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Bradford Houston
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W3F1-2004-0044

May 26, 2004

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

SUBJECT: Small Break Loss-of-Coolant Accident Emergency Core Cooling System Performance Analysis
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. Entergy Letter dated April 29, 2004, "Reporting of Information under 10 CFR 50.46, Newly Identified Single Failure for Small Break LOCA Analysis of Record"
2. Entergy Letter dated April 30, 1998, "Small Break Loss-of-Coolant Accident Emergency Core Cooling System Performance Analysis Using the ABB/CE Supplement 2 Model"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) reported, pursuant to 10 CFR 50.46(a)(3)(ii), an error discovered in the Waterford Steam Electric Station, Unit 3 (Waterford 3) small break loss-of-coolant accident (SBLOCA) analysis of record (see Reference 2). That letter described the nature of the error and its effect on the current Waterford 3 emergency core cooling system (ECCS) analysis. Also in that letter Waterford 3 committed to submit the small break LOCA re-analysis results to demonstrate compliance with 10 CFR 50.46 by May 31, 2004.

The attached description of a revision to the Waterford 3 ECCS analysis of record for the SBLOCA was performed using the Supplement 2 version (referred to as the S2M or Supplement 2 Model) of the Westinghouse SBLOCA evaluation model for Combustion Engineering (CE) designed Pressurized Water Reactors (PWRs). This is the same model as used for the current analysis of record. The S2M was approved by the NRC for use by CE-designed plants on December 16, 1997.

The revised analysis utilizes the same methodology used in the analysis of record. Two of the design inputs have been changed: credit for supplemental charging flow has been eliminated and the flow curve for the high pressure safety injection pump has been revised. As described in the attachment, the results of the revised Waterford 3 SBLOCA ECCS performance analysis conform to the ECCS acceptance criteria of 10 CFR 50.46. A table

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identifying the impact of various model and input changes on the SBLOCA analysis made since the analysis of record (Reference 2) is included in the attachment.

There are no new commitments contained in this letter. If you have any questions or require additional information, please contact Jerry Burford at 601-368-5755.

Sincerely,

Bradford L Houston

BLH/FGB/cbh

Attachment: Description of Analysis and Results

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Attachment to

W3F1-2004-0044

Description of Analysis and Results

Description of Analysis and Results

1.0 Introduction

On March 31, 2004, Entergy Operation, Inc. (Entergy) reported the identification of a different worst case single failure for the Waterford Steam Electric Station, Unit 3 (Waterford 3) small break loss-of-coolant accident (SBLOCA) emergency core cooling system (ECCS) performance analysis (Reference 1). This worst case single failure is that of a direct current (DC) power bus. The failure of a DC power bus results in the inability to start one emergency diesel generator (EDG) and the failure of a charging loop isolation valve to remain open.

The current Waterford 3 SBLOCA analysis assumes the failure of an EDG as the worst single failure. In addition, it credits the flow from one charging pump. The charging pumps inject into two reactor coolant pump (RCP) discharge legs. Therefore, after accounting for the assumption that charging flow to the broken RCP discharge leg will not reach the reactor vessel, the SBLOCA analysis credits 50% of the flow from one charging pump reaching the reactor vessel.

With an assumed failure of a DC bus and the consequential failure of a charging loop isolation valve to remain open, one RCP discharge leg receives 100% of the charging flow. However, if that discharge leg is postulated to be the location of the break, then no charging flow is assumed to reach the reactor vessel. This is contrary to the current Waterford 3 SBLOCA analysis, which credits 50% of the flow from one charging pump. Therefore, the consequences of this worst case single failure are not bounded by the analysis and thus this single failure represents an unanalyzed condition.

This attachment describes a new SBLOCA ECCS performance analysis for Waterford 3 that does not credit any injection flow from the charging pumps. Given the fact that the new analysis does not credit injection from the charging pumps, the failure of a DC bus and the failure of an EDG are functionally equivalent with respect to their impact on the availability of ECCS equipment. Consequently, the failure of an EDG is nominally selected as the most damaging single failure assumed in the analysis.

In order to compensate for the adverse impact of the removal of credit for charging flow, the new analysis credits additional flow from the high pressure safety injection (HPSI) pump. The additional flow was obtained by removing discretionary conservatism that was included in the calculation of the HPSI pump delivery curve (i.e., HPSI pump flow versus reactor coolant system (RCS) pressure) used in the current SBLOCA analysis.

The following sections of this attachment describe the methodology, changes in plant design data, results, and conclusions of the new analysis.

2.0 Methodology

The new analysis has been performed using the Supplement 2 version (referred to as the S2M or Supplement 2 Model) of the Westinghouse SBLOCA evaluation model for Combustion Engineering (CE) designed Pressurized Water Reactors (PWRs) (Reference 2). This is the same methodology used in the current Waterford 3 SBLOCA analysis, which was submitted to the NRC in Reference 12. It is also described in Sections 6.3.3.3 and 15.6.3.3.3.2 of the

Waterford 3 Final Safety Analysis Report (FSAR) (Reference 3). The S2M is accepted by the Nuclear Regulatory Commission (NRC) for use in CE design PWR licensing applications, including reference in plant technical specifications and core operating limits reports (Reference 4).

In the S2M evaluation model, the CEFLASH-4AS computer program (Reference 5) is used to perform the thermal hydraulic analysis of the RCS until the time the safety injection tanks (SITs) begin to inject. After injection from the SITs begins, the COMPERC-II computer program (Reference 6) is used to perform the thermal hydraulic analysis of the RCS. The hot rod cladding temperature and maximum cladding oxidation are calculated by the STRIKIN-II computer program (Reference 7) during the initial period of forced convection heat transfer and by the PARCH computer program (Reference 8) during the subsequent period of pool boiling heat transfer. Core-wide cladding oxidation is conservatively represented as the rod-average cladding oxidation of the hot rod. The initial steady state fuel rod conditions used in the SBLOCA analysis are determined using the FATES3B computer program (Reference 9).

