PSE&G

Public Service Electric and Gas Company 80 Park Place Newark, N.J. 07101 Phone 201/430-7000

November 8, 1978

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Olan D. Parr, Chief
Light Water Reactors Branch 3
Division of Project Management

Gentlemen:

RESPONSE TO REQUESTS FOR
ADDITIONAL INFORMATION
NO. 2 UNIT
SALEM NUCLEAR GENERATING STATION
DOCKET NO. 50-311

Public Service Electric and Gas Company hereby submits 40 copies of its responses to your request for additional information regarding the ECCS Analysis Question 5.65 and the RHR System Question 9.61. The information contained herein will be incorporated into the Salem FSAR in an amendment to our application.

Should you have any questions, please do not hesitate to contact us.

Very truly yours,

R. L Mittl
General Manager -
Licensing and Environment
Engineering and Construction

Attachment

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The Energy People

95-0942

Question 5.65

Section 50.34 of 10 CFR part 50 requires that an analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents be performed in accordance with the requirements of Section 50.46. Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 sets forth certain required and acceptable features of evaluation models. Appendix K states in part, that the containment pressure used for evaluating cooling shall not exceed a pressure calculated conservatively for this purpose. It further requires that the calculation include the effects of operation of all installed pressure reducing systems and processes. Branch Technical Position CSB 6-1, "Minimum Containment Pressure Wheel for PWR ECCS Performance Evaluation," provides additional guidance for the performance of a minimum containment pressure analysis and should be used when the analysis is performed. Therefore, state the minimum containment pressure that has been used in the analysis of the emergency core cooling system. Justify this value to be conservatively low by describing the conservatism in the assumptions of initial containment conditions, modeling of the containment heat sinks, heat transfer coefficients to the heat sinks, heat sink surface area and any other parameter assumed in the analysis. Provide the containment pressure, temperature and sump temperature response for the most conservative assumptions. Your November 2, 1977 submittal on this matter was incomplete.

Answer

The Loss of Coolant Accident (LOCA) has been reanalyzed for Salem Units 1 and 2. The following information amends the Safety Analysis Report section on Major Reactor Coolant System Pipe Ruptures.

The description of the various aspects of the LOCA analysis is given in WCAP-8339 [1]. The individual computer codes which comprise the Westinghouse Emergency Core Cooling System (ECCS) evaluation model are described in detail in separate reports [2-5] along with code modifications specified in references 6, 7 an 8. The analysis presented here was performed with the February 1978 version of the evaluation model which includes modifications delineated in references 9, 10, 11 and 12.

Results

The analysis of the loss of coolant accident is performed at 102 percent of the licensed core power rating. The peak linear power and total core power used in the analysis are given in Table 2. Since there is margin between the value of peak linear power density used in this analysis and the value of the peak linear power density expected during plant operation, the peak clad temperature calculated in this analysis is greater than the maximum clad temperature expected to exist.

Table 1 presents the occurrence time for various events throughout the accident transient.

Table 2 presents selected input values and results from the hot fuel rod thermal transient calculation. For these results, the hot spot is defined as the location of maximum peak clad temperatures. The location is specified in Table 2 for each break analyzed. The location is indicated in feet which presents elevation above the bottom of the active fuel stack.

Table 3 presents a summary of the various containment systems parameters and structural parameters which were used as input to the COCO computer code [5] used in this analysis.

Tables 4 and 5 present reflood mass and energy releases to the containment, and the broken loop accumulator mass and energy release to the containment, respectively.

The results of several sensitivity studies are reported [13]. These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

Figures 1 through 17 present the transients for the principle parameters for the break sizes analyzed. The following items are noted:

Figures 1A - 3C: Quality, mass velocity and clad heat transfer coefficient for the hotspot and burst locations

Figures 4A - 6C: Core pressure, break flow, and core pressure drop. The break flow is the sum of the flowrates from both ends of the guillotine break. The core pressure drop is taken as the pressure just before the core inlet to the pressure just beyond the core outlet

Figures 7A-9C: Clad temperature, fluid temperature and core flow. The clad and fluid temperatures are for the hot spot and burst locations

Figures 10A - 11C: Downcomer and core water level during reflood, and flooding rate

Question 5.65 (Continued)

Figures 12A - 13C: Emergency core cooling system flowrates, for both accumulator and pumped safety injection

Figures 14A - 15C: Containment pressure and core power transients

Figures 16, 17: Break energy release during blowdown and the containment wall condensing heat transfer coefficient for the worst break

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 (14) that is:

1. The calculated peak clad temperature does not exceed 2200°F based on a total core peaking factor of 2.32.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircalloy 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 long-lived radioactivity remaining in the core.

References

1. Bordelon, F. M., Massie, H. W., And Zordan, T. A., Westinghouse ECCS Evaluation Model-Summary, "WCAP-8339, July 1974.

2. Bordelon, F.M., et al., "SATAN-VI Program" Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary Version), WCAP-8306 (Non-Proprietary Version), June 1974.

3. Bordelon, F.M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary Version), WCAP-8305 (Non-Proprietary Version), June 1974.

4. Kelly, R.D., et al., "Calculation Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)." WCAP-8170 (Proprietary Version), WCAP-8171 (Non-Proprietary Version), June 1974.

5. Bordelon, F.M., and Murphy E.T., "Containment Pressure Analysis Code (COCO), "WCAP-8327 (Proprietary Version), WCAP-8326 (Non-Proprietary Version), June 1974.

6. Bordelon, F.M., et al., "The Westinghouse ECCS Evaluation Model: Supplementary Information," WCAP-8471 (Proprietary Version), WCAP 8472 (Non-Proprietary Version), January 1975.

7. Westinghouse ECCS Evaluation Model, October, 1975 Versions," WCAP-8622 (Proprietary Version), WCAP-8623 (Non-proprietary Version), January 1975.

8. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassalo of the Nuclear Regulatory Commission, letter number NS-CE-924, January 23, 1976.

9. Kelly, R. D., Thompson, C. M., et. al., "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCA's During Operation With One Loop Out of Service for Plants Without Loop Isolation Valves," WCAP-9166, February, 1978.

10. Eicheldinger C., "Westinghouse ECCS Evaluation Model, February 1978 Version," WCAP-9220 (Proprietary Version), WCAP-9221 (Non-Proprietary Version), February, 1978.

11. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMS-1830, June 16, 1978.

12. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMS-1834, June 20, 1978.

Question 5.65 (Continued)

13. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary Version), WCAP-8356 (Non-Proprietary Version), July 1974.

14. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10CFR50.46 and Appendix K of 10CFR50.46. Federal Register, Volume 39, Number 3, January 4, 1974.

TABLE 1

LARGE BREAK BREAK - TIME SEQUENCE OF EVENTS

EVENT	OCCURRENCE TIME (SECONDS)		
	DECLG, C_D = 0.6	DECLG, C_D = 0.8	DECLG, C_D = 1.0
Accident Initiation	0.0	0.0	0.0
Reactor Trip Signal	1.66	1.66	1.65
Safety Injection Signal	1.03	0.92	0.86
Start Accumulator Injection	16.8	14.6	14.1
End of ECC Bypass	27.51	26.0	25.4
End of Blowdown	30.46	28.8	28.1
Bottom of Core Recovery	42.5	40.95	40.34
Accumulators Empty	53.64	51.6	51.15
Start Pumped ECC Injection	26.03	25.92	25.86

TABLE 2

LARGE BREAK - ANALYSIS INPUT AND RESULTS

Quantities in the calculations:

Licensed core power rating	102% of 3411 MWt
Total core peaking factor	2.32
Peak linear power	12.63 kw/ft
Accumulator water volume	850 cubic feet per tank
Accumulator pressure	600 PSIA
Number of Emergency Core Cooling Pumps Operating	3
Steam Generator Tube Plugging Level	0 percent (uniform)
Fuel Parameters - Cycle 1 Region All	

Results	DECLG, C_D = 0.6	DECLG, C_D = 0.8	DECLG, C_D = 1.0
Peak clad temperature (^oF)	1968	2130	2108
Location (feet)	7.5	6.0	6.0
Maximum local clad/water reaction (%)	2.87	6.1	5.97
Location (feet)	7.5	6.0	6.0
Total core clad/water reaction (%)	<0.3	<0.3	<0.3
Hot rod burst time (seconds)	33.0	28.1	31.4
Location (feet)	6.25	6.0	6.0

TABLE 3
CONTAINMENT DATA

NET FREE VOLUME	2.62×10^6 ft.^{3}
INITIAL CONDITIONS	
Pressure	14.7 psia
Temperature	90°F
RWST Temperature	40°F
Service Water Temperature	32°F
Outside Temperature	0°F
SPRAY SYSTEM	
Number of Pumps Operating	2
Runout Flow Rate	3800 gpm each
Actuation Time	27 sec.
SAFEGUARDS FAN COOLERS	
Number of Fan Coolers Operating	5
Fastest Post Accident Initiation of fan coolers.	30 sec.

STRUCTURAL HEAT SINKS

Thickness (In.)	Area (Ft. 2)
.0075 Paint, .375 Steel, 54 concrete	49,923
2.5 Insulation, .375 Steel, 54 concrete	15,702
.0075 Paint, .5 Steel, 42 Concrete	32,327
.018 Paint, 42 Concrete	12,883
.018 Paint, 12 Concrete	10,912
.018 Paint, 20.5 Concrete	10,416
.014 Paint, 18 Concrete	35,000
.187 Steel, 23 Concrete	17,536
.0075 Paint, 0.1 Steel	73,870

TABLE 3 (Continued)
CONTAINMENT DATA

STRUCTURAL HEAT SINKS

Thickness (In.)	Area (Ft. 2)
.0075 Paint, .25 Steel	90,110
.0075 Paint, .5 Steel	23,688
.0075 Paint, .75 Steel	10,864
.0075 Paint, 1.0 Steel	9,441
.0075 Paint, 1.5 Steel	3,370
.0075 Paint, 2.5 Steel	1,916
.0625 Steel	53,460
1.125 Steel	1,832
.125 Steel	133,056
0.86 Steel	7,274
1.41 Steel	4,915

TABLE 4
REFLOOD MASS AND ENERGY
RELEASE TO THE CONTAINMENT

0.8 DECLG BREAK

Time (sec)	Mass (lbm/sec)	Energy (BTU/sec)
42.0	0.0	0.0
47.5	39.5	51,192
55.9	195	179,111
68.3	336.9	217,064
83.5	376.8	221,579
100.4	387.2	216,319
118.8	394.3	209,858
159.7	406.0	194,033

TABLE 5
BROKEN LOOP ACCUMULATOR MASS AND ENERGY
RELEASE TO THE CONTAINMENT

0.8 DECLG BREAK

Time (sec)	Mass (lb/sec)	Energy BTU/sec)
2.0	2,317	134,419
4.0	2,113	122,567
6.0	1,956	113,440
8.0	1,827	105,990
10.0	1,719	99,713
14.0	1,547	89,747
18.0	1,418	82,300
22.0	1,322	76,703
26.0	1,243	72,100
30.0	1,182	68,582
33.682	0.0	0.0

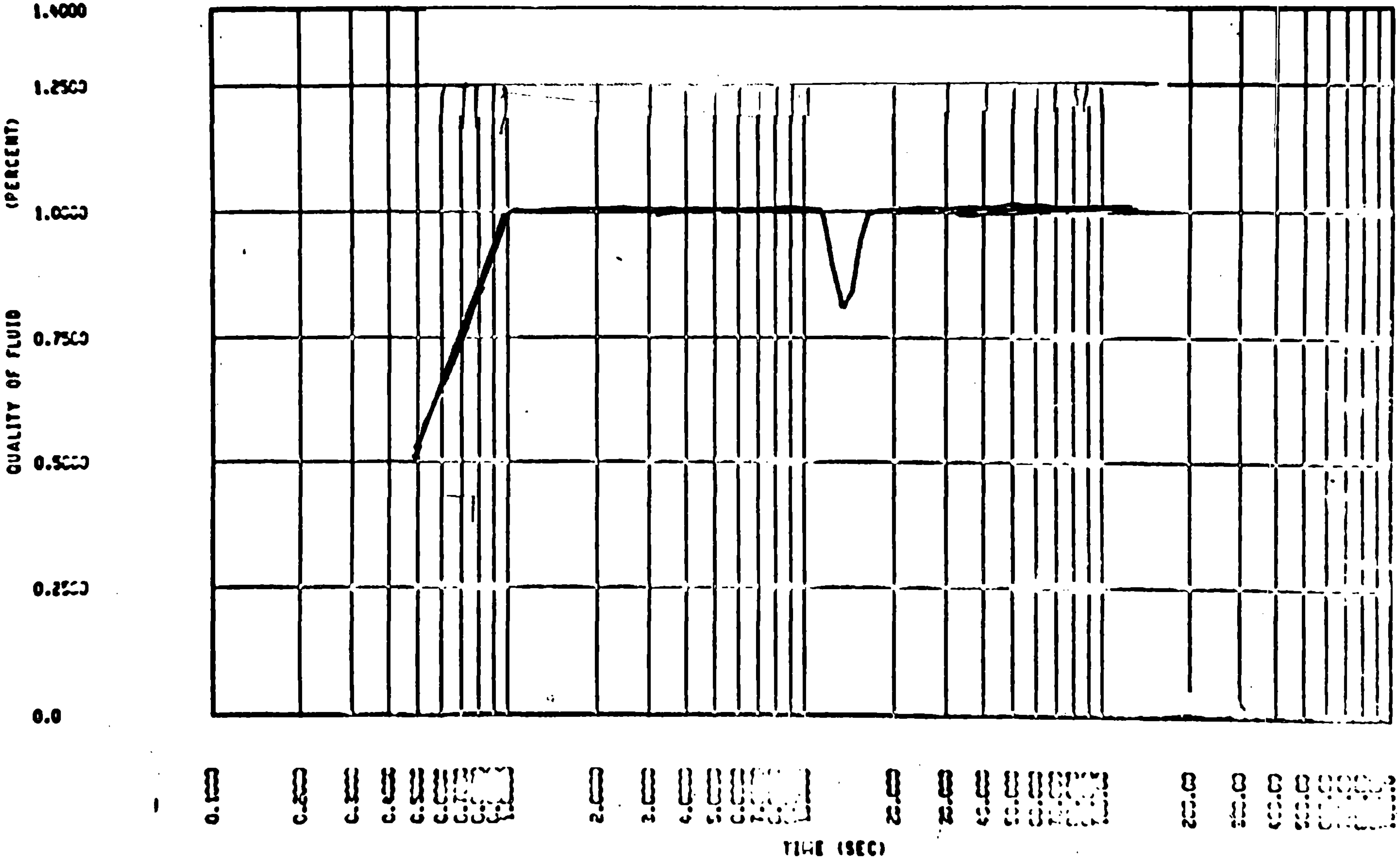


FIGURE 1a
FLUID QUALITY
- DECLG (C_D = 1.0)

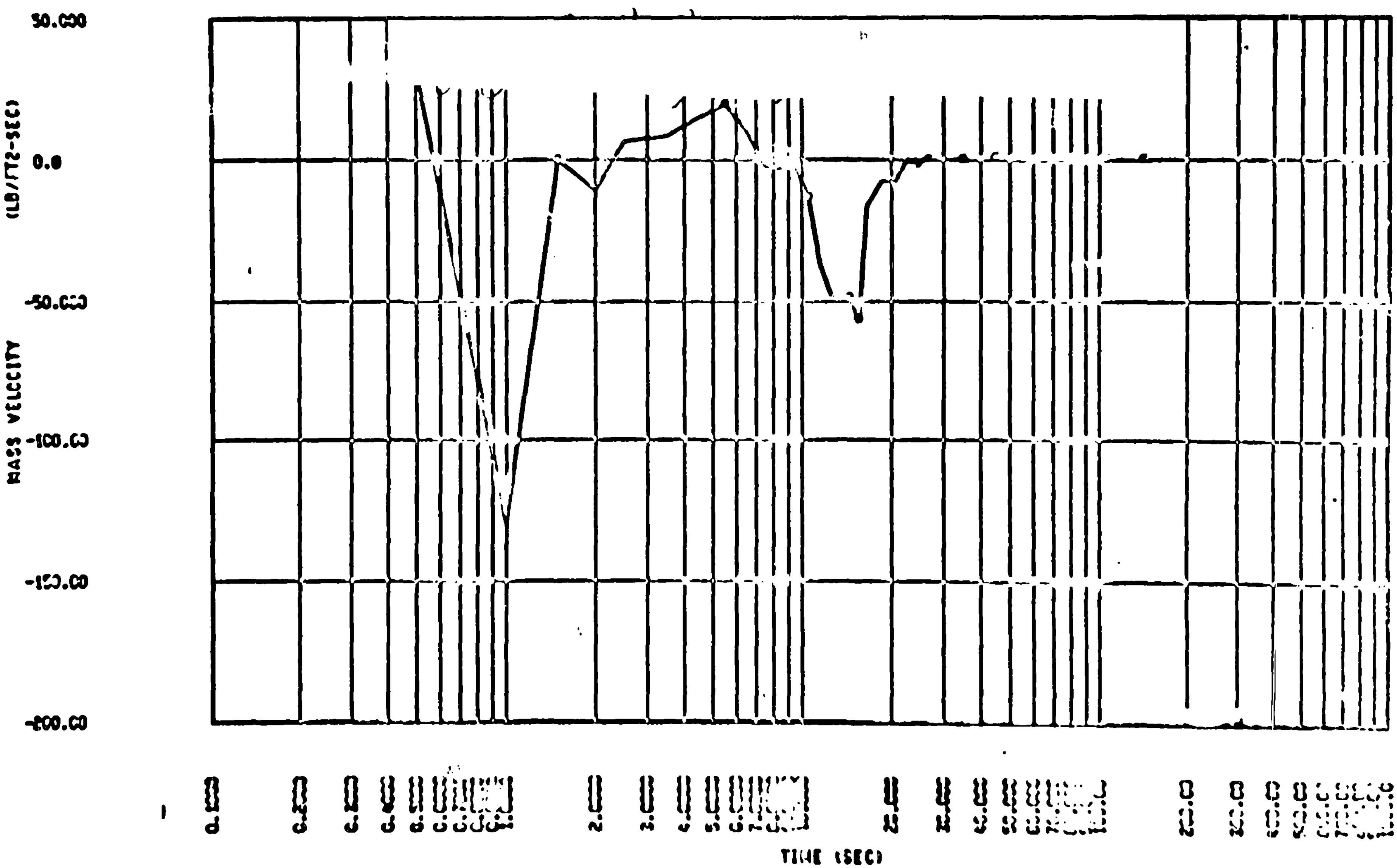


FIGURE 2a
MASS VELOCITY
DECL6 (C_D= 1.0)

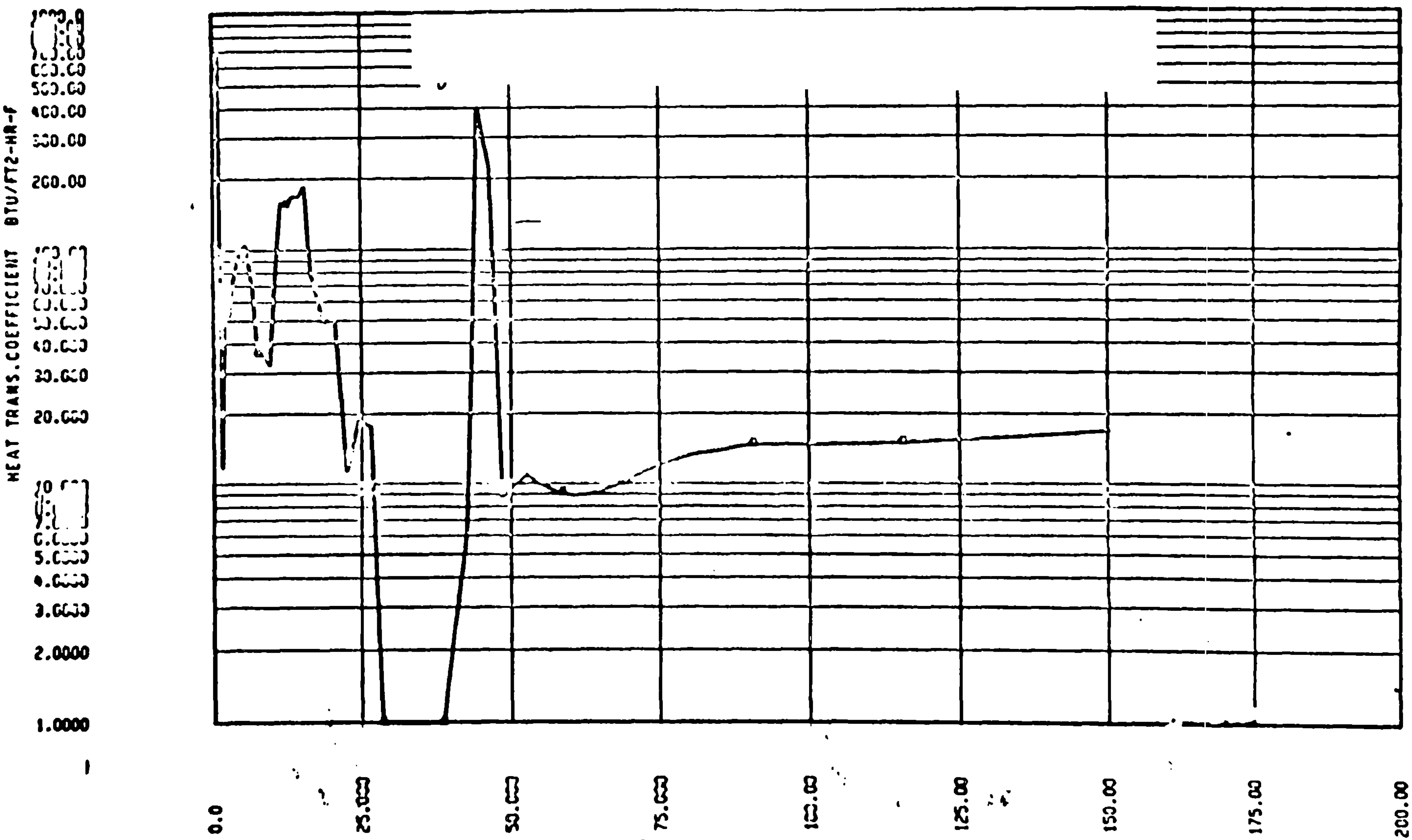


FIGURE 3a
HEAT TRANS. COEFFICIENT DECLG (CD=1.0)

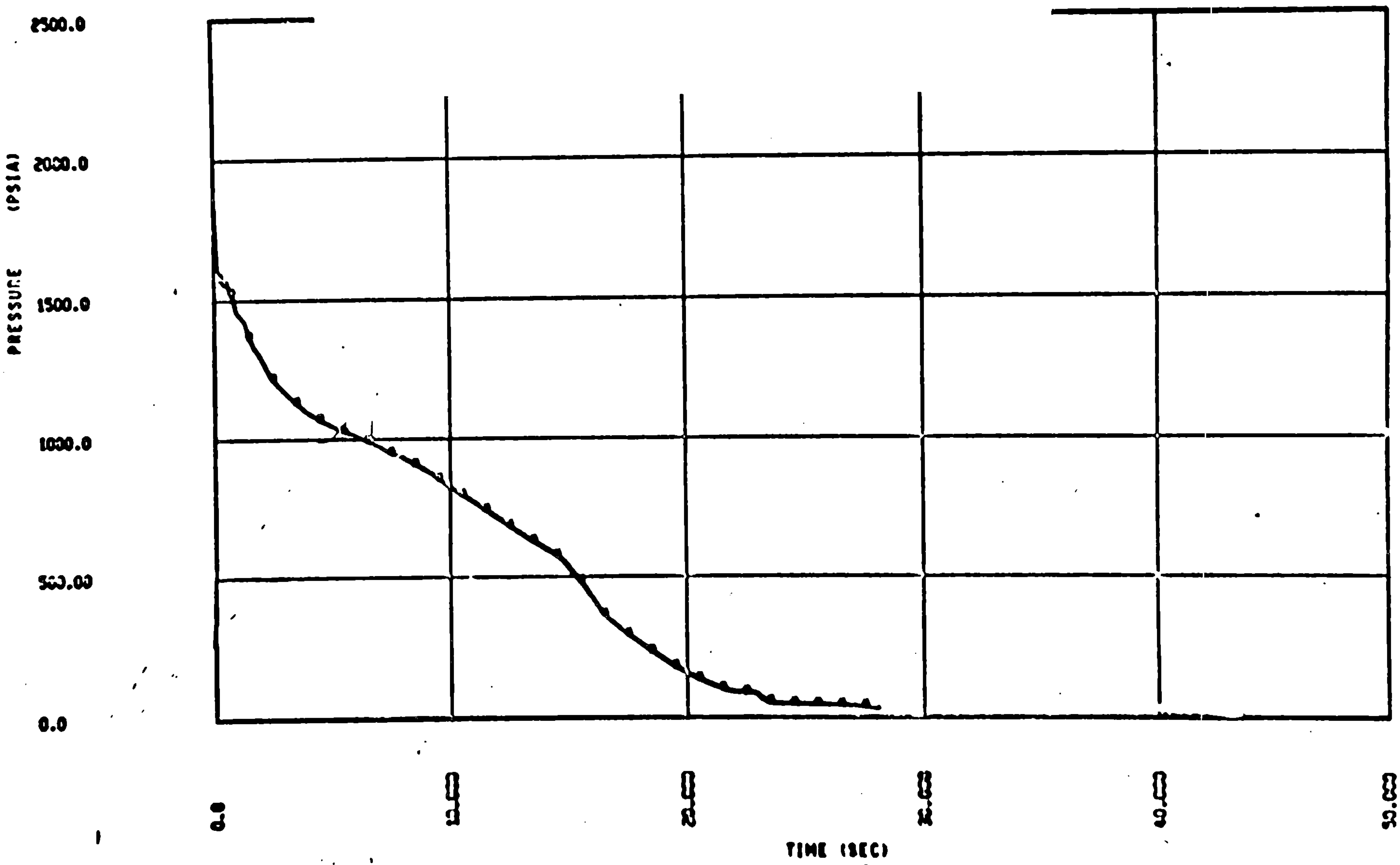


FIGURE 4a

CORE PRESSURE DECLG ($C_D = 1.0$)

