REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8001150426 DOC.DATE: 80/01/11 NOTARIZED: NO DOCKET #
FACIL:50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
AUTH.NAME AUTHOR AFFILIATION
HUNTER,R.S. Indiana & Michigan Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Forwards ECCS analysis using approved Westinghouse Feb 1978 evaluation model.ECCS will meet acceptance criteria presented in code for breaks up to & including double-ended severance of reactor cooling pipe.

DISTRIBUTION CODE: A039S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 72
TITLE: Resp to Lesson Learn Task Force - Westinghouse

NOTES: SEND 3 CYS ALL MATL TO I&E.

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
ACTION:	10 BC ORB #1	7	7			
INTERNAL:	1 REG FILE	1	1	17 I & E	2	2
	19 ~~TA/EDO~~	1	1	2 NRC PDR	1	1
	20 CORE PERF BR	1	1	21 ENG BR	1	1
	22 REAC SFTY BR	1	1	23 PLANT SYS BR	1	1
	24 EEB	1	1	25 EFLT TRT SYS	1	1
	3 LPDR	1	1	4 NSIC	1	1
	5 J OLSHINSKI	1	1	6 J KERRIGAN	1	1
	7 J BURDION	1	1	8 C WILLIS	1	1
	9 G IMBRO	1	1	J.T. TELFORD	2	2
	M FIELDS	1	1	N ANDERSON	1	1
	OELD	1	0	P O'REILLY	1	1
EXTERNAL:	26 ACRS	16	16			

JAN 16 1980

ECCS

TOTAL NUMBER OF COPIES REQUIRED: LTTR ~~47~~ 50 ENCL ~~46~~ 49

INDIANA & MICHIGAN ELECTRIC COMPANY

P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

January 11, 1980
AEP:NRC:00322C

Donald C. Cook Nuclear Plant Unit No. 2
Docket No. 50-316
License No. DPR-74
Subject: Peaking Factor Limits at Unit 2
In Light of Revised Analysis

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

The purpose of this letter is to formally transmit to you information which was presented at a meeting held on January 10, 1980 and which was attended by members of your staff, by AEP and by Westinghouse Electric Corporation. Attachment A contains an ECCS analysis for Unit 2 of the Cook Plant using the approved Westinghouse February 1978 evaluation model. This analysis resulted in a total F_Q of 2.02 while our previously submitted and approved analysis using the October 1975 model with the correct metal-water reaction description yielded an F_Q of 2.11. The data explaining this reduction in F_Q when using the newer model is contained in Attachment B. The February 1978 analysis and the change in peaking factors associated with its use was discussed in detail at the referenced meeting wherein all attendees agreed that the February 1978 analysis is valid.

In light of the data presented in draft NUREG-0630, the total F_Q for Unit 2 was further reduced by 0.03 as explained in our letters of January 8, 1980 (AEP:NRC:00322A) and January 9, 1980 (AEP:NRC:00322B). We have accepted this further reduction in F_Q to 1.99 voluntarily.

8001150426

A039
S 1/1

[illegible]

[illegible]

Finally, we have reviewed the revised Technical Specification Figure 3.2-2 submitted in AEP:NRC:00322B. This review confirms that the submitted figure is properly based on the February 1978, Unit 2 - specific large LOCA analysis and the Unit 2 small break analysis previously submitted and approved (FSAR Section 14.3.2), and is therefore correct.

Very truly yours,

R. S. Hunter
Vice President

RSH:em

cc: R. C. Callen
G. Charnoff
J. E. Dolan
R. W. Jurgensen
D. V. Shaller-Bridgman

ATTACHMENT A
TO
AEP:NRC:00322C

The Loss of Coolant Accident (LOCA) has been reanalyzed for Donald C. Cook Unit 2 (AMP). The following information amends the Safety Analysis Report section on Major Reactor Coolant System Pipe Ruptures. The results are consistent with acceptance criteria provided in reference 1.

The description of the various aspects of the LOCA analysis is given in WCAP-8339[2]. The individual computer codes which comprise the Westinghouse Emergency Core Cooling System (ECCS) evaluation model are described in detail in separate reports [3-6a] along with code modifications specified in references 7, 9 and 10. The analysis presented here was performed with the February 1978 version of the evaluation model which includes modifications delineated in references 11, 12, 13 and 14.

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 occurence 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. That 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 LOTIC computer code[6-6a] 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. 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

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.[1] That is:

1. The calculated peak clad temperature does not exceed 2200°F based on a total core peaking factor of 2.02.

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.

(American Electric Power should add a discussion here referring to the W and NRC trust and blockage issues leadership to a I.P.; [illegible] greater and refer to their 60 day response letter)

References for Section 15.4.1

1. "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.

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

3. 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.

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

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

6.) 6a) Hsieh, T. & Raymond M. ," Long Term Ice Condenser Containment-LOTIC Code Supplement 1," WCAP 8355 Supplement 1, May 1975. WCAP-8354 (Proprietary), July 1974.

Letter from K. Kniel to J. Tillinghast, December 27, 1974- "Order for Modification of License for the Donald C. Cook Nuclear Plant Unit 1."

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

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

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

10. 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.

11. 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.

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

13. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-1981, Nov. 1, 1978.

14. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-2014, Dec. 11, 1978.

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	DECL C_D = 1.0 (Sec)	DECL C_D = 0.8 (Sec)	DECL C_D = 0.6 (Sec)
START	0.0	0.0	0.0
Rx Trip Signal	0.70	0.71	0.72
S. I. Signal	29.78	29.83	29.95
Acc. Injection	14.0	14.6	16.8
End of Blowdown	31.88	39.07	36.83
Bottom of Core Recovery	42.13	48.97	43.28
Acc. Empty	55.52	56.42	58.65
Pump Injection	29.78	29.83	29.95
End of Bypass	27.62	33.18	28.64

TABLE 2

LARGE BREAK

Results	DECL C_D = 1.0	DECL C_D = 0.8	DECL C_D = 0.6
Peak Clad Temp. °F	1934	2171	2023
Peak Clad Location Ft.	7.5	7.5	7.5
Local Zr/H_2O Rxn(max)%	3.07	6.36	4.09
Local Zr/H_2O Location Ft.	7.5	7.5	7.5
Total Zr/H_2O Rxn %	<0.3	<0.3	<0.3
Hot Rod Burst Time sec	42.0	40.6	37.3
Hot Rod Burst Location Ft.	6.0	6.0	6.0

Calculation	
NSSS Power Mwt 102% of	3391
Peak Linear Power kw/ft 102% of	11.152
Peaking Factor (At License Rating)	2.02
Accumulator Water Volume	950 ft^3/tank

Fuel region + cycle analyzed	Cycle	Region
UNIT 1		
UNIT 2 (If applicable)		

TABLE 3

CONTAINMENT DATA (ICE CONDENSER CONTAINMENT)

I) NET FREE VOLUME

	UPPER COMPARTMENT	746,829 ft^3
	LOWER COMPARTMENT	249,446 ft^3
	DEAD END COMPARTMENT	116,168 ft^3
	ICE CONDENSOR COMPARTMENT	122,400 ft^3

II) INITIAL CONDITIONS

PRESSURE		14.7 psia
TEMPERATURE	UPPER COMPARTMENT	100°F
	LOWER COMPARTMENT	120°F
	DEAD END COMPARTMENT	120°F
RWST TEMPERATURE		80°F
SERVICE WATER TEMPERATURE		40°F
TEMPERATURE OUTSIDE CONTAINMENT		-7°F
INITIAL SPRAY TEMPERATURE		80°F

Table 3 – Continued

III) Spray System

Runout Flow For a Spray Pump		3600 gpm
Number of Spray Pumps Operating		2
Post Accident Initiation of Spray System		40 sec
Distribution of Spray Flow to Upper	UC	2835 gpm
and Lower Compartment	LC	4365 gpm

IV) Deck Fan

Post Accident Initiation of Deck Fan	600 sec
Flow Rate Per Fan	39000 cfm/fan

V) Hydrogen Skimmer System Flow Rate	2800 cfm/fan
VI) Assumed Spray Efficiency Of Water From Ice Condensor Drains	100%

TABLE 3 – CONTINUED
STRUCTURAL HEAT SINKS

COMPARTMENT		AREA (Ft^2)	THICKNESS (Ft)	COMPOSITION
1	LC	12105	0.00108/0.0469/2.0	PAINT/STEEL/CONCRETE
2	LC	11700	2.0	Concrete
3	LC	65980	1.35	Concrete
4	LC	5481	0.000067/0.0833	Paint/Steel
5	LC	4735	0.000067/0.01147	Paint/Steel
6	LC	289	0.25	Lead
7	LC	14690	0.000067/0.0079	Paint/Steel
8	LC	3439	0.000067/0.1561	Paint/Steel
9	LC	5775	0.000067/0.009	Paint/Steel
10	LC	4966	0.000067/0.0096	Paint/Steel
11	LC	7013	0.000067/0.037	Paint/steel
12	LC	2457	0.0334	Steel
13	UC	378	0.000067/0.0365/0.1667	Paint/Steel/Concrete
14	UC	29772	0.00108/0.0092	Paint/Steel
15	UC	8033	0.000067/0.0209	Paint/Steel
16	UC	420	0.000067/0.0052	Paint/steel
17	UC	29330	1.47	Concrete
18	UC	34125	0.00108/0.0469/2.0	Paint/Steel/Concrete
19	UC	210	0.0052	Steel

TABLE 4

REFLOOD MASS & ENERGY RELEASES

D.C. COOK UNIT 2

DECLG $C_D = 0.8$

TIME (Sec)	$\dot{m}$ (TOTAL) (LBm/sec)	$\dot{m}h$ (TOTAL) (BTU/sec)
48.97	0.	0.
56.38	30.01	3.89 +4
68.46	63.99	7.84 +4
88.16	83.09	1.01 +5
108.61	380.28	1.94 +5
128.91	403.52	1.97 +5
150.81	409.65	1.92 +5
199.01	420.41	1.81 +5
253.91	430.77	1.70 +5
317.81	441.99	1.58 +5
391.61	450.50	1.46 +5

TABLE 5

BROKEN LOOP ACCUMULATOR FLOW

TIME	$\dot{m}$	$\dot{m}h$
0.	0.0	0.0
0.2	288.4	1.72×10^4
1.0	2774.8	1.65×10^5
2.0	2585.7	1.54×10^5
3.0	2438.1	1.45×10^5
4.0	2312.5	1.38×10^5
5.0	2203.7	1.31×10^5
6.0	2111.5	1.26×10^5
8.0	1958.8	1.17×10^5
10.0	1836.4	1.09×10^5
12.0	1736.2	1.04×10^5
14.0	1651.7	9.85×10^5
16.0	1579.1	9.41×10^5
20.0	1459.9	8.70×10^5
24.0	1365.0	8.14×10^5
30.0	1258.9	7.51×10^5
33.0	1213.4	7.23×10^5
35.54	1186.0	7.07×10^5

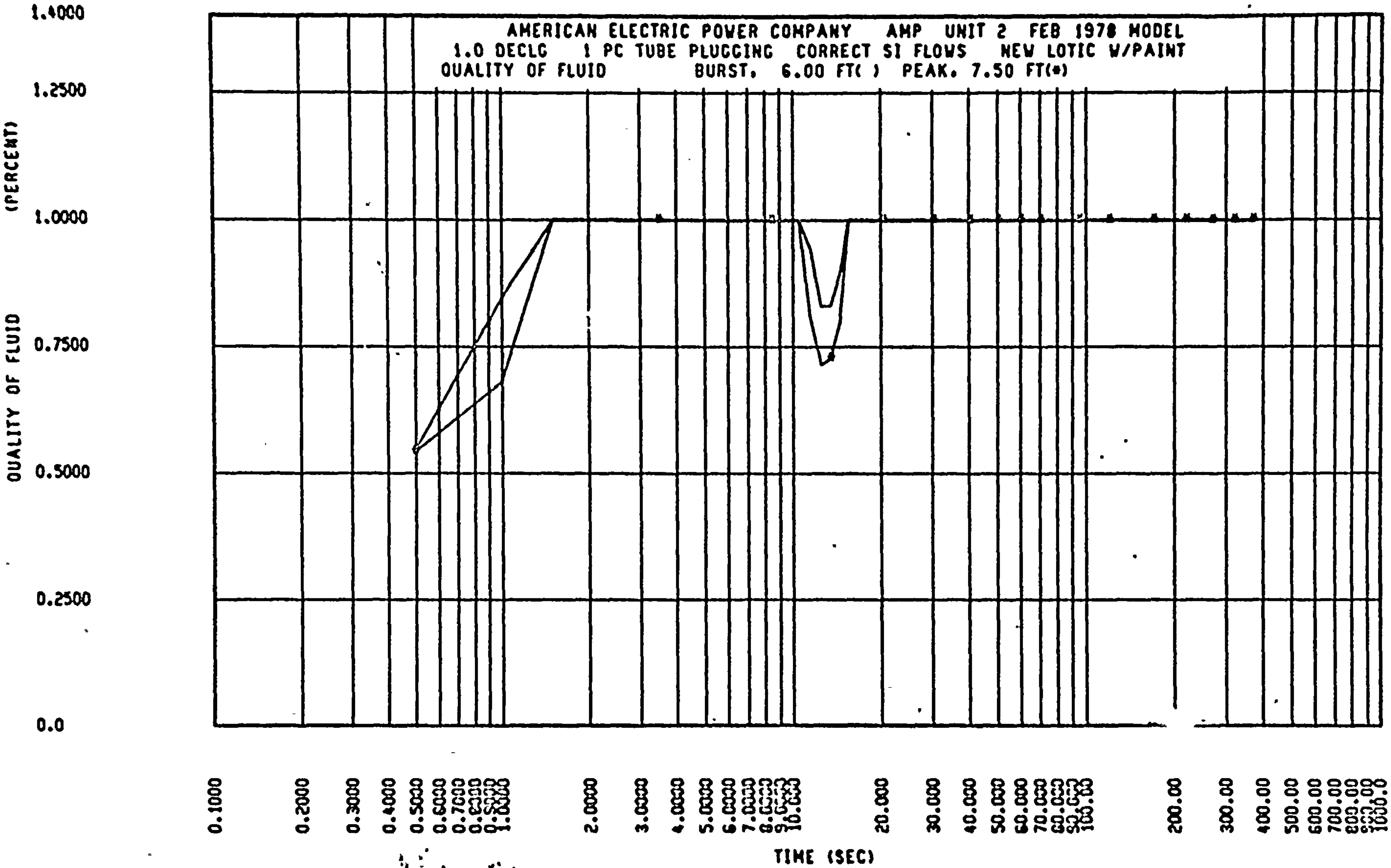


FIGURE 1a FLUID QUALITY - DECLG(CD = 1.0)

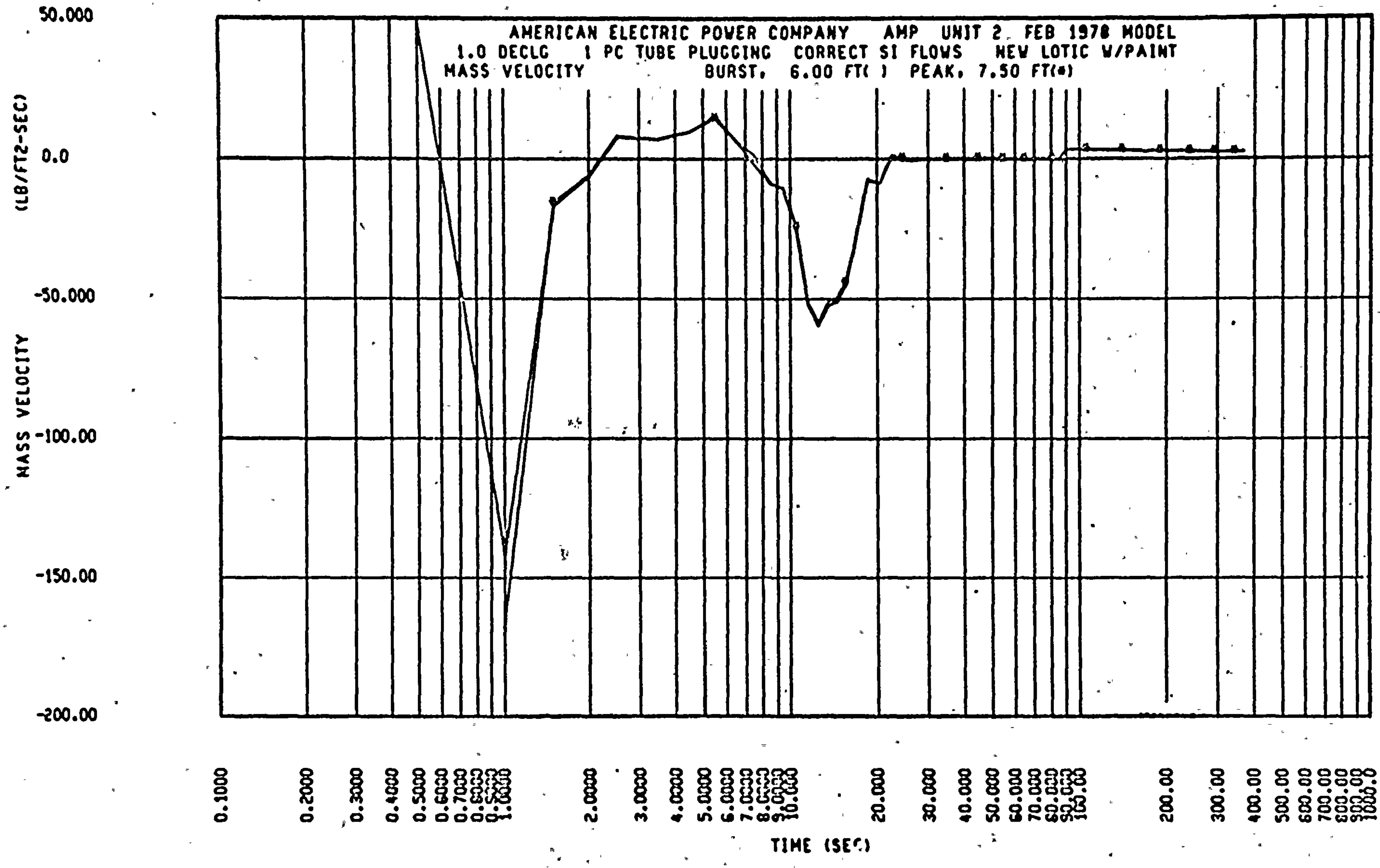


FIGURE 2 a MASS VELOCITY - DECLG(CD = 1.0)

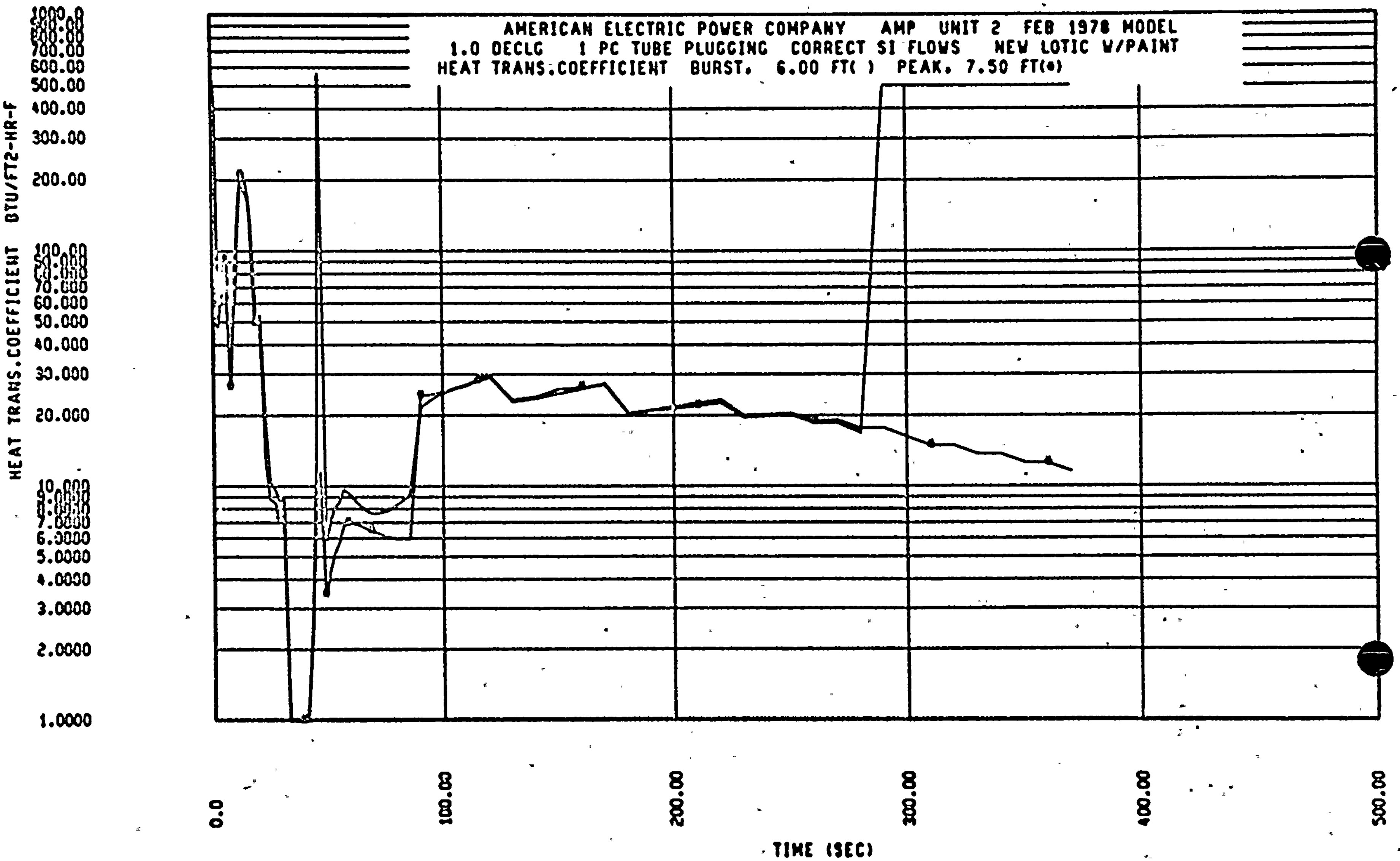


FIGURE 3 a HEAT TRANSFER COEFFICIENT - DECLG(CD = 1.0)

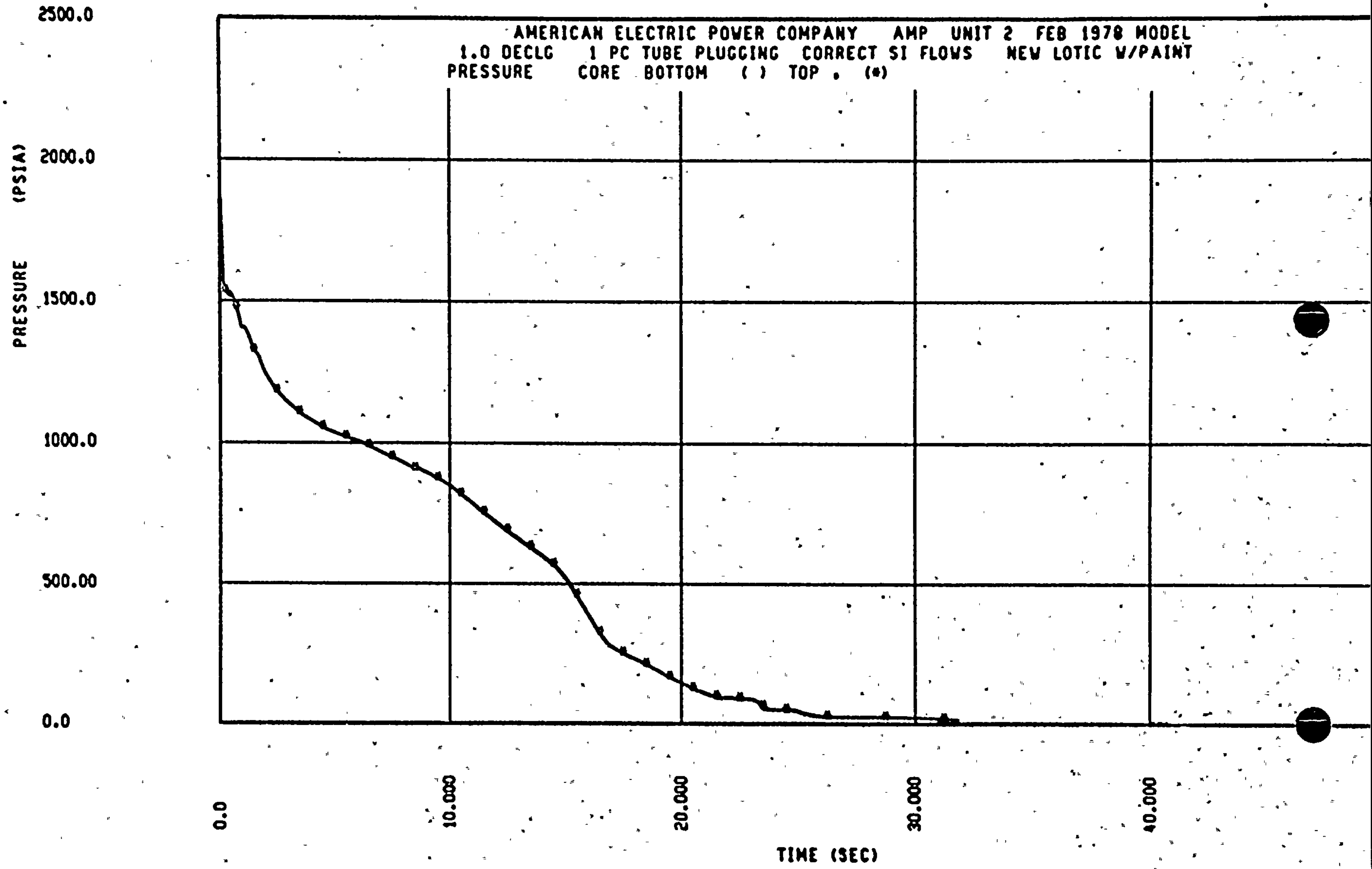


FIGURE 4 a CORE PRESSURE - DECLG(CD = 1.0)

