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FPL

FLORIDA POWER & LIGHT COMPANY

Regulatory Docket File

June 8, 1977
L-77-172

RECEIVED
JUN 13 1977
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Office 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 No. 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 (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.

This proposal is being submitted as a result of a re-evaluation of ECCS cooling performance calculated in accordance with an approved Westinghouse Evaluation Model. The proposed change 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 is revised such that the limit on the Heat Flux Hot Channel Factor ($F_q$) for both Units 3 and 4 is reduced from 2.22 to 2.20 for steam generator tube plugging in excess of 10%.

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

Pages B3.2-4 and B3.2-6 present the basis for the revised limit on $F_q$ for both Units 3 and 4.

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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. A safety evaluation is attached.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU/WAK/cmp

Attachments

cc: Mr. Norman C. Moseley, Region II
Robert Lowenstein, Esquire

reactivity insertion upon ejaction 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rod cannot be moved by the drive mechanism, shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the inoperable rod.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. 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. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_q(Z) \leq (2.22/P)^* \times K(Z)$ for $P > .5$

$F_q(Z) \leq (4.44)^* \times K(Z)$ for $P \leq .5$

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

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 and Z is the core height location of Fq.

* For tube plugging in excess of 10%, these values become (2.20/P) and (4.40) respectively.

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,

An upper bound envelope of 2.22* times 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) 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

*For steam generator tube plugging in excess of 10%, this value becomes 2.20.

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 of 2.22* times 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 ±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-

* For steam generator tube plugging in excess of 10%, this value becomes 2.20.

# SAFETY EVALUATION

## I. Introduction

This safety evaluation and the attached Westinghouse ECCS re-evaluation support the following proposed change to the Technical Specifications:

(1) The maximum allowable nuclear peaking factor ($F_q$) is decreased from 2.22 to 2.20, for steam generator tube plugging in excess of 10%.

## II. Discussion

A re-evaluation of ECCS cooling performance calculated in accordance with an approved Westinghouse Evaluation Model has been performed. The re-evaluation shows that for breaks up to and including the double ended severence of a reactor coolant pipe, the ECCS will meet the Acceptance Criteria presented in 10 CFR 50.46. The detailed re-evaluation is attached, and shows that, at a core power level of 102% of 2200 Mwt and a minimum accumulator water volume of 875 $ft^3$ per accumulator, the maximum allowable nuclear peaking factor is 2.20 for steam generator tube plugging in excess of 10%.

The attached Westinghouse ECCS re-evaluation assumed:

1. 15% steam generator tube plugging
2. $F_q = 2.20$
3. 875 $ft^3$ accumulator minimum water volume
4. 2200 Mwt core power level

## III. Conclusions

Based on these considerations, (1) the proposed change does not increase the probability or consequences of accidents or malfunctions of equipment important to safety and does not reduce the margin of safety as defined in the basis for any technical specification; therefore, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

| | DECL (CD=0.4) (Sec) |
|---|---|
| START | 0.0 |
| Rx Trip Signal | 0.595 |
| S. I. Signal | 0.67 |
| Acc. Injection | 16.0 |
| End of Bypass | 27.16 |
| End of Blowdown | 27.31 |
| Bottom of Core Recovery | 46.24 |
| Acc. Empty | 60.8 |
| Pump Injection | 25.67 |

TABLE 2

LARGE BREAK

| | DECL (CD=0.4) | |
|---|---|---|
| **Results** | | |
| Peak Clad Temp. °F | 2173 | |
| Peak Clad Location Ft. | 6.5 | |
| Local $Zr/H_2O$ Reaction (max)% | 11.655 | |
| Local $Zr/H_2O$ Location Ft. | 6.0 | |
| Total $Zr/H_2O$ Reaction % | <0.3 | |
| Hot Rod Burst Time sec | 22.6 | |
| Hot Rod Burst Location Ft. | 6.0 | |
| **Calculation** | | |
| Core Power Mwt 102% of | | 2200 |
| Peak Linear Power kw/ft 102% of | | 12.499 |
| Peaking Factor | | 2.20 |
| Accumulator Water Volume ($ft^3$) | | 875 (per accumulator) |
| Fuel region + cycle analyzed | Cycle | Region |
| UNITS 3 & 4 | 3 | 3 |

TABLE 3

LARGE BREAK

CONTAINMENT DATA (DRY CONTAINMENT)

| | | |
|---|---|---|
| NET FREE VOLUME | $1.55 \times 10^6$ | $Ft^3$ |
| 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)

LARGE BREAK
CONTAINMENT DATA
(DRY CONTAINMENT)

STRUCTURAL HEAT SINKS

| Thickness (In) | | Area ($Ft^2$) |
|---|---|---|
| Steel | 0.03 | 31,400 |
| Steel | 0.063 | 107,158 |
| Steel | 0.1 | 56,371 |
| Steel | 0.2 | 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 | 1268.7 |
| Steel | 5.5 | 1277.4 |
| Steel | 9.0 | 260.4 |
| Stainless | 0.14 | --- |
| Concrete | 24.0 | 14,392 |
| Stainless | 0.44 | 768 |
| Stainless | 2.126 | 3,704 |
| Stainless | 0.007 | 102,400 |
| Concrete | 24.0 | 59,132 |

TABLE 4

REFLOOD MASS AND ENERGY RELEASES FOR LIMITING BREAK (DECLG CD = 0.4)

| Time, Sec | Total Mass Flowrate LBm/Sec | Total Energy Flowrate ($10^5$ BTU/Sec) |
|---|---|---|
| 46.235 | 0.0 | 0.0 |
| 48.36 | 0.0 | 0.0 |
| 53.982 | 35.09 | 0.4562 |
| 64.197 | 93.67 | 1.16 |
| 76.997 | 96.36 | 1.20 |
| 92.197 | 117.2 | 1.30 |
| 108.097 | 238.3 | 1.61 |
| 124.697 | 267.3 | 1.65 |
| 160.397 | 276.7 | 1.57 |
| 199.397 | 283.3 | 1.48 |

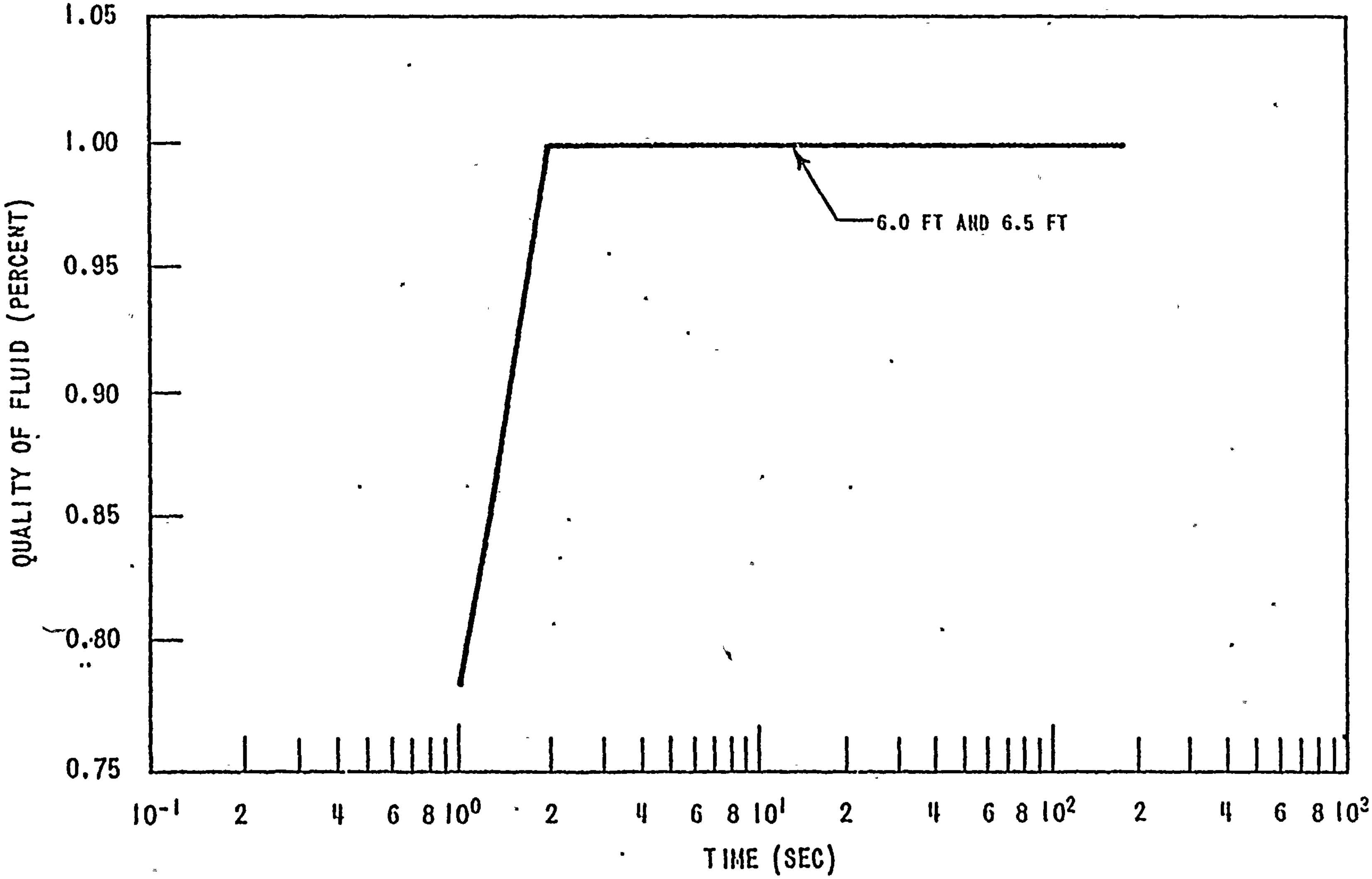


Figure 1. Fluid Quality - DECLG ($C_D$ = 0.4)

