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OHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: Application for amend to License DPR-31 & DPR-41 revising Tech Specs to reflect results of ECCS analysis on power distribution limits for steam generator tube plugging level of 25%.Supplements 800213 amend request.

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FLORIDA POWER & LIGHT COMPANY

April 29, 1980
L-80-129

Director of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed Amendment to Facility
Operating Licenses DPR-31 and DPR-41

In accordance with 10 CFR 50.30, Florida Power & Light Company submits herewith three (3) signed originals and forty (40) copies of a request to amend Appendix A of Facility Operating Licenses DPR-31 and DPR-41. This request supplements our amendment request of February 13, 1980 (L-80-51).

Our NSSS vendor (Westinghouse) has completed a revised ECCS analysis for Turkey Point Units 3 and 4. February, 1978 Appendix K evaluation models were used for the worst DECLG break (C_D=0.4), assuming a steam generator plugging level of 25%, a 5% reduction in thermal design flow, and removal of a 65°F fuel temperature conservatism. The limiting break was reanalyzed at an Fq of 1.97 with a resulting peak clad temperature of 2136°F. The results of the analysis are presented in Attachment 1.

Our NSSS vendor is currently investigating the impact on LOCA evaluation results of fuel rod models proposed by the NRC in draft NUREG-0630. Compensation for possible penalties from use of these fuel rod models has been demonstrated by available improvements in the ECCS evaluation model (See Attachment 2). Until final resolution of the overall fuel rod model concern, the procedure used to perform this analysis is believed to be suitably conservative and acceptable.

During our review of the ECCS analysis for 25% steam generator tube plugging, it was determined that recent in-containment structural modifications were not explicitly included in the structural heat sink data table (Table 3 of Attachment 1). We have evaluated the effect of the structural modifications (0.5% increase in the total amount of steel in containment) and determined that this will have an insignificant effect on the calculated containment backpressure and subsequently on the peak clad temperature, and that there is peak clad temperature margin available to cover this effect. Accordingly, Florida Power & Light concludes that the attached 25% analysis and the 22% analysis submitted on February 13, 1980 remain valid for the current heat sink configuration in both the Unit 3 and Unit 4 containments.

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The proposed amendment is described below and shown on the accompanying Technical Specification pages bearing the date of this letter in the lower right hand corner.

Page 3.2-3

Specification 3.2.6.a (1) is revised to reflect the results of a recent ECCS analysis on the power distribution limits for a steam generator tube plugging level of 25%.

Figure 3.2-3b

The normalized hot channel factor operating envelope for a steam generator tube plugging level of 25% is revised to reflect the results of a recent ECCS analysis.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/MAS/cph

Attachment

cc: Mr. James P. O'Reilly, Region II
Harold F. Reis, Esquire

reactivity insertion upon ejection greater than 0.3% k/k at rated power. Inoperable rod worth shall be determined within 4 weeks.

b. A control rod shall be considered inoperable if
   (a) the rod cannot be moved by the CRDM, or
   (b) the rod is misaligned from its bank by more than 15 inches, or
   (c) the rod drop time is not met.

c. If a control rod cannot be moved by the drive mechanism, shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the inoperable rod.

5. CONTROL ROD POSITION INDICATION

If either the power range channel deviation alarm or the rod deviation monitor alarm are not operable rod positions shall be logged once per shift and after a load change greater than 10% of rated power. If both alarms are inoperable for two hours or more, the nuclear overpower trip shall be reset to 93% of rated power.

6. POWER DISTRIBUTION LIMITS

a. Hot channel factors:

(1) With steam generator tube plugging >22% and ≤25%, the hot channel factors (defined in the basis) must meet the following limits at all times except during low power physics tests:

$$F_q\ (Z) \le (1.97/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \le (3.94) \times K(Z), \text{ for } P \le .5$$

$$F^N_{\Delta H} \le 1.55\ [1.+0.2\ (1-P)]$$

Where P is the fraction of rated power at which the core is operating; K(Z) is the function given in Figure 3.2-3b; Z is the core height location of F_q.

If F_q, as predicted by approved physics calculations, exceeds 1.97, the power will be limited to the rated power multiplied by the ratio of 1.97 divided by the predicted F_q, or augmented surveillance of hot channel factors shall be implemented.

(2) With steam generator tube plugging ≤22%, the hot channel factors (defined in the basis) must meet the following limits at all times except during low power physics tests:

$$F_q\ (Z) \le (1.99/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \le (3.98) \times K(Z), \text{ for } P \le .5$$

$$F^N_{\Delta H} \le 1.55\ [\ 1.+0.2\ (1-P)]$$

Where P is the fraction of rated power at which the core is operating; K(Z) is the function given in Figure 3.2-3a; Z is the core height location of F_q.

HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

(for steam generator tube plugging 25% and F_q=1.97)

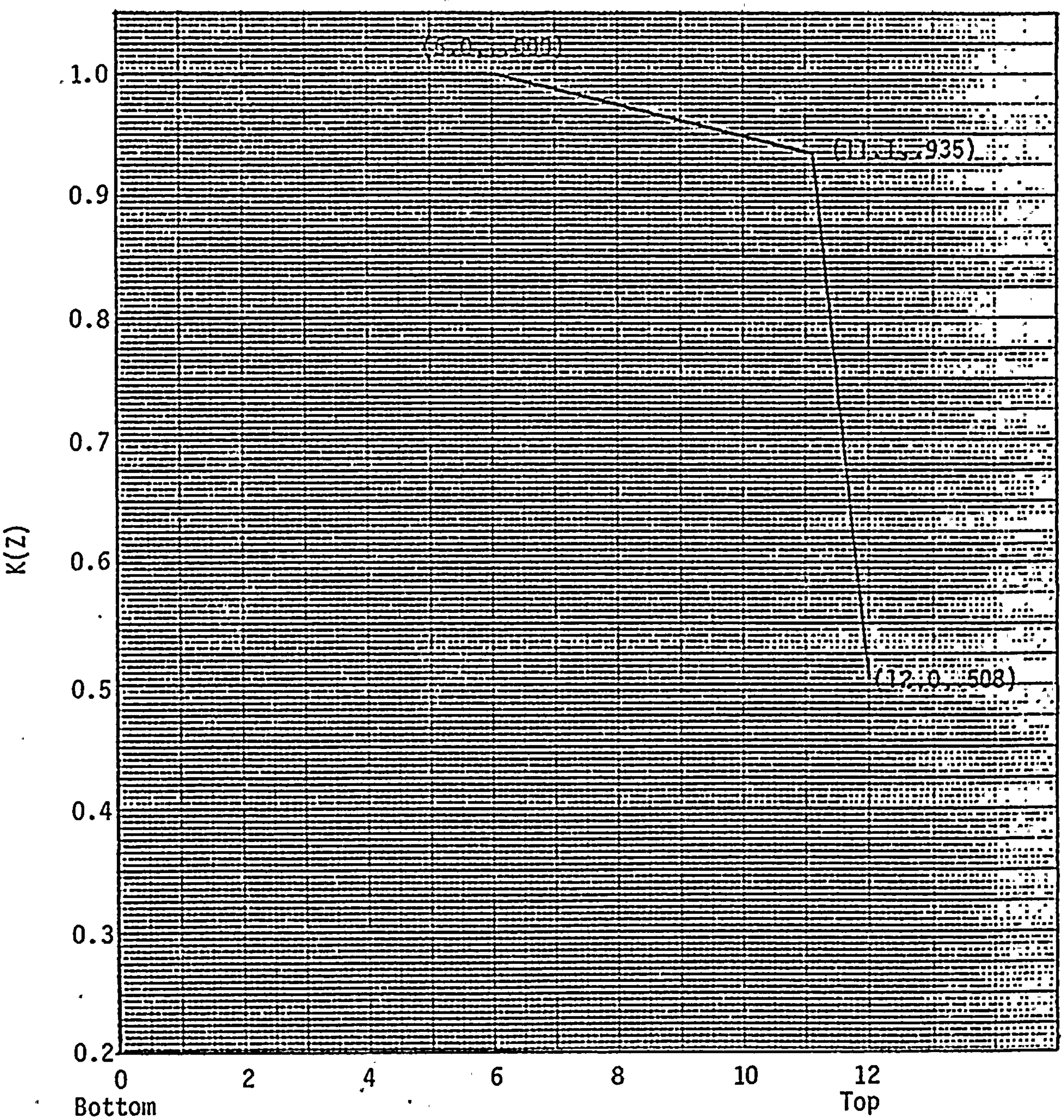

Figure 3.2-3b

ATTACHMENT 1

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	DECL C_D=0.4 (Sec)
START	0.0
Rx Trip Signal	0.669
S.I. Signal	0.73
Acc. Injection	15.5
End of Bypass	27.83
End of Blowdown	29.12
Bottom of Core Recovery	46.6
Acc. Empty	59.67
Pump Injection	25.73

TABLE 2

LARGE BREAK

Results	DECL
Peak Clad Temp. °F	2136
Peak Clad Location Ft.	6.0
Local Zr/H_2O Rxn(max)%	6.945
Local Zr/H_2O Location Ft.	6.0
Total Zr/H_2O Rxn %	<0.3
Hot Rod Burst Time sec	34.8
Hot Rod Burst Location Ft.	6.0

Calculation	
Core Power Mwt 102% of	2200
Peak Linear Power kw/ft 102% of	11.19
Peaking Factor	1.97
Accumulator Water Volume (ft^3)	875 (per accumulator)

Fuel region + Cycle analyzed	Cycle	Region
Unit 3 and Unit 4	All	All

TABLE 3

LARGE BREAK

CONTAINMENT DATA (DRY CONTAINMENT)

NET FREE VOLUME	1.55×10^6	Ft^3
INITIAL CONDITIONS		
Pressure	14.7	psia
Temperature	90	°F
RWST Temperature	39	°F
Service Water Temperature	63	°F
Outside Temperature	39	°F
SPRAY SYSTEM		
Number of Pumps Operating	2	
Runout Flow Rate	1450	gpm
Actuation Time	26	secs
SAFEGUARDS FAN COOLERS		
Number of Fan Coolers Operating	3	
Fastest Post Accident Initation of Fan Coolers	26	secs

CONTAINMENT DATA (DRY CONTAINMENT)

STRUCTURAL HEAT SINKS	THICKNESS (INCH)	AREA (FT^2)
Paint Carbon steel	0.006996 0.20	51824.69
Carbon steel	0.006996	996054.9
Paint Carbon steel	0.006996 0.4896	35660.11
Carbon steel	0.4896	11886.7
Paint Carbon steel Concrete	0.006996 0.2898 24.0	102000.0
Carbon steel Concrete	0.2898 24.0	34000.0
Paint Carbon steel	0.006996 1.56	4622.69
Carbon steel	1.56	1540.89
Paint Carbon steel	0.006996 5.496	1277.87
Carbon steel	5.496	425.93
Paint Carbon steel	0.006996 2.748	951.525
Carbon steel	2.748	317.175
Paint Carbon steel	0.006996 0.03	23550.0
Paint Carbon steel	0.006996 0.063	80368.5
Paint Carbon steel	0.006996 0.10	42278.25
Alluminum	0.006996	102400.0
Stainless steel	0.4404	768.0
Stainless steel	2.1264	3704.0
Stainless steel Concrete	0.1398 24.0	14392.0
Concrete	24.0	59132.0

