CORE OPERATING LIMITS REPORT

6.9.5 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or any remaining part of a reload cycle.

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1) "The ROCS and DIT Computer Codes for Nuclear Design", CENPD-266-P-A, April 1983 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.4 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, and 3.2.4.b for DNBR Margin).

2) "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6 for Regulating and Group P CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

3) "Modified Statistical Combination of Uncertainties, CEN-356(V)-P-A, Revision 01-P-A, May 1988 (Methodology for Specification 3.2.4.c and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

4) "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

5) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 1, February 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 2-P, July 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

7) "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CEN-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

8) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

9) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 1-P, January 1977 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

CORE OPERATING LIMITS REPORT

10) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating CEA and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

11) Letter: O.D. Parr (NRC) to F.M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for 6.9.5.1.4, 6.9.5.1.5, and 6.9.5.1.8 methodologies.

12) Letter: O.D. Parr (NRC) to A.E. Scherer (CE), dated December 9, 1975 (NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model changes). NRC approval for 6.9.5.1.6 methodology.

13) Letter: K. Kniel (NRC) to A.E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.5.1.9 methodology.

14) Letter: 2CNA038403, dated March 20, 1984, J.R. Miller (NRC) to J.M. Griffin (AP&L), "CESEC Code Verification." NRC approval for 6.9.5.1.10 methodology.

6.9.5.2 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.5.3 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

MARKUP OF CURRENT ANO-2 TECHNICAL SPECIFICATIONS

(FOR INFO ONLY)

CORE OPERATING LIMITS REPORT

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3) "Modified Statistical Combination of Uncertainties, CEN-356(V)-P-A, Revision 01-P-A, May 1988 (Methodology for Specification 3.2.4.c and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

4) "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

5) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 1, February 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 2-P, July 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

7) "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CEN-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

8) "Calculati~~onal~~ve Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

9) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 1-P, January 1977 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

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~~9~~10) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating CEA and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

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1~~2~~4) Letter: 2CNA038403, dated March 20, 1984, J.R. Miller (NRC) to J.M. Griffin (AP&L), "CESEC Code Verification." NRC approval for 6.9.5.1.~~9~~10 methodology.

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DRAFT ANO-2 SAR REWRITE

6.3.3.2.3 Small Break Analysis

6.3.3.2.3.1 Evaluation Model

The small break LOCA analysis was performed using the ABB CE small break LOCA evaluation model (Reference 8, Supplement 1-P). The evaluation model was approved by the NRC in Reference 82. In the ABB CE small break LOCA evaluation model, the CEFLASH-4AS computer program (Reference 83) is used to perform the hydraulic analysis of the RCS until the time the SITs begin to inject. After injection from the SITs begins, the COMPERC-II computer program (Reference 11) is used to perform the hydraulic analysis. The hot rod cladding temperature and maximum cladding oxidation are calculated by the STRIKIN-II computer program (Reference 30) during the initial period of forced convection heat transfer and by the PARCH computer program (Reference 27) 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 analysis are determined using the FATES3B computer program (Reference 10).

6.3.3.2.3.2 Safety Injection System Parameters

As described in Section 6.3.2, the ANO-2 ECCS consists of three HPSI pumps, two LPSI pumps, and four SITs. Each HPSI pump injects to one of two high pressure injection headers which feed each cold leg. The LPSI pumps inject to a common header which feeds each cold leg. Each SIT injects to a single cold leg. Two HPSI pumps and the LPSI pumps are automatically actuated by a safety injection actuation signal that is generated by either low pressurizer pressure or high containment pressure. The SITs automatically discharge when the RCS pressure decreases below the SIT pressure.

In the small break LOCA analysis it is assumed that offsite power is lost coincident with reactor trip and, therefore, the HPSI and LPSI pumps must await emergency diesel generator startup and load sequencing before they start. The total delay time assumed is 40 seconds from the time the pressurizer pressure reaches the SIAS setpoint to the time that the HPSI pumps are at speed and aligned to the RCS. For breaks in the reactor coolant pump discharge leg all safety injection flow delivered to the broken discharge leg is modeled to spill out the break.

An analysis of the possible single failures that can occur within the ECCS has shown that the most damaging single failure of ECCS equipment is the failure of an emergency diesel generator to start (Reference 8). This failure causes the loss of both a HPSI and LPSI pump and results in a minimum of safety injection water being available to cool the core.

Based on the above, the following safety injection flows are credited in the small break LOCA analysis for a break in the reactor coolant pump discharge leg: 75% of the flow from one HPSI pump, 50% of the flow from one LPSI pump and 100% of the flow from three SITs. Table 6.3-16 presents the HPSI pump flow rate versus RCS pressure used in the small break LOCA analysis.

6.3.3.2.3.3 Core and System Parameters

The significant core and system parameters used in the small break LOCA analysis are presented in Table 6.3-17. For the 0.05 ft^2 and the 0.06 ft^2 break sizes, the MSSV first bank opening pressure was assumed to be 1125 psia. For the 0.02 ft^2 and 0.04 ft^2 break sizes, the MSSV first bank opening pressure was assumed to be 1103.5 psia. The low pressurizer pressure reactor trip and SIAS setpoints were assumed to be 1400 psia for the 0.02 ft^2, 0.05 ft^2, and 0.06 ft^2 break sizes. The low pressurizer pressure reactor trip was assumed to be 1625 psia, and the low pressurizer pressure SIAS setpoint was assumed to be 1578 psia for

the 0.04 ft^2 break size. The fuel rod initial conditions were taken at the burnup that produced the maximum initial stored energy. The analysis accounts for up to 30% steam generator tube plugging per steam generator.

6.3.3.2.3.4 Containment Parameters

The small break LOCA analysis does not use a detailed containment model. Therefore, other than the containment volume and the initial containment pressure, which are assumed to be 1,820,000 ft^3 and 14.7 psia, respectively, no containment parameters are employed in the analysis.

6.3.3.2.3.5 Break Spectrum

The break spectrum consisted of four reactor coolant pump breaks ranging in size from 0.02 ft^2 to 0.06 ft^2. Table 6.3-18 lists the specific break sizes that were analyzed.

The reactor coolant pump discharge leg was previously determined to be the limiting break location (Reference 8). It is limiting because it maximizes the amount of spillage from the safety injection system.

The break size range of 0.02 ft^2 to 0.06 ft^2 encompasses the breaks sizes for which hot rod cladding heatup is terminated solely by injection from the HPSI pump. It is within this range that the limiting small break LOCA, the 0.05 ft^2 break, resides. Breaks outside this range are either too small to experience any significant core uncovery or are sufficiently large such that injection from the SITs will recover the core and terminate cladding heatup before the cladding temperature approaches the peak cladding temperature calculated for the limiting small break LOCA.

6.3.3.2.3.6 Results and Conclusions

The peak cladding temperatures and cladding oxidation percentages for the small break LOCA analysis are summarized in Table 6.3-19. Table 6.3-20 lists times of interest for the breaks analyzed. As noted in Table 6.3-18, results for the variables listed in Table 6.3-23 are plotted as a function of time in Figures 6.3-22a through 6.3-25h for the breaks analyzed. Peak cladding temperature versus break size is presented in Figure 6.3-26.

Based on the results of the analysis, it is concluded that the ANO-2 ECCS design satisfies the Acceptance Criteria of 10CFR50.46 for a spectrum of small break LOCAs.

References:

8. CENPD-137-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," August 1974.

CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977.

10. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report," 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.

11. CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

27. CENPD-138P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.

CENPD-138P, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," February 1975.

CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.

30. CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

CENPD-135P, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.

CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

82. Letter from K. Kniel (NRC) to A.E. Scherer (CE), "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P," September 27, 1977.

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

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

Table 6.3-16

HIGH PRESSURE SAFETY INJECTION PUMP MINIMUM DELIVERED FLOW TO RCS (ASSUMING ONE EMERGENCY GENERATOR FAILED)

RCS Pressure, psia	Flow Rate, gpm
1348	0.0
1321	82.6
1284	138.6
1248	186.5
1142	264.4
1071	314.1
990	361.5
899	407.6
800	458.5
692	507.7
577	554.7
456	602.6
327	651.6
191	702.3
46	750.6
31	755.1
22	757.8
14.7	760.0

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 6.3-17

SYSTEM PARAMETERS AND INITIAL CONDITIONS FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION

Quantity	Value	Units
Reactor power level (103% of rated power)	2900	MWt
Peak linear heat generation rate (PLHGR)	13.5	kW/ft
Axial shape index	-0.3	asiu
Gap conductance at PLHGR	1582	BTU-hr-ft^2-°F
Fuel centerline temperature at PLHGR	3334	°F
Fuel average temperature at PLHGR	2115	°F
Hot rod gas pressure	1123	psia
Moderator temperature coefficient at initial density	0.0×10^{-4}	$\Delta\rho$/°F
RCS flow rate	108.4×10^6	lbm/hr
Core flow rate	104.6×10^6	lbm/hr
RCS pressure	2250	psia
Cold leg temperature	556.7	°F
Hot leg temperature	622.7	°F
Plugged tubes per steam generator	30	%
MSSV first bank opening pressure	1103.5	psia
Low pressurizer pressure reactor trip setpoint	1625	psia
Low pressurizer pressure SIAS setpoint	1578	psia
Safety injection tank pressure	550	psia

Table 6.3-18

BREAK SPECTRUM
FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION

Break Size and Location	Abbreviation	Figure No.
0.06 ft^2 Break in Pump Discharge Leg	0.06 ft^2/PD	6.3-22
0.05 ft^2 Break in Pump Discharge Leg	0.05 ft^2/PD	6.3-23
0.04 ft^2 Break in Pump Discharge Leg	0.04 ft^2/PD	6.3-24
0.02 ft^2 Break in Pump Discharge Leg	0.02 ft^2/PD	6.3-25

Table 6.3-19

PEAK CLADDING TEMPERATURES AND OXIDATION PERCENTAGES FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION

Break	Peak Cladding Temperature (°F)[a]	Maximum Cladding Oxidation (%)[b]	Hot Rod Oxidation (%)[c]
0.06 ft^2/PD	2003	4.78	<0.726
0.05 ft^2/PD	2011	5.47	<0.835
0.04 ft^2/PD	1870	3.37	<0.567
0.02 ft^2/PD	1671	1.73	<0.318

(a) Acceptance criterion is 2200°F.
(b) Acceptance criterion is 17%.
(c) Acceptance criterion is 1.0% core-wide cladding oxidation. Rod-average oxidation of the hot rod is given as a conservative representation of the core-wide cladding oxidation.

Table 6.3-20

TIMES OF INTEREST
FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION
(Seconds after Break)

Break	HPSI Flow Delivered to RCS (sec)	LPSI Flow Delivered to RCS (sec)	SIT Flow Delivered to RCS (sec)	Peak Cladding Temperature Occurs (sec)
0.06 ft^2/PD	169	(a)	1290[b]	1541
0.05 ft^2/PD	192	(a)	1592[b]	1624
0.04 ft^2/PD	179	(a)	(c)	1943
0.02 ft^2/PD	389	(a)	(c)	3411

(a) Calculation completed before LPSI flow delivery to RCS begins.
(b) SIT injection calculated to begin but not credited in analysis.
(c) Calculation completed before SIT injection begins.

Table 6.3-23

VARIABLES PLOTTED AS A FUNCTION OF TIME FOR EACH BREAK OF THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION

Variable	Figure Designation
Normalized Total Core Power	a
Inner Vessel Pressure	b
Break Flow Rate	c
Inner Vessel Inlet Flow Rate	d
Inner Vessel Two-Phase Mixture Level	e
Heat Transfer Coefficient at Hot Spot	f
Coolant Temperature at Hot Spot	g
Cladding Temperature at Hot Spot	h