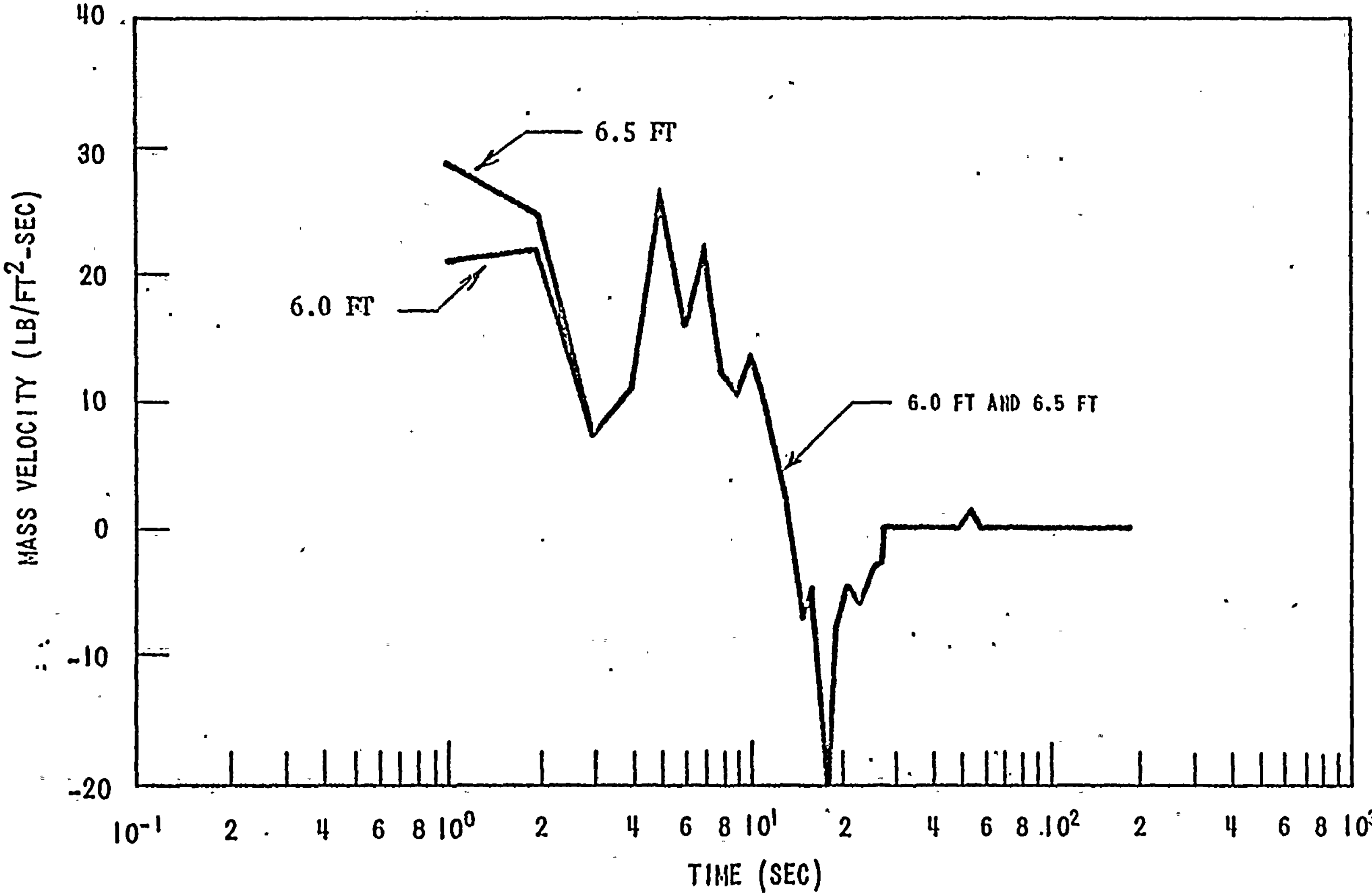


Figure 2. Mass Velocity - DECLG ($C_D$ = 0.4)

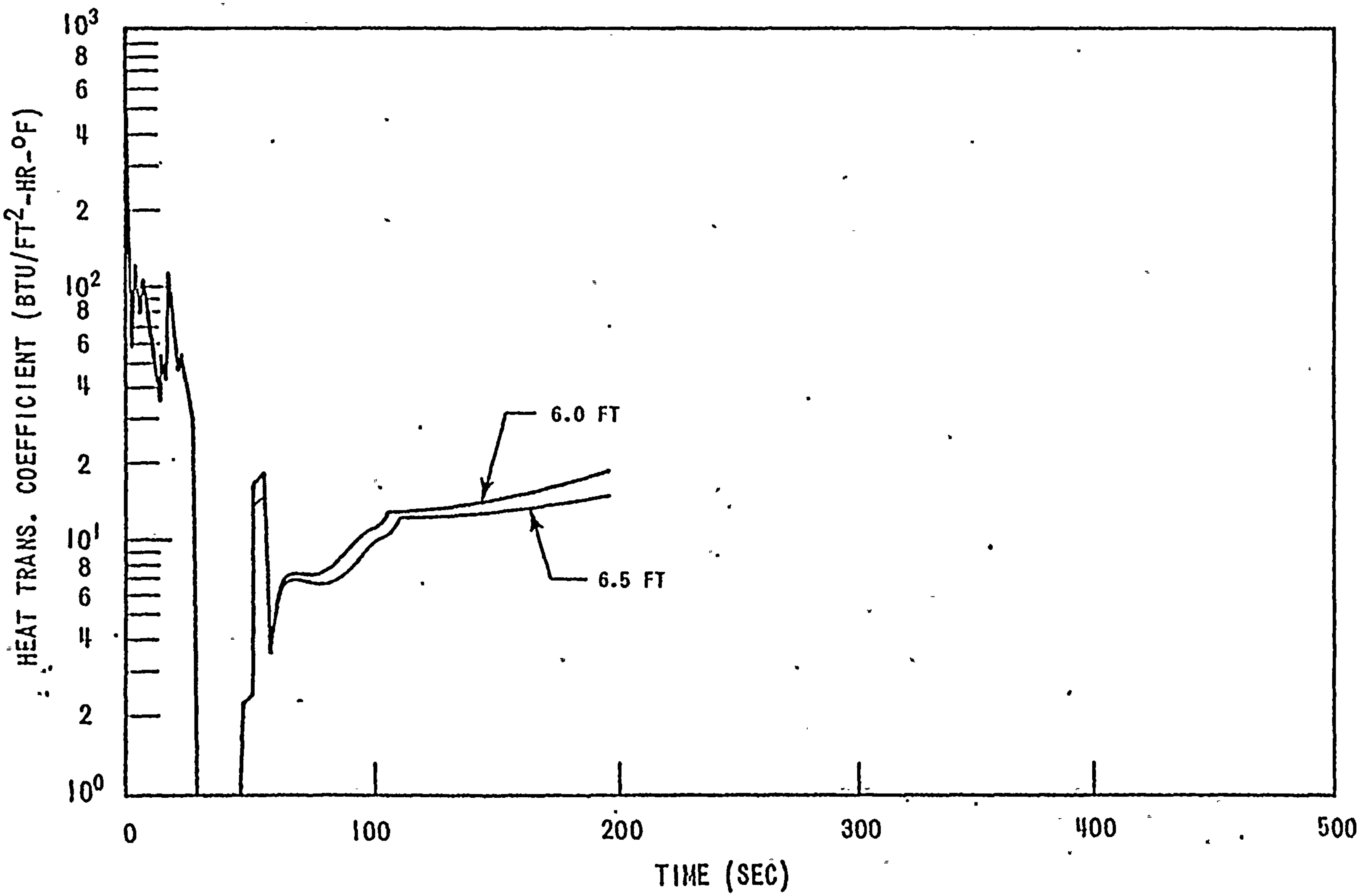


Figure 3. Heat Transfer Coefficient - DECLG ($C_D$ = 0.4)

