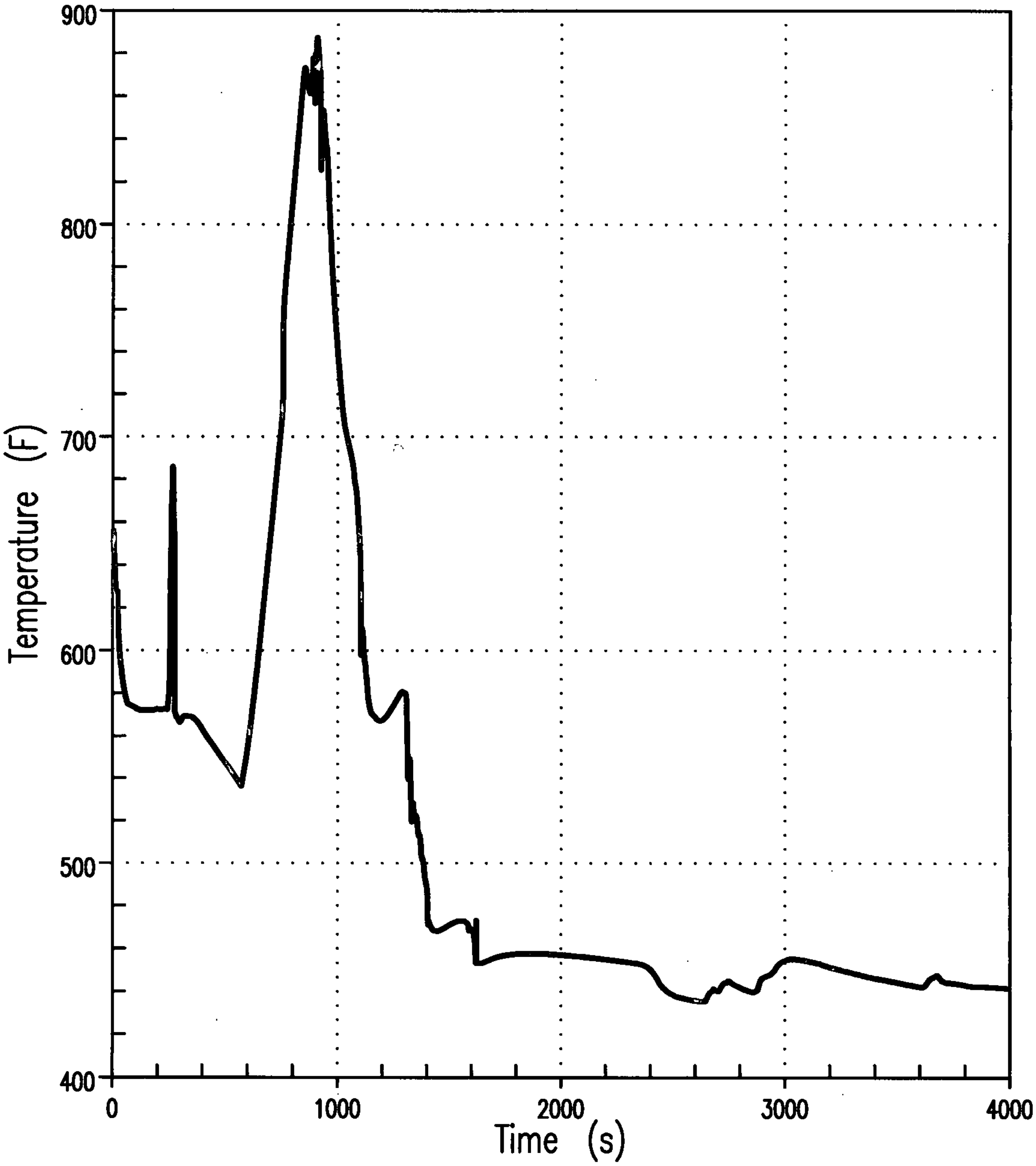

Figure 3F Core Exit Vapor Temperature 4-Inch Break (Unit 2)

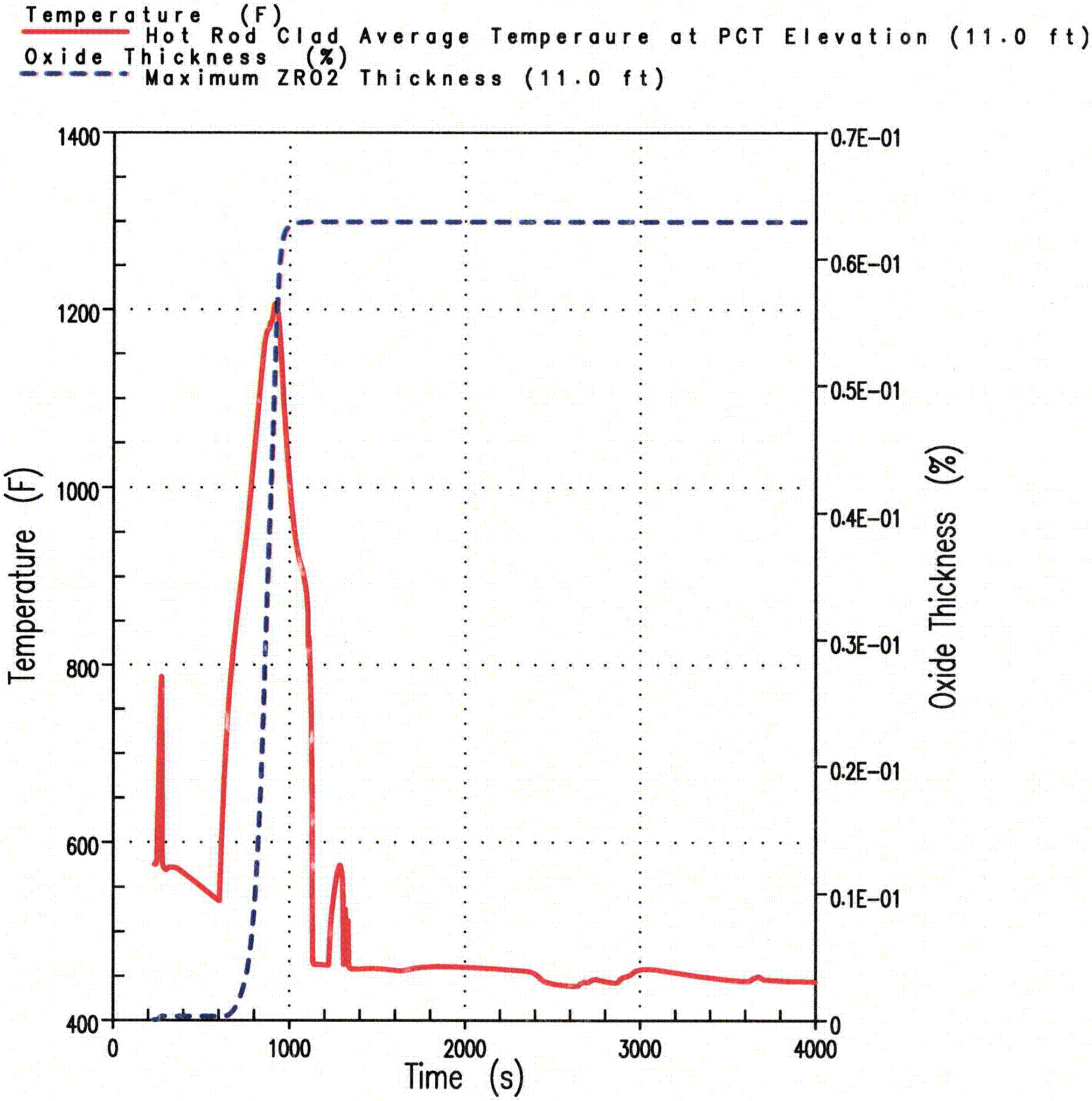

Figure 3G Hot Rod Clad Temperature and ZrO_2 Thickness at PCT Elevation 4-Inch Break (Unit 2)

