50-280/281

ATTACHMENT 1 - LARGE BREAK LOCA - ECCS REANALYSIS dtd 10-29-76

ATTACHMENT 2 - CHANGE NO. 47 to Tech Specs dtd 10-29-76

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Large Break LOCA-ECCS Reanalysis

Surry Power Station
Units No. 1 and 2

October 29, 1976

Attachment 1

1.0 INTRODUCTION

The reanalysis of ECCS cooling performance for the postulated large break Loss of Coolant Accident (LOCA), as required by the Order for Modification of License for Surry Power Station Units No. 1 and 2 issued by the Nuclear Regulatory Commission on August 27, 1976 [18] is presented herein.* This reanalysis is in compliance with 10CFR50.46 [1], Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors. This reanalysis was performed with the October, 1975 version of the Westinghouse Evaluation Model. The analytical techniques utilized in the reanalysis are in compliance with Appendix K to 10CFR50, and are documented in References (2,13, 15, and 16).

As required by Appendix K of 10CFR50, certain conservative assumptions were made for the LOCA analysis. These assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs, and include such items as the core peaking factors, the containment pressure, and the performance of the emergency core cooling system (ECCS). All assumptions and initial operating condition input data used in this reanalysis were the same as was used in the previously applicable LOCA-ECCS analysis (see our letter of June 6, 1975 - Serial No. 500-S) [8] except for (1) the limiting value of heat flux hot channel factor (changed from 2.1 to 2.0), (2) the minimum allowable value of the containment temperature (changed from 75°F to 90°F), (3) the number of steam generator tubes assumed to be plugged (see our letters of May 14, 1976 (Serial No. 017/043073) [9], August 18, 1976 (Serial No. 194) [10], and October 19, 1976 (Serial No. 260/092276)) [11], (4) the temperature of the fluid in the reactor vessel upper head region (changed from a temperature equal to the temperature in the cold leg to a temperature equal to the temperature in the hot leg [14,17,18], i.e.,

*It should be noted that a reanalysis of the small break LOCA is not necessary and therefore the analysis of this accident submitted in our letter of June 6, 1975 (Serial No. 500-S) [8] remains applicable.

100% of T_{hot}), and (5) for one case, as discussed below, the minimum level of accumulator water volume (changed from 975 ft^3 to 1075 ft^3).

2.0 DESCRIPTION OF POSTULATED MAJOR REACTOR COOLANT PIPE RUPTURE (LOSS OF COOLANT ACCIDENT - LOCA)

A LOCA is the result of a rupture of the Reactor Coolant System (RCS) piping or of any line connected to the system. The boundary considered for the LOCA as applicable to this connected piping is defined in the FSAR. Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A Safety Injection System (SIS) signal is actuated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. (It should be noted, however, that no credit is taken in the analysis for the insertion of control rods to shut down the reactor.)
2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the reactor coolant system. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10CFR50. Thereafter the core heat transfer is based on local conditions with transition boiling and forced convection of steam as the major heat transfer mechanisms. During the refill period, it is assumed that rod-to-rod radiation is the only core heat transfer mechanism. The heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and the main safety valves may actuate to reduce the pressure. Make-up to the secondary side is automatically provided by the auxiliary feedwater system. The safety injection signal stops normal feedwater flow by closing the main feedwater control valves, trips the main feedwater pumps and initiates emergency feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of reactor coolant system pressure. When the reactor coolant system depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is then made that injected accumulator water bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10CFR50. In addition, the reactor coolant pumps are assumed to be tripped at the initialization of the accident and effects of pump coastdown are included in the blowdown analyses.

The water injected by the accumulators cools the core and subsequent operation of the low head safety injection pumps supply water for long term cooling. After the contents of the refueling water storage is emptied, long term cooling of the core is accomplished by switching to the recirculation mode of core cooling, in which the spilled borated water is drawn from the containment sump by the low head safety injection pumps and returned to the reactor vessel.

The containment spray system and the recirculation spray system operate to return the containment to subatmospheric pressure.

3.0 ANALYSIS

The large break LOCA transient is divided, for analytical purposes, into three phases: blowdown, refill, and reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of inter-related computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339.[2] This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in WCAP-8306[3], WCAP-8171[5], WCAP-8326[6], and WCAP-8305[4], respectively. These codes are able to assess whether sufficient heat transfer geometry and core amenability to cooling are preserved during the time spans applicable to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code is used to calculate this transient during the refill and reflood phases of the accident. The COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analyais. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.

SATAN-VI is used to determine the RCS pressure, enthalpy, and density, as well as the mass and energy flow rates in the RCS and steam generator secondary as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator mass and pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown phase, these data are transferred to the WREFLOOD code. Also at the end of blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end of blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow rates that are assumed to be vented to the containment. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. WREFLOOD is also linked to the LOCTA-IV code in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature.

LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel and clad temperature of the hottest rod in the core. The input to LOCTA-IV which consists of appropriate thermal-hydraulic output from SATAN-VI and WREFLOOD and the conservatively chosen initial core operating conditions summarized in Tables 3a, 3b, and 3c and Figure 18. (The axial power shape assumed for LOCTA-IV (curve 1, Figure 18) is a cosine curve which has been previously verified to be the shape that produces the maximum peak clad temperature.[7, 12, 13])

The COCO code, which is also used throughout the LOCA analysis, calculates the containment pressure. Input to COCO is obtained from the mass and energy flow rates assumed to be vented to the containment as calculated by the

SATAN-VI and WREFLOOD codes. In addition, conservatively chosen initial containment conditions and an assumed mode of operation for the containment cooling system are input to COCO.

4.0 RESULTS

Based on the results of the LOCA sensitivity studies,[7,12,14] the limiting large break was found to be the double-ended cold leg guillotine (DECLG) break of the RCS. Therefore only the DECLG break results are reported.

The results of three sets of initial operating conditions are provided for the present reanalysis. Table 1 defines Cases A, B, and C for the three sets of initial operating conditions that have been assumed for this reanalysis. Tables 2a, 2b, and 2c present the time sequence of events for Cases A, B, and C, respectively. Tables 3a, 3b, and 3c present the results and the major initial conditions for the three cases which are discussed in more detail below.

The base case, Case A, assumed a steam generator tube plugging level of 7 percent per steam generator and minimum accumulator water level of 975 ft^3. Table 3a presents results for the DECLG for the values of three discharged coeffcients (C_D). This range of discharge coefficients was determined to include the limiting case for peak clad temperature from sensitivity studies reported in WCAP-8356[7], WCAP-8572[12], and WCAP-8853[14]. The limiting base case (Case A) break, as in the previous analysis, was found to be the $C_D = 0.4$ break and resulted in a peak clad temperature of 2074^oF, a maximum local metal-water reaction of 5.6 percent, and a total core metal-water reaction of less than 0.3 percent.

An additional analysis was then conducted with the limiting ($C_D = 0.4$) break size in order to determine the sensitivity of increased steam generator tube plugging on peak clad temperature. The Case B assumptions were the same as for Case A except that the steam generator tube plugging level was assumed to be 10 percent. These results are presented in Table 3b and incate a peak clad temperature of 2091°F, a maximum local metal-water reaction of 5.9 percent, and a total core metal-water reaction of less than 0.3 percent.

Case C, the final case, assumed a change in accumulator water volume from 975 ft^3 to 1075 ft^3. This was done because it was found that for steam generator tube plugging approaching 11 percent and an accumulator water volume of 975 ft^3 the peak clad temperature slightly exceeded 2200°F. In order to obtain more margin in peak clad temperature to accomodate the potential need for higher steam generator tube plugging levels, the effect of an increase in accumulator water volume was investigated and found to be beneficial. Table 3c provides results for Case C which assumed a steam generator tube plugging level of 12 percent and an accumulator water volume of 1075 ft^3. The results indicate a peak clad temperature of 2107°F, a maximum local metal-water reaction of 6.2 percent, and a total core metal-water reaction of less than 0.3 percent.

Finally, for information purposes, an analysis was conducted to determine the impact of asymmetric steam generator tube plugging on peak clad temperature. Limiting asymmetric conditions were investigated and the result indicated no adverse impact on the limiting peak clad temperature. Currently, less than 12% of the steam generator tubes are plugged in the Surry units and the plugging distribution is essentially symmetric.

The detailed results of the LOCA reanalysis for Cases A, B, and C are provided in Tables 1 through 7 and Figures 1 through 18.

5.0 CONCLUSIONS

For breaks up to and including the double ended severance of a reactor coolant pipe and for the operating conditions specified by Cases A, B, and C, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10CFR50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.
3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.
4. The core remains amenable to cooling during and after the break.
5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining the core.

REFERENCES

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

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

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

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

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

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

7. Buterbaugh, T. L., Johnson, W. J. and Kopelic, S. D., "Westinghouse ECCS-Plant Sensitivity Studies," WCAP-8356, July 1974.

8. Letter from C. M. Stallings (Vepco) to K. R. Goller (NRC), Serial No. 500-S, dated June 6, 1975.

9. Letter from C. M. Stallings (Vepco) to B. C. Rusche (NRC), Serial No. 017/043073, dated May 14, 1976.

10. Letters from C. M. Stallings (Vepco) to B. C. Rusche (NRC), Serial No. 194, dated August 18, 1976, and Serial No. 211, dated August 26, 1976.

11. Letter from C. M. Stallings (Vepco) to B. C. Rusche (NRC), Serial No. 260/092276, dated October 19, 1976.

12. Buterbaugh, T. L., Julian, H. V., Tome, A. E., "Westinghouse ECCS-Three Loop Plant (17 x 17) Sensitivity Studies," WCAP-8572 July 1975 (Proprietary) and WCAP-8573, July 1975 (Non-Proprietary).

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

14. Julian, H. V., Tabone, C. J., and Thompson, C. M., "Westinghouse ECCS-Three Loop Plant (17 x 17) Sensitivity Studies, WCAP-8853, September 1976 (Non-Proprietary).

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

16. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. F. Vassallo of the Nuclear Regulatory Commission, Letter Number NS-CE-9, dated January 23, 1976.

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

18. Letter from R. W. Reid (NRC) to W. L. Proffitt, Serial No. 219/082776, dated August 27, 1976.

Table 1

INITIAL OPERATING ASSUMPTIONS FOR THE SURRY UNITS 1 AND 2

LOCA-ECCS REANALYSIS

CASE A

This is the base case reanalysis which assumed the following:

1) total peaking factor (F_Q) of 2.0
2) minimum temperature of the containment of 90°F
3) uniform steam generator tube plugging of 7 percent
4) accumulator water volume of 975 ft^3 (per accumulator)
5) temperature of the fluid in the reactor vessel upper head region equal to 100 percent of T_{HOT}

CASE B

This case extended the results of Case A (limiting break (C_D=0.4)) to indicate the effect on peak clad temperature of increased steam generator tube plugging. The following assumptions were made:

1) total peaking factor (F_Q) of 2.0
2) minimum temperature of the containment of 90°F
3) uniform steam generator tube plugging of 10 percent
4) accumulator water volume of 975 ft^3 (per accumulator)
5) temperature of the fluid in the reactor vessel upper head region equal to 100 percent of T_{HOT}

CASE C

This case extended the results of Cases A and B to indicate the effect on peak clad temperature of increased accumulator water volume and further increased steam generator tube plugging. The following assumptions were made:

1) total peaking factor (F_Q) of 2.0
2) minimum temperature of the containment of 90°F
3) uniform steam generator tube plugging of 12 percent
4) accumulator water volume of 1075 ft^3 (per accumulator)
5) temperature of the fluid in the reactor vessel upper head region to 100 percent of T_{HOT}

Table 2a

TIME SEQUENCE OF EVENTS

CASE A

	DECLG (C_D=1.0)	DECLG (C_D=0.6)	DECLG (C_D=0.4)
START	0.0 sec.	0.0 sec.	0.0 sec.
Reactor Trip Signal	0.623	0.635	0.648
S. I. Signal	1.45	1.81	2.25
Acc. Injection	10.5	12.9	16.6
End of Bypass	21.71	23.84	24.94
End of Blowdown	23.16	26.91	31.20
Bottom of Core Recovery	35.66	37.62	37.75
Acc. Empty	42.3	44.75	43.64
Pump Injection	26.45	26.81	27.25

Table 2b

TIME SEQUENCE OF EVENTS

CASE B

	DECLG (C_D=0.4)
START	0.0 sec.
Reactor Trip Signal	0.648
S. I. Signal	2.23
Acc. Injection	16.4
End of Bypass	24.67
End of Blowdown	29.0
Bottom of Core Recovery	37.45
Acc. Empty	48.32
Pump Injection	27.23

Table 2c

TIME SEQUENCE OF EVENTS

CASE C

	DECLG (C_D=0.4)
START	0.0 sec.
Reactor Trip Signal	0.648
S. I. Signal	2.23
Acc. Injection	16.2
End of Bypass	24.26
End of Blowdown	27.81
Bottom of Core Recovery	37.88
Acc. Empty	55.99
Pump Injection	27.23

Table 3a

RESULTS - CASE A

Results	DECLG (C_D=1.0)	DECLG (C_D=0.6)	DECLG (C_D=0.4)
Peak Clad Temp., °F	1927	1981	2074
Peak Clad Location, ft.	9.0	9.0	9.0
Local Zr/H_2O Reaction,(max) %	2.748	4.354	5.601
Local Zr/H_2O Location, ft.	9.0	9.0	9.0
Total Zr/H_2O Reaction, %	<0.3	<0.3	<0.3
Hot Rod Burst Time, sec.	63.2	33.6	26.3
Hot Rod Burst Location, ft.	6.0	6.0	6.0

Initial Conditions	
Core Power, Mwt, 102% of	2441
Peak Linear Power, kw/ft, 102% of	12.49
Peaking factor	2.00
Accumulator Water Volume, ft^3	975

Most Limiting Fuel Region	Cycle	Region
UNIT 1	All	4
UNIT 2	All	4

Table 3b

RESULTS - CASE B

Results	DECLG (C_D=0.4)
Peak Clad Temp., ^oF	2091
Peak Clad Location, ft.	9.0
Local Zr/H_2O Reaction (max), %	5.939
Local Zr/H_2O Location, ft.	9.0
Total Zr/H_2O Reaction, %	< 0.3
Hot Rod Burst Time, sec.	26.3
Hot Rod Burst Location, ft.	6.0

Initial Conditions	
Core Power, Mwt, 102% of	2441
Peak Linear Power, kw/ft, 102% of	12.49
Peaking Factor	2.00
Accumulator Water Volume, ft^3	975

Most Limiting Fuel Region	Cycle	Region
UNIT 1	All	4
UNIT 2	All	4

Table 3c

RESULTS - CASE C

Results	DECLG (C_D=0.4)
Peak Clad Temp., ^oF	2107
Peak Clad Location, ft.	9.0
Local Zr/H_2O Reaction (max), %	6.234
Local Zr/H_2O Location, ft.	9.0
Total Zr/H_2O Reaction, %	< 0.3
Hot Rod Burst Time, sec.	28.2
Hot Rod Burst Location, ft.	6.0

Initial Conditions	
Core Power, Mwt, 102% of	2441
Peak Linear Power, kw/ft, 102% of	12.49
Peaking factor	2.00
Accumulator Water Volume, ft^3	1075

Most Limiting Fuel Region	Cycle	Region
UNIT 1	All	4
UNIT 2	All	4

Table 4

CONTAINMENT DATA (DRY CONTAINMENT)

NET FREE VOLUME	1.863×10^6 Ft^3
INITIAL CONDITIONS	
Pressure	9.35 psia
Temperature	90 ^oF
RWST Temperature	40 ^oF
Service Water Temperature	35 ^oF
Outside Temperature	9 ^oF
SPRAY SYSTEM I	
Number of Pumps Operating	2
Runout Flow Rate	3200 gpm
Actuation Time	20 secs
SPRAY SYSTEM II--RECIRCULATION SPRAY FROM PUMP	
Number Pumps Operating	2
Runout Flow Rate (each)	3200 gpm
Actuation Time	125 secs
Heat Exchanger {UA(per pump)}	3.5×10^6 Btu/hr-^oF
Service Water Flow (per exchanger)	6100 gpm

STRUCTURAL HEAT SINKS

Thickness(in.)	Area(Ft^2), w/uncertainty
Concrete, 6	6,972
Concrete, 12	57,960
Concrete, 18	40,470
Concrete, 24	10,500
Concrete, 36	4,410
Carbon Steel, 0.375 / Concrete, 54	46,887
Carbon Steel, 0.50 / Concrete, 30	25,075
Concrete, 24	11,284
Carbon Steel, 0.366	167,165
Stainless Steel, 0.426	3,399

TABLE 5a

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TABLE 5b

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Table 6a

REFLOOD MASS AND ENERGY RELEASES FOR LIMITING CASE

AT 10 PERCENT PLUGGING - DECLG (C_D=0.4)

Time (sec)	Total Mass Flow Rate (lb/sec)	Total Energy Flow Rate (10^5 Btu/sec)
37.448	0.0	0.0
38.373	0.0	0.0
38.473	0.0435	0.0006
43.382	35.53	0.4620
52.145	206.8	1.436
65.545	243.2	1.482
81.845	255.8	1.464
100.045	263.0	1.426
119.645	269.2	1.382
215.045	301.8	1.124
278.945	320.9	0.9728
366.754	333.3	0.8697

Table 6b

BROKEN LOOP ACCUMULATOR FLOW RATE

FOR LIMITING CASE AT 10 PERCENT PLUGGING - DECLG (C_D=0.4)

Time (sec)	Mass Flow Rate (lbm/sec)
0.0	0.0
1.0	3.74
2.0	3.47
3.0	3.26
4.0	3.07
5.0	2.91
6.0	2.77
7.0	2.65
8.0	2.54
9.0	2.44
10.0	2.35
11.0	2.27
12.0	2.19
13.0	2.11
14.0	2.04
15.0	1.99
16.0	1.94
18.0	1.86
20.0	1.78
22.0	1.71
24.0	1.78
26.0	1.72
28.0	0.0
30.63	0.0

* For energy mass flow rate, multiply mass flow rate by a constant of 60 Btu/lbm.

Table 7b

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT

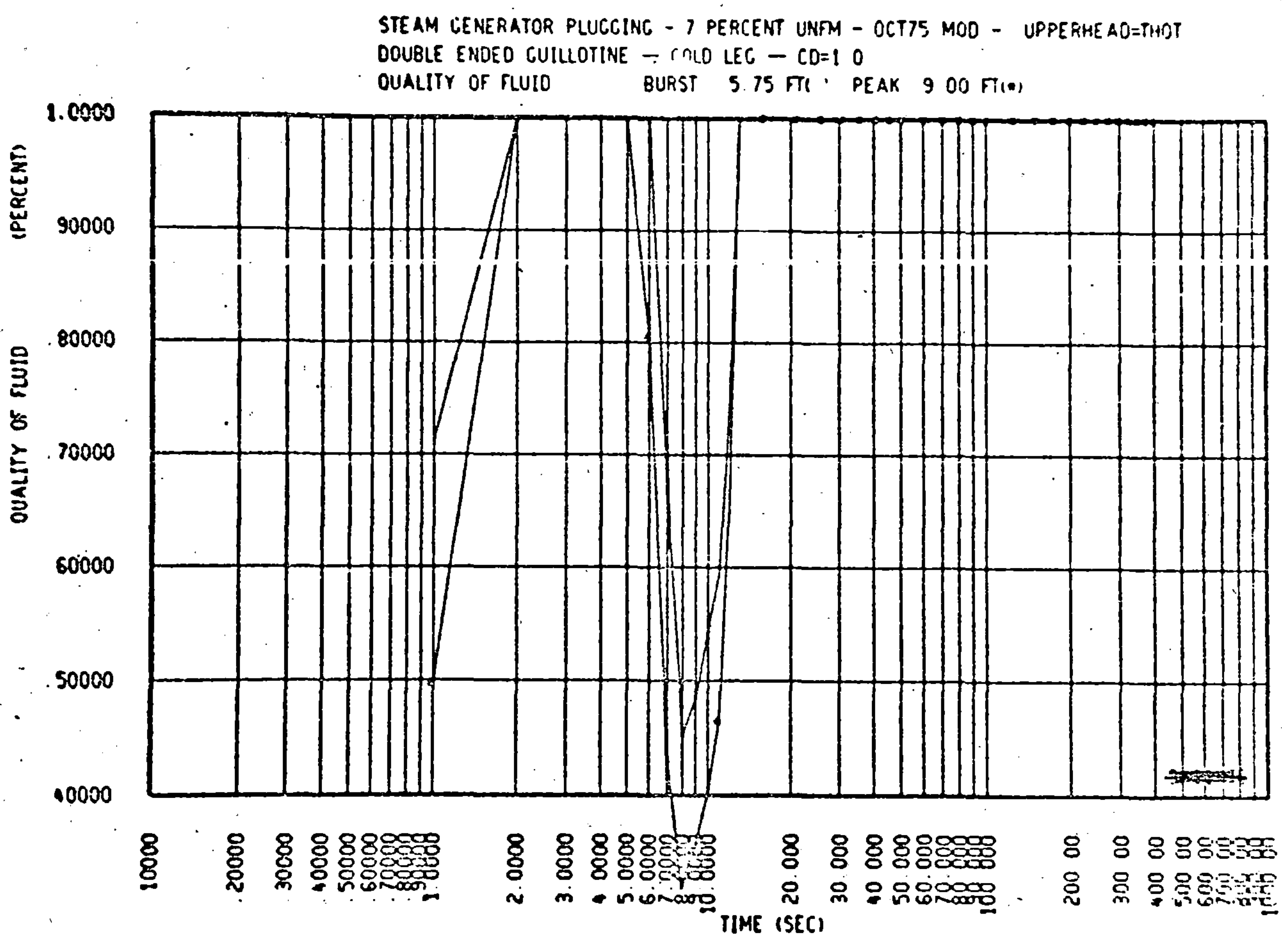
For Limiting Case At 12 Percent Plugging - DECLG (C_D = 0.4)

TIME (SEC)	MASS FLOWRATE* (LBm/SEC)
0.0	4107
1.0	3738
3.0	3232
5.0	2891
7.0	2639
10.0	2355
15.0	2024
20.0	1800
23.01	1703
32.27	0.0

*For energy mass flowrate multiply mass flowrate by a constant of 58 BTU/lbm.

FIGURE 1a

CASE A

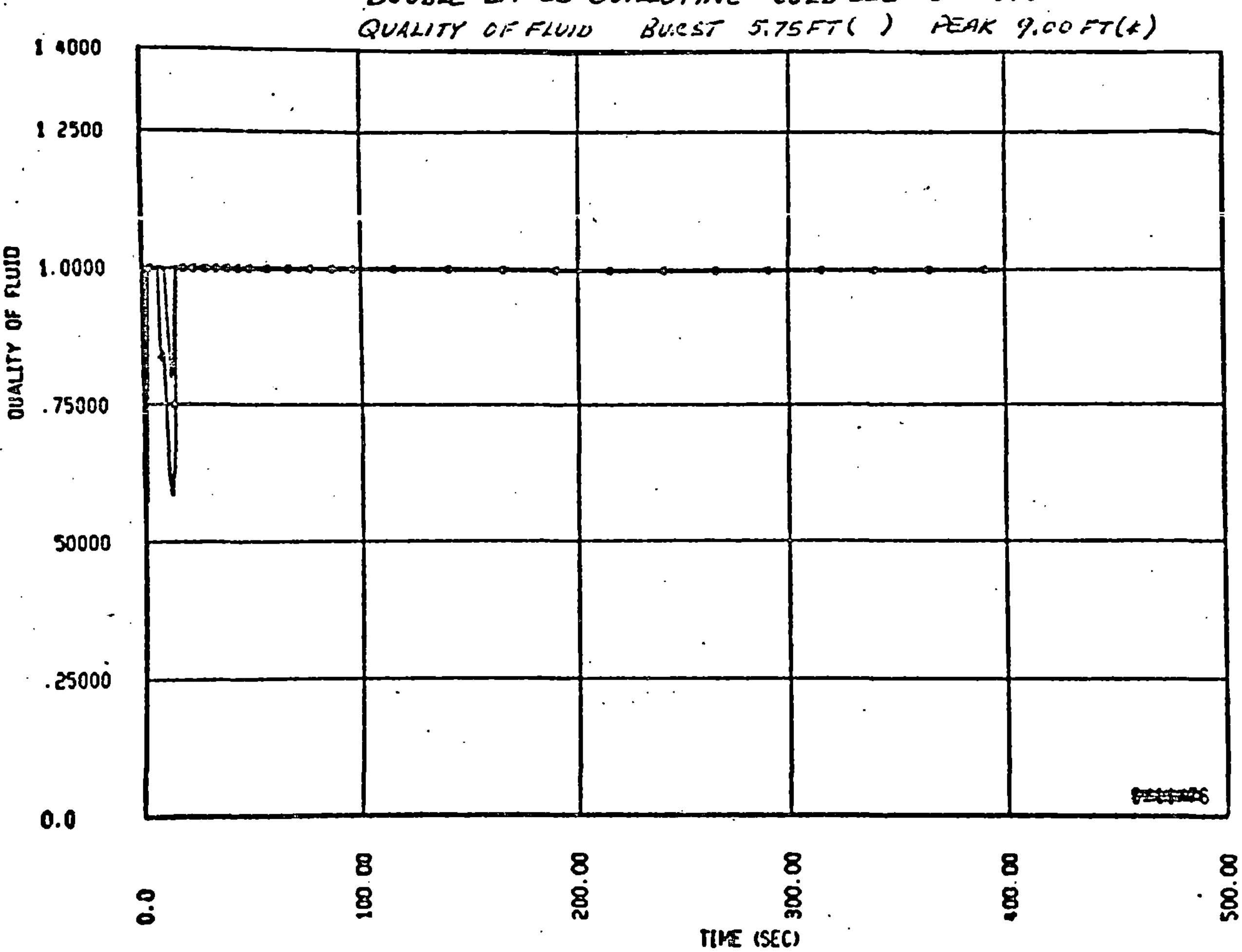


FLUID QUALITY - DECLG (C_D =1.0)

FIGURE 1b

CASE A

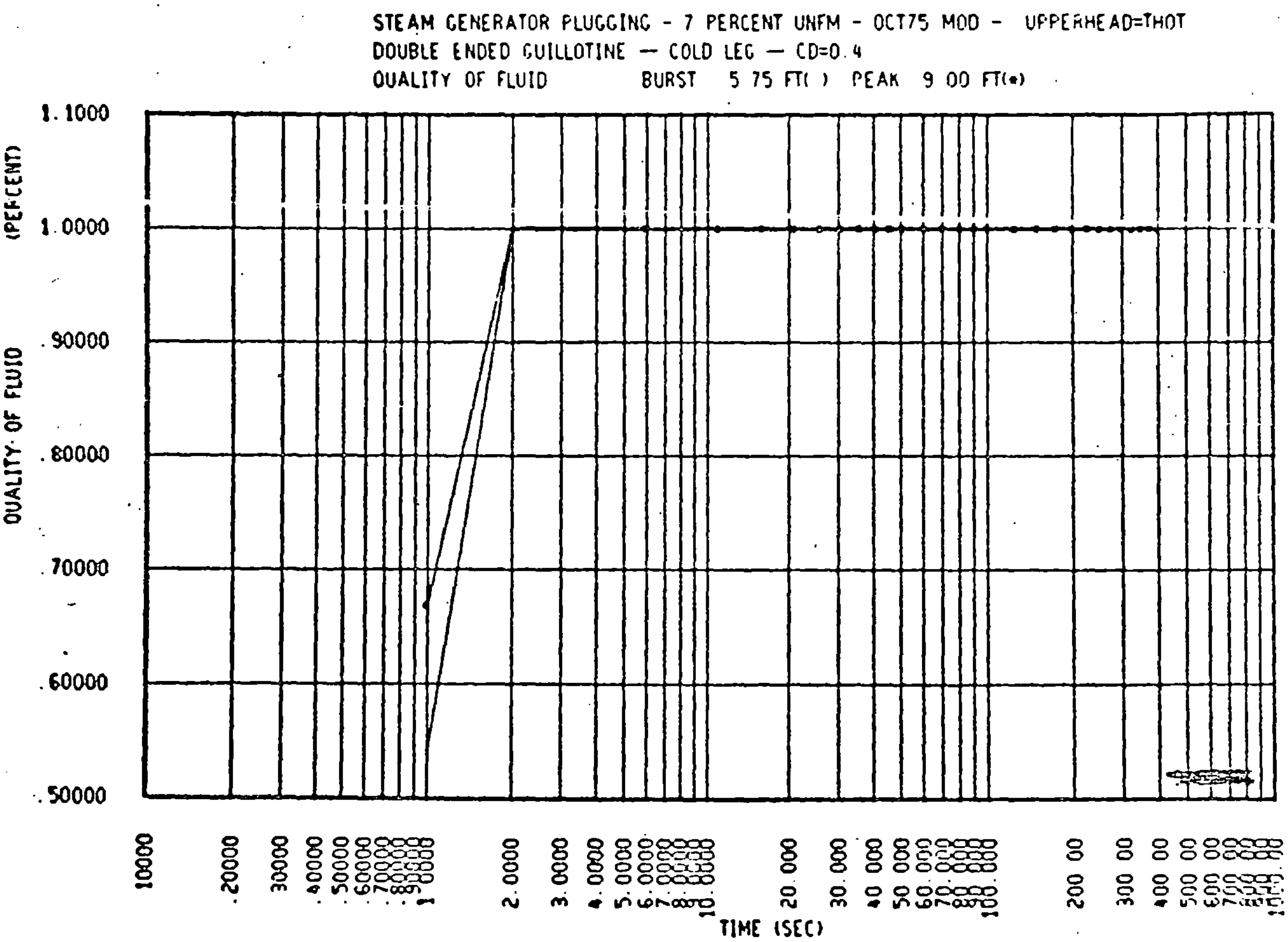
STEAM GENERATOR PLUGGING - 7 PERCENT UNFM - OCT75 MOD - UPPERHEAD=THOT
DOUBLE ENDED GUILLOTINE - COLD LEG - CD=0.6



FLUID QUALITY - DECLG (C_D = 0.6)

FIGURE 1c

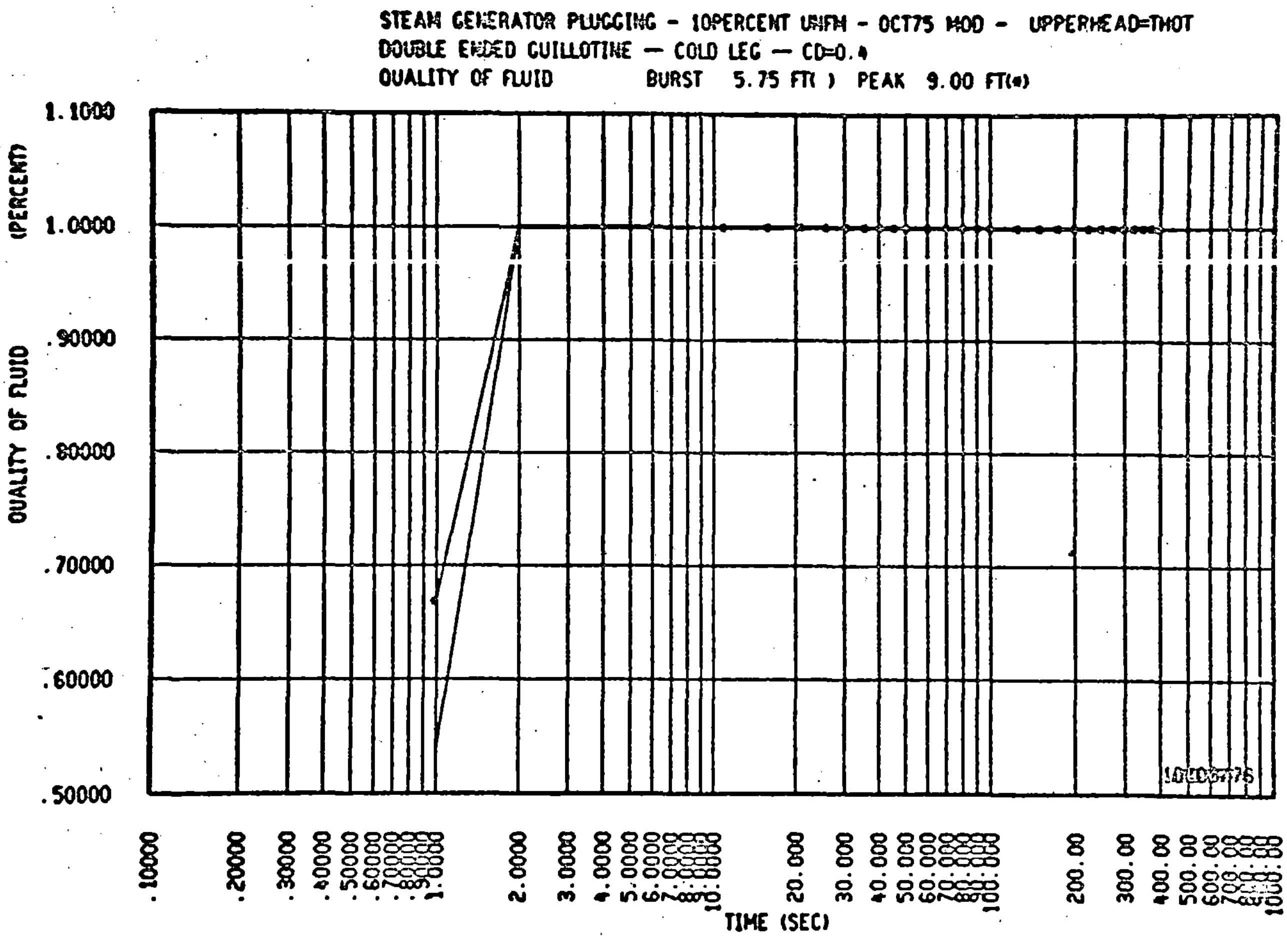
CASE A



FLUID QUALITY - DECLG (CD = 0.4)

FIGURE 1d

CASE B



FLUID QUALITY - DECLG (CD = 0.4)

STEAM GENERATOR PLUGGING - 12PERCENT UNFM - OCT75 MOD - UPPERHEAD=THOT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD=0.4
QUALITY OF FLUID BURST 5.75 FT( ) PEAK 9.00 FT(*)

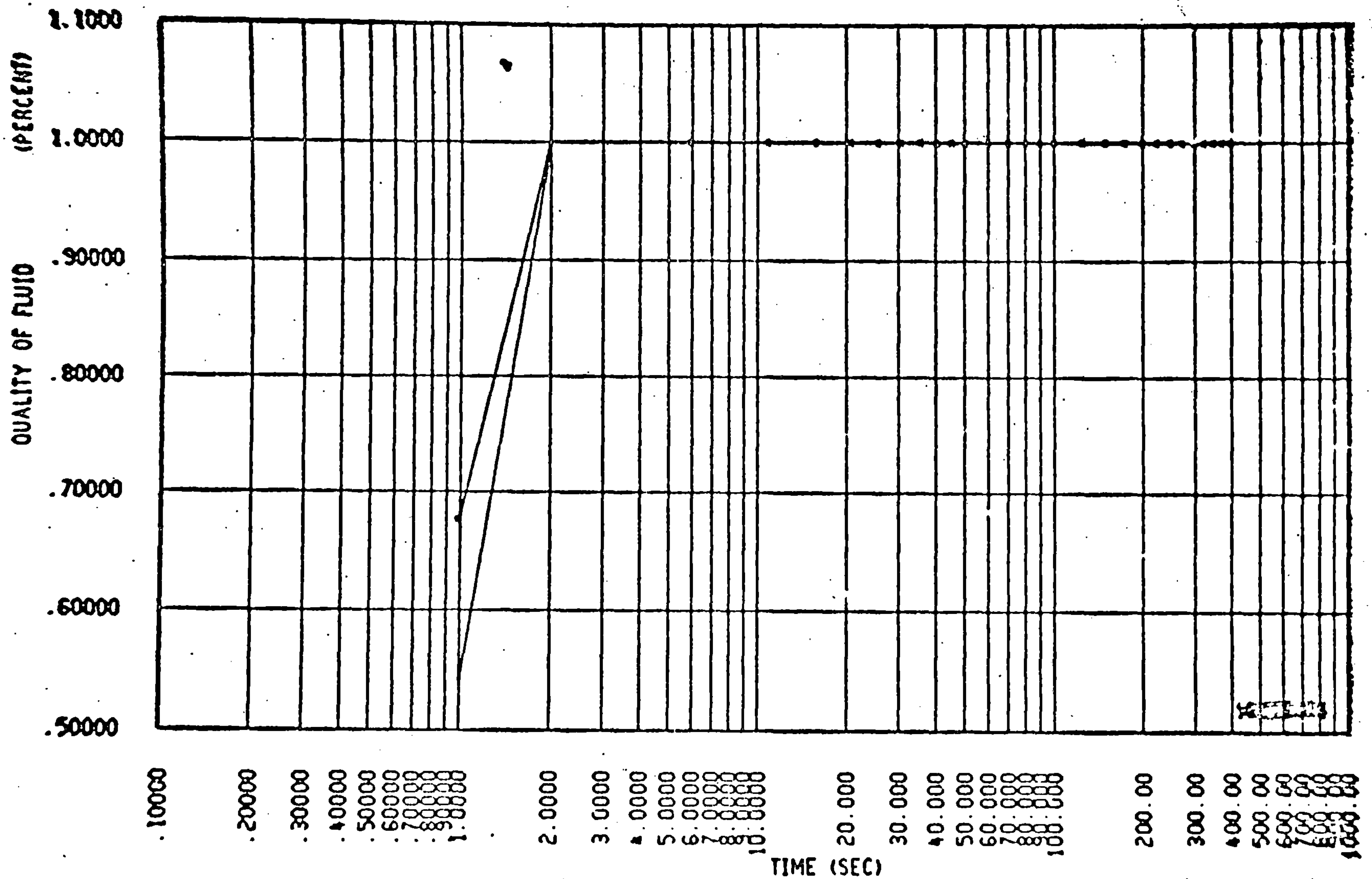


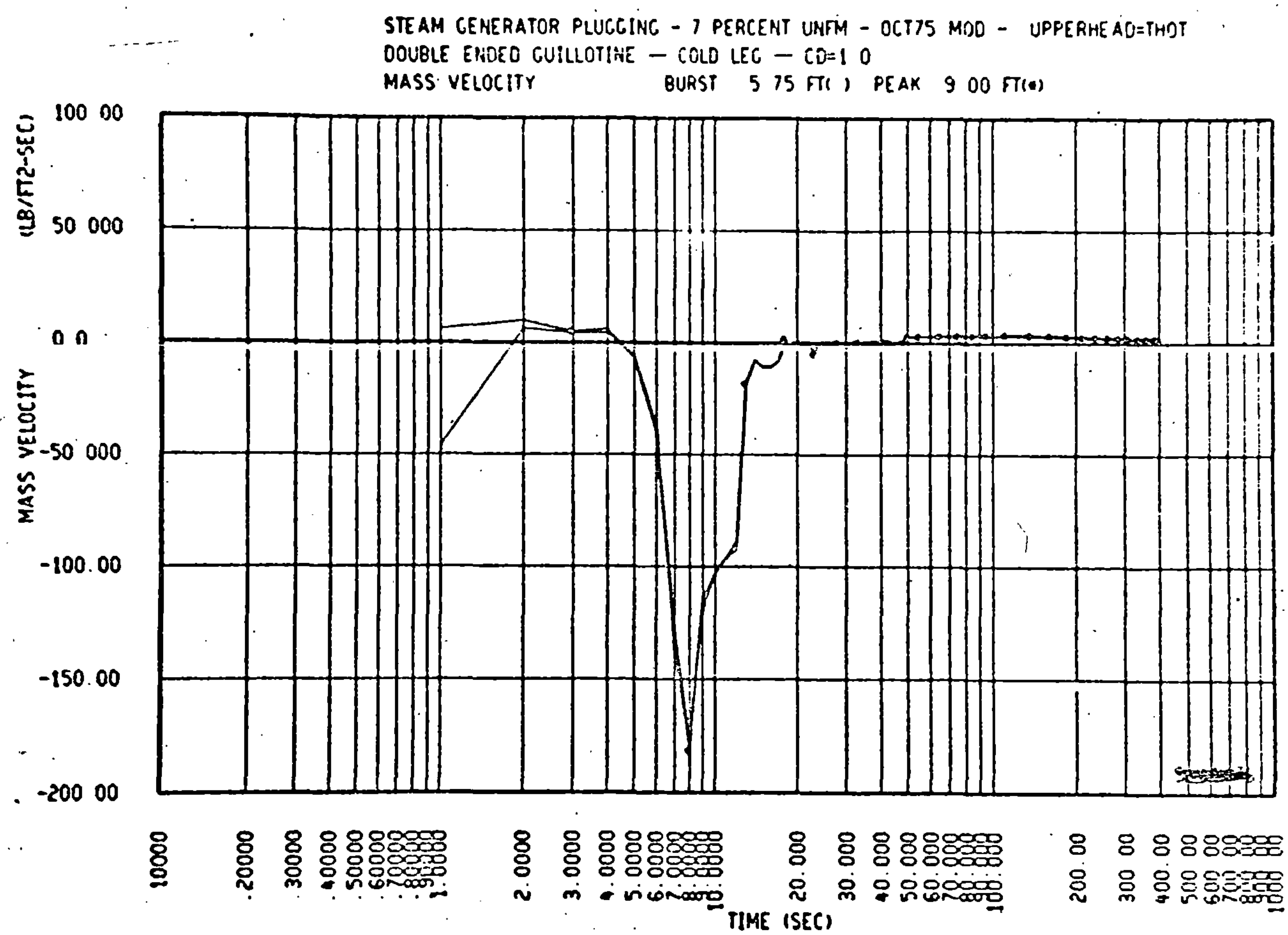
FIGURE 1e

CASE C

FLUID QUALITY - DECLG (CD = 0.4)

FIGURE 2a

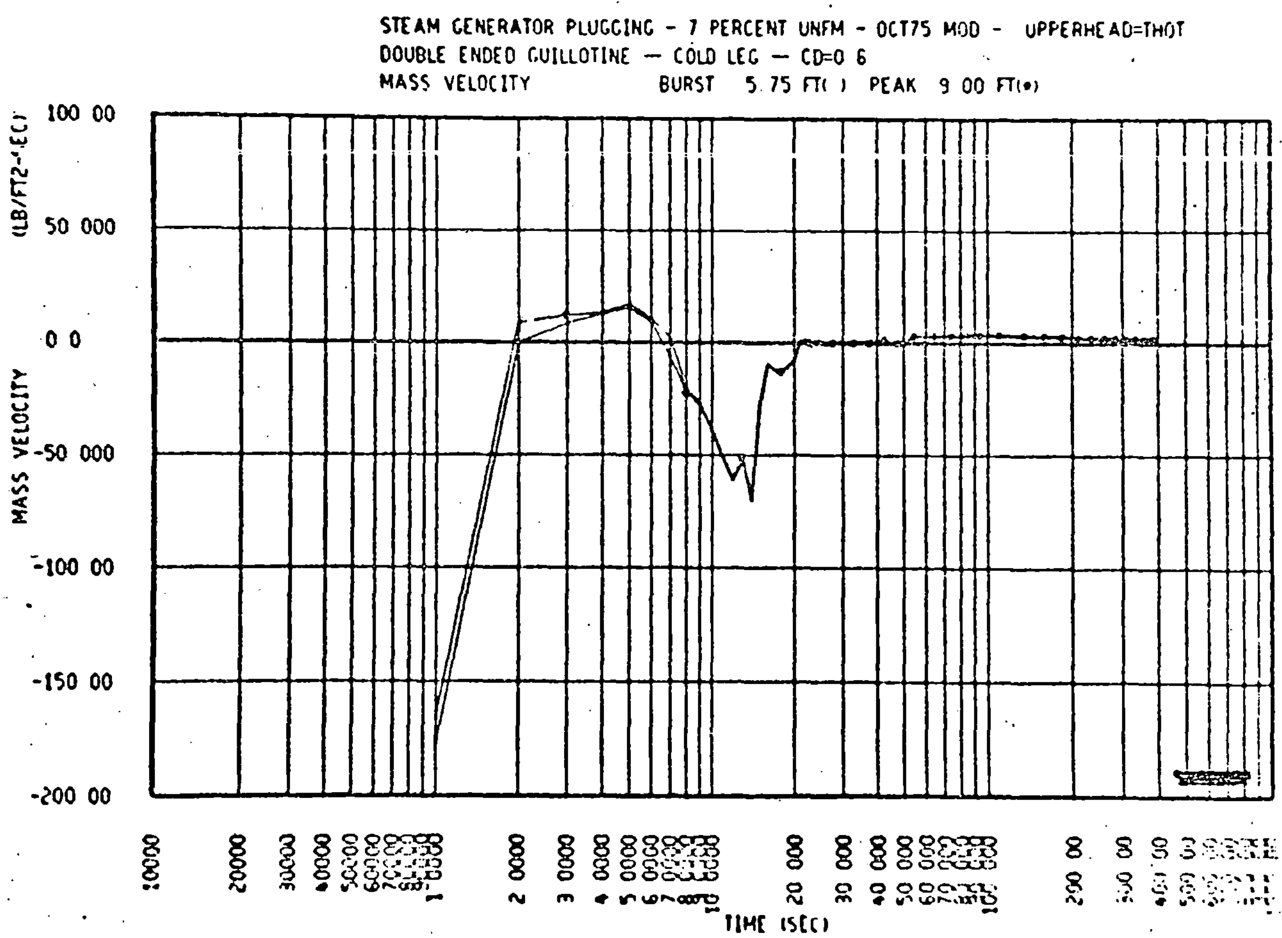
CASE A



MASS VELOCITY - DECLG (CD=1.0)

