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50-251 TURKEY POINT PLANT, UNIT 4, FLORIDA POWER AND LIGHT C 05000251

AUTH.NAME AUTHOR AFFILIATION
UHRIG,R.E. FLORIDA POWER & LIGHT CO.
RECIP.NAME RECIPIENT AFFILIATION
STELLO,V. DIVISION OF OPERATING REACTORS

SUBJECT: FORWARDS REQUEST FOR AMEND TO APP A OF LICENSE DPR-31 & DPR-41.AMEND COVERS CHANGING TECH SPEC LIMITS RE ECCS COOLING PERFORMANCE IN LIGHT OF WESTINGHOUSE REEVALUATION.

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FLORIDA POWER & LIGHT COMPANY

May 18, 1979
L-79-124

Director of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Director
Division of Operating Reactors
U. S. Nuclear Regulator Commission
Washington, D. C. 20555

Dear Mr. Stello:

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 and 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.

Westinghouse has reevaluated ECCS cooling performance for Turkey Point Units 3 and 4. The reevaluation supports a Technical Specification limit for total nuclear peaking factor (F_q) of 2.10 for both Units 3 and 4, assuming 22% steam generator tube plugging.

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

Pages 3.2-3 and 3.2-4

A separate set of "hot channel factor" limits is added to Specification 3.2.6.a. The current Specification applies when steam generator tube plugging is less than or equal to 25%. The proposed addition provides a separate set of limits when steam generator tube plugging is less than or equal to 22%.

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Figures 3.2-2, 3.2-2a, and 3.2-2b

The two existing figures are renumbered and a third figure is added to provide successive Hot Channel Factor operating envelopes for 19%, 22%, and 25% steam generator tube plugging.

Pages B3.2-4 and B3.2-6

Bases Section B3.2 (Control and Power Distribution Limits) is revised to include reference to Figure 3.2-2a and 3.2-2b.

The proposed amendment has been reviewed by the Turkey Point Plant Nuclear Safety Committee and the Florida Power & Light Company Nuclear Review Board. They have concluded that it does not involve an unreviewed safety question.

Very truly yours,

E. A. Adomat

for Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/MAS/gm

Attachment

cc: Mr. James P. O'Reilly, Region II
Robert Lowenstein, Esquire

SAFETY EVALUATION

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 & 50-251
Revised ECCS Analysis

The following Tables and Figures provide the results of a revised ECCS analysis performed by the NSSS vendor assuming 22% steam generator tube plugging at Turkey Point Units 3 & 4. The revised analysis supplements an existing approved analysis that assumed 25% tube plugging.

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:

$$F_q\ (Z) \leq (2.03/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \leq (4.06) \times K(Z), \text{ for } P \leq .5$$

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

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 predicted F_q exceeds 2.03, the power will be limited to the rated power multiplied by the ration of 2.03 divided by the predicted F , 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:

$$F_q\ (Z) \leq (2.10/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \leq (4.20) \times K(Z), \text{ for } P \leq .5$$

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

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

If predicted F_q exceeds 2.10, the power will be limited to the rated power multiplied by the ration of 2.10 divided by the predicted F_q, or augmented surveillance of hot channel factors shall be implemented.

b. Following initial loading before the reactor is operated above 75% of rated power and at regular effective full rated power monthly intervals thereafter, power distribution maps, using the movable detector system shall be made, to confirm that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

(1) 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.

(2) The measurement of the enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error.

If either measured hot channel factor exceeds its limit specified under Item 6a, the reactor power shall be reduced so as not to exceed a fraction of the rated value equal to the ration of the F_q or $F_{\Delta H}^N$ limit to measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio. If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing, The reactor may be returned to higher power levels when measurements indicate that hot channel factors are within limits.

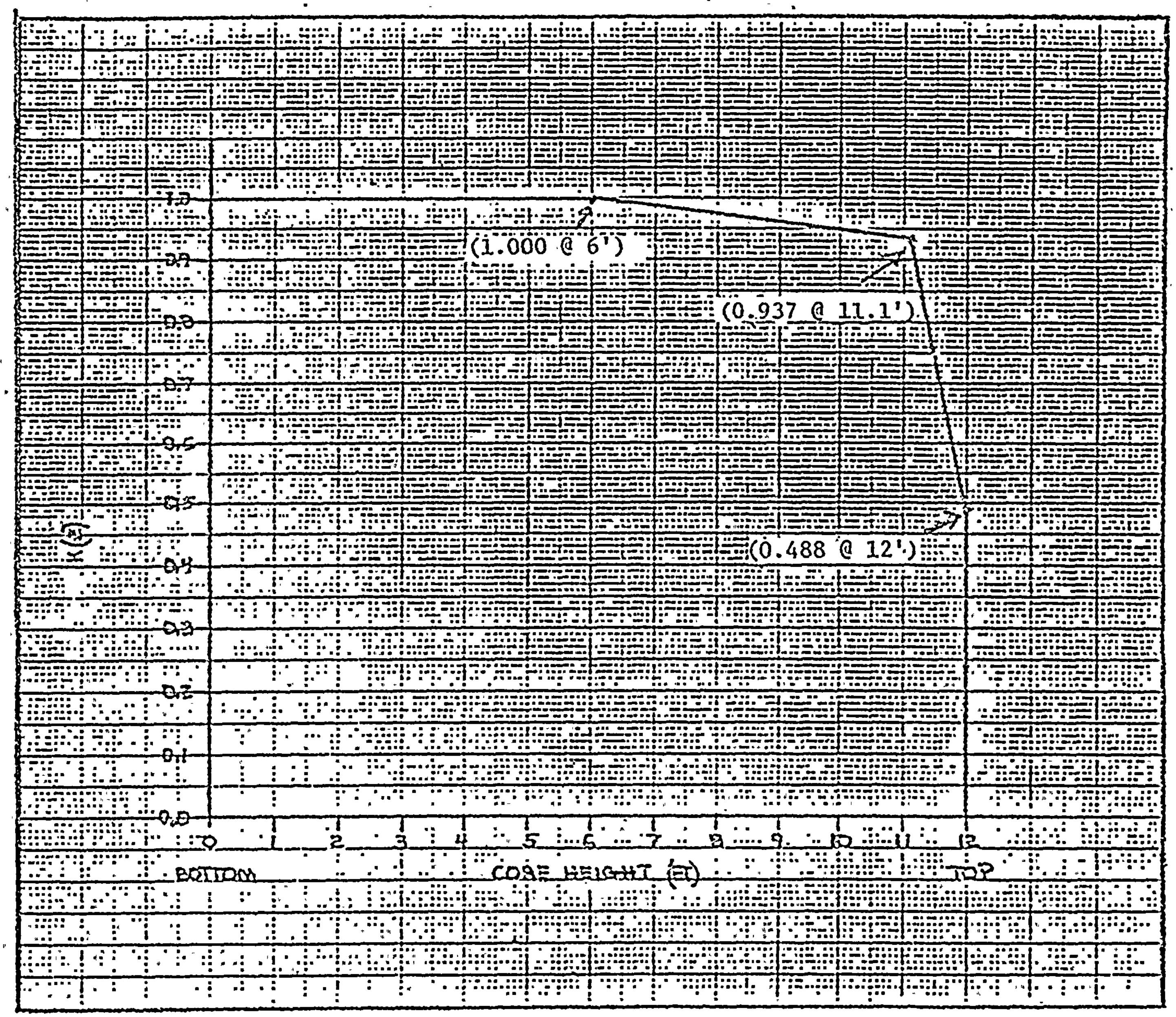
c. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter. If the axial flux difference has not been measured in the last effective full power month, the target flux difference must be updated monthly by linear interpolation using the most recent measured value and the value predicted for the end of the cycle life.

d. Except during physics tests or during excore calibration procedures and as modified by items 6e through 6g below, the indicated axial flux difference shall be maintained within a $\pm$ 5% band about the target flux difference (this defines the target band on axial flux difference).

e. If the indicated axial flux difference at a power level greater than 90% of rated power deviates:

FIGURE 3.2-3

HOT CHANNEL FACTOR
NORMALIZED OPERATING
ENVELOPE (FOR TUBE PLUGGING
>15% ≤19%)



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HOT CHANNEL FACTOR - NORMALIZED OPERATING ENVELOPE (for steam generator tube plugging 22% and F_q = 2.10)

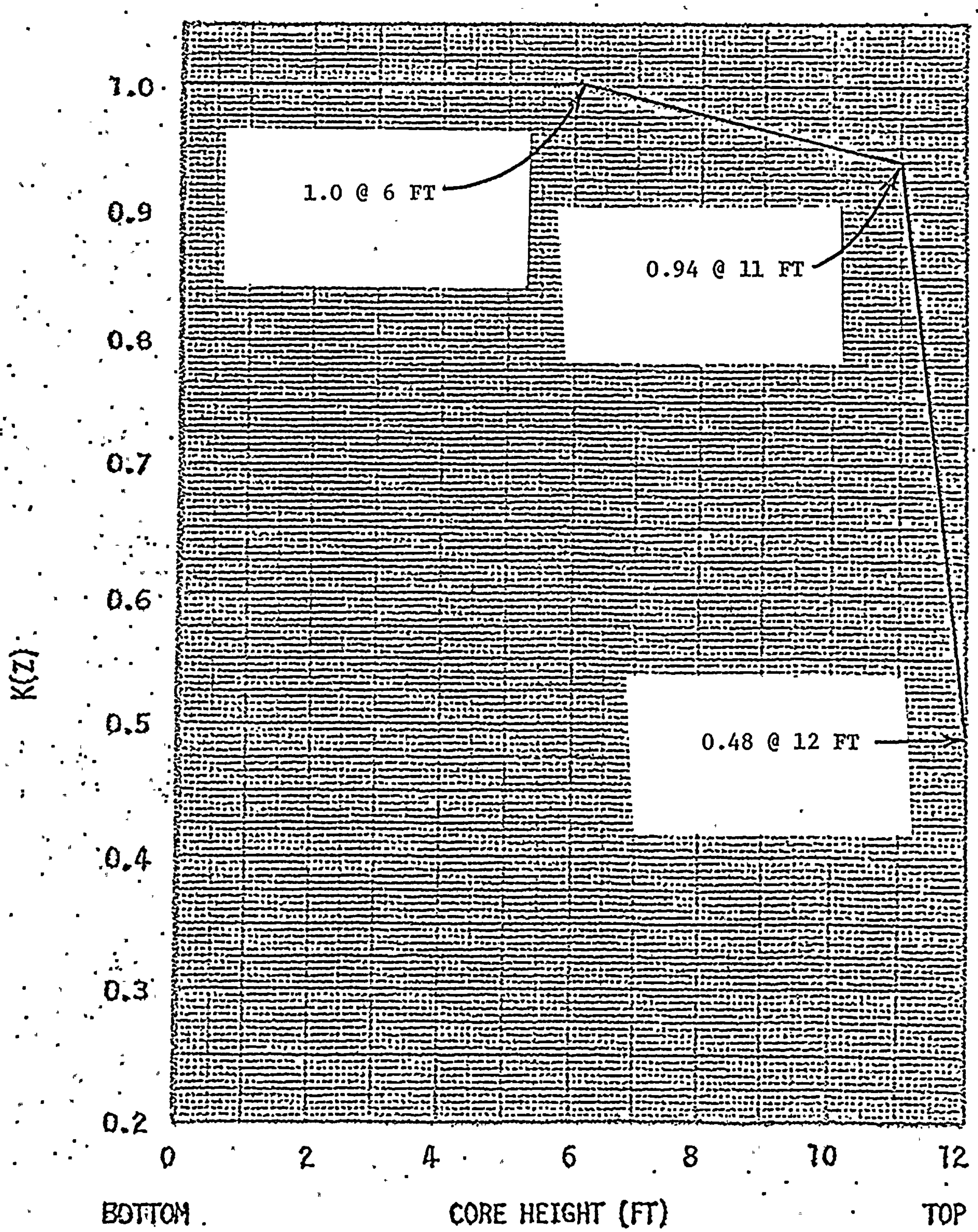

Figure 3.2-3a

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HOT CHANNEL FACTOR-NORMALIZED OPERATING ENVELOPE (FOR STEAM GENERATOR TUBE PLUGGING ≤25% and F_q=2.03)

