REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS,
THREE LOOP OPERATION

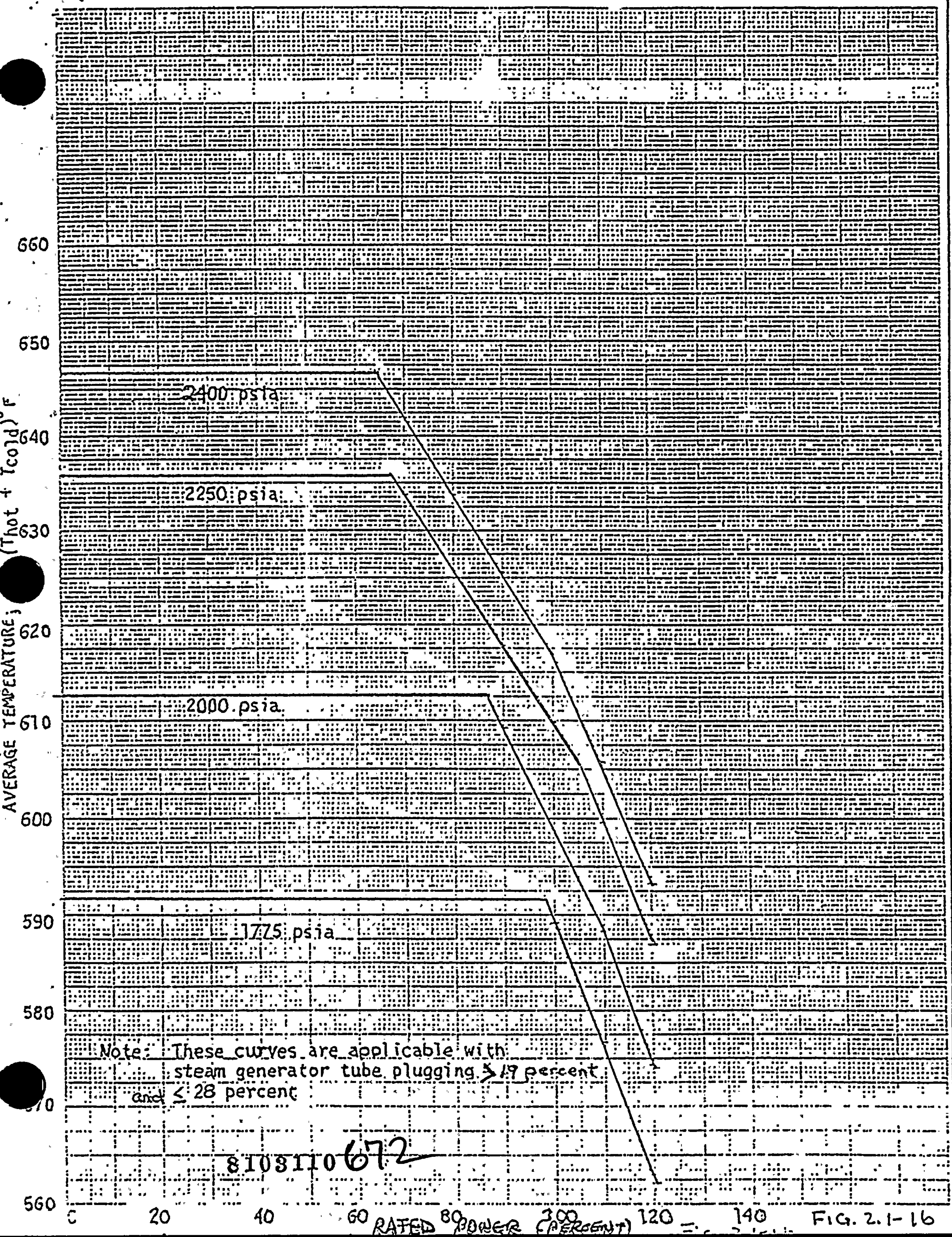

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FIG. 2.1-16

REACTOR COOLANT TEMPERATURE

Overtemperature $\Delta T \leq \Delta T_o$ [K_1 - 0.0107 (T-574) + 0.000453 (P-2235) - f (Δq)]

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

P = Pressurizer pressure, psig

f(Δq)= 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(Δq) = 0.

For each percent that the magnitude of (q_t - q_b) exceeds + 10 percent, the Delta-T trip setpoint 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 setpoint 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 $\leq$ 28 percent.

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

ΔT_o = Indicated T 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 Presssurizer 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.

Undervoltage 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 15% of narrow range instrument scale.

---

*This factor is 1.11 for steam generator tube plugging $\leq$15%.
This factor is 1.10 for steam generator tube plugging >15% and $\leq$19%.
This factor is 1.08 for steam generator tube plugging >19% and $\leq$28%.

[+]This factor is 0.00106 for steam generator tube plugging >19% and $\leq$28%.

6. DNB PARAMETERS

The following DNB related parameter 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 termal 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 pube plugging $\leq$15%.

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

Reactor Coolant Flow $\geq$ 255,075 gpm for steam generator tube plugging >19% and $\leq$28%.

reactivity insertion upon injection 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

(1) the rod cannot be moved by the CRDM, or
(2) the rod is misaligned from its bank by more than 15 inches, or
(3) 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 DISTRBUTION LIMITS

a. Hot channel factors:

With steam generator tube plugging $\leq$ 28%, the hot channel factors (defined in the basis) must meet the following limits at all times except during low power physics tests:

$$F_q\ (Z) \leq (2.125/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \leq (4.25) \times K(Z), \text{ for } P \leq .5$$

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

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 F_q, as predicted by approved physics calculations, exceeds 2.125 the power will be limited to the rated power multiplied by the ratio of 2.125 divided by the predicted F_q, or augmented surveillance of hot channel factors shall be implemented.

HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

(for $\leq$ 28% steam generator tube plugging and F_q = 2.125)

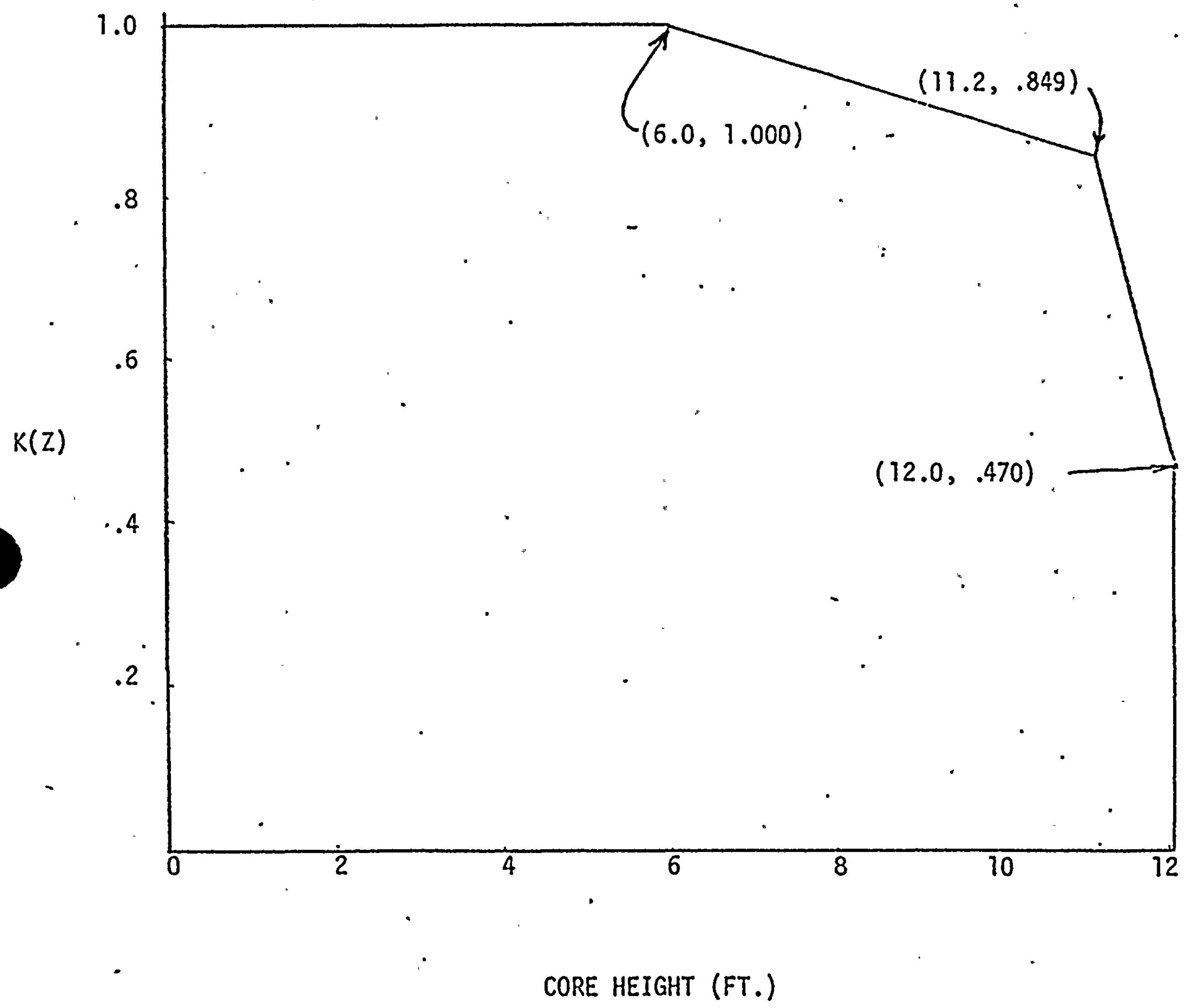


FIG. 3.2-3

ATTACHMENT A

LOCA ANALYSIS

An overview of the Westinghouse Emergency Core Cooling System (ECCS) Evaluation Model which was developed in accordance with the requirements of 10 CFR Part 50, Appendix K [1], is presented in WCAP-8339 [2]. The individual computer codes which comprise the Westinghouse ECCS Evaluation Model are described in detail in separate reports (all dated June, 1974) in references 3, 4, 5, and 6. Since that initial development of the Appendix K ECCS Evaluation Model, several model changes were made, submitted to the NRC for review and approved for use in design LOCA analyses. The "October, 1975" version of the Westinghouse ECCS Evaluation Model incorporates modifications specified in referenes 7, 8 and 9. Additional modifications delineated in references 10, 11, 12 and 13 update the model to the "February, 1978" version which is the model currently used and accepted for plant licensing calculations.

The LOCA analysis presented in this report was performed with an evaluation model which includes improvements made subsequent to the submittal of the "February, 1978" version model changes. These model changes include use of "UHI Software Technology" and addresses the problem of interaction between the accumulator and pumped safety injection flow previously described in reference 14 and 15. There are, however, several differences between the Reference 14 method and that used in this analysis due to its application to a 3 loop plant. First, the split downcomer (elements 11-14 and 47-50) is divided into two azimuthal sections equal to 2/3 of the downcomer volume attached to the two intact loops and 1/3 of the downcomer volume attached to the broken loop. This modelling method gives a more realistic downcomer flow behavior during the transient than splitting the volumes equally. Second, the intact loop was not split at the pump junction into two legs as described in Reference 14. Instead, by attaching the two intact loops into the same downcomer region, the standard 3-loop plant loop nodalization can be retained. The nodalization used for the SATAN-VI calculation has been modified because of the differences presented above. The modified nodalization is shown in Figure 19.

In addition, Turkey Point Units 3 and 4 have the safety injection lines connected to the accumulator surge line between the check valves. Due to the magnitude of the accumulator injection flow and the hydraulic resistance of the check valve and piping from the accumulator line between the check valves to the cold leg, the pressure in the accumulator line will be significantly higher than the RCS cold leg pressure when the accumulutors are injecting. Thus, the interaction between accumulator flow and the safety injection flow increases the injection pressure above RCS pressure. Current Westinghouse Evaluation Models assume that the safety injection pumps inject directly to the RCS presssure and not the accumulator line pressure. When the higher injection section pressure is accounted for, less safety injection delivery occurs and the calculated peak clad temperature will increase. Reference 15 presents details of the model changes required to allow safety injection into the accumulator line.

This analysis includes revised component volume and heat tansfer area calculations based on methods developed for incorporation into an automated analysis input processor. Credit was also taken for revised accumulator line resistance including a general 6% reduction, and for the accumulator line volume. This analysis includes the removal of the 65°F conservatism on the initial steady state fuel pellet temperature. This input change is discusssed

in WCAP-8720, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations", and approved by the NRC in reference 16. The modifications, in essence, update the "February, 1978" version of the SATAN-VI code to include analytical techniques currently approved for use in the LOCA analyses of Westinghouse plants equipped with an Upper Head Injection (UHI) system.

The modifications described in this report are, in the judgment of Westinghouse, in conformance with the requirements of 10CFR Part 50, Appendix K. As such, an evaluation model which includes those modifications is suitable for determining a core peaking factor limit which demonstrates compliance with the acceptance criteria set forth in 10CFR50.46. The Loss of Coolant Accident for Turkey Point Units 3 and 4 has been reanalyzed with the above model modifications.

RESULTS

Table 1 presents the occurrence time for various events throughout the loss of coolant accident transient.

The peak linear power and total core power used in the analysis are given in Table 2. Table 2 also presents selected input values and results from the hot fuel rod thermal transient calculation. For these results, the hot spot location is specified for each break analyzed. The location is indicated in feet, which is elevation above the bottom of the active fuel stack.

Table 3 presents a summary of the various containment systems parameters and structural parameters which were used as input to the COCO computer code [6] used in this analysis.

Tables 4 and 5 present reflood mass and energy releases to the containment and the broken loop accumulator mass and energy release to the containment, respectively.

Figures 1 through 17 present the transients for the principal parameters for the break sizes analyzed. The following items are noted:

Figures 1 - 3: Quality, mass velocity and clad heat transfer coefficient for the clad location exhibiting the maximum temperature (hot spot) and for the section of clad that bursts, if applicable.

Figures 4 - 6: Core pressure, break flow, and core pressure drop. The break flow is the sum of the flowrates from both ends of the guillotine break. The core pressure drop is taken as the pressure just before the core inlet to the pressure just beyond the core outlet.

Figures 7 - 9: Clad temperature, fluid temperature and core flow. The clad and fluid temperatures are for the hot spot and burst locations.

Figures 10 - 11: Downcomer and core water level during reflood, and flooding rate.

Figures 12 - 13: Emergency core cooling system flowrates, for both accumulator and pumped safety injection.

Figures 14 - 15: Containment pressure and core power transients.

Figures 16 - 17: Break energy release during blowdown and the containment wall condensing heat transfer coefficient for the worst break.

Figure 18: K(z) versus core height for Fq = 2.125.

Figure 19: SATAN Model for Turkey Point

CONCLUSIONS - THERMAL ANALYSIS

For breaks up to and including the double ended severance of a reactor coolant pipe, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10CFR50.46[1]. That is:

1. The calculated peak clad temperature does not exceed 2200°F based on a total core peaking factor of 2.25.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of zircaloy in the reactor

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The local cladding oxidation limit of 17% is not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

5. As shown in Appendix A, it is estimated that a reduction in F_Q of 0.125 is required to maintain the 2200°F clad temperature as a result of the fuel rod models in NUREG-0630. This leaves a net F_Q of 2.125.

REFERENCES

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10CFR50.46 and Appendix K of 20CFR50.46. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Bordelon, F. M., Massie, H. W., and Zordan, T. A., "Westinghouse ECCS Evaluation Model-Summary", WCAP-8339, July, 1974.

3. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant", WCAP-8302 (Proprietary Version), WCAP-8306 (Non-Proprietary Version), June 1974.

4. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8301 (Proprietary Version), WCAP-8305 (Non-Proprietary Version), June 1974.

5. Kelly, R. D. et al. "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)". WCAP-8170 (Proprietary Version), WCAP-8171 (Non-Proprietary Version), June 1974.

6. Bordelon, F.M., and Murphy E.T., "Containment Pressure Analysis Code (COCO)", WCAP-8327 (Proprietary Version), WCAP-8326 (Non-Proprietary Version), June 1974.

7. Bordelon, F. M. et al. "The Westinghouse ECCS Evaluation Model: Supplementary Information". WCAP-8471 (Proprietary Version), WCAP-8472 (Non-Proprietary Version), January 1975.

8. "Westinghouse ECCS Evaluation Model, October, 1975 Versions", WCAP-8622 (Proprietary Version), WCAP-8623 (non-Proprietary Version), November, 1975.

9. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassalo of the Nuclear Regulatory Commission, letter number NS-CE-924, January 23, 1976.

10. Kelly, R. D. Thompson, C. M., et al. "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCA's During Operation with One Loop Out of Service for Plants without Loop Isolation Valves", WCAP-9166, February, 1978.

11. Eicheldinger, C. "Westinghouse ECCS Evaluation Model, February 1978 Version", WCAP-9220-P-A (Proprietary Version), WCAP-9221-A (Non-Proprietary Version), February, 1978.

12. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-1981, Nov. 1, 1978.

13. Letter from T. M. Anderson of Westinghouse Electric Corporation to Tedesco of the Nuclear Regulatory Commissin, letter number NS-TMA-2014, Dec. 11, 1978.

14. Letter from T. M. Anderson of Westinghouse Electric Company to J. M. Miller, NRC, NS-TMA-2311, September 15, 1980.

15. Letter from T. M. Anderson of Westinghouse Electric Company to D. F. Ross, NRC, NS-TMA-2354, December 22, 1980.

16. Letter from J. F. Stol, NRC to T. M. Anderson, Westinghouse, dated March 27, 1980.

TABLE 1

LARGE BREAK - UHI TECHNOLOGY MODEL

TIME SEQUENCE OF EVENTS

	DECL(C_D = 0.4) (Sec)	DECL(C_D = 0.6) (Sec)
START	0.0	0.0
RX Trip Setpoint Reached	0.69	0.68
S.I. Setpoint Reached	0.78	0.60
Acc. Injection	14.6	11.2
End of Blowdown	30.31	25.07
Bottom of Core Recovery	50.213	44.867
Acc. Empty	61.35	56.731
Pump Injection	25.78	25.60
End of Bypass	30.31	25.07

TABLE 2

LARGE BREAK - UHI TECHNOLOGY MODEL

	DECL C_D = 0.4	DECL C_D = 0.6
Results		
Peak Clad Temp. °F	2183	1962
Peak Clad Location Ft.	7.5	7.25
Local Zr/H_2O Rxn(max)percent	7.39	2.991
Local Zr/H_2O Location Ft.	6.0	7.25
Total Zr/H_2O Rxn percent	<0.3	<0.3
Hot Rod Burst Time sec	41.8	72.3
Hot Rod Burst Location Ft.	6.0	6.0

Calculation	
Core Power Mwt 102 percent of	2200
Peak linear Power kw/ft 102 percent of	12.77
Peaking Factor	2.25
Accumulator Water Volume	875 ft^3/accumulator

Fuel region + cycle analyzed	Cycle	Region
UNIT 3 and 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 Initiation of Fan Coolers	26 secs

TABLE 3 (continued)

CONTAINMENT DATA

(DRY CONTAINMENT)

STRUCTIONAL HEAT SINKS	THICKNESS (INCH)	AREA (FT^2)
Paint	0.006996	-
Carbon steel	0.2898	87335.8
Carbon steel	0.006996	1000086.0
Paint	0.006996	-
Carbon steel	0.4896	35660.11
Carbon steel	0.4896	12367.5
Paint	0.006996	-
Carbon steel	0.2898	-
Concrete	24.0	50430.0
Carbon steel	0.2898	-
Concrete	24.0	16810.0
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

TABLE 3 (continued)

CONTAINMENT DATA

(DRY CONTAINMENT)

STRUCTIONAL HEAT SINKS	THICKNESS (INCH)	AREA (FT^2)
Carbon steel	0.2898	17190.0
Stainless steel	0.032	113253.4
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 C_D=0.4 UHI Technology Model

TIME (Sec)	$\dot{M}$ (Total) (LBM/Sec)	$\dot{M}$H (Total) (BTU/Sec)
50.213	0.0	0.0
50.938	2.698-02	34.90
58.741	31.34	4.056 + 4
70.244	43.62	5.570 + 4
86.644	55.09	6.953 + 4
105.541	67.37	8.431 + 4
124.044	78.10	9.722 + 4
142.844	250.04	1.449 + 5
182.144	287.05	1.437 + 5
225.744	294.06	1.345 + 5

TABLE 4A

REFLOOD MASS AND ENERGY RELEASES
DECLG C_D=0.6 UHI Technology Model

TIME (Sec)	$\dot{M}$ (Total) (LBM/Sec)	$\dot{M}H$ (Total) (BTU/Sec)
44.862	0.0	0.0
45.587	0.0267	34.47
53.315	31.64	4.090 + 4
64.618	49.98	6.340 + 4
80.418	60.16	7.592 + 4
98.318	71.18	8.938 + 4
116.718	80.91	1.013 + 5
135.118	252.6	1.474 + 5
173.818	285.7	1.448 + 5
216.718	292.8	1.360 + 5

TABLE 5

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT

C_D=0.4 UHI Technology Model

TIME (Sec)	MASS FLOW RATE (LBM/Sec)
0.0	3195.049
1.01	2872.266
2.01	2635.418
4.01	2298.696
6.01	2064.529
8.01	1886.928
10.01	1744.883
15.01	1487.450
20.01	1313.054
25.01	1190.337
29.01	1126.647
34.32	1009.557

FOR ENERGY FLOW, MULTIPLY MASS FLOW BY 59.62 BTU/LBM.

