ANALYSIS OF THE EMERGENCY CORE COOLING SYSTEM

IN ACCORDANCE WITH THE ACCEPTANCE CRITERIA OF

10CFR50.46 AND APPENDIX K OF 10CFR50

CONSOLIDATED EDISON COMPANY
OF NEW YORK, INC. and
POWER AUTHORITY OF THE
STATE OF NEW YORK
INDIAN POINT UNIT NO. 3
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The analysis specified by the Nuclear Regulatory Commission in the "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors" per 10CFR50.46 and Appendix K of 10CFR50, as published in the Federal Register, January 4, 1974, for small breaks in the reactor coolant piping is presented in References 1 and 2 for Indian Point Unit No. 3 (IP3).

For major Reactor Coolant System pipe ruptures (Loss of Coolant Accident) the analysis specified by 10CFR50.46 is presented here in. The analytical techniques used are in compliance with Appendix K of 10CFR50 and are described in Reference 3. The results of the loss of coolant accident analysis are shown in Tables IP3-1 and show compliance with the Acceptance Criteria. The analyses were performed using the October 1975 version of the Westinghouse ECCS Evaluation Model. This version of the evaluation model is documented in Reference 4,5 and 6 and includes the modifications of the models specified by the Nuclear Commission in Reference 7.

The boundary considered for loss of coolant accidents as related to connecting pipings is defined in Section 4.1.3 of the IP3 FSAR.

Should a major break occur, depressurization of the Reactor Coolant System results in a pressure decrease in the pressurizer. Reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. A Safety Injection System signal is actuated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

b. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

At the beginning of the blowdown phase, the entire Reactor Coolant System contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10CFR50 . Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major mechanisms. During the refill period, rod-to-rod radiation is the only mechanism.

When the Reactor Coolant System pressure falls below 600 psia, the accumulators begin to inject borated water. The conservative assumption is made that accumulator water injection bypasses the core and goes out through the break until the termination of bypass. This conservatism is consistent with Appendix K of 10CFR50.

Thermal Analysis

Westinghouse Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss of coolant accident including the double ended severance of the largest

Reactor Coolant system pipe. The reactor core and internals together with the Emergency Core Cooling System are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident.

The Emergency Core Cooling System even when operating during the injection mode with the most severe single active failure is designed to meet the Acceptance Criteria.

Method of Thermal Analysis

The description of the various aspects of the LOCA analysis is given in Reference 3. This document describes the major phenomena modeled, the interface among the computer codes and features of the codes which maintain compliance with the Acceptance Criteria. The individual codes are described in detail in Reference 8,9,10, and 11. The containment parameters used in Reference 11 to determine the ECCS backpressure are presented in Table IP3-3. As noted in Table IP3-3, a conservatively low value of 35^{o} F for service water temperature was used.

The analysis presented in this submittal is performed for the DECLG breaks with CD = 1.0 and 0.8. The analysis considers the reactor vessel upper head fluid temperature to be equal to the Reactor Coolant System hot leg fluid temperature. The effect of using the hot leg temperature in the reactor vessel upper head region is described in Reference 12. Reference 13 presents a break spectrum sensitivity study using the increased upper head fluid temperature. This study shows that the limiting break size is unchanged due to the increased upper head fluid temperature.

As reported in Reference 14, the limiting break for Indian Point Unit No. 3 was the CD =0.8 DECLG. Both the 0.8 and 1.0 DECLG breaks were reanalyzed, and the results now show the CD = 1.0 DECLG as limiting. This shift in the limiting break for Indian Point Unit No. 3 is attributed to changes included in the October 1975 version of the ECCS codes.

The Indian Point Unit No. 3 plant differs from the generic 4 loop 15x15 plant analysis presented in Reference 13 with respect to accumulator water volume and accumulator total volume. These differences have an effect on core cooling during the refill and reflood transient, causing the 1.0 DECLG break to be limiting, rather than the limiting break size established in the generic analysis.

Results

Table IP3-1 presents the peak clad temperature and hot spot metal water reaction for two break sizes (CD = 1.0 and 0.8). These cases are sufficient to identify the worst case as the CD = 1.0 case. The time sequence of events during the large breaks is shown in Table IP3-2.

The SATAN VI blowdown and reflood analysis of the loss of coolant accident is performed at 102 percent of the Engineered Safeguards Design Rating of 3216 MWT. The equivalent core parameters at the license application power level of 3025 MWT are shown in Table IP3-1, and the peak linear power used in the rod heat up calculations (LOCTA IV code) is based on 102 percent of the licensed core power rating. Since there is margin between the value of the peak linear power density used in this analysis and the value expected in operation, a lower peak clad temperature would be obtained by using the peak linear power density expected during operation.

For the results discussed below, the hot spot is defined to be the location of maximum peak clad temperature. This location is given in Table IP3-1 for each break size analyzed.

Figures IP3-1 through IP3-16 present the transients for the following principal parameters:

Figures IP3-1a through IP3-3b	These figures show the fluid quality, the mass velocity and the heat transfer coefficient (as calculated by the LOCTA IV code) at the hot spot (location of maximum clad temperature) and burst location, on the hottest fuel rod (hot rod).
Figures IP3-4a through IP3-6b	These figures show core pressure, the flow rate out of the break (plotted as the sum of both ends for the guillotine break), and the core pressure drop (from the lower plenum near the core, to the upper plenum at the core outlet).
Figures IP3-7a through IP3-9b	These figures show the clad temperature transient at the hot spot and burst location, the fluid temperature (also for the hot spot and burst location), and the core flow (top and bottom).
Figures IP3-10a through IP3-10b	These figures show the core reflood transient.

Figures IP3-11a through IP3-12b — These figures show the Emergency Core Cooling flow for all cases analyzed. The accumulator flow assumed is the sum of that injected in the intact cold legs.

Figures IP3-13a,b — These figures show the containment pressure transient.

Figures IP3-14a,b — These figures show the core power transient.

Figures IP3-15 — This figure shows the break energy released to the containment.

Figure IP3-16 — This figure shows the containment wall condensing heat transfer coefficient.

In addition to the above, Tables IP3-4 and IP3-5 present the reflood mass and energy releases to the containment, and the broken loop accumulator mass and energy flow rate to the containment, respectively, for the limiting break cases (CD = 1.0).

The clad temperature analysis is based on a total peaking factor of 2.32. The hot spot metal water reaction reached is 7.59%, which is well below the embrittlement limit of 17% as required by 10CFR50.46. In addition, the total core metal water reaction is less than 0.3% for all the breaks as compared with the 1 percent criterion of 10CFR50.46

The results of several sensitivity studies are reported in Reference 15. These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

Conclusions-Thermal Analysis

For breaks up to and including the double ended severance of a reactor coolant pipe, the Emergency Core Cooling System will meet the Acceptance

Criteria as presented in 10CFR50.46. That is:

1. The calculated peak fuel element clad temperature provides a substantial margin to the requirements of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17% are not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the longlived radioactivity remaining in the core.

References

1. R. Salvatori, "ECCS Acceptance Criteria Analysis Indian Point Nuclear Generating Station Unit 2", WCAP-8399, September 1974; Westinghouse Non-Proprietary Class 3.

2. Letter from Carl L. Newman of Consolidated Edison Co. of New York to D. B. Vassallo of the Nuclear Regulatory Commission dated January 29, 1976. Indian Point Unit No. 3 Small Break LOCA Analysis.

3. Bordelon, F.M., et al., "Westinghouse ECCS Evaluation Model-Summary", WCAP-8339, July 1974.

4. Bordelon, F.M., et al., "Westinghouse ECCS Evaluation Model-Supplementary Information", WCAP-8472, January 1975

5. "Westinghouse ECCS Evaluation Model, October 1975 Version", WCAP-8622, November 1975 (Proprietary) and WCAP-8623, November 1975 (Non-Proprietary).

6. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassallo of the Nuclear Regulatory Commission, Letter Number NS-CE-924 dated January 23, 1976.

7. Federal Register, "Supplement to the Status Report by the Directorate of Licensing in the matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10CFR50, Appendix K", November 1974.

8. Bordelon, F.M., et al., SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant, "WCAP-8306, June 1974.

9. Bordelon, F.M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8305, June 1974.

10. Kelly, R.D., et al, "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)", WCAP-8171, June 1974.

11. Bordelon, F.M., and Murphy, E.T., "Containment Pressure Analysis Code (COCO)", WCAP-8326, June 1974.

12. Letter from C. Eicheldinger of Westinghouse Electric Corporation to V. Stello of the Nuclear Regulatory Commission, Letter Number NSCE-1163 dated August 13, 1976.

13. Beck, H.S., et al. "Westinghouse ECCS - Four Loop Plant (15x15) Sensitivity Studies with Upper Head Fluid Temperature at T Hot",. WCAP-8855, October 1976 (Non-Proprietary).

14. Appendix 14 of the IP-3 FSAR, Supplement 30 dated May 1975.

15. Buterbaugh, T.L., et al., "Westinghouse ECCS - Plant Sensitivity Studies", WCAP-8356, July 1974.

TABLE IP3-1

LARGE BREAK

	DECLG (C_D=1.0)	DECLG (C_D=0.8)
Results		
Peak Clad Temp. °F	2125°F	2075°F
Peak Clad Location Ft.	6.25	6.5
Local Zr/H_2O Rxn(max)%	7.59	6.39
Local Zr/H_2O Location Ft.	6.0	6.0
Total Zr/H_2O Rxn %	<0.3	<0.3
Hot Rod Burst Time sec	26.8	27.5
Hot Rod Burst Location Ft.	6.0	6.0
Calculation		
NSSS Power Mwt 102% of	3025	
Peak Linear Power kw/ft 102% of	14.5	
Peaking Factor (At License Rating)	2.32	
Accumulator Water Volume (ft^3)	800	
Fuel region + cycle analyzed	Cycle	Region
UNIT 3	1	Limiting

TABLE IP3-2

LARGE BREAK TIME SEQUENCE OF EVENTS

	DECLG (C_D=1.0) (Sec)	DECLG (C_D=0.8) (Sec)
START	0.0	0.0
Rx Trip Signal	0.611	0.614
S. I. Signal	0.80	0.87
Acc. Injection	13.7	14.3
Pump Injection	25.80	25.87
End of Bypass	27.06	26.9
End of Blowdown	29.64	29.77
Bottom of Core Recovery	44.53	43.94
Acc. Empty	55.19	55.82

TABLE IP3-3

LARGE BREAK CONTAINMENT DATA

NET FREE VOLUME	2.61×10^6 ft^3
INITIAL CONDITIONS	
Pressure	14.7 psia
Temperature	90 °F
RWST Temperature	40 °F
Service Water Temperature	35 °F
Outside Temperature	-20 °F
SPRAY SYSTEM	
Number of Pumps Operating	2
Runout Flow Rate	3000 gpm
Actuation Time	20 secs
SAFEGUARDS FAN COOLERS	
Number of Fan Coolers Operating	5
Fastest Post Accident Initiation of Fan Coolers	30 secs

STRUCTURAL HEAT SINKS

Thickness (In)	Area (Ft^2)
1. 0.375 steel, 36.0 concrete	49,838
2. 0.5 steel, 36.0 concrete	32,072
3. 12.0 concrete	15,000
4. 0.375 stainless steel, 12.0 concrete	10,000
5. 12.0 concrete	61,000
6. 0.5 steel	68,792
7. 0.375 steel	81,704

TABLE IP3-3 (Continued)

LARGE BREAK CONTAINMENT DATA

	Thickness (In)	Area (Ft^2)
8.	0.25 steel	27,948
9.	0.1875 steel	69,800
10.	0.125 steel	3,000
11.	0.138 steel	22,000
12.	0.0625 steel	10,000
13.	.75 steel, 36.0 concrete	565
14.	.019 stainless over insulation	7,634

TABLE IP3-4

REFLOOD MASS AND ENERGY RELEASES

DECLG (C_D = 1.0)

Time(sec)	Total Mass Flowrate (lb/sec)	Total Energy Flowrate (10^5 Btu/sec)	Spilling Mass Flowrate (lb/sec)	Spilling Energy* Flowrate (10^5 Btu/sec)
44.53	0.0	0.0	0.0	0.0
45.28	4.78	0.062	0.0	0.0
45.58	4.76	0.062	0.0	0.0
45.88	4.71	0.061	0.0	0.0
51.92	43.8	0.57	0.0	0.0
60.48	131.6	1.61	0.0	0.0
73.68	138.1	1.69	0.0	0.0
89.2	148.1	1.81	0.0	0.0
105.68	298.0	2.30	144.3	0.0259
122.38	357.1	2.43	207.9	0.0374
158.2	370.7	2.32	237.2	0.0427
197.8	379.0	2.19	262.3	0.0472

* Assuming enthalpy = 18 Btu/lbm

TABLE IP3-5

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT

DECLG (C_D=1.0)

Time (Sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec) x 10^{-4}
0.0	0.0	0.0
0.02	4682.2	27.16
0.98	3942.8	22.87
2.00	3498.5	20.29
2.98	3189.7	18.5
5.0	2746.2	15.93
10.0	2128.0	12.34
15.0	1776.0	10.3
20.0	1535.5	8.91
21.78	1465.8	8.5

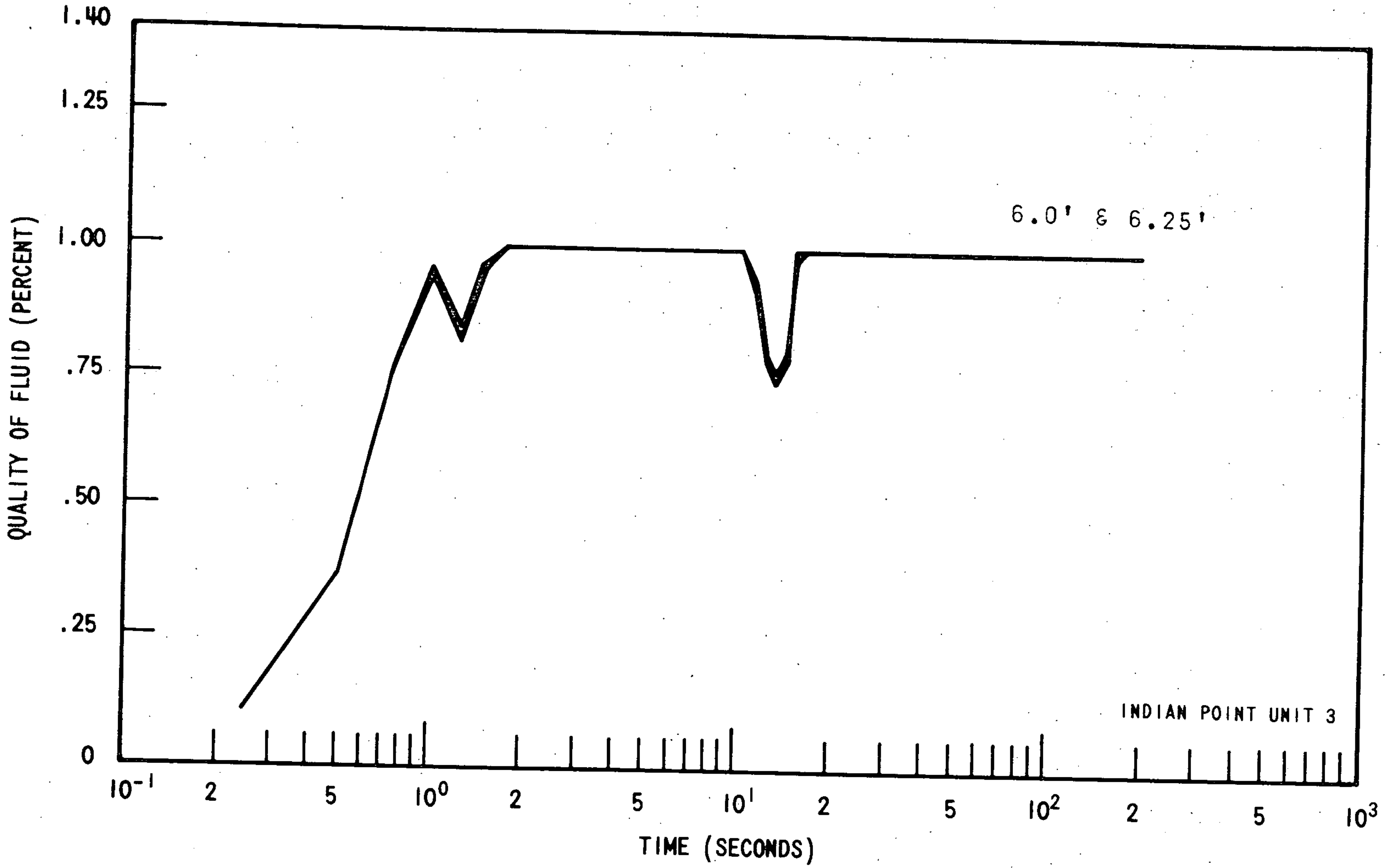

Figure IP3-1a. Fluid Quality – DECLG (C_D = 1.0)