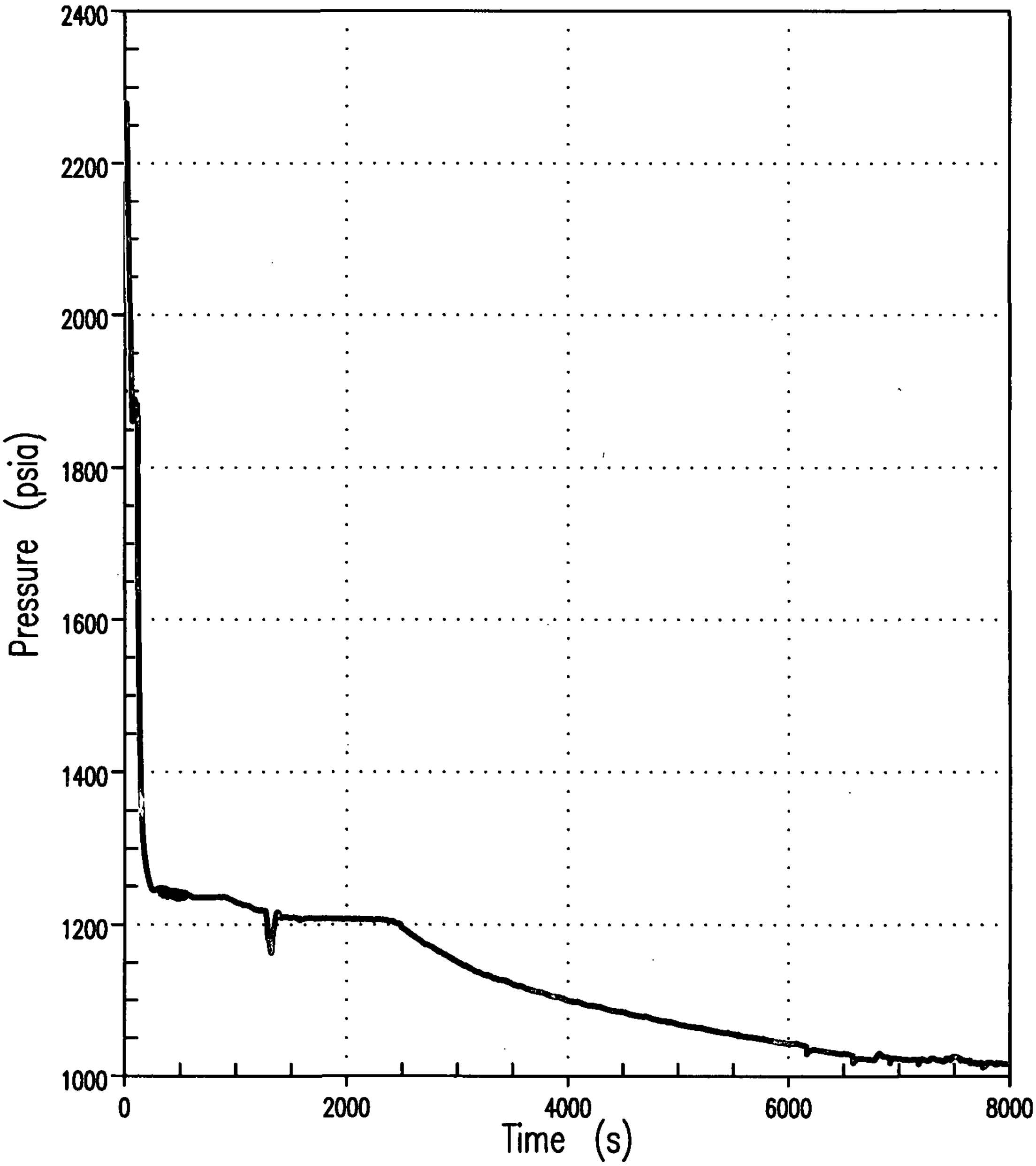

Figure 4A RCS Pressure 2-Inch Break (Unit 1)

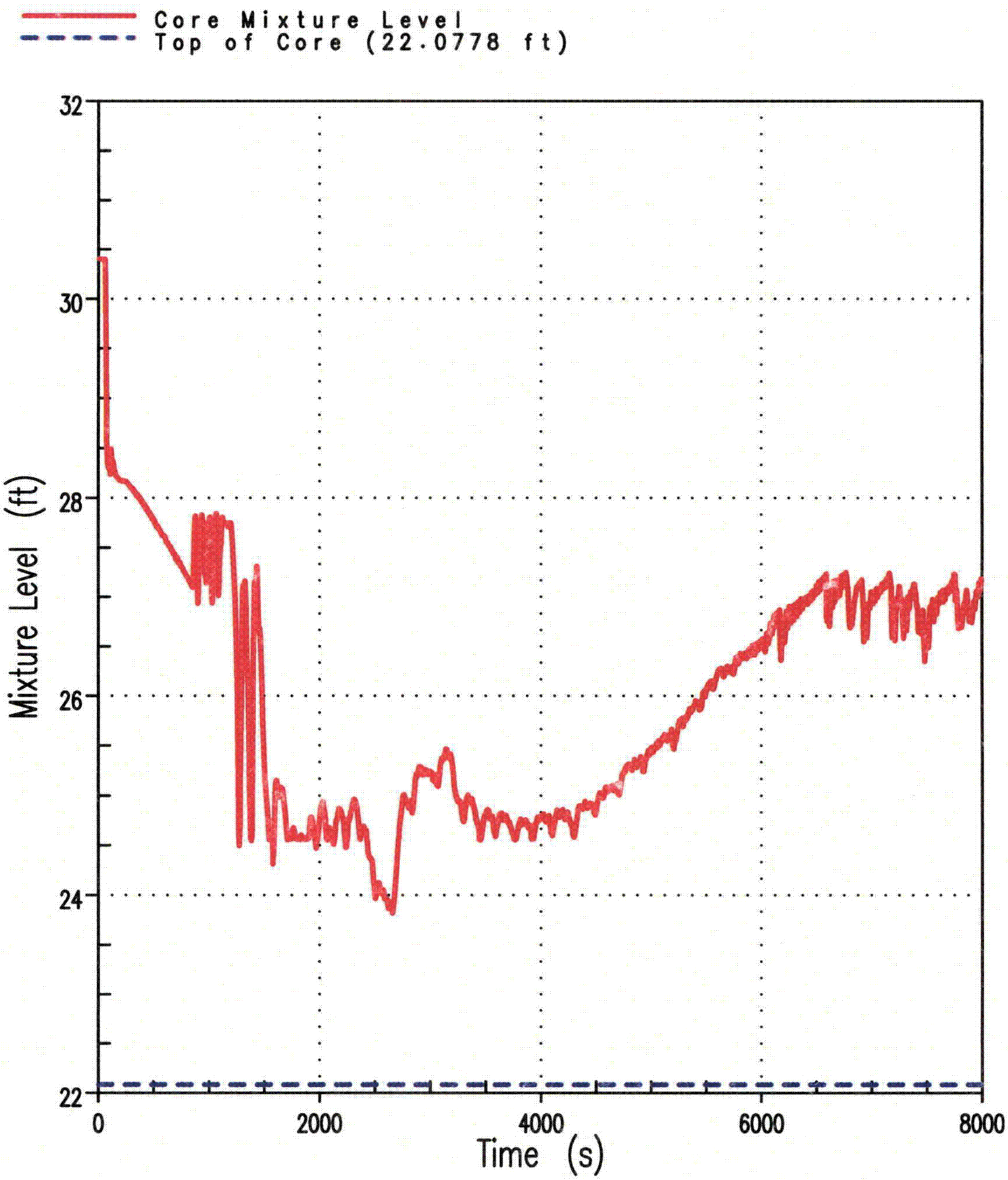

Figure 4B Core Mixture Level 2-Inch Break (Unit 1)

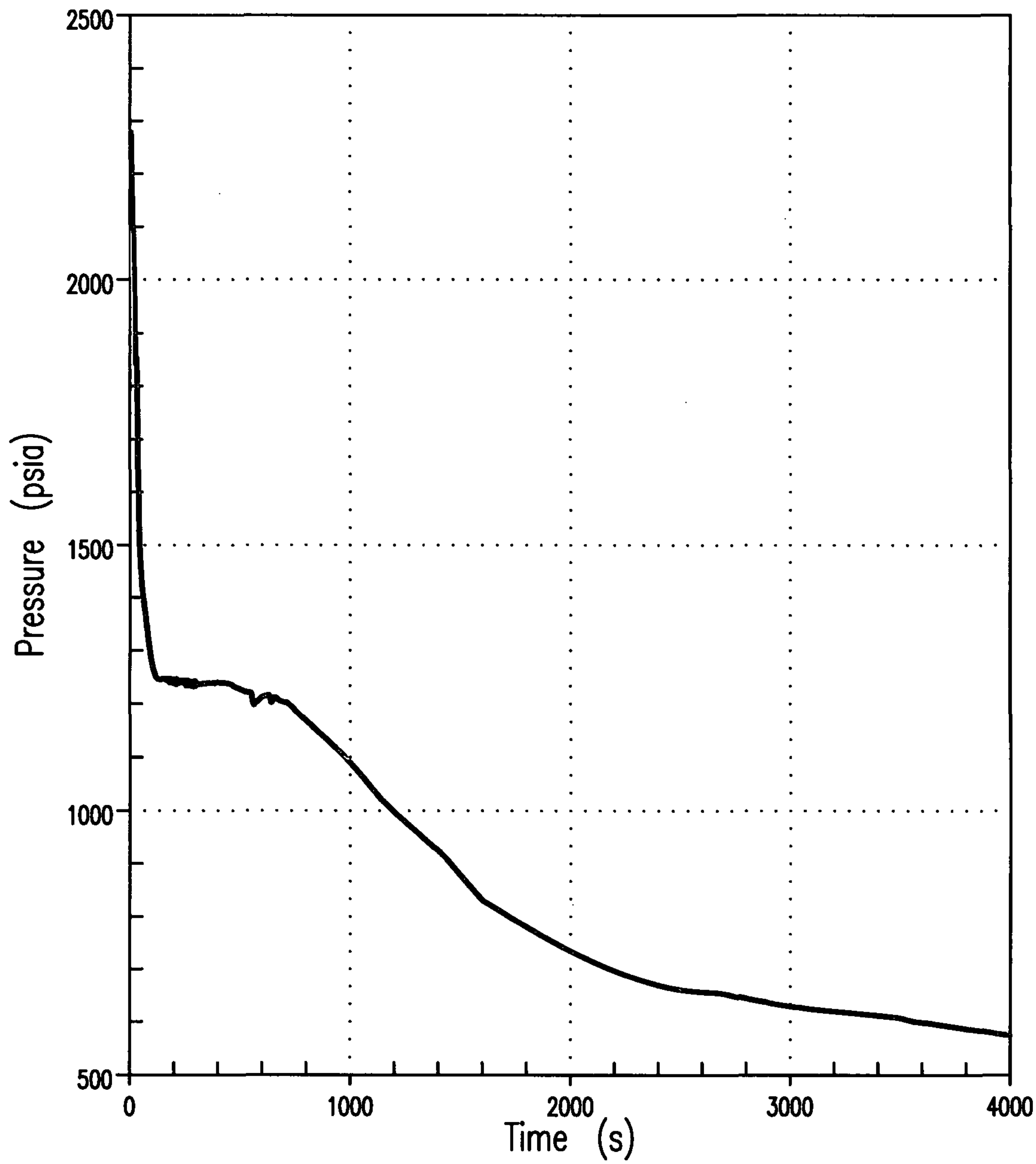

Figure 5A RCS Pressure 3-Inch Break (Unit 1)