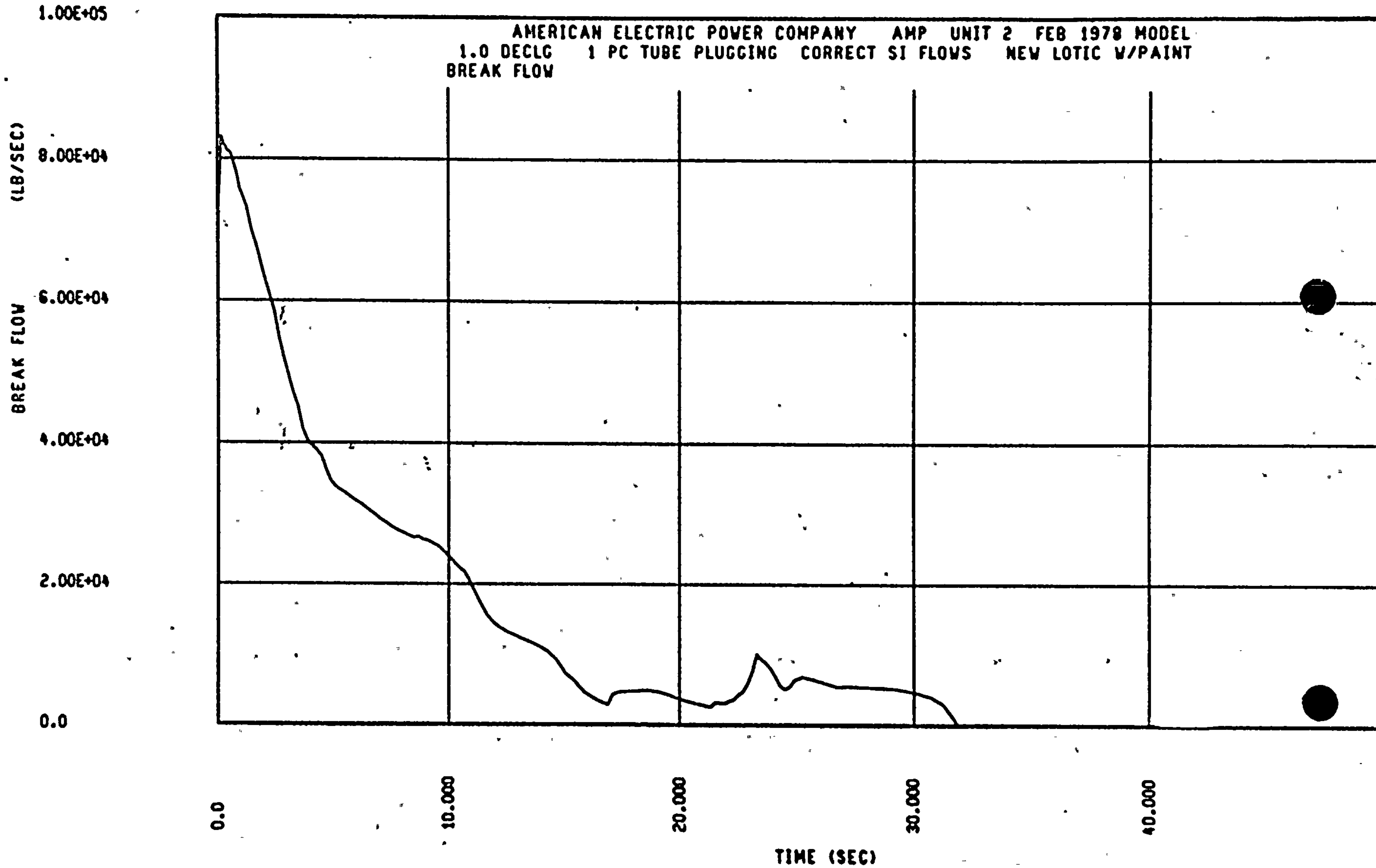


FIGURE 5a BREAK FLOW RATE - DECLG(CD = 1.0)

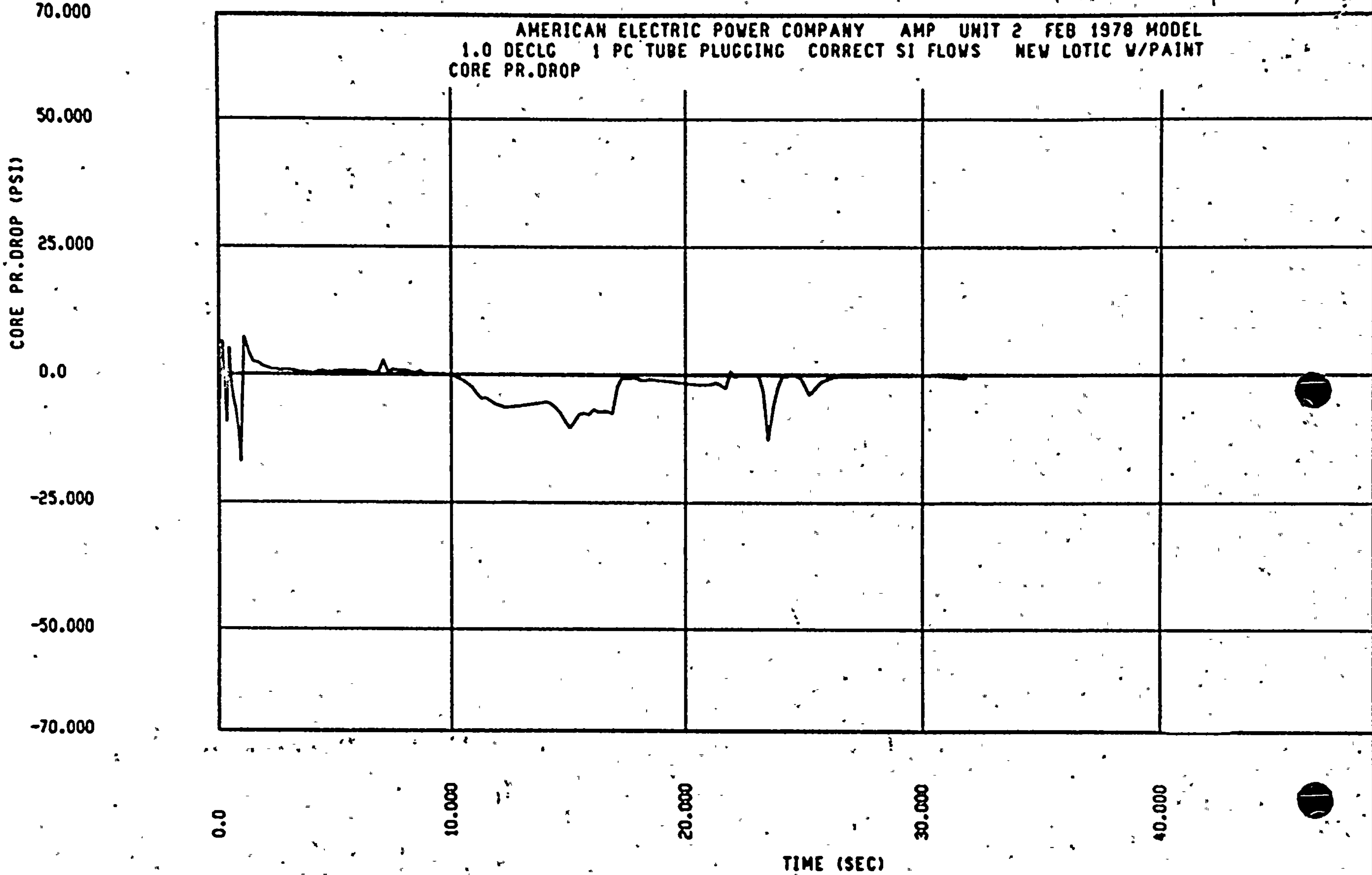


FIGURE 6 a . CORE PRESSURE DROP - DECLG(CD = 1.0)

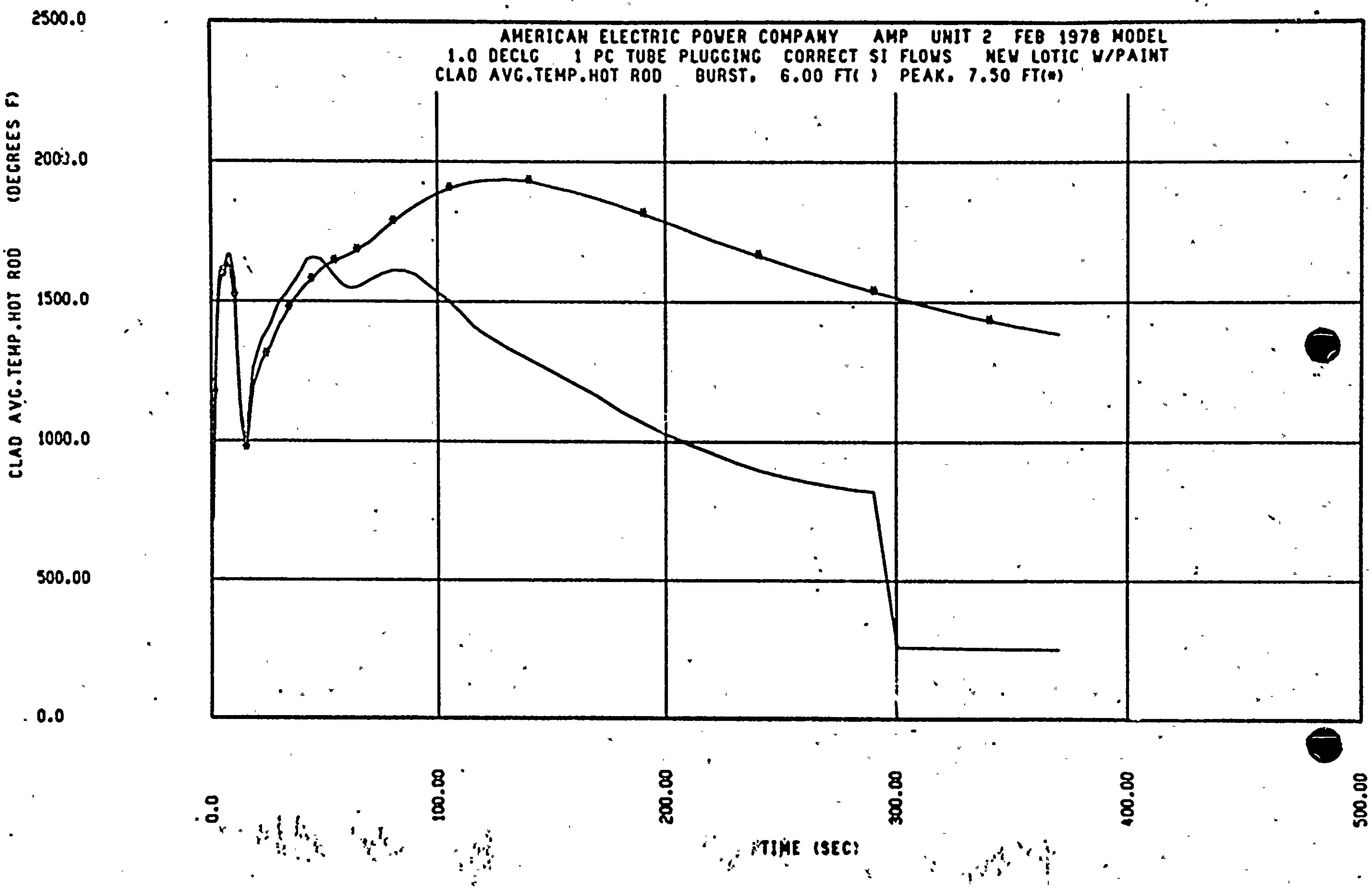


FIGURE 7a - PEAK CLAD TEMPERATURE - DECLG(CD = 1.0)

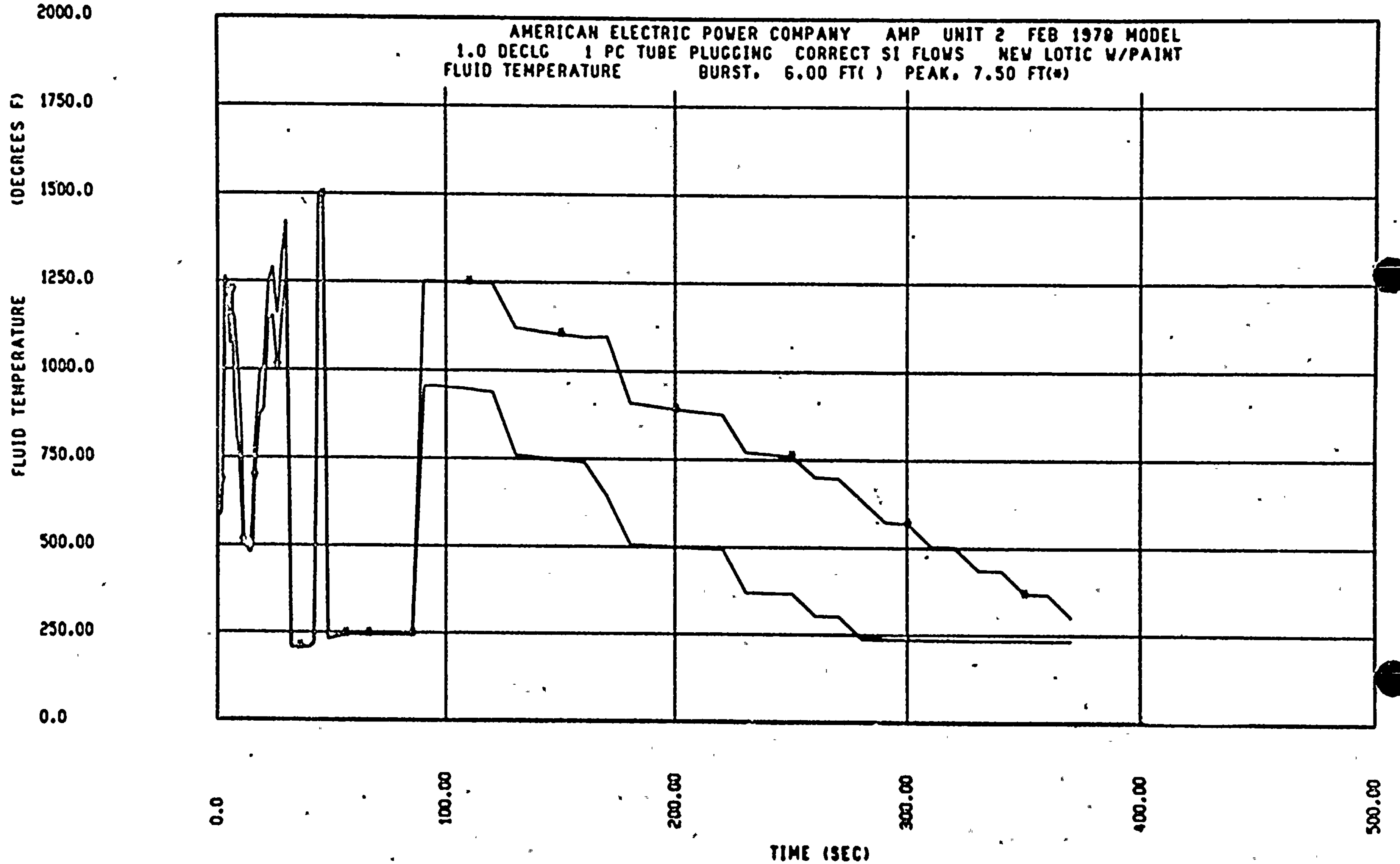


FIGURE 8 a FLUID TEMPERATURE - DECLG(CD = 1.0)

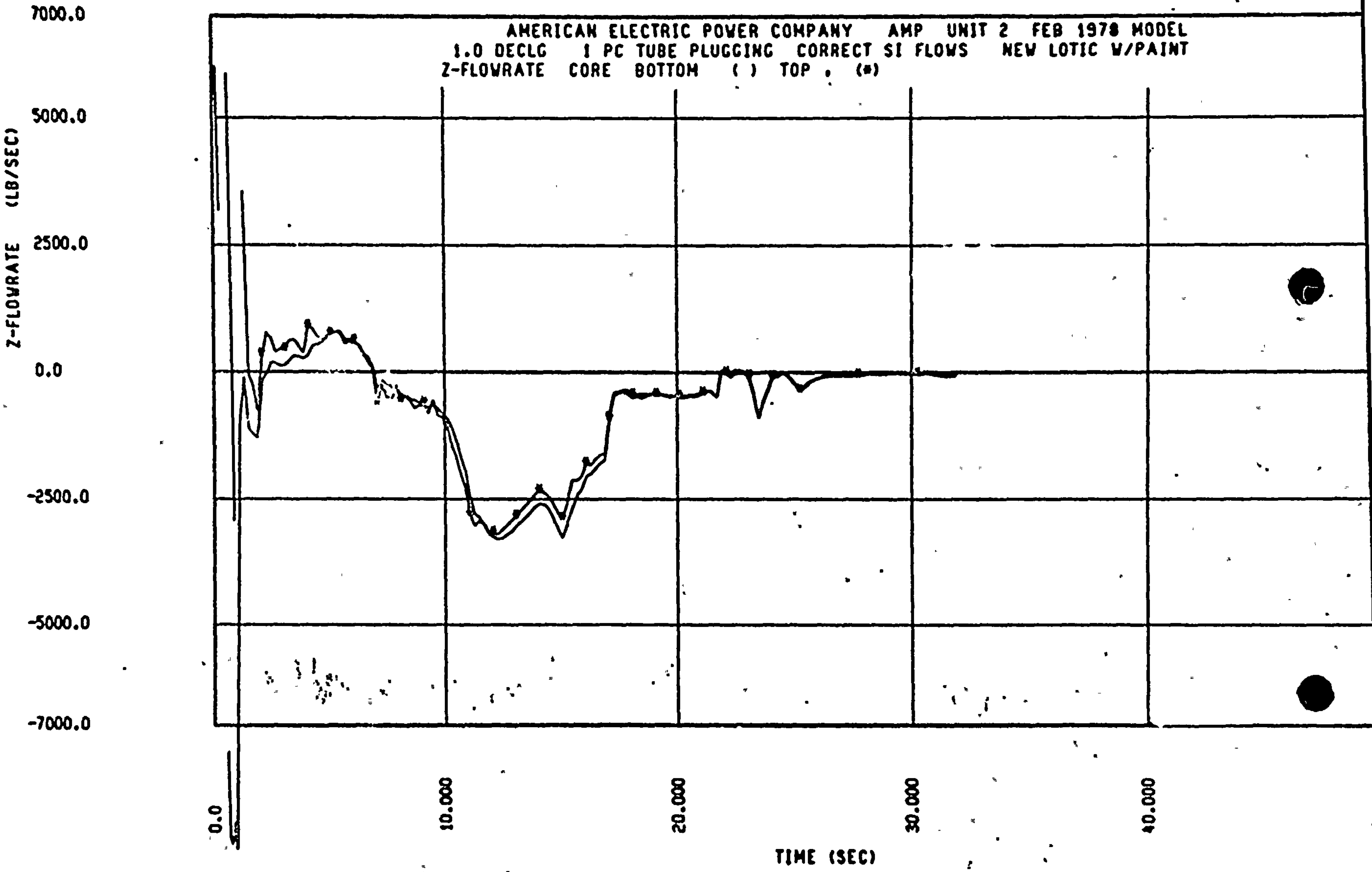


FIGURE 9 a CORE FLOW (TOP AND BOTTOM) - DECLG(CD = 1.0)