FIGURE 2b

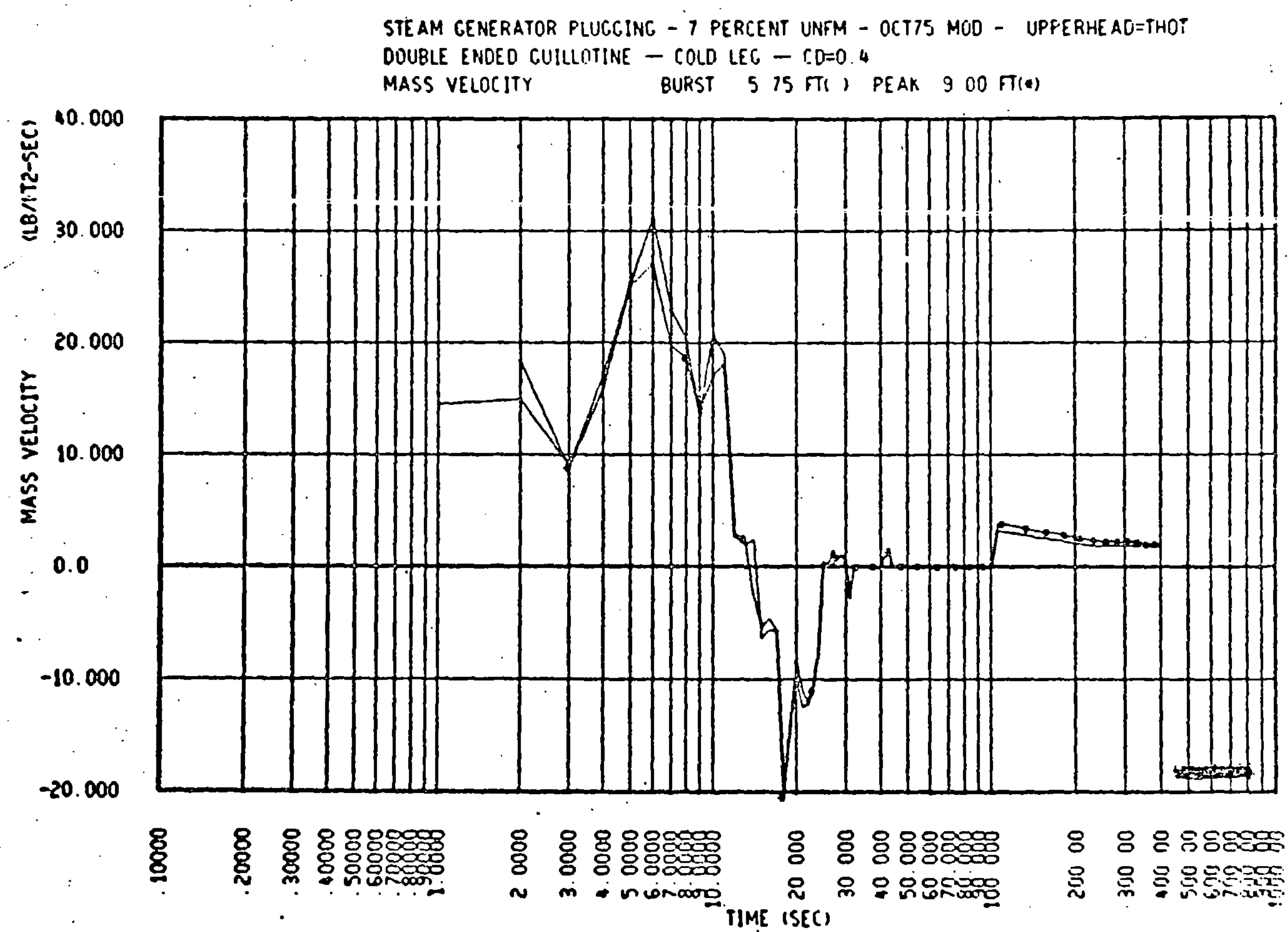
CASE A



MASS VELOCITY - DECLG (CD=0.6)

FIGURE 2c

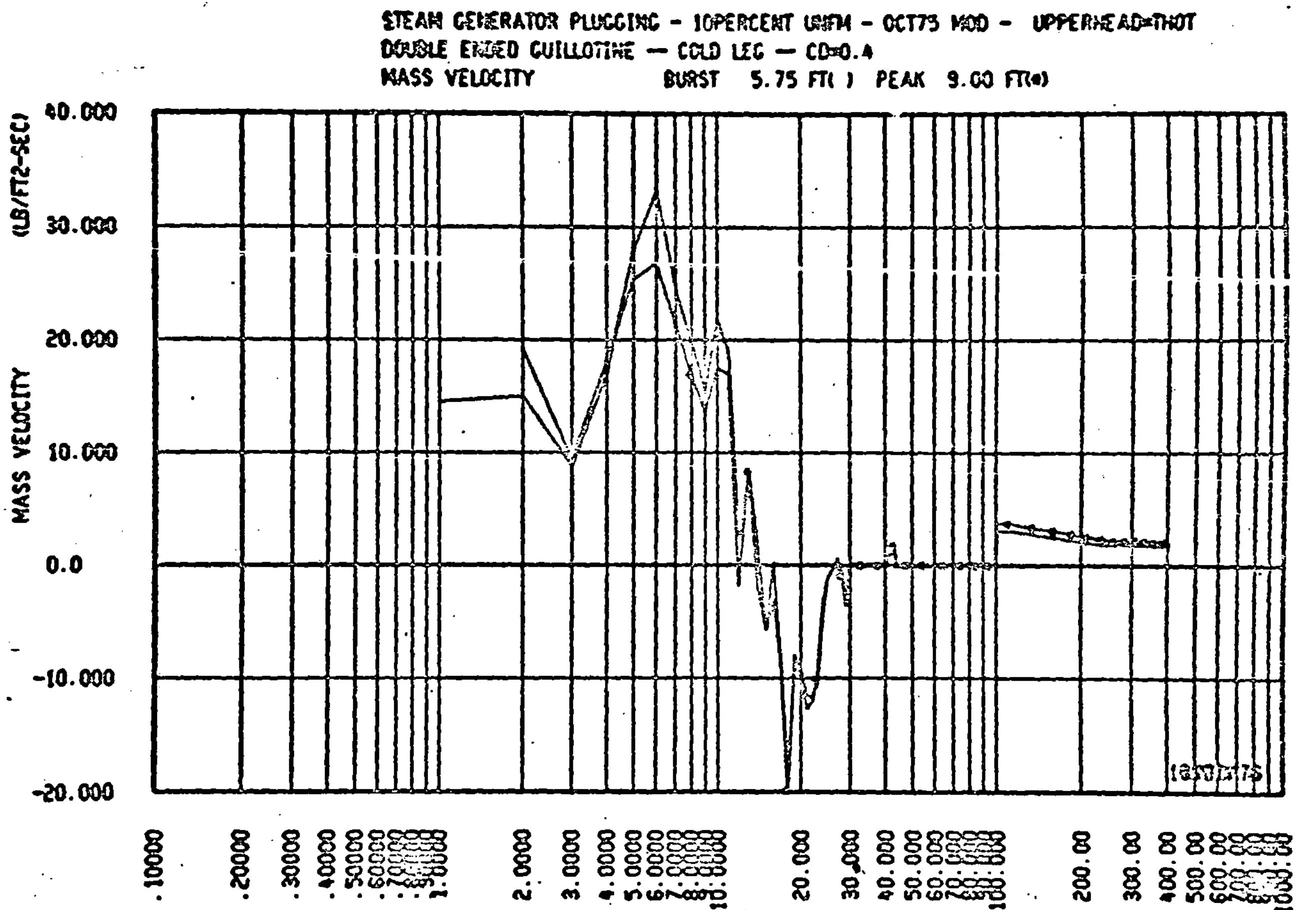
CASE A



Mass Velocity - DECLG (Cd=0.4)

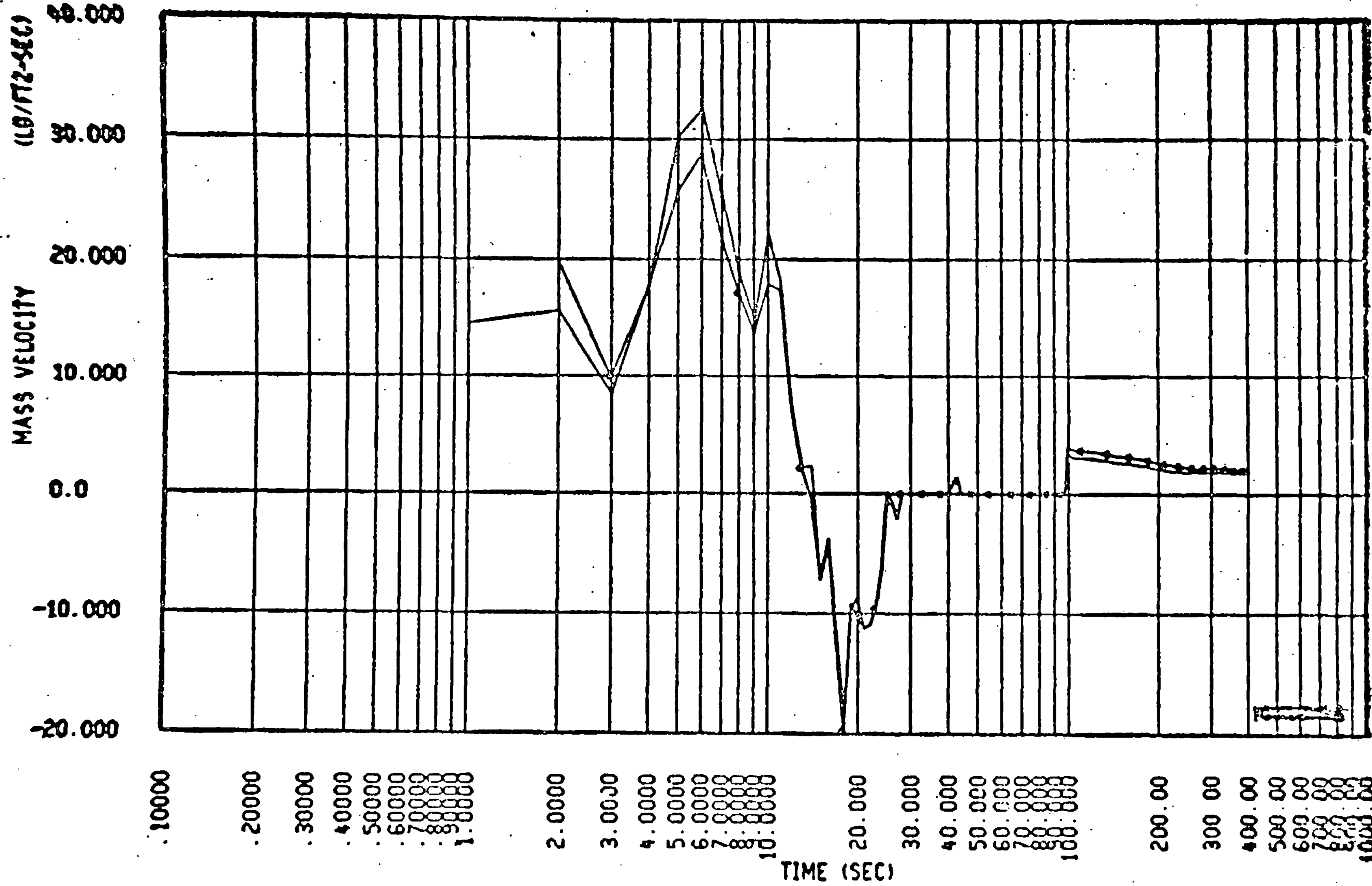
FIGURE 2d

CASE B



MASS VELOCITY - DECLG (CD = 0.4)

STEAM GENERATOR PLUGGING - 12PERCENT UNFM - OCT75 MOD - UPPERHEAD=THOT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD=0.4
MASS VELOCITY BURST 5.75 FT( ) PEAK 9.00 FT(*)



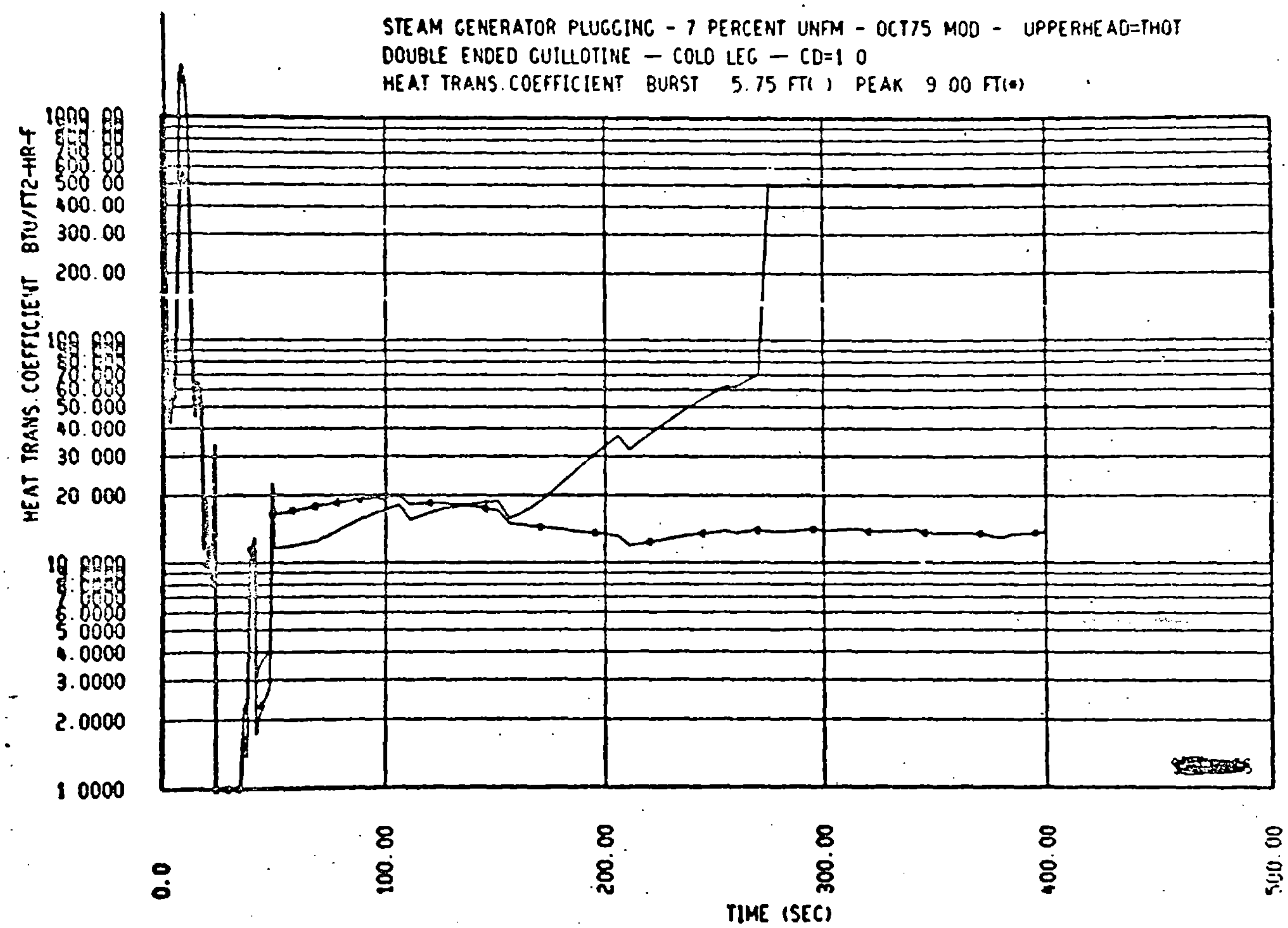
Mass Velocity - DECLG ($C_D = 0.4$)

FIGURE 2e

CASE C

FIGURE 3a

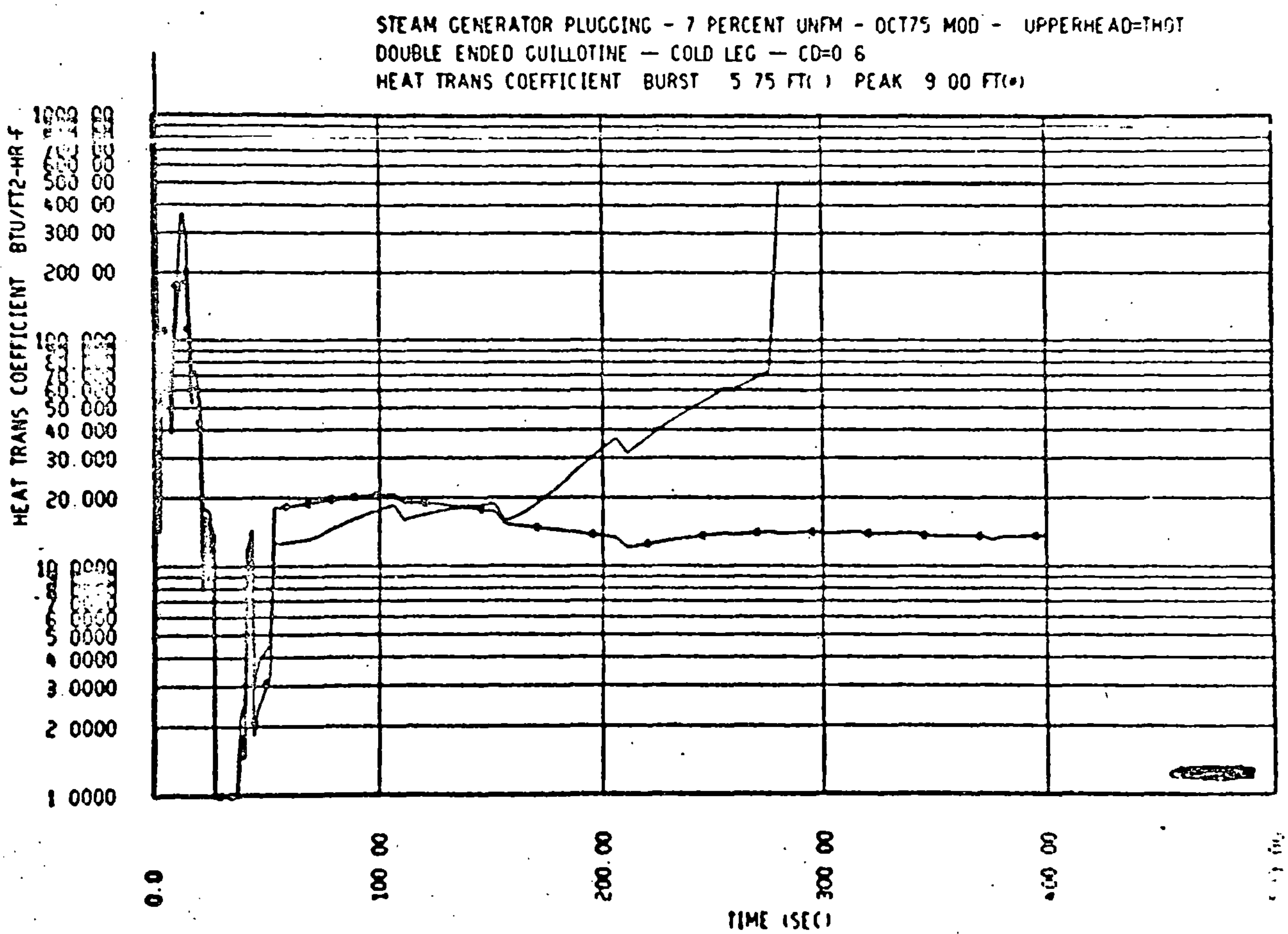
CASE A



HEAT TRANSFER COEFFICIENT - DECLG (C_D = 1.0)

FIGURE 3b

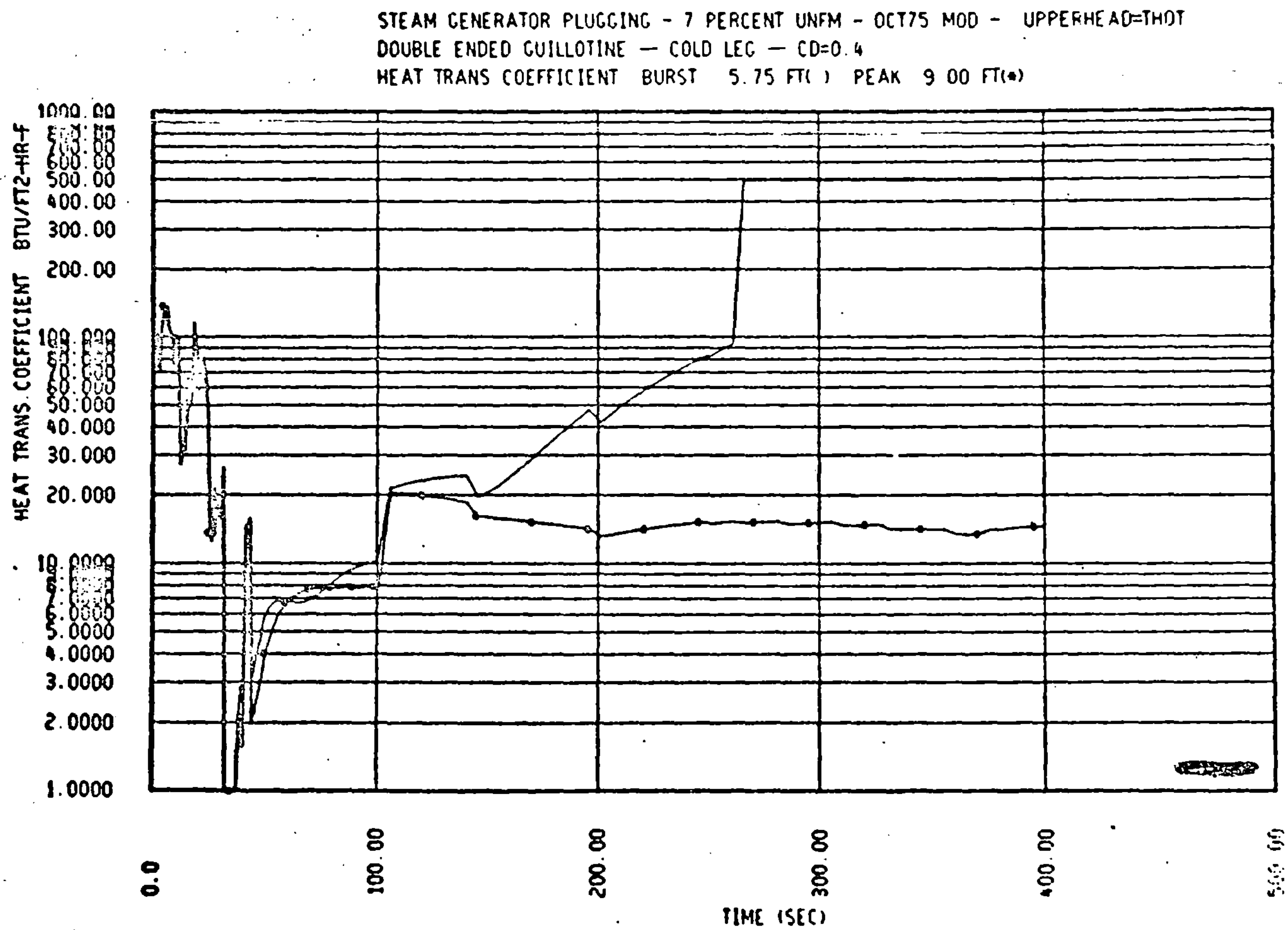
CASE A



HEAT TRANSFER COEFFICIENT - DECLG ($C_D = 0.6$)

FIGURE 3c

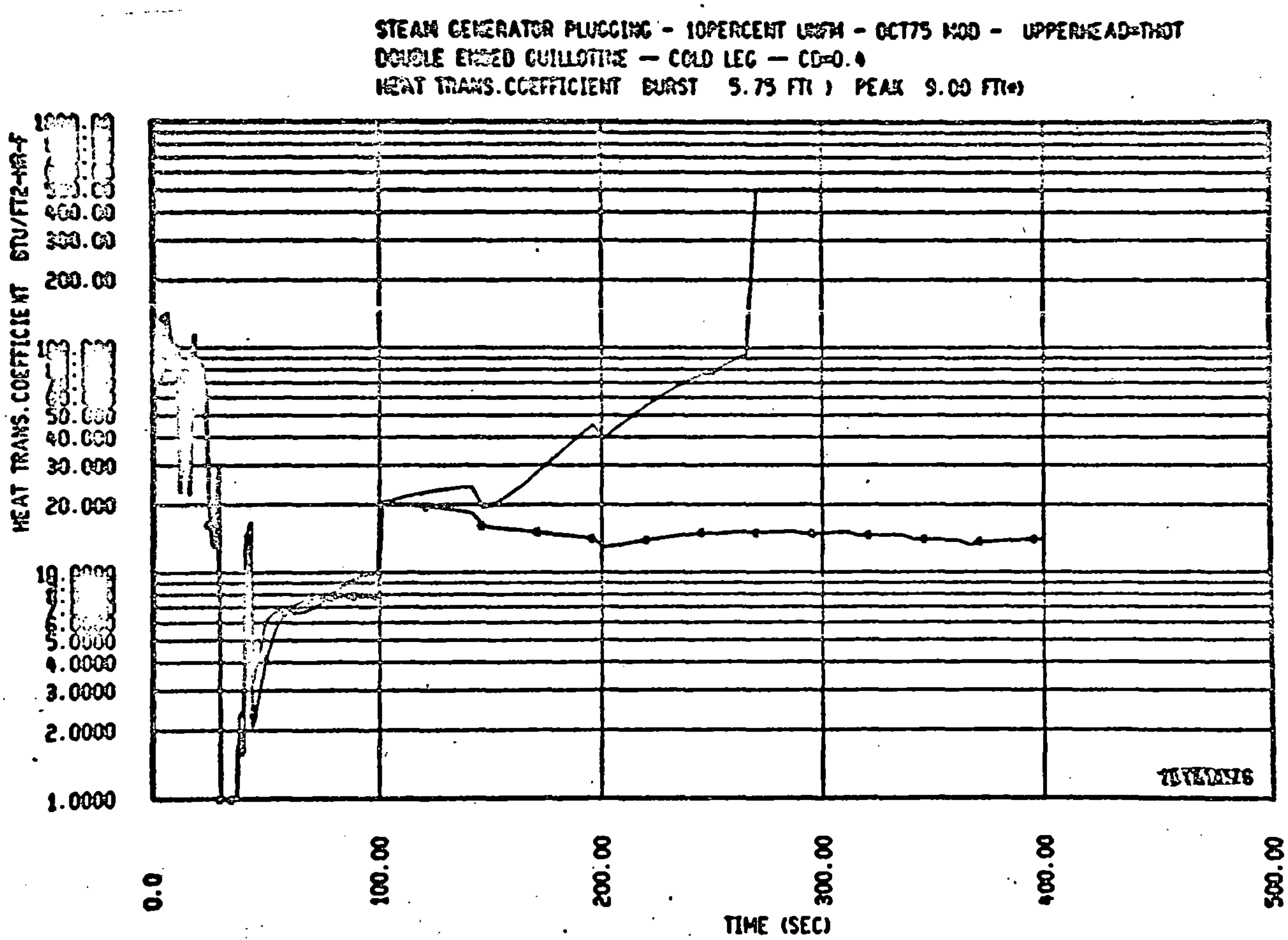
CASE A



HEAT TRANSFER COEFFICIENT - DECLG (C_D = 0.4)

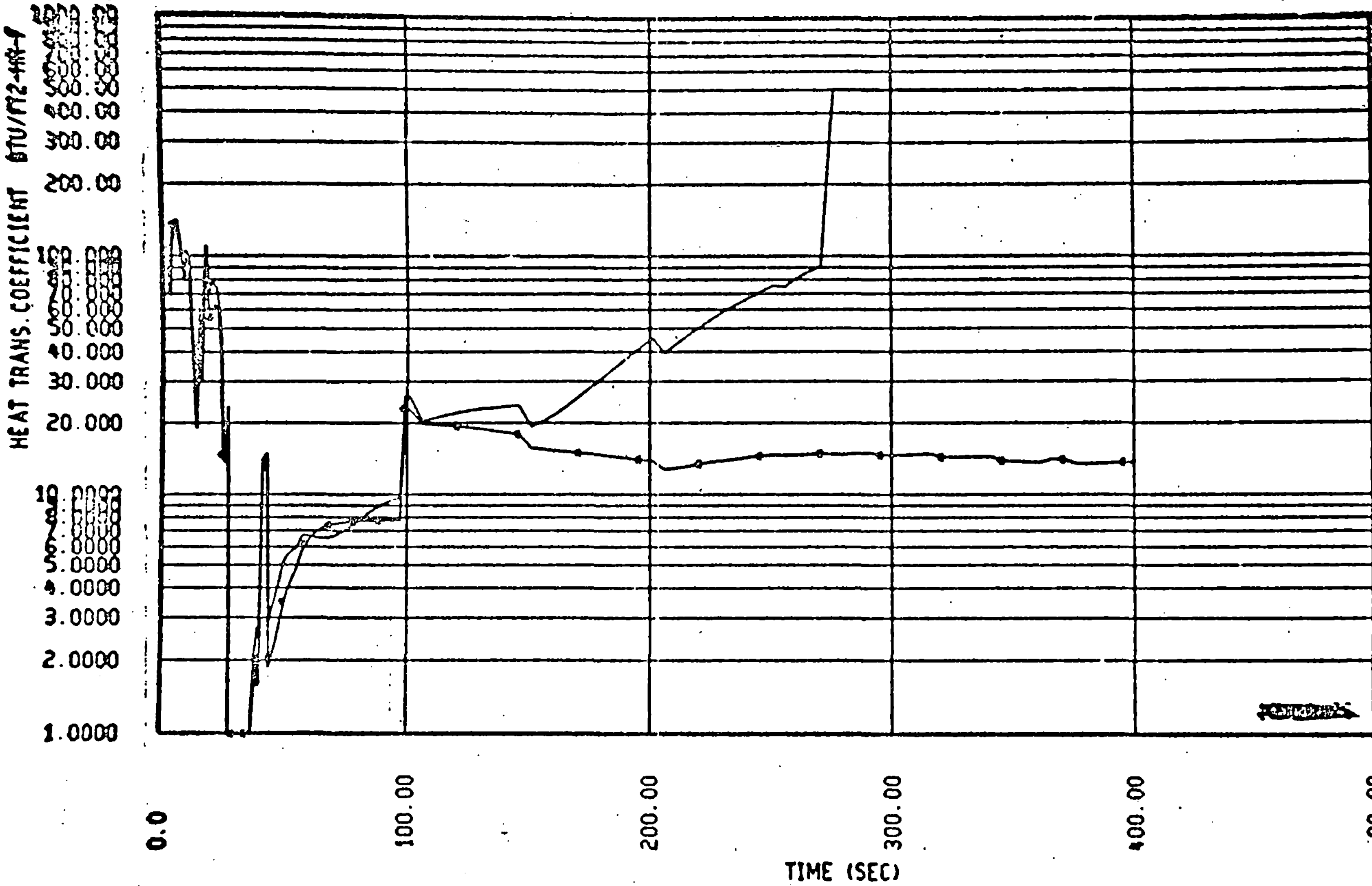
FIGURE 3d

CASE B



HEAT TRANSFER COEFFICIENT - DECLG (C_D =0.4)

STEAM GENERATOR PLUGGING - 12PERCENT UNFM - OCT75 MOD - UPPERHEAD=THOT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD=0.4
HEAT TRANS. COEFFICIENT BURST 5.75 FT( ) PEAK 9.00 FT(*)



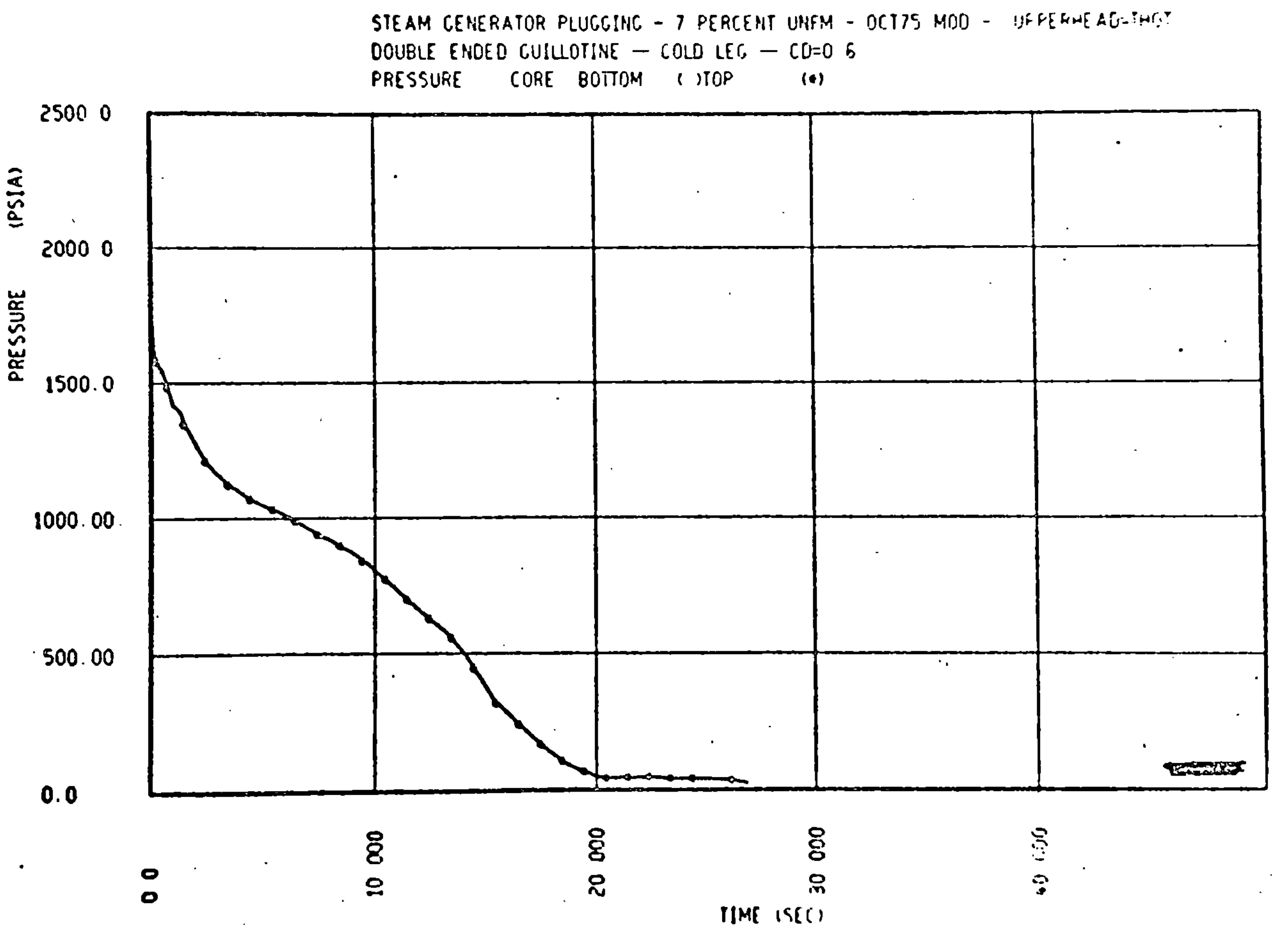
HEAT TRANSFER COEFFICIENT - DECLG (C_D = 0.4)

CASE C

FIGURE 3e

FIGURE 4b

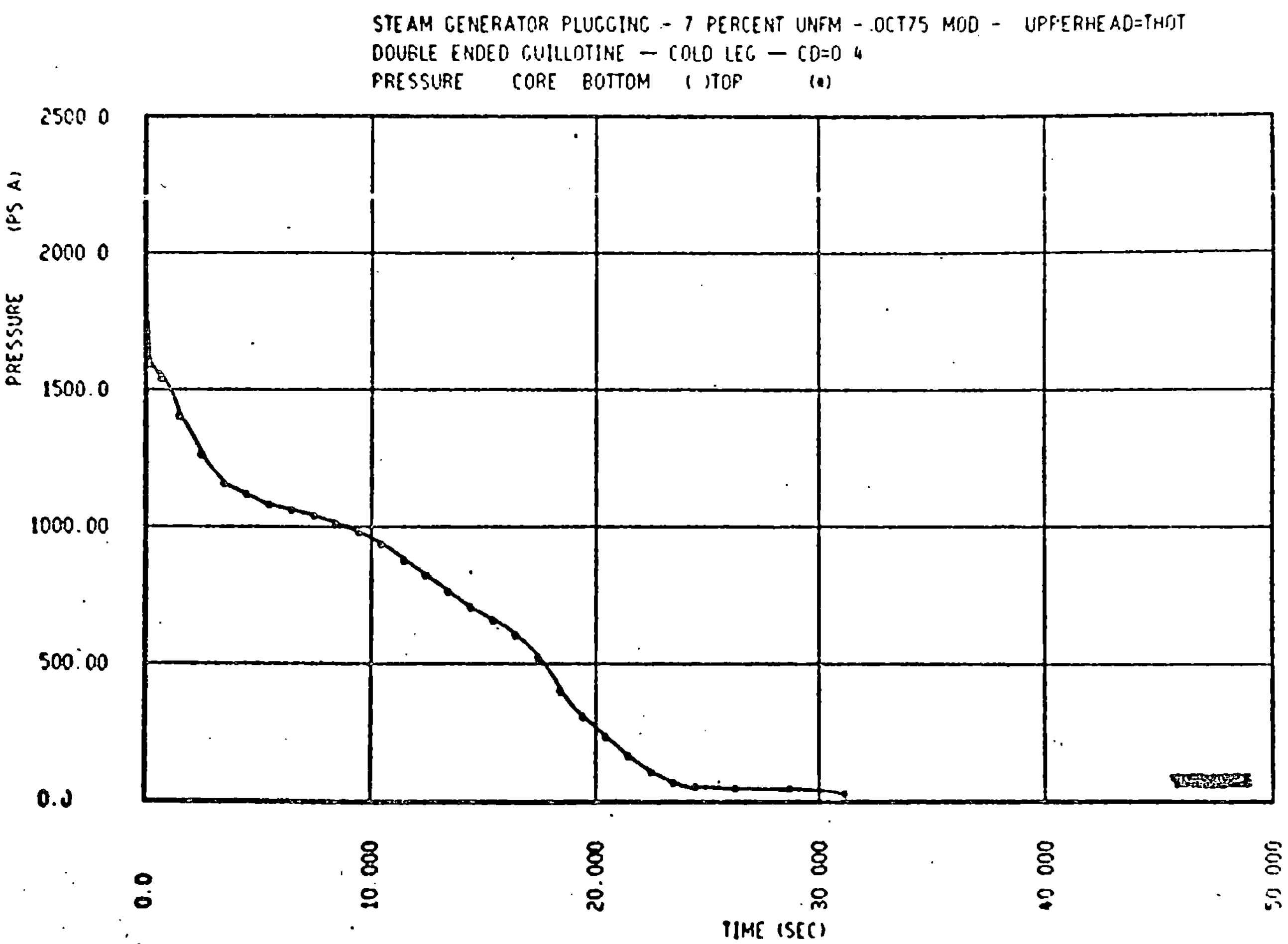
CASE A



CORE PRESSURE - DECLG (C_D = 0.6)

FIGURE 4c

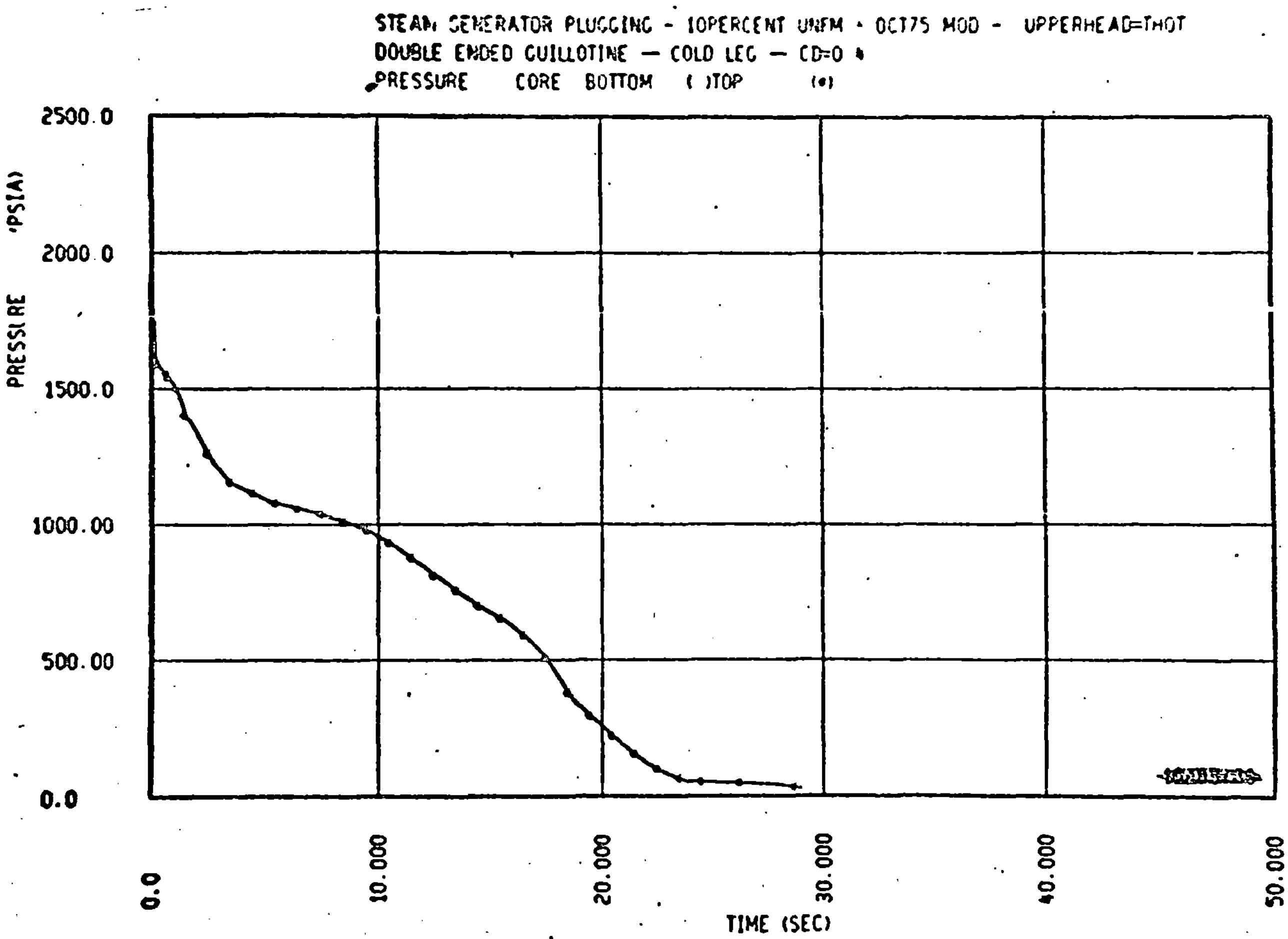
CASE A



Core Pressure - DECLG (C_D = 0.4)

FIGURE 4d

CASE B



CORE PRESSURE - DECLG (C_D = 0.4)