10,807-1

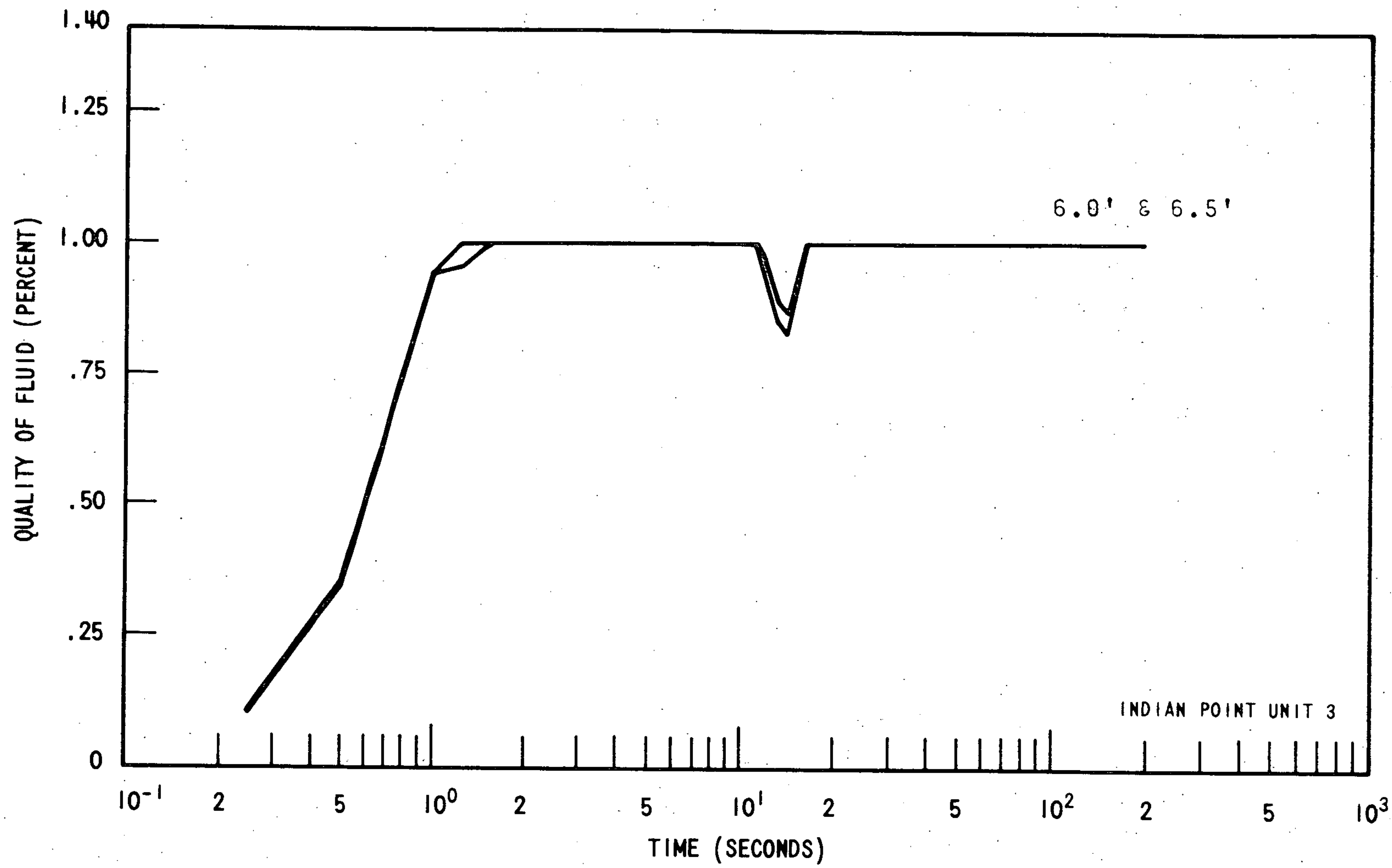

Figure IP3-1b. Fluid Quality – DECLG (C_D = 0.8)

10,807-2

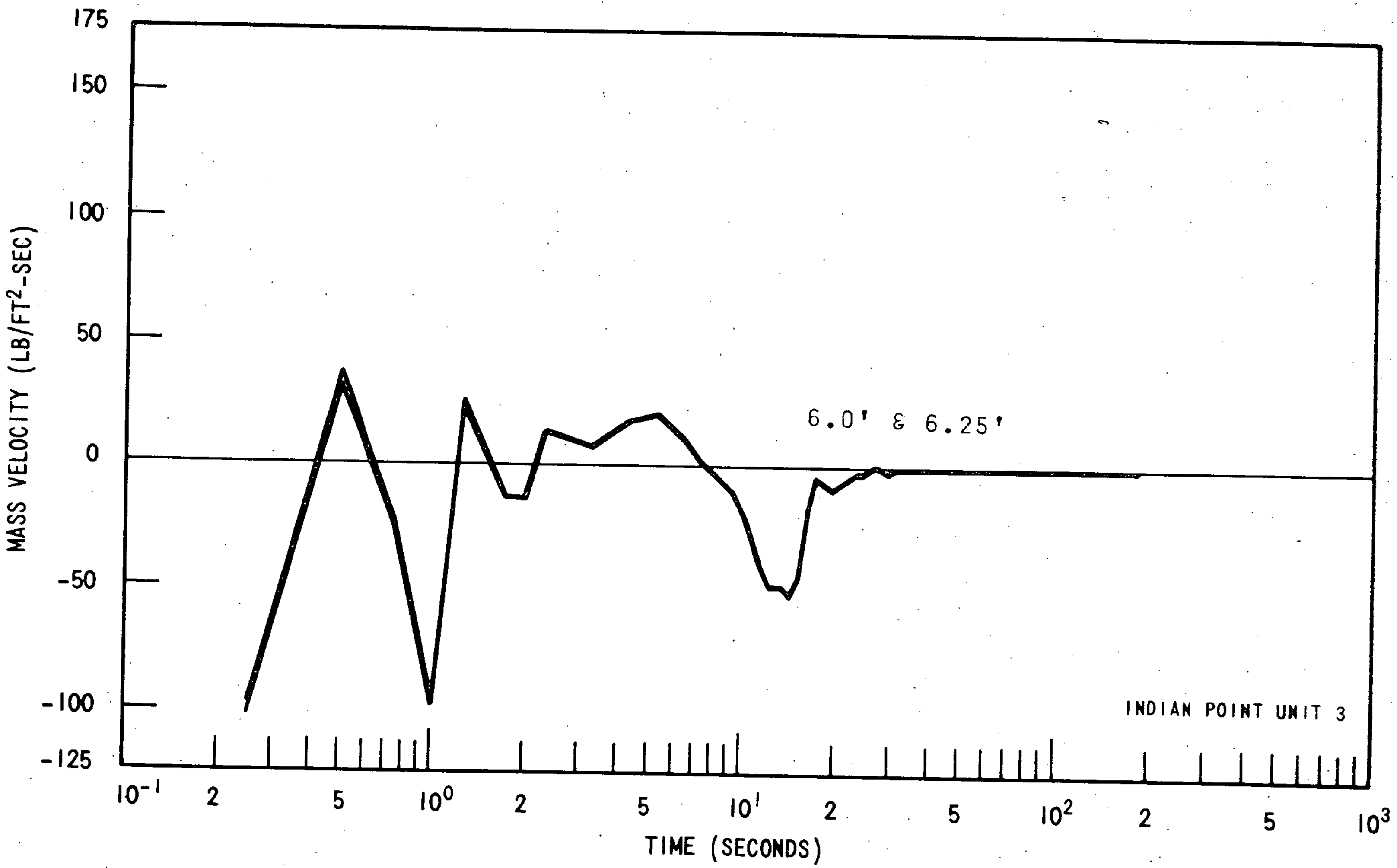

Figure IP3-2a. Mass Velocity – DECLG (C_D = 1.0)

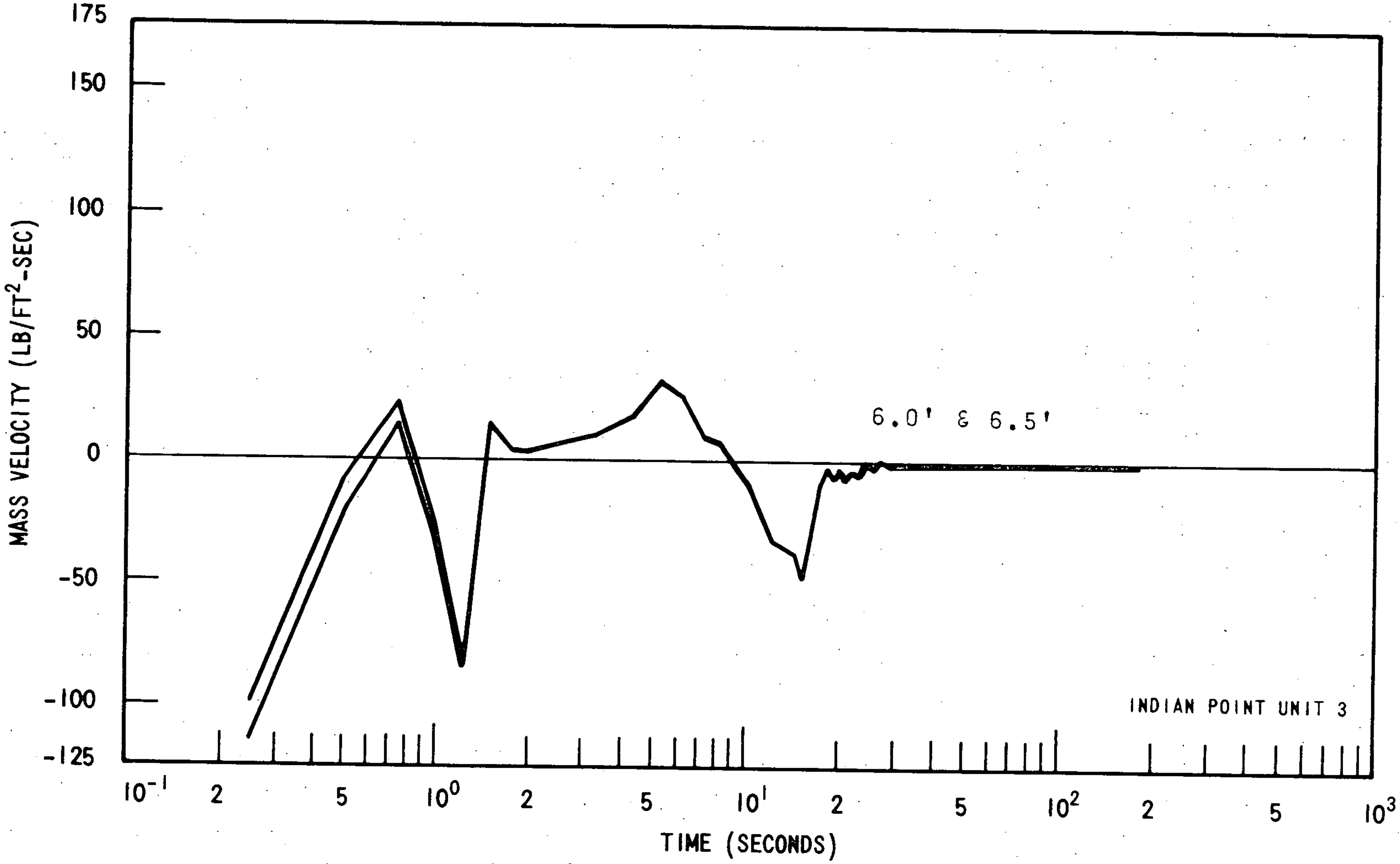

Figure IP3-2b Mass Velocity – DECLG (C_D = 0.8)

10,807-4

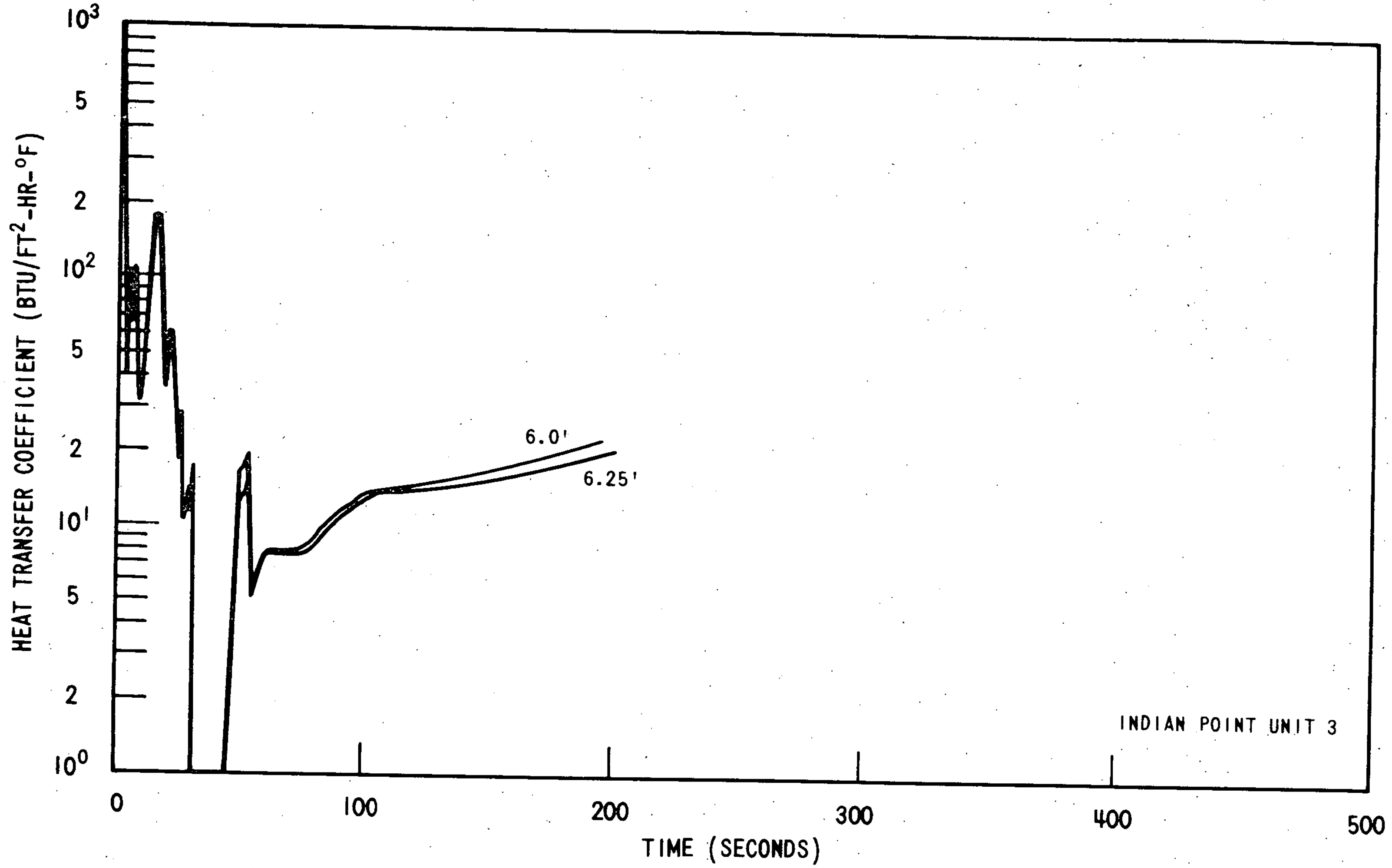

Figure IP3-3a Heat Transfer Coefficient – DECLG (C_D = 1.0)