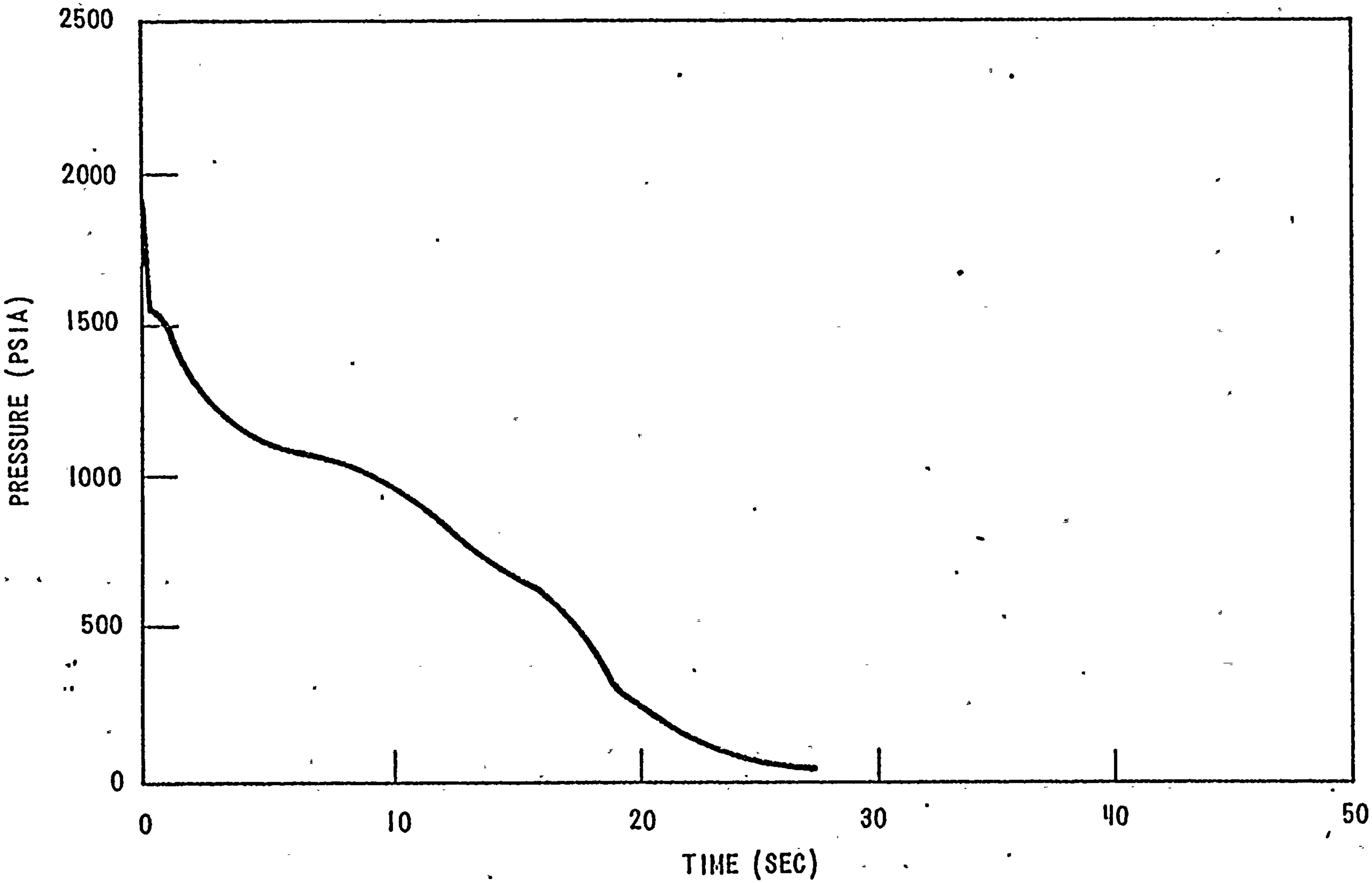


Figure 4. Core Pressure - DECLG ($C_D$ = 0.4)

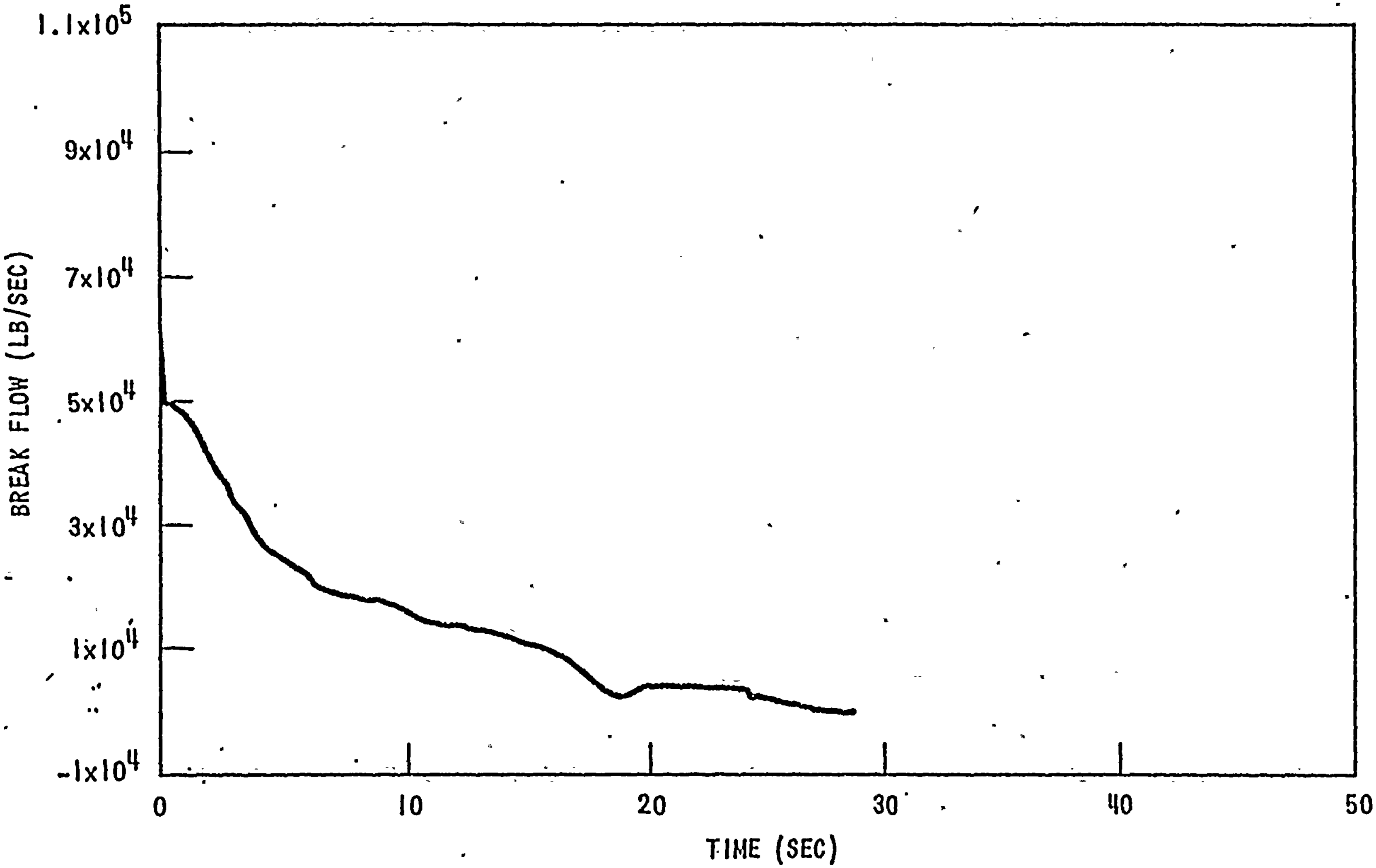


Figure 5. Break Flow Rate - DECLG ($C_D$ = 0.4)