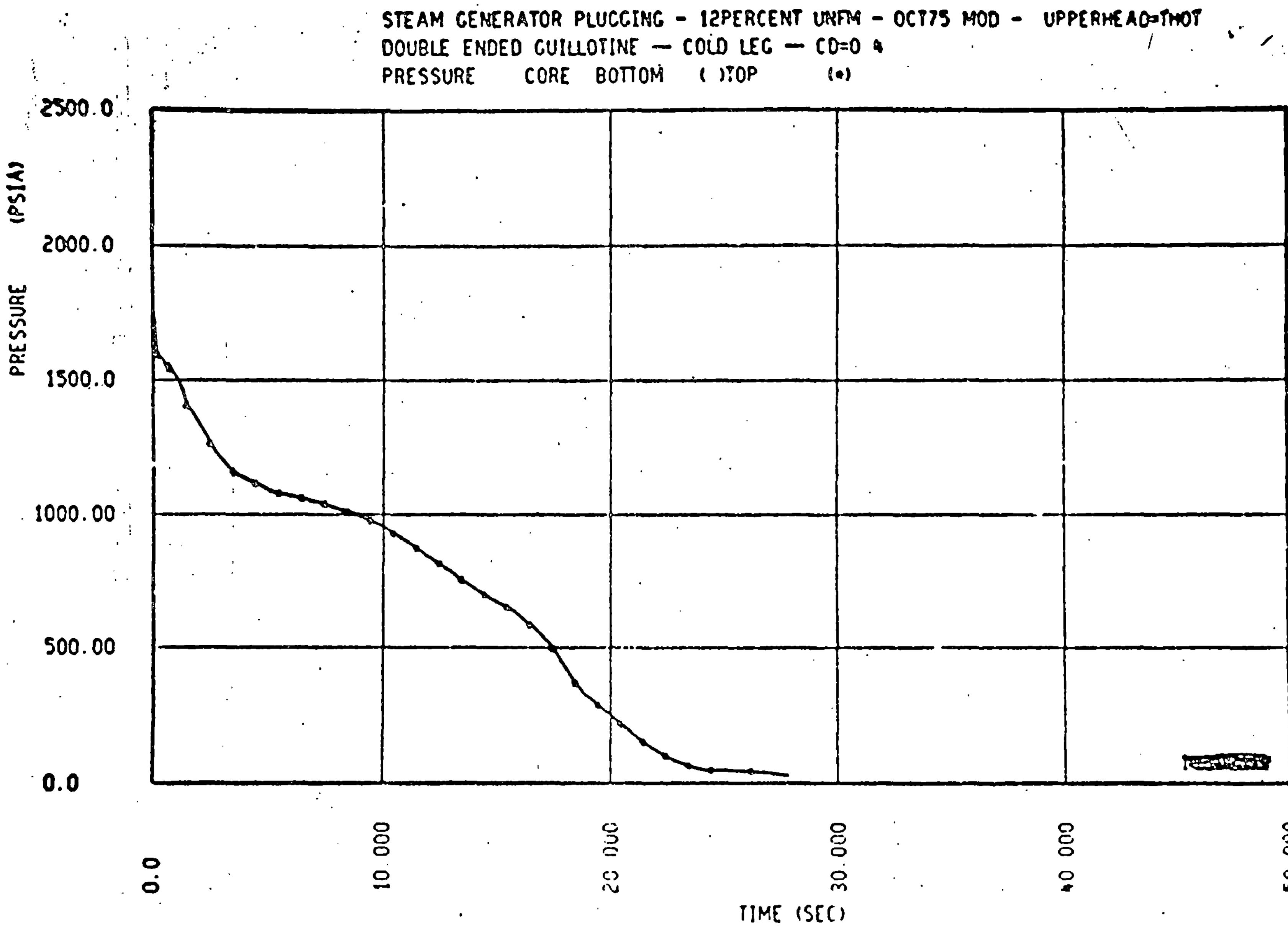


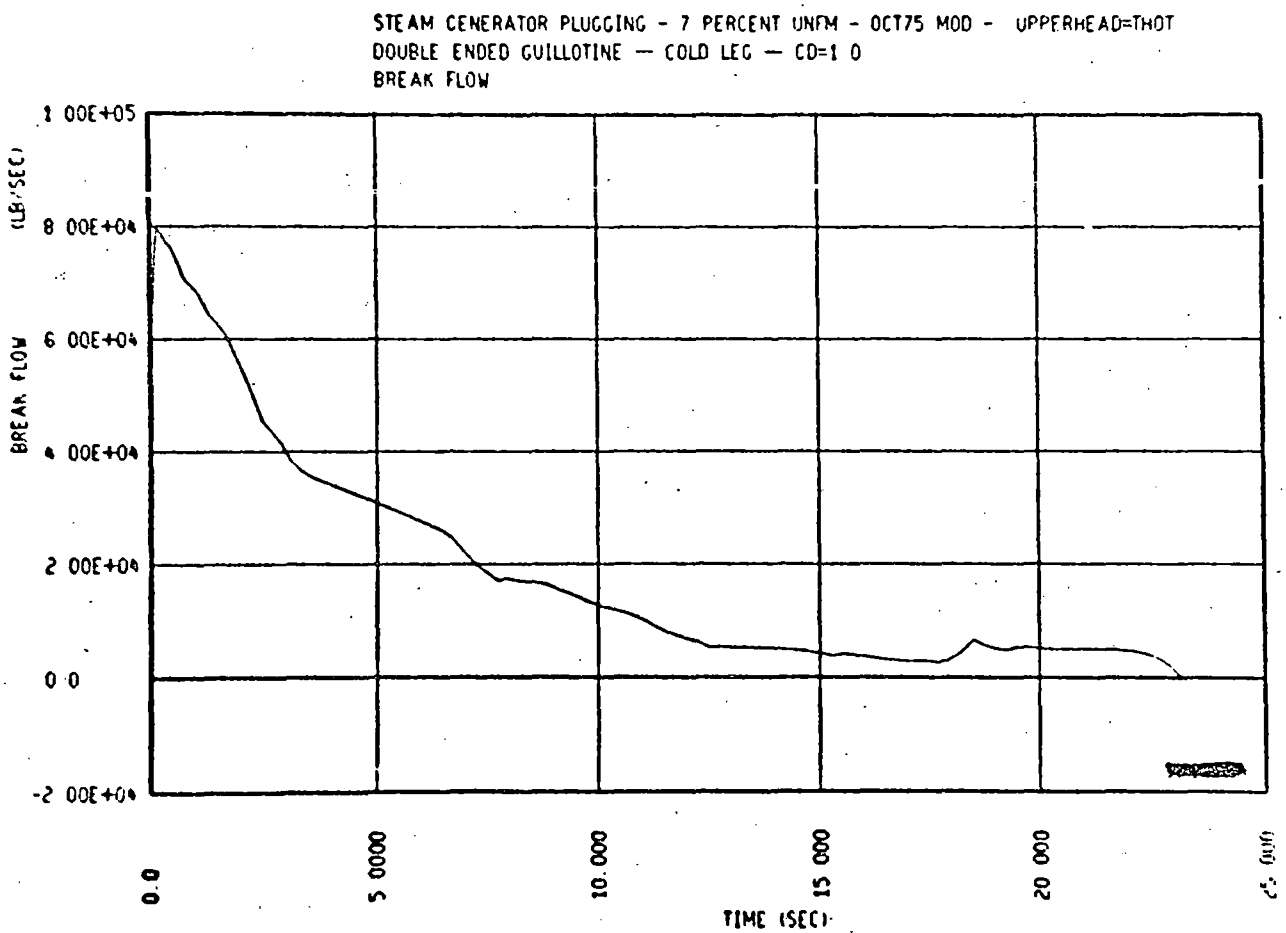
FIGURE 4e

CASE C

Core Pressure - DECLG (C_D = 0.4)

FIGURE 5a

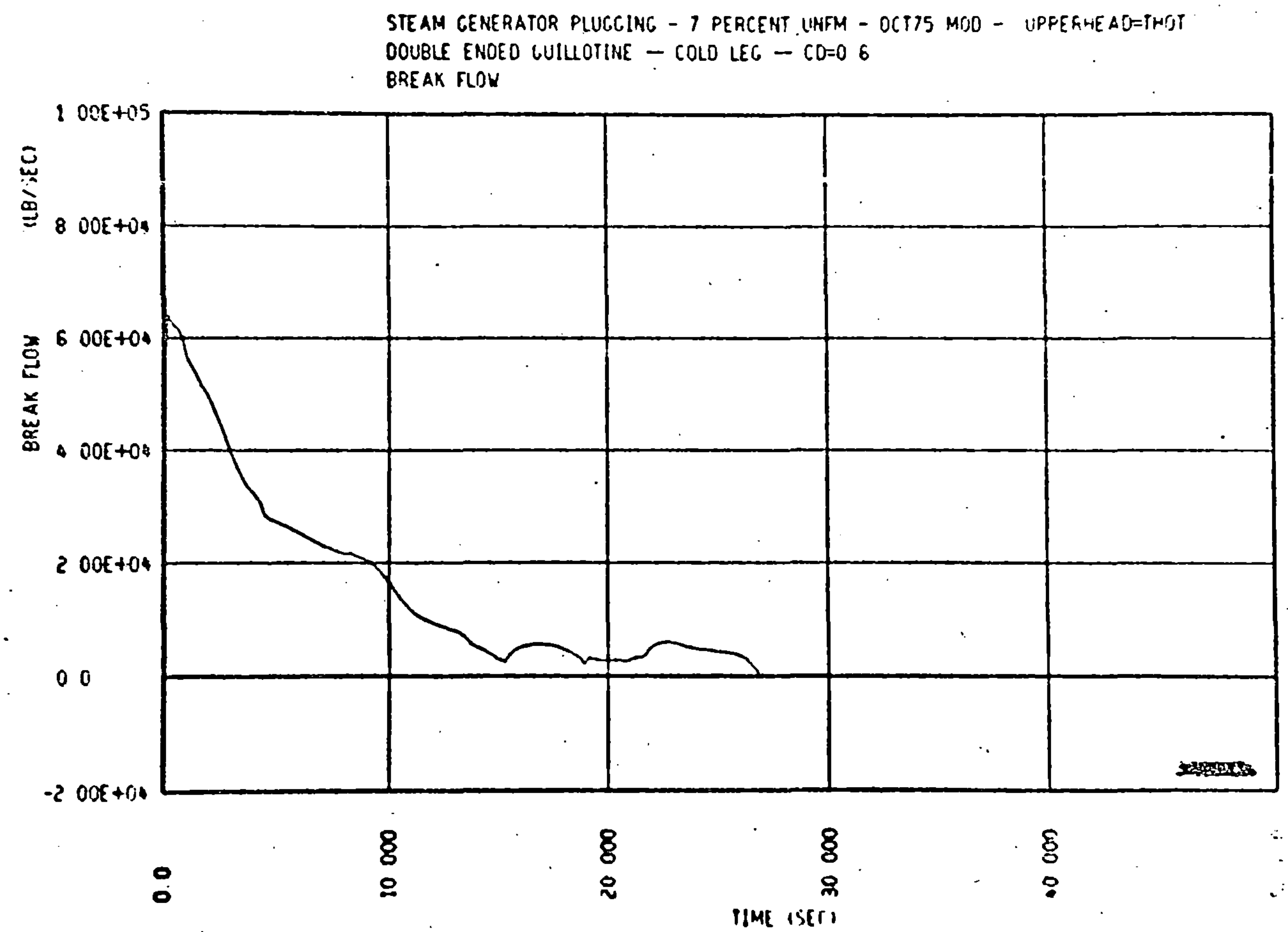
CASE A



BREAK FLOW RATE - DECLG (CD - 1.0)

FIGURE 5b

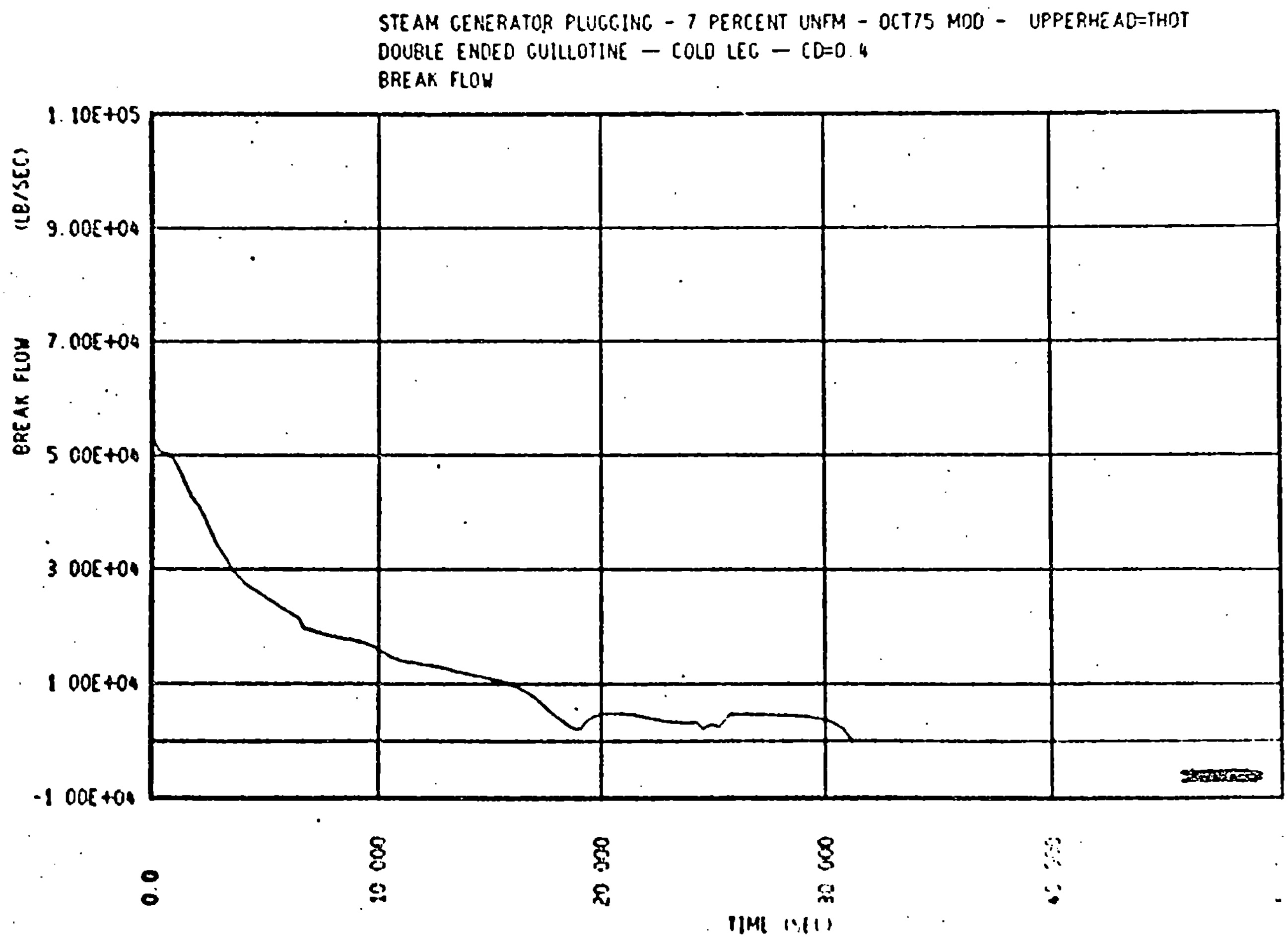
CASE A



BREAK FLOW RATE - DECLG (CD - 0.6)

FIGURE 5c

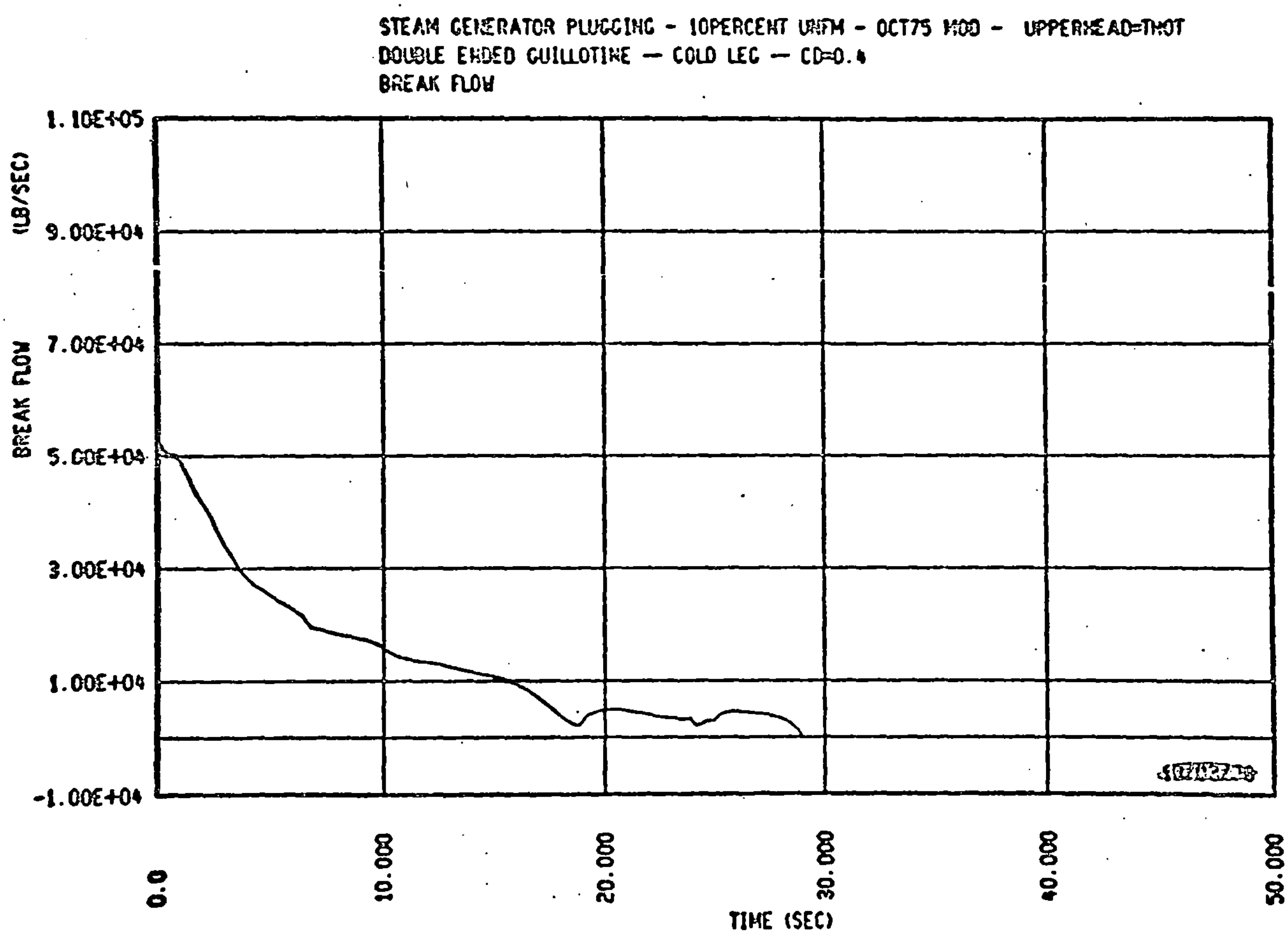
CASE A



BREAK FLOW RATE - DECLG (CD - 0.4)

FIGURE 5d

CASE B



BREAK FLOW RATE - DECLG (CD - 0.4)

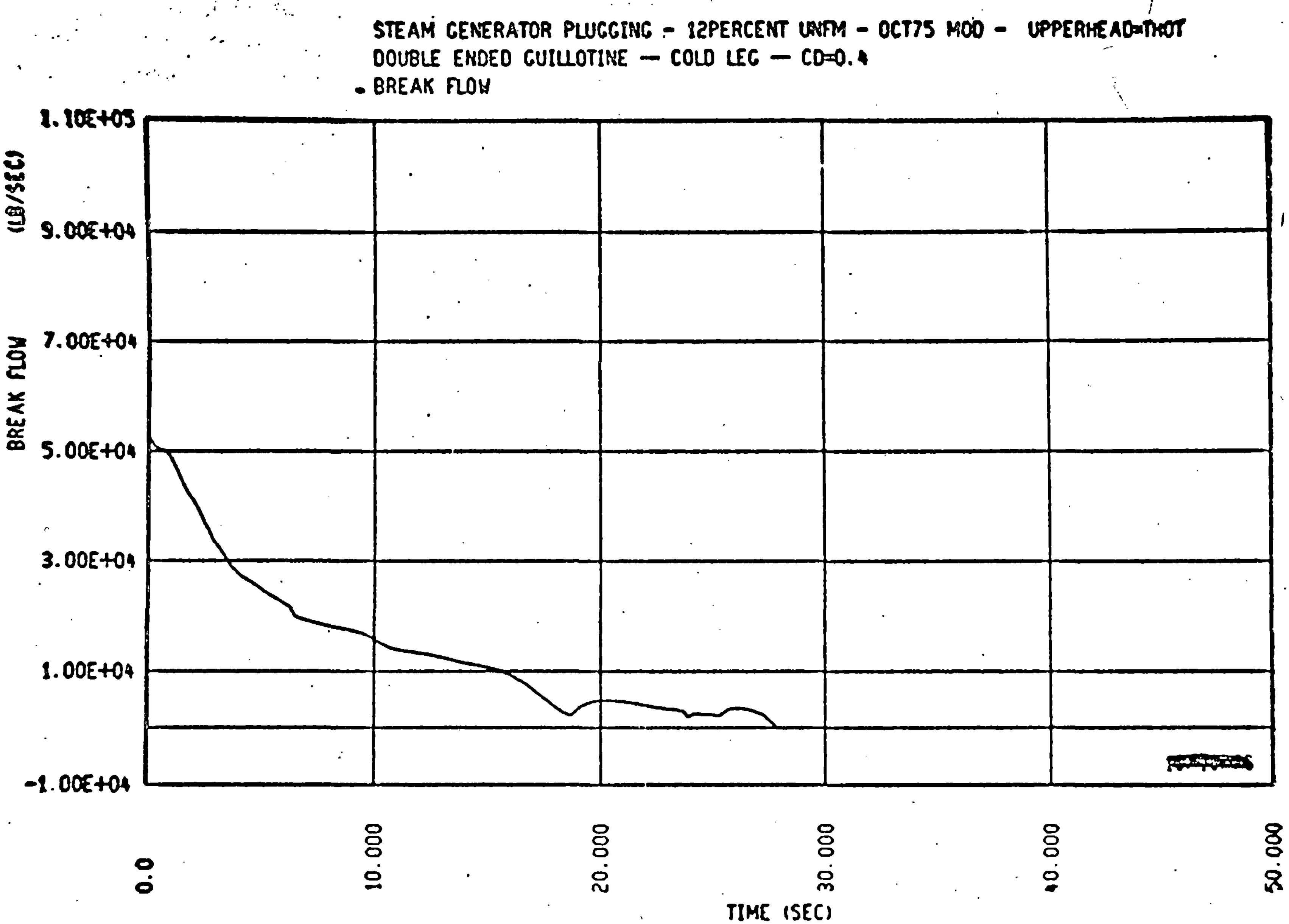


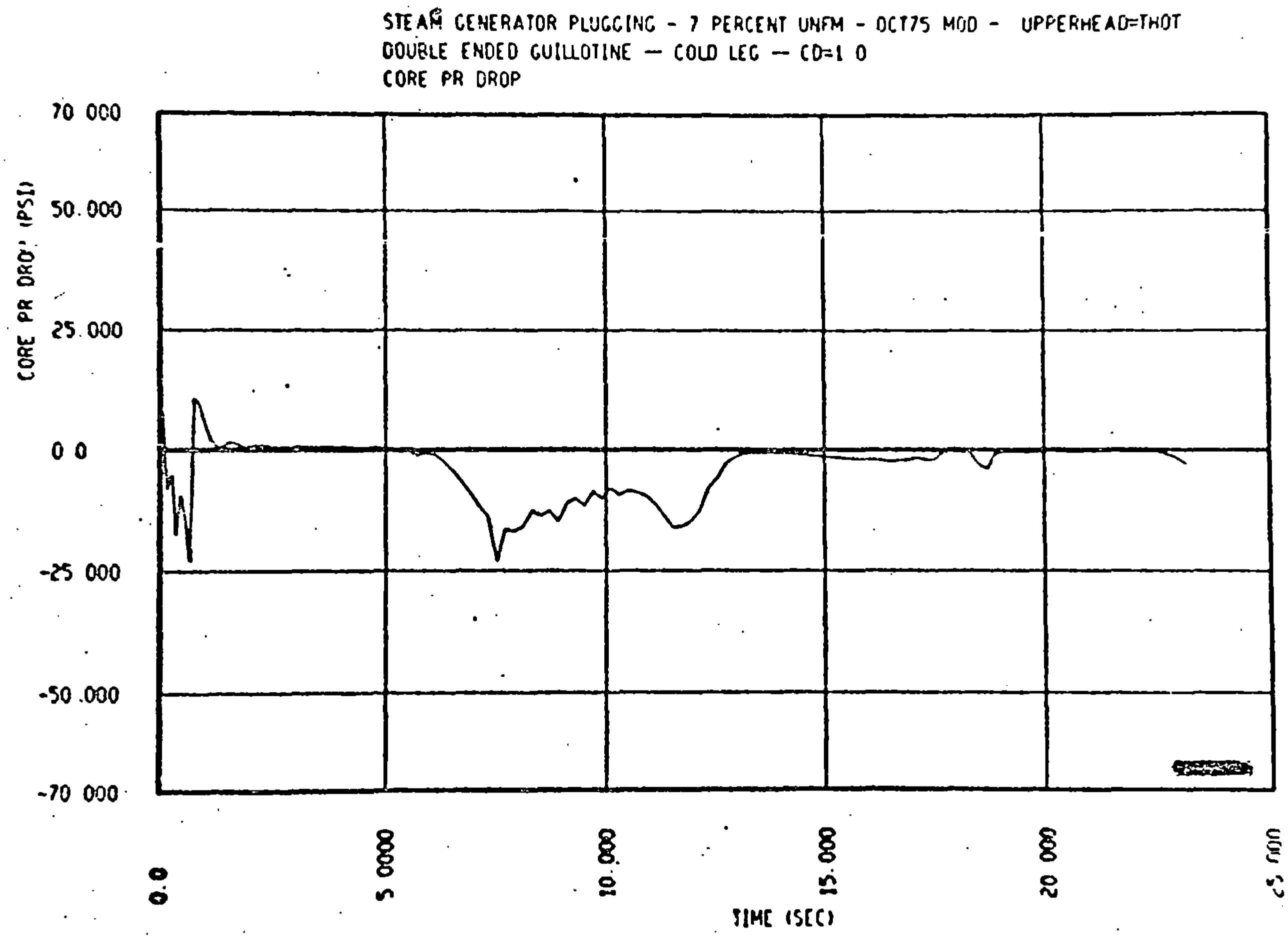
FIGURE 5e

CASE C

BREAK FLOW RATE - DECLG (CD - 0.4)

FIGURE 6a

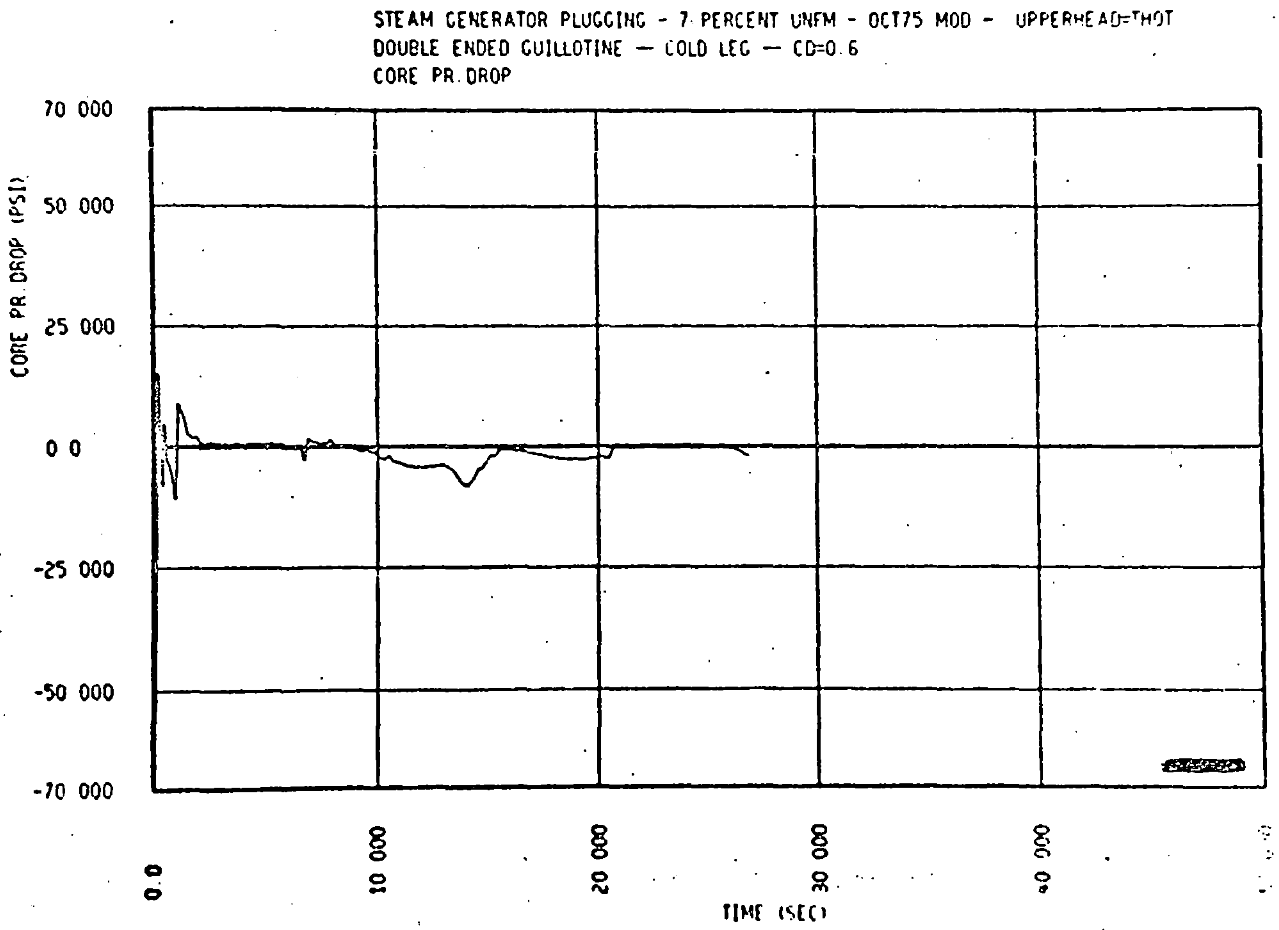
CASE A



Core Pressure Drop - DECLG (C_D=1.0)

FIGURE 6b

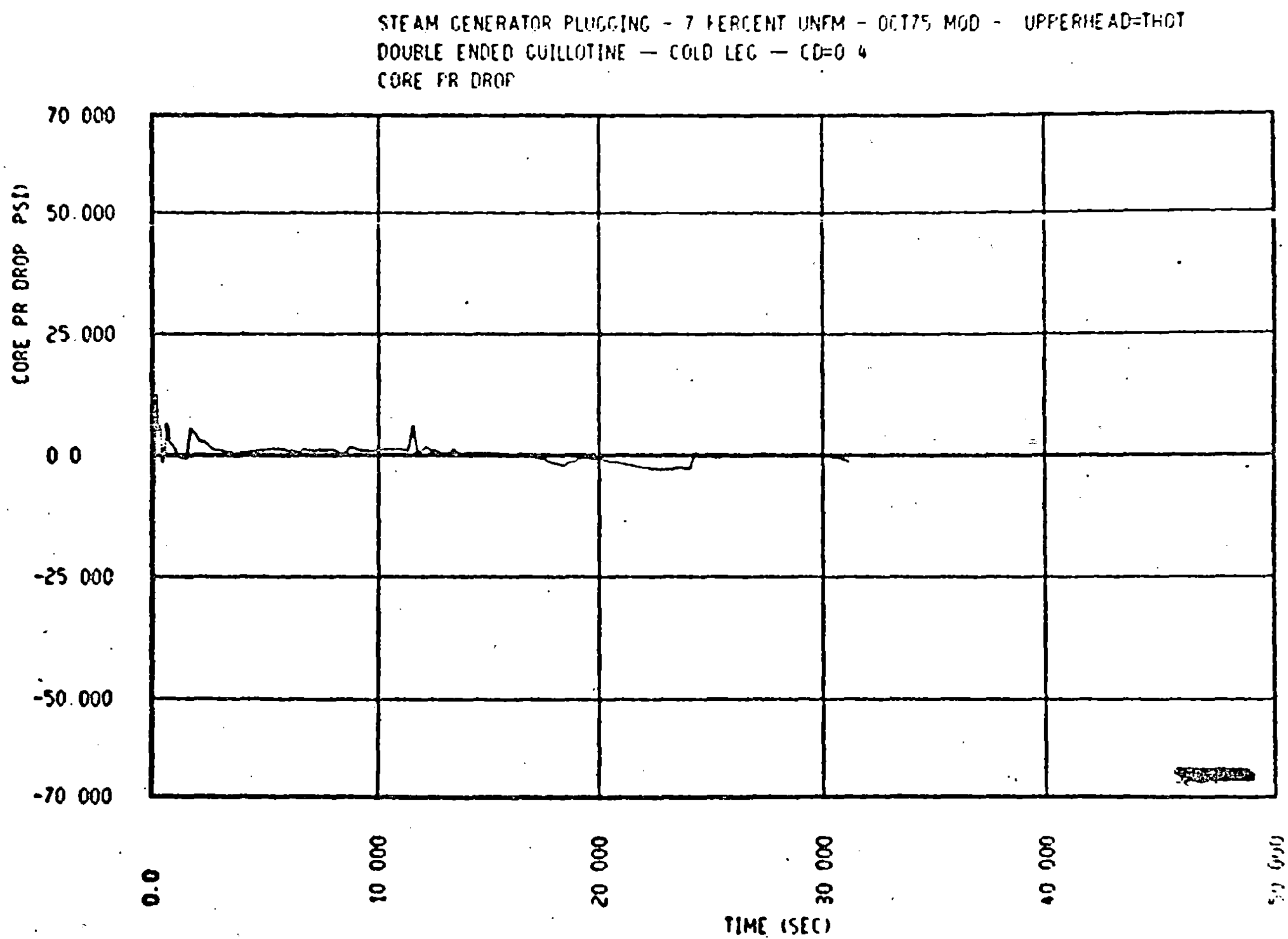
CASE A



Core Pressure Drop - DECLG (C_D = 0.6)

FIGURE 6c

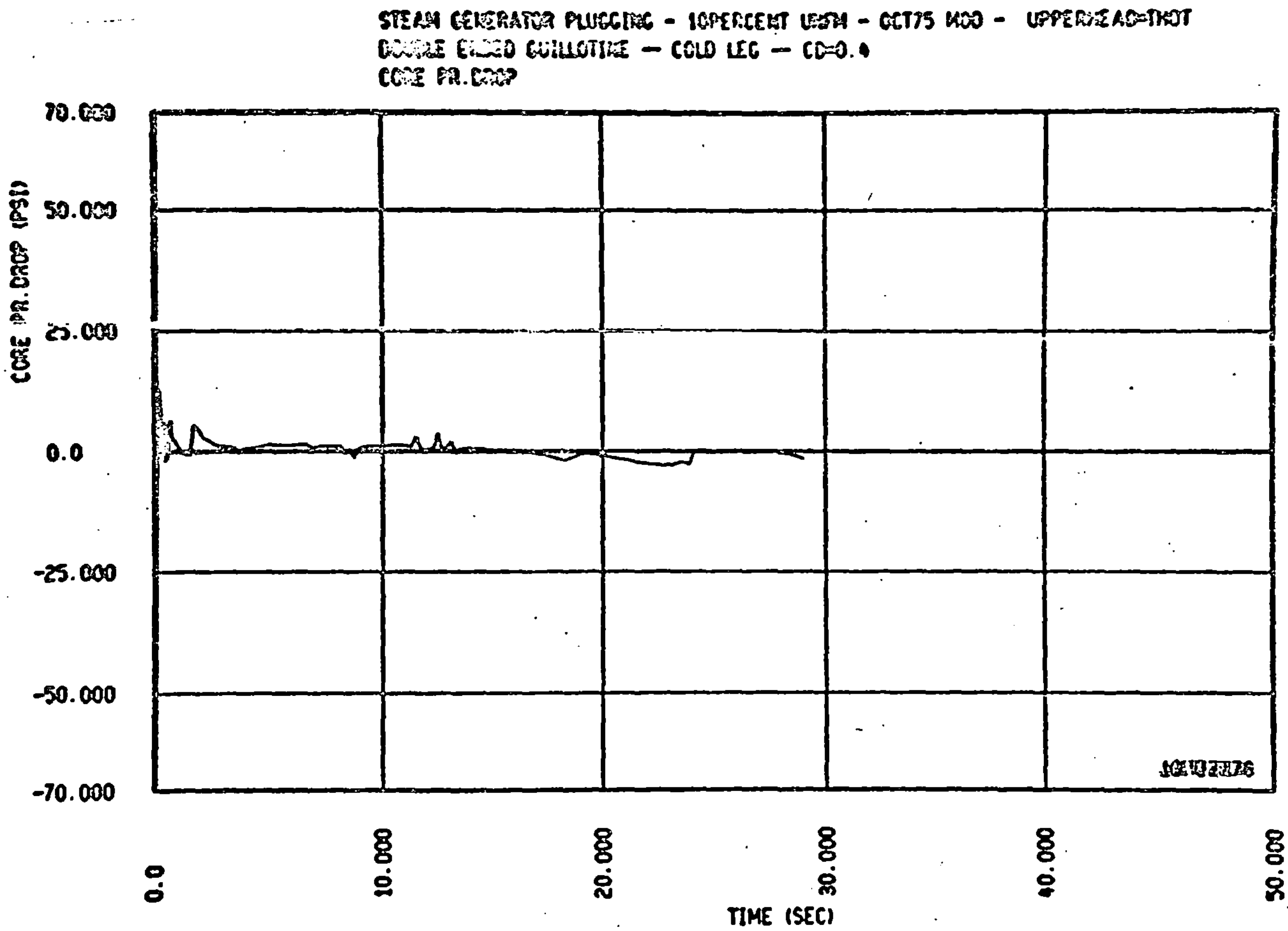
CASE A



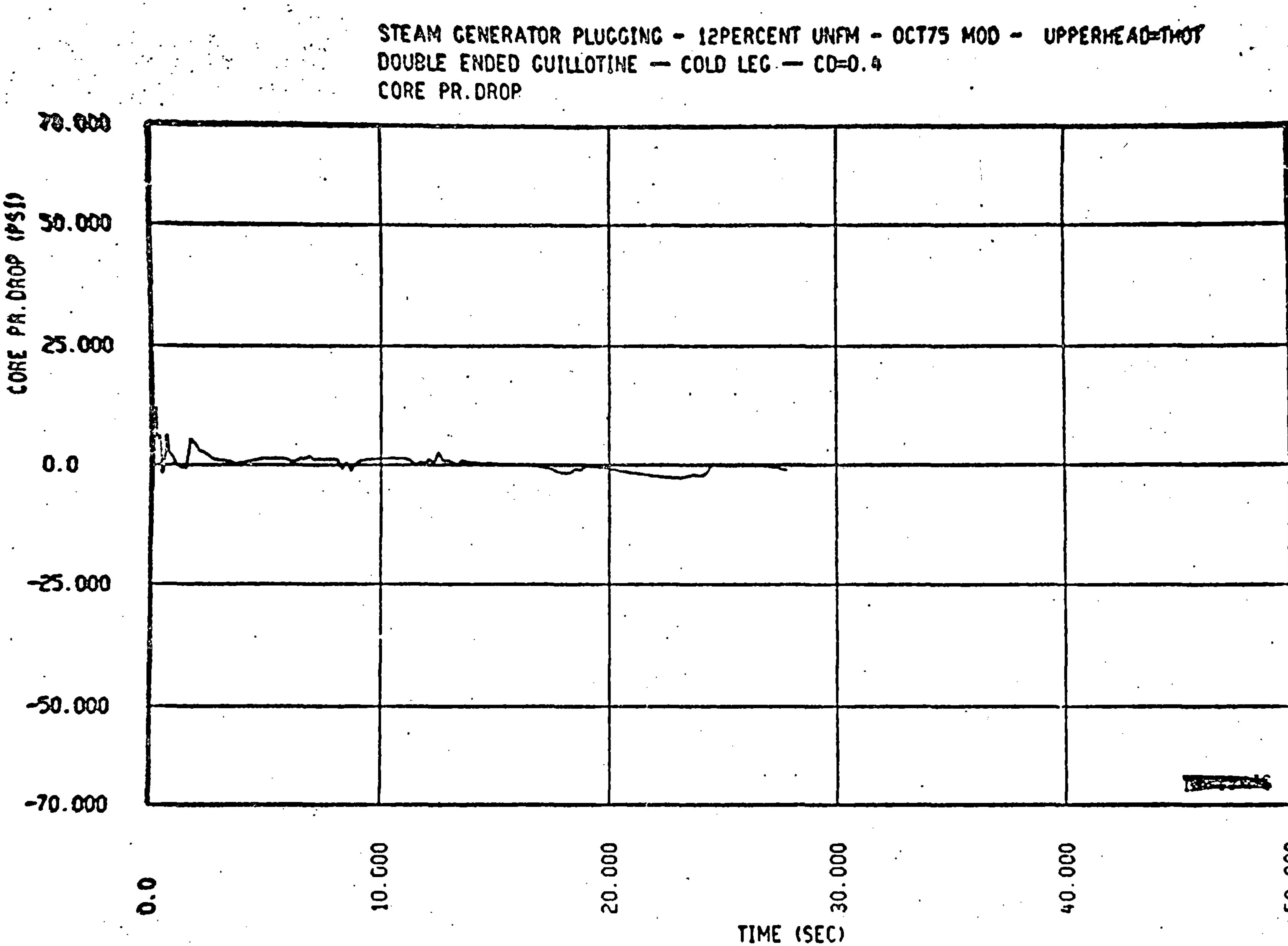
Core Pressure Drop - DECLG ($C_D = 0.4$)

FIGURE 6d

CASE B



Core Pressure Drop - DECLG ($C_D = 0.4$)



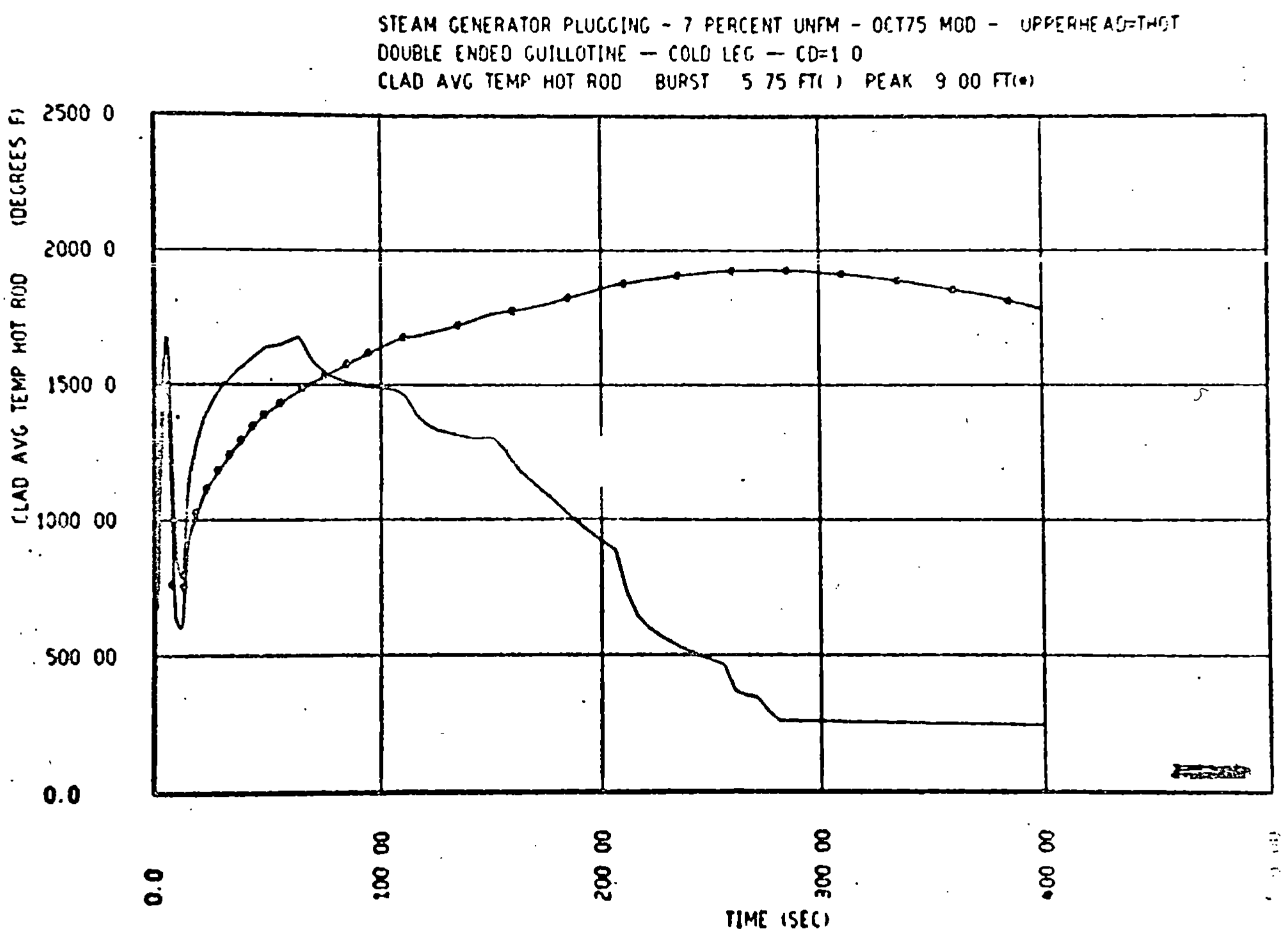
CORE PRESSURE DROP - DECLG (C_D = 0.4)

FIGURE 6e

CASE C

FIGURE 7a

CASE A



PEAK CLAD TEMPERATURE - DECLG (C_D = 1.0)