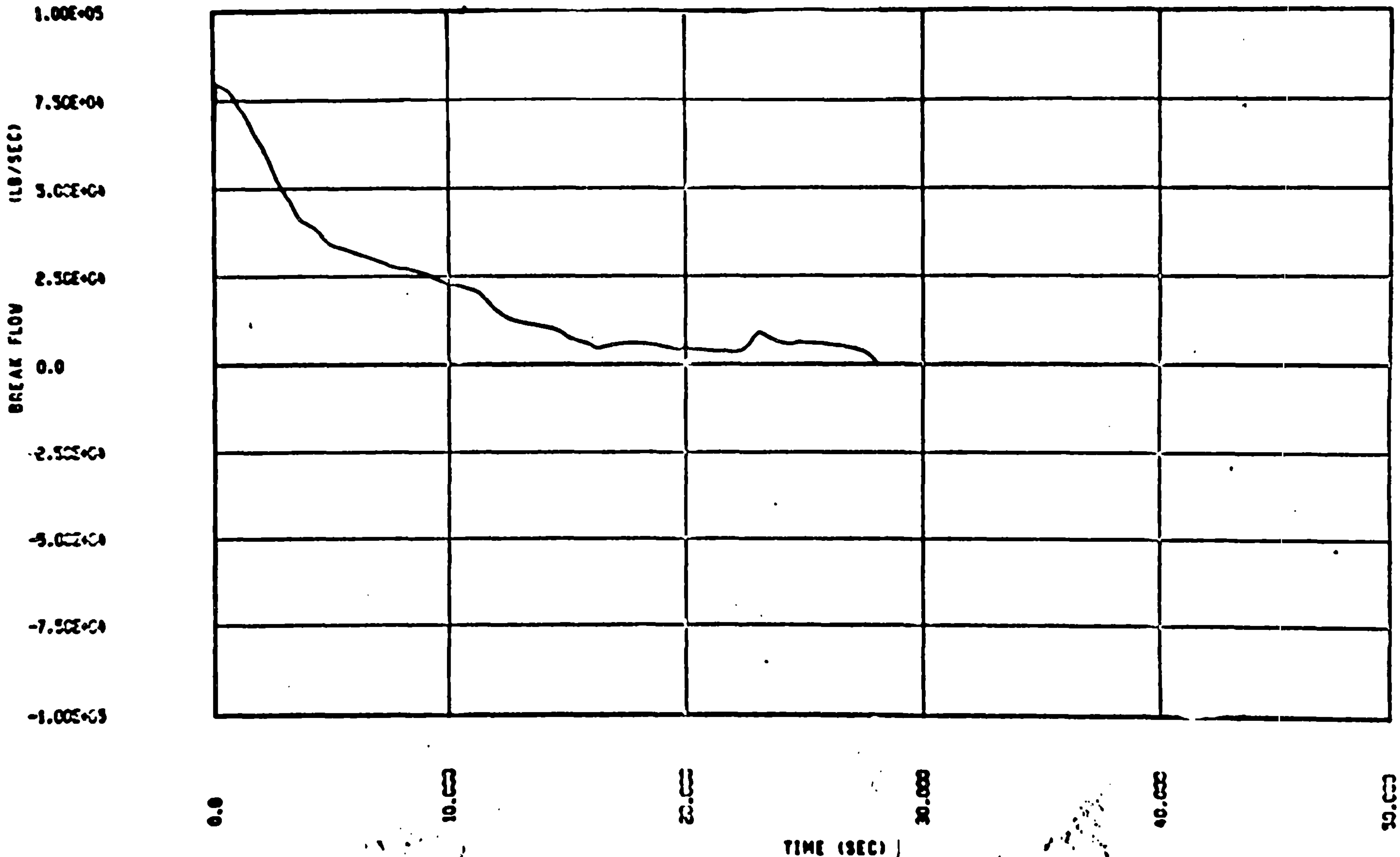


FIGURE 5a

BREAK FLOW RATE - DECLG (C_D = 1.0)

EAR RGY

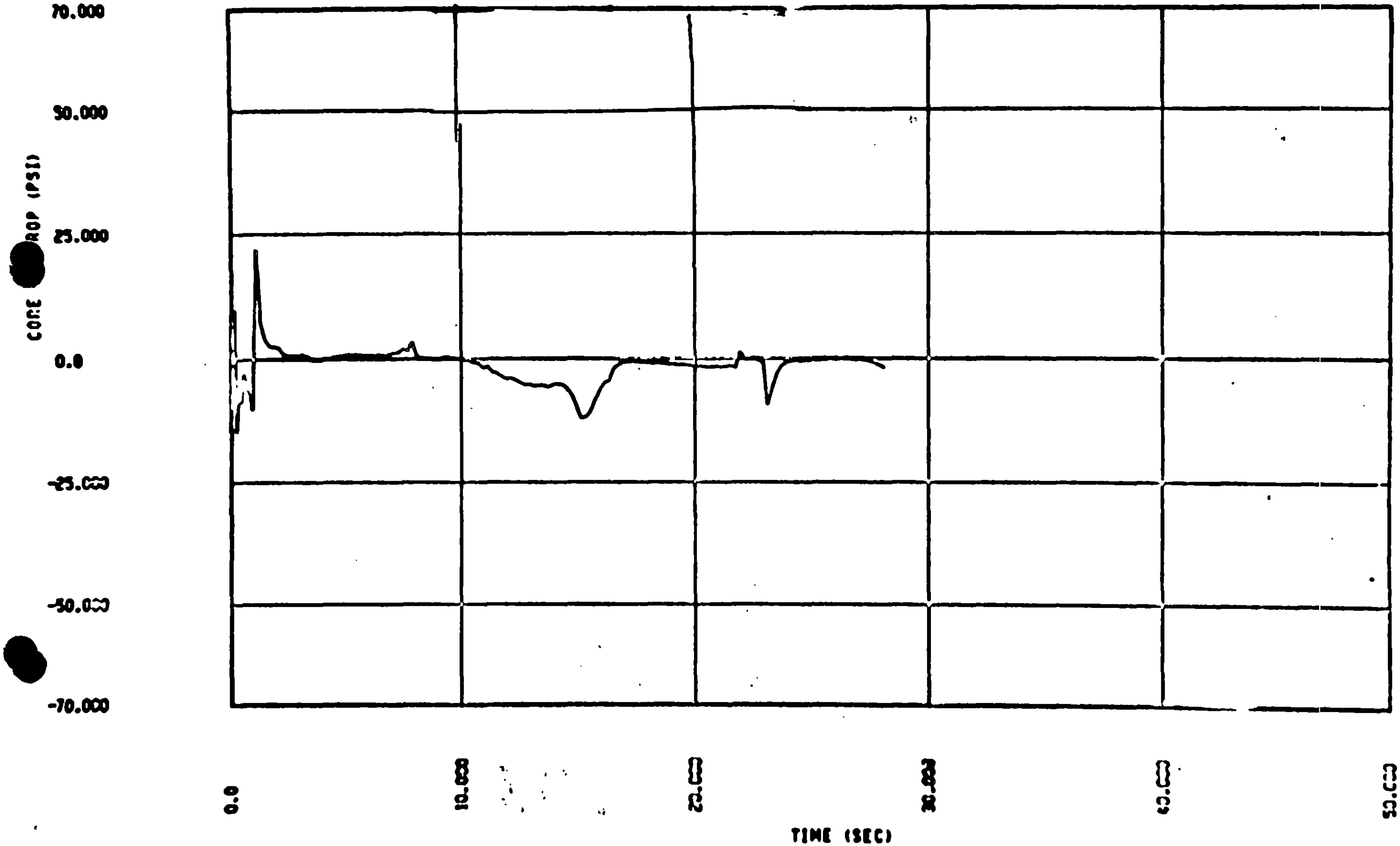


FIGURE 6a

E4R RAY

CORE PRESSURE - DECLG (C_D = 1.0)
DROP

FIGURE 7a

PEAK CLAD TEMP - DECLG ($C_D = 1.0$)

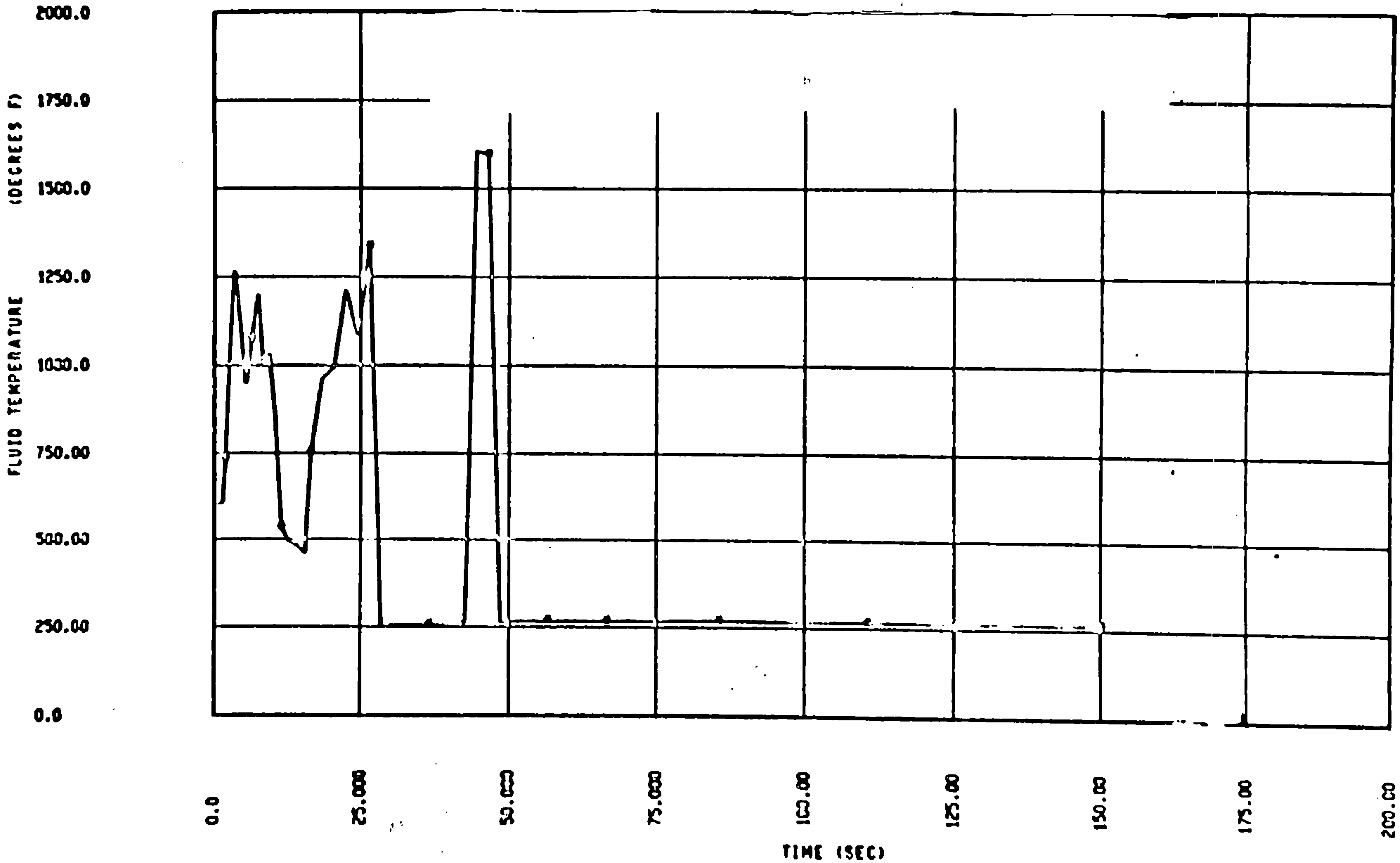
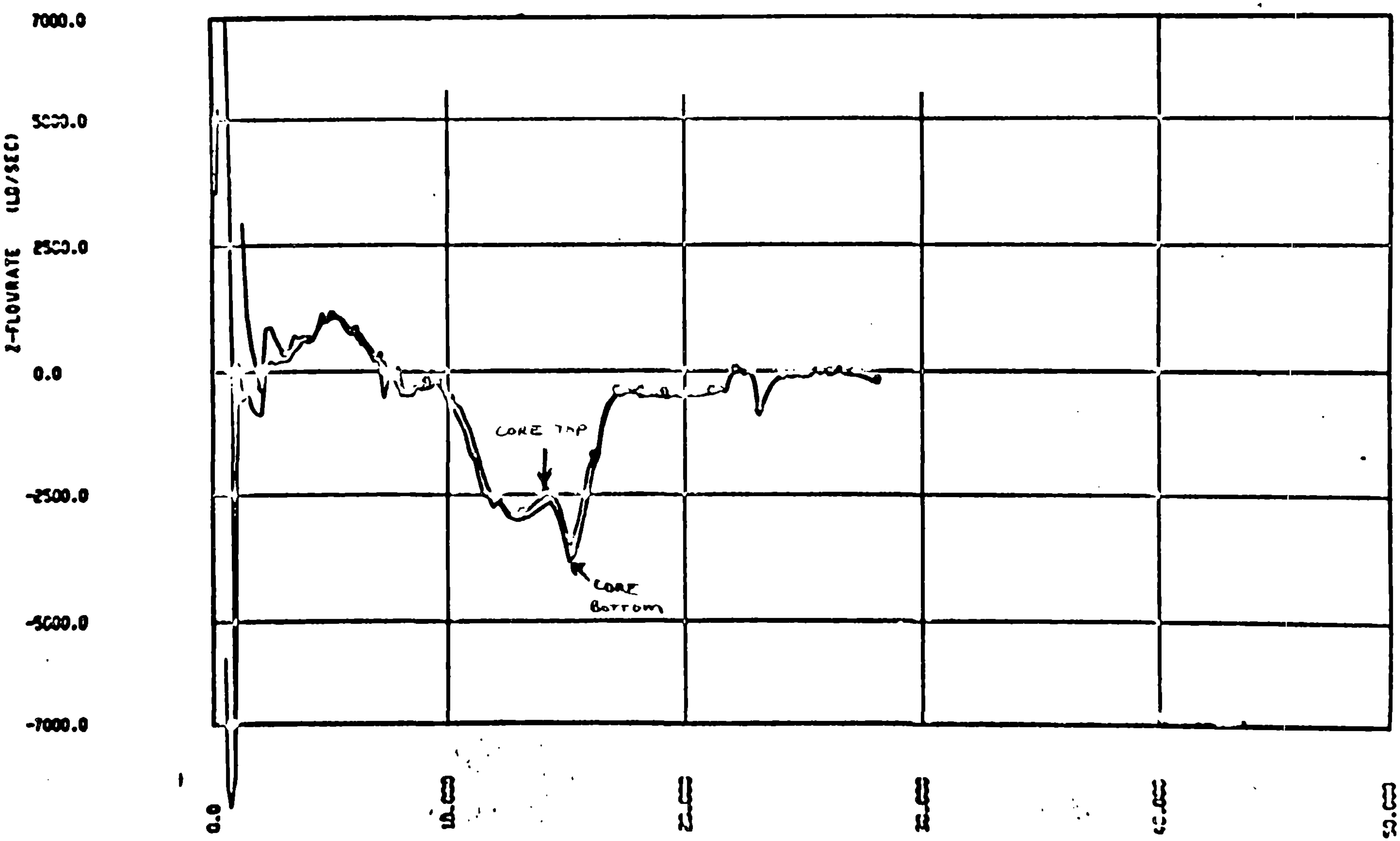


FIGURE 8a

E4R RAY

FLUID TEMP - DECLG (C_D = 1.0)



CORE FLOW RATE - TOP & BOTTOM FIGURE 9a

DECL6 ($C_D = 1.0$)

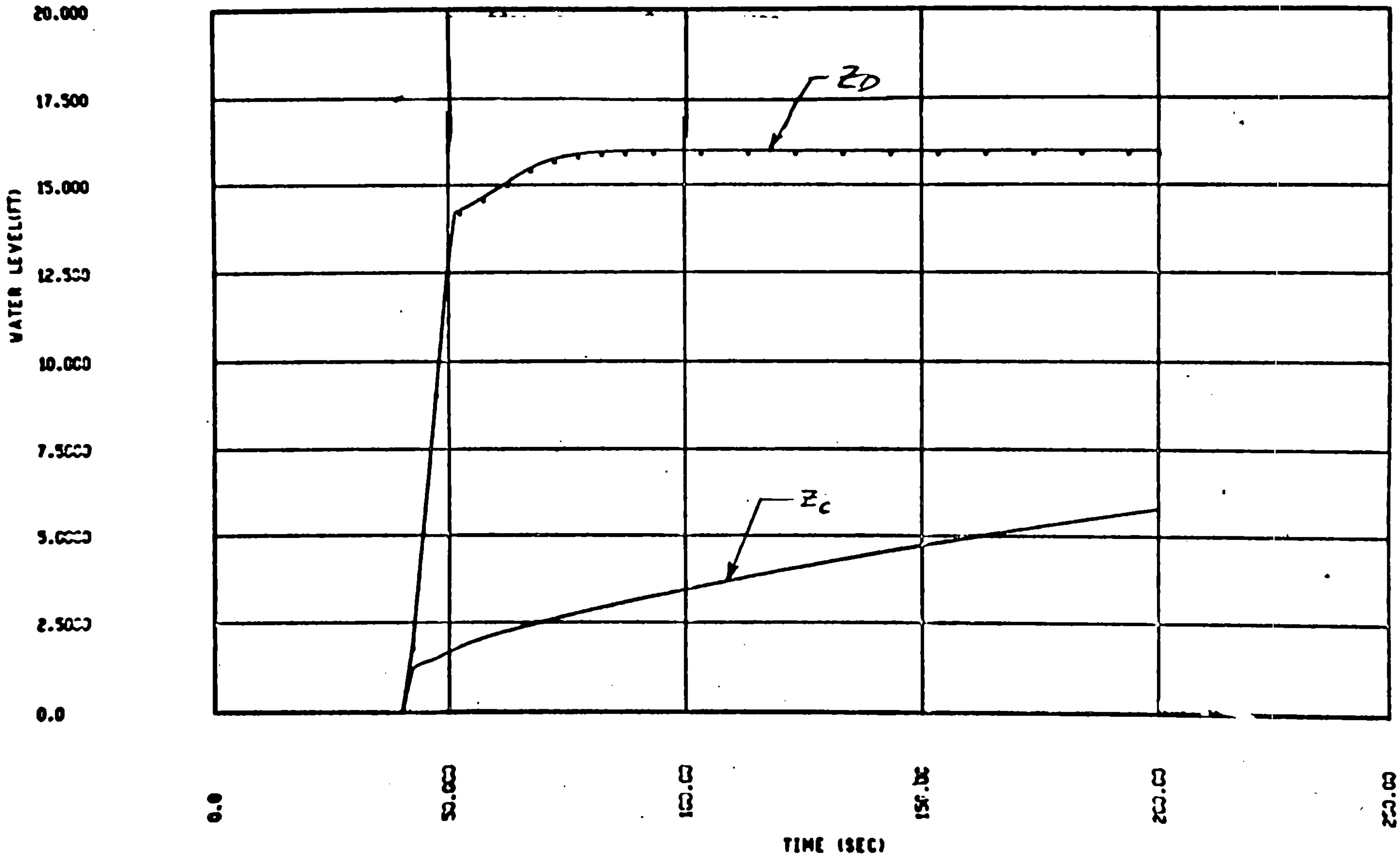


FIGURE 10a
REFLOOD TRANSIENT DOWNCOMER
AND CORE WATER LEVEL

DECIG (C_D = 1.0)

E4R RAV

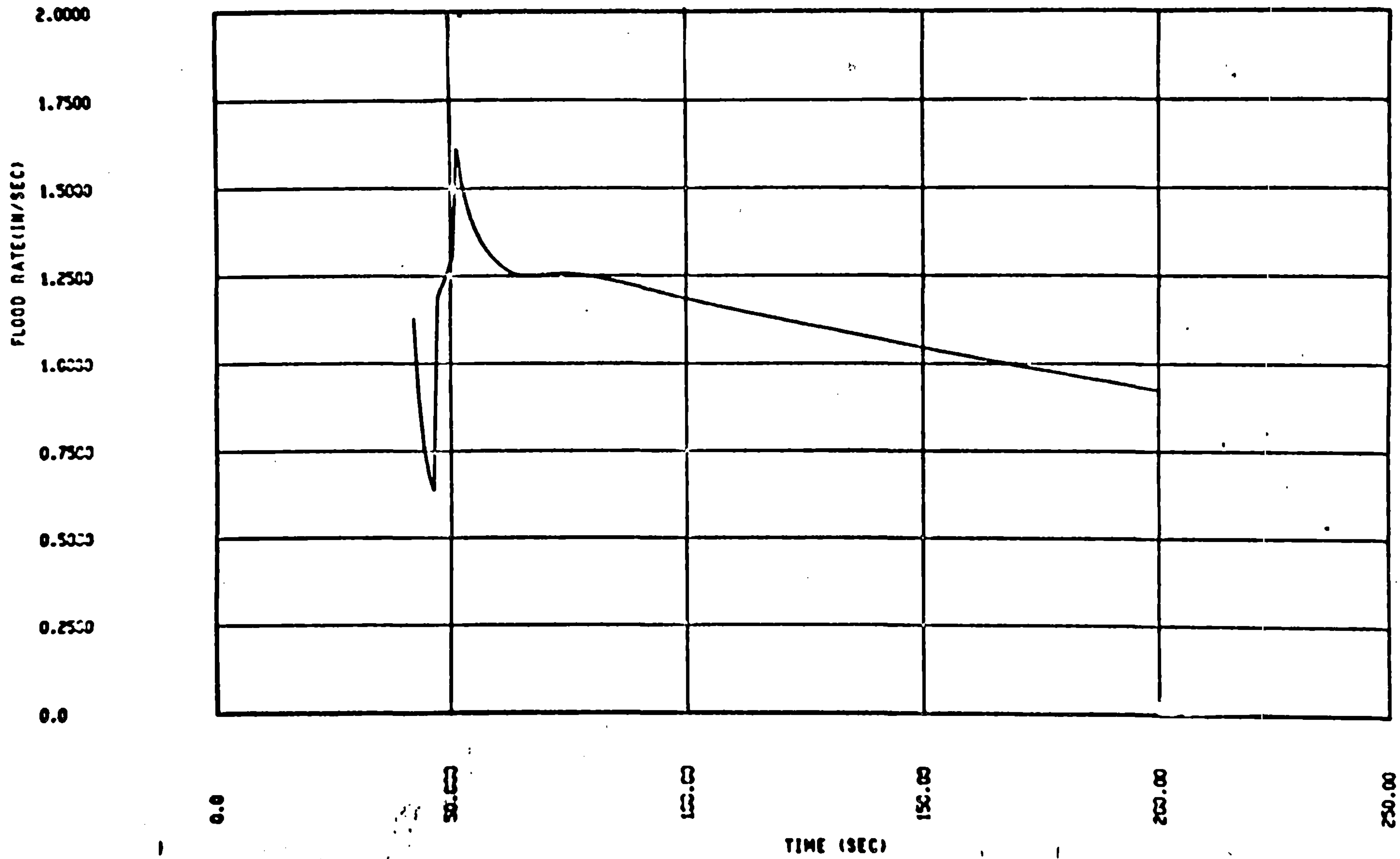


FIGURE 11a
REFLOOD TRANSIENT CORE INLET VELOCITY

DECLG (C_d = 1.0)

ER? RAY

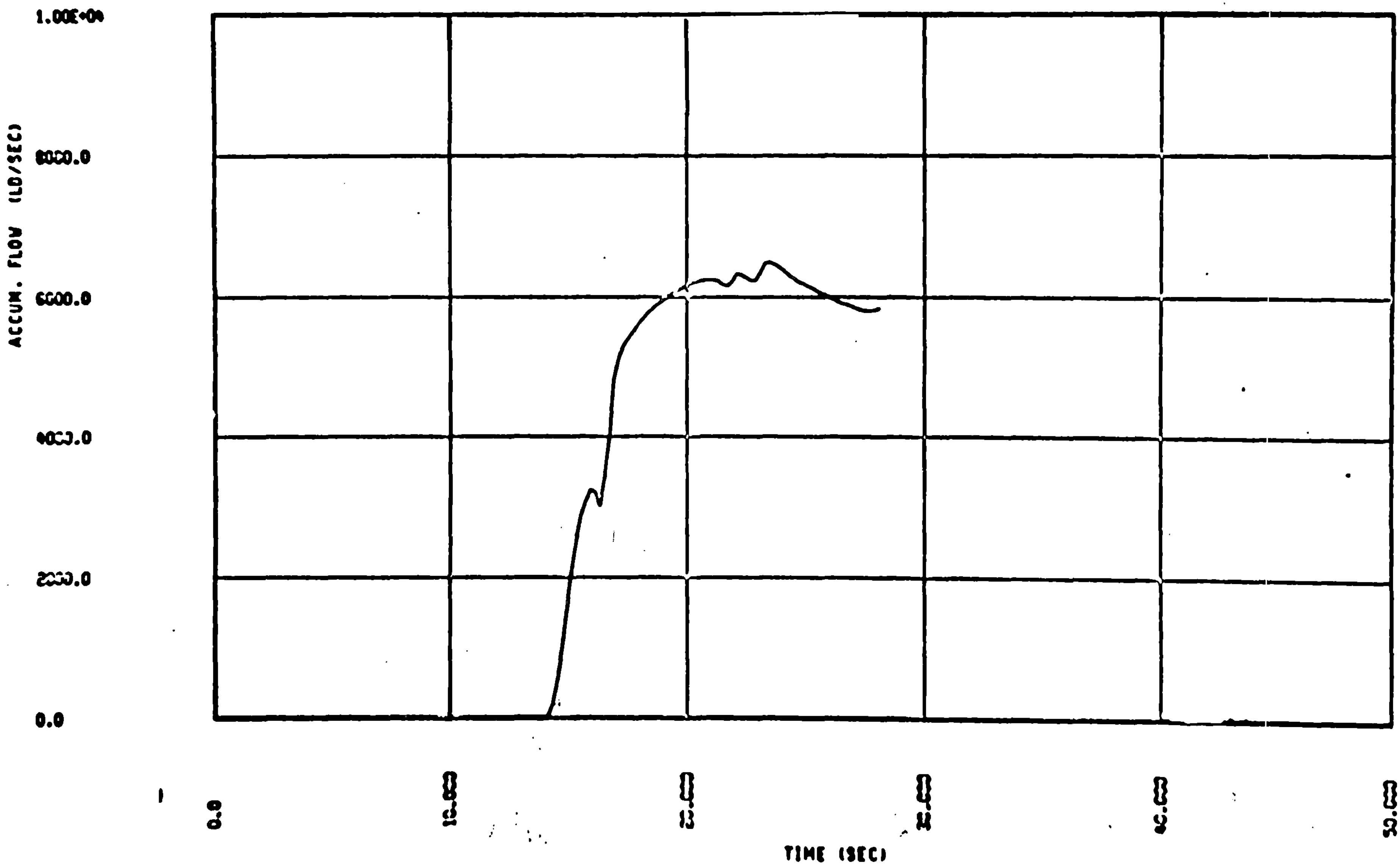


FIGURE 12a
ACCUMULATOR FLOW (BLOWDOWN) DECLG (C_D=1.0)

