REGULATORY DOCKET FILE COPY

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23209

December 26, 1978

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Serial No. 736
FR/MLB:jab

Docket Nos. 50-280
50-281

License Nos. DPR-32
DPR-37

Dear Mr. Denton:

AMENDMENT TO OPERATING LICENSE
SURRY POWER STATION UNIT NOS. 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGE NO. 75

Pursuant to 10 CFR 50.90, the Virginia Electric and Power Company hereby requests an amendment, in the form of changes to the Technical Specifications, to Operating License Nos. DPR-32 and DPR-37 for the Surry Nuclear Power Station, Units Nos. 1 and 2. The proposed changes are attached and have been designated as Change No. 75.

The proposed amendment is in response to the Exemption and Order for Modification of License for Surry Units No. 1 and 2 (reference your letter of September 13, 1978). As indicated in our letter of October 11, 1978 (Serial No. 535/091378), a LOCA-ECCS analysis and any required Technical Specifications changes, would be provided to demonstrate compliance to the requirements of 10 CFR 50.46 with a calculational model which fully conforms to the provisions of Appendix K, 10 CFR 50. This analysis and required Technical Specifications are provided in Attachments 1 and 2, respectively.

The proposed changes have been determined to be a Class III amendment for Unit 1. The amendment involves a single safety issue and does not involve a significant hazards consideration. The proposed change is Class 1 for Unit 2, since it is duplicated. Accordingly, a check in the amount of $4400.00 is attached in payment of the $4000.00 and $400.00 fees.

This proposed change has been reviewed and approved by both the Station Nuclear Safety and Operating Committee, and the System Nuclear Safety and Operating Committee. It has been determined that this request does not involve an unreviewed safety question, as defined in 10 CFR 50.59.

A001
S
40/40
w/CHECK
$4,400.00

P

7812280132

Your review and approval of the attached Technical Specifications Change is requested by January 2, 1979. Should you have questions, we would like to meet with you at your earliest convenience.

Very truly yours,

C. M. Stallings
Vice President-Power Station
and Production Operations

Attachments:
1. Safety Analysis
2. Proposed Technical Specifications
3. Voucher check no. 8493, in the amount of $4400.00

cc: Mr. James P. O'Reilly, Director
Office of Inspections and Enforcement
Region II

COMMONWEALTH OF VIRGINIA)
) S. S.
CITY OF RICHMOND)

Before me, a Notary Public, in and for the City and Commonwealth aforsaid, today personally appeared Sam C. Brown, Jr., who being duly sworn, made oath and said (1) that he is Vice President-Power Station Engineering and Construction, of the Virginia Electric and Power Company, (2) that he is duly authorized to execute and file the foregoing Amendment in behalf of that Company, and (3) that the statements in the Amendment are true to the best of his knowledge and belief.

Given under my hand and notarial seal this 26th day of December, 1978.

My Commission expires January 26, 1981.

Robert M. [illegible]
Notary Public

(SEAL)

ATTACHMENT 1

1.0 INTRODUCTION

A reanalysis of the ECCS cooling performance for the postulated large break Loss Of Coolant Accident (LOCA) has been performed which is in compliance with Appendix K to 10 CFR 50. The results of this reanalysis are presented herein* and are in compliance with 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors. This reanalysis was performed with the NRC approved(1) February, 1978 version of the Westinghouse LOCA-ECCS evaluation model. The analytical techniques used are in full compliance with 10CFR50, Appendix K and satisfy the requirements of Reference 2.

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur and include such items as the core peaking factors, the containment pressure, and the performance of the emergency core cooling system (ECCS). All assumptions and initial operating condition input data used in this reanalysis were the same as was used in the previously applicable LOCA-ECCS analysis(3) except for 1) the limiting value of the heat flux hot channel factor was increased to 2.05, 2) the core inlet temperature value was decreased to 534.5°F, 3) the value of initial fuel temperature was increased in that the temperature was calculated based on generic values of fuel characteristics rather than as-built values, 4) the change of several ECCS containment parameters was made to reflect the containment response in a more realistic, but still conservative, manner, and 5) the assumed steam generator tube plugging level was increased to 28 percent.