Table 7a

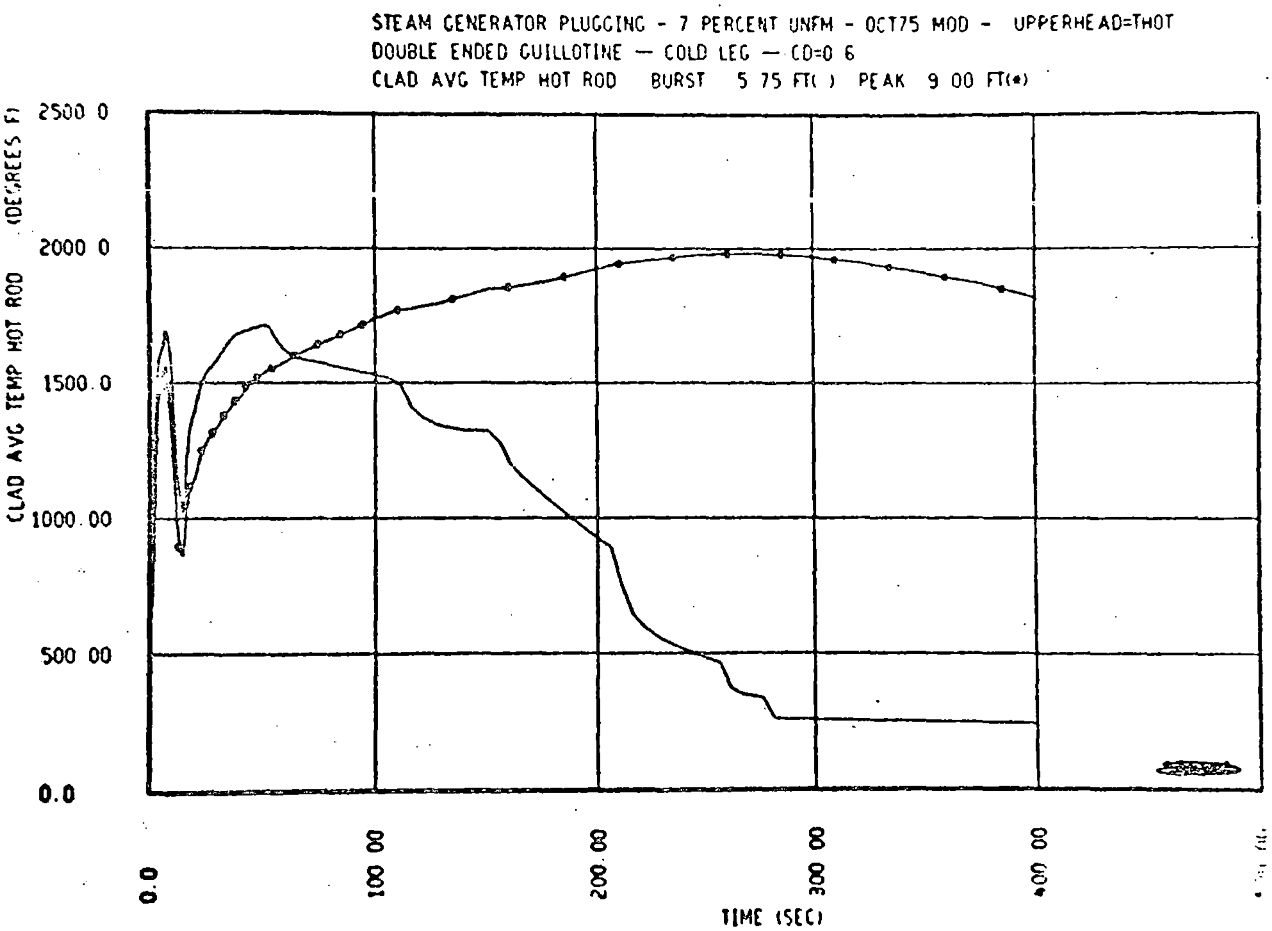
REFLOOD MASS AND ENERGY RELEASES FOR LIMITING CASE

AT 12 PERCENT PLUGGING - DECLG (CD=0.4)

TIME	TOTAL MASS FLOWRATE (LBm/SEC)	TOTAL ENERGY FLOWRATE (10^5 BTU/SEC)
37.88	0.0	0.0
38.805	0.0	0.0
39.005	1.032	0.0134
44.277	34.58	0.4497
51.99	34.58	0.4497
51.996	2892	3.126
53.796	2892	3.126
55.996	2892	3.126
56.0	246.4	1.471
67.44	246.4	1.471
84.046	257.6	1.449
102.346	264.8	1.411
122.146	270.9	1.366
166.446	286.9	1.244
218.946	303.9	1.106
284.046	322.4	0.9596
373.846	333.9	0.8584

FIGURE 7b

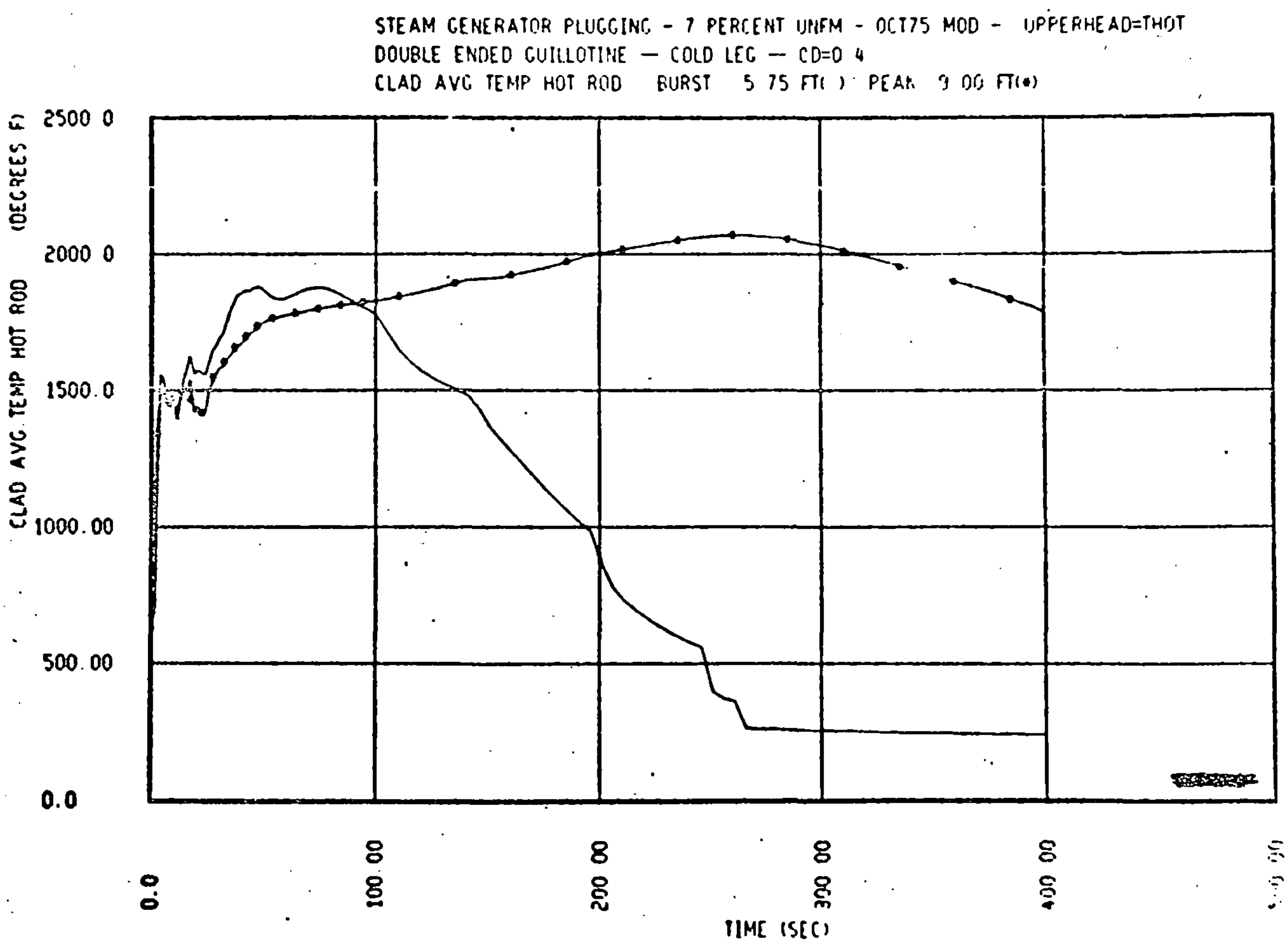
CASE A



PEAK CLAD TEMPERATURE - DECLG (C_D = 0.6)

FIGURE 7c

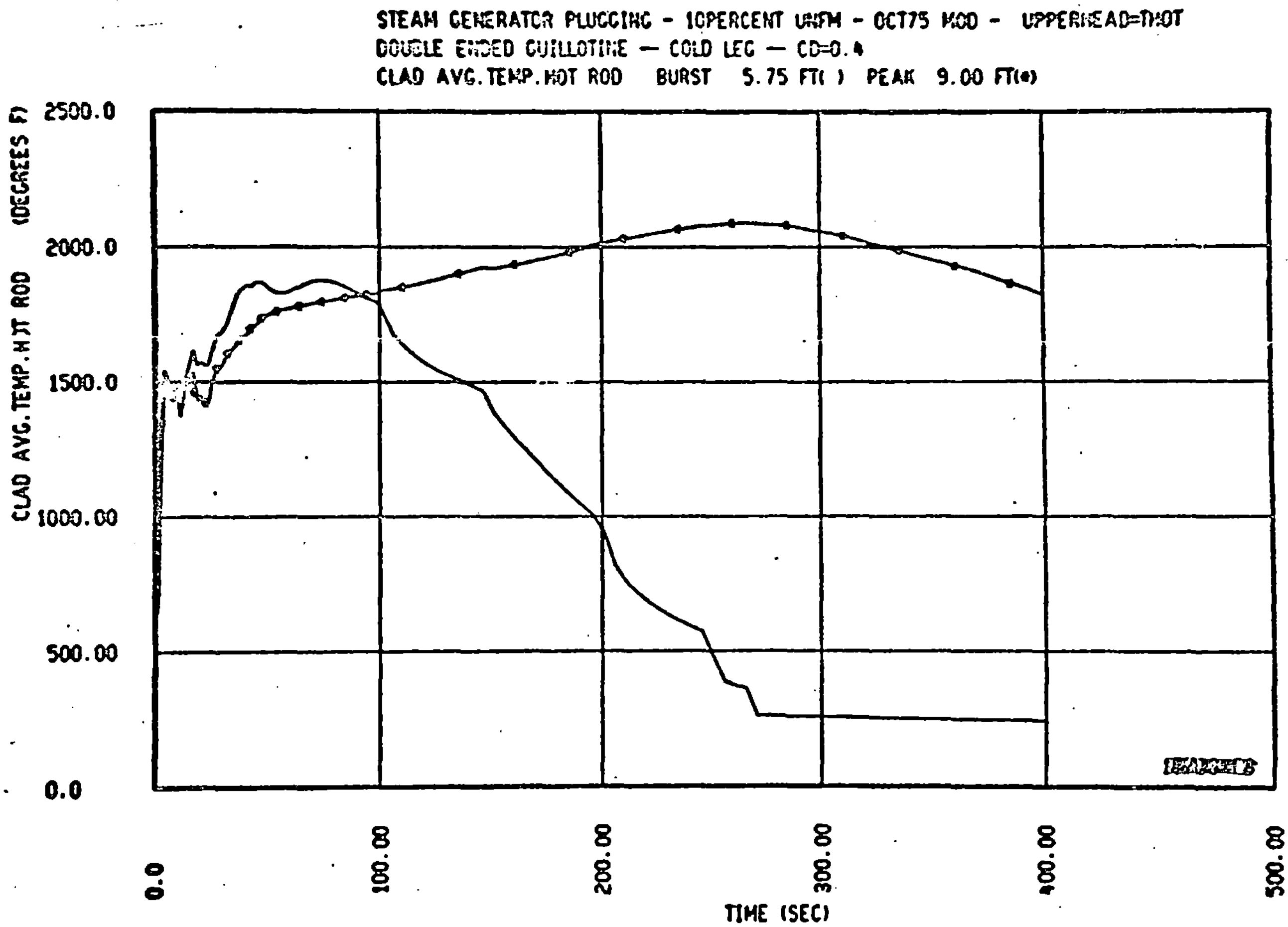
CASE A



PEAK CLAD TEMPERATURE - DECLG (C_D = 0.4)

FIGURE 7d

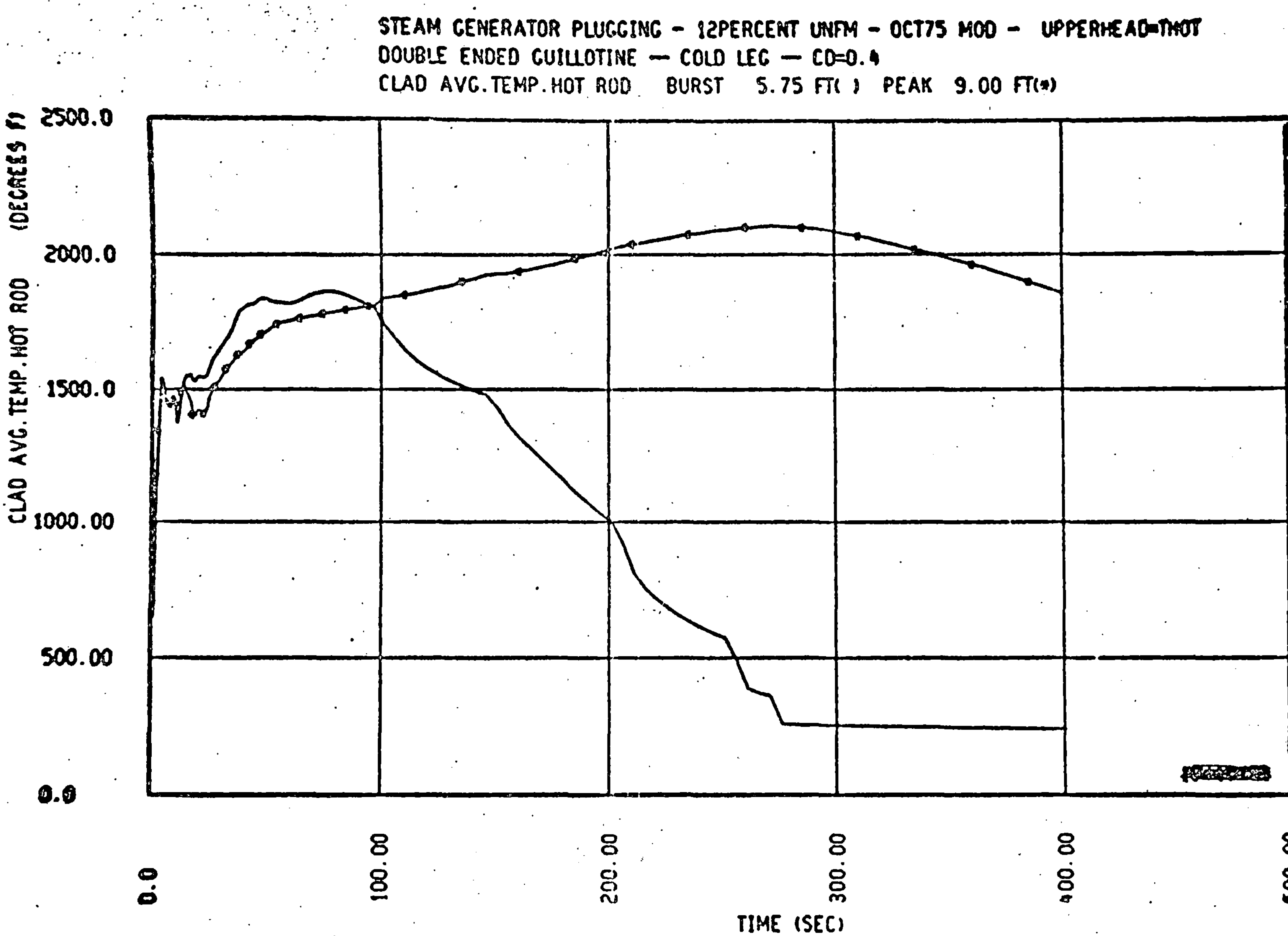
CASE B



PEAK CLAD TEMPERATURE - DECLG (C_D = 0.4)

FIGURE 7e

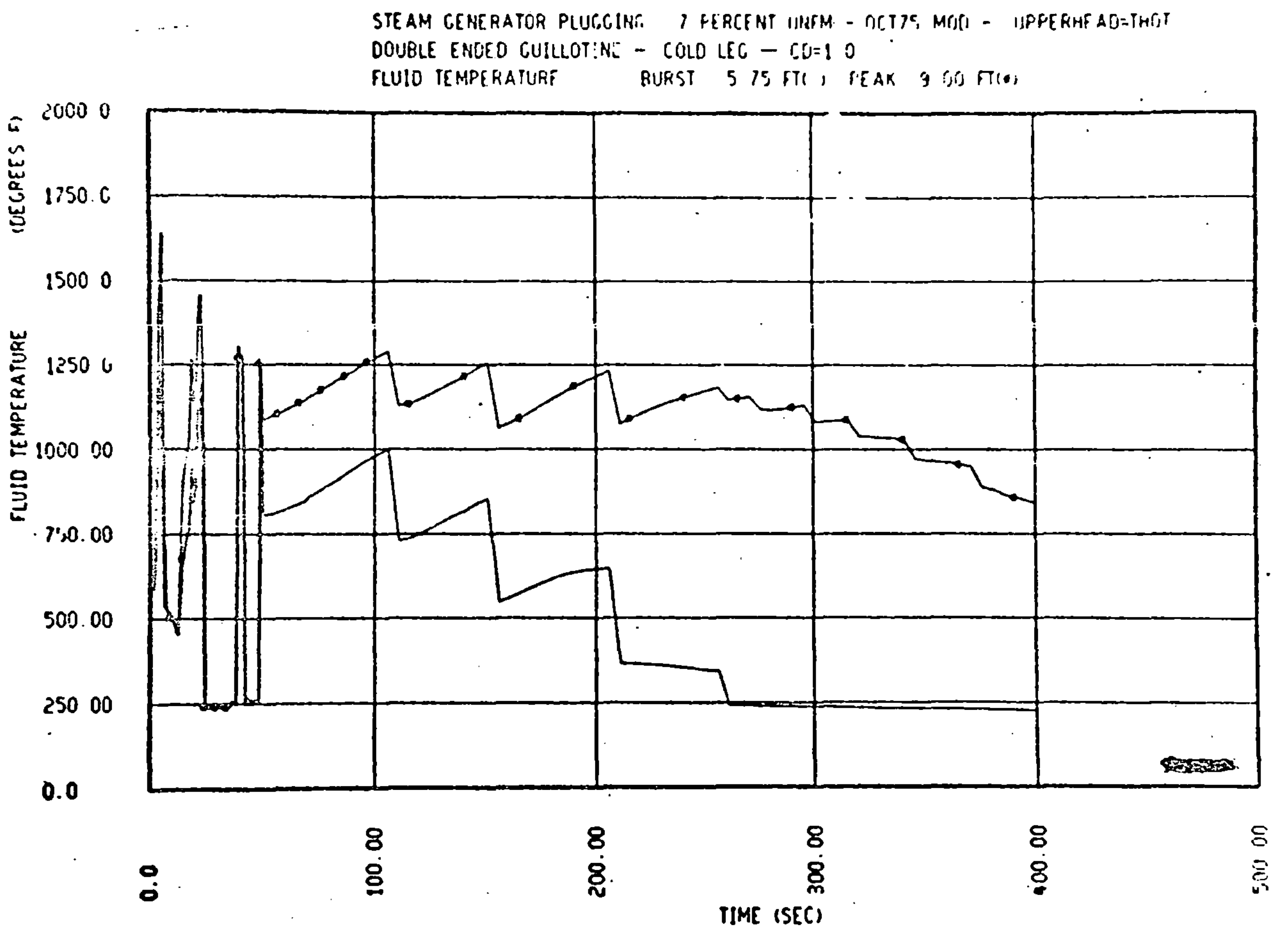
CASE C



PEAK CLAD TEMPERATURE - DECLG (C_D = 0.4)

FIGURE 8a

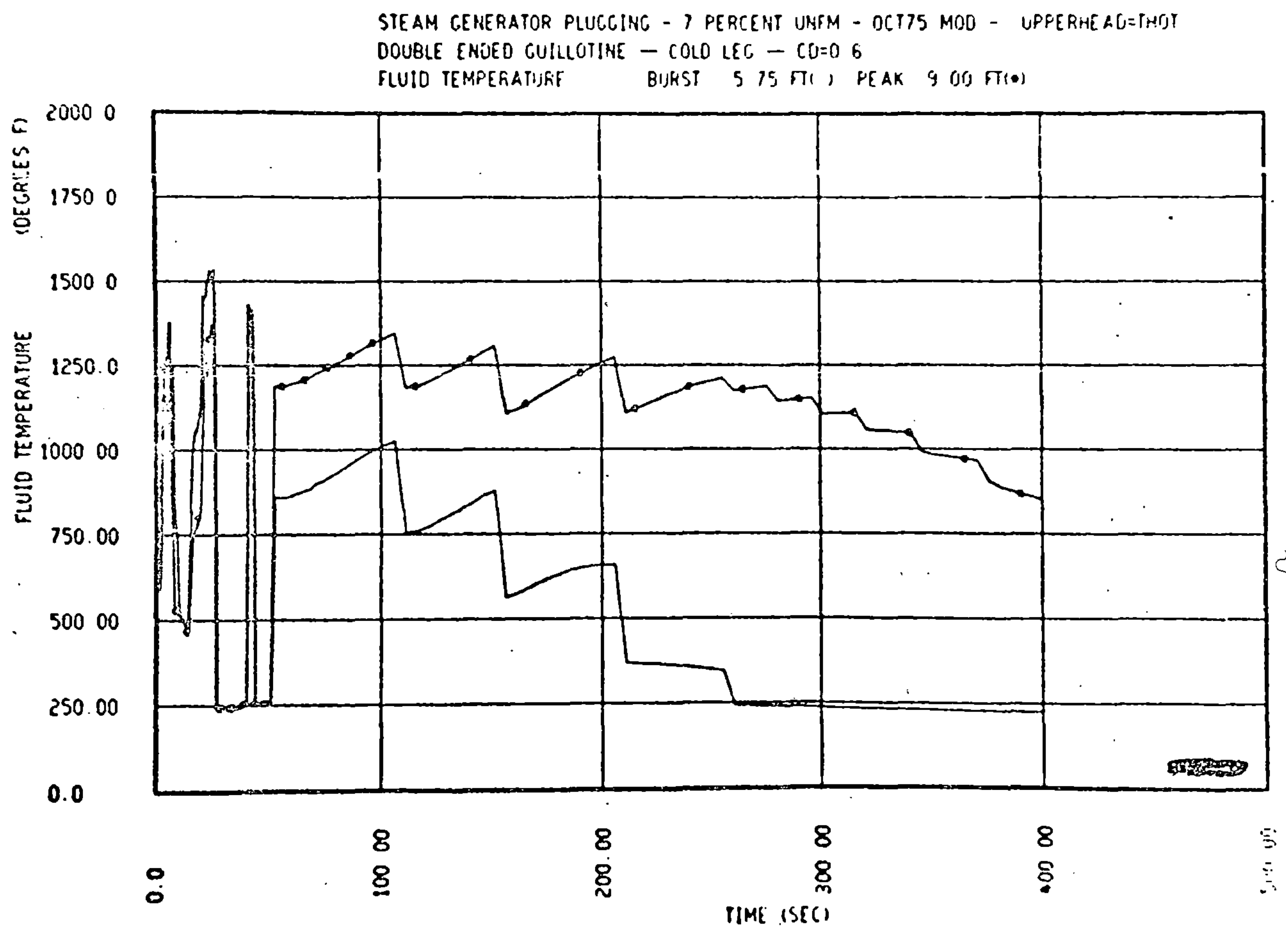
CASE A



FLUID TEMPERATURE – DECLG (C_D = 1.0)

FIGURE 8b

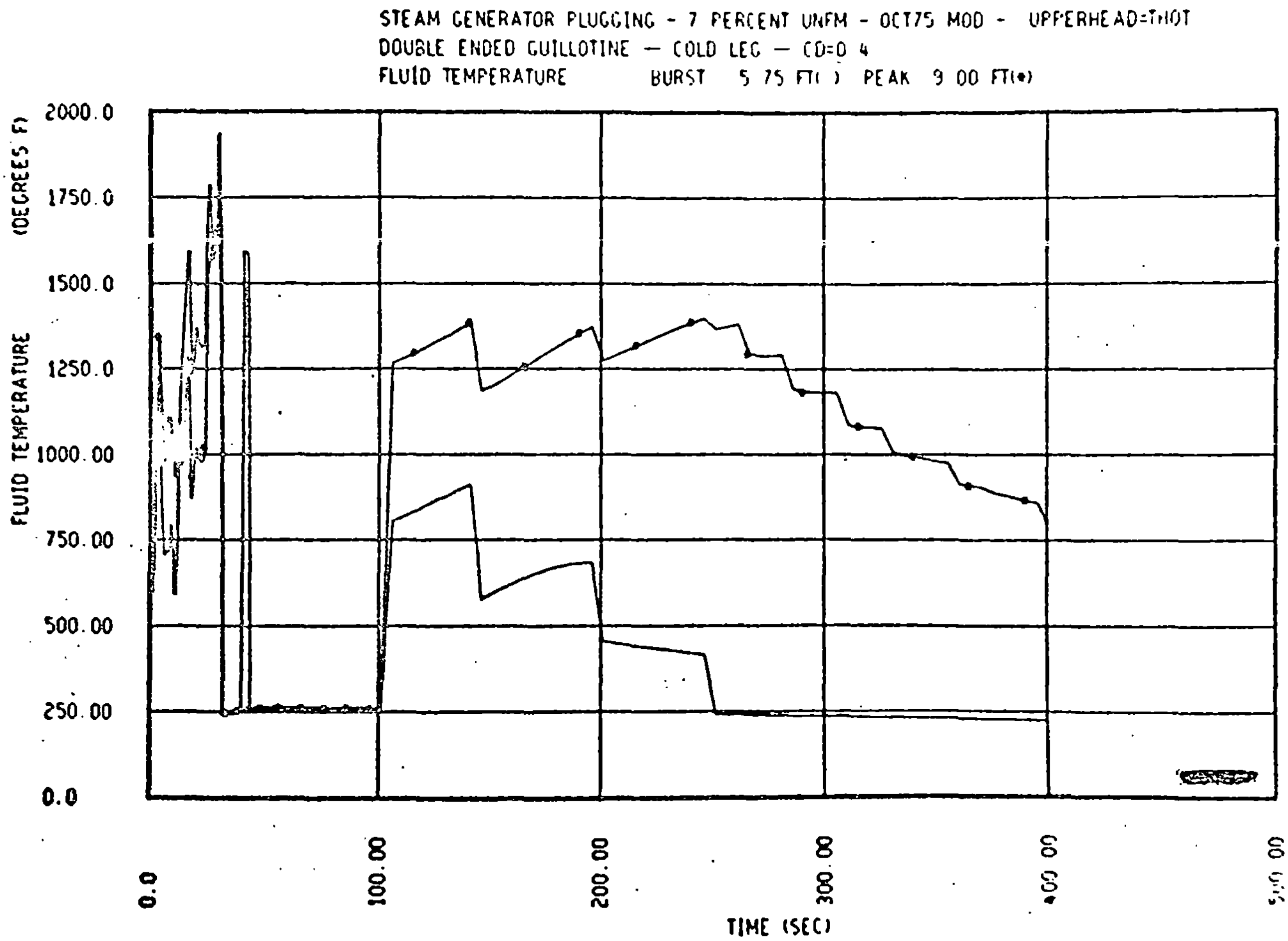
CASE A



FLUID TEMPERATURE - DECLG (CD = 0.6)

FIGURE 8c

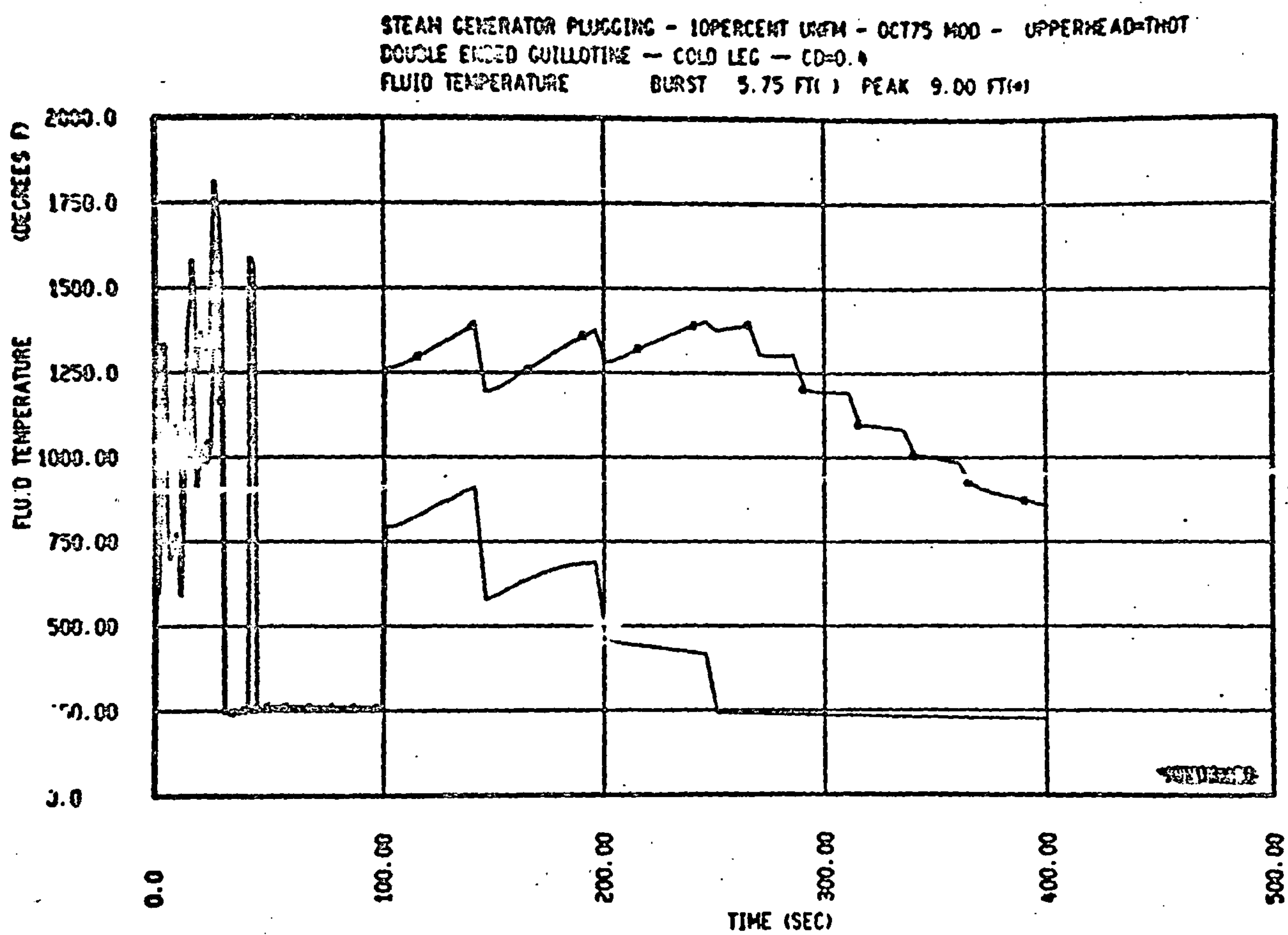
CASE A



FLUID TEMPERATURE – DECLG (CD = 0.4)

FIGURE 8d

CASE B



FLUID TEMPERATURE – DECLG (C_D = 0.4)

STEAM GENERATOR PLUGGING - 12PERCENT UNFM - OCT75 MOD - UPPERHEAD=THOT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD=0.4
FLUID TEMPERATURE BURST 5.75 FT( ) PEAK 9.00 FT(•)

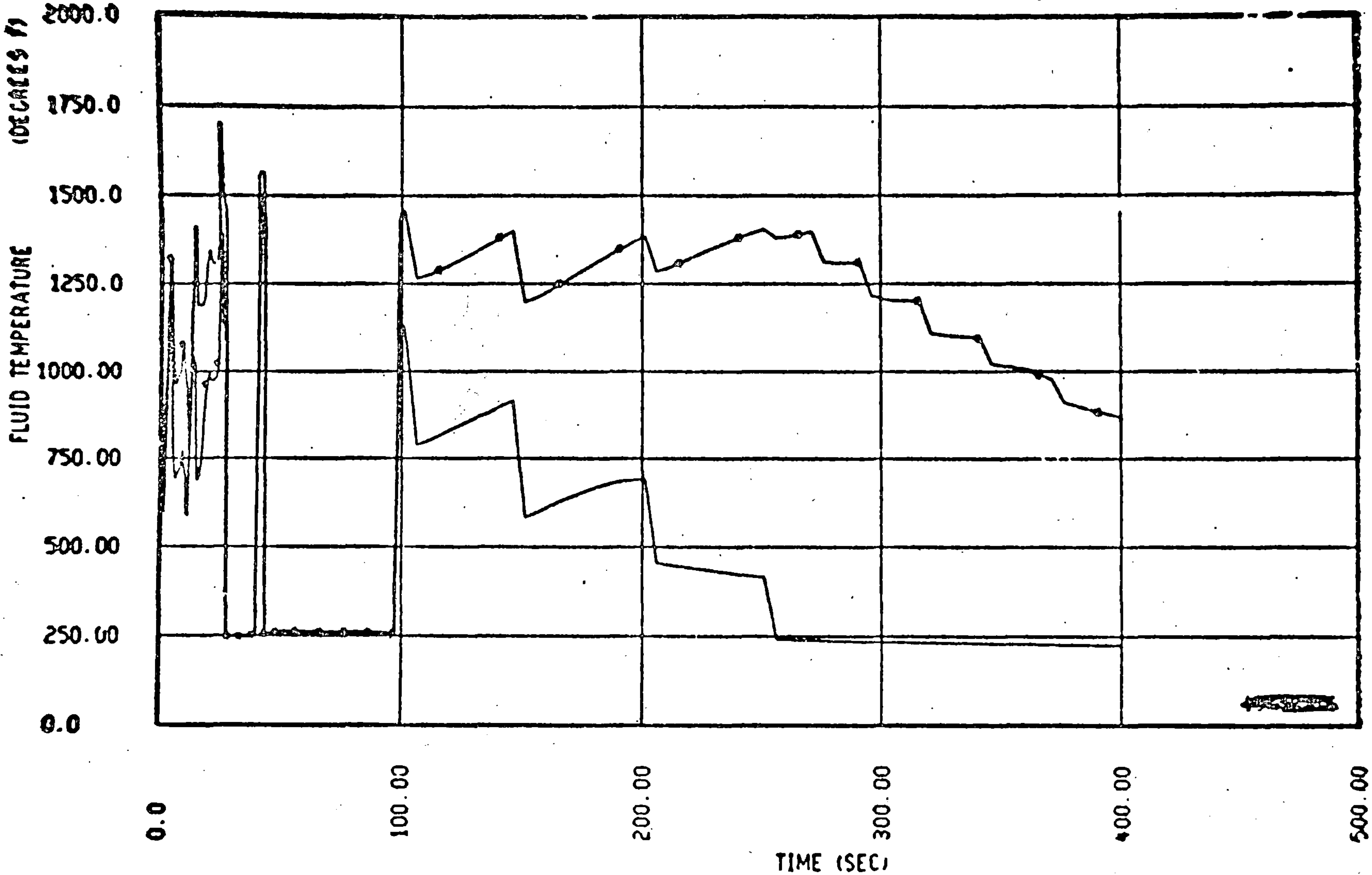
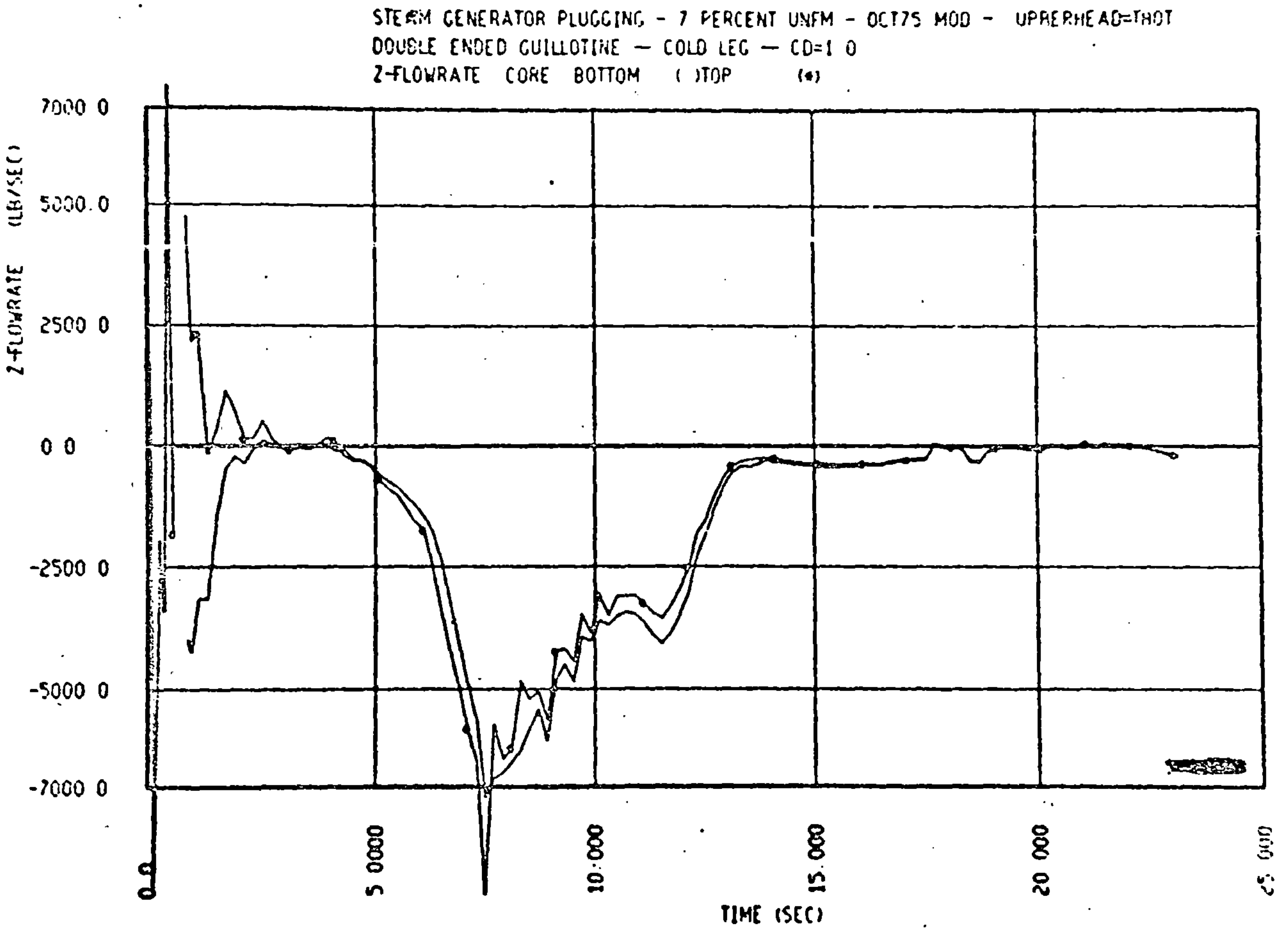


FIGURE 8e
CASE C

FLUID TEMPERATURE – DECLG (C_D = 0.4)