Figure 6.3-22a

0.06 ft^2/PD BREAK

NORMALIZED TOTAL CORE POWER

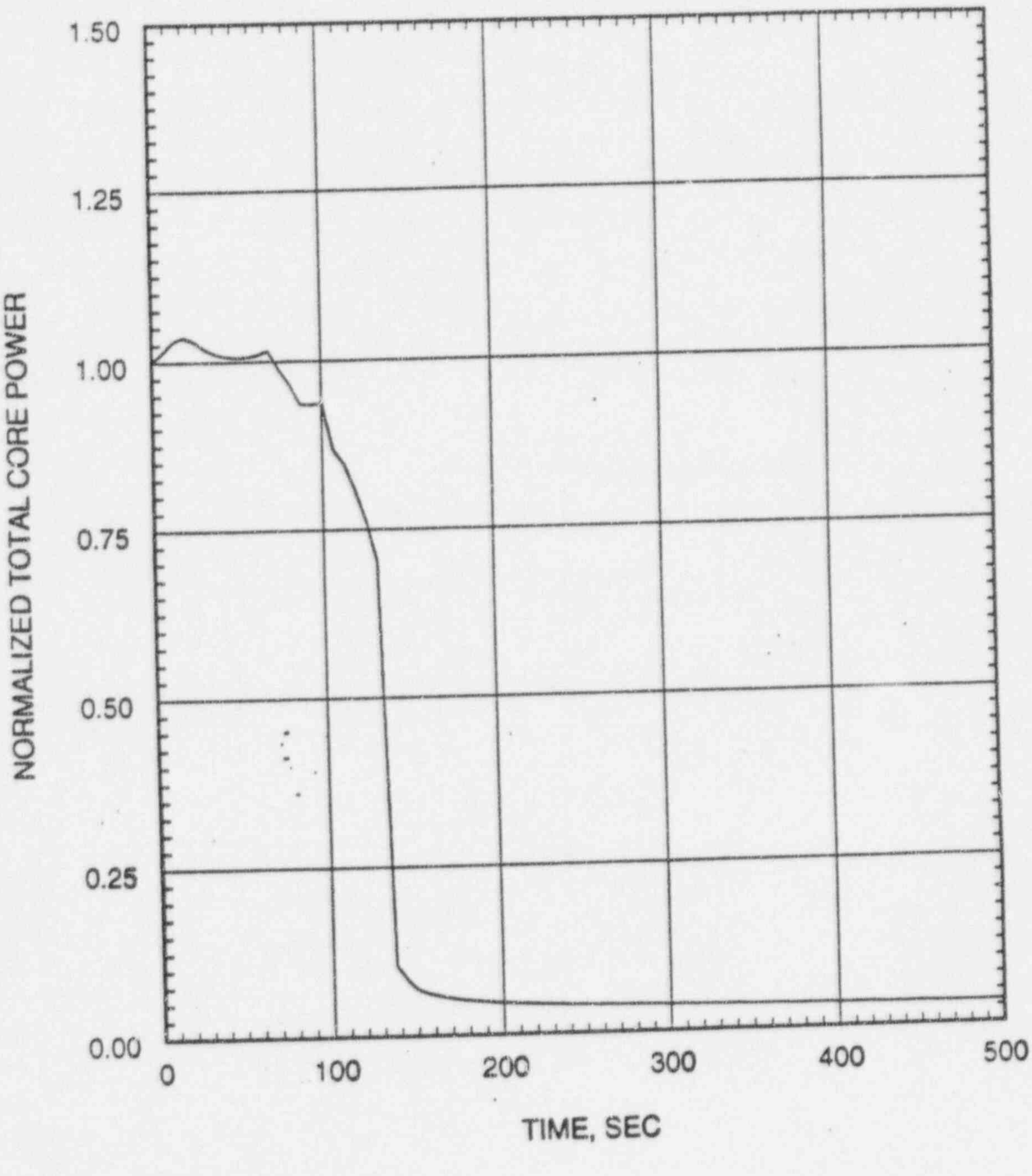

Figure 6.3-22b

0.06 ft^2/PD BREAK

INNER VESSEL PRESSURE

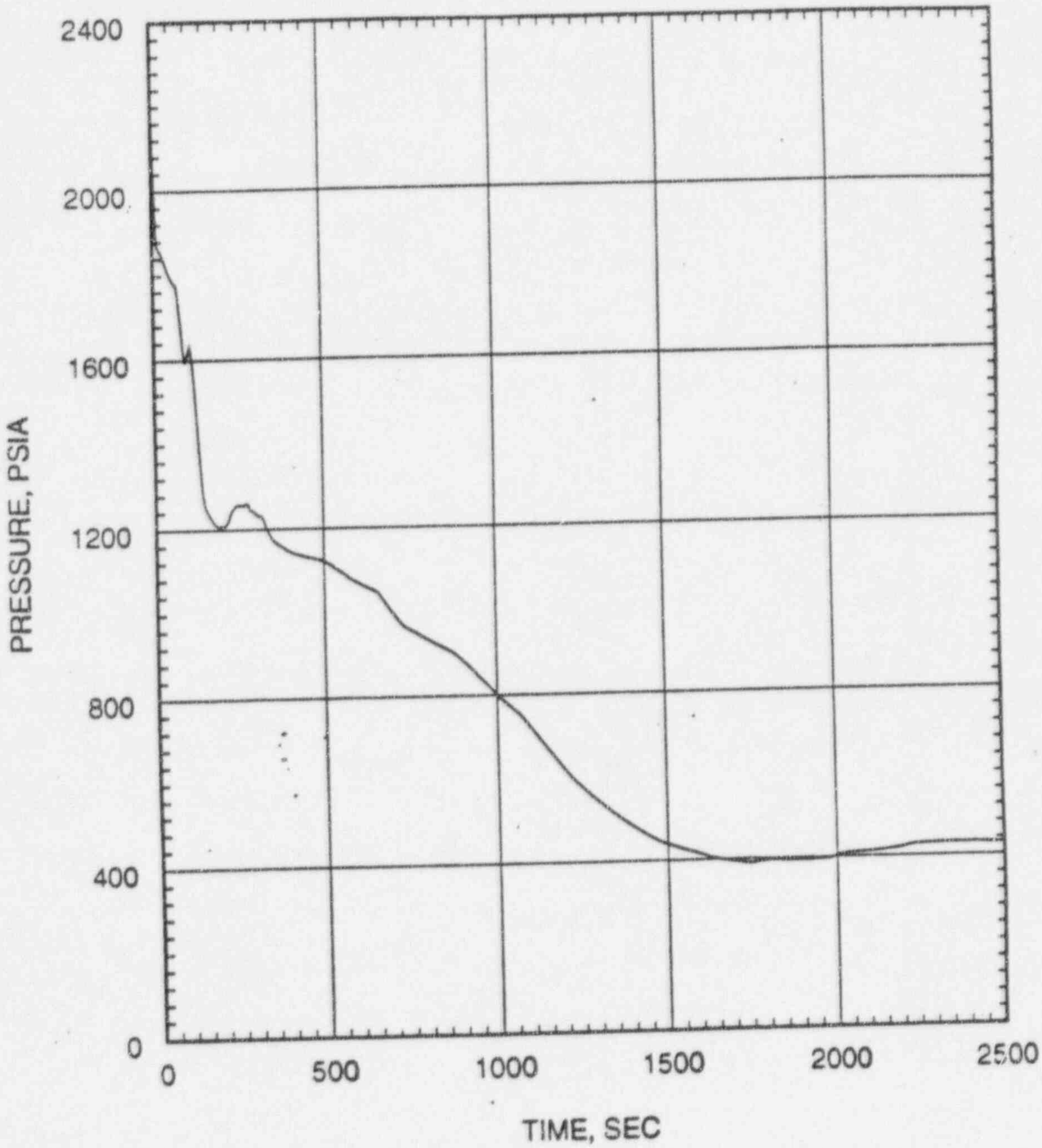

Figure 6.3-22c

0.06 ft^2/PD BREAK

BREAK FLOW RATE

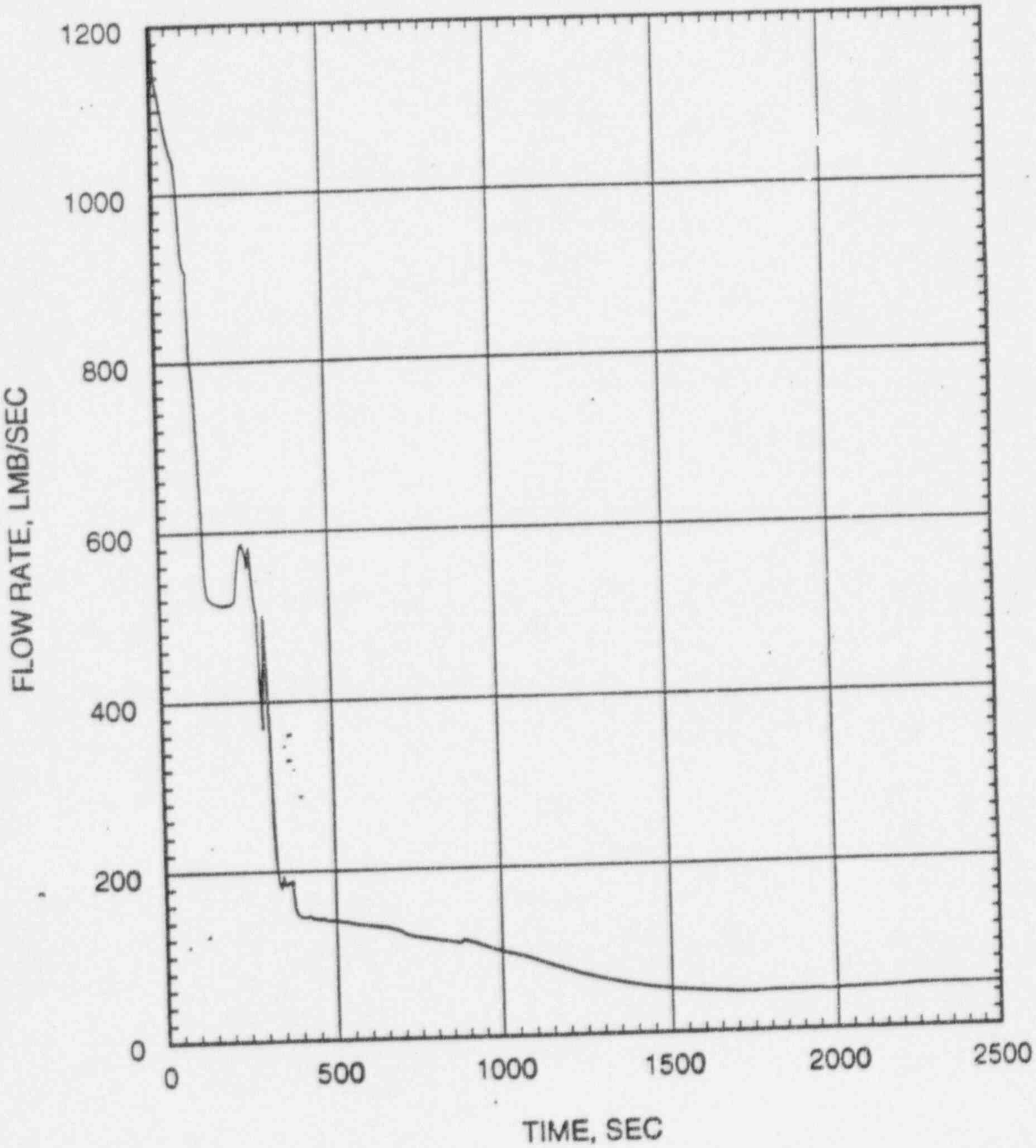

Figure 6.3-22d

0.06 ft^2/PD BREAK

INNER VESSEL INLET FLOW RATE

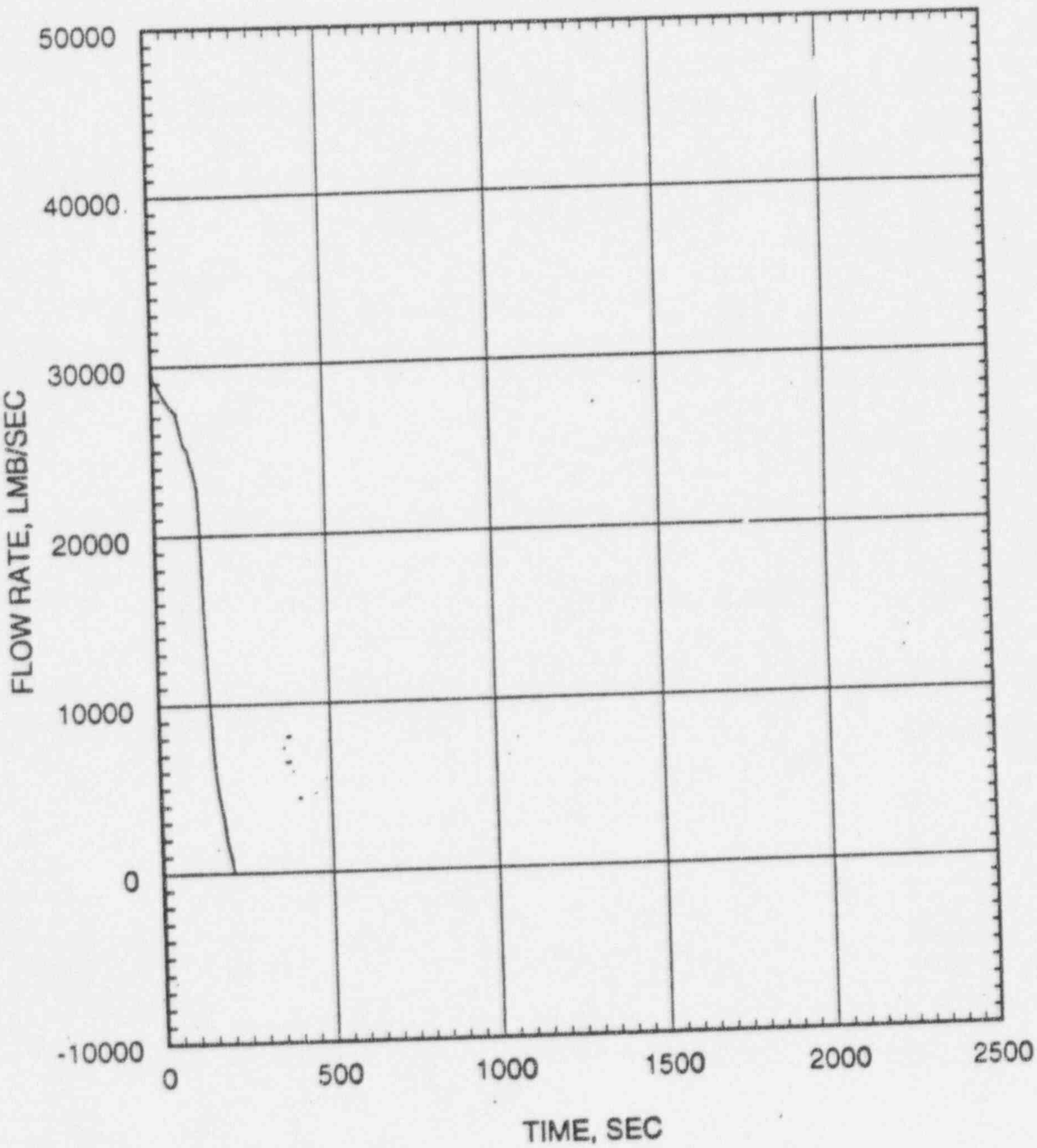

Figure 6.3-22e

0.06 ft^2/PD BREAK

INNER VESSEL TWO-PHASE MIXTURE LEVEL

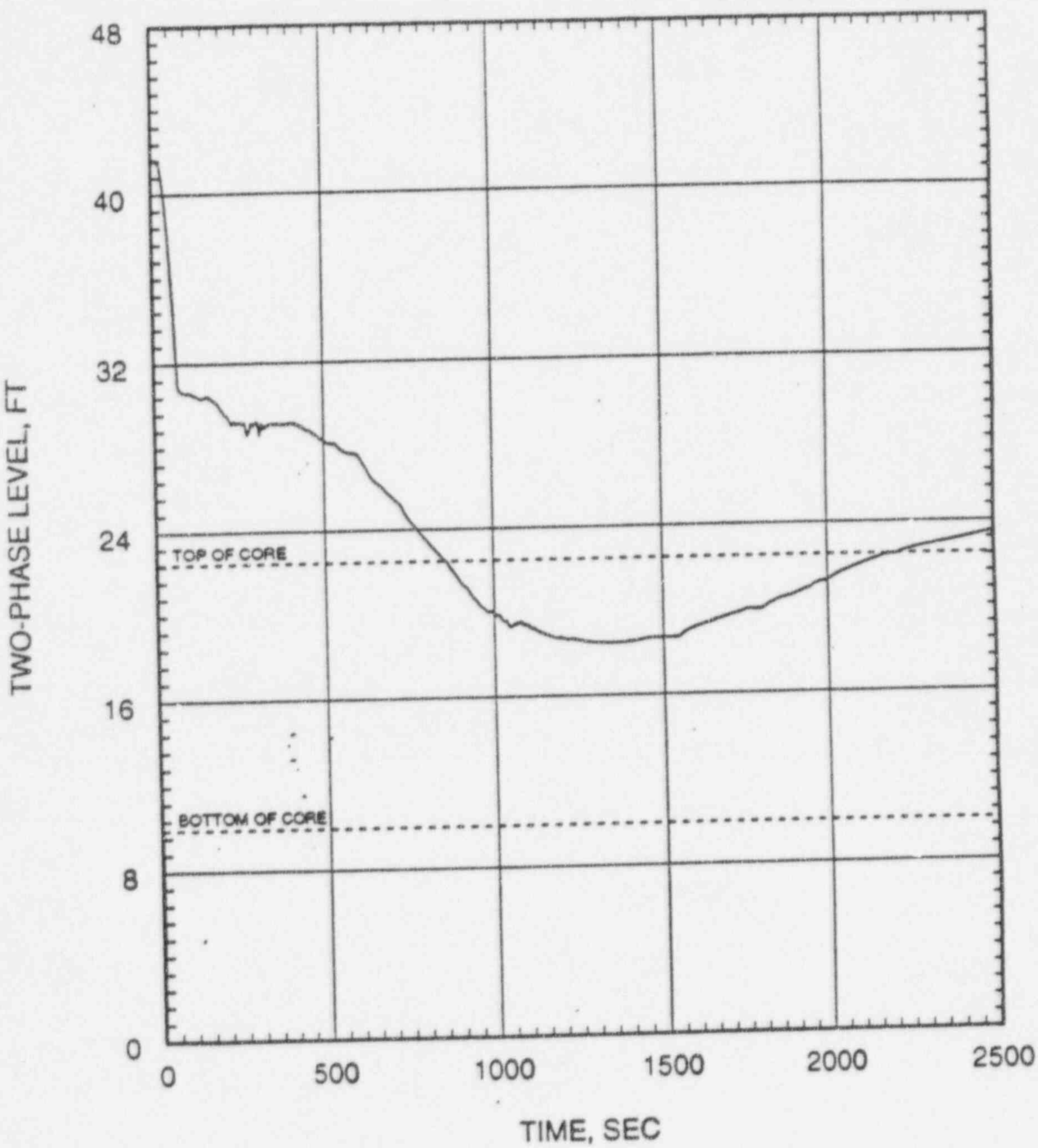