As described in Section 3.0, removal of the credit for charging flow and crediting additional HPSI pump flow results in a very small net change in safety injection flow relative to the current SBLOCA analysis. Because of this, there is very little difference in the RCS thermal hydraulic transient and the hot rod heatup transient between the new analysis and the current analysis. Consequently, only the limiting break of the current analysis (i.e., the 0.05 ft^2/PD break in the RCP Discharge leg) was reanalyzed in the new analysis. Also, STRIKIN-II was not run in the new analysis since there is an insignificant difference between the current and new analyses during the forced convection portion of the transient, which lasts for approximately 300 seconds for the 0.05 ft^2/PD break. Since STRIKIN-II was not run, the PARCH computer program was initialized with the STRIKIN-II results from the current analysis.

The current analysis was performed for the fuel rod conditions at the burnup that resulted in the maximum initial stored energy in the fuel. In the new analysis, additional studies were performed using PARCH to determine the fuel rod internal pressure that causes cladding rupture to occur at the time that results in the maximum peak cladding temperature (PCT) and maximum cladding oxidation.

3.0 Plant Design Data

The new SBLOCA analysis uses the same plant design data used in the current analysis with the two exceptions noted in Section 1.0, namely, no credit for charging flow and the use of a revised HPSI pump delivery curve. Table 1 lists important input parameters and initial conditions used in the analysis. Except for the charging and HPSI flows, the values are the same as listed in Table 15.6-13a of the Waterford 3 FSAR.

Table 2 lists the HPSI pump delivery curve used in the new analysis. Figure 1 provides a comparison of the new HPSI pump delivery curve to that of the HPSI pump and charging pump flow used in the current analysis. As shown in the figure, for the RCS pressure range of interest (i.e., above approximately 500 psia), there is very little difference between the new HPSI pump delivery curve and the total safety injection (i.e., HPSI plus charging flow) credited in the current analysis.

The current and the new analyses were performed for up to 500 plugged tubes per steam generator (see Table 1). In order to accommodate up to 700 plugged tubes per steam generator, a PCT adder of +53°F was previously determined. Reference 10 identified a +3°F PCT adder for a minor correction to the RCP suction leg geometry. Also, Reference 11 identified a -38°F PCT adjustment for errors identified in CEFLASH-4AS.

4.0 Results

Table 3 lists the peak cladding temperature and oxidation percentages that were calculated in the new analysis for the limiting break (i.e., 0.05 ft^2/PD break). Times of interest are listed in Table 4. The variables listed in Table 6 are plotted as a function of time for the 0.05 ft^2/PD break in Figures 2 through 9.

The results for the 0.05 ft^2/PD break demonstrate conformance to the ECCS acceptance criteria as summarized below.

Parameter	Criterion	Results
Peak Cladding Temperature	≤2200°F	1959°F
Maximum Cladding Oxidation	≤17%	9.0%
Maximum Core-Wide Oxidation	≤1%	<0.58%
Coolable Geometry	Yes	Yes

Table 5 has been included to summarize the PCT impact of various changes since the current analysis of record. Included in this table are the effects of the two revised inputs in the new analysis.

5.0 Conclusions

The results of the Waterford 3 SBLOCA ECCS performance analysis described in this attachment conform to the ECCS acceptance criteria of 10 CFR 50.46. The analysis uses the same NRC-accepted evaluation model previously used in the current analysis of record for Waterford 3. The results of the analysis are applicable to Waterford 3 for a power level of 3478 MWt (including power measurement uncertainty), a peak linear heat generation rate (PLHGR) of 13.5 kW/ft, and up to 700 plugged tubes per steam generator.

6.0 References

1. Event No. 40632, NRC Daily Events Report for April 1, 2004, "New Worst Case Single Failure May Exceed 10 CFR 50.46 Acceptance Criterion for SBLOCA."
2. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.
3. Final Safety Analysis Report, Waterford Steam Electric Station, Unit No. 3, Facility Operating License No. NPF-38, Docket No. 50-382.
4. T.H. Essig (NRC) to I.C. Rickard (ABB CE), "Acceptance for Referencing of the Topical Report CENPD-137(P), Supplement 2, 'Calculative Methods for the C-E Small Break LOCA Evaluation Model' TAC No. M95687)," December 16, 1997.

5. CENPD-133P, Supplement 1, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," August 1974; Supplement 3-P, January 1977.

6. CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974; Supplement 1, February 1975; Supplement 2-A, June 1985.

7. CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974; Supplement 2, February 1975; Supplement 4-P, August 1976; Supplement 5, April 1977.

8. CENPD-138P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974; Supplement 1, February 1975; Supplement 2-P, January 1977.

9. CENPD-139-P-A, "C-E Fuel Evaluation Model," July 1974; CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989; CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.

10. Entergy letter dated April 5, 2004, "Annual Report on Westinghouse Electric Company LLC Combustion Engineering Emergency Core Cooling System Performance Evaluation Models" (W3F1-2004-0021).

11. Entergy letter dated May 7, 2002, "Annual Report on Westinghouse Electric Company LLC Combustion Engineering Emergency Core Cooling System Performance Evaluation Models" (W3F1-2002-0044).

12. Entergy letter dated April 30, 1998, "Small Break Loss-of-Coolant Accident Emergency Core Cooling System Performance Analysis Using the ABB/CE Supplement 2 Model" (W3F1-98-0090).