TABLE 5A

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT

C_D=0.6 UHI Technology Model

TIME (Sec)	MASS FLOW RATE (LBM/Sec)
0.00	3195.049
1.01	2868.804
2.01	2628.985
4.01	2287.787
6.01	2048.893
8.01	1867.277
10.01	1723.220
15.01	1469.498
20.01	1304.673
24.01	1210.650
34.362	1010.478

FOR ENERGY FLOW, MULTIPLY MASS FLOW BY 59.62 BTU/LBM.

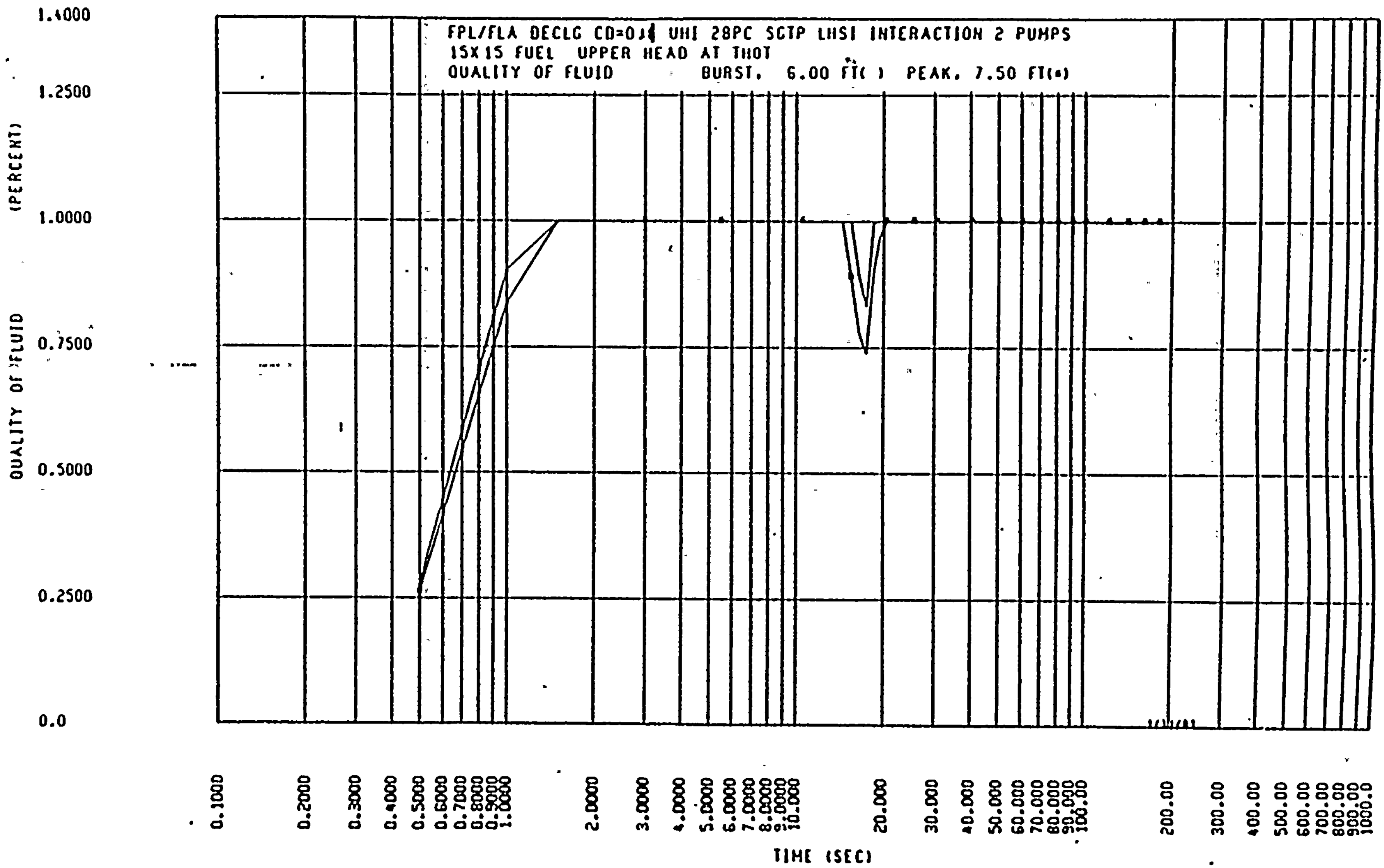


FIGURE 1 FLUID QUALITY DECLG (C_D = 0.4)

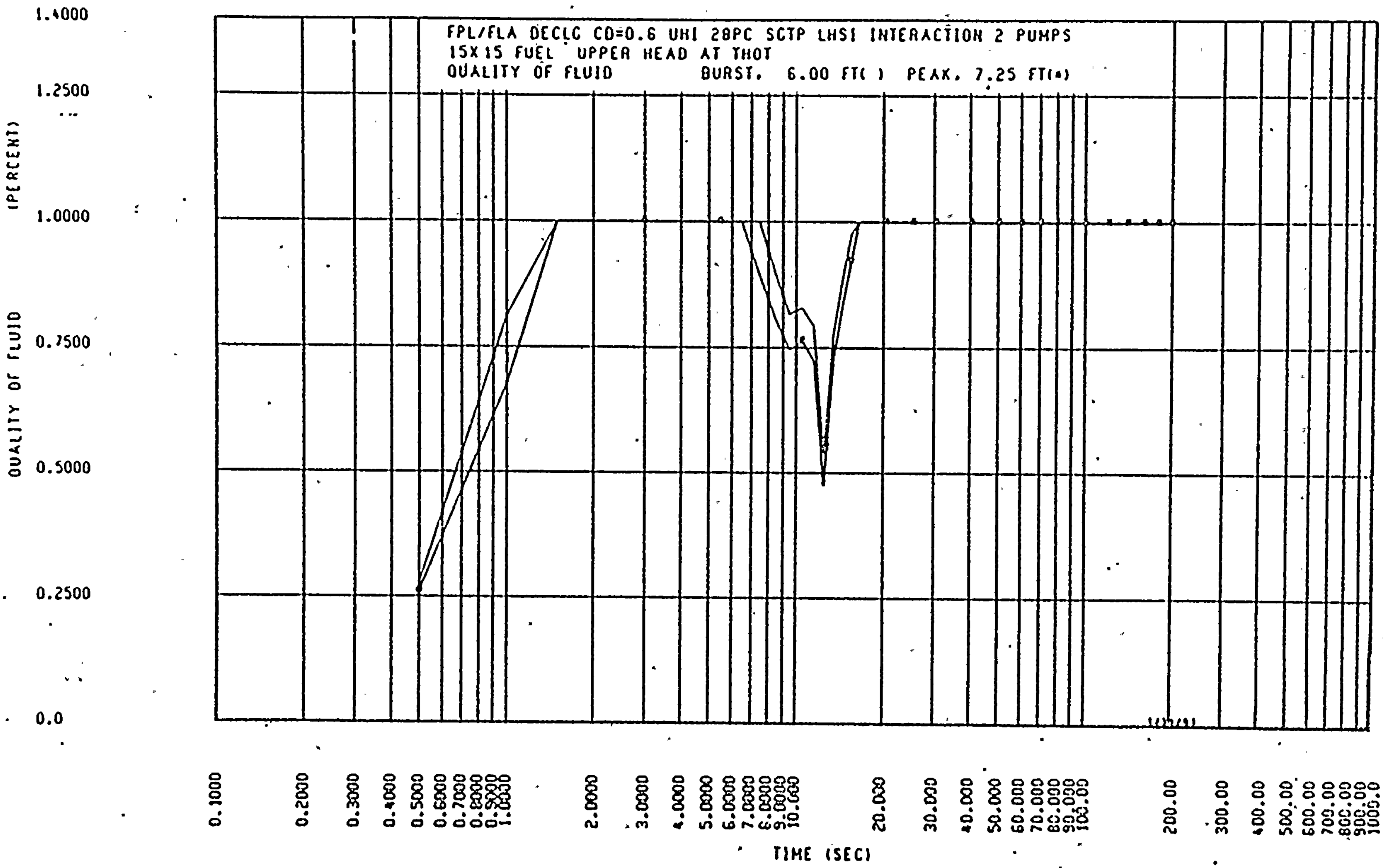


FIGURE 1A FLUID QUALITY DECLG (C_D = 0.6)

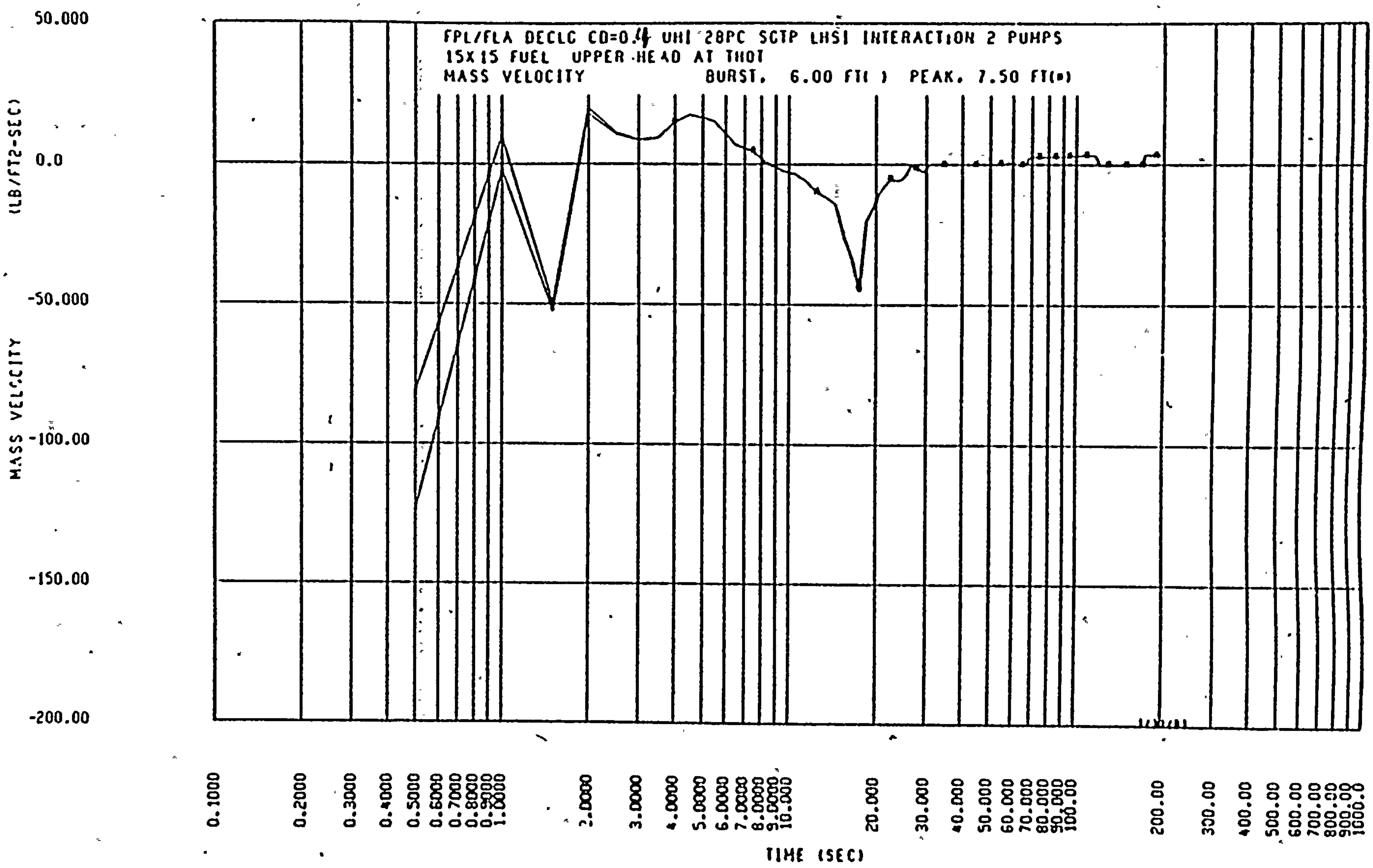


FIGURE 2 MASS VELOCITY DECLG (C_D = 0.4)

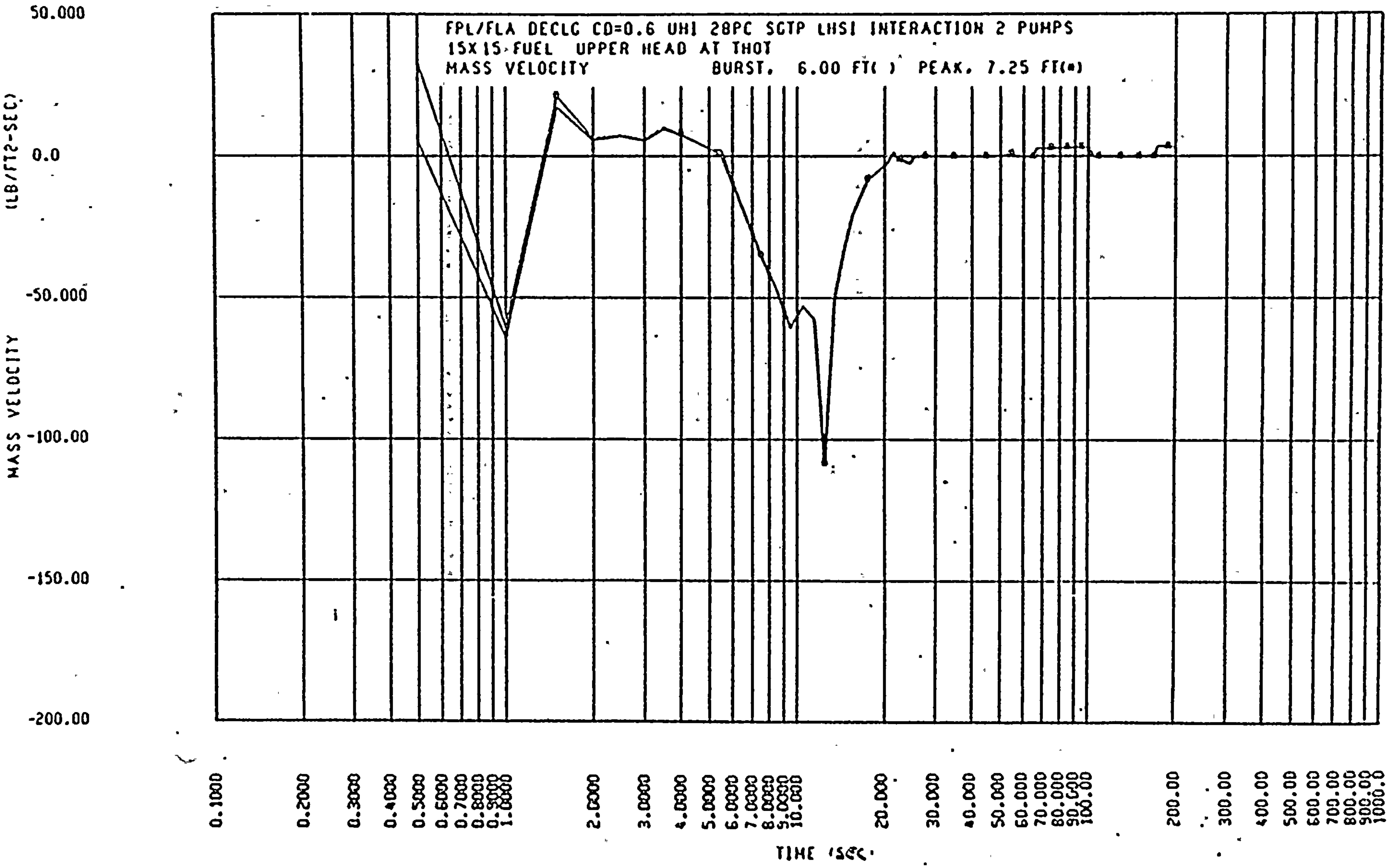


FIGURE 2A MASS VELOCITY DECLG (C_D = 0.6)

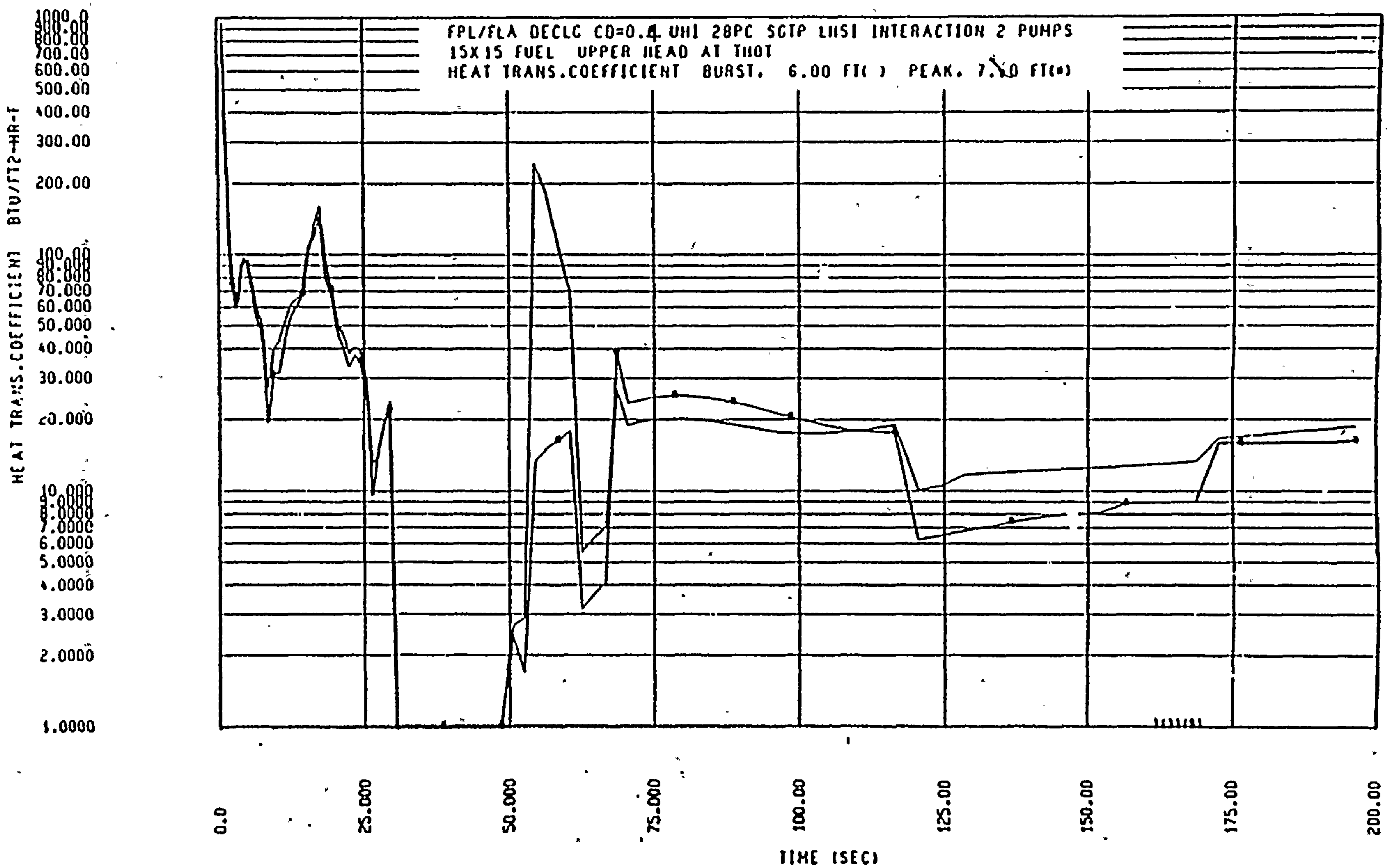


FIGURE 3 HEAT TRANSFER COEFFICIENT DECLG (C_D = 0.4)