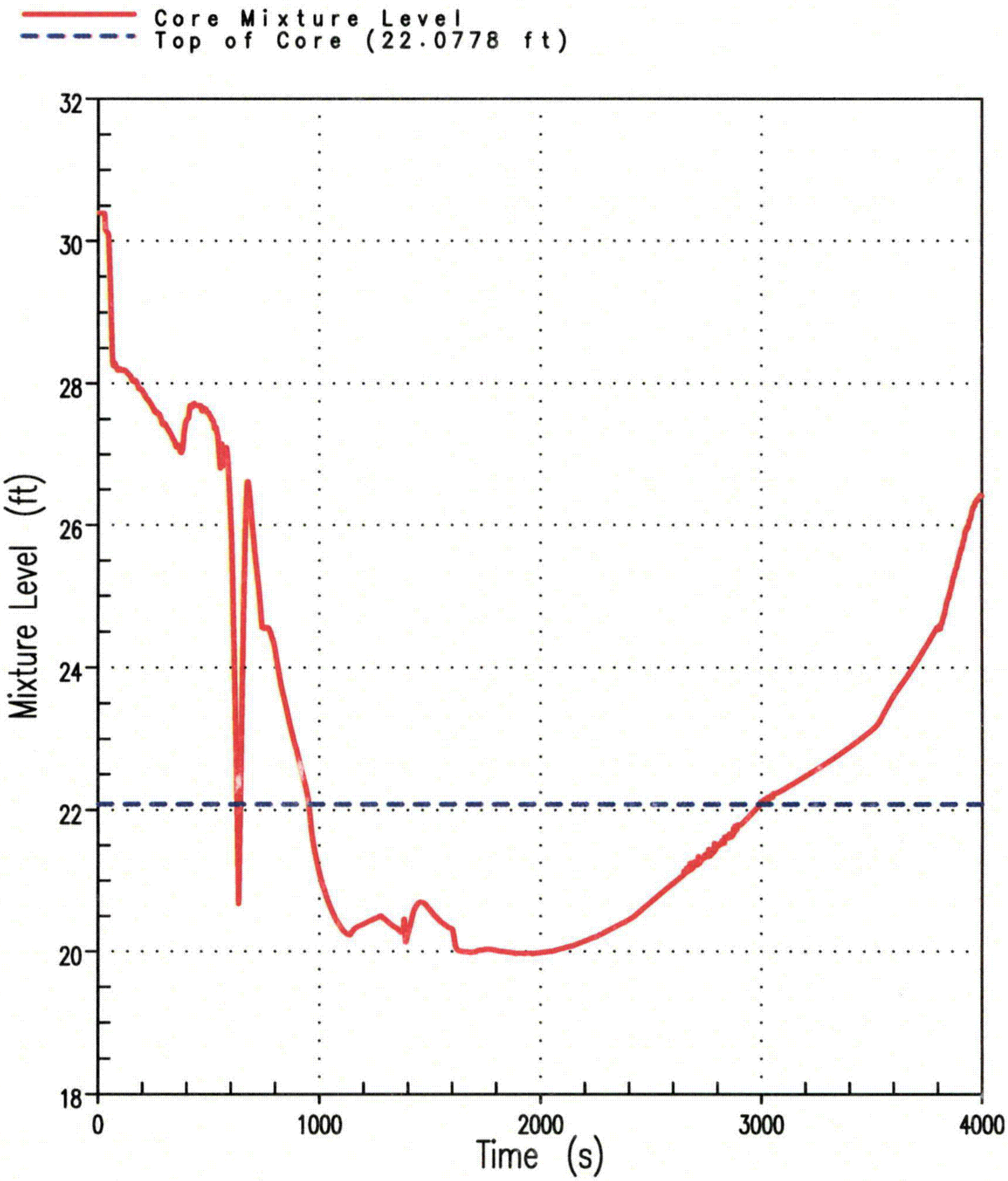

Figure 5B Core Mixture Level 3-Inch Break (Unit 1)

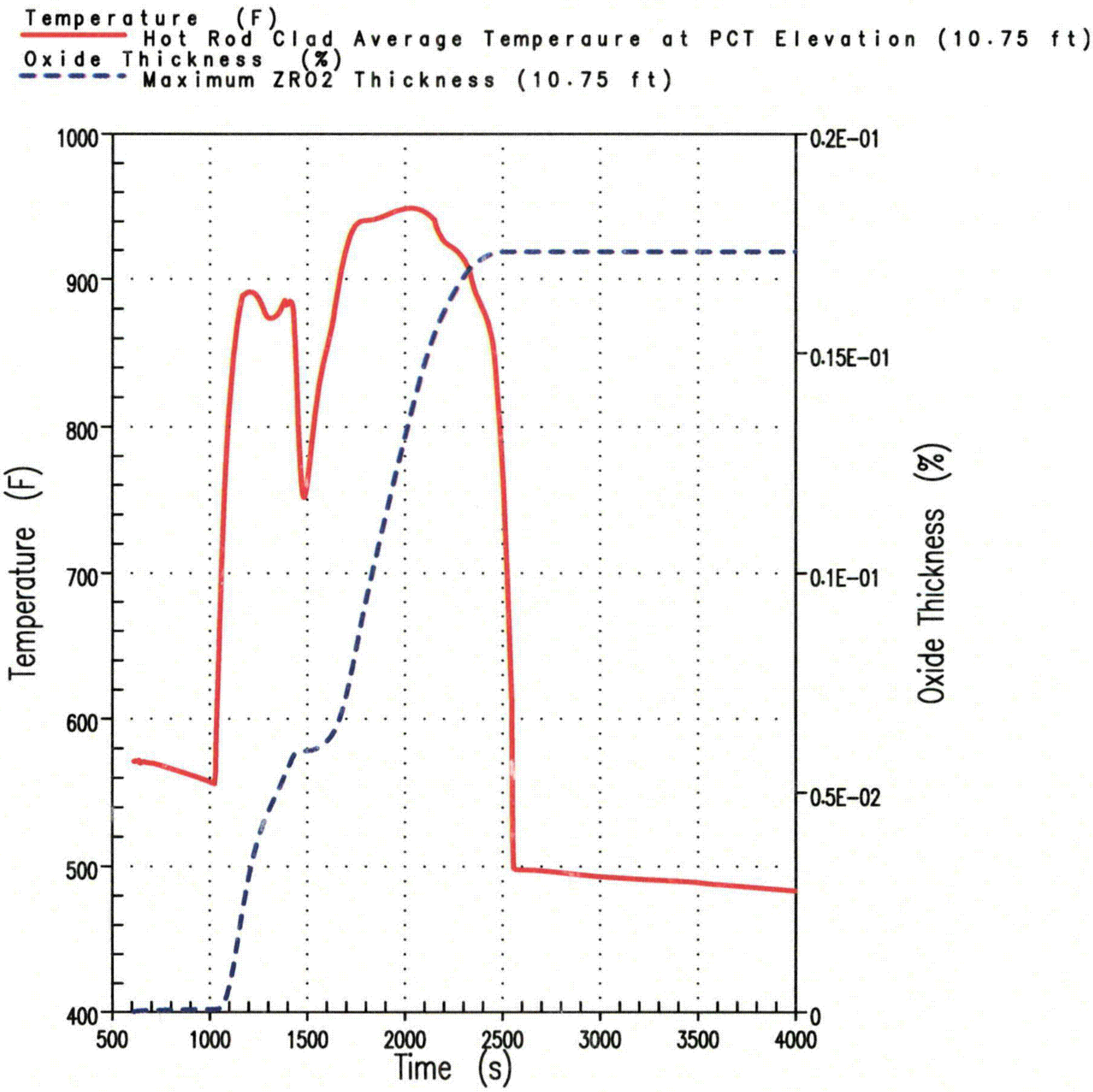

Figure 5C Hot Rod Clad Temperature and ZrO_2 Thickness at PCT Elevation 3-Inch Break (Unit 1)