10.807-5

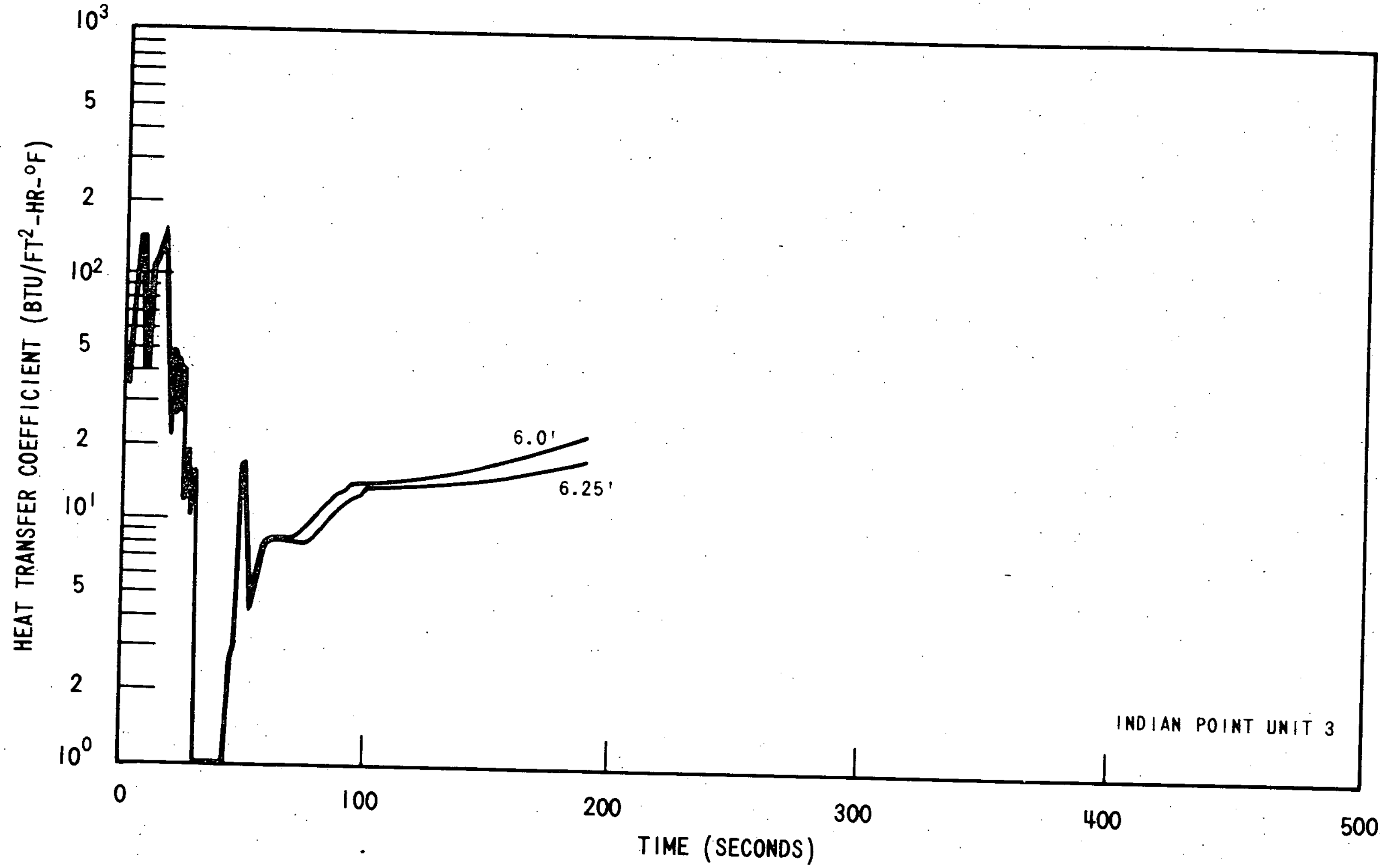

Figure IP3-3b. Heat Transfer Coefficient – DECLG (C_D = 0.8)

10,807-6

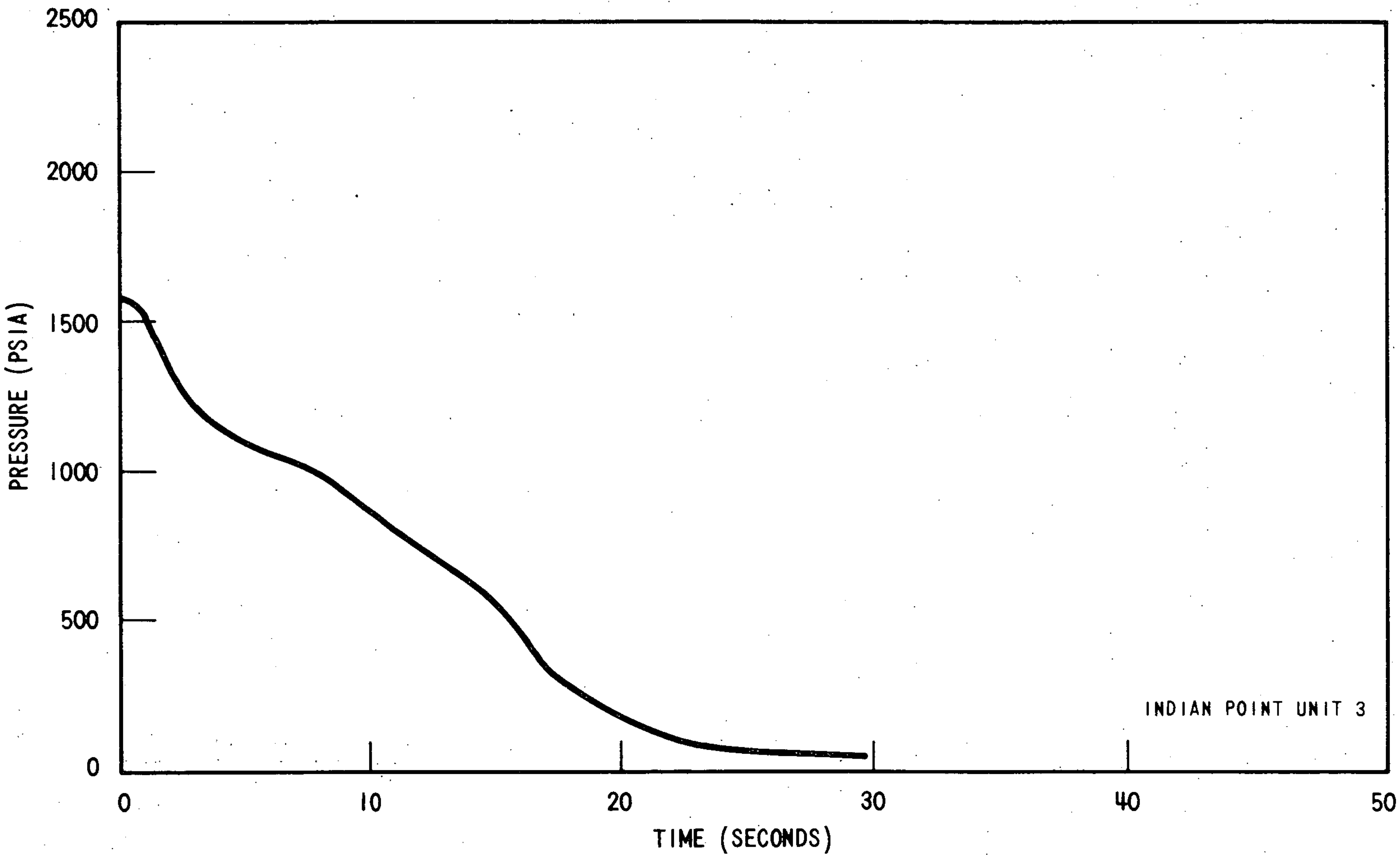

Figure IP3-4b. Core Pressure — DECLG (C_D=0.8)

10,807-7

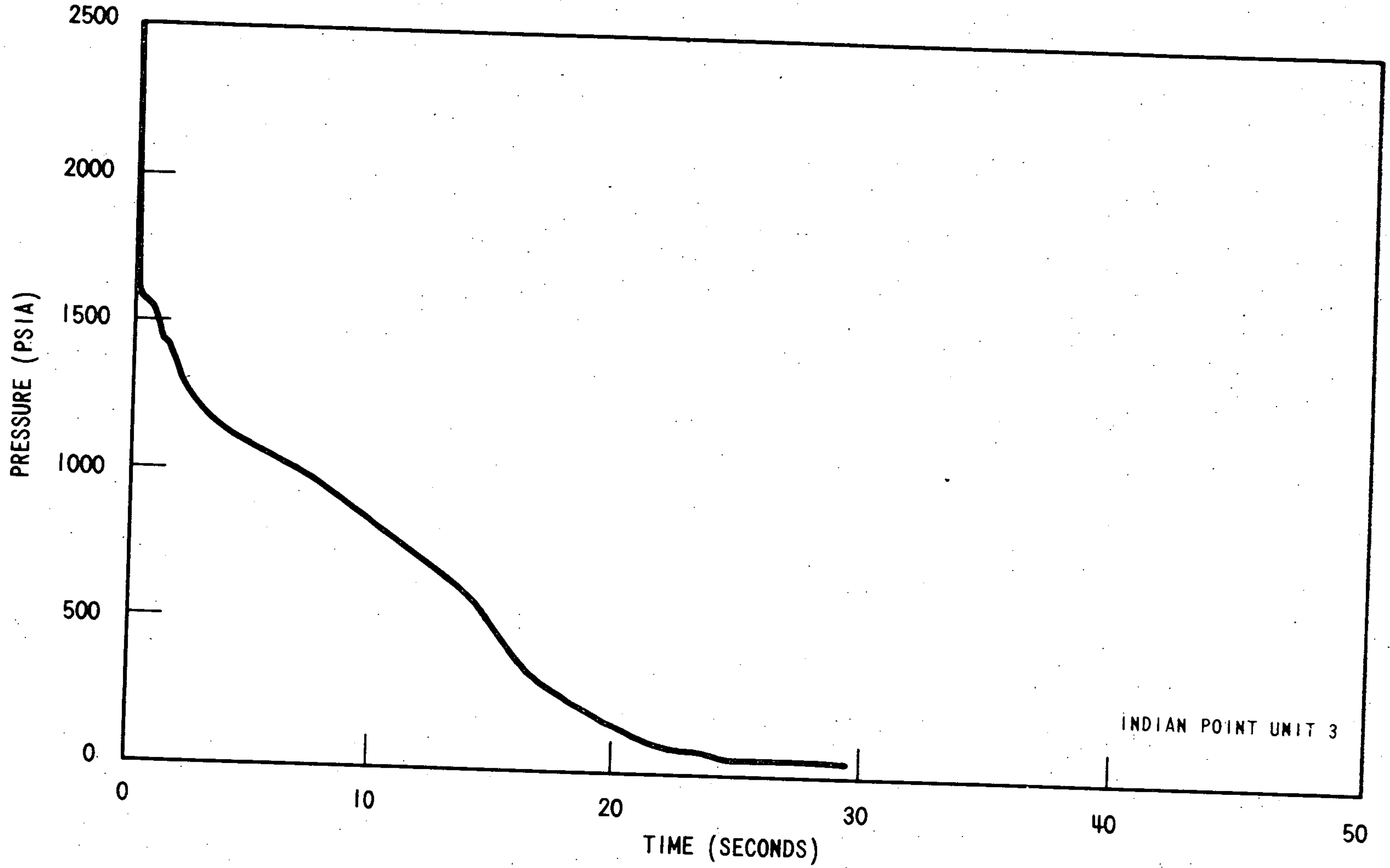

Figure IP3-4a. Core Pressure – DECLG (C_D=1.0)

10.807-8

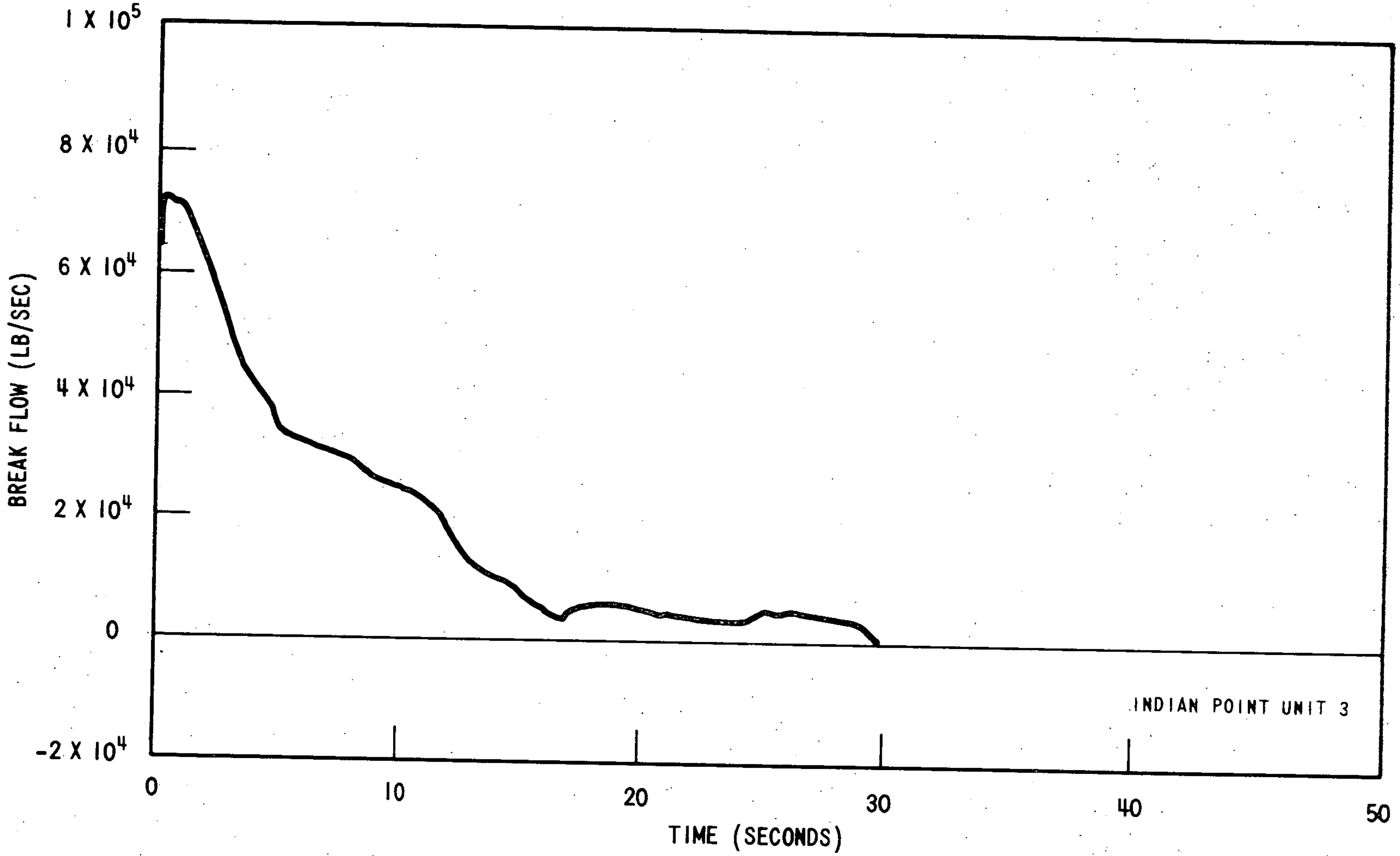

Figure IP3-5b. Break Flow Rate – DECLG (C_D=0.8)