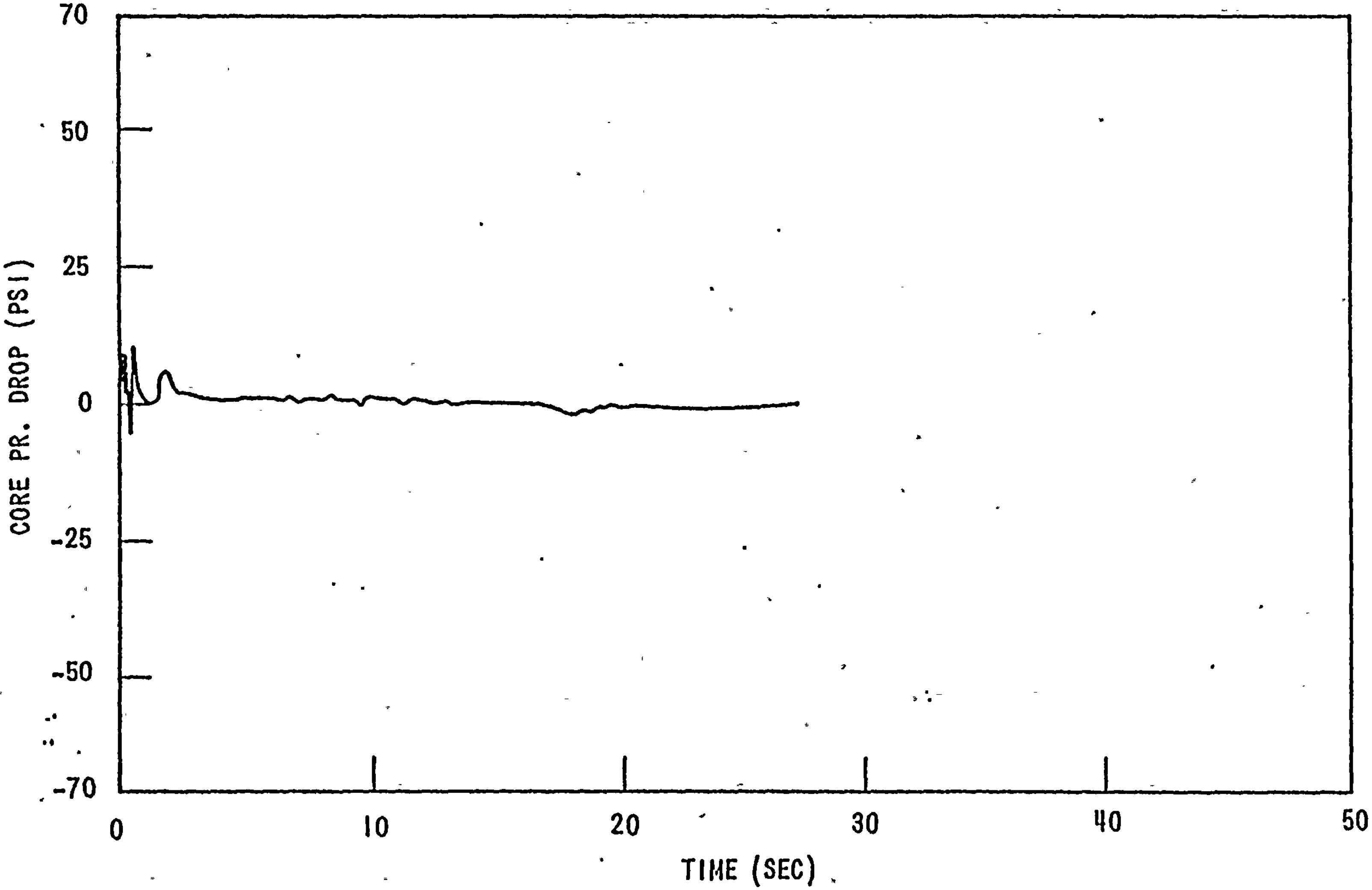


Figure 6. Core Pressure Drop - DECLG ($C_D$ = 0.4)

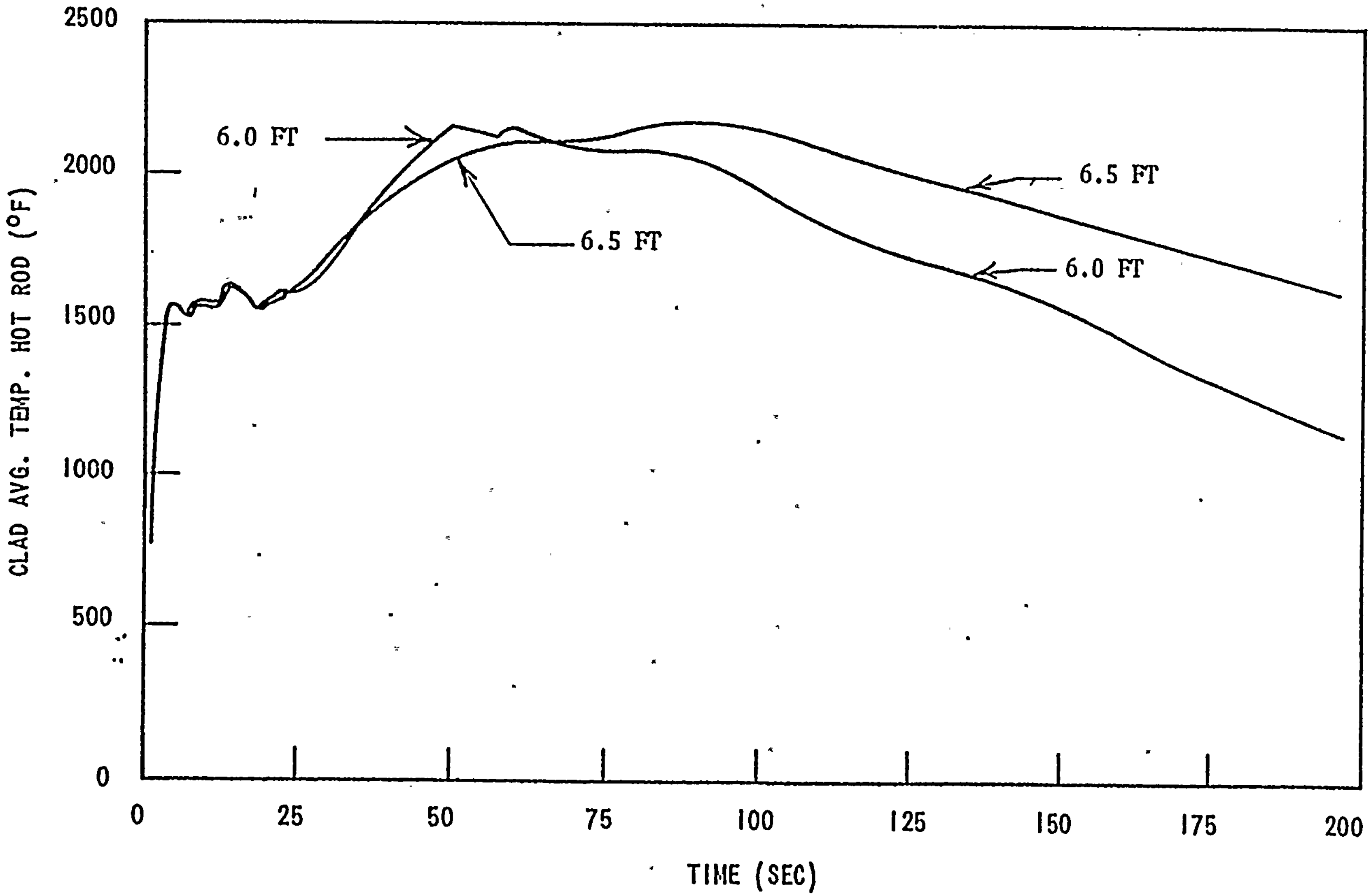


Figure 7. Peak Clad Temperature - DECLG ($C_D$ = 0.4)

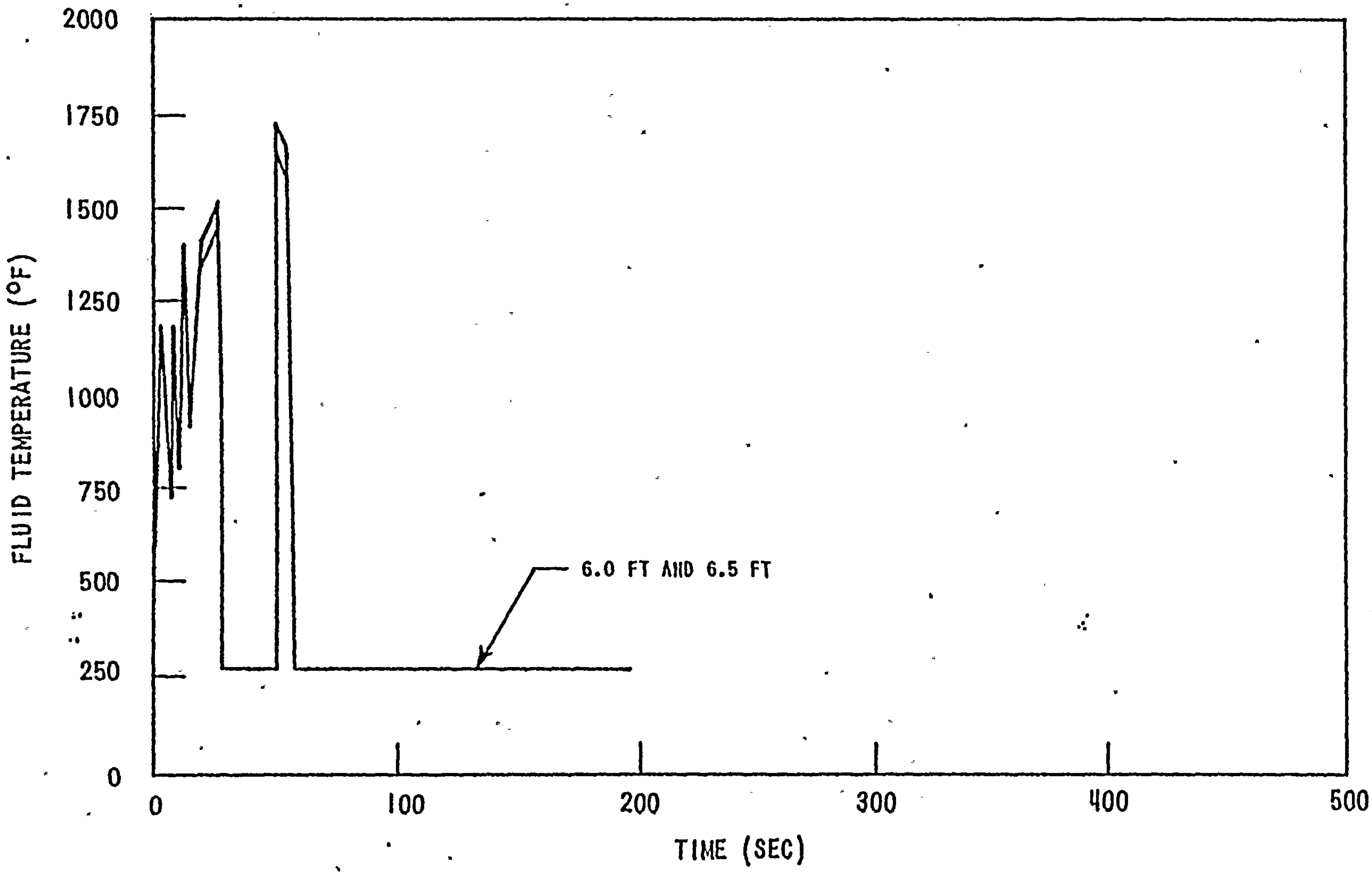


Figure 8. Fluid Temperature - DECLG ($C_D$ = 0.4)