PUMPED ECCS FLOW DURING REFLOOD
DECLG $C_D = 1.0$

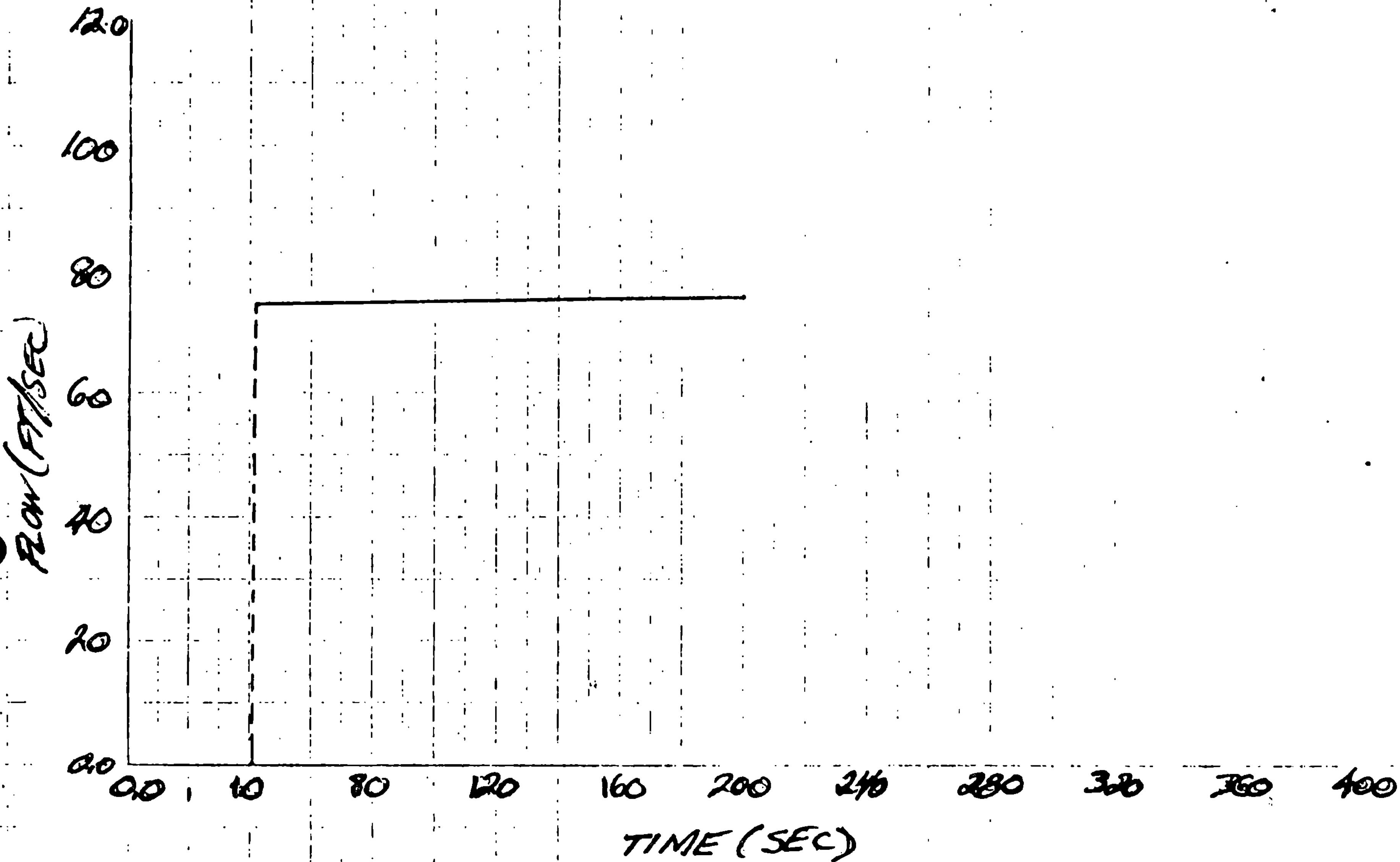


FIGURE 13a

K+E 10 X 10 TO THE CENTIMETER 18 X 25 CM. KEUFFEL & ESSER CO. MADE IN U.S.A. 46 1510

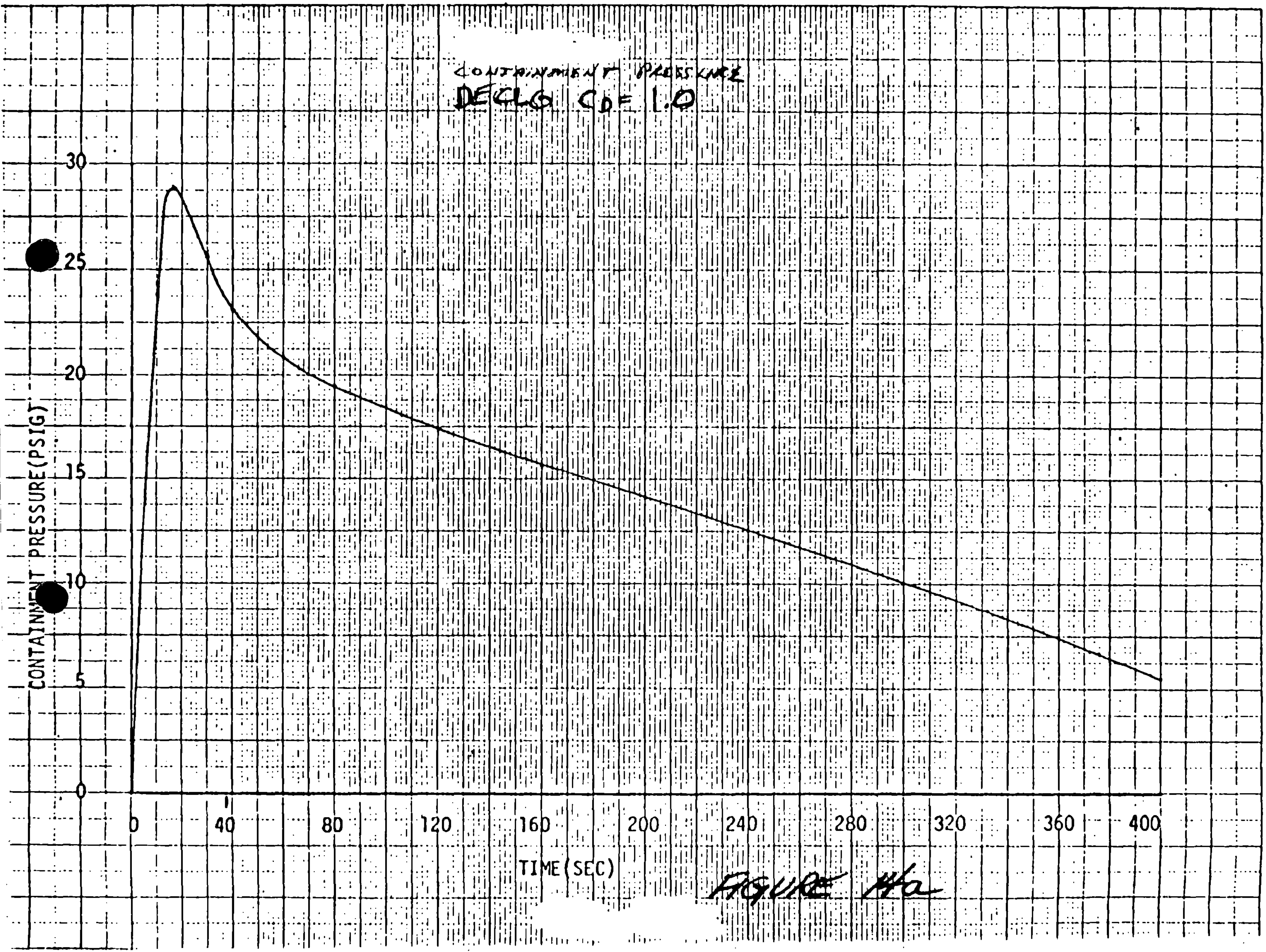


FIGURE 14a

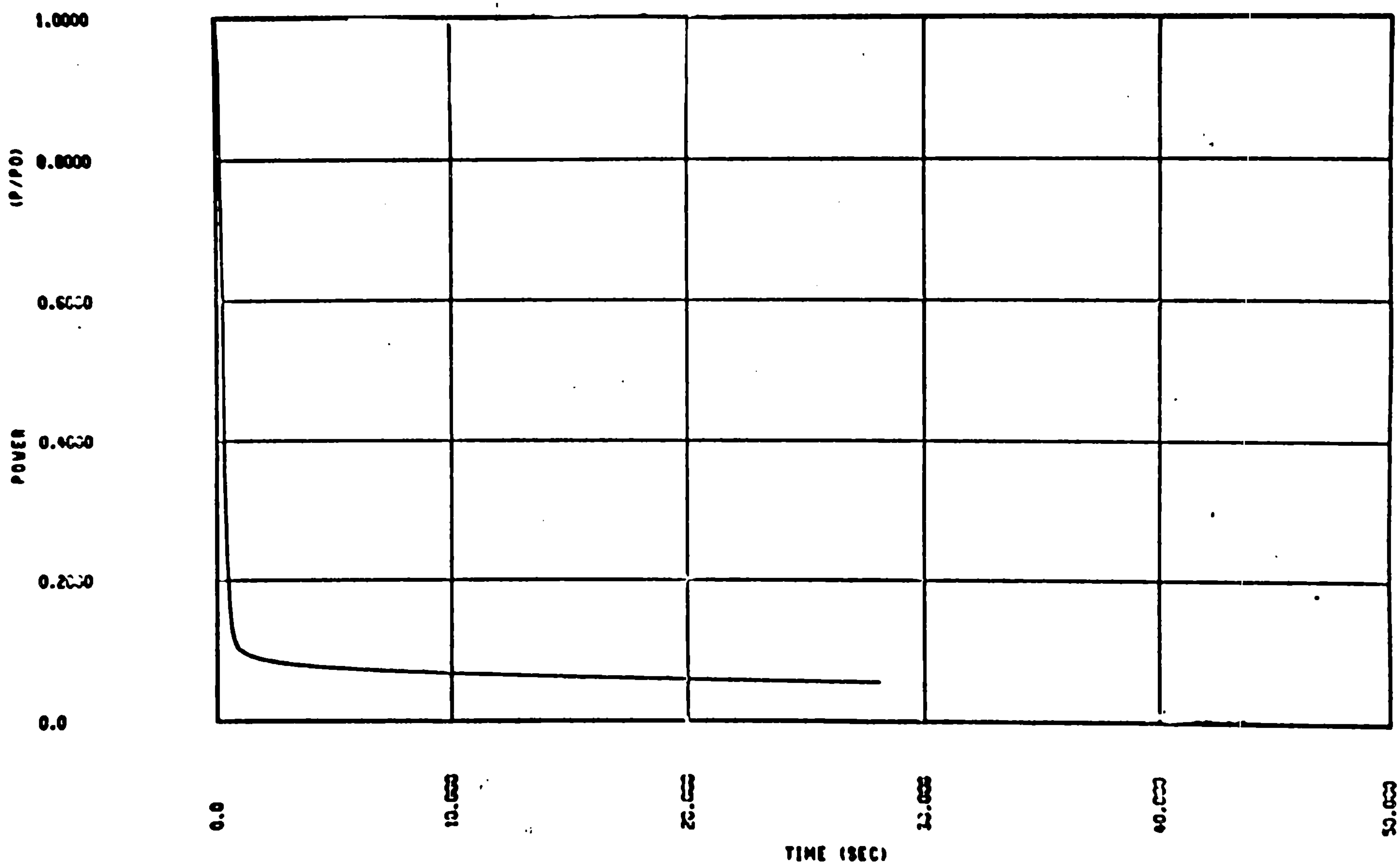


FIGURE 15a
CORE POWER TRANSIENT
DECLG (C_D=1.0)

E4R RAY

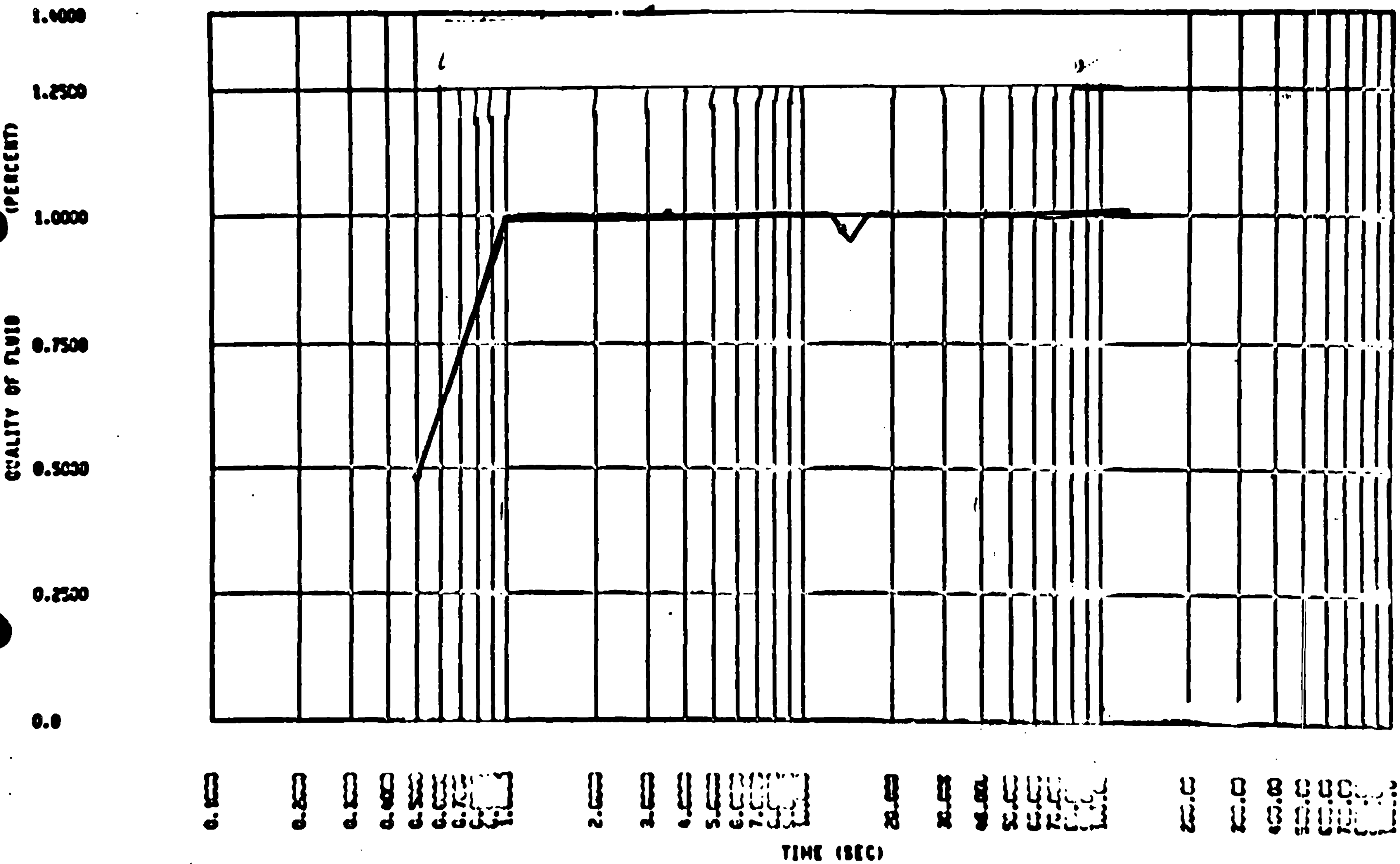


FIGURE 16

FLUID QUALITY - DECLG ($C_D = 0.8$)

E4R RAV

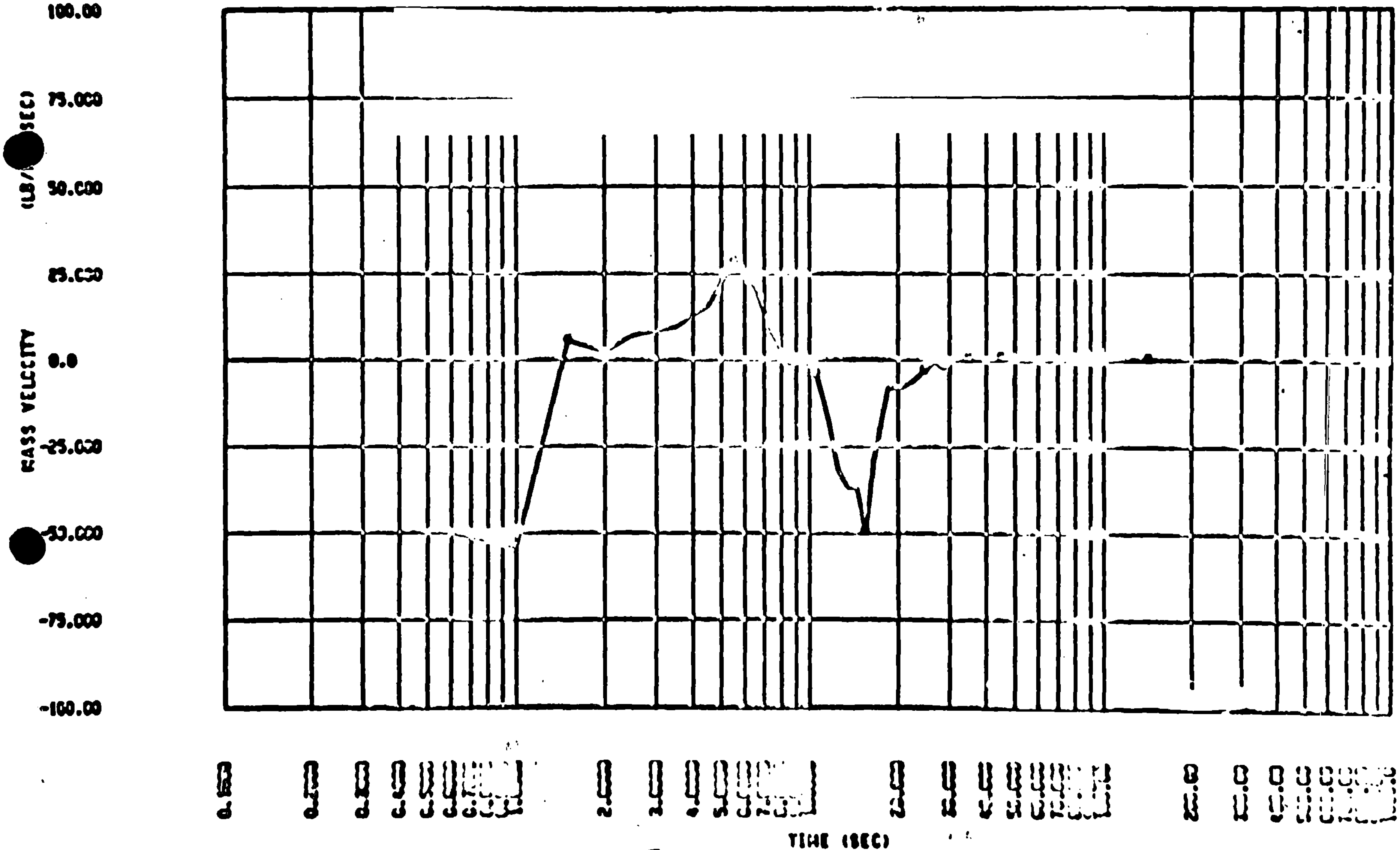


FIGURE 26
MASS VELOCITY DECLG (CD=0.8)

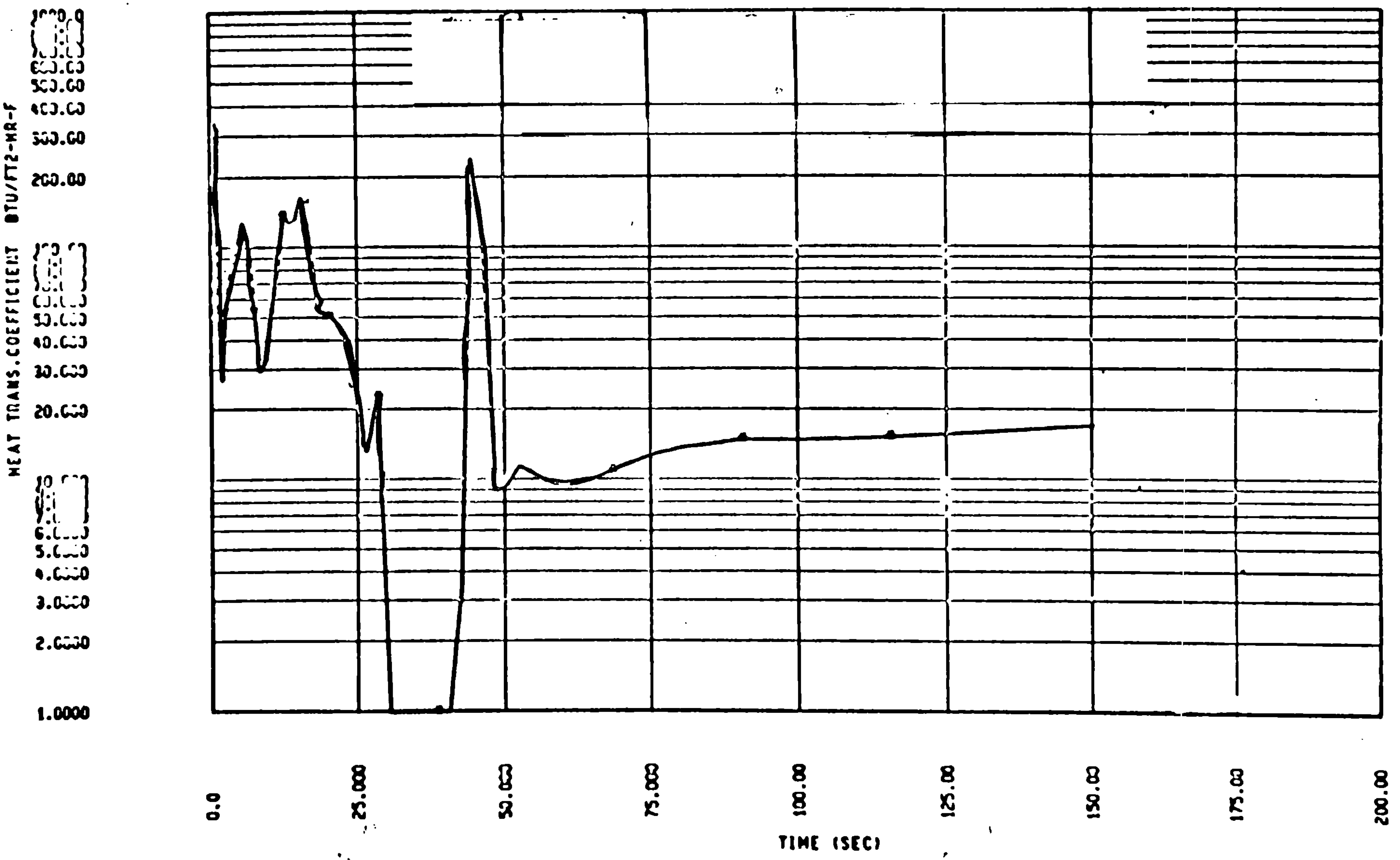


FIGURE 36

EVR RAV

HEAT TRANS COEFFICIENT
DECLG ($C_D = 0.8$)