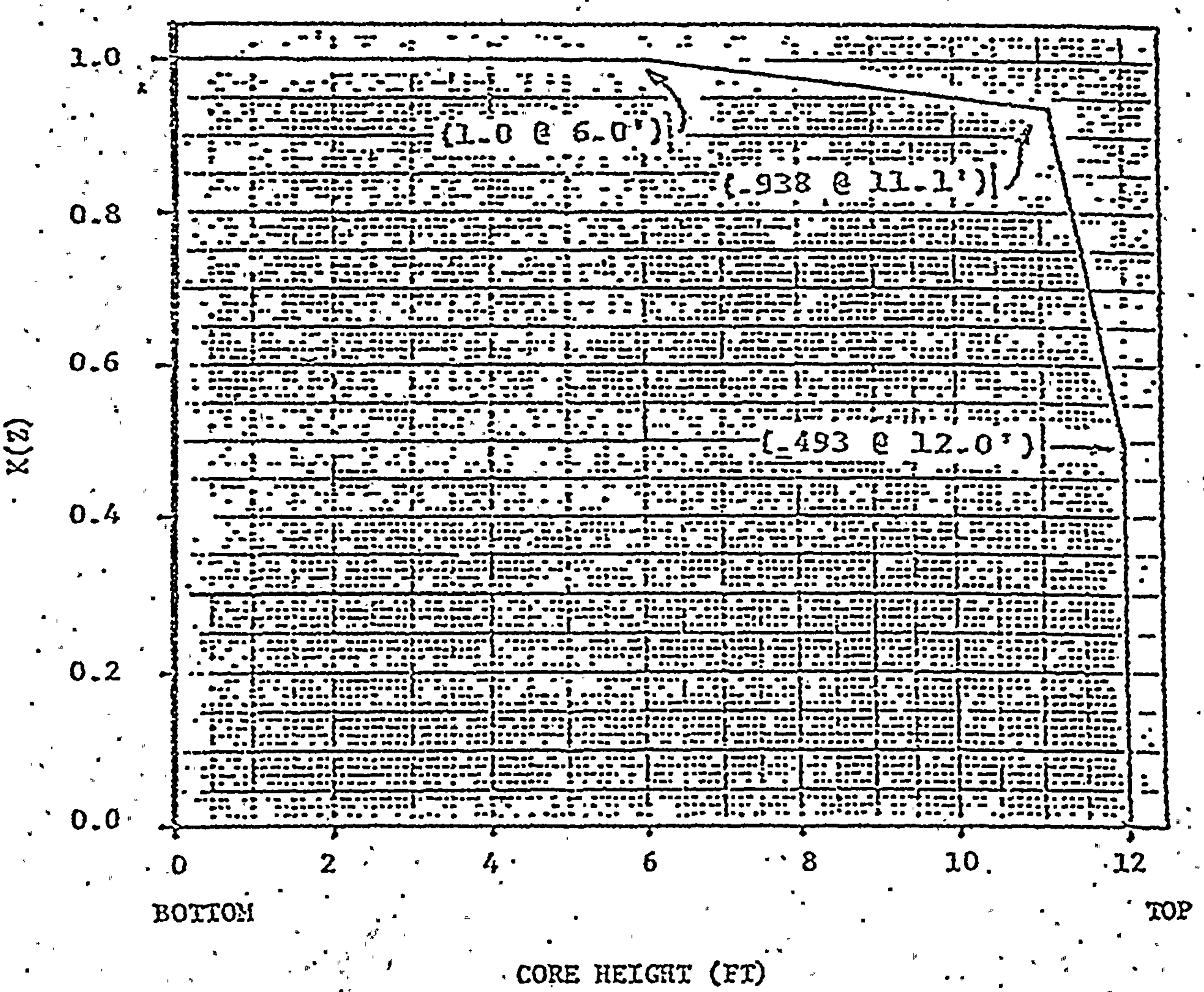

FIGURE 3.2-3b

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An upper bound envelope as defined by the normalized peaking factor axial dependence of Figures 3.2-3, a, & b has been determined to be consistent with the technical specifications on power distribution control as given in Section 3.2.

When an F_q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate experimental uncertainty allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. 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) although 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 in F_q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of start-up physics tests, at least once each full rated 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 initial loading provides confirmation of the basic nuclear

Flux Difference ($\Delta\phi$) and a reference value which corresponds to the full design power equilibrium value of Axial Offset (Axial Offset = $\Delta\phi$/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 assure that the F_q upper bound envelope as defined by Figures 3.2-3, a, & b. is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

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 part length+ rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal rated power operating position appropriate for the time in life. Control rods are usually withdrawn farther as burnup proceeds). This value, divided by the fraction of design power at which the core was operating is the design power value of the target flux difference. Values for all other core power levels are obtained by multiplying the design 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 ±5% 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 measuring the target flux difference every rated power month. For this reason, methods are permitted by Item 6c of Section 3.2 for updating the target flux differences. Figure B3.2-1 shows a typical construction of the target flux difference band at BOL and Figure B3.2-2 shows the typical variation of the full power value with burnup.

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 the required, periodic excore calibra-

+Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	DECL (Sec)
START	0.0
Rx Trip Signal	0.679
S. I. Signal	0.73
Acc. Injection	15.7
End of Bypass	27.94
End of Blowdown	29.26
Bottom of Core Recovery	46.81
Acc. Empty	59.86
Pump Injection	25.73

TABLE 2

LARGE BREAK

	DECL
Results	
Peak Clad Temp. °F	2189
Peak Clad Location Ft.	6.0
Local Zr/H_2O Rxn(max)%	8.047
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	11.93
Peaking Factor	2.10
Accumulator Water Volume (ft^3).	875 (per accumulator)

Cycle Analyzed	Cycle	Region
Unit 3 and Unit 4	6	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
Fastests Post Accident Initiation of Fan Coolers	26 secs

TABLE 3 (CONTINUED)
CONTAINMENT DATA
(DRY CONTAINMENT)

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

TABLE 3 (CONTINUED)
CONTAINMENT DATA
(DRY CONTAINMENT)

STRUCTURAL HEAT SINKS	THICKNESS (INCH)	AREA (FT^2)
Stainless Steel	2.1264	3704.0
Stainless Steel	0.1398	--
Concrete	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)
46.81	0.0	0.0
48.436	0.137	0.0018
54.517	34.55	0.4485
64.576	79.45	0.9914
73.176	84.08	1.047
94.076	92.62	1.128
110.676	240.8	1.518
127.976	274.6	1.561
165.376	284.0	1.480
206.676	291.2	1.387
252.776	298.9	1.277

TABLE 5

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT FOR LIMITING CASE DECLG (CD = 0.4) 22% STEAM GENERATOR TUBE PLUGGING

TIME (SEC)	MASS FLOW (LB/SEC)
0.0	0.0
0.01	2820.8
2.01	2367.1
4.01	2082.1
6.01	1878.9
8.01	1724.3
10.01	1599.2
15.01	1368.0
20.01	1213.1
25.01	1106.1
30.144	1034.7
31.711	1015.0

*For energy flow, multiply mass flow by an enthalpy of 59.62 BTU/LB

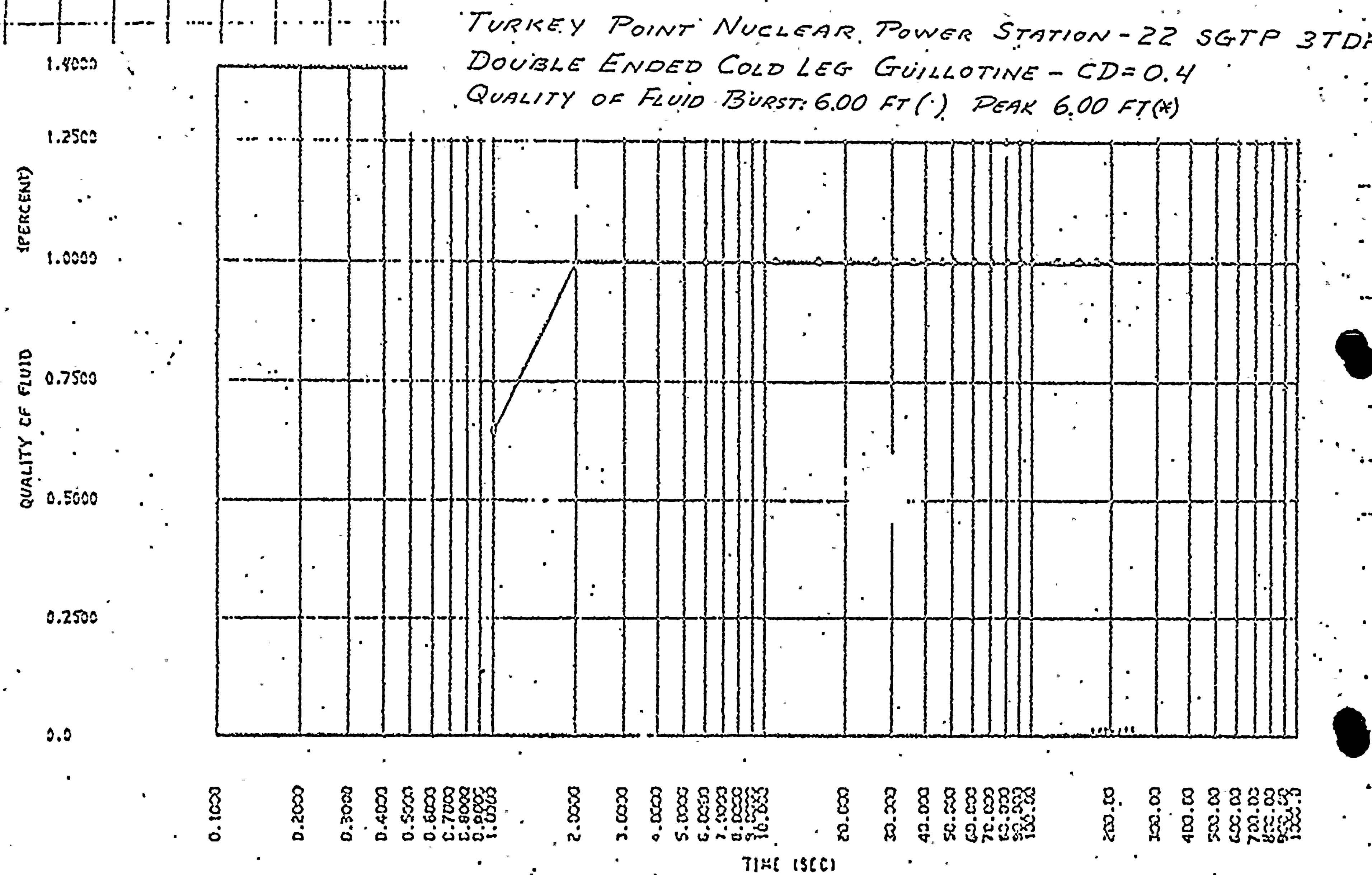

FIG. 1 FLUID QUALITY - DECLG (CD = 0.4)