Figure 6.3-22f

$0.06\ ft^2$/PD BREAK

HEAT TRANSFER COEFFICIENT AT HOT SPOT

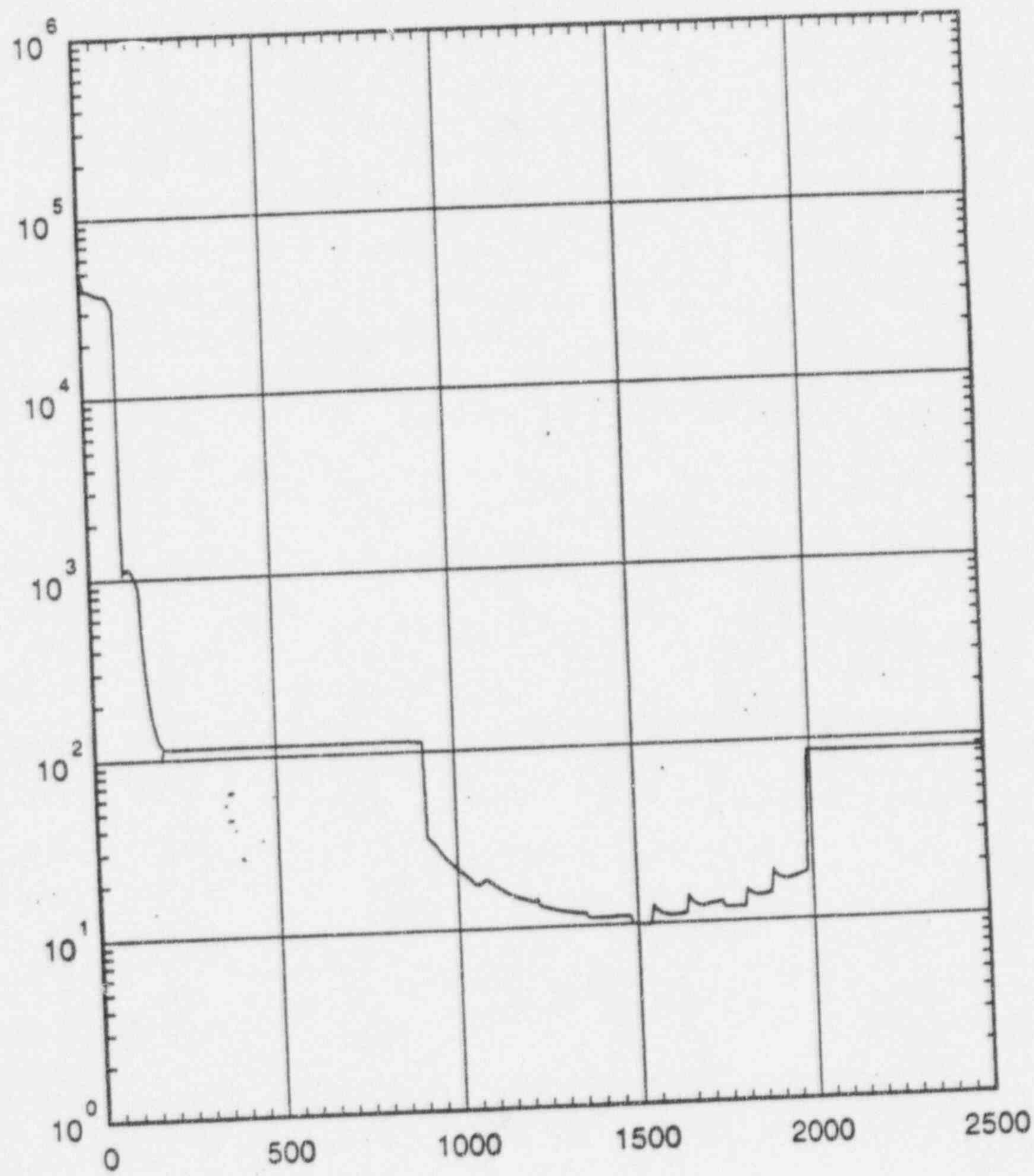

Figure 6.3-22g

0.06 ft^2/PD BREAK

COOLANT TEMPERATURE AT HOT SPOT

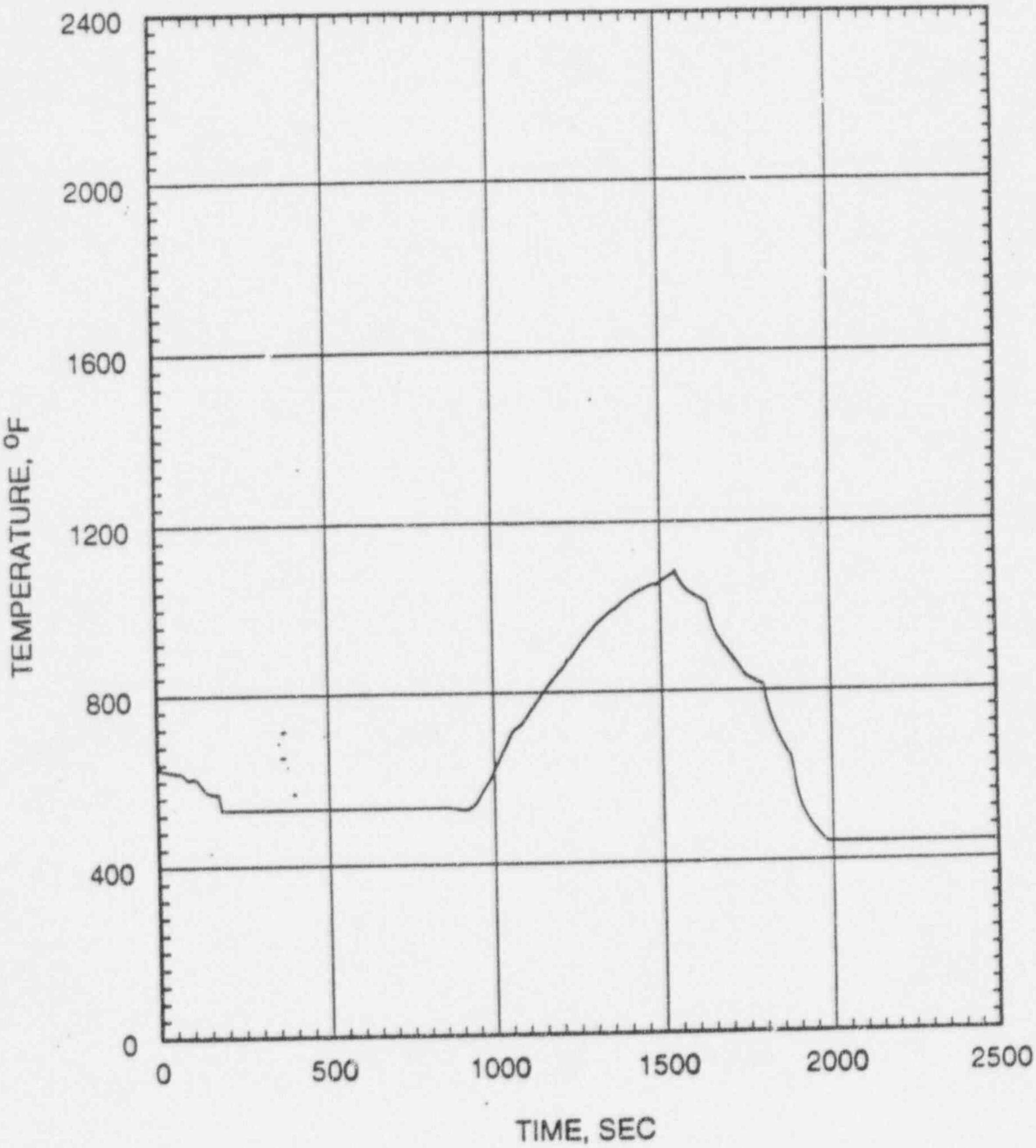

Figure 6.3-22h

0.06 ft^2/PD BREAK

CLADDING TEMPERATURE AT HOT SPOT

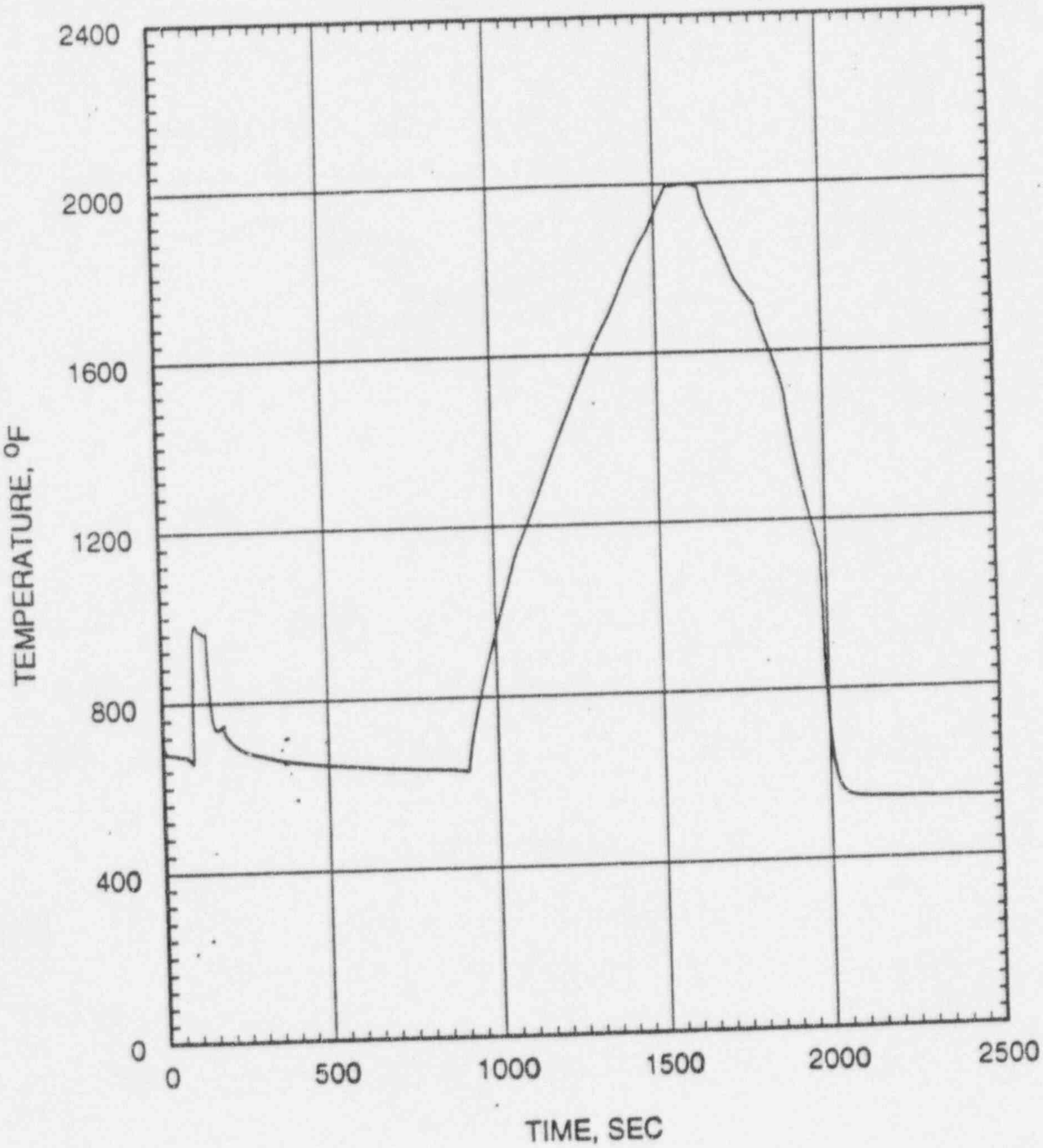

Figure 6.3-23a

0.05 ft^2/PD BREAK

NORMALIZED TOTAL CORE POWER

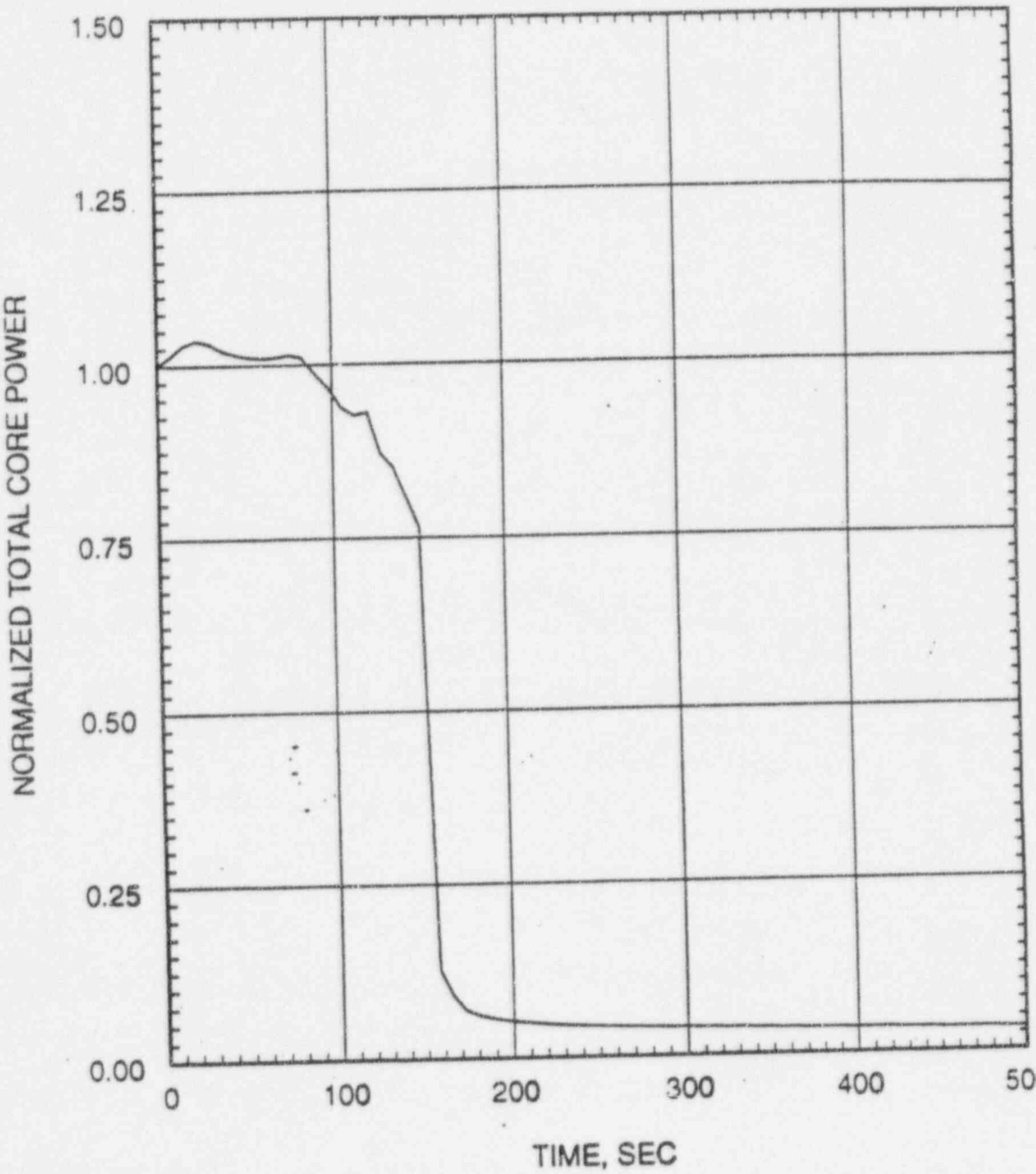

Figure 6.3-23b

0.05 ft^2/PD BREAK

INNER VESSEL PRESSURE