10,807-9

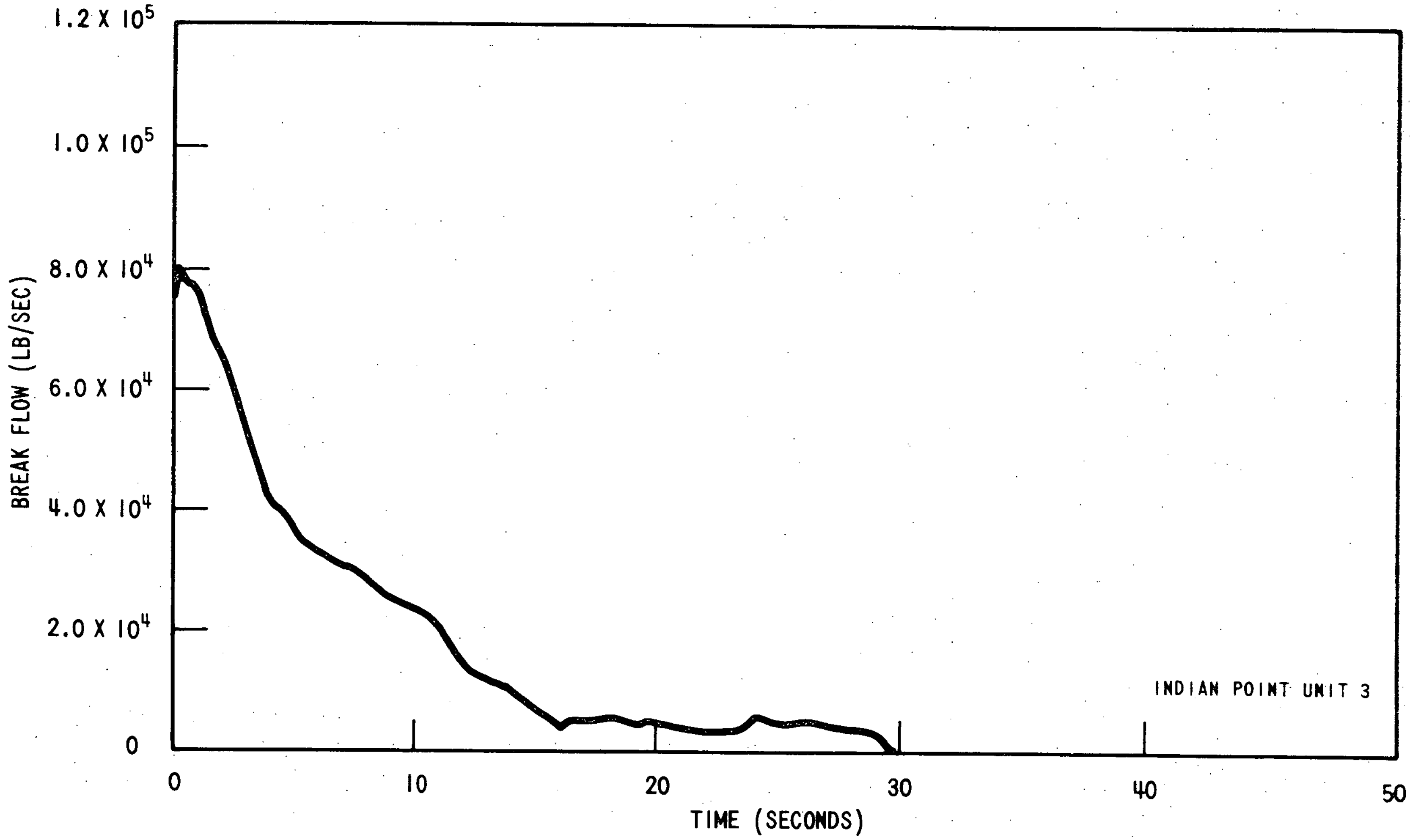

Figure IP3-5a. Break Flow Rate — DECLG (C_D=1.0)

10,807-10

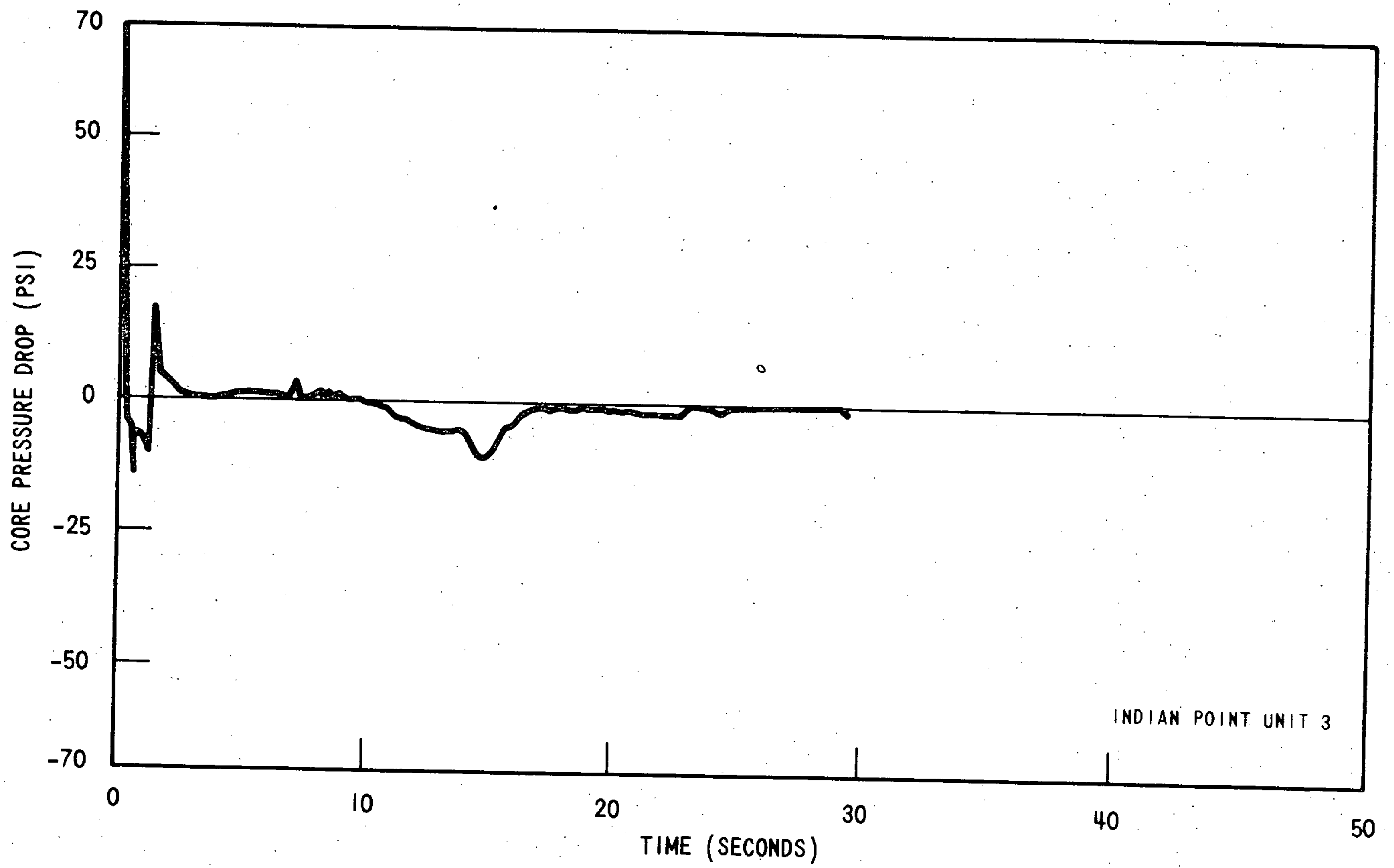

Figure IP3-6a. Core Pressure Drop – DECLG (C_D = 1.0)

10.807-12

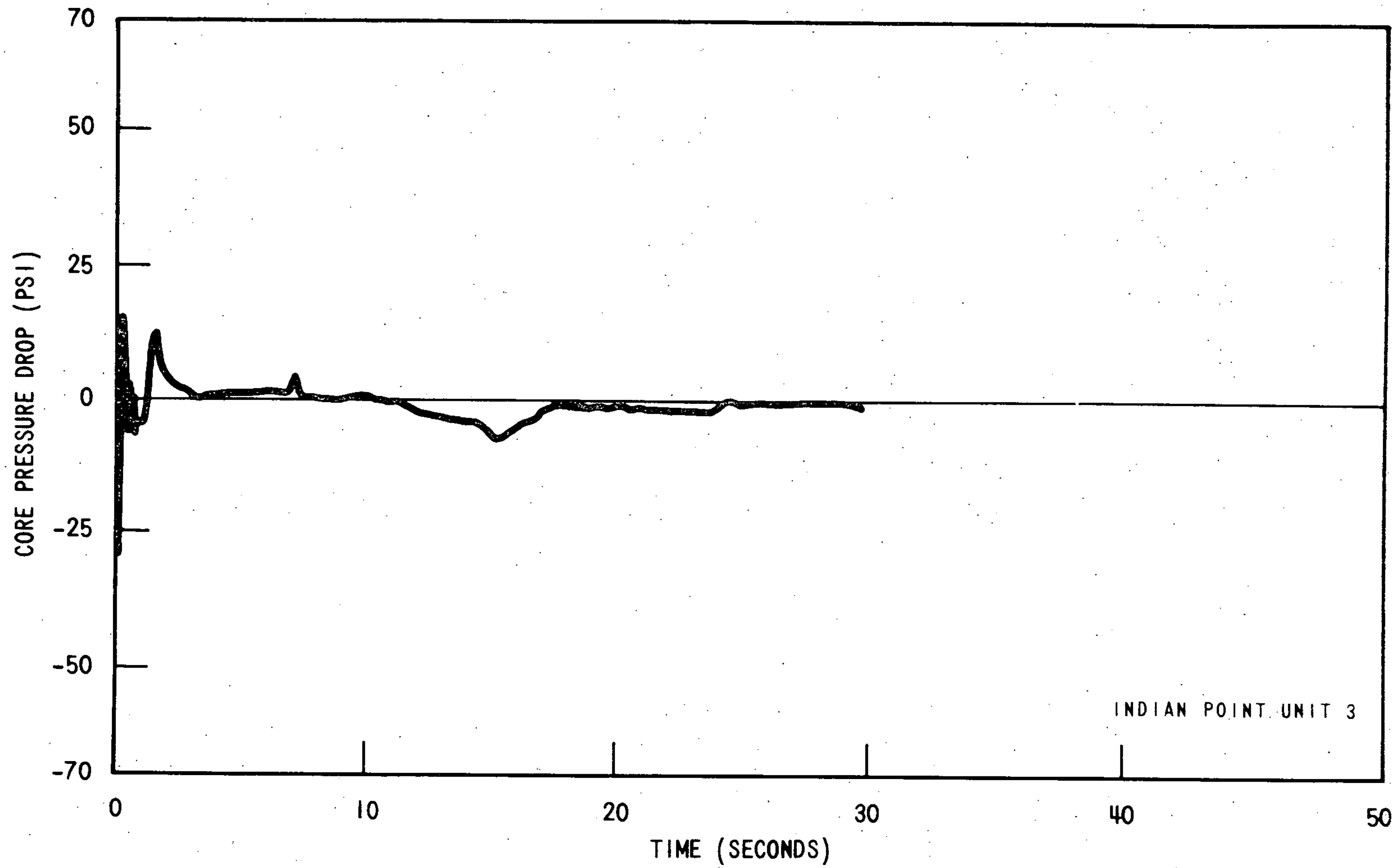

Figure IP3-6b. Core Pressure Drop – DECLG (C_D = 0.8)

10,807-11

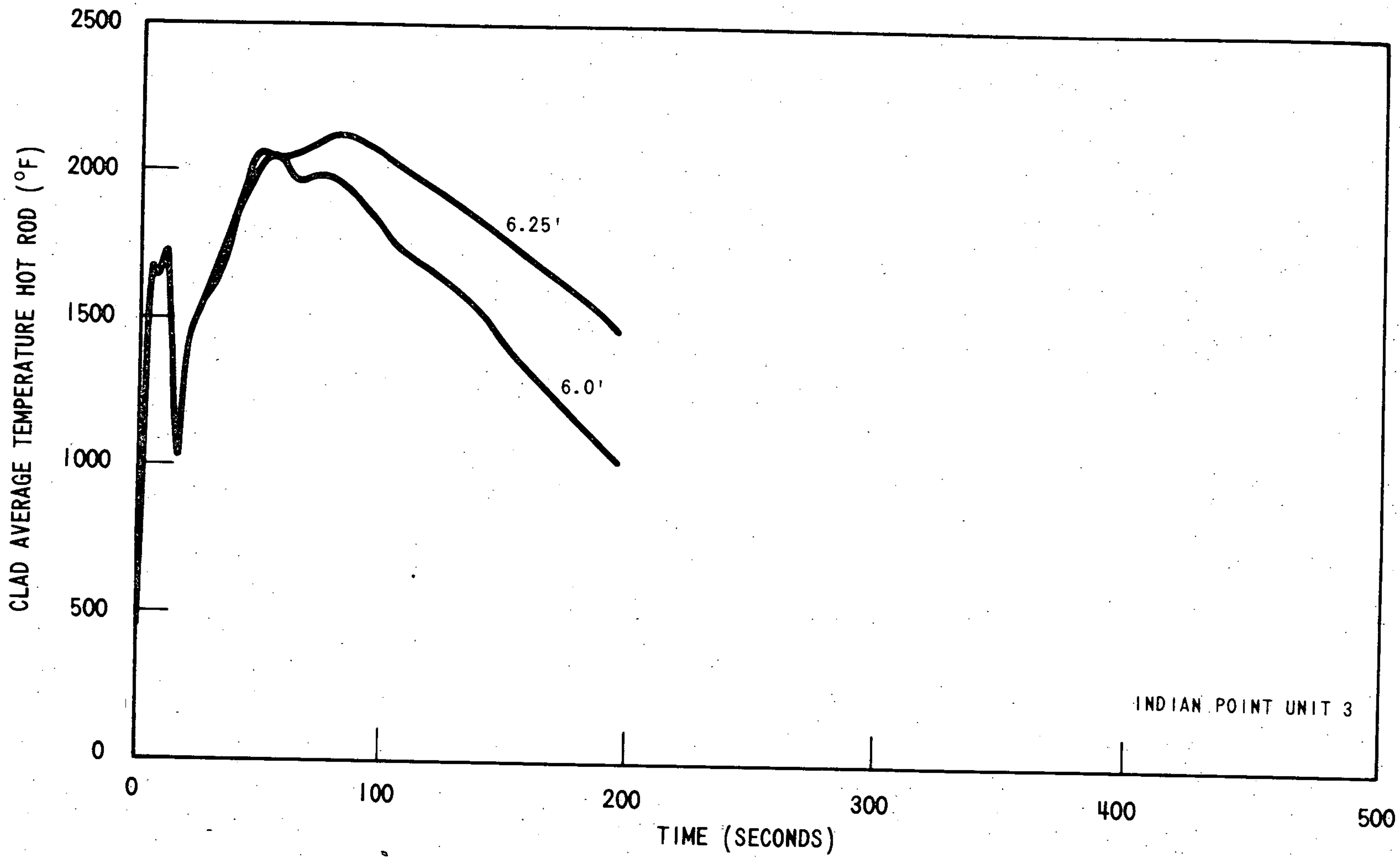

Figure IP3-7a. Peak Clad Temperature – DECLG (C_D = 1.0)