Table 1
SBLOCA ECCS Performance Analysis
Core and Plant Design Data

Parameter	Value	Units
Reactor power level (including measurement uncertainty)	3478	MWt
Peak linear heat generation rate (PLHGR)	13.5	kW/ft
Gap conductance at the PLHGR(1)	1584	BTU/hr-ft^2-°F
Fuel centerline temperature at the PLHGR(1)	3402	°F
Fuel average temperature at the PLHGR(1)	2159	°F
Hot rod gas pressure(1)	1113	psia
Moderator temperature coefficient at initial density	0.0×10^{-4}	$\Delta\rho$/°F
Axial shape index	-0.25	asiu
RCS pressure	2250	psia
RCS flow rate	148×10^6	lbm/hr
Core flow rate	144.15×10^6	lbm/hr
Cold leg temperature	557.5	°F
Hot leg temperature	615.5	°F
Number of plugged tubes per steam generator	500	—
Main steam safety valve first bank opening pressure	1117	psia
Low pressurizer pressure reactor trip setpoint	1560	psia
Low pressurizer pressure SIAS setpoint	1560	psia
High pressure safety injection pump flow rate	Table 2	gpm(psia)
Time delay for actuation of HPSI flow (with loss of offsite power	30	seconds
Charging pump flow rate (to intact discharge leg)	0	gpm
Safety injection tank pressure	615	psia

Notes:
(1) The values for these parameters are the values for the rod average burnup of the hot rod (1000 MWD/MTU) that yields the maximum initial fuel stored energy.

Table 2
High Pressure Safety Injection Pump
Minimum Delivered Flow to RCS
(Assuming Failure of an Emergency Diesel Generator)

RCS Pressure (psia)	Flow Rate (gpm)(1), (2)
0	775
92	745
231	698
352	655
486	605
609	556
798	473
901	423
978	382
1047	342
1183	249
1244	196
1287	152
1326	100
1366	0

Notes:
(1) The flow is assumed to be split equally to each of the four discharge legs.
(2) The flow to the broken discharge leg is assumed to spill out the break.

Table 3
SBLOCA ECCS Performance Analysis Results

Break Size	Peak Cladding Temperature (°F)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Cladding Oxidation (%)
0.05 ft^2/PD	1959	9.0	<0.58

Table 4
SBLOCA ECCS Performance Analysis
Times of Interest (seconds after break)

Break Size	Reactor Trip and SIAS	HPSI Flow Delivered to RCS	LPSI Flow Delivered to RCS	SIT Flow Delivered to RCS	PCT Occurs
0.05 ft^2/PD	131	161	n/a(1)	1740(2)	1802

Notes:
(1) Calculation completed before LPSI flow to the RCS begins.
(2) SIT injection calculated to begin but not credited in analysis.

Table 5
PCT Summary Table for the Waterford 3
SBLOCA ECCS Performance Analysis

	Temperature (°F)
Current Analysis of Record PCT	**1929**
Changes and Errors	
CEFLASH-4AS error	-38
Increase in SGTP to 700 plugged tubes/SG	+53
Suction leg geometry error	+3
Increase in HPSI pump flow	-204
Removal of credit for charging flow	+216
New Analysis of Record PCT	**1959**

Table 6
SBLOCA ECCS Performance Analysis
Variables Plotted as a Function of Time

Variable
Normalized Total Core Power
Inner Vessel Pressure
Break Flow Rate
Inner Vessel Inlet Flow Rate
Inner Vessel Two-Phase Mixture Level
Heat Transfer Coefficient at Hot Spot
Coolant Temperature at Hot Spot
Cladding Temperature at Hot Spot

Figure 1
Comparison HPSI Pump Delivery Curves
(Flow to the Three Intact RCP Discharge Legs)