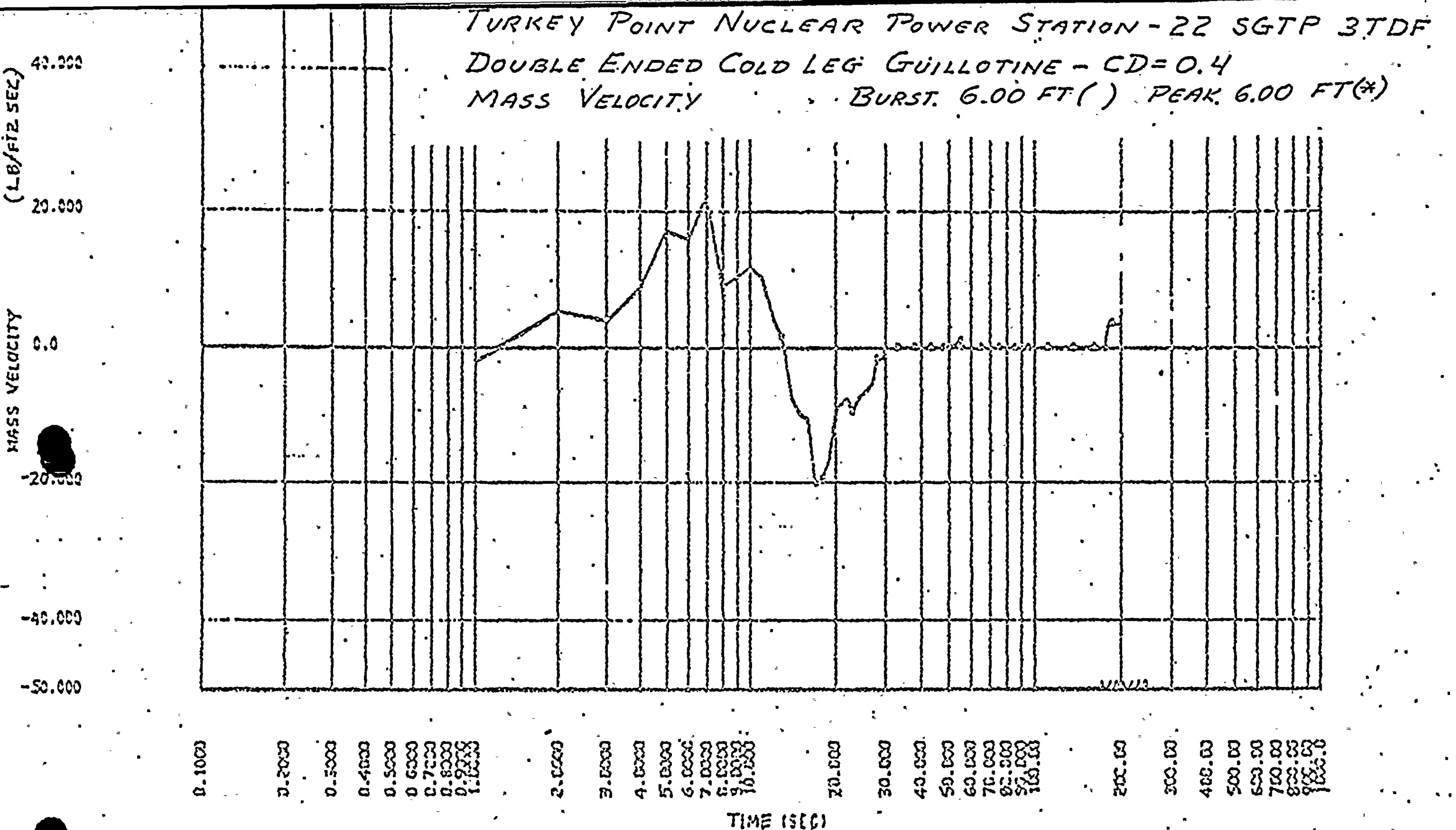

FIG. 2 MASS VELOCITY - DECLG (CD=0.4)