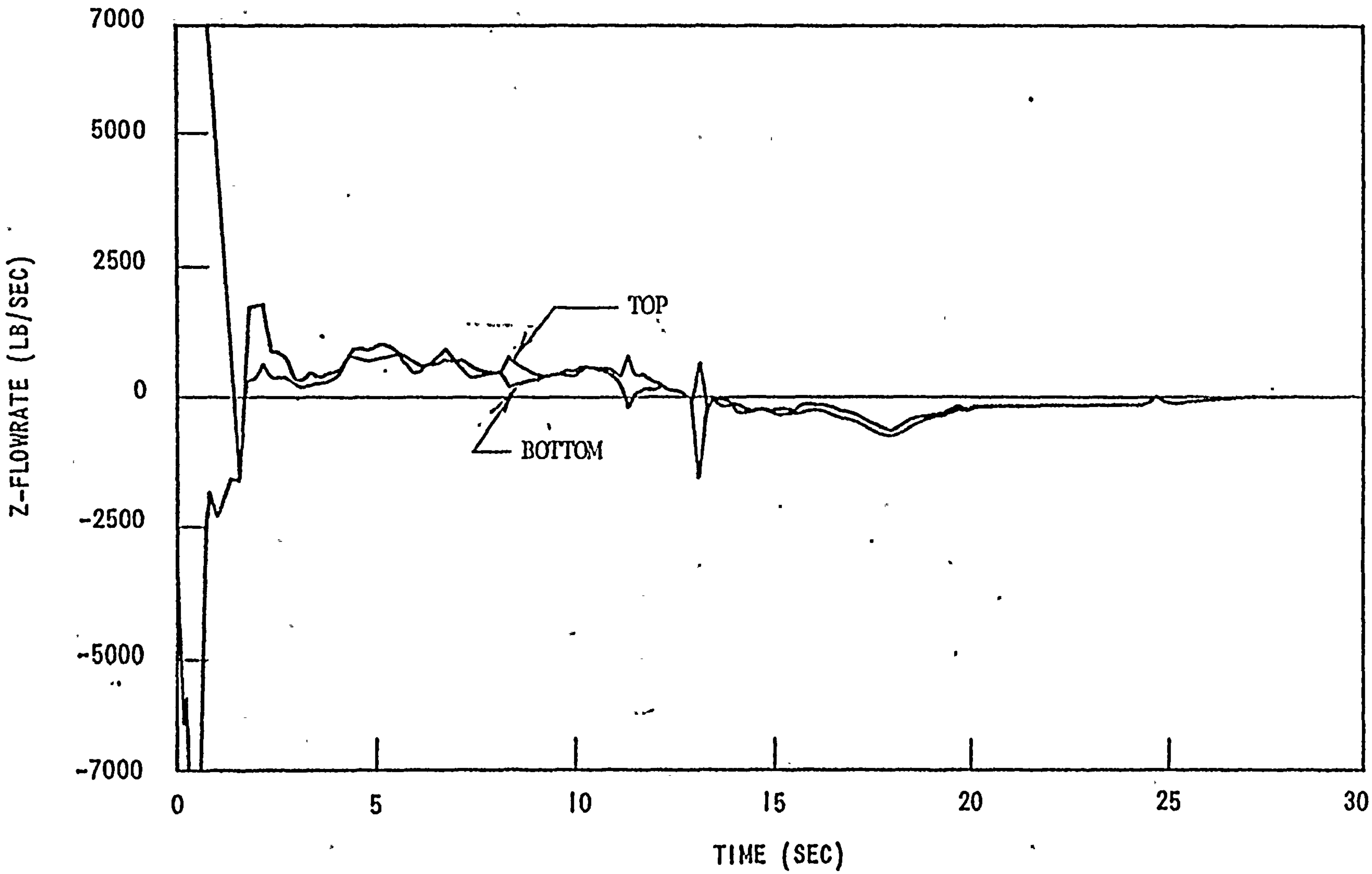


Figure 9. Core Flow - Top and Bottom - DECLG ($C_D$ = 0.4)

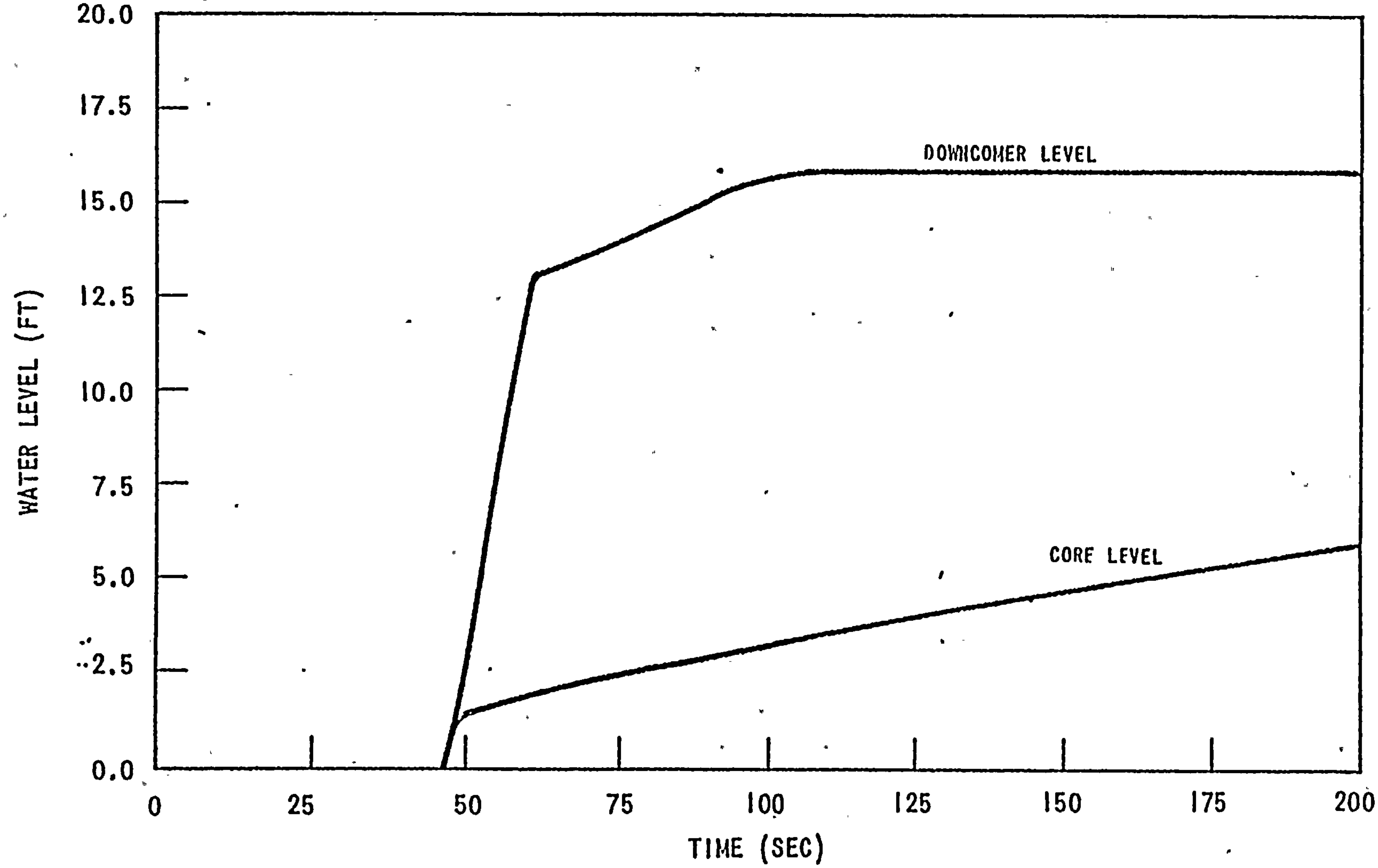


Figure 10. Reflood Transient - DECLG ($C_D$ = 0.4) Downcomer and Core Water Levels