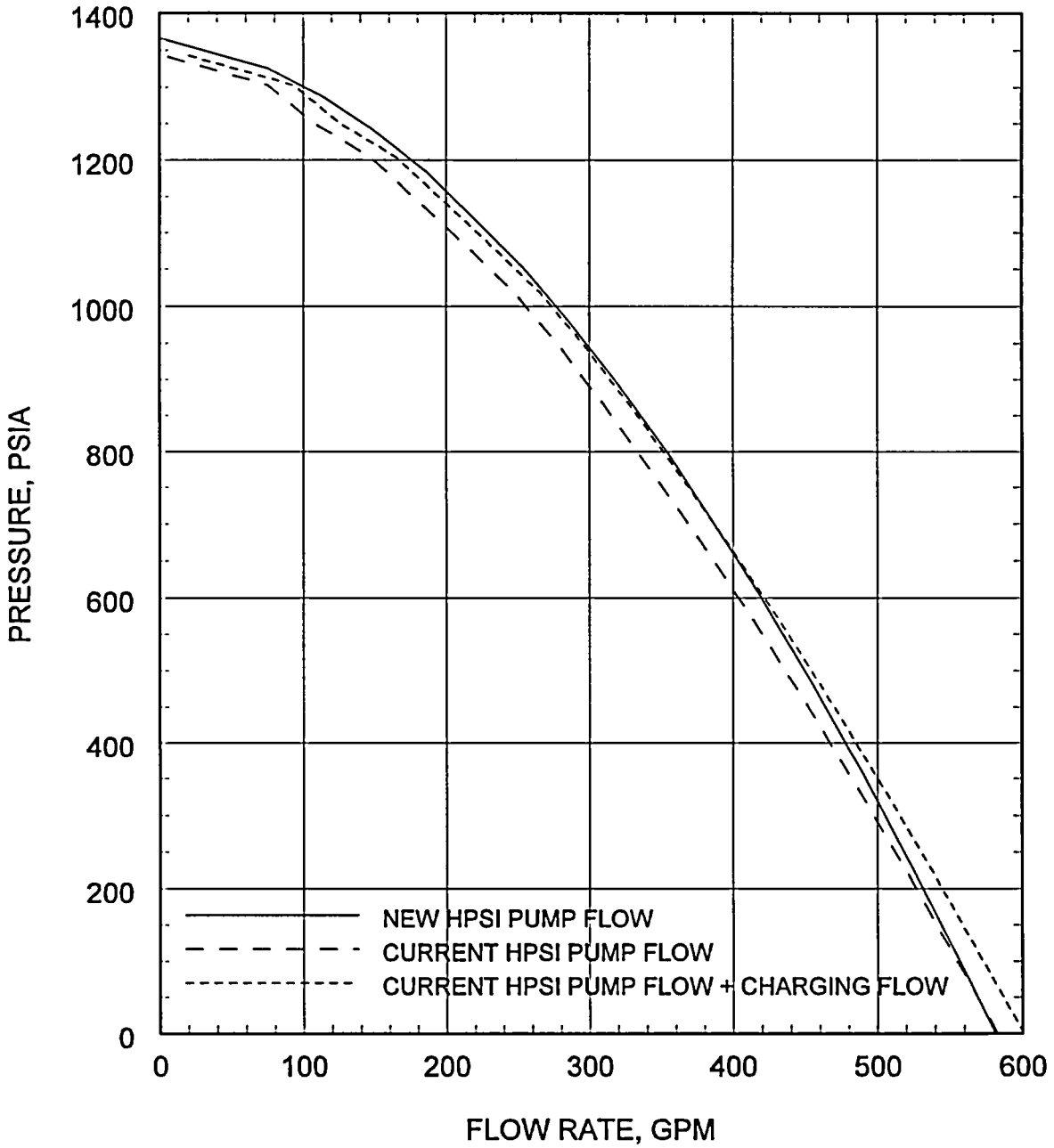

Figure 2
0.05 ft²/PD Break in the RCP Discharge Leg
Normalized Total Core Power

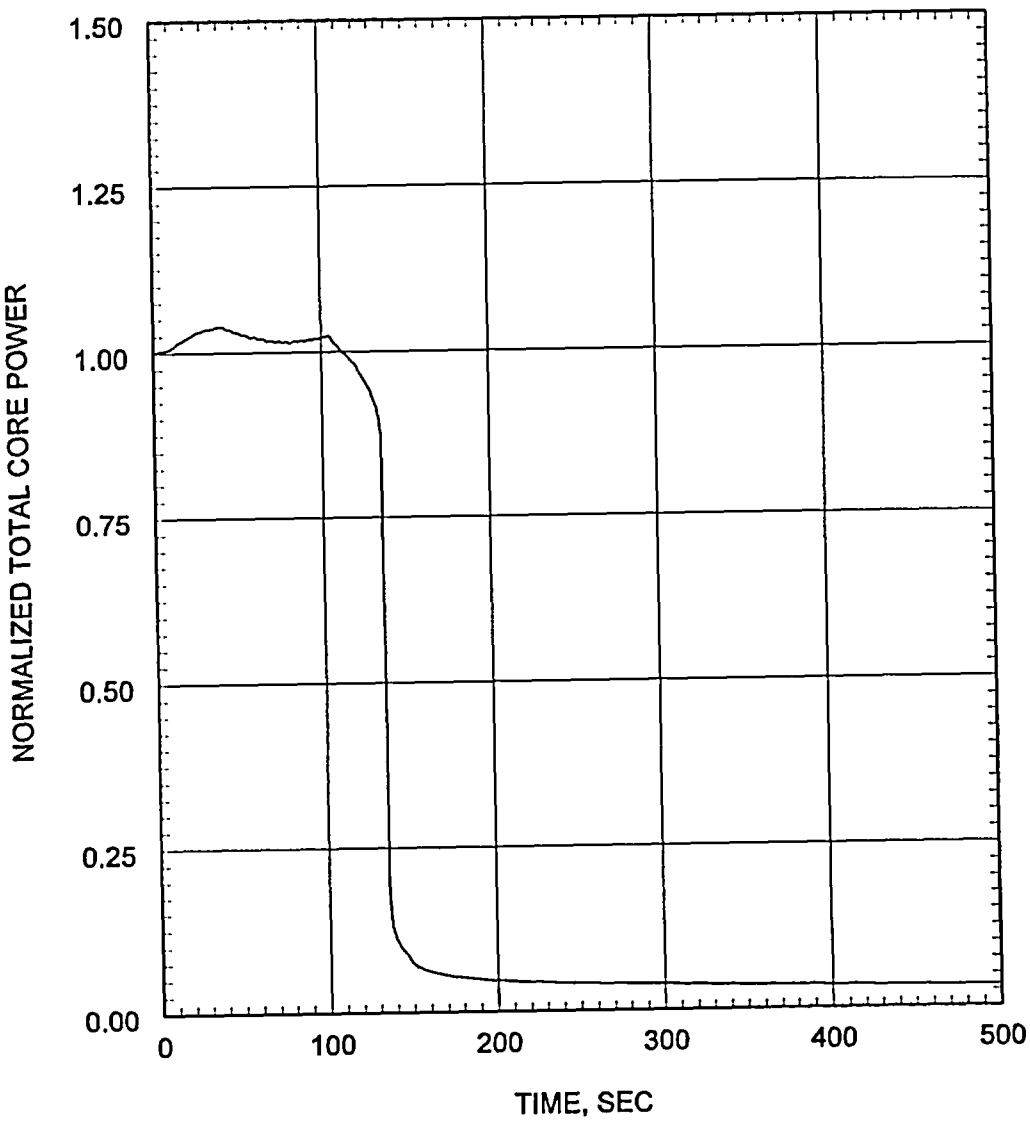

Figure 3
0.05 ft^2/PD Break in the RCP Discharge Leg