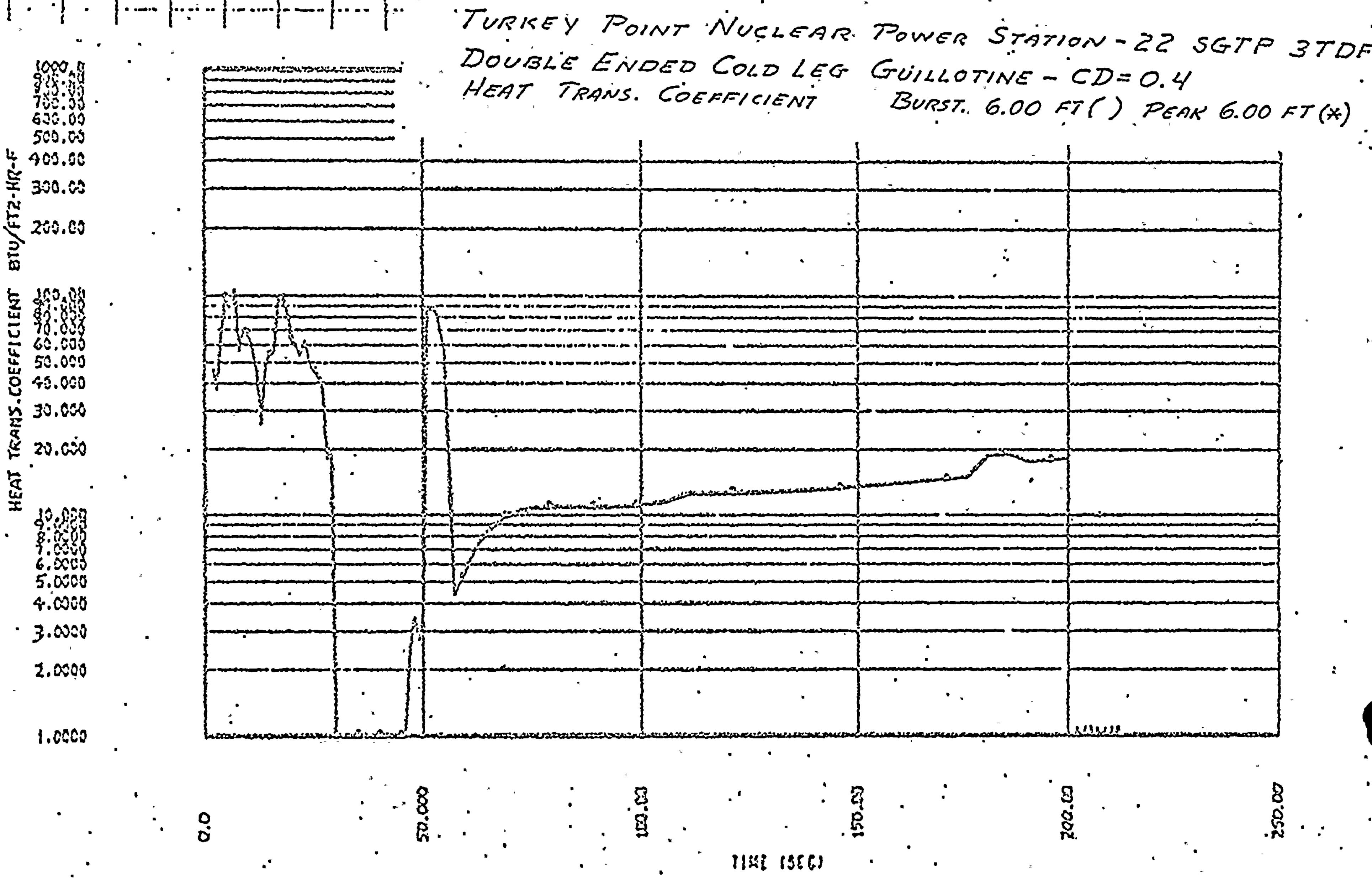

FIG-3 HEAT TRANSFER COEFFICIENT - DECLG (CD = 0.4

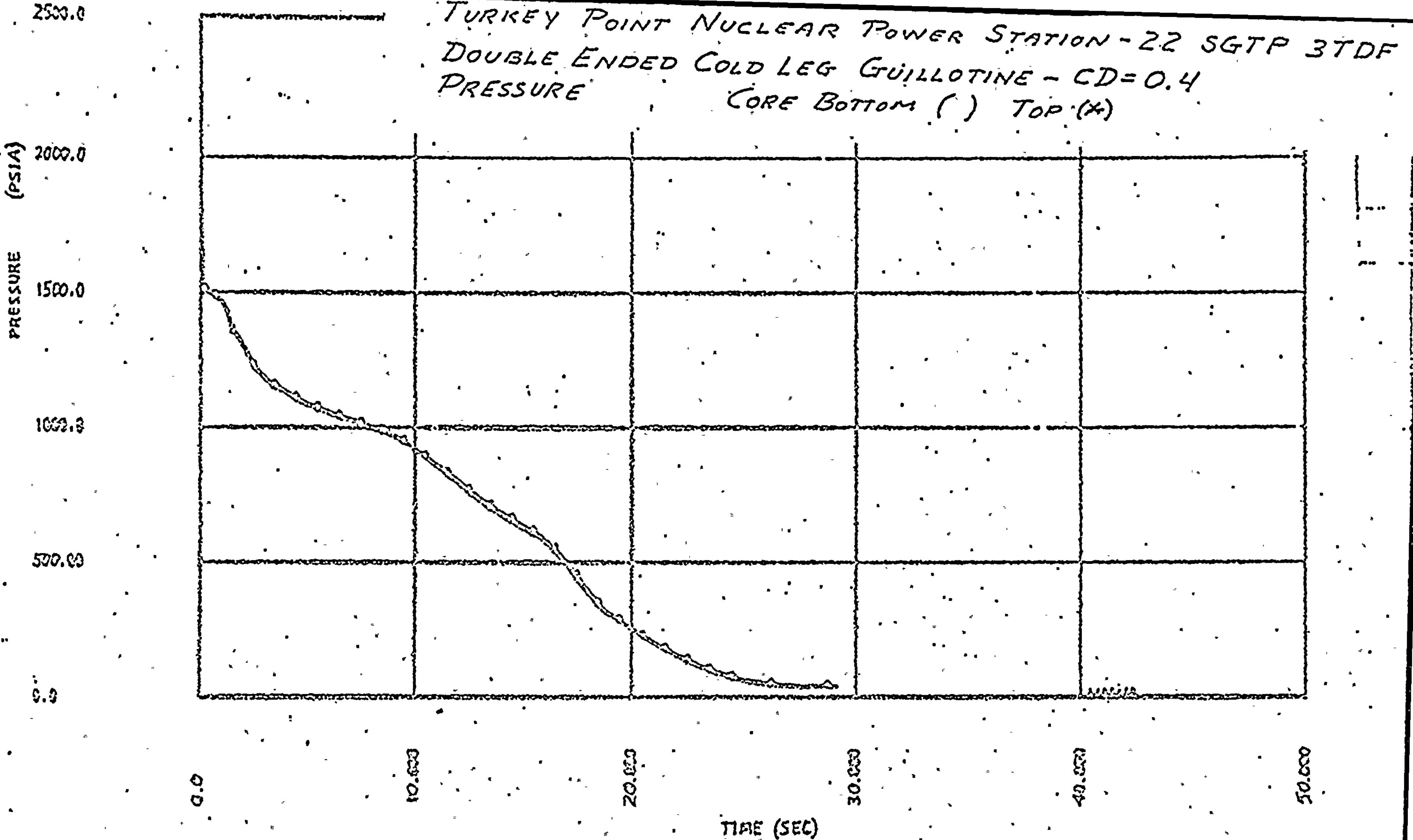

FIG-4 PRESSURE - DECLG (CD = 0.4)

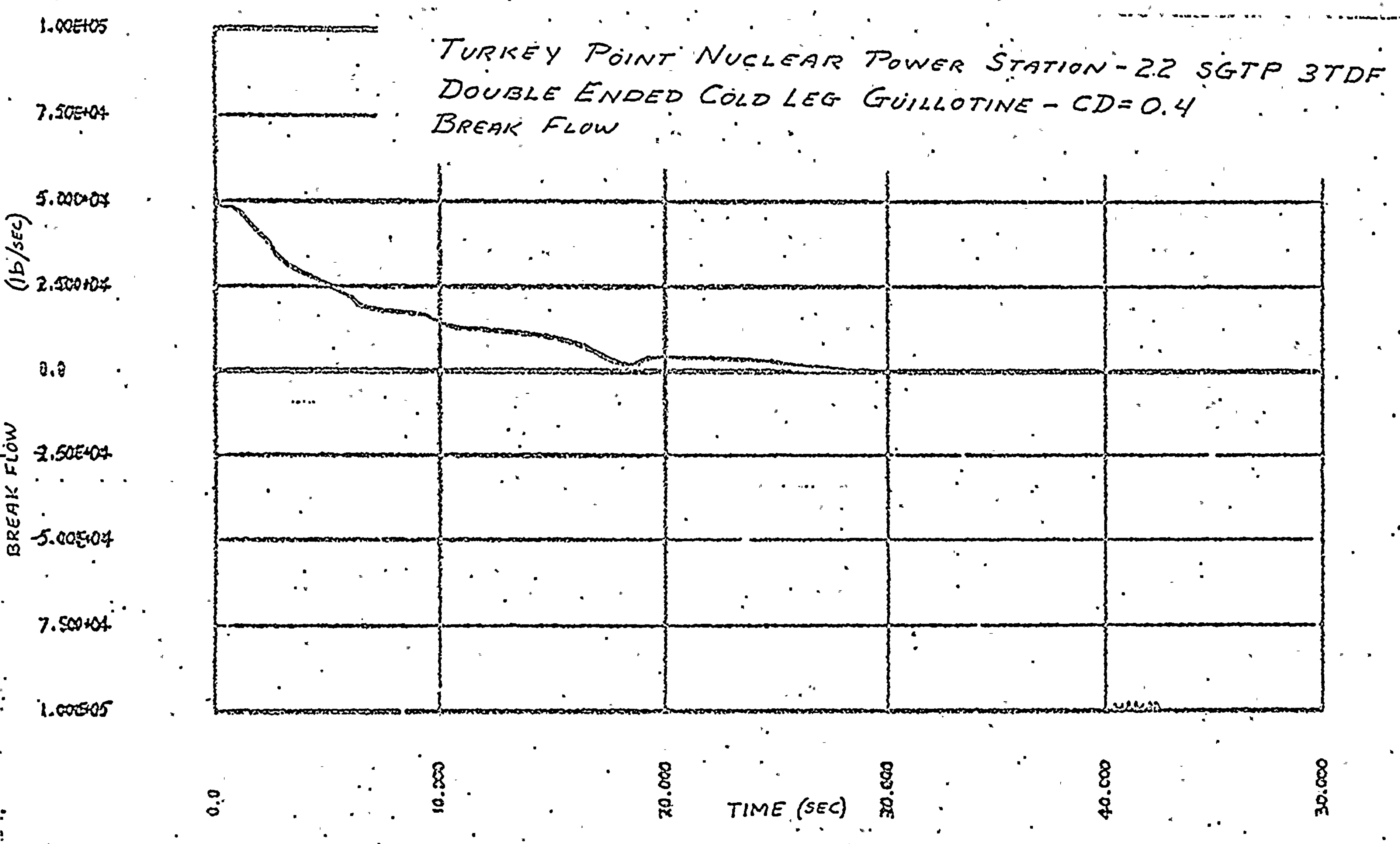

FIG-5 BREAK FLOW RATE - DECLG (CD=0.4)

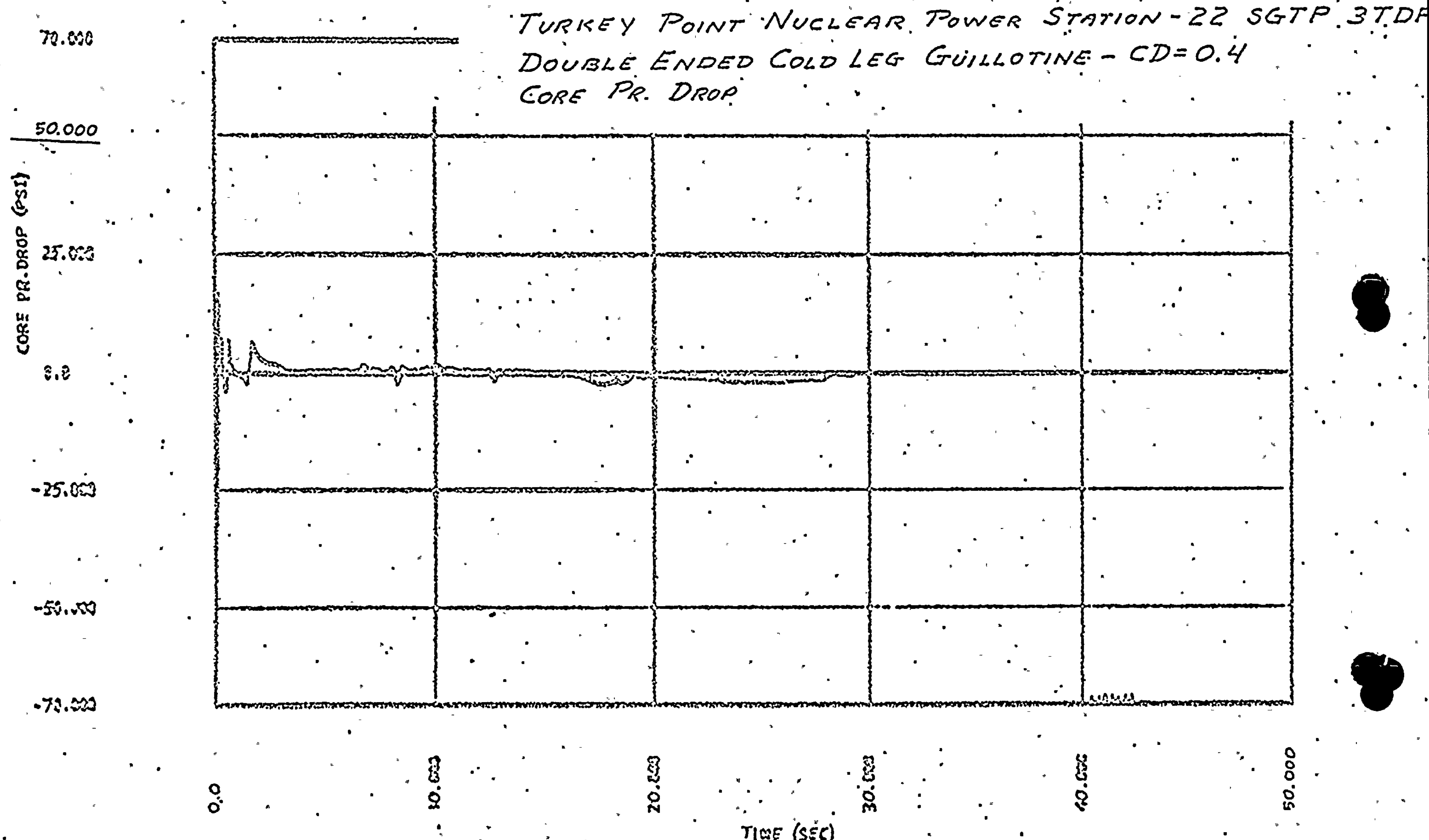

FIG-6 CORE PRESSURE DROP - DECLG (CD=0.4)