TABLE 4

REFLOOD MASS AND ENERGY RELEASES - DECLG (CD = 0.4)

TIME (SEC)	MASS FLOW (LB/SEC)	ENERGY FLOW (10^5 BTU/SEC)
46.597	0.0	0.0
47.822	0.0245	0.003
54.36	34.06	0.4418
64.488	77.45	0.9665
78.288	82.3	1.025
94.288	100.5	1.131
111.088	250.8	1.514
128.688	276.8	1.535
166.488	285.4	1.453
208.588	292.7	1.360
255.688	300.6	1.249

TABLE 5

Broken Loop Accumulator Flow To Containment
For Limiting Case Declg (CD = 0.4)

TIME (SEC)	MASS FLOW (LB/SEC)
0.0	0.0
0.01	2820.8
2.01	2367.2
4.01	2082.2
6.01	1879.4
8.01	1725.0
10.01	1600.2
15.01	1369.6
20.01	1215.1
25.01	1108.2
30.84	1026.4
31.567	1017.3

* FOR ENERGY FLOW, MULTIPLY MASS FLOW BY
AN ENTHALPY OF 59.62 BTU/LB

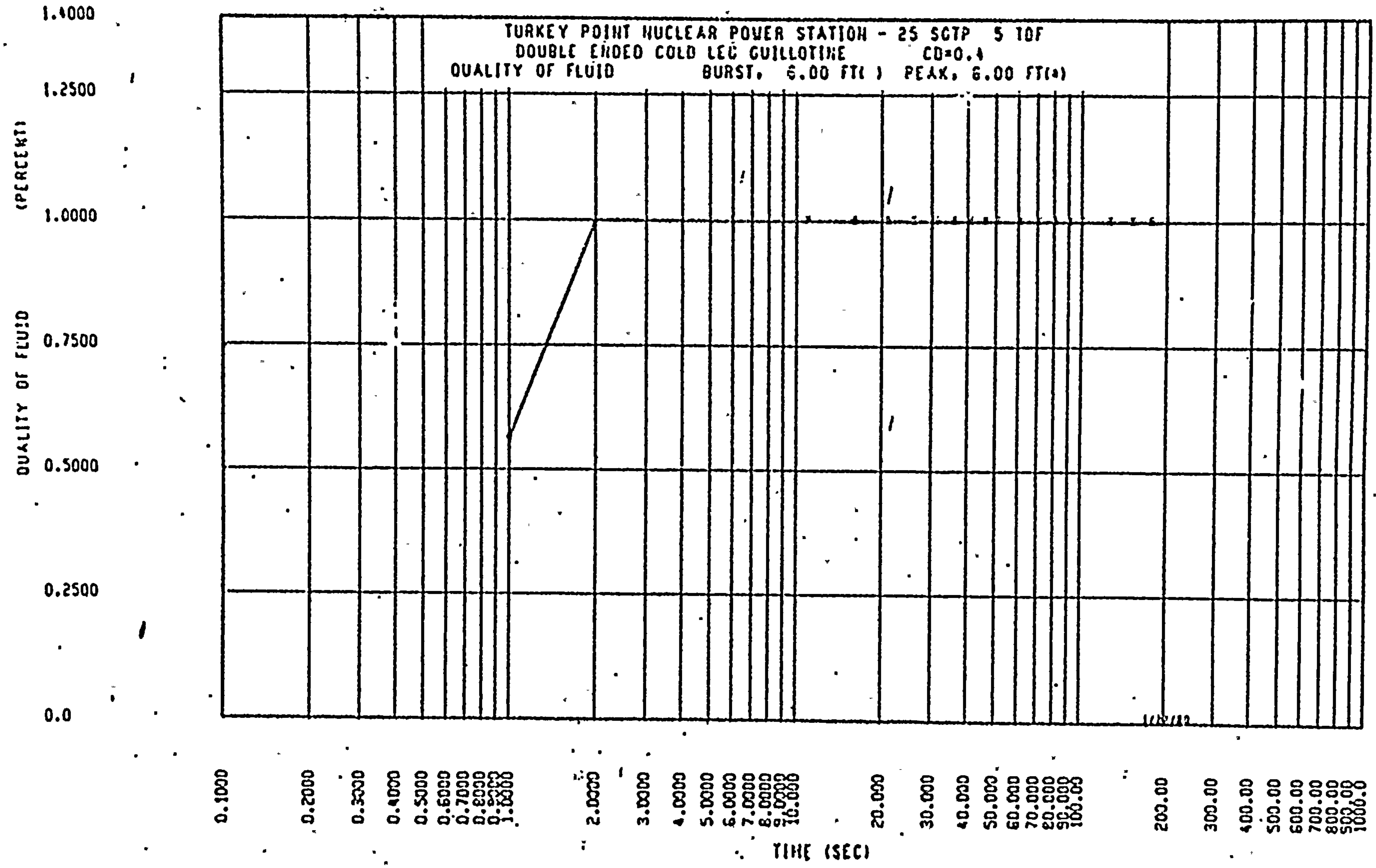

Figure 1 Fluid quality - DECLG (CD=0.4)

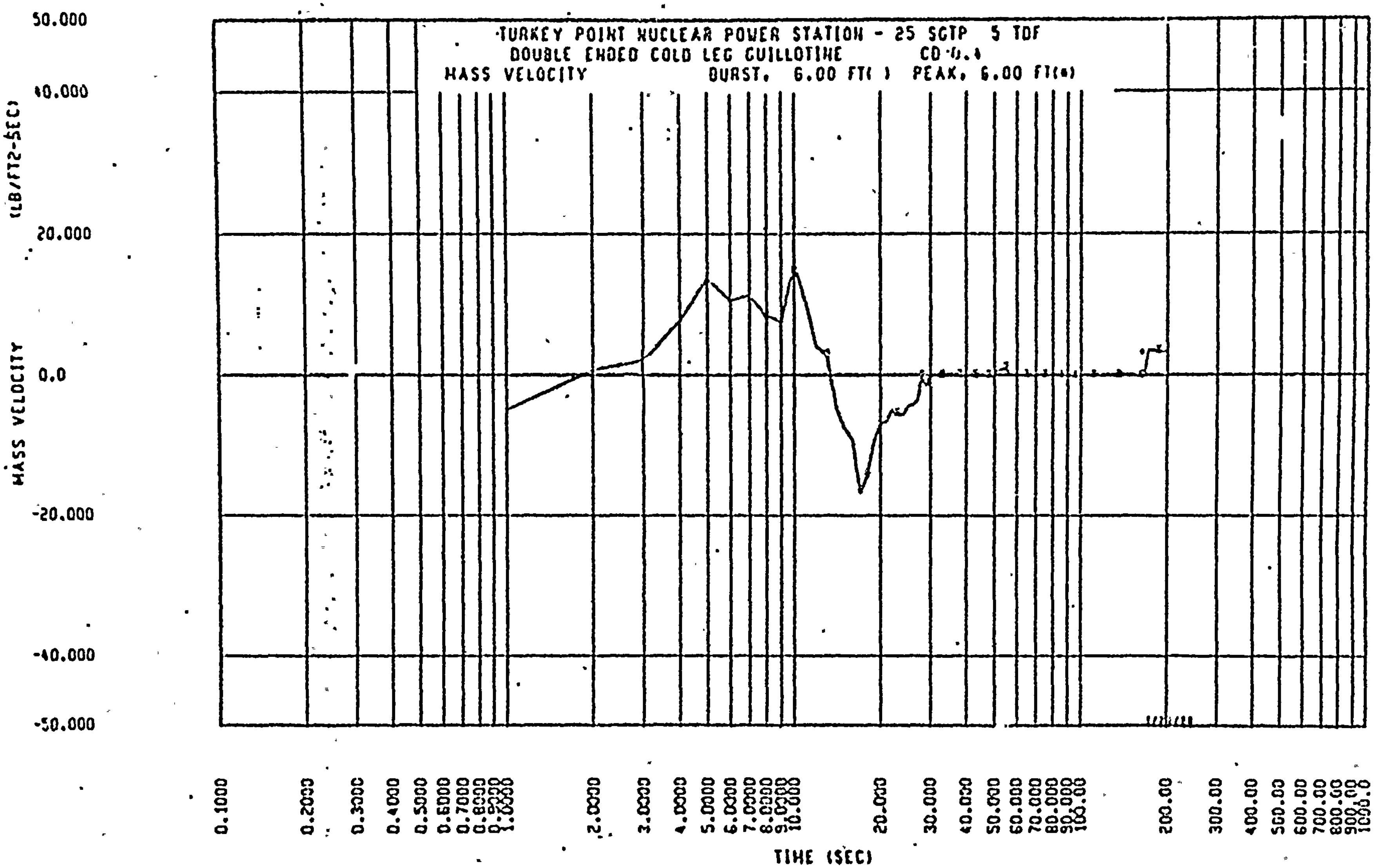

Figure 2 Mass Velocity - DECLG (CD=0.4)