Inner Vessel Pressure

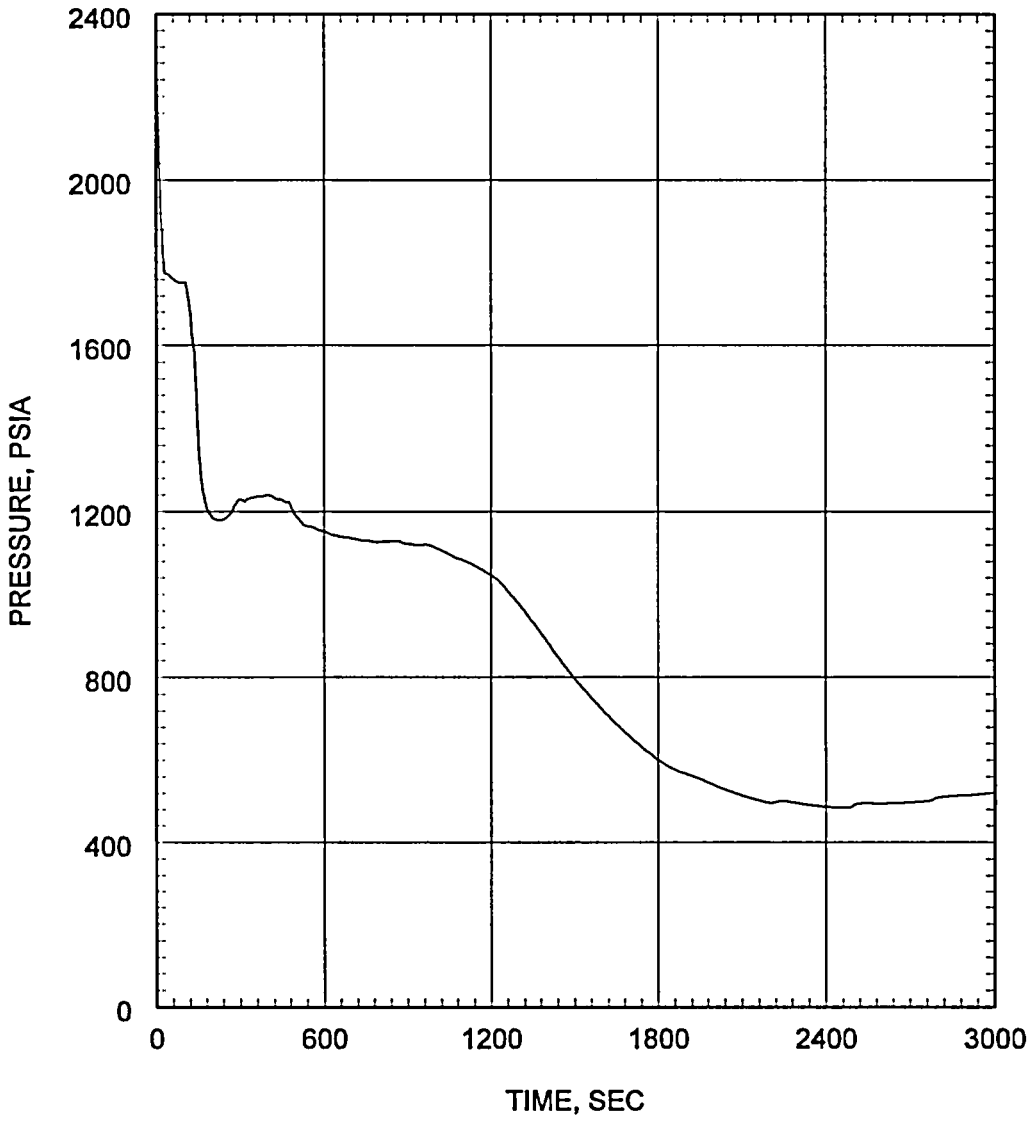

Figure 4
0.05 ft²/PD Break in the RCP Discharge Leg
Break Flow Rate

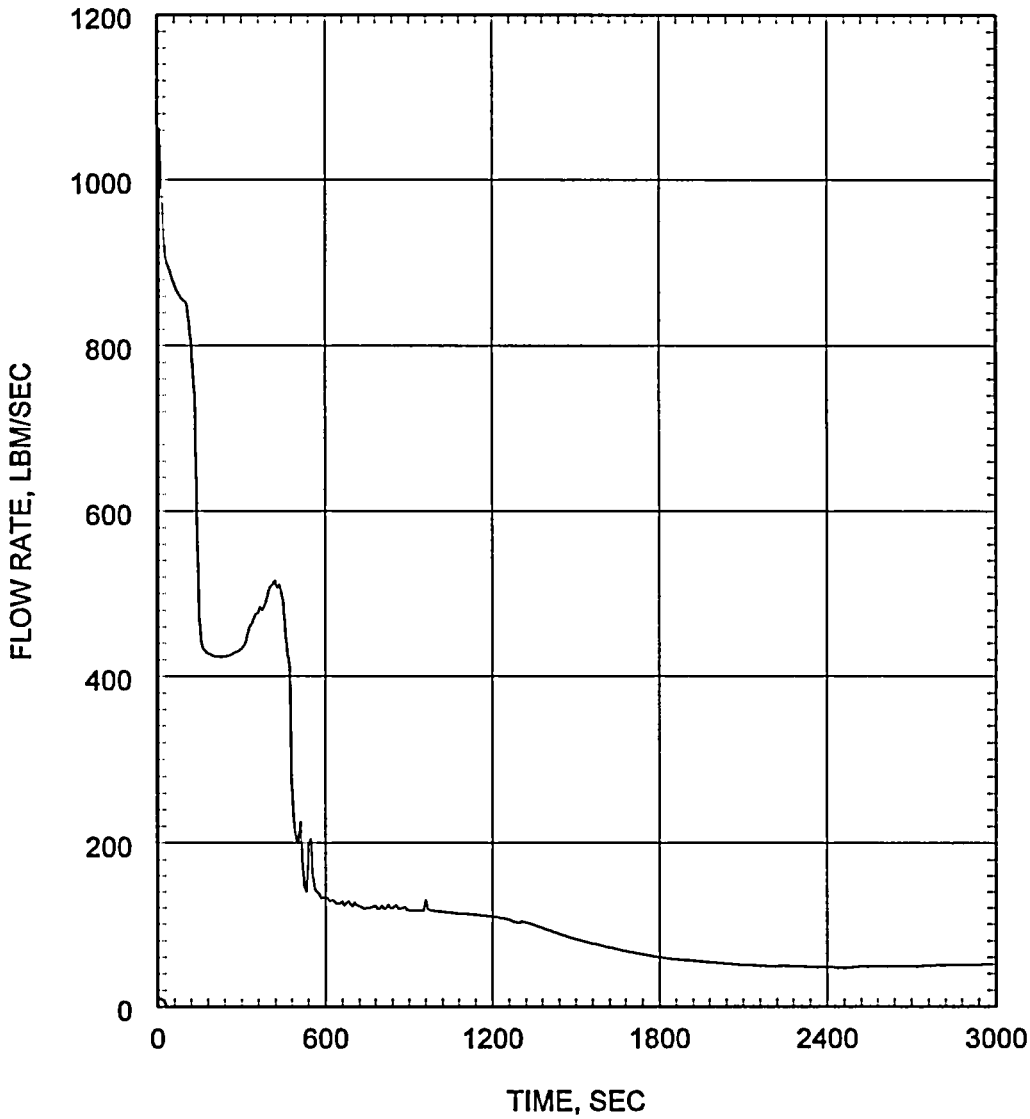

Figure 5