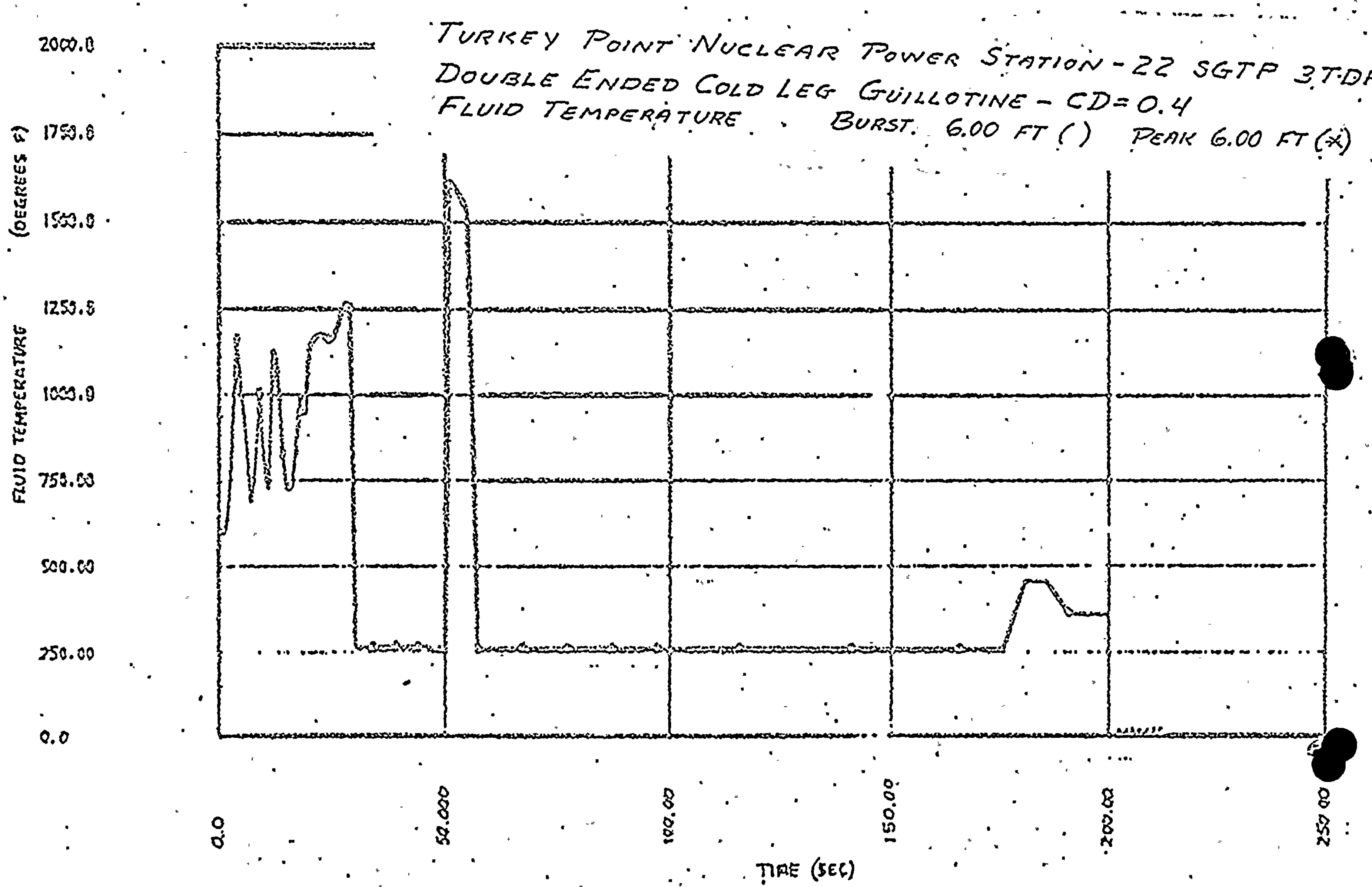

FIG-8 FLUID TEMPERATURE - DECLG (CD=0.4)

TURKEY POINT NUCLEAR POWER STATION - 22 SGTP 3TDF
DOUBLE ENDED COLD LEG GUILLOTINE - CD=0.4
CLAD AVE. TEMP. HOT ROD BURST 6.00 FT( ) PEAK 6.00 FT(*)

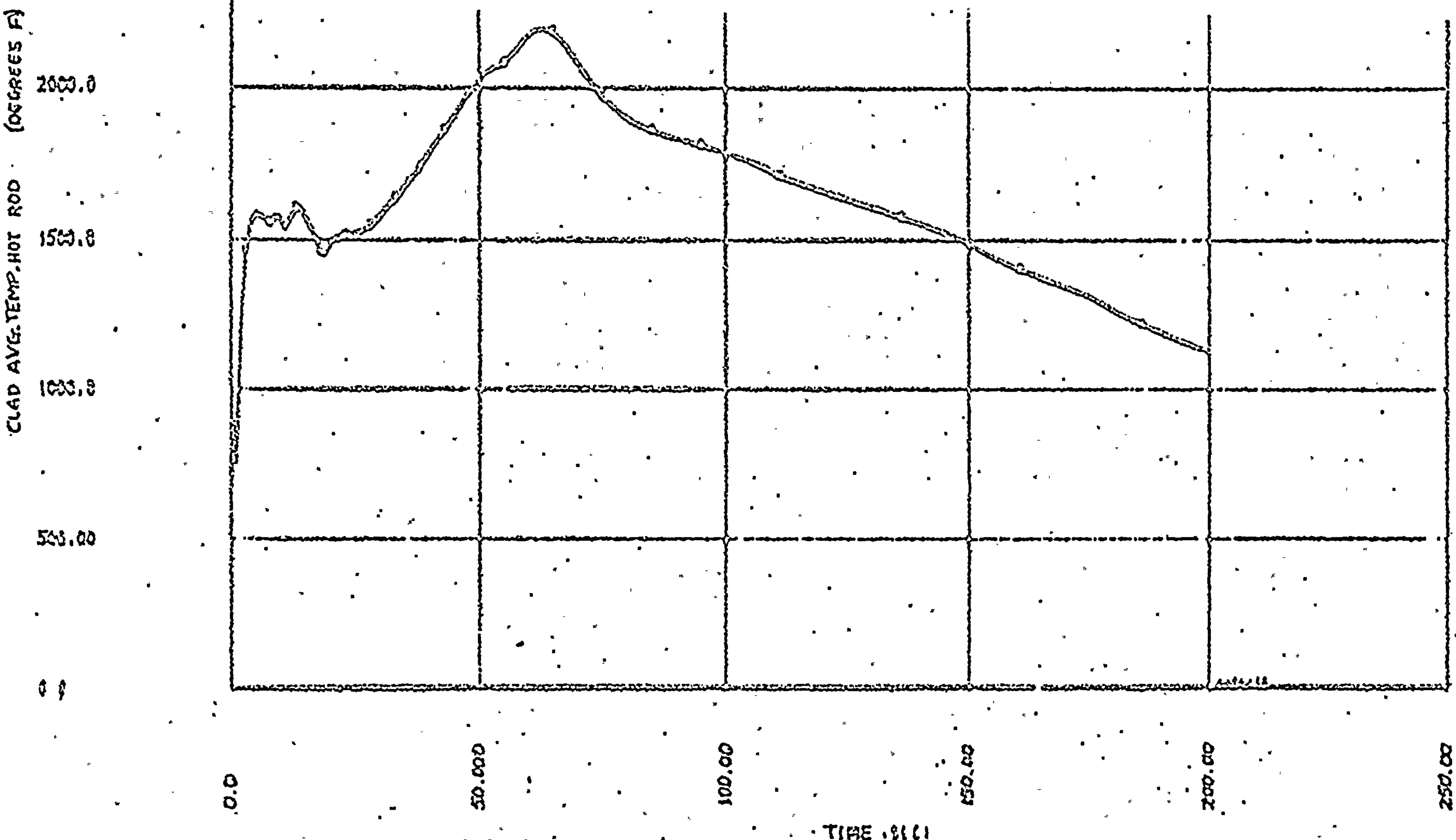

FIG - 7 PEAK CLAD TEMPERATURE - DECLG (CD=0.4)

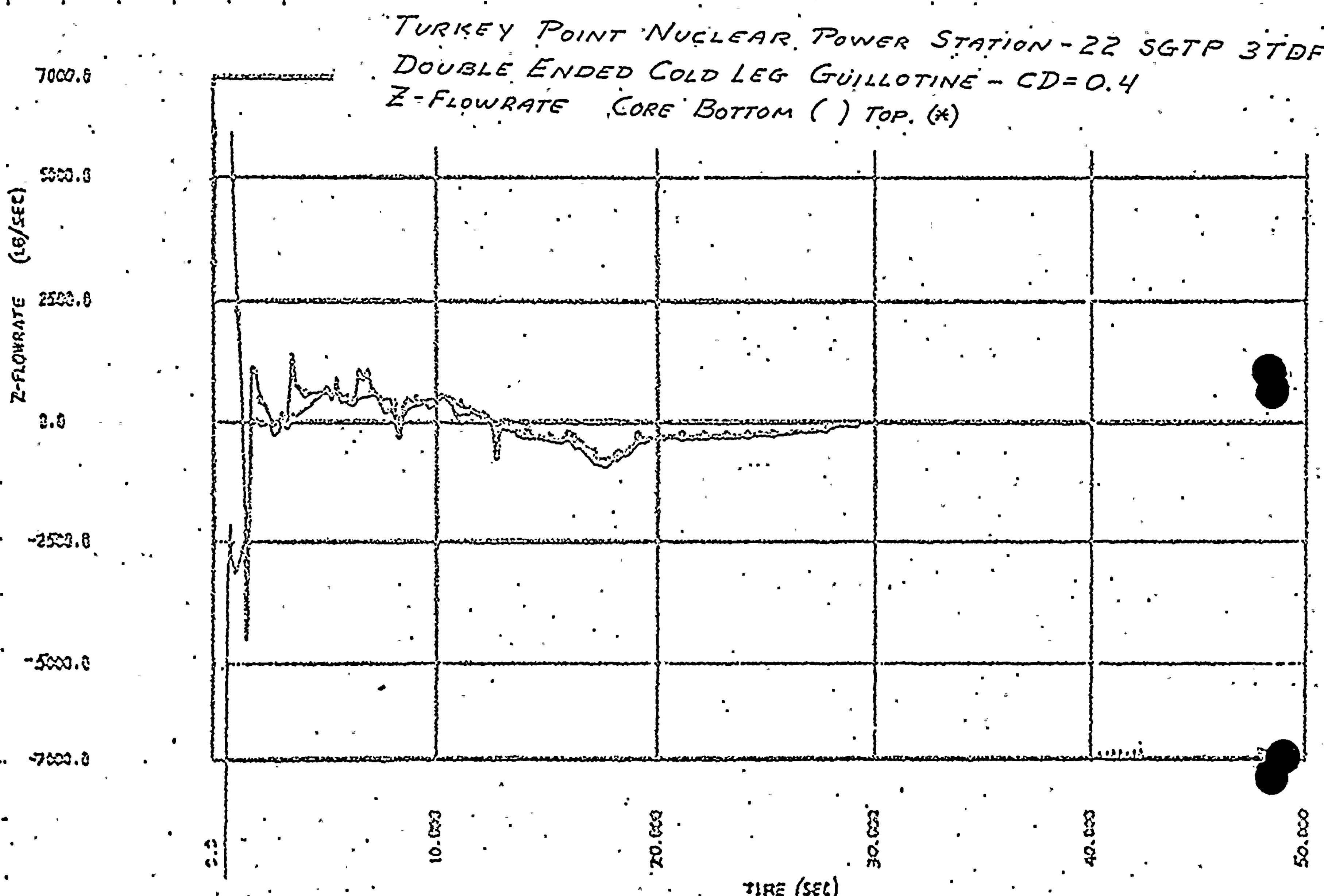

FIG-9 CORE FLOW - TOP AND BOTTOM - DECLG (CD=0.4)

