Wisconsin Electric POWER COMPANY
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November 27, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactor Branch No. 1

Gentlemen:

DOCKET NOS. 50-266 AND 50-301
LOW PRESSURE ECCS EVALUATION FOR
18% STEAM GENERATOR TUBE PLUGGING
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

Enclosed herewith are the results of an ECCS reanalysis for operation of the Point Beach Nuclear Plant Units 1 and 2 at a reduced primary system pressure of 2000 psia with up to 18% of the U-tubes plugged in each steam generator. On the basis of generic sensitivity studies, which showed that the limiting break for Westinghouse two-loop plants with 14x14 fuel is a double-ended, cold-leg, guillotine pipe break with a discharge coefficient of 0.4, only the analysis for this specific limiting break for the Point Beach Nuclear Plant is provided. The results of these generic sensitivity studies were provided to you previously with our letter dated March 20, 1979.

The enclosed reanalysis was conducted at a reduced primary system pressure of 2000 psia using a core inlet temperature of a nominal 544°F. As the Point Beach Nuclear Plant utilizes upper plenum injection, a 60°F increase in temperature should be added to the calculated peak clad temperature to account for explicit modeling of upper plenum injection. Westinghouse has determined for the Point Beach Nuclear Plant that fuel rod bursting and blockage is appropriately accounted for in the ECCS analysis results. These results demonstrate that the Emergency Core Cooling System continued to meet the Acceptance Criteria as presented in 10 CFR Part 50.46 of the Commission's Regulations.

As you know, an ECCS reanalysis for the 18% steam generator tube plugging limit and operation at 2250 psia was submitted to you by letter dated November 19, 1979. At that time, we stated that Westinghouse ECCS sensitivity studies have shown that operation at reduced pressure of 2000 psia will have an insignificant effect on the analysis results. The attached reanalysis confirms this statement.

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Should you have any questions regarding this analysis, please notify us as soon as possible so that we may provide you any necessary clarifications.

Very truly yours,

C. W. Fay, Director
Nuclear Power Department

The Loss of Coolant Accident (LOCA) has been reanalyzed for Point Beach Units 1 and 2. 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[3-6] 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 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. 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 COCO computer code[6] 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.[8] 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 principal parameters for the break sizes analyzed. The following items are noted:

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

Figures 4 - 6: 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 7 - 9:	Clad temperature, fluid temperature and core flow. The clad and fluid temperatures are for the hot spot and burst locations
Figures 10 - 11:	Downcomer and core water level during reflood, and flooding rate
Figures 12 - 13:	Emergency core cooling system flowrates, for both accumulator and pumped safety injection
Figures 14 - 15:	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 2.32.

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 percent 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.

The effects of upper plenum injection for Westinghouse-designed 2-loop plants has been discussed with the staff.[15,16,17,18,19] Based on interim calculations, a 60°F increase in calculated peak clad temperatures results from explicit modelling of upper plenum injection in the Point Beach power plant. In order to use the present Westinghouse ECCS evaluation model[13,14,15] to analyze a postulated LOCA in the

Point Beach plants and remain in compliance with 10CFR50.46, a limit of 2140°F on calculated peak clad temperatures must be observed. It can be seen from the results contained herein that this ECCS analysis for the Point Beach power plants remains in compliance with 10CFR50.46.

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

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 R. L. Tedesco of the Nuclear Regulatory Commission, letter number NS-TMA-2014, December 11, 1978.

15. "Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two-Loop Plants," November, 1977.

16.* Letter from V. J. Esposito to H. W. Gutzman of NSD Operations Support, both from Westinghouse Electric Corporation, letter number SE-SAI-2267, January 30, 1978.

17. "NRC Questions Regarding TAC 1/16/78 Submittal by Westinghouse Designed Two-Loop Plant Operators," February 1, 1978.

18.* Letter from V. J. Esposito to H. W. Gutzman of NSD Operations Support, both from Westinghouse Electric Corporation, letter number SE-SAI-2290, February 17, 1972.

19. "Safety Evaluation Report on Interim ECCS Evaluation Model for Westinghouse Two-Loop Plants," March 1978.

*Wisconsin Electric Power should supply the proper references by which these references were formally transmitted to the NRC.

TABLE 1

LARGE BREAK - TIME SEQUENCE OF EVENTS

Event	Occurrence Time (Seconds) DECLG, C_D = 0.4
Accident Initiation	0.0
Reactor Trip Signal	~~[illegible]~~ .37
Safety Injection Signal	.8
Start Accumulator Injection	~~[illegible]~~ 9.8
End of ECC Bypass	~~[illegible]~~ 21.7
End of Blowdown	~~[illegible]~~ 21.7
Bottom of Core Recovery	~~[illegible]~~ 42.1
Accumulators Empty	~~[illegible]~~ 57.2
Start Pumped ECC Injection	25.8

TABLE 2

LARGE BREAK - ANALYSIS INPUT AND RESULTS

Quantities in the Calculations:

Licensed core power rating	102 percent of 1518.5 MWt
Total core peaking factor	2.32
Peak linear power	102 percent of 13.23 kw/ft
Accumulator water volume	1100 cubic feet per tank
Accumulator pressure	700 psia
Number of safety injection pumps operating	2
Steam generator tube plugging level	18 percent (uniform)
Fuel parameters - Cycle: Generic	Region: Generic

Results	DECLG, $C_D = 0.4$
Peak clad temperature (°F)	~~2053~~ 2062
Location (feet)	7.5
Maximum local clad/water reaction (percent)	~~5.1~~ [illegible]3
Location (feet)	7.5
Total core clad/water reaction (percent)	<0.3
Hot rod burst time (seconds)	~~22.8~~ 29.6
Location (feet)	5.75

TABLE 3

CONTAINMENT DATA (DRY CONTAINMENT)

Net Free Volume	1.065×10^6 ft^3
Initial Conditions	
Pressure	14.7 psia
Temperature	90 °F
RWST Temperature	34 °F
Service Water Temperature	33 °F
Outside Temperature	-25 °F
Spray System	
Number of Pumps Operating	2
Runout Flow Rate	1950 gpm each
Actuation Time	10 secs
Safeguards Fan Coolers	
Number of Fan Coolers Operating	4
Fastest Post Accident Initiation of Fan Coolers	35 secs

TABLE 3 (Cont)

STRUCTURAL HEAT SINK DATA'

Thickness (in)	Material	Area, ft^2
.322	Steel/concrete, 12	56020
.188	Steel/concrete, 12	6230
.25	Steel/concrete, 12	2480
.188	Steel/concrete, 12	690
.094	Steel	103724
.3[illegible]4	Steel	11710
.443	Steel	4730
.584	Steel	5441
.712	Steel	4490
1.0	Steel	957
2.634	Steel	3667
.125	Steel	10221
.20[illegible]	Steel	16551
.5	Steel	2707
.322	Steel	13835
.065	Steel	1411[illegible]
.036	Steel	38250
.1	Steel/concrete, 3	19500
3.0	Concrete	19500
30.0	Concrete	61500

TABLE 3 (Cont)

PAINTED STRUCTURAL HEAT SINK DATA

Structural Heat Sink Surface Area (Ft^2)	Structural Heat Sink Thickness (In)	Paint Thickness (Mils)
56020	.322	7.5
2400	.25	7.5
103724	.094	7.5
11710	.304	7.5
4730	.443	7.5
5441	.584	7.5
4490	.712	7.5
957	1.0	7.5
3057	2.634	6.0
10221	.125	6.0
16551	.209	6.0
2707	.5	6.0