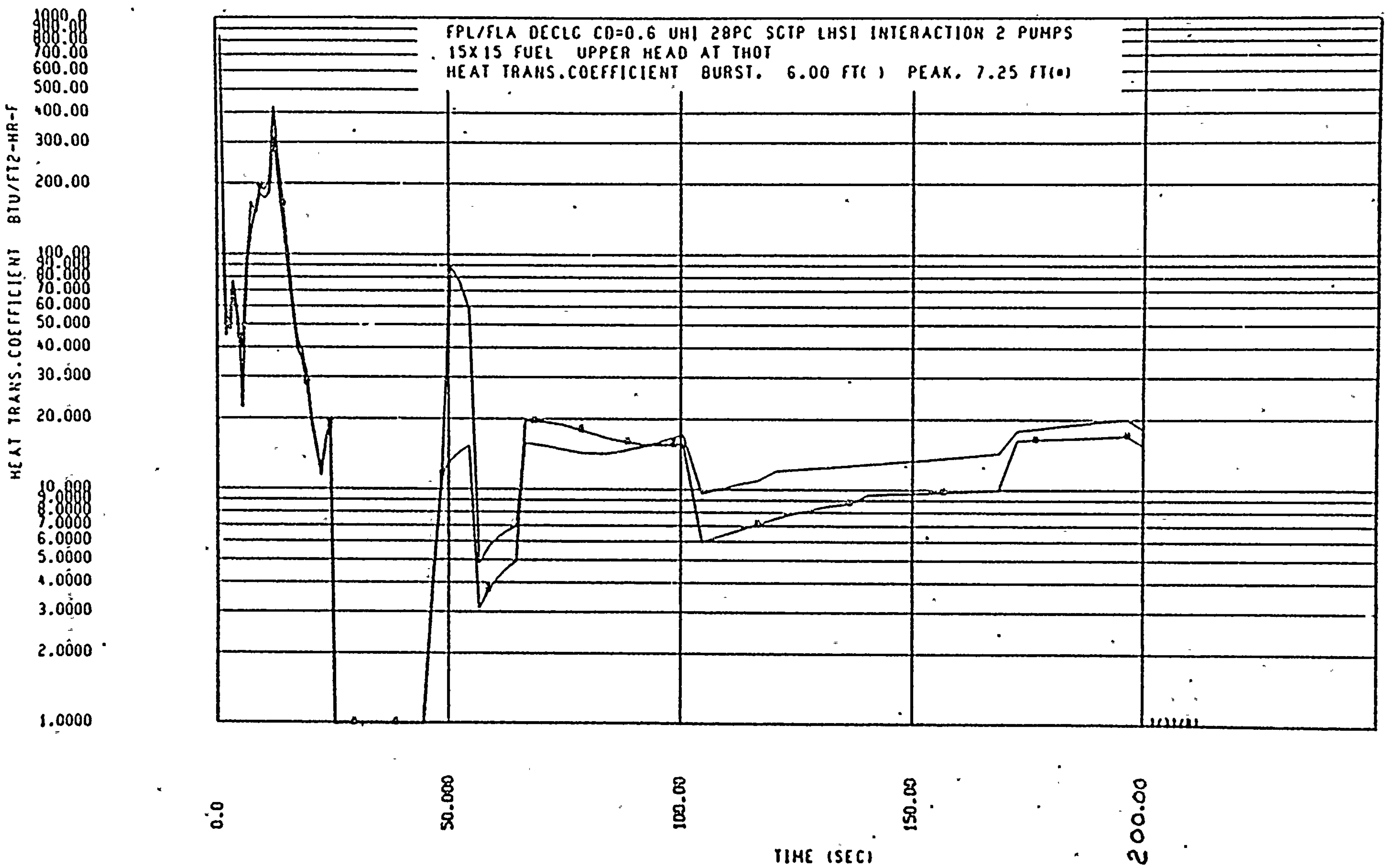


FIGURE 3A HEAT TRANSFER COEFFICIENT DECLG (C_D = 0.6)

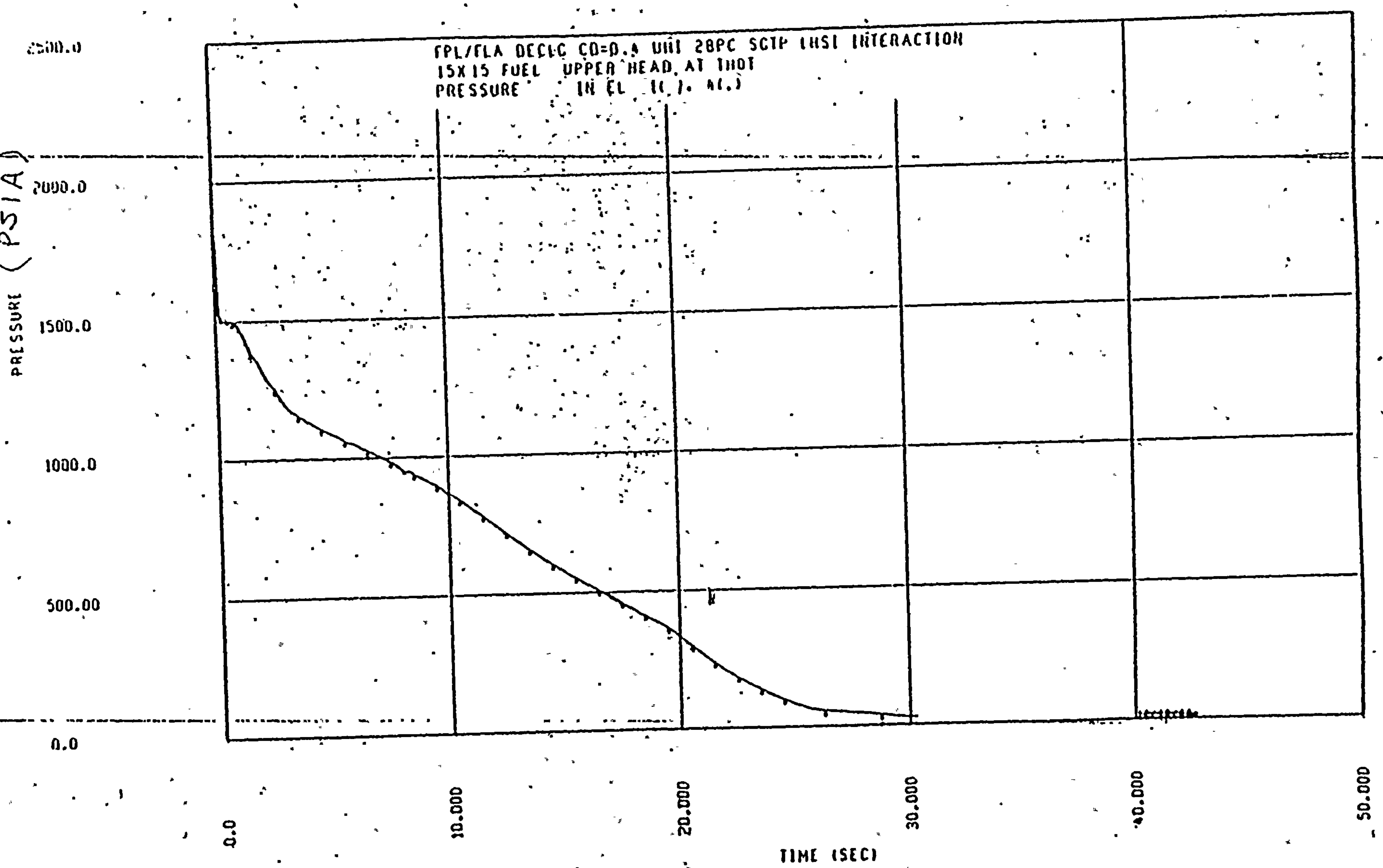


FIGURE 4 PRESSURE DECLG (C_D = 0.4)

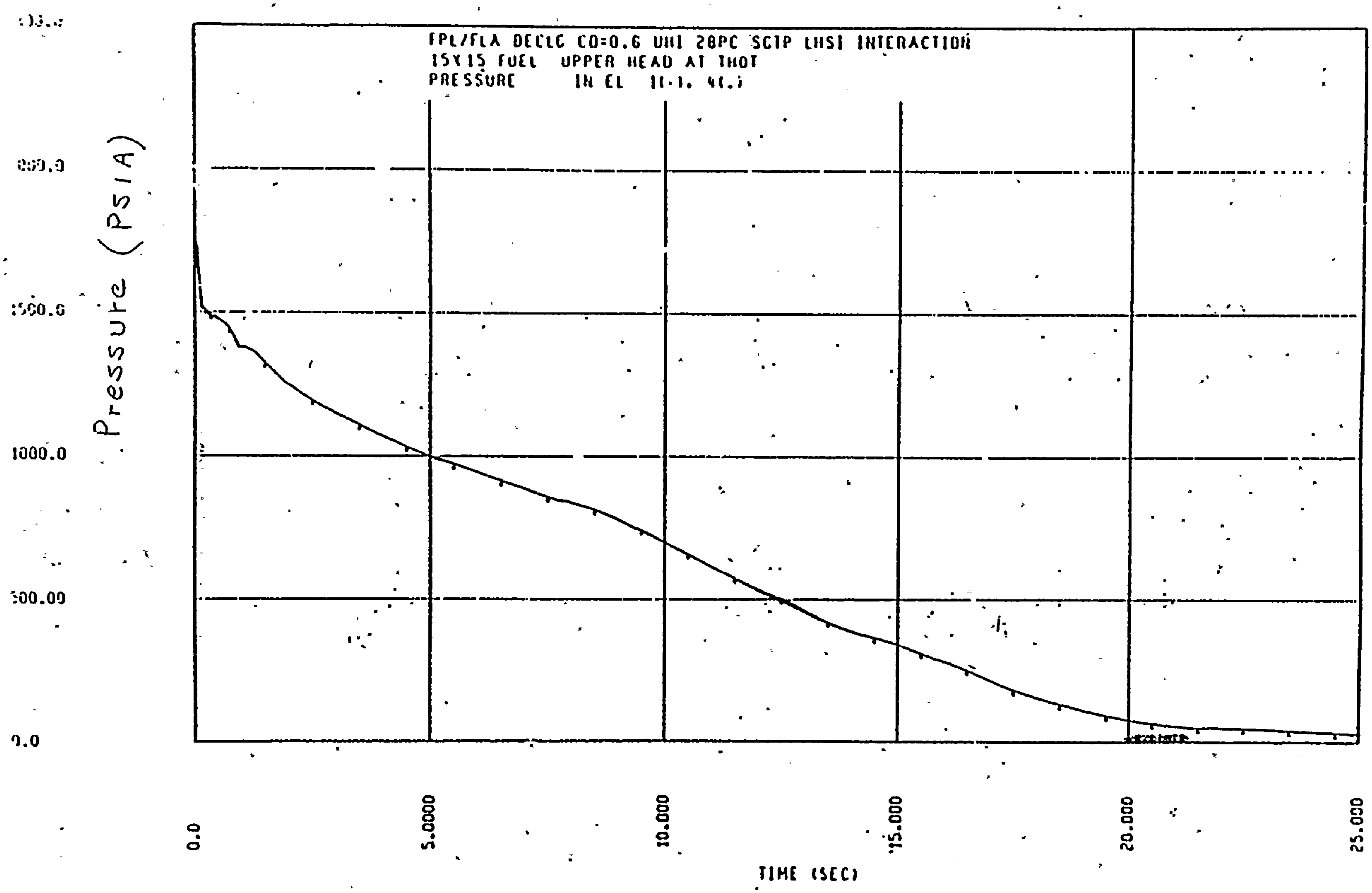


FIGURE 4A PRESSURE DECLG (C_D = 0.6)

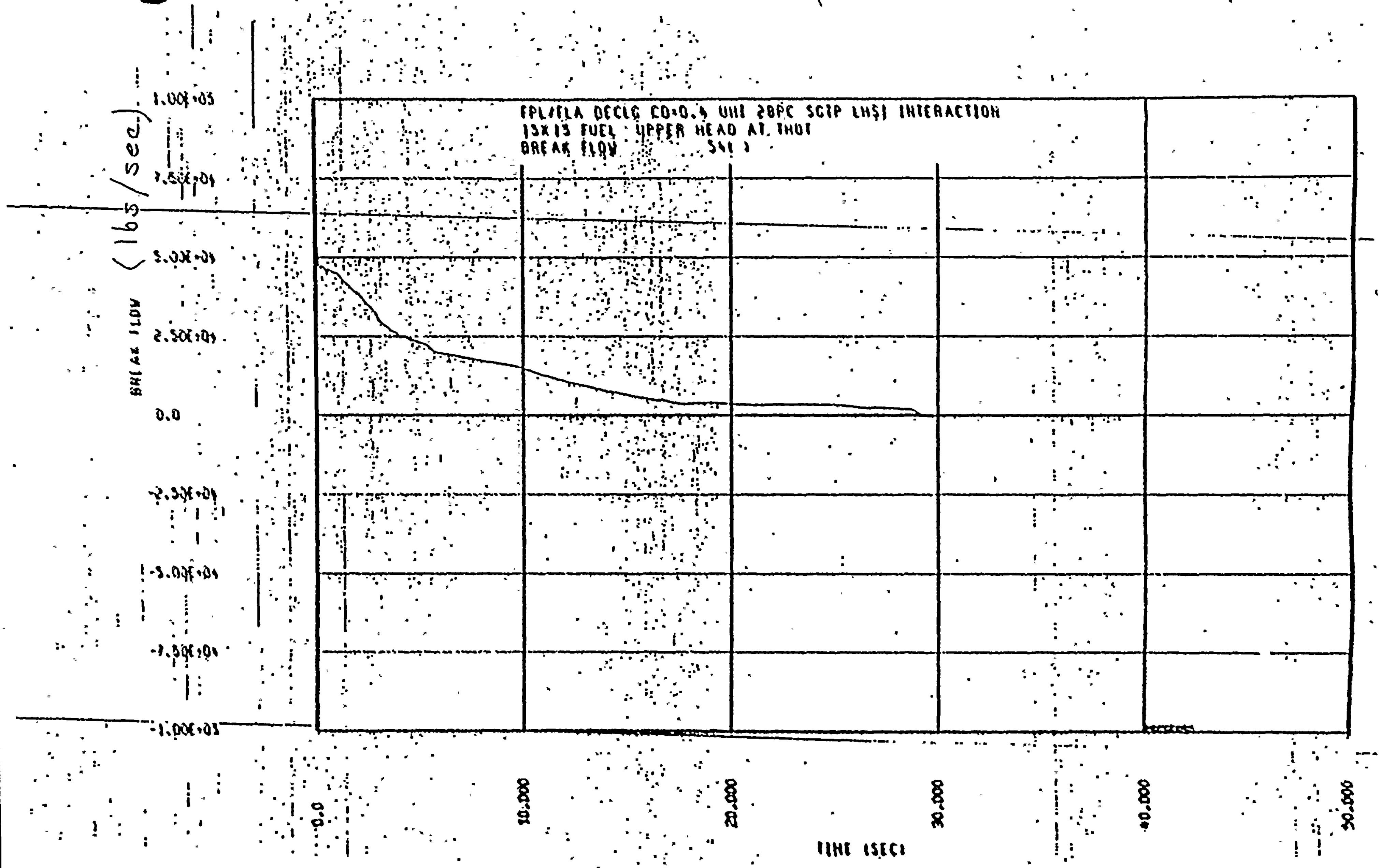


FIGURE 5 BREAK FLOW DECLG (C_D = 0.4)

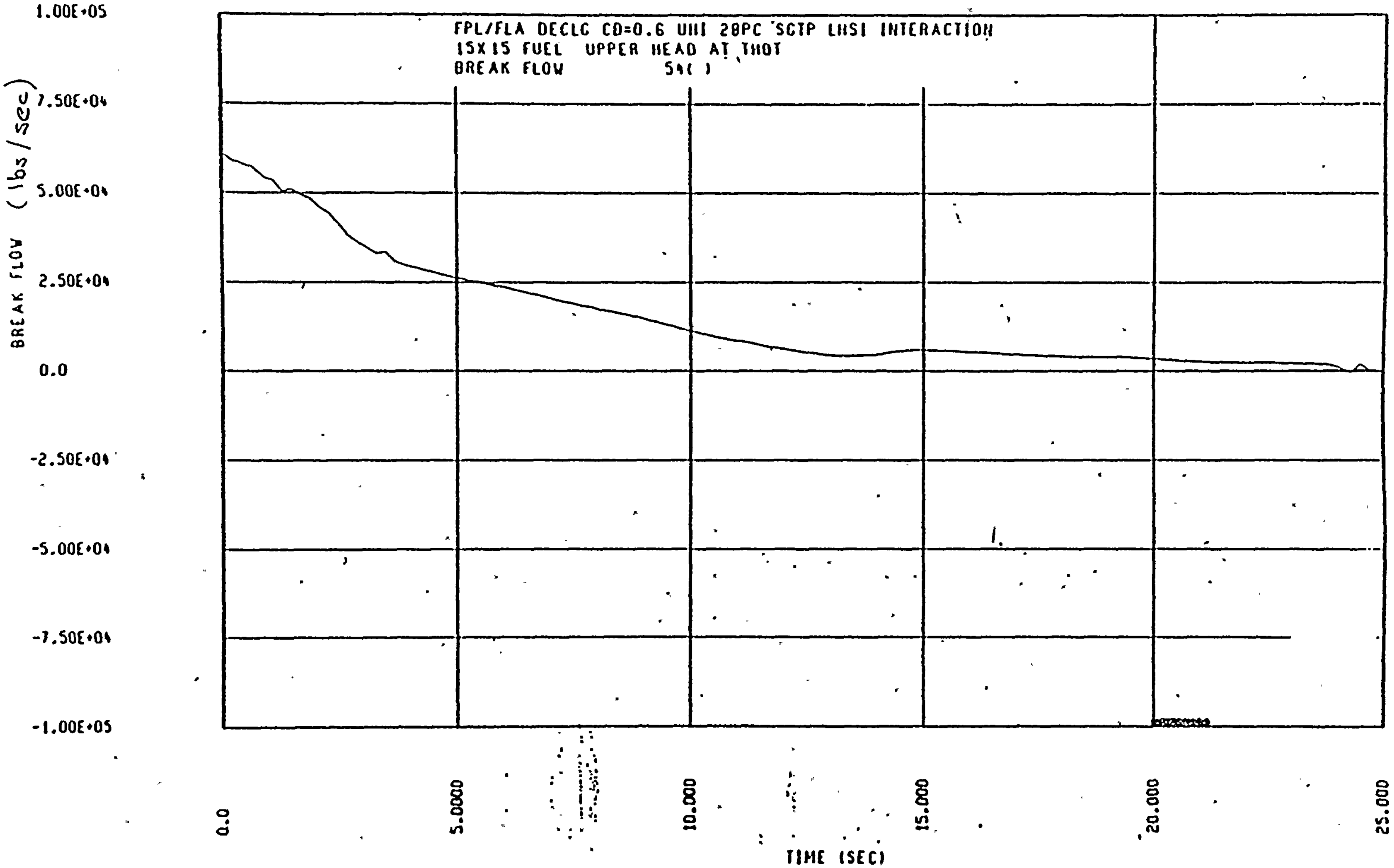


FIGURE 5A BREAK FLOW DECLG (C_D = 0.6)

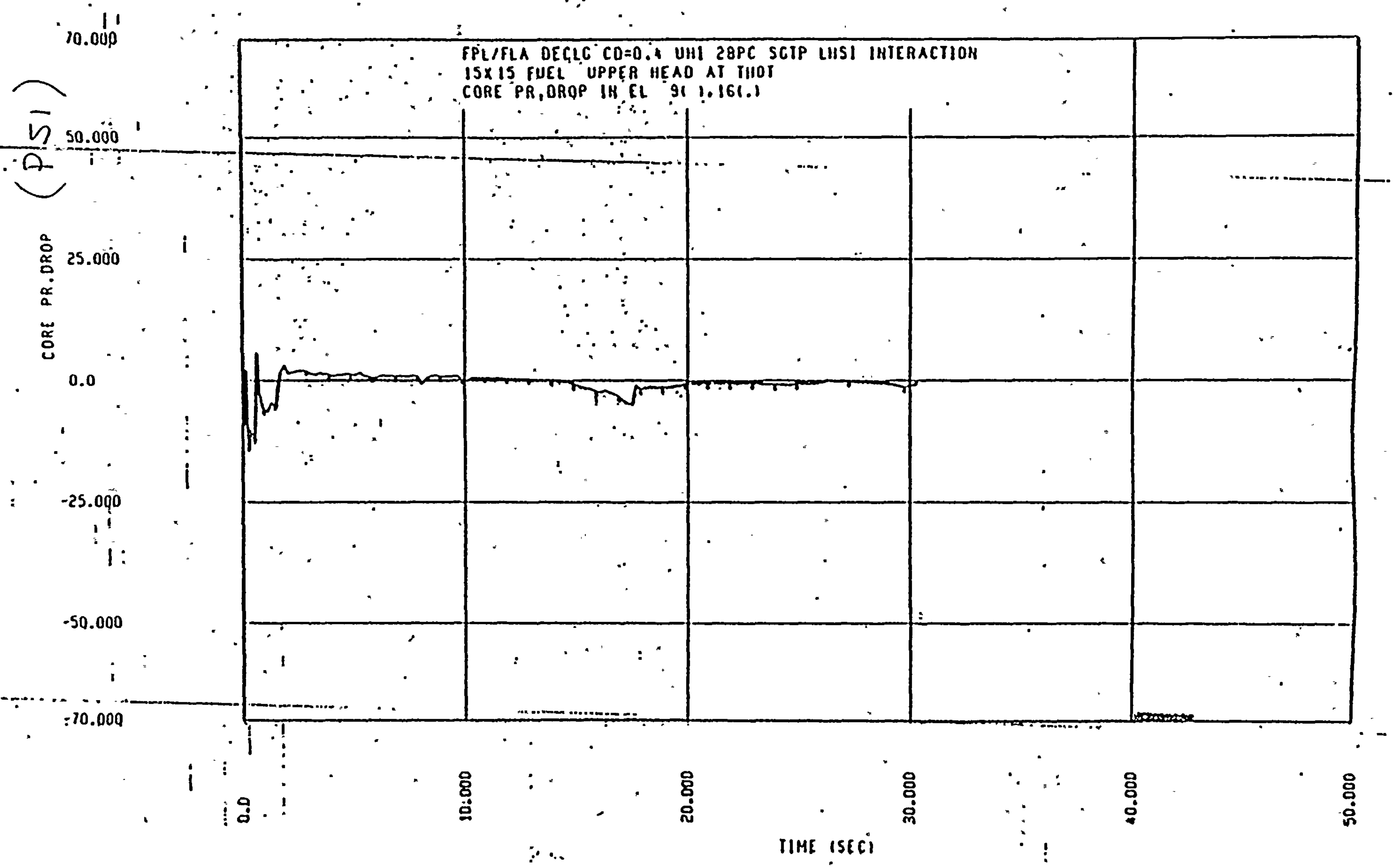


FIGURE 6 CORE PRESSURE DROP DECLG (C_D = 0.4)