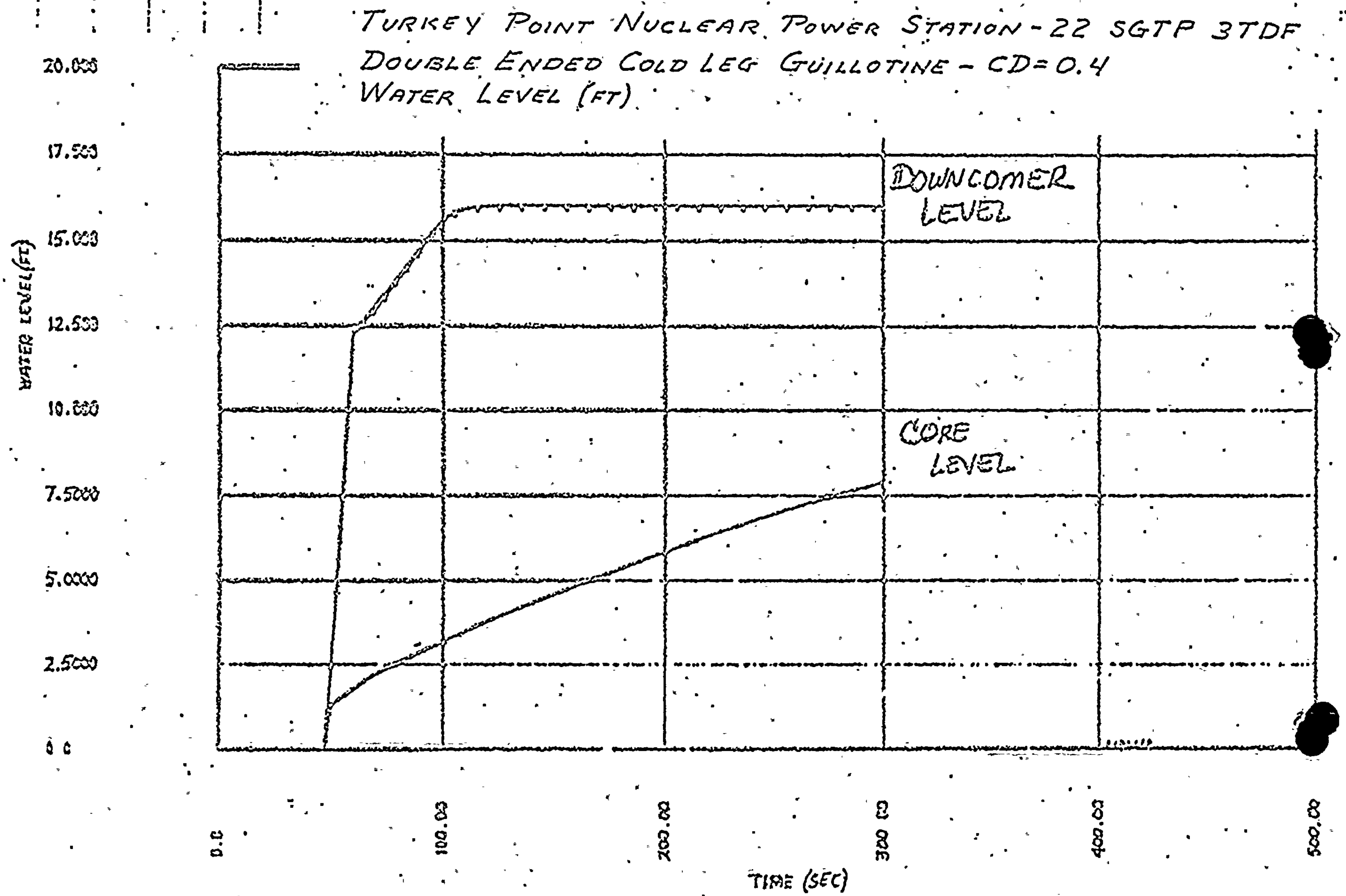

FIG - 10a REFLOOD TRANSIENT - DECLG (CD = 0.4)

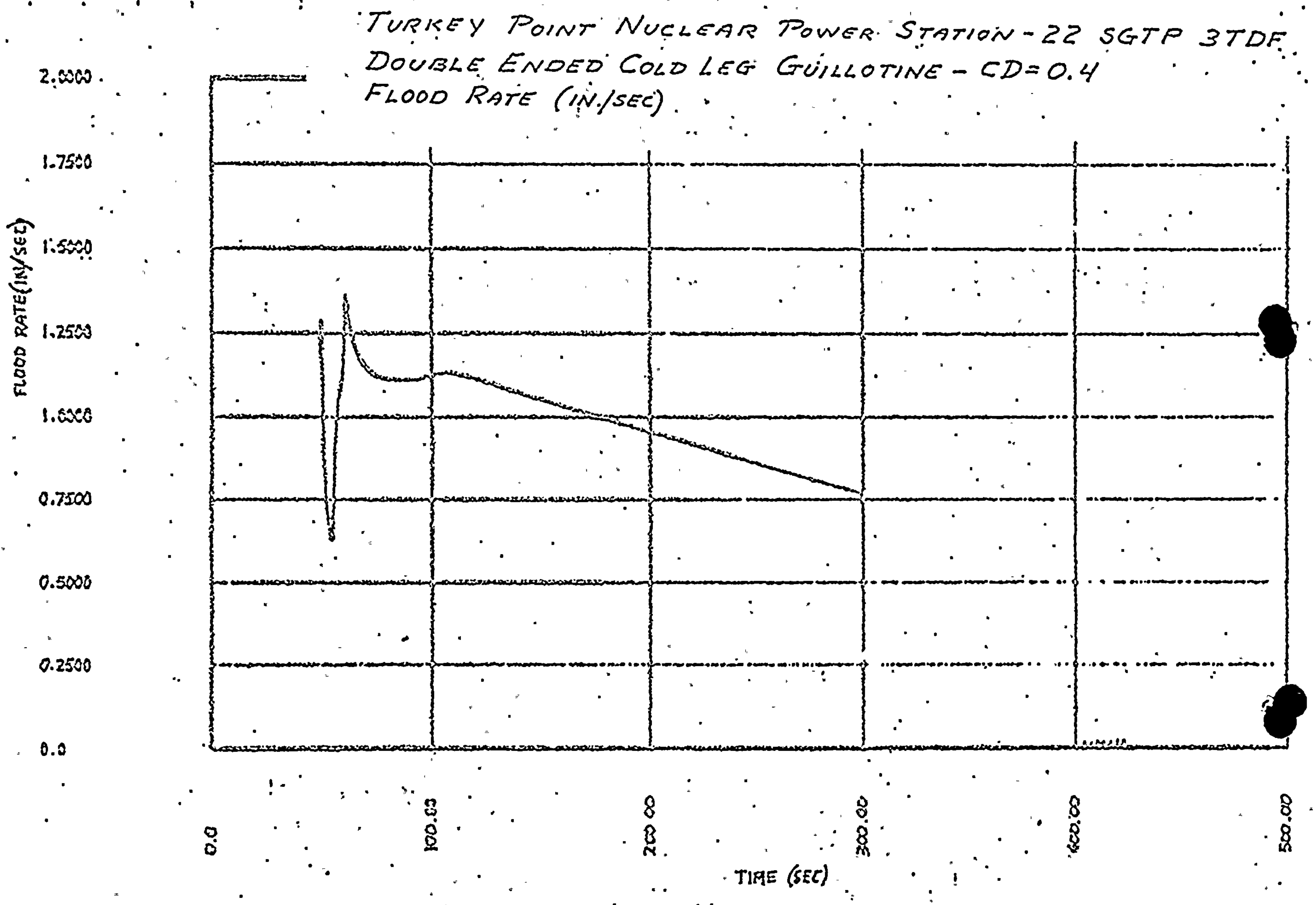

FIG - 10 b REFLOOD TRANSIENT - DECLG (CD=0.4)

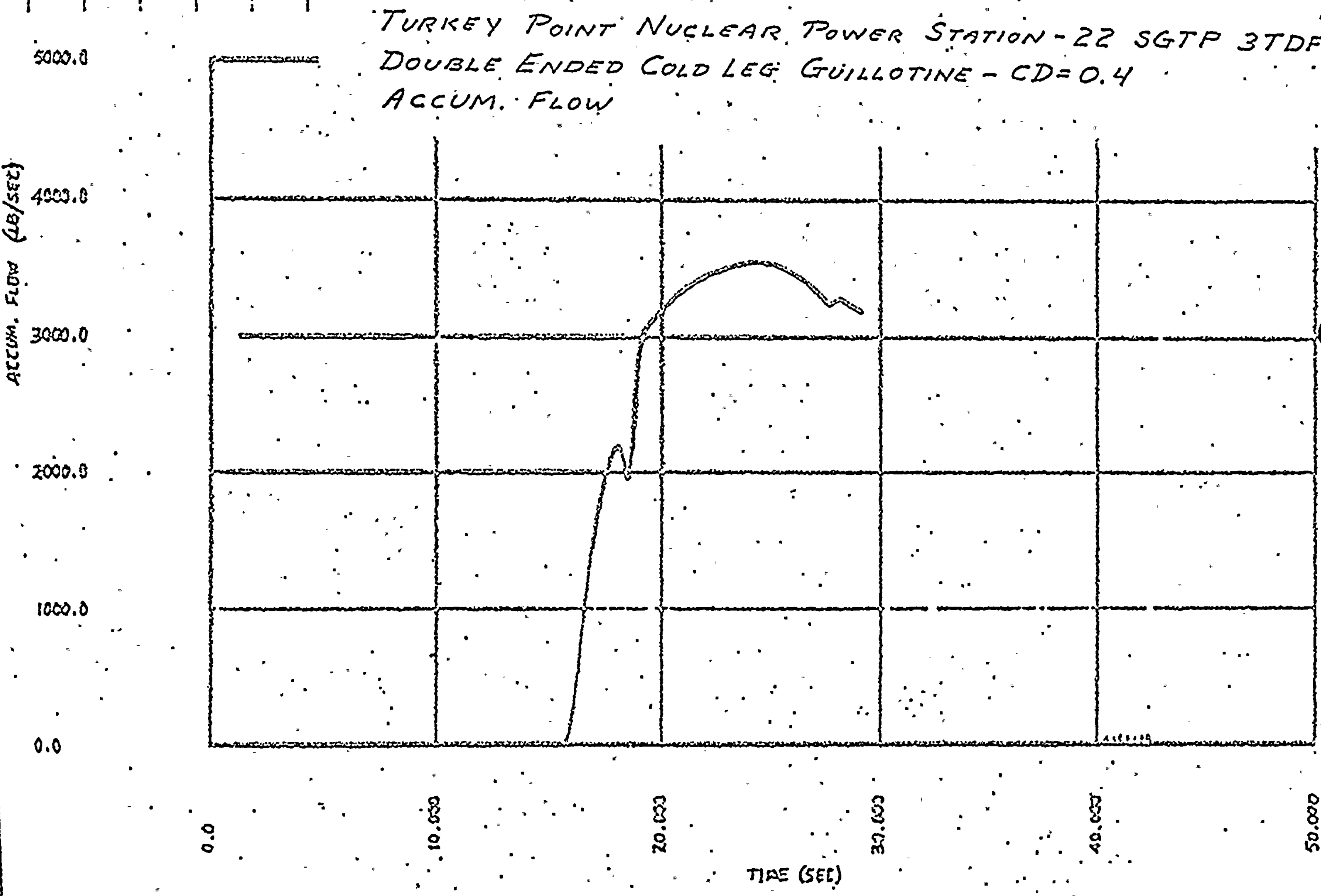

FIG-11 ACCUMULATOR FLOW (BLOWDOWN) - DECLG (CD=0.