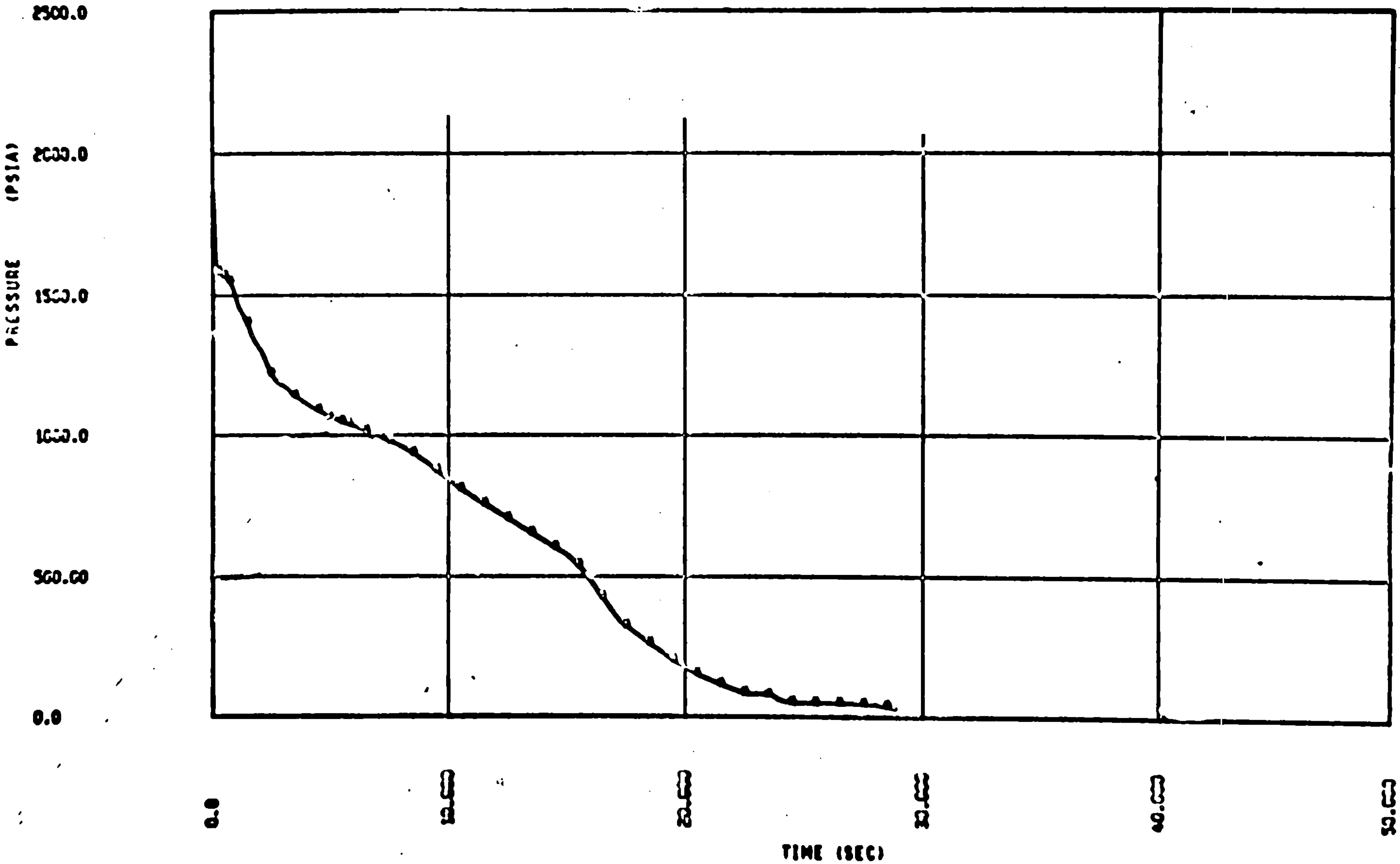


FIGURE 46

CORE PRESSURE DECLG (C_D = 0.8)

E4R RAY

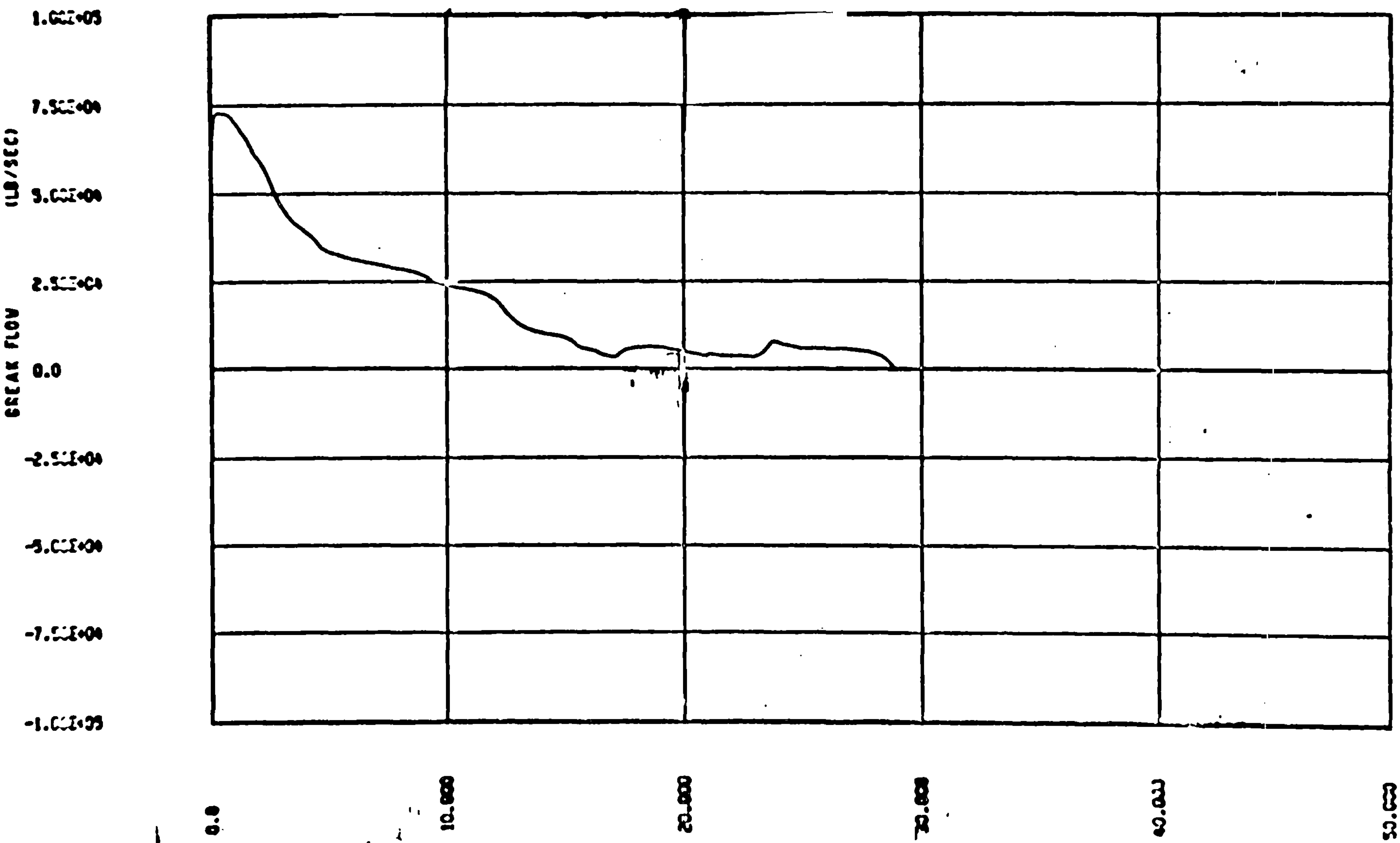


FIGURE 56

BREAK FLOW RATE DECLG (CD = 0.8)

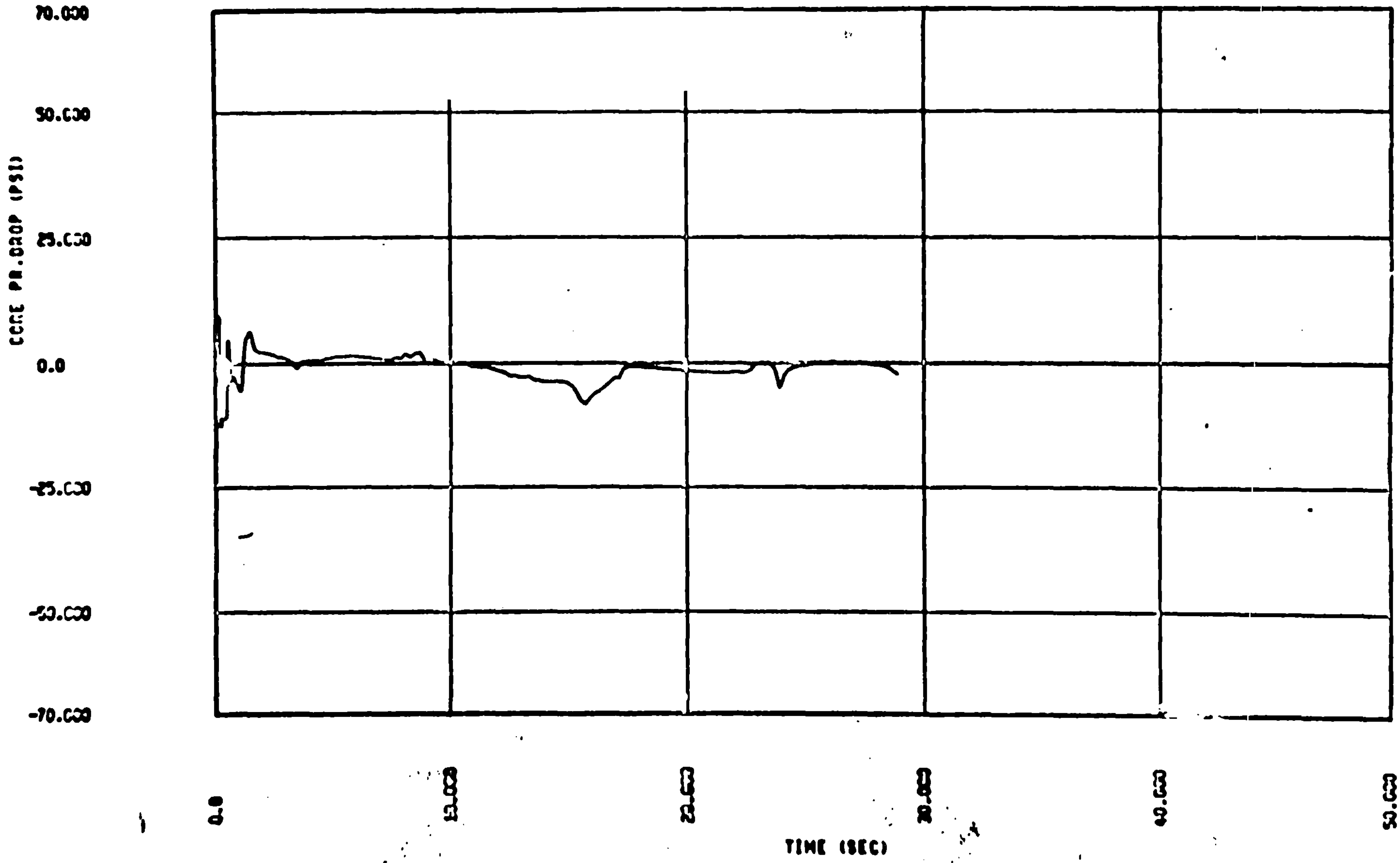


FIGURE 6b

CORE PRESSURE DROP - DECLG ($C_D = 0.8$)

E4R · RAY

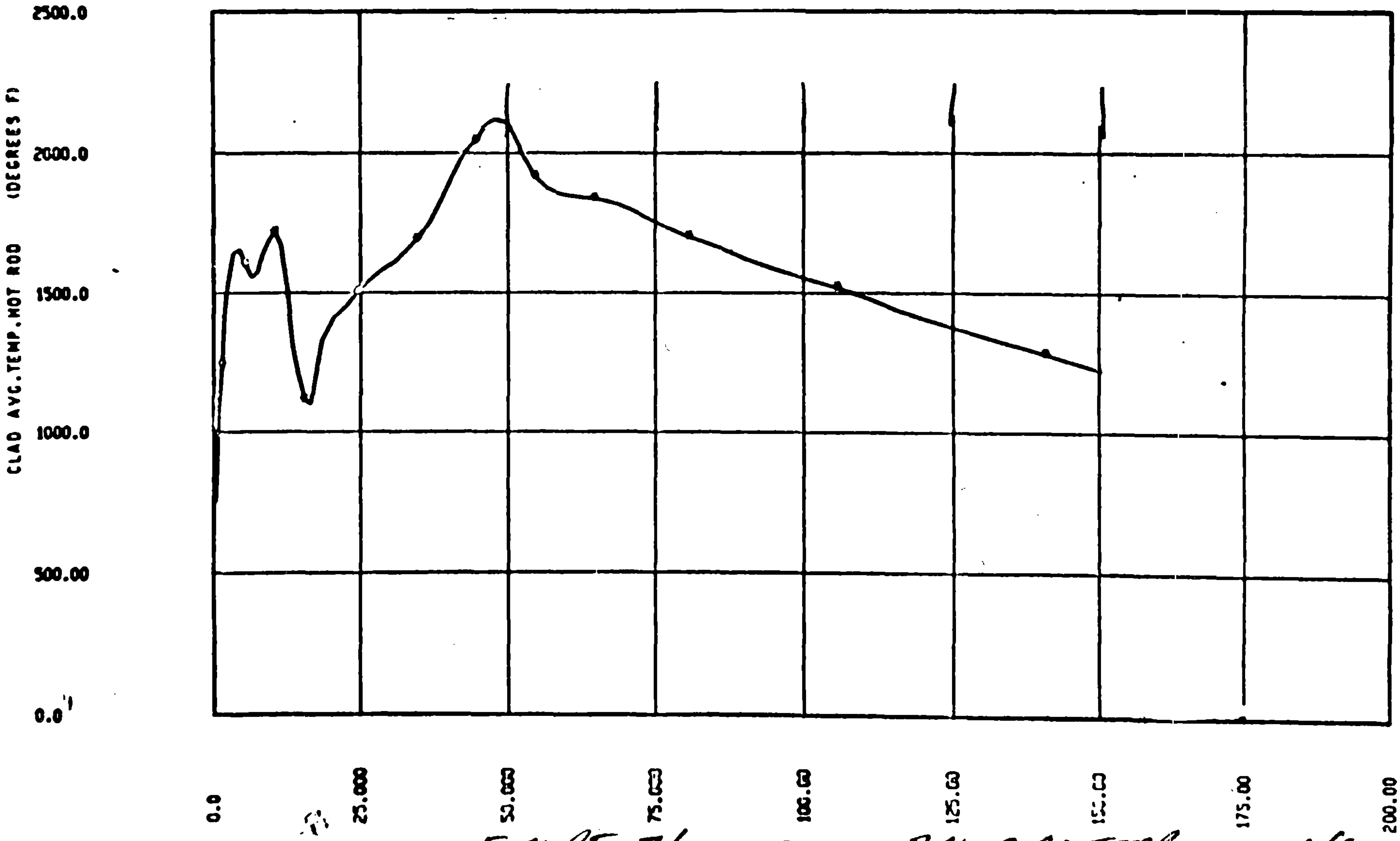


FIGURE 7b PEAK CLAD TEMP - DECLG (CD=0

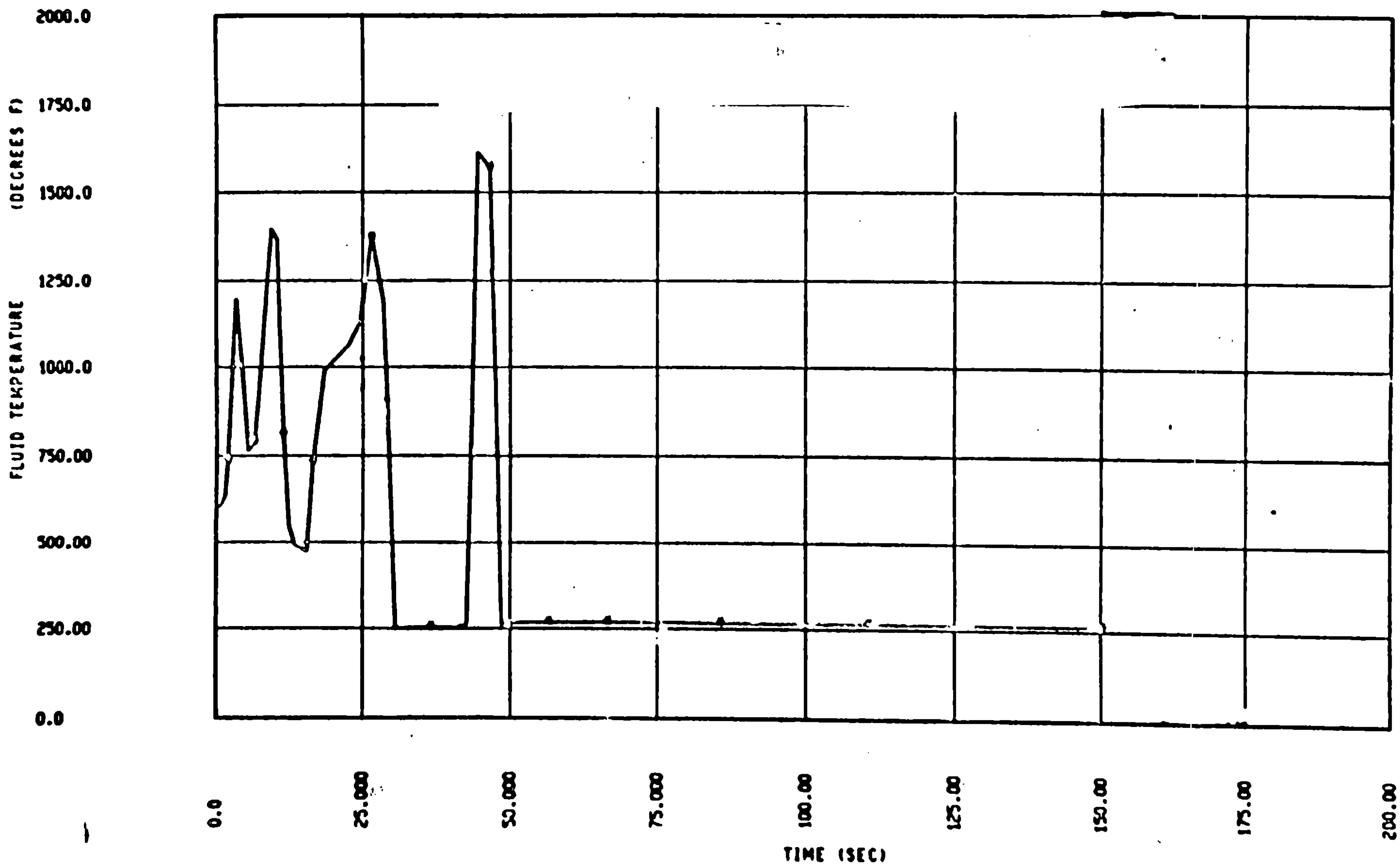


FIGURE 8b

FLUID TEMP – JECLG $(C_D = 0.8)$

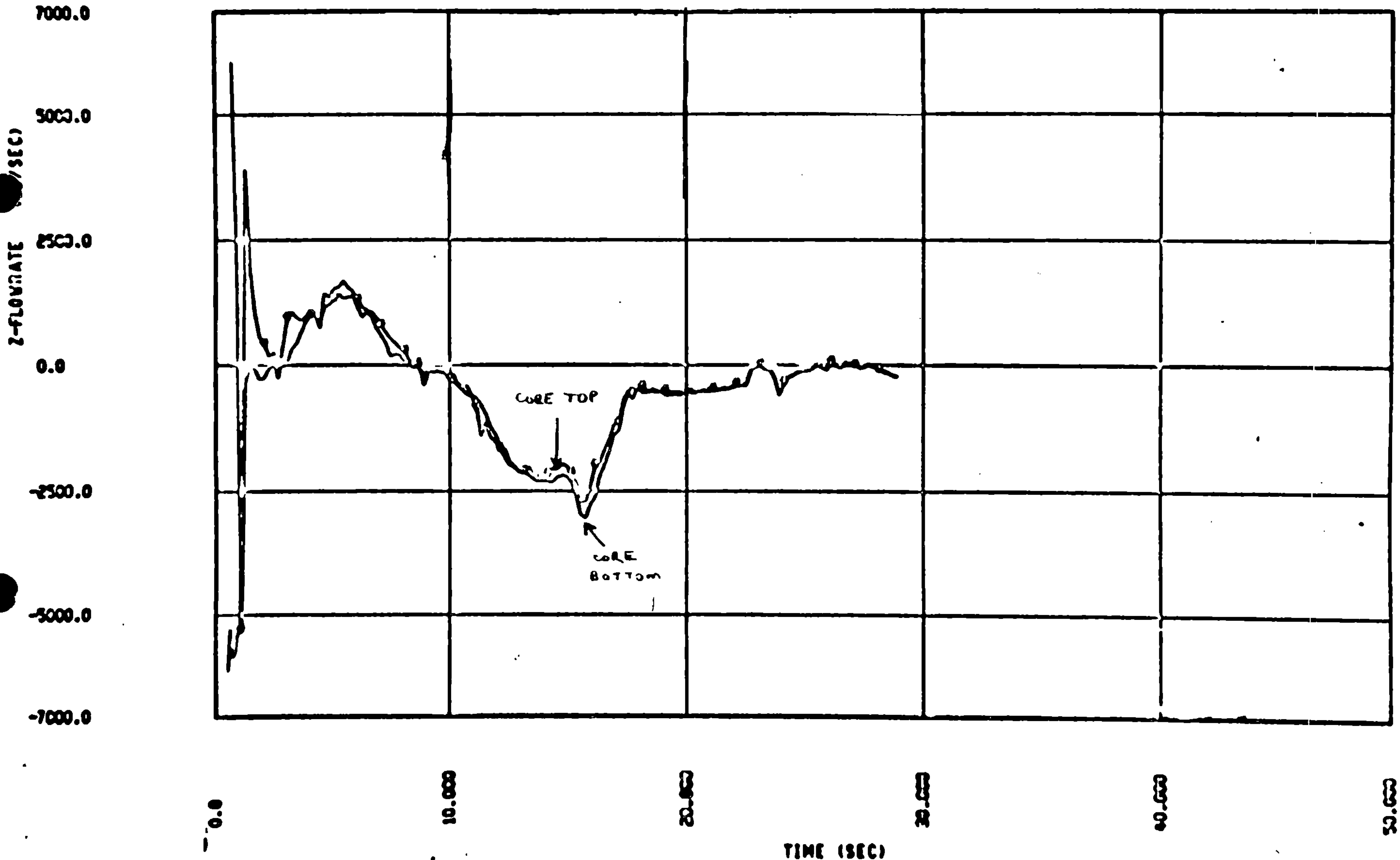


FIGURE 9b

CORE FLOWRATE - TOP AND BOTTOM DECLG ($C_D = 0.8$)

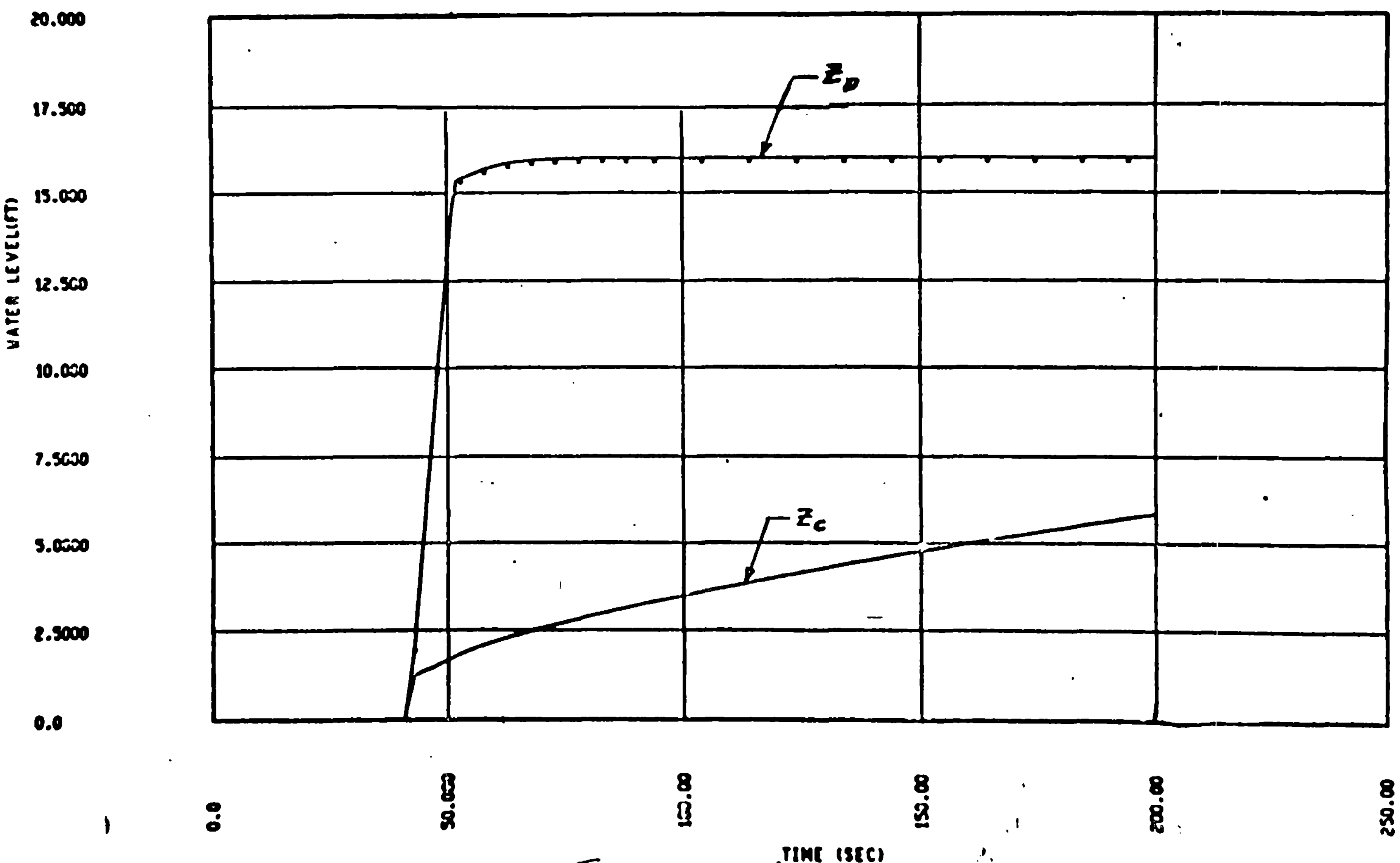


FIGURE 10b

REFLOOD TRANSIENT DECLG ($C_D = 0.8$)