0.05 ft²/PD Break in the RCP Discharge Leg
Inner Vessel Inlet Flow Rate

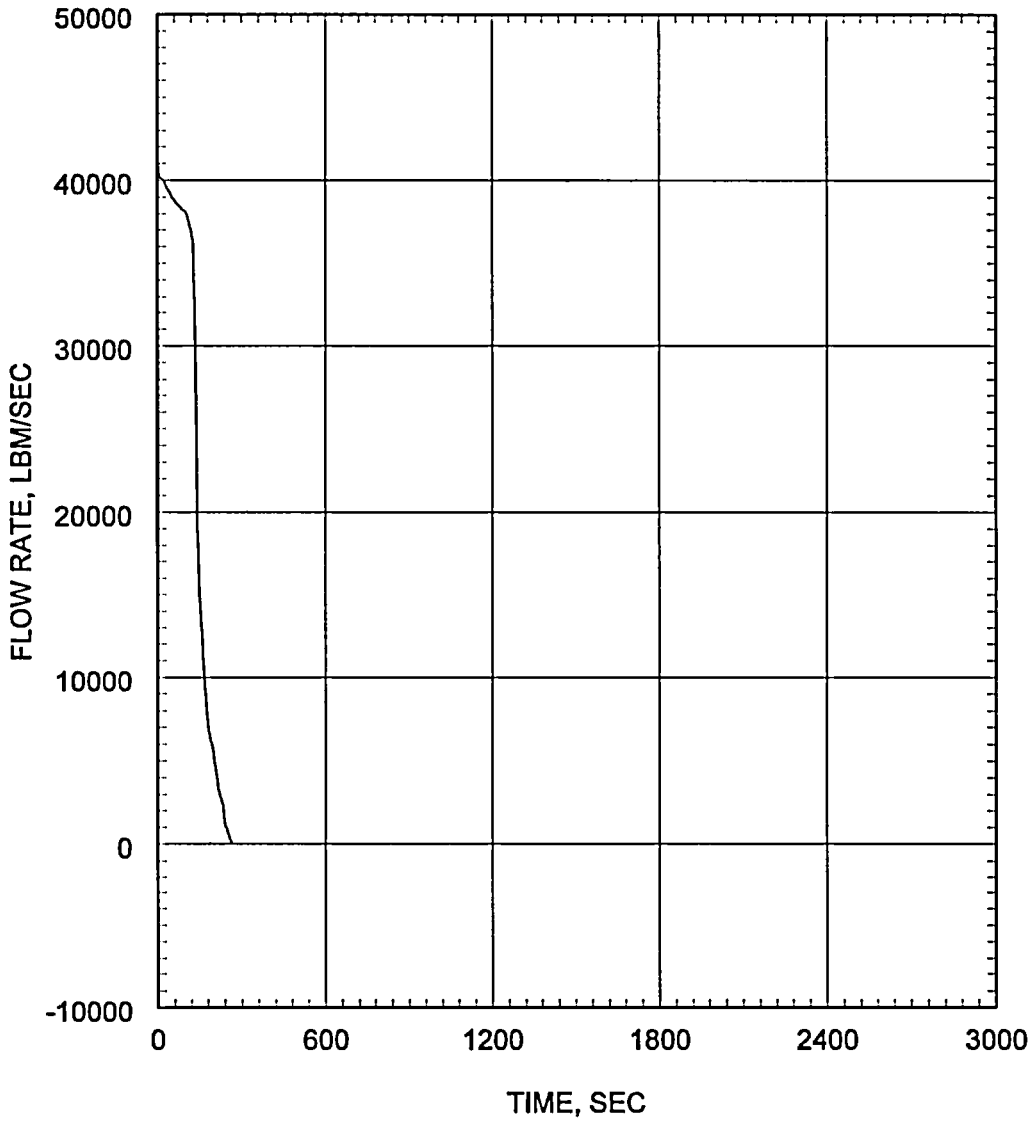

Figure 6
0.05 ft²/PD Break in the RCP Discharge Leg
Inner Vessel Two-Phase Mixture Level