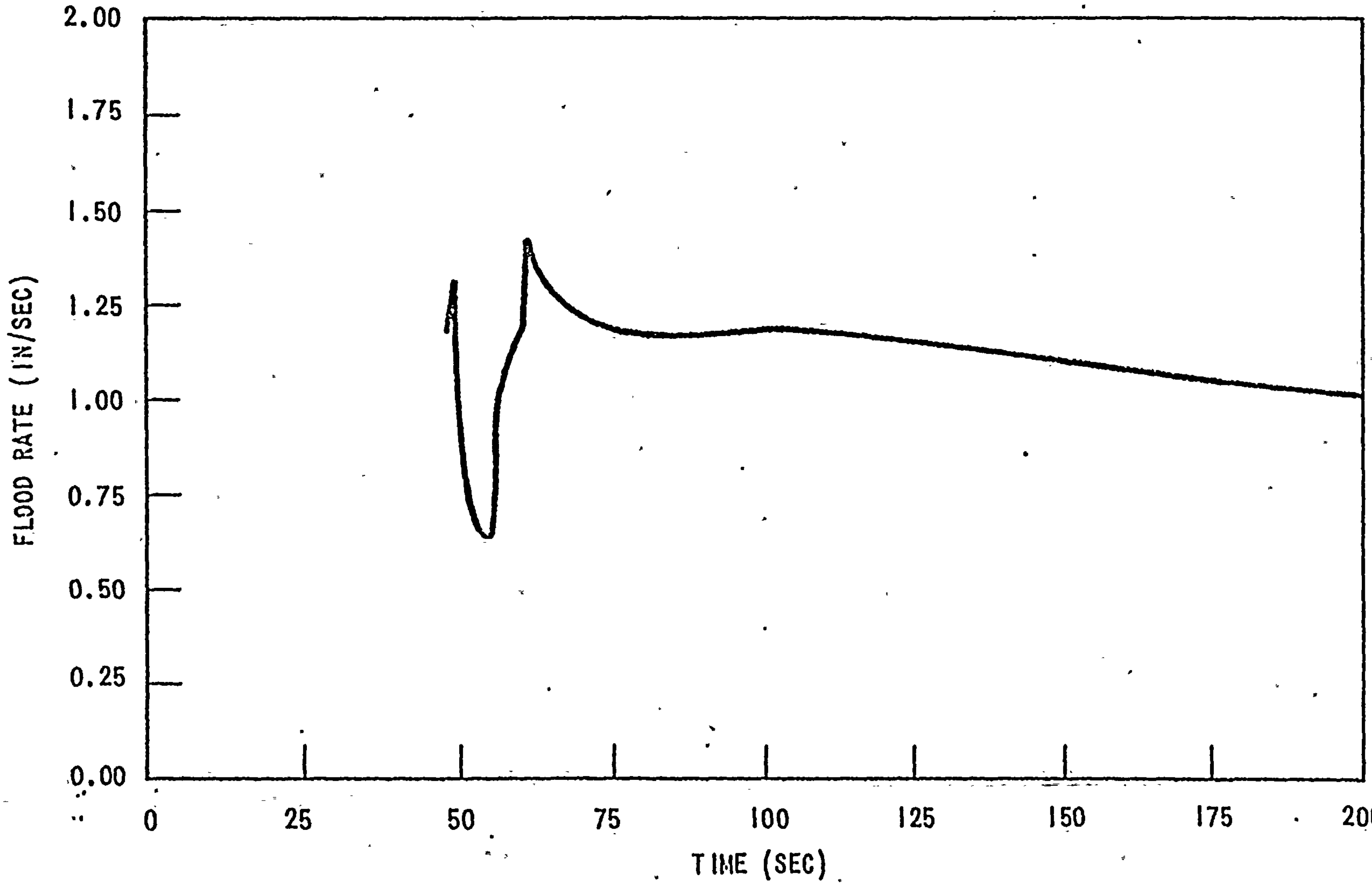


Figure 10a. Reflood Transient - DECLG ($C_D$ = 0.4) Core Inlet Velocity

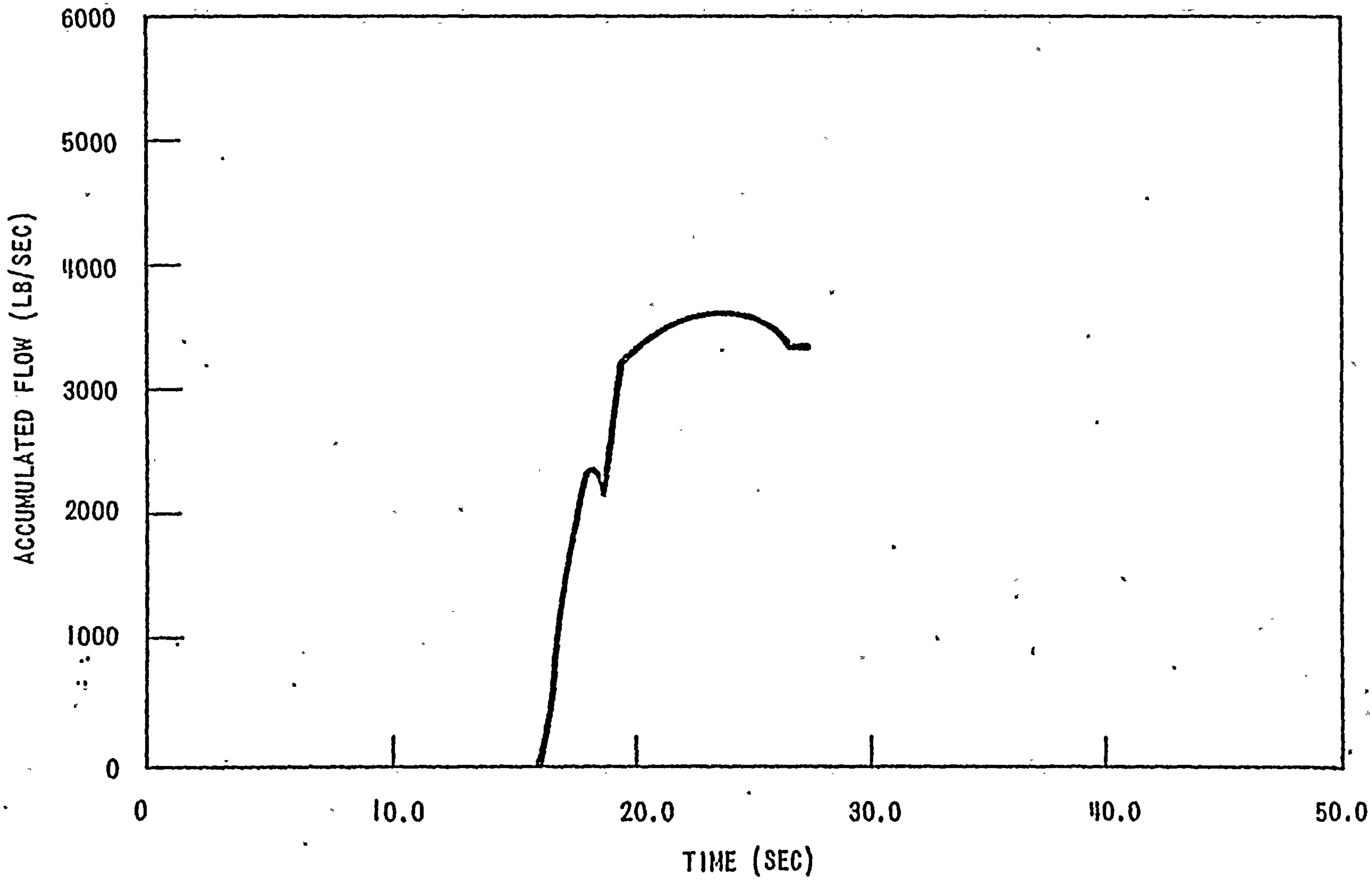


Figure 11. Accumulator Flow (Blowdown) - DECLG ($C_D$ = 0.4)