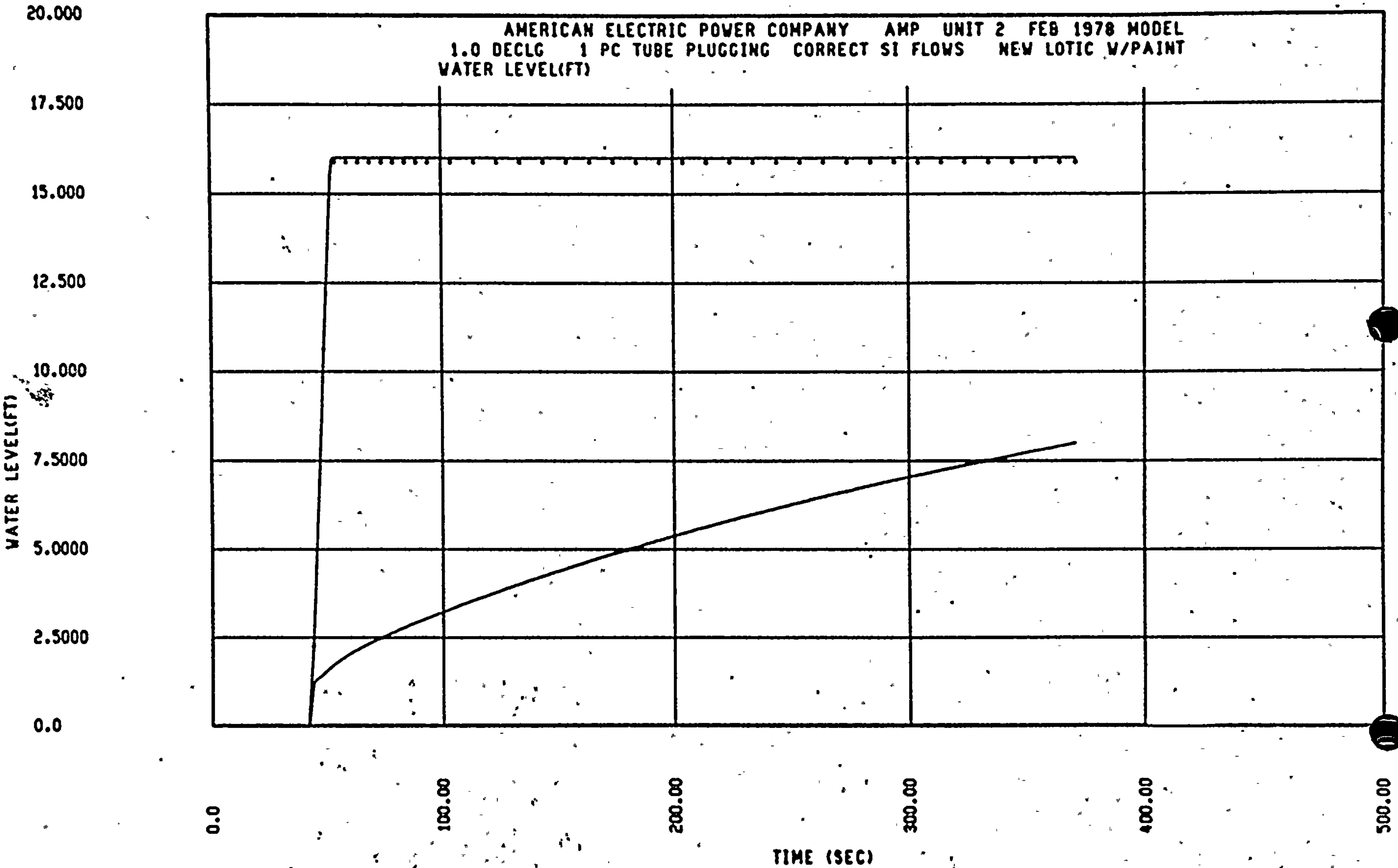


FIGURE 10 a REFLOOD TRANSIENT - DECLG(CD = 1.0)
DOWNCOMER AND CORE WATER LEVELS

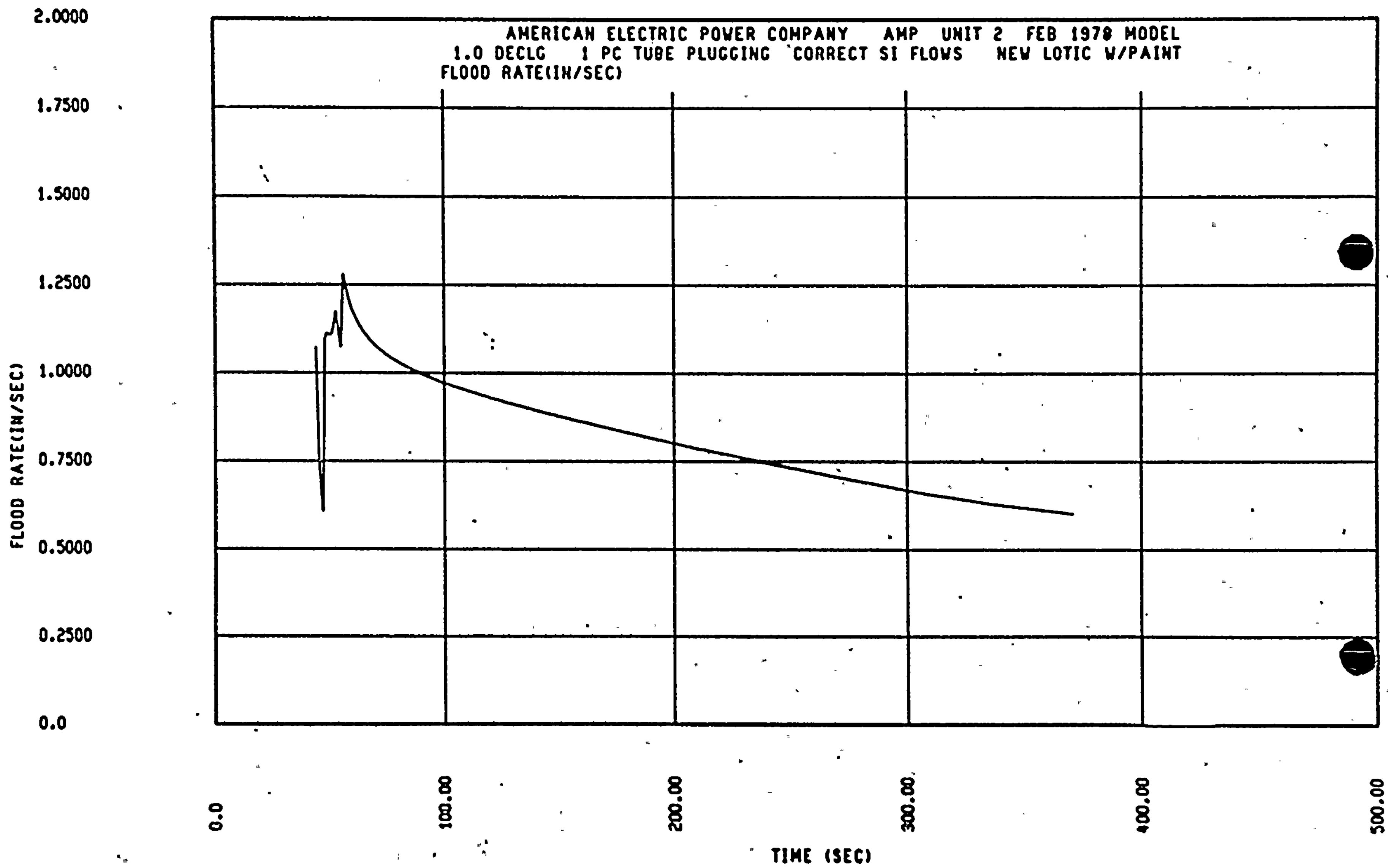


FIGURE 11 a REFLOOD TRANSIENT - DECLG(CD = 1.0)
CORE INLET VELOCITY

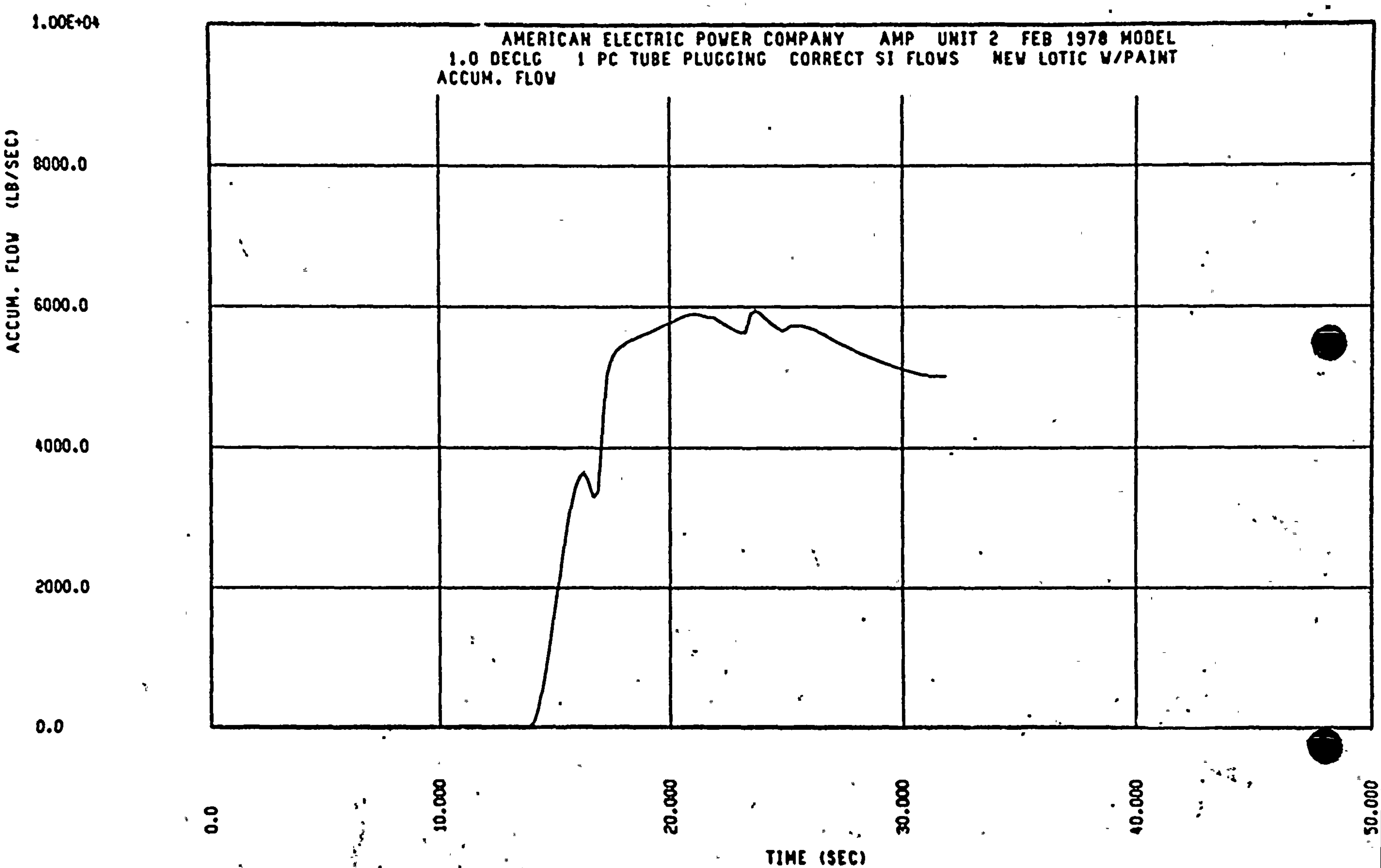


FIGURE 12 a ACCUMULATOR FLOW (BLOWDOWN) - DECLG(CD = 1.0)

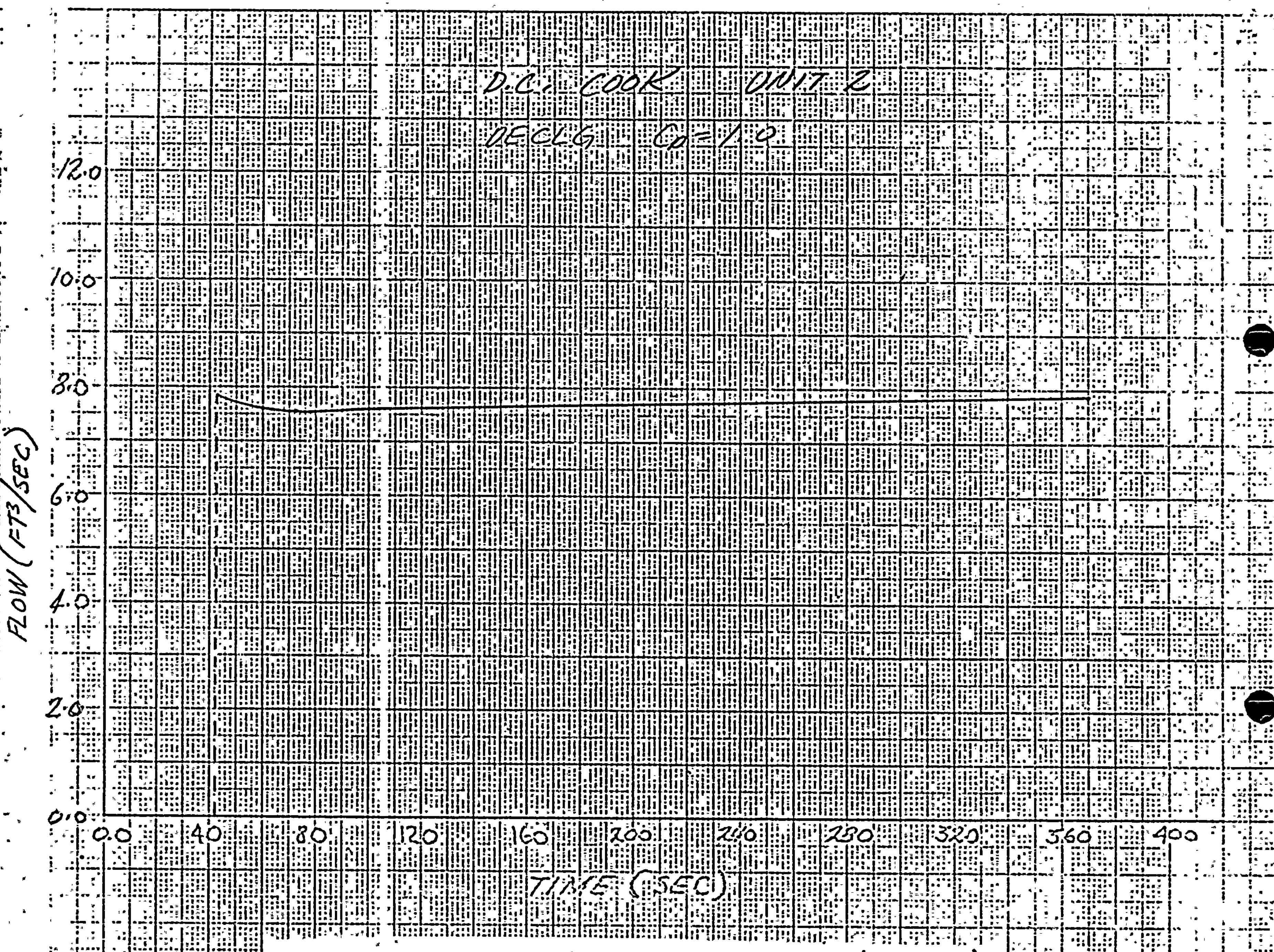


FIGURE 13 a PUMPED ECCS FLOW (REFLOOD) - DECLG(CD = 1.0)

D.C. COOK UNIT 2

DECLG CD - 1.0

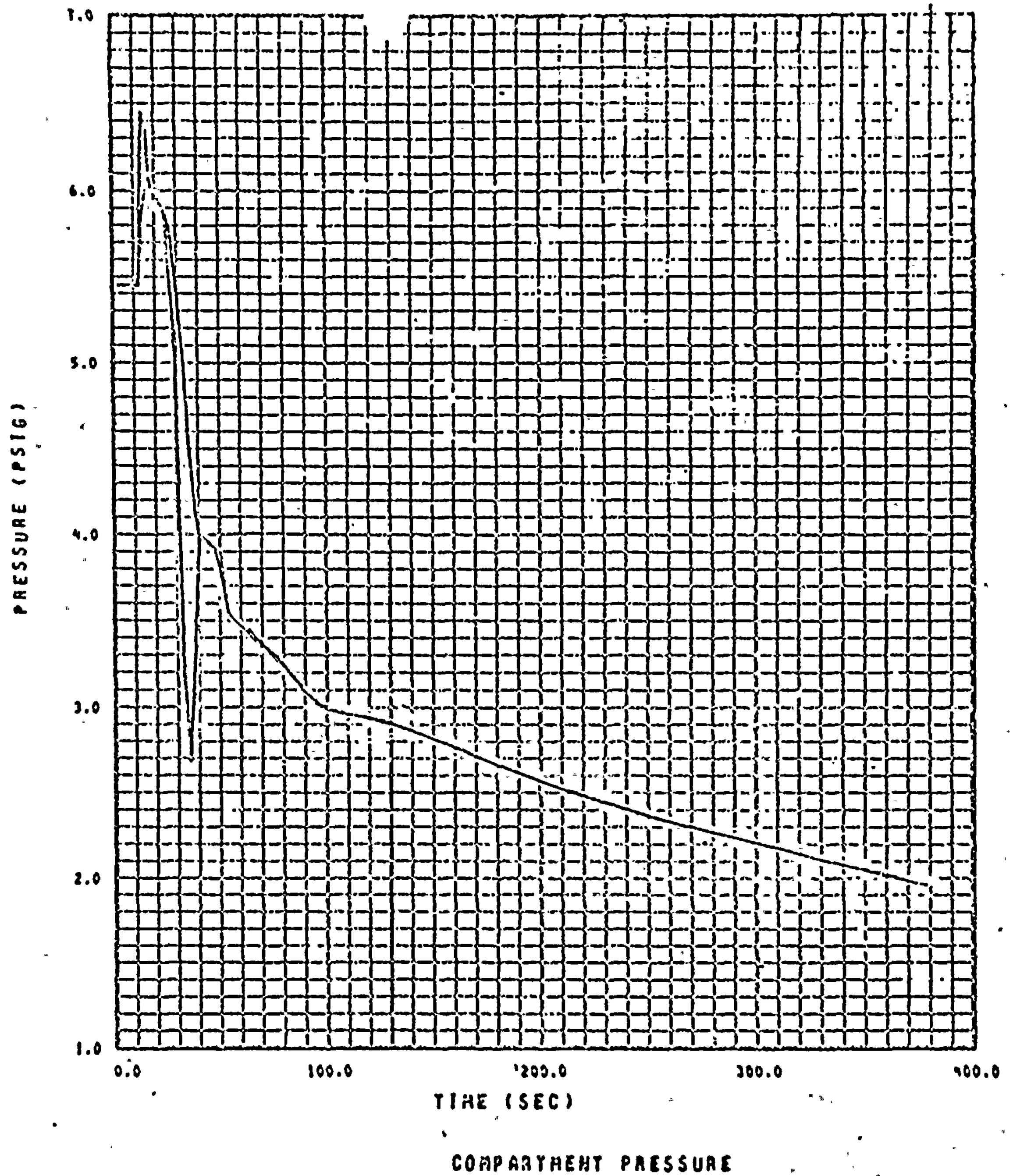


FIGURE 14 a · CONTAINMENT PRESSURE - DECLG(CD = 1.0)

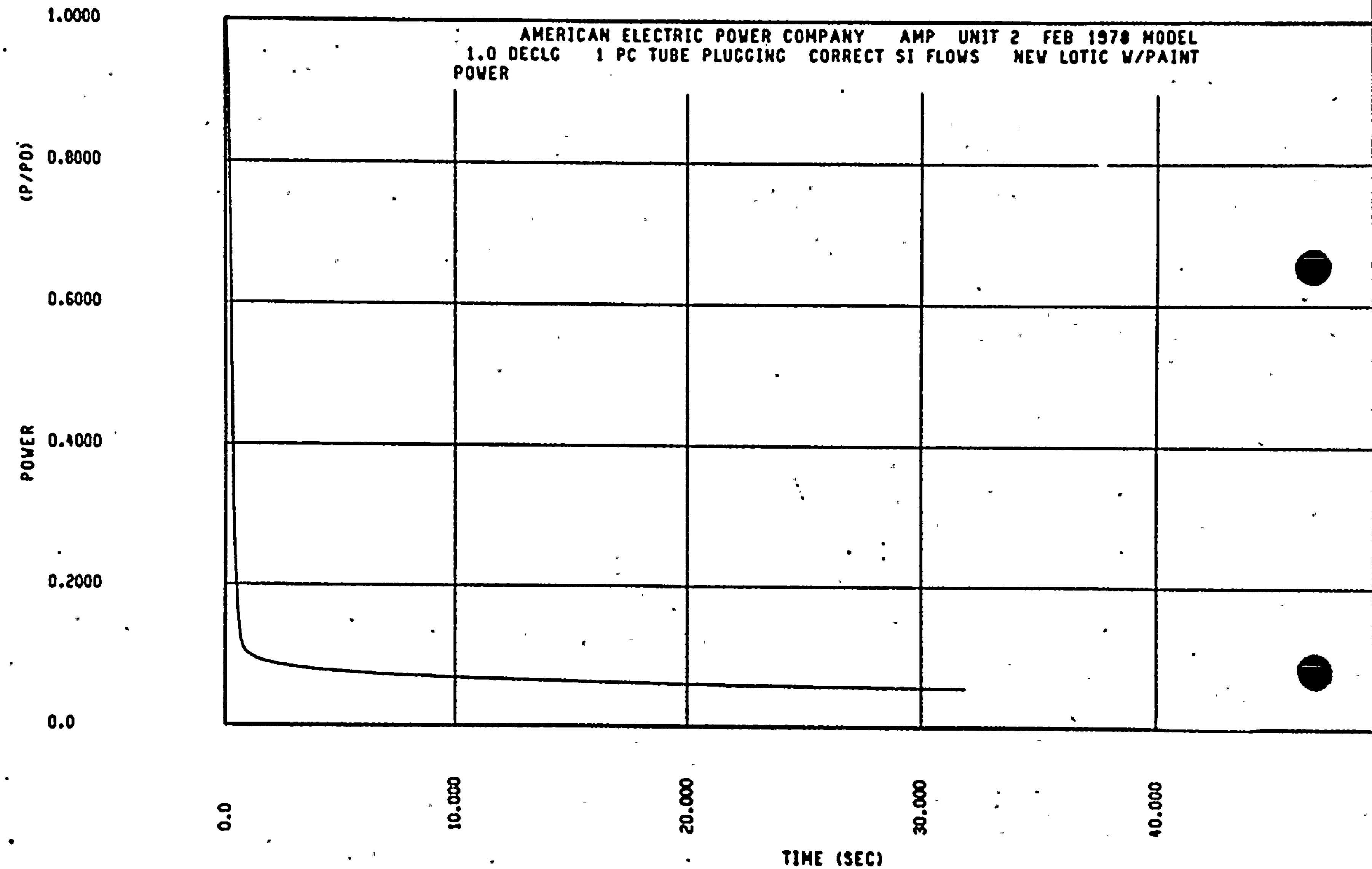


FIGURE 15 a CORE POWER TRANSIENT - DECLG(CD = 1.0)

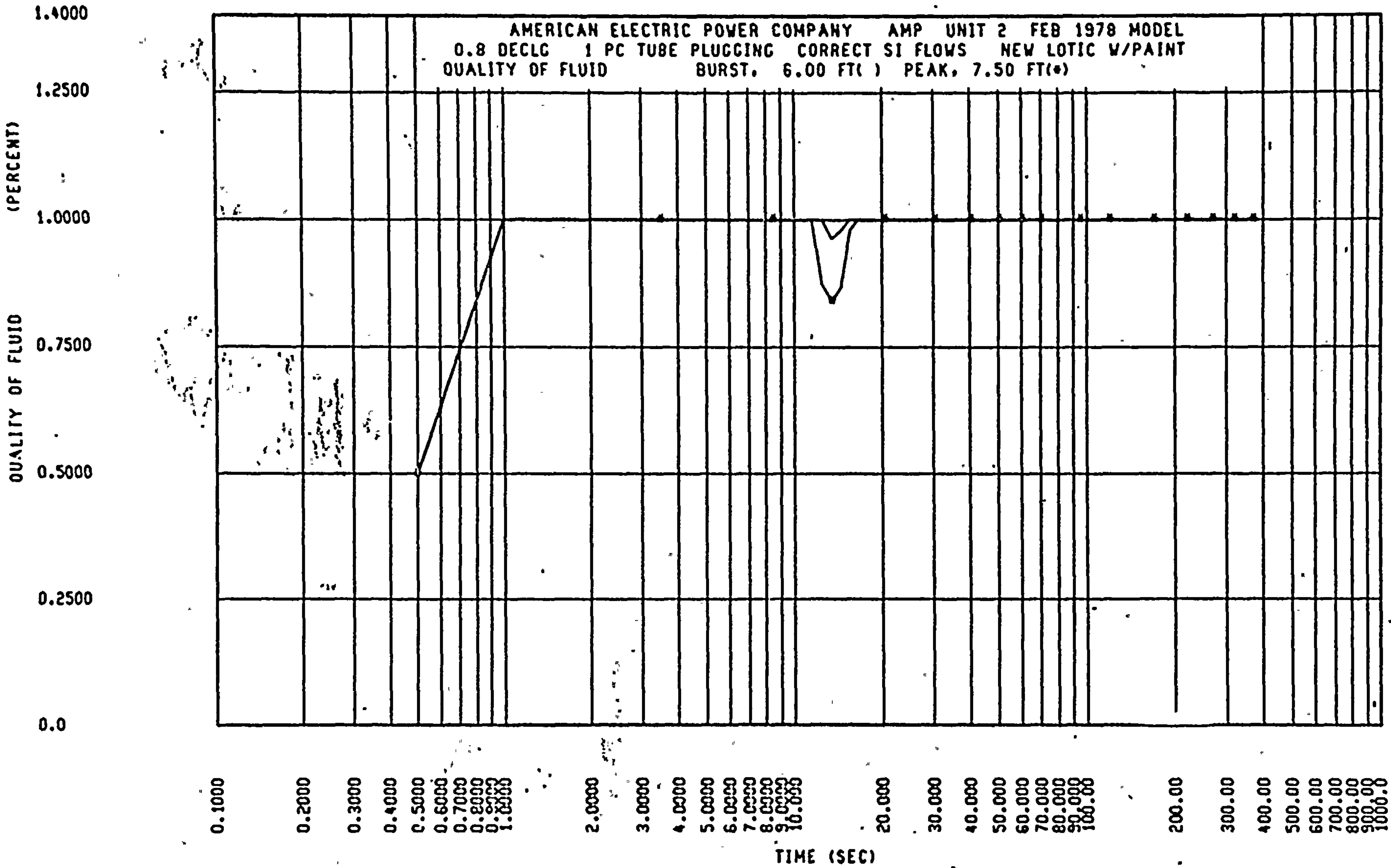


FIGURE 1 b FLUID QUALITY - DECLG(CD = 0.8)

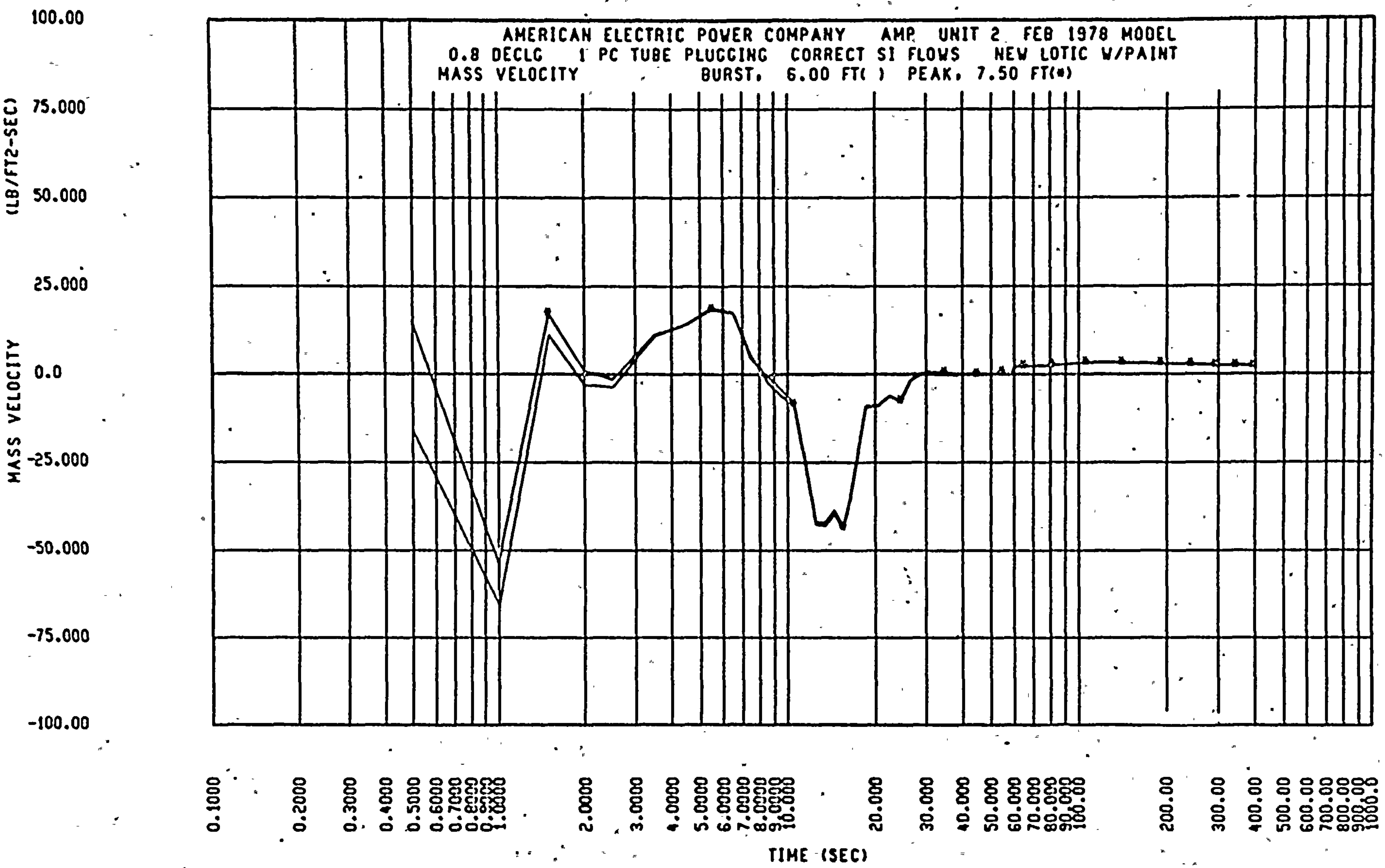


FIGURE 2 b MASS VELOCITY - DECLG(CD = 0.8)