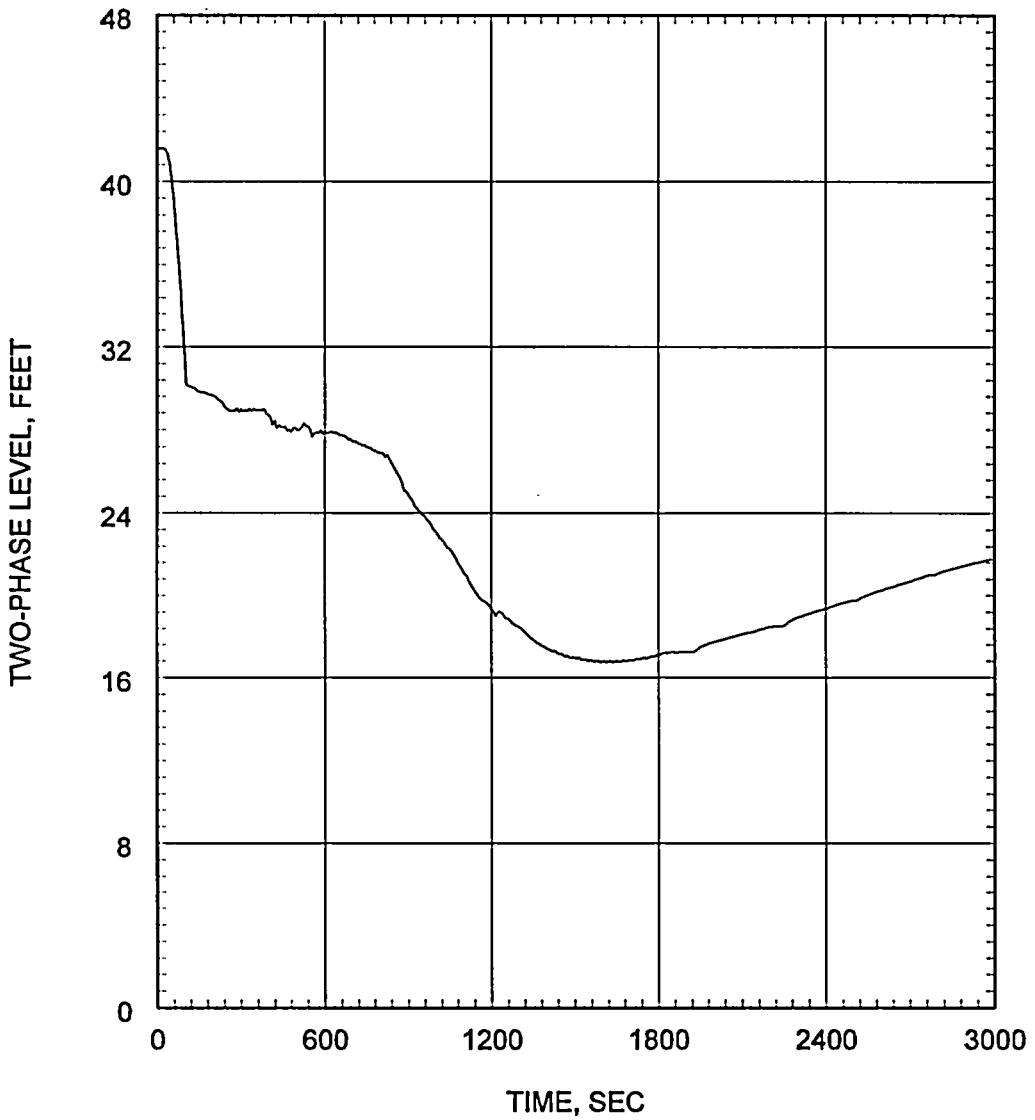

Figure 7
0.05 ft²/PD Break in the RCP Discharge Leg
Heat Transfer Coefficient at Hot Spot