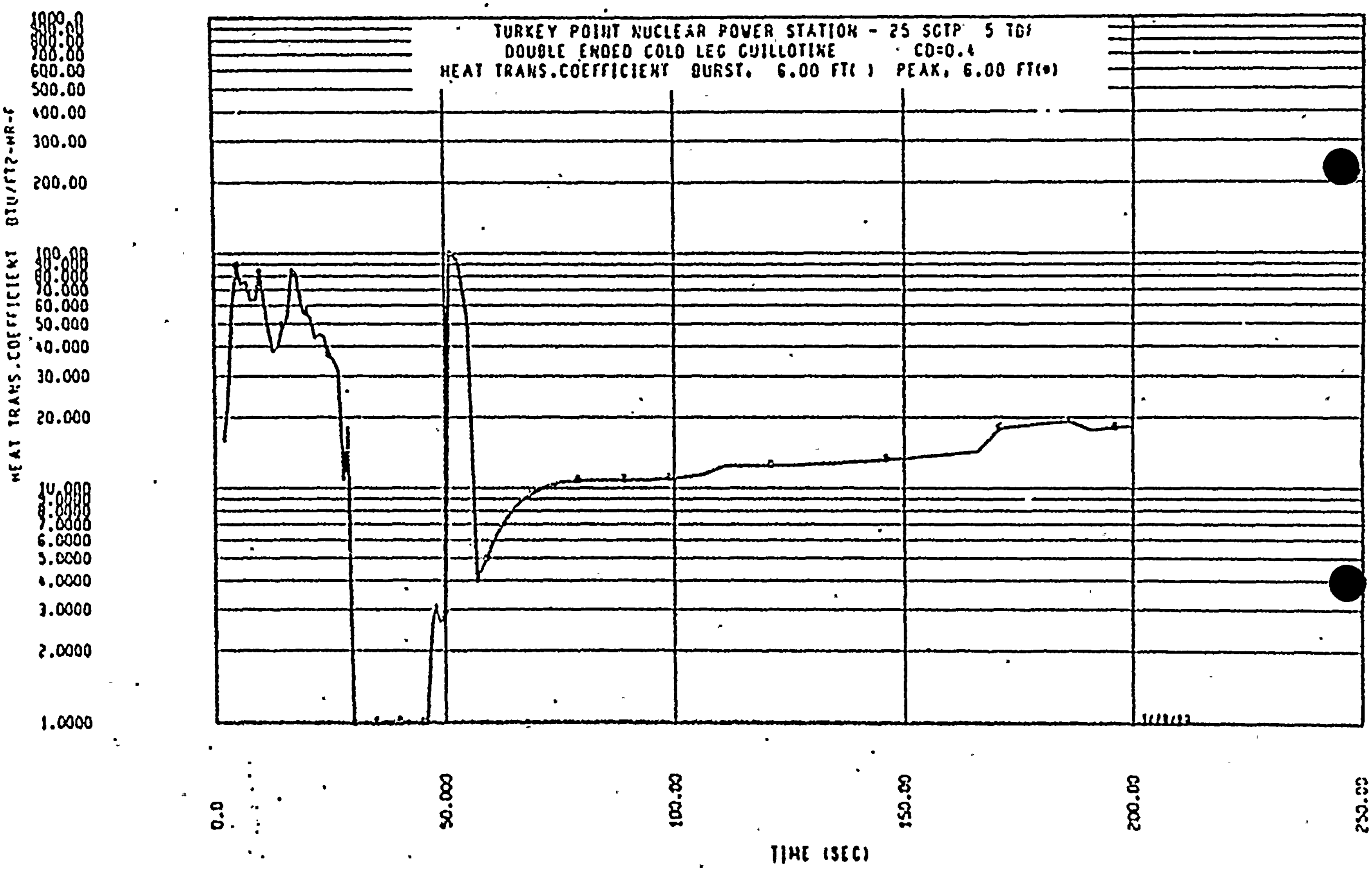

Figure 3 Heat Transfer Coefficient - DECLG (CD=0.4)

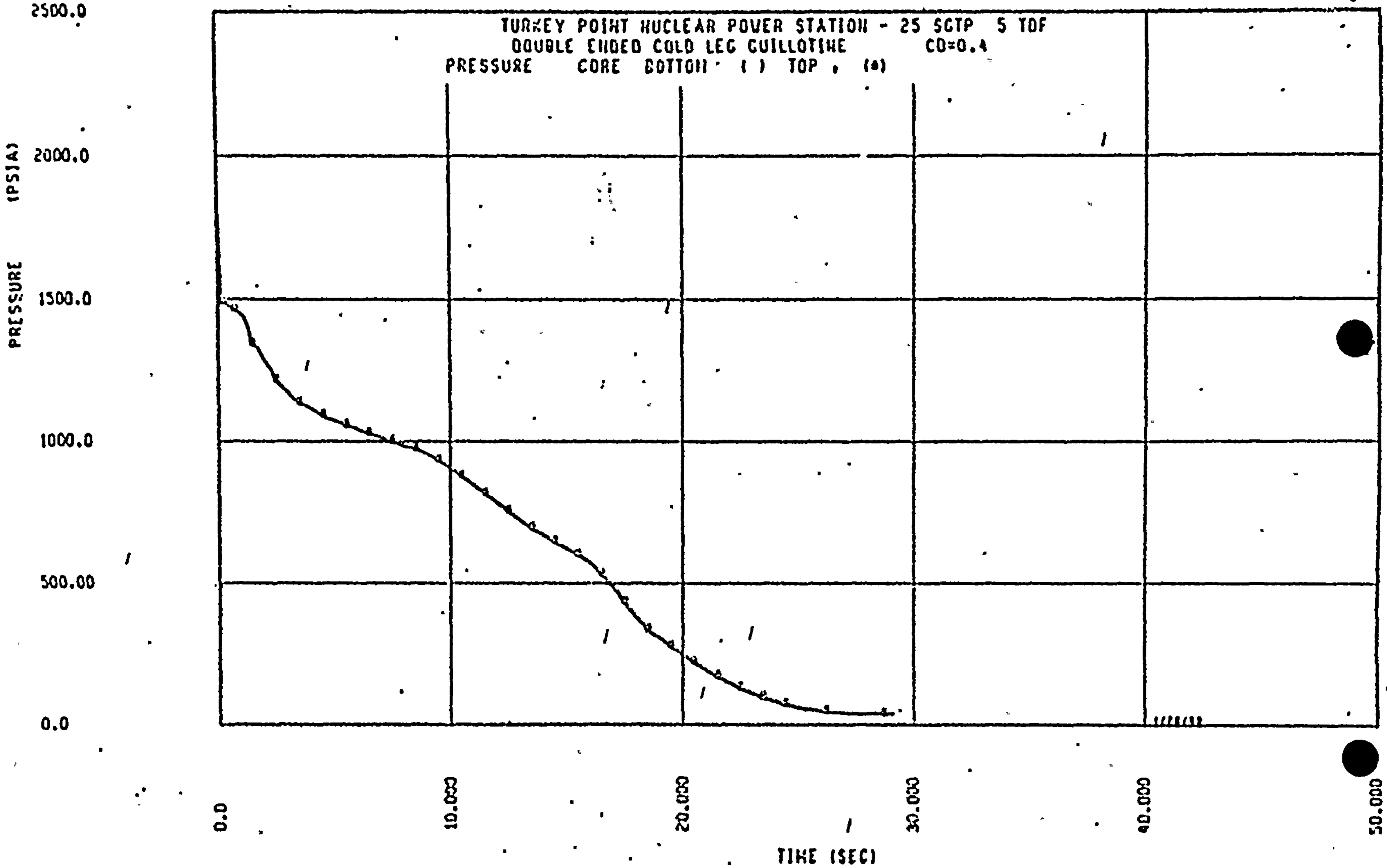

Figure 4 Pressure - DECLG (CD=0.4)

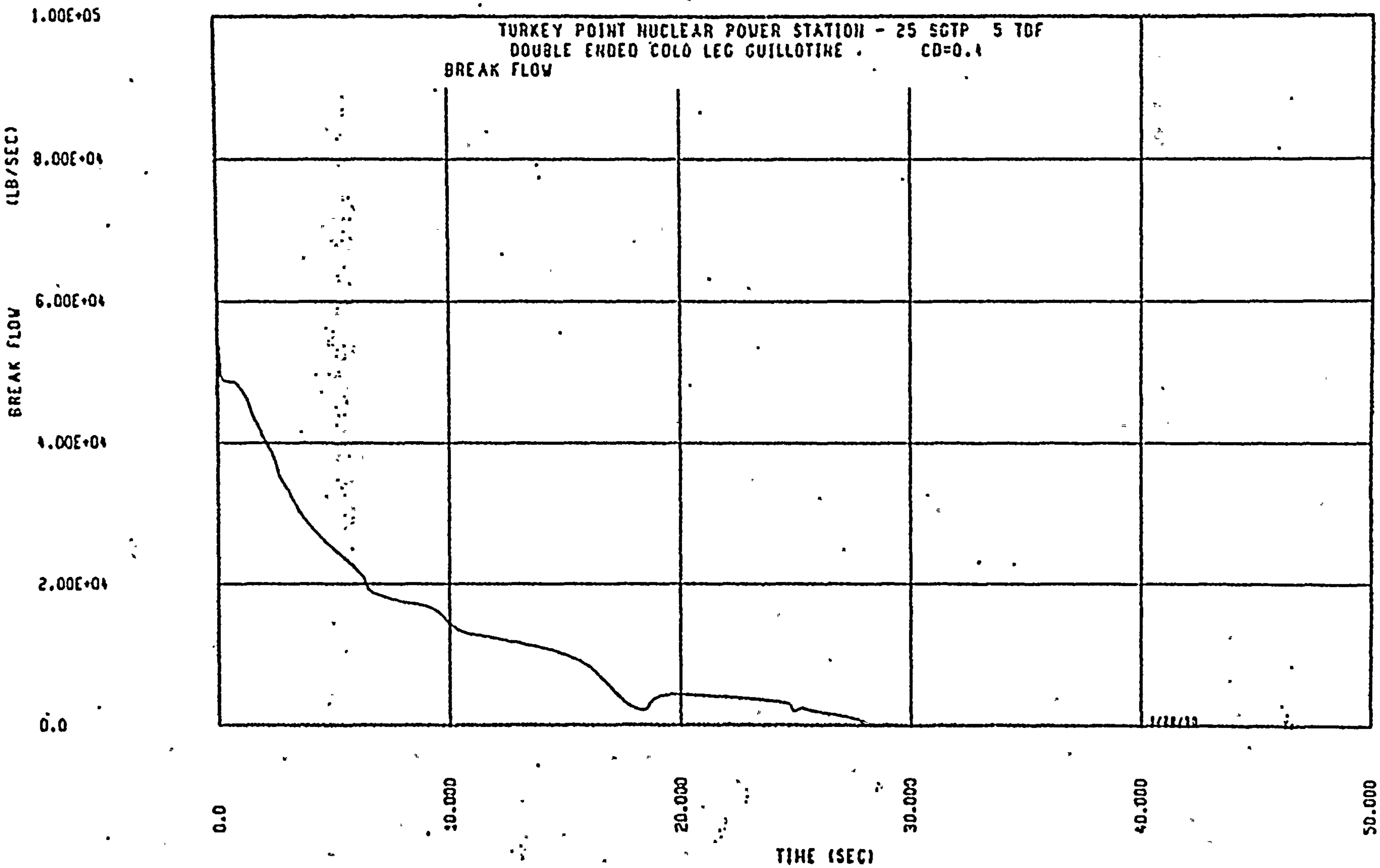

Figure 5 Break Flow Rate - DECLG (CD=0.4)