TABLE 4

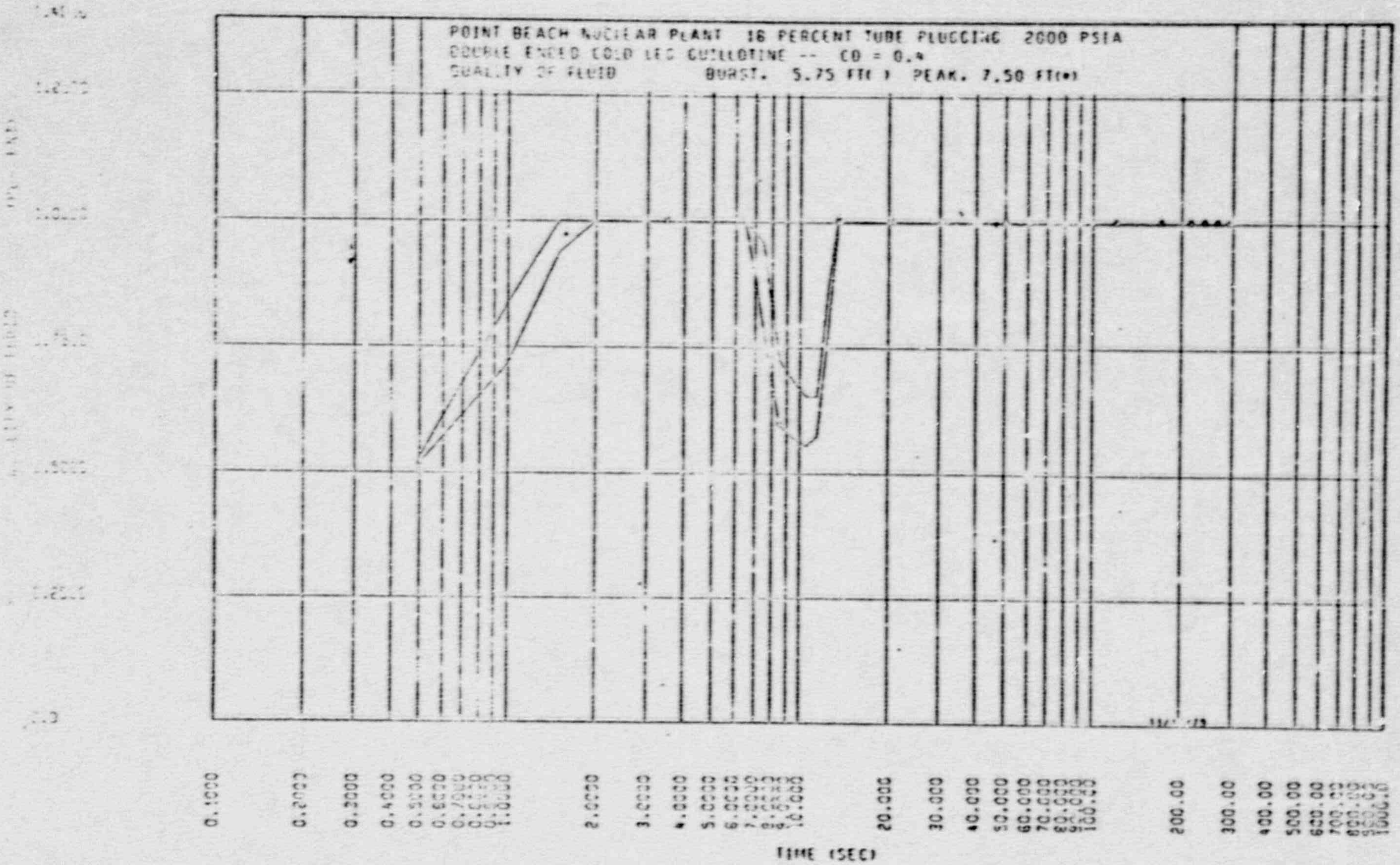
REFLOOD MASS AND ENERGY RELEASE

Time (Sec)	Mass Flow (Lb/Sec)	Energy Flow (Lb/Sec)
42.7	.013	16.9
44.3	.42	543.9
48.9	27.3	26722.
59.8	47.4	64159.
73.4	54.5	66487.
89.6	173.8	96369.
107.8	211.73	[illegible]
127.2	216.2	94049.
173.2	222.7	[illegible]
222.2	221.8	99922.

TABLE 5

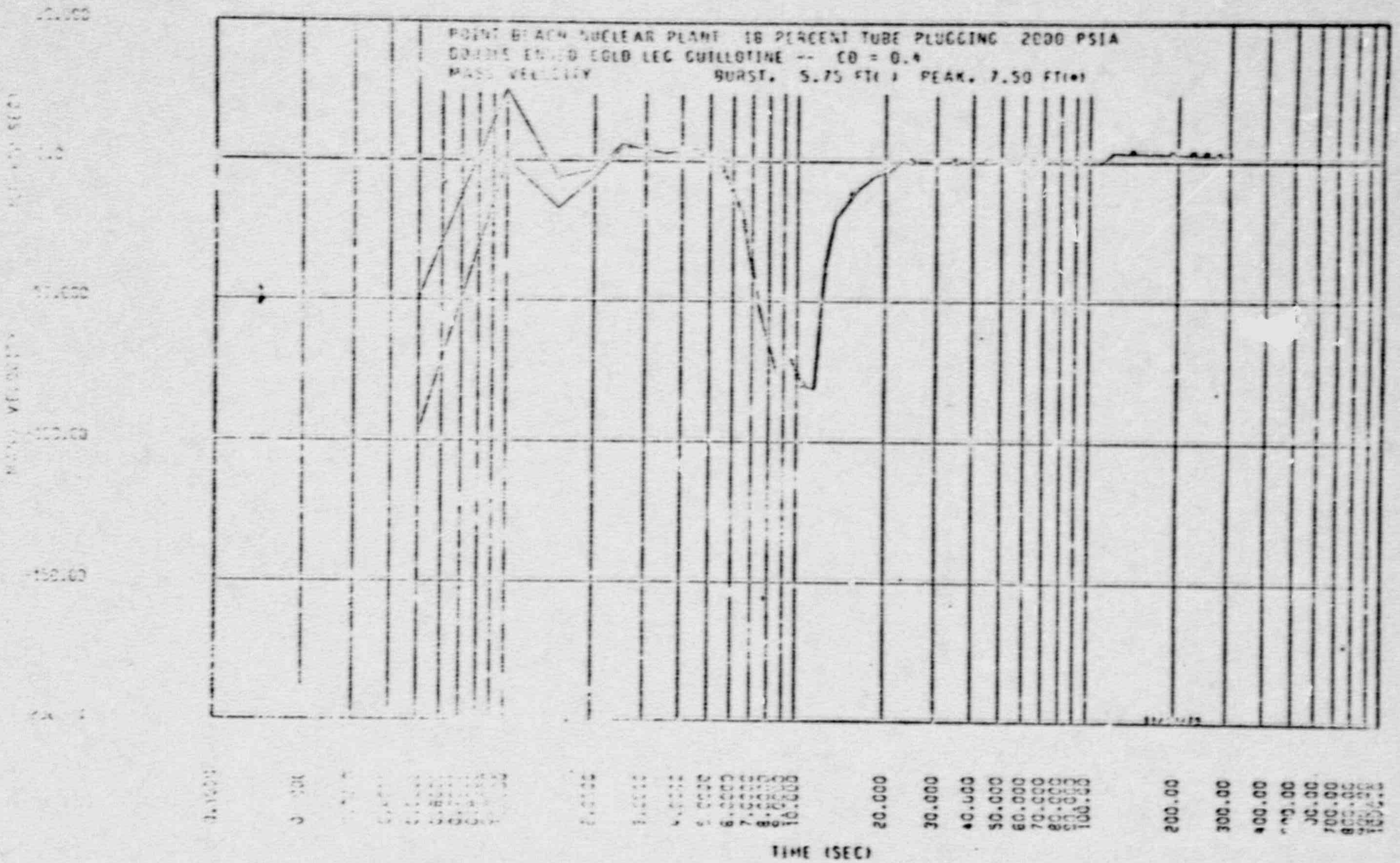
Broken Loop Accumulator Mass and Energy Release

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)
1.010	2523.301	151120.523
2.010	2412.643	144493.209
3.010	2317.511	138795.761
4.010	2234.541	133826.601
5.010	2160.050	129235.007
6.010	2092.160	125299.909
7.010	2029.721	121560.017
0.010	1972.231	118116.917
9.010	1919.719	114971.994
10.010	1071.409	112003.454
11.010	1827.145	109427.719
12.010	1786.237	106977.723
13.010	1740.010	104693.128
14.010	1712.747	102576.444
15.010	1600.074	100619.639
16.010	1649.766	98804.472
17.010	1621.515	97112.515
18.010	1595.055	95530.454
19.010	1570.170	94037.401
20.010	1546.630	92628.129
21.010	1524.390	91295.731
21.692	1507.994	90313.772
21.903	1507.977	90300.742
21.909	1507.556	90287.533
22.009	1503.052	90035.692
22.122	1503.015	90015.546
22.122	1503.015	90015.546
22.202	1499.028	89824.736
22.402	1495.650	89576.601
22.602	1491.537	89331.167
22.802	1487.514	89087.241
23.002	1483.475	88845.309
23.202	1479.464	88605.100
23.402	1475.489	88367.013
23.602	1471.540	88131.029
23.802	1467.644	87897.209
24.002	1463.747	87663.814
[illegible]	[illegible]	[illegible]33.156
24.402	1456.001	[illegible]04.775
[illegible]	[illegible]	[illegible]5.220
25.002	1444.468	86509.165
25.502	1434.570	85916.419
26.002	1424.068	85335.233
26.502	1415.540	84765.210
27.002	1406.011	84205.995
27.502	1396.648	83657.215
28.002	1387.053	83118.523
[illegible]	1379.023	82589.627
29.002	1370.369	82070.169
29.502	1361.[illegible]	81559.982
30.002	1353.497	81058.365
30.502	1345.229	80565.765
31.002	1337.161	80081.305
31.502	1329.103	79605.097
32.002	1321.367	79136.682
32.502	1313.674	78675.915
33.002	1306.104	78222.512
33.502	1298.655	77776.413
34.002	1291.323	77337.349
[illegible]	[illegible].108	[illegible]05.098
[illegible]	1276.998	[illegible]79.405
[illegible]	[illegible].997	[illegible]0.213
36.[illegible]	[illegible].10[illegible]	[illegible]47.506
[illegible]	[illegible].3[illegible]	[illegible]0.51[illegible]
37.[illegible]	[illegible]	[illegible]9.6[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]92.[illegible]
[illegible]	[illegible]	[illegible]19.727
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]



POINT BEACH NUCLEAR PLANT 16 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
QUALITY OF FLUID BURST. 5.75 FT( ) PEAK. 7.50 FT(*)
0.1000
0.2000
0.3000
0.4000
0.5000
2.0000
3.0000
4.0000
5.0000
20.000
30.000
40.000
50.000
60.000
70.000
200.00
300.00
400.00
500.00
TIME (SEC)