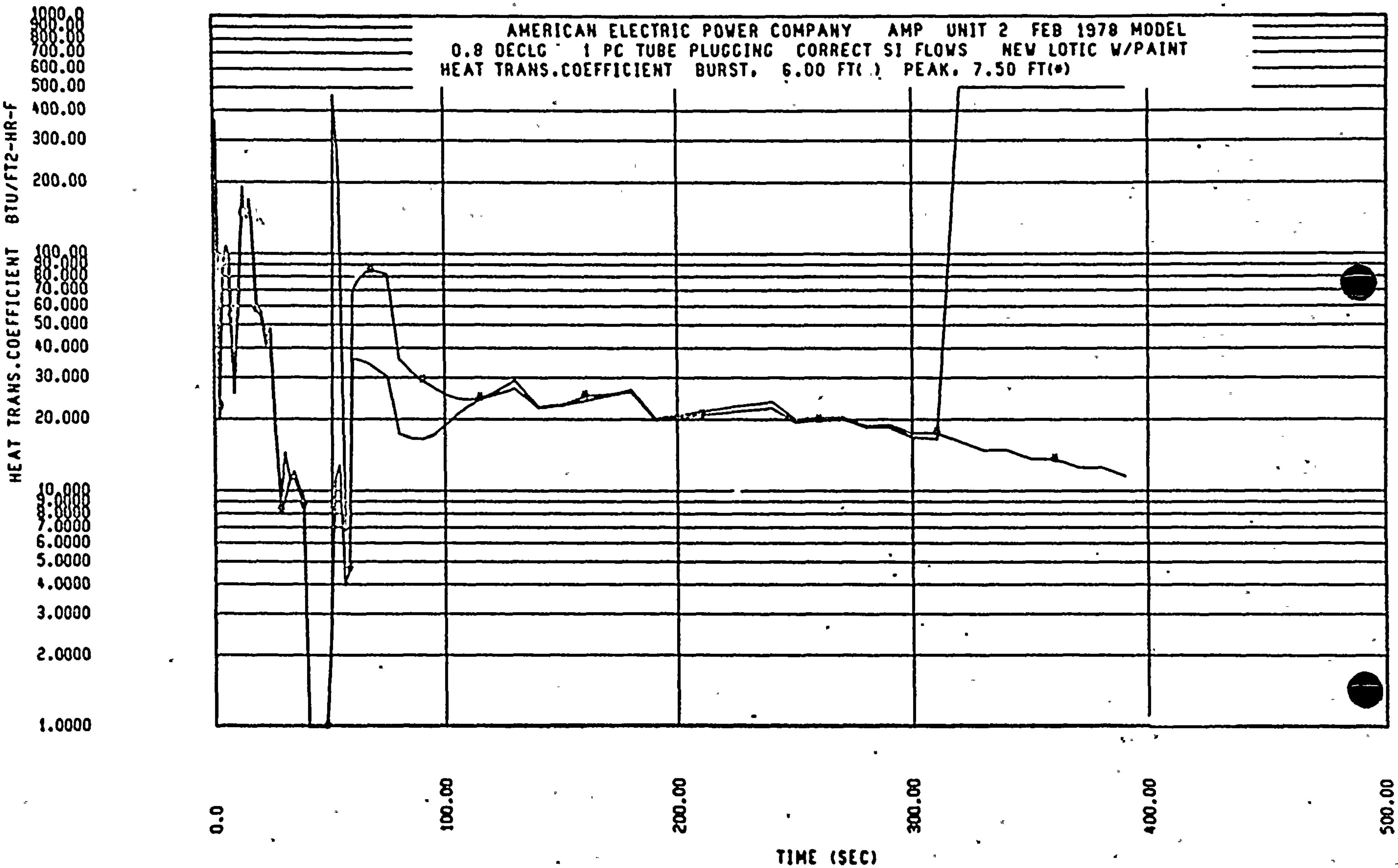


FIGURE 3 b HEAT TRANSFER COEFFICIENT - DECLG(CD = 0.8)

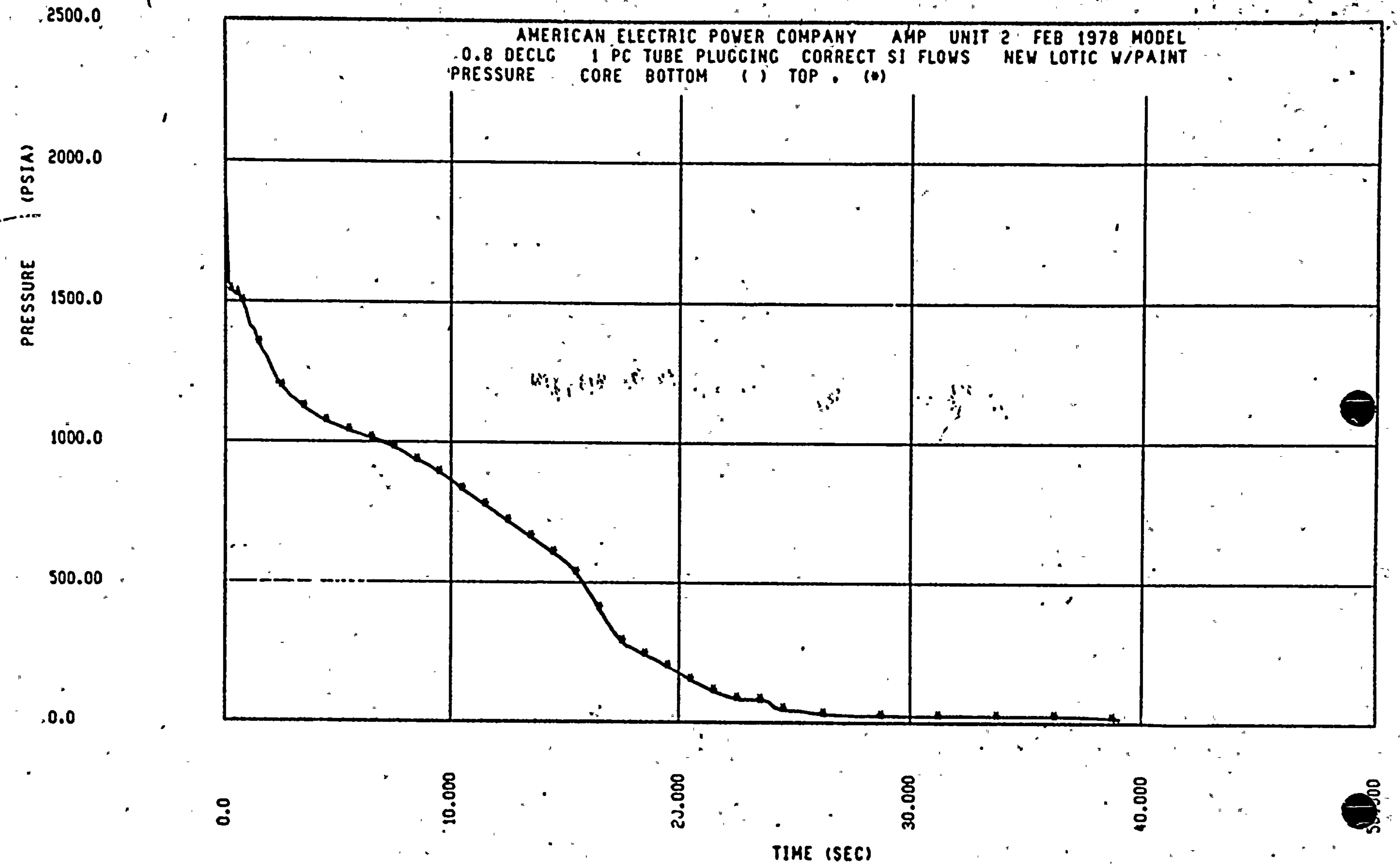


FIGURE 4 b CORE PRESSURE - DECLG(CD = 0.8)

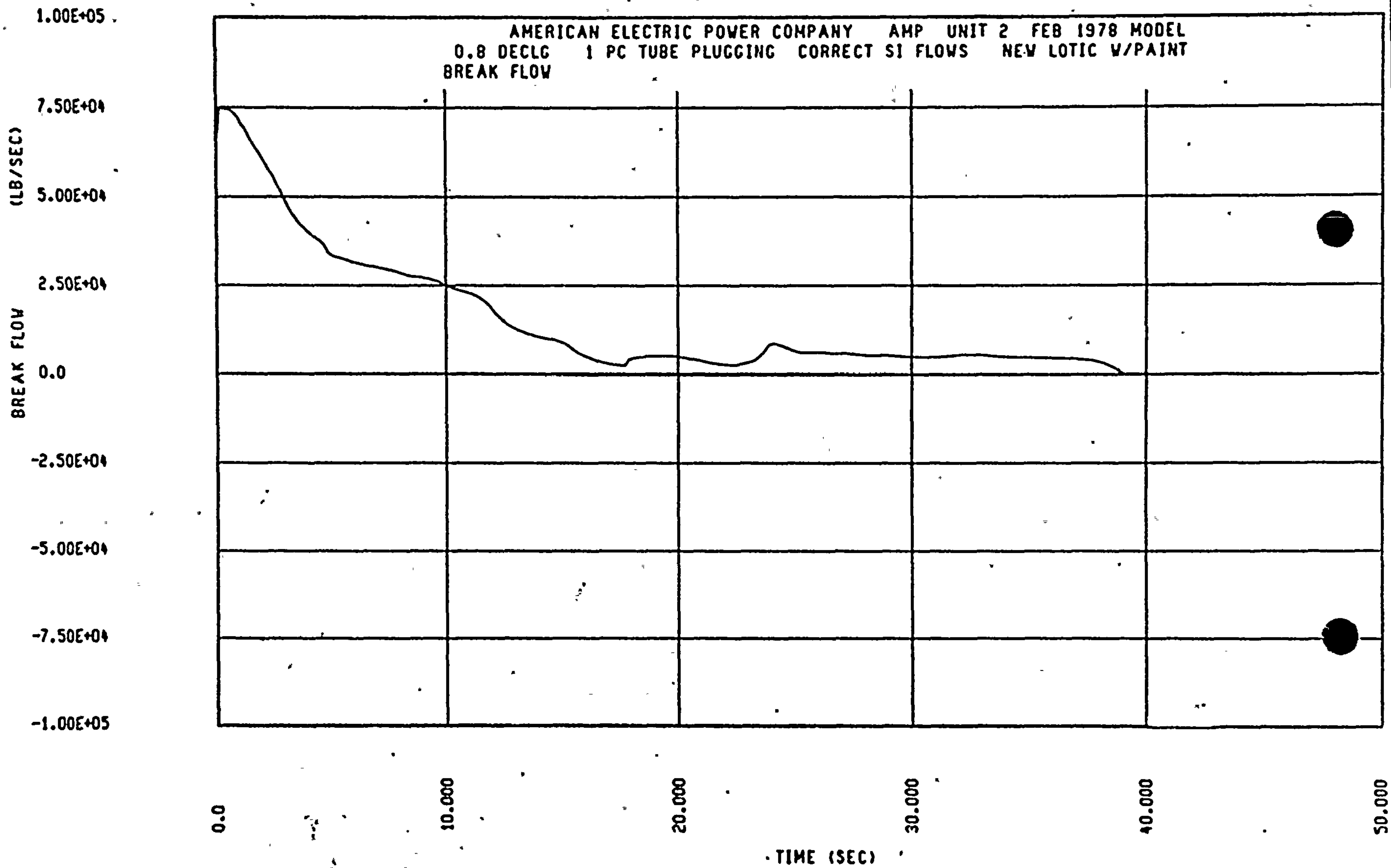


FIGURE 5 b BREAK FLOW RATE - DECLG(CD = 0.8)

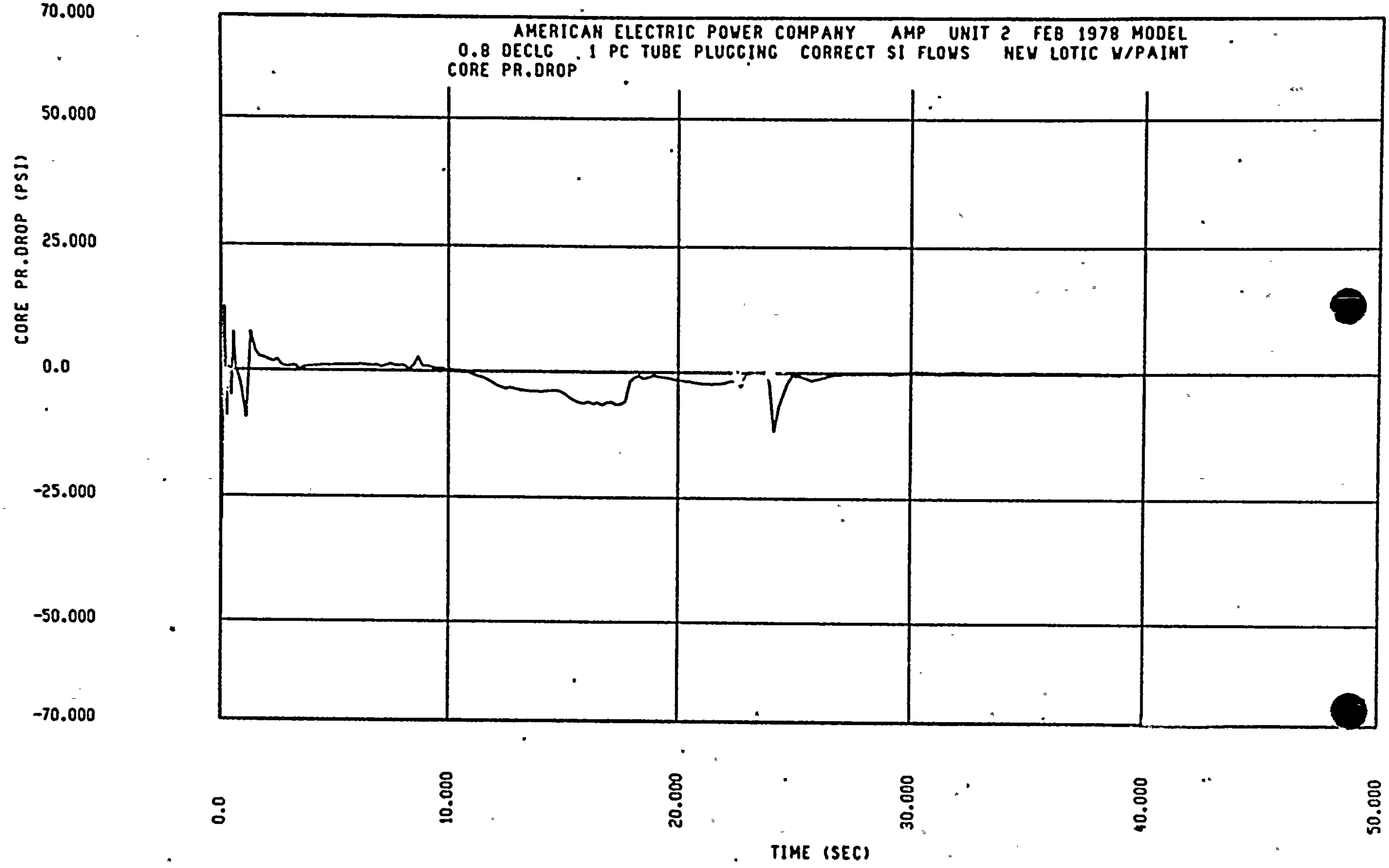


FIGURE 6 b CORE PRESSURE DROP - DECLG(CD = 0.8)

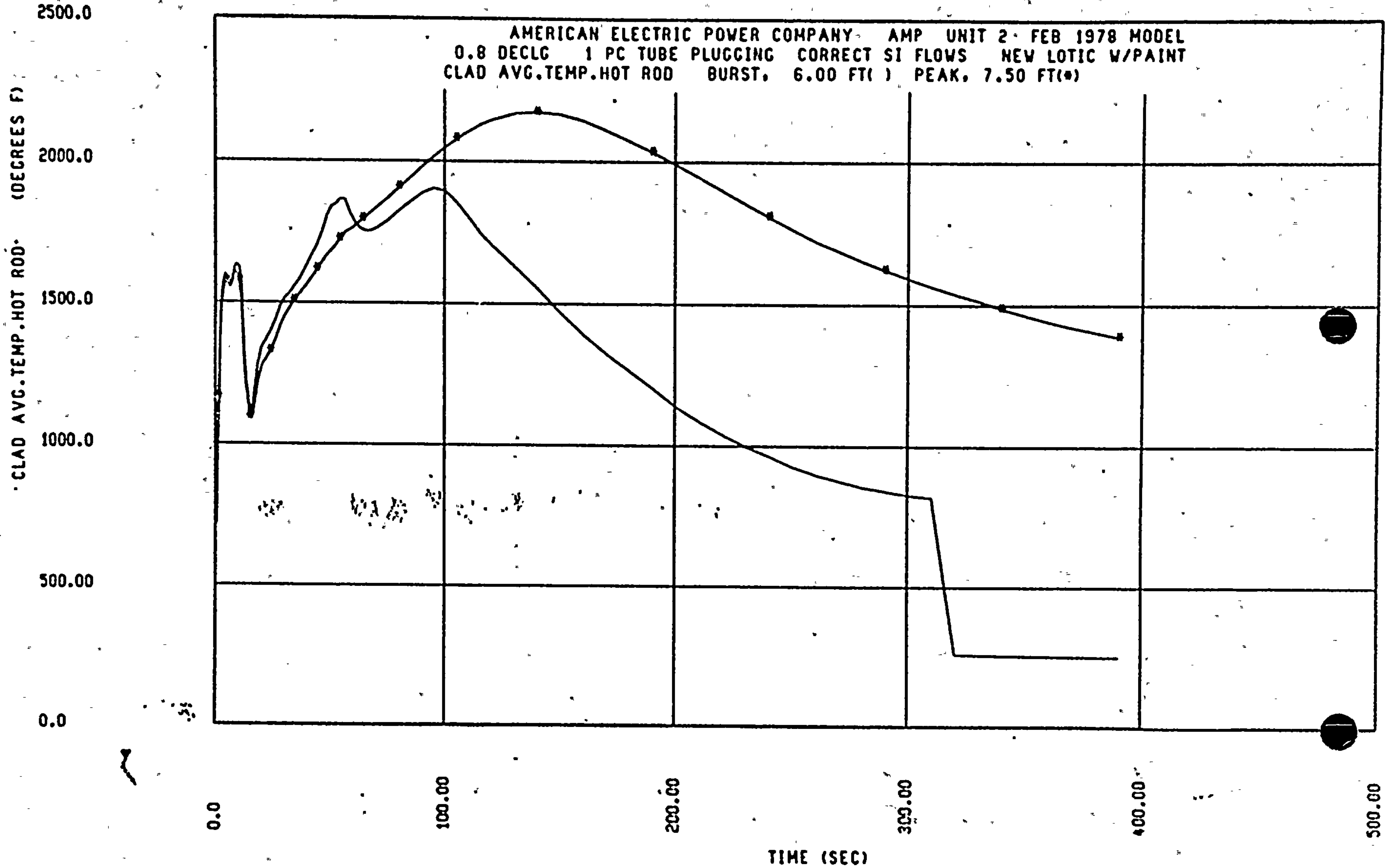


FIGURE 7 b - PEAK CLAD TEMPERATURE - DECLG(CD = 0.8)

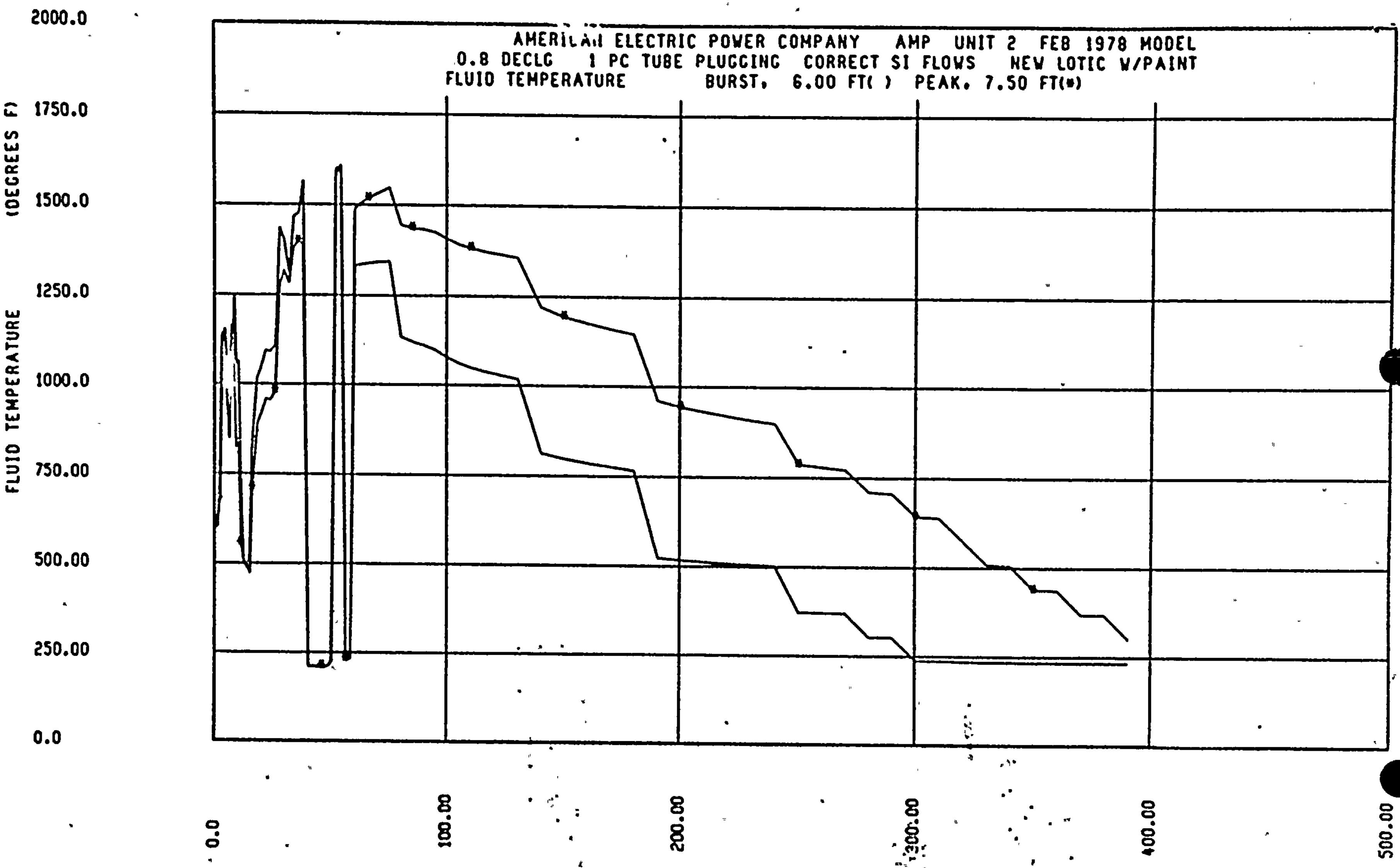


FIGURE 8 b FLUID TEMPERATURE - DECLG(CD = 0.8)