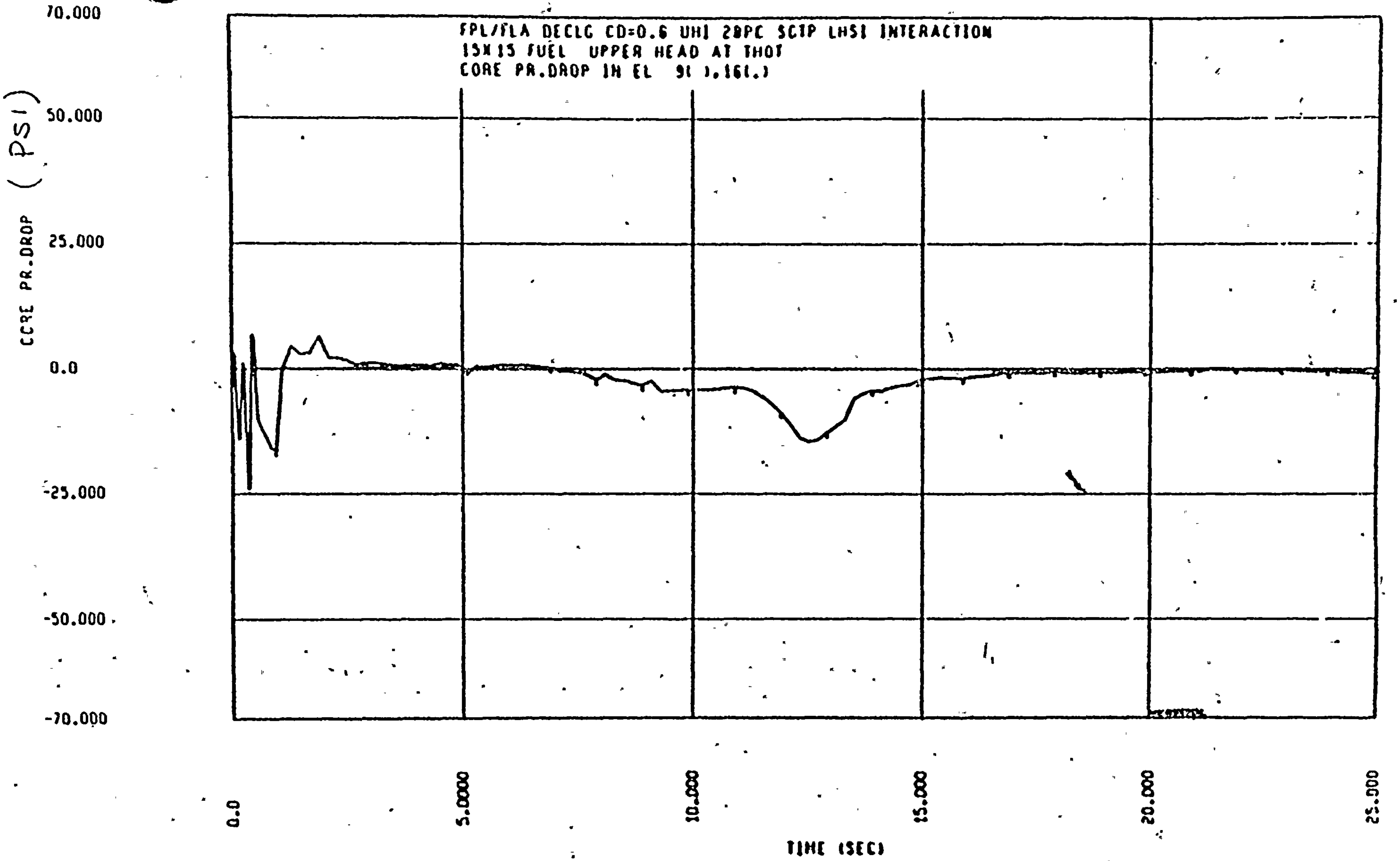


FIGURE 6A CORE PRESSURE DROP DECLG (C_D = 0.6)

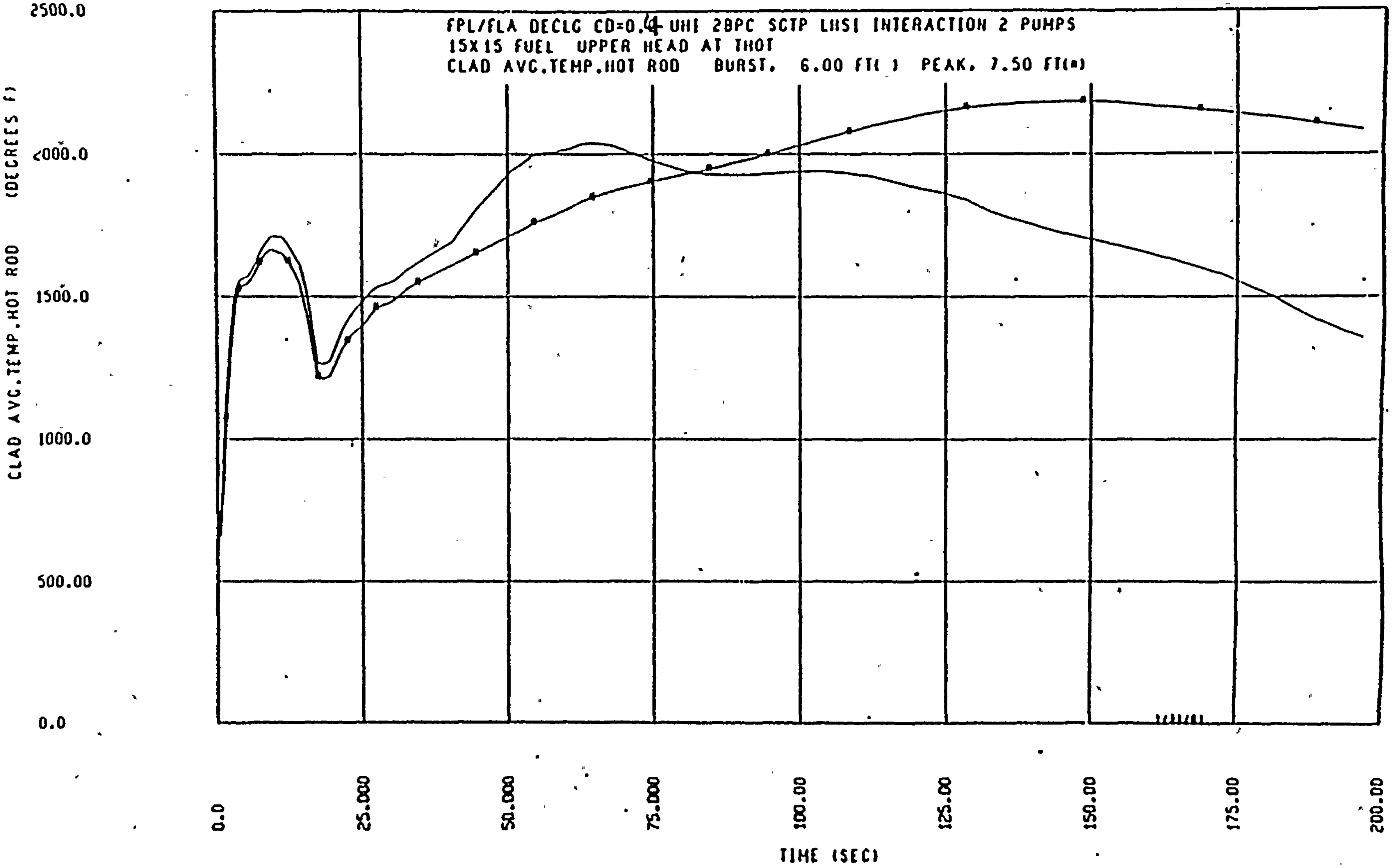


FIGURE 7 PEAK CLAD TEMPERATURE DECLG (C_D = 0.4)

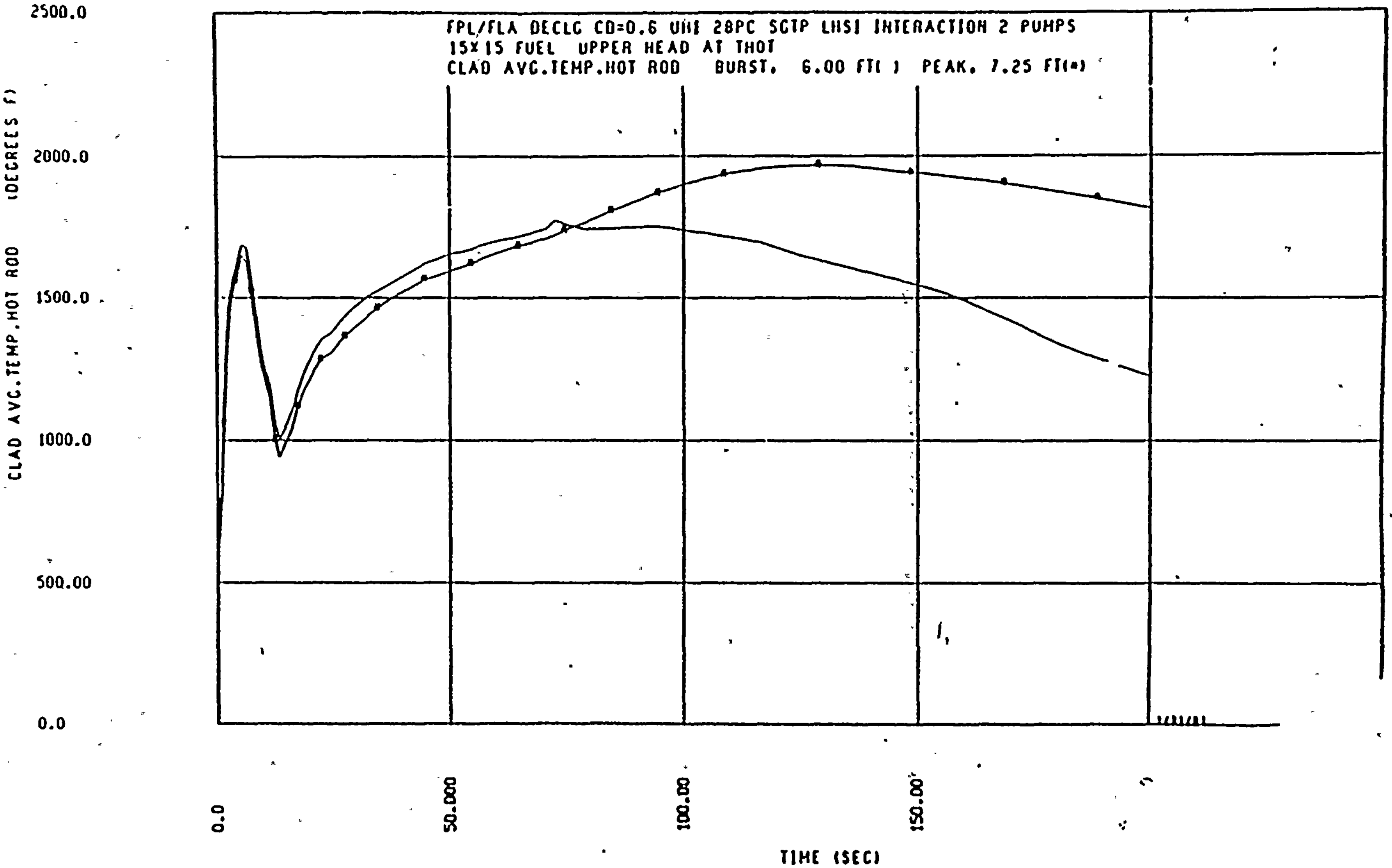


FIGURE 7A PEAK CLAD TEMPERATURE DECLG (C_D = 0.6)

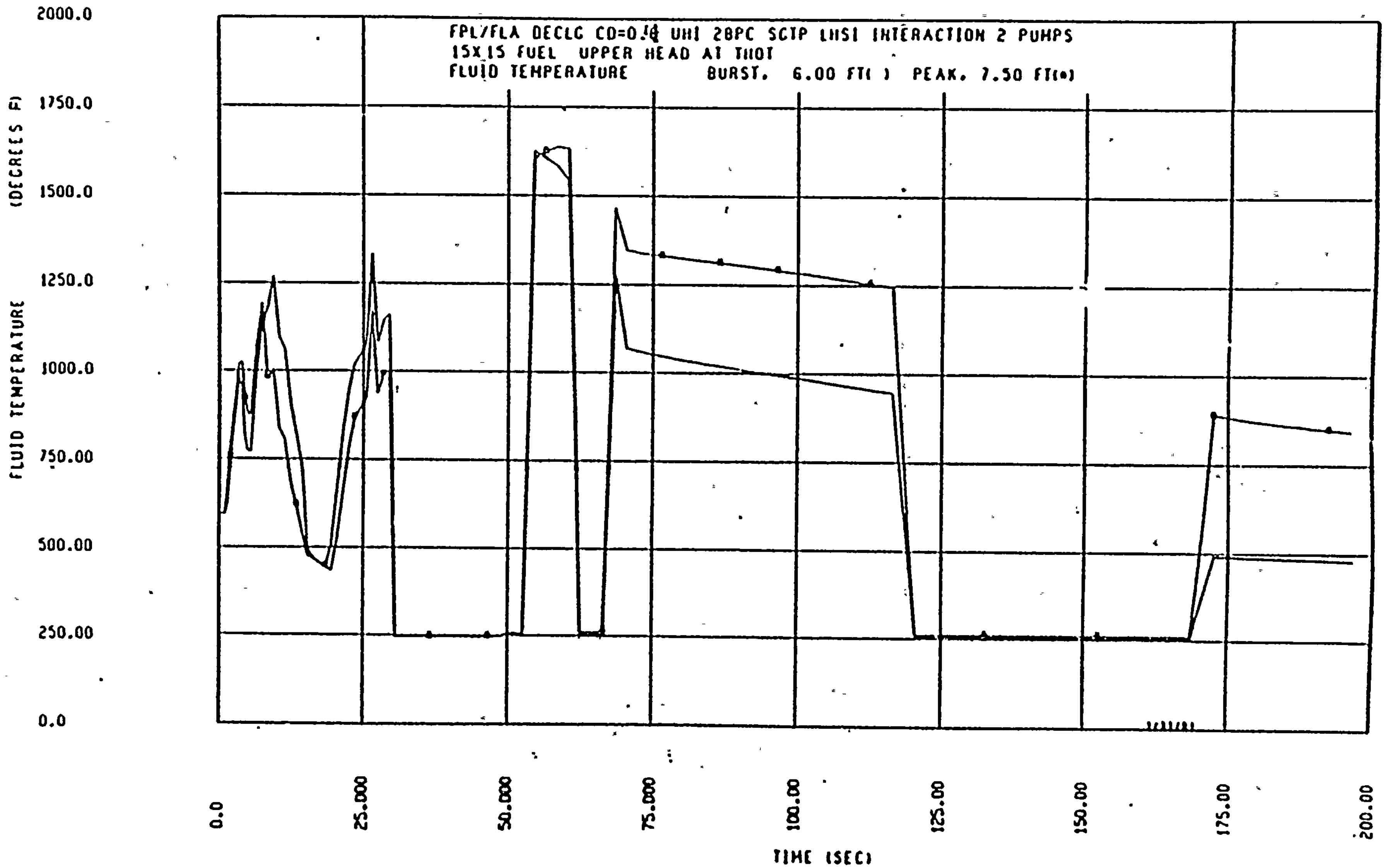


FIGURE 8 FLUID TEMPERATURE DECLG (C_D = 0.4)

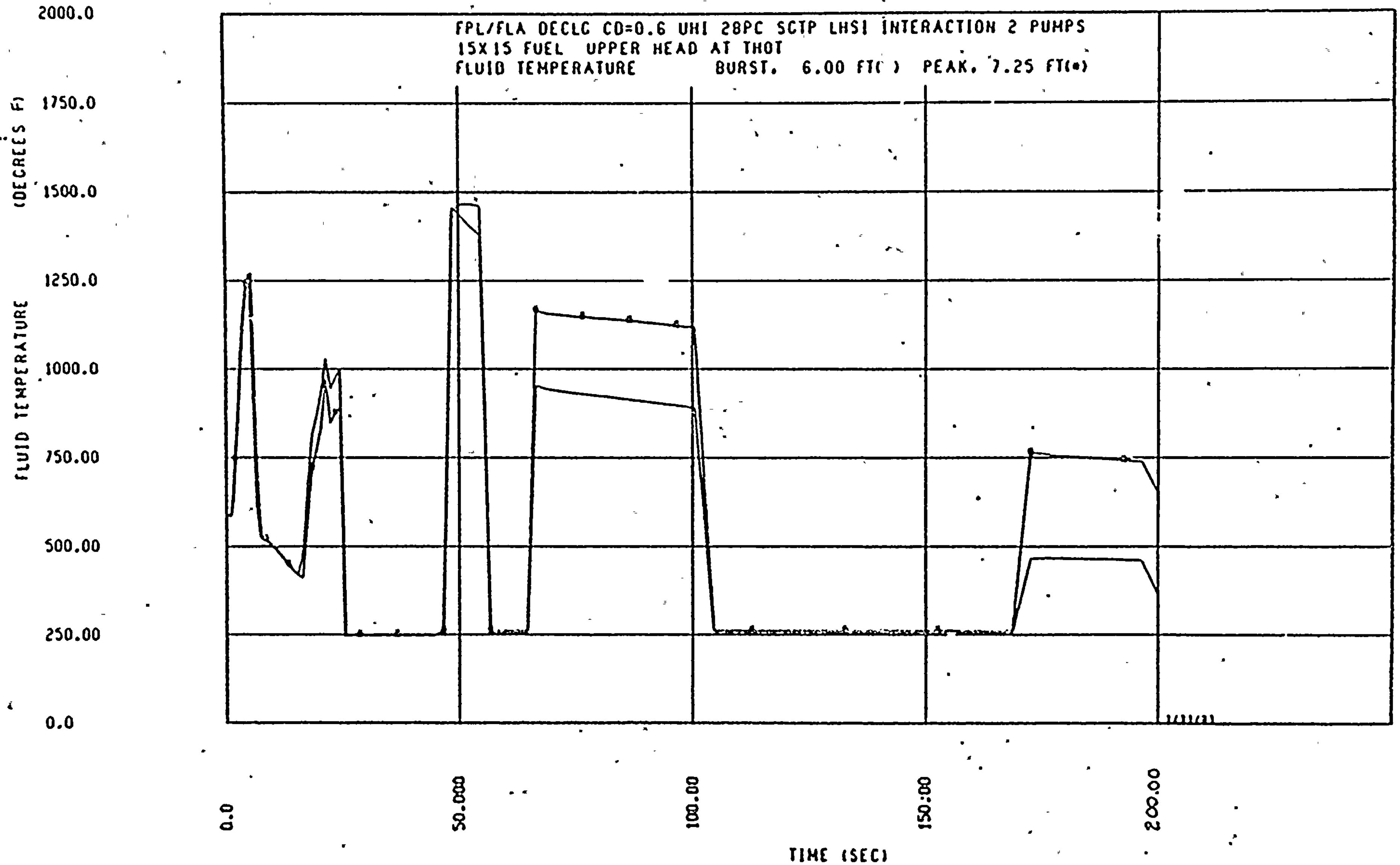


FIGURE 8A FLUID TEMPERATURE DECLG (C_D = 0.6)