FIGURE 9a

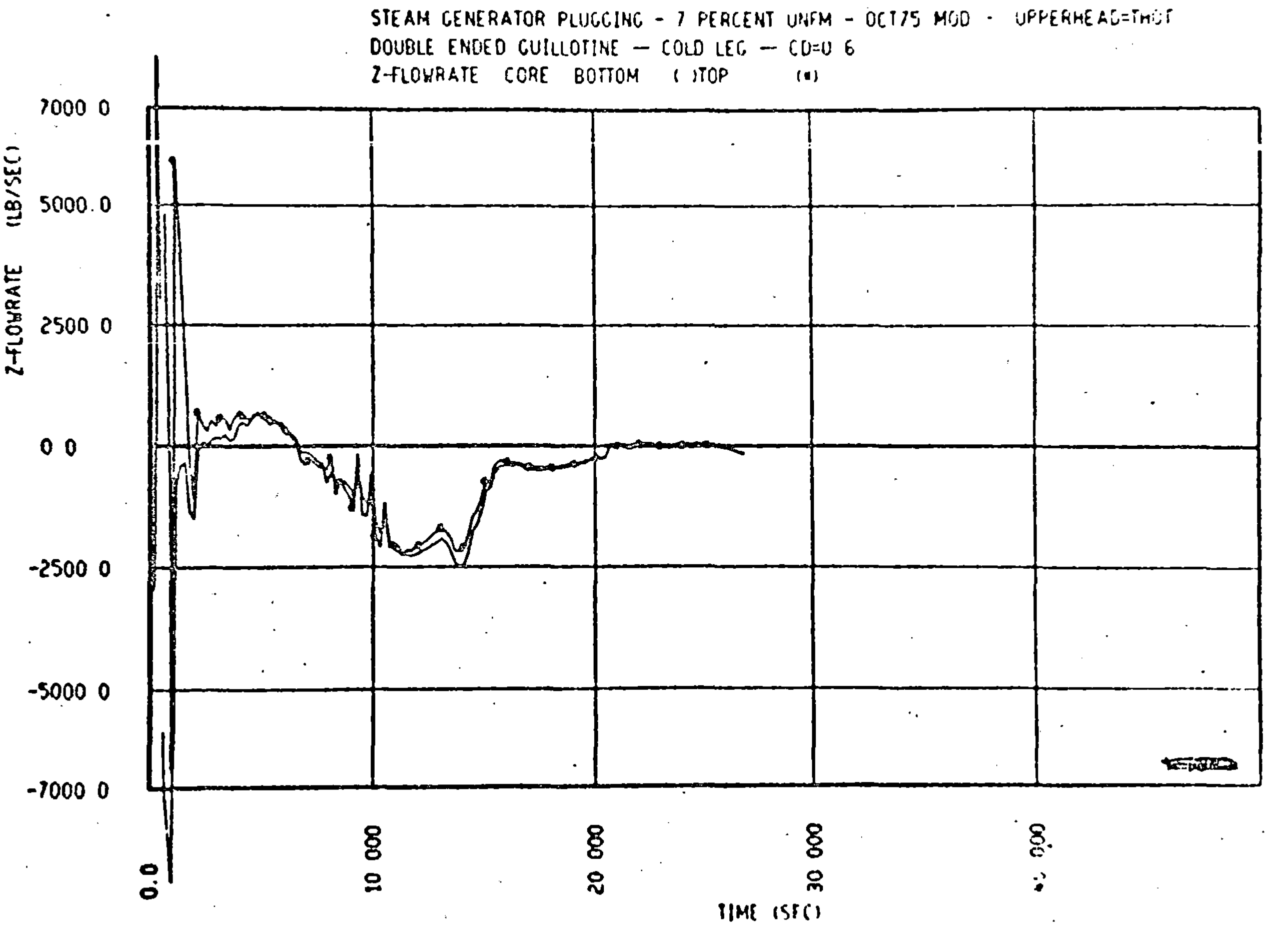
CASE A



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

FIGURE 9b

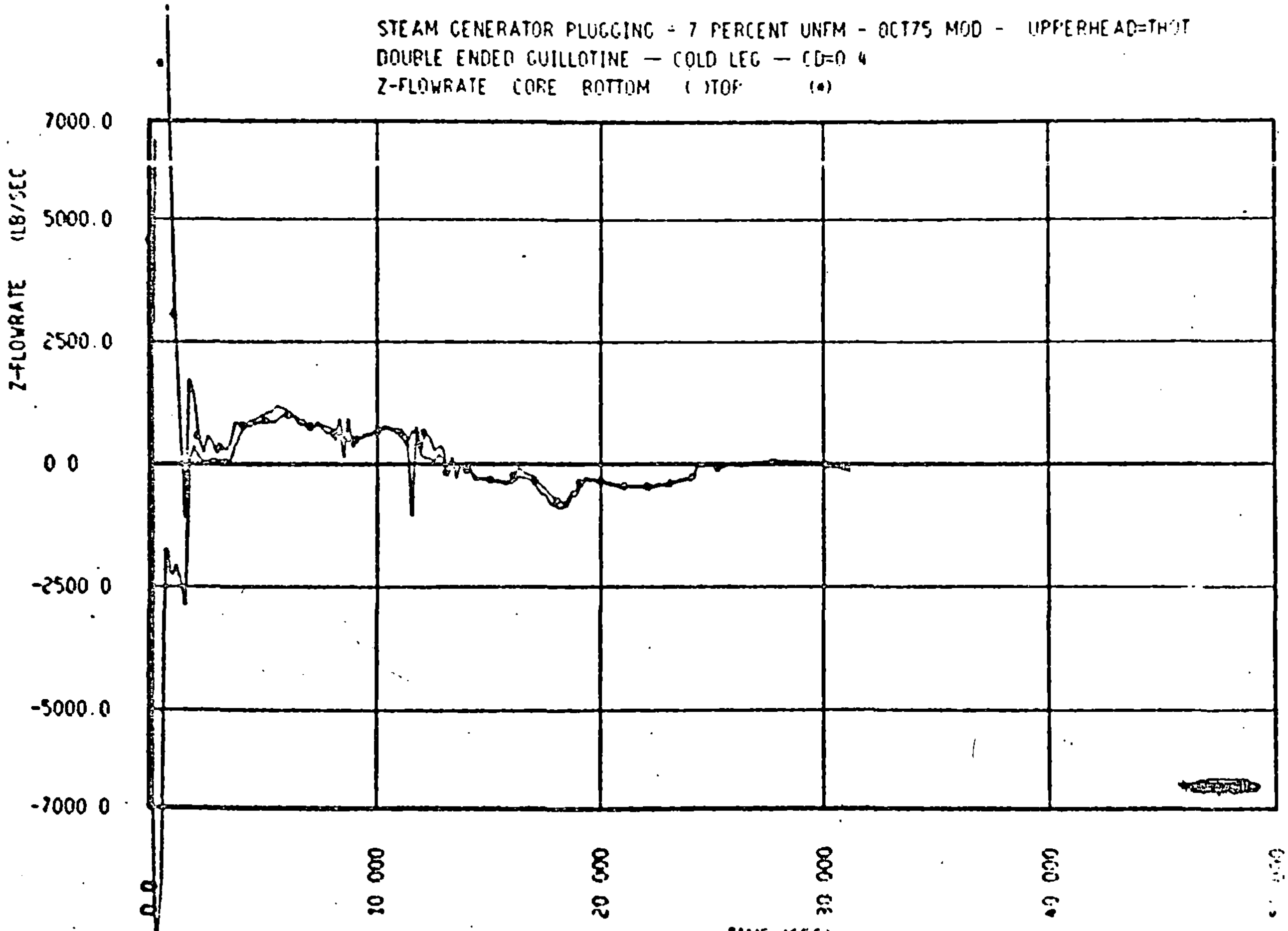
CASE A



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

FIGURE 9c

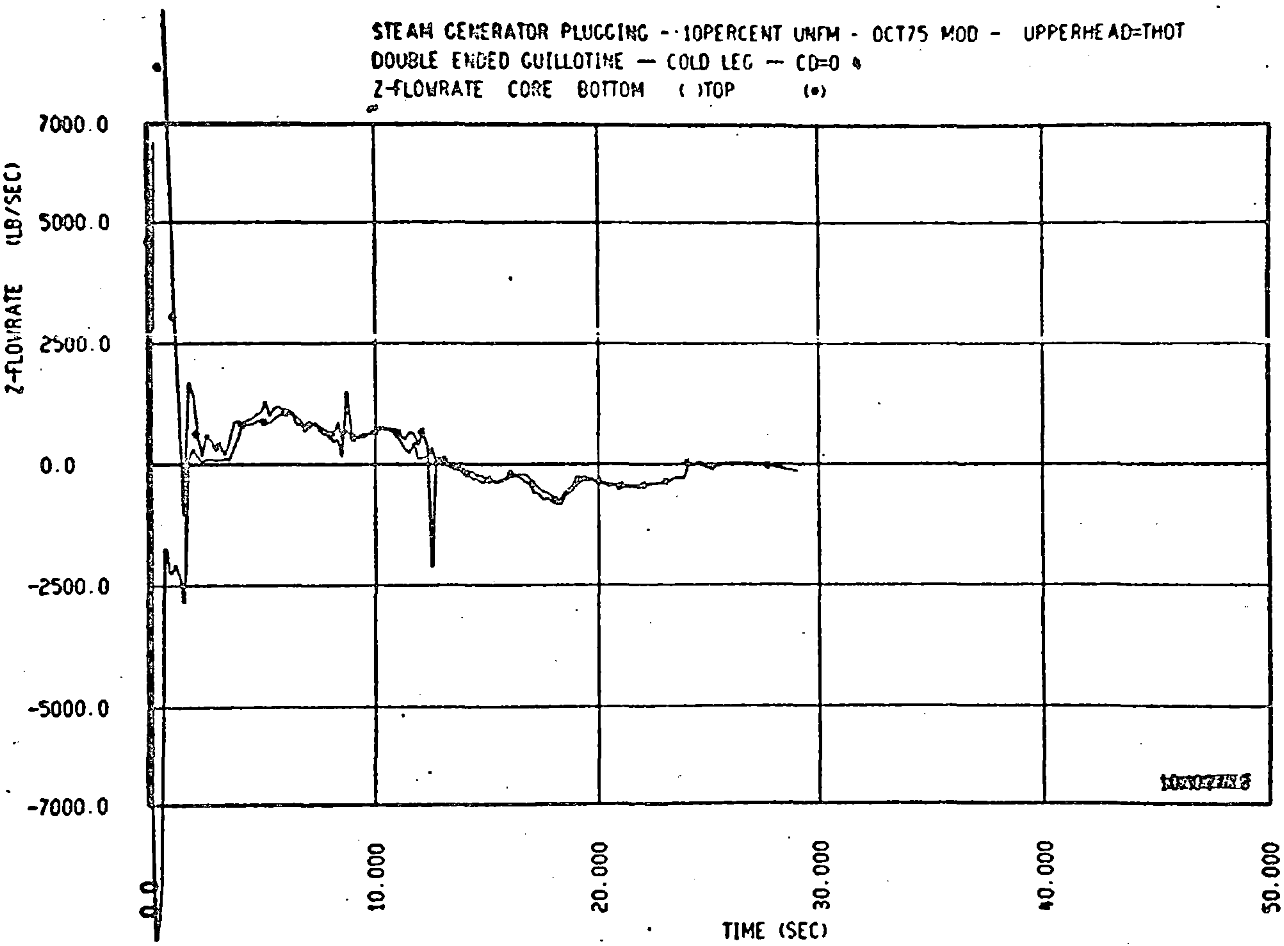
CASE A



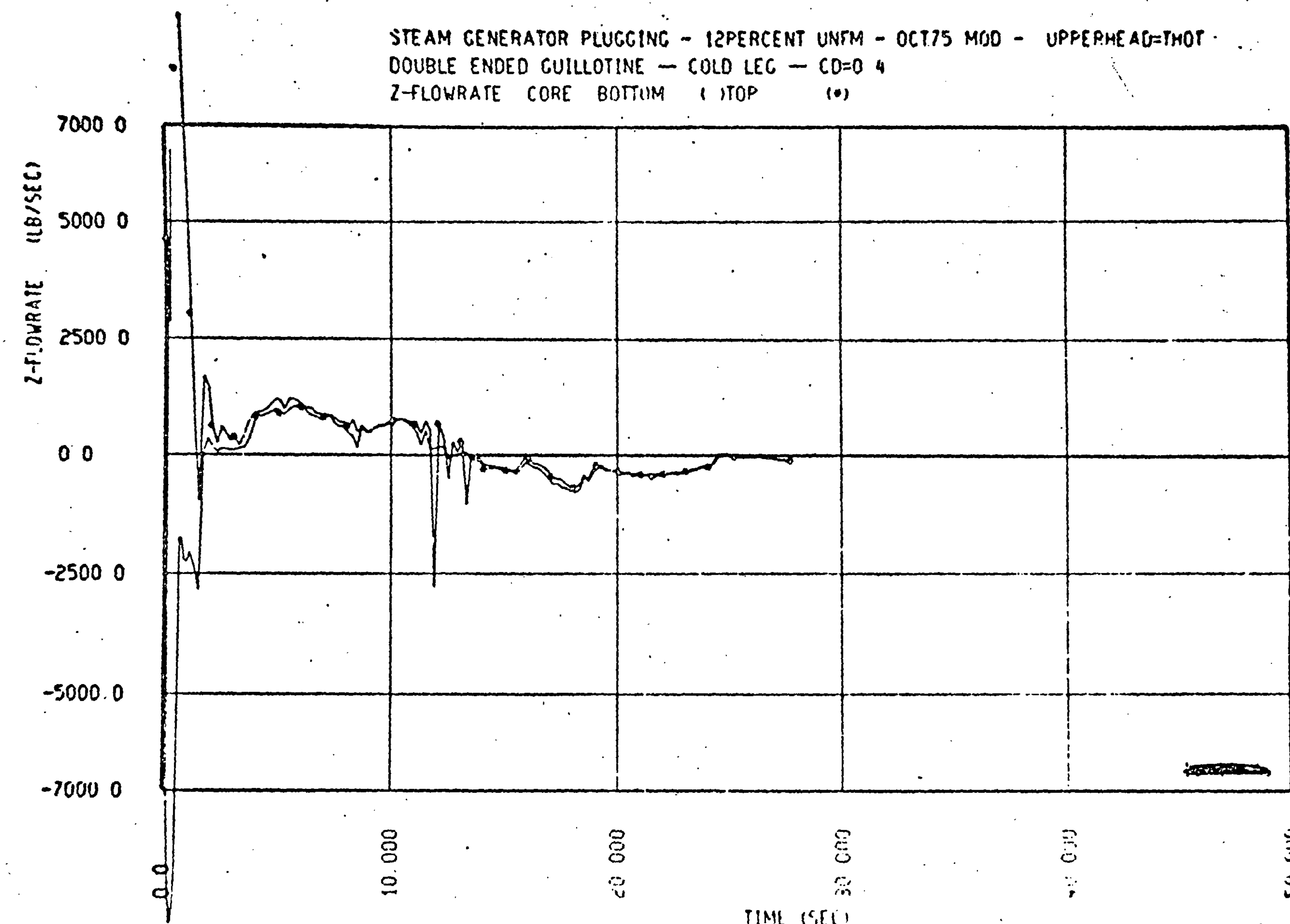
CORE FLOW - TOP AND BOTTOM - DECLG (C_D = 0.4)

FIGURE 9d

CASE B



CORE FLOW - TOP AND BOTTOM - DECLG (C_D = 0.4)



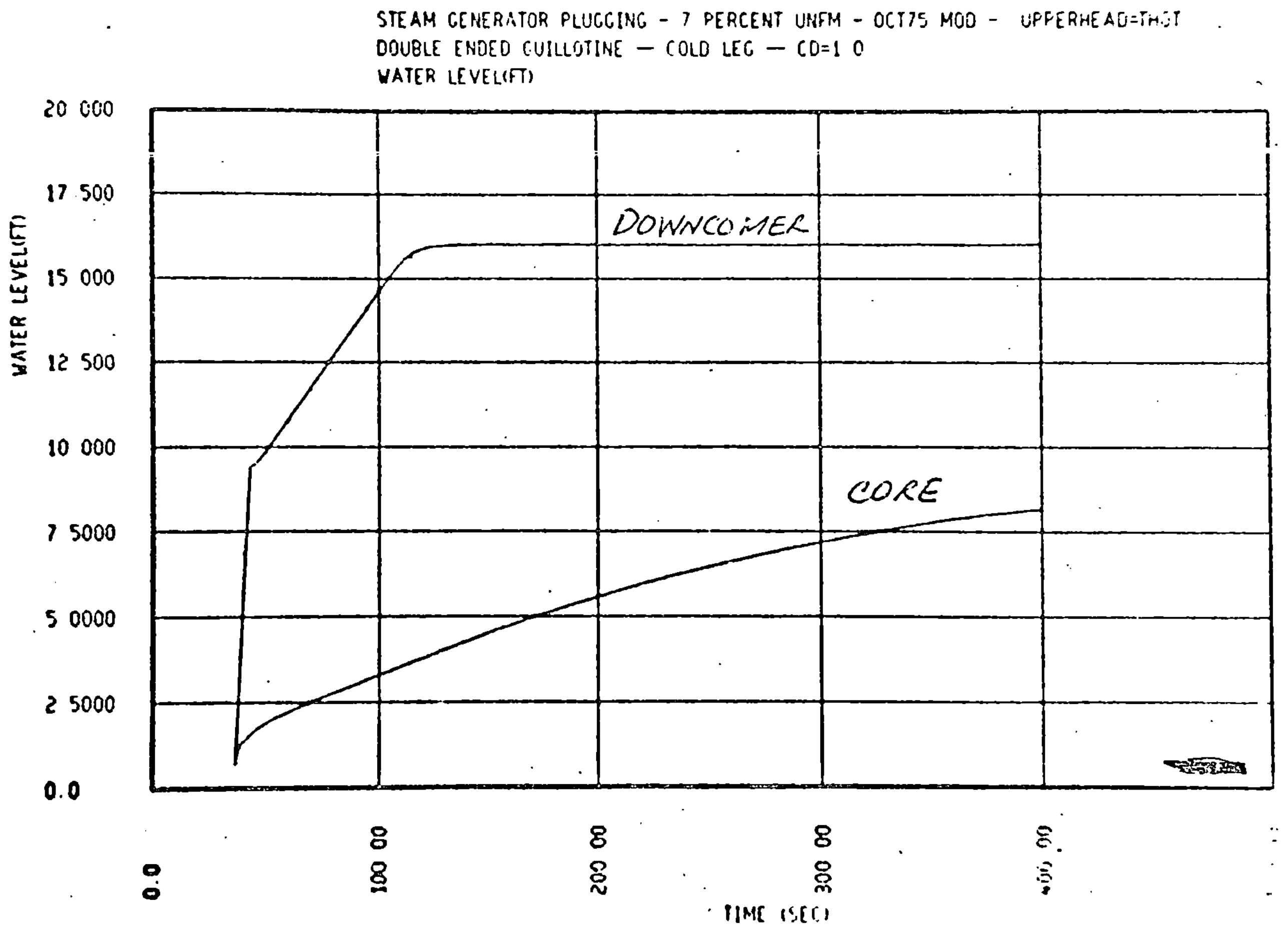
CORE FLOW - TOP AND BOTTOM - DECLG (CD = 0.4)

FIGURE 9e

CASE C

FIGURE 10a

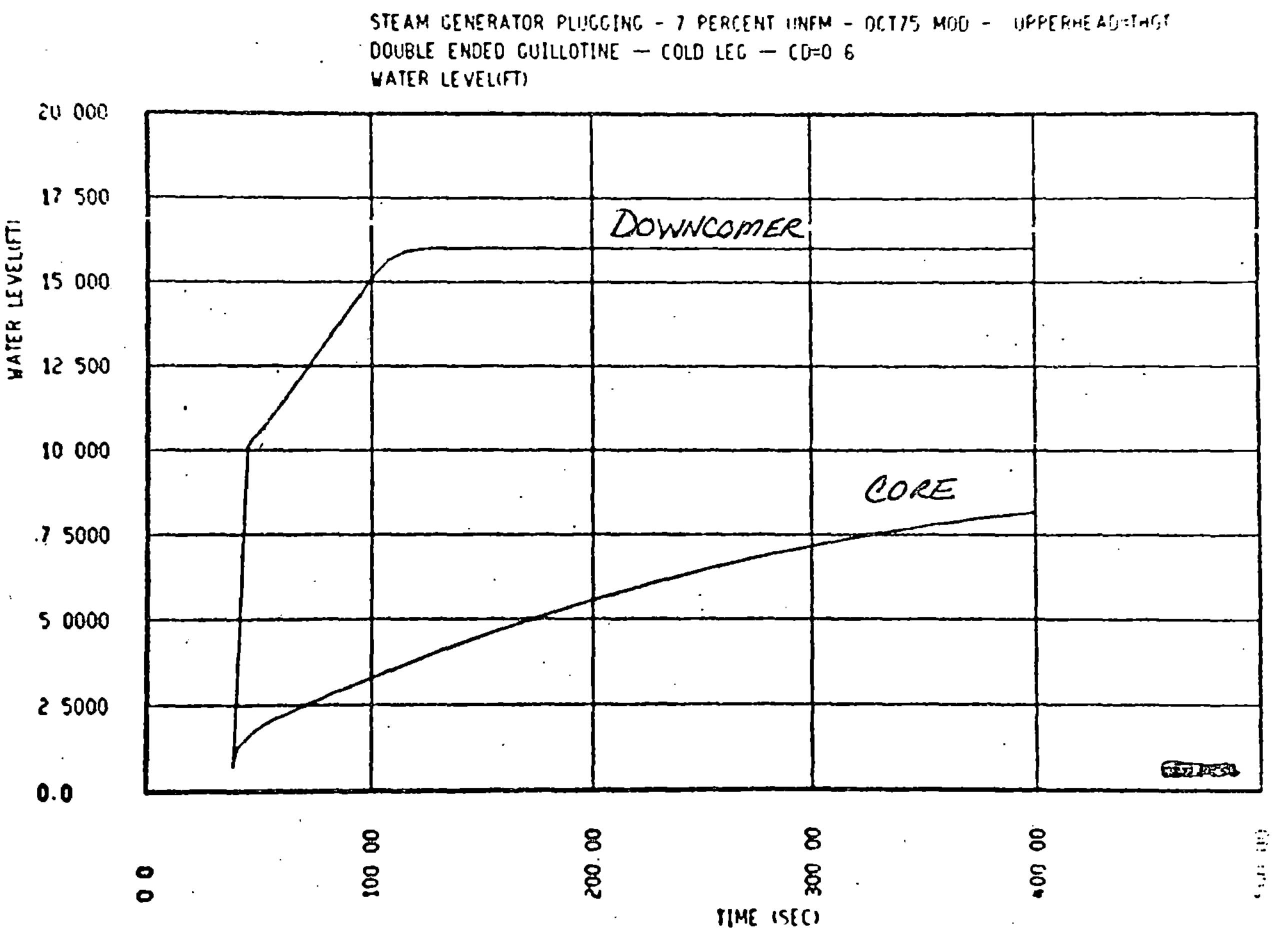
CASE A



REFLOOD TRANSIENT - DECLG ($C_D = 1.0$)
DOWNCOMER AND CORE WATER LEVELS

FIGURE 10b

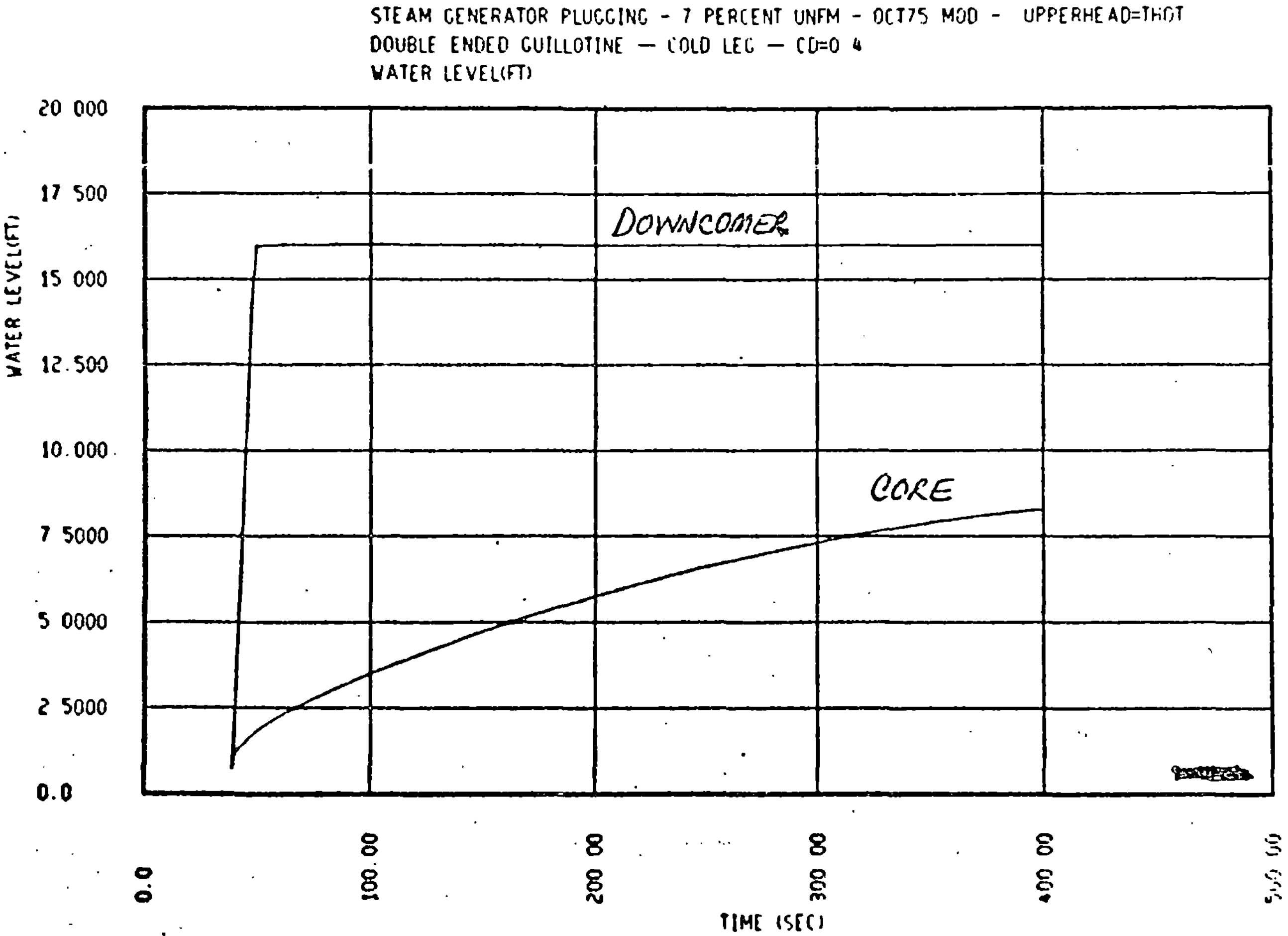
CASE A



REFLOOD TRANSIENT - DECLG ($C_D = 0.6$)
DOWNCOMER AND CORE WATER LEVELS

FIGURE 10c

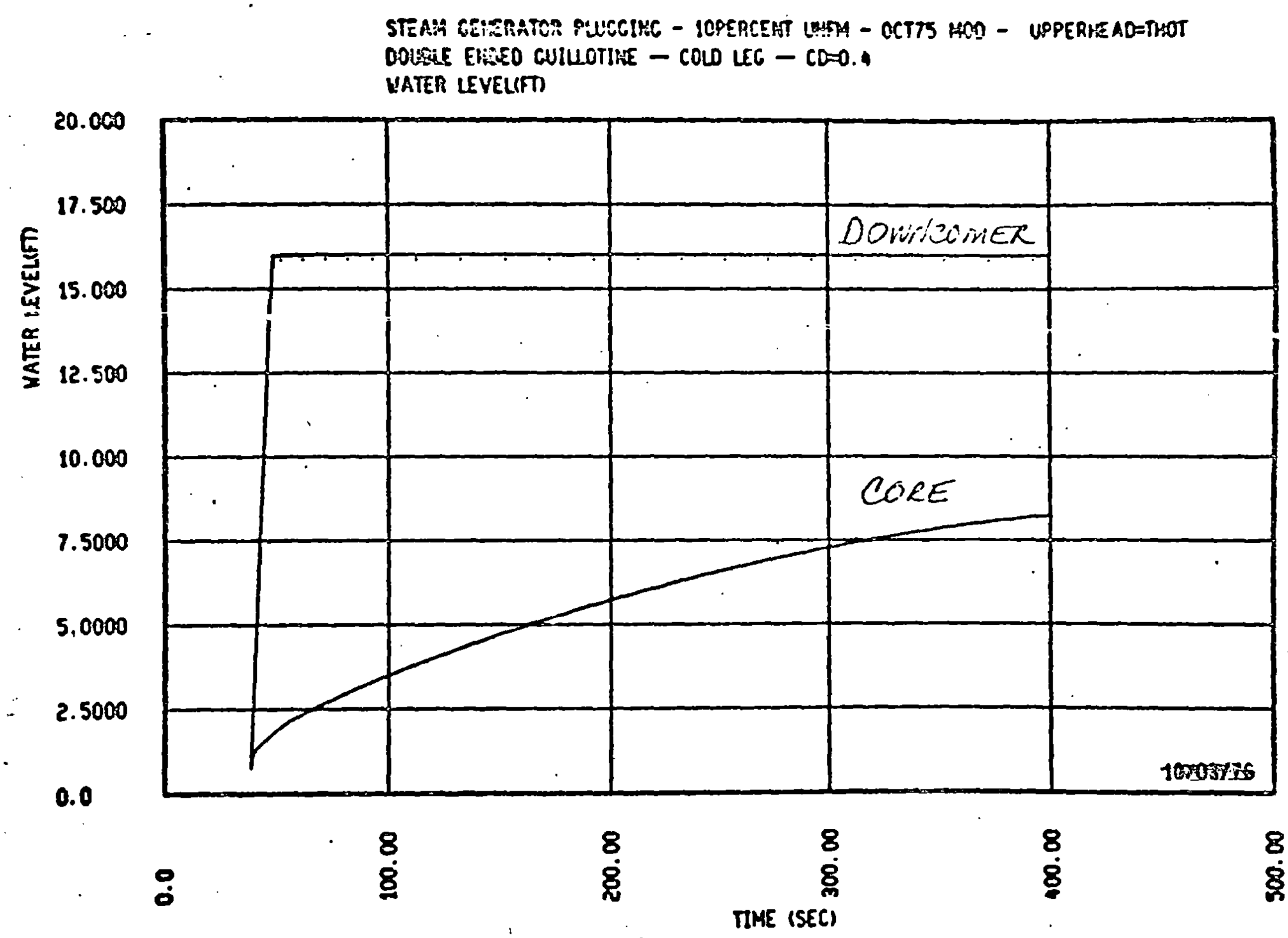
CASE A



REFLOOD TRANSIENT - DECLG (C_D = 0.4)
DOWNCOMER AND CORE WATER LEVELS

FIGURE 10d

CASE B



REFLOOD TRANSIENT - DECLG (CD =0.4)
DOWNCOMER AND CORE WATER LEVELS

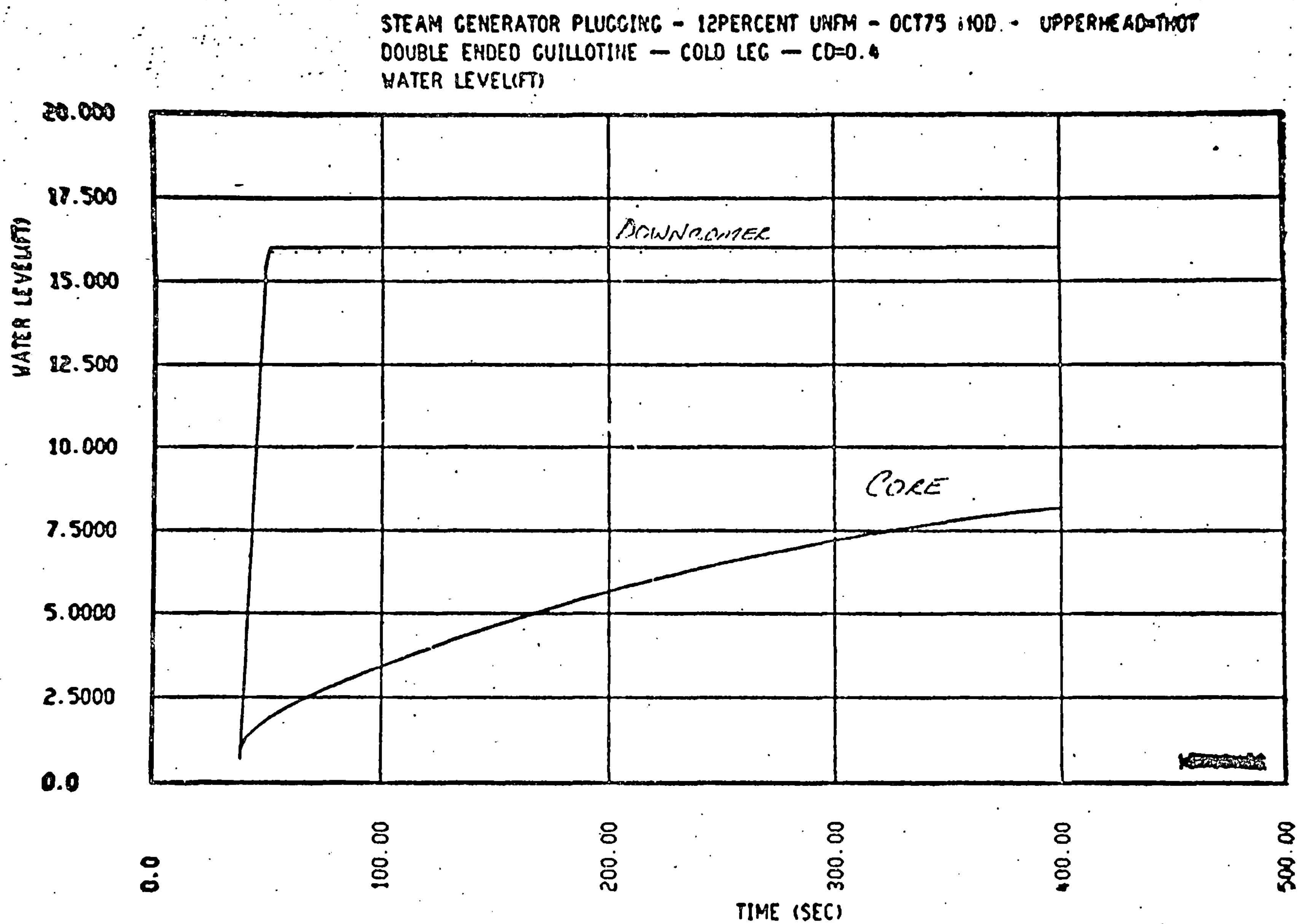


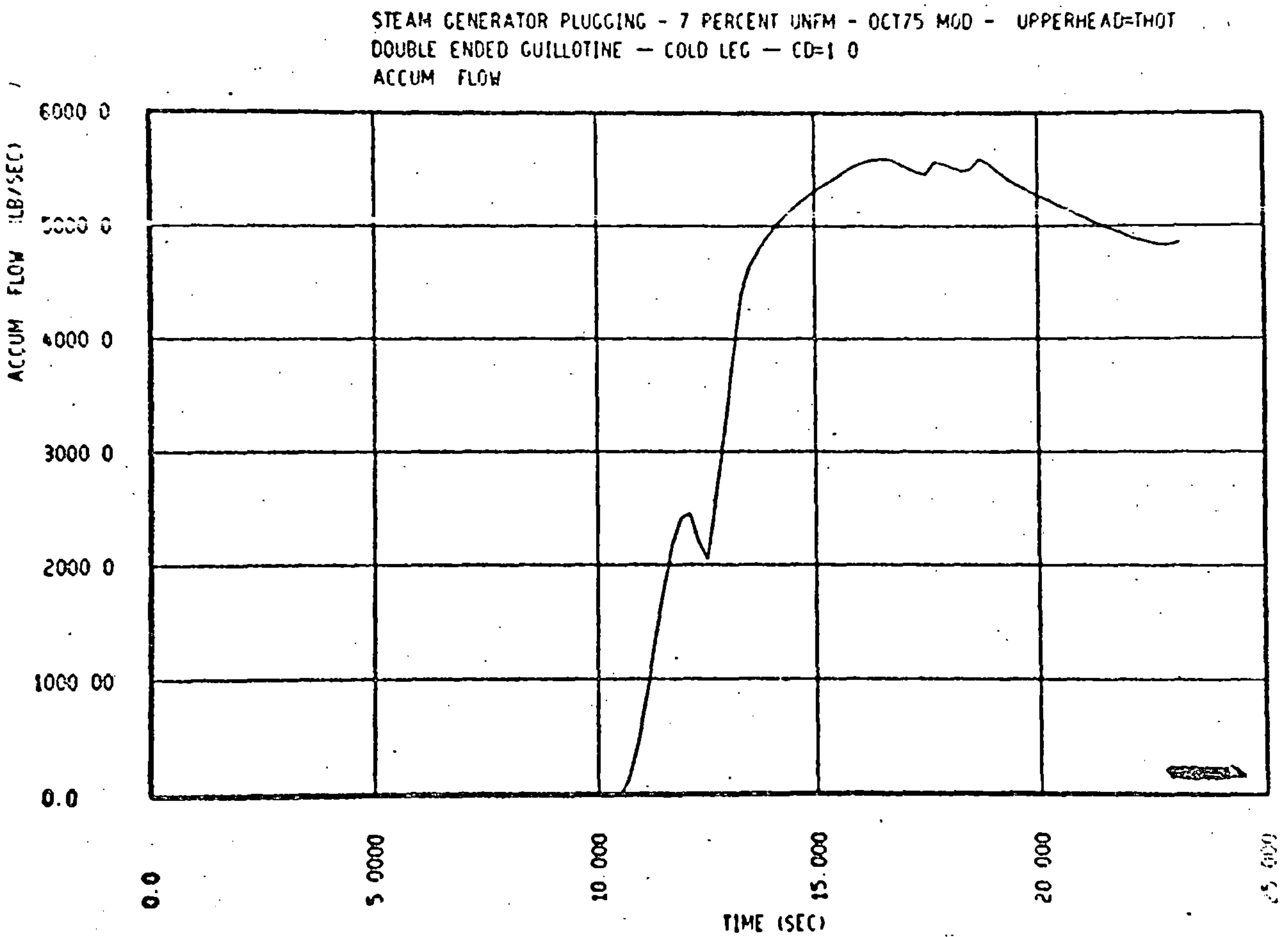
FIGURE 10e

CASE C

REFLOOD TRANSIENT - DECLG (C_D = 0.4)
DOWNCOMER AND CORE WATER LEVELS

FIGURE 11a

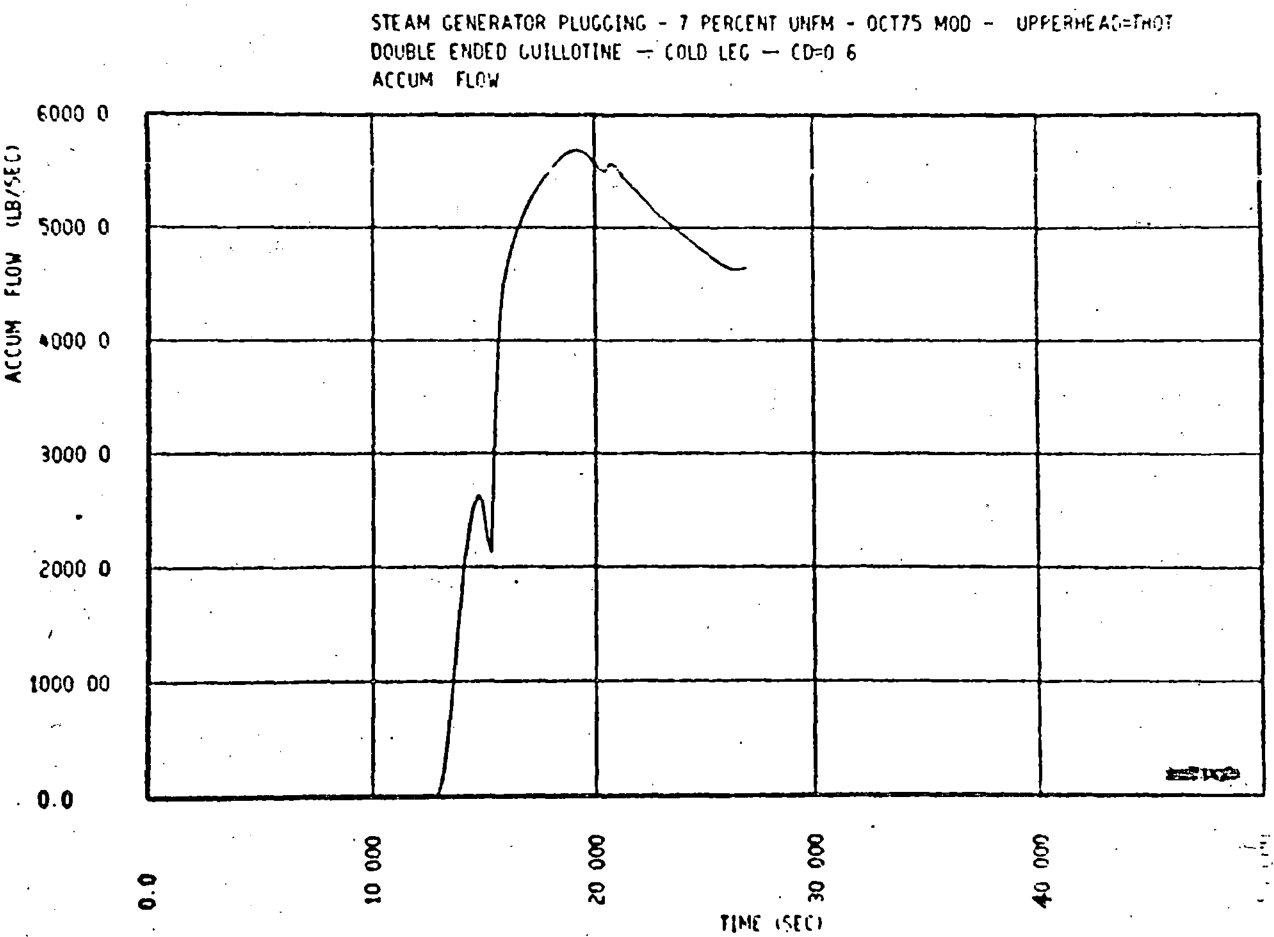
CASE A



ACCUMULATOR FLOW (BLOWDOWN) - DECLG ($C_D = 1.0$)

FIGURE 11b

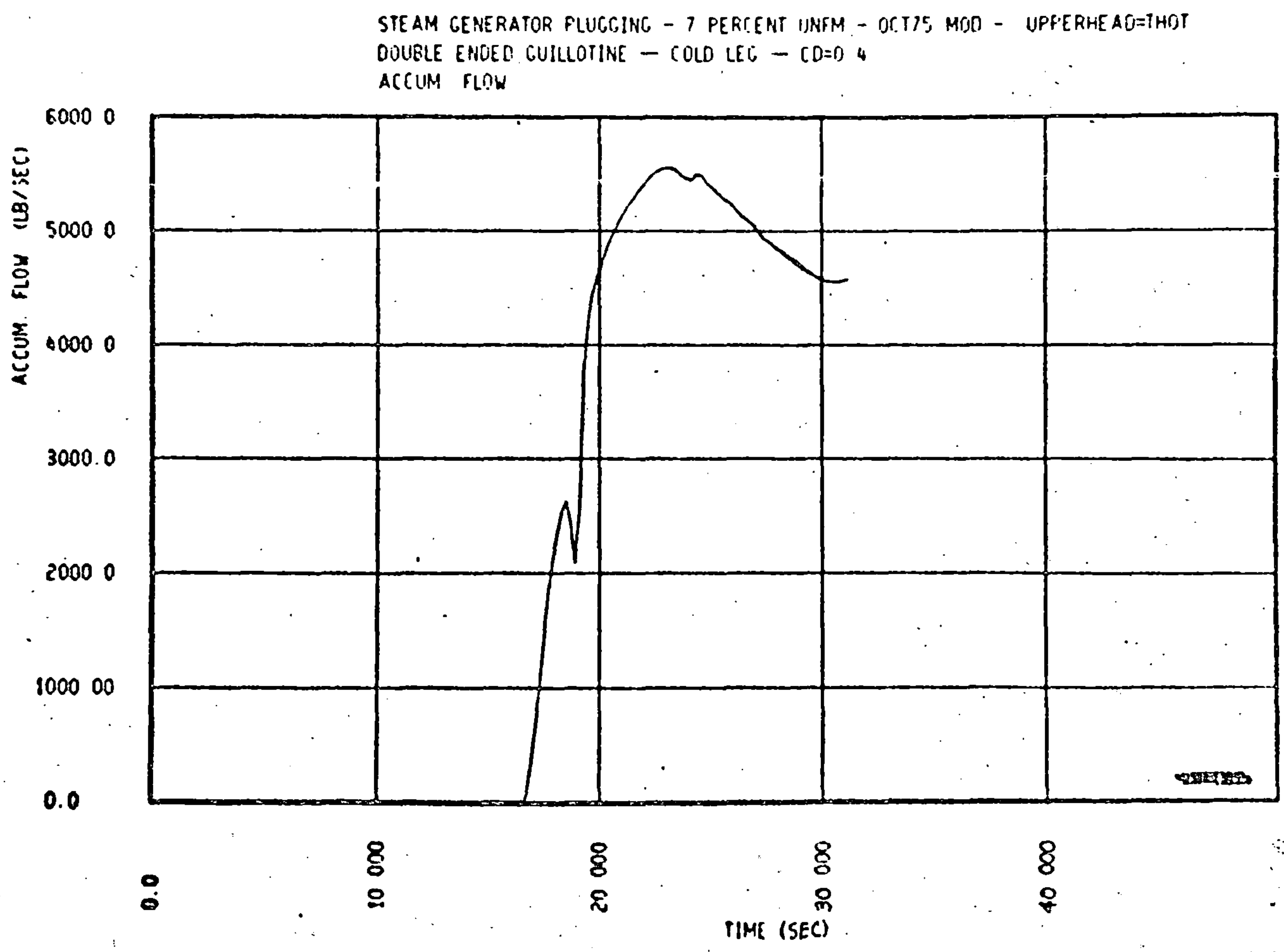
CASE A



ACCUMULATOR FLOW (BLOWDOWN) - DECLG ($C_D = 0.6$)

FIGURE 11c

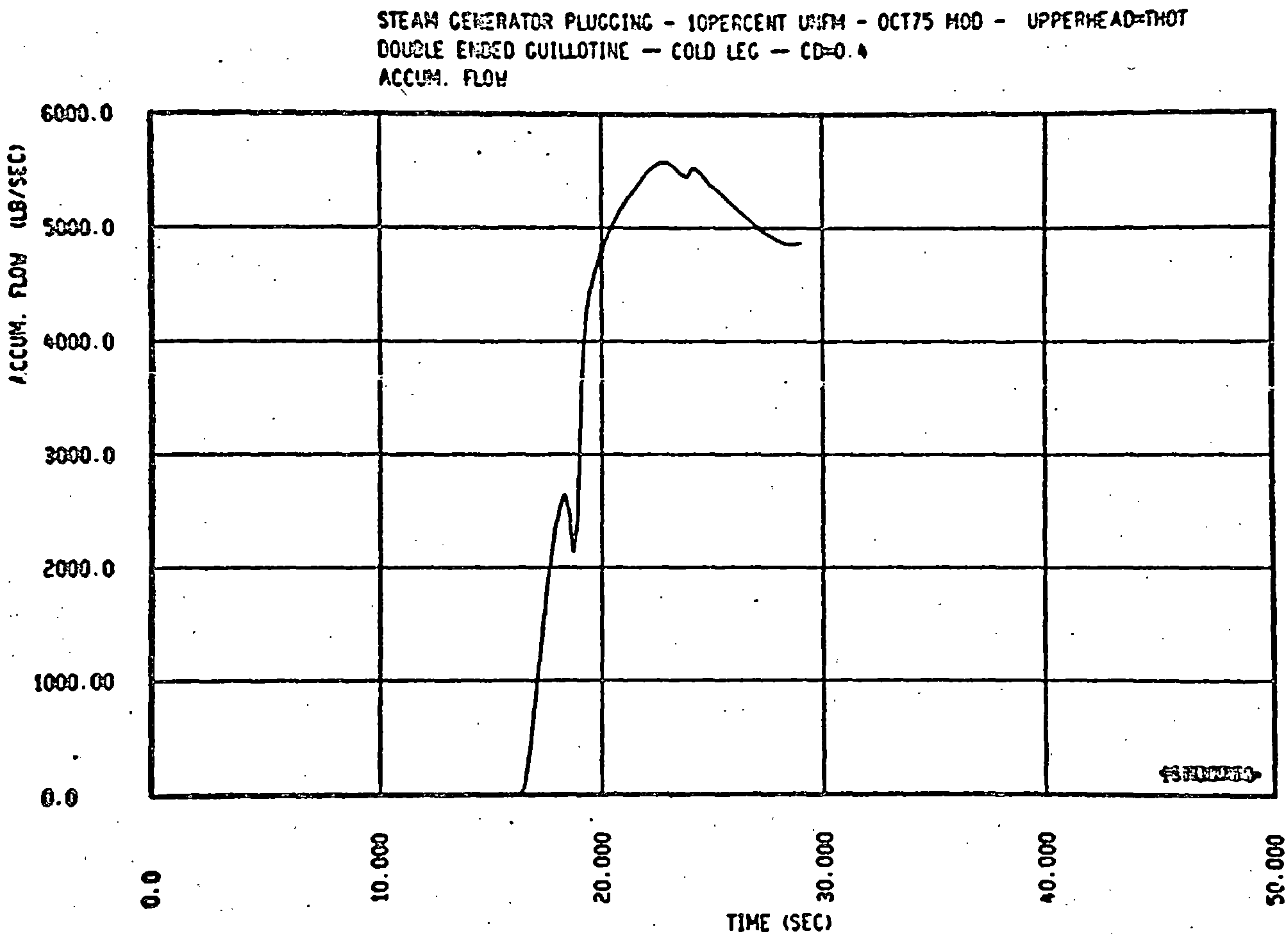
CASE A



ACCUMULATOR FLOW (BLOWDOWN) - DECLG (C_D = 0.4)

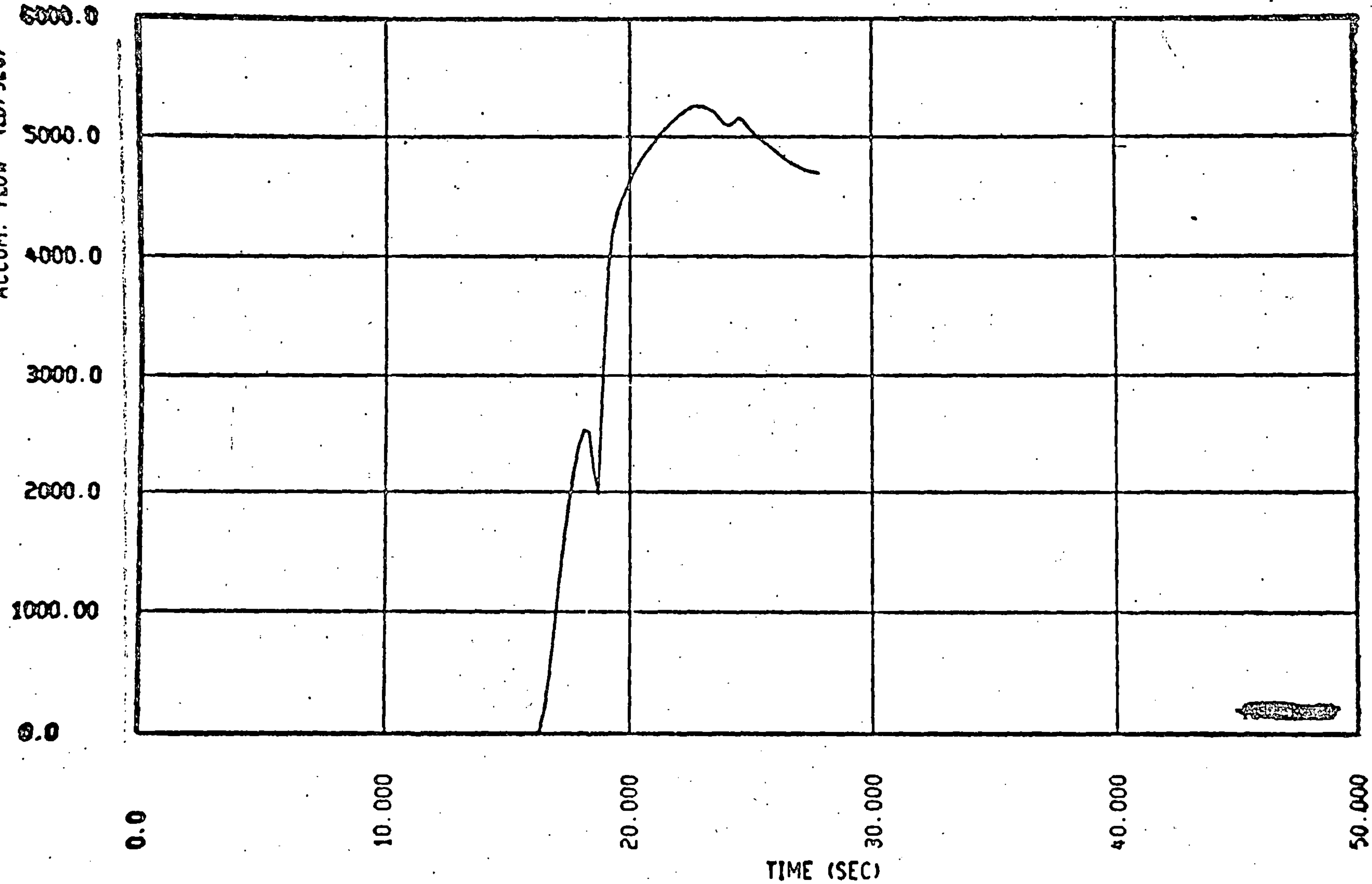
FIGURE 11d

CASE B



ACCUMULATOR FLOW (BLOWDOWN) - DECLG (C_D = 0.4)