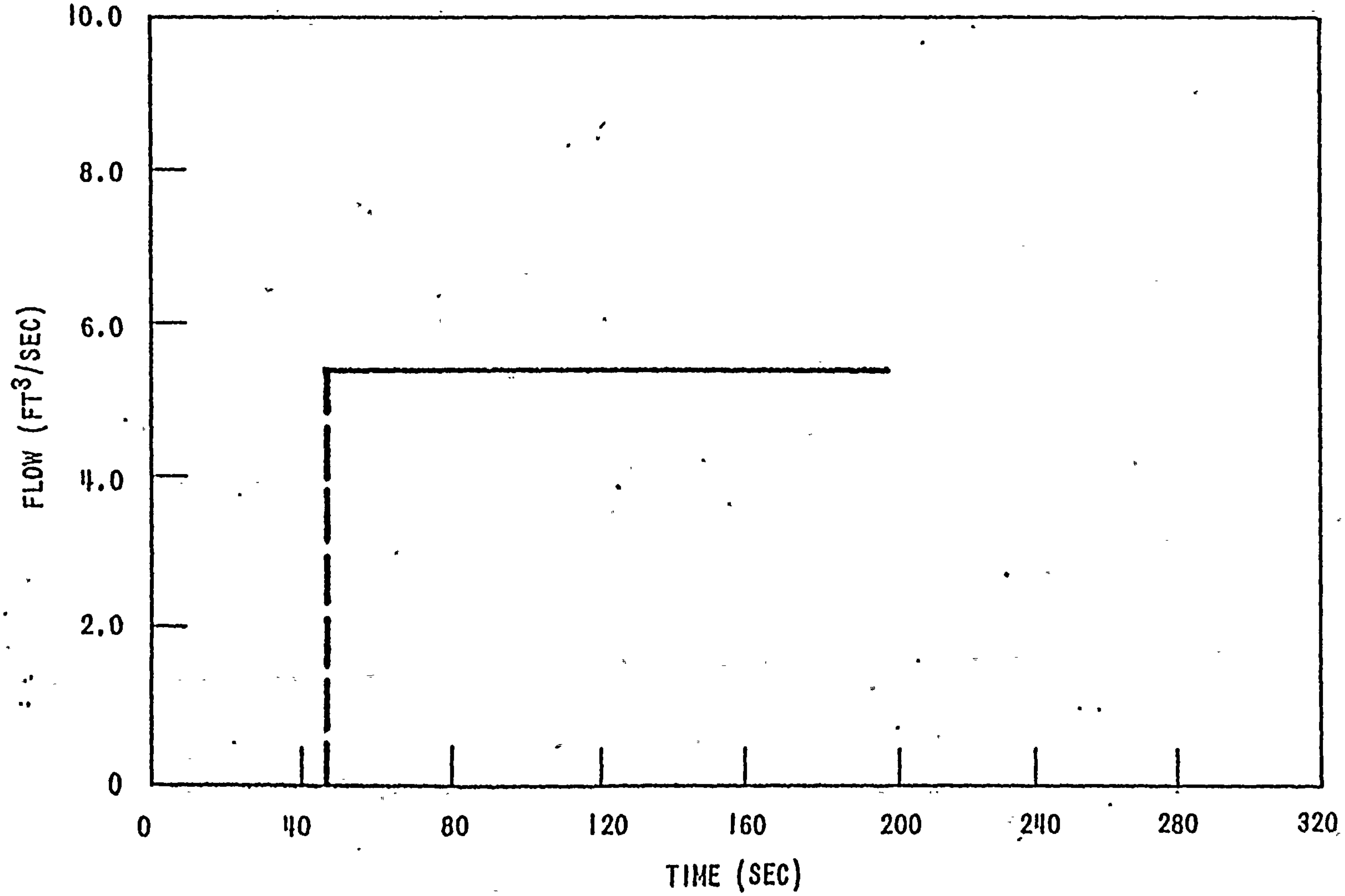


Figure 12. Pumped ECCS Flow (Reflood) - DECLG ($C_D$ = 0.4)

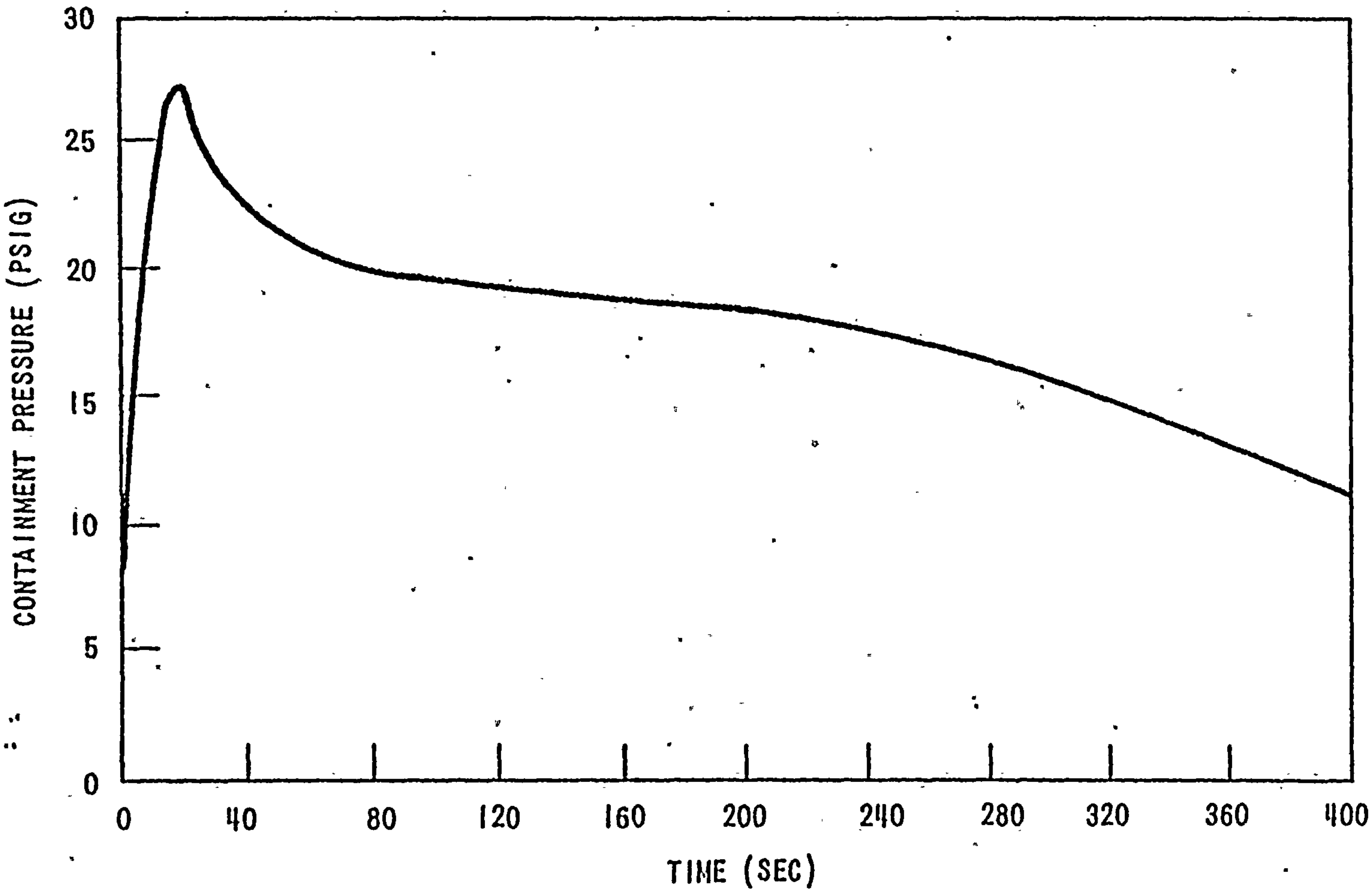


Figure 13. Containment Pressure - DECLG ($C_D$ = 0.4)

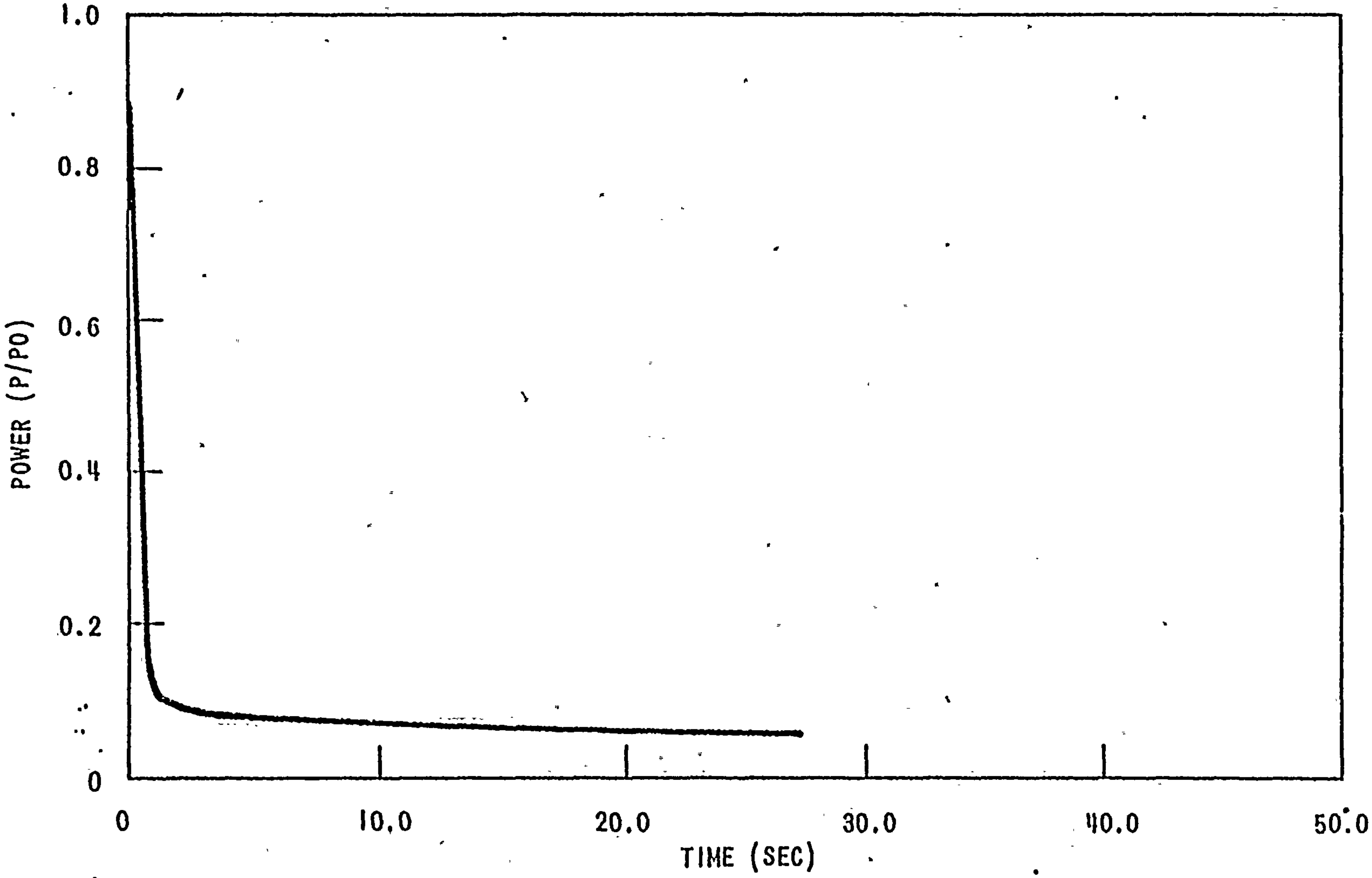


Figure 14. Core Power Transient - DECLG ($C_D$ = 0.4)