Figure 1



POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
MASS VELOCITY BURST, 5.75 FT( ) PEAK, 7.50 FT(•)
TIME (SEC)
20.000
30.000
40.000
50.000
60.000
70.000
80.000
100.000
200.00
300.00
400.00
700.00
800.00

Figure 2

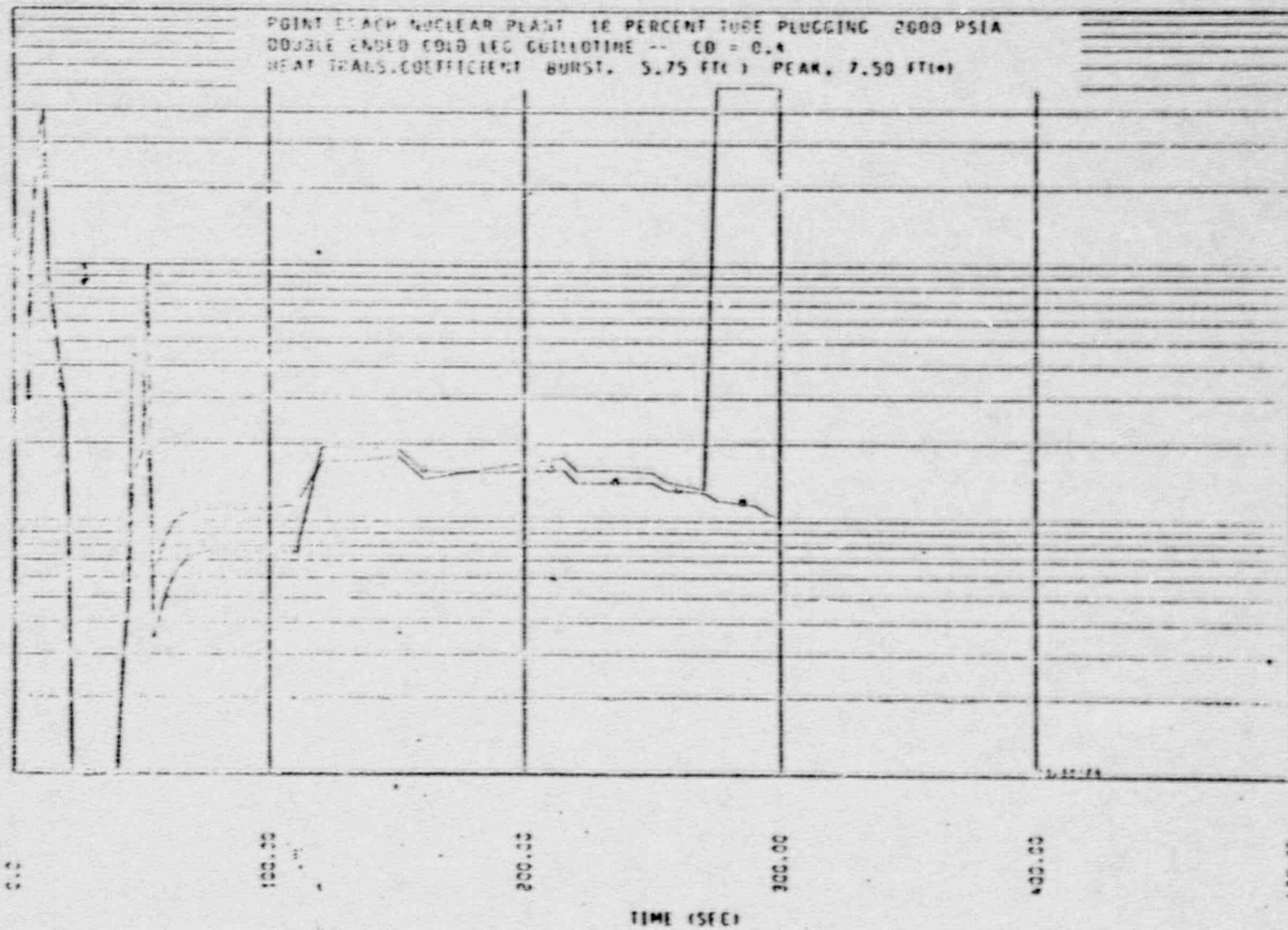

POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
HEAT TRANS.COEFFICIENT BURST, 5.75 FT( ) PEAK, 7.50 FT(*)
100.00
200.00
300.00
400.00
TIME (SEC)

Figure 3

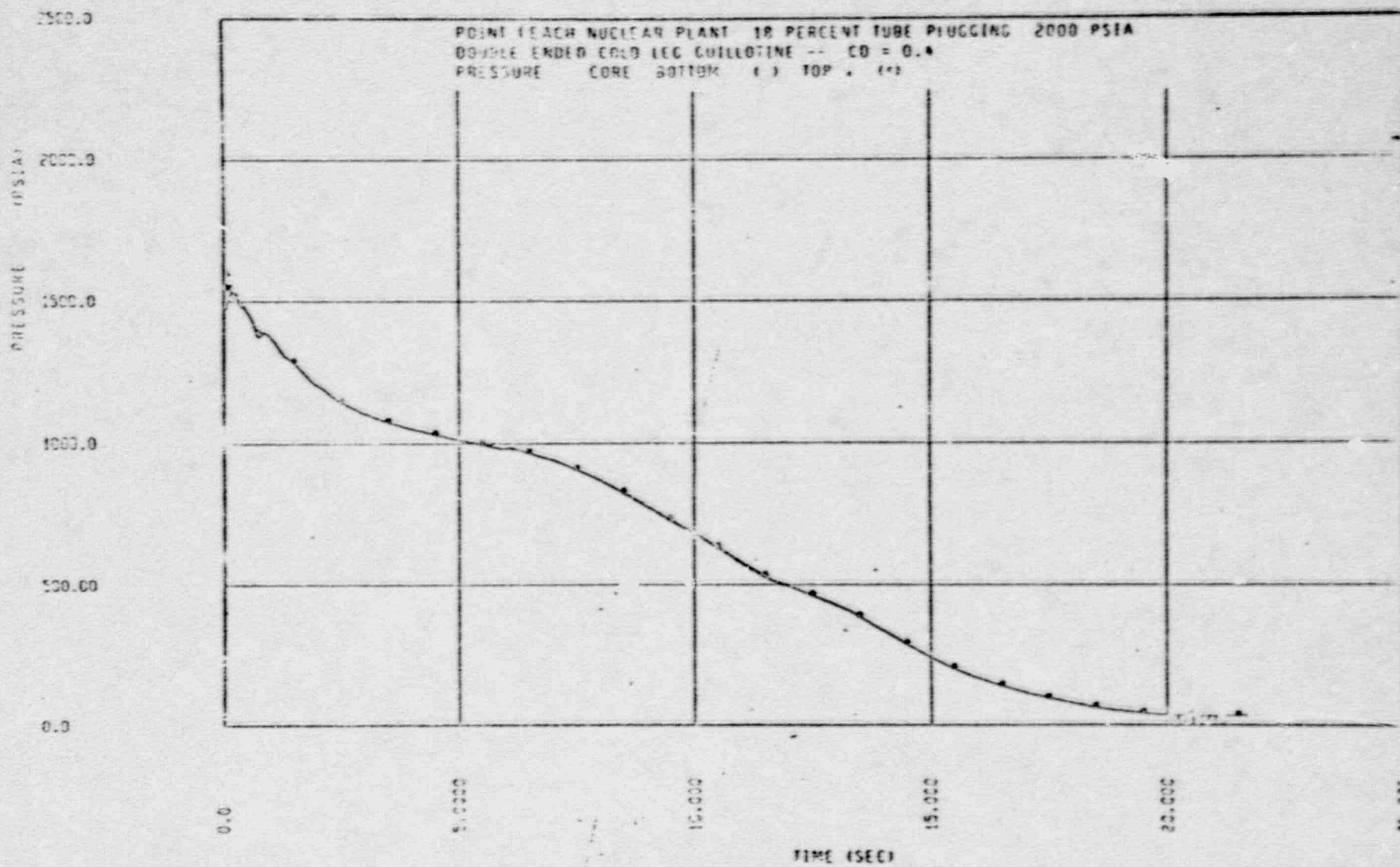

POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
PRESSURE CORE BOTTOM ( ) TOP . (*)
PRESSURE (PSIA)
2500.0
2000.0
1500.0
1000.0
500.00
0.0
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 4