10.807-13

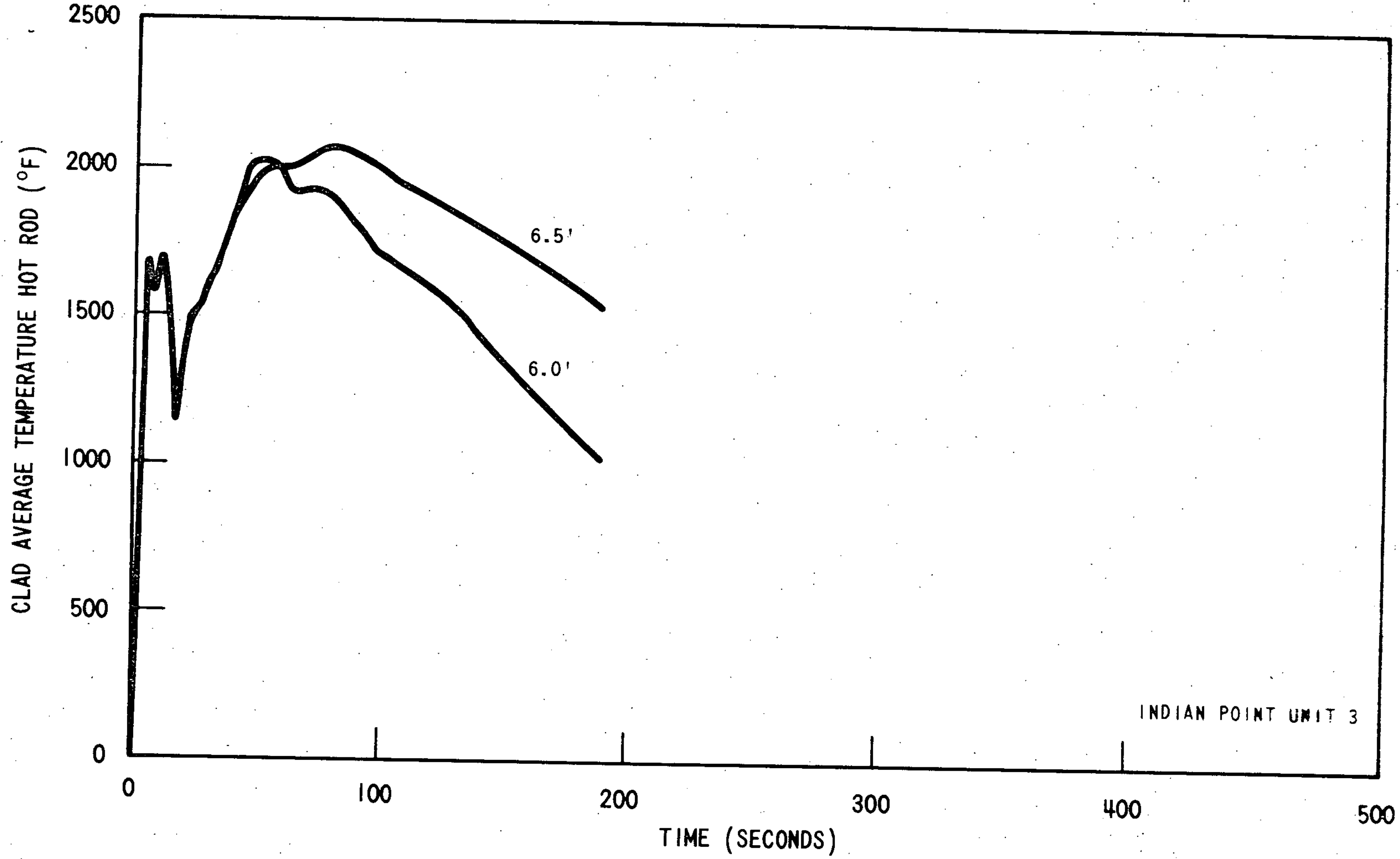

Figure IP3-7b. Peak Clad Temperature – DECLG (C_D = 0.8)

10,807-14

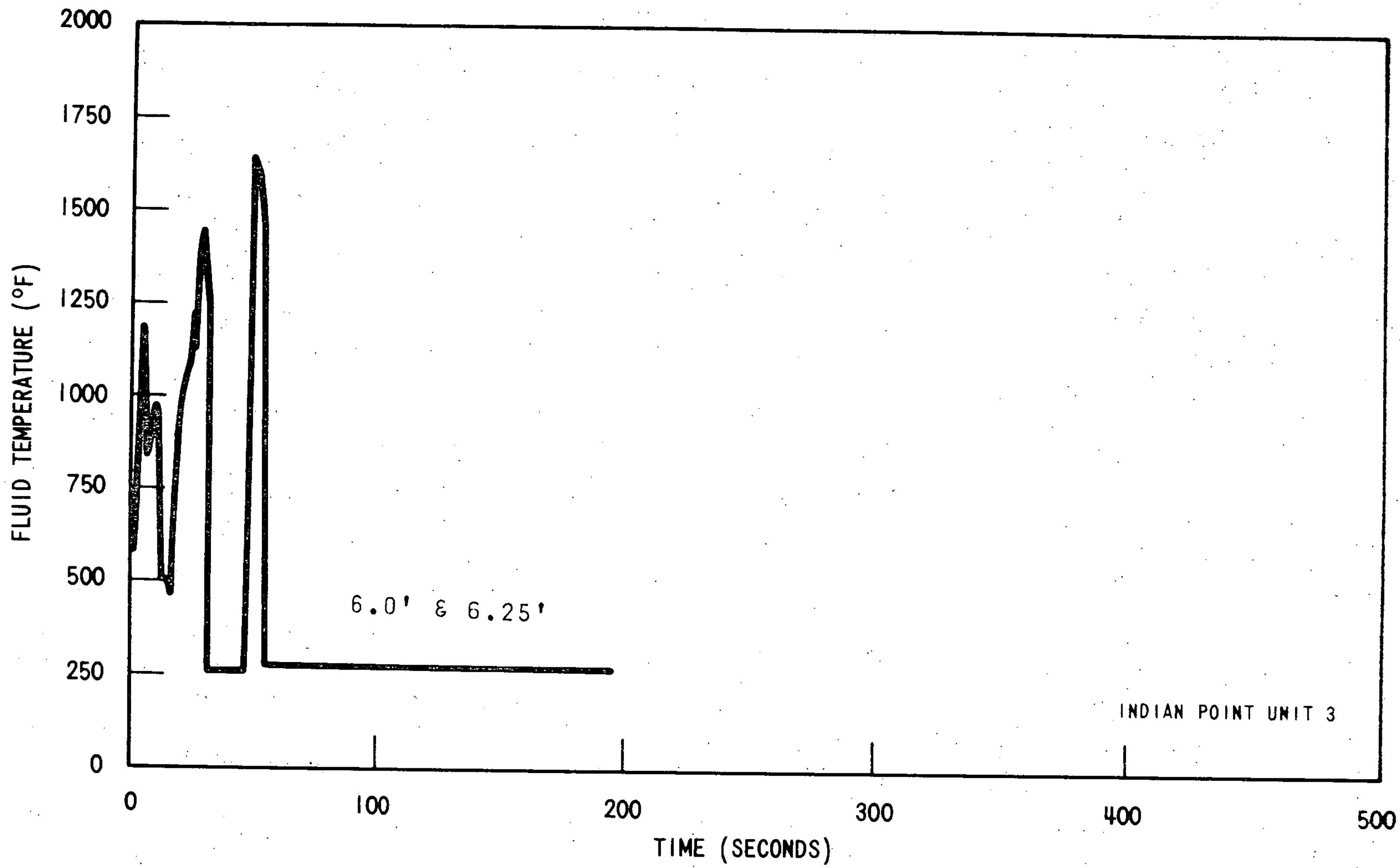

Figure IP3-8a. Fluid Temperature – DECLG (C_D = 1.0)

10,807-15

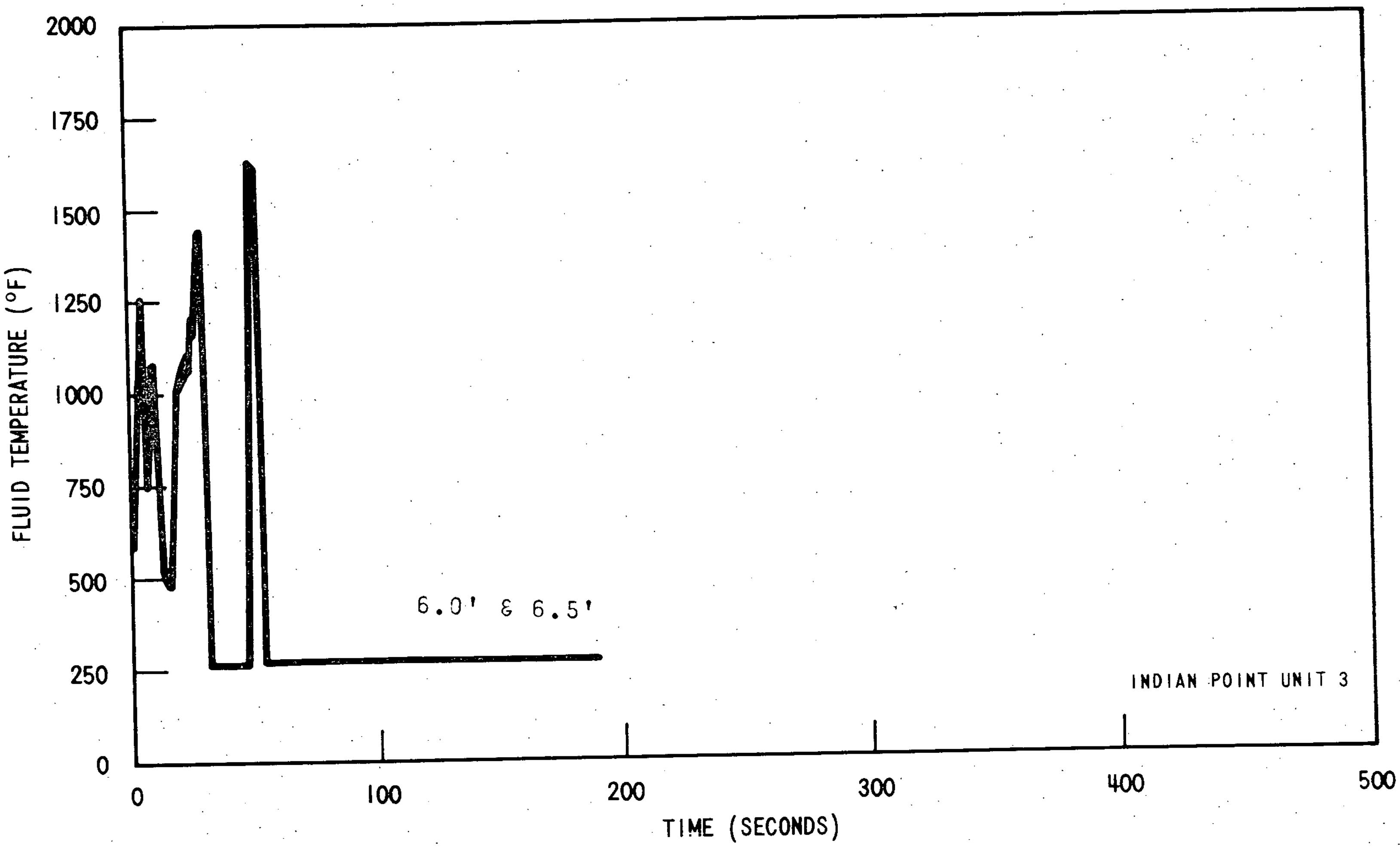

Figure IP3-8b. Fluid Temperature – DECLG (C_D = 0.8)

10,807-16

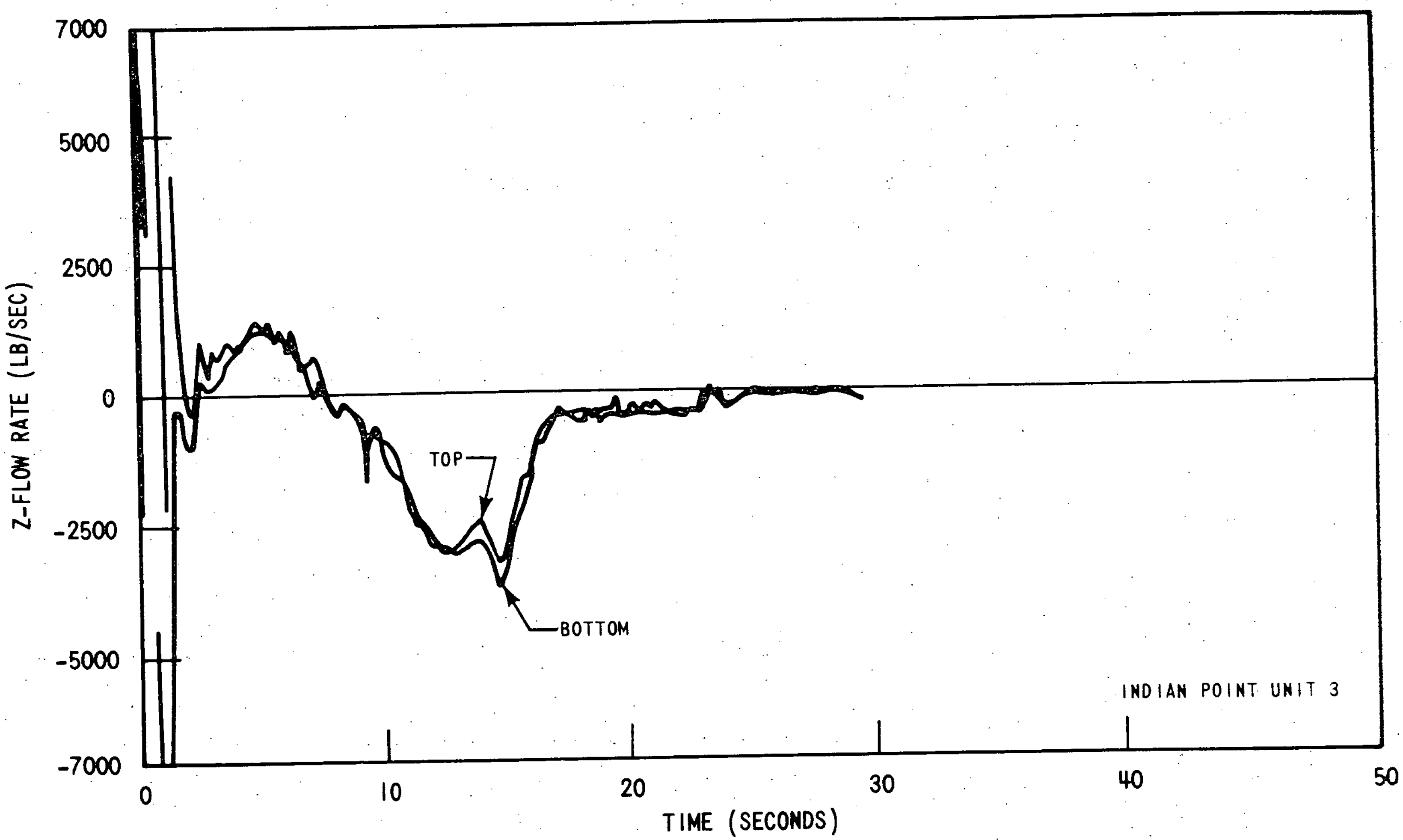

Figure IP3-9a. Core Flow – Top and Bottom – DECLG (C_D = 1.0)

10,807-18

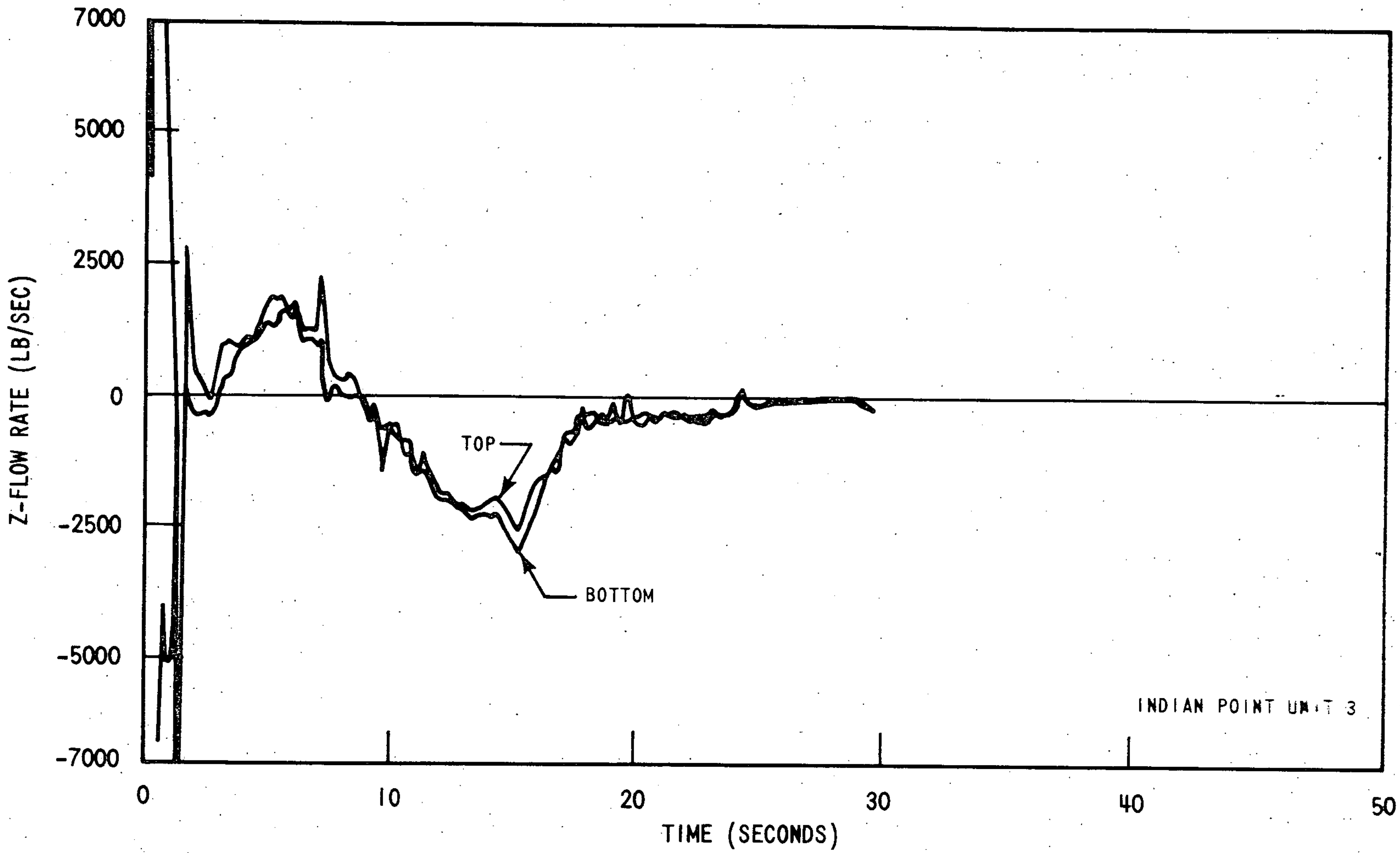

Figure IP3-9b. Core Flow – Top and Bottom – DECLG (C_D = 0.8)