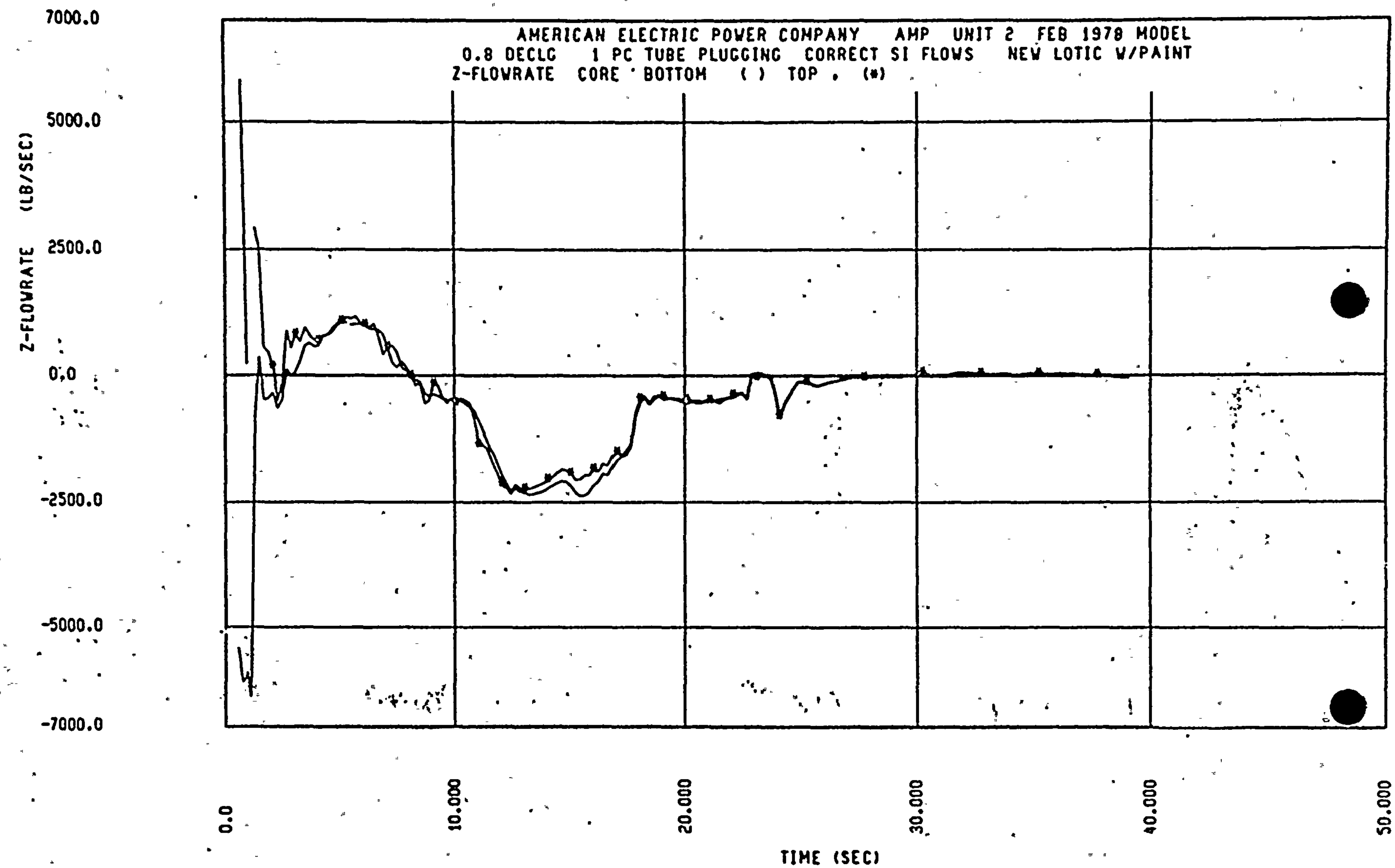


FIGURE 9 b CORE FLOW (TOP AND BOTTOM) - DECLG(CD = 0.8)

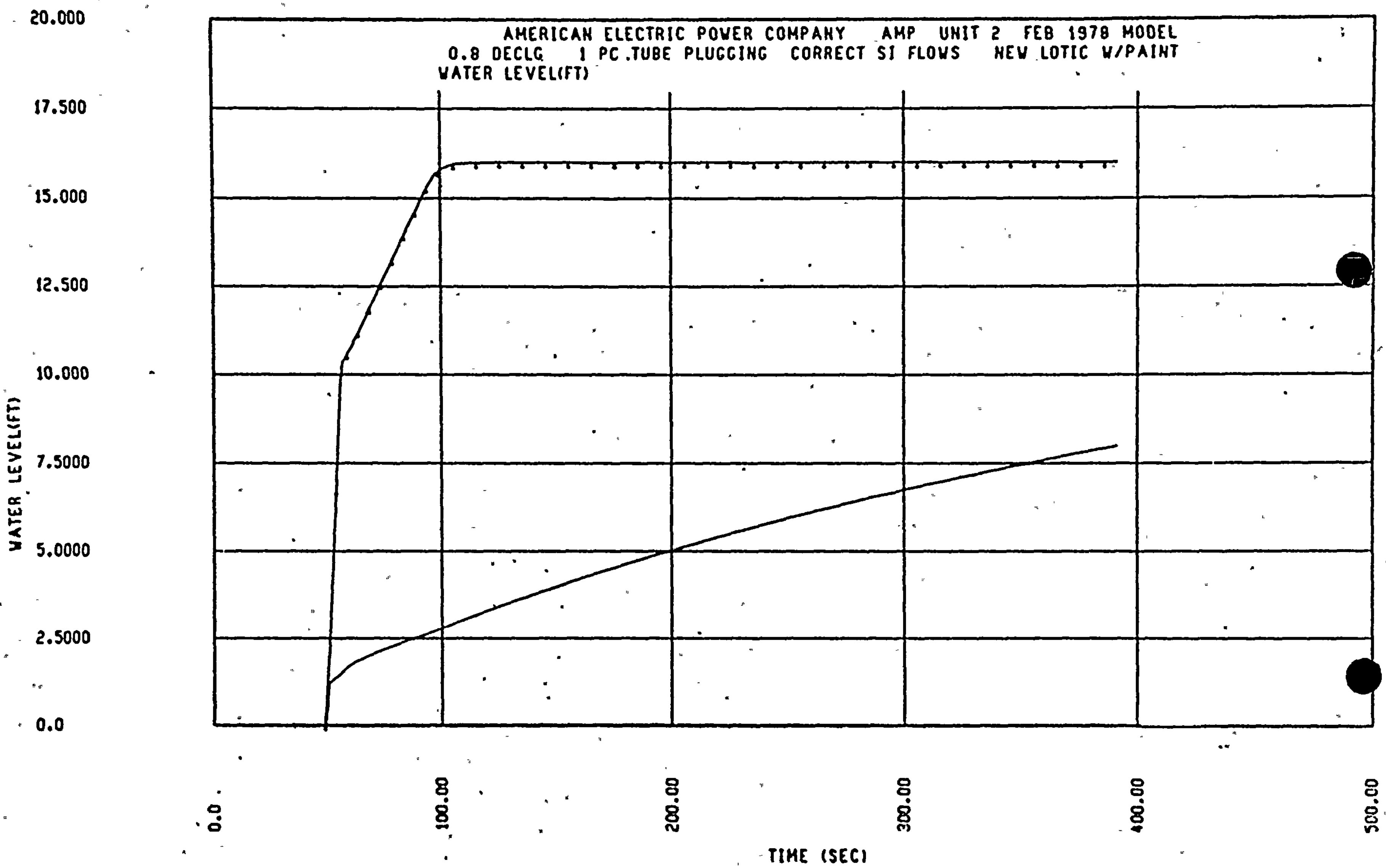


FIGURE 10 b REFLOOD TRANSIENT - DECLG(CD = 0.8)
DOWNCOMER AND CORE WATER LEVELS

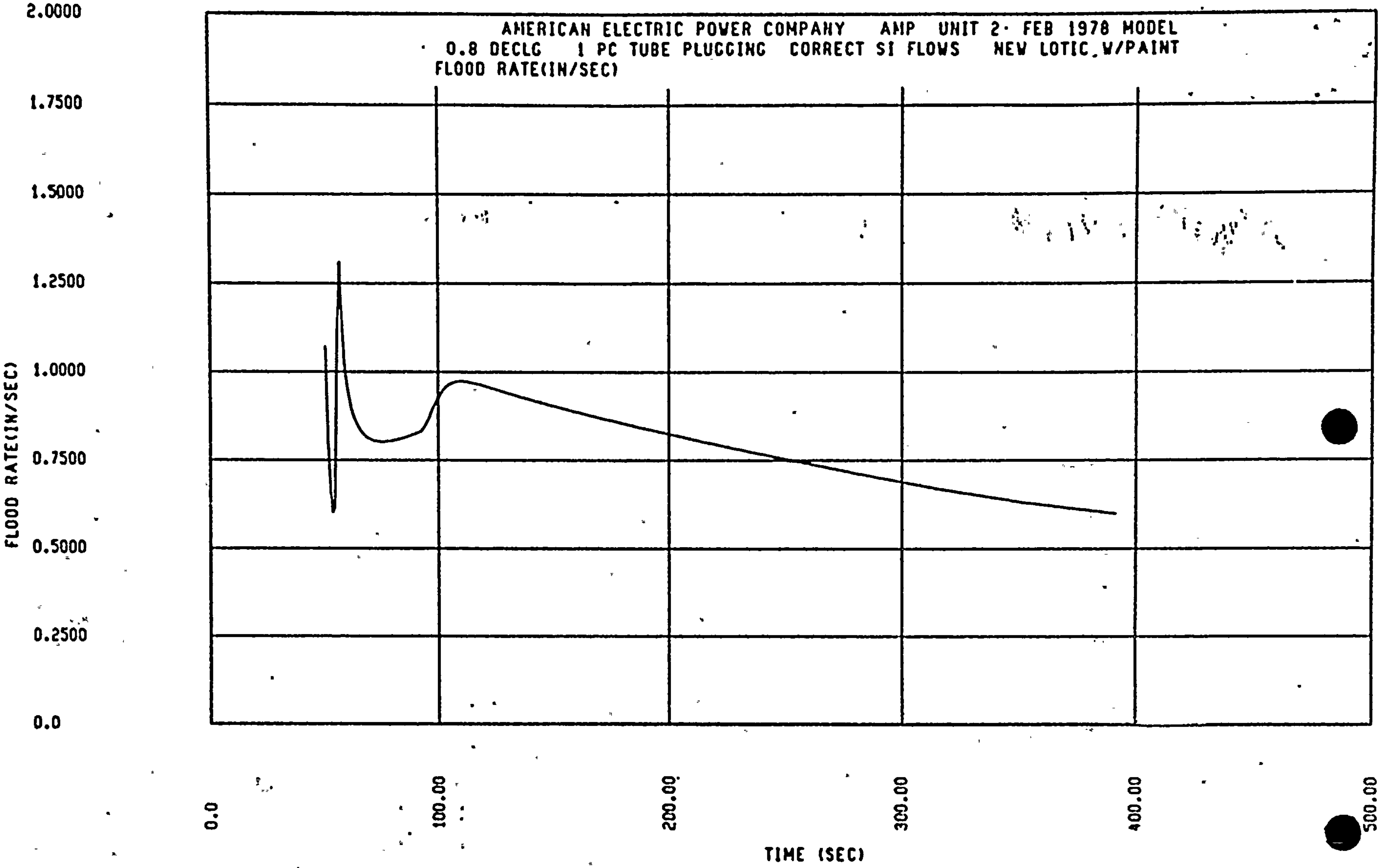


FIGURE 11 b REFLOOD TRANSIENT - DECLG(CD = 0.8)
CORE INLET VELOCITY

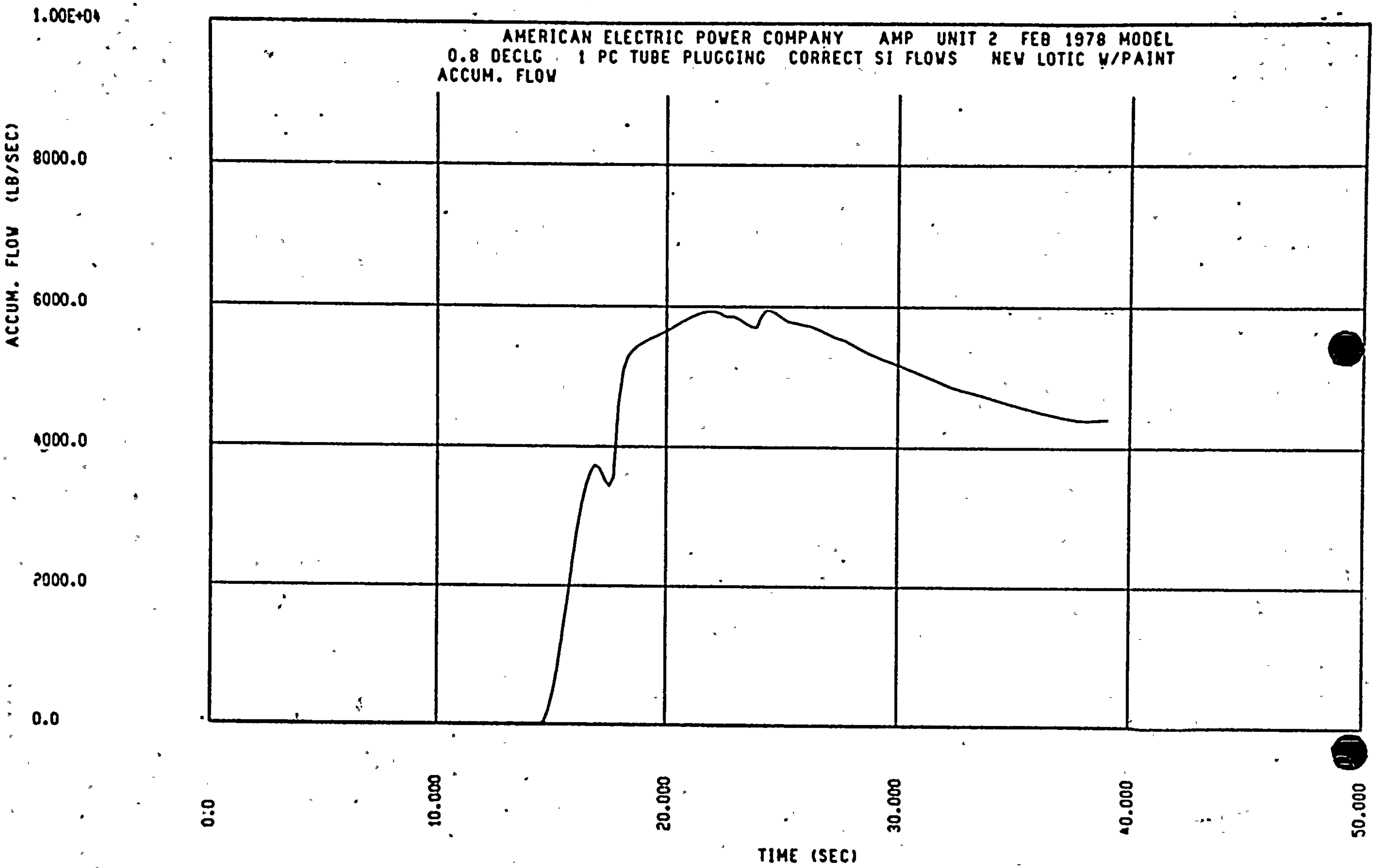


FIGURE 12 b ACCUMULATOR FLOW (BLOWDOWN) - DECLG(CD = 0.8)

46 1510

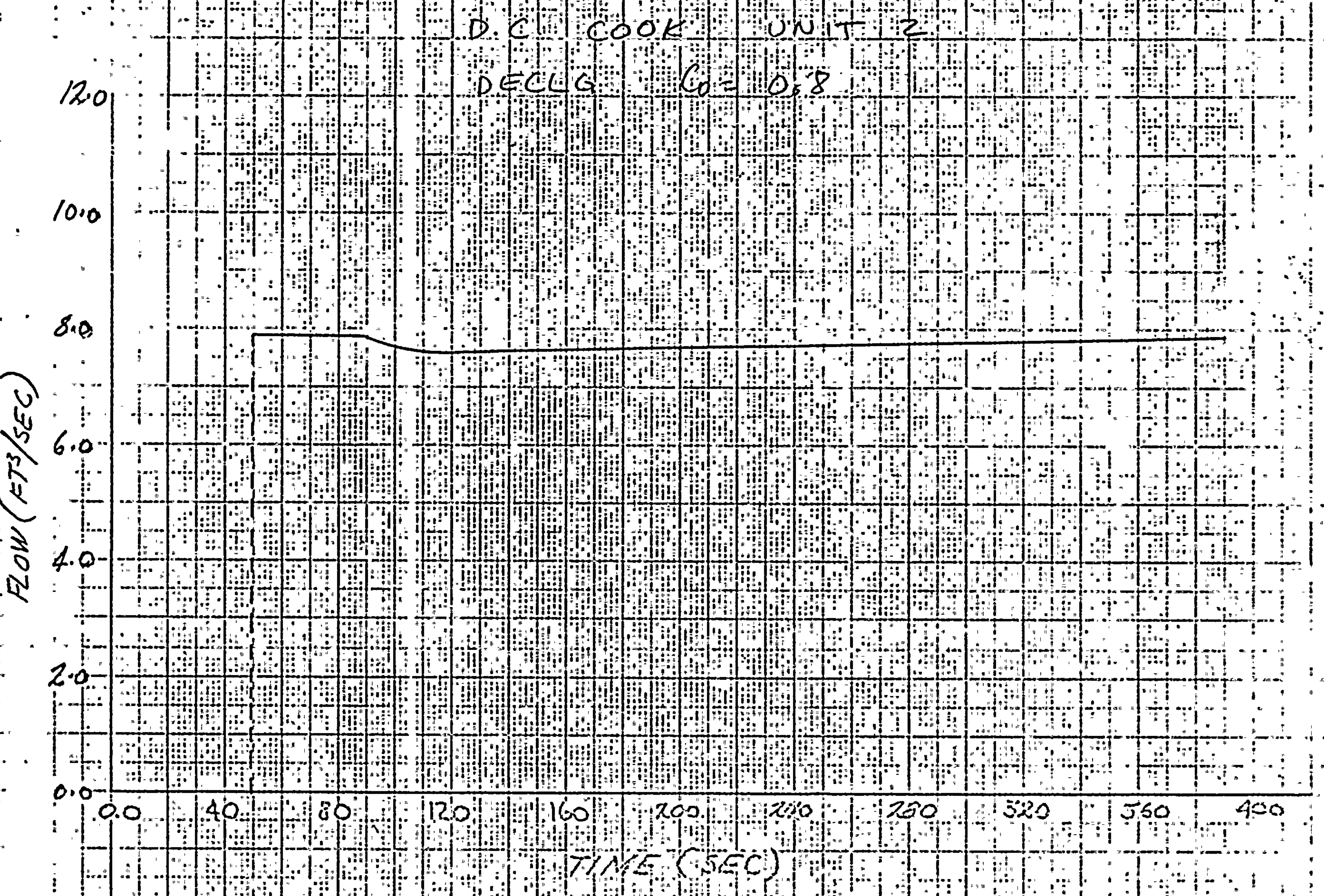


FIGURE 13 b PUMPED ECCS FLOW (REFLOOD) - DECLG(CD = 0.8)

D.C. COOK UNIT 2
DECLG $C_D = 0.8$

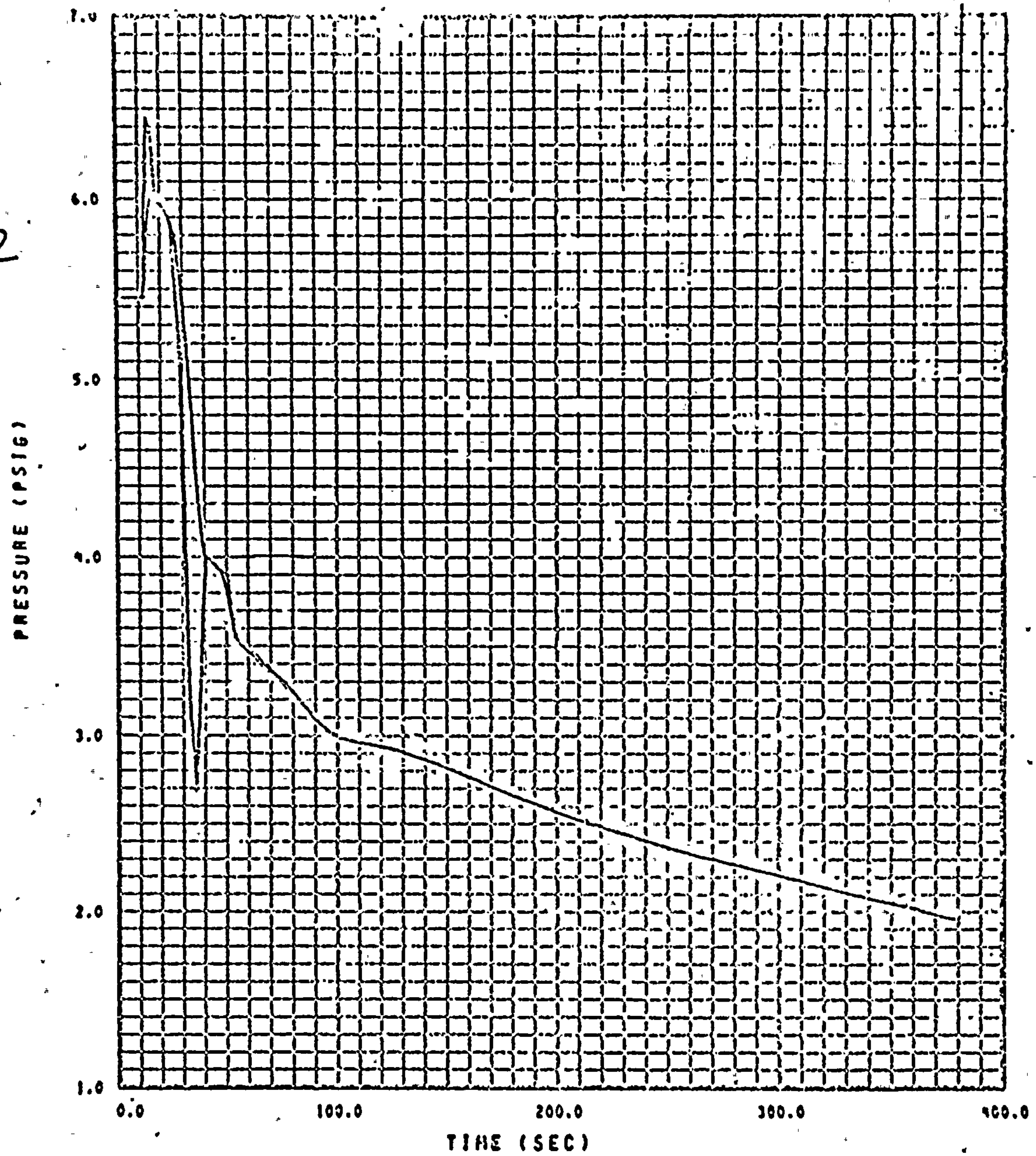


FIGURE 14 b CONTAINMENT PRESSURE - DECLG(CD = 0. 8)

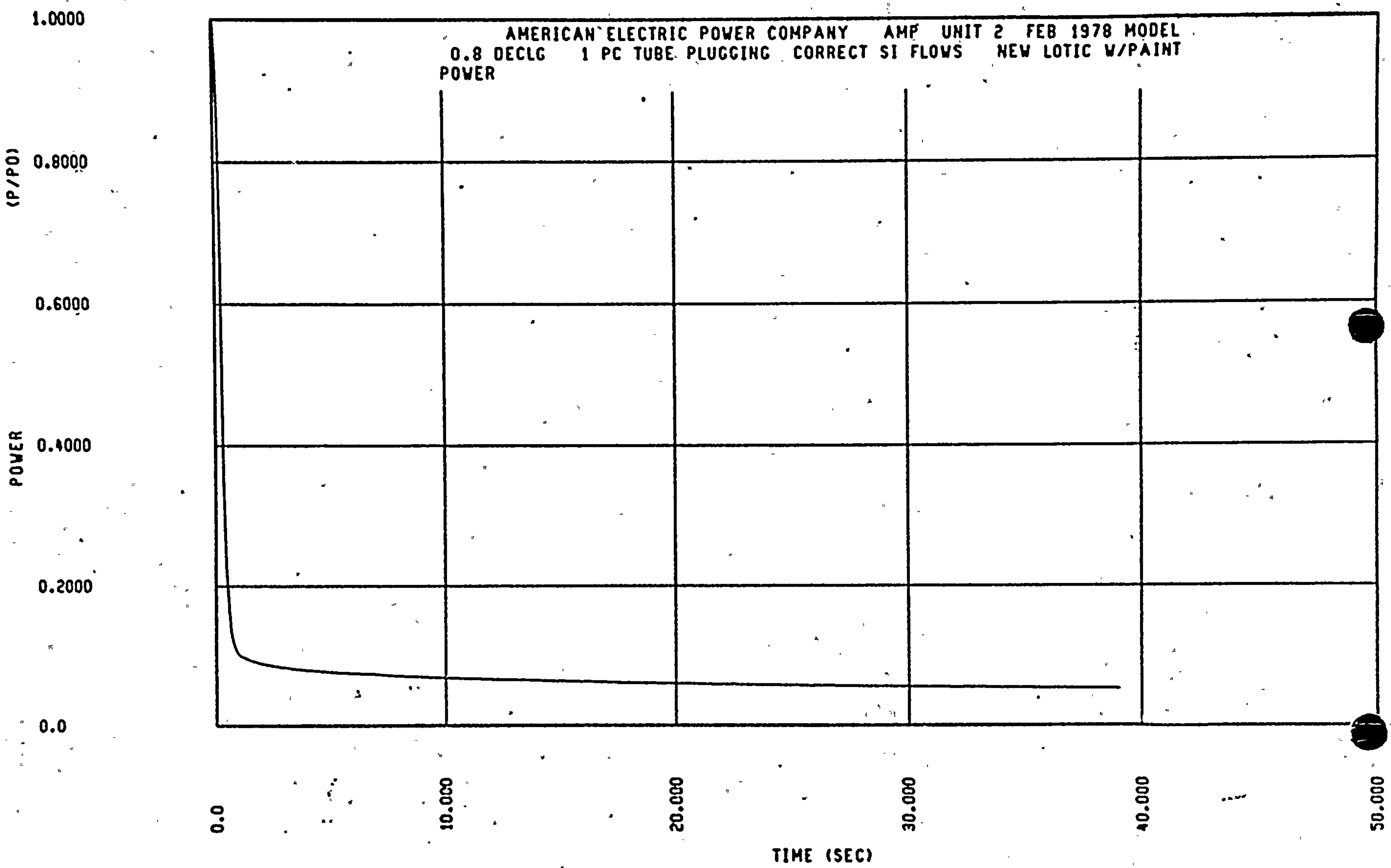


FIGURE 15 b CORE POWER TRANSIENT - DECLG(CD = 0.8)