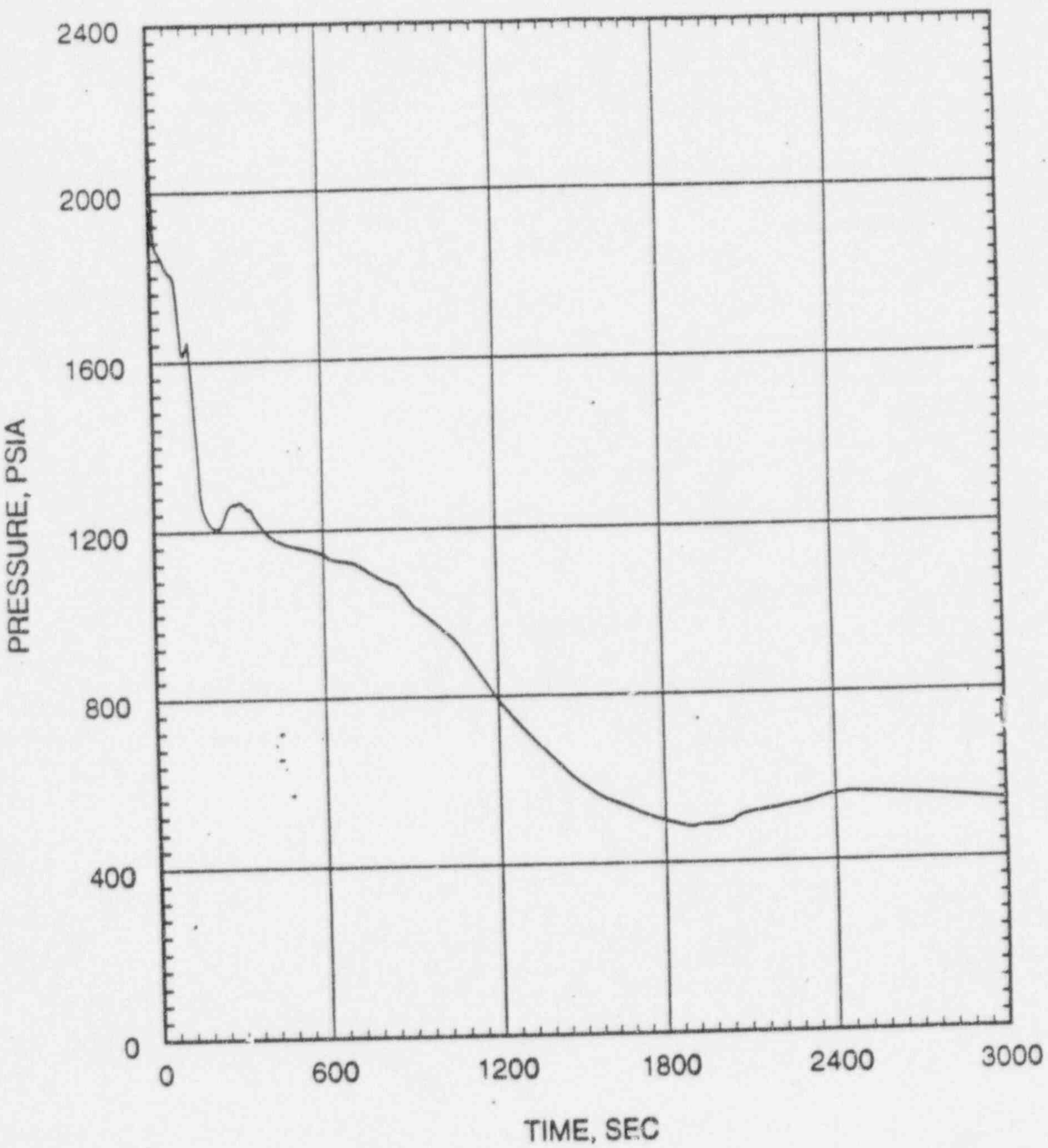

Figure 6.3-23c
0.05 ft^2/PD BREAK
BREAK FLOW RATE

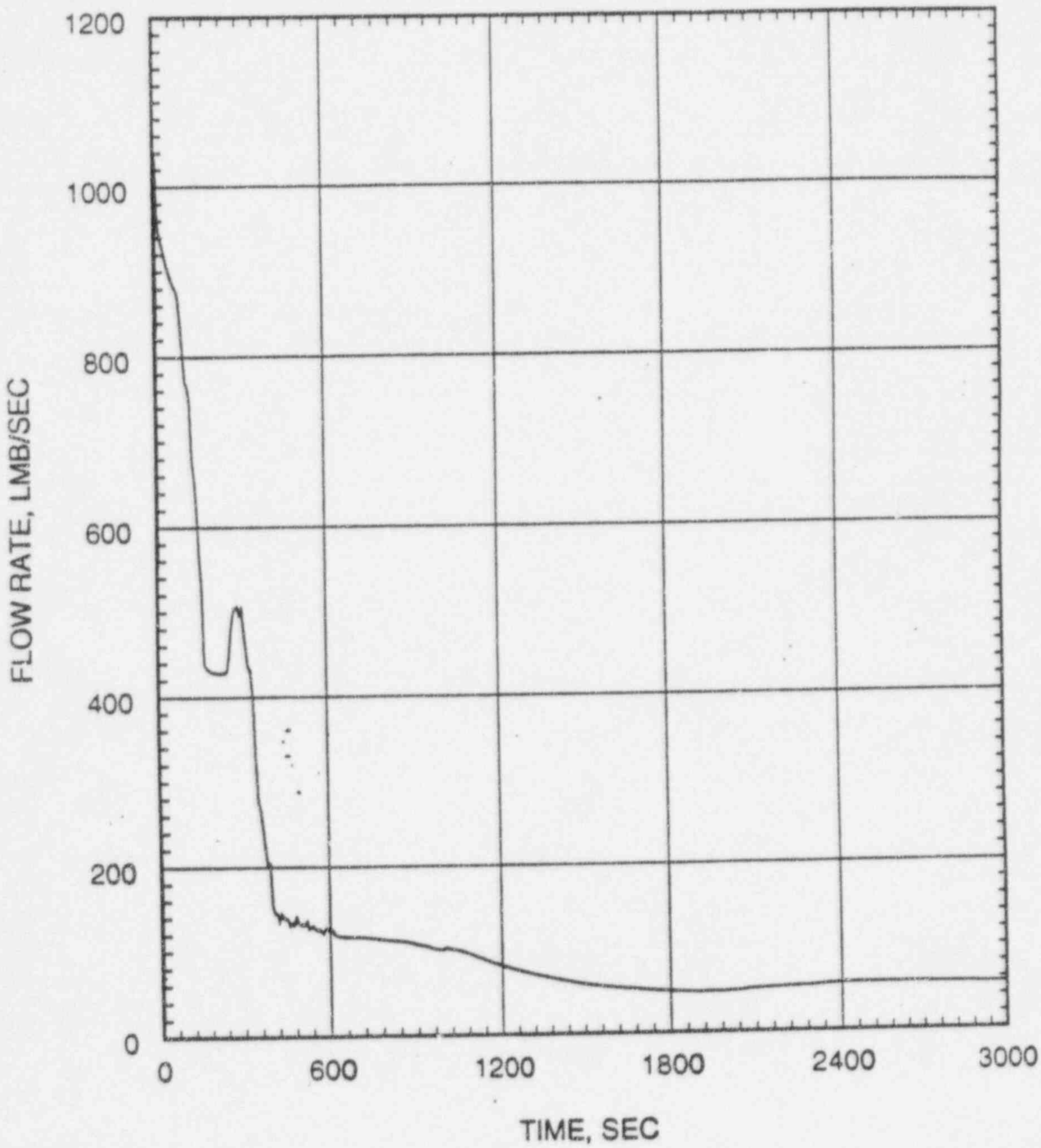

Figure 6.3-23d

0.05 ft^2/PD BREAK

INNER VESSEL INLET FLOW RATE

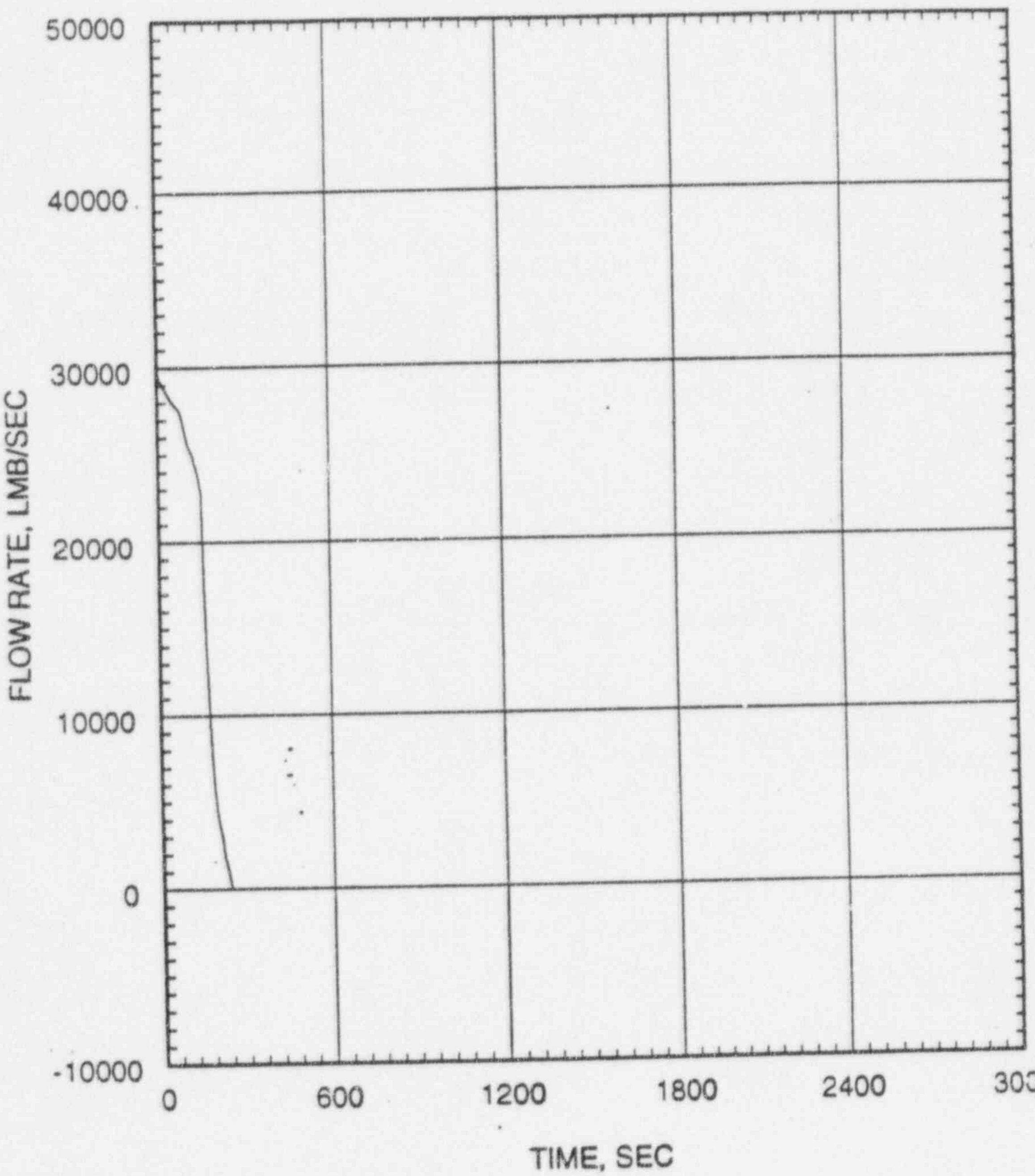

Figure 6.3-23e

0.05 ft^2/PD BREAK

INNER VESSEL TWO-PHASE MIXTURE LEVEL

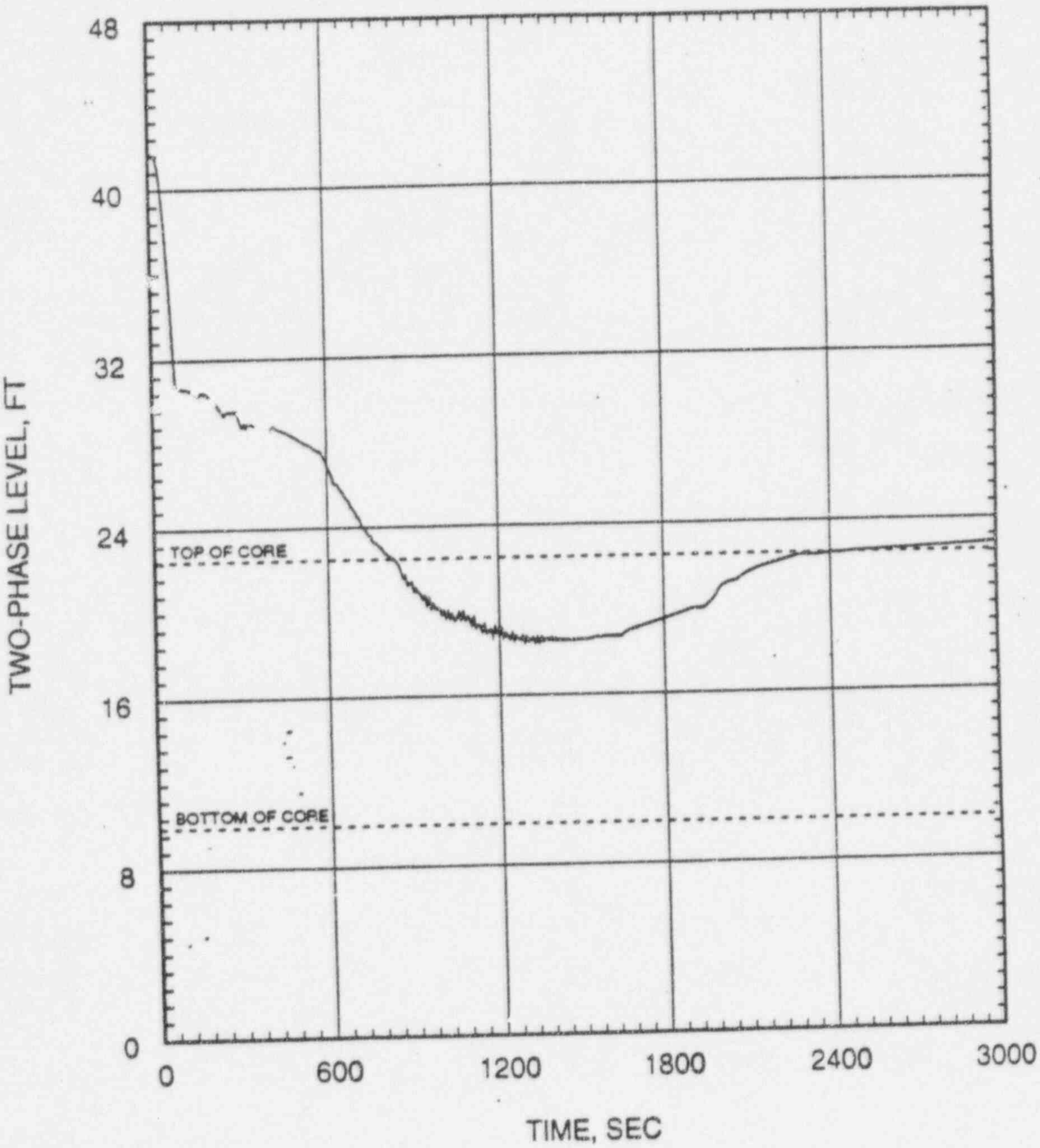

Figure 6.3-23f

0.05 ft^2/PD BREAK

HEAT TRANSFER COEFFICIENT AT HOT SPOT

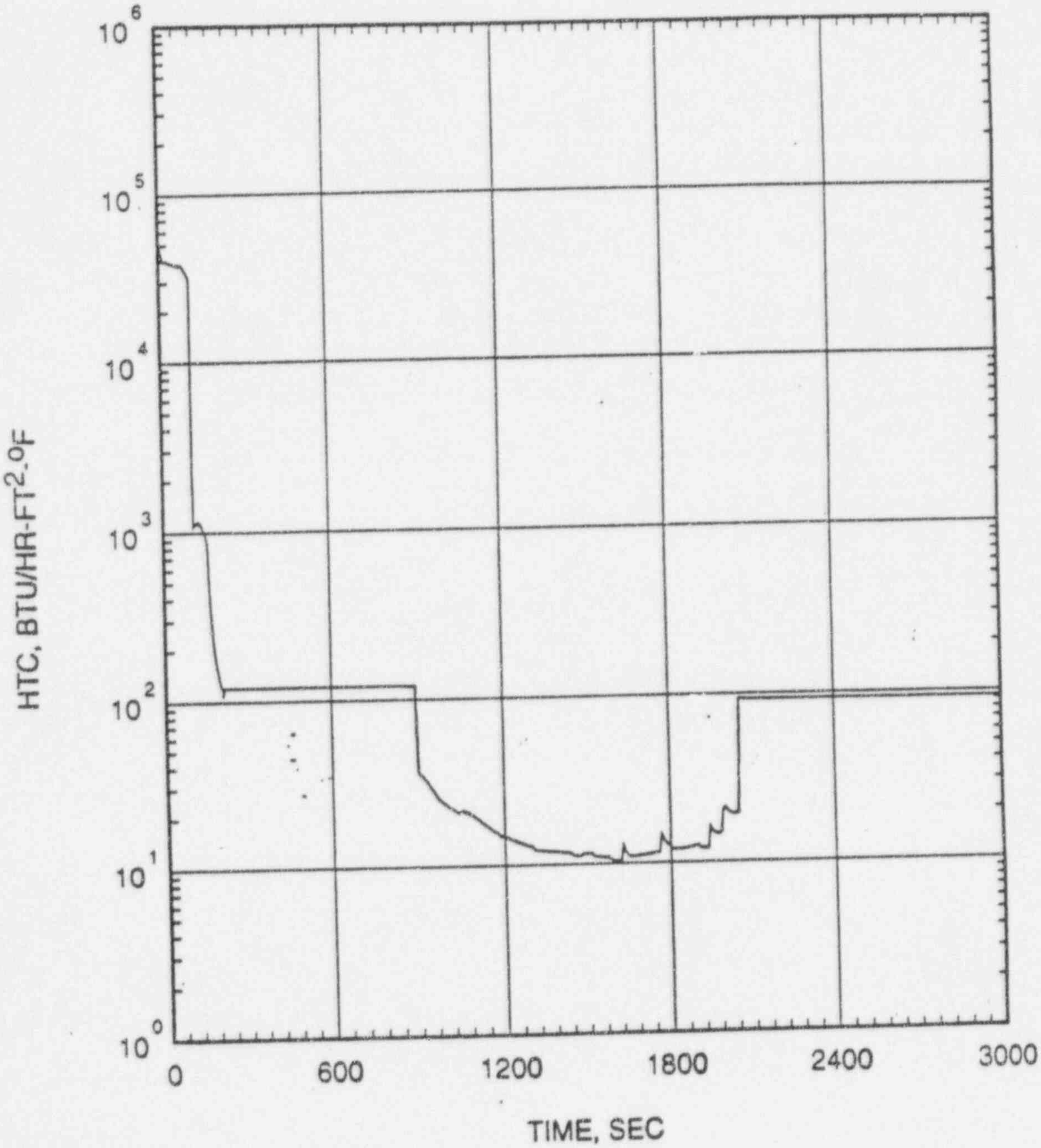

Figure 6.3-23g