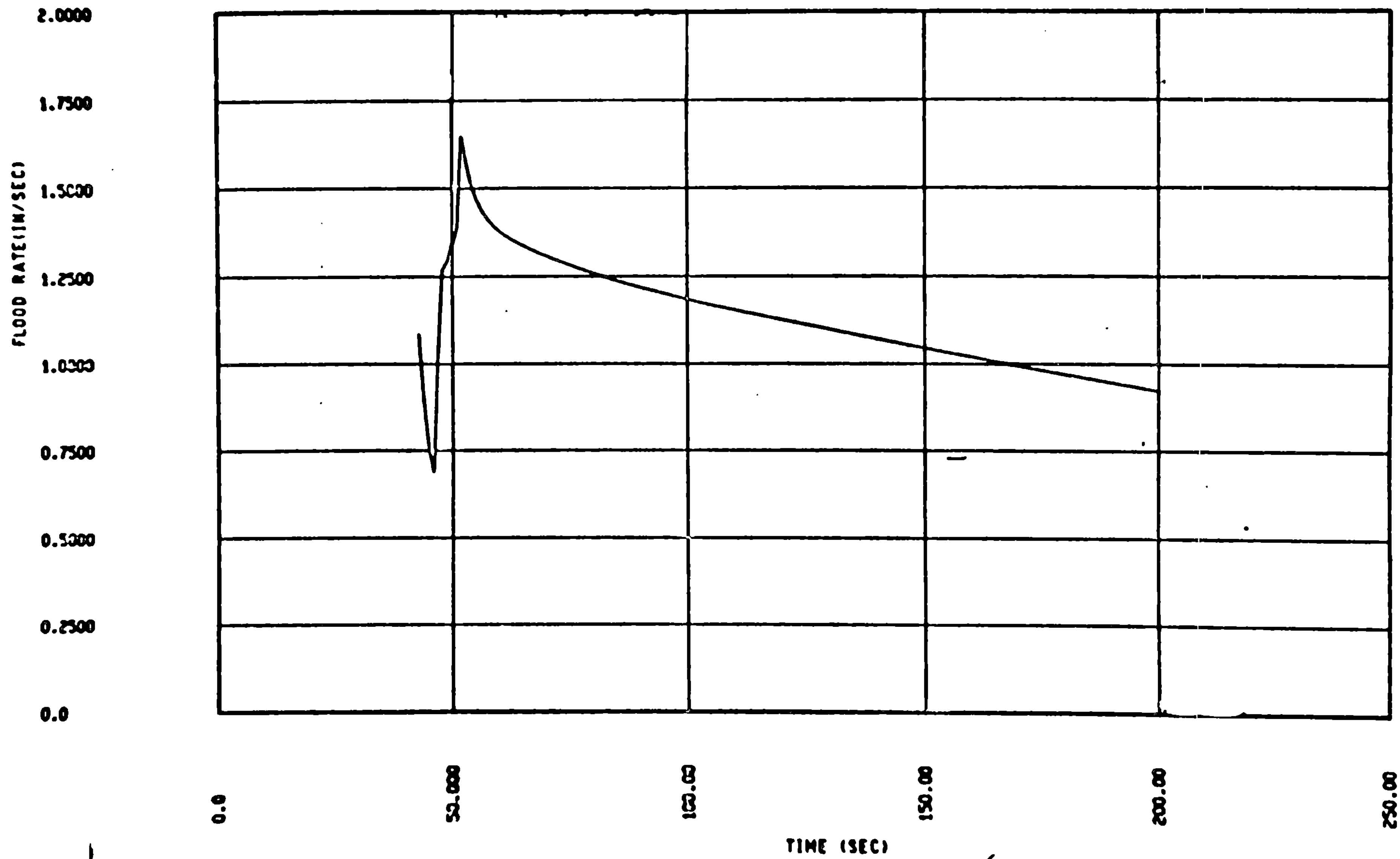


FIGURE 11 b
REFLOOD TRANSIENT DECLG (C_D = 0.8)
DOWNCOMER AND CORE WATER LEVEL

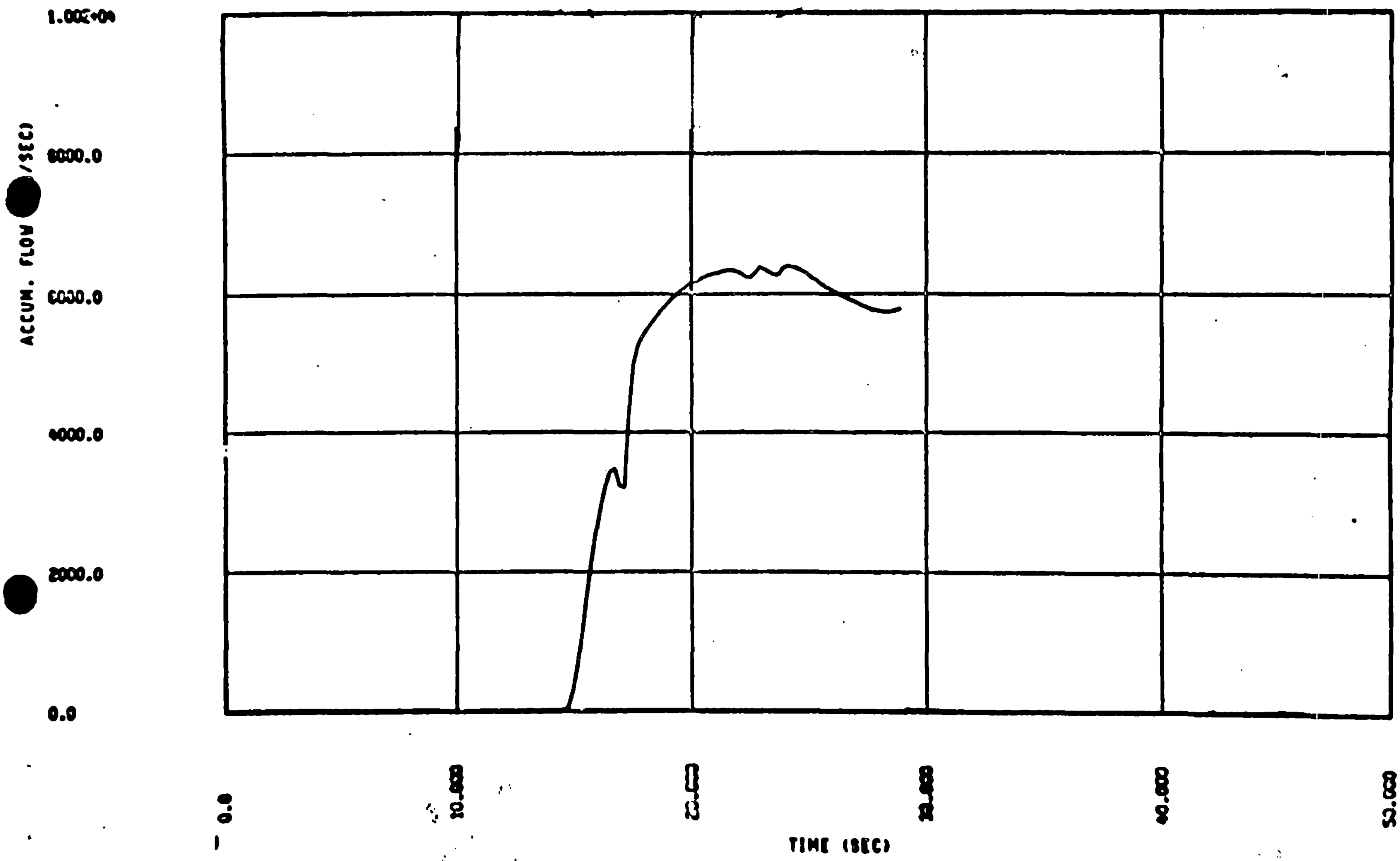


FIGURE 12b
ACCUMULATOR FLOW (BLOWDOWN)
DECLG (C_D = 0.8)

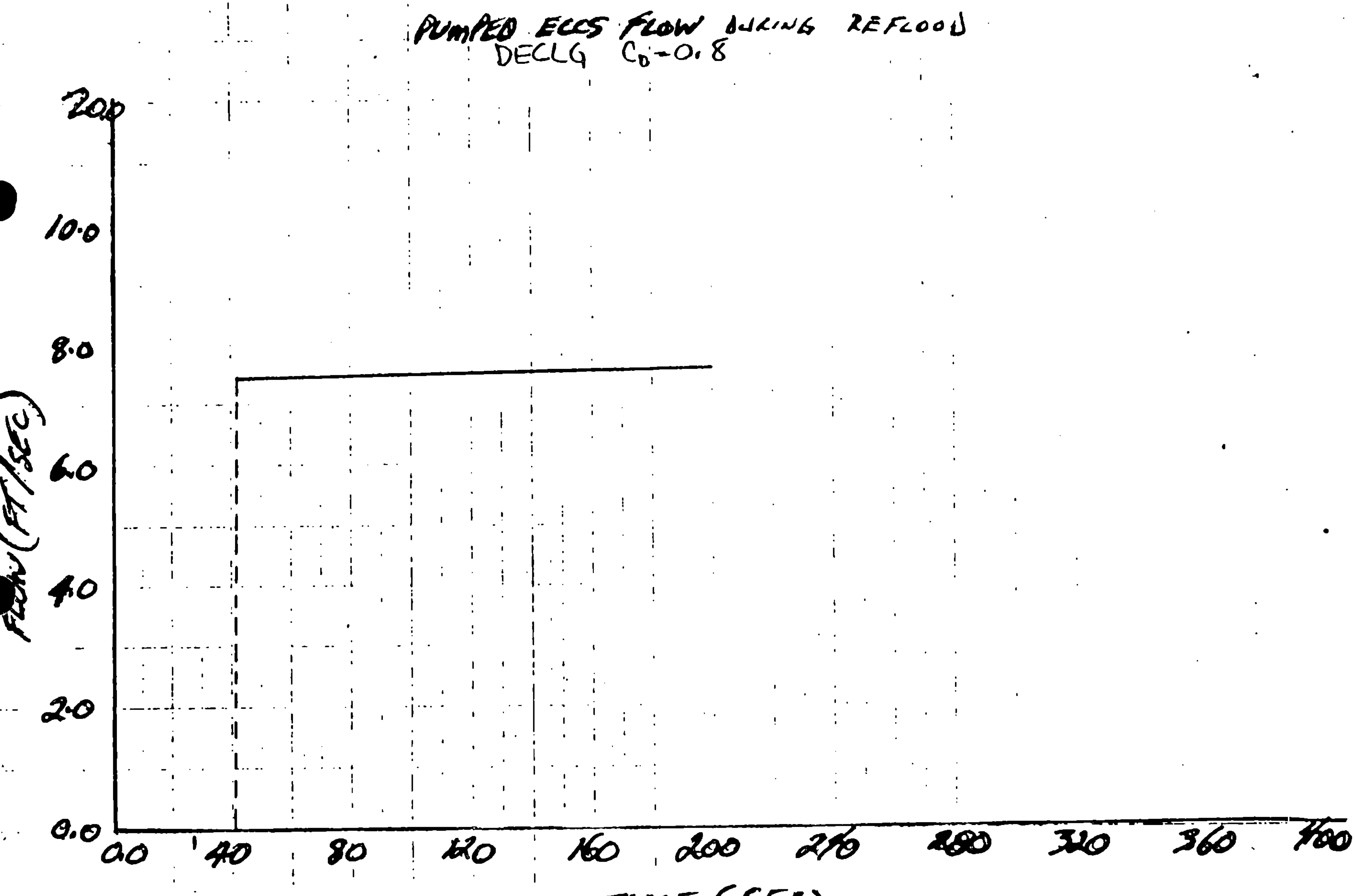

PUMPED ECCS FLOW DURING REFLOOD
DECLG C_D-0.8
20.0
10.0
8.0
6.0
4.0
2.0
0.0
FLOW (FT/SEC)
0.0
40
80
120
160
200
240
280
320
360
400
TIME (SEC)

FIGURE 136

K•E 10 X 10 TO THE CENTIMETER 18 X 25 CM. KEUFFEL & ESSER CO. MADE IN U.S.A. 46 1510

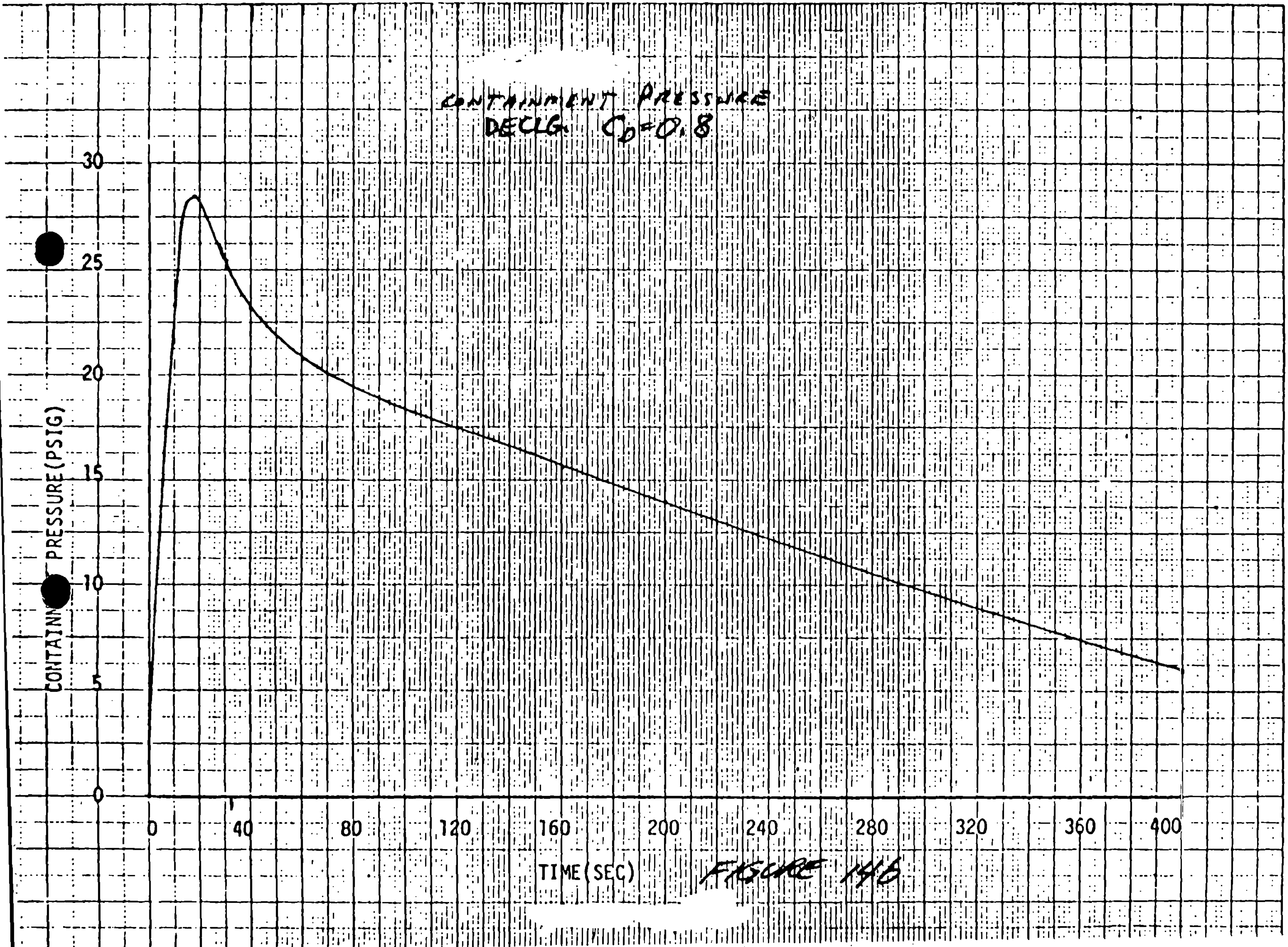


FIGURE 14b

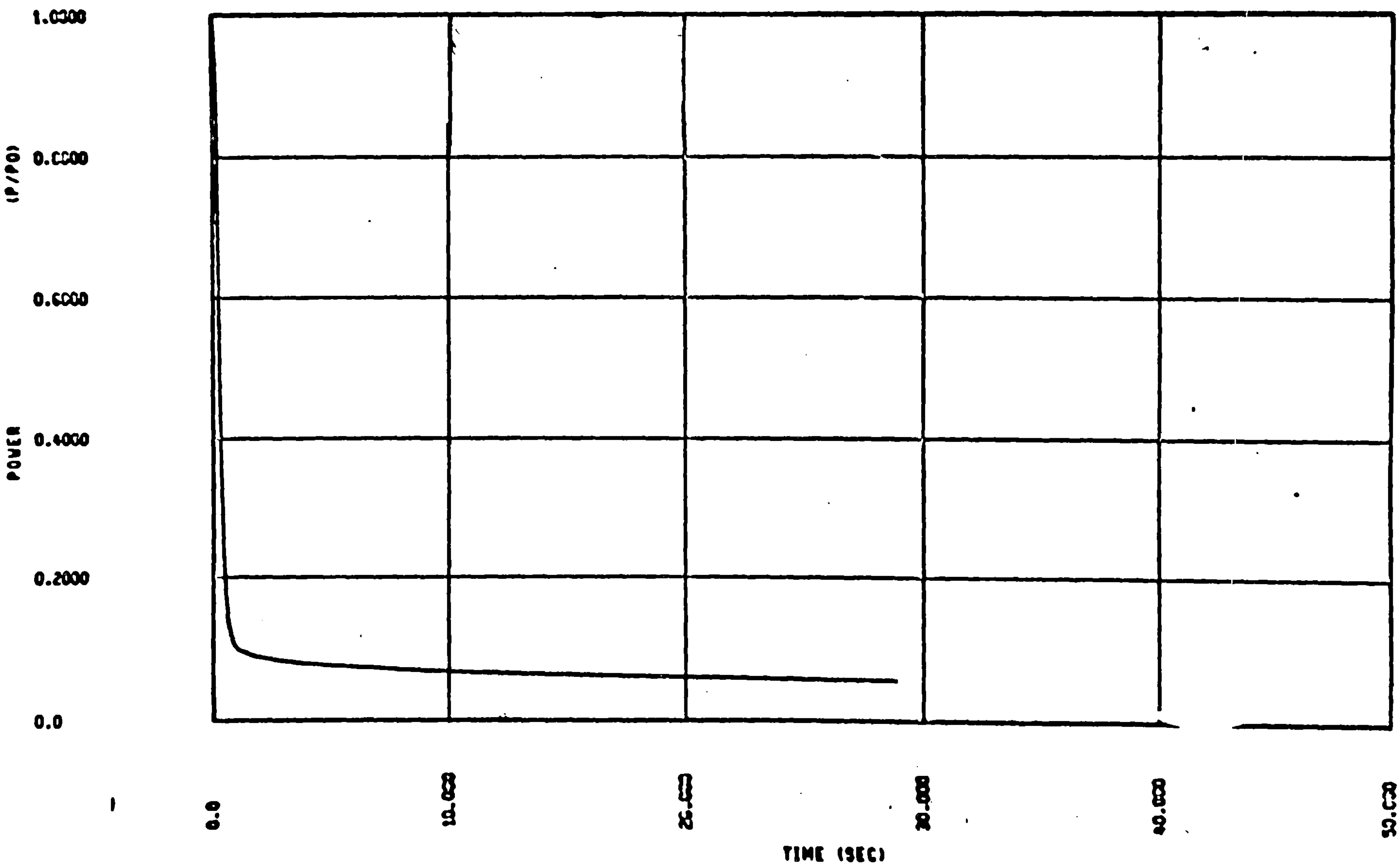


FIGURE 15b

CORE POWER TRANSIENT DECLG (C_D = 0.8)

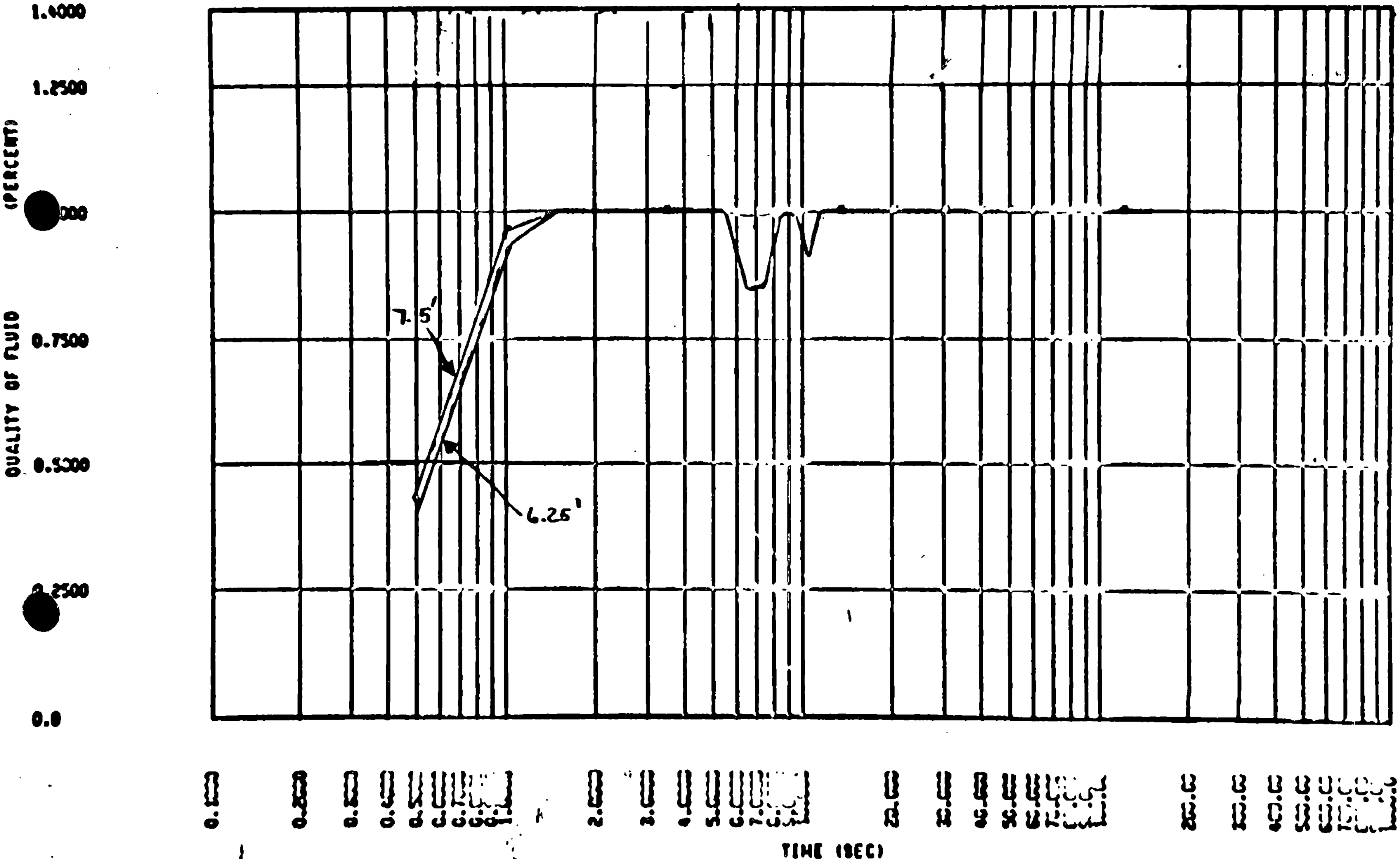


FIGURE 1c
FLUID QUALITY - DECLG (CD=0.6)

E4R RAV

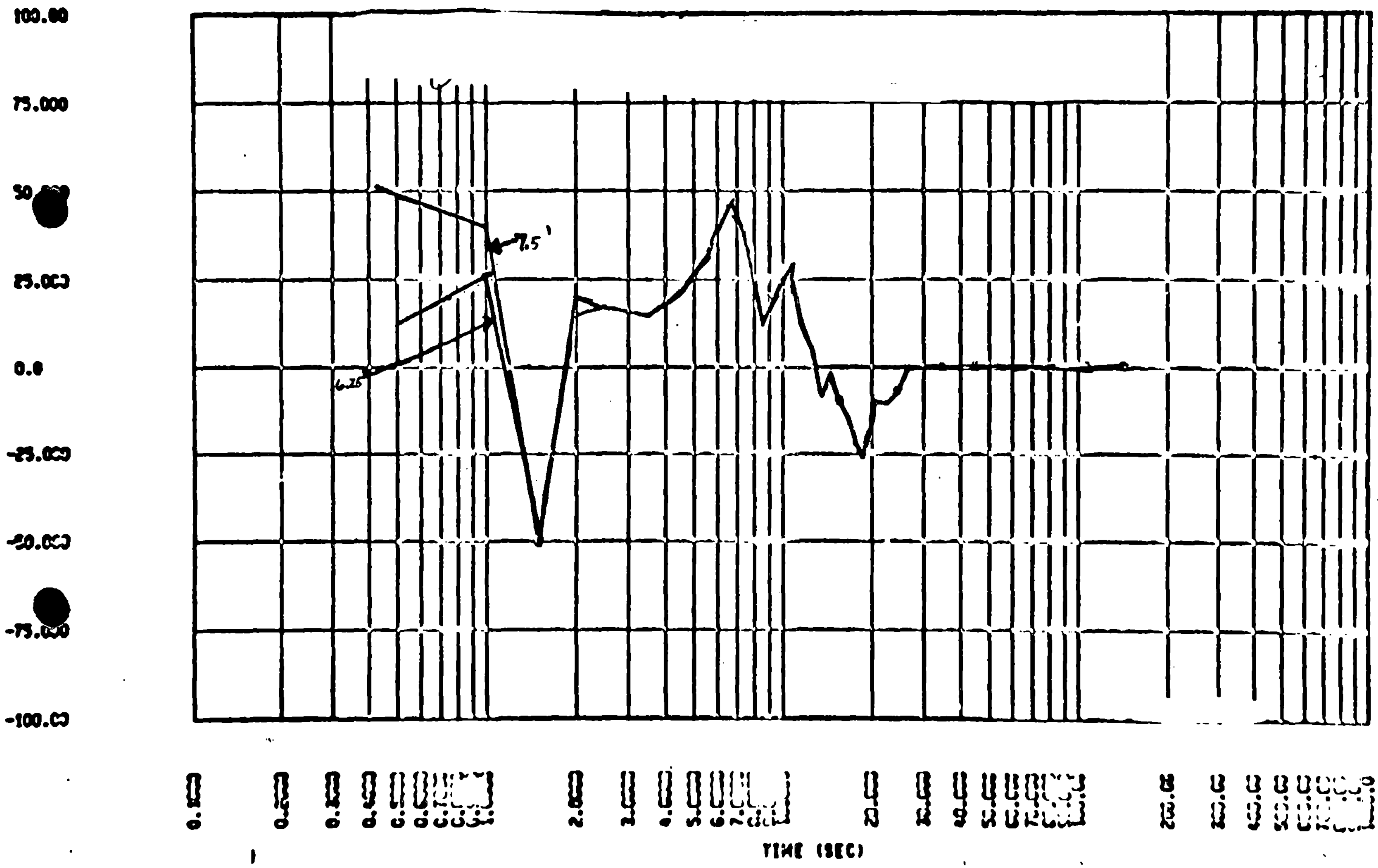


FIGURE 2c
MASS VELOCITY DECLG (C_D=0.6)