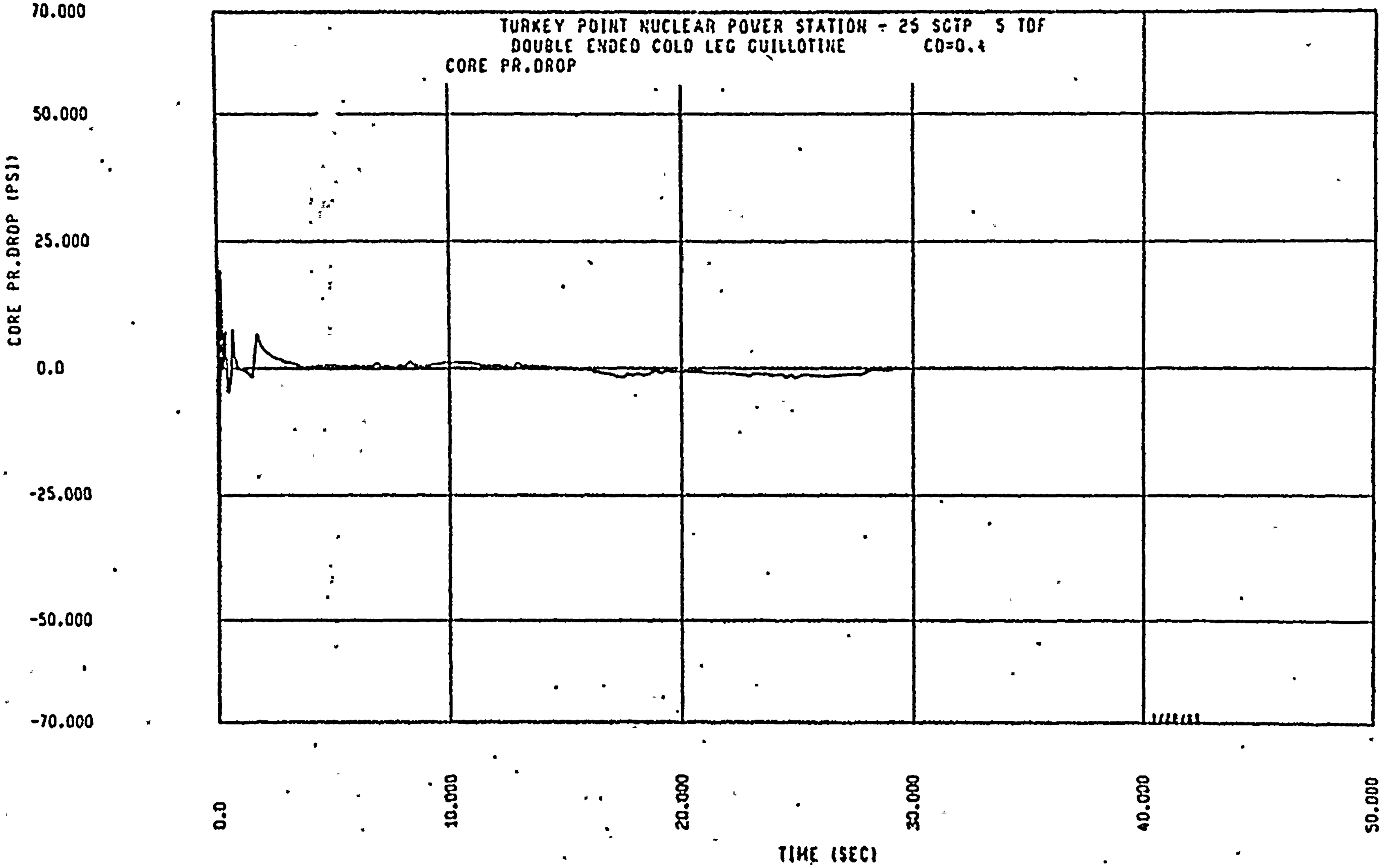

Figure 6 Core Pressure Drop - DECLG (CD=0.4)

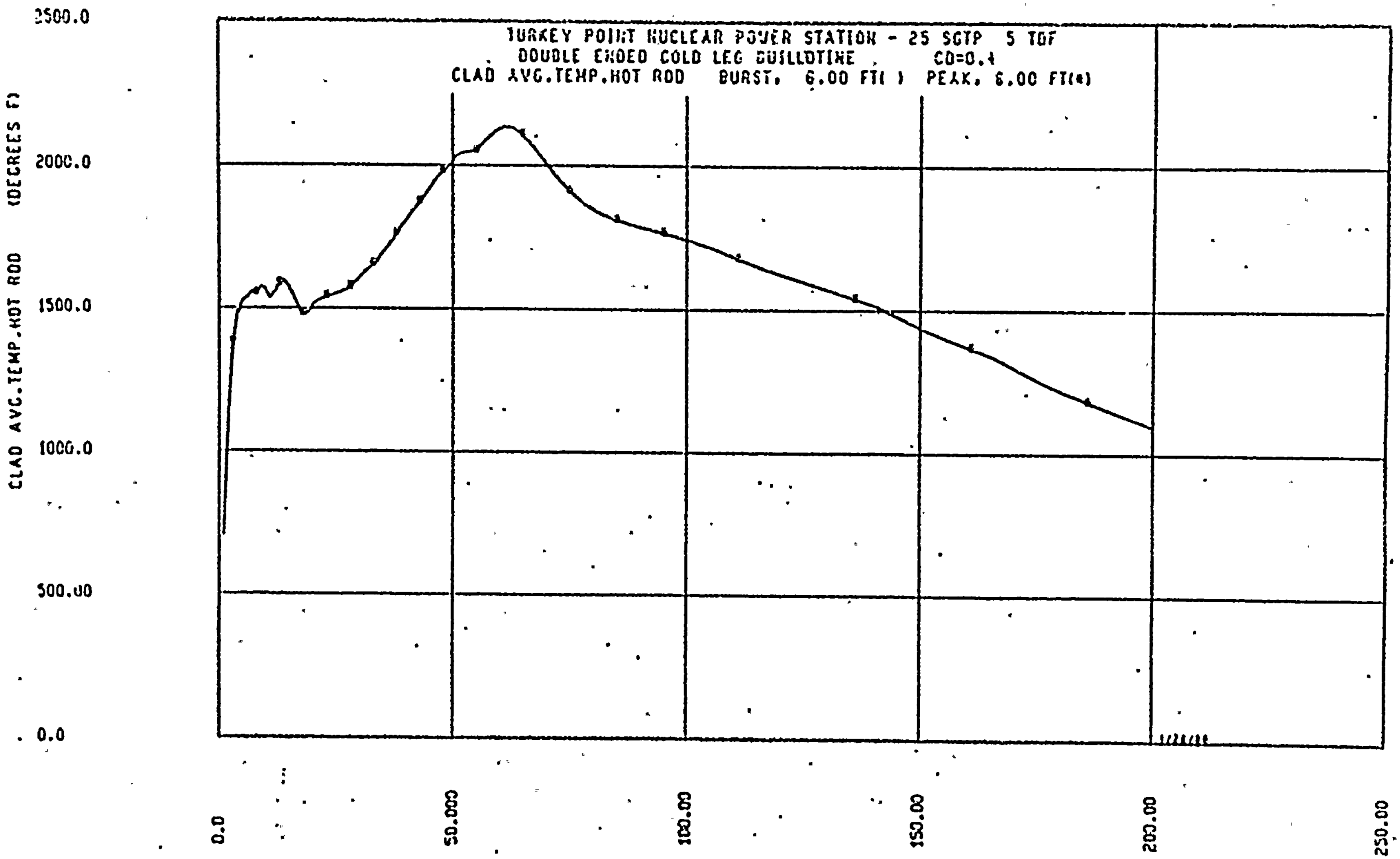

Figure 7 Peak Clad Temperature - DECLG (CD=0.4)

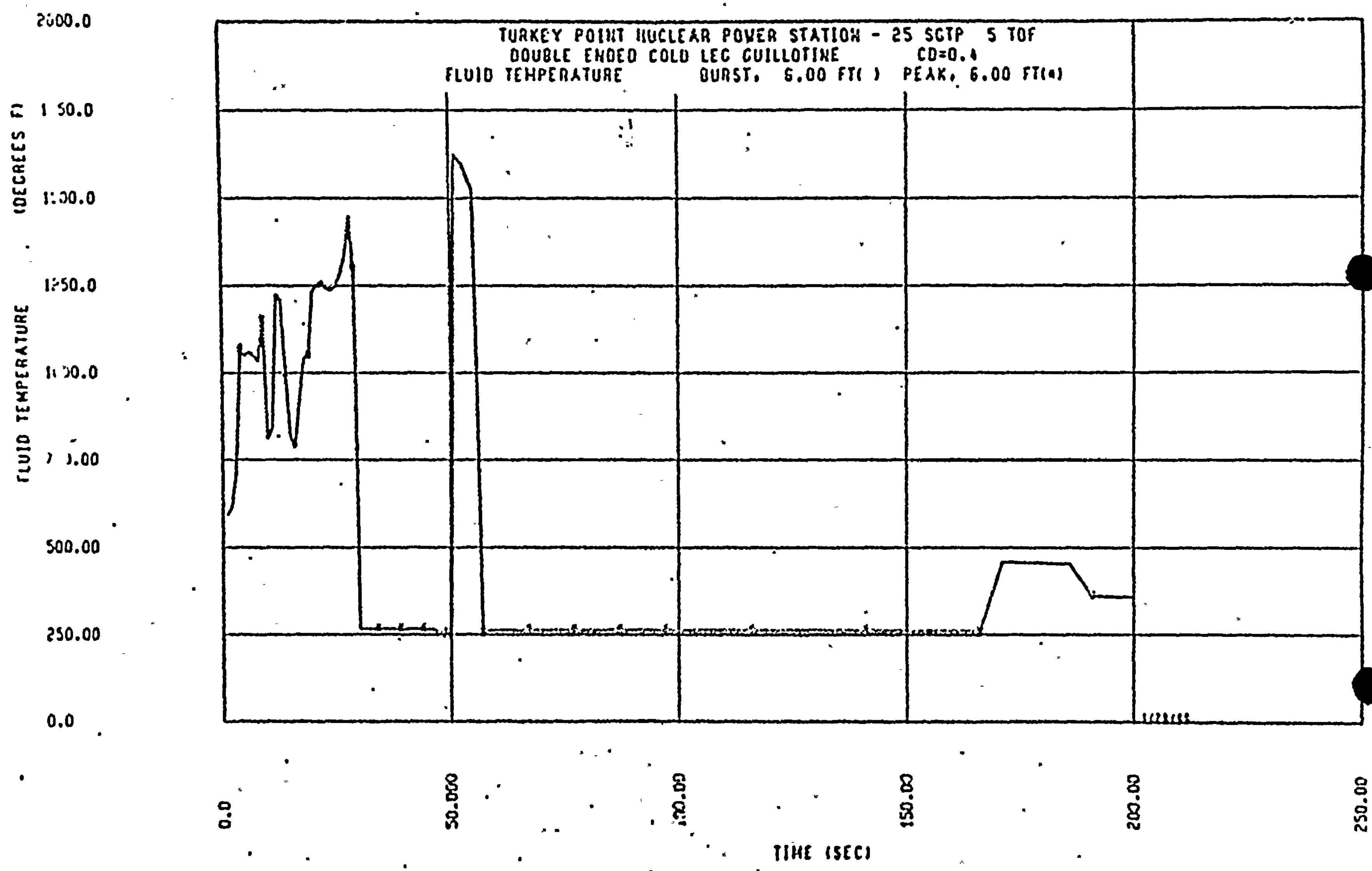

Figure 8 Fluid Temperature - DECLG (CD=0.4)