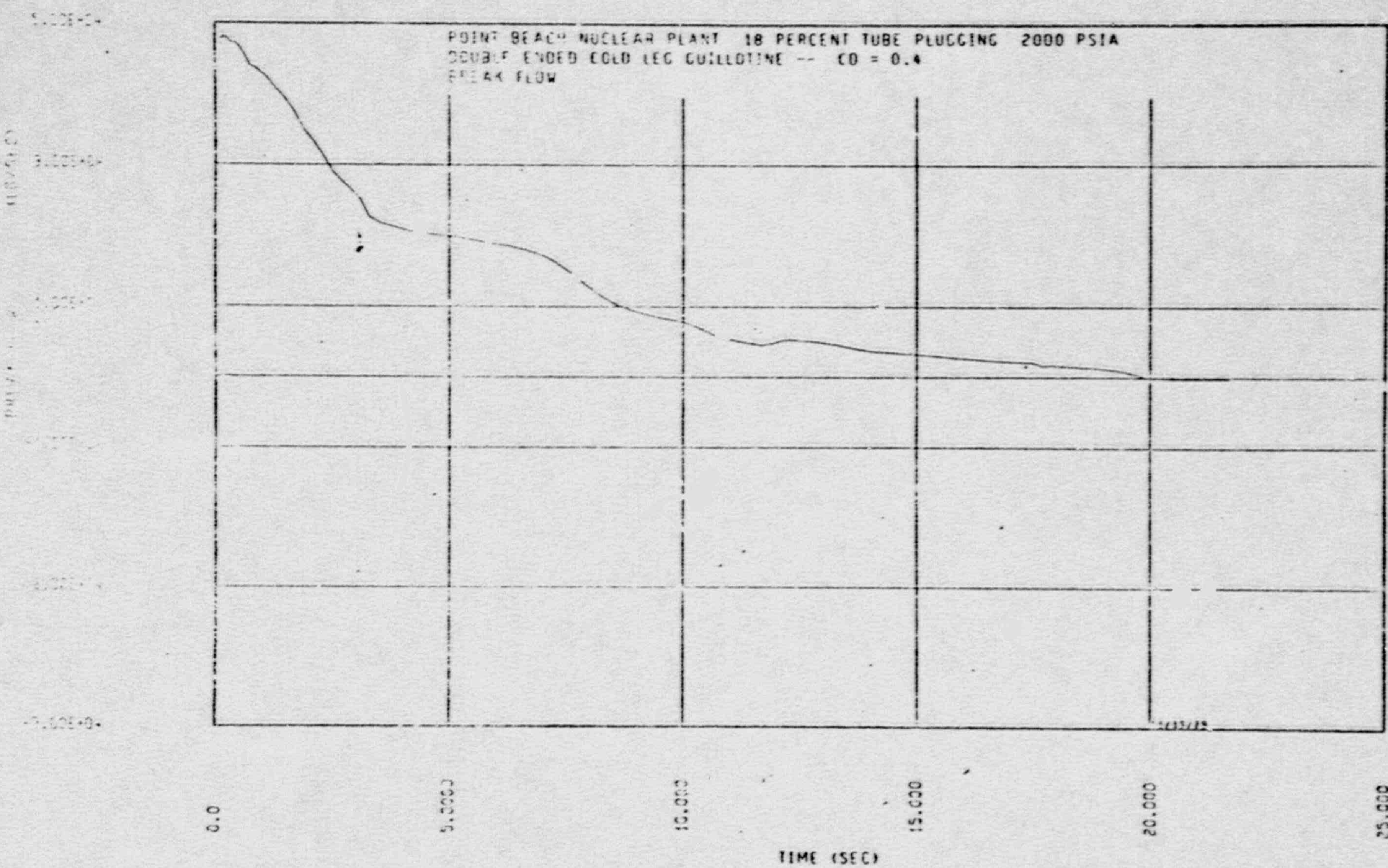

POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
BREAK FLOW
0.0
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 5

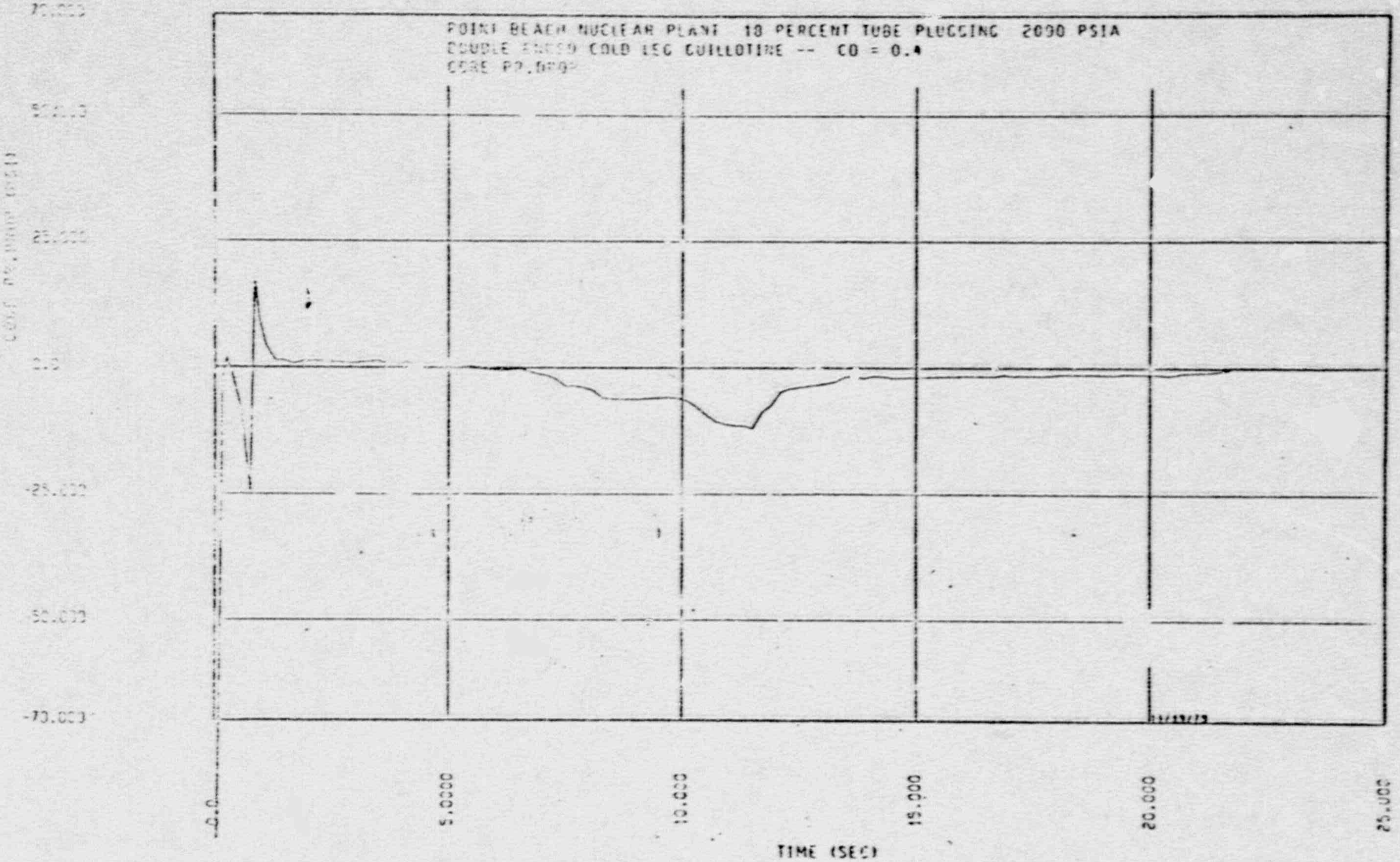

POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
COLD LEG GUILLOTINE -- CD = 0.4
CORE PR.DROP
25.000
0.0
-25.000
0.0
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 5