STEAM GENERATOR PLUGGING - 12PERCENT UNFM - OCT75 MOD - UPPERHEAD=THOT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD=0.4
ACCUM. FLOW



ACCUMULATOR FLOW (BLOWDOWN)- DECLG (C_D = 0.4)

CASE C

FIGURE 11e

FIGURE 12a

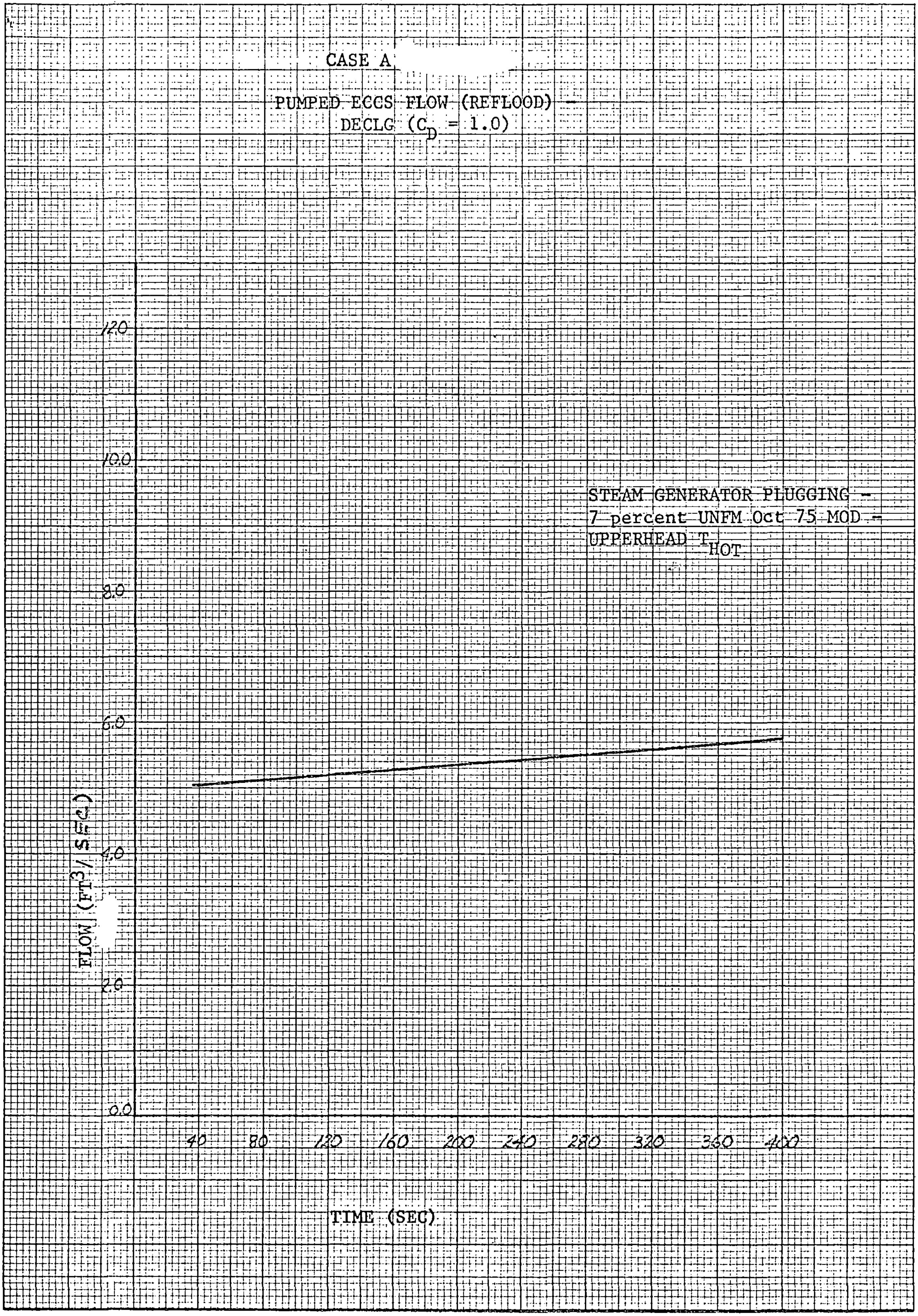
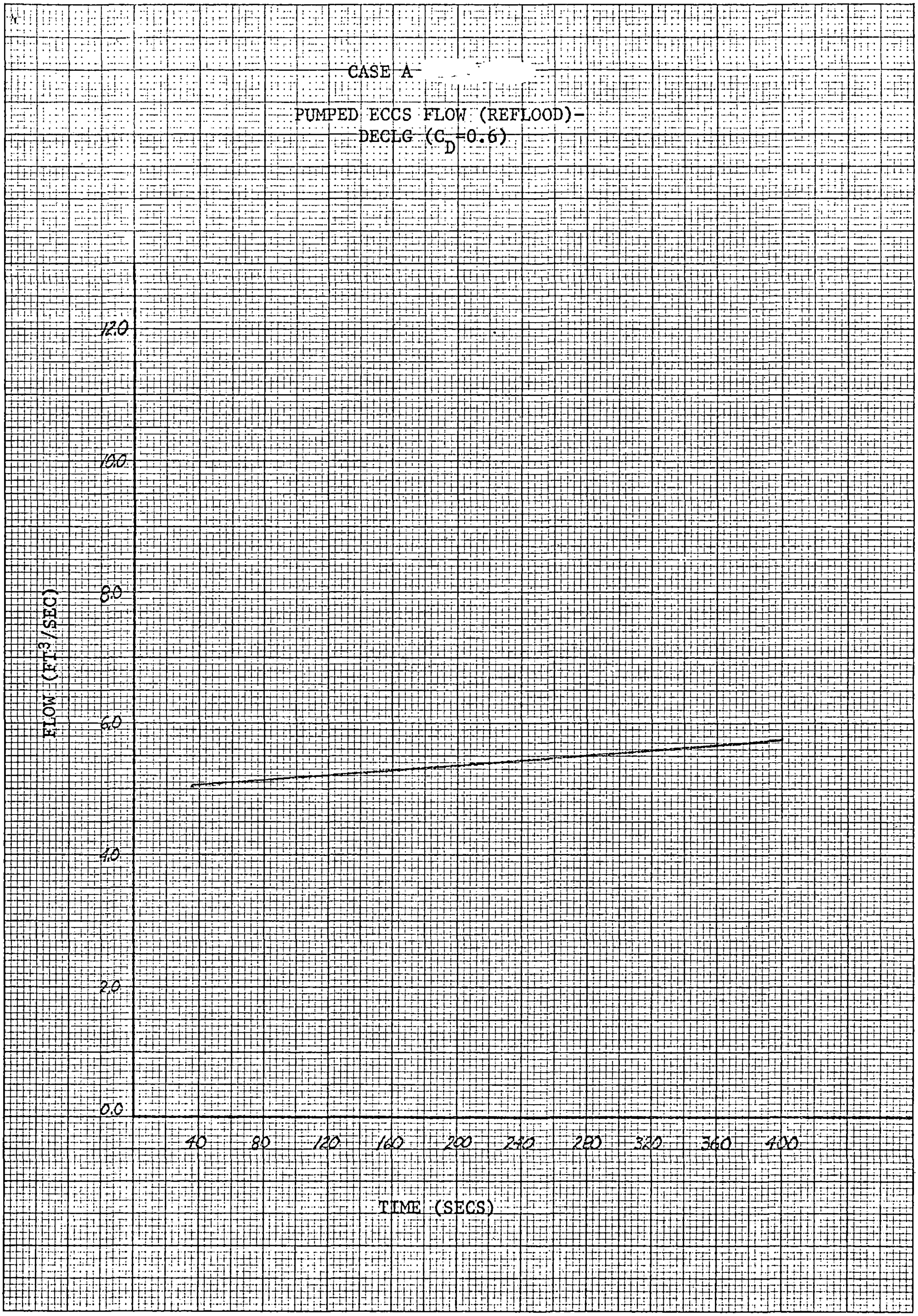
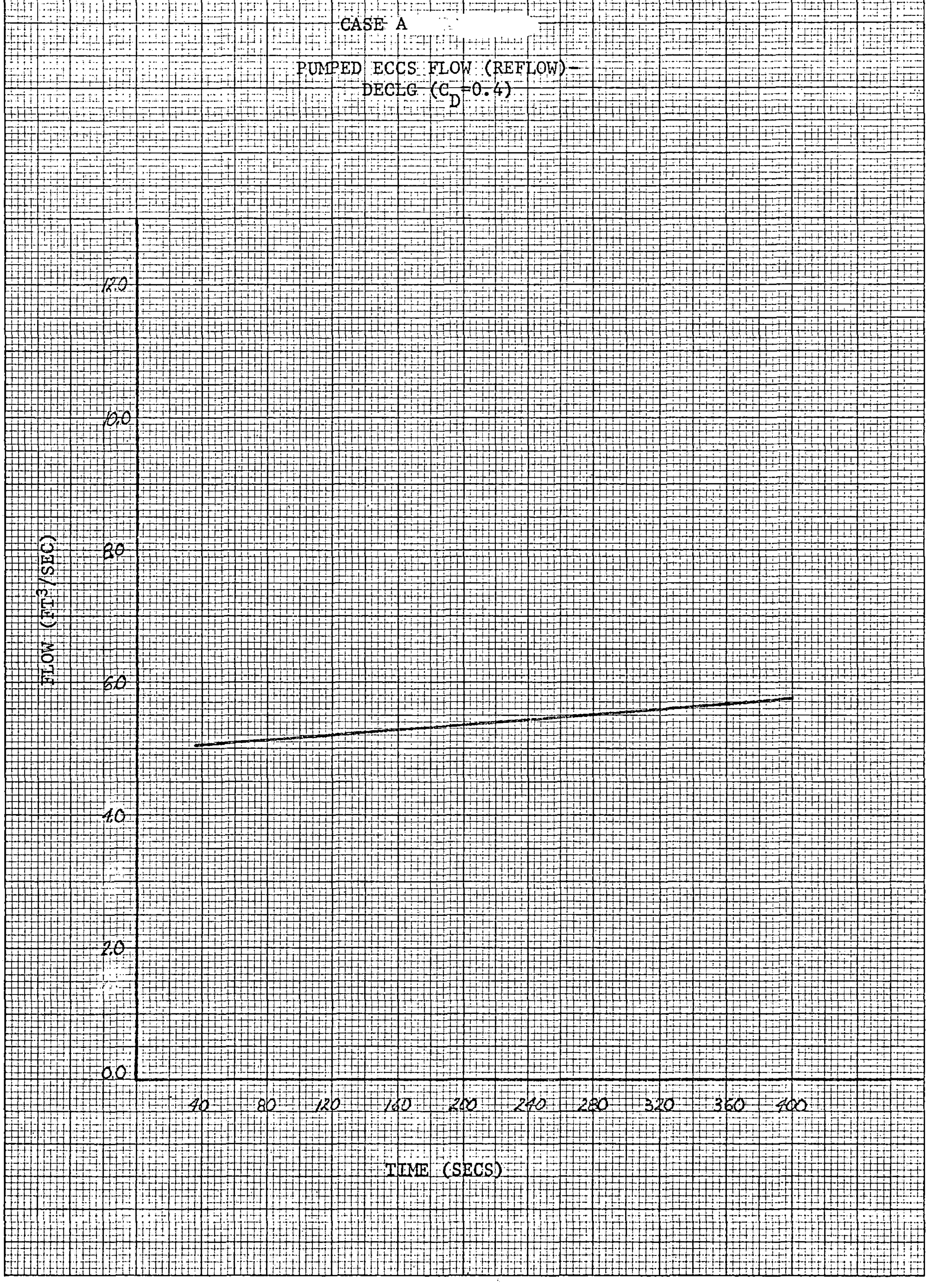

CASE A
PUMPED ECCS FLOW (REFLOOD) -
DECLG (C_D = 1.0)
STEAM GENERATOR PLUGGING -
7 percent UNFM Oct 75 MOD -
UPPERHEAD T_{HOT}
FLOW (FT^3/SEC)
12.0
10.0
8.0
6.0
4.0
2.0
0.0
40
80
120
160
200
240
280
320
360
400
TIME (SEC)

FIGURE 12b



CASE A
PUMPED ECCS FLOW (REFLOOD)-
DECLG (C_D=0.6)
FLOW (FT^3/SEC)
12.0
10.0
8.0
6.0
4.0
2.0
0.0
40
80
120
160
200
240
280
320
360
400
TIME (SECS)

FIGURE 12c



10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

46 1320

FIGURE 12d

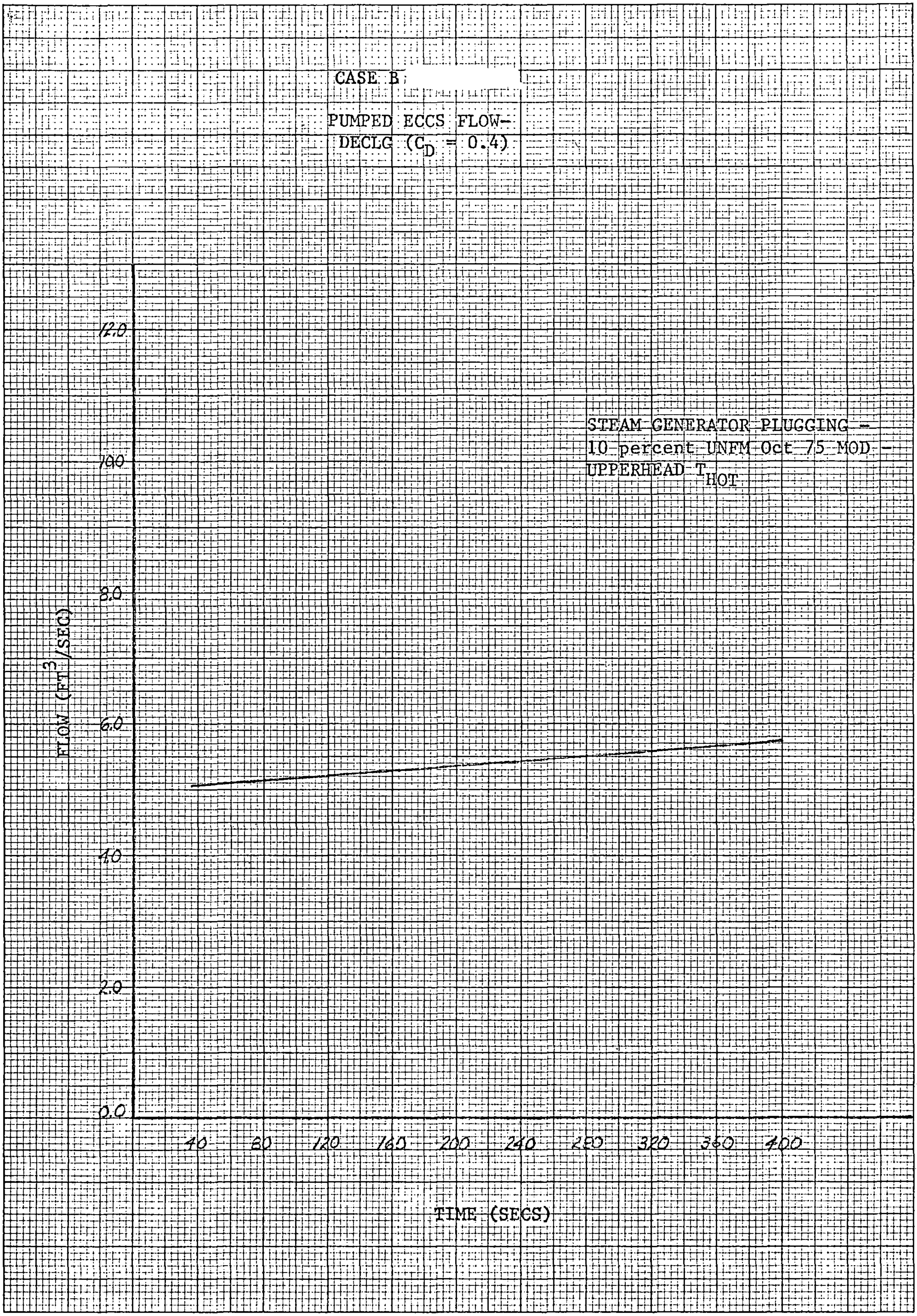
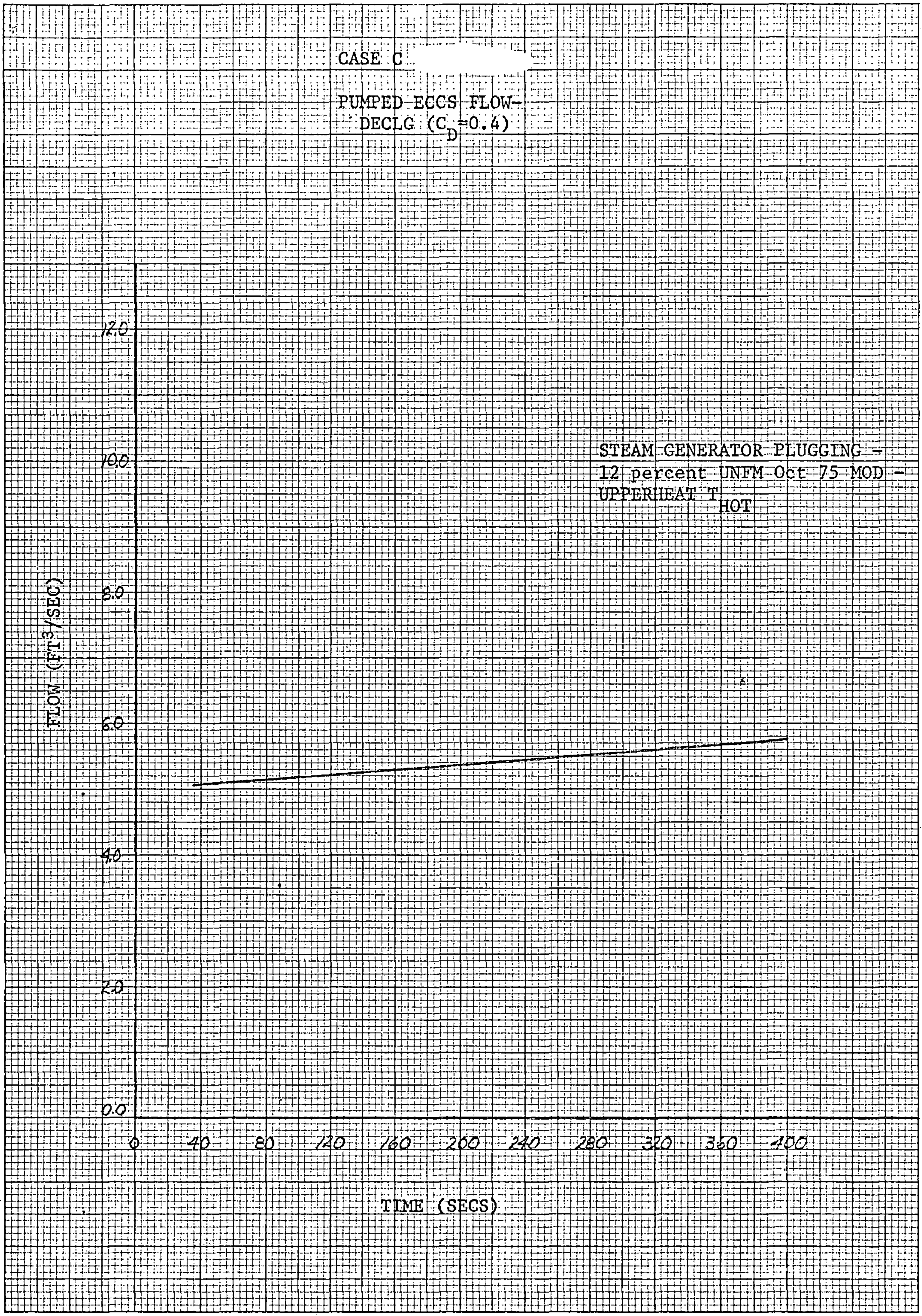

CASE B
PUMPED ECCS FLOW-
DECLG (C_D = 0.4)
STEAM GENERATOR PLUGGING -
10 percent UNFM Oct 75 MOD -
UPPERHEAD T_{HOT}
FLOW (FT^3/SEC)
12.0
10.0
8.0
6.0
4.0
2.0
0.0
40
80
120
160
200
240
280
320
360
400
TIME (SECS)

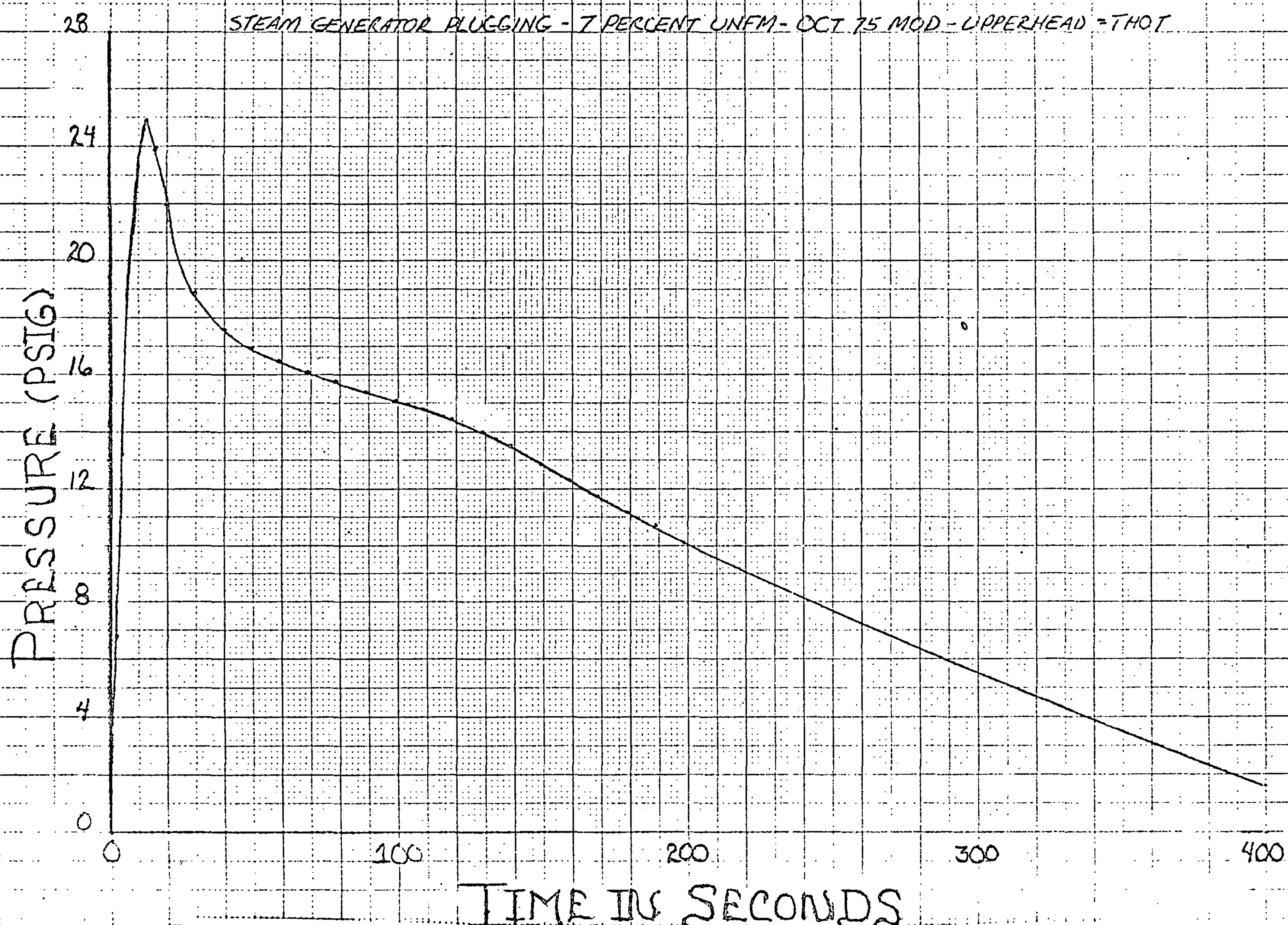
FIGURE 12e



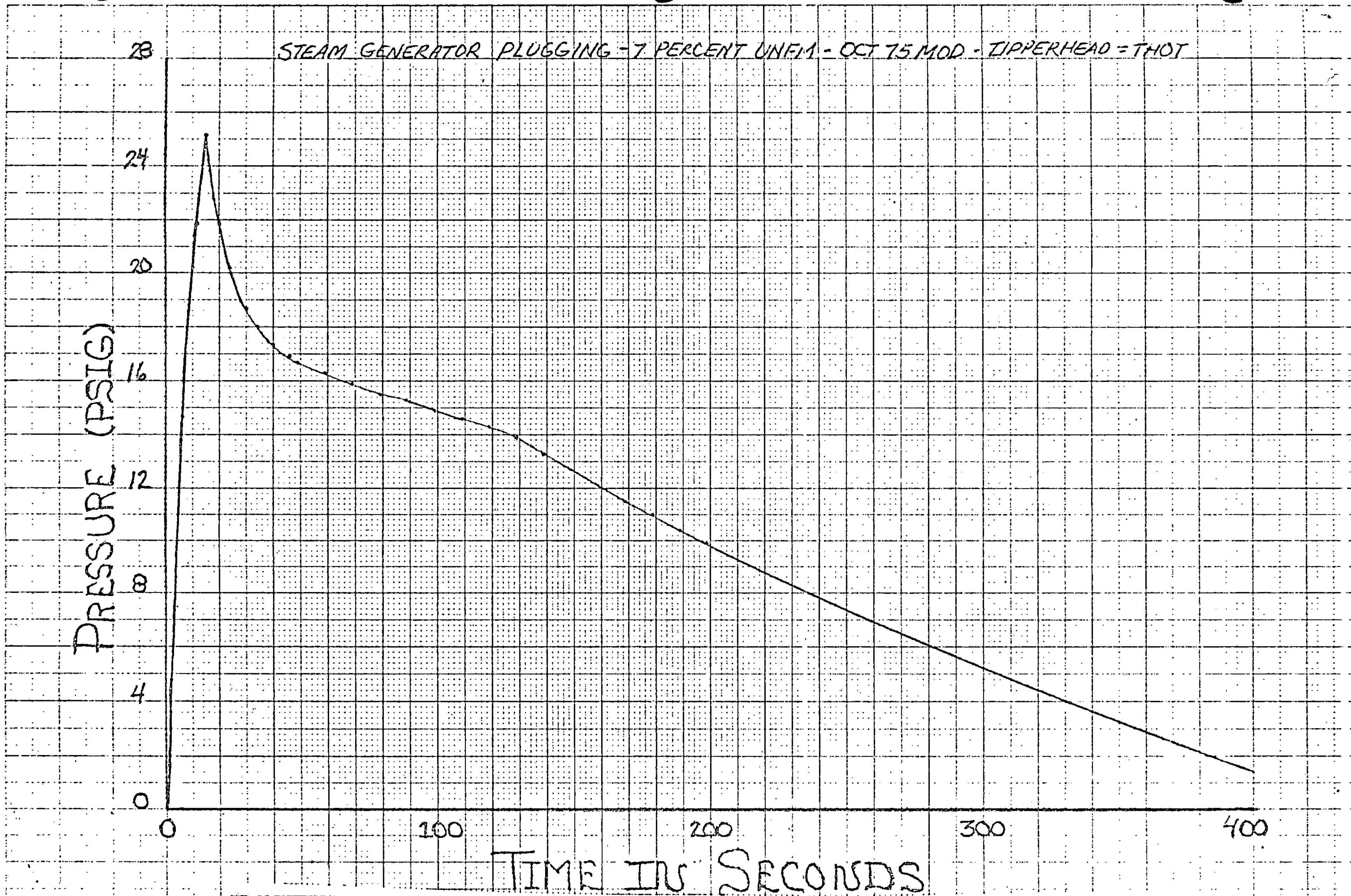
CASE C
PUMPED ECCS FLOW-
DECLG (C_D=0.4)
STEAM GENERATOR PLUGGING -
12 percent UNFM Oct 75 MOD -
UPPERHEAT T_{HOT}
FLOW (FT^3/SEC)
12.0
10.0
8.0
6.0
4.0
2.0
0.0
0
40
80
120
160
200
240
280
320
360
400
TIME (SECS)

FIGURE 15a

CASE A



CONTAINMENT PRESSURE - DECLG (C_D = 1.0)



CONTAINMENT PRESSURE - DECLG (C_D = 0.6)

CASE A

461510

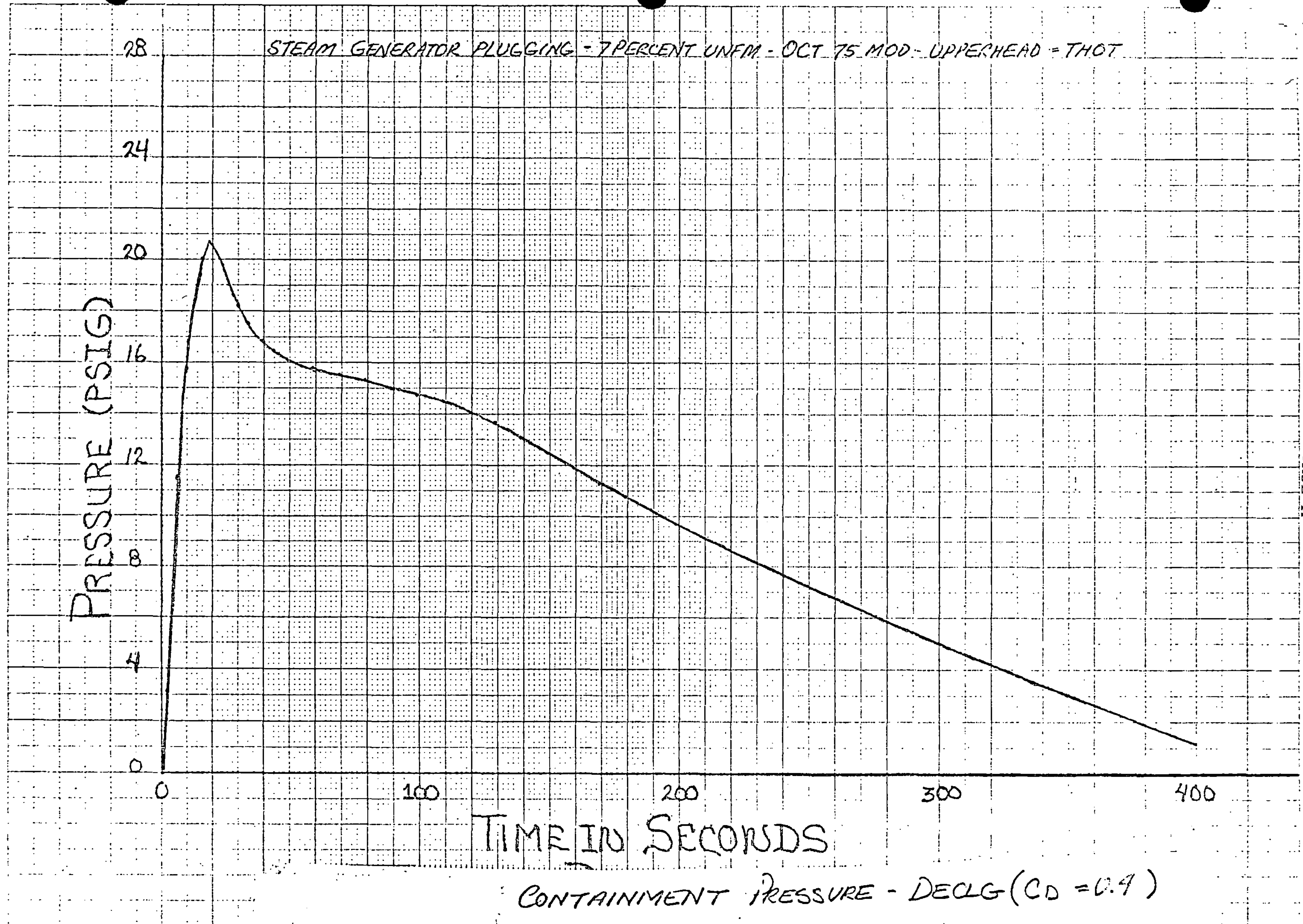
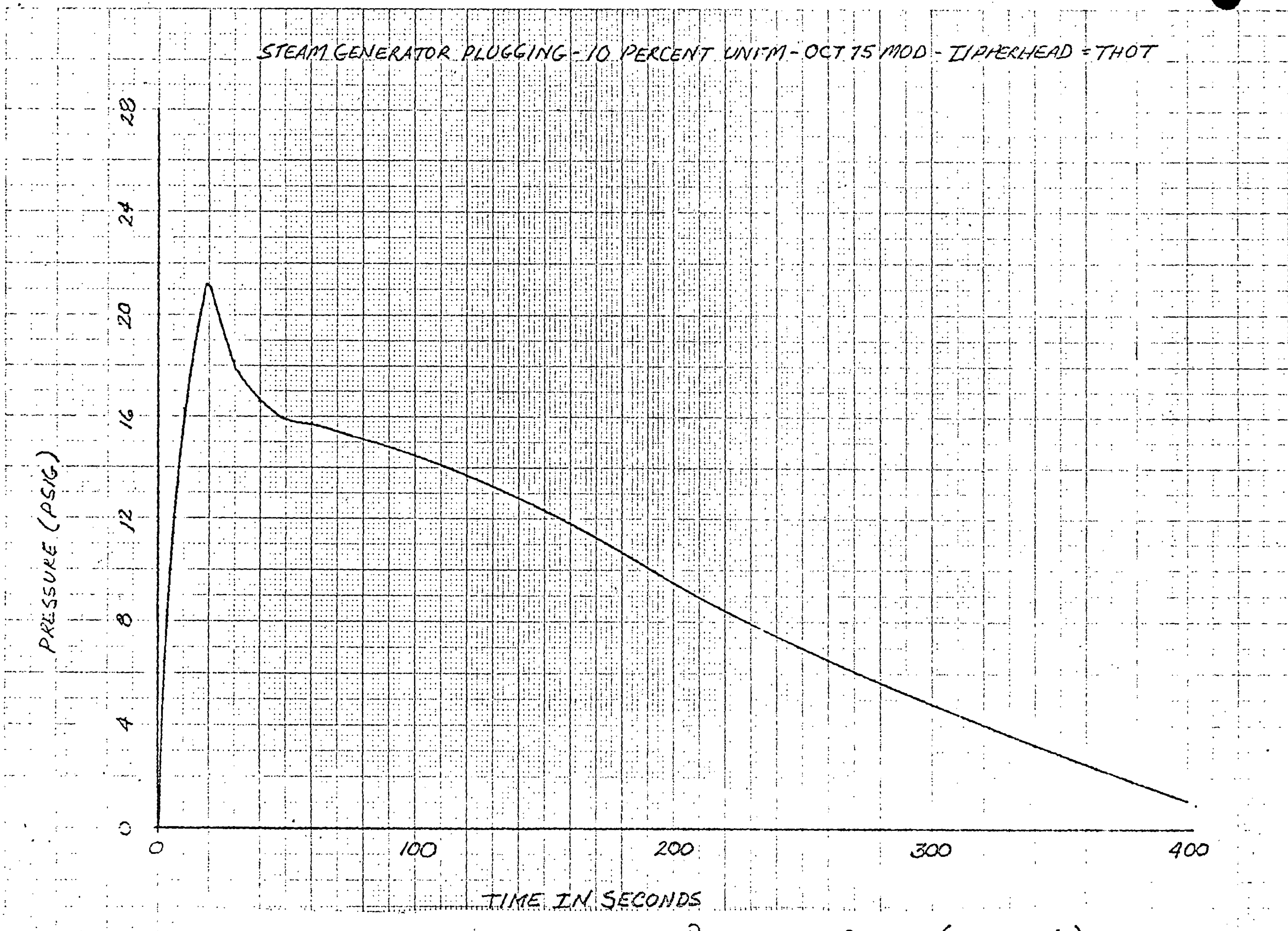


FIGURE 13c
CASE A

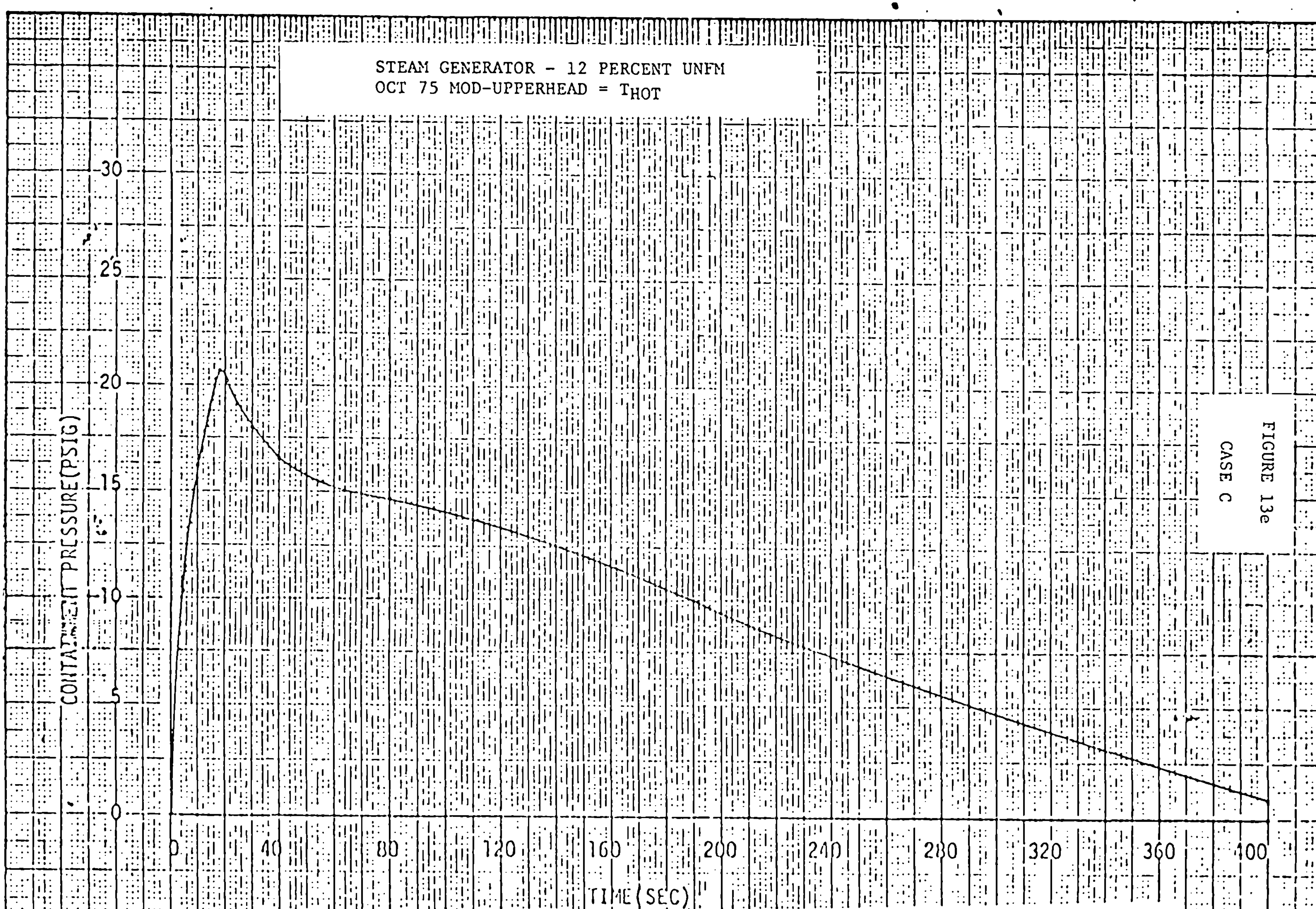
461510



CONTAINMENT PRESSURE - DECLG (C_D = 0.4)

FIGURE 13d

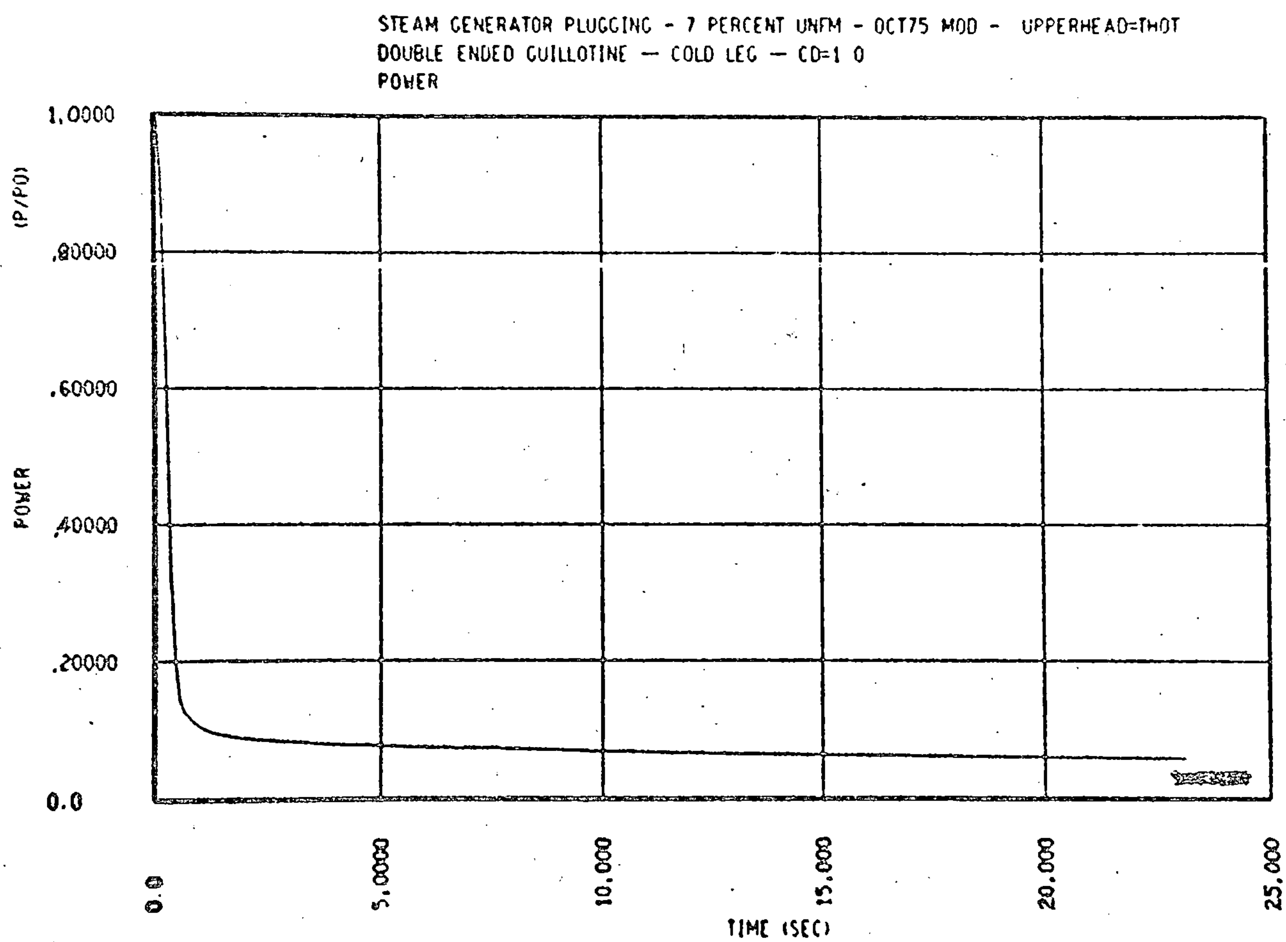
CASE B



CONTAINMENT PRESSURE - DECLG (C_D = 0.4)