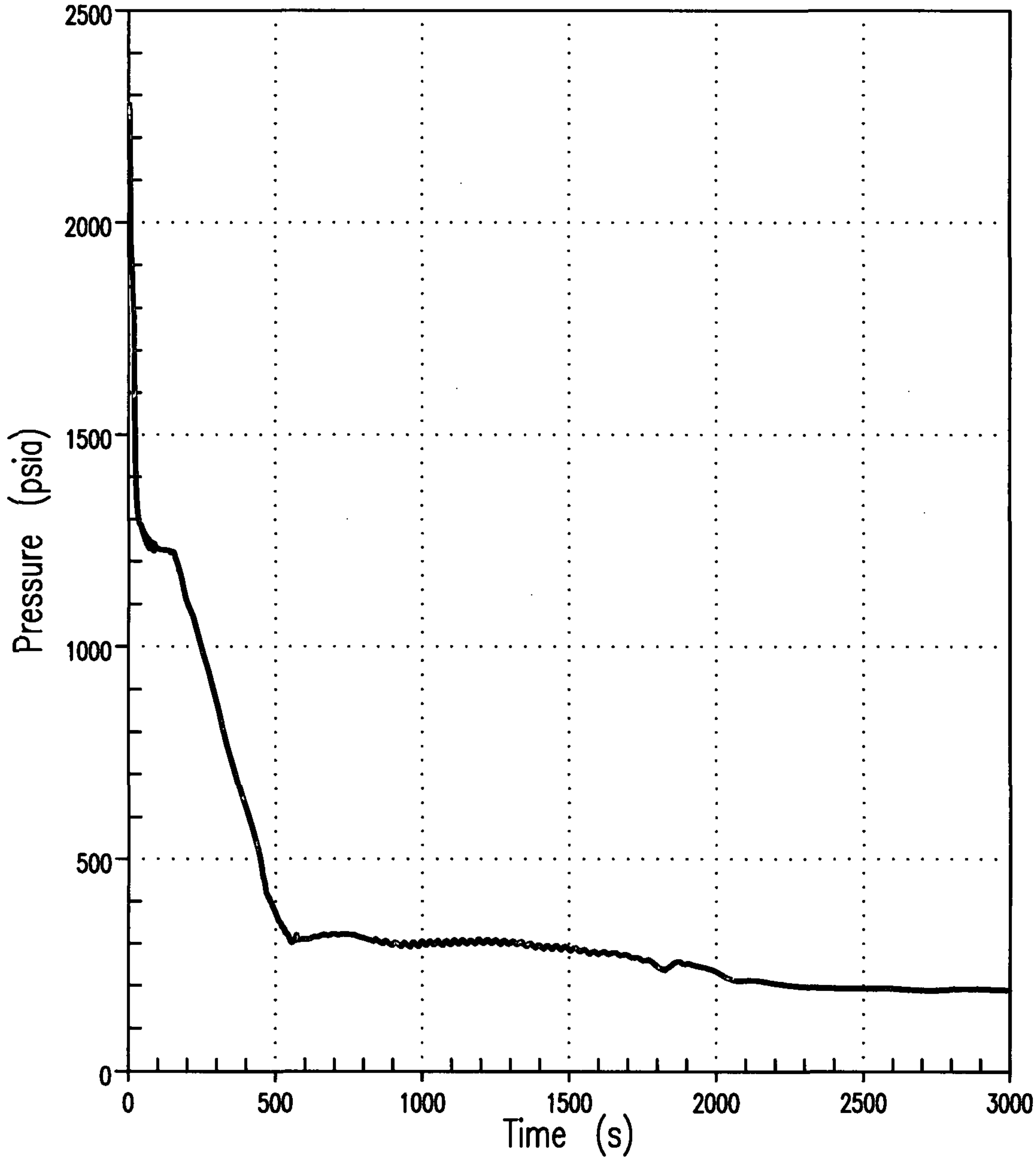

Figure 6A RCS Pressure 6-Inch Break (Unit 1)

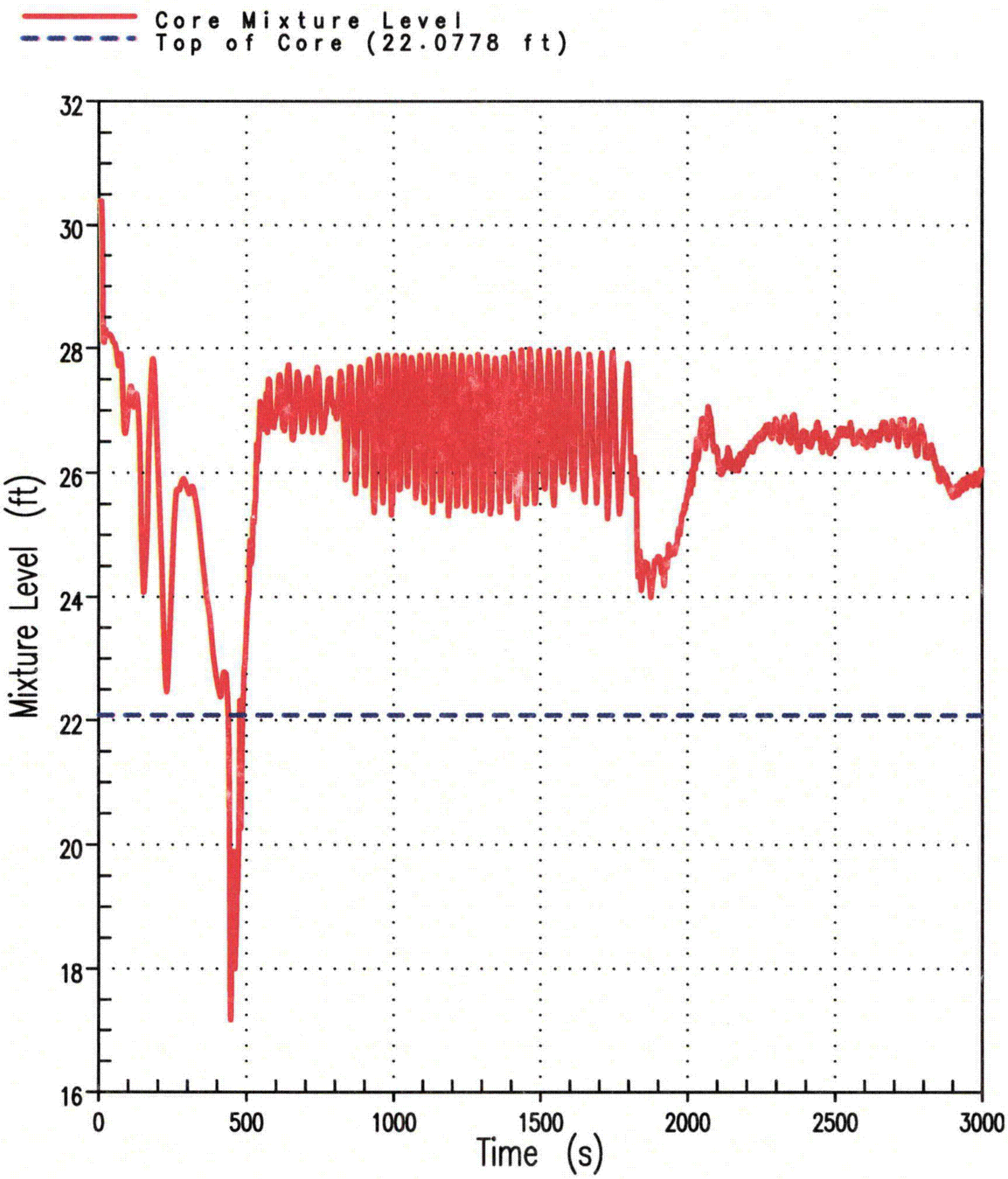

Figure 6B Core Mixture Level 6-Inch Break (Unit 1)

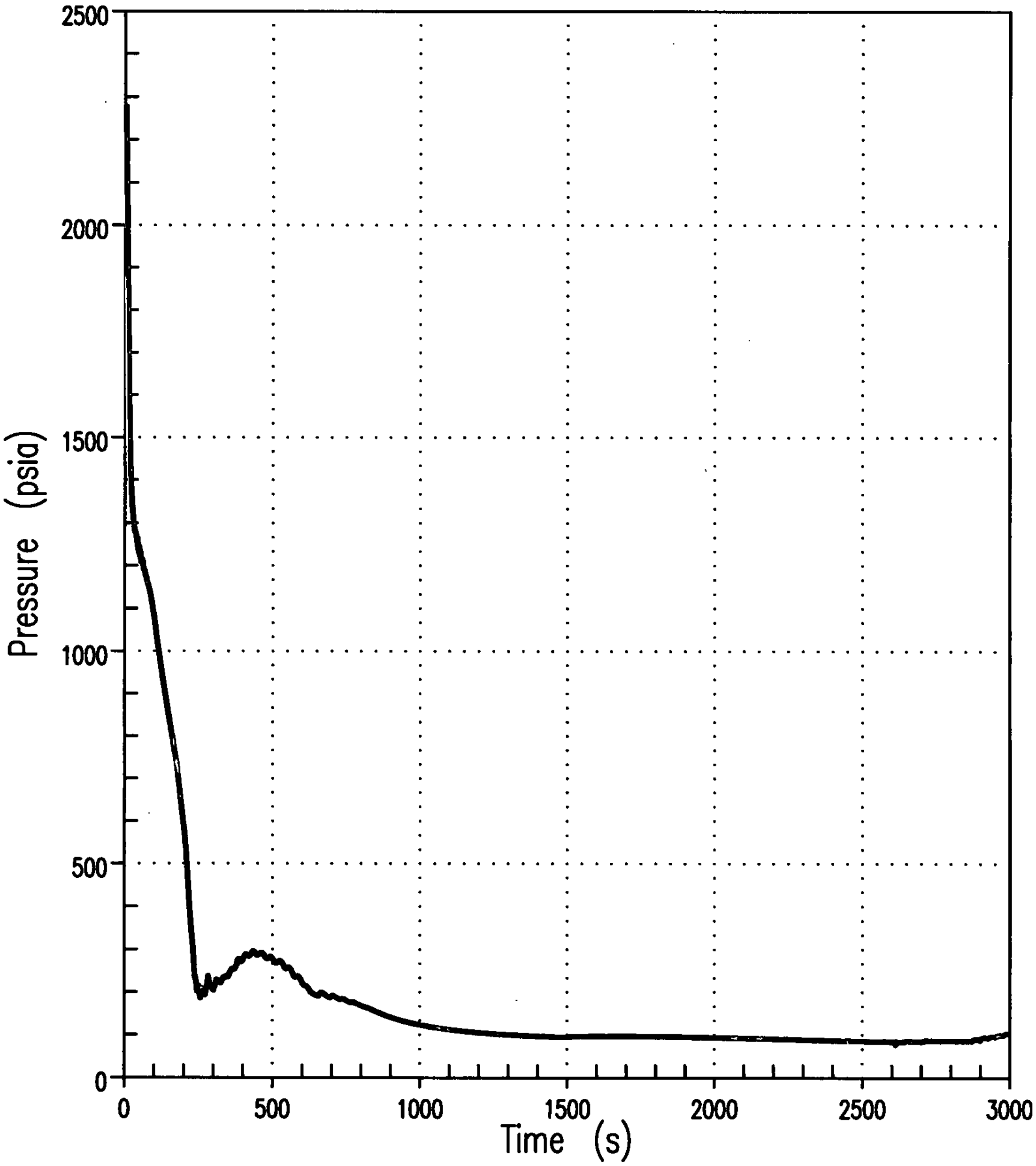

Figure 7A RCS Pressure 8.75-Inch Break (Unit 1)