RAV -

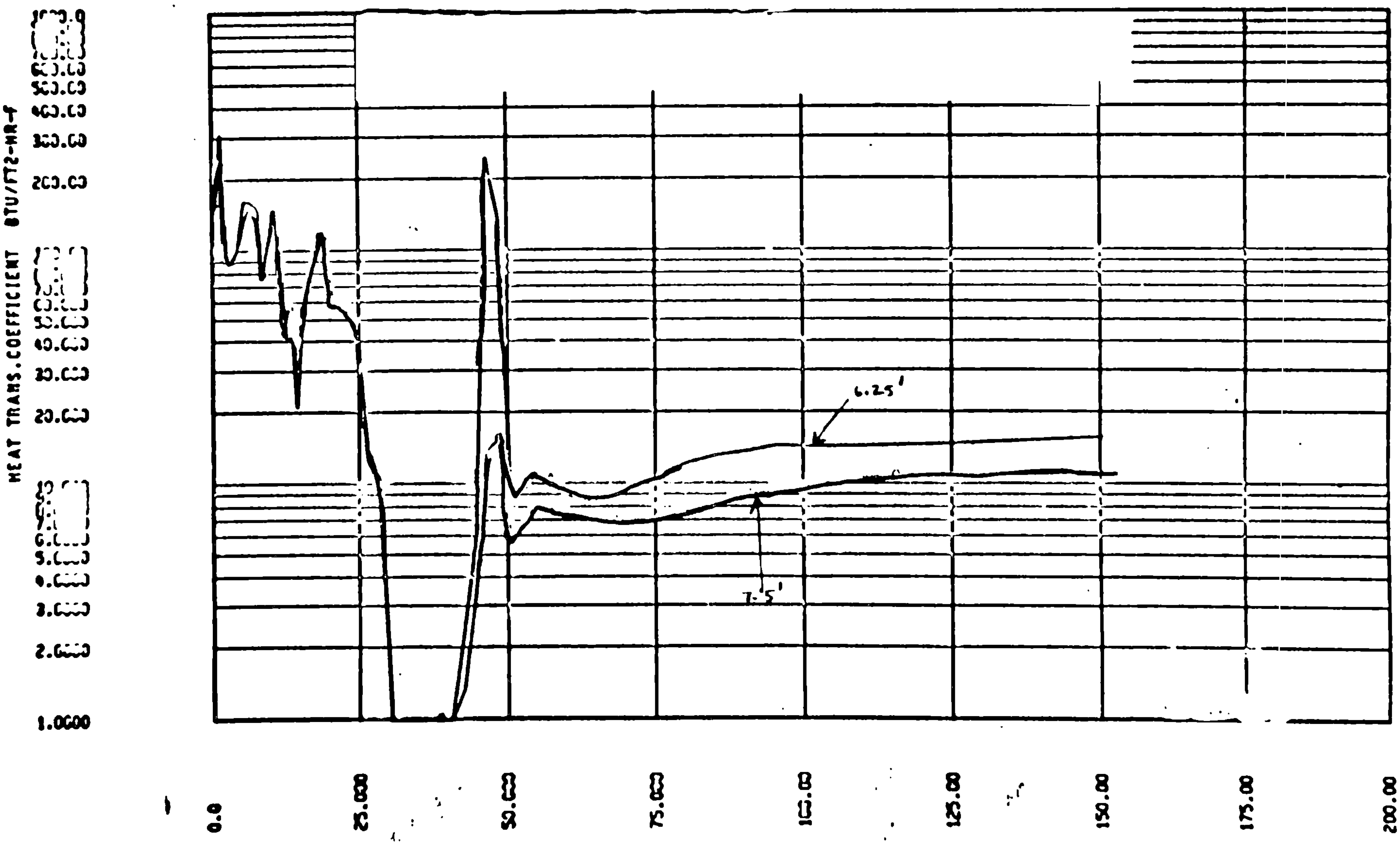


FIGURE 3C
HEAT TRANS COEFFICIENT
DECLG (CD=0.6)

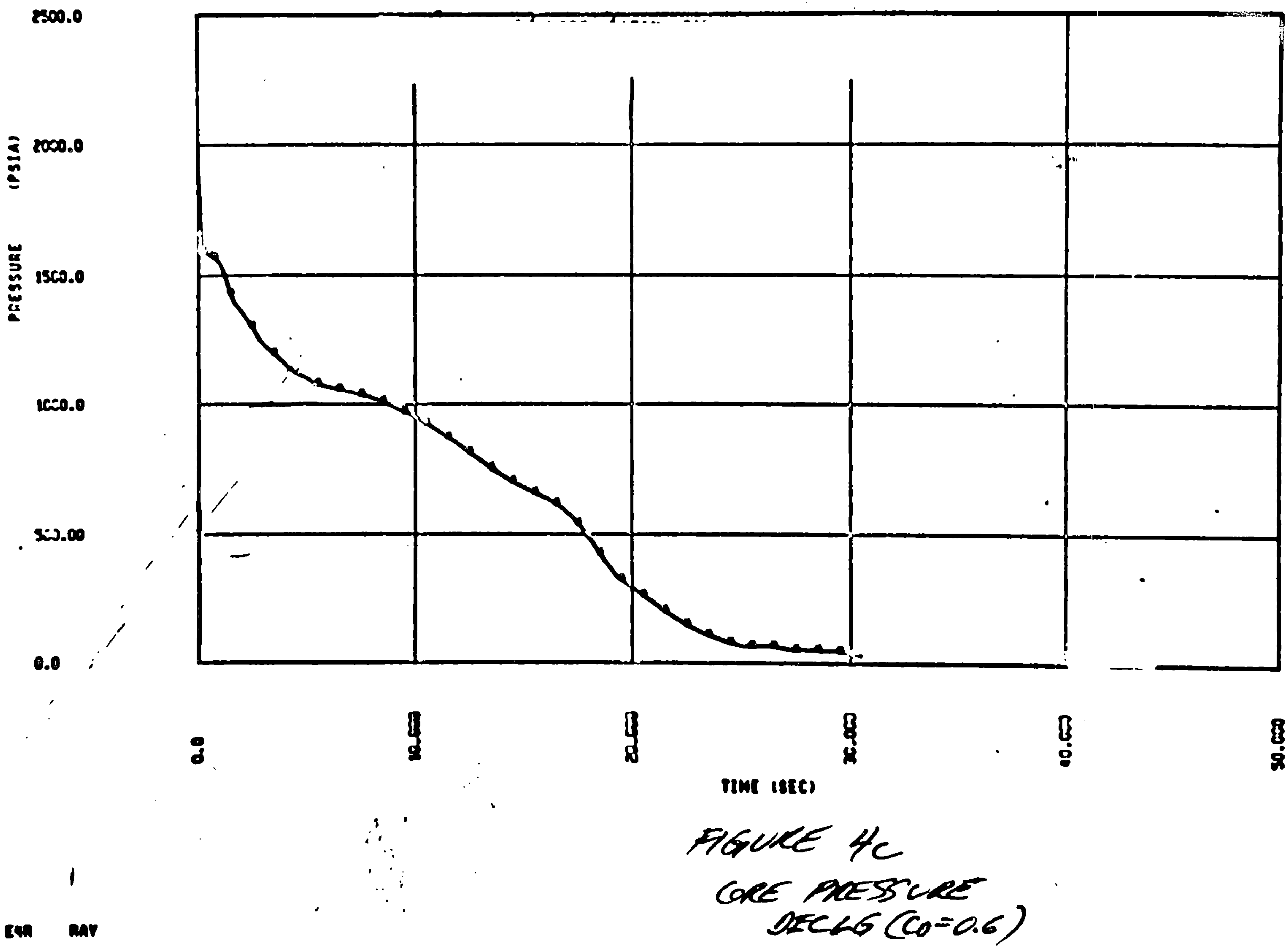


FIGURE 4c
CORE PRESSURE
DECLG (Co=0.6)

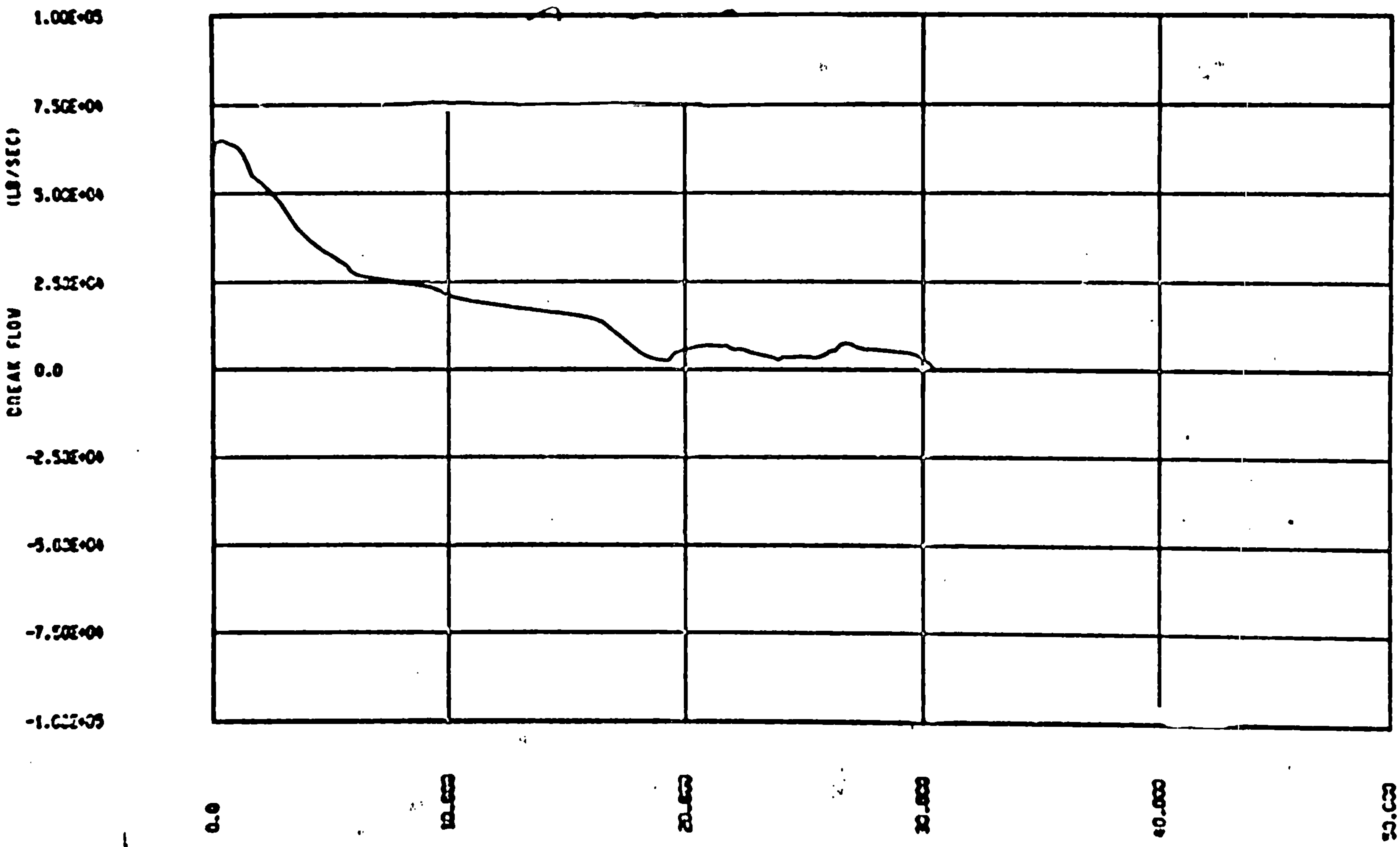


FIGURE 5c
BREAK FLOW RATE DECLG (CD=0.6)

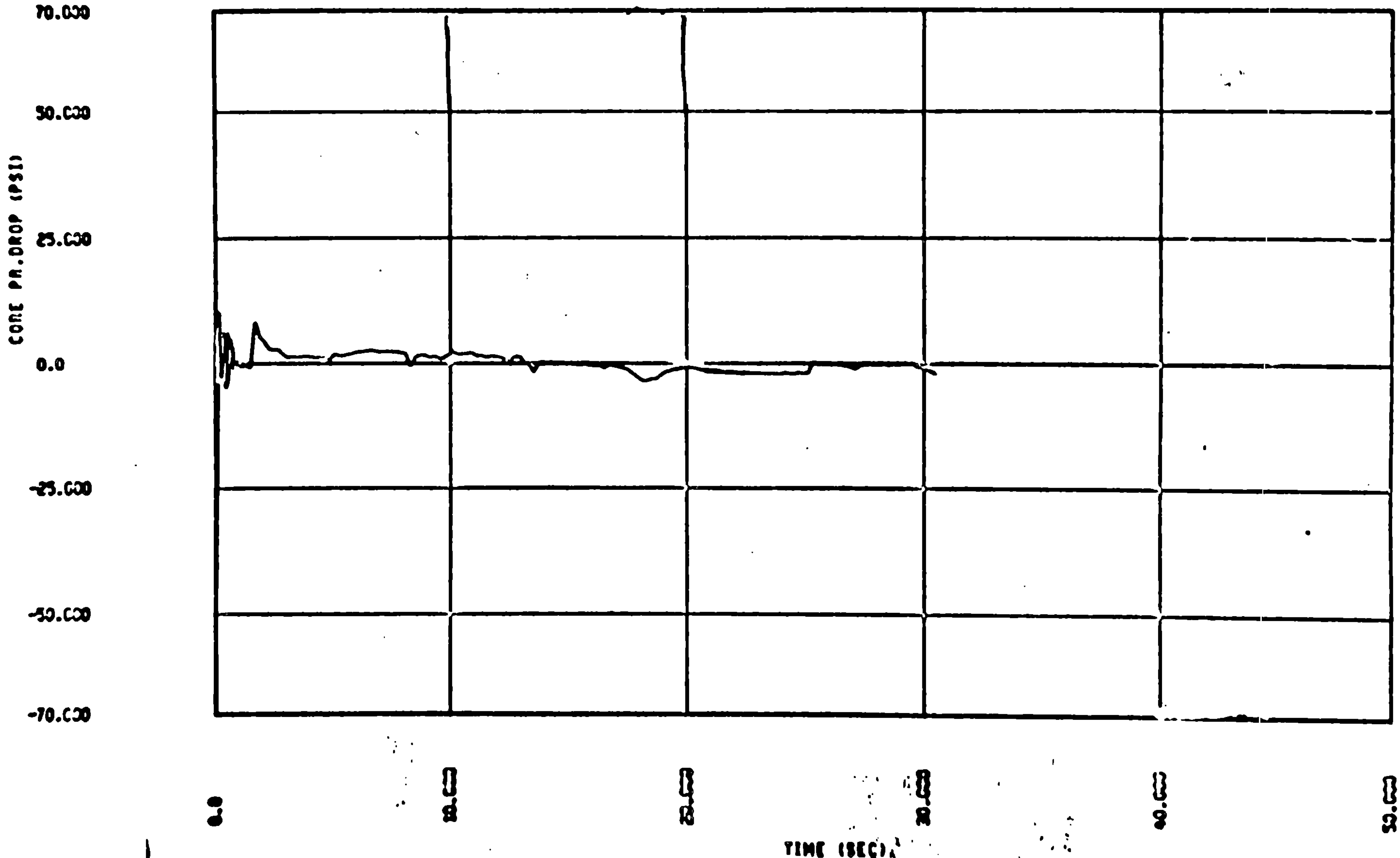


FIGURE 6c
CORE PRESSURE DROP DECLG (CD=0.6)

E4R RAY

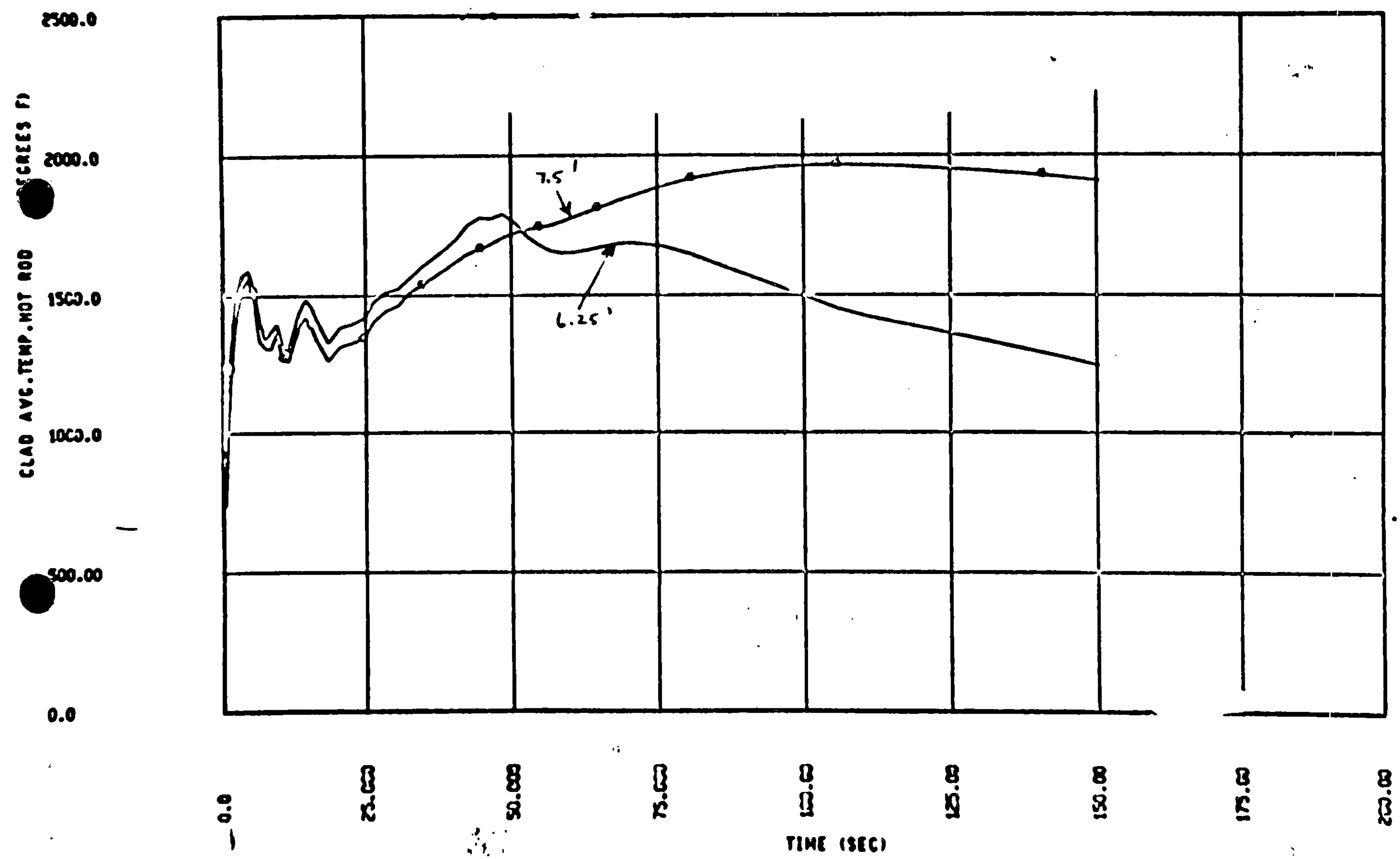


FIGURE 7C
~~PEAK~~ CLAD TEMPERTURE DECLG (C_D=0.6)

ERR RAY

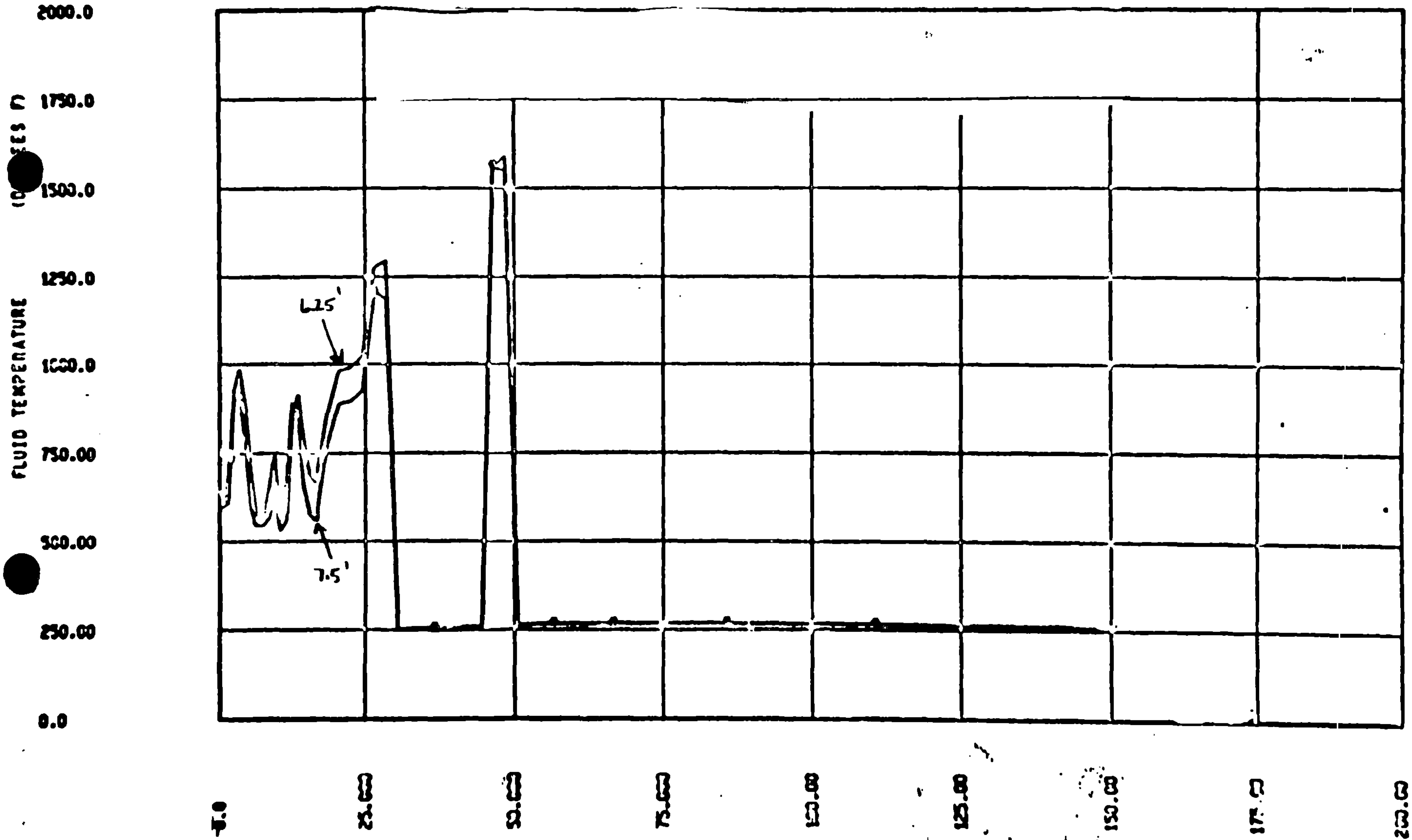


FIGURE 8C
FLUID TEMP. LBCLG (CD=0.6)