0.05 ft^2/PD BREAK

COOLANT TEMPERATURE AT HOT SPOT

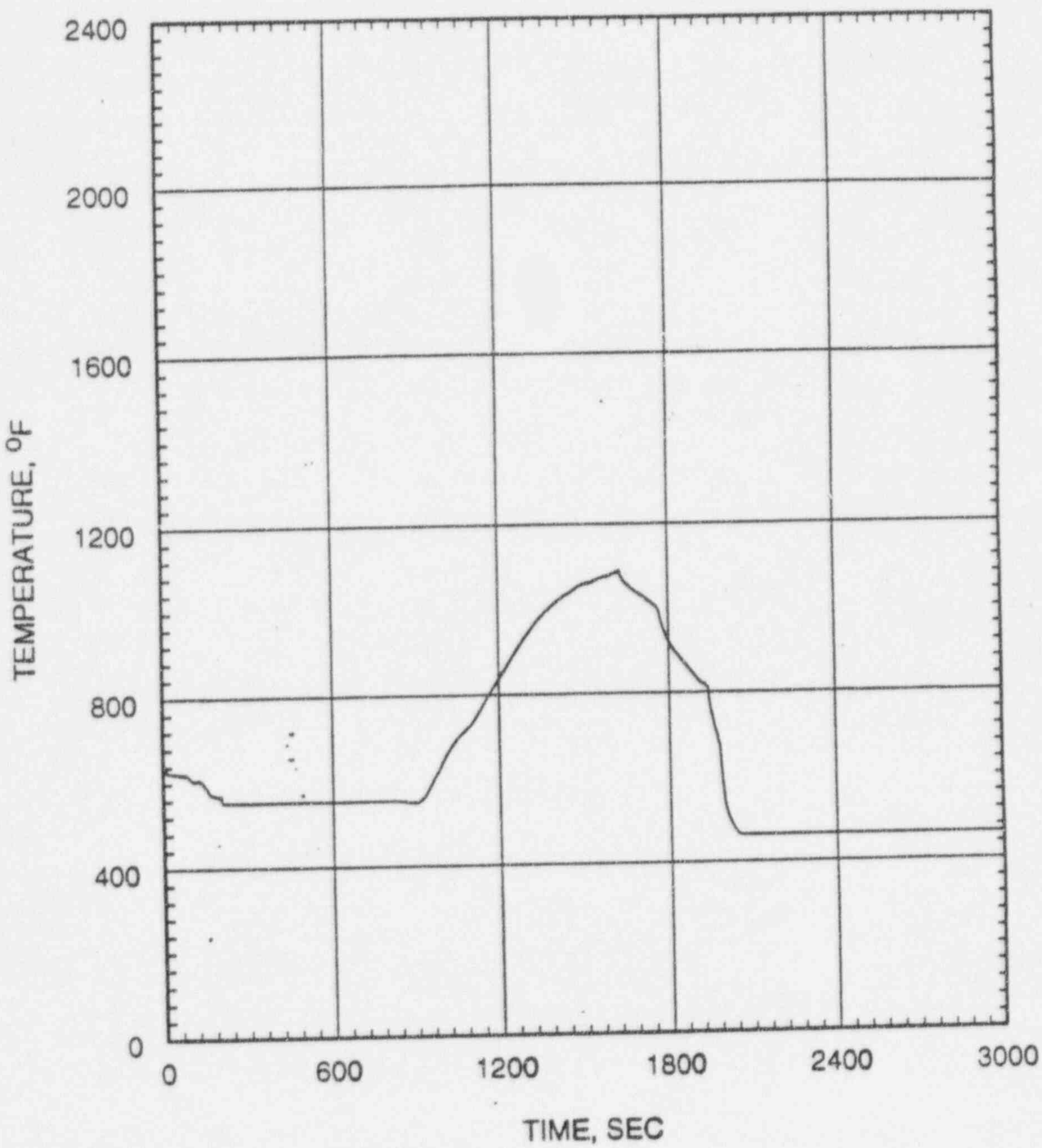

Figure 6.3-23h

0.05 ft^2/PD BREAK

CLADDING TEMPERATURE AT HOT SPOT

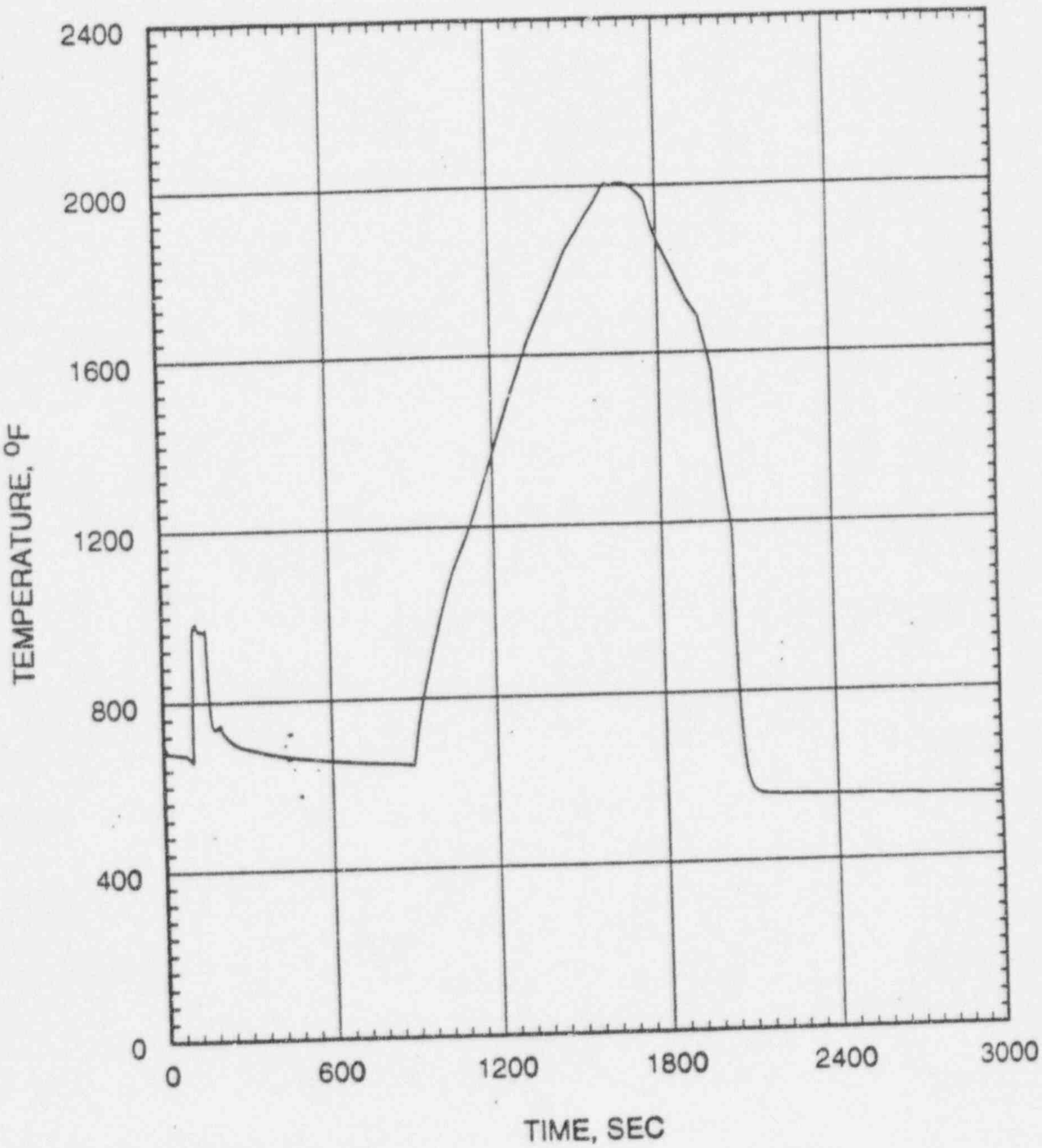

Figure 6.3-24a

0.04 ft^2/PD BREAK

NORMALIZED TOTAL CORE POWER

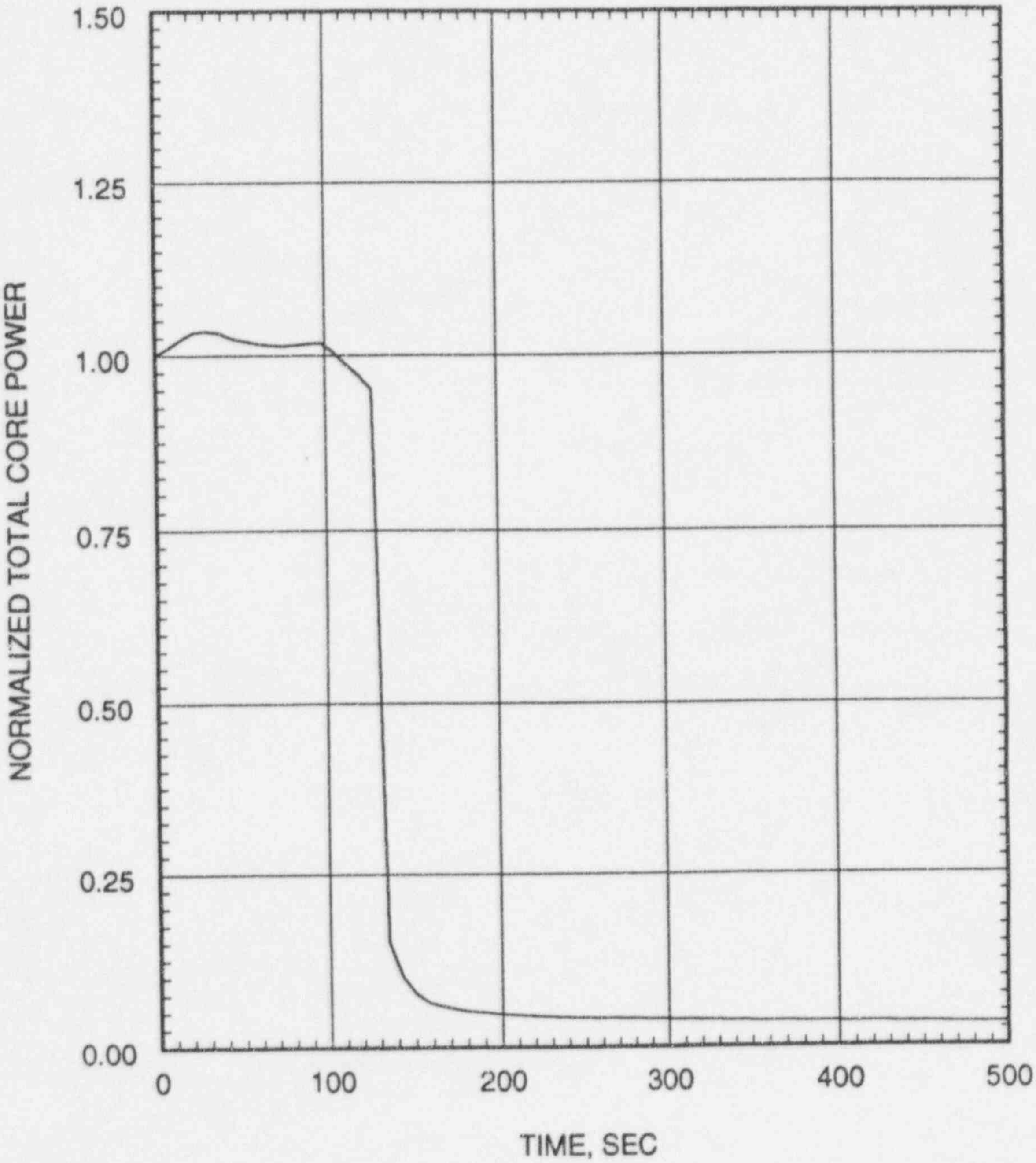

Figure 6.3-24b

0.04 ft^2/PD BREAK

INNER VESSEL PRESSURE

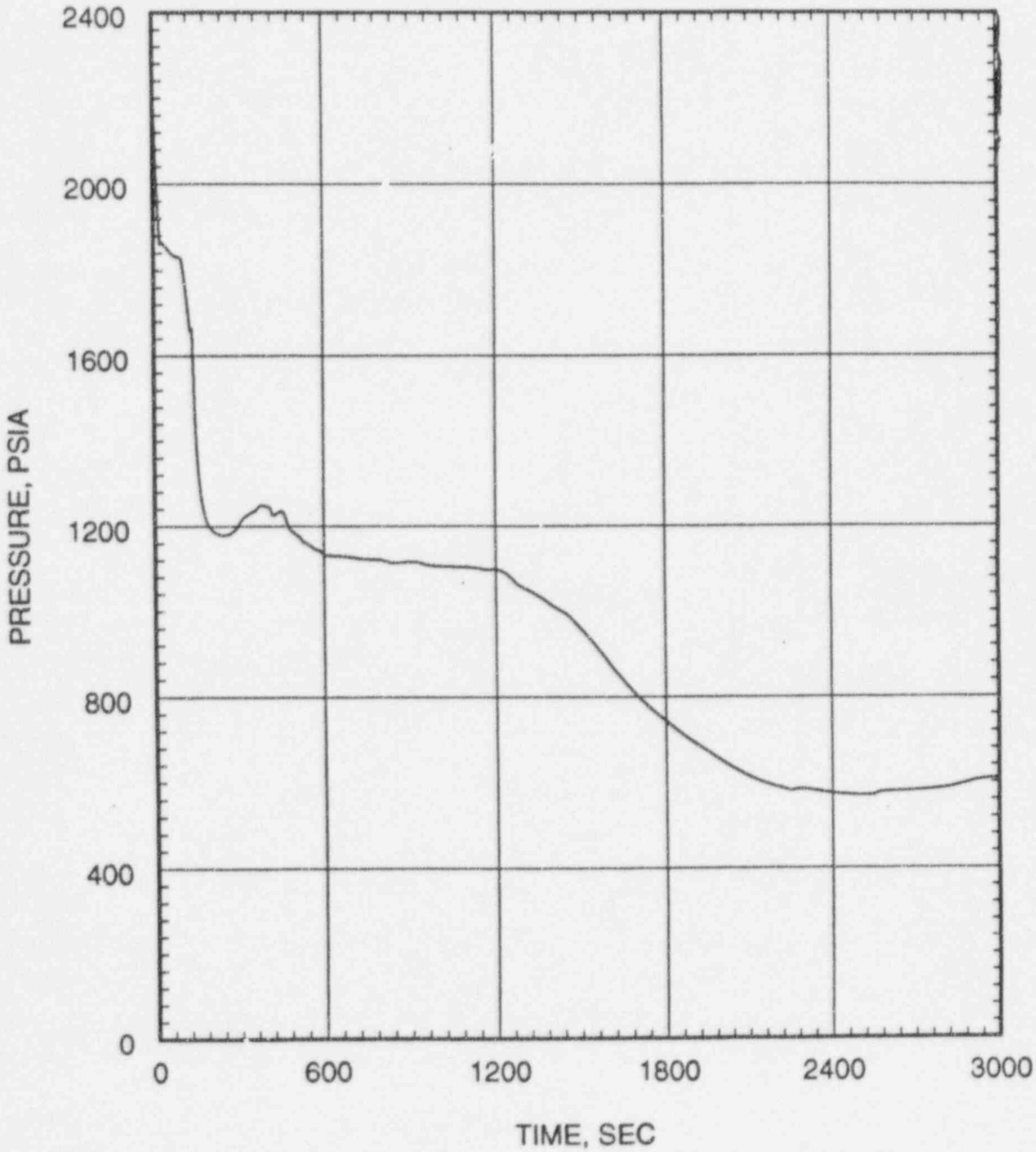

Figure 6.3-24c
0.04 ft^2/PD BREAK
BREAK FLOW RATE

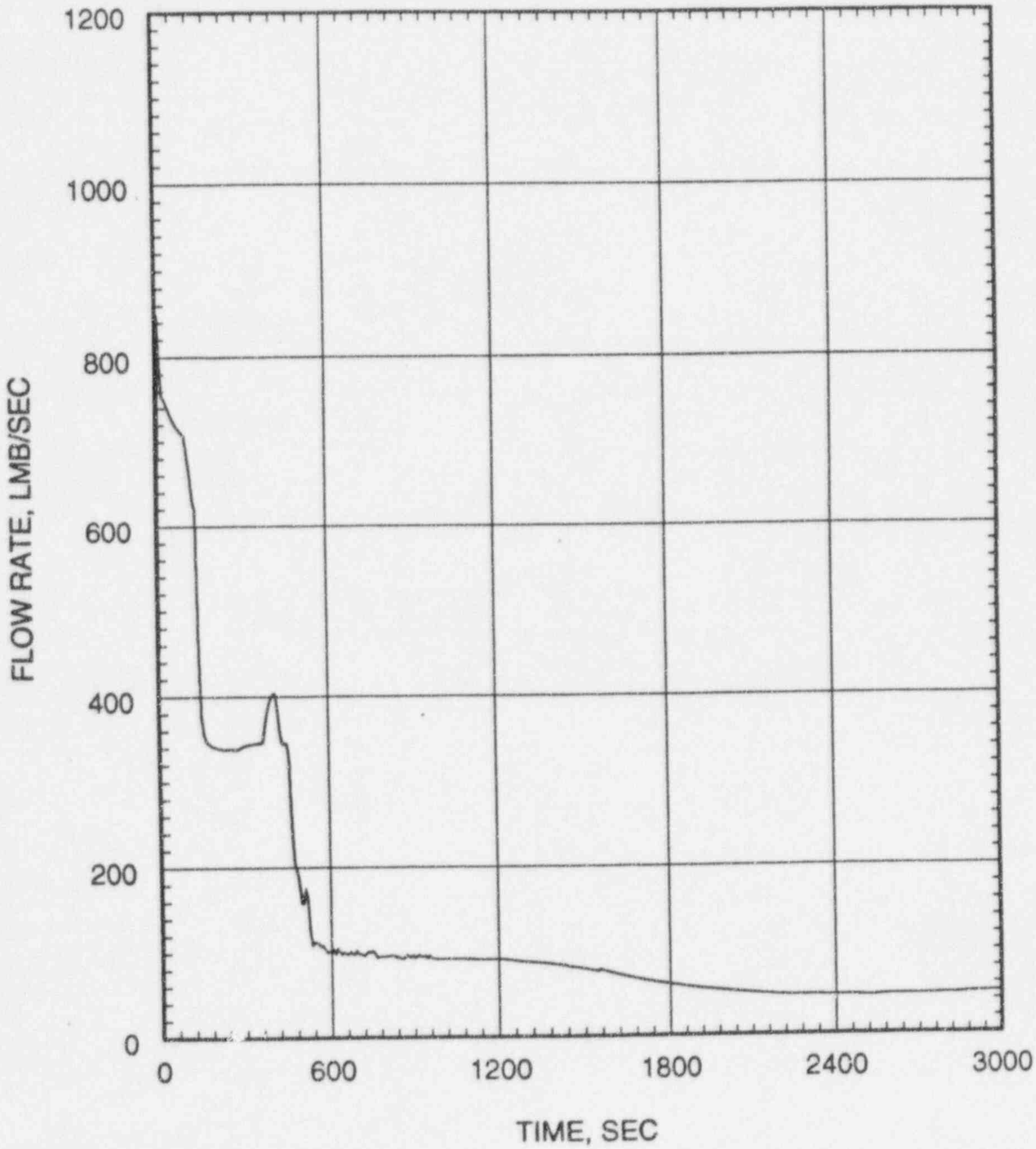

Figure 6.3-24d