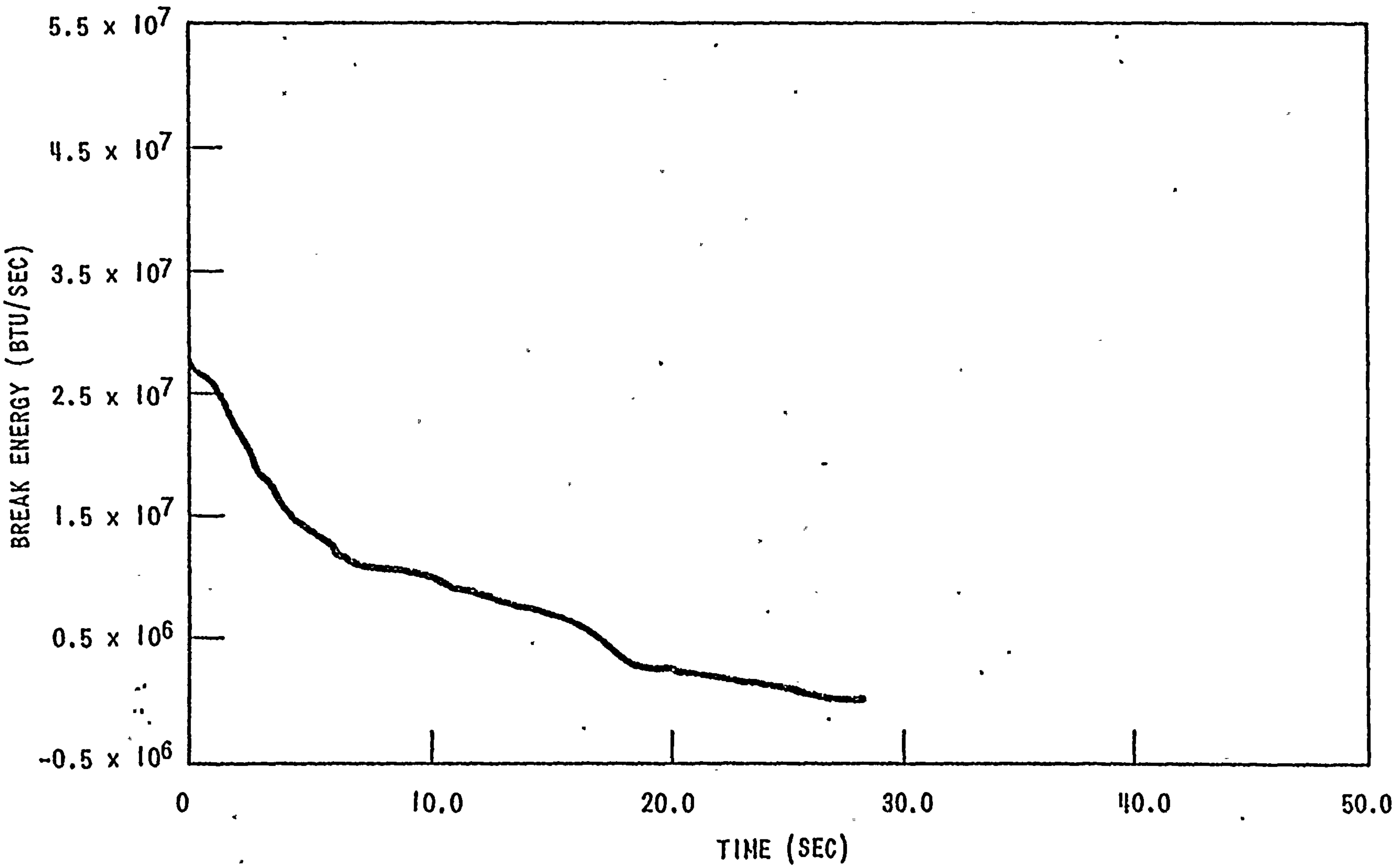


Figure 15. Break Energy Released to Containment

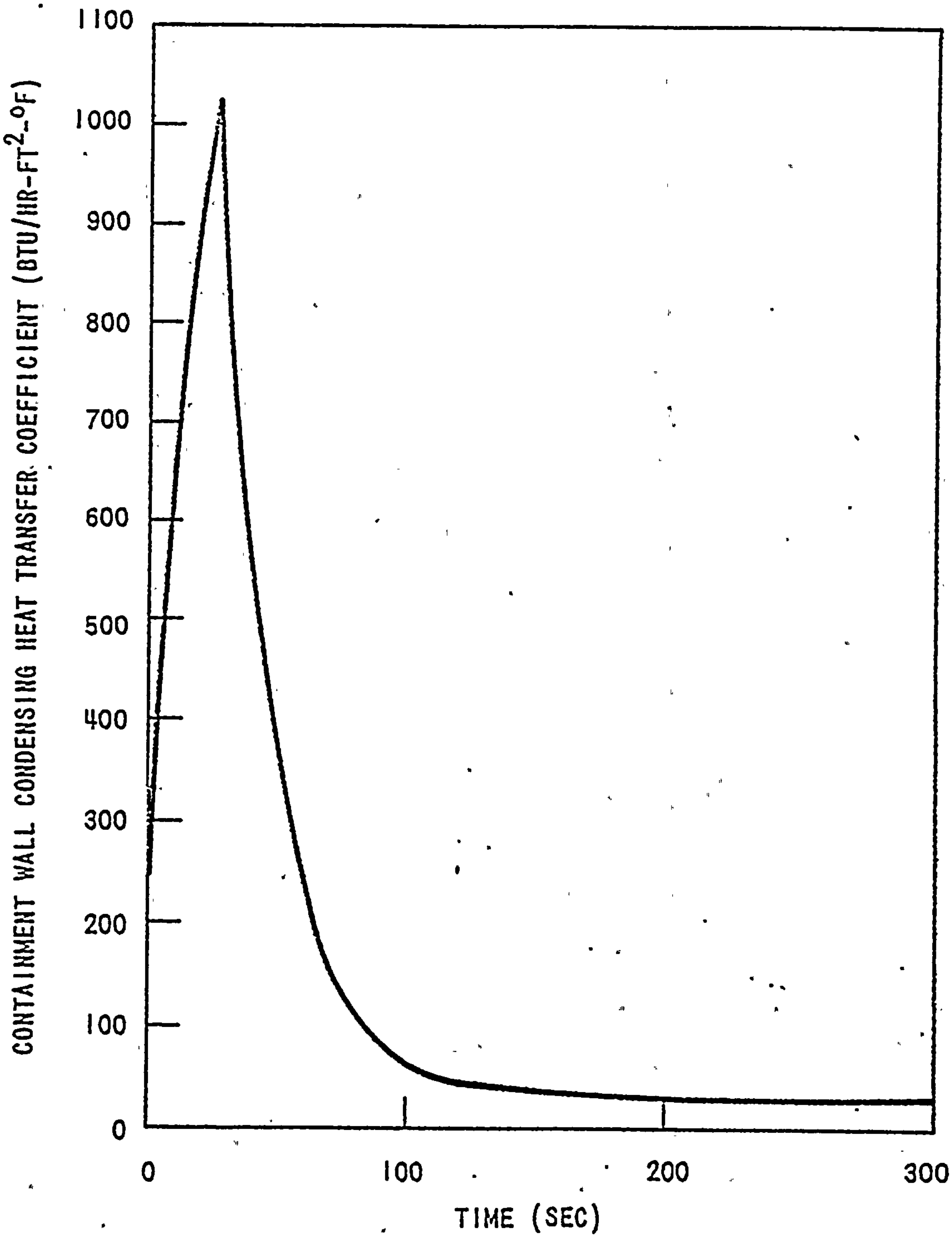


Figure 16. Containment Wall Condensing Heat Transfer Coefficient

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

8 day of June, 1977

Theresa M. Miranda

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 MAY 5, 1981 BONDED THRU MAYNARD BONDING AGENCY