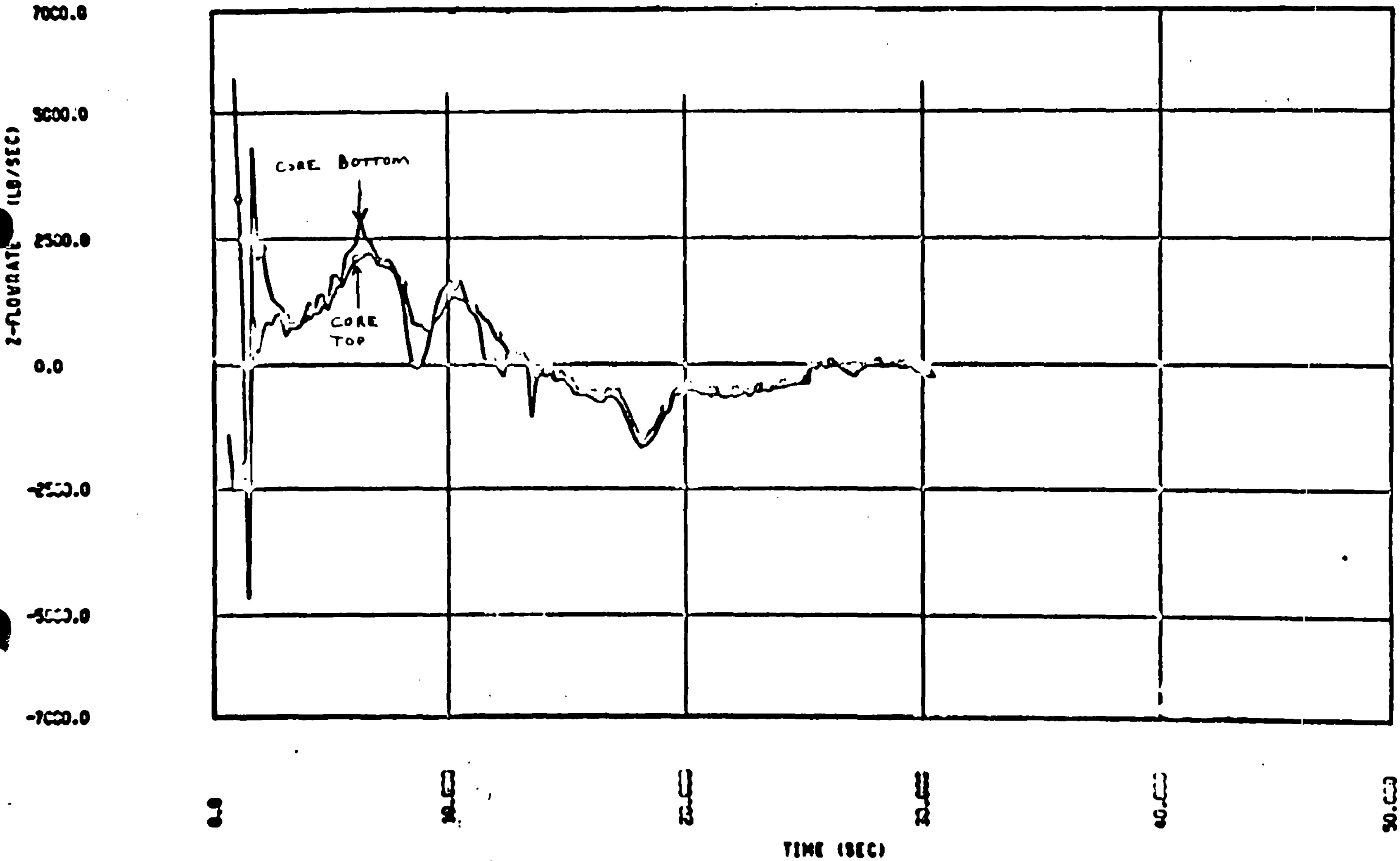


FIGURE 9L
CORE FLOWRATE - TOP AND BOTTOM
DECLG (CD = 0.6)

EAR RAY

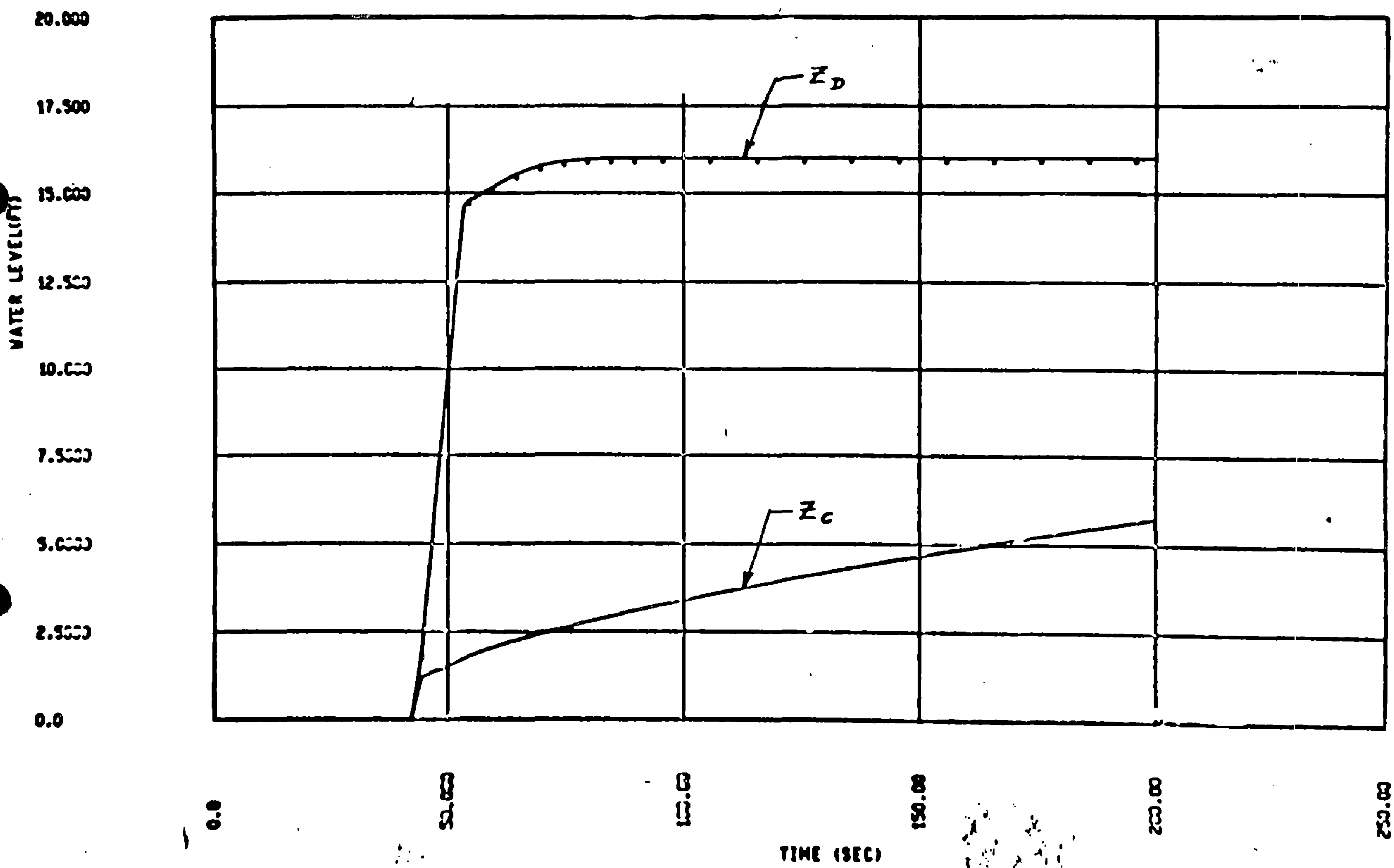


FIGURE 10C
REFLOOD TRANSIENT DOWNCOMER
AND CORE WATER LEVEL DECLG (L_D=0.6)

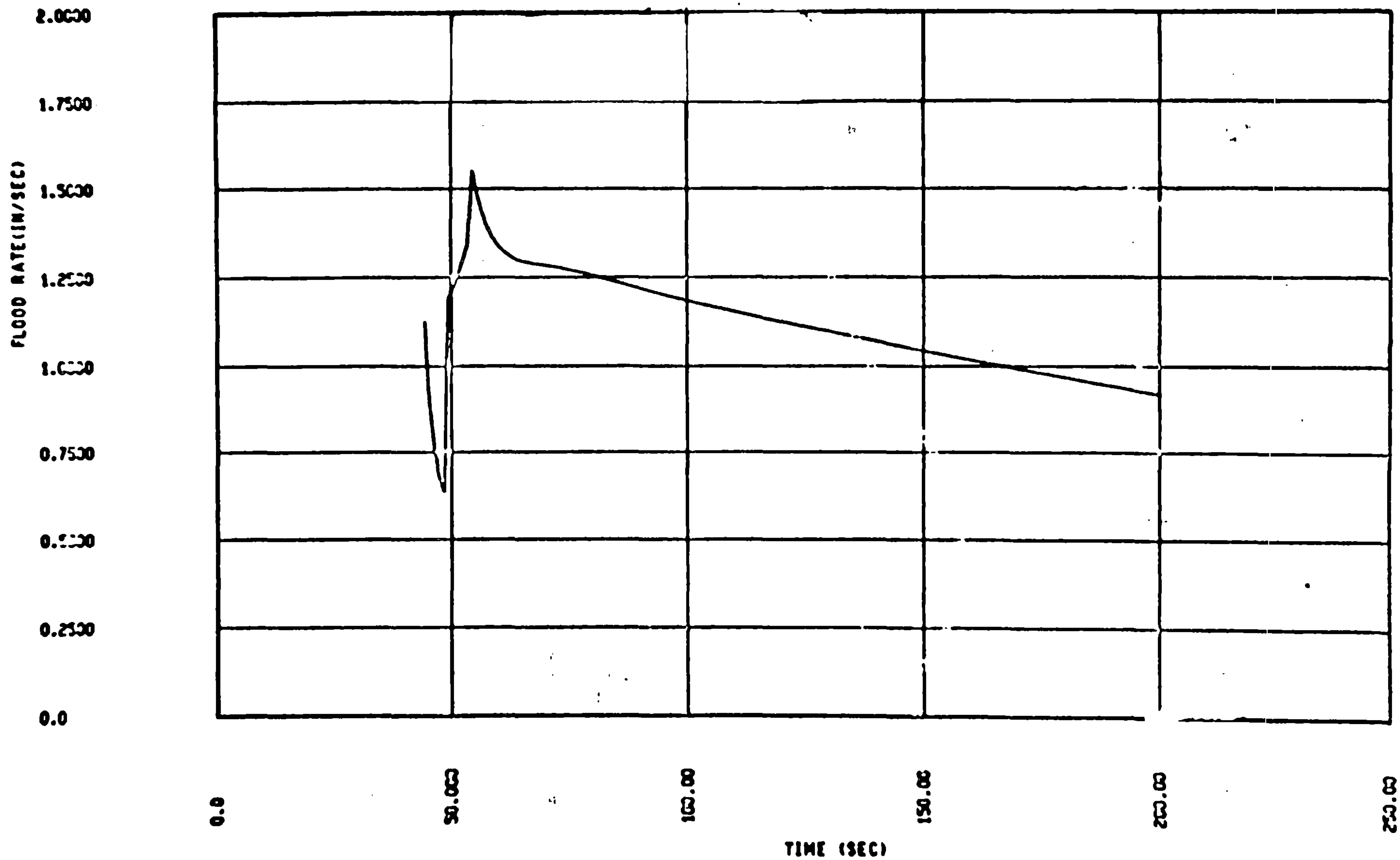


FIGURE 11C

E4R RAY

REFLOOD TRANSIENT DECLG ($C_D = 0.6$)
CORE INLET VELOCITY

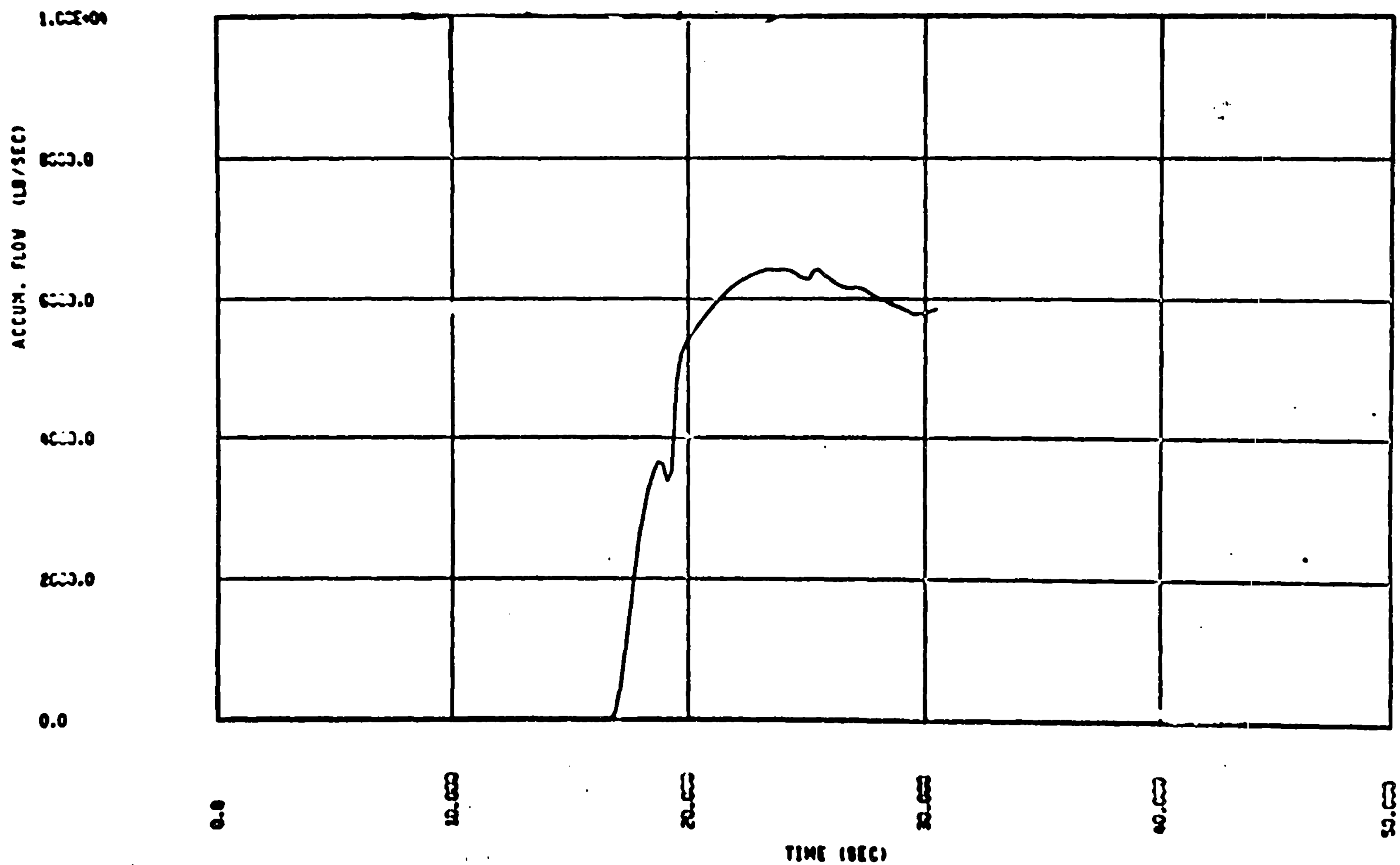


FIGURE 12c

E4R RAV

ACCUMULATOR FLOW (BLOWDOWN) DECLG (C_D = 0.6)

PUMPED ECCS FLOW DURING REFLOOD

DECLG C_D=0.6

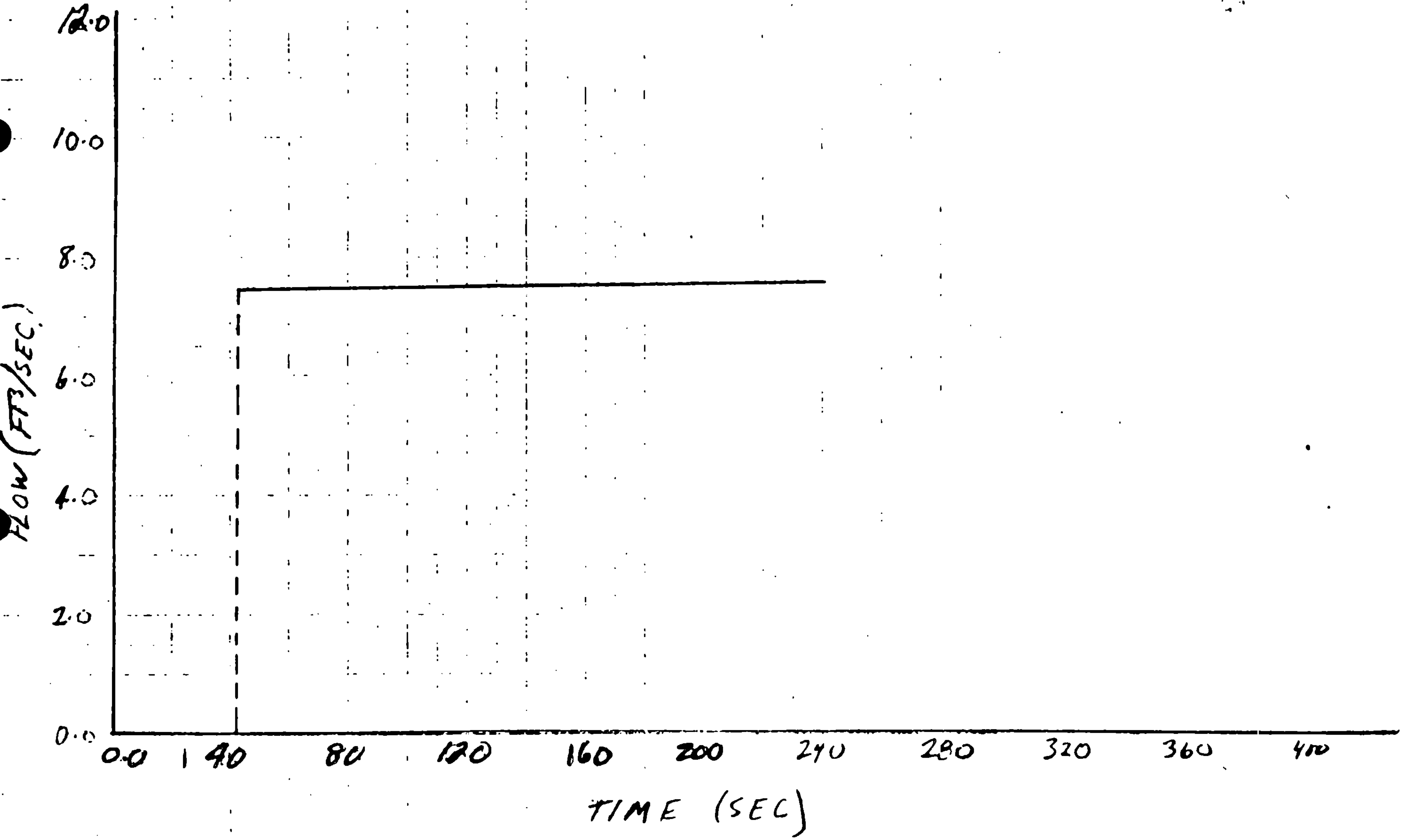


FIGURE 13c

K•E 10 X 10 TO THE CENTIMETER 18 X 25 CM.
KEUFFEL & ESSER CO. MADE IN U.S.A.

46 1510

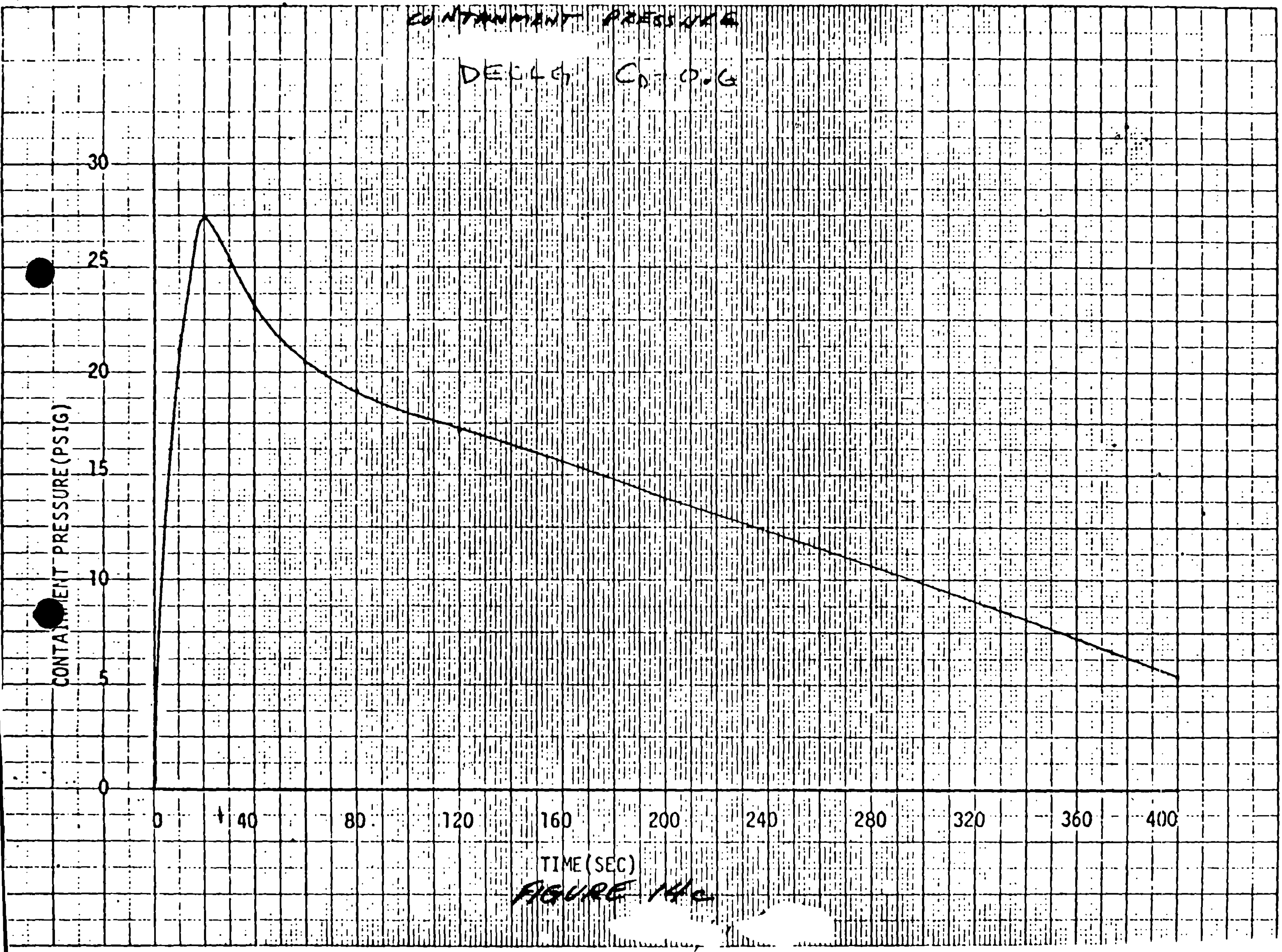


FIGURE 14c

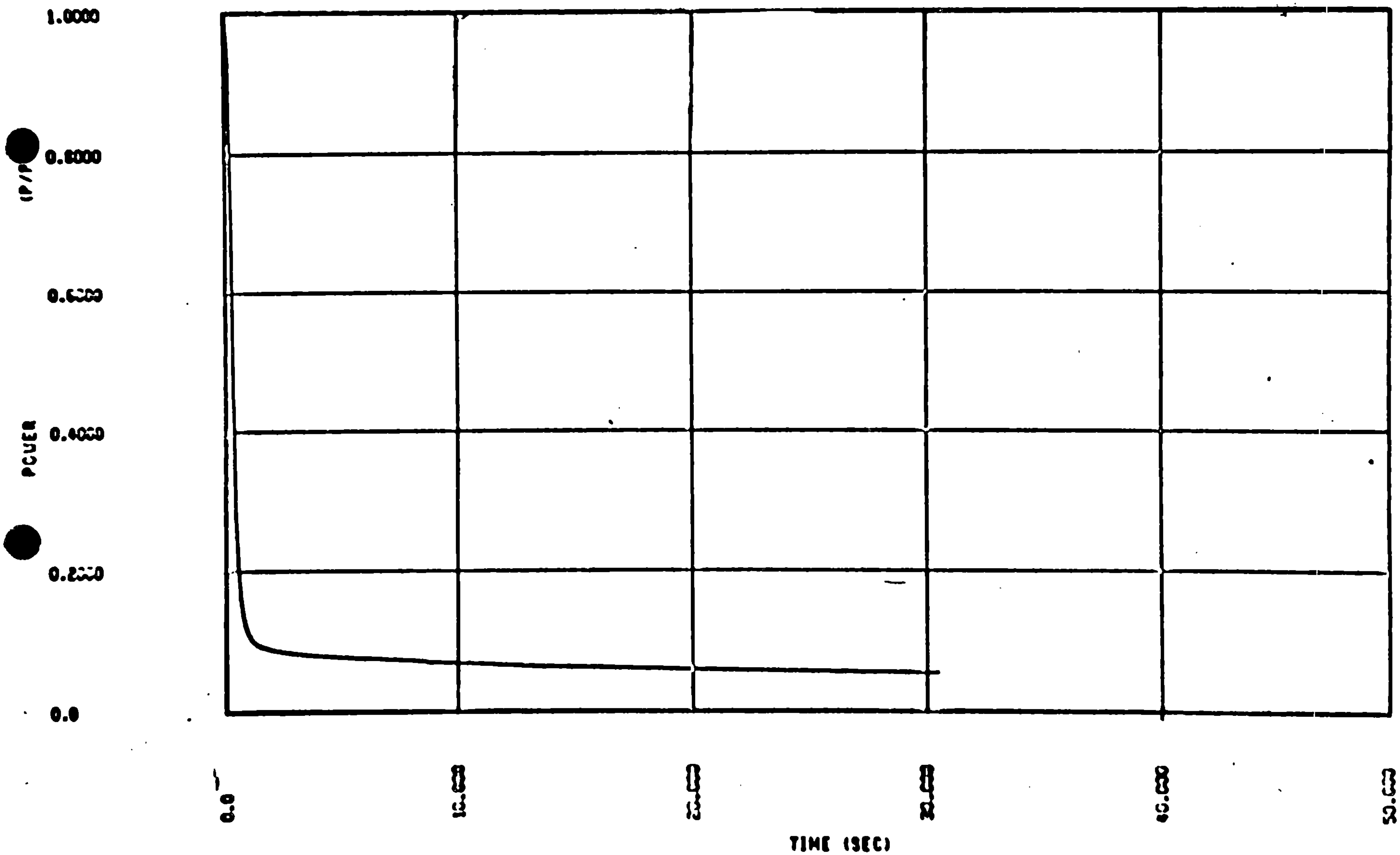


FIGURE 15c
CORE POWER TRANSIENT DECLG (C_D=0.6)