0.04 ft^2/PD BREAK

INNER VESSEL INLET FLOW RATE

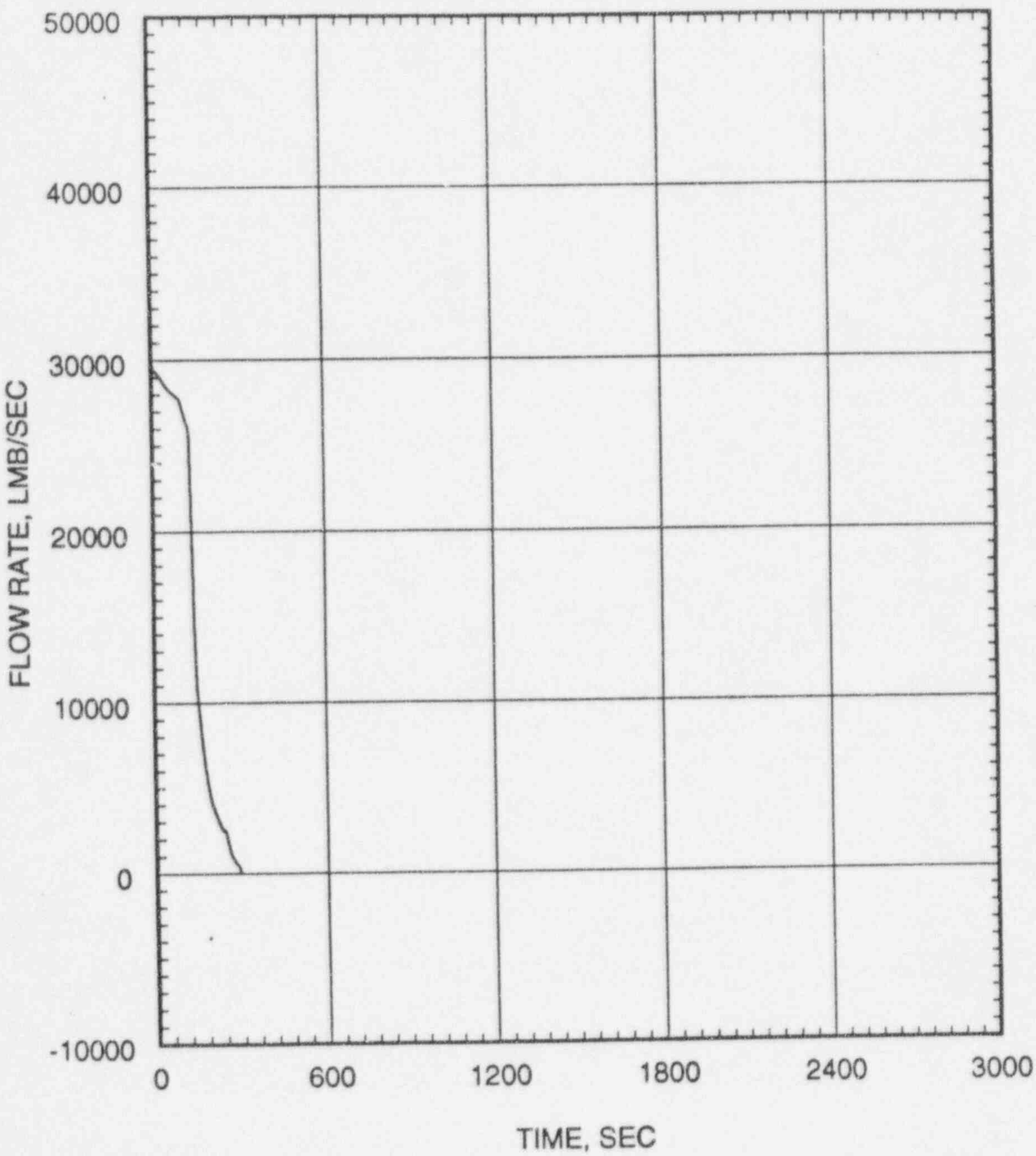

Figure 6.3-24e

0.04 ft^2/PD BREAK

INNER VESSEL TWO-PHASE MIXTURE LEVEL

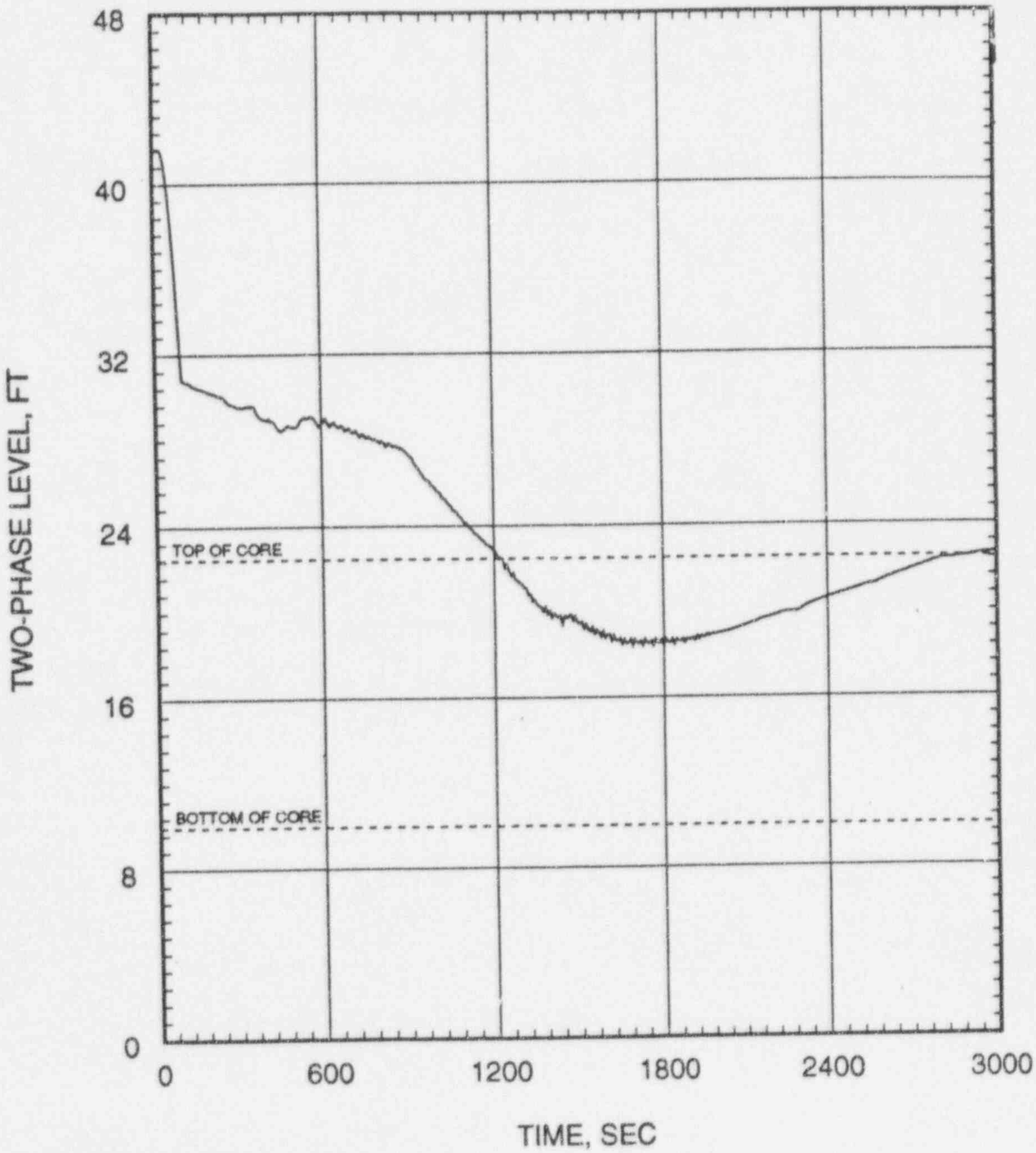

Figure 6.3-24f

0.04 ft^2/PD BREAK

HEAT TRANSFER COEFFICIENT AT HOT SPOT

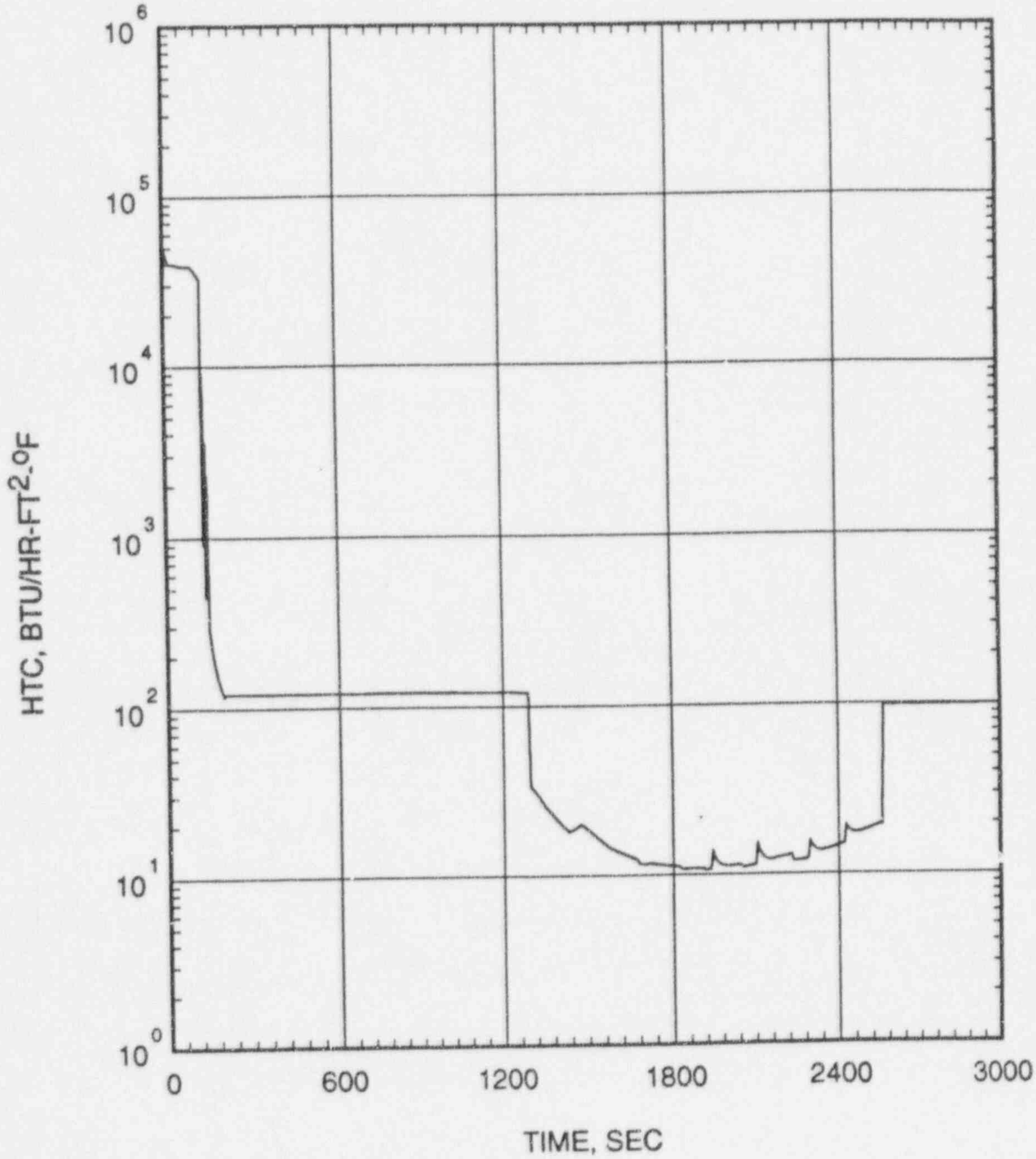

Figure 6.3-24g

0.04 ft^2/PD BREAK

COOLANT TEMPERATURE AT HOT SPOT

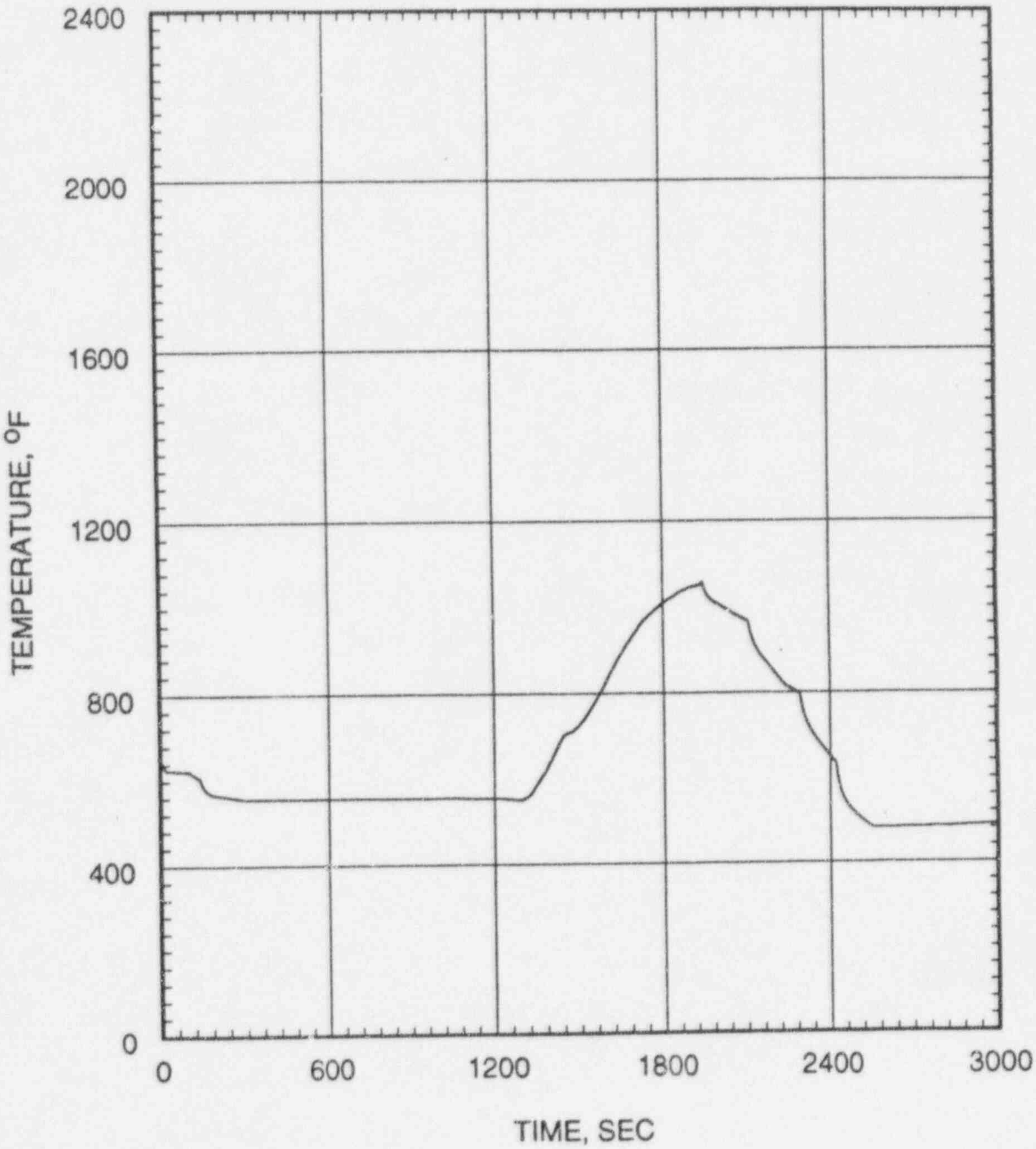

Figure 6.3-24h

$0.04\ ft^2$/PD BREAK

CLADDING TEMPERATURE AT HOT SPOT