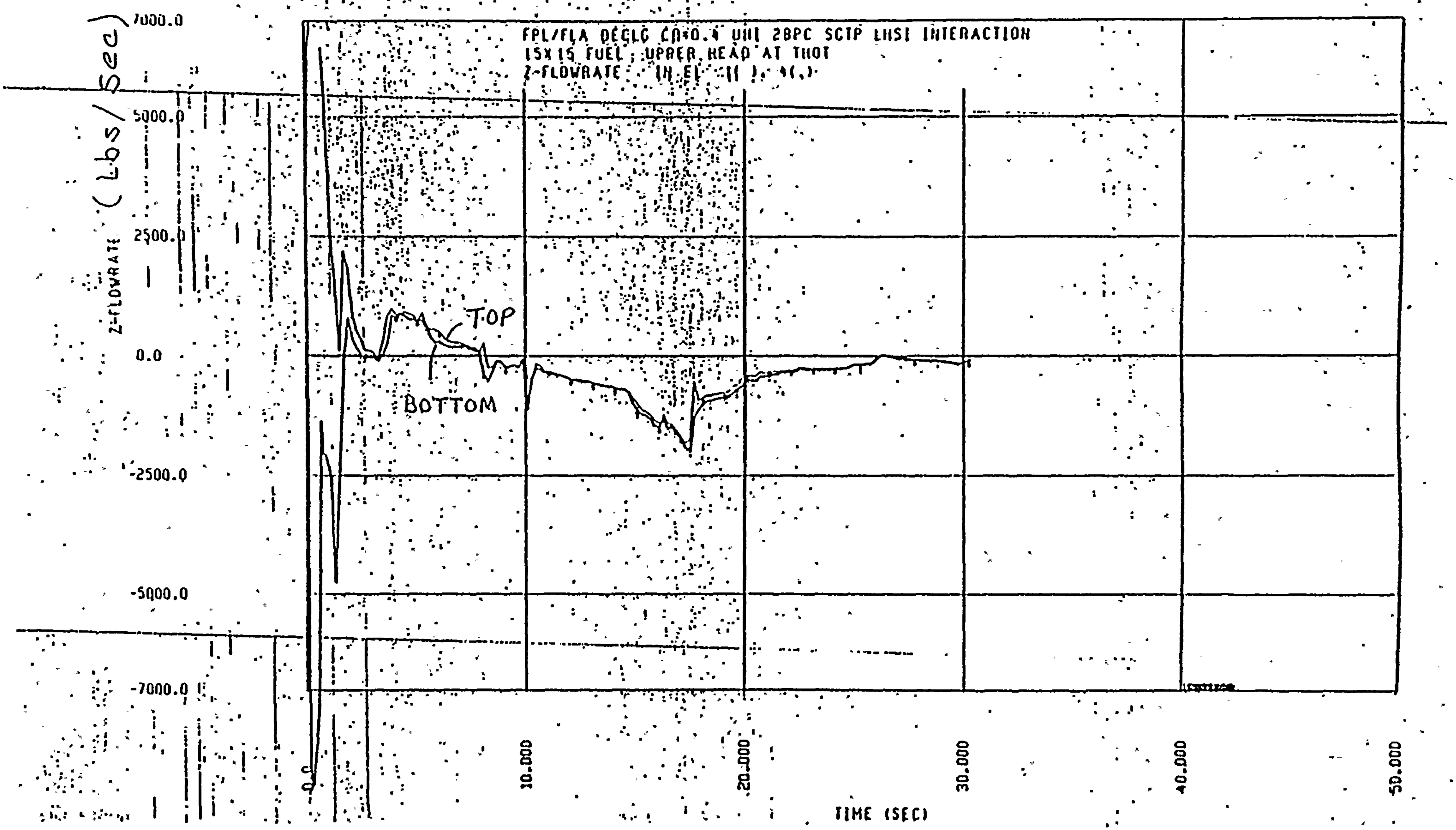


FIGURE 9 CORE FLOW RATE (TOP AND BOTTOM) DECLG ($C_D' = 0.4$)

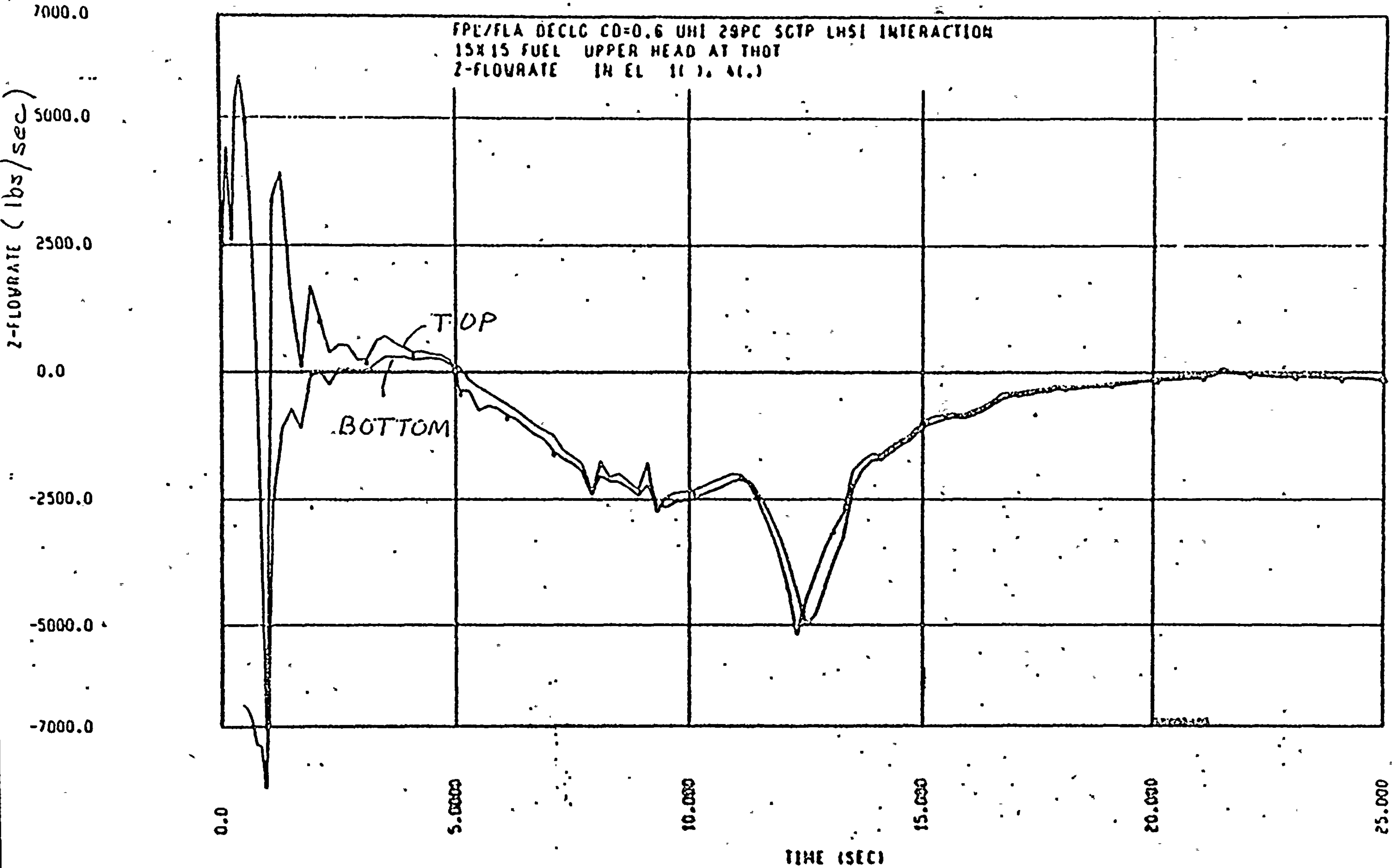


FIGURE 9A CORE FLOW RATE (TOP AND BOTTOM) DECLG (C_D = 0.6)

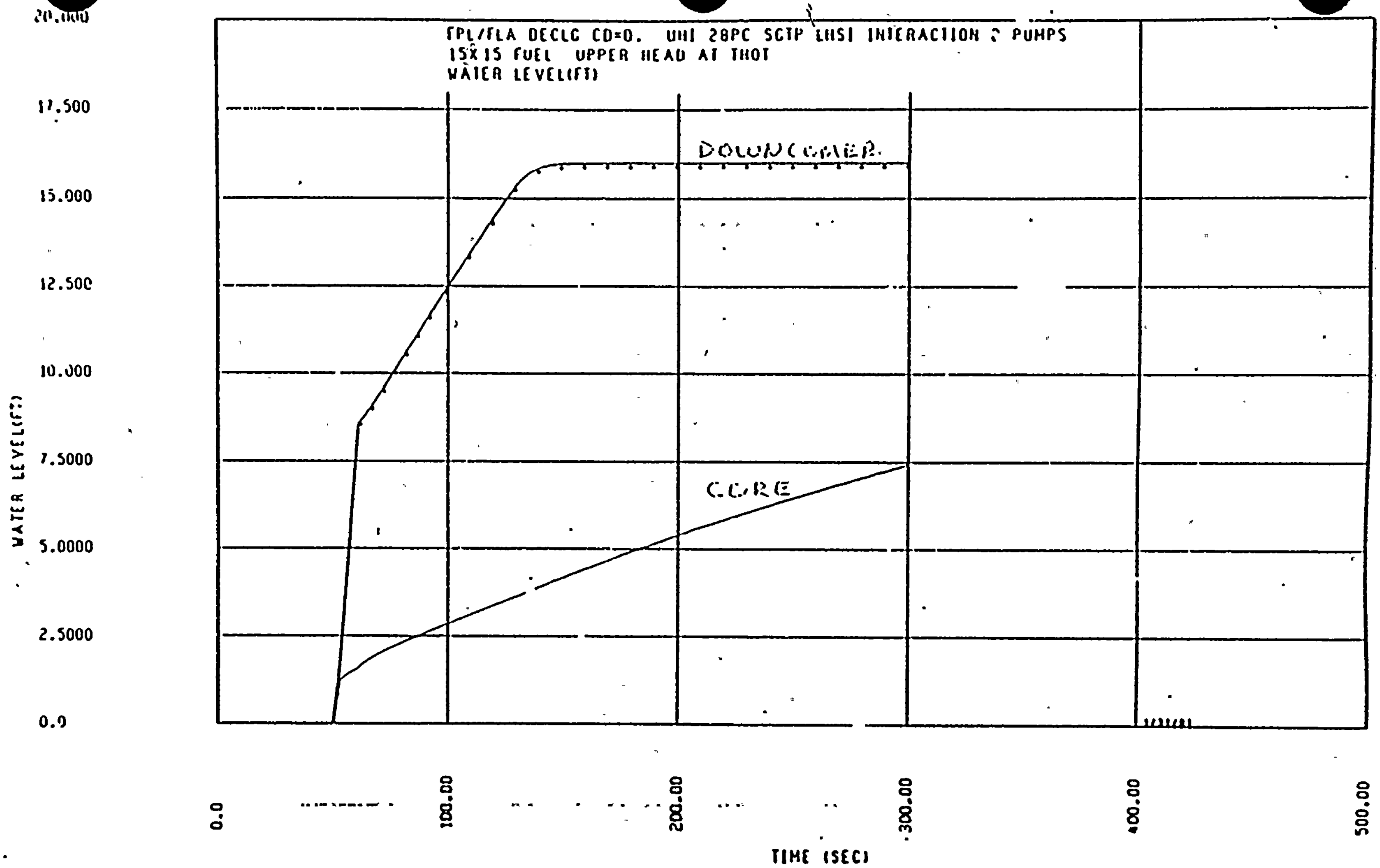


FIGURE 10 DOWNCOMER AND CORE WATER LEVEL
DURING REFLOOD TRANSIENT DECLG (C_D = 0.4)

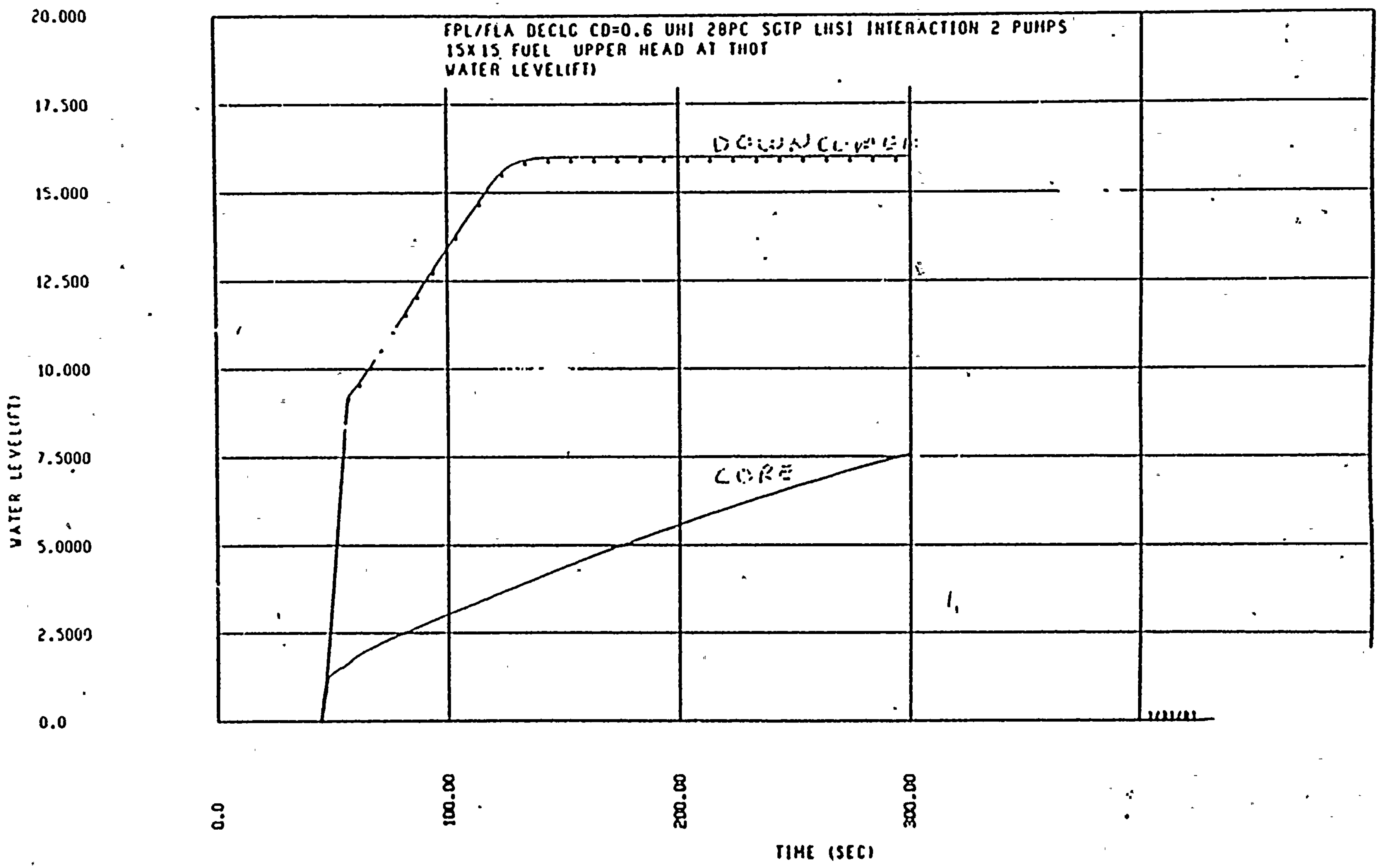


FIGURE 10A DOWNCOMER AND CORE WATER LEVEL DURING REFLOOD TRANSIENT DECLG (C_D = 0.6)

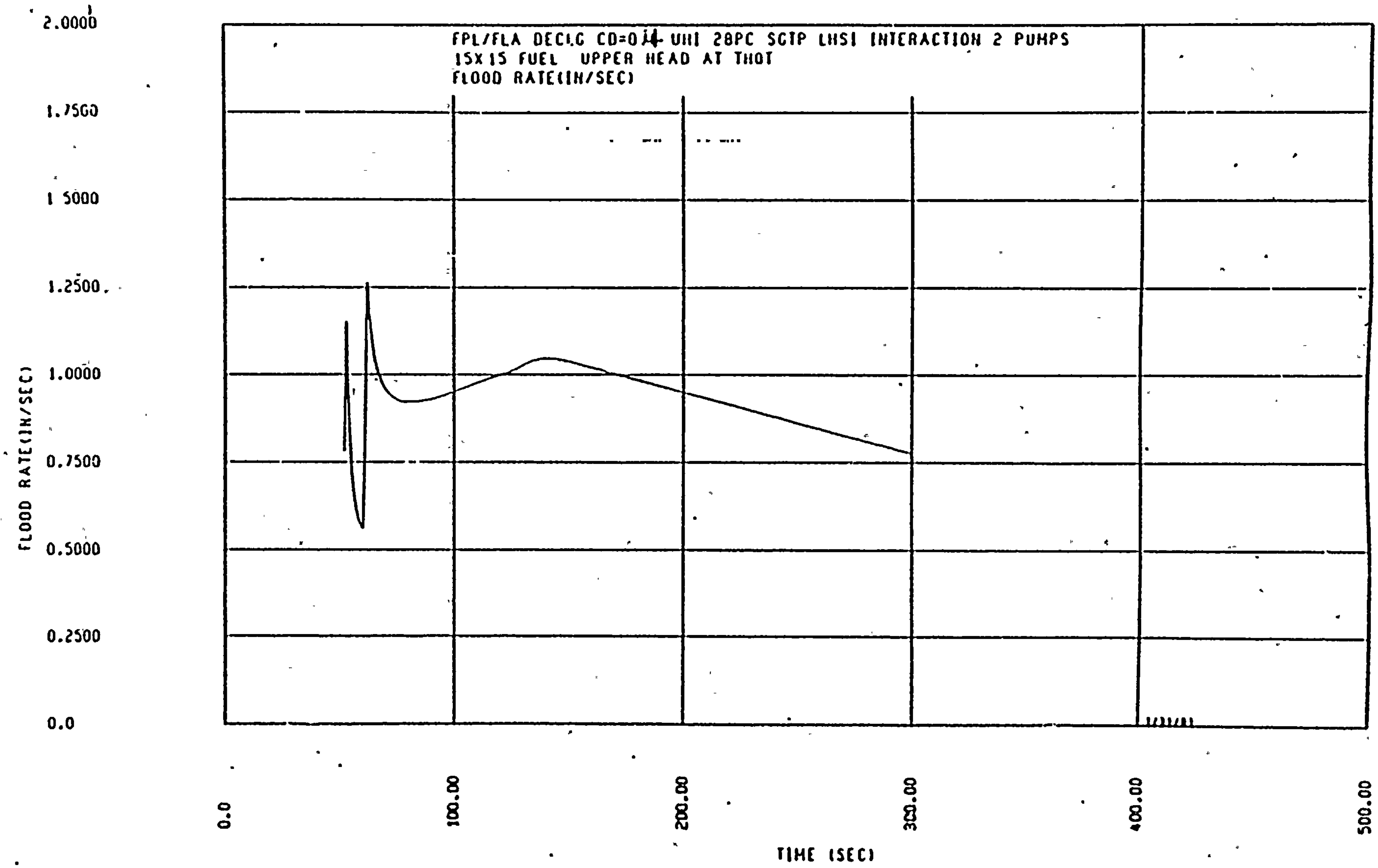


FIGURE 11 CORE INLET VELOCITY DURING REFLOOD TRANSIENT DECLG (C_D = 0.4)

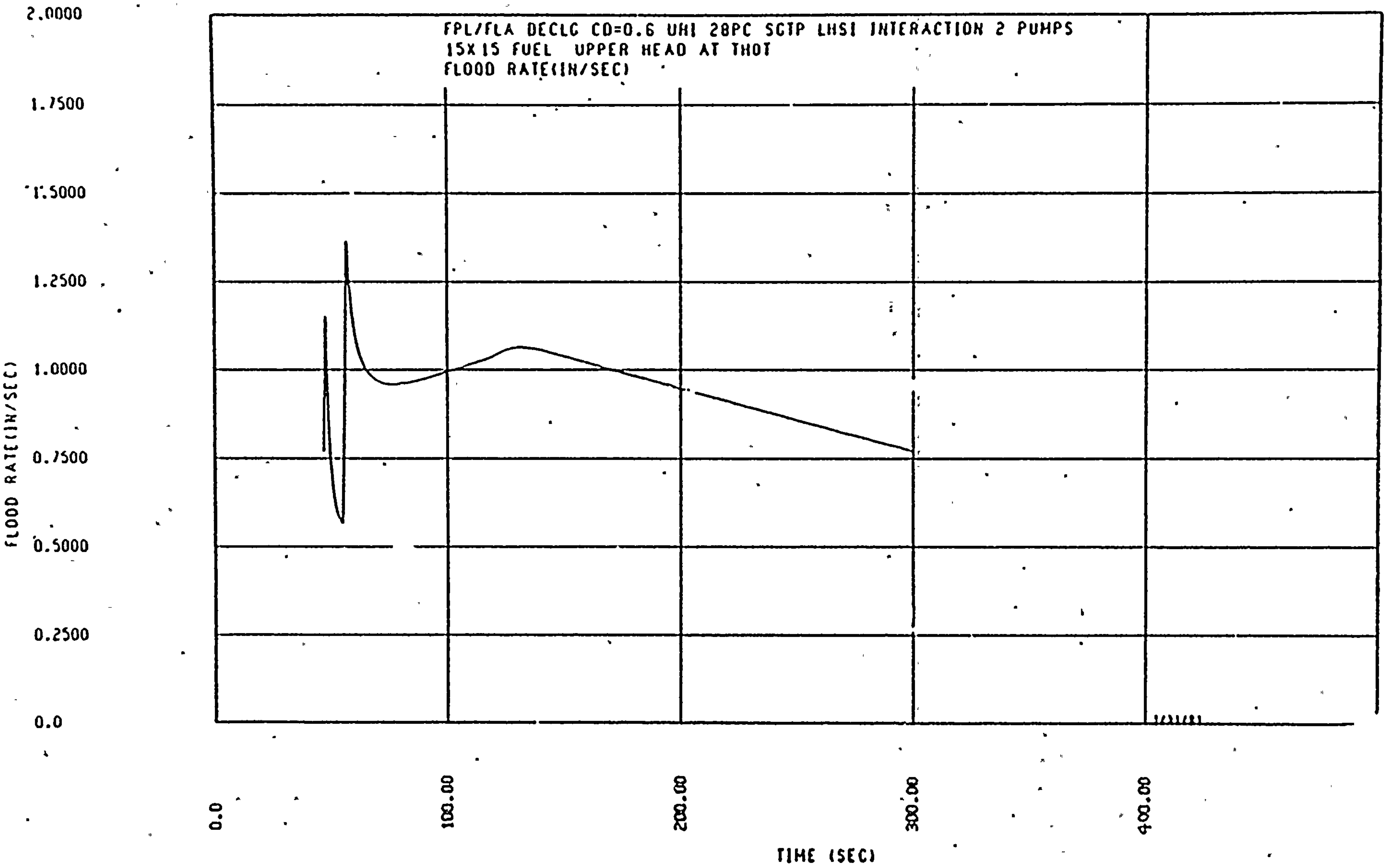


FIGURE 11A CORE INLET VELOCITY DURING REFLOOD TRANSIENT DECLG (C_D = 0.6)