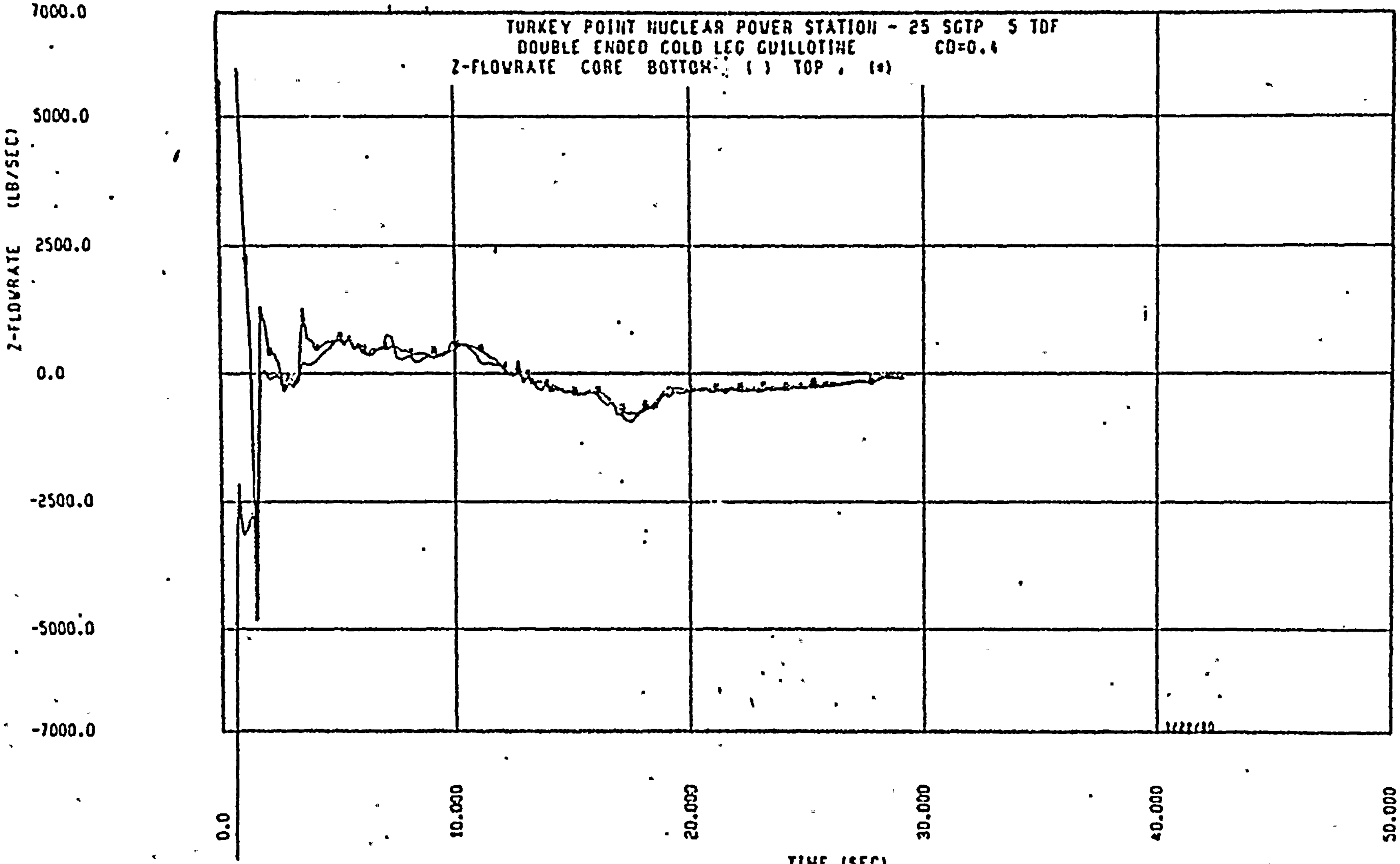

Figure 9 Core Flow - Top and Bottom - DECLG (CD=0.4)

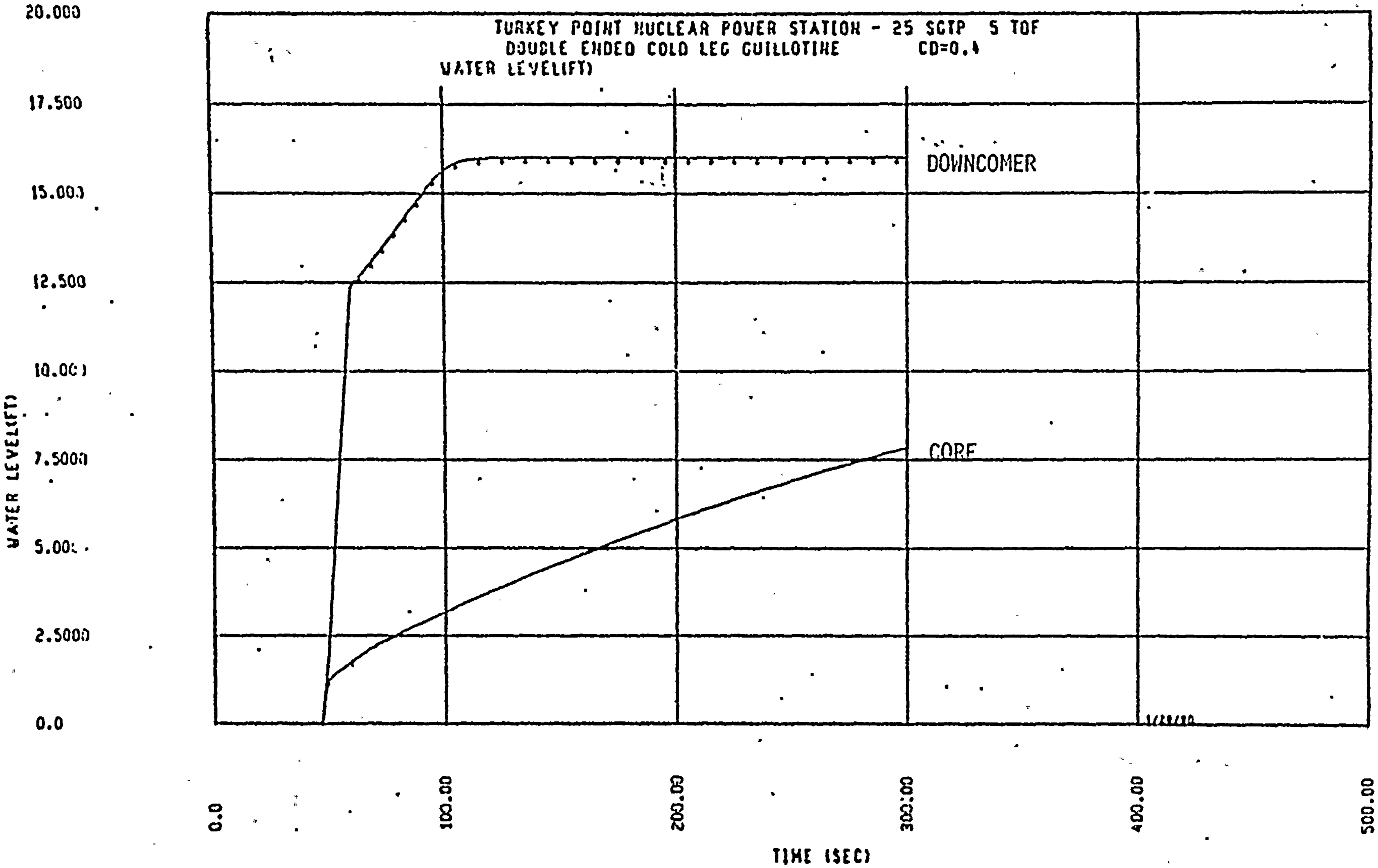

Figure 10a Reflood Transient - DECLG (CD=0.4)

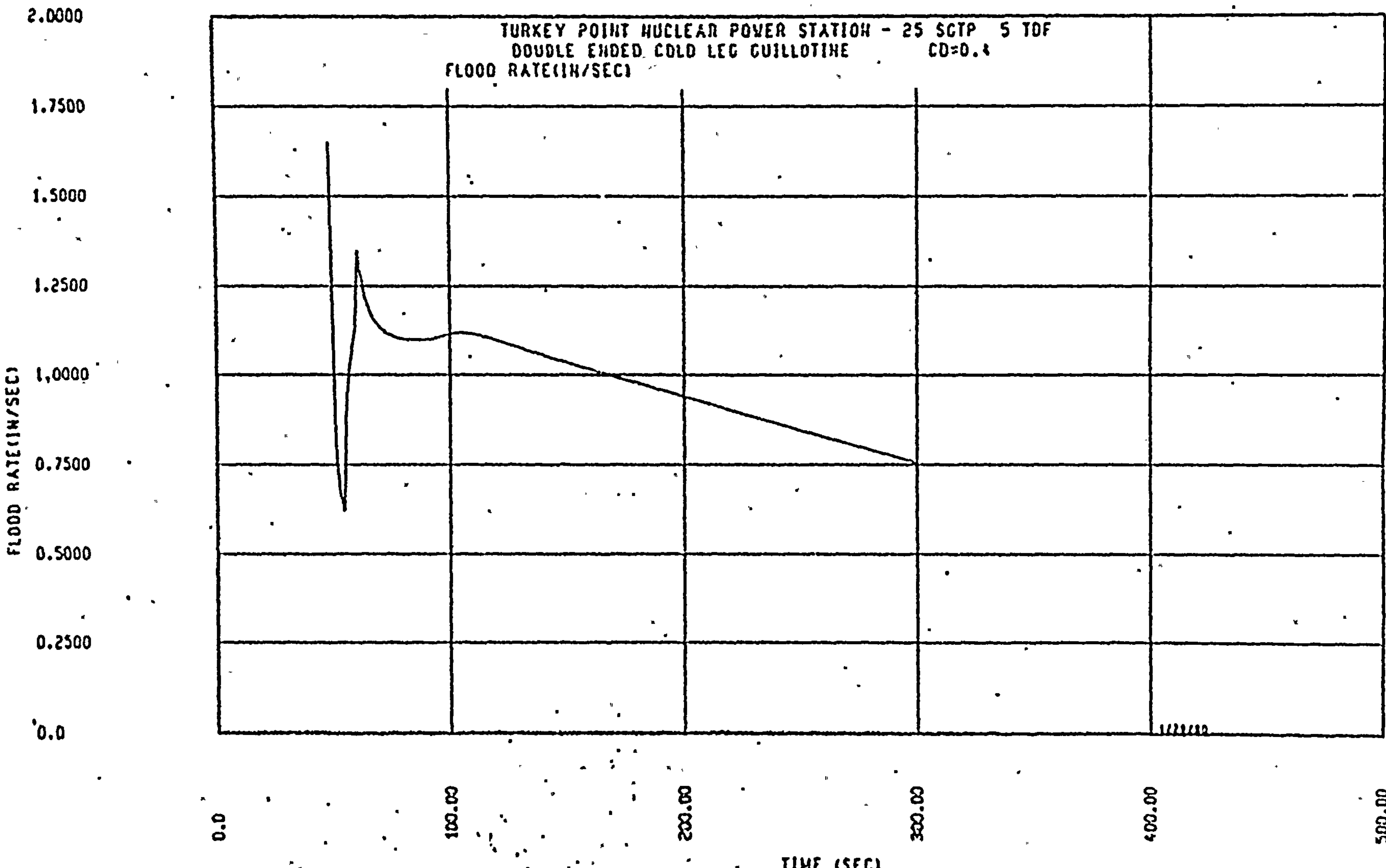

Figure 10b Reflood Transient - DECLG (CD=0.4).