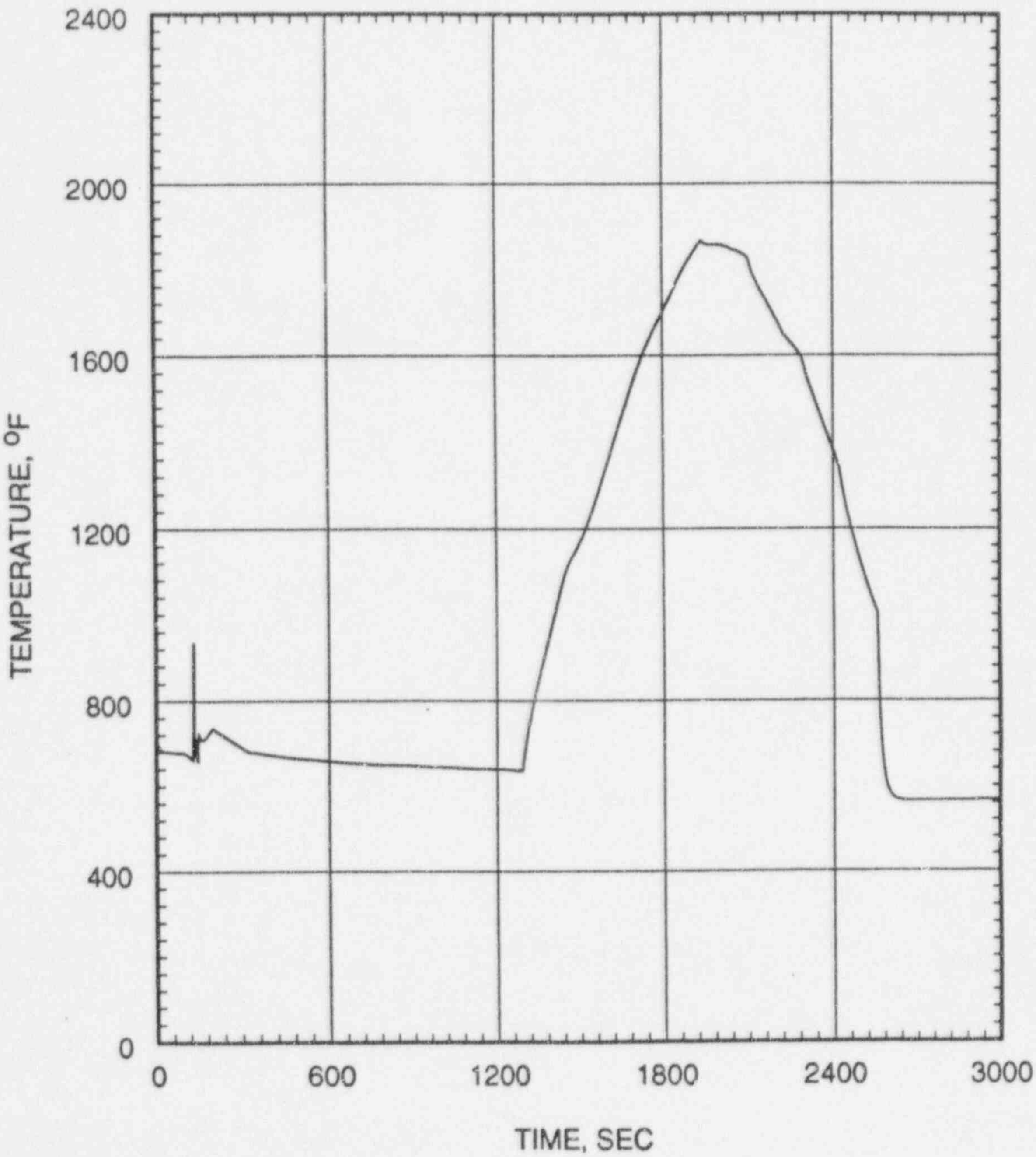

Figure 6.3-25a

0.02 ft^2/PD BREAK

NORMALIZED TOTAL CORE POWER

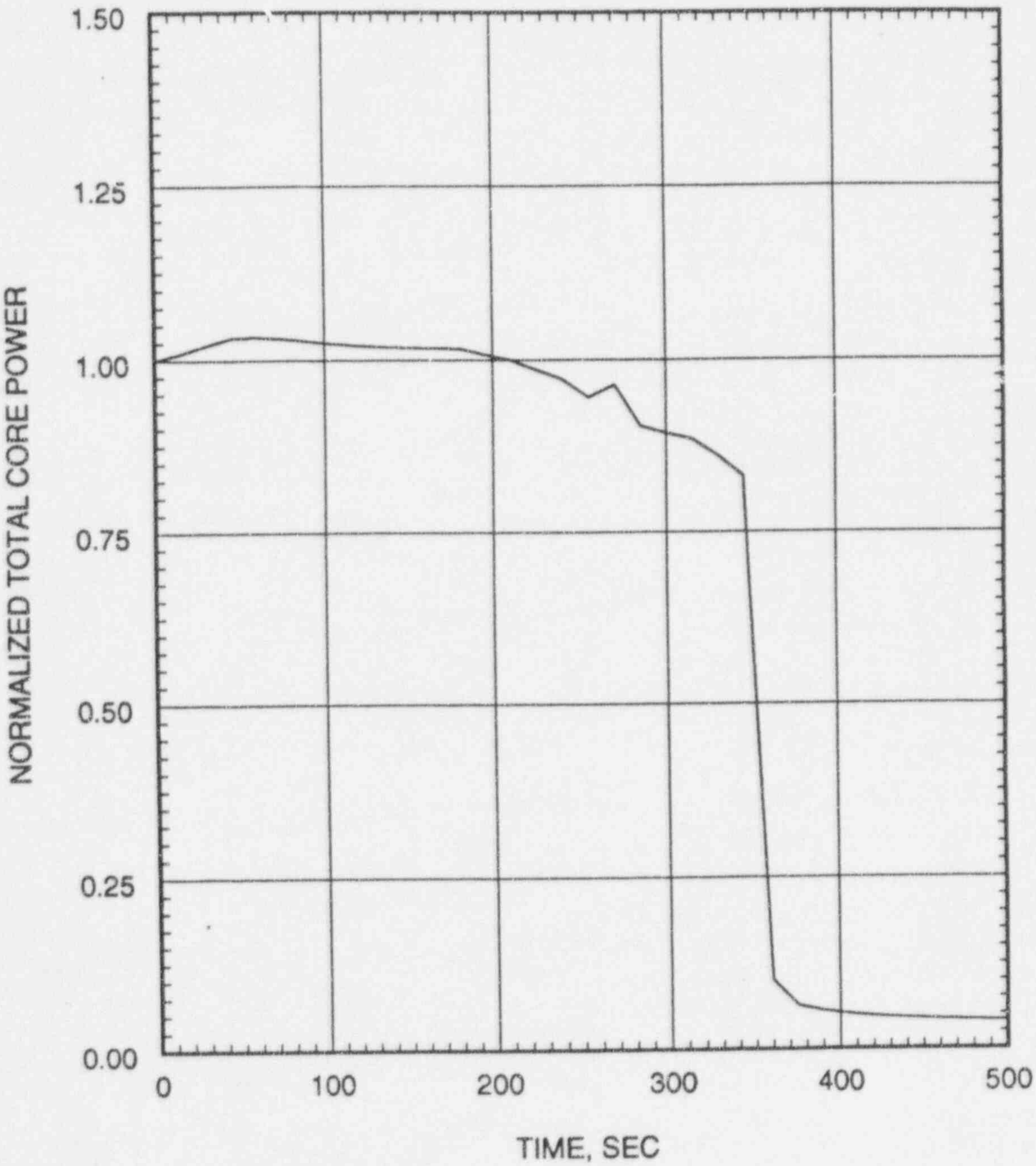

Figure 6.3-25b

0.02 ft^2/PD BREAK

INNER VESSEL PRESSURE

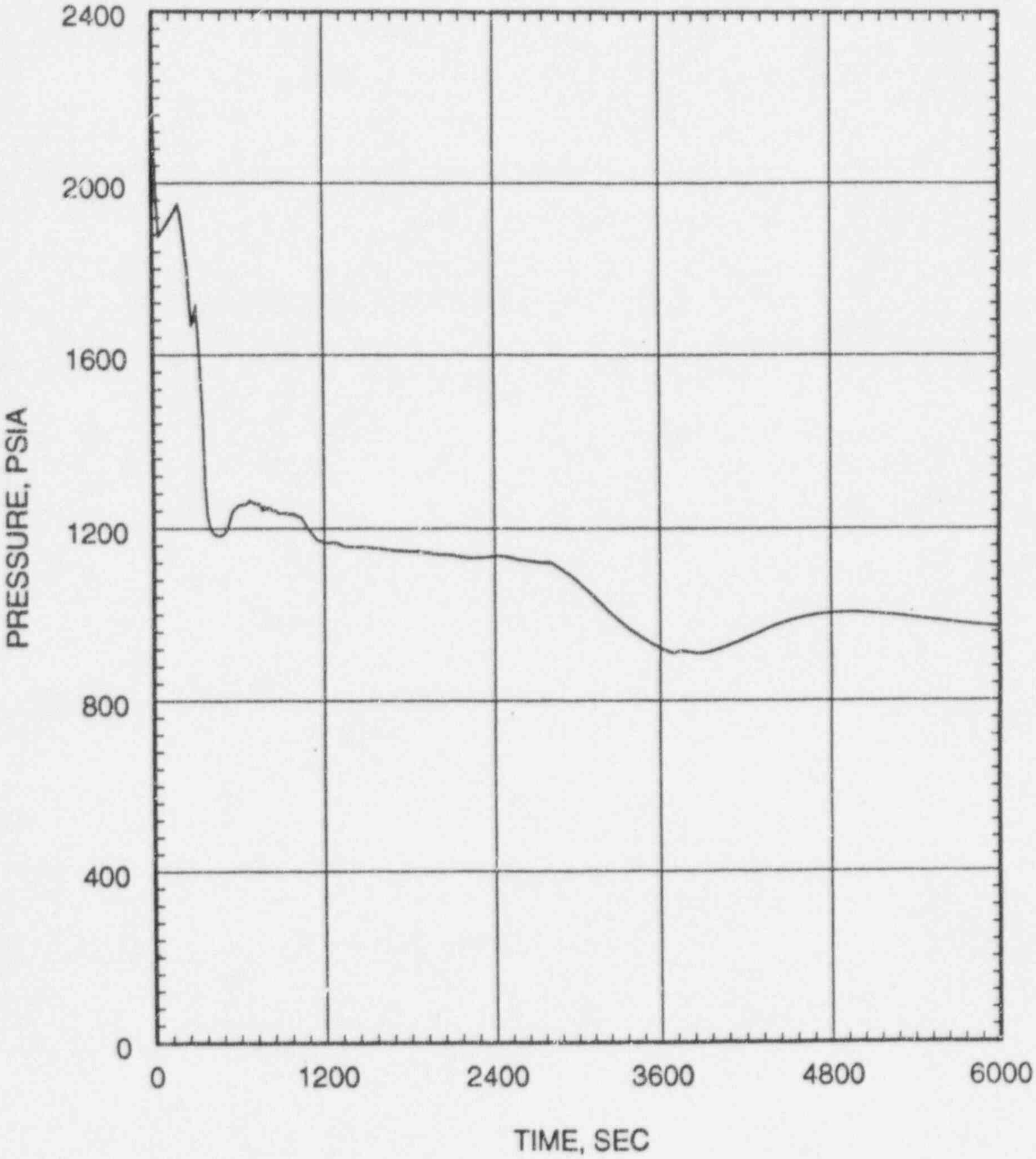

Figure 6.3-25c

0.02 ft^2/PD BREAK

BREAK FLOW RATE

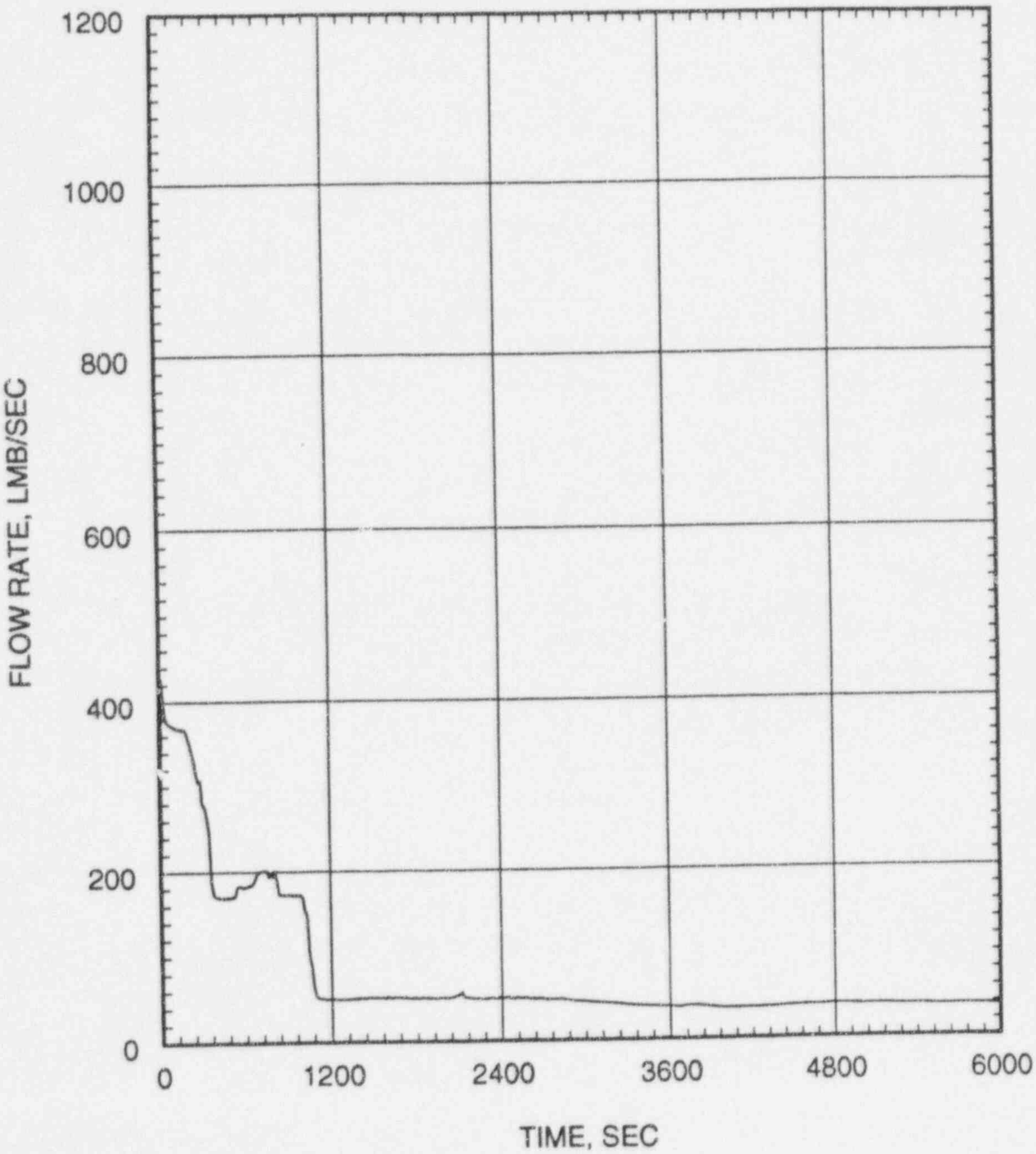

Figure 6.3-25d

0.02 ft^2/PD BREAK

INNER VESSEL INLET FLOW RATE

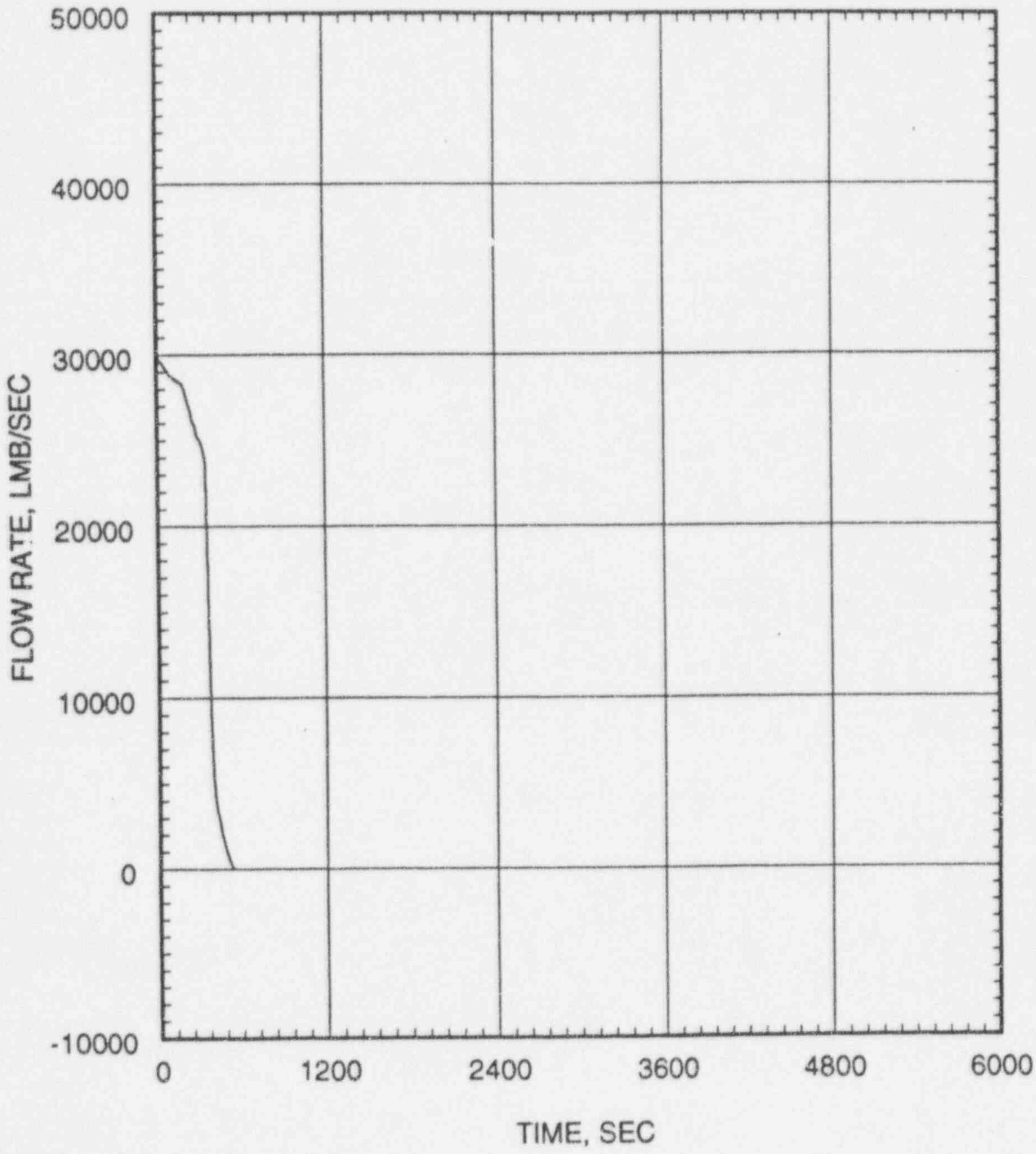

Figure 6.3-25e

0.02 ft²/PD BREAK

INNER VESSEL TWO-PHASE MIXTURE LEVEL