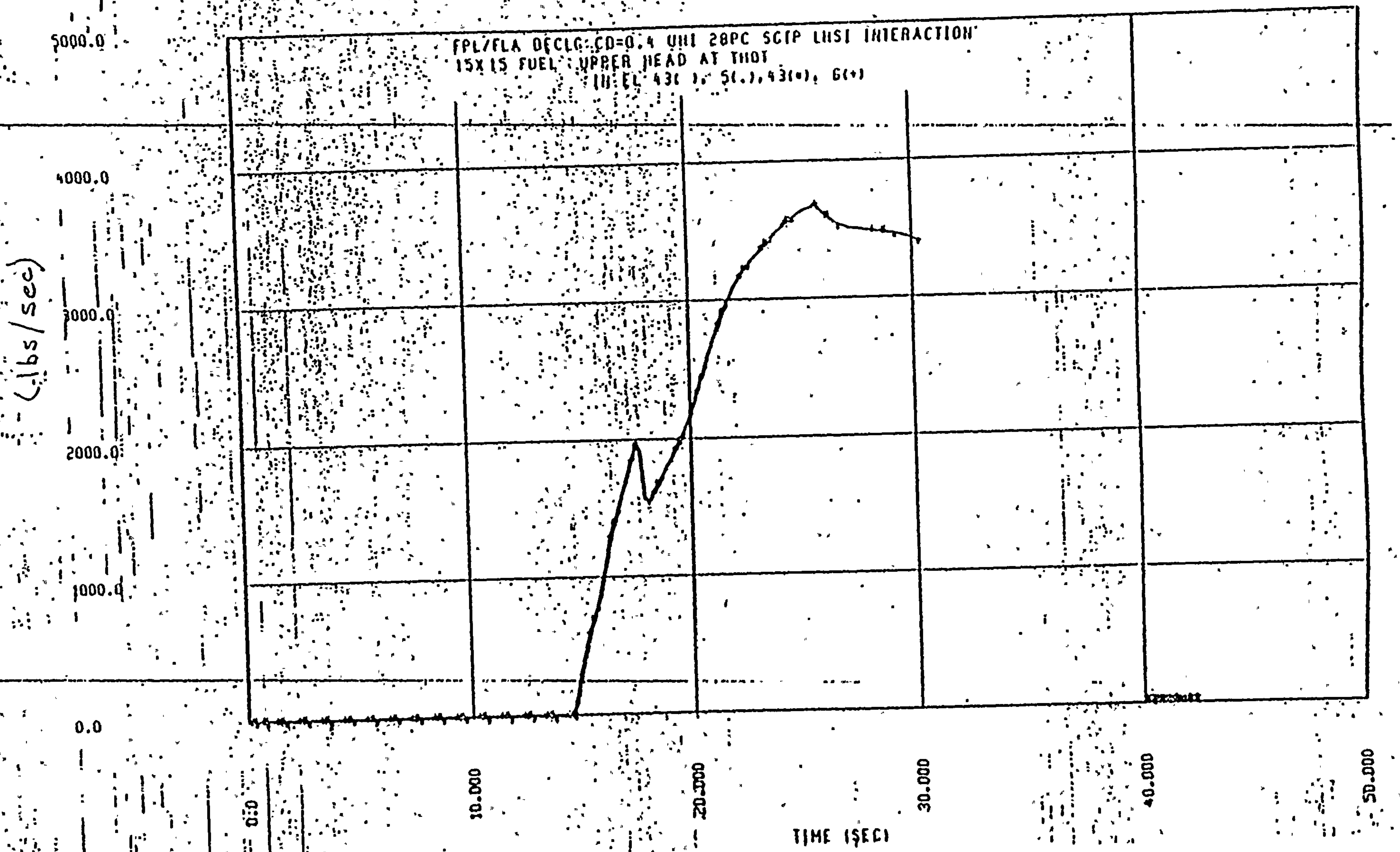


FIGURE 12 ACCUMULATOR INJECTION DURING BLOWDOWN

DECLG (C_D = 0.4)

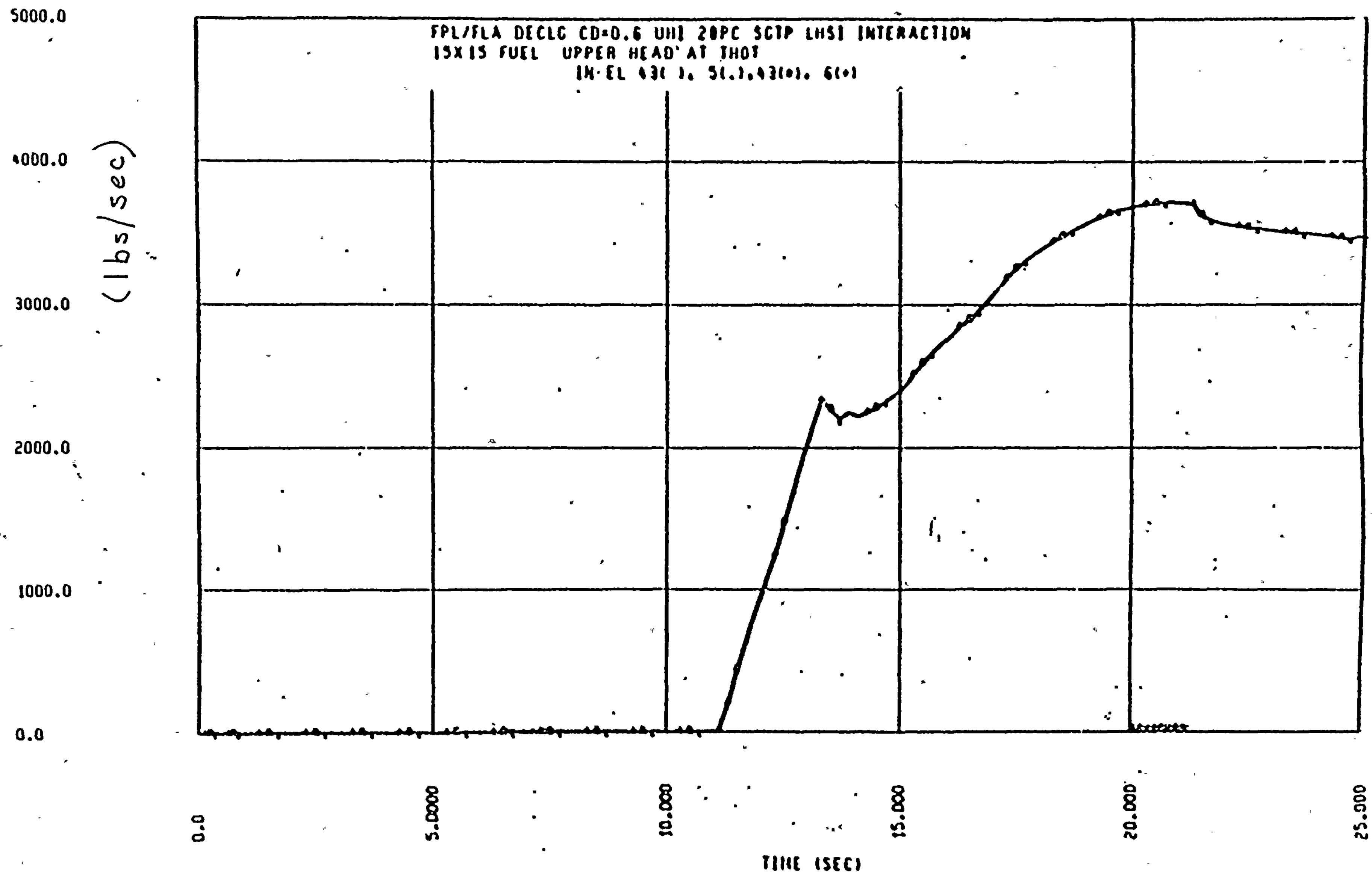


FIGURE 12A ACCUMULATOR INJECTION DURING BLOWDOWN
DECLG (C_D = 0.6)

FIGURE 13 PUMPED ECCS FLOW (REFLOOD)

DECLG (C_D = 0.4)

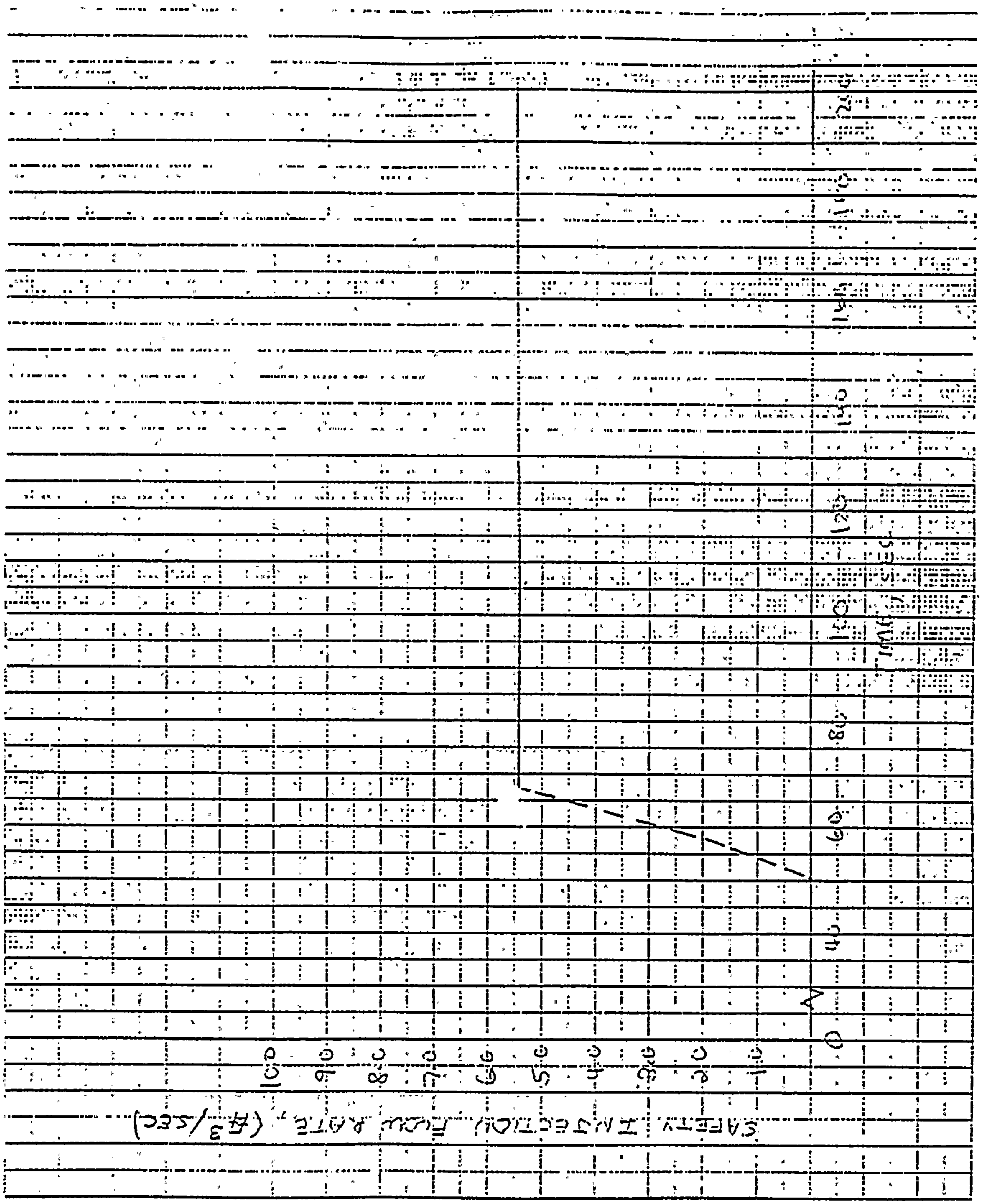
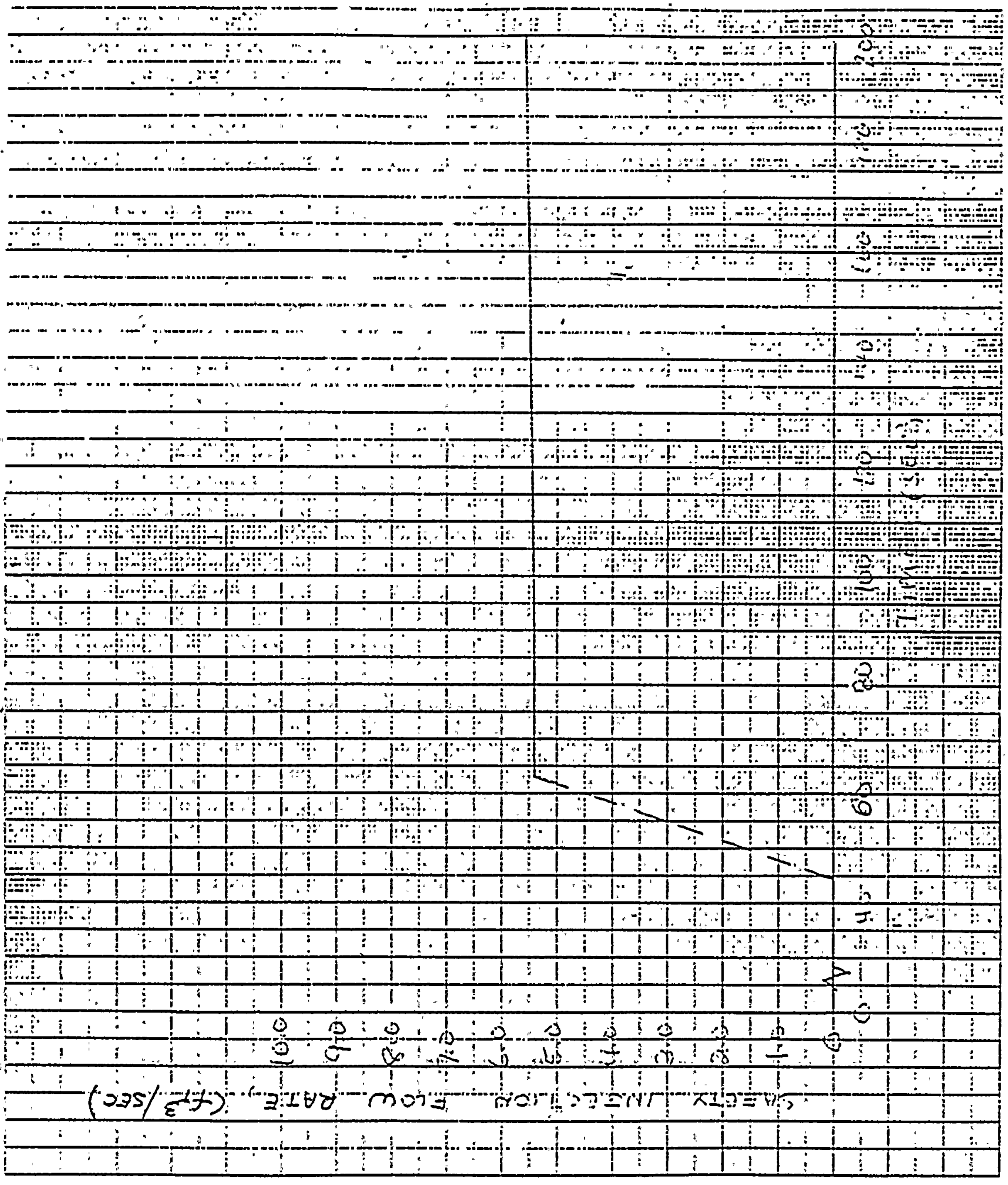


FIGURE 13A . PUMPED ECCS FLOW (REFLOOD)

DECLG (C_D = 0.6)



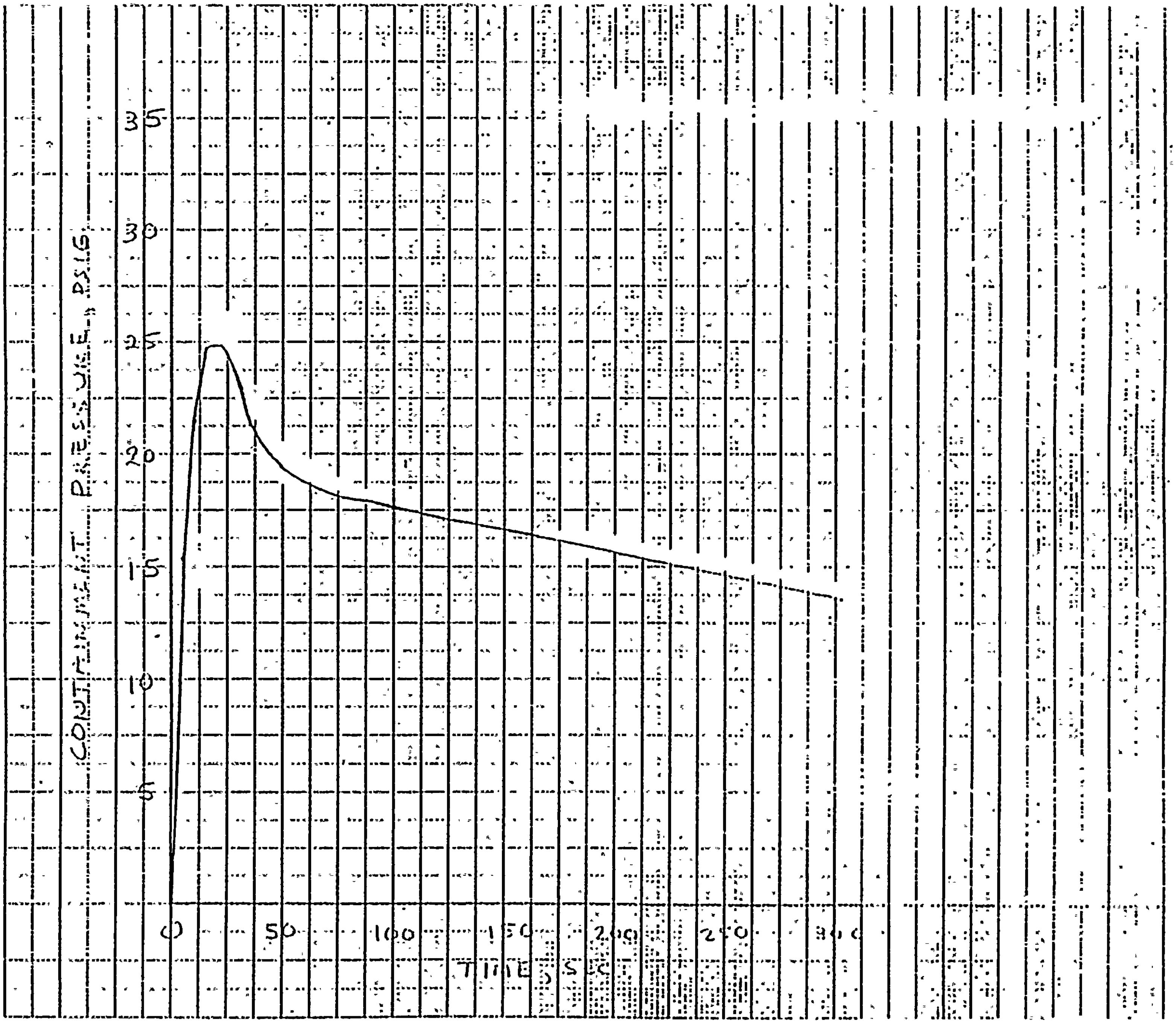
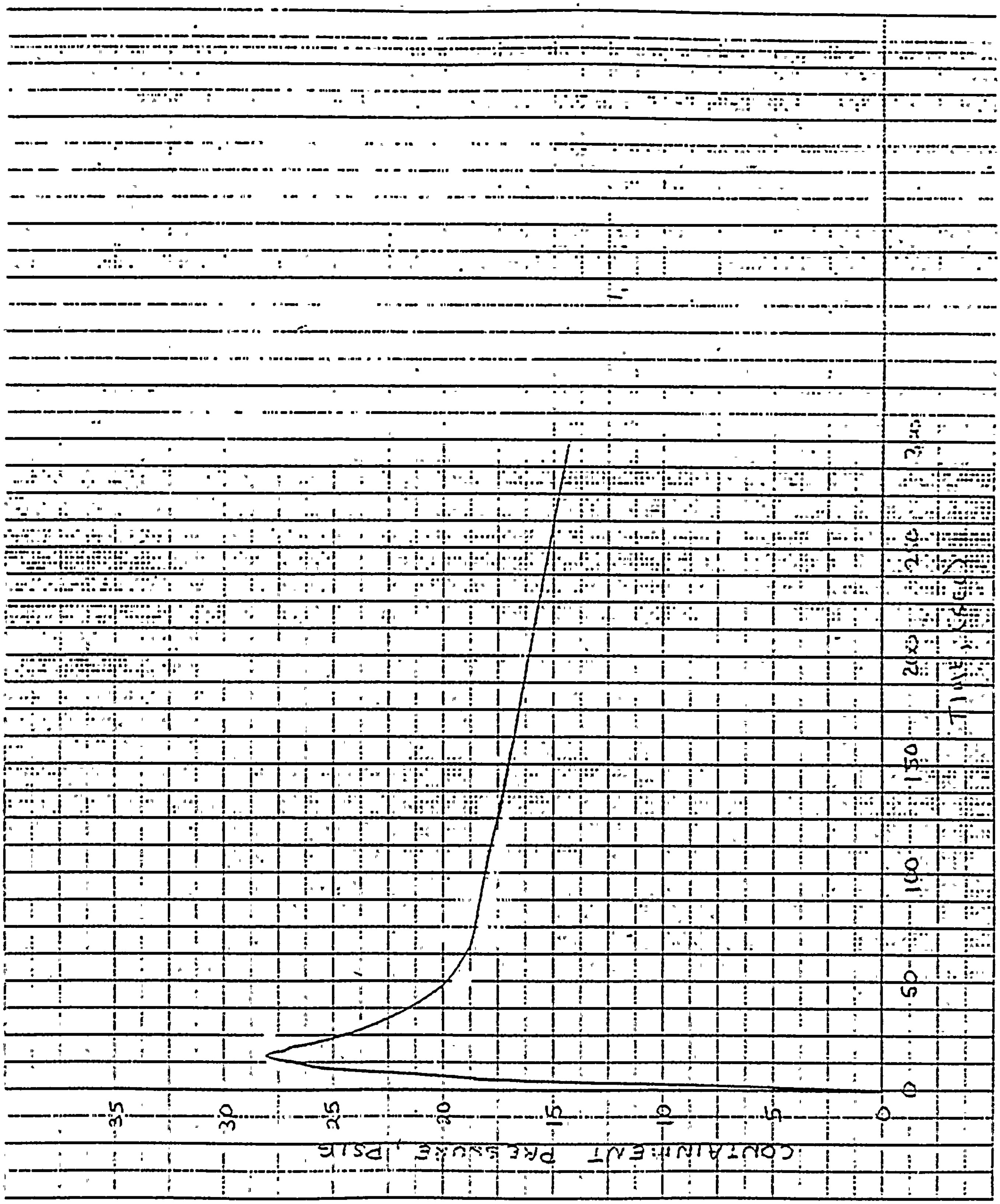