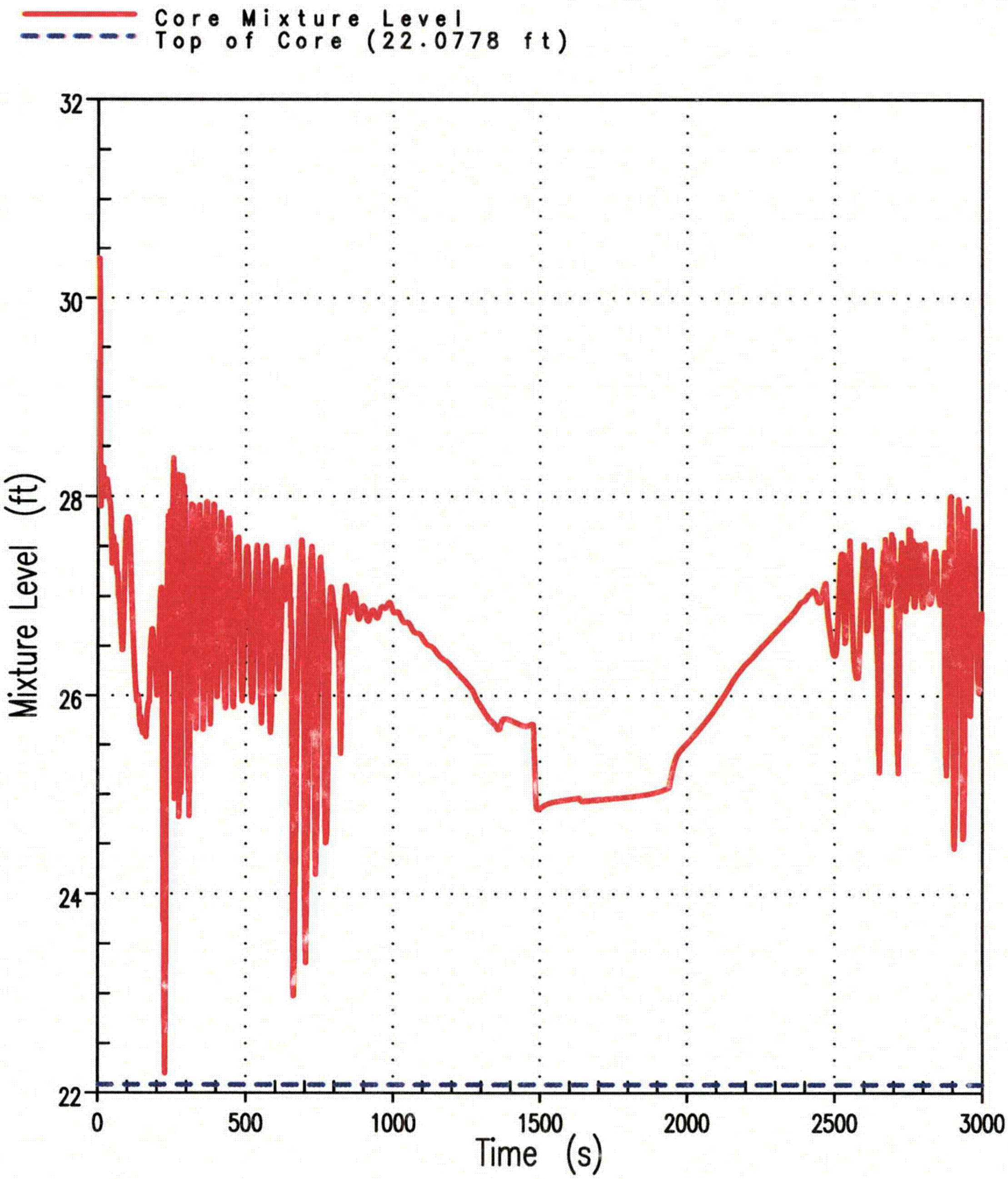

Figure 7B Core Mixture Level 8.75-Inch Break (Unit 1)

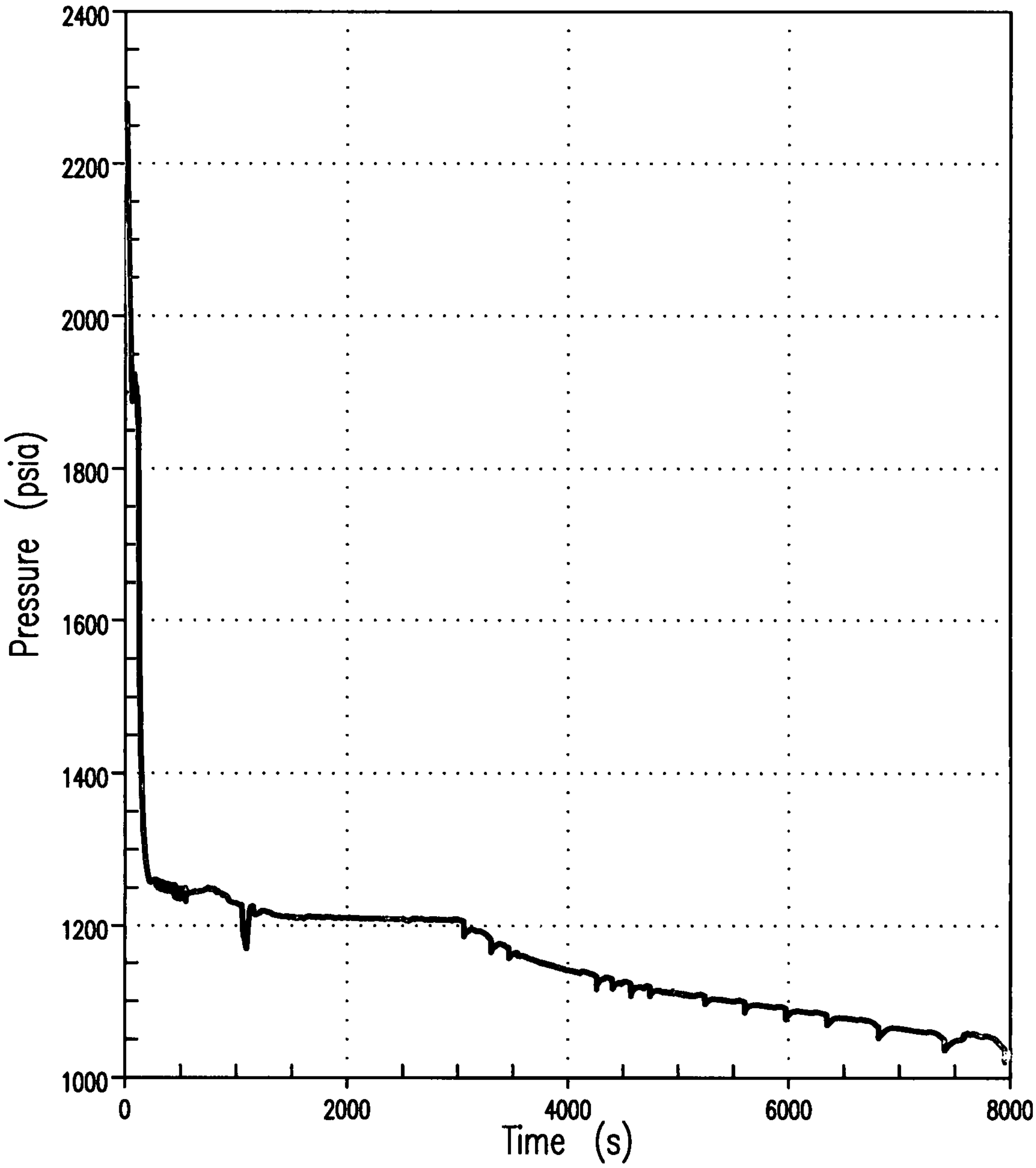

Figure 8A RCS Pressure 2-Inch Break (Unit 2)