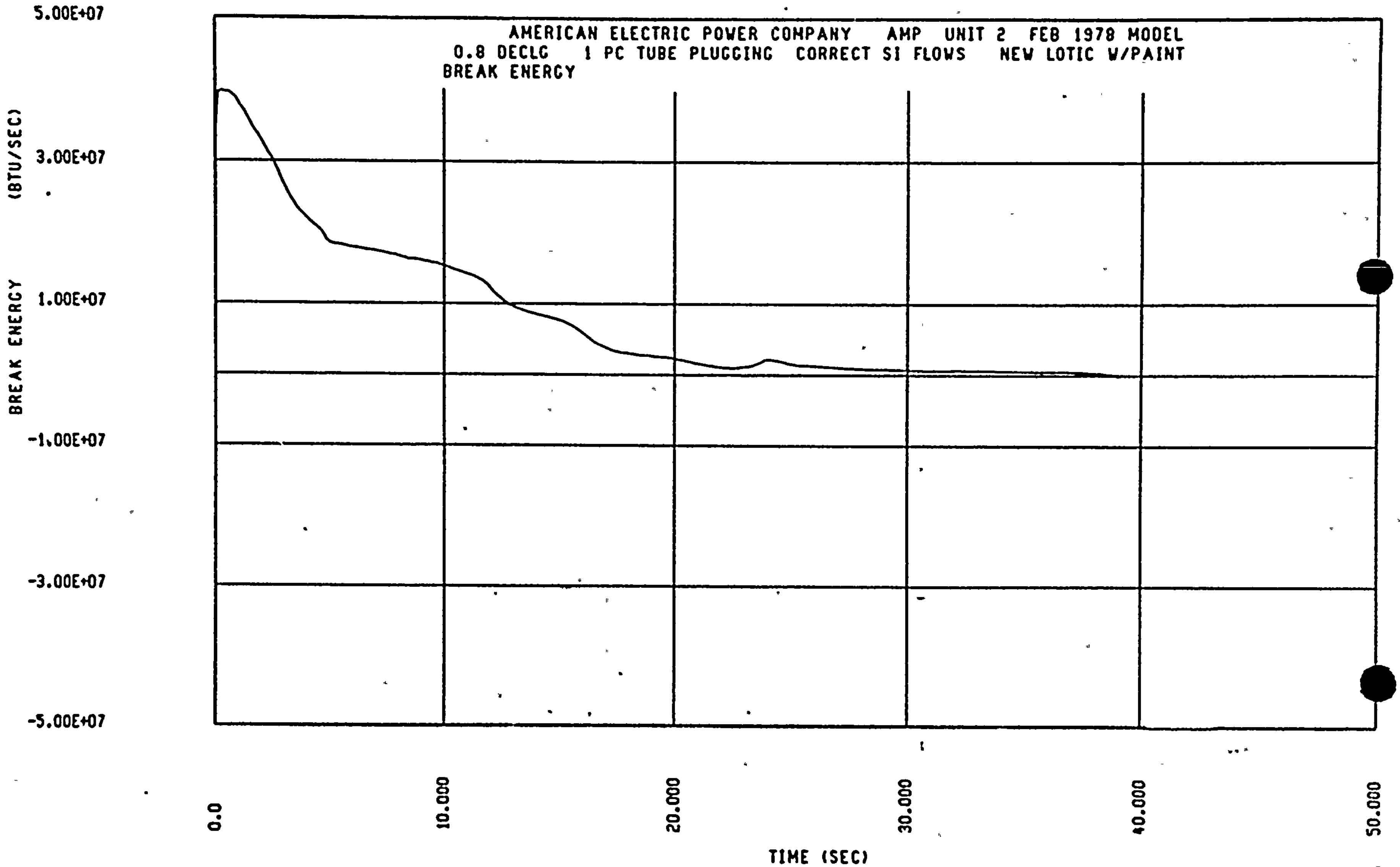


FIGURE 16 BREAK ENERGY RELEASED TO CONTAINMENT

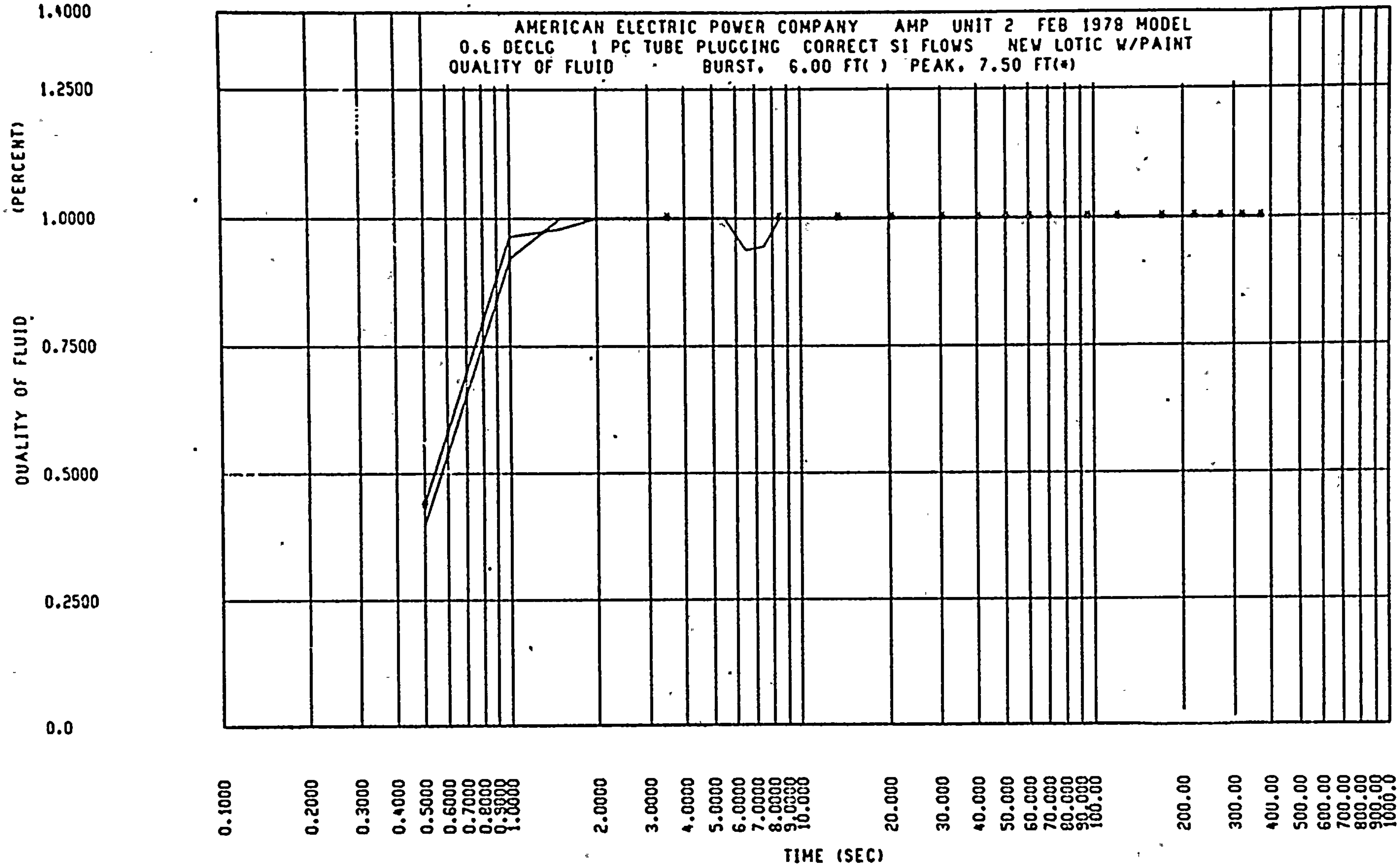


FIGURE 1 c FLUID QUALITY - DECLG(CD = 0.6)

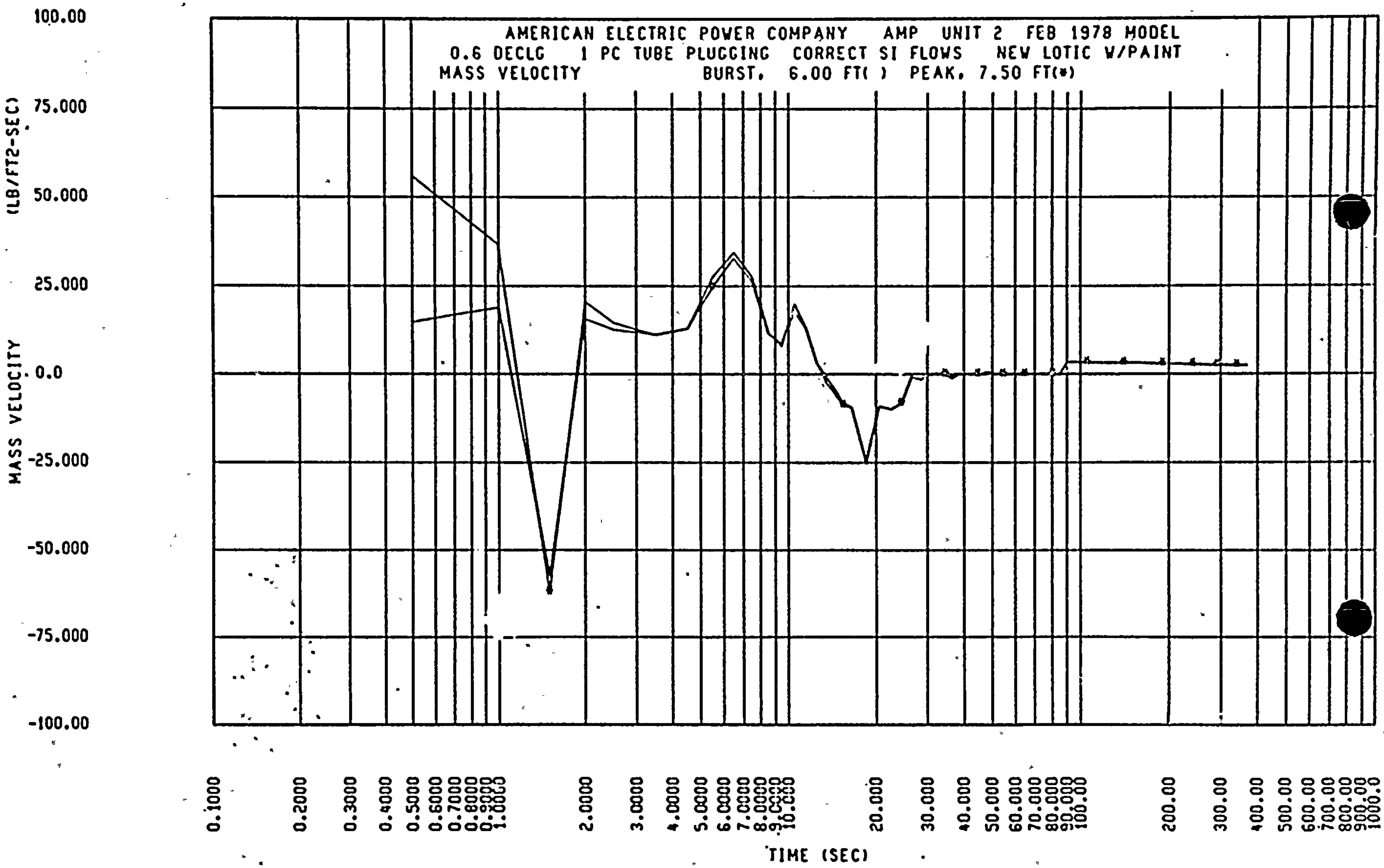


FIGURE 2 c MASS VELOCITY - DECLG(CD = 0.6)

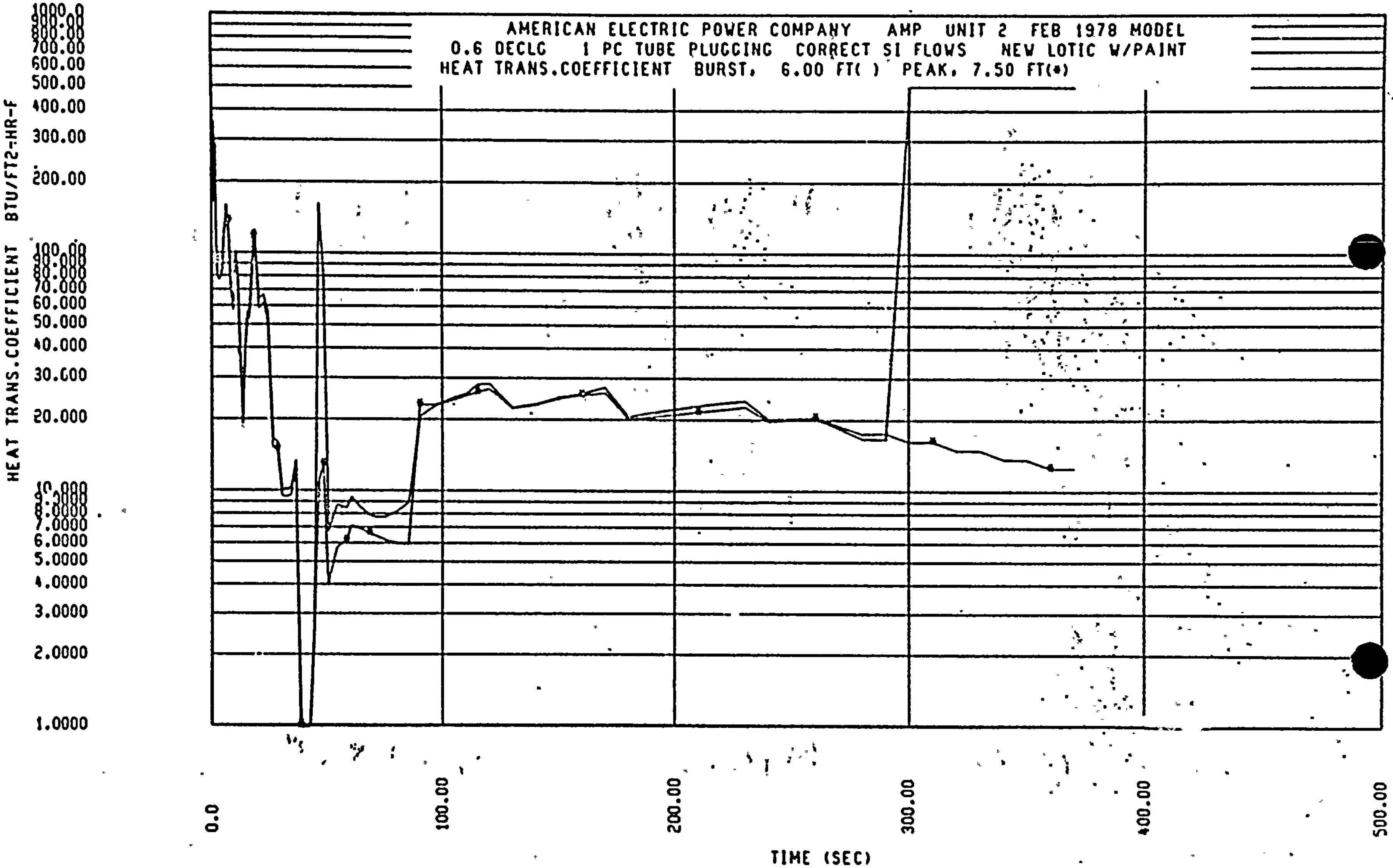


FIGURE 3 c HEAT TRANSFER COEFFICIENT - DECLG(CD = 0.6)

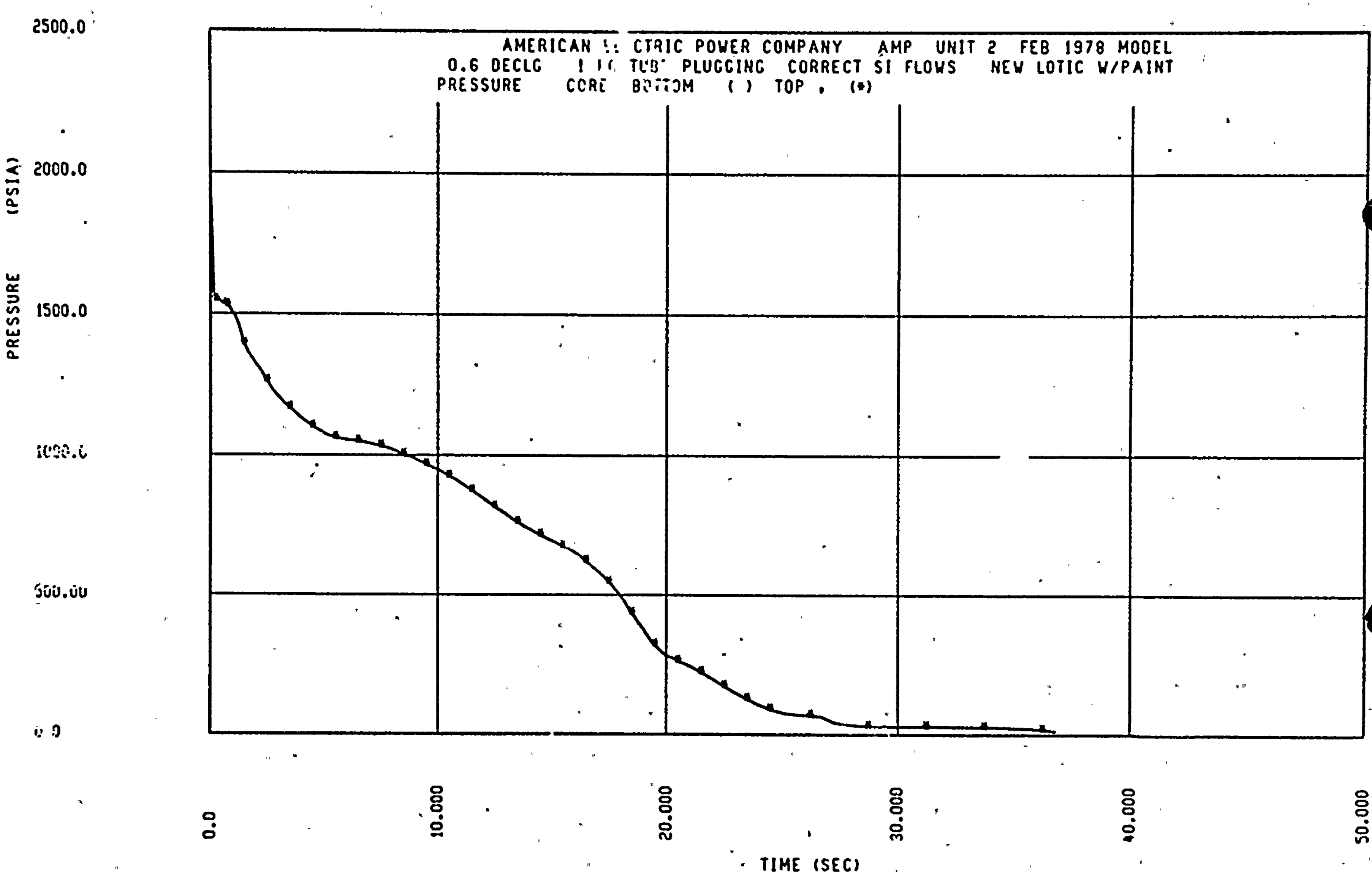


FIGURE 4 c CORE PRESSURE - DECLG(CD = 0.6)

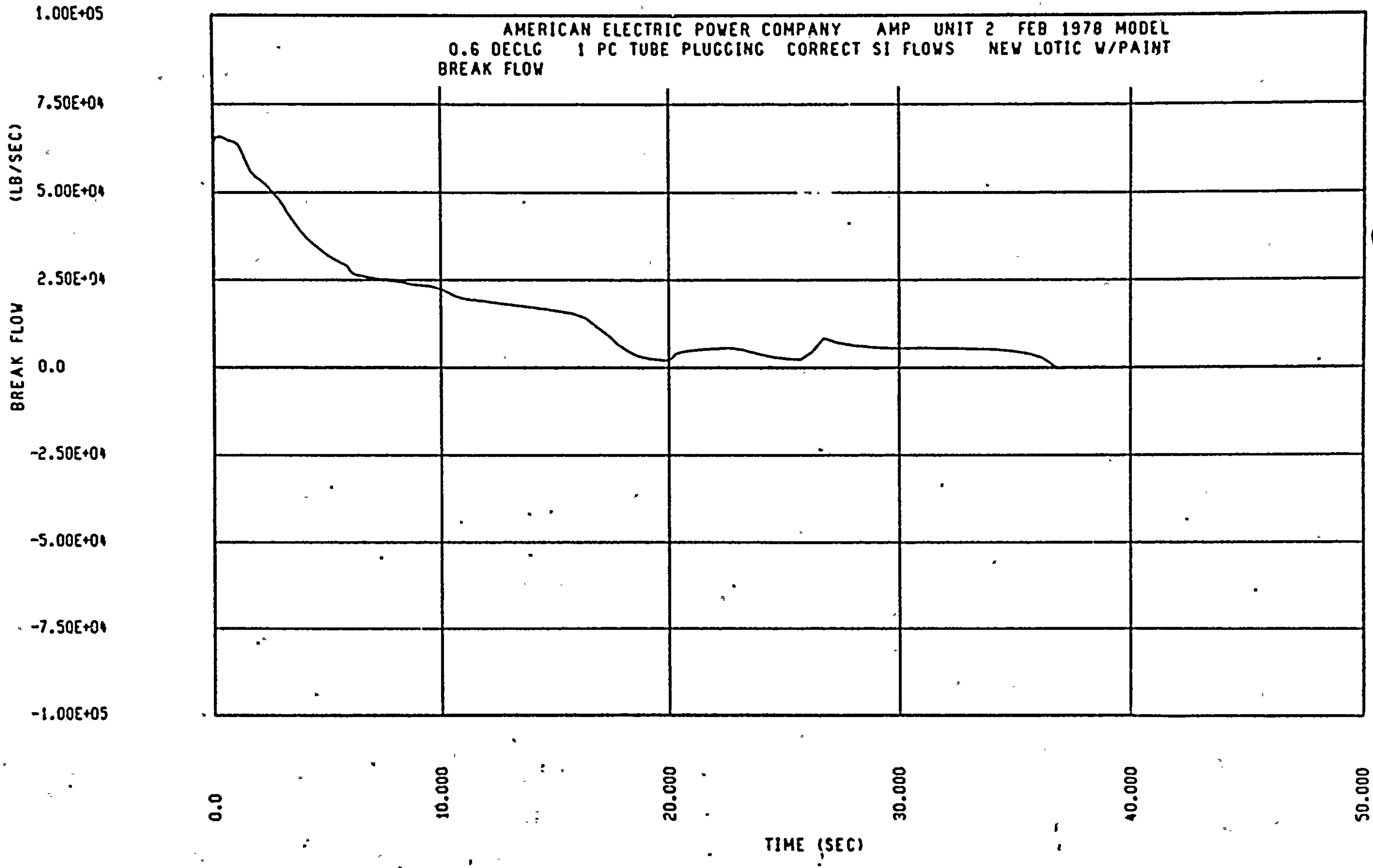


FIGURE 5c. BREAK FLOW RATE -- DECLG(CD = 0.6)