FIGURE 14. CONTAINMENT PRESSURE, PSIG
FPL/FLA DECLG (C_D = 0.4)

FIGURE 14A CONTAINMENT PRESSURE, PSIG

FPL/FLA DECLG (C_D = 0.6)



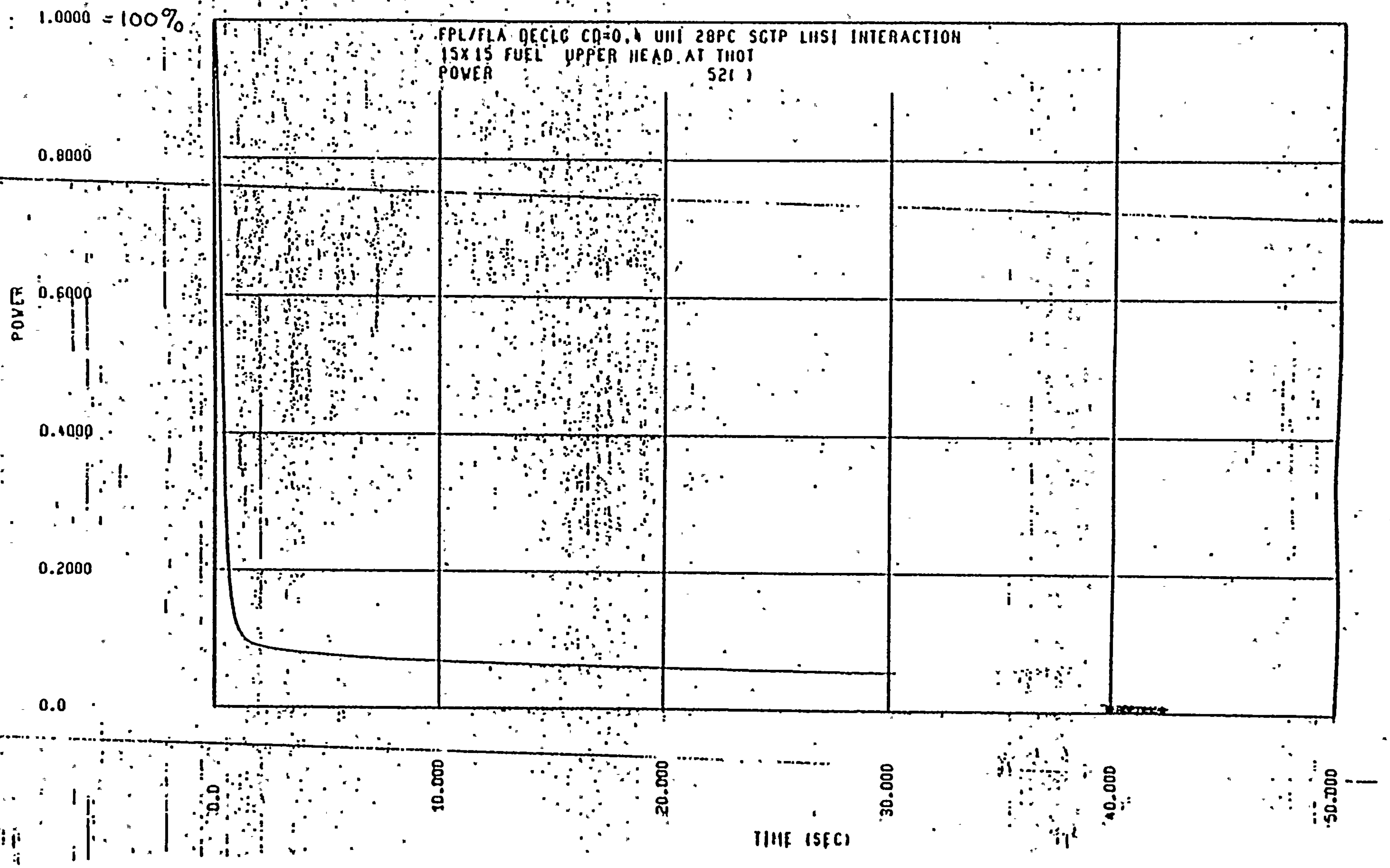


FIGURE 15 CORE POWER TRANSIENT DECLG (C_D = 0.4)

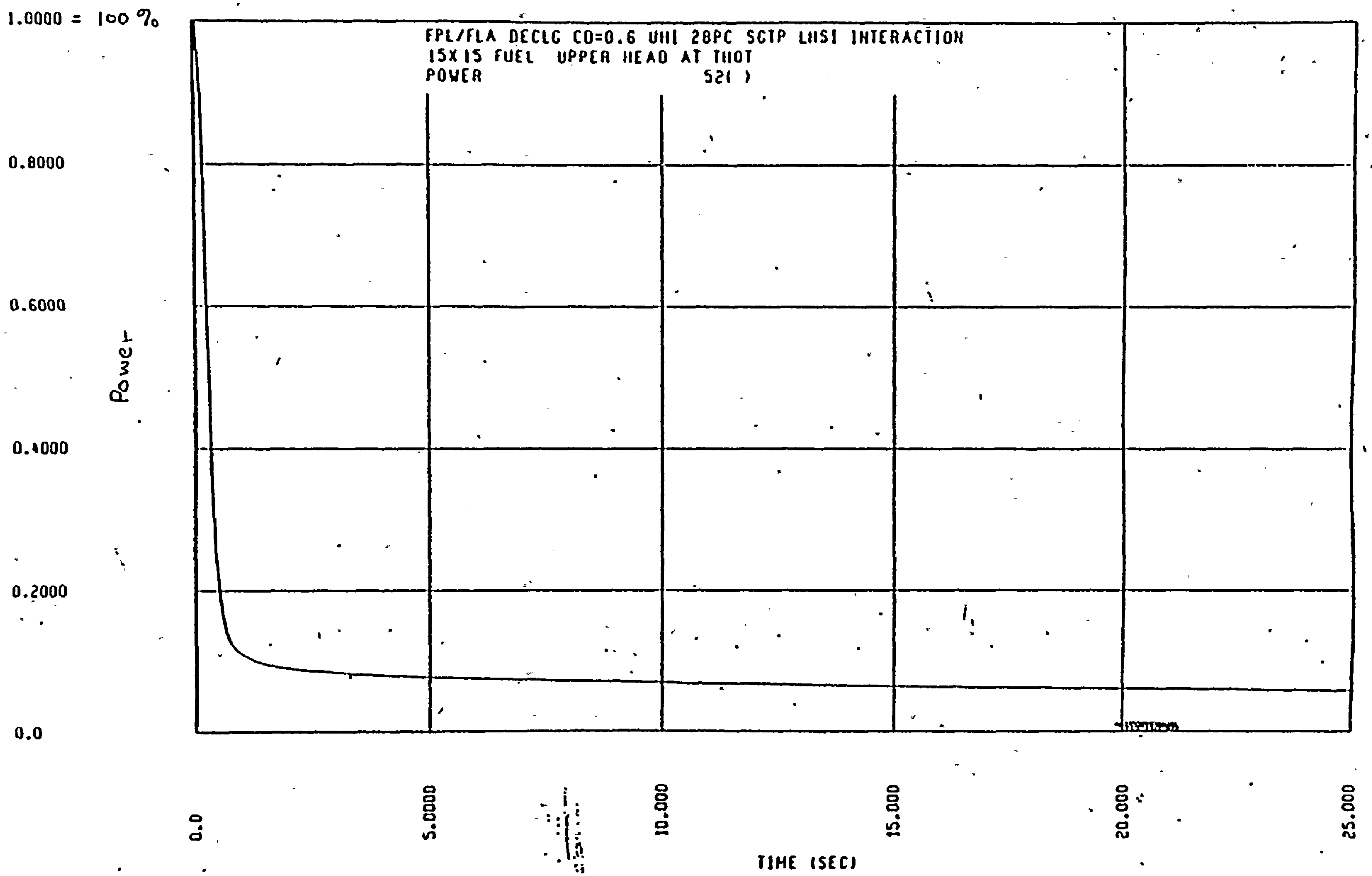


FIGURE 15A CORE POWER TRANSIENT DECLG (C_D = 0.6)

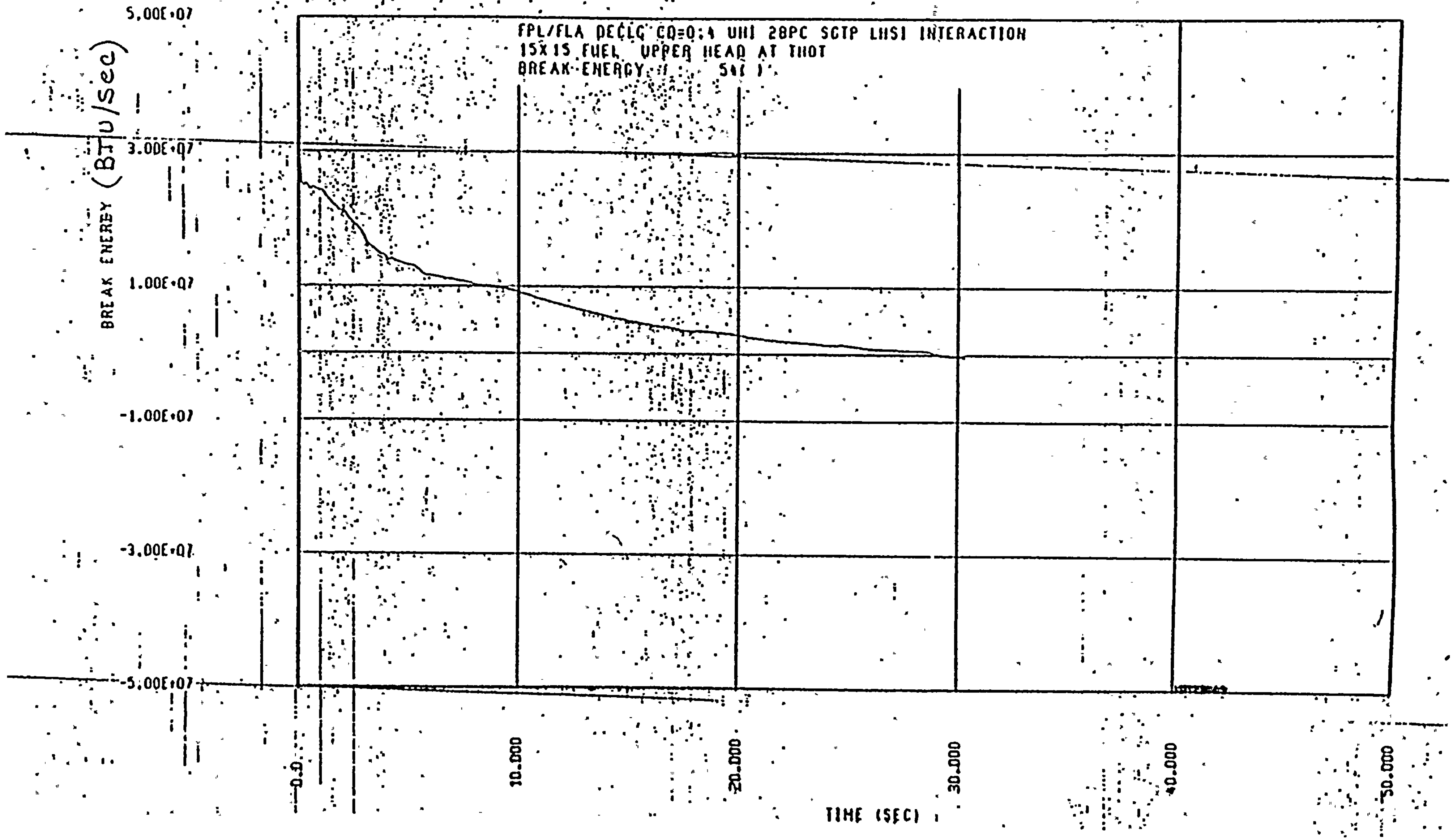


FIGURE 16 BREAK ENERGY DECLG (C_D = 0.4)

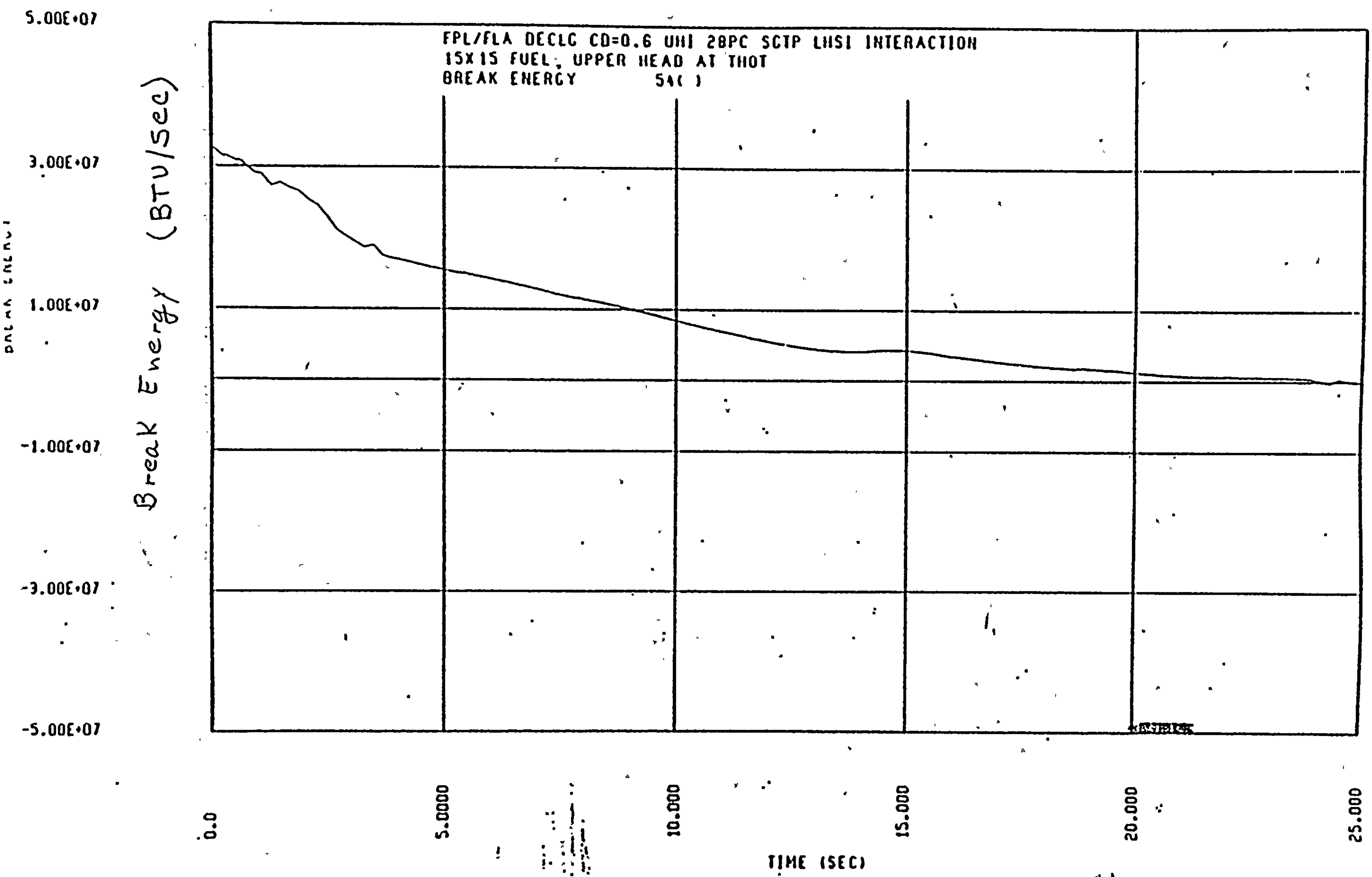


FIGURE 16A BREAK ENERGY DECLG (C_D = 0.6)

FIGURE 17 CONTAINMENT WALL HEAT TRANSFER COEFFICIENT
FPL/FLA DECLG ($C_D = 0.4$)