10,807-17

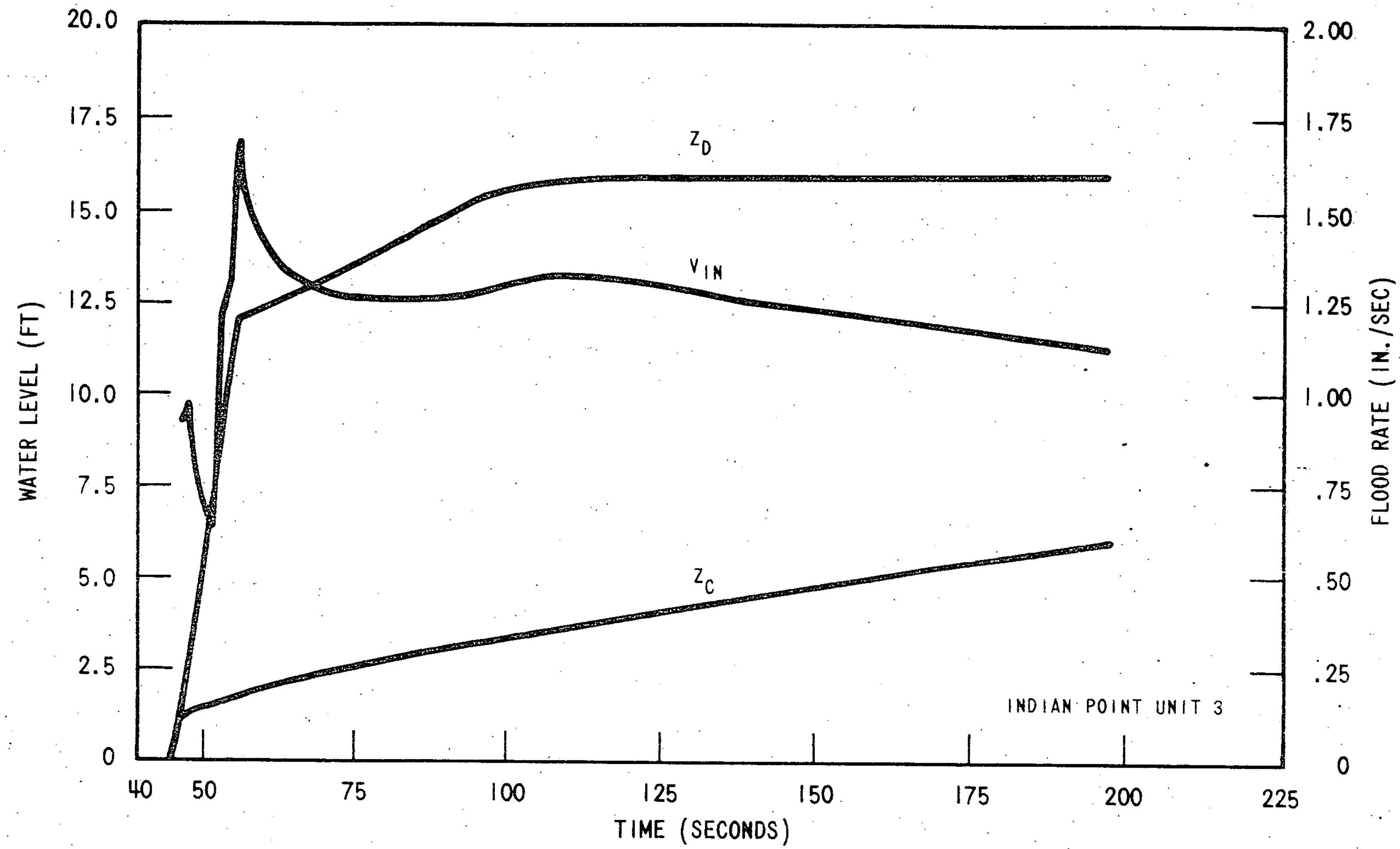

Figure IP3-10a. Reflood Transient – DECLG (C_D = 1.0)
Downcomer and Core Water Levels

10.807-19

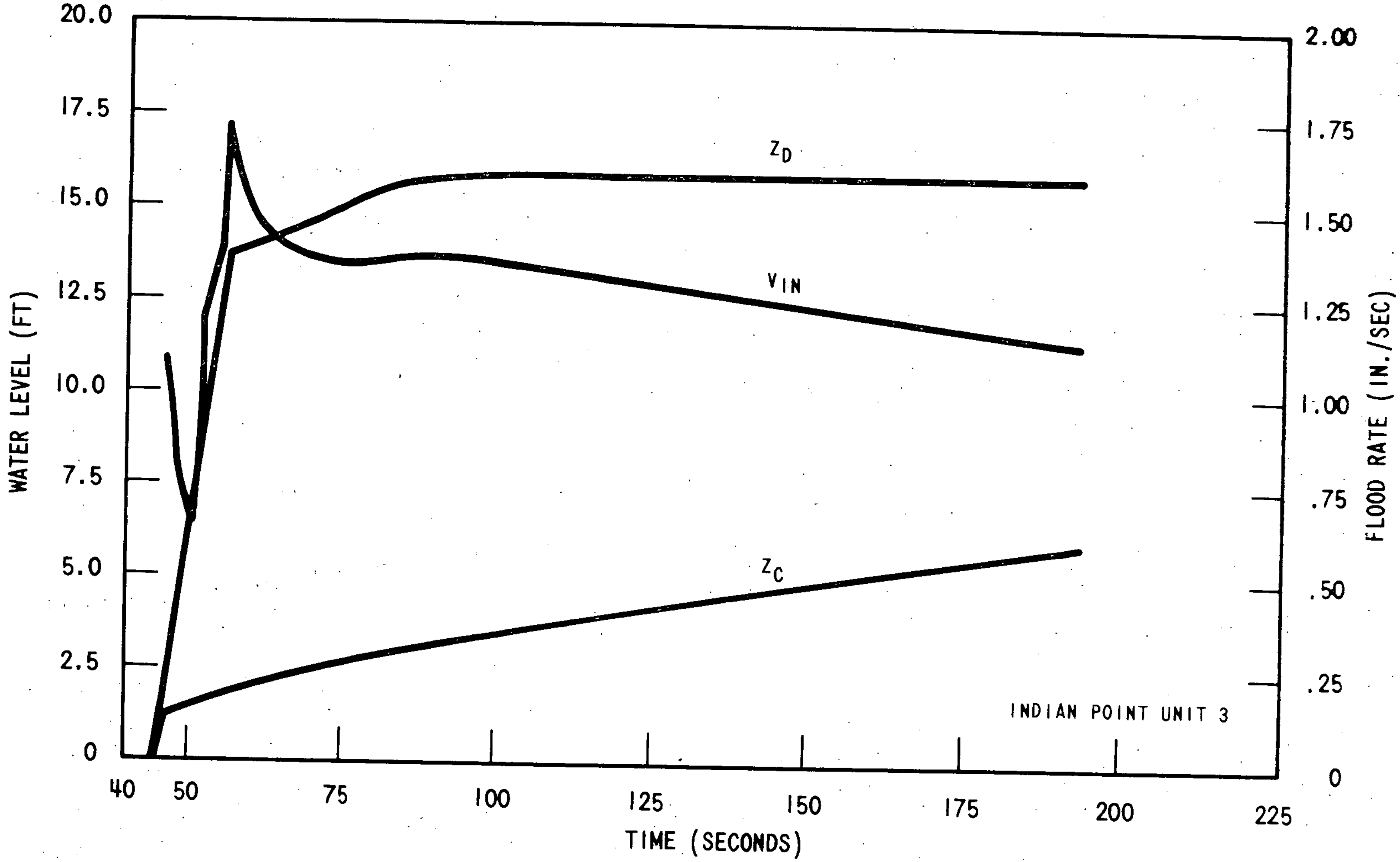

Figure IP3-10b. Reflood Transient – DECLG (C_D = 0.8) Downcomer and Core Water Levels

10.807-20

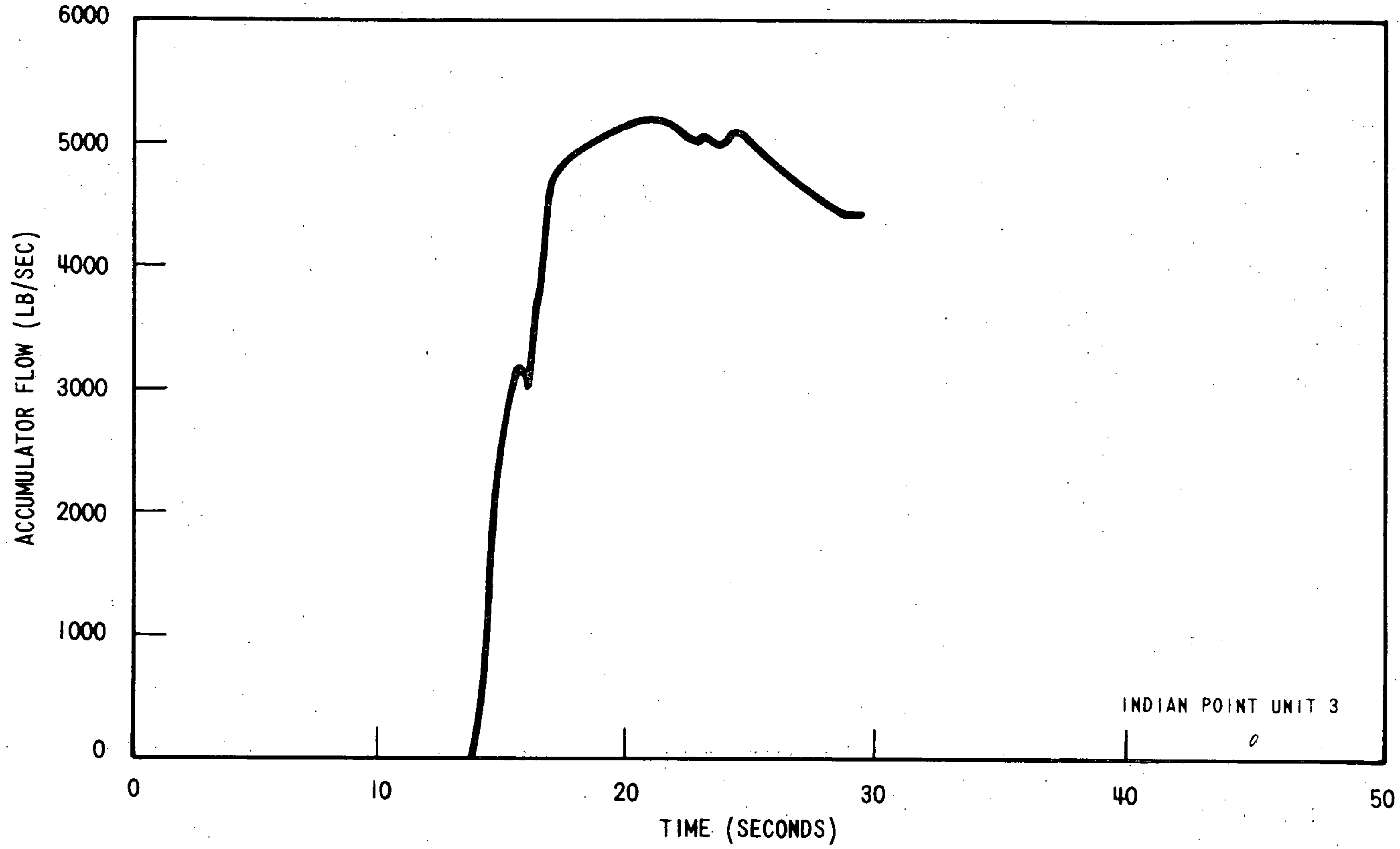

Figure IP3-11a. Accumulator Flow (Blowdown) – DECLG (C_D = 1.0)

10.807-22

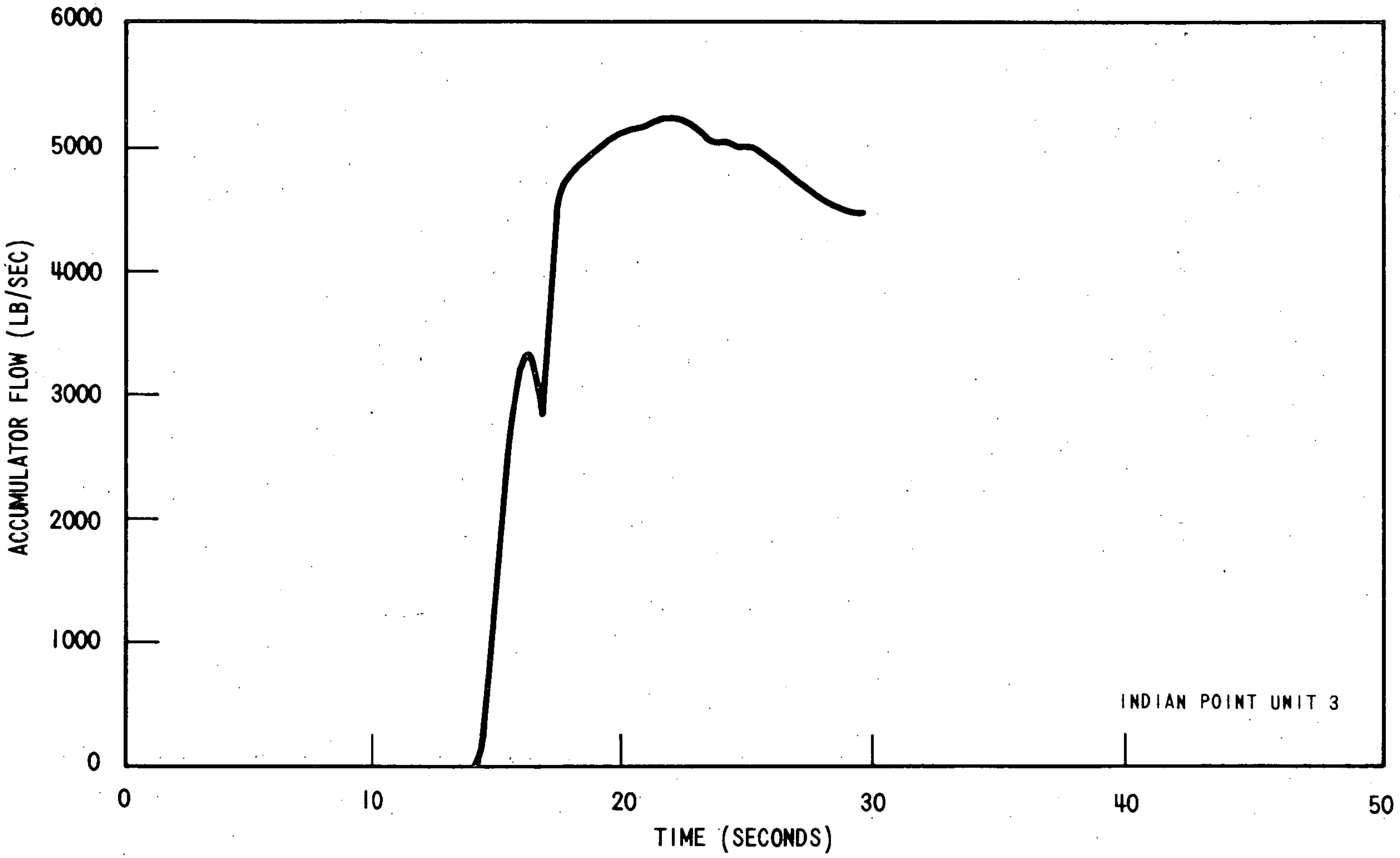

Figure IP3-11b. Accumulator Flow (Blowdown) – DECLG (C_D = 0.8)