(19)

TURKEY POINT NUCLEAR POWER STATION - 22 SGTP 3TDF
DOUBLE ENDED COLD LEG GUILLOTINE - CD=0.4

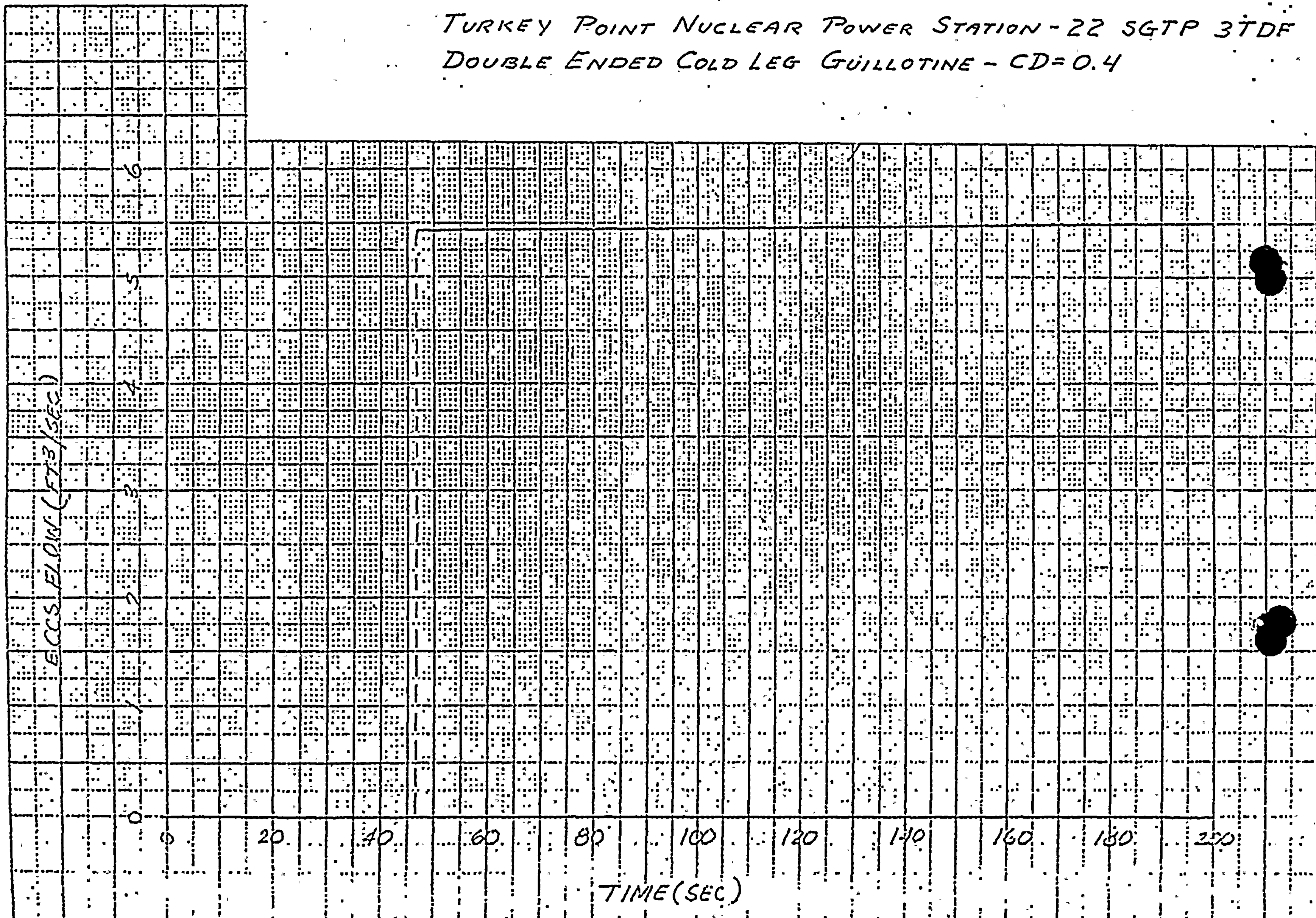

FIGURE-12 PUMP ECCS FLOW (REFLOOD) - DECLG (CD=0.4)

(20)

TURKEY POINT NUCLEAR POWER STATION - 22 SGTP3TD
DOUBLE ENDED COLD LEG GUILLOTINE - CD = 0.4.

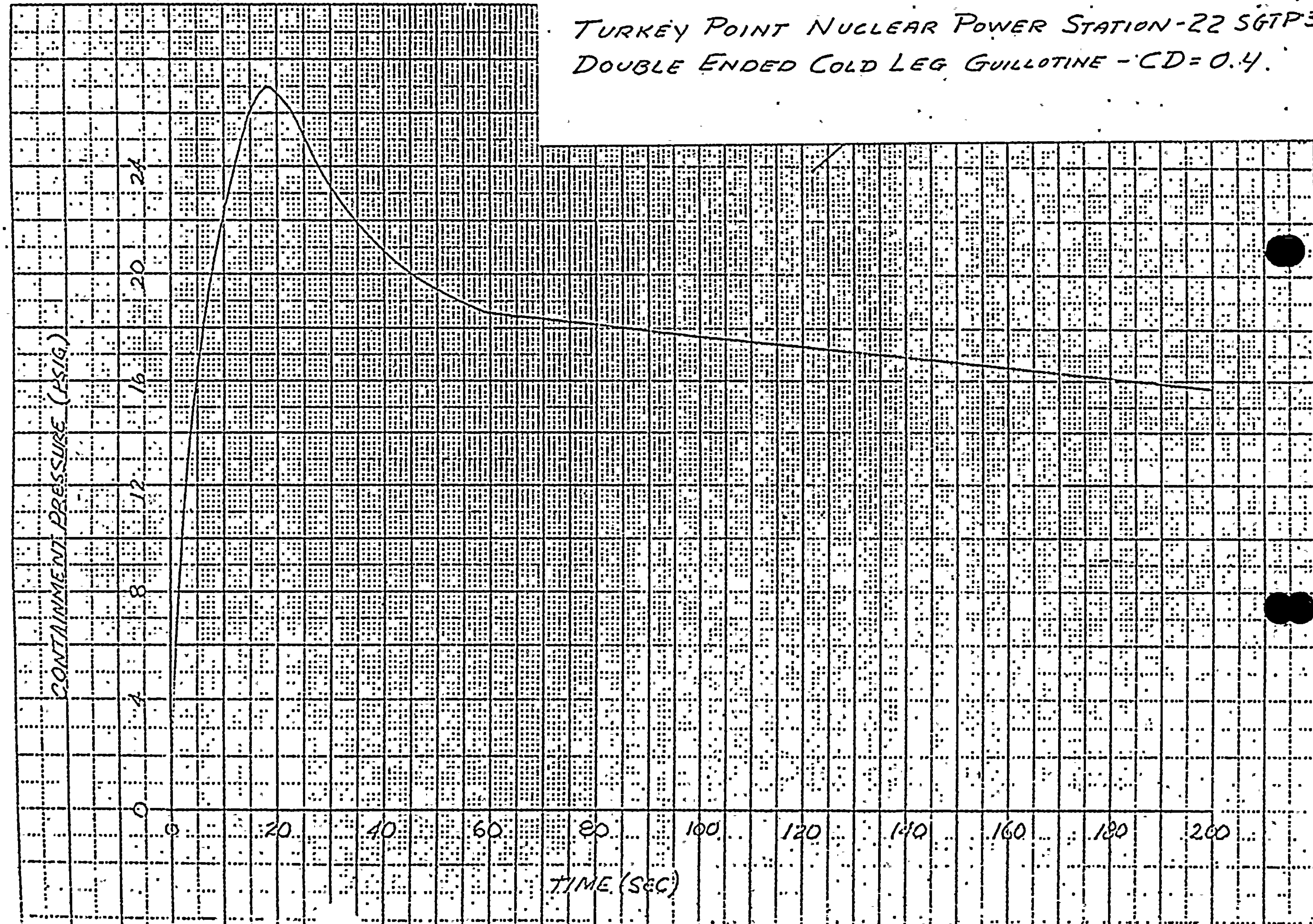

FIGURE - 13 CONTAINMENT PRESSURE - DECLC (CD = 0.4)