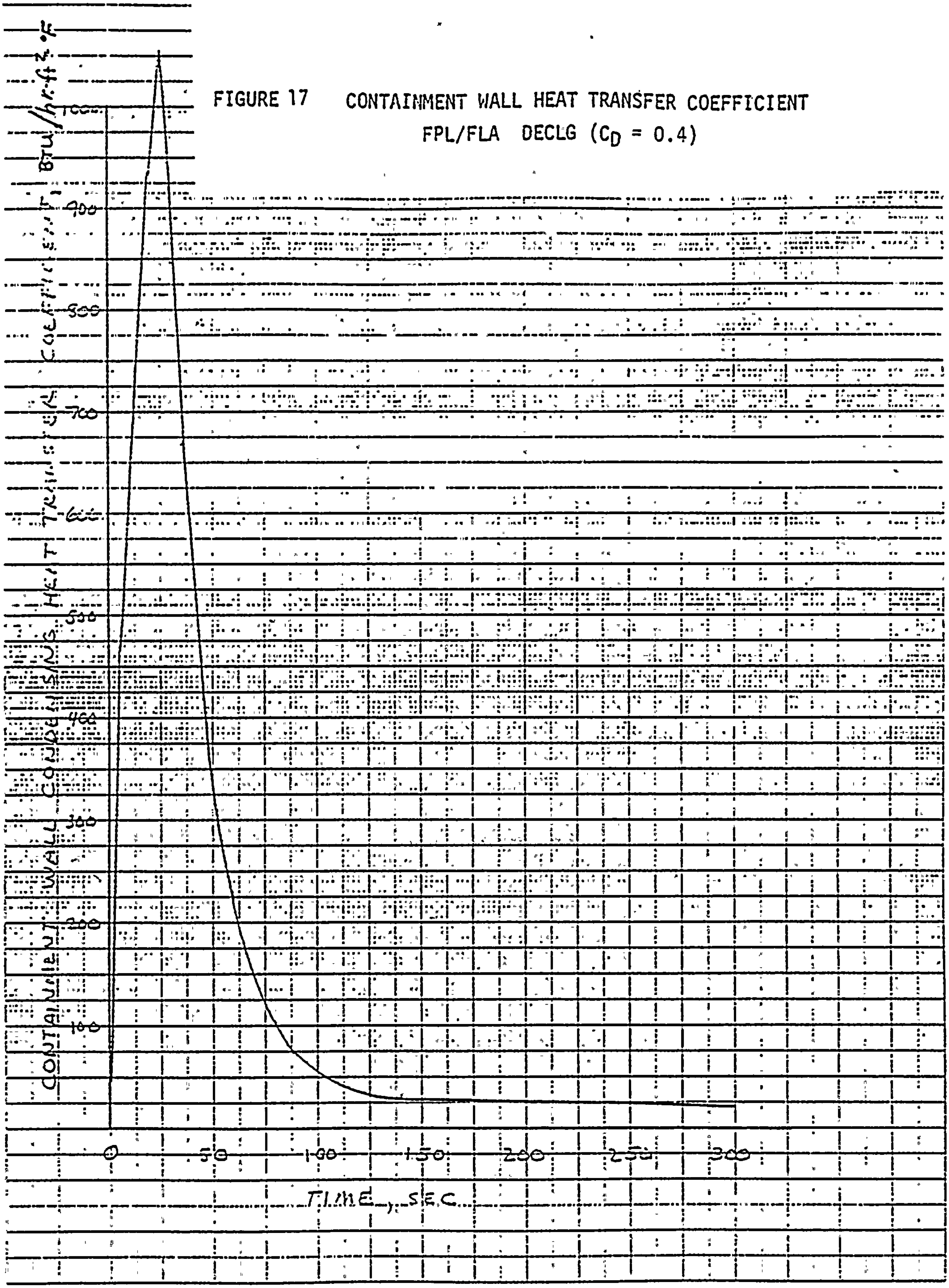
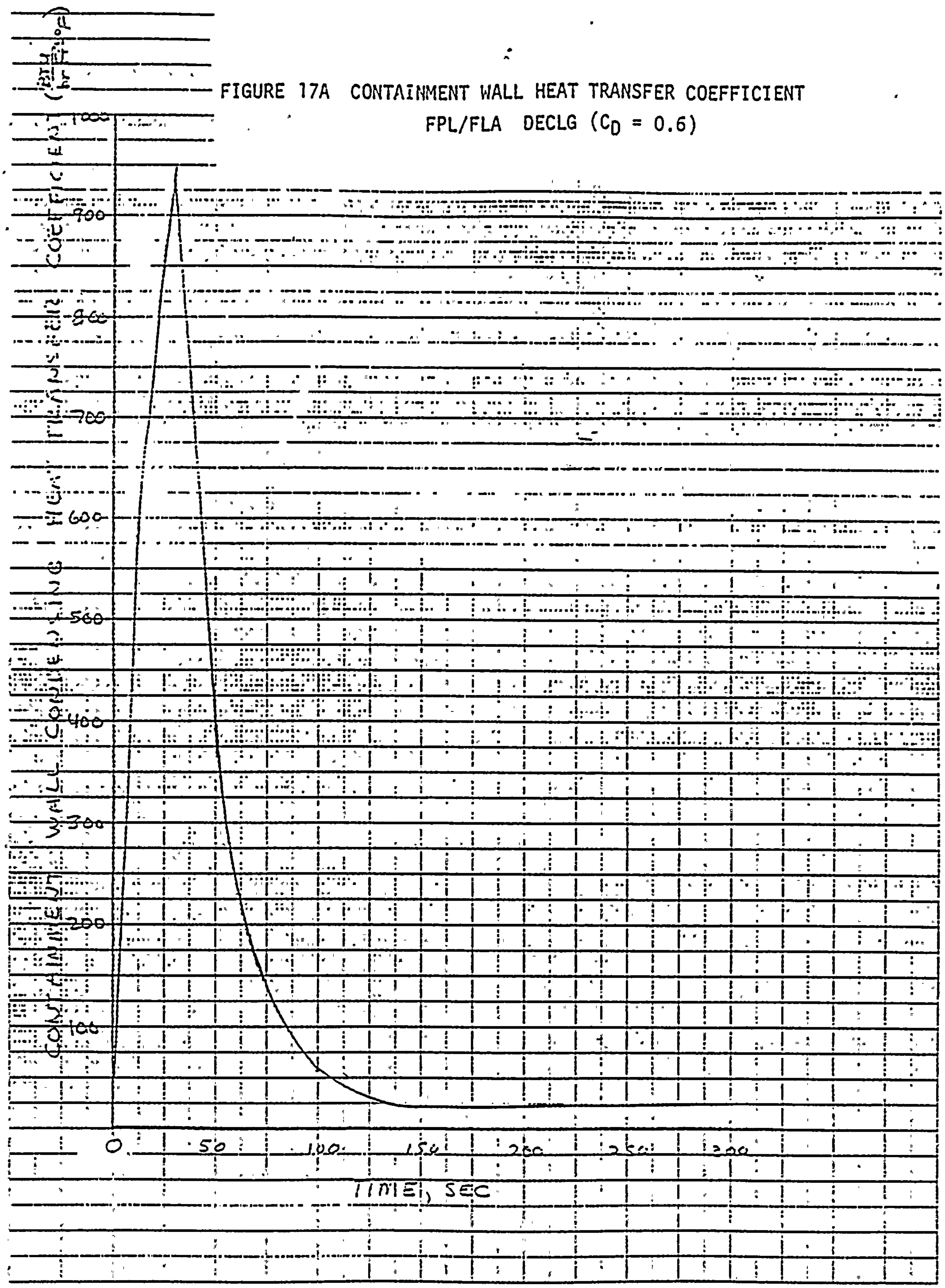


FIGURE 17A CONTAINMENT WALL HEAT TRANSFER COEFFICIENT

FPL/FLA DECLG (C_D = 0.6)



HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

(for ≤ 28% steam generator tube plugging and F_q = 2.125)

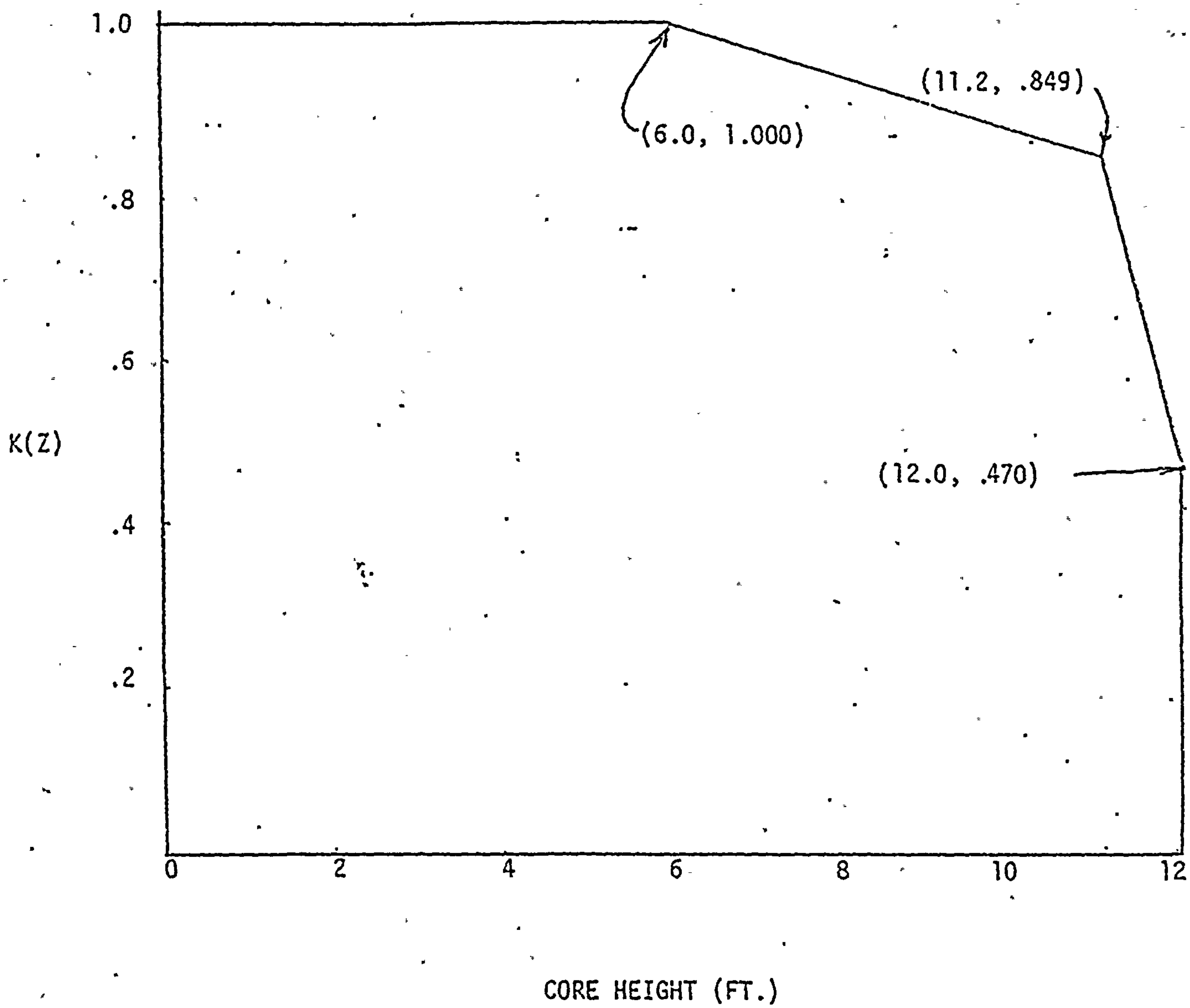


FIG. 18

SATAN MOD OR PTP 3 & 4 (54 ELEMENTS)

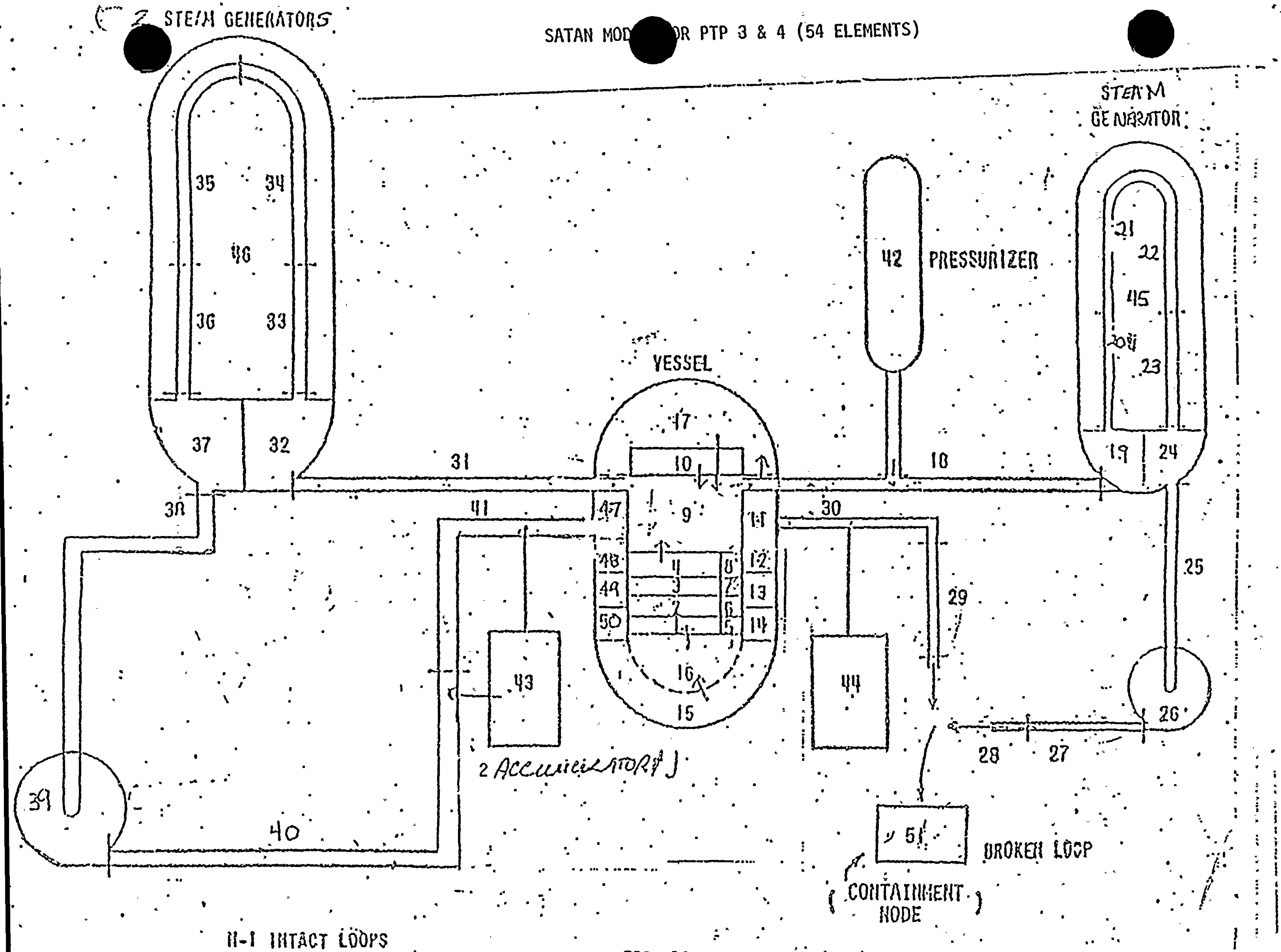


FIG. 19

APPENDIX A

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 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 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, 1979 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 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 anlytical 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 the fuel rod models presented in draft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for Turkey Point Units 3 and 4 with 28 percent SGTP.

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 Revised February 1978 with Safety Injection Interaction Modifications and UHI software technology

CORE PEAKING FACTOR 2.25

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 2041 °F = PCT_B

ELEVATION - 6.0 Feet.

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 2183 °F = PCT_N

ELEVATION - 7.5 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION 1.93 Percent

MAXIMUM CLAD STRAIN AT THIS ELEVATION - 9.04 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 ΔFQ → ~ 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:

$$\Delta PCT_1 = (0.027 + .03)\ (150^\circ F/.01) = 855^\circ F$$

Margin to the 2200°F limit is:

$$\Delta PCT_2 = 2200.^\circ F - PCT_B = \underline{159}^\circ F$$

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

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

$$= (\underline{855} - \underline{159}\ \left(\frac{.01}{150}\right)$$

$$= \underline{0.0464}\ \text{(but not less than zero).}$$

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 burst is calculated. Three sets of LOCA analysis results were studied to establish an acceptable sensitivity to apply generically in this evaluation. 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:

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

$$= \left(\frac{20}{.01}\right) (\underline{0.0904} - \underline{0.0193})$$

$$= \underline{142.2}^\circ F$$

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,

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

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

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

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

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = \underline{142.2 + 0 = 142.2}$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200°F = PCT_N = \underline{2200 - 2183 = 17°F}$$

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

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

ΔFQ_N = $\underline{0.125}$ 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_N,

or; $\Delta\ FQ_{PENALTY} = 0.125$

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

FQ ADJUSTMENT = $\underline{0.125}$

Final FQ Value = 2.25 - 0.125 = 2.125

ATTACHMENT B

NON-LOCA ANALYSIS

INTRODUCTION

The NSSS vendor has performed an evaluation of the impact of increasing the steam generator tube plugging level to 28 percent and of increasing the peaking factor F_Q to 2.175* on the non-LOCA accident analyses presented in Chapter 14 of the Turkey Point Unit 3 and Unit 4 FSAR.

The current Turkey Point safety analyses are valid for steam generator tube plugging levels up to 25%, with $\geq$ 95% of RCS thermal design flow, and a peaking factor of 1.93.

CHANGE IN STEAM GENERATOR TUBE PLUGGING LEVEL

Changes in the number of steam generator tubes plugged result in lower reactor coolant flow rates. Many of the non-LOCA accidents are strongly dependent on the flow rate available. Recent flow testing has shown that the previously assumed flow rate of 85025 gpm per loop,[1] corresponding to 95% of RCS thermal design flow rate, conservatively bounds the predicted flow with 28% tube plugging. Therefore, effects due a change in flow rate need not be addresssed. Because the time duration of the non-LOCA transients are very short, changes in pressure drops and time response due to tube plugging have little effect on the course of transients/accidents. Increasing the tube plugging level to 28% decreases the heat transfer area available in the steam generator. The impact of this, on the non-LOCA accident analysis also is not significant.

CHANGES IN F_Q LIMIT

The impact of higher F_Q values, on the non-LOCA accident analyses presented in Chapter 14 of the FSAR was analyzed.

The accident/transients considered were:

-Uncontrolled RCCS Withdrawal from Subcritical Conditions.

-Uncontrolled RCCS Withdrawal at Power

-Malpositioning of a Part Length Rod

-Rod Cluster Control Assembly (RCCA) Drop

-Start-up of an Inactive Reactor Coolant Loop

-Reduction in Feedwater Enthalpy Incident

-Exessive Load Increase Incident

-Loss of Reactor Coolant Flow (Locked Rotor Accident)

-Loss of Reactor Coolant Flow (Flow Coastdown Accident)

-Loss of External Electrical Load

-Loss of Normal Feedwater

-Loss of AC Power

-Rupture of Steam Pipe

* Proposed LOCA analysis requirement sets the limit on F_Q at $\leq$2.125, as per Attachment A.

- Rupture of Control Rod Mechanism Housing
- Chemical and Volume Control System Malfunction

The effect of flow reductions and changes in the value of the peaking factor, F_Q, on the above accidents are discussed in detail in a previous analysis (reference 1), which was performed for 25 percent steam generator tube plugging and 95% reactor coolant flow.

As the flow rate with 28% steam generator tube plugging is not expected to result in a flow rate less than 95% of design flow, changes in the flow rate need not be addressed and previous discussions and conclusions presented, (reference 1) are still valid. However, changing F_Q from 1.93 to 2.175* needs to be considered.

From the accidents presented in Chapter 14 of the FSAR, only the following accidents are affected by changing the value of F_Q:

LOSS OF REACTOR COOLANT FLOW (LOCKED ROTOR ACCIDENT)

The current applicable analysis of this event is based on a hot spot heat transfer calculation which utilizes heat flux and fuel temperatures associated with an F_Q of 2.55[2]. An F_Q limit of $\leq$ 2.175 results in a 14% reduction in total energy input to the hot spot which will compensate for the reduction of coolant flow to 95% of thermal design flow of 85025 gpm per loop, due to 28% tube plugging. This will result in peak fuel and clad temperatures well below the safety limit. Hence a change of F_Q from 1.93 to 2.175 has little effect on the safety limit margins and the conclusions of reference 1 are valid for 28% level tube plugging.

RUPTURE OF CONTROL ROD DRIVE MECHANISM HOUSING

The applicable current analysis was performed for an F_Q of 2.32 at 100% flow rate, (reference 2). From LOCA considerations F_Q has to be below 2.175*. Therefore, changing the value of F_Q from 1.93 to 2.175 still results in a reduction in initial fuel temperature which translates into a reduction in peak transient fuel temperatures compared to those in reference 2. Also according to reference 1 reducing the coolant flow to 95% of thermal design flow (85025 gpm), due to 28% tube plugging results in a 50°F increase of clad temperature which is still more than 400°F below the peak allowable temperature of 2700°F.

CONCLUSION

Increasing the value of the peaking factor, F_Q from 1.93 to 2.175* for LOCA considerations, with steam generator tube plugging $\leq$ 28% provided reactor coolant flow stays $\geq$ 255,075 gpm, does not have a significant effect on the

* Proposed LOCA analysis requirement sets the limit on F_Q at $\leq$2.125, as per Attachment A.

non-LOCA analysis due to the conservative inputs and large margin to the safety limit. In addition, since the current non-LOCA analysis was done for an F_Q of 2.32 any reduction from this value is considered an additional safety margin.

With regards to Overtemperature ΔT, Overpower ΔT and thermal hydraulic safety limit, the results presented for 25% tube plugging are still valid, with the reactor coolant flow greater or equal to 255075 gpm.

The arguments presented here are based on the fact that the reactor coolant flow rate is maintained within the limits set by the previous analysis for 25 percent level steam generator tube plugging (reference 1).

REFERENCES

1. Non-LOCA Accident Safety Evaluation for Higher Levels of Steam Generator Tube Plugging, for Turkey Point 3 and 4, June 1978 submitted with FPL letter L-78-242 dated July 20, 1978.

2. Reload Safety Evaluation, Turkey Point Plant Unit 4 Cycle 4.