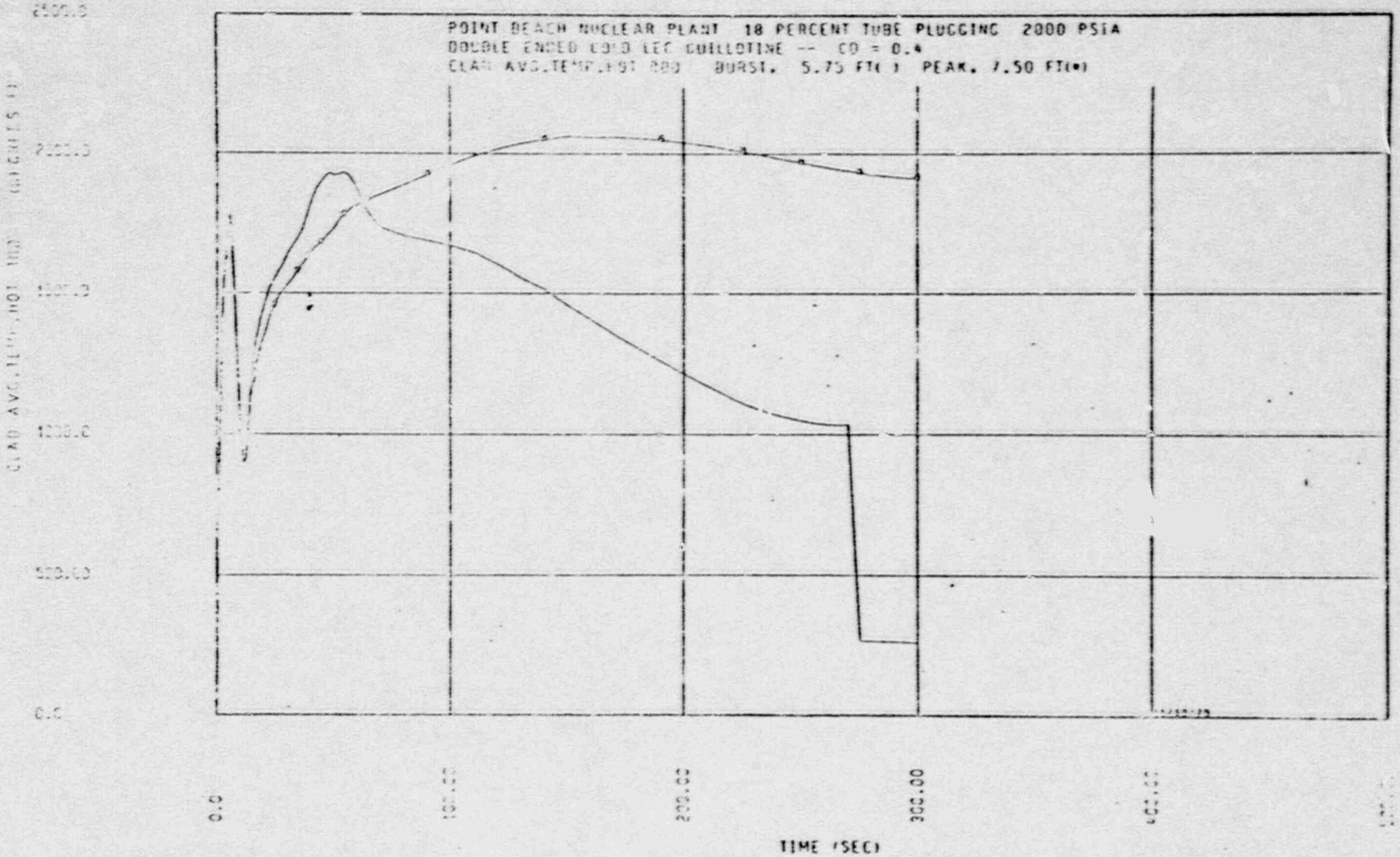
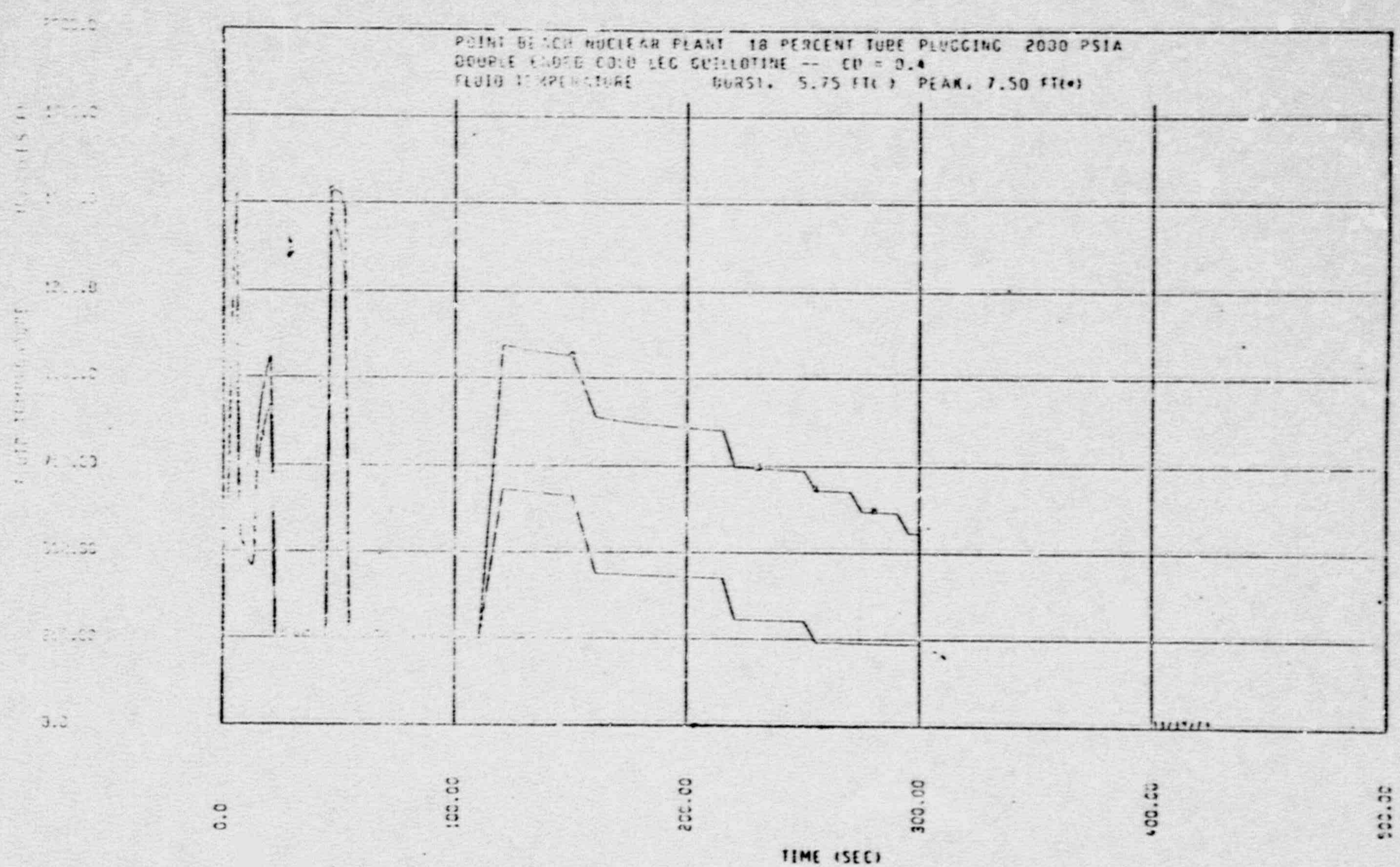
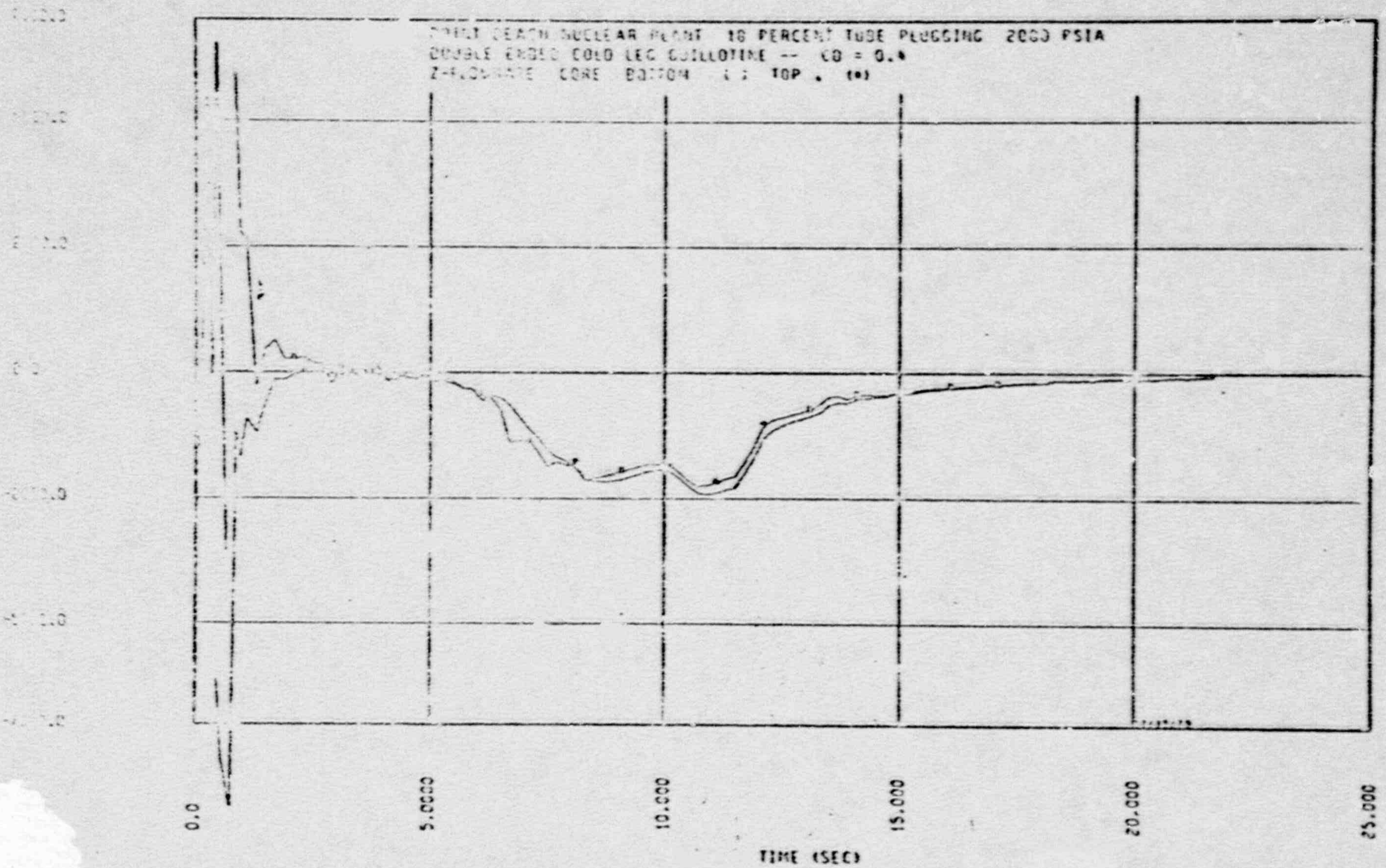

POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
BURST, 5.75 FT( ) PEAK, 7.50 FT(*)
0.0
200.00
300.00
TIME (SEC)