10,807-21

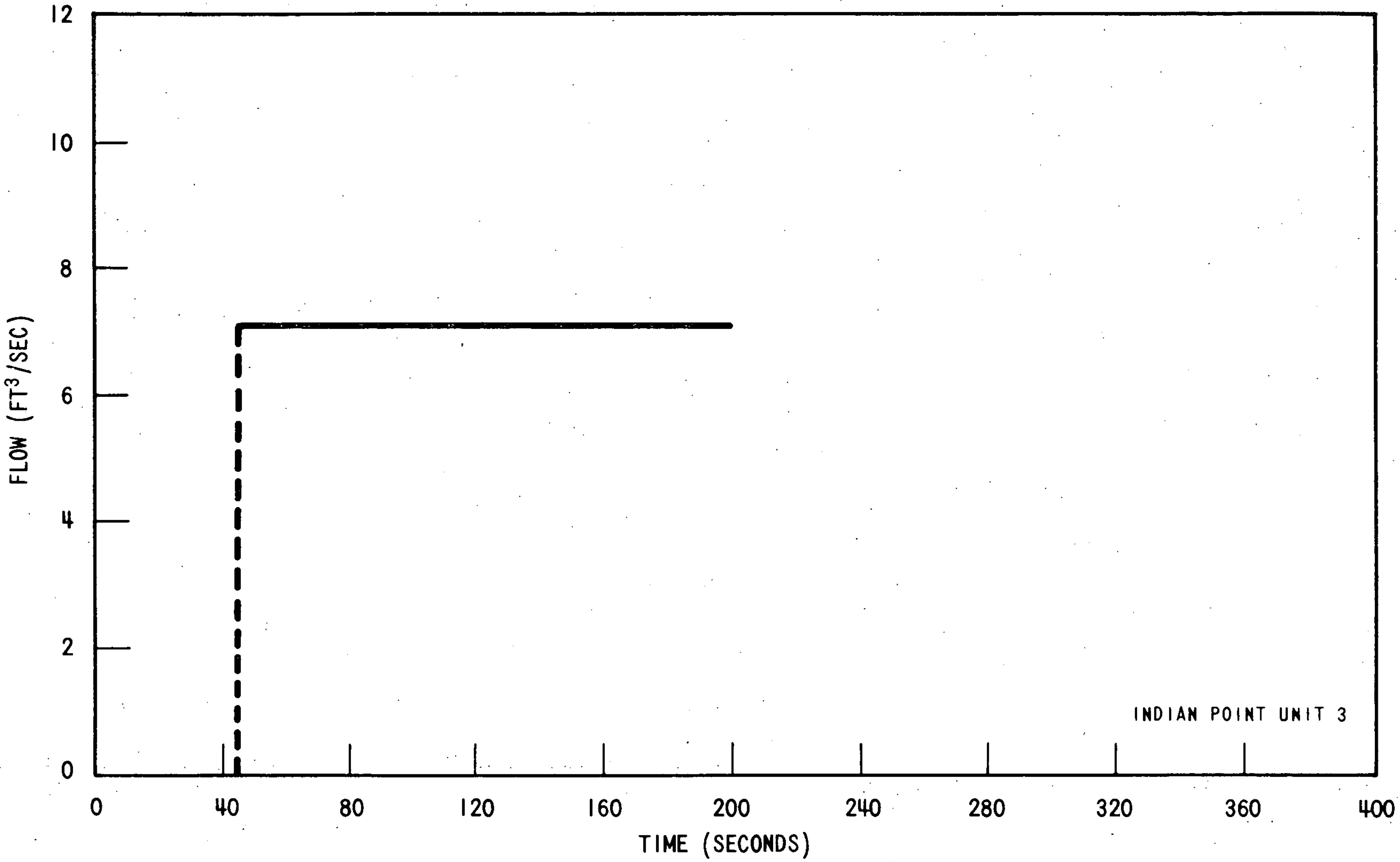

Figure IP3-12a. Pumped ECCS Flow (Reflood) — DECLG (C_D = 1.0)

10.807-23

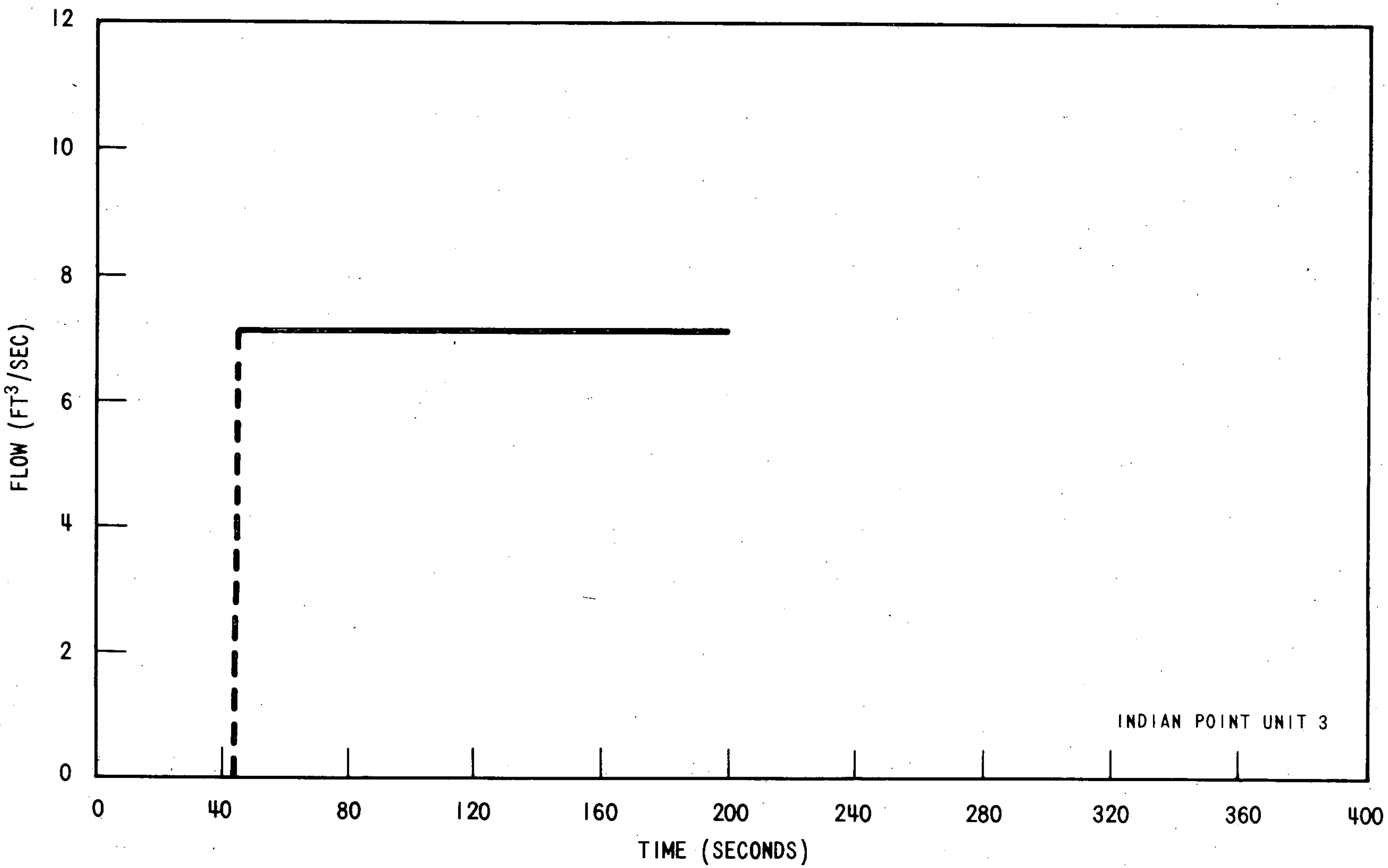

Figure IP3-12b. Pumped ECCS Flow (Reflood) – DECLG (C_D = 0.8)

10,807-24

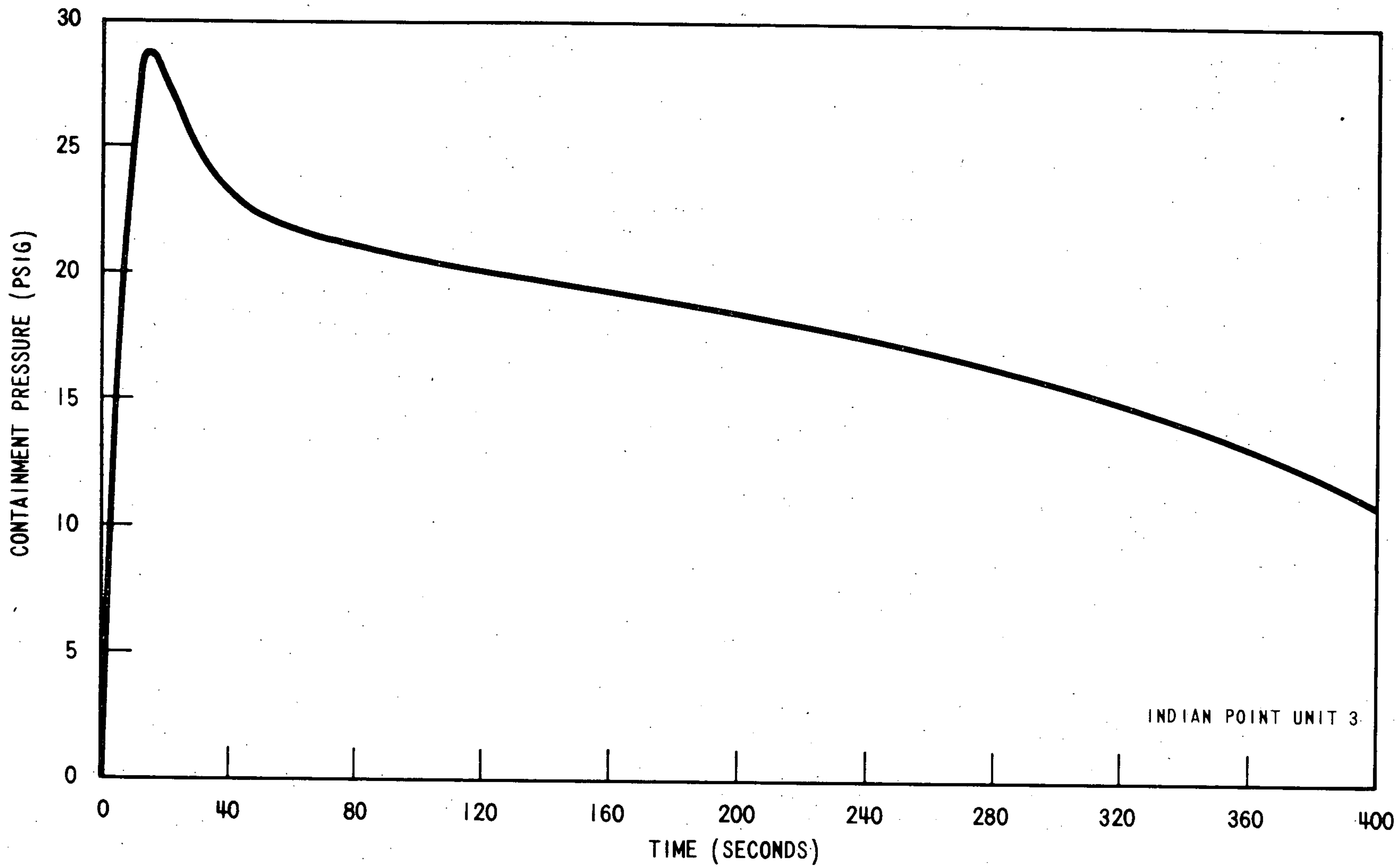

Figure IP3-13a. Containment Pressure — DECLG (C_D = 1.0)

10.807-25

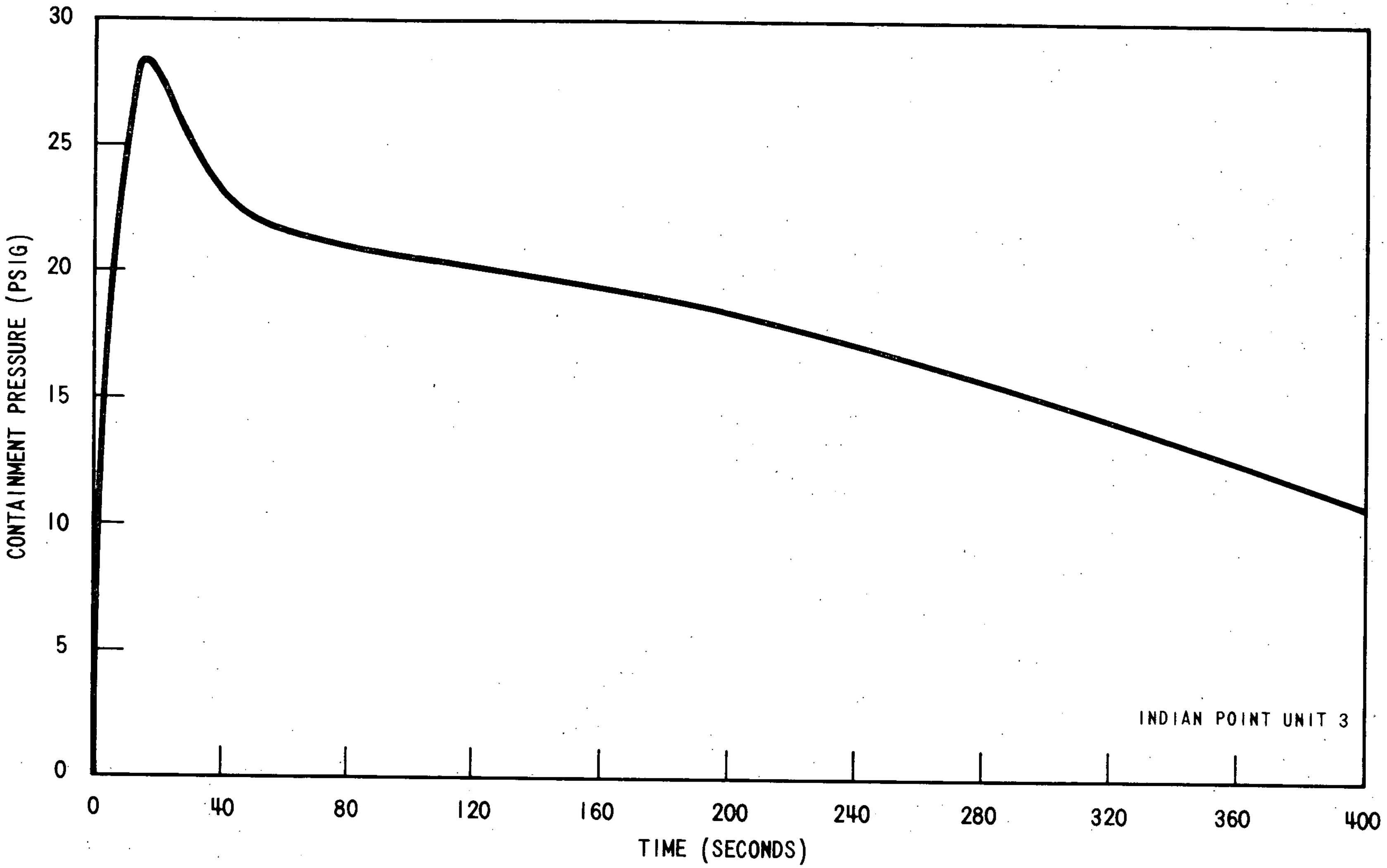

Figure IP3-13b Containment Pressure — DECLG (C_D = 0.8)