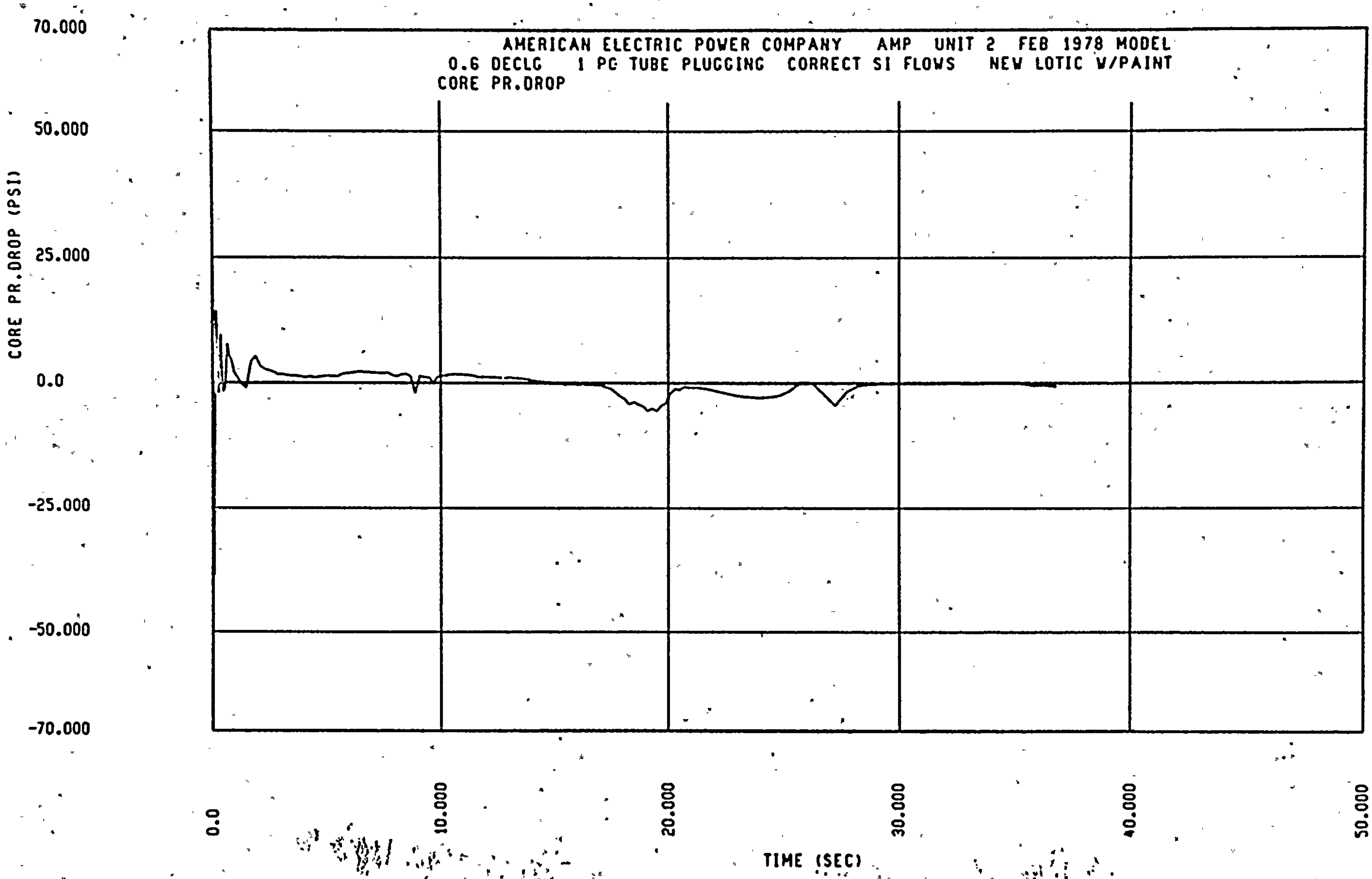


FIGURE 6 c CORE PRESSURE DROP - DECLG(CD = 0.6)

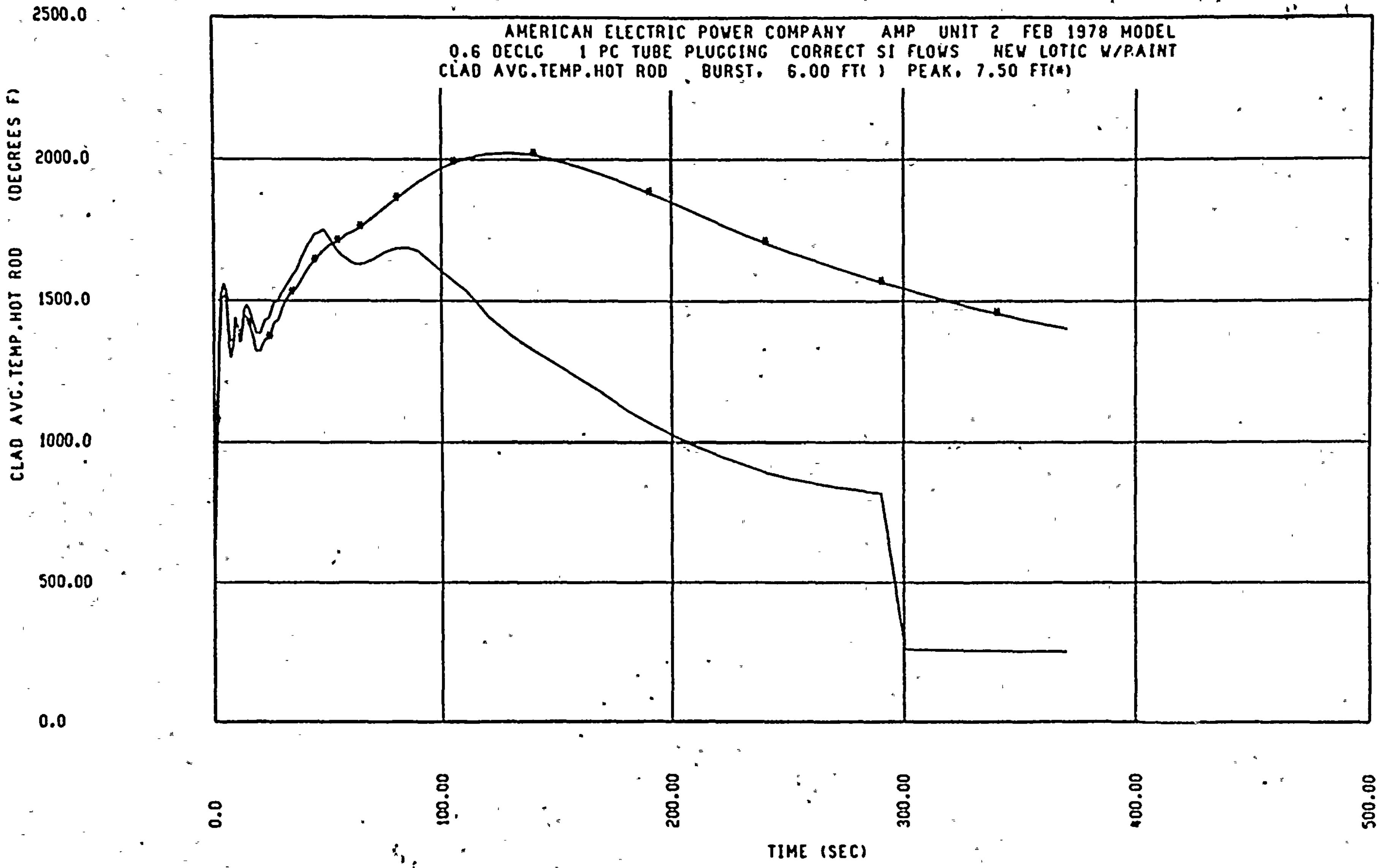


FIGURE 7 c - PEAK CLAD TEMPERATURE - DECLG(CD = 0.6)

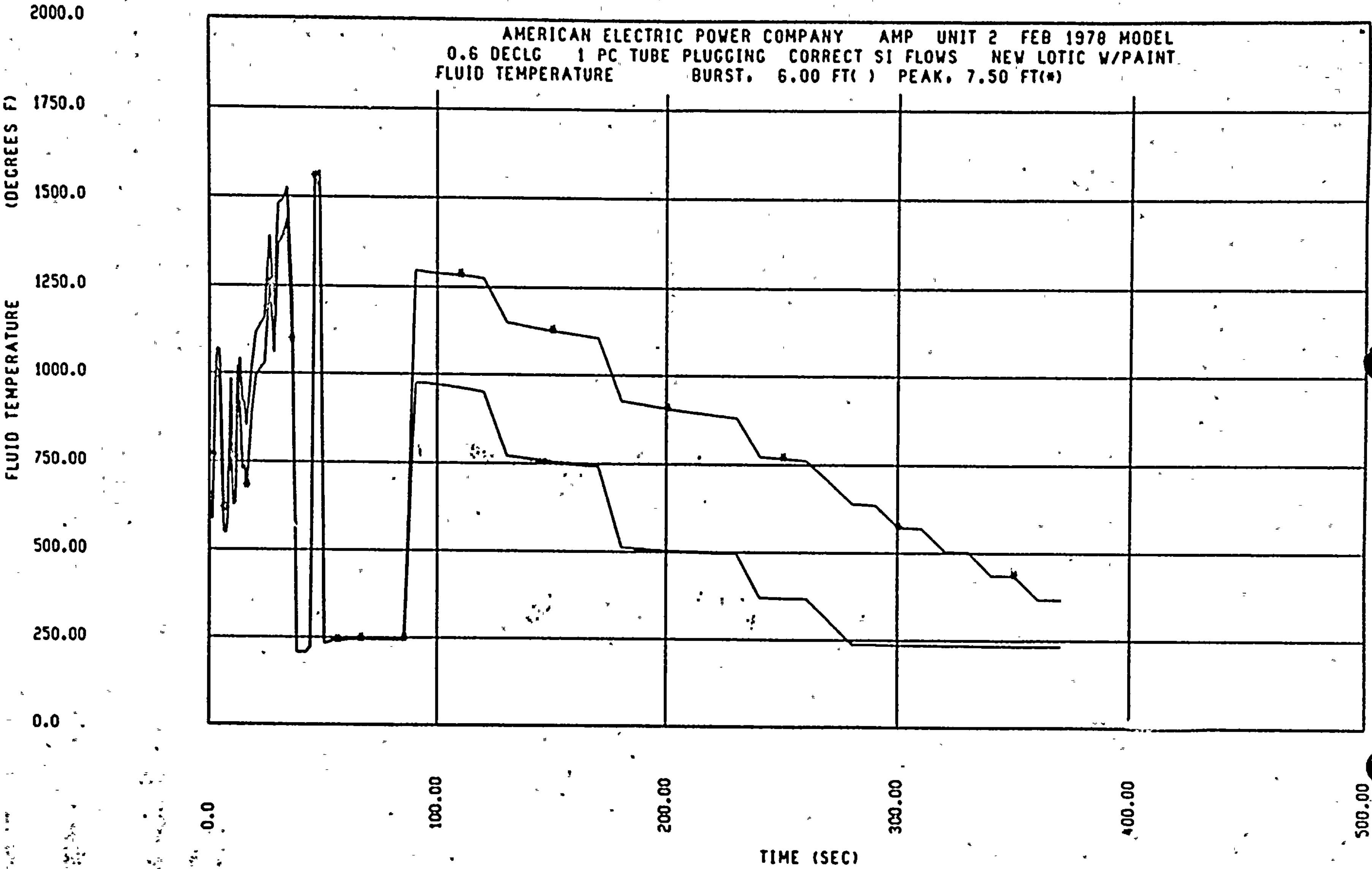


FIGURE 8 C FLUID TEMPERATURE - DECLG(CD = 0.6)

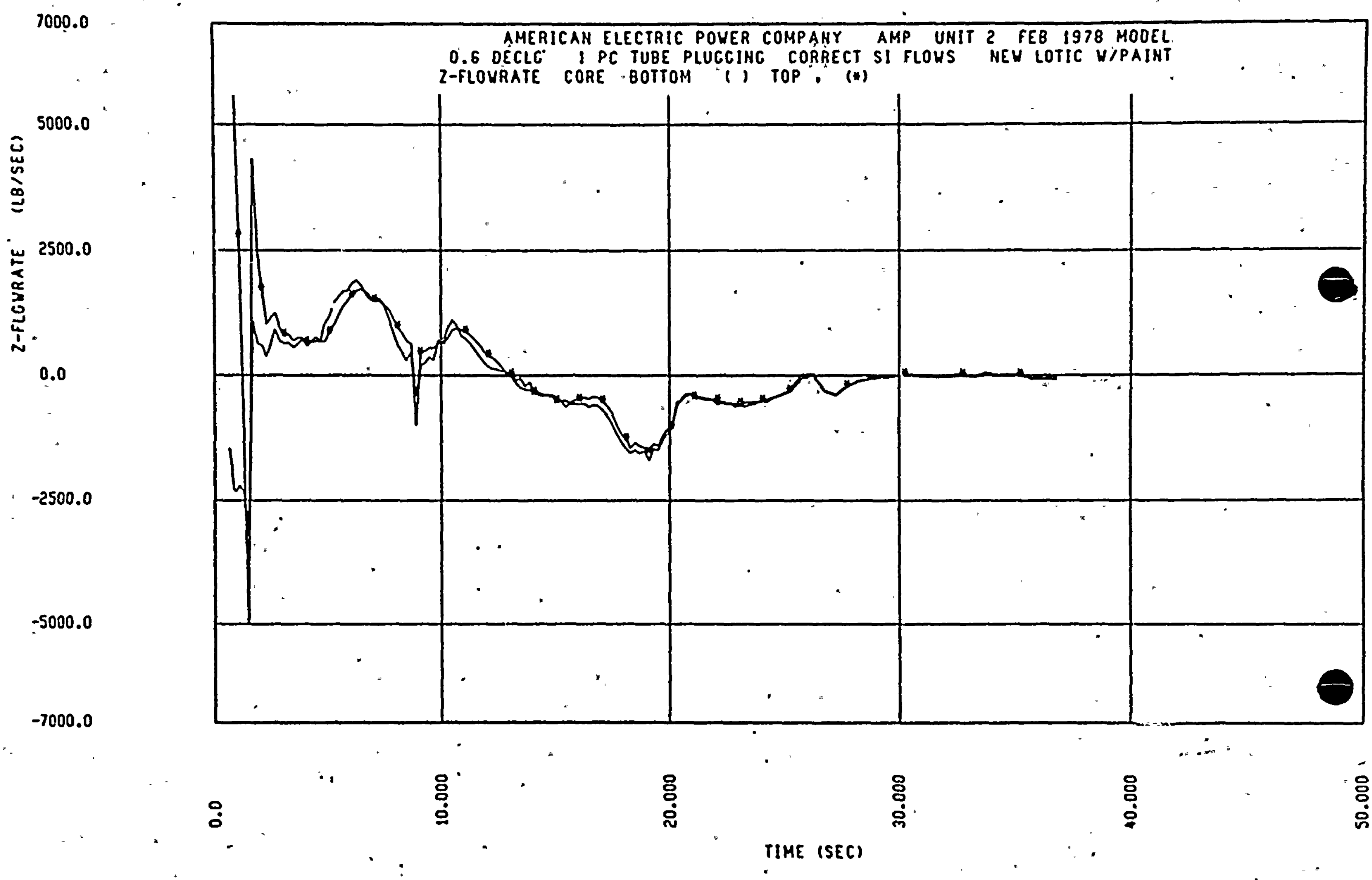


FIGURE 9c CORE FLOW (TOP AND BOTTOM) - DECLG(CD = 0.6)

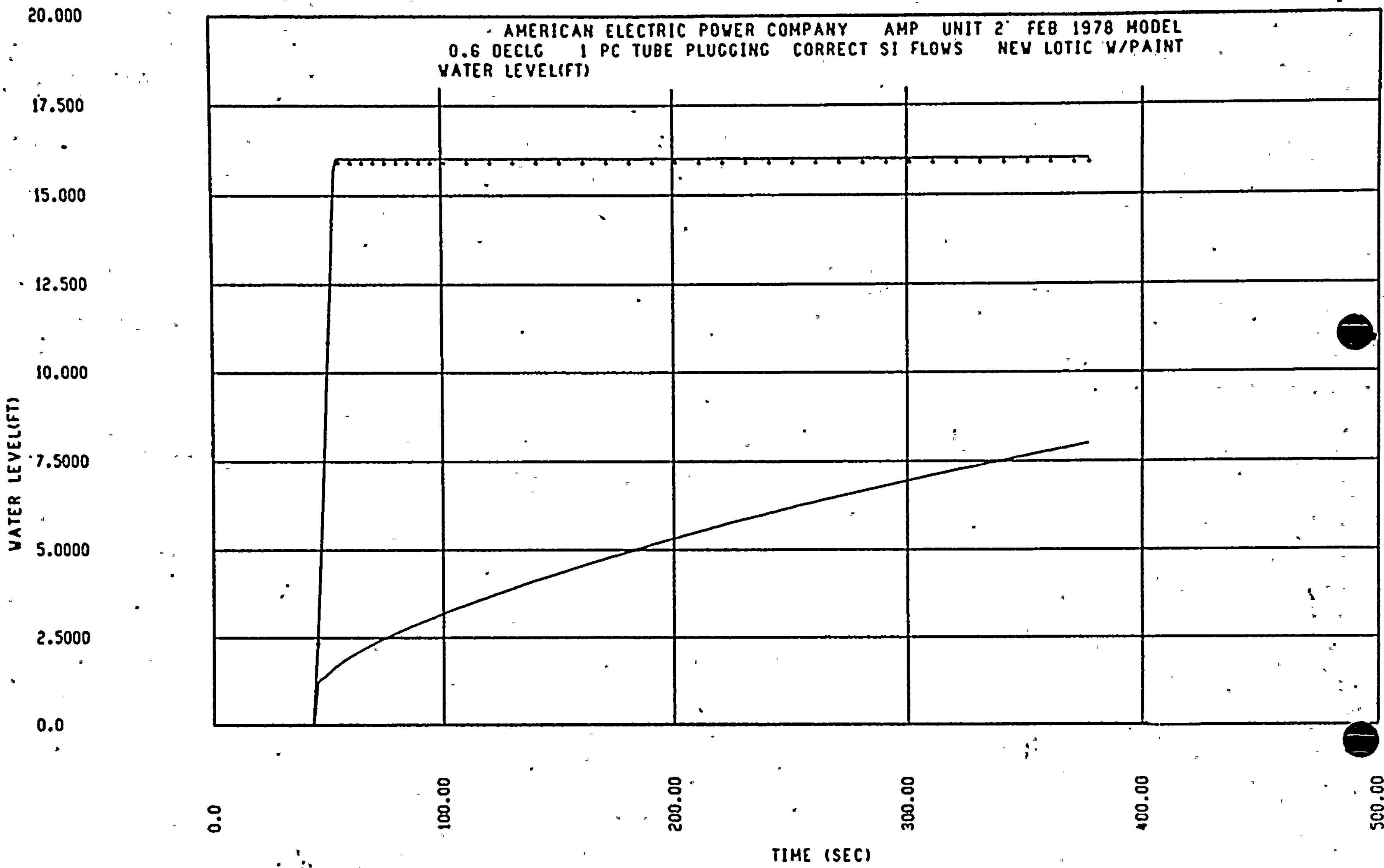


FIGURE 10 c REFLOOD TRANSIENT - DECLG(CD = 0.6)
DOWNCOMER AND CORE WATER LEVELS

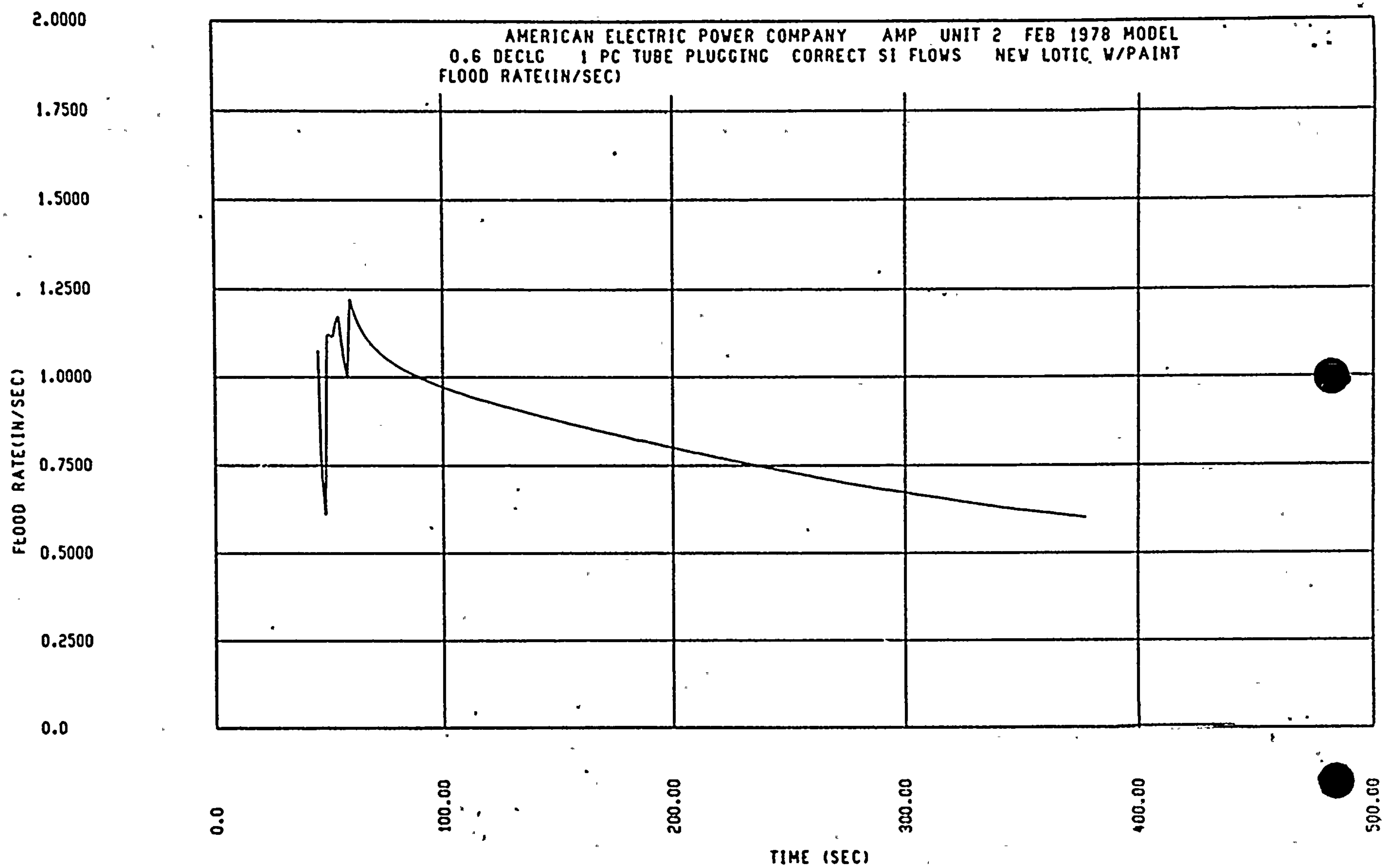


FIGURE 11 c REFLOOD TRANSIENT - DECLG(CD = 0.6)
CORE INLET VELOCITY

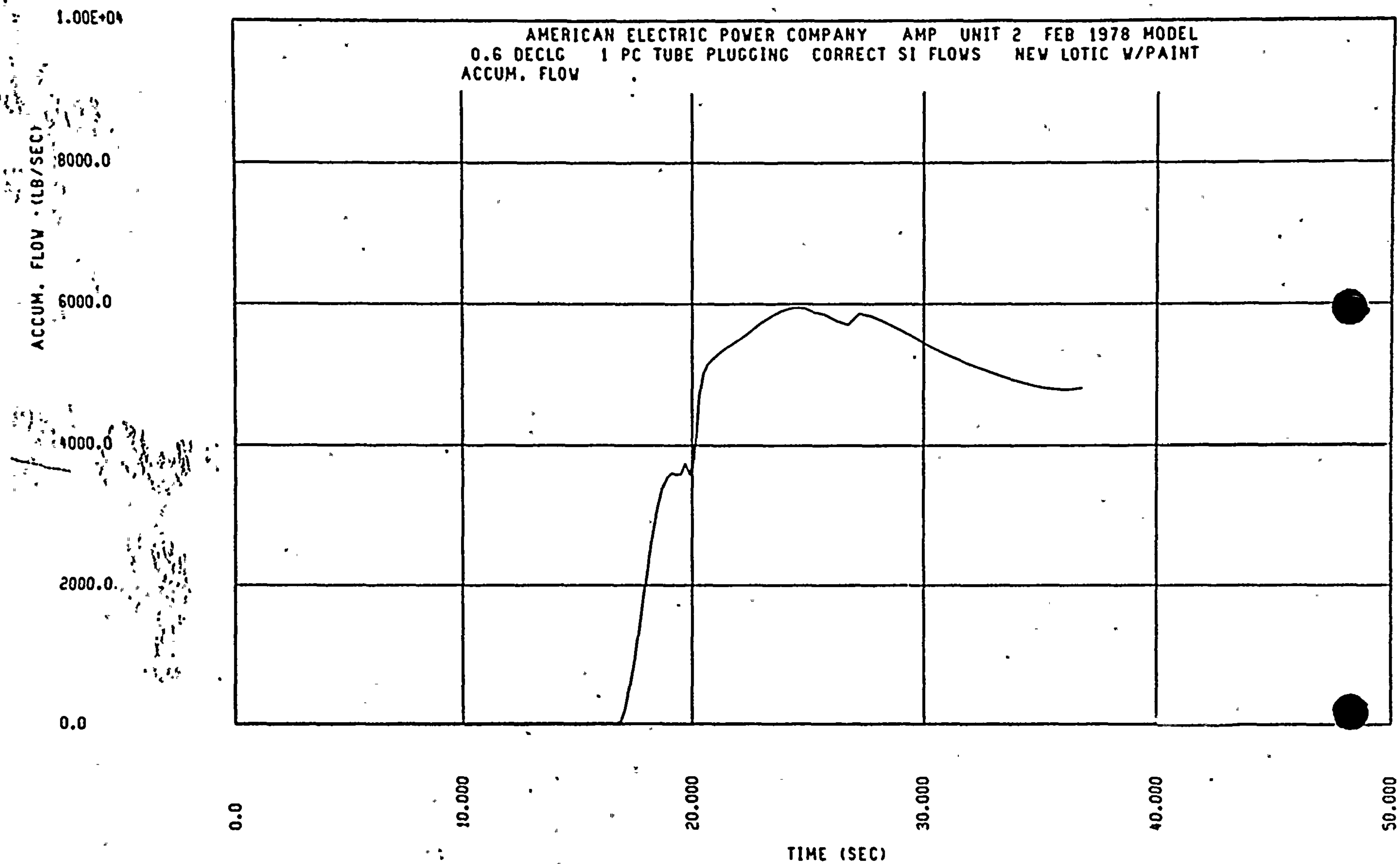


FIGURE 12 c ACCUMULATOR FLOW (BLOWDOWN) - DECLG(CD = 0.6)