Figure 7



POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2030 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
BURST, 5.75 FT( ) PEAK, 7.50 FT(•)
0.0
100.00
200.00
300.00
400.00
500.00
TIME (SEC)

Figure 8



POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
CORE BOTTOM TOP
0.0
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 9

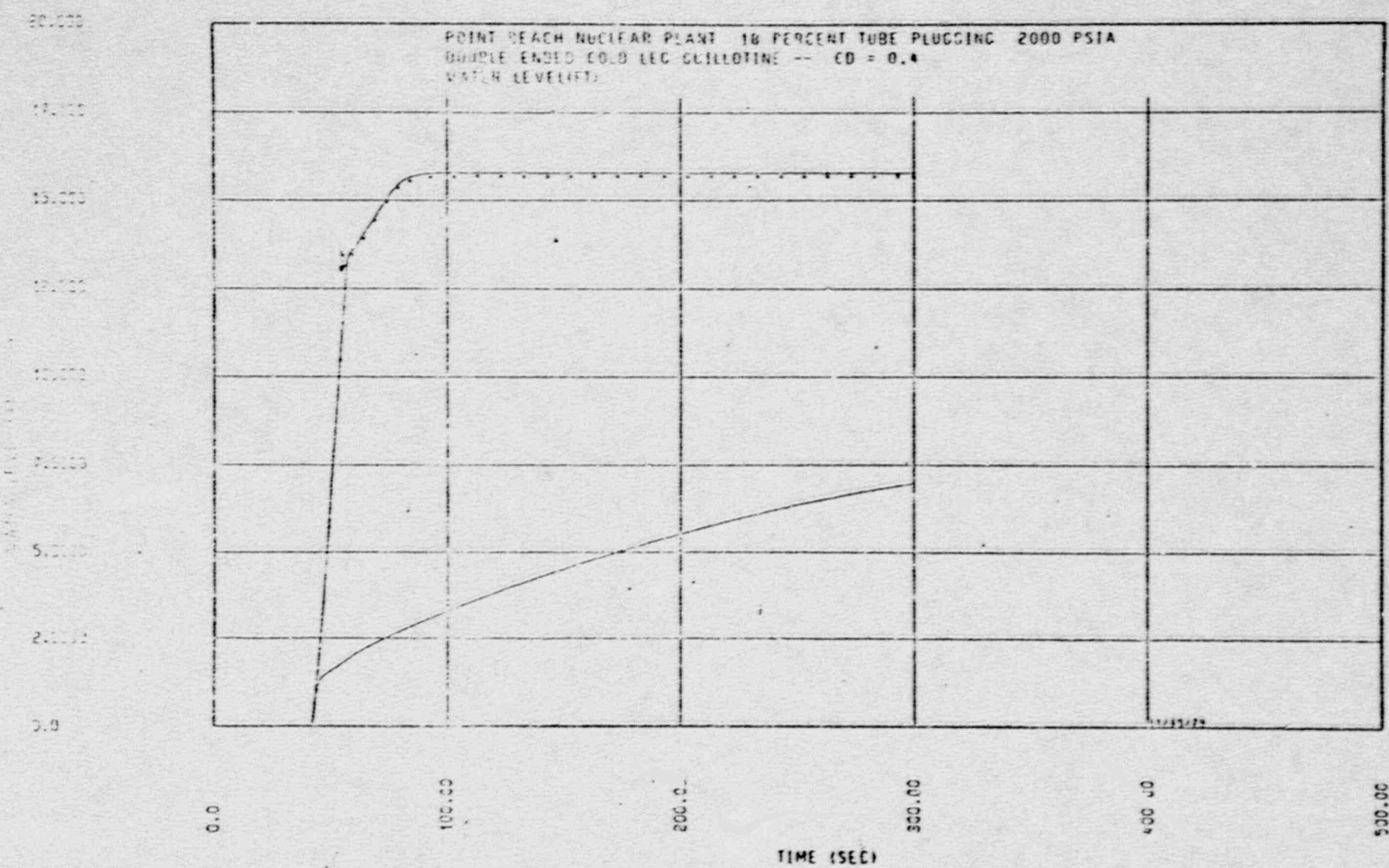
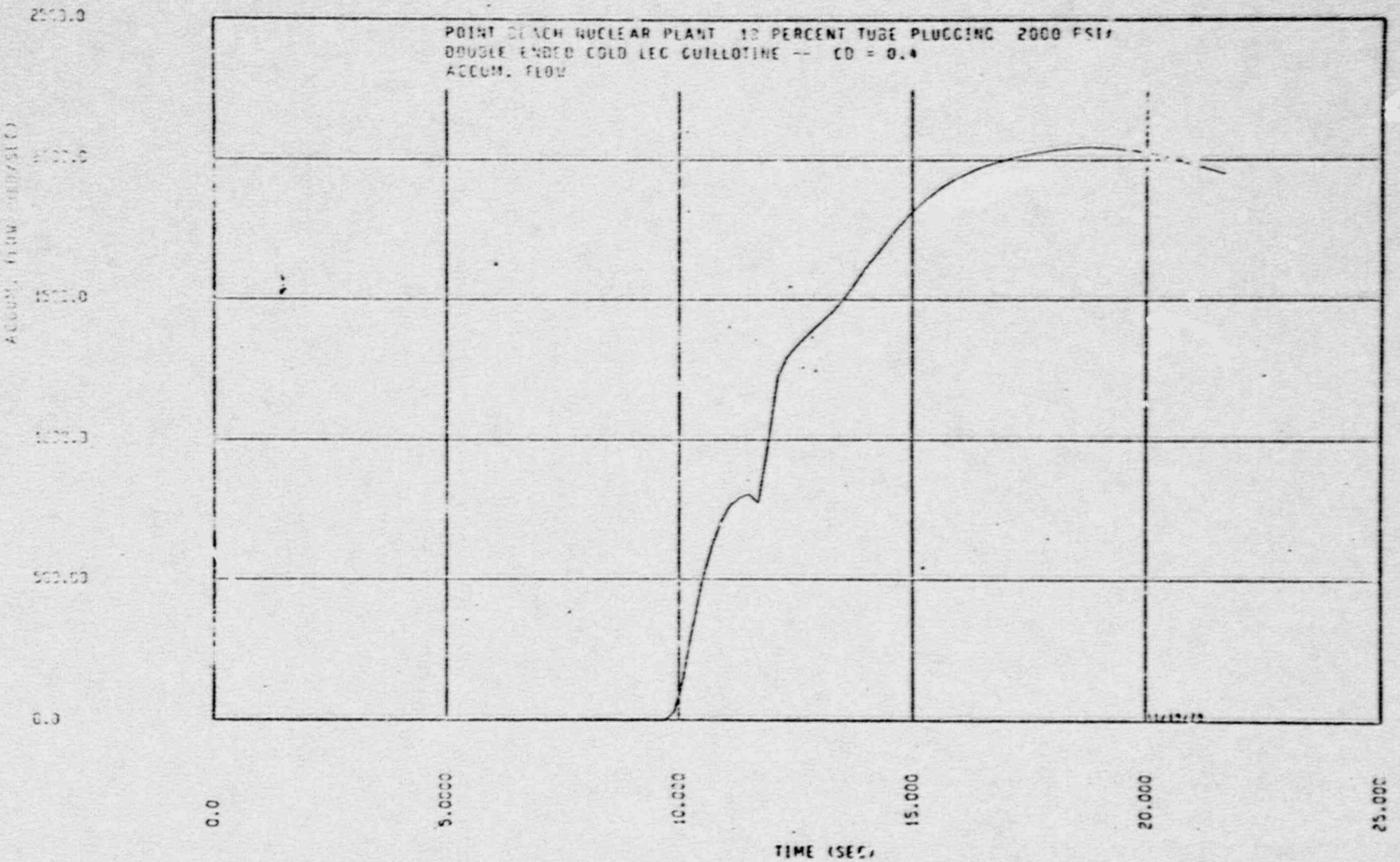

POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
WATER LEVEL(FT)
20.000
15.000
5.000
2.500
0.0
0.0
100.00
300.00
400.00
500.00
TIME (SEC)

figure 10



POINT BEACH NUCLEAR PLANT 10 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
ACCUM. FLOW
2500.0
2000.0
1500.0
1000.0
500.00
0.0
0.0
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 12