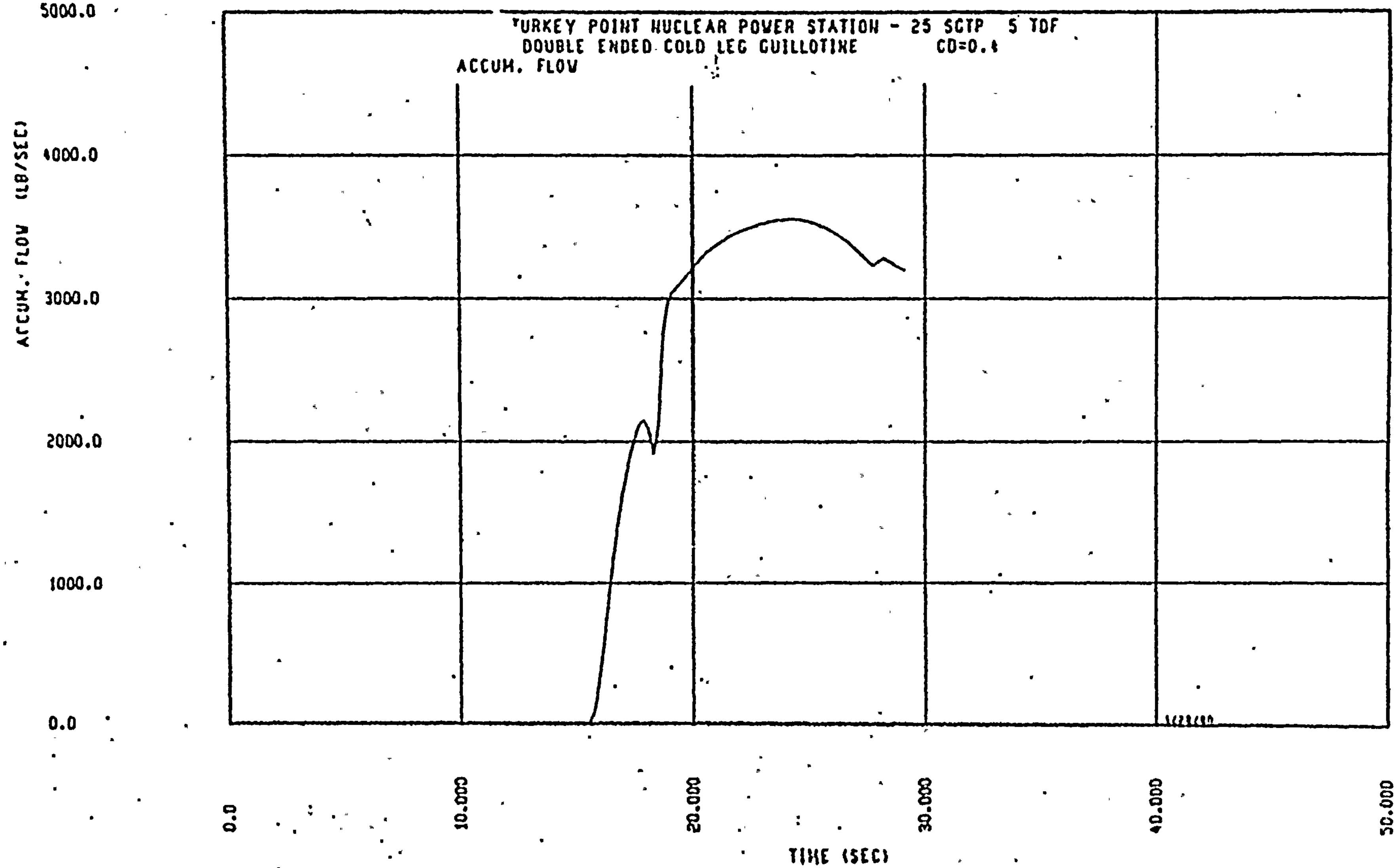

Figure 11 Accumulator Flow (Blowdown) - DECLG (CD=0.4)

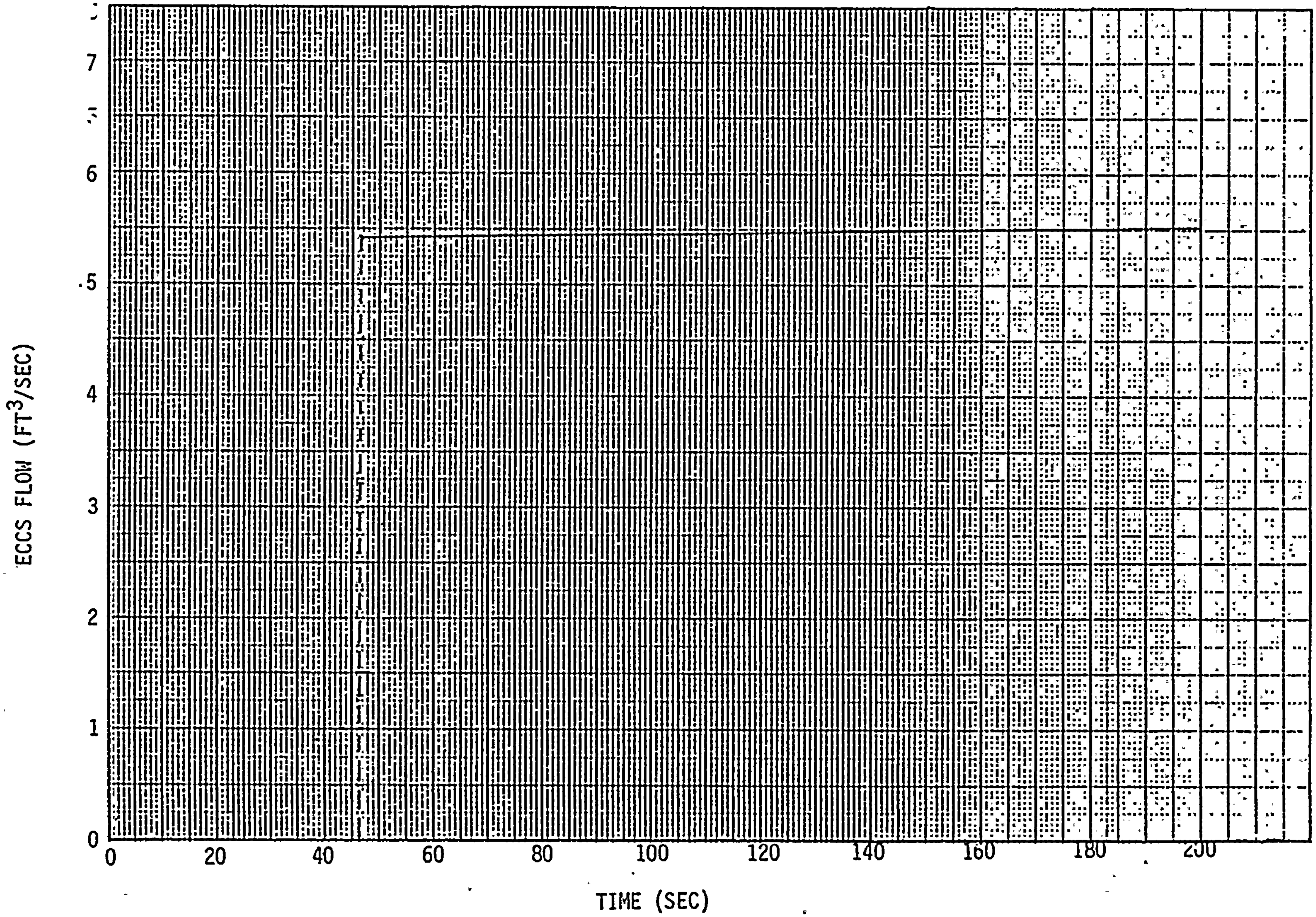

Figure 12 Pump ECCS Flow (Reflood) - DECLG (CD=0.4)

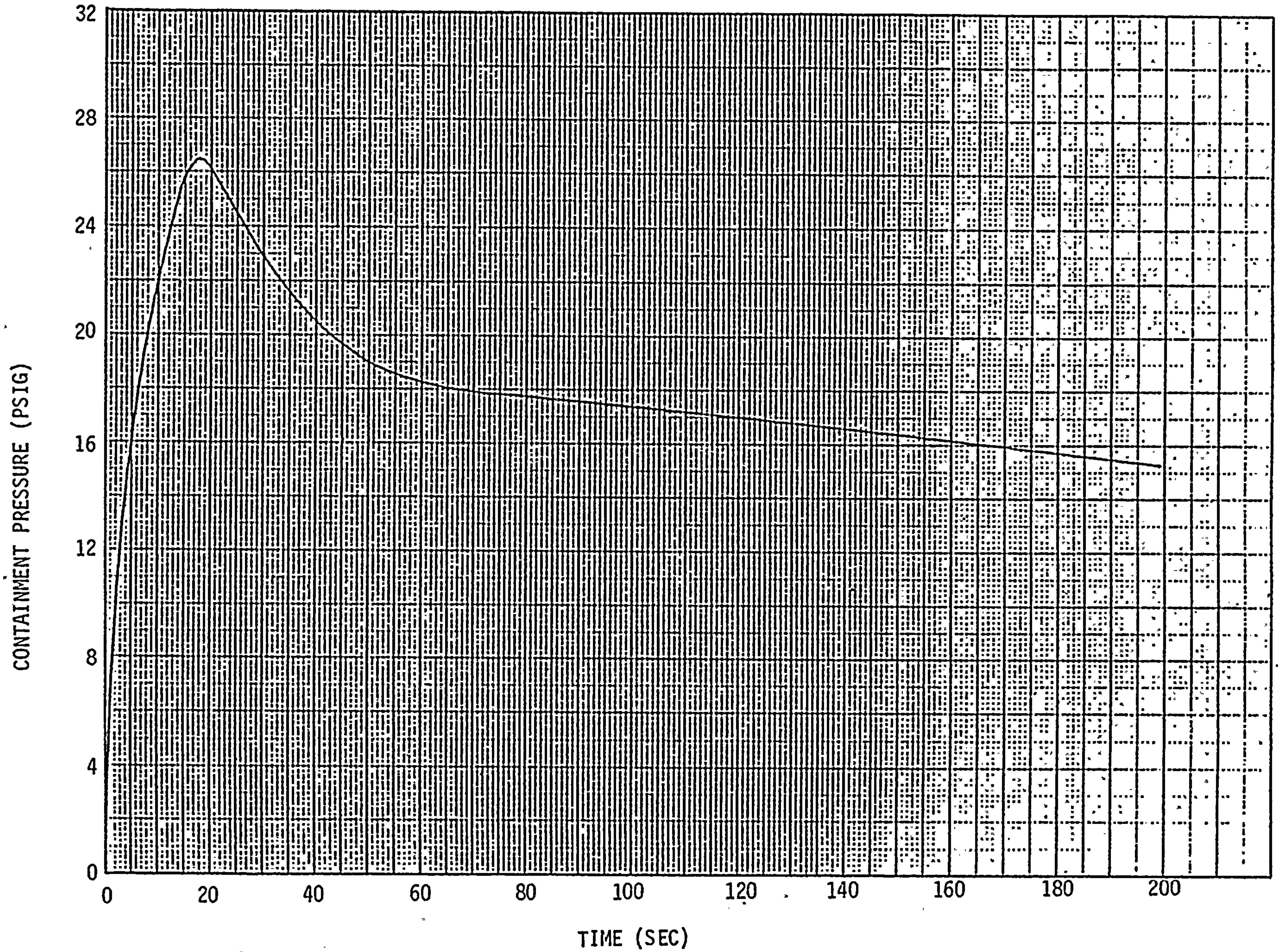

Figure 13 Containment Pressure - DECLG (CD=0.4)