FIGURE 14a

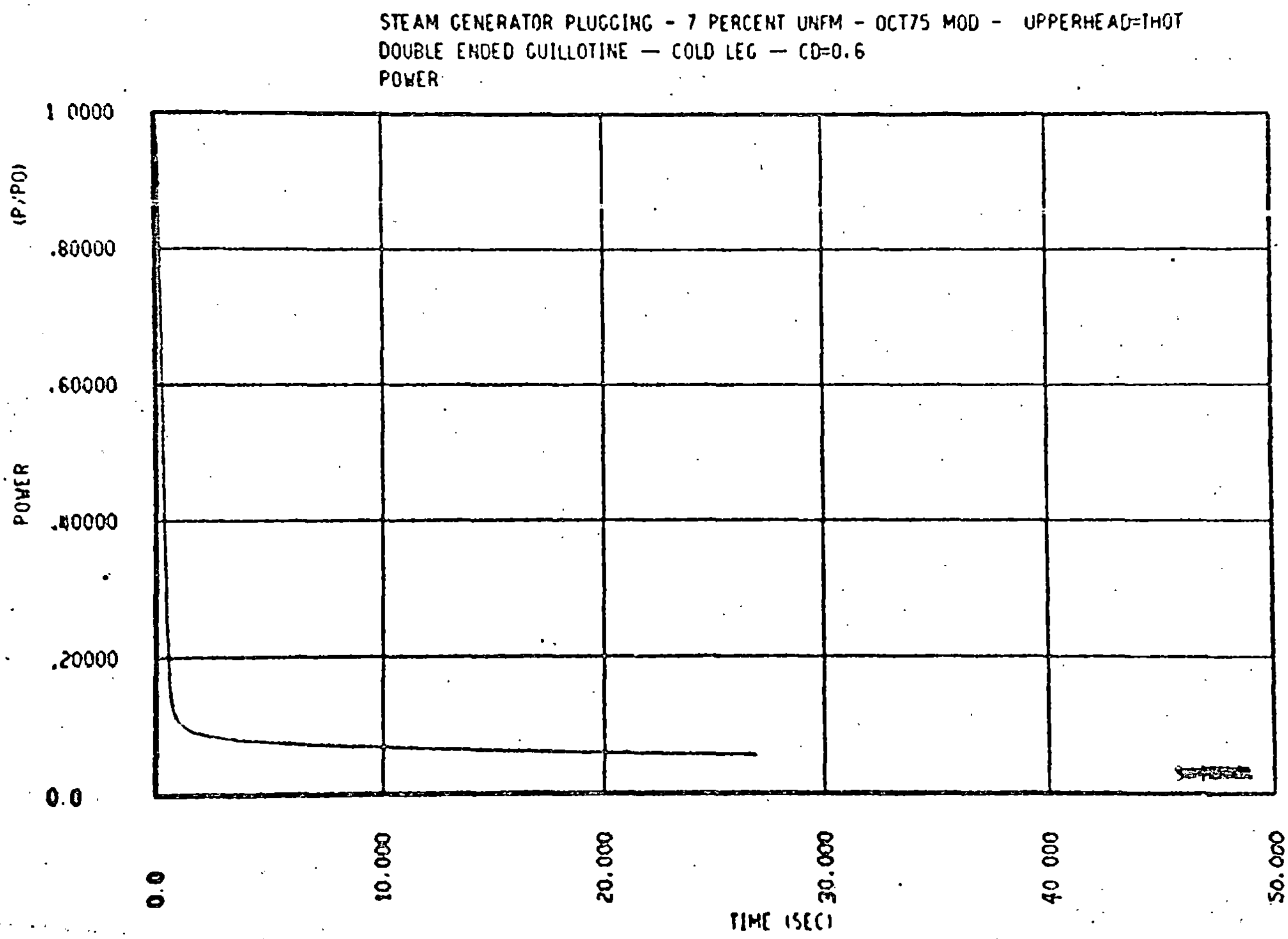
CASE A



CORE POWER TRANSIENT - DECLG (CD = 1.0)

FIGURE 14b

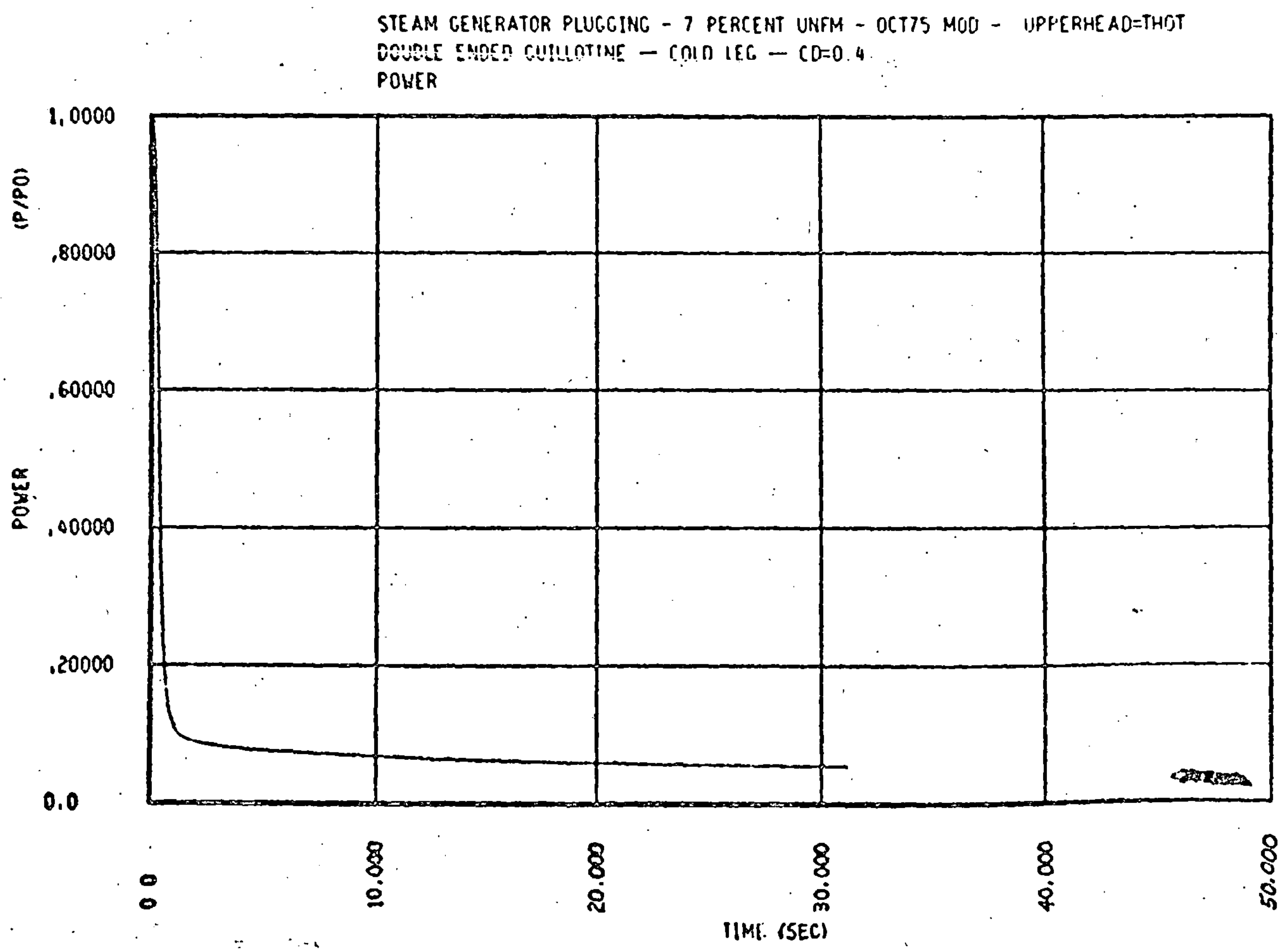
CASE A



CORE POWER TRANSIENT - DECLG (CD = 0.6)

FIGURE 14c

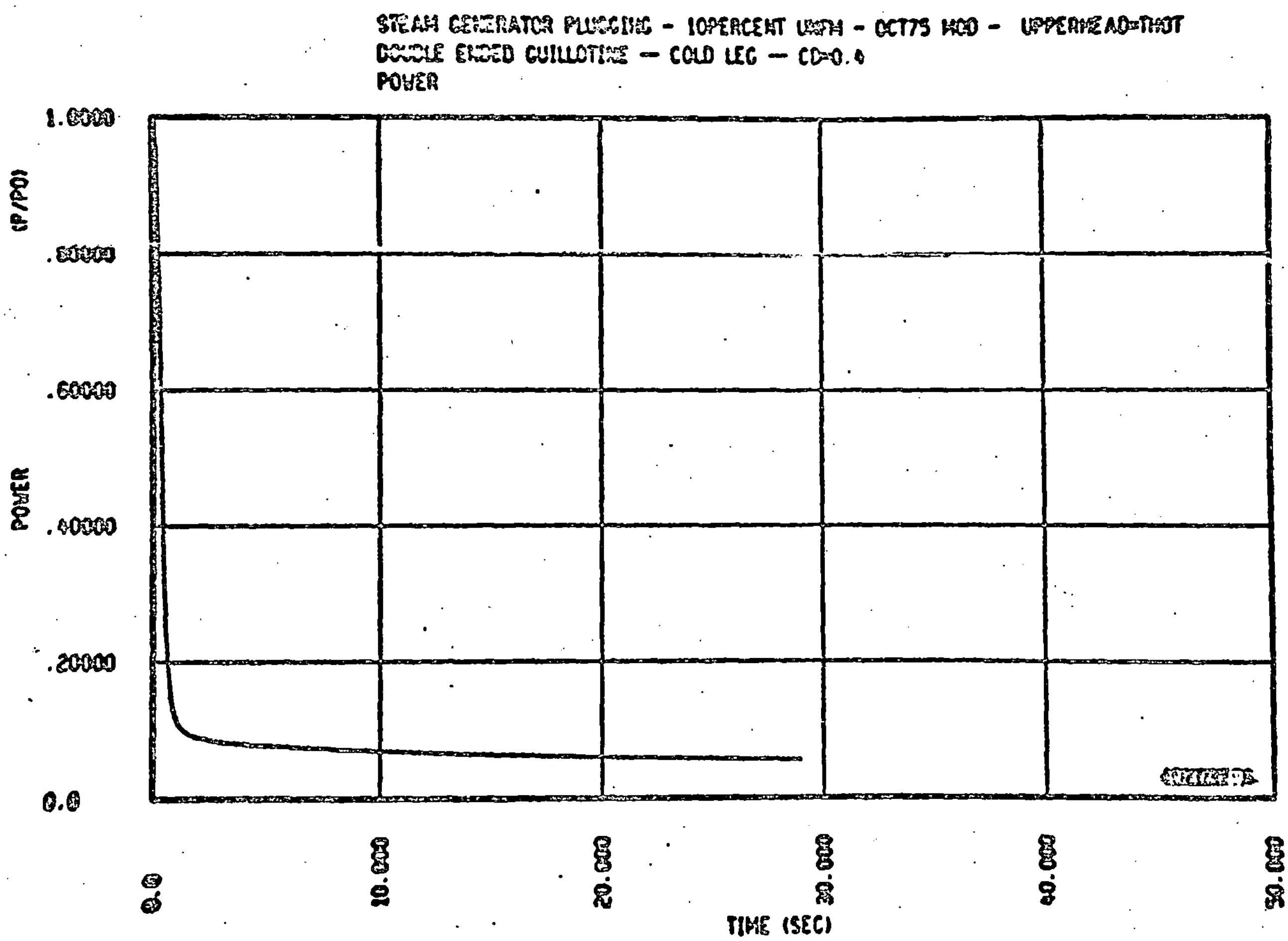
CASE A



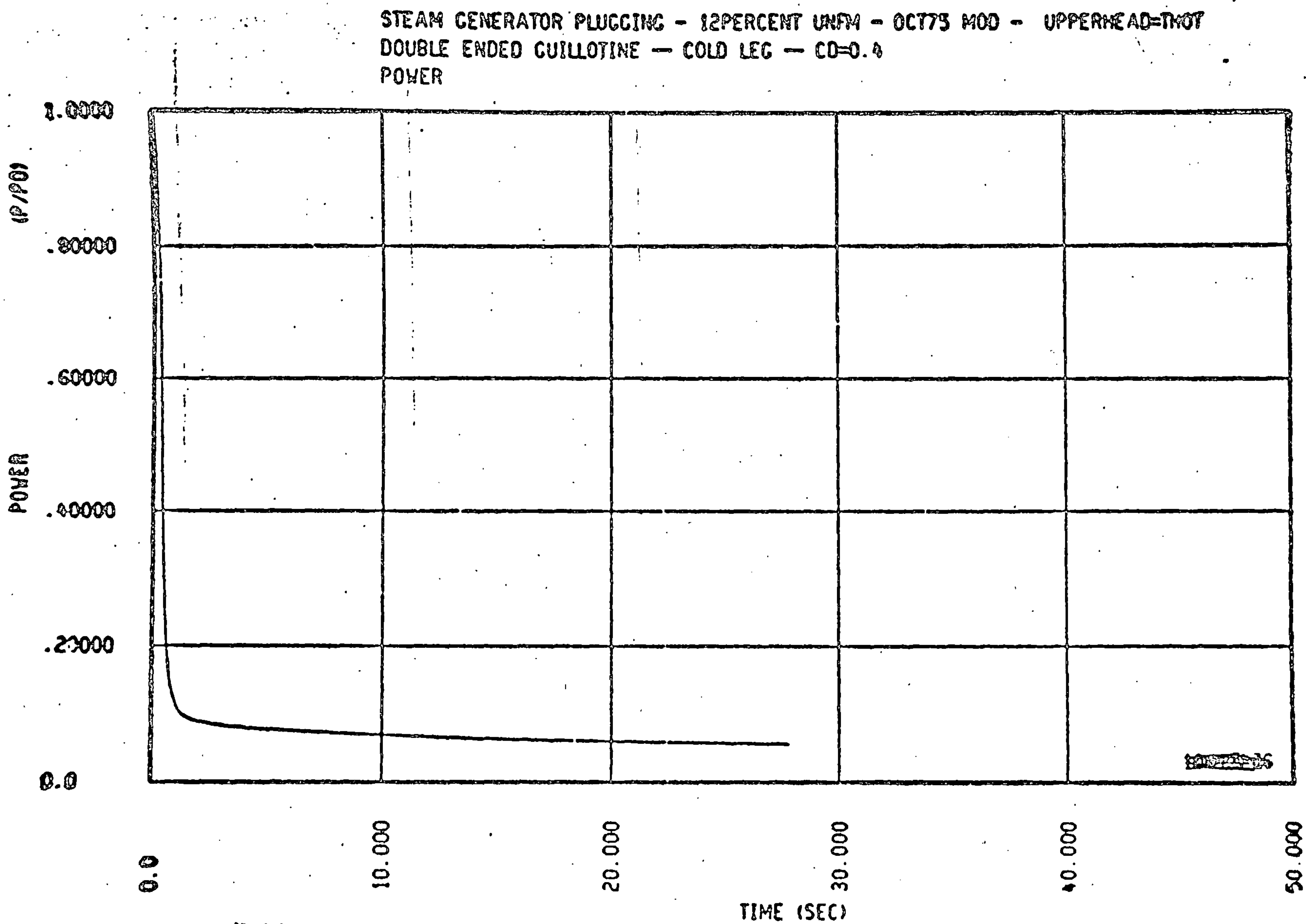
CORE POWER TRANSIENT - DECLG (CD = 0.4)

FIGURE 14d

CASE B



CORE POWER TRANSIENT - DECLG (CD = 0.4)



CORE POWER TRANSIENT - DECLG (CD = 0.4)

FIGURE 14e

CASE C

FIGURE 15a - CASE A

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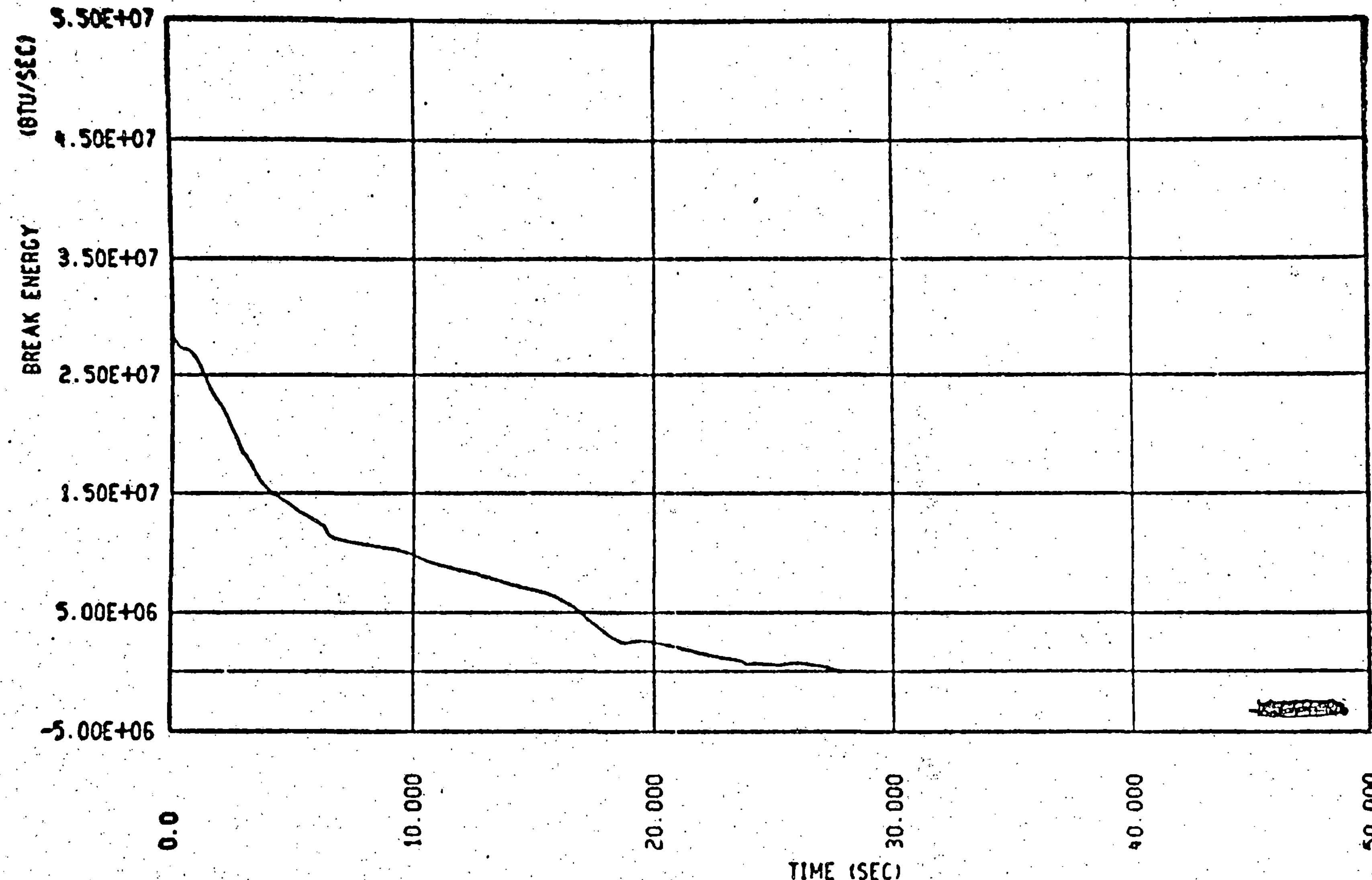
STEAM GENERATOR PLUGGING - 12PERCENT UNFM - OCT75 MOD - UPPERHEAD=THOT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD=0.4
BREAK ENERGY



BREAK ENERGY RELEASED TO CONTAINMENT

FIGURE 15c

CASE C

FIGURE 16a - CASE A

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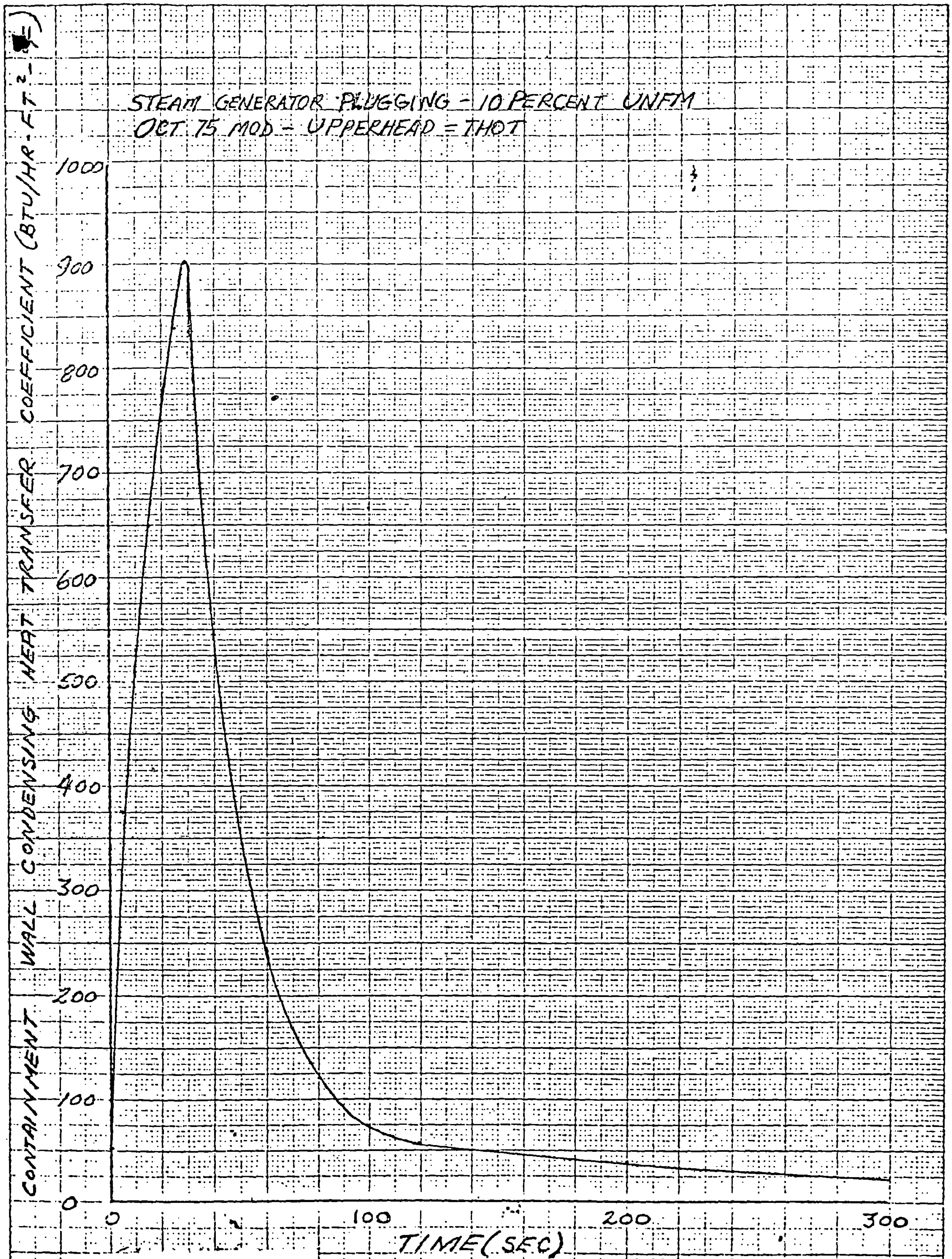
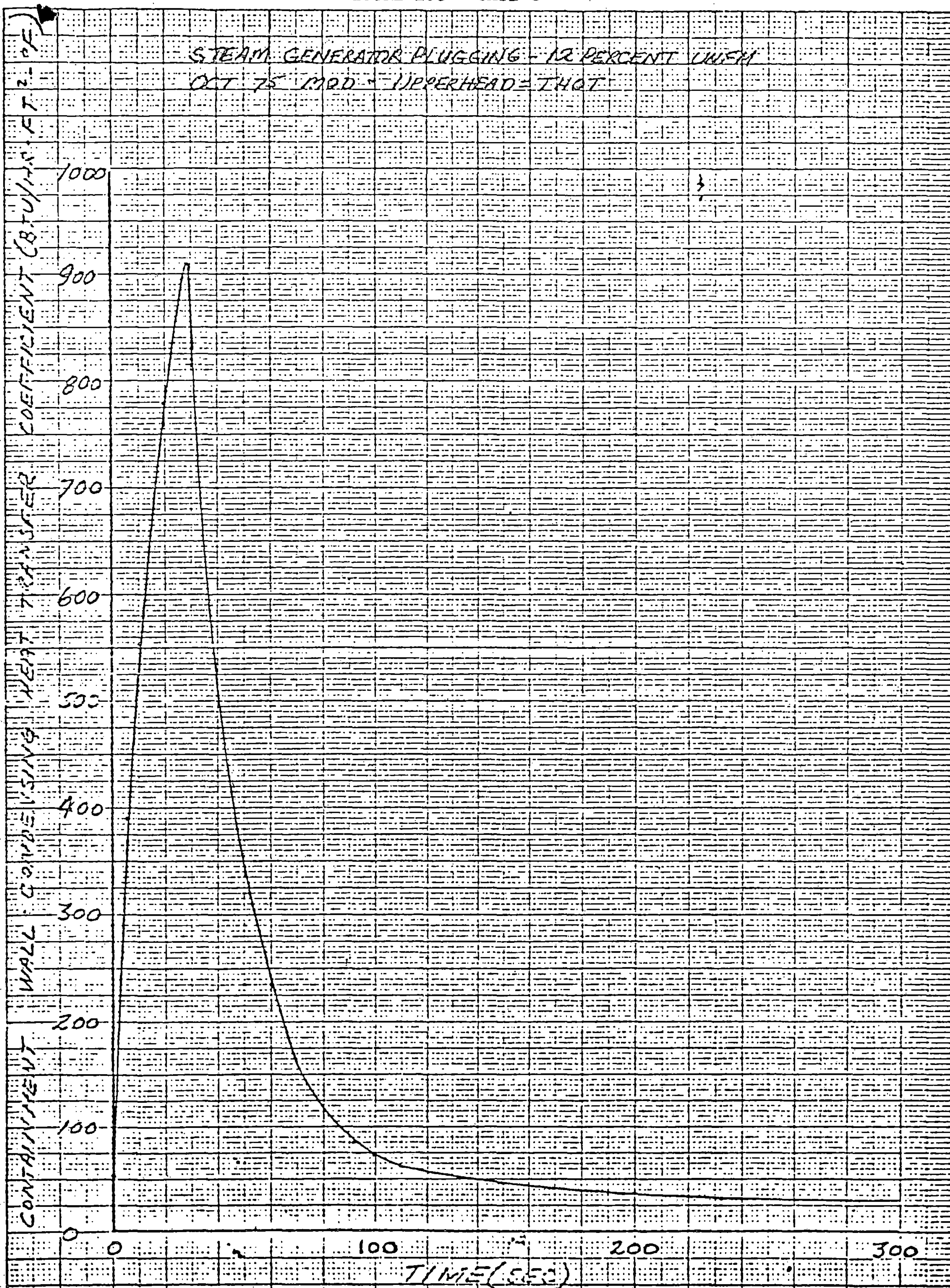
FIGURE 16b - CASE B



CONTAINMENT WALL CONDENSING
HEAT TRANSFER COEFFICIENT

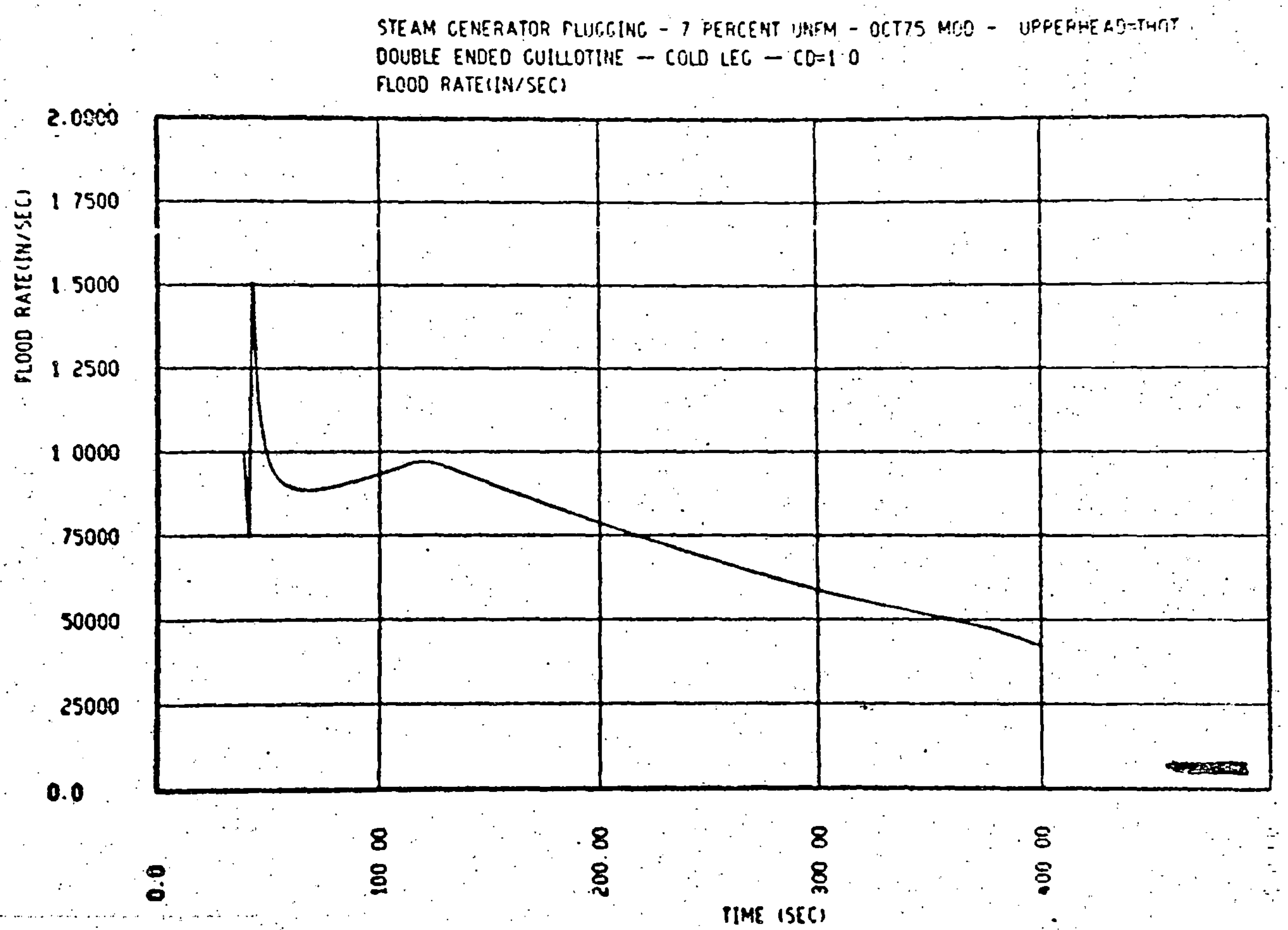
FIGURE 16c - CASE C



CONTAINMENT WALL CONDENSING

FIGURE 17a

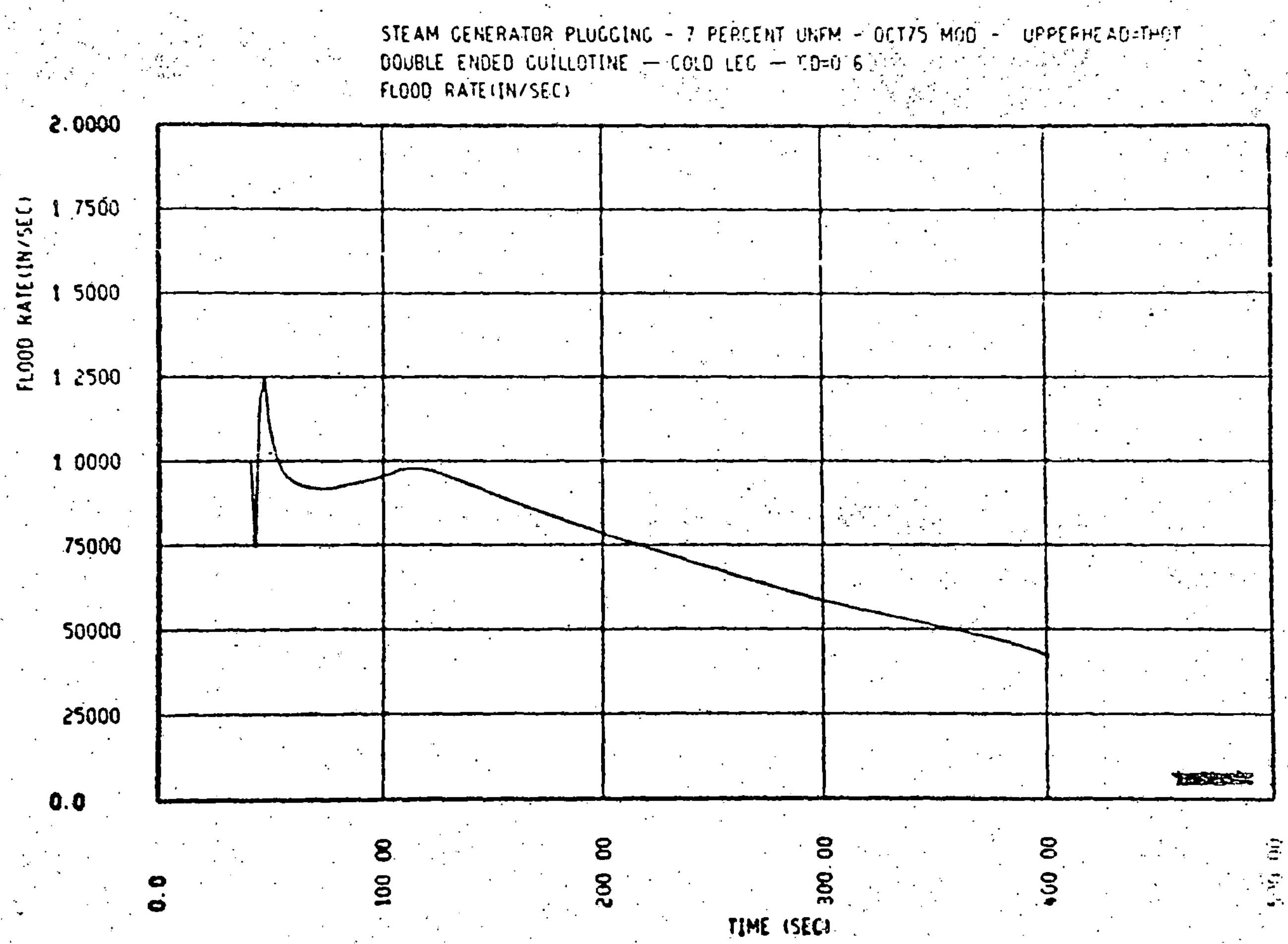
CASE A



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

FIGURE 17b

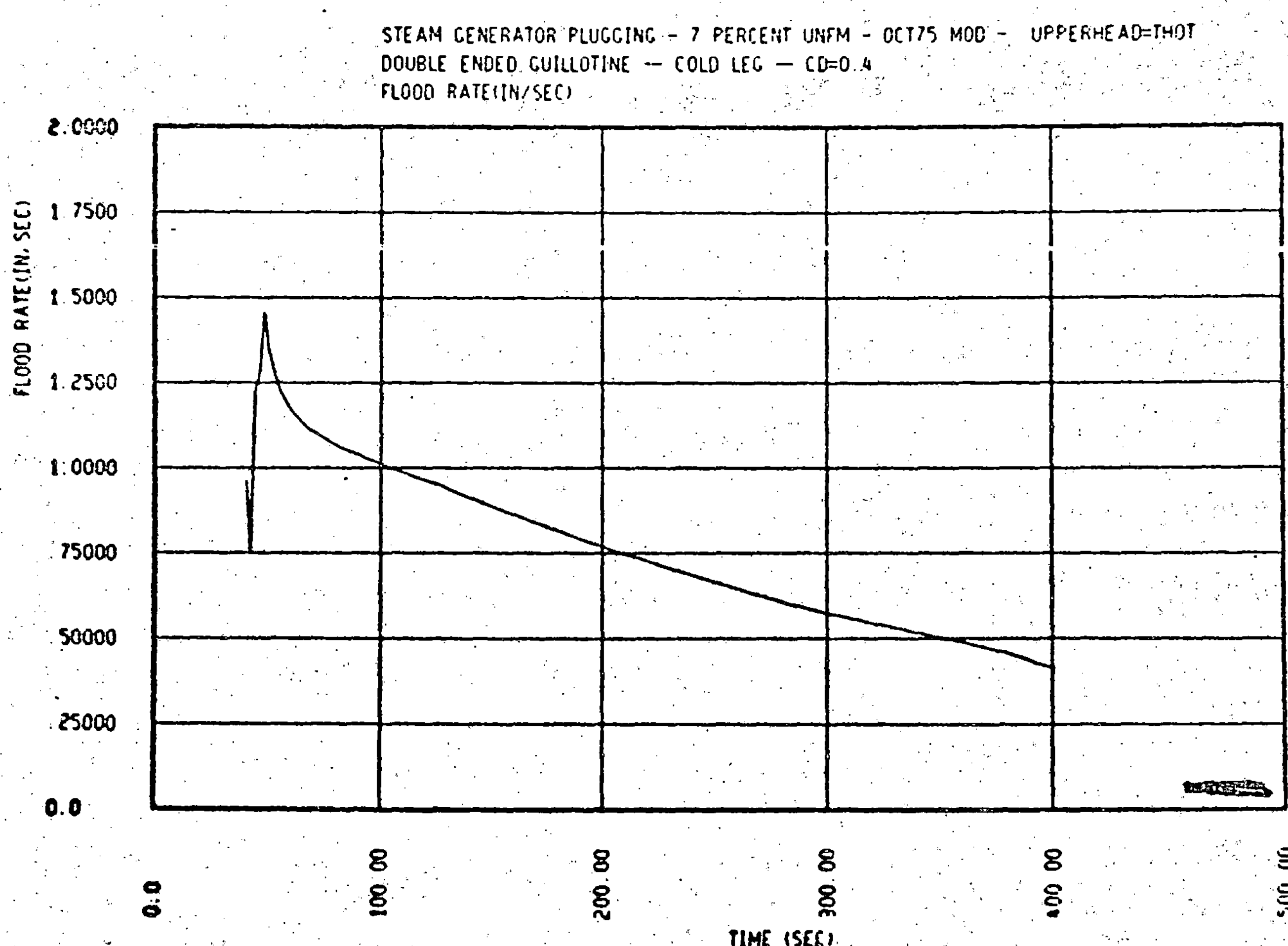
CASE A



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

FIGURE 17c

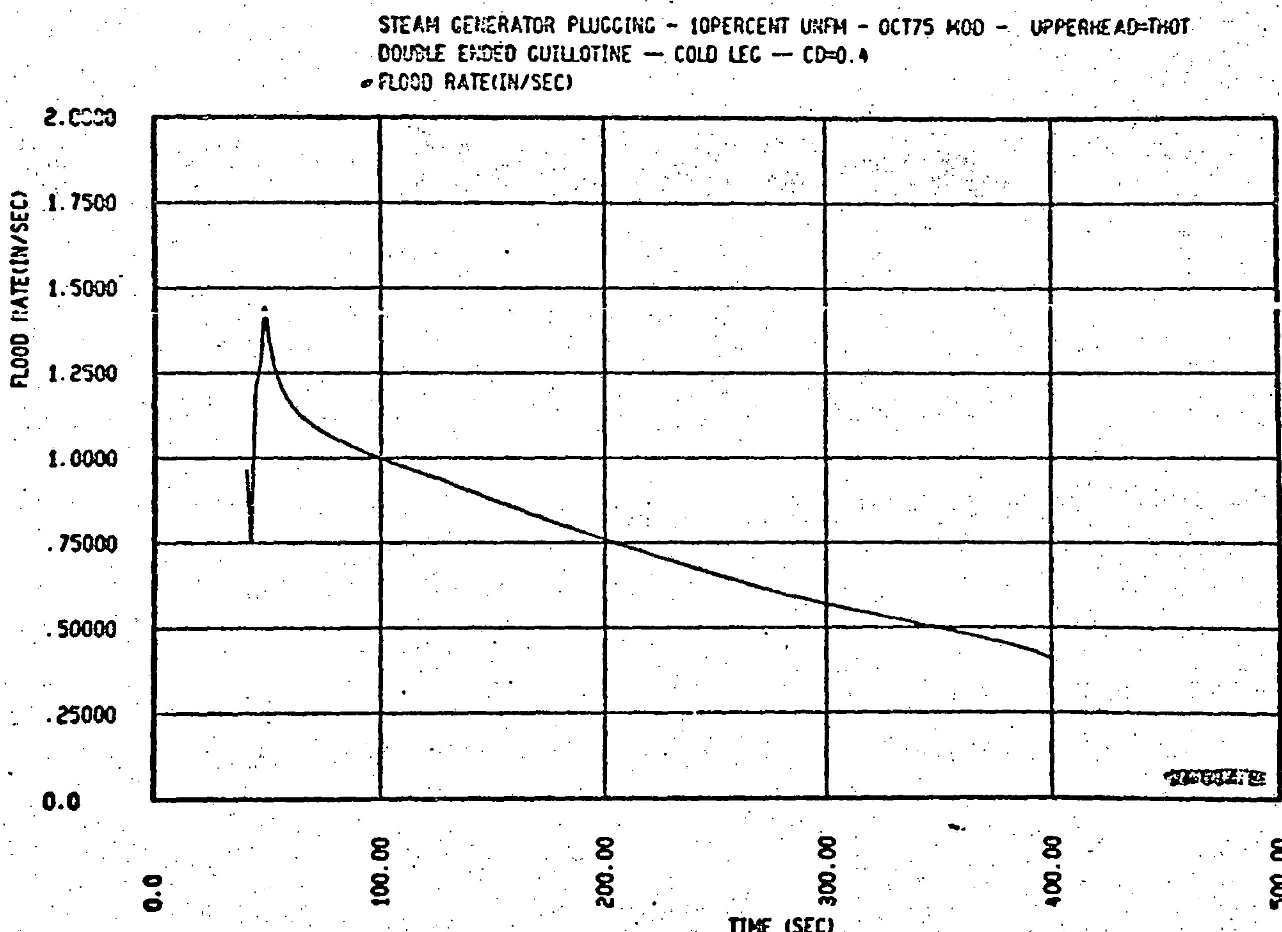
CASE A



REFLOOD TRANSIENT - DECLG (C_D = 0.4)
CORE INLET VELOCITY

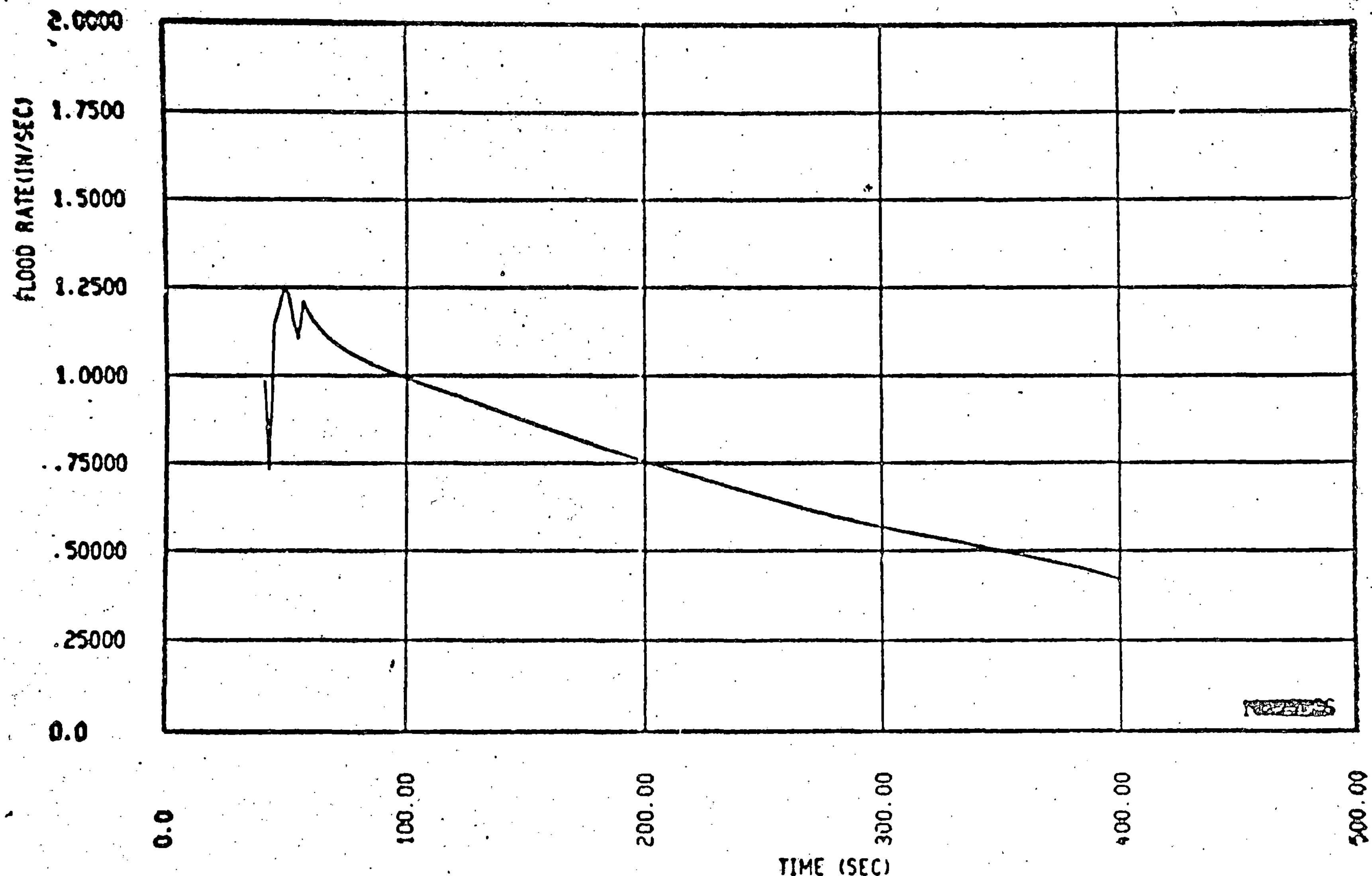
FIGURE 17d

CASE B



REFLOOD TRANSIENT - DECLG (CD=0.4)
CORE INLET VELOCITY

STEAM GENERATOR PLUGGING - 12PERCENT UNFM - OCT75 MOD - UPPERHEAD=THOT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD=0.4
FLOOD RATE(IN/SEC)



REFLOOD TRANSIENT - DECLG (C_D = 0.4)
CORE INLET VELOCITY

CASE C

FIGURE 17e

FIGURE 18

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Change No. 47 To The
Technical Specifications

Surry Power Station
Units No. 1 and 2

October 29, 1976

Attachment 2

4. The reactor thermal power level shall not exceed 118% of rated power.

B. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in TS Figures 2.1-1, 2.1-2 or 2.1-3; or the core thermal power exceeds 118% of the rated power.