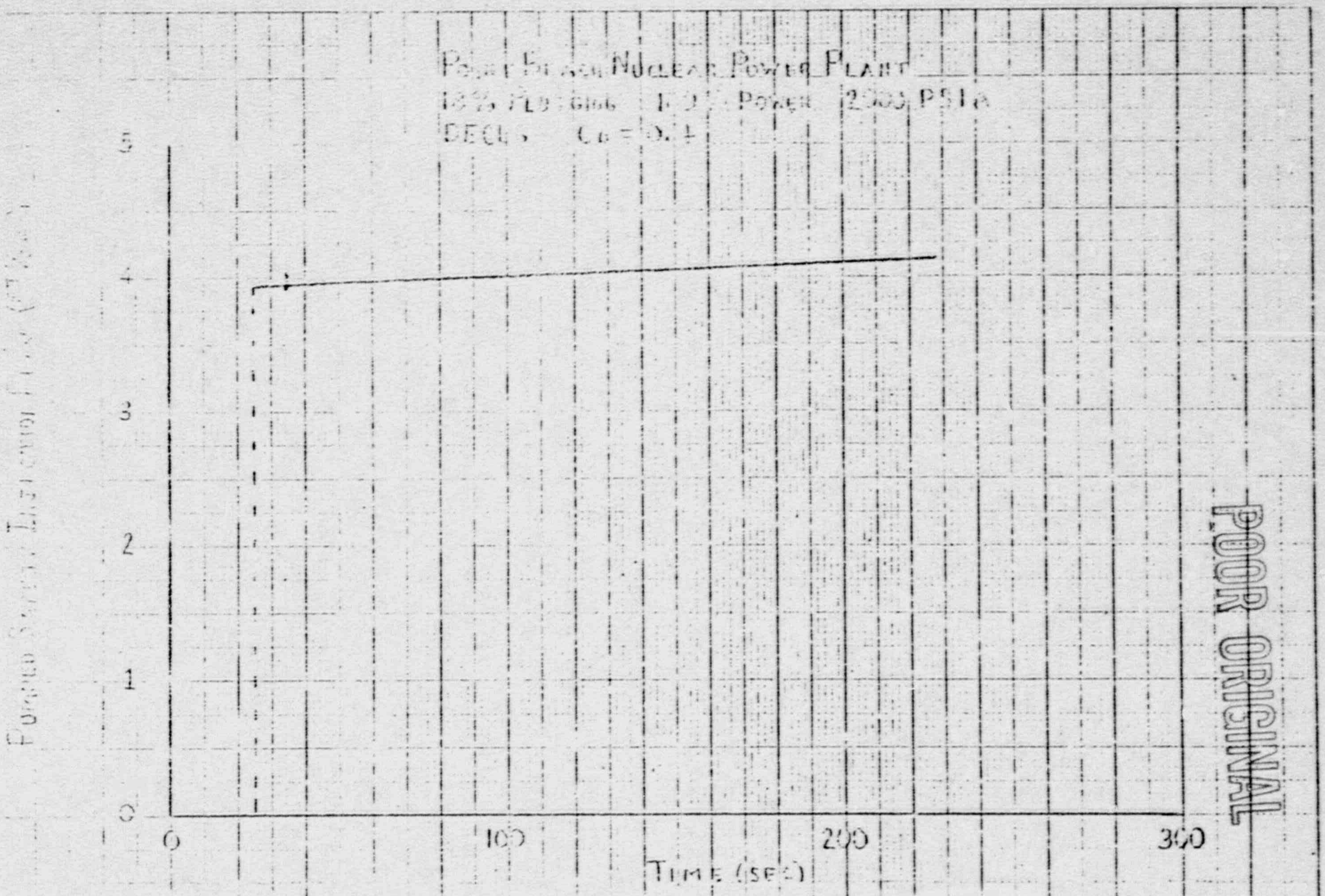

Point Beach Nuclear Power Plant
DECL,
0
1
2
3
4
5
0
100
200
300
Time (sec)

Figure 13

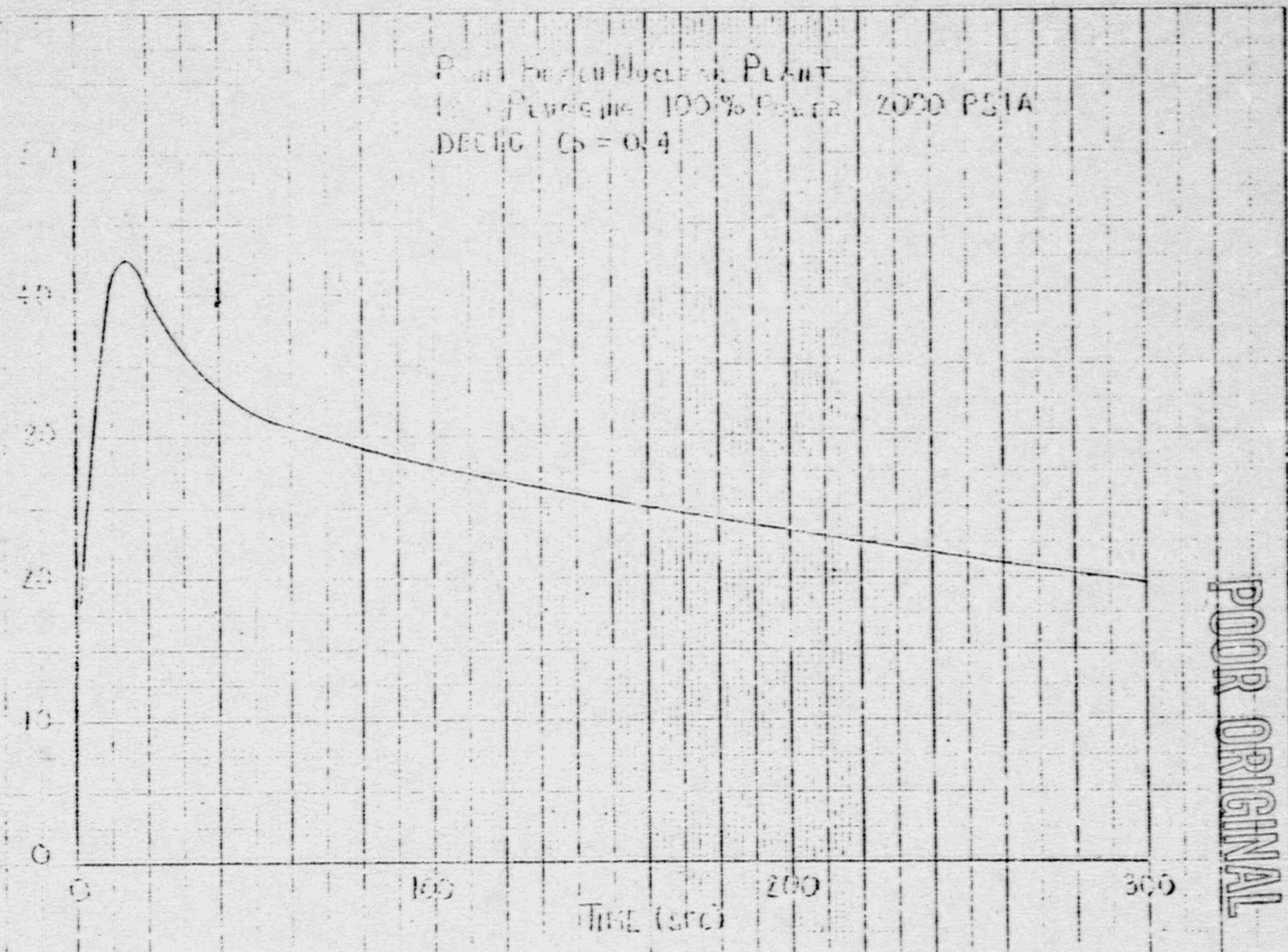
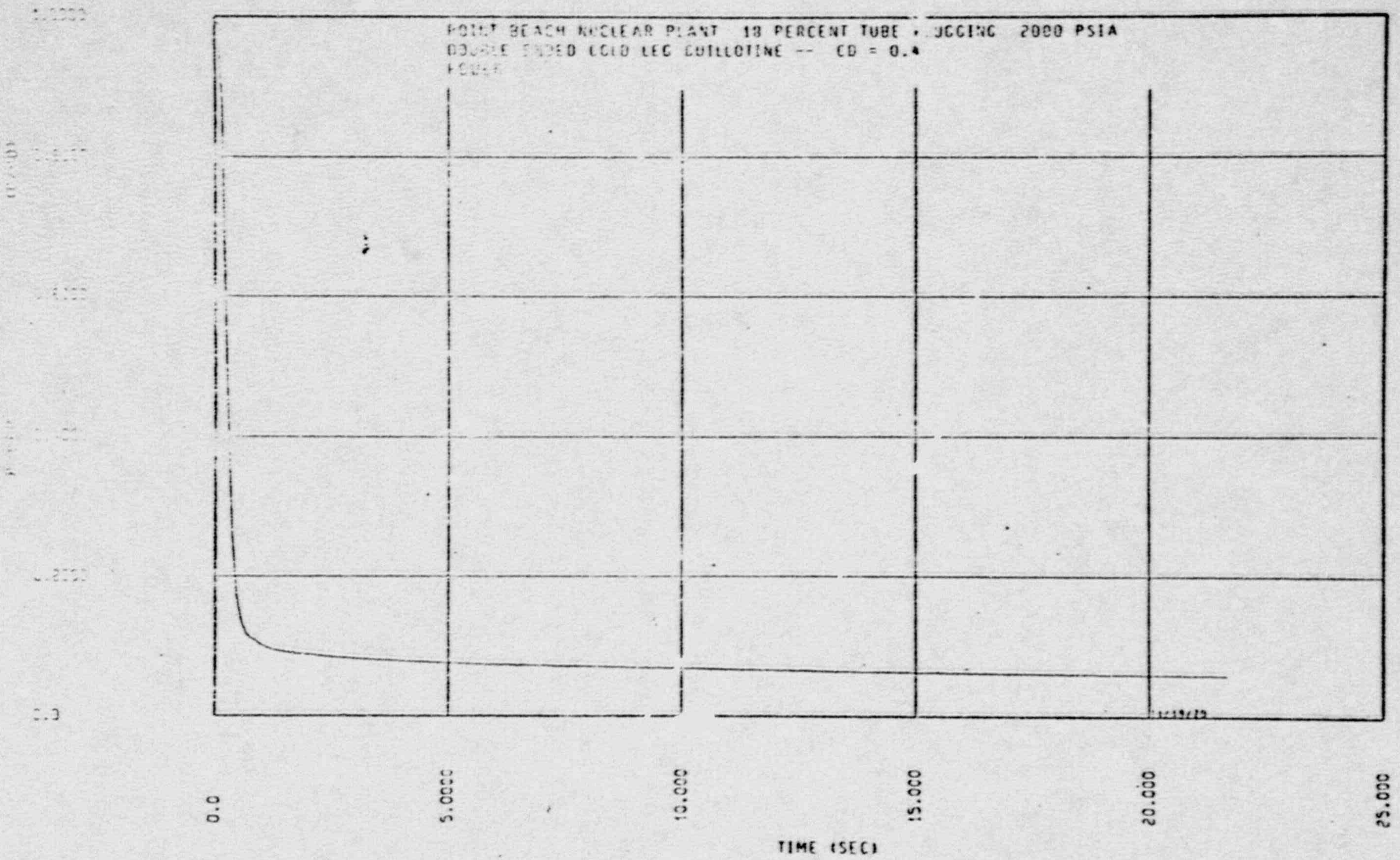