---

* It should be noted that reanalysis of the small break LOCA is not necessary, and therefore, the analysis of this accident submitted by Reference 4 remains applicable.

2.0 DESCRIPTION OF POSTULATED MAJOR REACTOR COOLANT PIPE RUPTURE (LOSS OF COOLANT ACCIDENT - LOCA)

A LOCA is the result of a rupture of the Reactor Coolant System (RCS) piping or of any line connected to the system. The system boundaries considered in the LOCA analysis are defined in the FSAR. Sensitivity studies(5) have indicated that a double-ended cold leg guillotine (DECLG) major break is limiting. Should a DECLG break occur, rapid depressurization of the RCS occurs. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A Safety Injection System (SIS) signal is actuated when the appropriate setpoint is reached and the high head safety injection pumps are activated. The actuation and subsequent activation of the ECCS, which occurs with the SIS signal, assumes the most limiting single failure event. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. (It should be noted, however, that no credit is taken in the analysis for the insertion of control rods to shut down the reactor.)
2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the reactor coolant system. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10CFR50. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection of steam as the major heat transfer mechanisms. During the refill period, it is assumed that rod-to-rod radiation is the only core heat transfer mechanism. The heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures. For the case of continued heat addition to the secondary side, secondary side pressure increases and the main safety valves may actuate to reduce the pressure. Make-up to the secondary side is automatically provided by the auxiliary feedwater system. Coincident with the Safety Injection Signal, normal feedwater flow is stopped by closing the main feedwater control valves and tripping the main feedwater pumps. Emergency feedwater flow is initiated by starting the auxiliary feedwater pumps. The secondary side flow aids in the reduction of reactor coolant system pressure. When the reactor coolant system depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is then made that injected accumulator water bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10CFR50. In addition, the reactor coolant pumps are assumed to be tripped at the initiation of the accident and effects of pump coastdown are included in the blowdown analysis.

The water injected by the accumulators cools the core and subsequent operation of the low head safety injection pumps supply water for long term cooling. When the RWST is nearly empty, long term cooling of the core is accomplished by switching to the recirculation mode of core cooling, in which the spilled borated water is drawn from the containment sump by the low head safety injection pumps and returned to the reactor vessel.

The containment spray system and the recirculation spray system operate to return the containment environment to a subatmospheric pressure.

The large break LOCA transient is divided, for analytical purposes, into three phases: blowdown, refill, and reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of inter-related computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339.(6) This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with 10 CFR 50, Appendix K. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in WCAP-8306(7), WCAP-8326(8), WCAP-8171(9) and WCAP-8305(10), respectively. These codes are able to assess whether sufficient heat transfer geometry and core amenability to cooling are preserved during the time spans applicable to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code is used to calculate the transient during the refill and reflood phases of the accident. The COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compare the thermal transient of the hottest fuel rod during the three phases.

SATAN-VI is used to determine the RCS pressure, enthalpy, and density, as well as the mass and energy flow rates in the RCS and steam generator secondary, as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator mass and pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end of blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow rates that are assumed to be vented to the containment. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. WREFLOOD is also linked to the LOCTA-IV code in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature.

LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel and clad temperature of the hottest rod in the core. The input to LOCTA-IV consists of appropriate thermal-hydraulic output from SATAN-VI and WREFLOOD and conservatively selected initial RCS operating conditions. These initial conditions are summarized in Table 1 and Figure 1. (The axial power shape of Figure 1 assumed for LOCTA-IV is a cosine curve which has been previously verified to be the shape that produces the maximum peak clad temperature(11).)

The COCO code, which is also used throughout the LOCA analysis, calculates the containment pressure. Input to COCO is obtained from the mass and energy flow rates assumed to be vented to the containment as calculated by the SATAN-VI and WREFLOOD codes. In addition, conservatively chosen initial containment conditions and an assumed mode of operation for the containment cooling system are input to COCO. These initial containment conditions and assumed modes of operation are provided in Table 2.

3.0 DISCUSSION OF SIGNIFICANT INPUT

Significant changes in the input used in this reanalysis from those used in the currently applicable analyses are delineated in Section 1.0 and are discussed in more detail below.