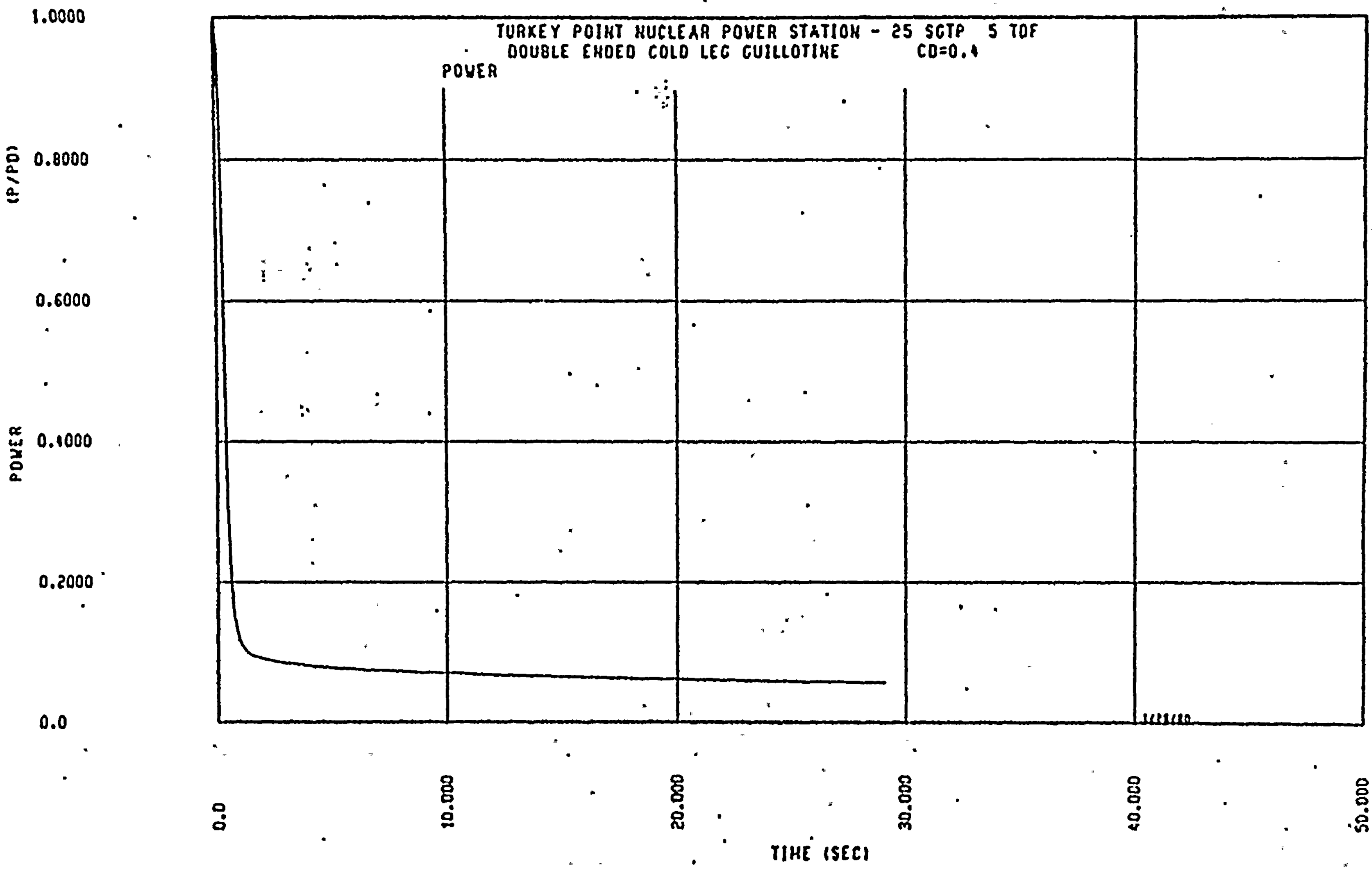

Figure 14 Core Power - DECLG (CD=0.4)

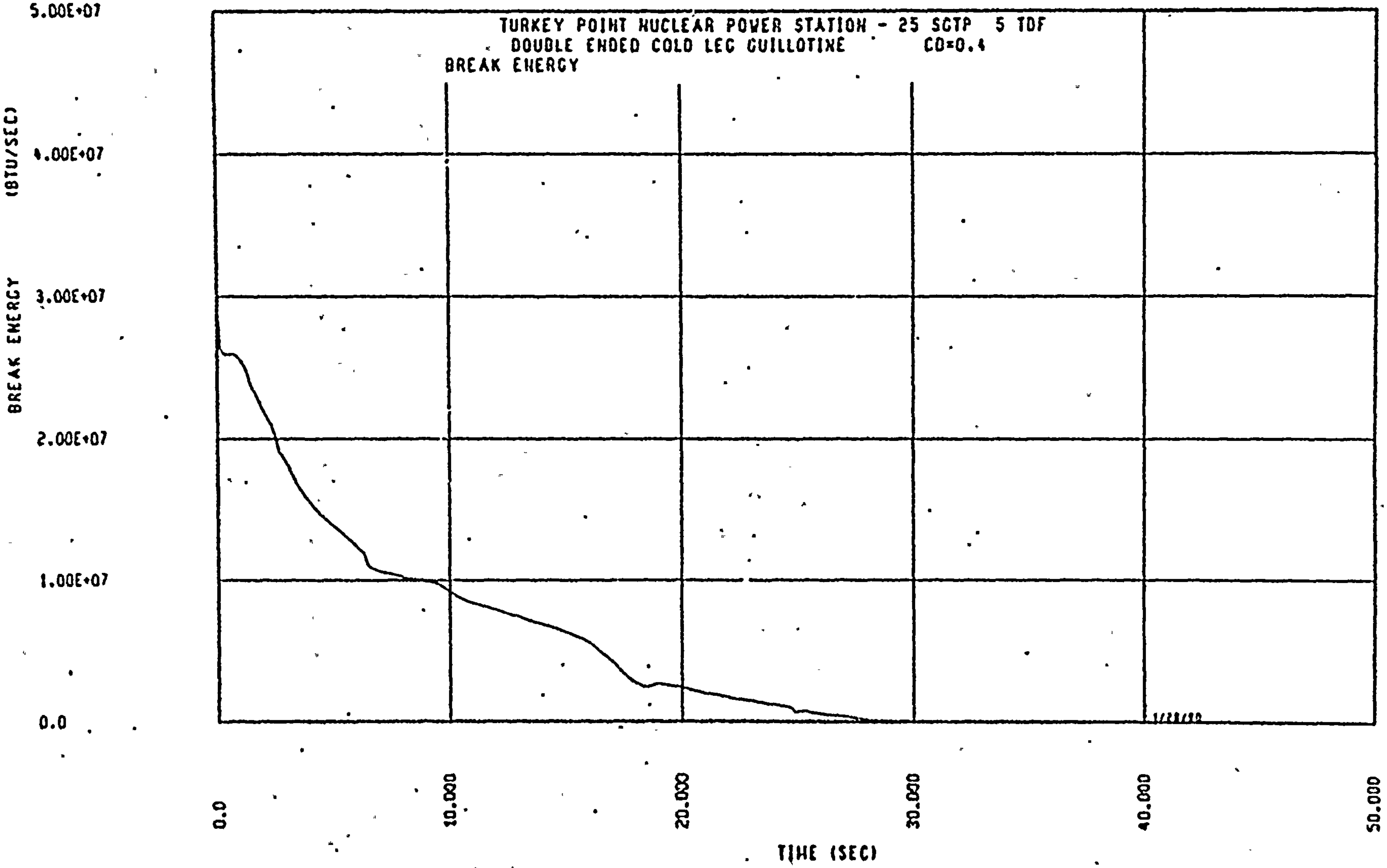

Figure 15. Break Energy Released To Containment - DECLG (CD=0.4)