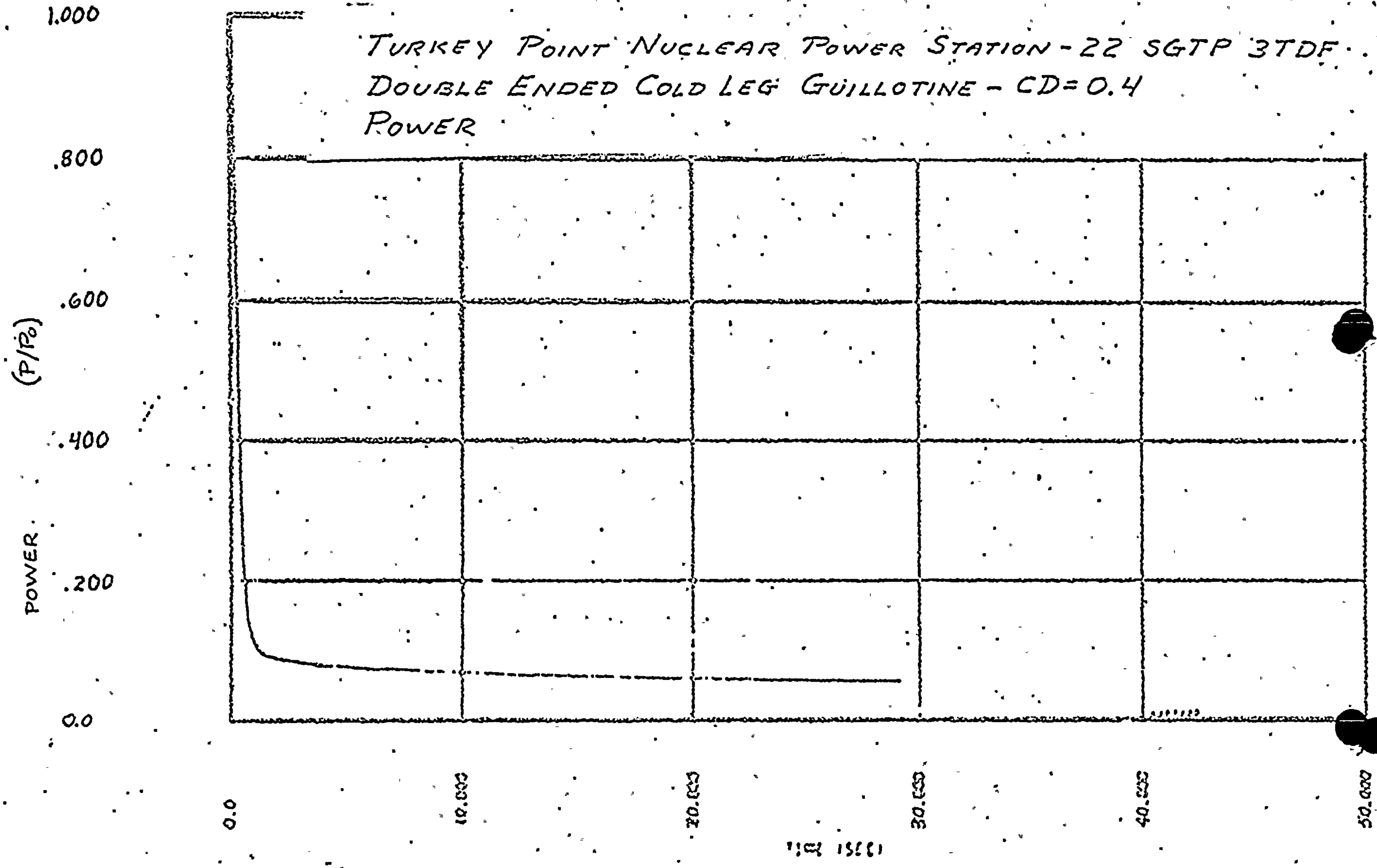

FIGURE -14 CORE POWER TRANSIENT - DECLG (CD=0.4)

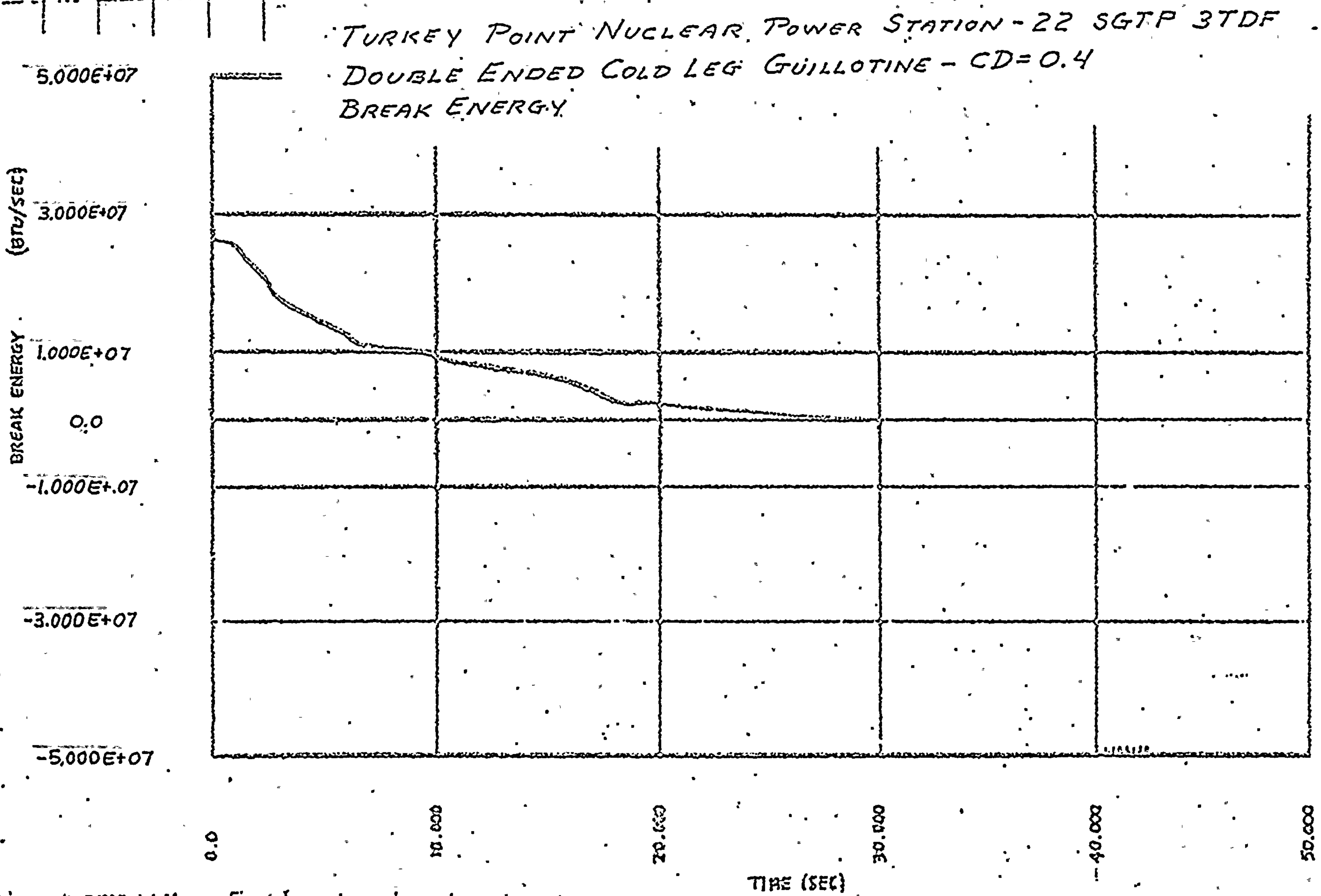

FIGURE 15 BREAK ENERGY RELEASED TO CONTAINMENT DECLG (

(23)

TURKEY POINT NUCLEAR POWER STATION - 22 SGTP 3TDF
DOUBLE ENDED COLD LEG GUILLOTINE - CD=0.4

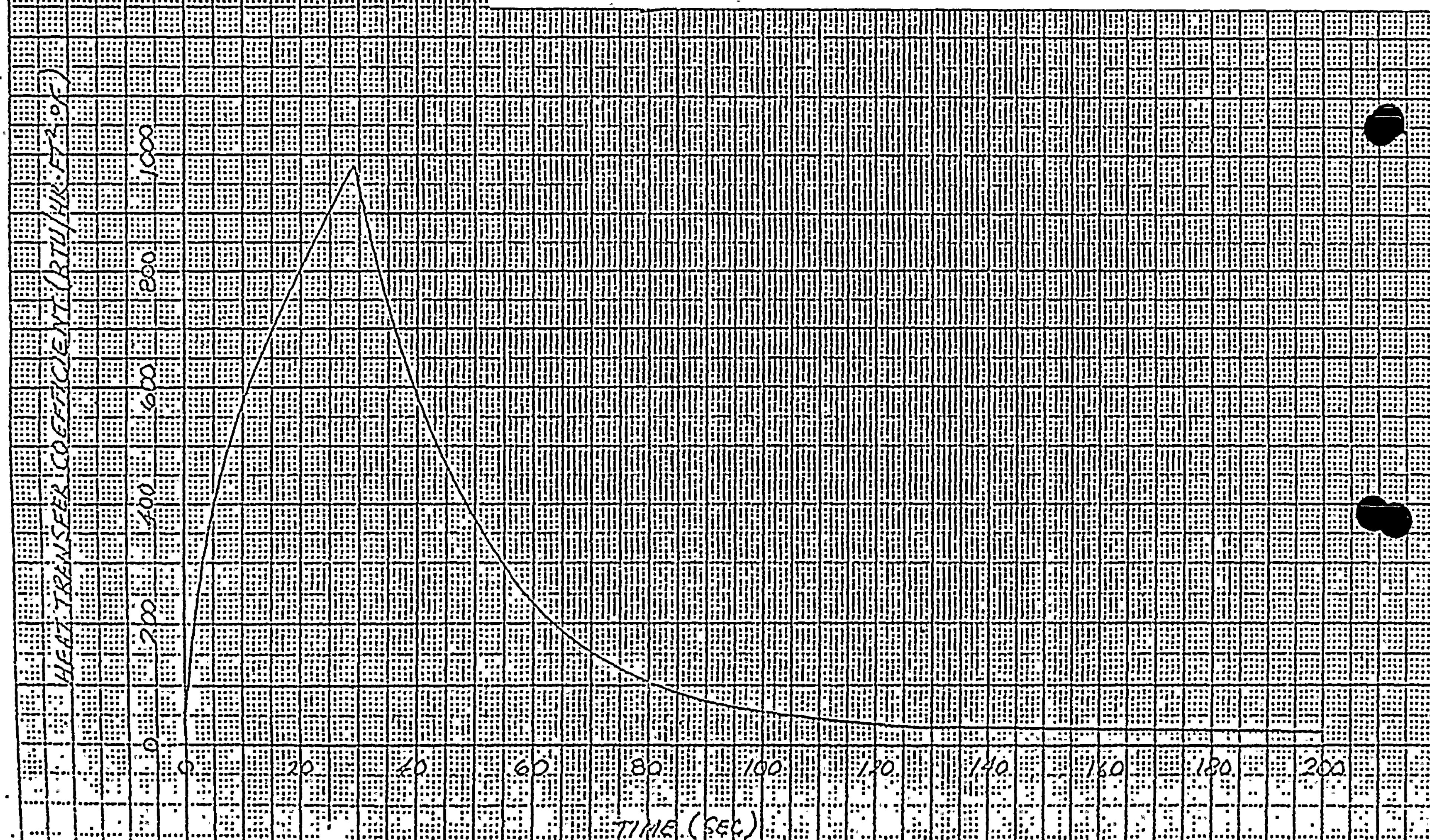

FIGURE 16 CONTAINMENT WALL HEAT TRANSFER COEFFICIENT - DECLG (CD=0.4)

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

E. A. Adomat, being first duly sworn, deposes and says:

That he is Executive 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

E. A. Adomat

E. A. Adomat

Subscribed and sworn to before me this

18th day of May, 1979

Bettie Brittain

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

NOTARY PUBLIC STATE OF FLORIDA AT LARGE
MY COMMISSION EXPIRES MARCH 27, 1982

My commission expires: ____________________