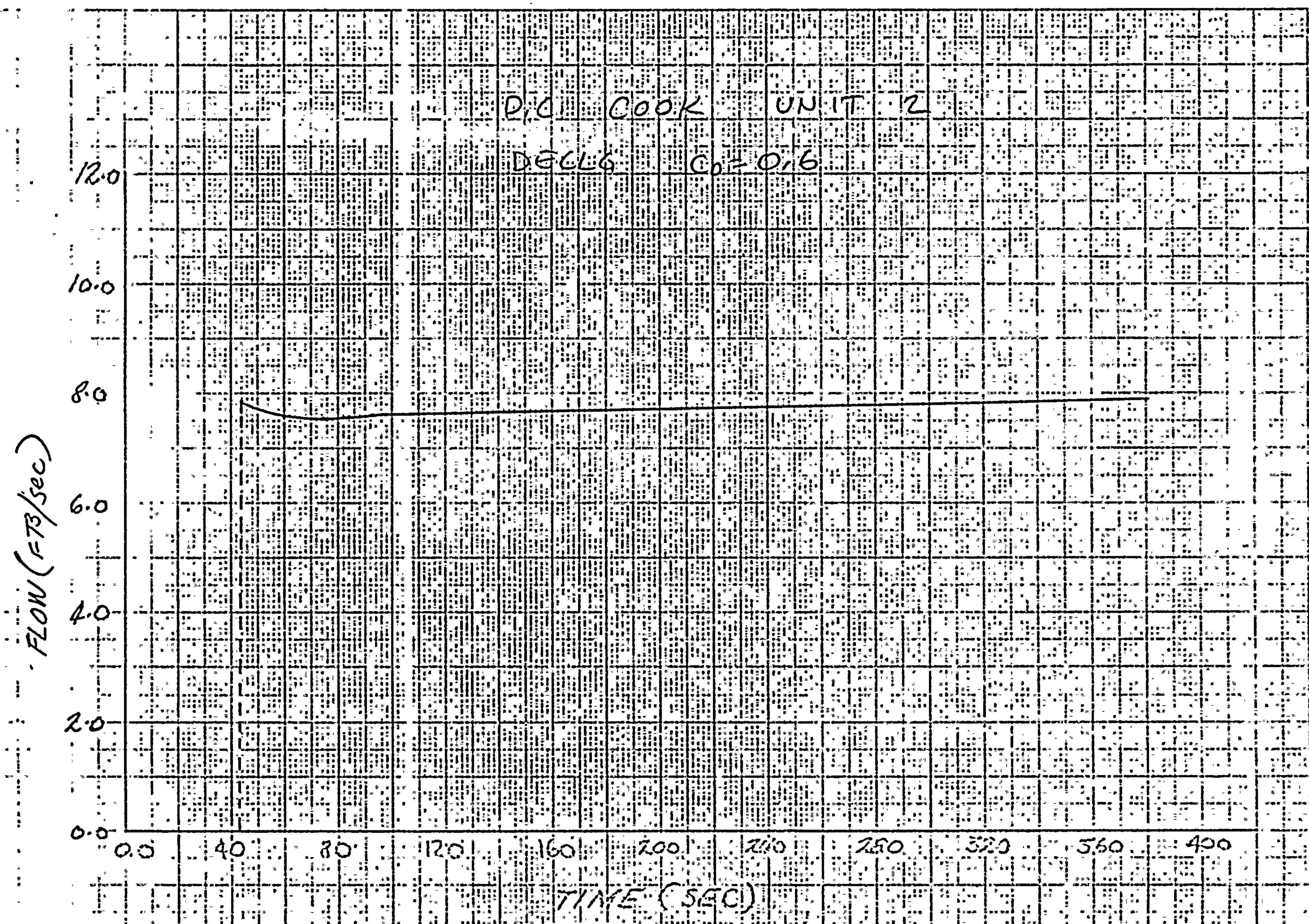


FIGURE 13 C PUMPED ECCS FLOW (REFLOOD) - DECLG(CD = 0.6)

D.C. COOK UNIT 2

DECLG $C_D = 0.6$

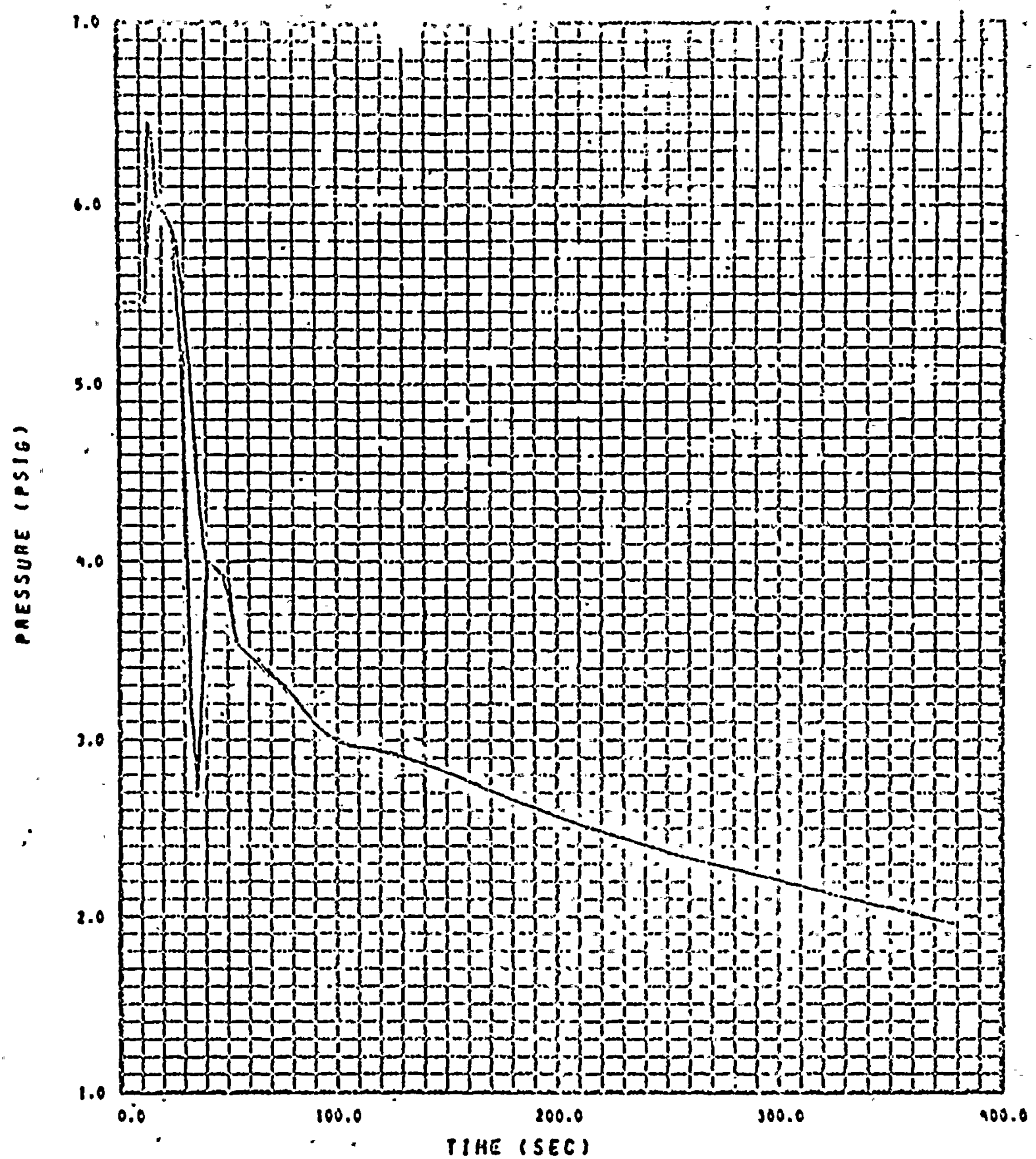


FIGURE 14 c CONTAINMENT PRESSURE - DECLG(CD = 0.6)

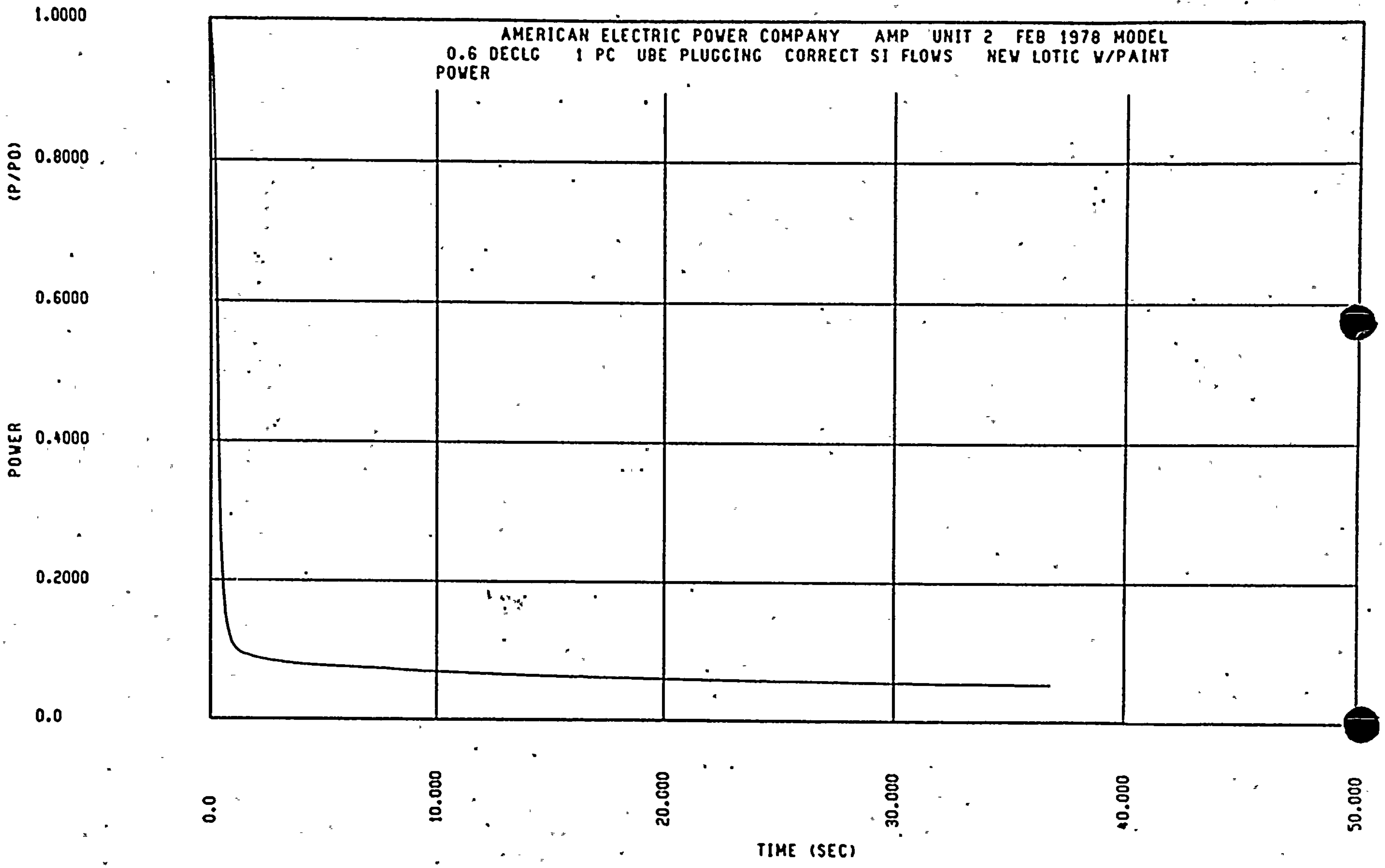


FIGURE 15 c CORE POWER TRANSIENT - DECLG(CD = 0.6)

ATTACHMENT B
TO
AEP:NRC:00322C

TABLE 1

TIME SEQUENCE OF EVENTS

	OCT'75 W/ZIRC FIX	FEB'78 ANALYSIS
REACTOR TRIP SIGNAL	0.72	0.71
SI SIGNAL	4.64	4.83
ACCUMULATOR INJECT	14.5	14.6
SI INJECTS	29.64	29.83
END OF BYPASS	27.72	33.18
END OF BLOWDOWN	32.04	39.07
BOTTOM OF CORE RECOVERY	44.48	48.97
ACCUMULATOR EMPTY	56.83	56.42

MASS DEFICIT IMPACT

	OCT'75 W/ZIRC FIX	FEB'78 ANALYSIS
LIQUID REMAINING @ EOB	25163.4 lbs.	35299.9 lbs.
BYPASS DEFICIT	28324.3 lbs.	30487.5 lbs.
LIQUID REMAINING FOR REFILL	0.0	4812.4 lbs.

EOB ACCUMULATOR DELIVERY RATE		4600.0 lbs./sec.
AVAILABLE REFILL TIME DIFFERENCE	1.05 sec.	
EOB DIFFERENCE	5.46 sec.	
BOC DIFFERENCE	4.49 sec.	

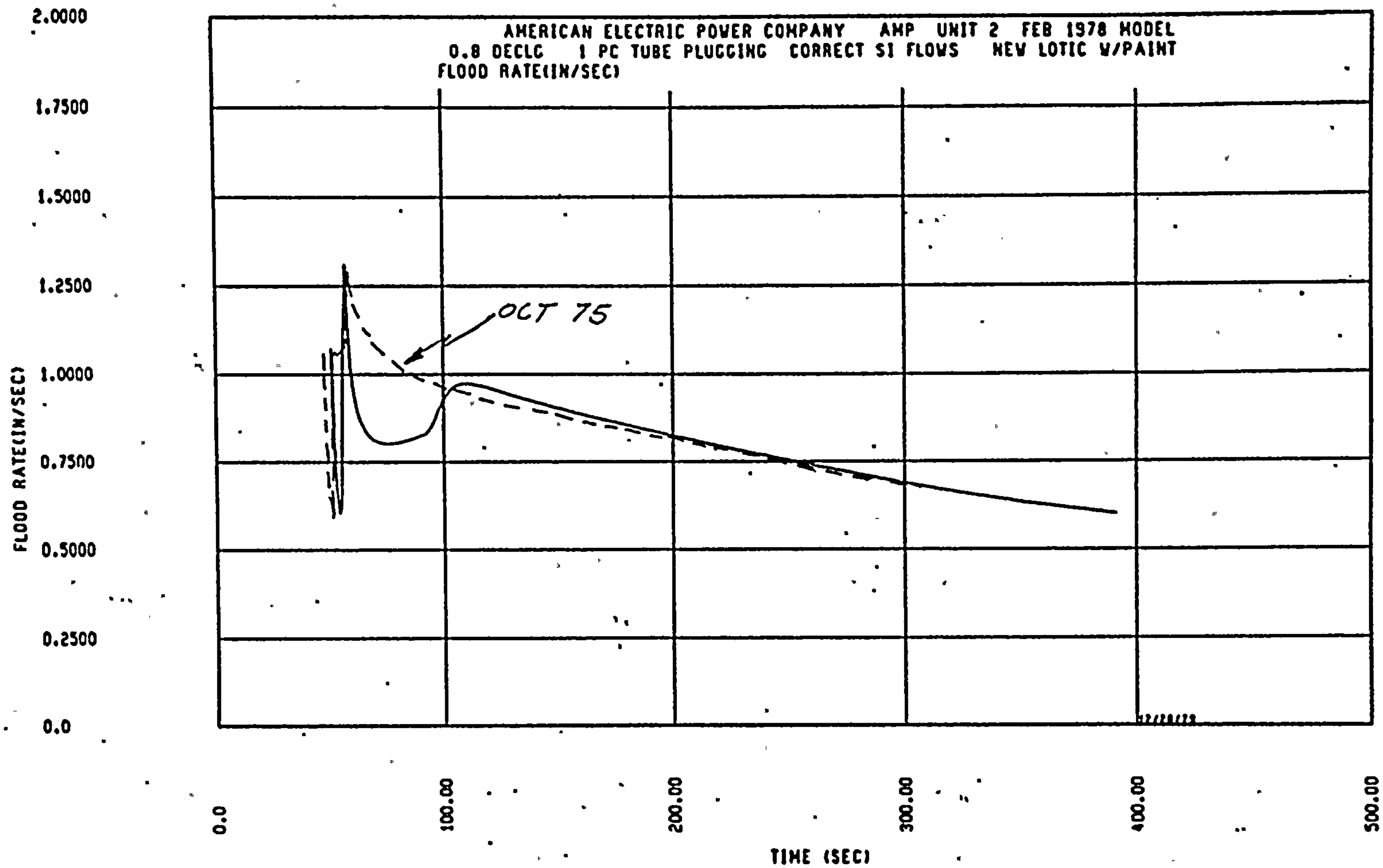
TABLE 2

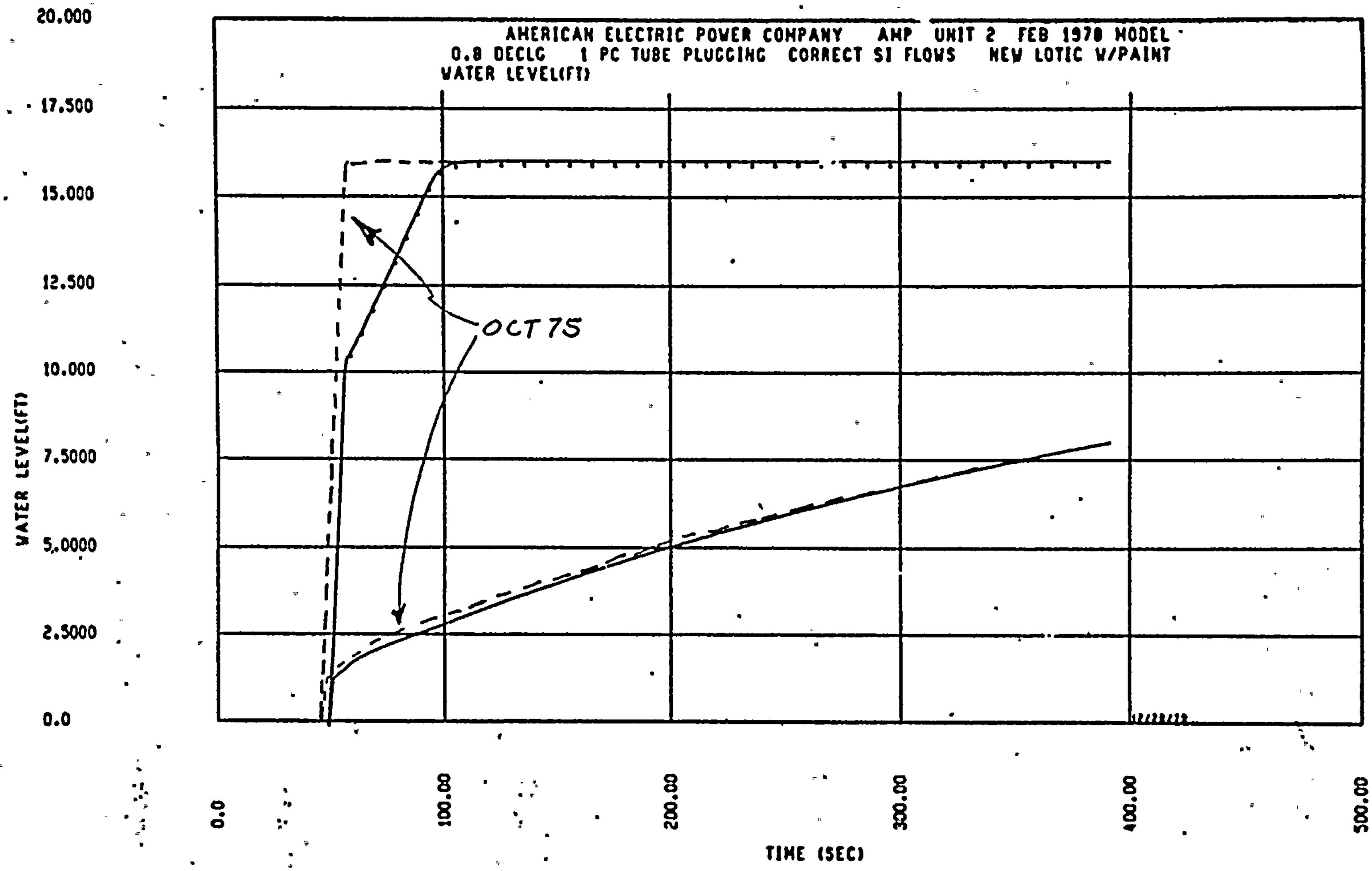
RESULTS

	OCT'75 W/ZIRC FIX	FEB'78 ANALYSIS
FQT	2.10	2.02
PCT	2188°F	2171°F
ZPCT	8.0 FT	7.5 FT
ZR/H_2O RXM	7.71%	6.36%
Z ZRH_2O RXM	8.0 FT	7.5 FT
HOT ROD BURST	36.8 SEC	40.6 SEC
Z HRBURST	6.0 FT	6.0 FT

TABLE 3

	OCT'75 W/ZIRC	FEB'78
ACCUMULATOR EMPTY TIME	56.83 SEC	56.42 SEC
Z_D @ ACCUMULATOR EMPTY TIME	15.99 FT	10.35 FT
Z_C @ ACCUMULATOR EMPTY TIME	2.01 FT	1.55 FT
STEAM COOLING STARTS	86.63 SEC	59.57 SEC
Z_D @ STEAM COOLING	16.0 FT	10.7 FT
Z_C @ STEAM COOLING	2.72 FT	1.715 FT
FULL DOWNCOMER TIME	63.6 SEC	128 SEC
ACCUMULATOR WATER VOLUME PER TANK @ BOC	377.67 FT^3	223.37 FT^3


AMERICAN ELECTRIC POWER COMPANY AMP UNIT 2 FEB 1978 MODEL
0.8 DECLG 1 PC TUBE PLUGGING CORRECT SI FLOWS NEW LOTIC W/PAINT
FLOOD RATE(IN/SEC)
OCT 75
FLOOD RATE(IN/SEC)
2.0000
1.7500
1.5000
1.2500
1.0000
0.7500
0.5000
0.2500
0.0
0.0
100.00
200.00
300.00
400.00
500.00
TIME (SEC)



AMERICAN ELECTRIC POWER COMPANY AMP UNIT 2 FEB 1978 MODEL
0.8 DECLG 1 PC TUBE PLUGGING CORRECT SI FLOWS NEW LOTIC W/PAINT
WATER LEVEL(FT)
20.000
17.500
15.000
12.500
10.000
7.5000
5.0000
2.5000
0.0
WATER LEVEL(FT)
OCT 75
0.0
100.00
200.00
300.00
400.00
500.00
TIME (SEC)

MAJOR MODEL CHANGE IMPACTS

A) 15 X 15 FLECHT - NO IMPACT

B) WATER PACKING - AS NOTED EARLIER

C) DYNAMIC STEAM COOLING - 60°F [WCAP 9220 -P-A]

D) γ = 1.25 DURING REFLOOD - REFILL & REFLOOD BENEFIT

E) CODE MAINTENANCE & ANALYTICAL IMPROVEMENT - LESS THAN -12°F

INPUT CHANGES IMPACT

A) <u>SATAN</u>

1) CORE POWER IN SATAN*

2) STEAM GENERATOR INITIALIZATION/T-INLET*

3) INCREASED SAFETY INJECTION

4) MICELLANEOUS*

B) <u>REFLOOD</u>

1) BROKEN LOOP ACCUMULATOR SPILL TO SPRAY**

2) INCREASED SAFETY INJECTION

3) LOWER PLENUM VOLUME REVISION

C) <u>LOTIC</u>

1) PAINT ON CONTAINMENT SURFACES**

D) <u>LOCTA</u>

1) AXIAL NODING SCHEME - NO IMPACT

2) INITIAL ROD GAP GAS BACKFILL PRESSURE

* LESS THAN 25°F IMPACT

** LESS THAN 3°F IMPACT