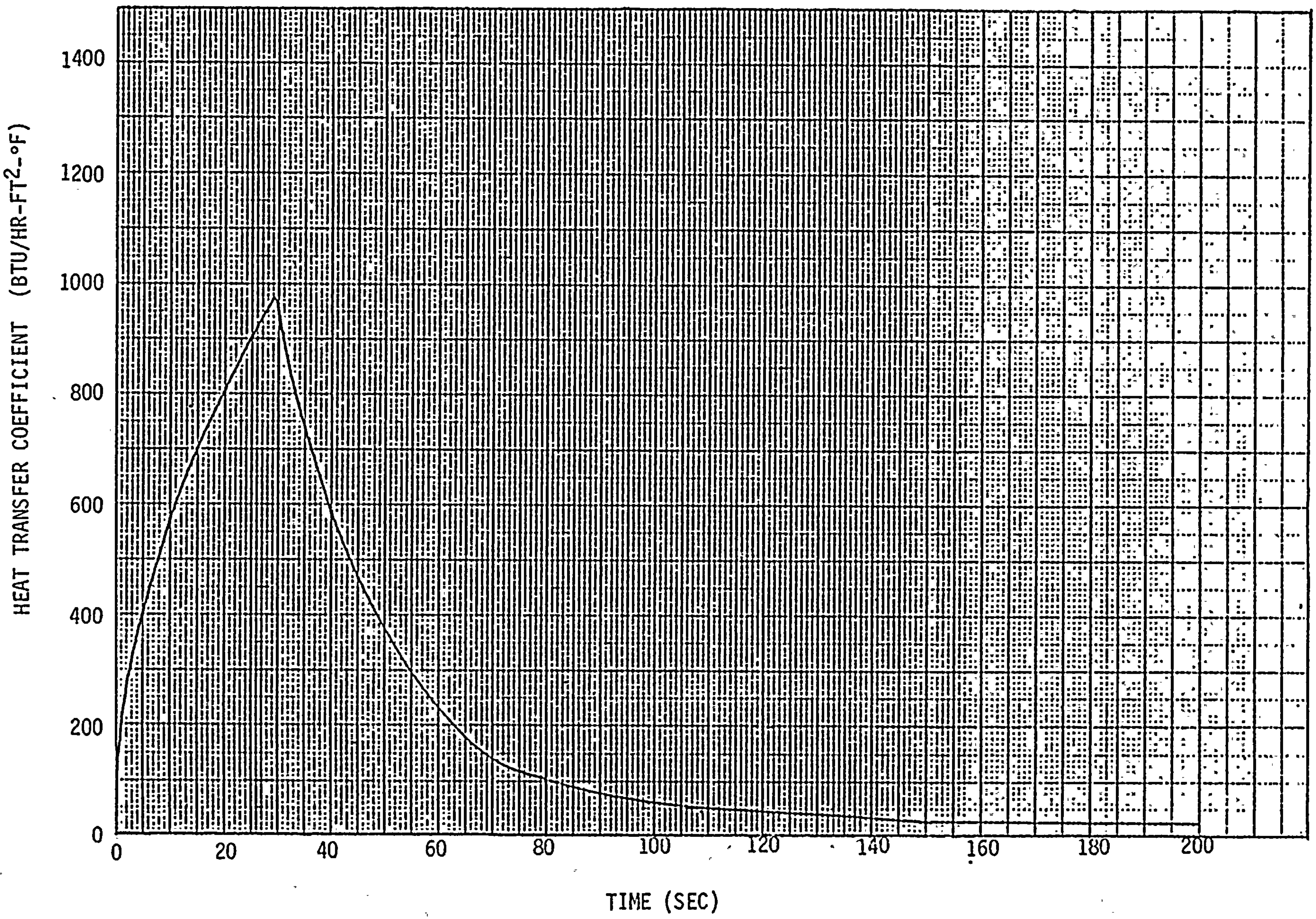

Figure 16 Containment Wall Heat Transfer Coefficient - DECLG (CD=0.4)

The Nuclear Regulatory Commission (NRC) issued a letter dated November 9, 1979 to operators of light water reactors regarding fuel rod models used in Loss of Coolant Accident (LOCA) ECCS evaluation models. That letter describes a meeting called by the NRC on November 1, 1979 to present draft report NUREG 0630, "Cladding Swelling and Rupture Models for LOCA Analysis." At the meeting, representatives of NSSS vendors and fuel suppliers were asked to show how plants licensed using their LOCA/ECCS evaluation model continued to conform to 10 CFR Part 50-46 in view of the new fuel rod models presented in draft NUREG 0630. Westinghouse representatives presented information on the fuel rod models used in analyses for plants licensed with the Westinghouse ECCS evaluation model and discussed the potential impact of fuel rod model changes on results of those analyses. That information was formally documented in letter NS-TMA-2147, dated November 2, 1979, and formed the basis for the Westinghouse conclusion that the information was presented in draft NUREG 0630 did not constitute a safety problem for Westinghouse plants and that all plants conformed with NRC regulations. In the November 9, 1979 letter, the NRC requested that operators of light water reactors provide, within sixty (60) days, information which will enable the staff to determine, in light of the fuel rod model concerns, whether or not further action is necessary.

As a result of compiling information for letter NS-TMA-2147, Westinghouse recognized a potential discrepancy in the calculation of fuel rod burst for cases having clad heatup rates (prior to rupture) significantly lower than 25 degrees F per second. This issue was reported to the NRC staff, by telephone, on November 9, 1979, and although independent of the NRC fuel rod model concern, the combined effect of this issue and the effect of the NRC fuel rod models had to be studied. Details of the work done on this issue were presented to the NRC on November 13, 1969 and documented in letter NS-TMA-2163 dated November 16, 1979. That work included development of a procedure to determine the clad heatup rate prior to burst and a reevaluation of operating Westinghouse plants with consideration of a modified Westinghouse fuel rod burst model. As part of this reevaluation, the Westinghouse position on NUREG-0630 was reviewed and it was still concluded that the information presented in draft NUREG-0630 did not constitute a safety problem for plants licensed with the Westinghouse ECCS evaluation model.

On December 6, 1979, NRC and Westinghouse personnel discussed the information thus far presented. At the conclusion of that discussion, the NRC staff requested Westinghouse to provide further detail on the potential impact of modifications to each of the fuel rod models used in the LOCA analysis and to outline analytical model improvements in other parts of the analysis and the potential benefit associated with those improvements. This additional information was compiled from various LOCA analysis results and documented in letter NS-TMA-2174 dated December 7, 1979.

Another meeting was held in Bethesda on December 20, 1979 where NRC and Westinghouse personnel established: 1) The currently accepted procedure for assessing the potential impact on LOCA analysis results of using the

fuel rod models presented in draft NUREG-0630 and 2) Acceptable benefits resulting from analytical model improvements that would justify continued plant operation for the interim until differences between the fuel rod models of concern are resolved.

Part of the Westinghouse effort provided to assist in the resolution of these LOCA fuel rod model differences is documented in letter NS-TMA-2175, dated December 10, 1979, which contains Westinghouse comments on draft NUREG-0630. As stated in that letter, Westinghouse believes the current Westinghouse models to be conservative and to be in compliance with Appendix K.

A. Evaluation of the potential impact of using fuel rod models presented in draft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for Turkey Point units 3 & 4 with 25% SGTP and 5% red. TDF.

This evaluation is based on the limiting break LOCA analysis identified as follows:

BREAK TYPE - DOUBLE ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT CD=0.4

WESTINGHOUSE ECCS EVALUATION MODEL VERSION February, 1978

CORE PEAKING FACTOR 1.97

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 2136 OF = PCT_B

ELEVATION - 6.0 Feet.

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 1976 OF = PCT_N

ELEVATION - 7.75 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION 4.00 Percent
MAXIMUM CLAD STRAIN AT THIS ELEVATION - 8.52 Percent

Maximum temperature for this non-burst node occurs when the core reflood rate is GREATER than 1.0 inch per second and reflood heat transfer is based on the FLECHT calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - N/A Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - 0.0 Percent

1. BURST NODE

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in terms of the change in the peaking factor limit (FQ) required to maintain a peak clad temperature (PCT) of 2200°F and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above 2200°F, individual effects (such as ΔPCT due to changes in several fuel rod models) indicated here may not accurately apply over large ranges,

but a simultaneous change in FQ which causes the PCT to remain in the neighborhood of 2200.°F justifies use of this evaluation procedure.

From NS-TMA-2174:
For the Burst Node of the clad:

- 0.01 $\Delta FQ \rightarrow \sim 150°F$ BURST NODE ΔPCT

- Use of the NRC burst model and the revised Westinghouse burst model could require an Fq reduction of 0.027

- The maximum estimated impact of using the NRC strain model is a required FQ reduction of 0.03.

Therefore, the maximum penalty for the Hot Rod burst node is:

$$\Delta PCT_1 = (.027 + .03)\ (150°F/.01) = 855°F$$

Margin to the 2200.°F limit is:

$$\Delta PCT_2 = 2200.°F - PCT_B = \underline{64°F}$$

The FQ reduction required to maintain the 2200°F clad temperature limit is:

$$\Delta FQ_B = (\Delta PCT_1 - \Delta PCT_2)\ \left(\frac{.01\ \Delta FQ}{150°F}\right)$$

$$= (\underline{855} - \underline{64})\ \left(\frac{.01}{150}\right)$$

$$= \underline{0.053}\ \text{(but not less than zero).}$$

2. NON-BURST NODE

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspects of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which brust is calculated. Three sets of LOCA analysis results were studied to established an acceptable sensitivity to apply generically in this evalution. The possible PCT increase resulting from a change in strain (in the Hot Rod) is +20.°F. per percent decrease in strain at the maximum clad temperature

locations. Since the clad strain calculated during the reactor coolant system blowdown phase of the accident is not changed by the use of NRC fuel rod models, the maximum decrease in clad strain that must be considered here is the difference between the "maximum clad strain" and the "clad strain at the end of RCS blowdown" indicated above.

Therefore:

$$\Delta PCT_3 = \left(\frac{20^{o}F}{.01 \text{ strain}}\right) (\text{MAX STRAIN - BLOWDOWN STRAIN})$$

$$= \left(\frac{20}{.01}\right) (\underline{0.08}52-\underline{0.04})$$

$$= \underline{\quad 90.4 \qquad\qquad}$$

The second aspect of the analysis that can increase PCT is the flow blockage calculated. Since the greatest value of blockage indicated by the NRC blockage model is 75 percent, the maximum PCT increase can be estimated by assuming that the current level of blockage in the analysis (indicated above) is raised to 75 percent and then applying an appropriate sensitivity formula shown in NS-TMA-2174.

Therefore,

$$\Delta PCT_4 = 1.25^{o}F \text{ (50 - PERCENT CURRENT BLOCKAGE)} + 2.36^{o}F \text{ (75-50)}$$

$$= 1.25 \ (50 - \underline{\ 0.0 \ }) + 2.36 \ (75-50)$$

$$= \underline{\ 121.5 \ }^{o}F$$

If PCT_N occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 90.4 + 0 = 90.4$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200^{o}F - PCT_N = 224.[illegible]^{o}F$$

The FQ reduction required to maintain this 2200°F clad temperature limit is (from NS-TMA-2174)

$$\Delta FQ_N = (\Delta PCT_5 - \Delta PCT_6) \left(\frac{.01 \Delta FQ}{10^{o}F \ \Delta PCT}\right) = -.134$$

$$\Delta FQ_N = \underline{\qquad 0 \qquad} \text{ but not less than zero.}$$

The peaking factor reduction required to maintain the 2200 °F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_H,

or; $\Delta FQ_{PENALTY}$ = 0.053

B. The effect on LOCA analysis results of using improved analytical and modeling techniques (which are currently approved for use in the Upper Head Injection plant LOCA analyses) in the reactor coolant system blowdown calculation (SATAN computer code) has been quantified via an analysis which has recently been submitted to the NRC for review. Recognizing that review of that analysis is not yet complete and that the benefits associated with those model improvements can change for other plant designs, the NRC has established a credit that is acceptable for this interim period to help offset penalties resulting from application of the NRC fuel rod models. That credit for two, three and four loop plants is an increase in the LOCA peaking factor limit of 0.12, 0.15 and 0.20 respectively.

C. The peaking factor limit adjustment required to justify plant operation for this interim period is determined as the appropriate ΔFQ credit identified in section (B) above, minus the $\Delta FQ_{PENALTY}$ calculated in section (A) above (but not greater than zero).

FQ ADJUSTMENT = 0.15 - 0.053

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

Robert E. Uhrig, being first duly sworn, deposes and says:

That he is a Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

Robert E Uhrig

Robert E. Uhrig

Subscribed and sworn to before me this

29th day of April, 1980

Lois T. Marino

NOTARY PUBLIC, in and for the county of Dade, State of Florida

My commission expires: NOTARY PUBLIC STATE OF FLORIDA at LARGE MY COMMISSION EXPIRES AUGUST 24, 1983 BONDED THRU MAYNARD BONDING AGENCY