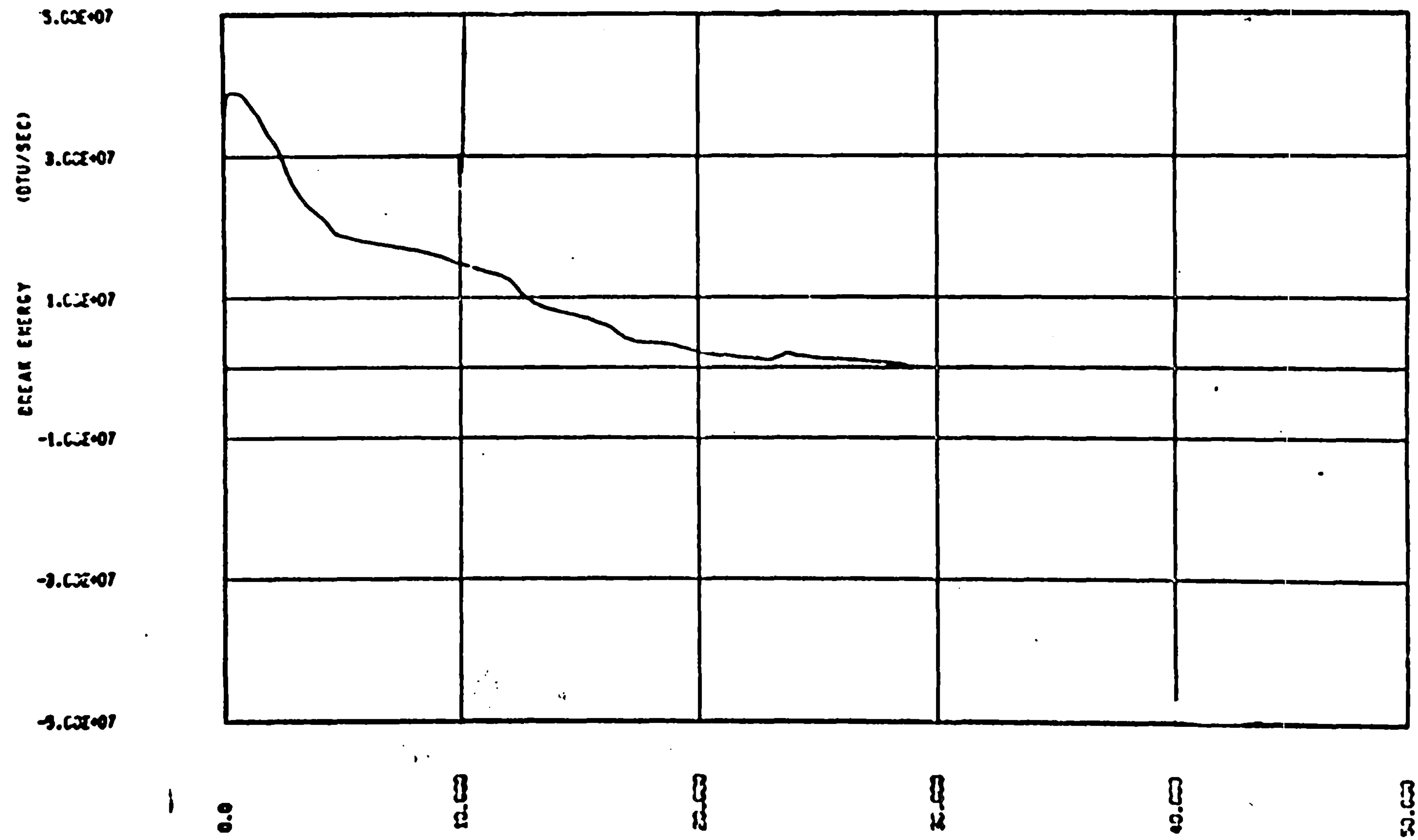


FIGURE 16
BREAK ENERGY RELEASED TO CONTAINMENT DECLG (C_D=0.8)

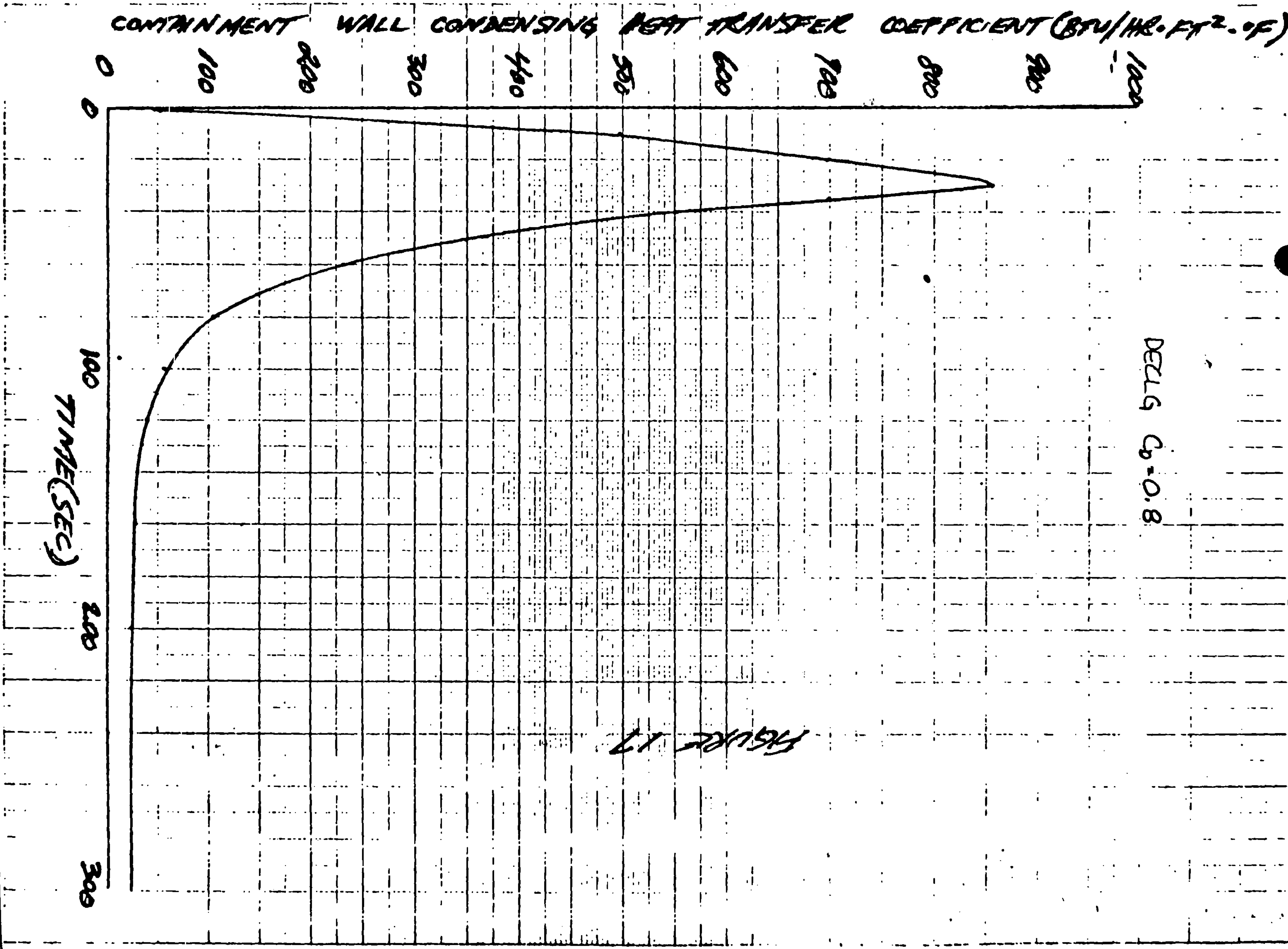

CONTAINMENT WALL CONDENSING HEAT TRANSFER COEFFICIENT (BTU/HR.FT2.°F)
0
100
200
300
400
500
600
700
800
900
1000
TIME (SEC)
0
100
200
300
DECLG Cd=0.8

FIGURE 17

Question 9.61

Show that Salem, Unit 2, complies with the requirements of the enclosed RHR Branch Technical Position (RSB5-1). This discussion should include each of the items tabulated in the enclosure: Impact of Revision 2 to SRP 5.4.7 on PWR Class 2 plants.

Answer

This response addresses the items tabulated in Table II, "Impact of Revision 2 to SRP 5.4.7 on PWR Class 2 Plants."

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
1. Double drop line (or valves in parallel) may not be provided from reactor to RHR system.	Compliance not required if manual actions inside or outside containment or return to hot standby until manual action (or repair) are found acceptable.

A single RHR suction line with two suction isolation valves in series is provided. Compliance is not required since the plant can be maintained in a safe hot standby condition while any required manual actions are taken. See the cold shutdown scenario and single failure evaluation provided below.

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
2. Safety grade dump valves, operators, air and power	Compliance required. Manual action outside control room is

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
supplies, etc., may not be provided and may not meet single failure.	acceptable to meet single failure. Applicant must also verify that air supplies (if used), including leakage, are adequate to operate dump valves for the time duration this system is required.

One safety grade steam generator power operated relief valve is provided for each of the four steam generators. Safety grade remote operators and power supplies are not required since hot standby can be achieved and maintained using the safety grade steam generator safety valves. The steam generator power operated relief valves are provided with handwheels and can be operated locally to permit plant cooldown. See the cold shutdown scenario and single failure evaluation provided below.

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
3. Capability to cooldown to shutdown assuming most limiting single failure in less than 36 hours may not be provided.	Compliance required. Compliance not required if manual actions inside or outside containment or remaining at hot standby unitl manual action (or repair) are found acceptable.

Compliance is not required since the plant can be maintained in a safe hot standby condition while any required manual actions are taken. The plant is capable of reaching RHR initiation conditions in approximately 36 to 48 hours, including time required to perform any manual actions. See the cold shutdown scenario and single failure evaluation provided below.

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
4. Depressurization may not be able to be achieved with only safety-grade systems assuming single failure.	Same as above.

Compliance is not required since the plant can be maintained in a safe hot standby condition while any required manual actions are taken. See the cold shutdown scenario and single failure evaluation provided below.

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
5. Boration with only safety grade systems assuming single failure may not be provided.	Same as above. (If backup method using injection of highly borated water with charging pumps, assuming a letdown line failure is proposed, an analysis of this approach must be performed).

Compliance is not required since the plant can be maintained in a safe hot standby condition while any required manual actions are taken. See the cold shutdown scenario and single failure evaluation provided below.

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
6. Provisions for collection and containment of RHR pressure relief discharge may not be provided.	Complance not required if adequate alternate methods of disposing or discharge available.

The RHR relief valves discharge to the pressurizer relief tank (inside containment).

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
7. Additional tests to study mixing of the added borated water and cooldown under natural circulation conditions with and without a single failure of an atmospheric dump valve not conducted.	Compliance or justification required.

Salem Nuclear Generating Station is similar to Diablo Canyon Power Station in design, both being Westinghouse PWR. Due to the similarity of the two plant, no special tests will be conducted by Salem Unit to establish boron mixing and cooldown capability under natural circulation since Diablo Canyon Station has committed to perform these tests. The results of the tests on Diablo Canyon will be applicable for Salem Unit.

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
8. Specific operational procedures for cooldown under natural circulation may not be provided.	Compliance required.

Salem Nuclear Generating Station will generate specific operational procedures that will enable the operators to bring the plant from hot standby condition to cold shutdown status using the systems and operating functions given below (see Cold Shutdown Scenario).

PWR Areas of Potential Non-Compliance to BTP RSB 5-1	Solution
9. Seismic Category 1 AFW supply for at least four hours at hot shutdown plus cooldown to RHR cut-in based on longest time (for only onsite or off-site power and assuming worst single failure) may not be provided.	Compliance not required if an adequate alternate seismic Category 1 source is available.

A long term source of Auxiliary Feedwater is provided by a connection to the Seismic Category 1 Service Water System.

COLD SHUTDOWN SCENARIO (Assuming loss of all non-seismic Category 1 equipment)

The safe shutdown design basis of Salem Unit 2 is hot standby. The plant can be maintained in a safe hot standby condition while manual actions are taken to permit achievement of cold shutdown conditions following a safe shutdown earthquake with loss of offsite power. Under such conditions the plant is capable of achieving RHR initiation conditions (approximately 350°F, 400 psig) in approximately 36 to 48 hours, including the time required for any manual actions. To achieve and maintain cold shutdown, four key functions must be performed. These are: (1) circulation of the reactor coolant, (2) removal of residual heat, (3) boration and makeup, and (4) depressurization.

1. Circulation of Reactor Coolant

Circulation of the reactor coolant has two stages in a cooldown from hot standby to cold shutdown. The first stage is from hot standby to 350° F. During this stage, circulation of the reactor coolant is provided by natural circulation with the reactor core as the heat source and steam generators as the heat sink. Steam release from the steam generators is initially via the steam generator safety valves and occurs automatically as a result of turbine and reactor trip. Steam release for cooldown is via the steam generator power operated relief valves which are operated manually with their handwheels. The steam generator power operated relief valves are accessible for local operation. The status of each steam generator can be monitored using Class 1E instrumentation located on the console in the Control Room. Three separate channels of indications for both steam generator pressure and water level are available.

Feedwater to the steam generators is provided from the Auxiliary Feedwater System which has a 220,000 gallon Seismic Category 1 Auxiliary Feedwater storage tank as the primary source and two separate Seismic Category 1 piping sub-systems. The first sub-system is composed of two motor-driven pumps each powered from a different emergency power train, and the second sub-system incorporates a turbine driven pump which can receive motive steam from either of two steam generators. There are additional sources of feedwater backup which can be manually accessed. Initial backup is provided by the demineralized water storage tank, the Domestic Water Storage Tank and the Fire Protection Water Tank. Additional backup is from the Seismic Category 1 Service Water System. The operation of the auxiliary feedwater system can be monitored using Class 1E instrumentation located on the control console in the Control Room. There is a single indication of the flows into each steam generator, pump operating status lights for the motor driven pumps, discharge and suction pressure indicator for turbine driven pump. There are also two separate indications of the level in the Auxiliary Feedwater Storage Tank.

The second stage of Reactor Coolant circulation is from 350°F to cold shutdown. During this stage, circulation of the reactor coolant is provided by the Residual Heat Removal Pumps.

2. Removal of Residual Heat

Removal of residual heat also has two stages in a cooldown from hot standby to cold shutdown. The first stage is from hot standby to 350°F.

During this stage, the steam generators act as the means of heat removal from the reactor coolant system. Initially, steam is released from the steam generators via the steam generator safety valves to maintain hot standby conditions. When the operators are ready to begin the cooldown, the steam generator power operated relief valves are slightly opened by local operation with their handwheels. As the cooldown proceeds, the operators will occasionally adjust these valves to increase the amount they are open. This allows a reasonable cooldown rate to be maintained. Feedwater makeup to the steam generators is provided from the Auxiliary Feedwater System. The Auxiliary Feedwater System has the ability to remove decay heat by providing feedwater to all four steam generators for extended periods of operation.

The second stage is from 350°F to cold shutdown. During this stage the Residual Heat Removal (RHR) System is brought into operation. The Residual Heat Removal Heat Exchangers in the RHR system act as the means of heat removal from the Reactor Coolant System. In the RHR Heat Exchanger, the residual heat is transferred to the Component Cooling System which ultimately transfers the heat to the Service Water System. The Component Cooling and the Service Water systems are both designed to Seismic Category 1. The RHR system includes two Residual Heat Removal Pumps and two Residual Heat Removal Heat Exchangers. Each RHR Pump is powered from different emergency power trains and each RHR Heat Exchanger is cooled by a different Component Cooling loop. If any component in one RHR loop becomes inoperable, cooldown of the plant is not compromised, however, the time for cooldown would be extended.

The operation of the RHR system can be monitored using Class 1E instrumentation on the control console in the Control Room. For each RHR loop there is indication of the pump discharge flow, the pump operating status and the Component Cooling flow from the discharge of the RHR heat exchanger.

3. Boration and Makeup

Boration is accomplished using portions of the Chemical and Volume Control System (CVCS). Boric acid 12 wt. % from the Boric Acid Tanks is supplied to the suction of the Centrifugal Charging Pumps by the Boric Acid Transfer Pumps. The Centrifugal Charging Pumps inject the borated water into the Reactor Coolant System via the normal charging and reactor coolant pump seal injection flow paths. The two Boric Acid Tanks, two Boric Acid Transfer Pumps, and the associated piping are of Seismic Category 1 design. There is sufficient boric acid capacity to provide for a cold shutdown with the most reactive rod withdrawn. The Boric Acid Transfer Pumps are each powered from different emergency power trains. The Boric Acid Tank level can be monitored to verify the operability of the boration portion of the CVCS. For this, credit is taken for operator action in using a portable differential pressure indicator which can be connected to the level signal lines from the Boric Acid Tanks.

Makeup, in excess of that provided as 12 wt. % boric acid is provided from the Refueling Water Storage Tank (RWST) using Centrifugal Charging Pumps and the same injection flow paths as described for boration. Two motor operated valves, each powered from different emergency power trains and connected in parallel, will transfer the suction of the changing pumps to the RSWT. Makeup from the RWST can be monitored using Class 1E instrumentation on the control console in the Control Room. Two separate channels of RWST level indication exist.

4. Depressurization

Depressurization is accomplished using portions of the Chemical and Volume Control System (CVCS). Either 12 wt. % boric acid or refueling water can be used as desired for depressurization with the flow path being from the Centrifugal Charging Pumps to the auxiliary spray valve in the Pressurizer. The two Centrifugal Charging Pumps of the CVCS are of Seismic Category I, and are powered from different emergency power trains. The pumps can be operated from and its operating status monitored in the Control Room. The depressurization of the reactor coolant system can be monitored using Class 1E instrumentation on the control console in the Control Room. Available to the operator are four channels of Pressurizer pressure, three channels of Pressurizer level and two channels of reactor coolant pressure.

Maintaining RCS Temperature and Pressure Without Letdown

In performing the cooldown, the operator will integrate the functions of heat removal, boration and makeup, and depressurization so that these functions can be accomplished without letdown from the reactor coolant system. Boration, cooldown, and depressurization will be accomplished in a series of short steps arranged to keep Reactor Coolant System temperature and pressure and Pressurizer level in the desired relationships. However, to demonstrate that boration and depressurization can be done without letdown, a simpler scenario can be used. First, the operators borate the RCS to the cold shutdown conditions, taking advantage of the steam space available in the pressurizer. Second, the operators use the cooldown contraction to lower the pressurizer water level. Finally, the operators use auxiliary spray from the CVCS to depressurize the plant to 425 psia.

The assumed initial conditions following plant trip are:

RCS Temperature = 547°F
RCS Pressure = 2250 psia
Pressurizer Water Volume = 500 ft^3
Pressurizer Steam Volume = 1300 $ft.^3$

To calculate if boration can be accomplished without letting down and without taking the plant water solid, worst case conditions of end of life and maximum peak Xenon were assumed. These result in a requirement for 600 cubic feet of 12 wt. % boric acid at 165°F to reach cold shutdown conditions. When added to the RCS, the boric acid would be heated to 547°F and would expand to 800 cubic feet. Since this volume is less than the 1300 cubic feet available in the pressurizer steam space, boration to cold shutdown concentrations can be accomplished without letdown, without taking the plant water solid, and without cooling down.

The cooldown from 547°F to 350° F decreases the volume of water in the RCS by approximately 1700 cubic feet. Some of this contraction is used to reduce the pressurizer water level to the no-load water level (following the increase caused by the boration) and the remainder is compensated for by makeup from the refueling water storage tank.

To calculate if depressurization can be accomplished without letting down and without taking the plant water solid, it was assumed that the Pressurizer was at saturated conditions with 500 cubic feet of water, 1300 cubic feet of steam, and the Pressurizer metal, all at 653°F (2250 psia). It was further assumed that no additional water would be removed from the pressurizer by the cooldown contraction. With these assumptions, and including the effect of heat input from the pressurizer metal, it was determined that spraying in approximately 820 cubic feet of 165°F water would produce saturated conditions at 425 psia (450°F) with a water volume of 1550 cubic feet and a steam volume of 250 cubic feet.

The results of the calculations described above demonstrate that boration and depressurization can be accomplished without letdown, without taking the plant water solid, and without taking full credit for the available volume created by the cooldown contraction.

SINGLE FAILURE EVALUATION

I. Circulation of the Reactor Coolant

A. From Hot Standby to 350°F (Refer to FSAR Figures 4.2-1, 10.2-1, and 10.2-4).- Four reactor coolant loops and steam generators are provided, any one of which can provide sufficient natural circulation flow to provide adequate core cooling. Even with the most limiting single failure (of a steam generator power operated relief valve), three of the reactor coolant loops and steam generators remain available.

B. From 350°F to cold shutdown (Refer to FSAR Figure 9.2-1) - Two RHR pumps are provided, either one of which can provide adequate circulation of the reactor coolant.

II. <u>Removal of Residual Heat</u>

A. From Hot Standby to 350°F (Refer to FSAR Figures 10.2-1, 10.2-4, and 9.9-1).

1. Steam generator power operated relief valves - Four are provided (one per steam generator), any one of which is sufficient for residual heat removal. In the event of a single failure, three power operated relief valves remain available.

2. Auxiliary Feedwater Pumps - Two motor driven and one steam driven auxiliary feedwater pumps are provided. In the event of a single failure, two pumps remain available, either of which can provide sufficient feedwater flow.

3. Flow control valves - Air operated, fail open valves. In the event of a single failure of one flow control valve (which effects flow to one steam generator from either a motor driven pump or the steam driven pump) auxiliary feed flow can still be provided to all four steam generators from the other pumps.

4. Backup source - A backup source of auxiliary feedwater can be provided via a spool piece from either train of the Seismic Category 1 Service Water System.

B. From 350°F to 200°F (Refer to FSAR Figures 9.3-1, 9.5-1 and 9.9-1).

1. RHR Suction Isolation Valves 1RH1 and 1RH2 - These valves are each powered from different emergency power trains. Failure of either power train or of either valve operator could prevent initiation of RHR cooling in the normal manner from the control room. In the event of such a failure, operator action could be taken to open the affected valve manually. The mechanical failure of the disc separating from the stem has been investigated (WCAP-9207) and its probability has been found to be in the range of 10^{-4} to 10^{-3} per year. The probability of an earthquake larger than the OBE is less than $8X10^{-5}$ per year. The combined probability of valve stem failure coincident with the earthquake ($<8X10^{-8}$) is so low that it need not be considered in the single failure analysis. In the event of a failure, the plant would remain in a safe hot standby condition with heat removal via the steam generators.

2. Isolation Valves 11RH4 and 12RH4 - If either of these normally open motor operated valves, which are powered from different emergency power trains, were to close spuriously, RHR cooling would be provided by the uneffected RHR pump and heat exchanger. The affected valve could be deenergized and opened with its handwheel.

3. RHR Pumps 11 and 12 - Each pump is powered from a different emergency power train. In the event of a single failure, either pump provides sufficient RHR flow.

4. RHR Heat Exchangers 11 and 12 - If either heat exchanger is unavailable for any reason, the remaining heat exchanger provides sufficient heat removal capability.

5. RHR Flow Control Valves 11RH18 and 12RH18 - If either of these normally open fail open valves should close spuriously, sufficient RHR cooling would be provided by the unaffected RHR train.

6. RHR/SIS Cold Leg Isolation Valves 11SJ49 and 12SJ49 - If either of these normally open, motor operated valves, which are powered from different emergency power trains, should close spuriously, sufficient RHR cooling would be

provided by the unaffected RHR train. The affected valve could be deenergized and opened with its handwheel.

7. Component Cooling Water System - Two redundant subsystems provided for safety related loads. Either subsystem can provide sufficient heat removal via one of the RHR heat exchangers.

8. Service Water System - Two redundant subsystems provided for safety related loads. Either subsystem can provide sufficient heat removal via one of the CCW heat exchangers.

III. Boration and Makeup (Refer to FSAR Figures 4.2-1, 6.2-1 and 9.2-1)

A. Boric Acid Tanks 11 and 12 - Two boric acid tanks are provided. Each tank contains sufficient 12% boric acid to borate the reactor coolant system for cold shutdown.

B. Boric Acid Transfer Pumps 11 and 12 - Each pump is powered from a different emergency power train. In the event of a single failure, either pump will provide sufficient boric acid flow.

C. Isolation Valve 1CV175 - If valve 1CV175, which is supplied from emergency power and is normally closed, cannot be opened due to power train or operator failure it can be opened locally with its handwheel. If valve 1CV175 cannot be opened with its handwheel, an alternate flow path is available via air operated, fail open valve 1CV172 and normally closed manual valve 1CV174.

D. Isolation Valves 1SJ1 and 1SJ2 - Each valve is powered from a different emergency power train, only one of these normally closed motor operated valves needs to be opened to provide a makeup flow path from the RWST to the Centrifugal Charging Pumps.

E. Centrifugal Charging Pumps 11 and 12 - Each pump is powered from a different emergency power train. In the event of a single failure, either pump provides sufficient boration or makeup flow.

F. Flow Control Valve 1CV55 - This normally open valve fails closed on loss of air or power. If 1CV55 closed spuriously, the charging pumps would operate on their miniflow circuits until operator action could open bypass valves 1CV81 and 1CV82.

G. Flow Control Valve 1CV71 - This normally open valve fails closed on loss of air or power. Use of a portable nitrogen bottle would allow 1CV71 to be reopened. If 1CV71 was stuck closed as a result of a single failure, manual bypass valve 1CV73 could be opened locally.

H. Isolation Valves - 1CV68 and 1CV69 - If either of these normally open, motor operated valves, each of which is powered from a different emergency power train, should close spuriously, operator action could be used to deenergize the valve operator and reopen the valve with its handwheel.

I. Isolation Valve 1CV77 - If the normally open valve should close spuriously, alternate charging valve 1CV79, which fails open, could be used.

V. <u>Depressurization</u>

A. Auxiliary Spray Valve 1CV75 - This normally closed valve fails closed on loss of air or power. Use of a portable nitrogen bottle would allow 1CV75 to be opened. If 1CV75 was stuck closed as a result of a single failure, the redundant Seismic Category 1 overpressure protection system valves can be used to depressurize the RCS by venting the pressurizer to the PRT.

B. Charging Valves 1CV77 and 1CV79 - These valves fail open on loss of air or power. Use of portable nitrogen bottles would allow 1CV77 and 1CV79 to be closed. If either was stuck open, the redundant seismic category 1 overpressure protection system valves can be used to depressurize the RCS by venting the pressurizer to the PRT.

Environmental Qualification of the RHR Suction Isolation Valves

The RHR suction isolation valves are qualified for the steam line break environment. Therefore, they are qualified for the less severe environment which would result from venting the pressurizer to depressurize the RCS.

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