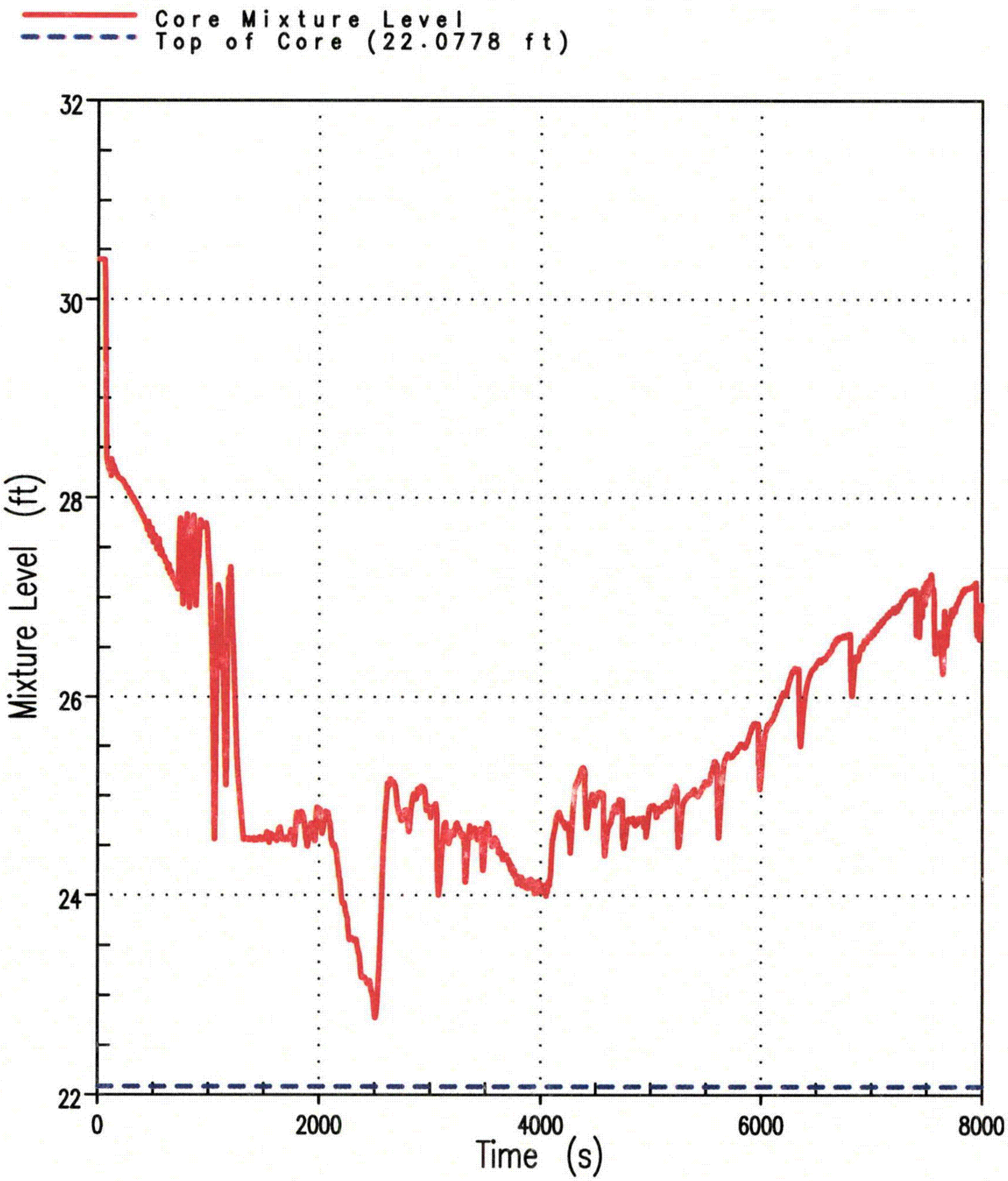

Figure 8B Core Mixture Level 2-Inch Break (Unit 2)

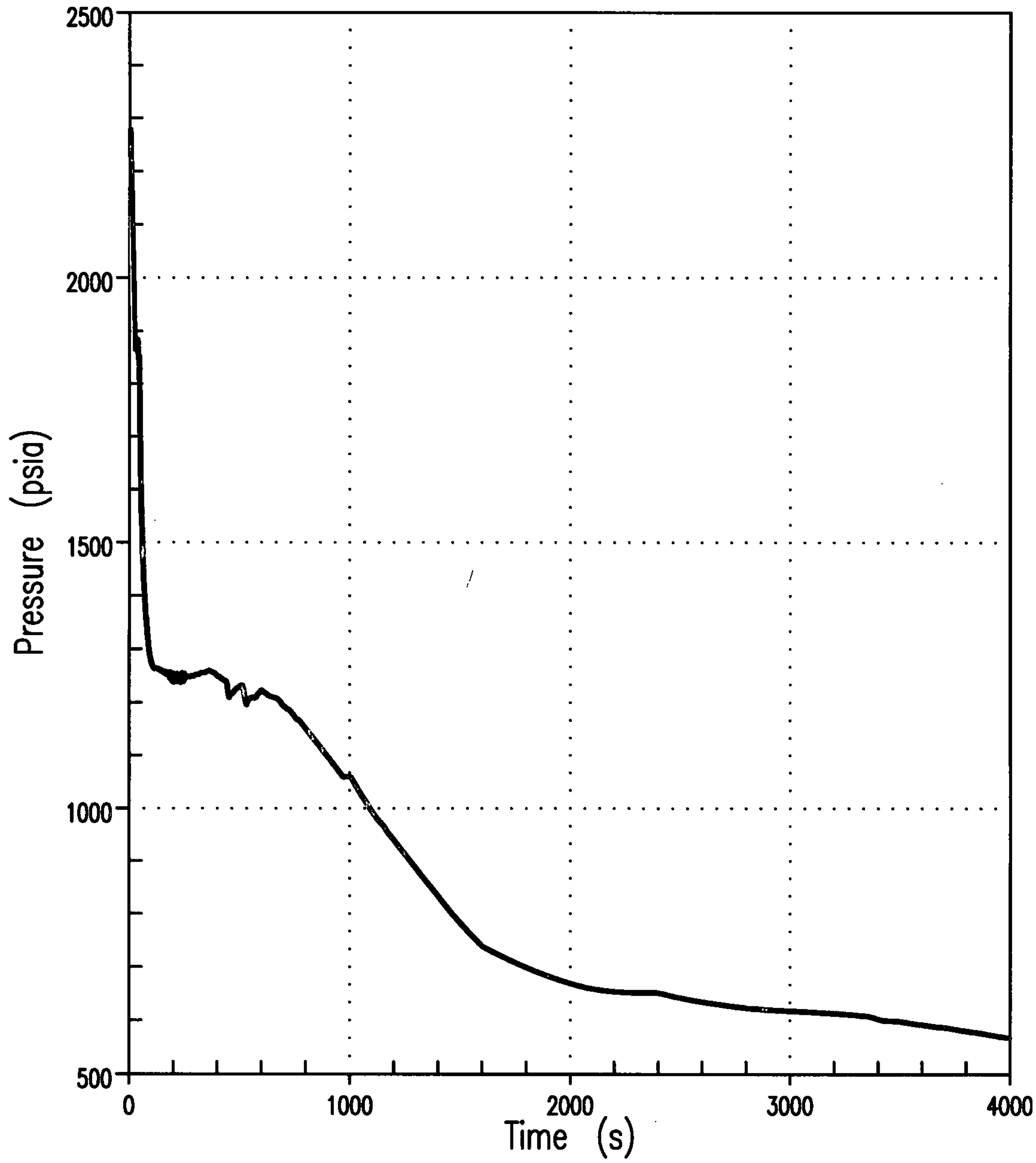

Figure 9A RCS Pressure 3-Inch Break (Unit 2)

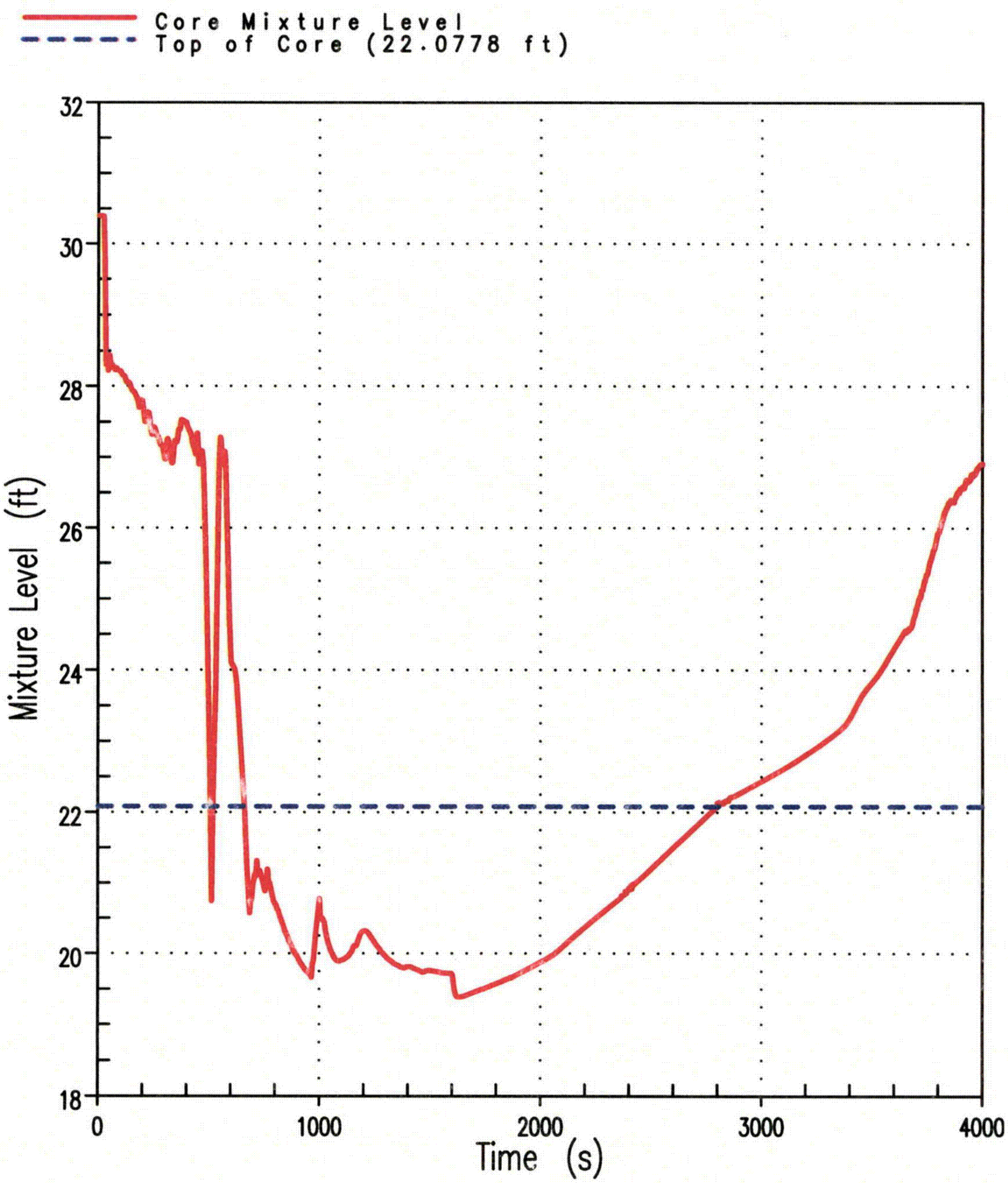

Figure 9B Core Mixture Level 3-Inch Break (Unit 2)