C. The fuel residence time shall be limited to 21,348 effective full power hours(EFPH) for Cycle 4 of Unit 1 and to 6699 EFPH for Cycle 3 of Unit 2.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially than the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches 1.30 and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. The plant conditions required to violate these limits are precluded by the protection system and the self-actuated safety valves on the steam generator. Upper limits of 70% power for loop stop valves open and 75% with loop stop valves closed are shown to completely bound the area where clad integrity is assured. These latter limits are arbitrary but cannot be reached due to the Permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service.

Operation with natural circulation or with only one loop in service is not allowed since the plant is not designed for continuous operation with less than two loops in service.

TS Figures 2.1-1 through 2.1-3 are based on a $F^N_{\Delta H}$ of 1.55, a 1.55 cosine axial flux shape and a DNB analysis as described in Section 4.3 of the report Fuel Densification Surry Power Station, Unit 1 dated December 6, 1972 (including the effects of fuel densification). They are also valid for the following limit of the enthalpy rise hot channel factor: $F^N_{\Delta H} = 1.55\ (1 + 0.2\ (1-P)) \times T(BU)$ where P is fraction of rated power and T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies fully withdrawn to to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

The fuel residence time is limited to 21,348 EFPH for Cycle 4 of Unit 1 and to 6699 EFPH for Cycle 3 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the Regulatory Staff.

References

1) FSAR Section 3.4

2) FSAR Section 3.3

3) FSAR Section 14.2

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated cooling water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless the following conditions are met:

1. The refueling water tank contains not less than 350,000 gal. of borated water with a boron concentration of at least 2000 ppm.

2. Each accumulator system is pressurized to at least 600 psia and contains a minimum of 1075 ft^3 and a maximum of 1089 ft^3 of borated water with a boron concentration of at least 1950 ppm.

3. The boron injection tank and isolated portion of the inlet and outlet piping contains no less than 900 gallons of water with a boron

3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rod assemblies and power distribution limits.

Objective

To ensure core subcriticality after a reactor trip, a limit on potential reactivity insertions from hypothetical control rod assembly ejection, and an acceptable core power distribution during power operation.

Specification

A. Control Bank Insertion Limits

1. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the shutdown control rods shall be fully withdrawn.
2. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the full length control rod banks shall be inserted no further than the appropriate limit determined by core burnup shown on TS Figures 3.12-1A, 3.12-1B, 3.12-2, or 3.12-3 for three-loop operation and TS Figures 3.12-4A, 3.12-4B, 3.12-5, or 3.12-6 for two-loop operation.
3. The limits shown on TS Figures 3.12-1A through 3.12-6 may be revised on the basis of physics calculations and physics data obtained during unit startup and subsequent operation, in accordance with the following:
    a. The sequence of withdrawal of the controlling banks, when going from zero to 100% power, is A, B, C, D.
    b. An overlap of control banks, consistent with physics cal-

culations and physics data obtained during unit startup and subsequent operation, will be permitted.

c. The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown on TS Figure 3.12-7 under all steady-state operation conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions ($T_{avg} \geq 547^{o}F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

4. Whenever the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod position at which criticality would be achieved if the control rod assemblies were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

5. Operation with part length rods shall be restricted such that except during physics tests, the part length rod banks are withdrawn from the core at all times.

6. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in TS Figure 3.12-7 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod, expected to have the highest worth, inserted and part length rods fully withdrawn.

7.

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B. Power Distribution Limits

1. At all times except during low power physics tests and implementation of 3.12.B.2.b.(2), the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.00/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.00) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1 - P)) \times T(BU)$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, Z is the core height location of F_Q, and T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9.

2. Prior to exceeding 75% power following each core loading, and during each effective full power month of operation thereafter, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

   a. The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error, and the measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error. If either measured hot channel factor exceeds its limit specified under 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits $F_Q \leq 2.00 \times K(Z)$ and $F^N_{\Delta H} \leq 1.55 \times T(BU)$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similiarly reduced.

b. $F_Q(Z)$ shall be evaluated for normal (Condition I) operation of each unit by combining the measured values of $F_{xy}(Z)$ with the design Condition I axial peaking factor values, $F_Z(Z)$, as listed in TS Table 3.12-1A and TS Table 3.12-1B. For the purpose of this specification $F_{xy}(Z)$ shall be determined between 1.5 feet and 10.5 feet elevations of the core exclusive of grid strap locations. The measured values of $F_{xy}(Z)$ shall be increased by three percent to account for radial xenon redistribution effects associated with normal (Condition I) operation. (In addition, the value of $F_{xy}(Z)$ for Unit 1 shall be increased by two and one half percent to account for the predicted increase in the values of $F_{xy}(Z)$ during each effective full power month. This additional percent penalty on the values of $F_{xy}(Z)$ for Unit 1 shall be applicable up to 9000 MWD/MTU burnup.) The resulting $F_Q(Z)$ shall then be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error. If the results of this evaluation predict that $F_Q(Z)$ could potentially violate its limiting values as established in Specification 3.12.B.1, either:

(1) the thermal power and high neutron flux trip setpoint shall be reduced at least 1% for each 1% of the potential violation (for the purpose of this specification, this power level shall be called $P_{THRESHOLD}$), or

(2) movable detector surveillance shall be required for operation when the reactor thermal power exceeds $P_{THRESHOLD}$. This surveillance shall be performed in accordance with the following:

(a) The normalized power distribution, $F_Q(Z)\Big|^{j}_{APDM}$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux system for which $\overline{R}_j$, as defined in the Basis, has been determined. This shall be done immediately following and as a minimum at 30, 60, 90, 120, 240, and 480 minutes following the events listed below and every eight hours thereafter:

i. Raising the thermal power above $P_{THRESHOLD}$, or

ii. Movement of the control bank of rods more than an accumulated total of five steps in any one direction while reactor power is greater than $P_{THRESHOLD}$ except during control rod assembly exercises and excore detector calibrations.

(b) If $F_Q(Z)\Big|^{j}_{APDM}$ exceeds its limit, $F_Q(Z)$ as defined in 3.12.B.1, the reactor power shall be reduced until the limit is met or until thermal power is reduced to $P_{THRESHOLD}$.

3. The reference equilibrium indicated axial flux difference (called the target flux difference) at a given power level P_o, is that indicated axial flux difference with the core in equilibrium xenon conditions (small or no oscillation) and the control rods more than 190 steps withdrawn. The target flux difference at any other power level, P, is equal to the target value of P multiplied by the ratio, P/P_o. The target flux difference shall be measured at least once per equivalent full power quarter. The target flux difference must be updated during each effective full power month of operation either by actual measurement, or by linear interpolation using the most recent value and the value predicted for the end of the cycle life.

4. Except during physics tests, during excore detector calibration and except as modified by 3.12.B.4.a, b, or c below, the indicated axial flux difference shall be maintained within a +6 to -9% band about the target flux difference (defines the target band on axial flux difference).

   a. At a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band, or the reactor power shall immediately be reduced to a level no greater than 90 percent of rated power.

   b. At a power level no greater than 90 percent of rated power,

      (1) The indicated axial flux difference may deviate from its +6 to -9% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded

by -18 percent and +11.5 percent at 90% power. (One half of the time the indicated axial flux difference is out of its target band at power levels up to 50 percent of rated power is to be counted as contributed to the one hour cumulative maximum the flux difference deviates from its target band at a power level less than or equal 90 percent of rated power.) For every 4 percent below 90% power, the permissible positive flux difference boundary is extended by 1 percent. For every 5 percent below 90% power, the permissible negative flux difference boundary is extended by 2 percent.

(2) If 3.12.B.4.b(1) is violated then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint shall be reduced to no greater than 55% power.

(3) A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

c. At a power level no greater than 50 percent of rated power,

(1) The indicated axial flux difference may deviate from its target band.

(2) A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24 hour period in which the power level is no greater than 50 percent of rated power.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in 3.12.B.4.a and the flux difference time limits in 3.12.B.4.b. If the alarms are out of service temporarily, the axial flux difference shall be logged, and conformance to the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

5. The allowable quadrant to average power tilt is 2.0%.

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6. If, except for physics and rod exercise testing, the quadrant to average power tilt exceeds 2%, then:

    a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the specification of 3.12.B.1, or

    b. If the hot channel factors are not determined within two hours, the power level and high neutron flux trip setpoint shall be reduced from rated power, 2% for each percent of quadrant tilt.

    c. If the quadrant to average power tilt exceeds ±10%, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

7. If, except for physics and rod exercise testing, after a further period of 24 hours, the power tilt in 3.12.B.5 above is not corrected to less than 2%:

   a. If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as a reportable occurrence to the Nuclear Regulatory Commission.

   b. If the design hot channel factors for rated power are exceeded and the power is greater than 10%, the Nuclear Regulatory Commission shall be notified and the Nuclear Overpower, Overpower ΔT and Overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factor exceeds the rated power design values.

   c. If the hot channel factors are not determined the Nuclear Regulatory Commission shall be notified and the Overpower ΔT and Overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

C. Inoperable Control Rods

1. A control rod assembly shall be considered inoperable if the assembly cannot be moved by the drive mechanism, or the assembly remains misaligned from its bank by more than 15 inches. A full-length control rod shall be considered inoperable if its rod drop time is greater than 1.8 seconds to dashpot entry.

2. No more than one inoperable control rod assembly shall be permitted when the reactor is critical.

3. If more than one control rod assembly in a given bank is out of service because of a single failure external to the individual rod drive mechanisms, i.e. programming circuitry, the provisions

of 3.12.C.1 and 3.12.C.2 shall not apply and the reactor may remain critical for a period not to exceed two hours provided immediate attention is directed toward making the necessary repairs. In the event the affected assemblies cannot be returned to service within this specified period the reactor will be brought to hot shutdown conditions.

4. The provisions of 3.12.C.1 and 3.12.C.2 shall not apply during physics tests in which the assemblies are intentionally misaligned.
5. If an inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in TS Figure 3.12-2 apply.
6. If an inoperable full-length rod cannot be located, or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in TS Figure 3.12-3 apply.
7. No insertion limit changes are required by an inoperable part-length rod.
8. If a full-length rod becomes inoperable and reactor operation is continued the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the unit power level shall be reduced to an

analytically determined part power level which is consistent with the safety analysis.

D. If the reactor is operating above 75% of rated power with one excore nuclear channel out of service, the core quadrant power balance shall be determined.

1. Once per day, and
2. After a change in power level greater than 10% or more than 30 inches of control rod motion.

The core quadrant power balance shall be determined by one of the following methods:

1. Movable detectors (at least two per quadrant)
2. Core exit thermocouples (at least four per quadrant)

E. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service then:
   a. For operation between 50% and 100% of rated power, the position of the RCC shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to motion, of the non-indicating rod, exceeding 24 steps, whichever occurs first.
   b. During operation below 50% of rated power no special monitoring is required.
2. Not more than one rod position indicator (RPI) channel per group nor two RPI channels per bank shall be permitted to be inoperable at any time.

F. Misaligned or Dropped Control Rod

1. If the Rod Position Indicator Channel is functional and the associated part length or full length control rod is more than

15 inches out of alignment with its bank and cannot be realigned, then unless the hot channel factors are shown to be within design limits as specified in Section 3.12.B.1 within 8 hours, power shall be reduced so as not to exceed 75% of permitted power.

2. To increase power above 75% of rated power with a part-length or full length control rod more than 15 inches out of alignment with its bank an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 3.12.B.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated for by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition.

The control rod assembly insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical assembly ejection, and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in the analysis of the hypothetical steam break accident. The rod insertion limits are based on end of core life conditions. Early in core life, less shutdown margin is required, and TS Figure 3.12-7 shows the shutdown margin equivalent to 1.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

Relative positions of control rod banks are determined by a specified control rod bank overlap. This overlap is based on the consideration of axial power shape control.

The specified control rod insertion limits have been revised to limit the potential ejected rod worth in order to account for the effects of fuel densification.

The various control rod assemblies (shutdown banks, control banks A, B, C, and D and part-length rods) are each to be moved as a bank, that is, with all assemblies in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks and a linear position indicator, Linear Variable Differential Transformer, which indicates the actual assembly position. The position indication accuracy of the Linear Differential Transformer is approximately ±5% of span (±7.5 inches) under steady state conditions. The relative accuracy of the linear position indicator is such that, with the most adverse errors, an alarm is actuated if any two assemblies within a bank deviate by more than 14 inches. In the event that the linear position indicator is not in service, the effects of malpositioned control rod assemblies are observable from nuclear and process information displayed in the Main Control Room and by core thermocouples and in-core movable detectors. Below 50% power, no special monitoring is required for malpositioned control rod assemblies with inoperable rod position indicators because, even with an unnoticed complete assembly misalignment (part-length or full length control rod assembly 12 feet out of alignment with its bank) operation at 50% steady state power does not result in exceeding core limits.

The specified control rod assembly drop time is consistent with safety analyses that have been performed.

An inoperable control rod assembly imposes additional demands on the operators. The permissible number of inoperable control rod assmeblies is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable control rod assemblies upon reactor trip.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature and cladding mechanical properties. First, the peak value of linear power density must not exceed 21.1 kw/ft for Unit 1 and 20.4 kw/ft for Unit 2. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above, the peak linear power density must not exceed the limiting kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F on peak clad temperature. This is required to meet the initial conditions assumed for the loss of coolant accident. To aid in specifying the limits on power distribution the following hot channel factors are defined.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46 using an upper bound envelope of 2.00 times the hot channel factor normalized operating envelope of TS Figure 3.12-8.

When an F_Q measurement is taken, both experiemental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerances.

In the specified limit of $F^N_{\Delta H}$ there is an eight percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55(1 + 0.2(1-P)) \times T(BU)/1.08$ where T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for the F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system.

Measurement of the hot channel factors are required as part of startup physics tests, during each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor, $F_{\Delta H}^{N}$, limit will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differeing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.
2. Control rod banks are sequenced with overlapping banks as shown in TS Figures 3.12-1A, 3.12-1B, and 3.12-2.
3. The full length and part length control bank insertion limits are not violated.

DELETED

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F_{\Delta H}^{N}$ with decreasing power level allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, this hot channel factor limit is met.

A recent evaluation of DNB test data from experiments of fuel rod bowing in thimble cells has identified that it is appropriate to impose a penalty factor to the accident analyses DNBR results. This evaluation has not been completed, but in order to assure that this effect is accommodated in a conservative manner, an interim thimble cell rod bow penalty for 15 x 15 fuel, T(BU), is applied to the measured values of the enthalpy rise hot channel factor, $F^N_{\Delta N}$. It is anticipated that the values of this penalty will change after the evaluation of the test data has been completed.

DELETED

For normal (Condition I) operation, it may be necessary to perform surveillance to insure that the heat flux hot channel factor, $F_Q(Z)$, limit is met. To determine whether and at what power level surveillance is required, the potential (Condition I) values of $F_Q(Z)$ shall be evaluated monthly by combining the values of $F_{xy}(Z)$ obtained from the analysis of the monthly incore flux map with the values of the design Condition I axial peaking factors, $F_Z(Z)$. The product of these shall be increased by five percent to account for measurement uncertainty, three percent to account for manufacturing tolerances, three percent to account for the effects of the radial redistribution of xenon during normal (Condition I) operation, and for Unit 1, two and one half percent to account for the increase in the value of $F_{xy}(Z)$ as a function of burnup out to 9000 MWD/MTU burnup. $P_{THRESHOLD}$ is defined as the value of rated power minus one percent power for each percent of potential $F_Q(Z)$ violation. If the potential values of $F_Q(Z)$ for normal (Condition I) operation are greater than the $F_Q(Z)$ limit, then surveillance shall be performed at all power levels above $P_{THRESHOLD}$.

Movable incore instrumentation thimbles for surveillance are selected so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor $F_Q(Z)$ can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor, $\overline{R}$, can be determined as the ratio of the total peaking factor result from a full core flux map and the axial peaking factor in a selected thimble. Based on this approach, the operational value of the heat flux hot channel factor, $F_Q(Z)\Big|^{j}_{APDM}$ is derived as follows:

$$F_Q(Z)\Big|^{j}_{APDM} = F_j(Z)\ (\overline{R}_j)\ (1.03)\ (1 + \sigma_j)\ (1.07)$$

where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

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b. $F_Q(Z)\Big|^{j}_{APDM}$ is the operational value of the heat flux hot channel factor for the purpose of this surveillance.

c. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns for power levels for which this surveillance is required.

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij} .$$

where

$$R_{ij} = \frac{F_{Q_i}^{meas}}{(F_{ij}(Z))_{max}}$$

and $F_{ij}(Z)$ is the normalized axial power distribution from thimble at elevation Z in map i which had a measured peaking factor without uncertainties of densification allowance of $F_{Q_i}^{meas}$.

The full incore flux map used to update $\overline{R}_j$ shall be taken at least per every regular effective full power month. The continued accuracy and representativeness of the selected thimbles shall be verified by using the latest flux maps to update the $\overline{R}_j$ for each representative thimble.

e. σ_j is the standard deviation of $\overline{R}_j$ and is derived from n flux maps covering the full configuration of permissable rod patterns for power levels for which surveillance is required using the relationship below, or 0.02, whichever is greater:

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} (\overline{R}_j - R_{ij})^2\right]^{\frac{1}{2}}}{\overline{R}_j}$$

f. The factor 1.03 reduction in the (kw/ft) limit is the engineering uncertainty factor.

g. The factor 1.07 is the combined uncertainty associated with the measurement of F_{Q_i} and $F_{ij}(Z)_{max}$.

The procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically control of flux difference is required to limit the difference between the current value of flux difference (ΔI) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup, but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control given in 3.12.B.4 together with the surveillance requirements given in 3.12.B.2.b assure that the Limiting Condition for Operation for the heat flux hot channel factor is met.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of +6 to -9% ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measureing the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore detector calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore detector calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 90% power, in the range +14.5 to -21 percent (+11.5 percent to -18 percent indicated) where for every 4 percent below rated power, the permissible positive flux difference boundary is extended by 1 percent, and for every 5 percent below rated power, the permissible negative flux difference boundary is extended by 2 percent.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference.

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A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of intensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rods and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

SURRY UNIT 1

CYCLE 4

CORE HEIGHT (Feet)	$F_Z(Z)$
1.5	1.318
2.0	1.318
2.5	1.309
3.0	1.362
3.5	1.391
4.0	1.408
4.5	1.416
5.0	1.415
5.5	1.401
6.0	1.375
6.5	1.336
7.0	1.300
7.5	1.274
8.0	1.240
8.5	1.212
9.0	1.218
9.5	1.258
10.0	1.269
10.5	1.231

TABLE 3.12-1A: DESIGN CONDITION I AXIAL PEAKING FACTORS, $F_Z(Z)$ VS. CORE HEIGHT FOR SURRY UNIT I

SURRY UNIT 2

CYCLE 3

CORE HEIGHT (Feet)	$F_Z(Z)$
1.5	1.334
2.0	1.308
2.5	1.270
3.0	1.218
3.5	1.192
4.0	1.224
4.5	1.240
5.0	1.253
5.5	1.256
6.0	1.266
6.5	1.285
7.0	1.272
7.5	1.290
8.0	1.295
8.5	1.302
9.0	1.289
9.5	1.272
10.0	1.228
10.5	1.244

TABLE 3.12-1B: DESIGN CONDITION I AXIAL PEAKING FACTORS, $F_Z(Z)$ VS. CORE HEIGHT FOR SURRY 2

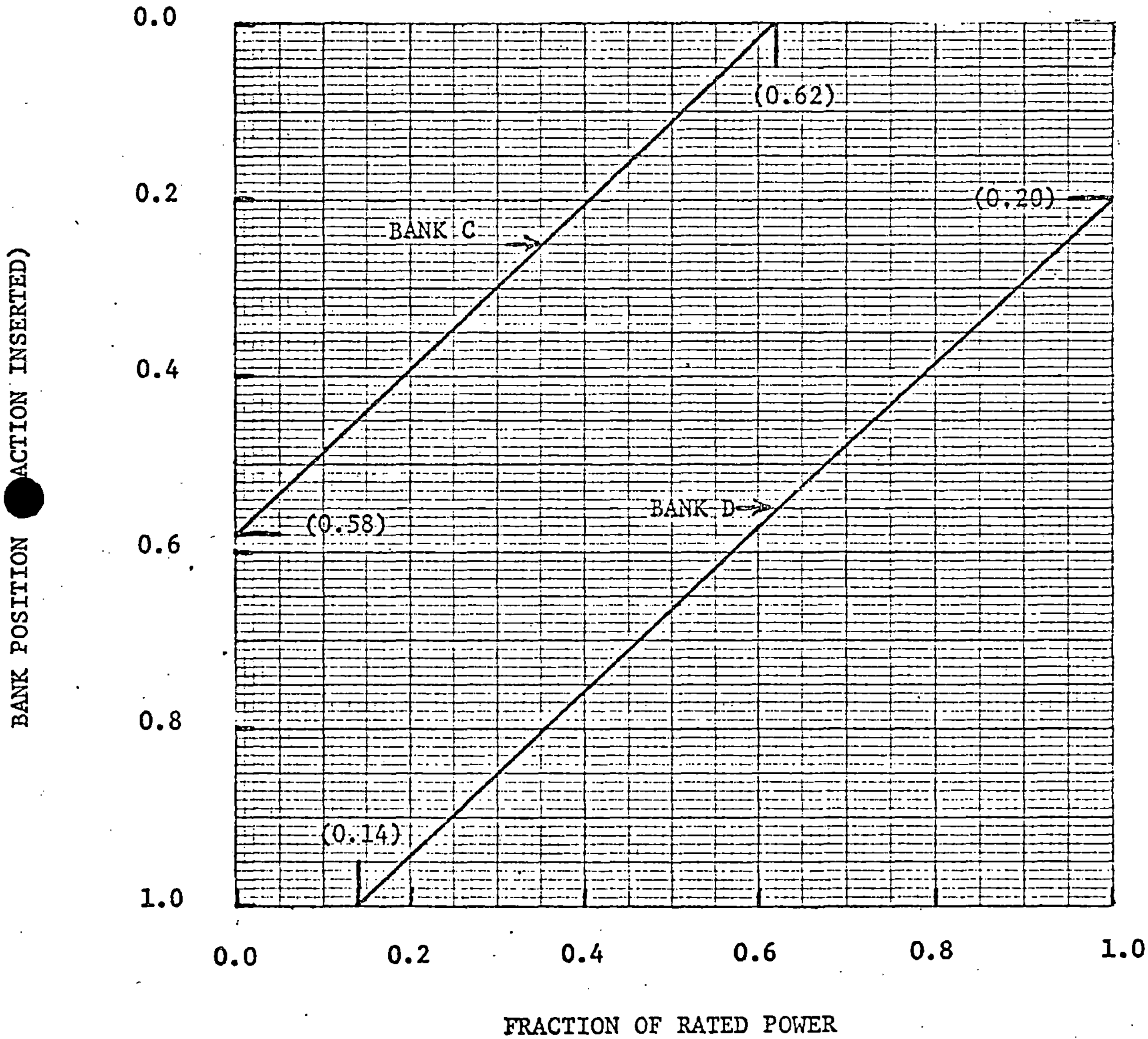
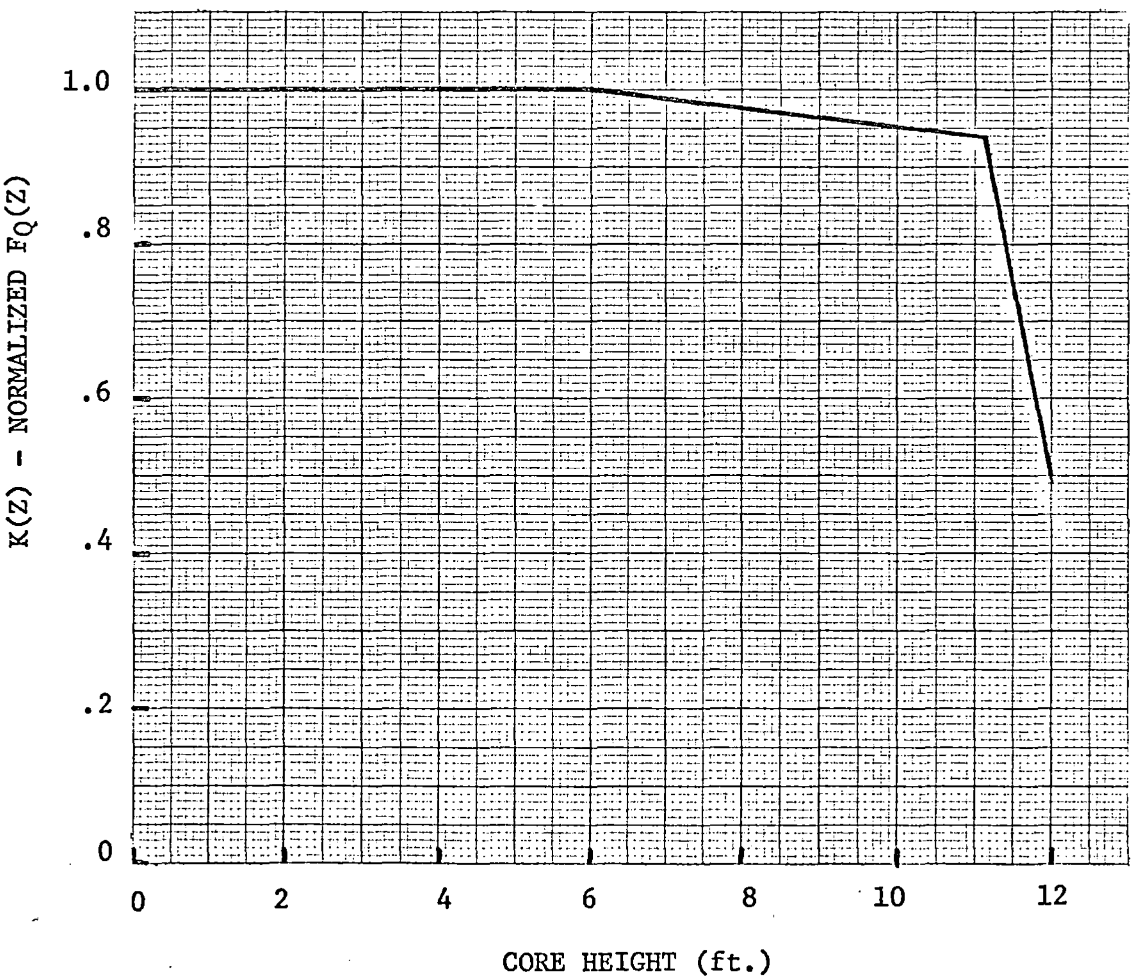


FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

HOT CHANNEL FACTOR NORMALIZED
OPERATING ENVELOPE

SURRY POWER STATION
UNIT NOS. 1 AND 2



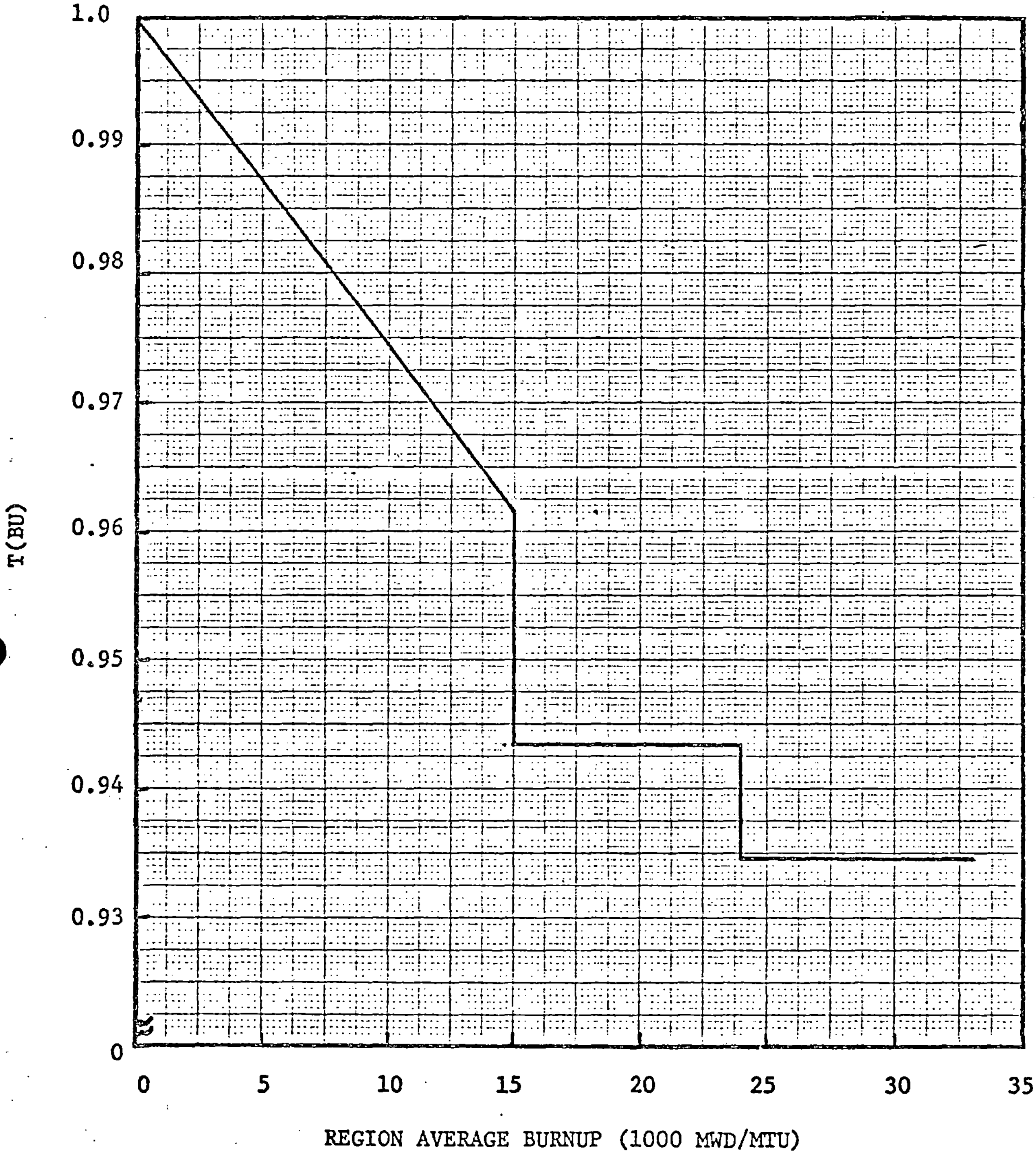


FIGURE 3.12-9 INTERIM THIMBLE CELL ROD BOW PENALTY ON $F^N_{\Delta H}$
SURRY UNITS NO. 1 AND 2

attack of the pipe. In order to insure the continued integrity of the pipe throughout the plant life, the affected lines are flushed periodically to remove stagnant water which may contain contaminants.

The flushing requirements delineated in TS Table 4.1-3A and TS Table 4.1-3B as appropriate for Unit No. 1 and Unit No. 2 respectively, ensure that a build up of contaminants will not occur. The specified minimum flush durations, with expected flow rates during flushing, insures that a volume of water greater than the volume contained in the stagnant flow paths listed in Table 4.1-3A and 4.1-3B will be flushed. The required sampling of flushed lines further ensures that the specified flushing procedures were effective in removing any undesirable contaminants that may have accumulated in the sensitized piping.

The control room ventilation system is required to establish a positive differential pressure in the control room for one hour following a design basis loss-of-coolant accident using a bottled air supply as the source of air. The ability of the system to meet this requirement is tested by pressurizing the control room using the ventilation system fans and comparing the volume of air required to that stored. The test is conducted each refueling interval normally coinciding with the refueling outage of Unit 1.

TABLE 4.1-2A

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	Description	Test	Frequency	FSAR Section Reference
1.	Control Rod Assemblies	Rod drop times of all full length rods at hot and cold conditions.	Each refueling shutdown or after disassembly or maintenace requiring the breech of the Reactor Coolant System integrity.	7
2.	Control Rod Assemblies	Partial movement of all rods	Every 2 weeks	7
3.	Refueling Water Chemical Addition Tank	Functional	Each refueling shutdown	6
4.	Pressurizer Safety Valves	Setpoint	Each refueling shutdown	4
5.	Main Steam Safety Valves	Setpoint	Each refueling shutdown	10
6.	Containment Isolation Trip	*Functional	Each refueling shutdown	5
7.	Refueling System Interlocks	*Functional	Prior to refueling	9.12
8.	Service Water System	*Functional	Each refueling shutdown	9.9
9.	Fire Protection Pump and Power Supply	*Functional	Monthly	9.10
10.	Primary System Leakage	*Evaluate	Daily	4
11.	Diesel Fuel Supply	*Fuel Inventory	5 days/week	8.5
12.	Boric Acid Piping Heat Tracing Circuits	*Operational	Monthly	9.1
13.	Main Steam Line Trip	Functional (1) Full closure (2) Partial closure	(1) Each cold shutdown (2) Before each startup but at least quarterly	10
14.	Service Water System Values in Line Supplying Recirculation Spray Heat Exchangers	Functional	Each refueling	9.9
15.	Control Room Ventilation System	*Ability to maintain positive pressure for 1 hour using a volume of air equivalent to or less than that stored in the bottled air supply	Each refueling interval (Unit One)	9.13

*See Specification 4.1.D

4.5 SPRAY SYSTEMS TESTS

Applicability

Applied to the testing of the Spray Systems.

Objective

To verify that the Spray Systems will respond promptly and perform their design function, if required.

Specification

A. Test and Frequencies

1.* The containment spray pumps shall be flow tested at a reduced flow rate at least once per month.

2.* All inside containment recirculation spray pumps shall be dry tested at least once per month.

3.* The recirculation spray pumps outside the containment shall be flow tested by determining the shut off head of the pump once per month.

4. The weight loaded check valves within the containment in the various subsystems shall be tested by pressurizing the pump discharge lines with air at least once each refueling period. Verification of seating the check valves shall be accomplished by applying a vacuum upstream of the valves.

5.* All motor operated valves in the containment spray and recirculation spray flow path shall be tested by stroking them at least once per month.

6. The containment spray nozzles and containment recirculation spray nozzles shall be checked for proper functioning at least every five years, coinciding with the closest refueling outage.

7.* The spray nozzles in the refueling water storage shall be checked for proper functioning at least monthly.

* During periods of extended reactor shutdown the monthly testing requirement may be waived provided the component is tested prior to reactor startup.

B. Acceptance Criteria

1. A dry-test of a recirculation spray pump shall be considered satisfactory if the motor and pump shaft rotates, starts on signal, and the ammeter readings for the motor are comparable to the original dry test ammeter readings.

2. A flow-test of a containment spray pump shall be considered satisfactory if the pump starts, and the discharge pressure and flow rate determine a point on the head curve. A check will be made to determine that no

particulate material from the refueling water storage tank clogs the test spray nozzles located in the refueling water storage tank.

3. The test of each of the weight loaded check valves shall be considered satisfactory if air flows through the check valve, and if sealing is achieve.

4. A test of a motor operated valve shall be considered satisfactory if its limit switch operates a light on the main control board demonstrating that the valve has stroked.

5. The test of the containment spray nozzles shall be considered satisfactory if the measured air flow through the nozzles indicates that the nozzles are not plugged.

6. The test of the spray nozzles in the refueling water storage tank shall be considered satisfactory if the monitored flow rate to the nozzles, when compared to the previously established flow rate obtained with the new nozzles, indicated no appreciable reduction in flow rate.

7. The test of the outside recirculation spray pump shall be considered satisfactory if the pump starts and the measured shutoff head of the pump is that specified on the head curve within instrument accuracy.

Basis

The flow testing of each containment spray pump is performed by opening the normally closed valve in the containment spray pump recirculation line returning water to the refueling water storage tank. The containment spray provisions to air test the nozzles every five years coinciding with the closest refueling outage is sufficient to indicate that plugging of the nozzles has not occurred.

The spray nozzles in the refueling water storage tank provide means to ensure that there is no particulate matter in the refueling water storage tank and the Containment Spray Subsystems which could plug or cause deterioration of the spray nozzles. The nozzles in the tank are identical to those used on the containment spray headers.

The monthly flow test of the contaiment spray pumps and recirculation to the refueling water storage will indicate any plugging of the nozzles by a reduction of flow through the nozzles.

References

FSAR Section 6.3.1 Containment Spray Pumps

FSAR Section 6.3.1 Recirculation Spray Pumps

b. Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment, initiated by a simulated loss of off-site power together with a simulated safety injection signal. This test will be conducted during reactor shutdown for refueling to assure that the diesel generator will start within 10 sec and assume load in less than 30 sec after the engine starting signal.

c. Availability of the fuel oil transfer system shall be verified by operating the system in conjunction with the monthly test.

d. Each diesel generator shall be given a thorough inspection during each refueling interval utilizing the manufacturer's recommendations for this class of stand-by service.

2. Acceptance Criteria

The above tests will be considered satisfactory if all applicable equipment operates as designed.

B. Fuel Oil Storage Tanks for Diesel Generators

1. A minimum fuel oil storage of 35,000 gal shall be maintained on-site to assure full power operation of one diesel generator for seven days.

C. Station Batteries

1. Tests and Frequencies

a. The specific gravity, electrolytic temperature, cell voltage of the pilot cell in each 60 cell battery, and the D.C. bus voltage of each battery shall be measured and recorded weekly.

b. Each month the voltage of each battery cell in each 60 cell battery shall be measured to the nearest 0.01 volts and recorded.

c. Every 3 months the specific gravity of each battery cell, the temperature reading of every fifth cell, the height of electrolyte of each cell, and the amount of water added to any cell shall be measured and recorded.

d. Twice a year, during normal operation, the battery charger shall be turned off for approximately 5 min and the battery voltage and current shall be recorded at the beginning and end of the test.

e. During the normal refueling shutdown each battery shall be subjected to a simulated load test without battery charger. The battery voltage and current as a function of time shall be monitored.

f. During the refueling outages connections shall be checked for tightness and anticorrosion coating shall be applied to interconnections.

2. Acceptance Criteria

a. Each test shall be considered satisfactory if the new data when compared to the old data indicate no signs of abuse or deterioration.

b. The load test in (d) and (e) above shall be considered satisfactory if the batteries perform within acceptable limits as established by the manufacturer's discharge characteristic curves.

Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of essential safeguards equipment. They also assure that the emergency diesel generator system controls and the control systems for the safeguards equipment will function automatically in the event of a loss of all normal station service power.

The testing frequency specified will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply and starting circuits and controls are continuously monitored and any faults are alarm indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater System.

Objective

To verify the operability of the auxiliary steam generator feedwater pumps and their ability to respond properly when required.

Specification

A. Tests and Frequency

1.* Each motor driven auxiliary steam generator feedwater pump shall be flow tested for at least 15 minutes on a monthly basis to demonstrate its operability.

2.* The turbine driven auxiliary steam generator feedwater pump shall be flow tested for at least 15 minutes on a monthly basis to demonstrate its operability.

3.* The auxiliary steam generator feedwater pump discharge valves shall be exercised on a monthly basis.

*During periods of extended reactor shutdown the monthly testing requirement may be waived provided the component is tested prior to startup.

4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anaomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of power, the hot channel factors, F_Q and $F^N_{\Delta H}$ shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed values of

$$F_Q(Z) \leq (2.00/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.00) \times K(Z) \text{ for } P \leq .5$$

$$F^N_{\Delta H} \leq 1.55\ (1 + 0.2\ (1 - P)) \times T(BU)$$

(where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, Z is the core height location of F_Q, and T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9), an evaluation as to the cause of the anomaly shall be made.

Basis

BORON CONCENTRATION

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reach initially, and with the control rod assembly groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration, and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive control rod assembly in the fully withdrawn position is always maintained.

2. The test will be considered satisfactory if control board indication and/or visual observations indicate that all the appropriate components have received the safety injection signal in the proper sequence. That is, the appropriate pump breakers shall have opened and closed, and all valves, required to establish a safety injection flow path to the Reactor Coolant System and to isolate other systems from this flow path, shall have completed their stroke.

B. Component Tests

Pumps

1. The low head safety injection pumps and charging pumps shall be operated at intervals not greater than one month. During periods of extended reactor shutdown the monthly testing requirement may be waived provided the component is tested prior to reactor startup.
2. Acceptable levels of performance for the low head safety injection pumps shall be that the pumps start, reach their required developed head on recirculation flow and the control board indications and/or visual observations indicated that the pumps are operating properly.
3. In addition to the Safety Injection System, the charging pumps form an integral part of the Chemical and Volume Control System (CVCS), and are operated on a routine basis as part of this system. If these pumps have performed their design function as part of the routine operation of the CVCS, their level of performance will be deemed acceptable as related to the Specification.

4.12 VENTILATION FILTER TESTS

Applicability

Applies to the testing of particulate and charcoal filters in safety related air filtration systems.

Objective

To verify that leakage efficiency and iodine removal efficiency are within acceptable limits.

Specification

A. Tests and Frequencies

1. The charcoal filters in the Auxiliary Building filter banks, control room emergency filter banks, and relay room emergency filter banks shall be tested for leakage efficiency during the refueling shutdown of Unit 1 using an in-place Freon-112 (or equivalent) test method.

2. The particulate filters in the Auxiliary Building filter, control room emergency filter banks, and relay room emergency filter banks shall be tested for leakage efficiency during the refueling shutdown of Unit 1 using an in-place DOP test method.

3. A carbon sample will be removed from one of the banks once every third year and subjected to chemical analysis to determine the iodine removal capability.

4. Instrumentation, equipment, and procedures shall generally conform to

4. In the event repairs of any welds are required following any examination during successive inspection intervals, the inspection schedule for the repaired welds will revert back to the first 10 year inspection program.

B. For all welds other than those identified in TS Figure 4.15:

1. Welds in the main steam lines including the safety valve headers and in the feedwater lines in the main steam valve house shall be examined in accordance with the requirements of subsection ISC 100 through 600 of the 1972 Winter Addenda of the ASME Section XI Code.

C. For all welds in the main steam valve house:

1. A visual inspection of the surface of the insulation at all weld locations shall be performed on a weekly basis when the reactor is greater than 350°F/450 psig for detection of leaks. Any detected leaks shall be investigated and evaluated. If the leakage is caused by a through-wall flaw, either the plant shall be shutdown, or the leaking piping isolated. Repairs shall be performed prior to return of this line to service.

2. Repairs, reexamination and piping pressure tests shall be conducted in accordance with the rules of ASME Section XI Code.

5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Safety Injection System.

Objective

To define those design features which are essential in providing for safe system operations.

Specifications

A. Reactor Core

1. The reactor core contains approximately 176,200 lbs of uranium dioxide in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. All fuel rods are pressurized with helium during fabrication. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods except for two demonstration fuel assemblies in Unit 2 which are part of Region 4 fuel. The demonstration assemblies each contain 264 fuel rods.

2. The average enrichment of the initial core is 2.51 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is 3.12 weight per cent of U-235.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight percent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 816 poison rods in the form of 12 rod clusters, which are located in vacant control rod assembly guide thimbles. The burnable poison rods consist of pyrex clad with stainless steel.

5. There are 48 full-length control rod assemblies and 5 part-length control rod assemblies in the reactor core. The full-length control rod assemblies contain a 144-inch length of silver-indium-cadmium alloy clad with stainless steel. The part-length control rod assemblies contain a 36-inch length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.

6. Surry Unit 1, Cycle 4, Surry Unit 2, Cycle 3, and subsequent cores will meet the following criteria at all times during the operating lifetime.

   a. Hot channel factors:

$$F_Q(Z) \leq (2.00/P) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq (4.00) \times K(Z) \text{ for } P \leq 0.5$$

$$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1-P)) \times T(BU)$$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, Z is the core height of F_Q, and T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9.

b. The moderator temperature coefficient in the power operating range is less than or equal to:

1) +3.0 pcm/°F at less than 50% of rated power, or

2) +3.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

c. Capable of being made subcritical in accordance with Specification 3.12 A.3.C

7. Up to 10 grams of enriched fissionable material may be used either in the core or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

B. Reactor Coolant System

1. The design of the Reactor Coolant System complies with the code requirements specified in Section 4 of the FSAR.

2. All piping, components, and supporting structures of the Reactor Coolant System are designed to Class 1 seismic requirements, and have been designed to withstand:

a. Primary operating stresses combined with the Operational seismic stresses resulting from a horizontal ground acceleration of 0.07g and a simultaneous vertical ground acceleration of 2/3 the horizontal, with the stresses maintained within code allowable working stresses.

b. Primary operating stresses when combined with the Design Basis Earthquake seismic stresses resulting from a horizontal ground acceleration of 0.15g and a simulataneous vertical ground

The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 2.b(1) and 2.b(2) need not be reported except where test results themselves reveal a degraded mode as described above. Specifically, the implementation of 3.12.B.2.b.(2) is not reportable.

(3) Observed inadequacies in the implementation of administration or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item 2.a(3) above designed to contain