C 08/14/78

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DOCDATE: 08/09/78
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DOCTYPE: LETTER NOTARIZED: YES

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LTR 3 ENCL 40

SUBJECT:
FORWARDING LIC NOS DPR-31 & 41 APPL FOR AMEND: TECH SPEC PROPOSED CHANGE CONCERNING REVISION TO THE ECCS COOLING PERFORMANCE AS LISTED AND REQUESTING PROVISIONS OF NRC"S 06/07/78 ORDER BE DELETED... NOTARIZED 08/09/78.

PLANT NAME: TURKEY PT #3
TURKEY PT #4

REVIEWER INITIAL: XJM
DISTRIBUTER INITIAL: M

****************** DISTRIBUTION OF THIS MATERIAL IS AS FOLLOWS ******************

GENERAL DISTRIBUTION FOR AFTER ISSUANCE OF OPERATING LICENSE.
(DISTRIBUTION CODE A001)

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INTERNAL:
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HANAUER**W/ENCL
AD FOR SYS & PROJ**W/ENCL
REACTOR SAFETY BR**W/ENCL
EEB**W/ENCL
J. MCGOUGH**W/ENCL
NRC PDR**W/ENCL
OELD**LTR ONLY
CORE PERFORMANCE BR**W/ENCL
ENGINEERING BR**W/ENCL
PLANT SYSTEMS BR**W/ENCL
EFFLUENT TREAT SYS**W/ENCL

EXTERNAL:
LPDR'S
MIAMI, FL**W/ENCL
TERA**W/ENCL
NSIC**W/ENCL
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App 2

ccp

DISTRIBUTION: LTR 40 ENCL 39
SIZE: 3P+29P

CONTROL NBR: 782190052

************************************ THE END ************************************

FLORIDA POWER & LIGHT COMPANY

August 9, 1978
L-78-264

Director of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Director
Division of Operating Reactors
U. S. Nuclear Regulatory 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

An Order for Modification of License, dated June 7, 1978, was issued by the Commission for Turkey Point Units 3 and 4. The Order amended Facility Operating Licenses DPR-31 and DPR-41 by adding the following provisions:

(1) As soon as possible, the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with the Westinghouse Evaluation Model, approved by the NRC staff and corrected for the errors described herein.

(2) Until further authorization by the Commission, the Technical Specification limit for total nuclear peaking factor (F_q) for the facility shall be limited to 2.03 and 1.91 for Unit Nos. 3 and 4, respectively.

(3) Until further authorization by the Commission, the licensee shall conduct the operating surveillance program described in its letter of April 10, 1978.

Westinghouse has reevaluated ECCS cooling performance in accordance with a corrected evaluation model that we understand has been approved by the NRC staff. The reevaluation supports a Technical Specification limit for total nuclear peaking factor (F_q) of 2.03 for both Units 3 and 4, assuming 25% steam generator tube plugging. Based on Westinghouse core physics calculations, the maximum values of F_q that could occur for the remainder of Unit 3, Cycle 5 and for the upcoming Unit 4, Cycle 5 are 2.02 and 2.00, respectively. Because the maximum predicted F_q values are less than the maximum allowable value derived from the corrected ECCS evaluation, the augmented surveillance procedures discussed in Florida Power & Light Company (FPL) letter L-78-127 of April 10, 1978 will not be necessary during Cycle 5 of both Units 3 and 4.

As a result of the ECCS reevaluation performed in compliance with the Commission's Order of June 7, FPL 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. We also request that the provisions of the June 7 Order be deleted. The proposed amendments to Appendix A are described below and shown on the accompanying Technical Specifications bearing the date of this letter in the lower right hand corner.

Page 3.2-3

Specification 3.2.6.a (Power Distribution Limits) is revised to reflect the results of the corrected Westinghouse ECCS evaluation.

Figures 3.2-3 and 3.2-3a

The current figures are deleted and replaced with a single Figure 3.2-3 (Hot Channel Factor Normalized Operating Envelope) that is valid for an F_q of 2.03 and steam generator tube plugging $\leq 25\%$.

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

Bases Section B3.2 (Control and Power Distribution Limits) is revised to delete superseded F_q values.

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,

Robert E. Uhrig
Vice President

REU/MAS/cpc

Attachment

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

reactivity insertion upon ejection greater than 0.3% $\Delta k/k$ 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. <u>CONTROL ROD POSITION INDICATION</u>

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. <u>POWER DISTRIBUTION LIMITS</u>

a. Hot channel factors:

With steam generator tube plugging $\leq$ 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 3.2-3; 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 ratio of 2.03 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,

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

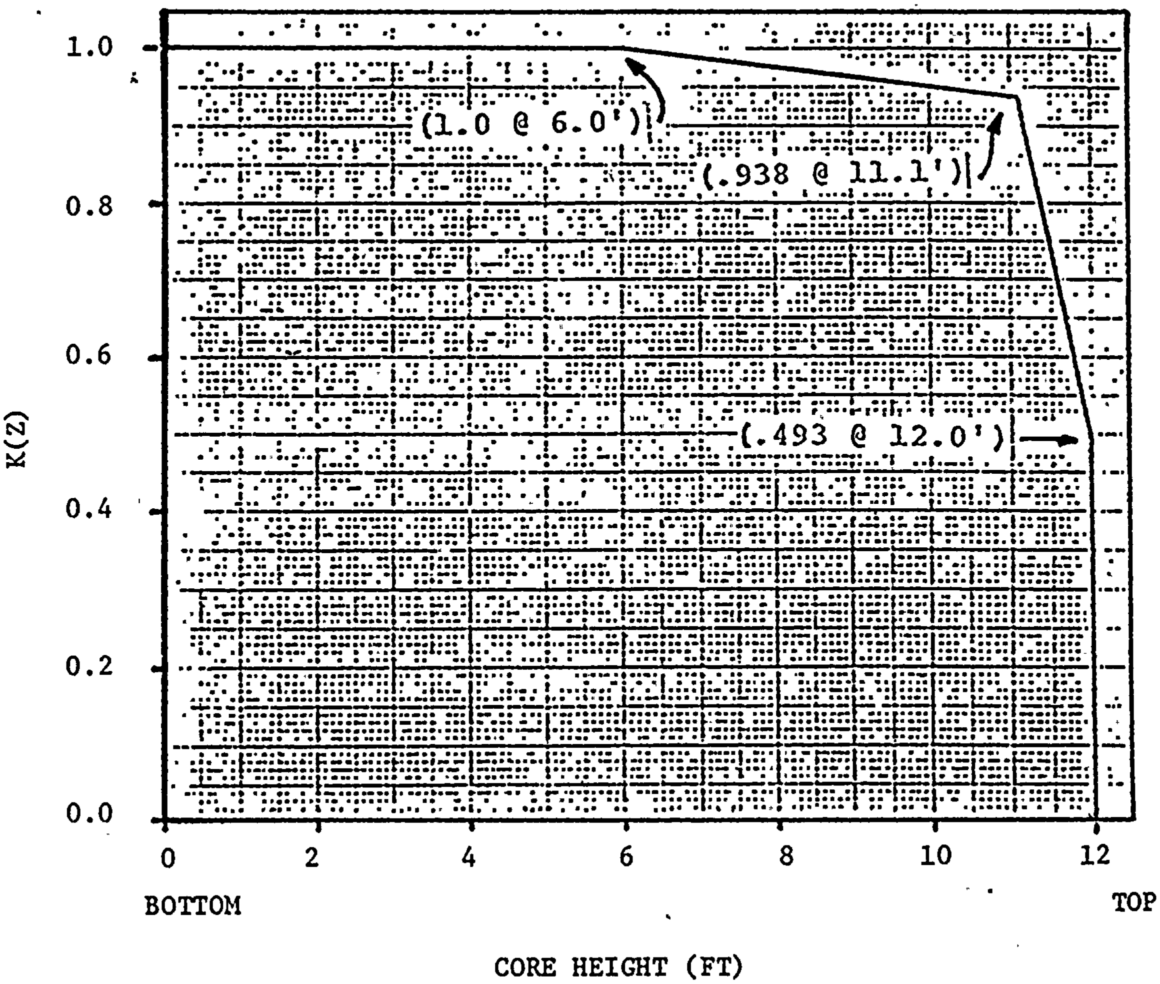

FIGURE 3.2-3

An upper bound envelope as defined by the normalized peaking factor axial dependence of Figure 3.2-3 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 Figure 3.2-3 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 $\pm5\%$ 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-

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	DECL (Sec)
START	0.0
Rx Trip Signal	0.535
S. I. Signal	0.77
Acc. Injection	16.1
End of Bypass	27.60
End of Blowdown	27.79
Bottom of Core Recovery	46.03
Acc. Empty	60.14
Pump Injection	25.77

TABLE 2

LARGE BREAK

	DECL
Results	
Peak Clad Temp. °F	2173
Peak Clad Location Ft.	6.25
Local Zr/H_2O Rxn(max)%	7.678
Local Zr/H_2O Location Ft.	6.25
Total Zr/H_2O Rxn %	<0.3
Hot Rod Burst Time sec	32.0
Hot Rod Burst Location Ft.	6.25
Calculation	
Core Power Mwt 102% of	2200
Peak Linear Power kw/ft 102% of	11.53
Peaking Factor	2.03
Accumulator Water Volume (ft^3)	875 (per accumulator)

Cycle Analyzed	Cycle
Unit 3 and Unit 4	5

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
Paint	0.006996	-
Carbon Steel	0.006996	76800.0
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	MASS FLOWRATE (LBm/SEC)	ENERGY FLOWRATE (10^5 BTU/SEC)
46.028	0.0	0.0
47.653	0.2476	0.0032
53.696	34.83	0.4511
63.653	89.23	1.108
76.653	92.04	1.142
91.953	187.08	1.416
108.153	261.56	1.579
125.253	273.11	1.562
162.253	280.75	1.482
202.853	287.45	1.392

TABLE 5

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT
FOR LIMITING CASE DECLG (CD=0.5) 25%
STEAM GENERATOR TUBE PLUGGING

TIME (SEC)	MASS FLOWRATE (LBm/SEC)
0.0	0.0
0.02	2770.1
2.00	2349.7
4.00	2083.2
6.00	1890.7
8.00	1748.9
10.0	1634.1
15.0	1417.4
20.0	1257.6
25.0	1157.3
30.0	1079.1
35.0	1014.1
37.9	981.7

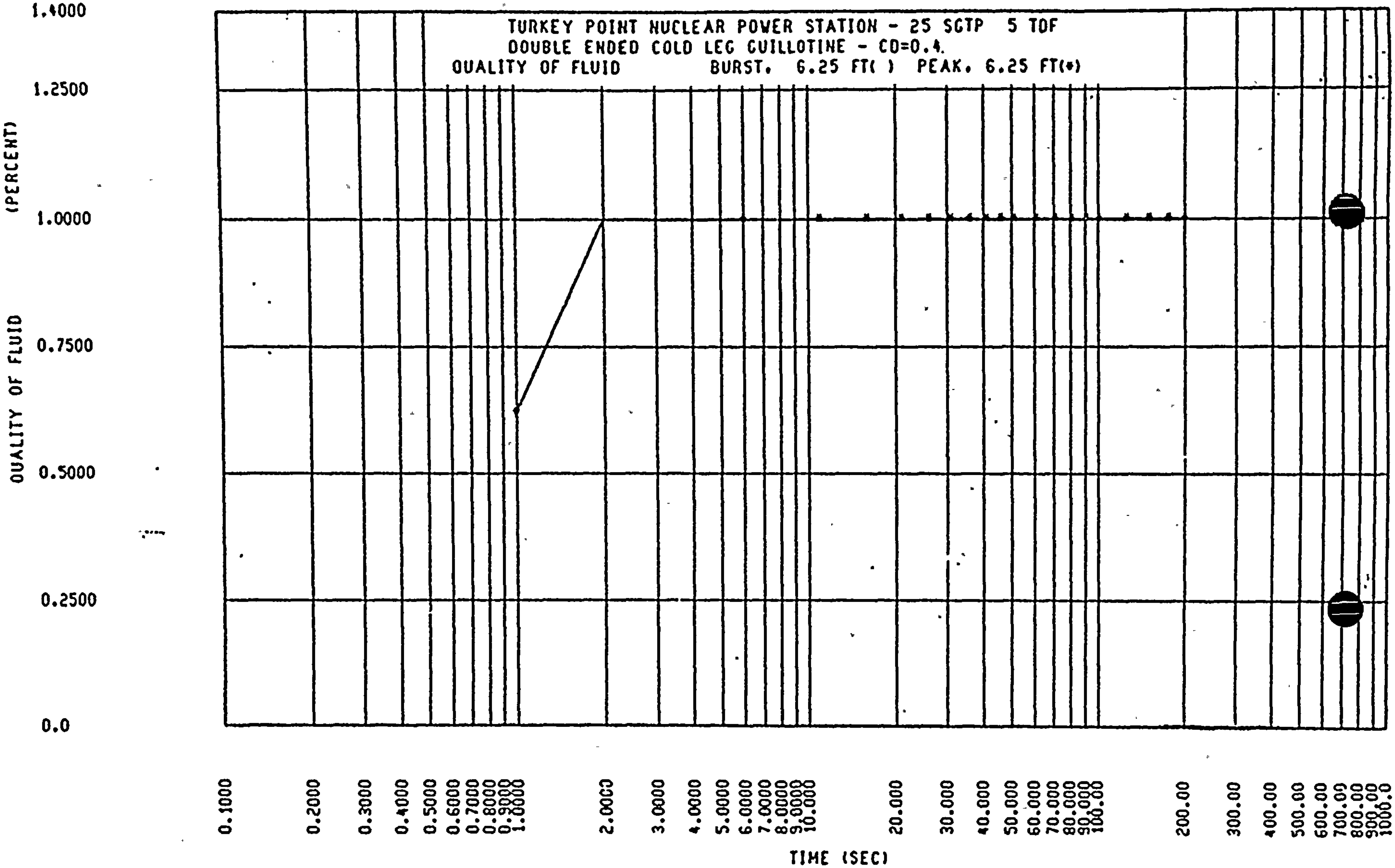

FIGURE 1 - FLUID QUALITY - DECLG (CD=0.4)

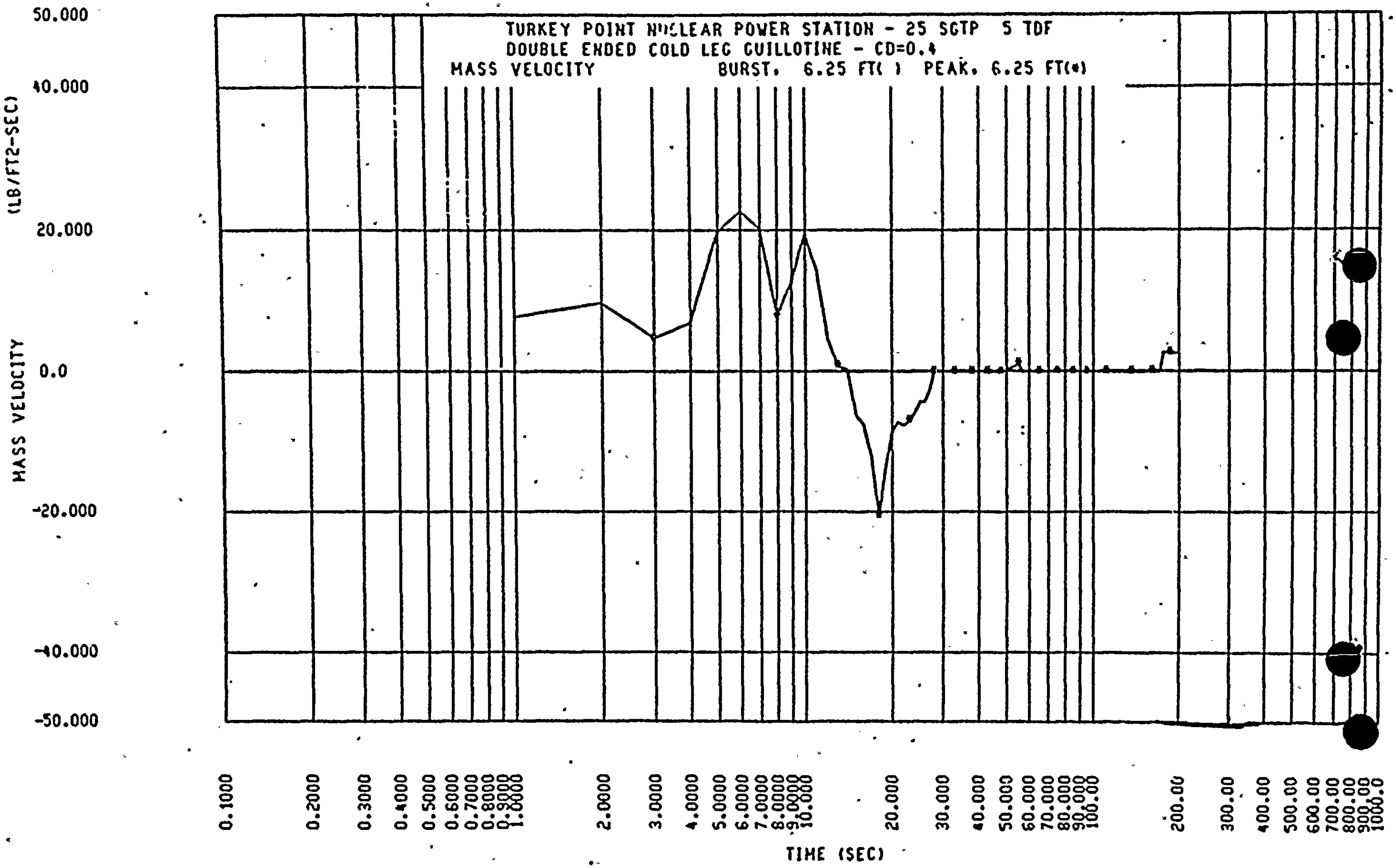

FIGURE 2 - MASS VELOCITY - DECLG (CD = 0.4)

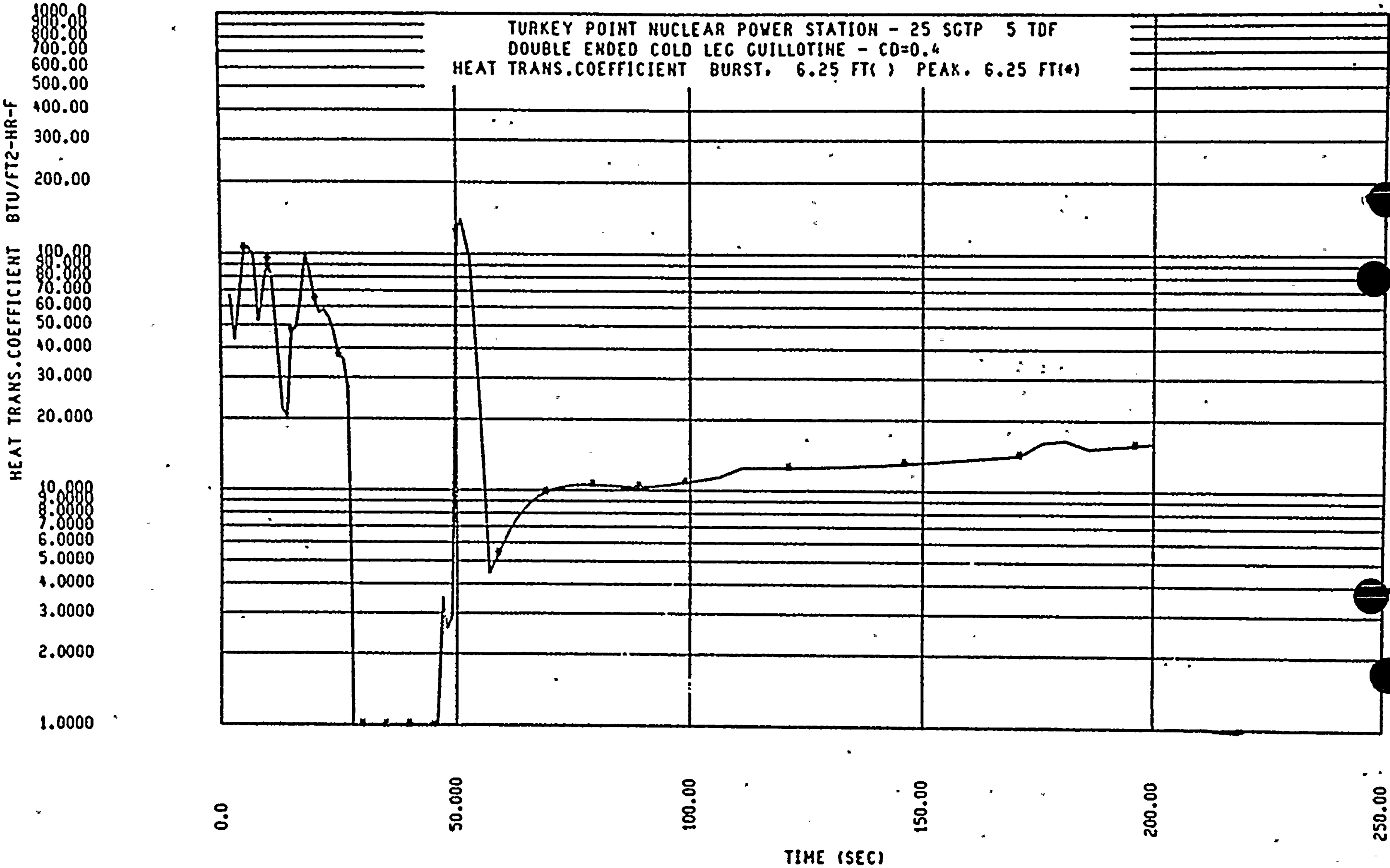

FIGURE 3 – HEAT TRANSFER COEFFICIENT – DECLG (CD=0.4)

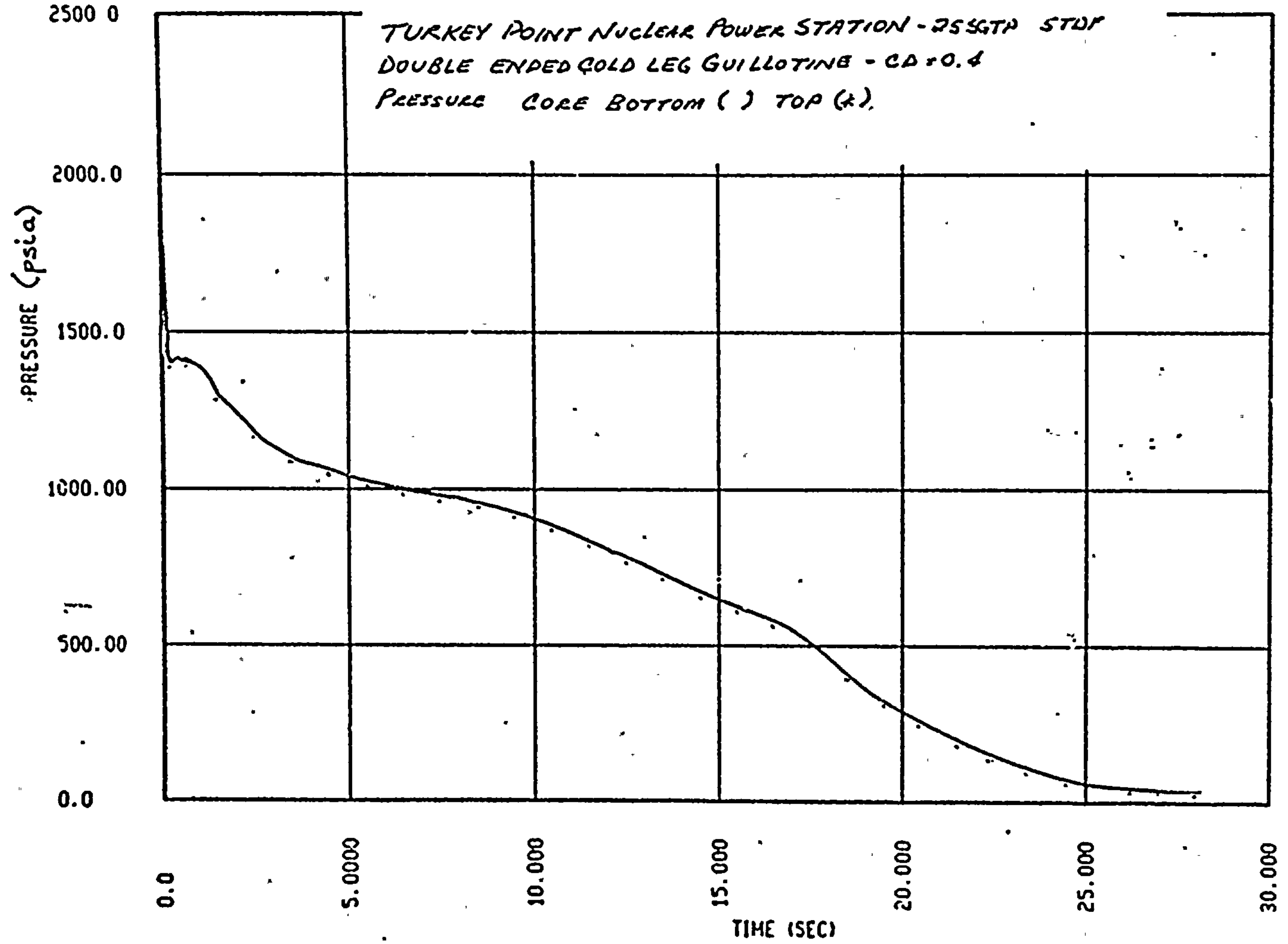

FIGURE 4 – PRESSURE - DECLG (CD = 0.4)

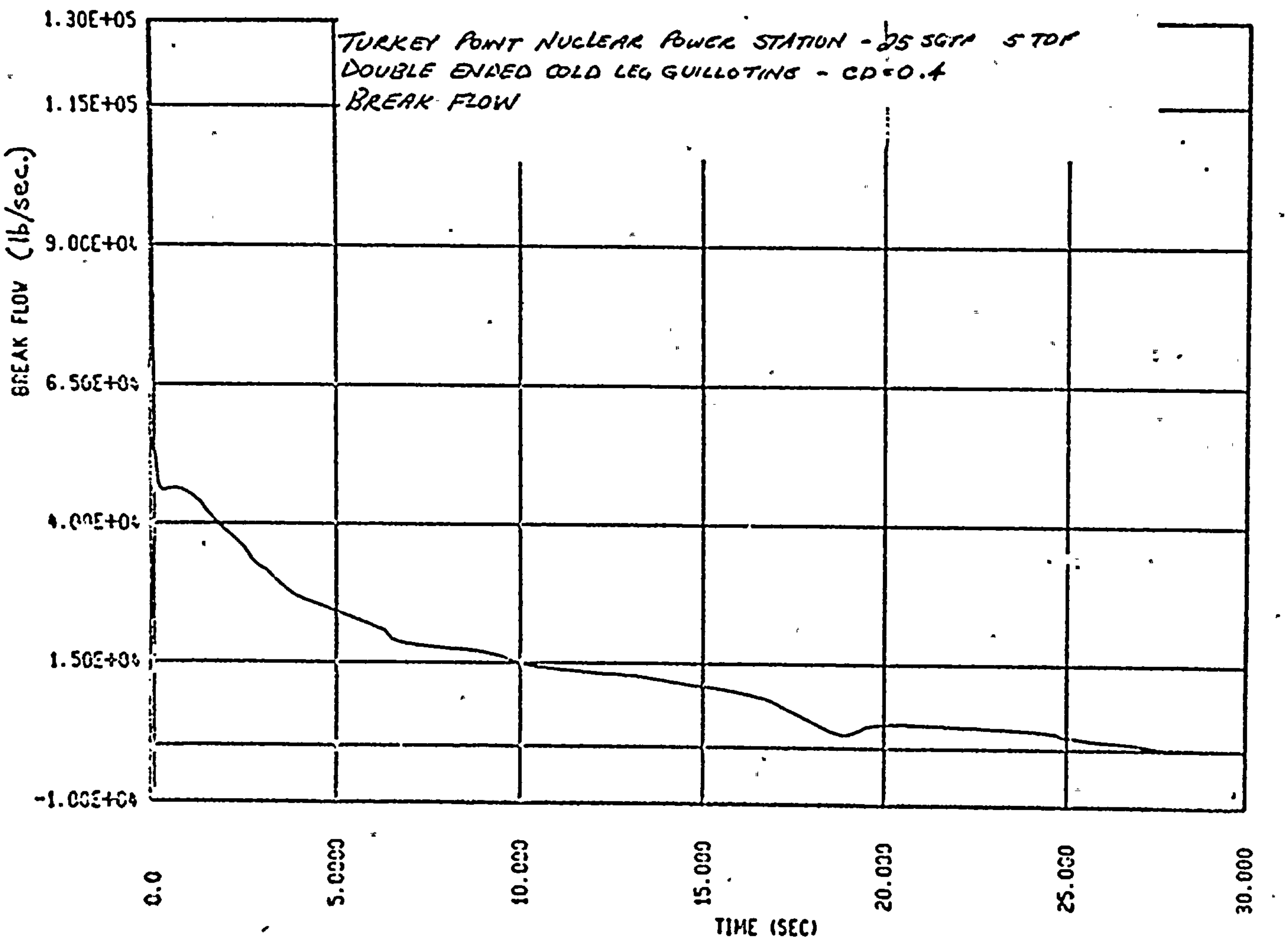

FIGURE 5 - BREAK FLOW RATE - DECLG (CD=0.4)

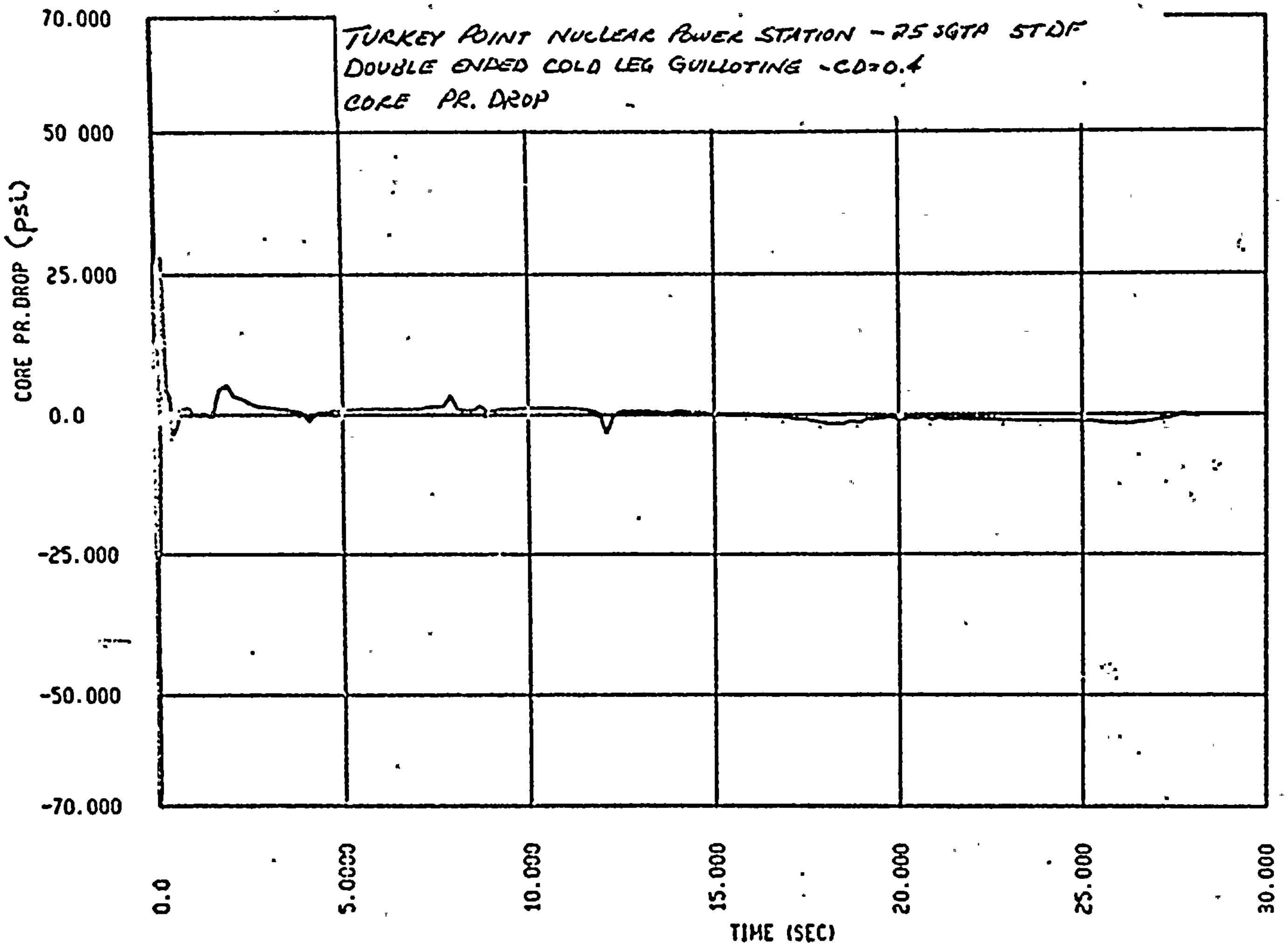

FIGURE 6 - CORE PRESSURE DROP - DECLG (CD=0.4)

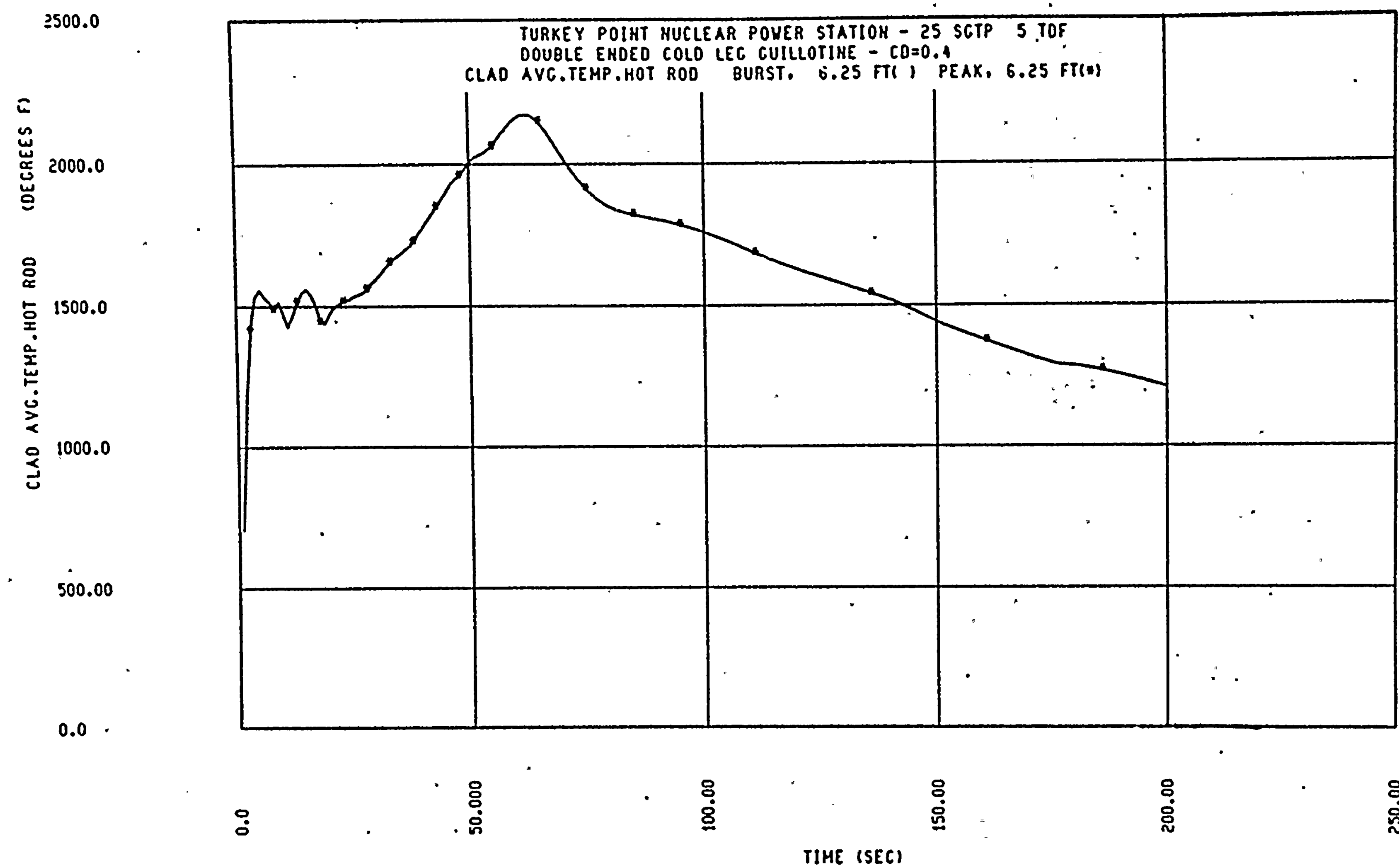

FIGURE 7 - PEAK CLAD TEMPERATURE - DECLG (CD=0.4)

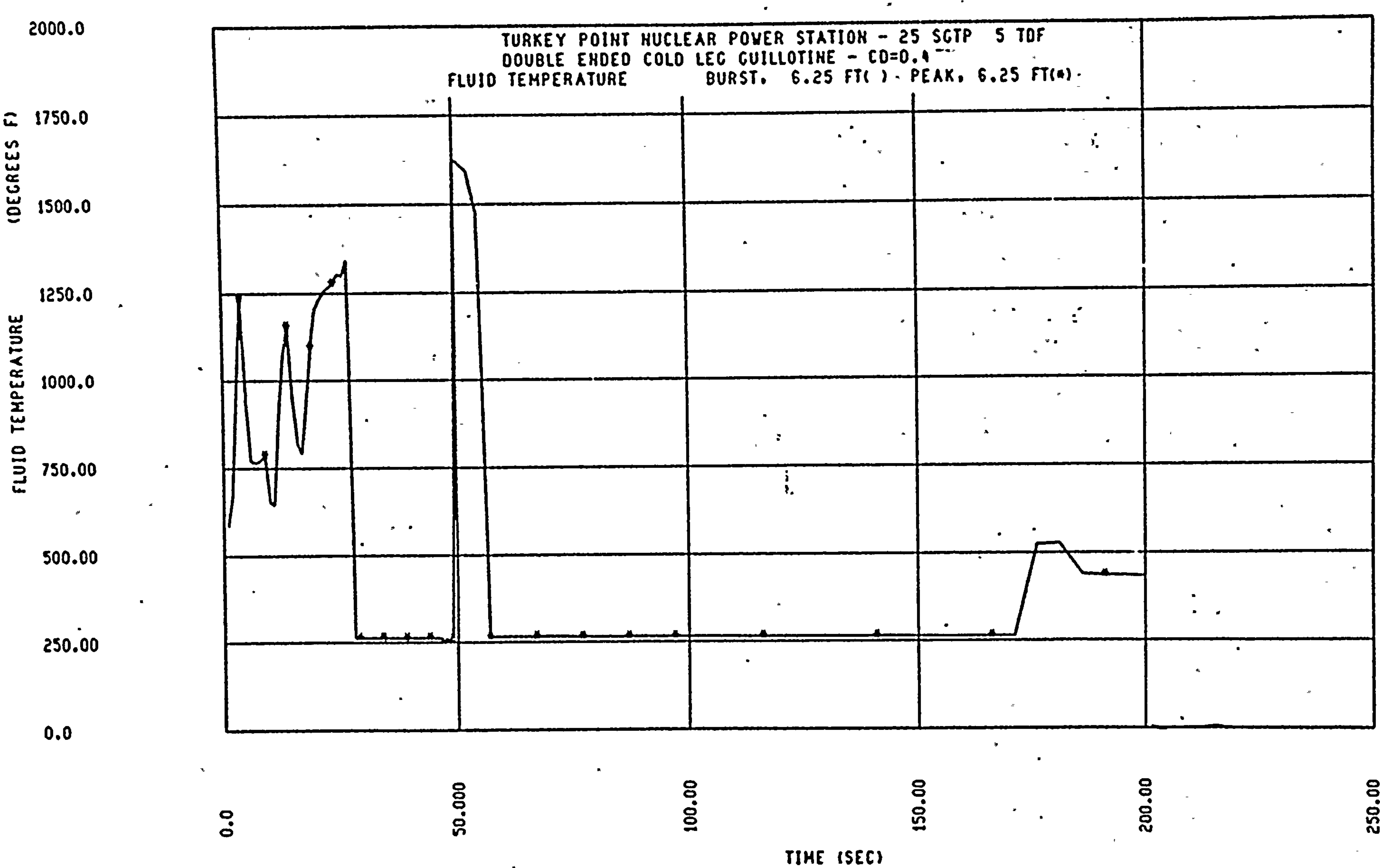

FIGURE 8 - FLUID TEMPERATURE - DECLG (CD=0.4)

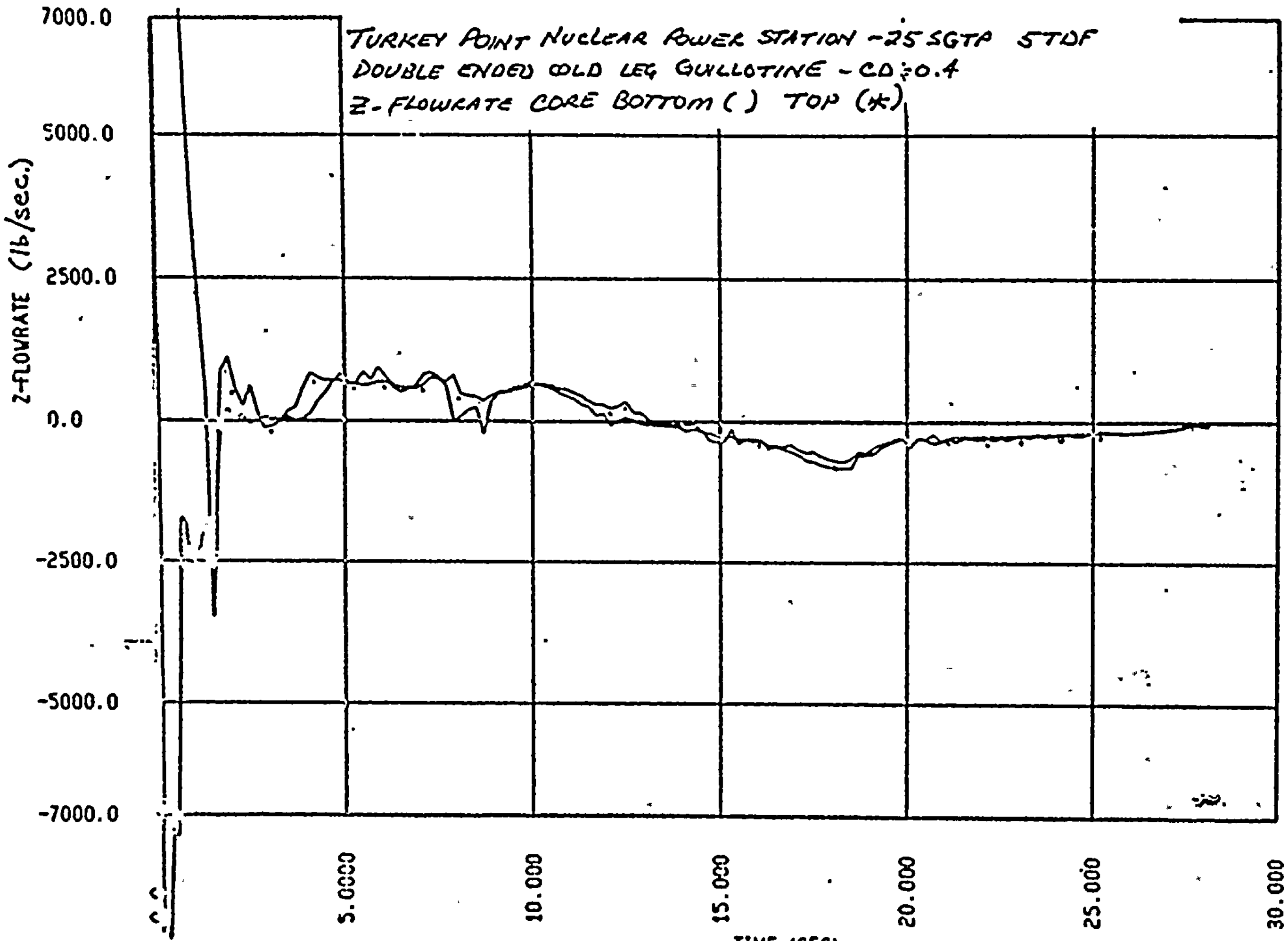

FIGURE 9-CORE FLOW - TOP AND BOTTOM - DECLG (CD=0.4)

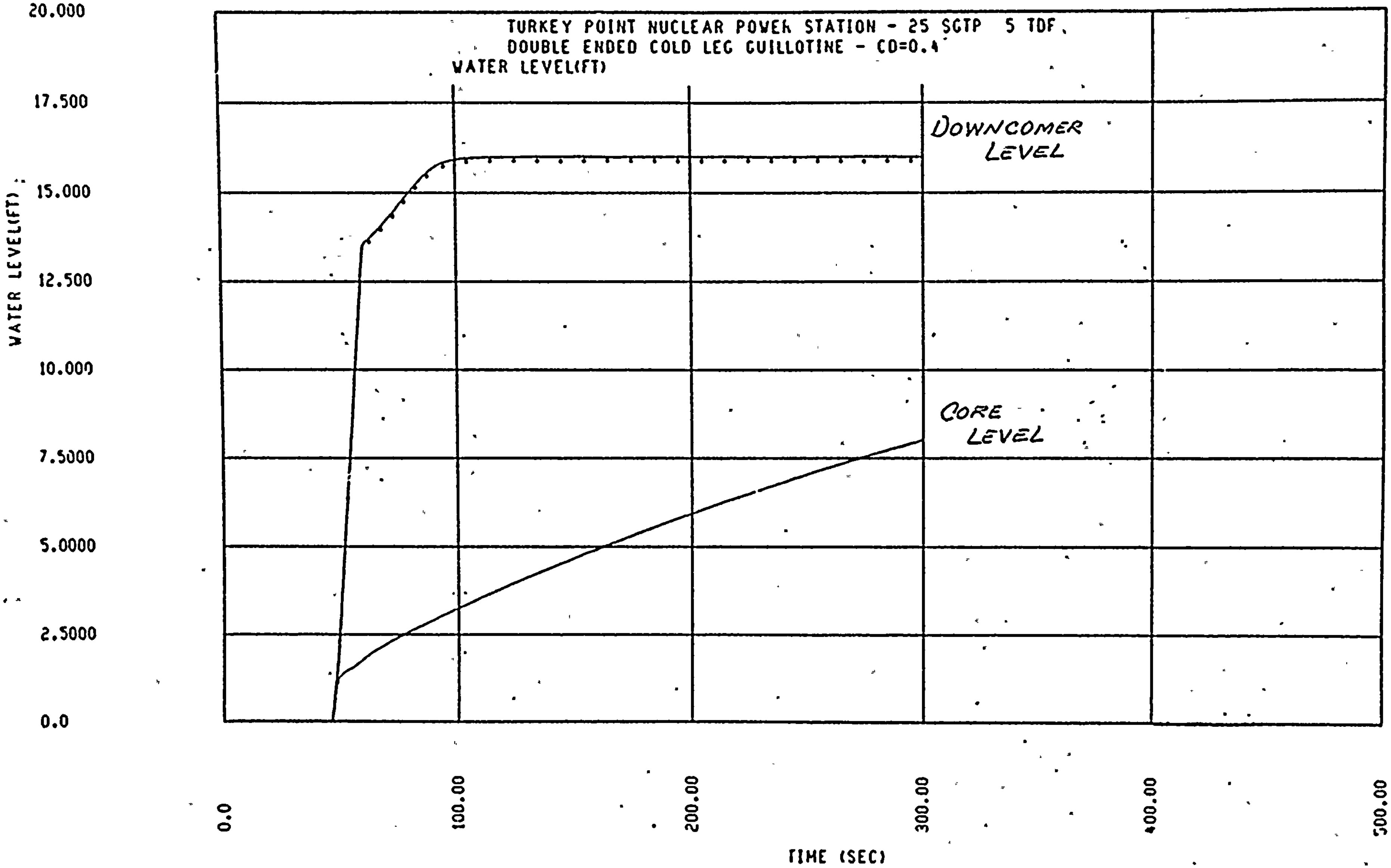

FIGURE. 10a-REFLOOD TRANSIENT-DECLG (CD=0.4)

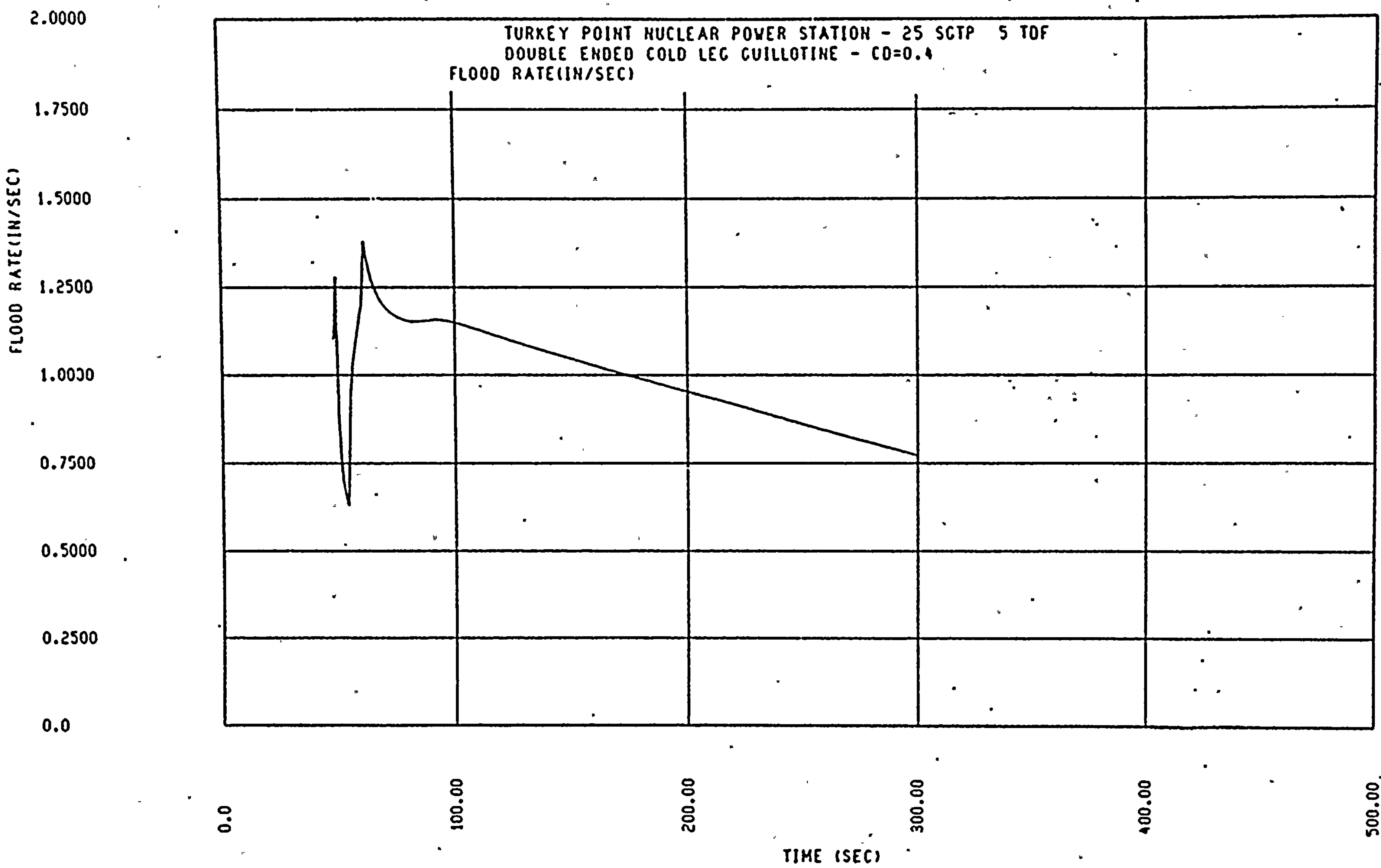

FIGURE 10b - REFLOOD TRANSIENT - DECLG (CD = 0.4)

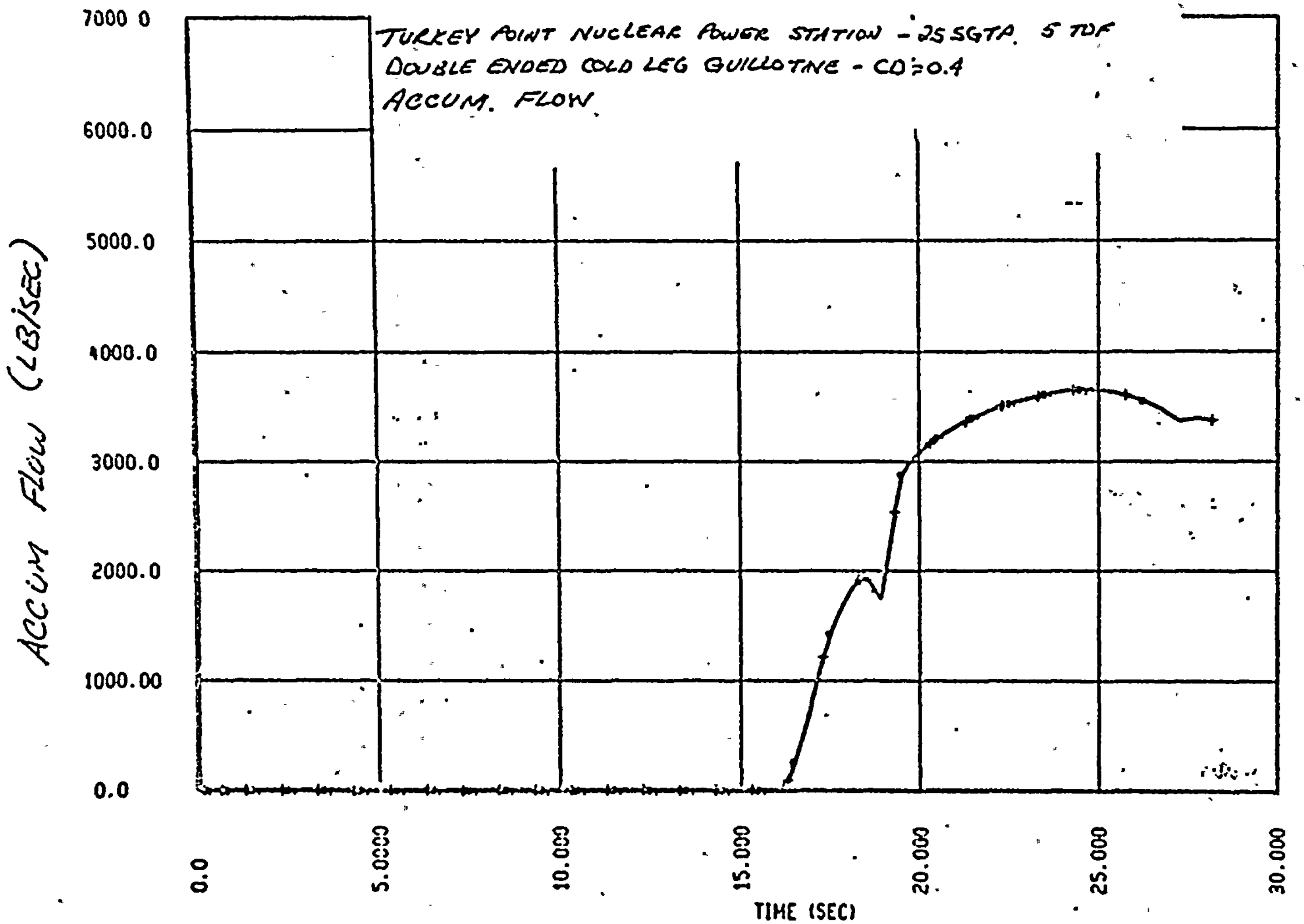

FIGURE 11 - ACCUMULATOR FLOW (BLOWDOWN) - DECLG (CD=0.4)

TURKEY POINT NUCLEAR POWER STATION - 25 SGTP 5 TDF
DOUBLE ENDED COLD LEG GUILLOTINE - CD=0.4

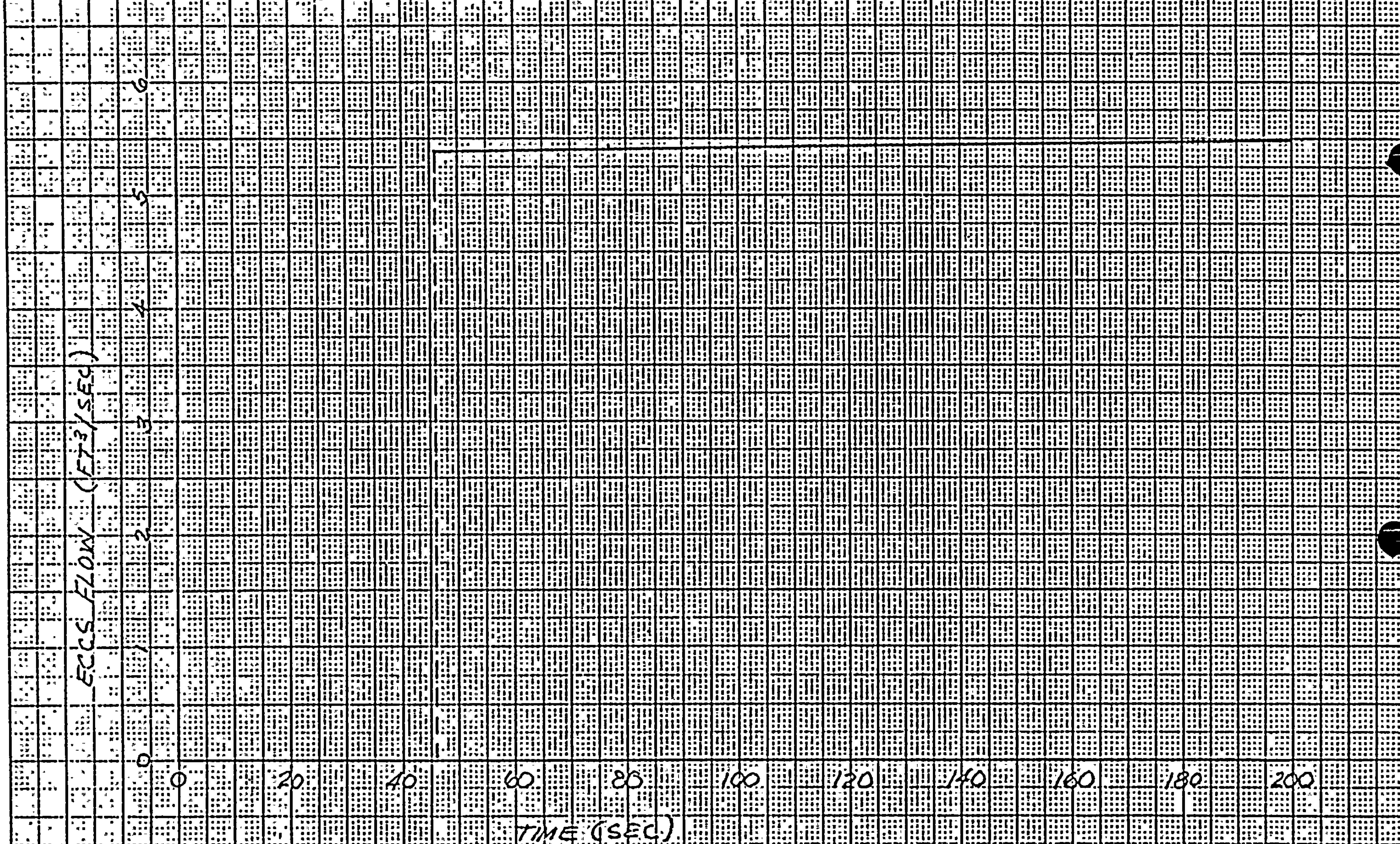

FIGURE 12 PUMPED ECCS FLOW (REFLOOD) - DECLG (CD=0.4)

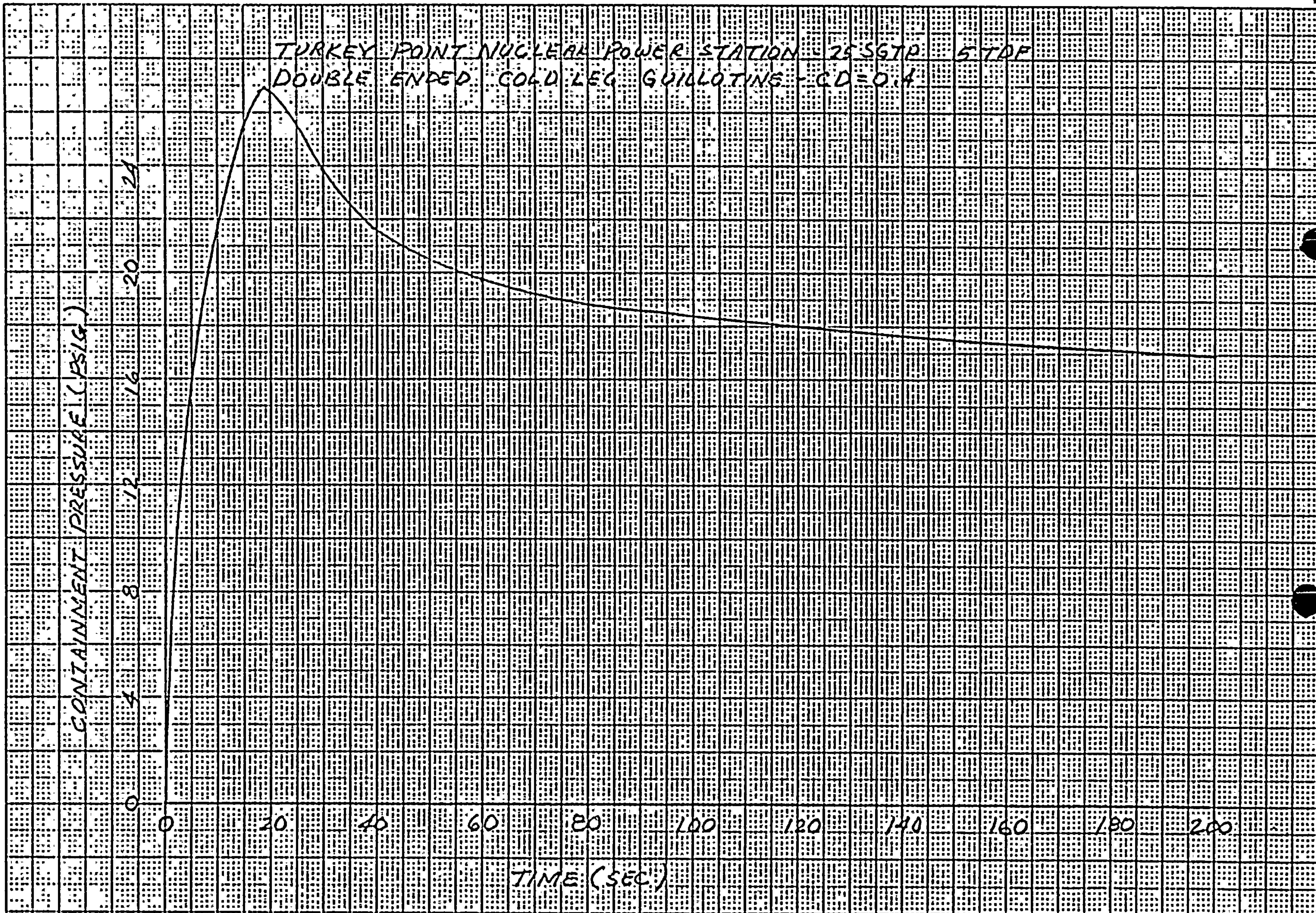

FIGURE 13 CONTAINMENT PRESSURE - DECLG (CD=0.4)