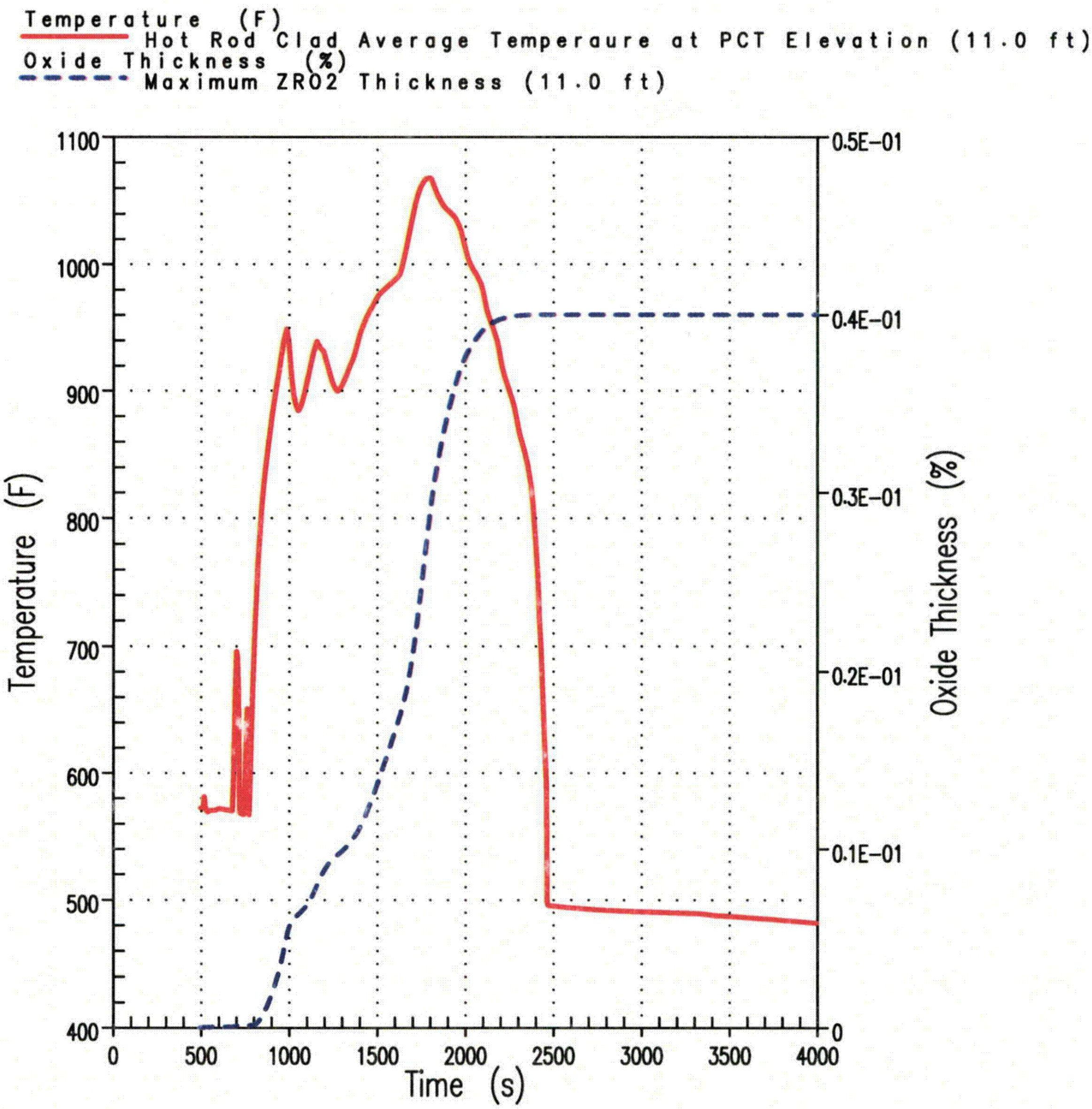

Figure 9C Hot Rod Clad Temperature and ZrO_2 Thickness at PCT Elevation 3-Inch Break (Unit 2)

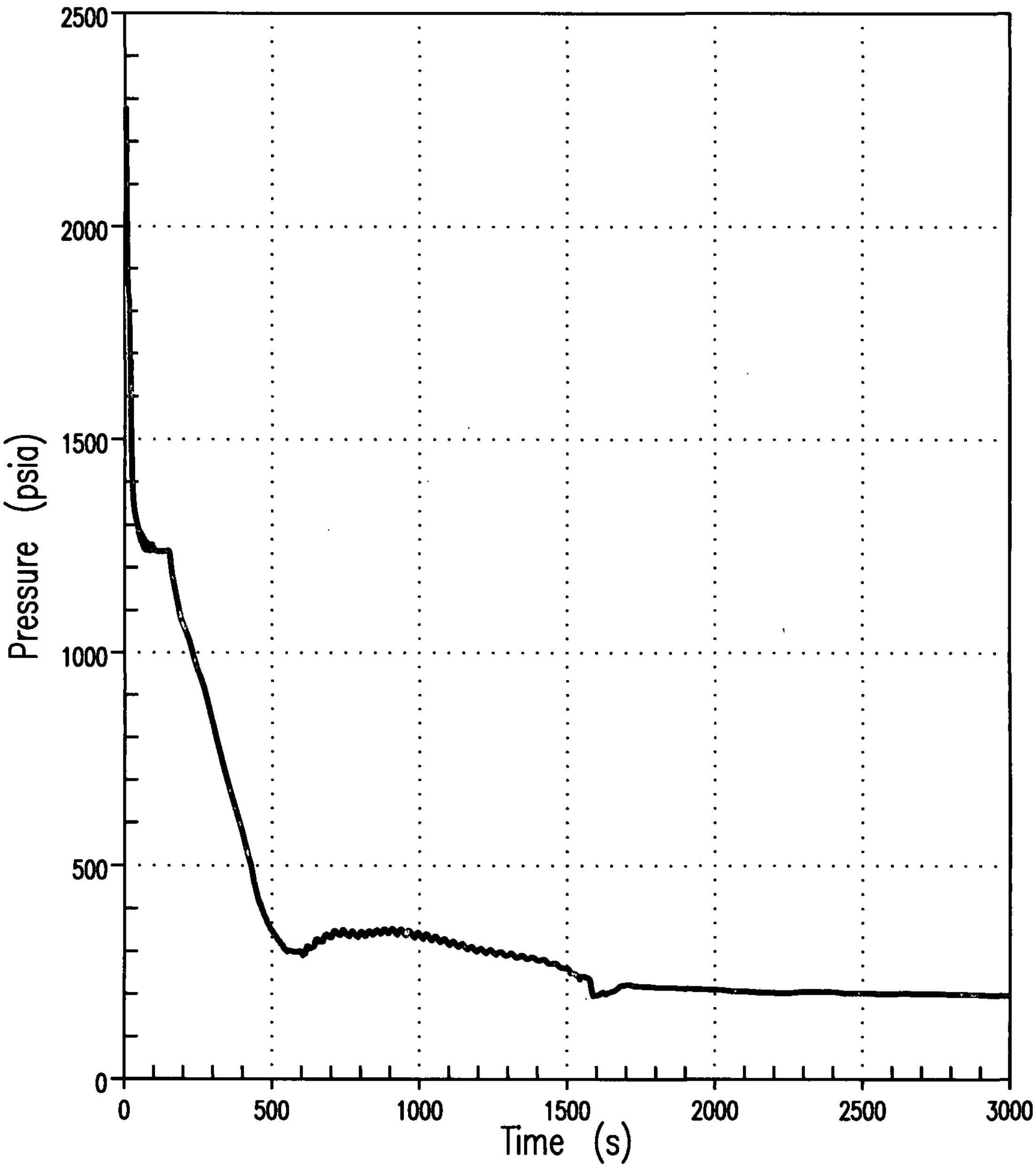

Figure 10A RCS Pressure 6-Inch Break (Unit 2)