The changes made in this analysis reflect the operational conditions and limits necessary to allow full power operation at a steam generator tube plugging level of up to 28%.* In order to ensure compliance with the 10 CFR 50.46 acceptance criteria, several changes to the operational limits assumed in the analysis were made. Specifically, the assumed value of the heat flux hot channel factor was increased from its current value of 1.94 in Unit 1 and 1.79 in Unit 2 to a value of 2.05 for both Units 1 and 2. Changes were also made to the fuel temperature, reactor coolant temperature, and containment structural heat sinks.

The previous analysis had been performed using initial fuel temperatures which were derived from as-built fuel parameters. While this assumption was conservative for the current operating cycles, the possibility did exist that future reload cycles would have different as-built fuel parameters which could result in slightly nonconservative initial fuel temperature. In order to insure conservative results for all future reload cycles, the initial fuel temperatures for this analysis were calculated based on limiting generic 15x15 fuel parameters.

The value assumed for the reactor coolant system core inlet temperature for this analysis was changed to be consistent with the overall conservatism inherent in the analysis. Specifically, a core inlet temperature of 534.5°F was used in the analysis. This value is the best-estimate core

---

*The percentage of steam generator tube plugging for each steam generator is assumed to be identical. This assumption accurately approximates the actual steam generator tube plugging distribution. The impact of non-symmetric plugging distribution on the LOCA-ECCS analysis has been found to be insignificant[12].

inlet temperature as determined from operational data and is adequate to encompass the applicable steam generator tube plugging range.

The amount of the various categories of structural heat sinks provided in Table 2 were reviewed in detail. Based on the as-built plant containment, these categories were conservatively revised and credit was taken for carbon steel painted surfaces. The remainder of the containment initial conditions and pressure reduction systems parameters were the same as used in the previous analyses.

Finally, this analysis was conducted with the February, 1978 version of the Westinghouse LOCA-ECCS Evaluation Model (13,14,15). This model version includes a modification to the SATAN VI and LOCTA IV codes to correct this calculation to properly account for the volumetric heat generation due to the metal-water reaction(16).

4.0 RESULTS

Table 3 presents the time sequence of events and Table 4 presents the results for the double-ended cold leg guillotine break (DECLG) for the C_D = 0.4 discharge coefficient. The DECLG has been determined to be the limiting break size and location based on the sensitivity studies reported in Reference 5. Based on all previous LOCA-ECCS submittals for the Surry units, the results obtained with a C_D = 0.4 discharge coefficient have always been limiting. The applicability of this conclusion (i.e. C_D=0.4 is the limiting break size) for this analysis was verified. Assuming the initial conditions and modes of operation presented in Tables 1 and 2 and Figure 1, the current analyses resulted in a peak clad temperature of 2172 °F, a maximum local cladding oxidation level of 7.81 percent, and a total core metal-water reaction of less than 0.3 percent. The detailed results of the LOCA reanalysis are provided in Tables 3 through 6 and Figures 2 through 18.

5.0 CONCLUSIONS

For breaks up to and including the double-ended severance of a reactor coolant pipe and for the operating conditions specified in Tables 1 and 2, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10CFR50.46. That is:

1. The calculated peak fuel rod clad temperature is below the requirement of 2200°F.
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 localized cladding oxidation limits of 17% are not exceeded during or after quenching.
4. The core remains amenable to cooling during and after the break.
5. The core temperature is reduced and the long-term decay heat is removed for an extended period of time.