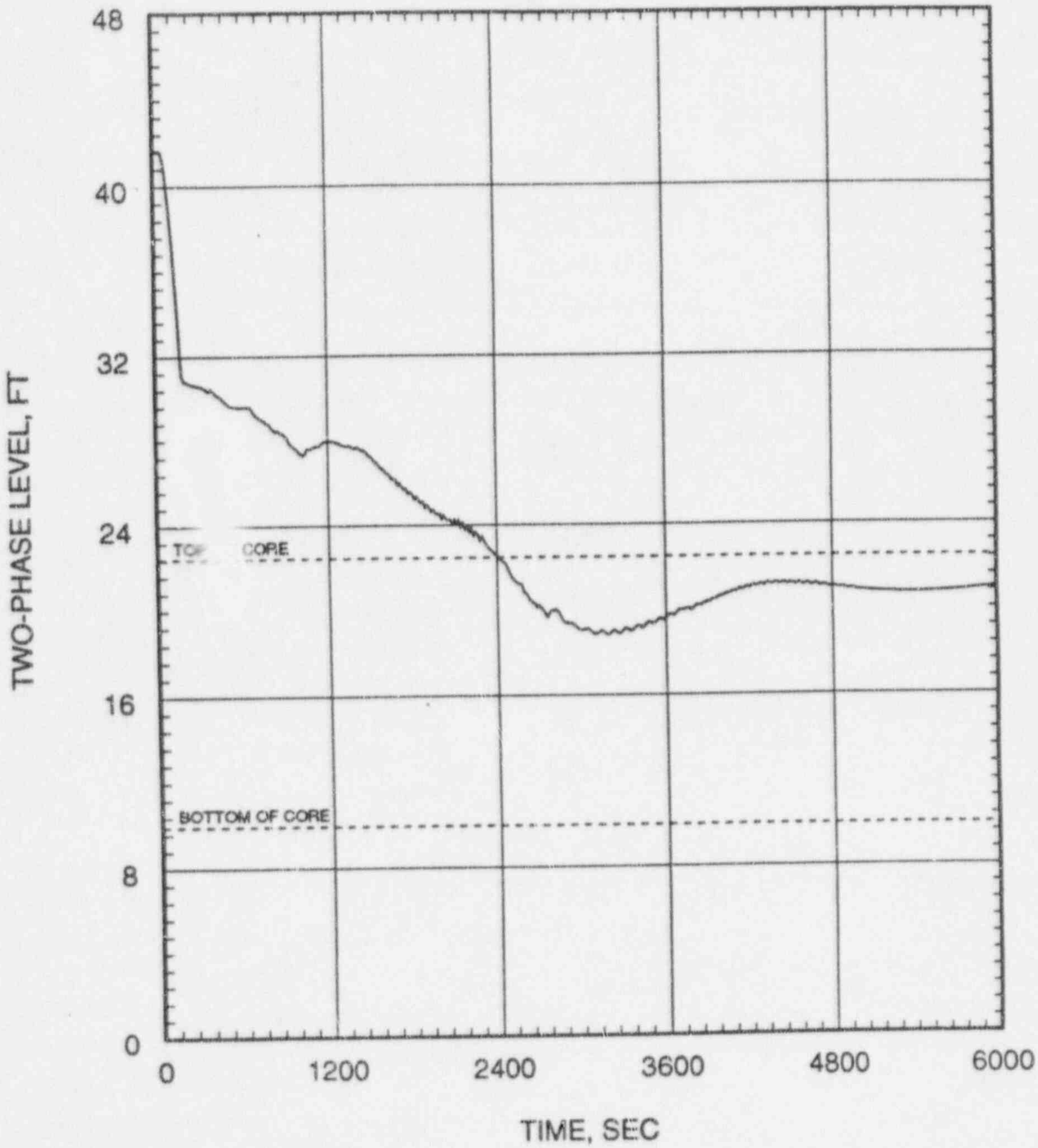

Figure 6.3-25f

0.02 ft^2/PD BREAK

HEAT TRANSFER COEFFICIENT AT HOT SPOT

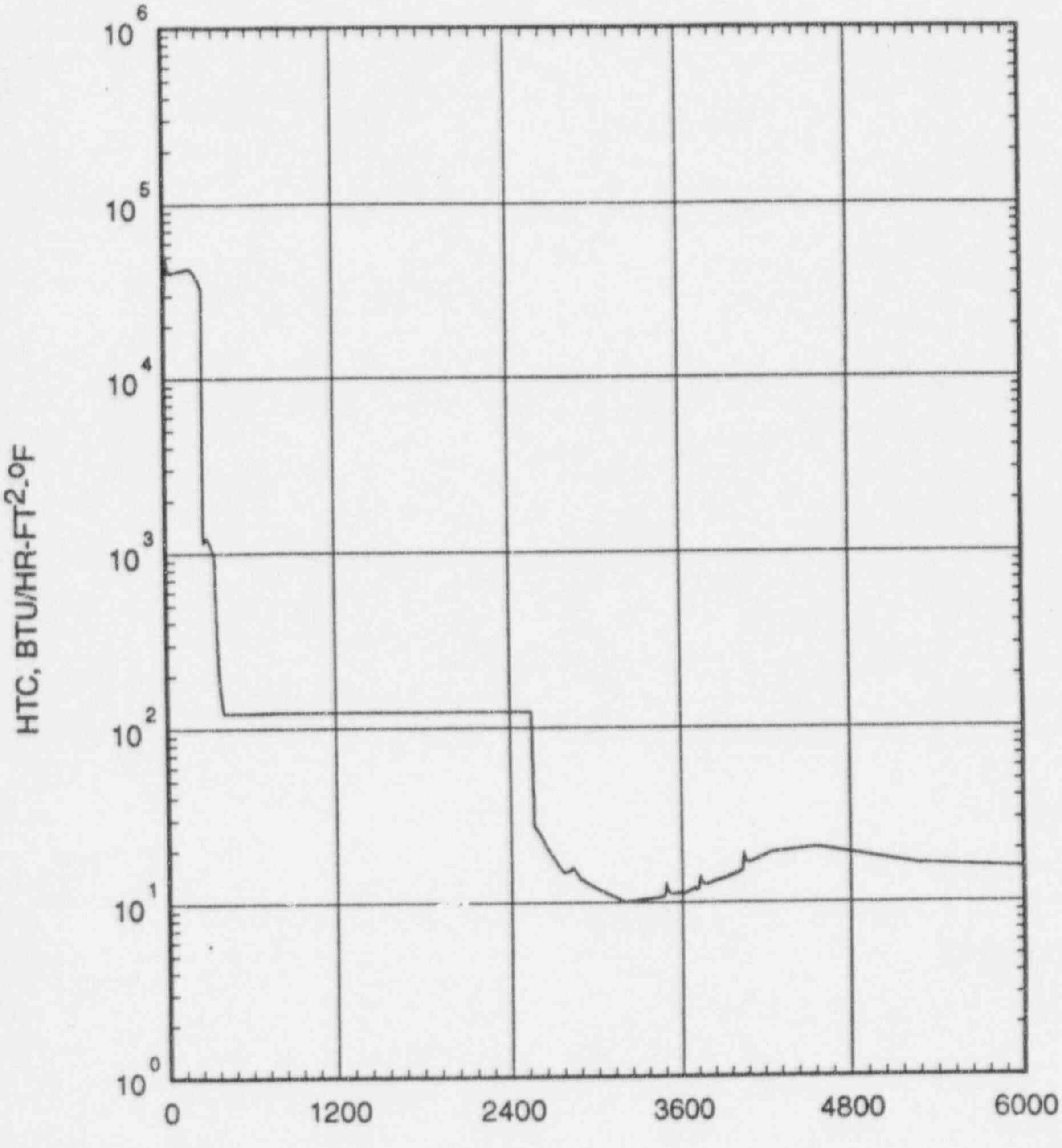

Figure 6.3-25g

0.02 ft^2/PD BREAK

COOLANT TEMPERATURE AT HOT SPOT

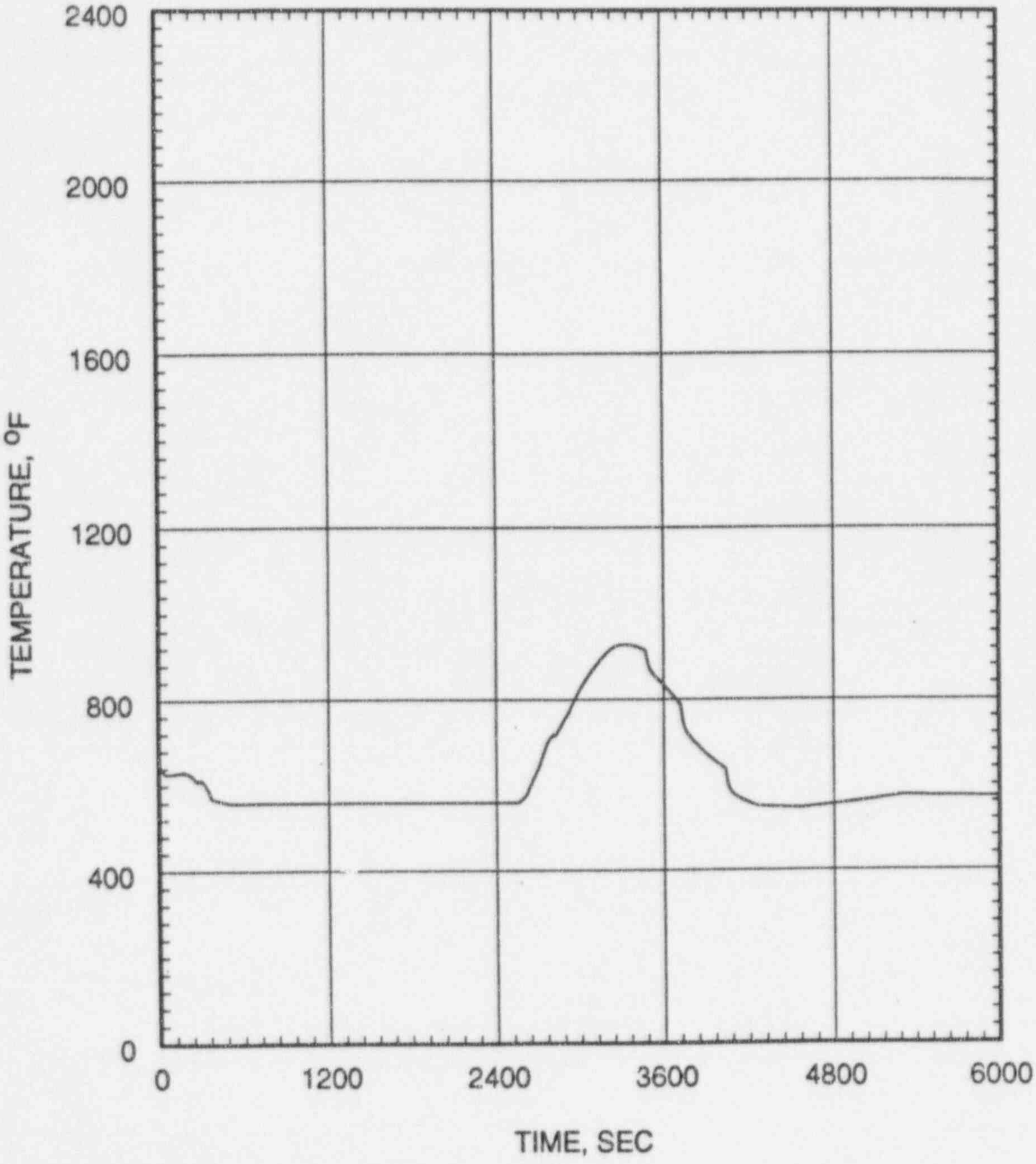

Figure 6.3-25h

0.02 ft^2/PD BREAK

CLADDING TEMPERATURE AT HOT SPOT

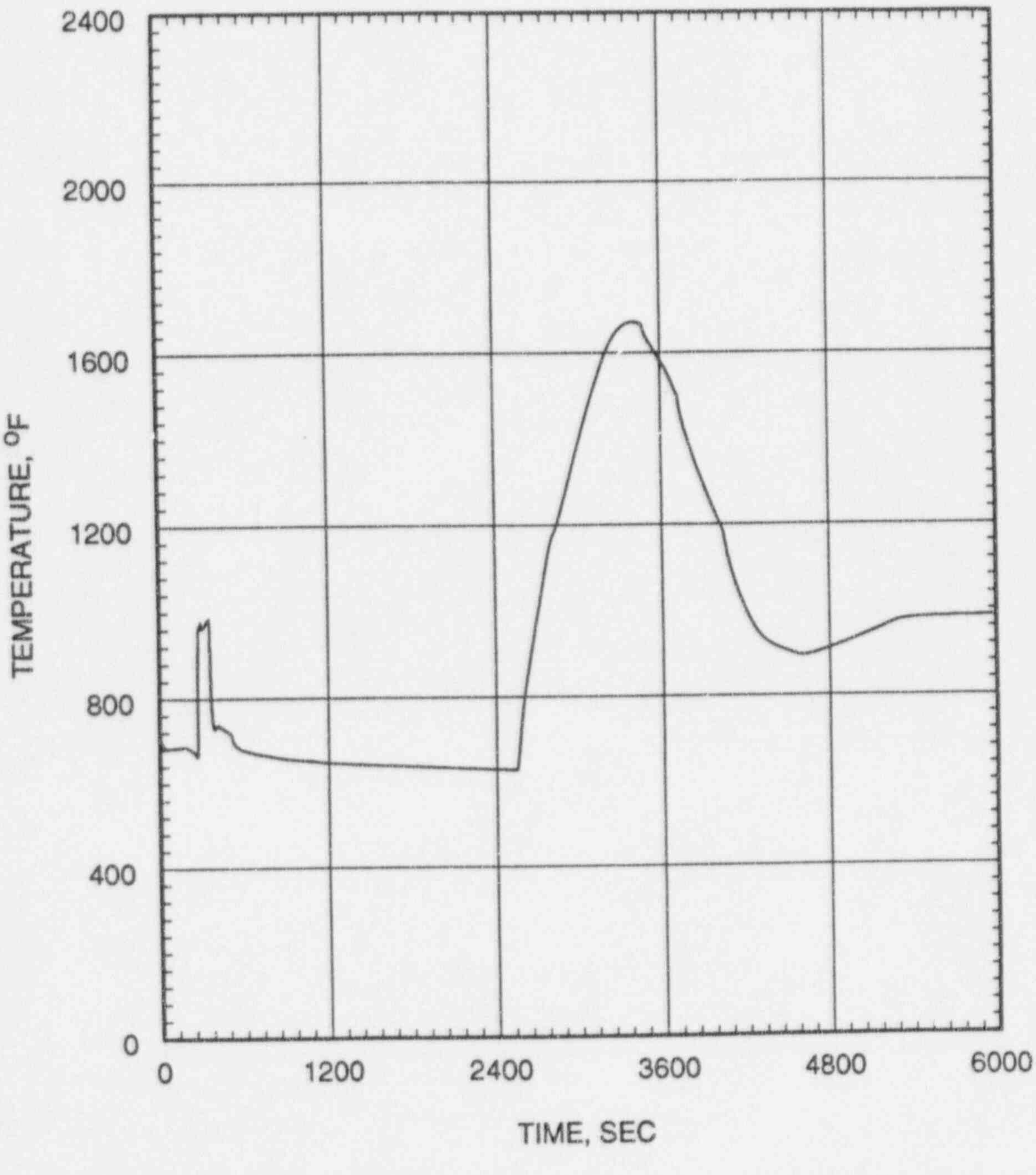

Figure 6.3-26

PEAK CLADDING TEMPERATURE VERSUS BREAK SIZE

FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION