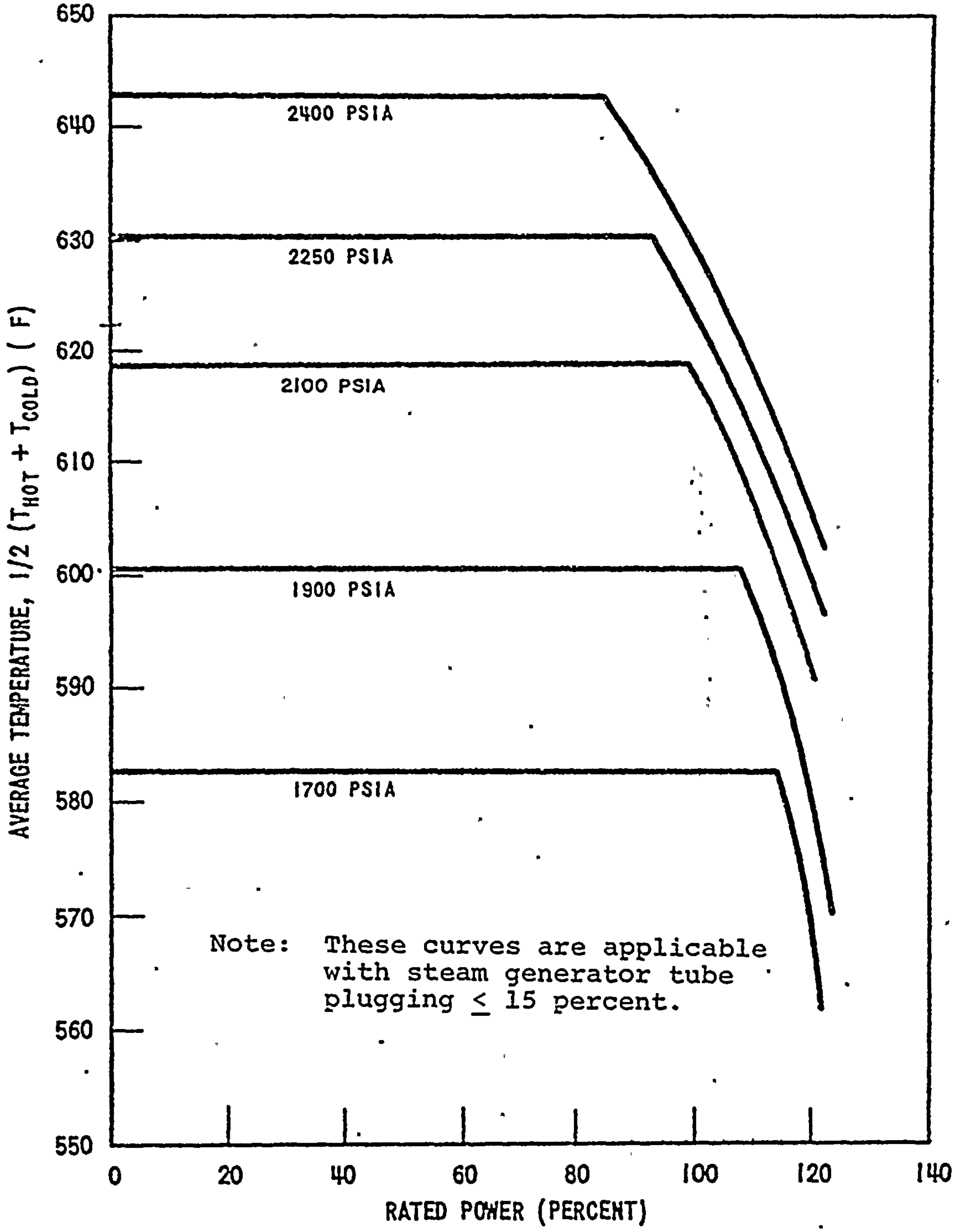


Figure 2.1-1. Reactor Core Thermal and Hydraulic Safety Limits, Three Loop Operation.

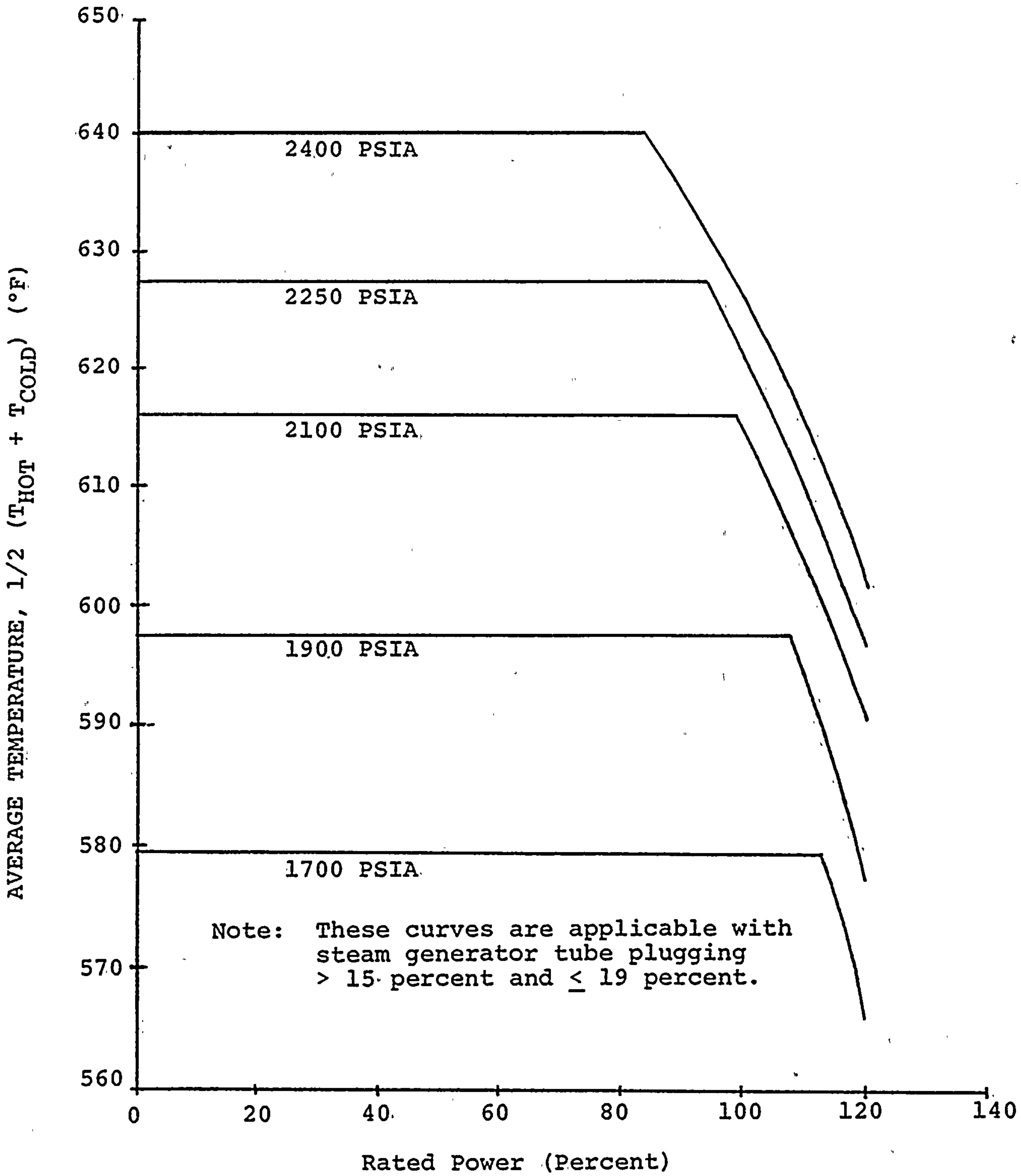


Figure 2.1-1a Reactor Core Thermal and Hydraulic Safety Limits, Three Loop Operation

9/27/78

Reactor Coolant Temperature

Overtemperature ΔT $\leq \Delta T_o \left[K_1 - 0.0107\,(T - 574) + 0.000453\,(P-2235) - f(\Delta q) \right]$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

P = Pressurizer pressure, psig

$f(\Delta q)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during startup tests such that:

For $(q_t - q_b)$ within +10 percent and -14 percent where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, $f(\Delta q) = 0$.

For each percent that the magnitude of $(q_t - q_b)$ exceeds +10 percent, the Delta-T trip set point shall be automatically reduced by 3.5 percent of its value at interim power.

For each percent that the magnitude of $(q_t - q_b)$ exceeds -14 percent, the Delta-T trip set point shall be automatically reduced by 2 percent of its value at interim power.

K_1 (Three Loop Operation) = 1.095*
(Two Loop Operation) = 0.88

---

*K_1 = 1.095 for steam generator tube plugging $\leq$ 15 percent ,
K_1 = 1.08 for steam generator tube plugging > 15 percent and $\leq$ 19 percent

$$\text{Over-power } \Delta T \leq \Delta T_o \left[1.11^{*} - K_1 \frac{dT}{dt} - K_2 (T - T') - f(\Delta q) \right]$$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

T' = Indicated average temperature at nominal conditions and rated power, F

K_1 = 0 for decreasing average temperature, 0.2 sec./F for increasing average temperature

K_2 = 0.00068 for T equal to or more than T'; 0 for T less than T'

$\frac{dT}{dt}$ = Rate of change of temperature, F/sec

$f(\Delta q)$ = As defined above

Pressurizer

Low Pressurizer pressure - equal to or greater than 1835 psig.

High Pressurizer pressure - equal to or less than 2385 psig.

High Pressurizer water level - equal to or less than 92% of full scale.

Reactor Coolant Flow

Low reactor coolant flow - equal to or greater than 90% of normal indicated flow

Low reactor coolant pump motor frequency - equal to or greater than 56.1 Hz

Under voltage on reactor coolant pump motor bus - equal to or greater than 60% of normal voltage

Steam Generators

Low-low steam generator water level - equal to or greater than 5% of narrow range instrument scale

---

*This factor is 1.11 for steam generator tube plugging ≤ 15 percent. This factor is 1.10 for steam generator tube plugging > 15 percent and ≤ 19 percent.

6. DNB PARAMETERS

The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg $\leq$ 578.2°F

b. Pressurizer Pressure $\geq$ 2220 psia*

c. Reactor Coolant Flow $\geq$ 268,500 gpm†

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limits after each refueling cycle.

---

* Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

† Reactor Coolant Flow $\geq$ 268,500 gpm for steam generator tube plugging $\leq$ 15 %.

Reactor Coolant Flow $\geq$ 263,130 gpm for steam generator tube plugging > 15% and $\leq$ 19%.

reactivity insertion upon ejection greater than 0.3% Δ 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 $\leq$ 15%, the hot channel factors (defined in the basis) must meet the following limits at all times except during low power physics tests:

$$F_q\ (Z) \leq (2.22/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \leq (4.44) \times K(Z), \text{ for } P \leq .5$$

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

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.

2. With steam generator tube plugging > 15% and $\leq$ 19%, the hot channel factors must meet the following limits at all times except during low power physics tests:

$$F_q\ (Z) \leq (2.05/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \leq (4.10) \times K(Z), \text{ for } P \leq .5$$

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

Where P, K(Z), and Z are defined in 1. above.

If predicted F_q exceeds 2.05 with tube plugging > 15% and $\leq$ 19%, then power will be limited to the rated power multiplied by the ratio of 2.05 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 conform that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

SAFETY EVALUATION

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

I. Introduction

This evaluation supports several changes to the Technical Specifications which have been brought about by the potential plugging of additional steam generator tubes at Turkey Point Units 3 and 4. Technical Specification 3.1.6.c (Reactor Coolant Flow) and 3.2.6.a (Hot Channel Factors), the Over-temperature ΔT and Overpower ΔT equations, and Figure 2.1-1 (Reactor Core Thermal and Hydraulic Safety Limits, Three Loop Operation) will be affected.

II. Revised ECCS Analysis

The attached ECCS Analysis (see Appendix A) constitutes a reanalysis of a hypothetical loss of coolant accident (LOCA) for Turkey Point Units 3 & 4. The previously demonstrated limiting break (DECLG, $C_D = 0.4$) was reanalyzed changing only the following parameters:

1) RCS flow = 98% of the thermal design value
   = 263,130 GPM
2) Steam generator tube plugging = 19% (uniform)
3) Total peaking factor (F_q^T) = 2.05
4) No change in nominal T_{avg}; however, instrument uncertainty was subtracted from T_{IN} rather than added.

III. RCS Flow

An evaluation has been performed to address the operation of Turkey Point Units 3 and 4 at 98% rated Thermal Design Flow. The evaluation was performed consistent with the following assumptions:

98% Thermal Design Flow, gpm	87,710
S.G. Tube Plugging, %	19
Maximum Power, Mwt	2,200
T_{avg} at 100% Power, °F	574.8

ΔT at 100% Power, °F	57.1
T_{inlet} at 100% Power, °F	546.2
F_q^T maximum	2.05

A reduction in the steady state primary flow will affect all of the FSAR Chapter 14 transients. However, by using excess margin available and technical specification reductions in allowed core peaking factors, a 2% change in flow will not change the safety conclusions in the FSAR. The FSAR transients can be divided into two categories: DNB Limited, and Fuel or Reactor Vessel Integrity Limited. These are discussed below with the method used to offset penalties associated with flow reductions.

A. DNB Limited Transients

The primary means of DNB protection for these transients is the Over-Temperature Delta-T Protection System. Although credit might not have been taken in the FSAR, this system assures DNB protection limits are not exceeded for the following transients: Rod Withdrawal at Power, Boron Dilution at Power, Excessive Heat Removal due to Feedwater Malfunction, Startup of an Inactive Loop, Excessive Load Increase and Loss of External Electrical Load. Revised Technical Specification core limits have been developed which incorporate a 2% reduction in thermal design flow. A reduction in the K_1 term of the Over-Temperature Delta-T setpoint equation from 1.09 to 1.08 will assure adequate protection. In addition, to the above there is considerable margin to DNB limits (DNBR = 1.24) in nearly all of the above transients. Since a 2% reduction in flow results in approximately a 2% reduction in DNBR, there is still adequate margin available.

The DNB transients not protected by the Over-Temperature Delta-T setpoints are: Rod Misalignment, Loss of Flow and Steamline Break. For all of these cases the flow reduction corresponds to less than a 2% reduction in minimum DNBRs, which can be accommodated with margin in the current design.

B. Fuel or Vessel Integrity Limited Transients

Rod Withdrawal from Subcritical - The current safety analysis shows large margins to safety limits with the peak heat flux being considerably less than 100% of rated power. Thus a 2% reduction in flow would have a negligible effect on peak fuel or clad temperatures.

Boron Dilution - The relatively long duration of the transient means that flow does not affect the operator action times during refueling or startup operation. In addition, the effect of 25% steam generator tube plugging on boron dilution has been analyzed. This analysis conservatively bounds the 19% plugging analysis. (Appendix B)

Locked Rotor - A reduction in flow will slightly increase peak system pressure (< 50 psia) from the value shown in the Cycle 3 RSE. However, the results are still considerably below the vessel faulted stress limits. The peak fuel and clad temperatures would also be affected. However, the hot spot peaking factor has been reduced, due to LOCA considerations, from 2.32 to 2.05. This 11% reduction in hot spot energy would more than compensate for the 2% reduction in flow. Thus vessel and fuel limits would not be exceeded due to a flow reduction.

Loss of Normal Feedwater/Station Blackout - The results of this accident are highly sensitive to the residual (decay) heat generation due to the long duration of the transient after trip. Residual heat generation is directly proportional to the initial power level preceding the trip. The analysis in the FSAR assumed the power to be 102% of the maximum turbine rating (2300 Mwt). Thus the total energy input to the system would be ∿ 5% less than originally assumed. Therefore this affect alone would more than compensate for a 2% flow reduction.

Rupture of a Control Rod Drive Mechanism - Sensitivity studies have shown that a 2% reduction in flow will result in less than a 40°F increase in fuel and clad peak temperatures. The current analysis shows that for a 40°F increase, all fuel and clad integrity limits can be met with margin.

Loss of Coolant Accident - An Appendix K LOCA analysis is attached (Appendix A) for 19% tube plugging and 98% thermal design flow.

Thus it has been shown that a 2% reduction in thermal design flow will not result in any safety limit violation.

APPENDIX A

REVISED ECCS ANALYSIS
TURKEY POINT UNITS 3 & 4

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	$C_D = 0.4$ DECLG (Sec)
START	0.0
Rx Trip Signal	.556
S.I. Signal	.7000
Acc. Injection	16.1
End of Blowdown	28.061
Bottom of Core Recovery	46.789
Acc. Empty	60.979
Pump Injection	25.7000
End of Bypass	27.815

TABLE 2

LARGE BREAK

	$C_D = 0.4$ DECLG
Results	
Peak Clad Temp. °F	2195.37
Peak Clad Location Ft.	6.0
Local Zr/H_2O Rxn(max)%	12.3951
Local Zr/H_2O Location Ft.	6.0
Total Zr/H_2O Rxn %	<0.3
Hot Rod Burst Time sec	22.80
Hot Rod Burst Location Ft.	6.0

Calculation

Core Power Mwt 102% of	2200
Peak Linear Power kw/ft 102% of	11.650
Peaking Factor (At License Rating)	2.05
Accumulator Water Volume (per tank)	875 ft^3

Fuel region + cycle analyzed	Cycle	Region
UNITS 3 and 4	3	3

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

LARGE BREAK

TABLE 3 (Continued)

CONTAINMENT DATA

(DRY CONTAINMENT)

STRUCTURAL HEAT SINKS

Thickness (In)	Area (Ft^2)
Steel 0.03	31,400
Steel 0.063	107,158
Steel 0.11	56,371
Steel 012	57,185
Steel 0.24	9,931
Steel 0.2898	---
Concrete 24.0	136,000
Steel 0.4896	23,677
Steel 0.6396	6,537
Steel 0.8904	4,915
Steel 1.256	27,802
Steel 1.56	5,307
Steel 2.0	668
Steel 2.75	1,268.7
Steel 5.5	1,277.4
Steel 9.0	260.4
Stainless 0.14	---
Concrete 24.0	14,392
Stainless 0.44	768
Stainless 2.126	3704
Stainless 0.007	102,400
Concrete 24.0	59,132

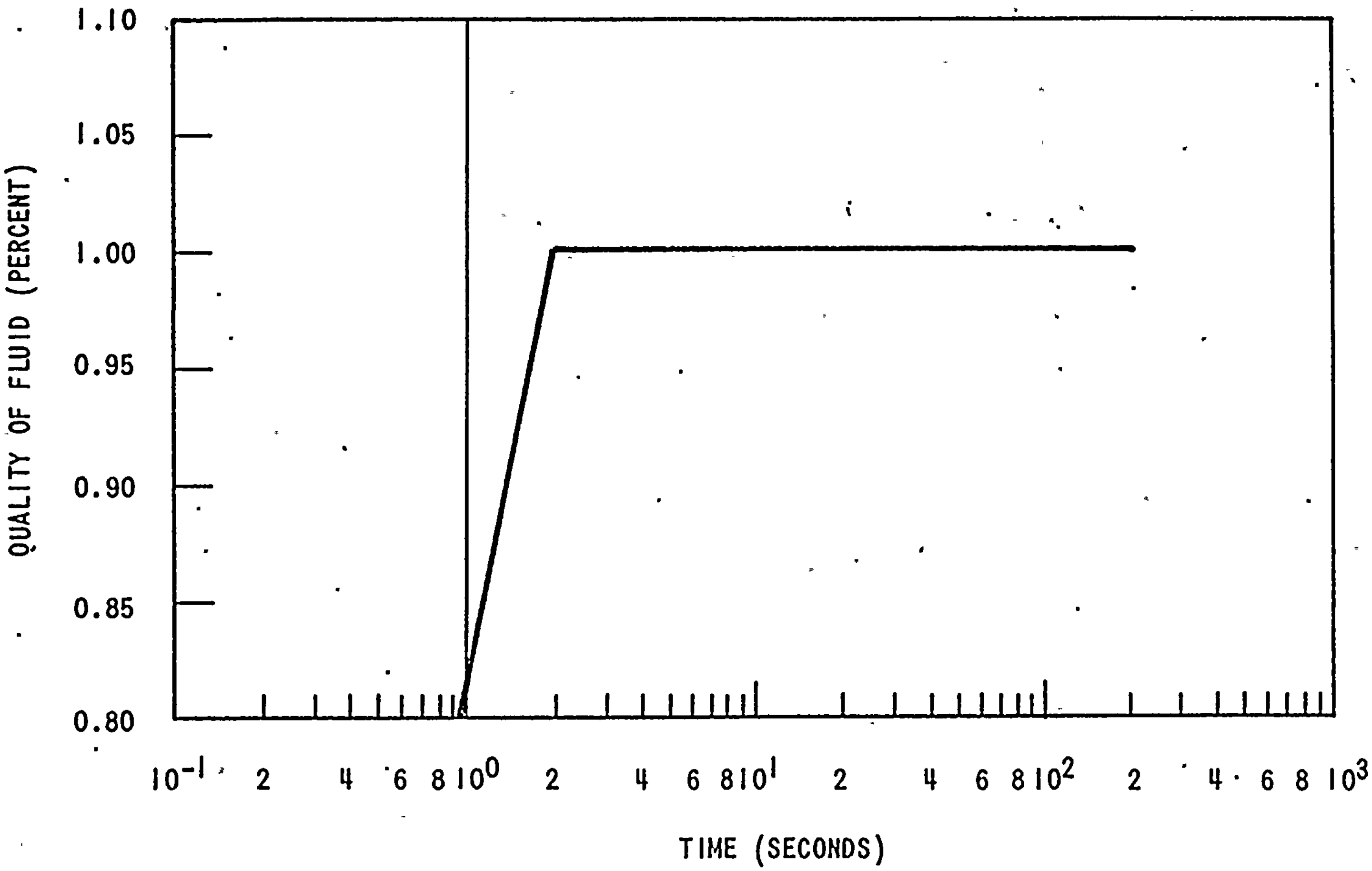


Figure 1 Fluid Quality - DECLG (C_D = 0 4)

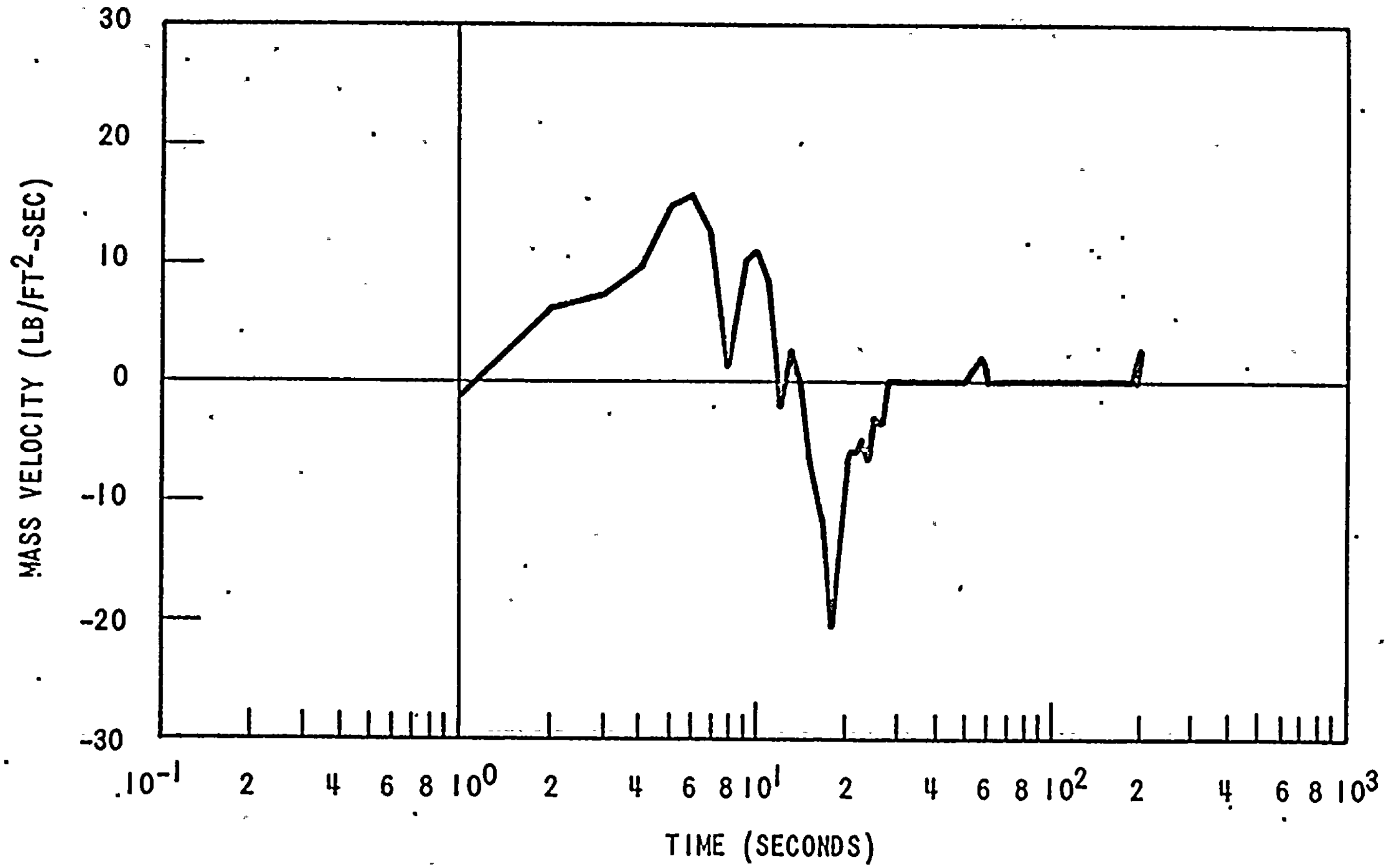


Figure 2 Mass Velocity - DECLG (C_D = 0.4)

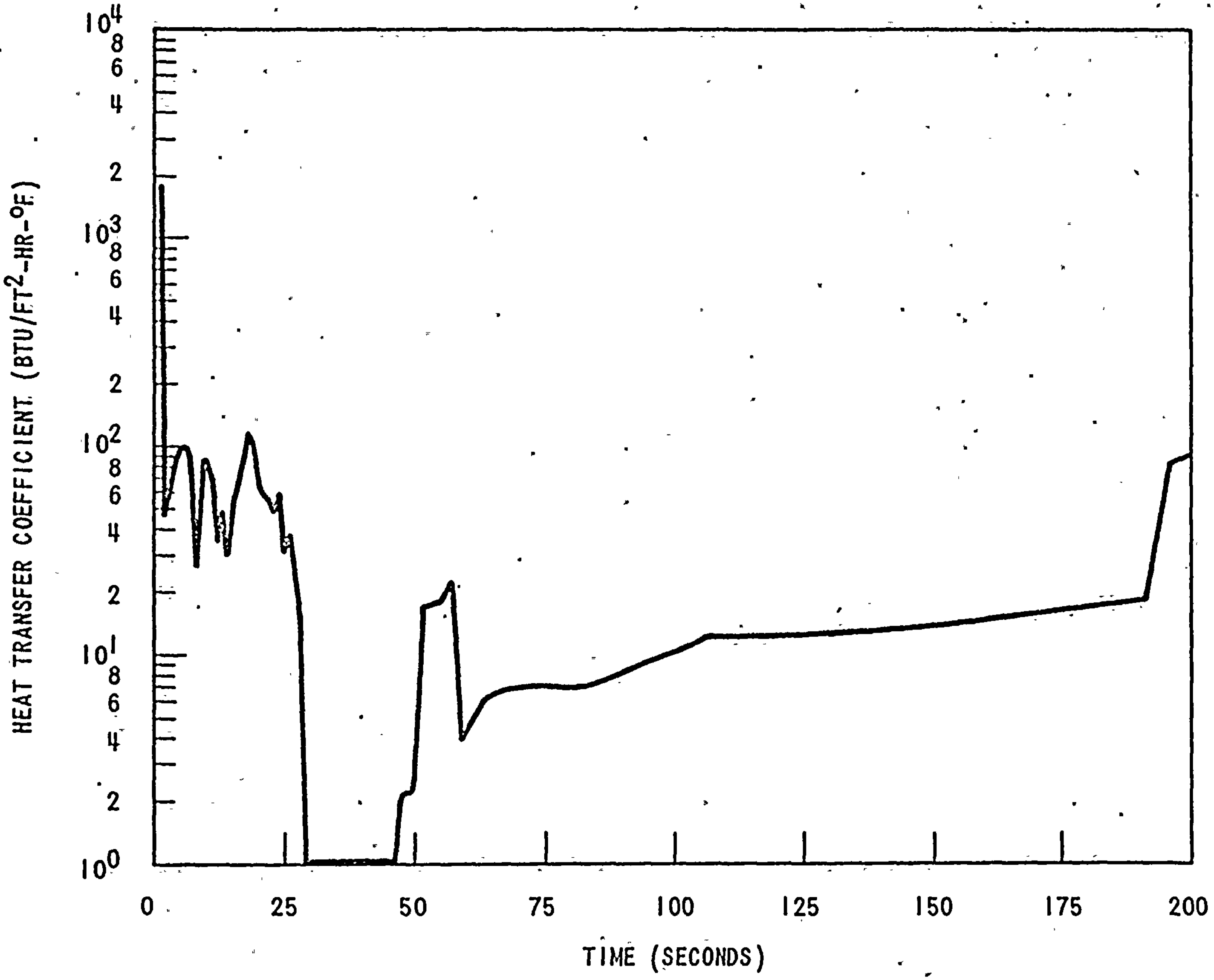


Figure 3 Heat Transfer Coefficient - DECLG (C_D = 0.4)

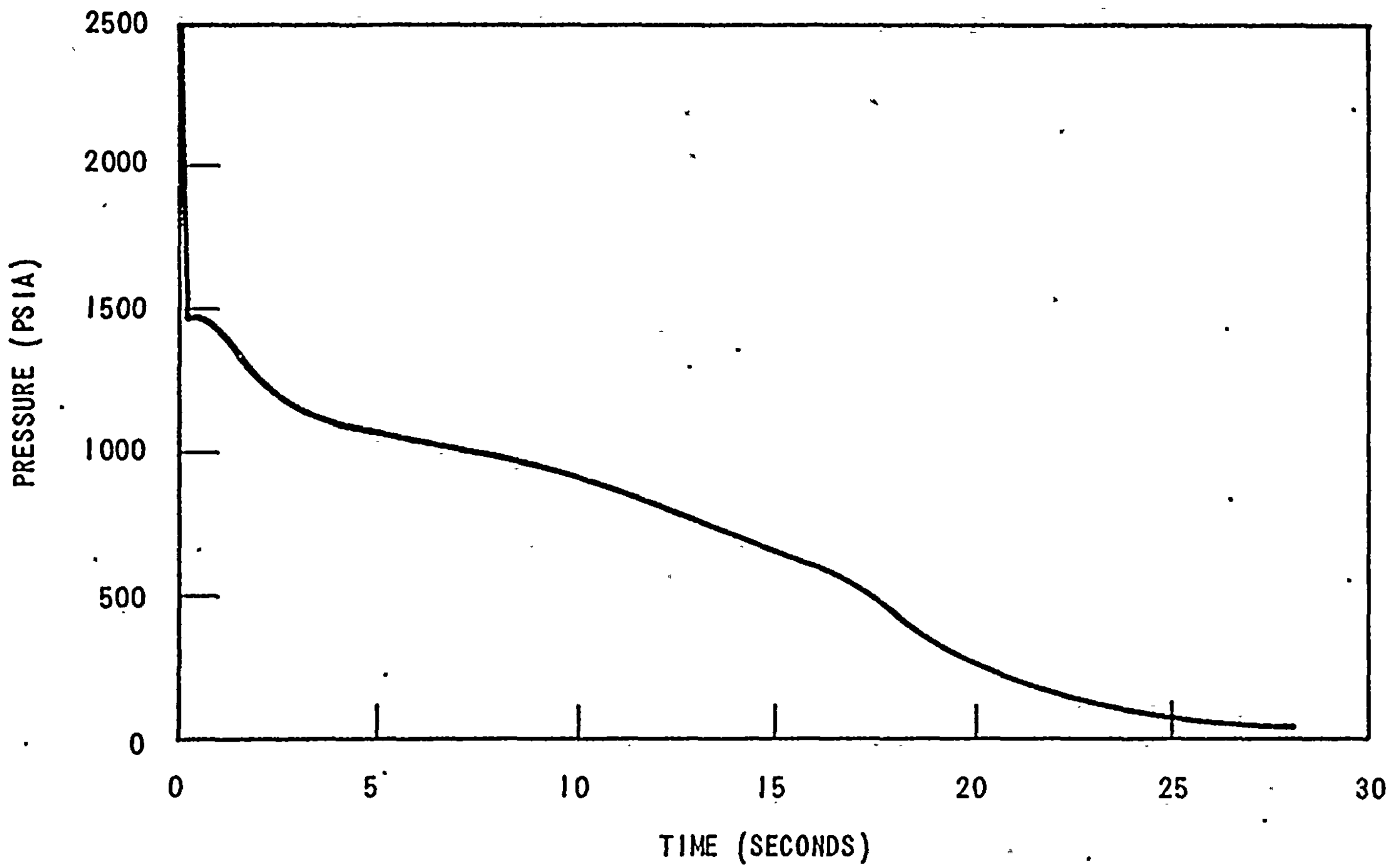


Figure 4 Core Pressure - DECLG (C_D = 0.4)

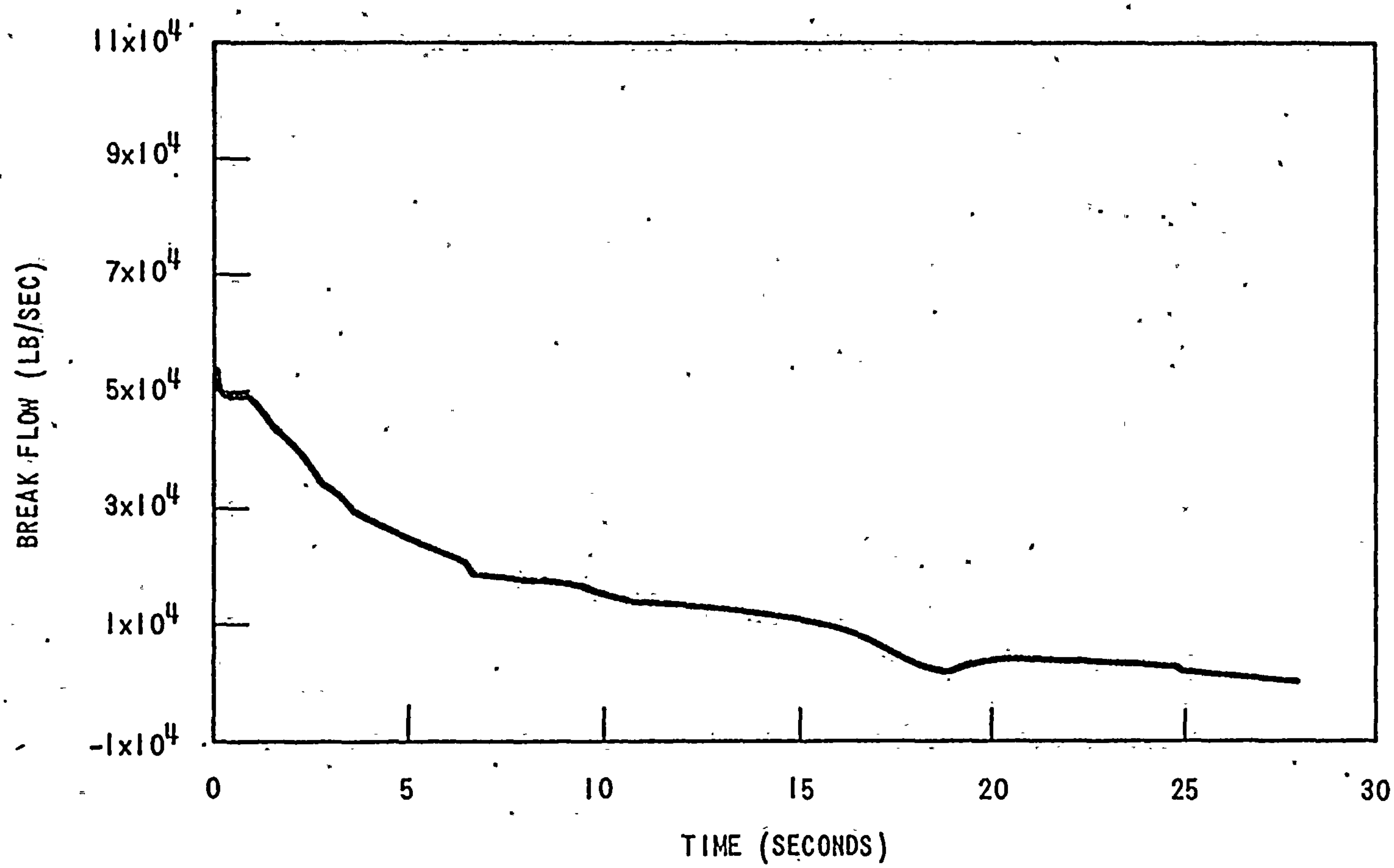


Figure 5 Break Flow Rate - DECLG ($C_D = 0.4$)

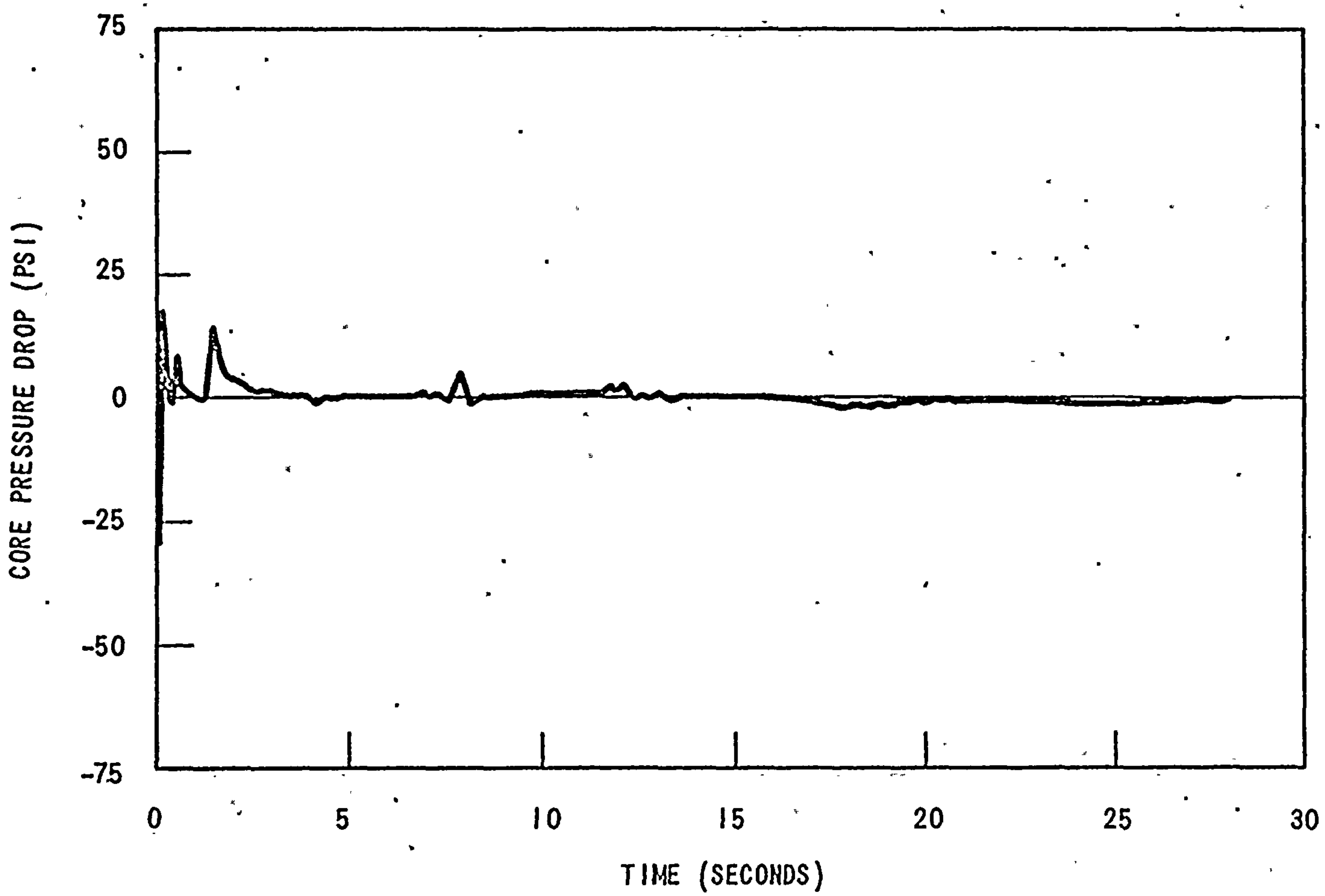


Figure 6 Core Pressure Drop – DECLG (C_D = 0.4)

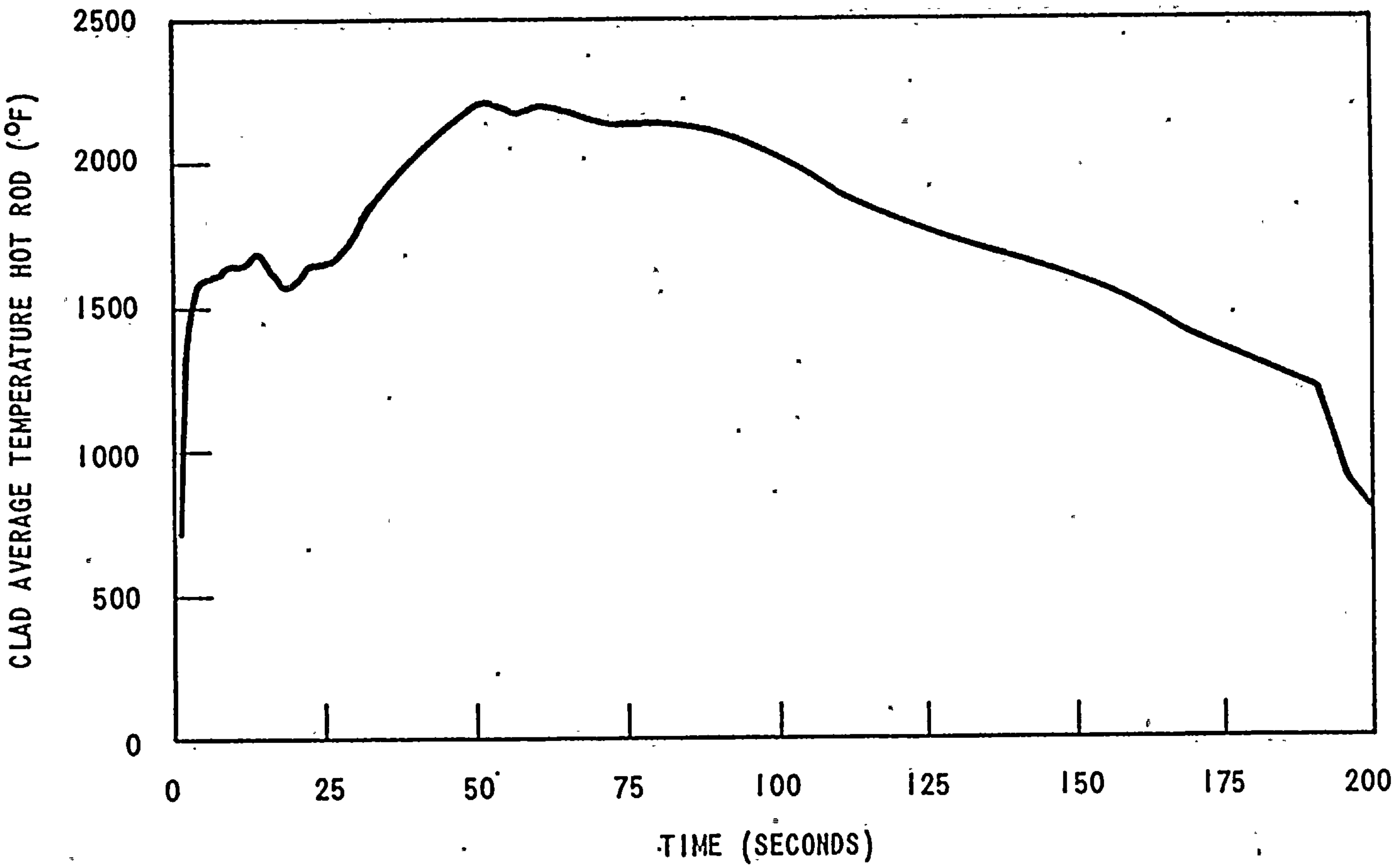


Figure 7 Peak Clad Temperature - DECLG (C_D = 0.4)

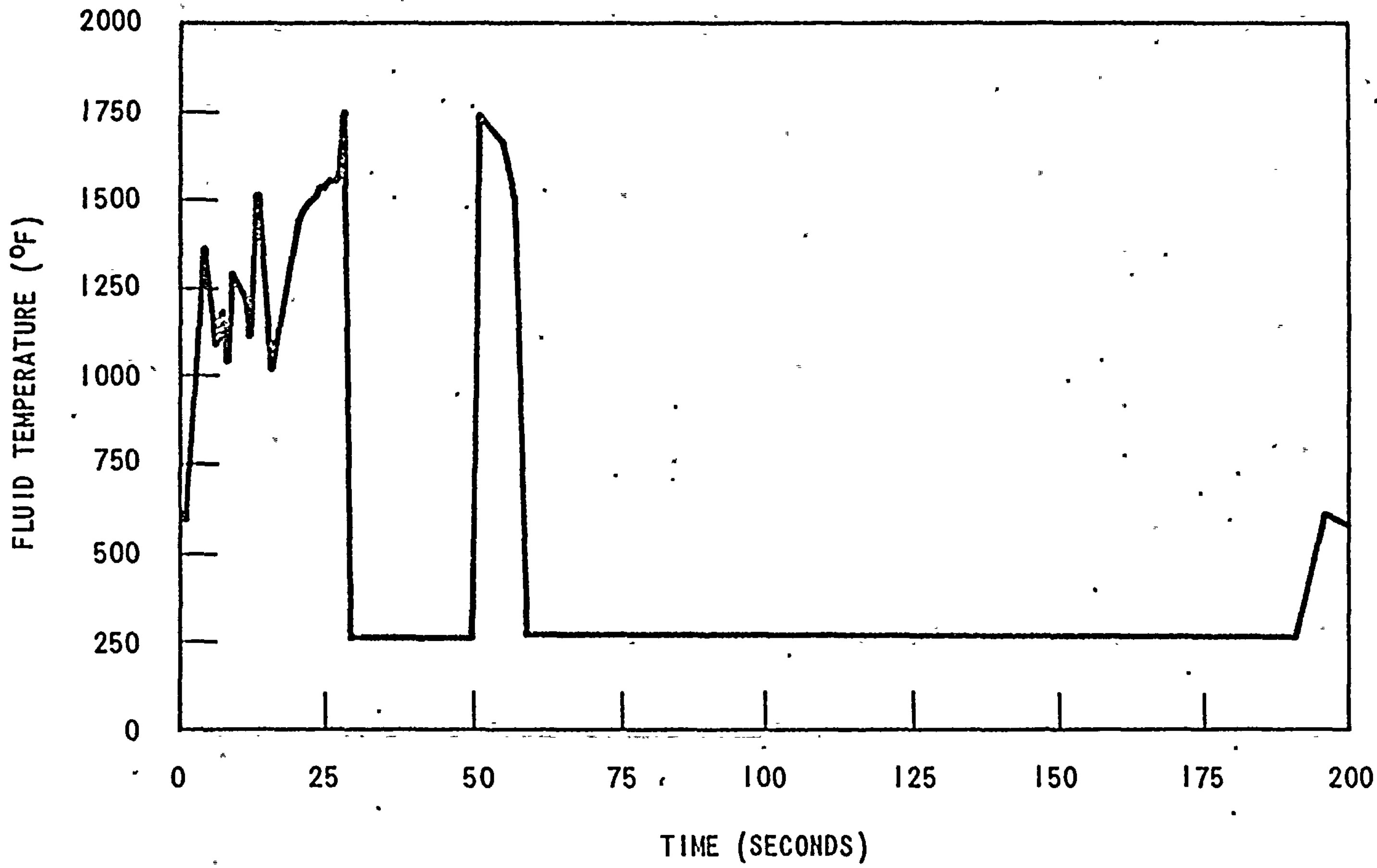


Figure 8 Fluid Temperature - DECLG (C_D = 0.4)

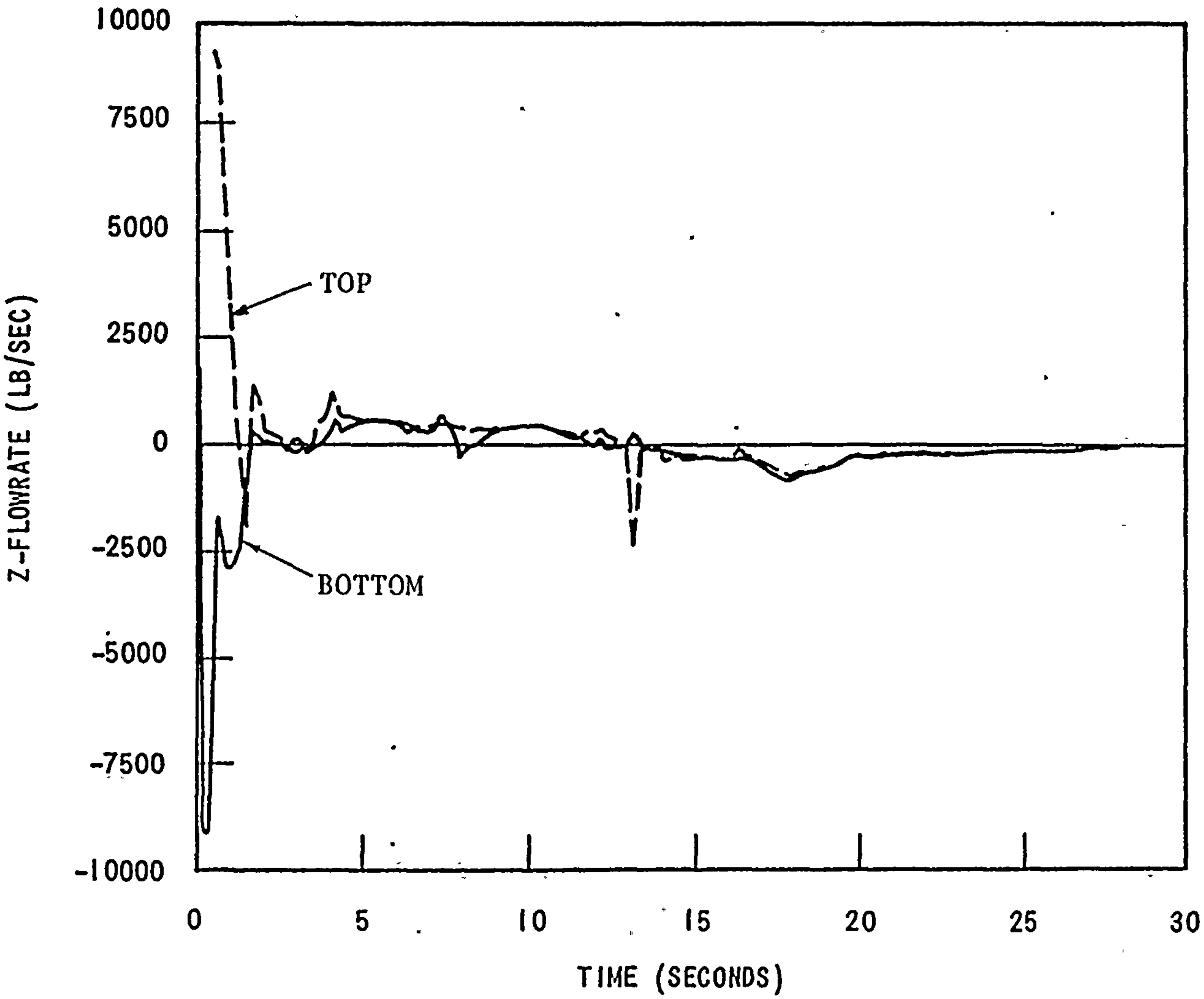


Figure 9 Core Flow - Top and Bottom - DECLG (C_D = 0.4)

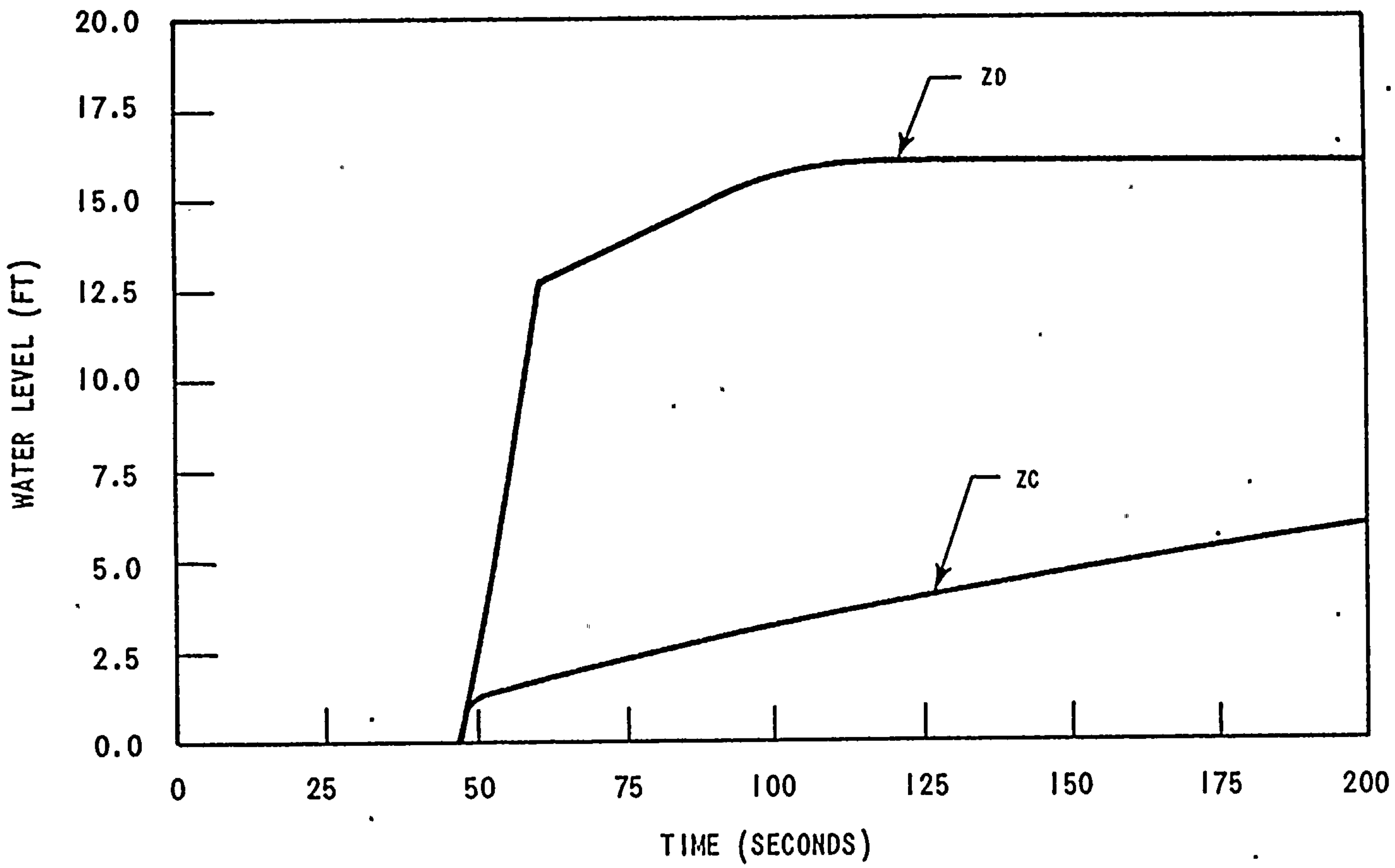


Figure 10a Reflood Transient - DECLG (C_D = 0.4)

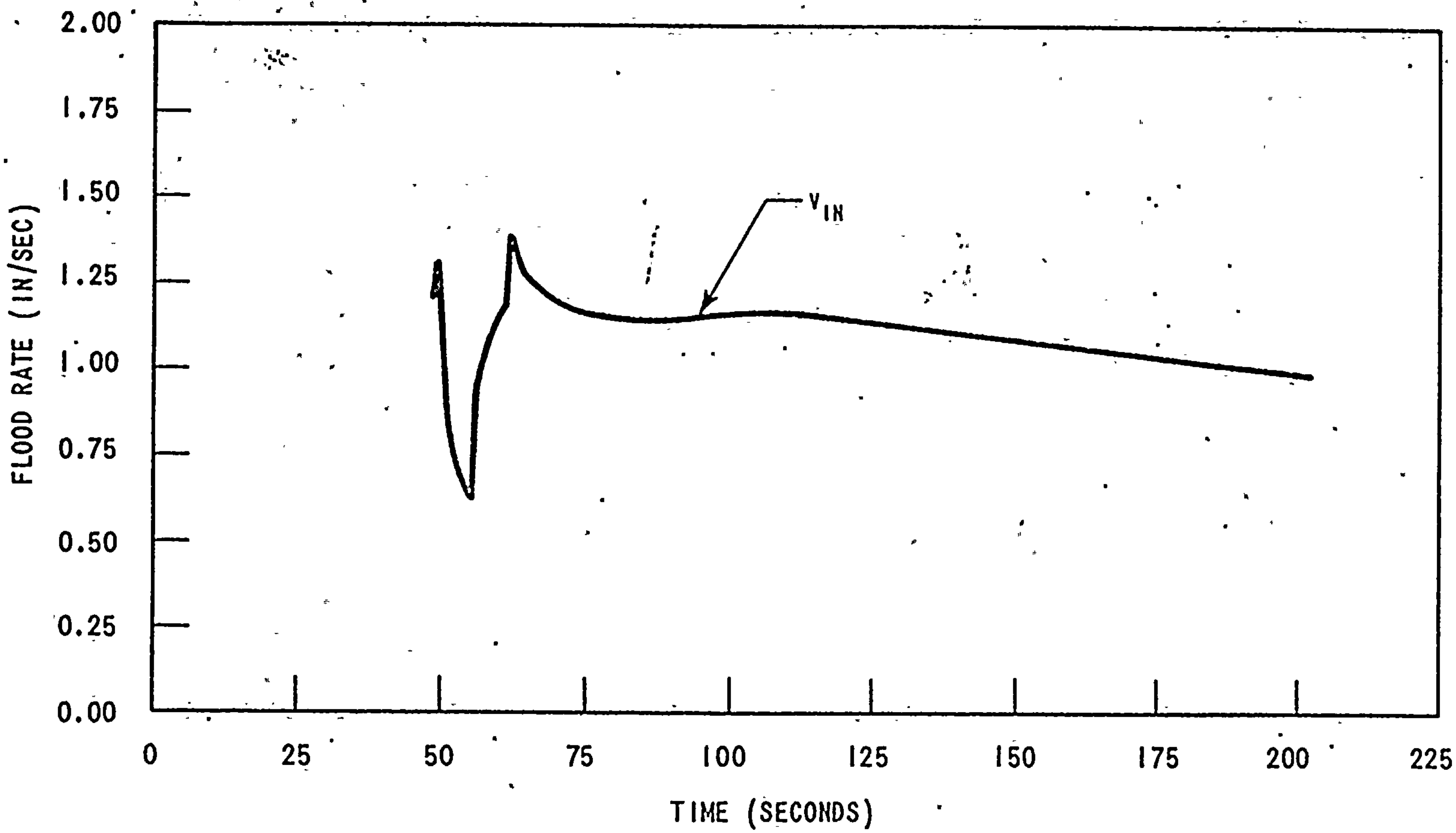


Figure 10b Reflood Transient - DECLG (C_D = 0.4)

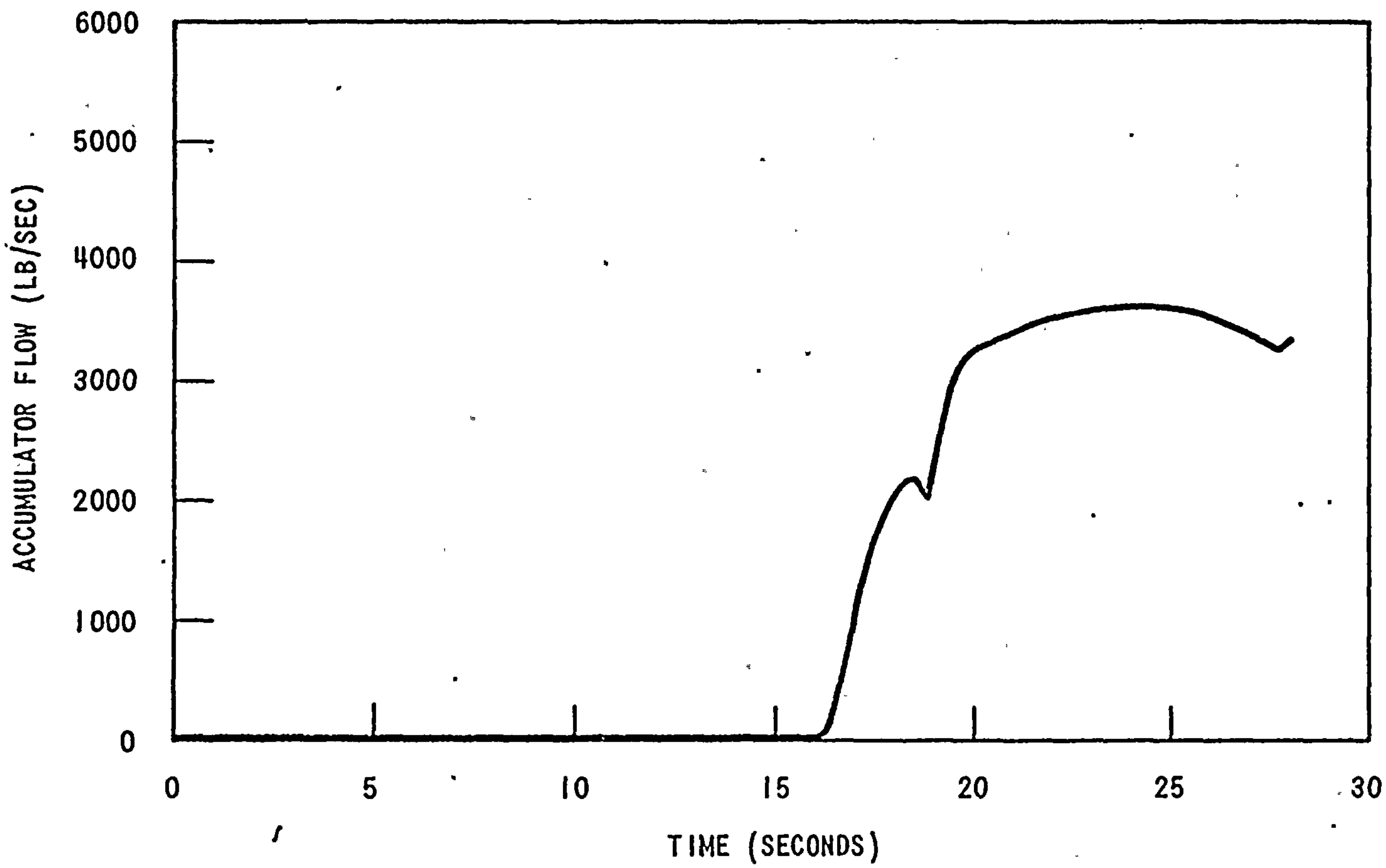


Figure 11 Accumulator Flow (Blowdown) - DECLG (C_D = 0.4)

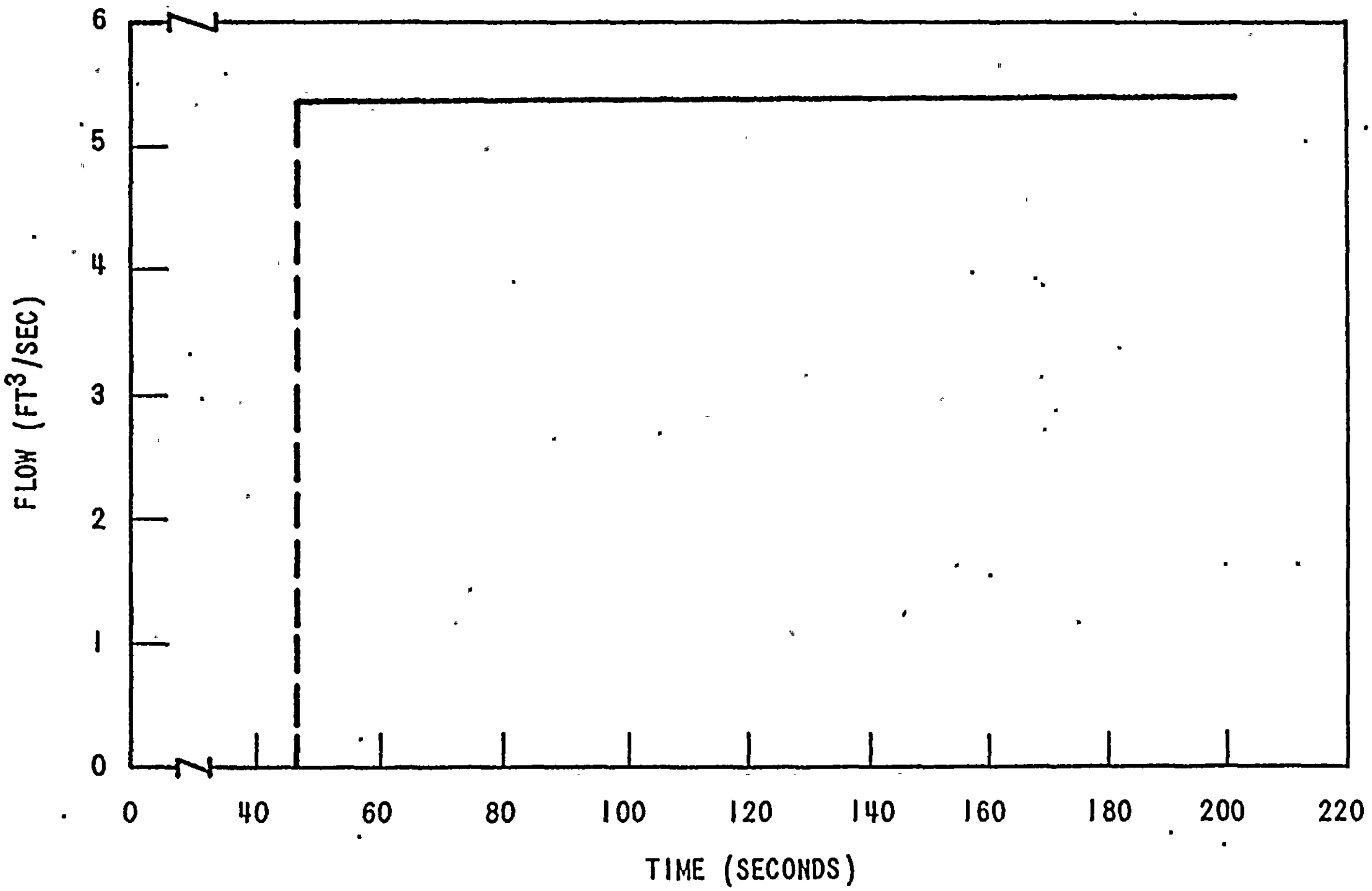


Figure 12 Pumped ECCS Flow (Reflood) · DECLG (C_D = 0.4)

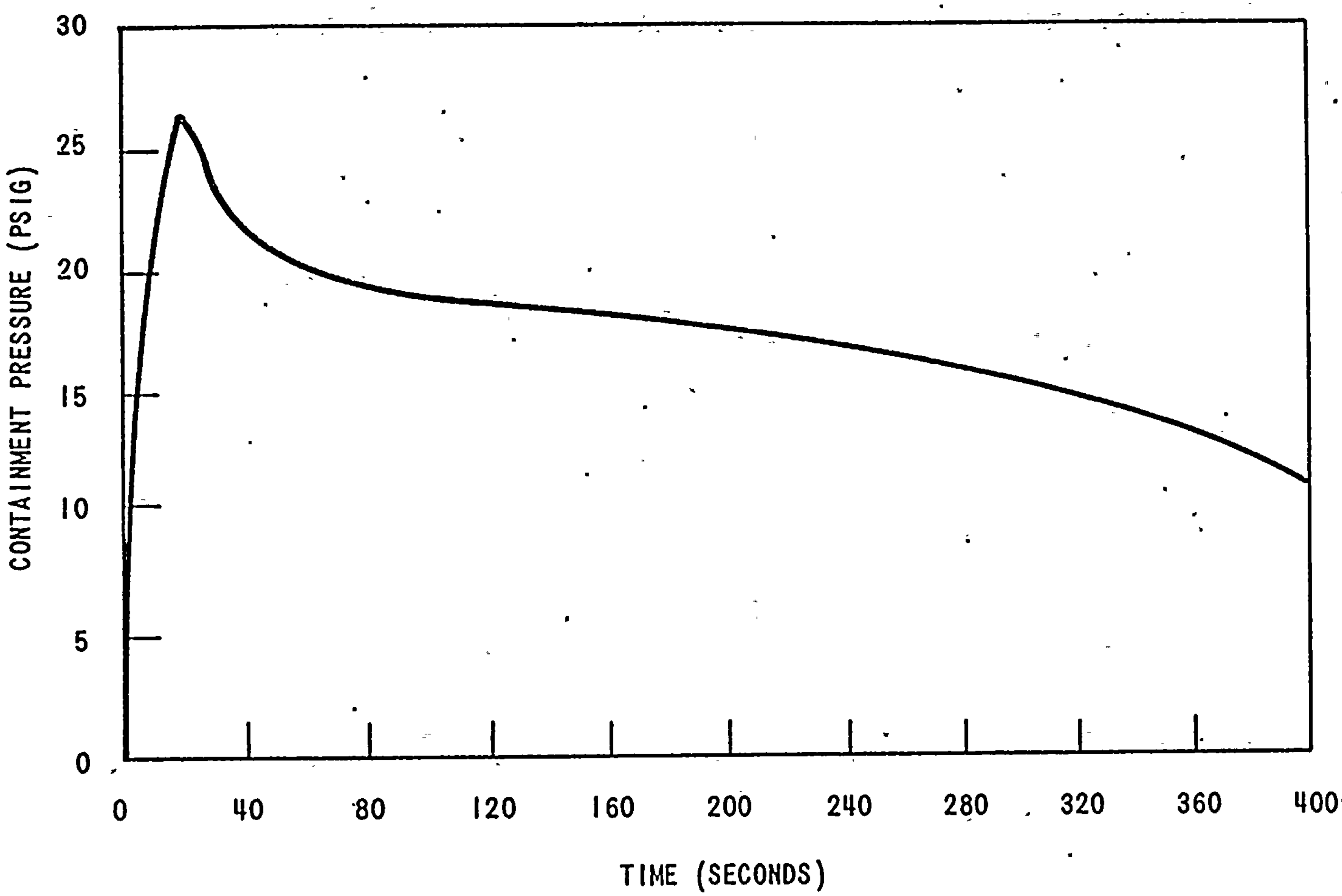


Figure 13 Containment Pressure - DECLG (C_D = 0.4)

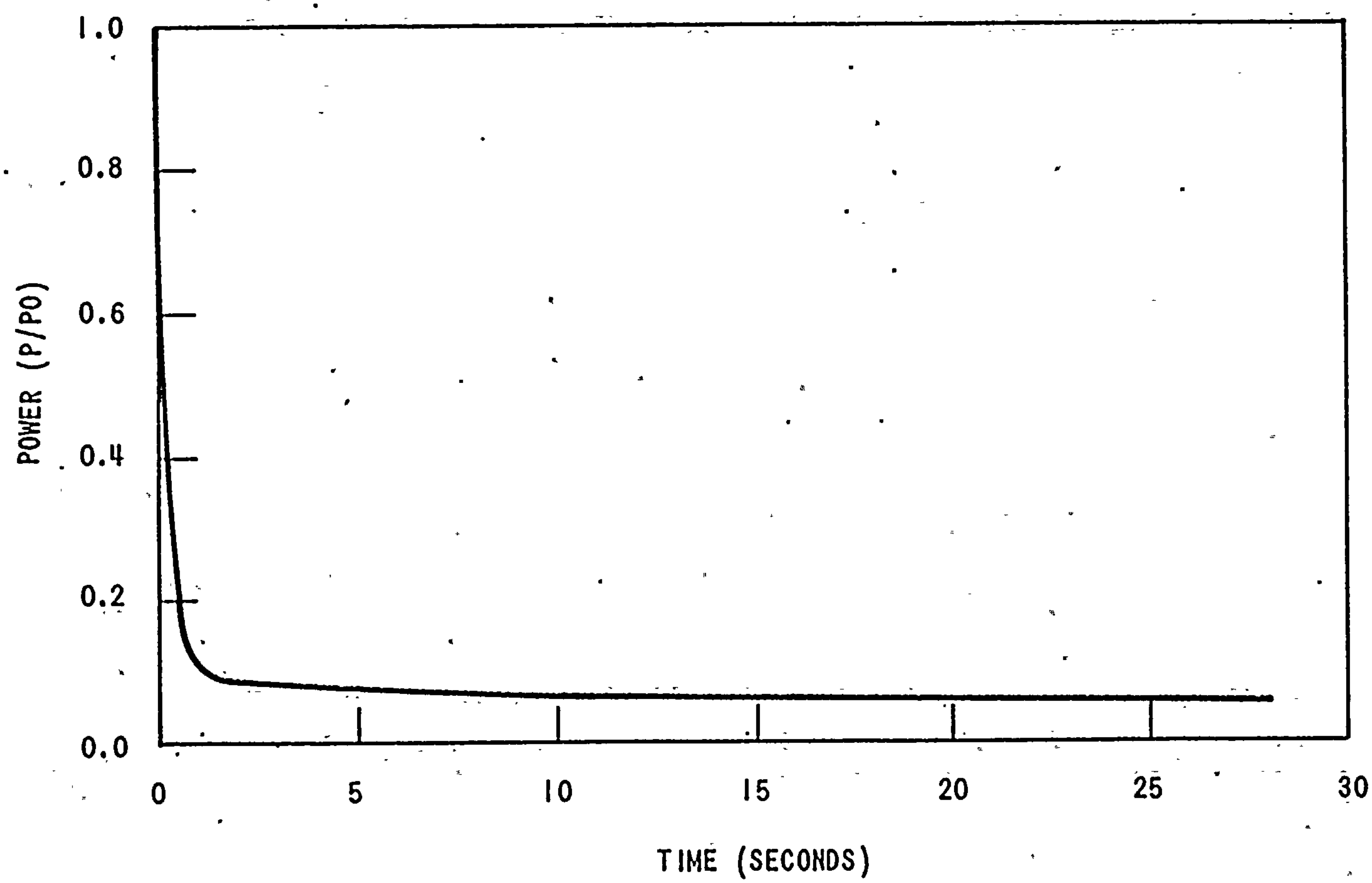


Figure 14 Core Power Transient - DECLG (C_D = 0.4)

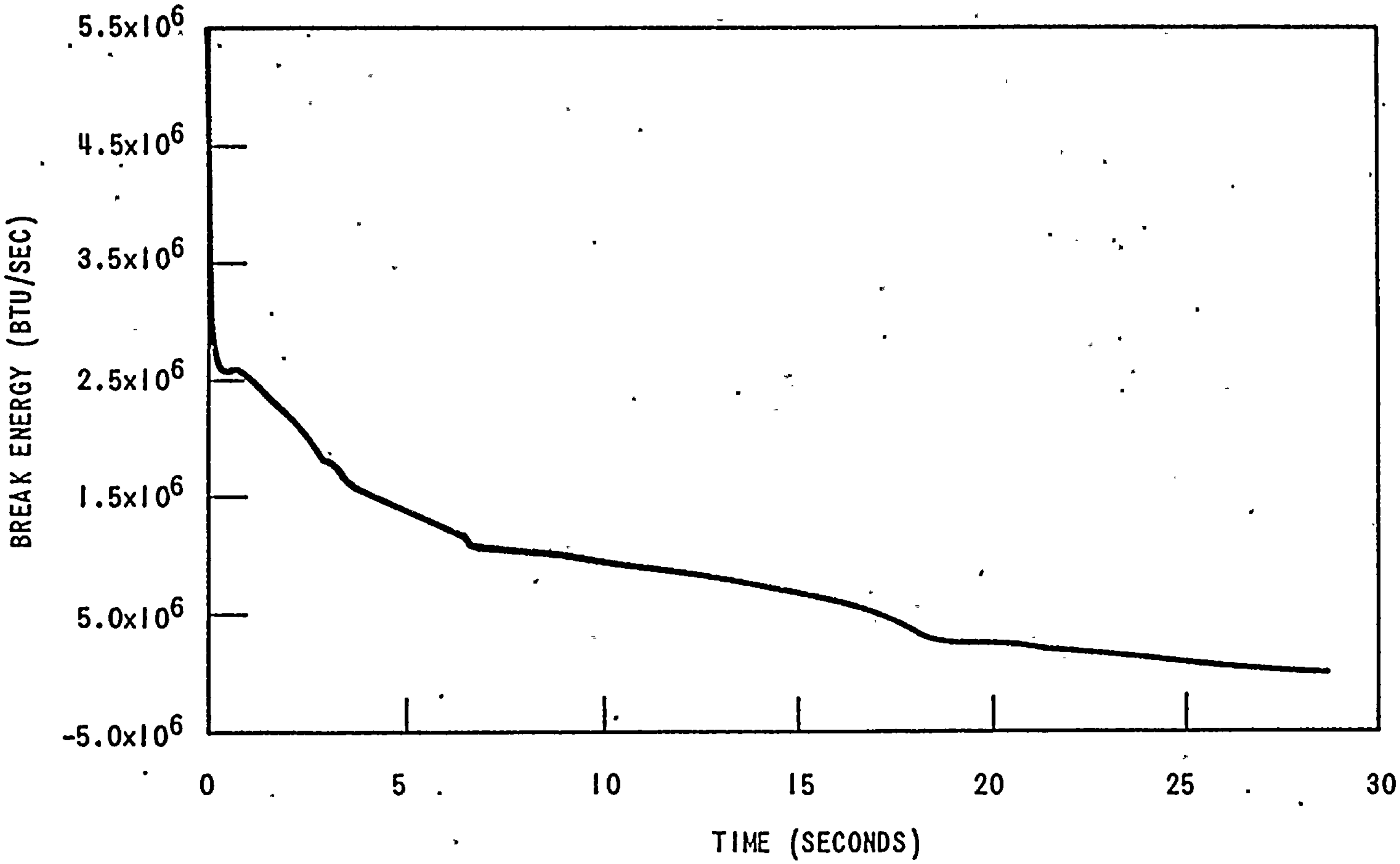


Figure 15 Break Energy Released to Containment - DECLG (C_D = 0.4)

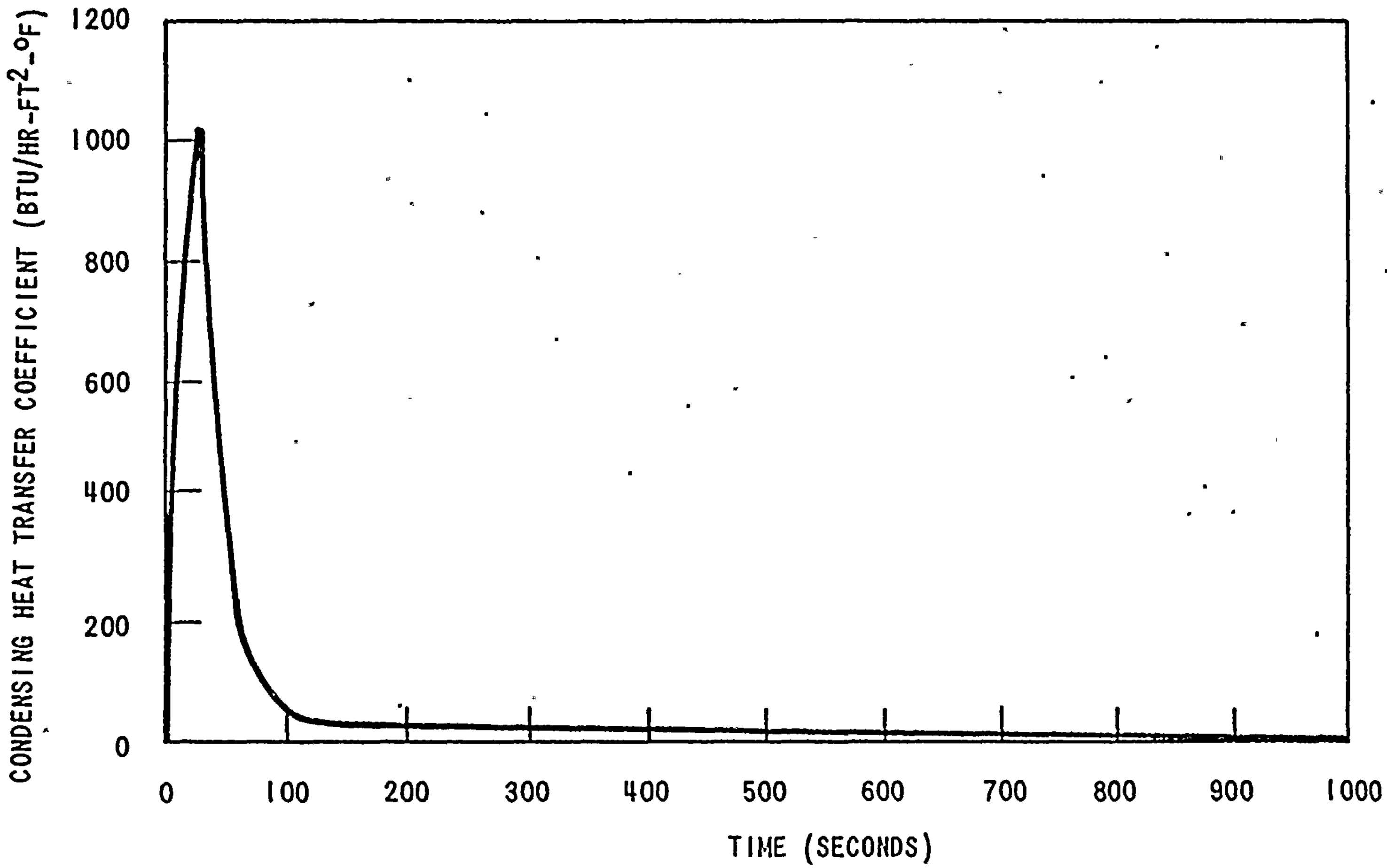


Figure 16 Containment Wall Heat Transfer Coefficient - DECLG (C_D = 0.4)

APPENDIX B

REACTIVITY INSERTION RATE vs. BORON CONCENTRATION
TURKEY POINT UNITS 3 & 4

Boron Dilution Analysis

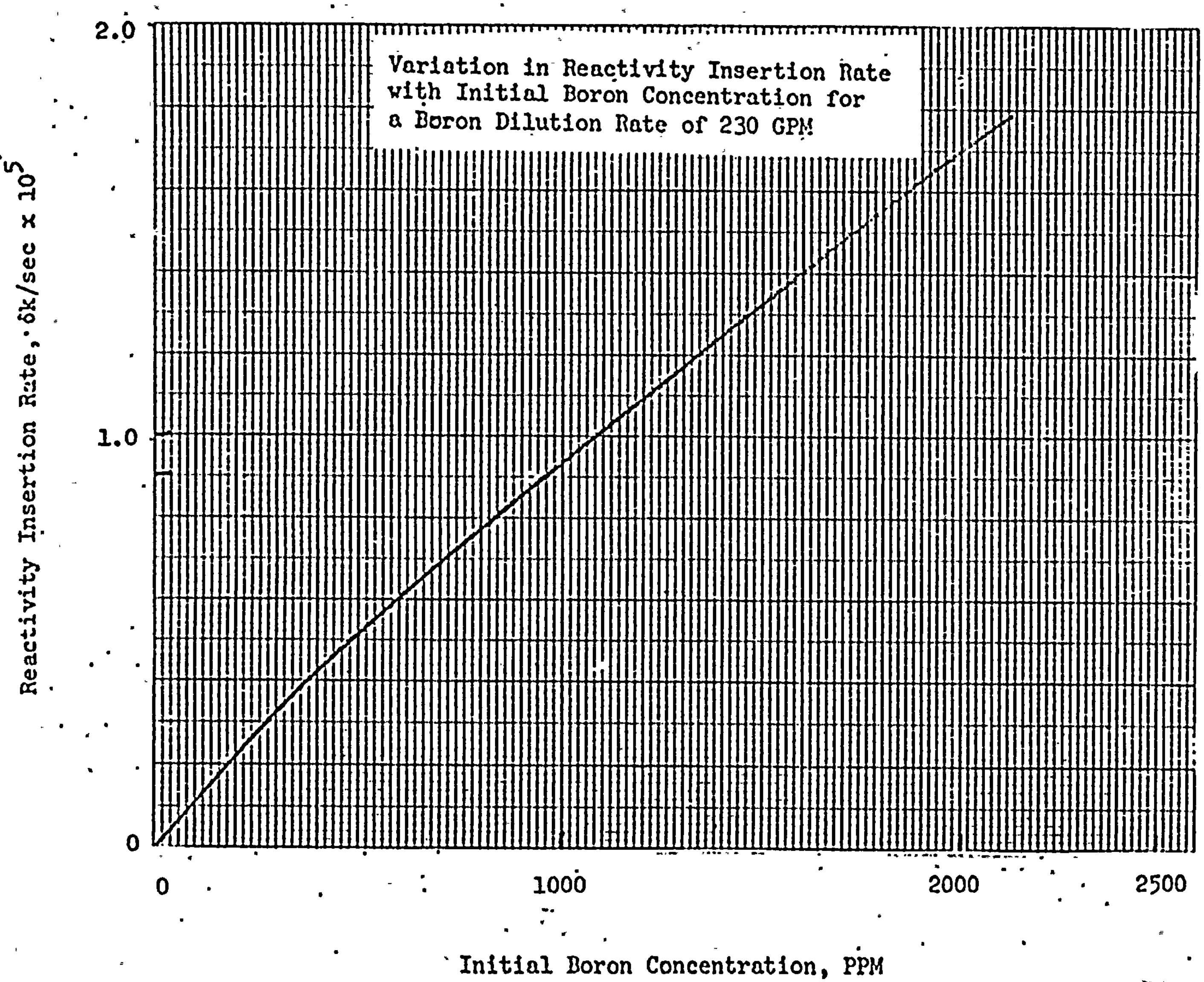
Section 14.1.5 of the Turkey Point Units 3 & 4 FSAR shows that for a boron dilution event the operator has sufficient time to identify the problem and terminate the dilution before the reactor returns critical or loses shutdown capability. The standard acceptance criteria and FSAR calculated values for operator action are summarized below:

MODE	FSAR (minutes)	ACCEPTANCE CRITERIA (minutes)
Refueling	> 120	30
Startup	> 240	15
Power		
a. Manual Control	> 15	15
b. Automatic Control	21	15

Steam generator tube plugging has no affect on the analysis at refueling conditions since only the reactor vessel volume is assumed active. The coolant loop volume is conservatively assumed stagnant.

For dilution during startup and at power, there is an effect due to the reduction in primary coolant volume. The effective volume of primary coolant in the steam generator tubes is conservatively assumed to be reduced by 25% (∿ 510 cubic feet). Thus the total volume assumed in the analysis has been reduced from 7800 cubic feet to 7290 cubic feet. This translates into approximately a 7% reduction in the originally calculated dilution time from startup conditions (240 minutes). This is still significantly greater than the required operator action time, therefore no safety concern exists.

For dilutions during power operation a highly conservative reactivity insertion rate of 1.1×10^{-5} δ k/sec was assumed in the FSAR consistent with an initial boron concentration of 1200 ppm. FSAR figure 14.1.5-1 (Reactivity Insertion Rate vs Boron Concentration) has been recalculated consistent with the lower primary value and is attached. The results show that the reactivity insertion rate assumed in the FSAR is still conservative. Therefore no additional analysis is required. It should be noted, however, that the FSAR analysis is still highly conservative with respect to the current cycles since the analysis assumed that only 1% shutdown margin is available. The Turkey Point Units have been designed such that > 2.5% shutdown margin is always available for BOL conditions. The result is that operator action times would be > 70 minutes with the more realistic value.


Variation in Reactivity Insertion Rate
with Initial Boron Concentration for
a Boron Dilution Rate of 230 GPM
Reactivity Insertion Rate, δk/sec x 10^5
2.0
1.0
0
0
1000
2000
2500
Initial Boron Concentration, PPM

FLORIDA POWER & LIGHT COMPANY

INTER-OFFICE CORRESPONDENCE

TO	R. E. Uhrig	LOCATION	Miami, Florida
		DATE	January 26, 1978
FROM	A. D. Schmidt	COPIES TO	J. R. Bensen/C. O. Woody G. E. Liebler/932.1 TP H. E. Yaeger/J. K. Hays R. J. Acosta N. F. Ajluni G. D. Whittier PRN-LI-78-20
SUBJECT:	TURKEY POINT UNITS 3 & 4 PROPOSED TECH SPEC CHANGE ECCS REANALYSIS (19% S/G PLUGGING)		

The subject proposal is attached for your review and forwarding to the NRC. It has been reviewed and approved by the PNSC and CNRB.

J R Bensen

for A. D. Schmidt

MAS/lah

Attachment