10.807-26

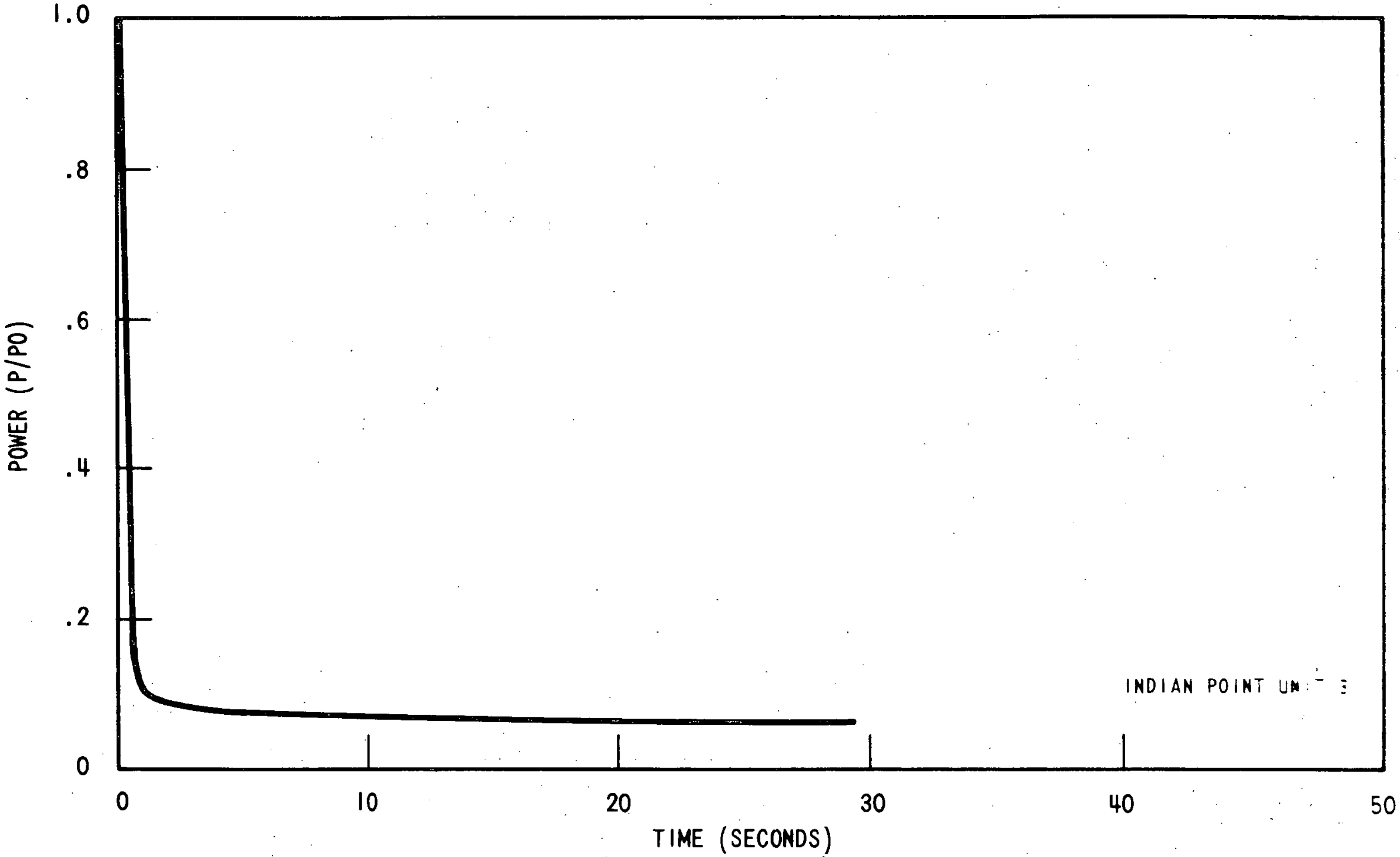

Figure IP3-14a. Core Power Transient — DECLG (C_D=1.0)

10,807-28

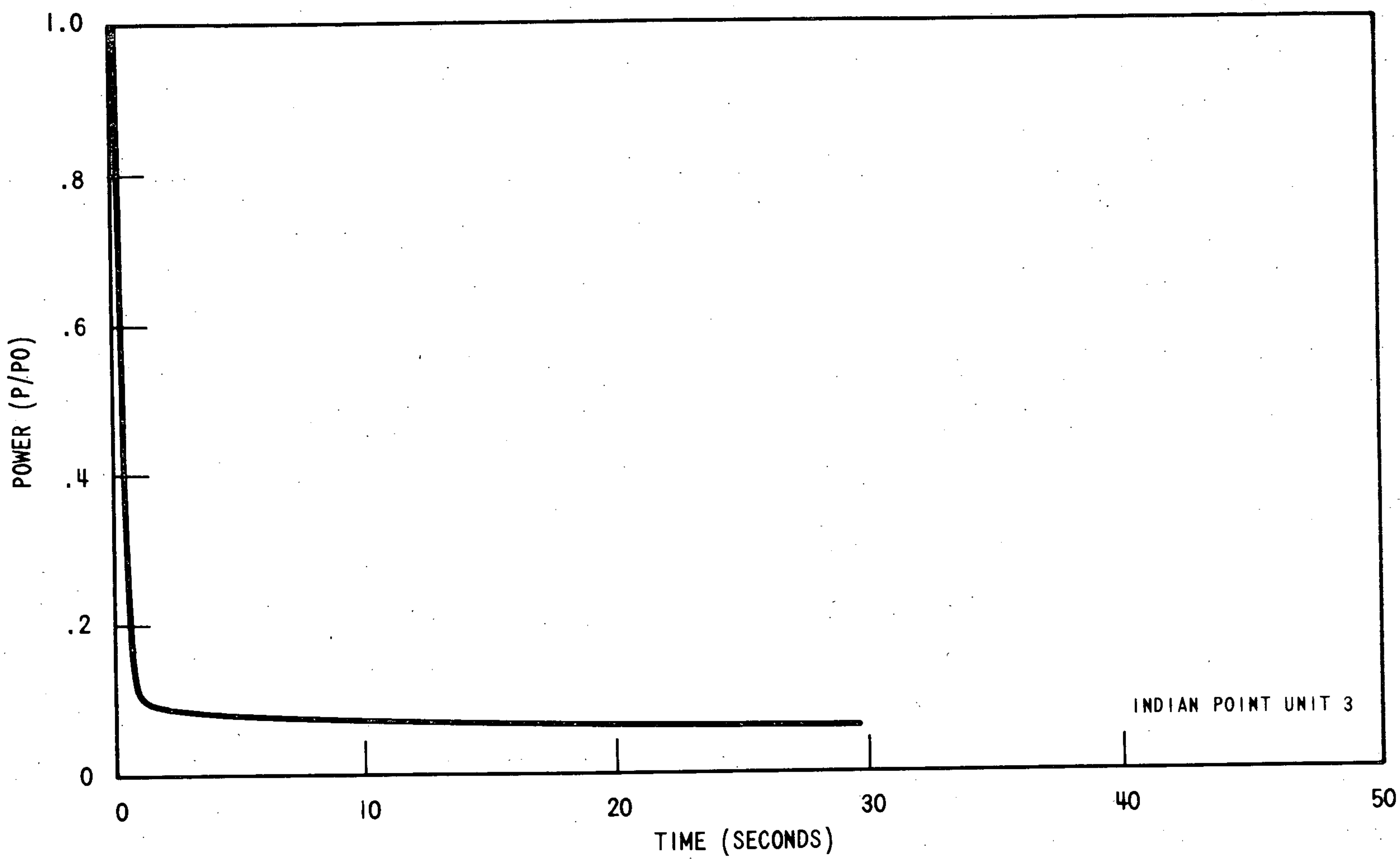

Figure IP3-14b. Core Power Transient – **DECLG** (C_D=0.8)

10.807-27

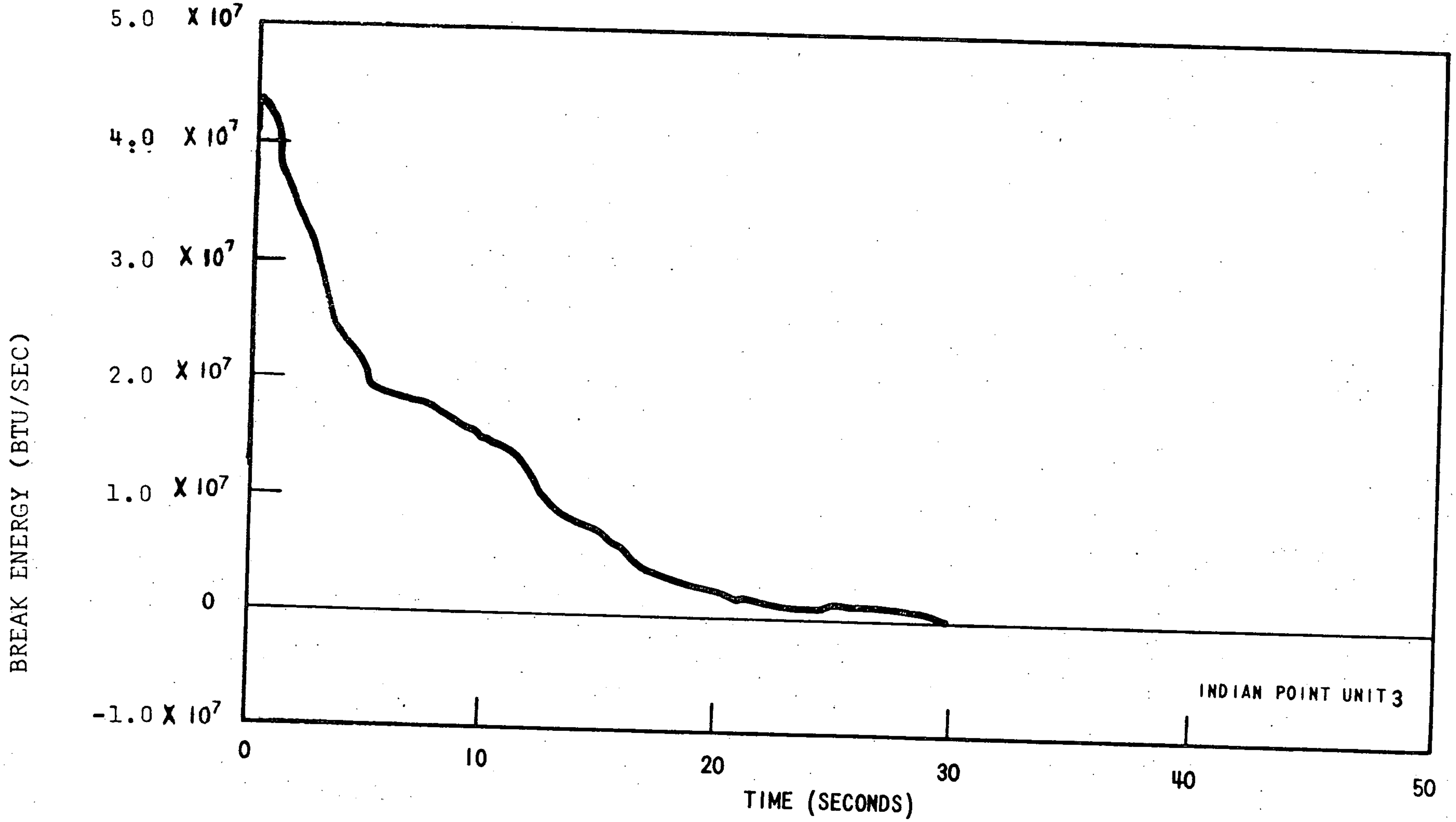

Figure IP3-15. Break Energy Released to Containment DECLG (C_D=1.0)

10,806-29

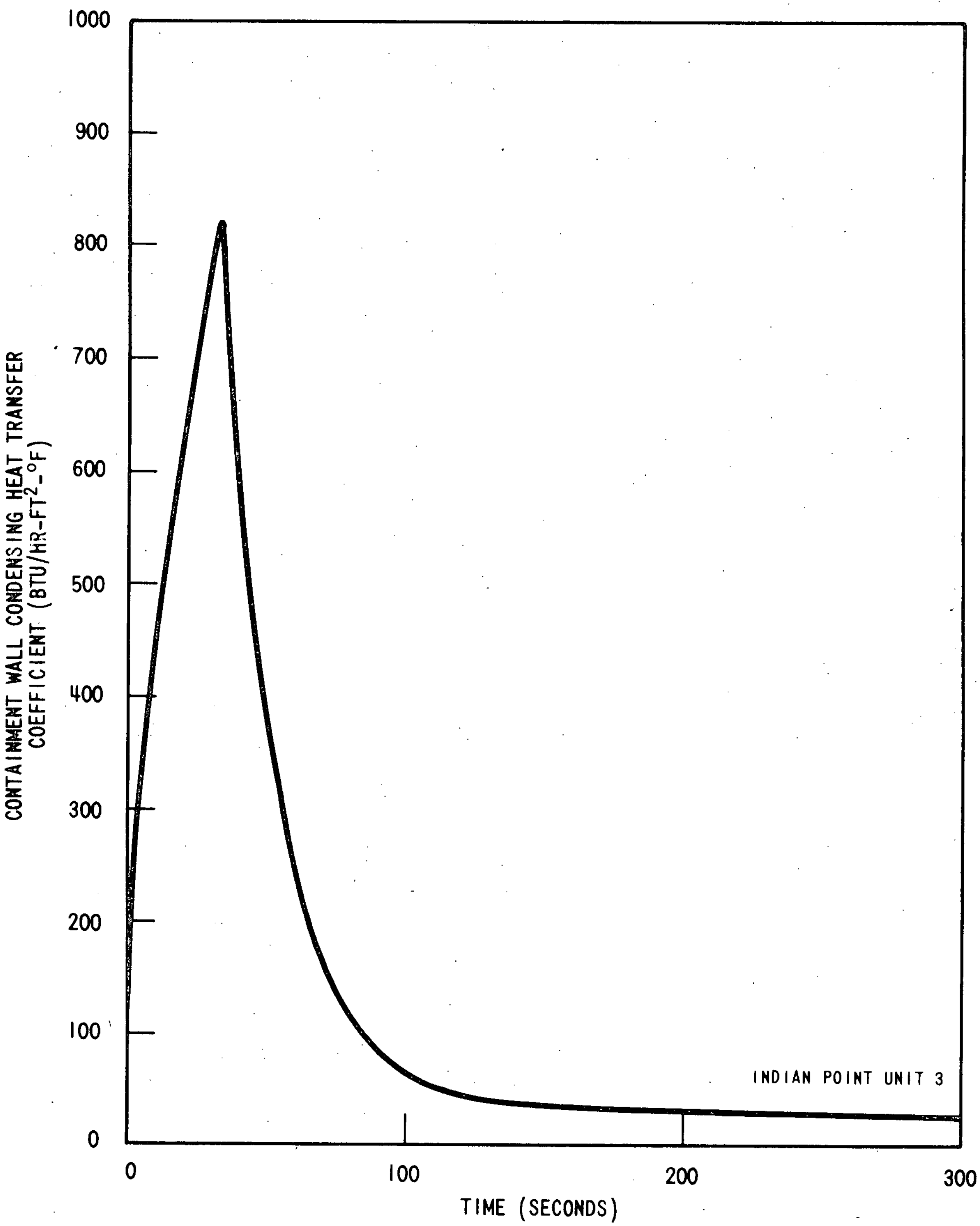

Figure IP3-16. Containment Wall Condensing Heat Transfer Coefficient

DECLG (C_D=1.0)