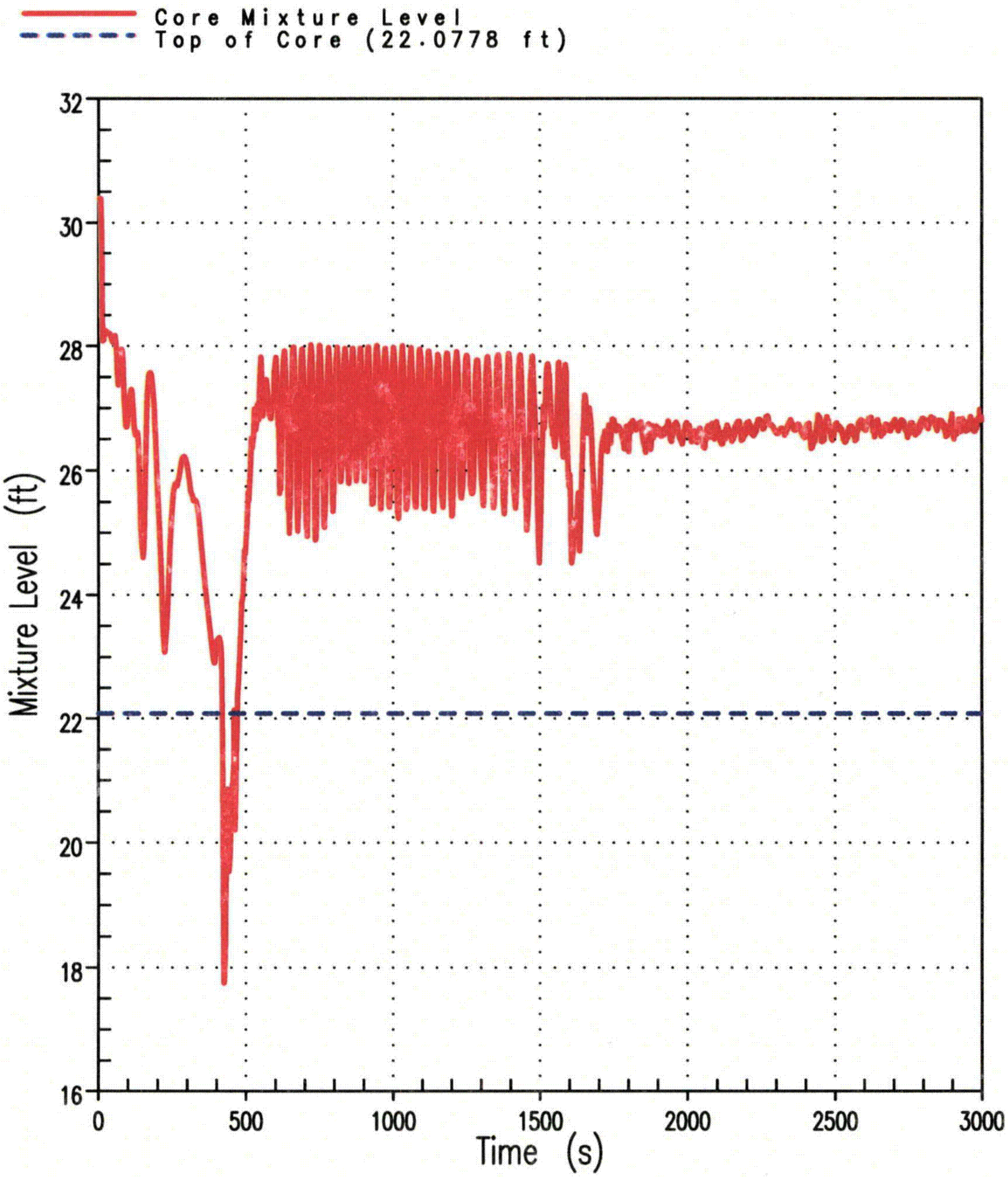

Figure 10B Core Mixture Level 6-Inch Break (Unit 2)

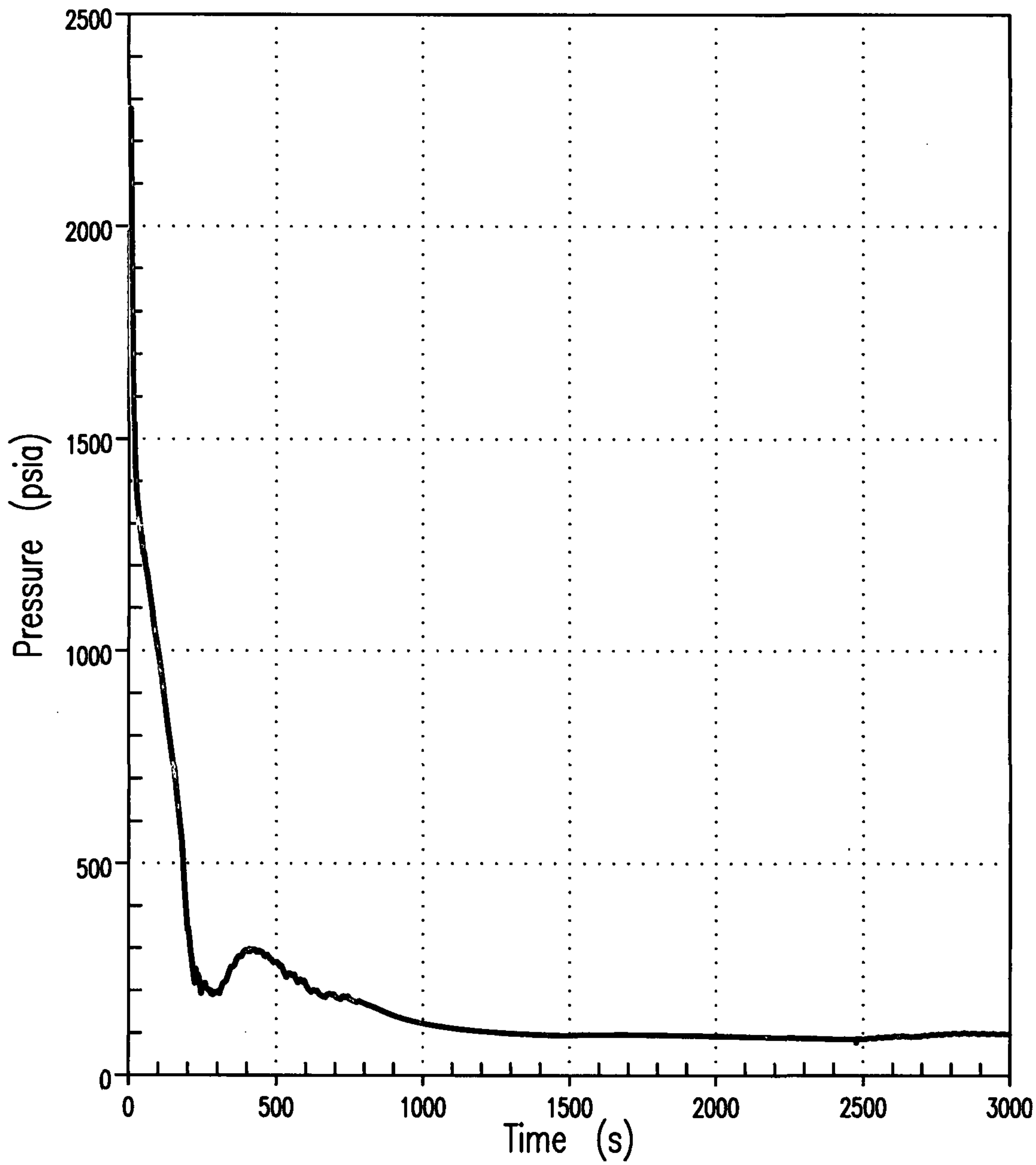

Figure 11A RCS Pressure 8.75-Inch Break (Unit 2)

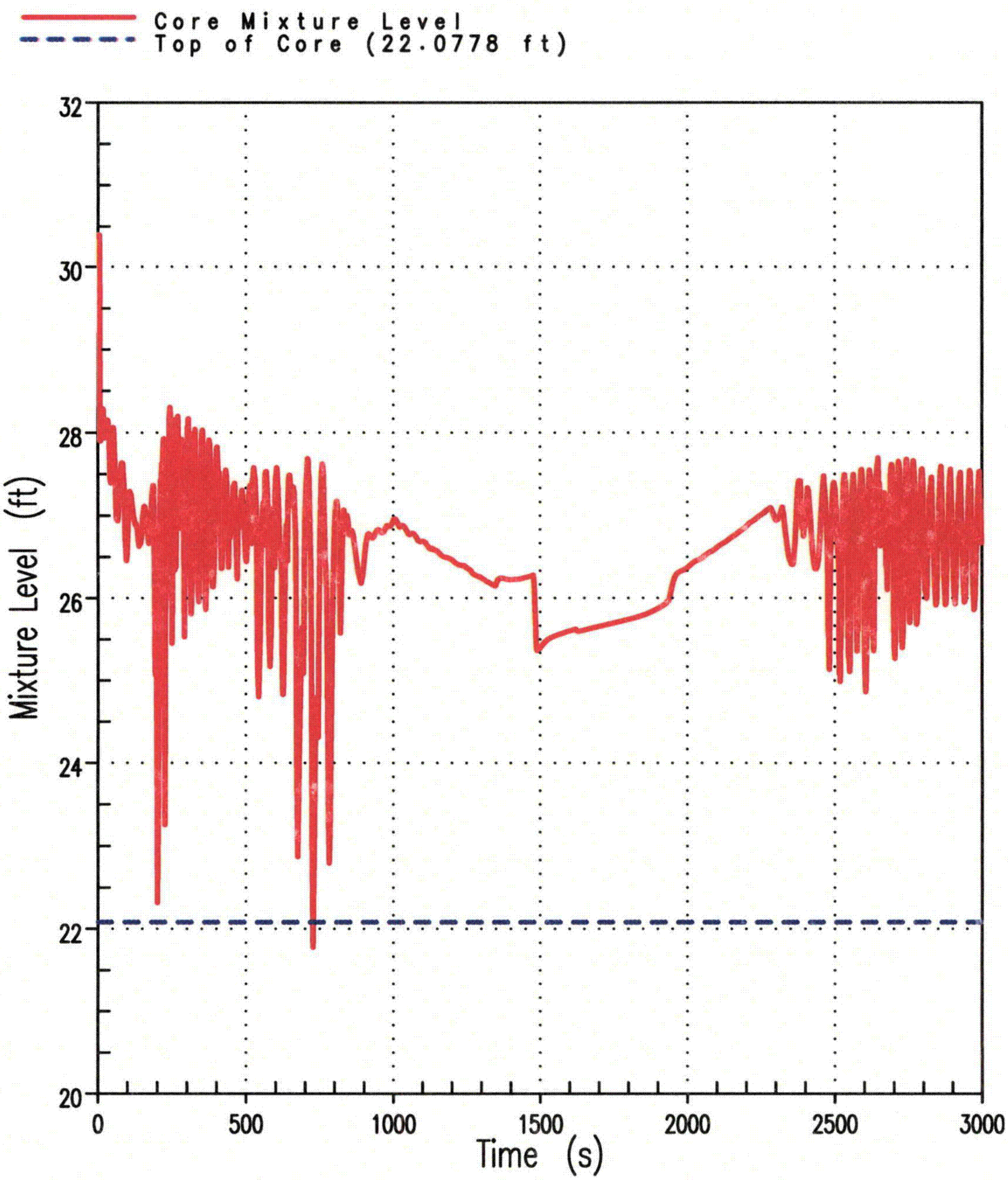

Figure 11B Core Mixture Level 8.75-Inch Break (Unit 2)