Figure 14



POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
0.0
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 15

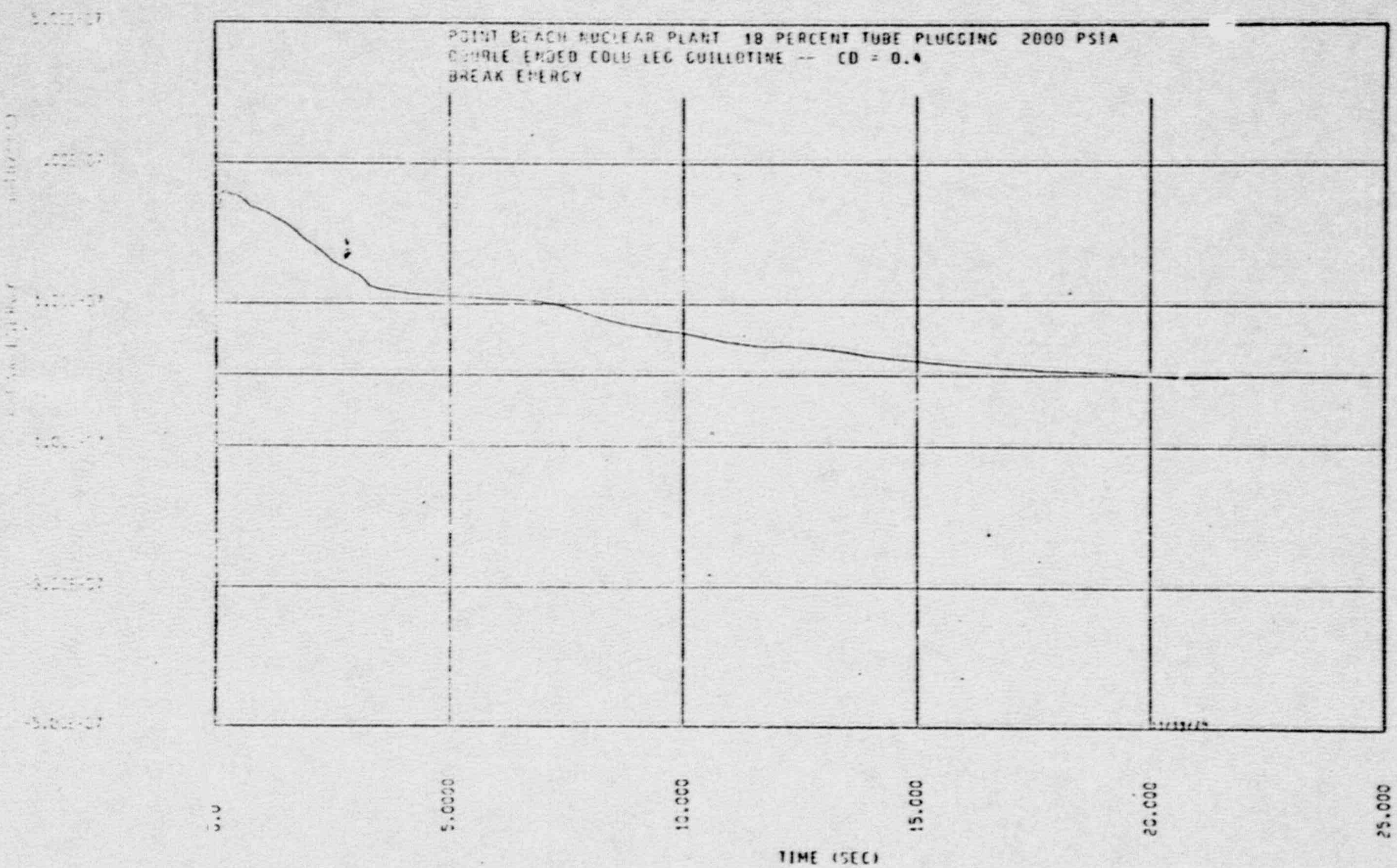

POINT BEACH NUCLEAR PLANT 18 PERCENT TUBE PLUGGING 2000 PSIA
DOUBLE ENDED COLD LEG GUILLOTINE -- CD = 0.4
BREAK ENERGY
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 16

46 1320

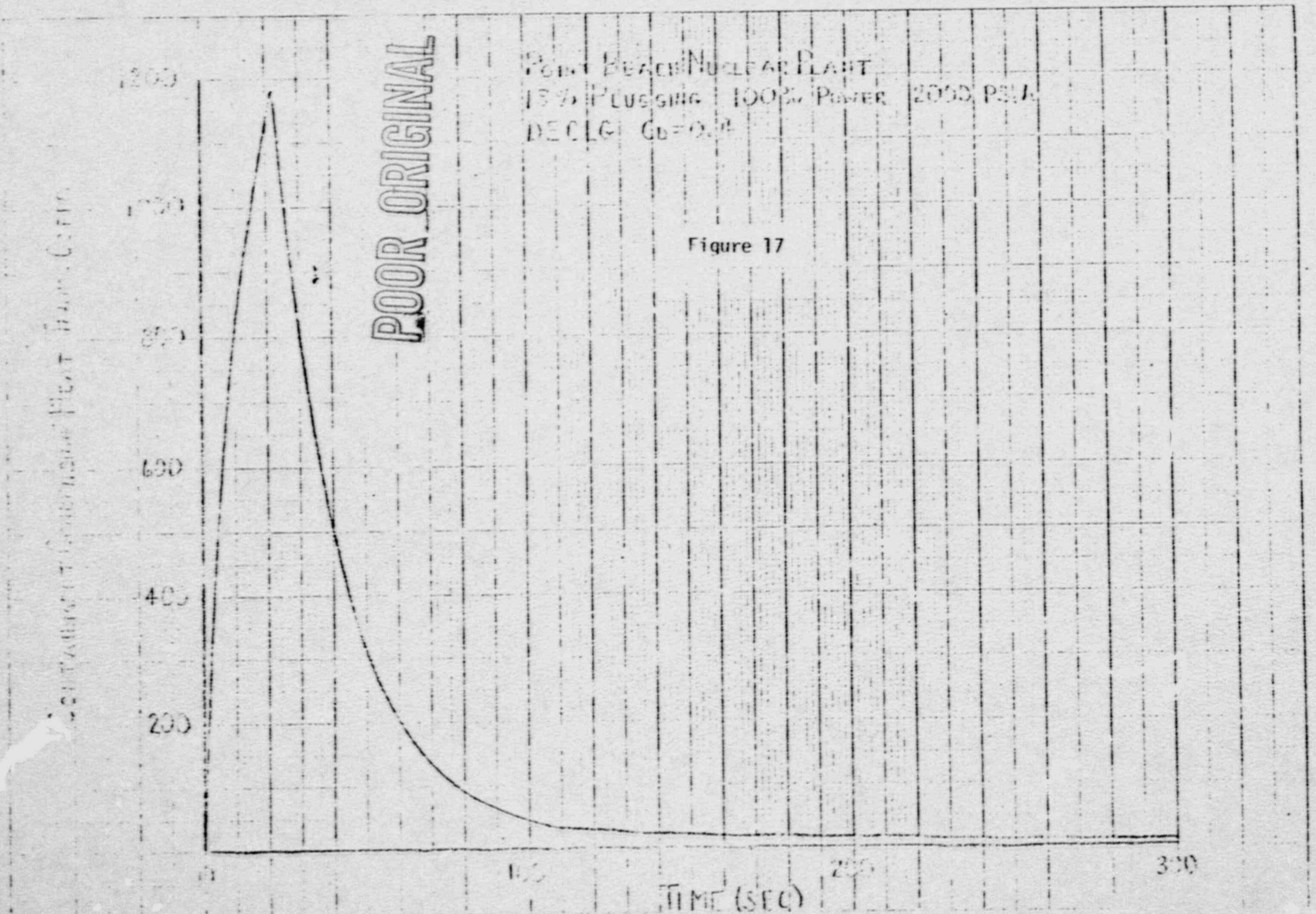


Figure 17