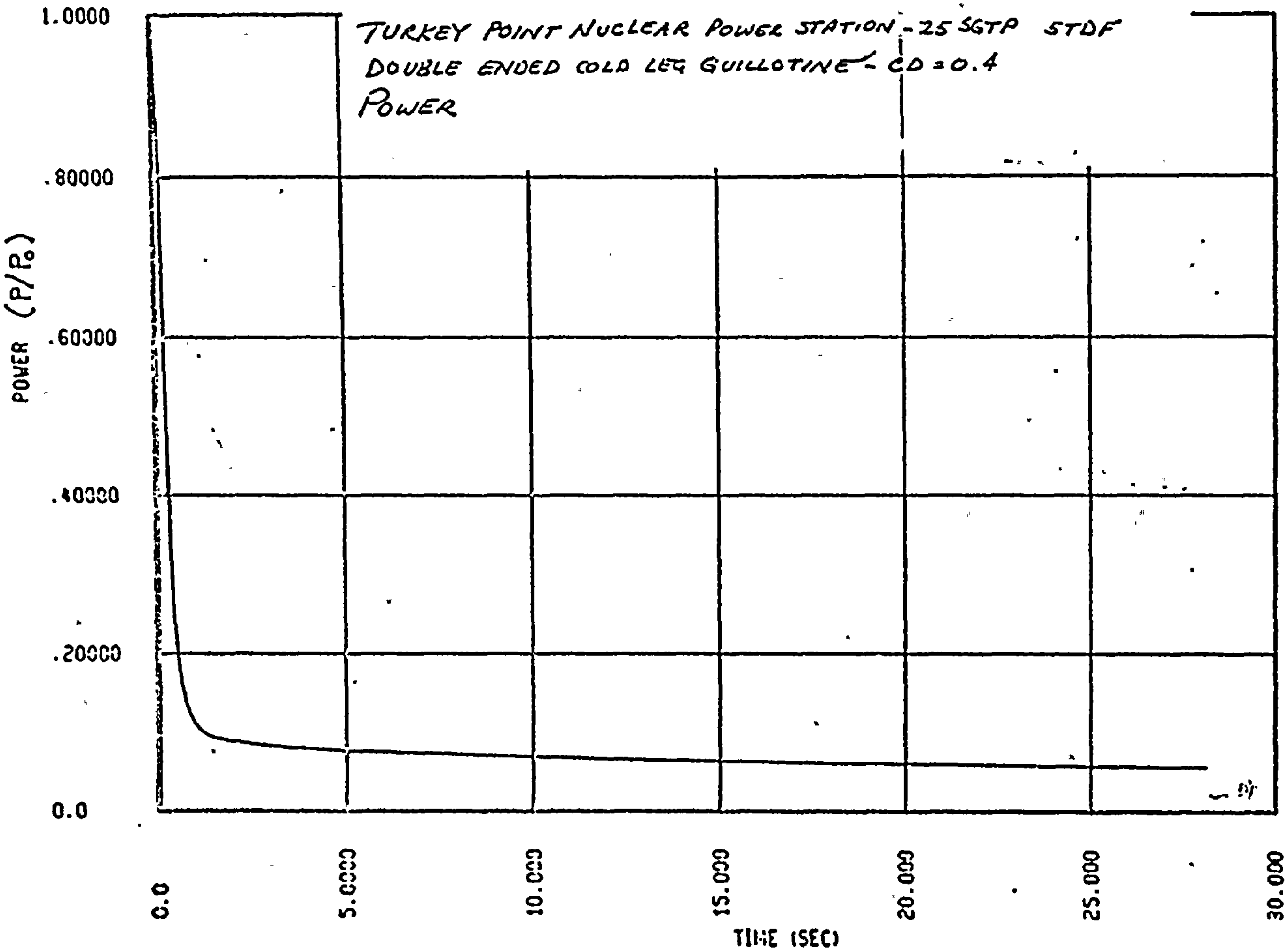

FIGURE 14 - CORE POWER TRANSIENT - DECLG (CD = 0.4)

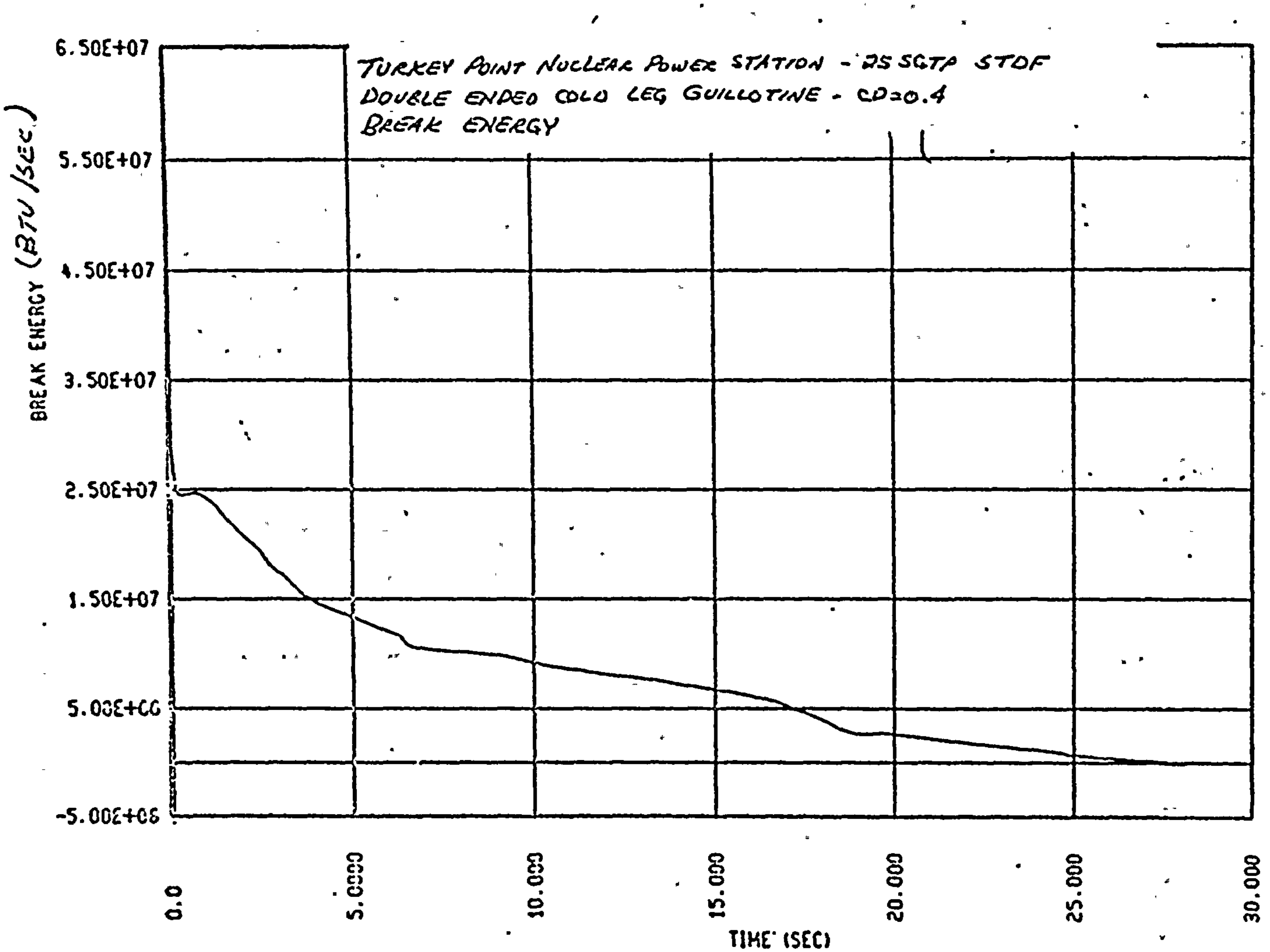

FIGURE 15 - BREAK ENERGY RELEASED TO CONTAINMENT - DECLG (CD=0.4)

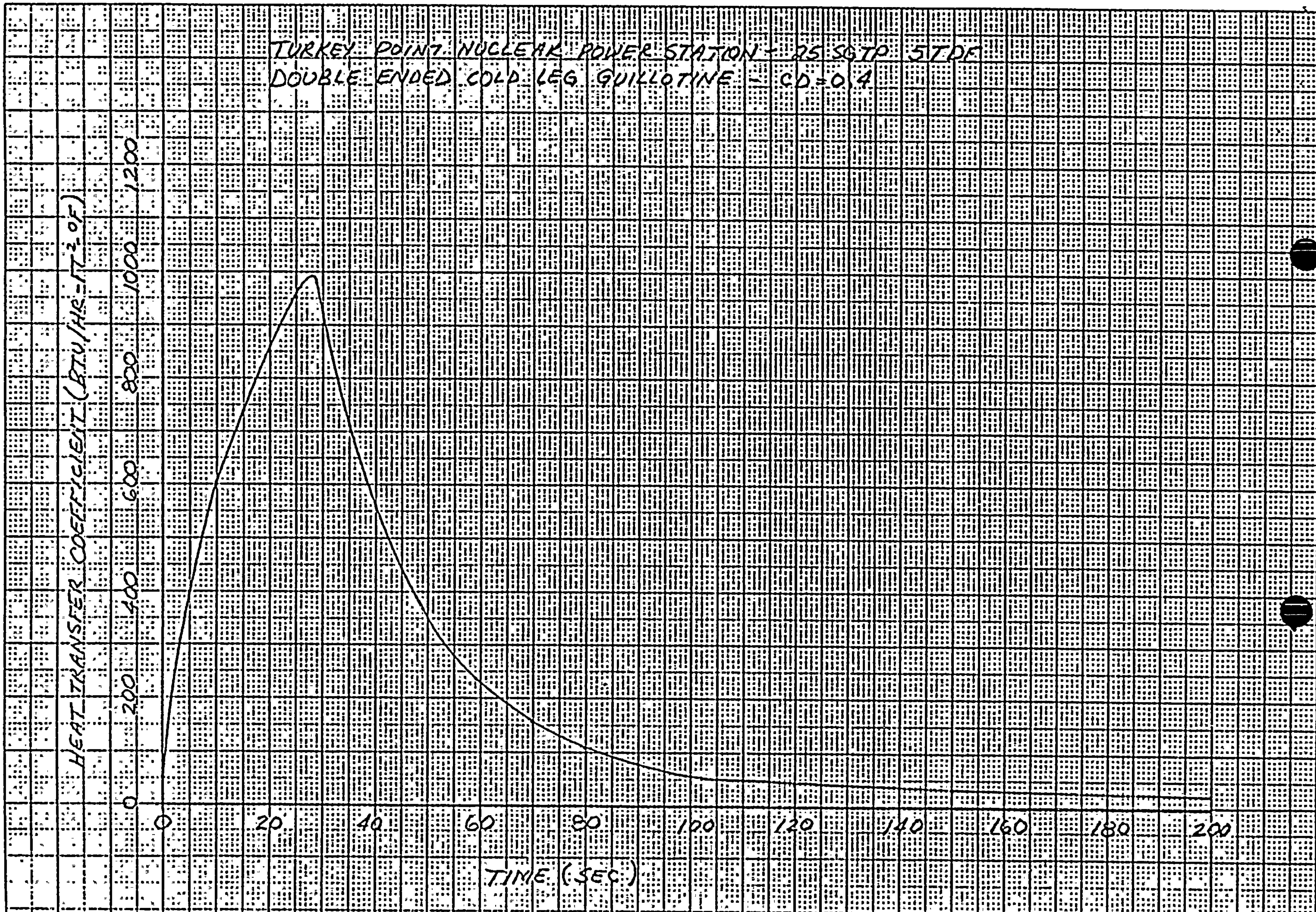

FIGURE 16 CONTAINMENT WALL HEAT TRANSFER COEFFICIENT - DECLG (CD=0.4)

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 Applicant 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 Applicant.

Robert E. Uhrig

Robert E. Uhrig

Subscribed and sworn to before me

this 9th day of August, 1978

Betty Brittain

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 MARCH 27, 1982 BONDED THRU MAYNARD BONDING AGENCY