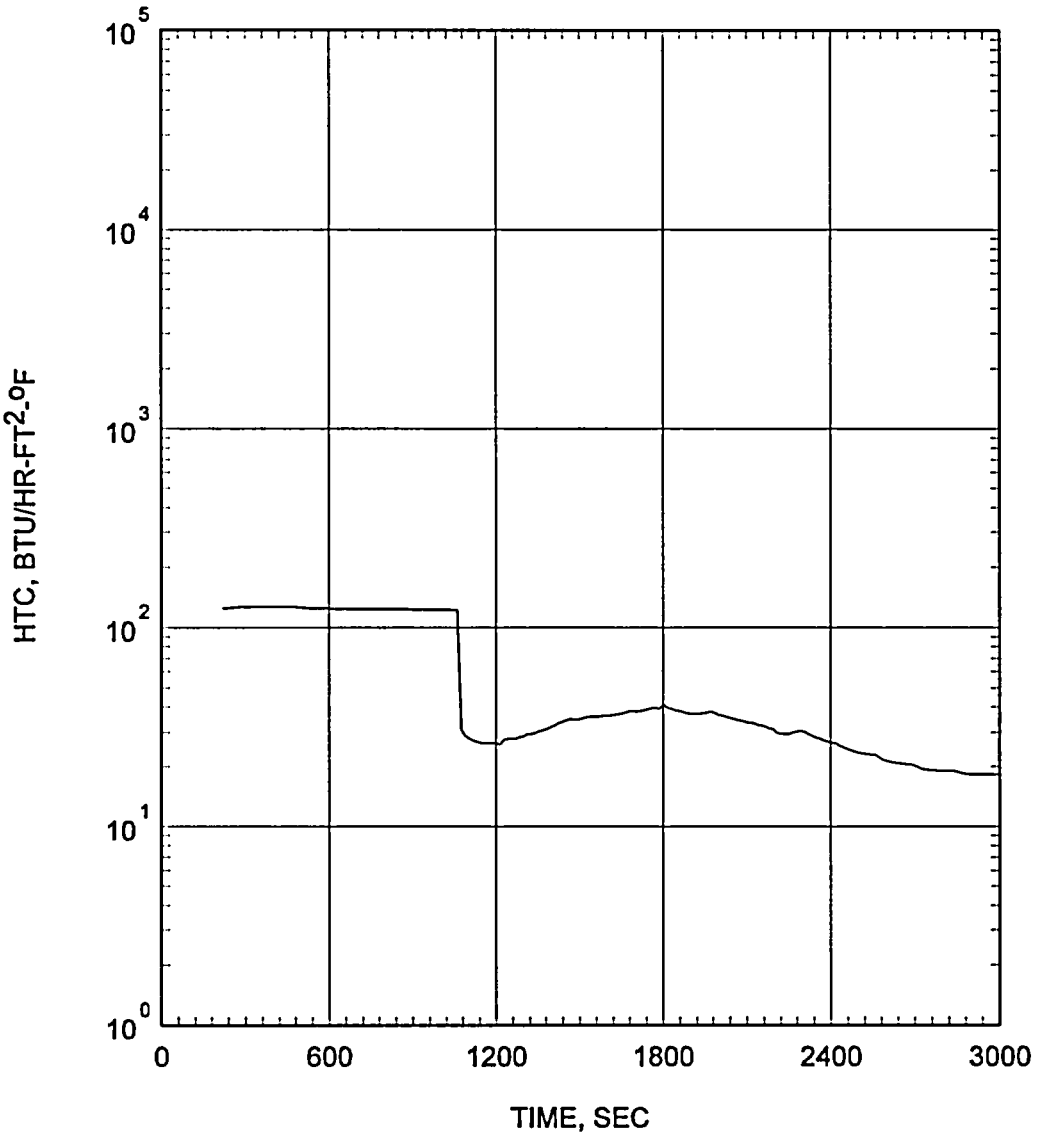

Figure 8
0.05 ft^2/PD Break in the RCP Discharge Leg
Coolant Temperature at Hot Spot

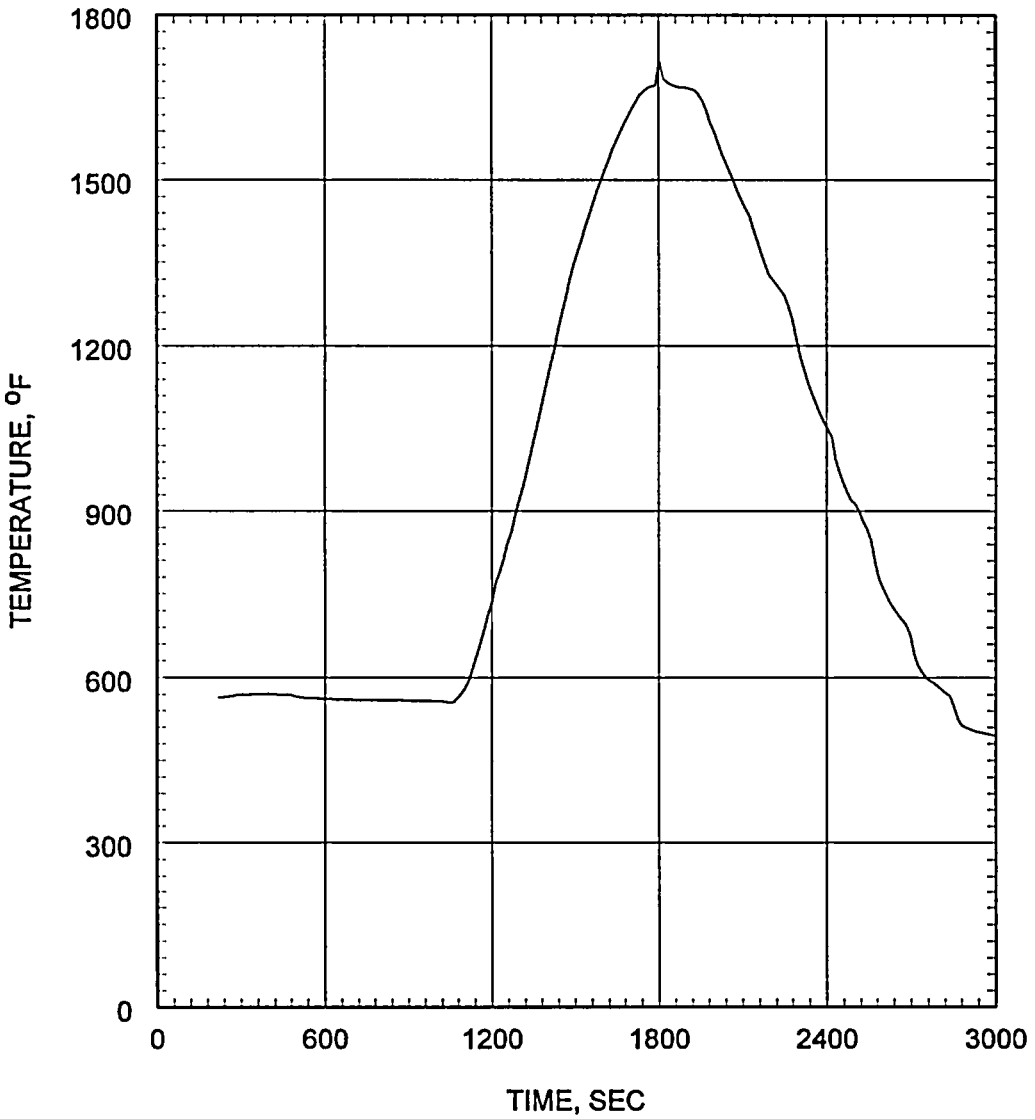

Figure 9
0.05 ft²/PD Break in the RCP Discharge Leg
Cladding Temperature at Hot Spot