6.0 REFERENCES

1. Letter from NRC (J. F. Stolz) to Westinghouse (T. M. Anderson) dated August 29, 1978.
2. Letter from NRC (A. Schwencer) to Vepco (W. L. Proffitt) dated September 13, 1978.
3. Letter from Vepco (C. M. Stallings) to NRC (E. G. Case) dated May 26, 1978, Serial No. 303.
4. Letter from Vepco (C. M. Stallings) to NRC (K. R. Goller), Serial No. 500-S,dated June 6, 1975.
5. Buterbaugh, T. L., Johnson, W. J. and Kopelic, S. D., "Westinghuse ECCS-Plant Sensitivity Studies," WCAP-8356, July 1974.
6. Bordelon, F. M., Massie, H. W., and Zordan, T. A., "Westinghouse ECCS Evaluation Model-Summary" WCAP-8339, July 1974.
7. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8306, June 1974.
8. Bordelon, F. M. and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8326, June, 1974
9. Kelly, R. D., et al., "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8171, June 1974.
10. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June, 1974.
11. Letter from Vepco (C. M. Stallings) to NRC (E. G. Case) dated February 17, 1978, Serial No. 092.
12. Letter from Vepco (C. M. Stallings) to NRC (B. C. Rusche), Serial No. 260/092276, dated October 19, 1976.
13. "Westinghouse ECCS Evaluation Model - February, 1978 Version", WCAP-9220-P-A (Proprietary) and WCAP-9221-P-A (Non-Proprietary), February 1978.
14. Letter from Westinghouse (T. M. Anderson) to NRC (J. F. Stolz), dated November 1, 1978, Serial No. NS-TMA-1981.
15. Letter from Westinghouse (T. M. Anderson) to NRC (R. Tedesco), dated December 11, 1978, Serial No. NS-TSM-2014.
16. Letter from Westinghouse (C. Eicheldinger) to NRC (J. F. Stolz), dated April 7, 1978, Serial No. NS-CE-1751.

TABLE 1

INITIAL RCS CONDITIONS

Core Power, Mwt, 102% of		2441
Peak Linear Power, Kw/ft, 102% of		12.72
Peaking Factor (F_Q)		2.05
Accumulator Water Volume, ft^3		975 (per accumulator)
Reactor Coolant System Flow, gpm (90% of Thermal Design)		79,650 (per loop)
Steam Generator Tube Plugging Level, %		28
Inlet Temperature, ^oF,		534.5
Temperature of the Fluid in the Upper Head Region of the Reactor Vessel		100% of T_{HOT}
Fuel Temperatures		Generic 15x15
Hot Assembly Radial Peaking Factor		1.38
Hot Rod Radial Peaking Factor		1.45
Most Limiting Fuel Region	Cycle	Region
Unit 1	All	All
Unit 2	All	All

TABLE 2

C[illegible]TAINMENT DATA (DRY CONTAINMENT[illegible]

NET FREE VOLUME	1.863×10^6 Ft^3
INITIAL CONDITIONS	
Pressure	9.35 psia
Temperature	90°F
RWST Temperature	40°F
Service Water Temperature	32.5°F
Outside Temperature	9°F
SPRAY SYSTEM I CONTAINMENT SPRAY SYSTEM	
Number of Pumps Operating	2
Runout Flowrate	3200 gpm
Actuation Time	52 secs
SPRAY SYSTEM II - INSIDE RECIRCULATION SPRAY SUBSYSTEM	
Number Pumps Operating	2
Runout Flowrate (each)	3500 gpm
Actuation Time	190 secs
Heat Exchanger (UA (per pump))	5.18×10^6 BTU/HR-°F
Service Water Flow (per exchanger)	6900 gpm
SPRAY SYSTEM II - OUTSIDE RECIRCULATION SPRAY SUBSYSTEM	
Number Pumps Operating	2
Runout Flowrate (each)	2250 gpm
Actuation Time	410 secs
Heat Exchanger (UA (per pump))	5.18×10^6 BTU/HR-°F
Service Water Flow (per exchanger)	6900 gpm

TABLE 2 (Continued)

STRUCTURAL HEAT SINKS

Type/Thickness (in.)	Area (ft^2), w/uncertainty
Concrete 6	8,393
Concrete 12	62,271
Concrete 18	55,365
Concrete 24	11,591
Concrete 27	9,404
Concrete 36	3,636
Carbon Steel 0.375	46,489*
Concrete 54	
Carbon Steel 0.50	25,652*
Concrete 30	
Concrete 26 (Floor)	12,110
Carbon Steel 0.239	158,059*
Stainless Steel 0.306	17,519
Aluminum 0.0091	3,911

---

*Credit for painted surfaces was taken only for the nominal surface area

TABLE 3

TIME SEQUENCE OF EVENTS

	DECLG CD=0.4 (Sec)
START	0.0
Reactor Trip	0.538
S. I. Signal	2.32
Acc. Injection	15.4
End of Bypass	24.45
Pump Injection	27.32
End of Blowdown	28.01
Bottom of Core Recovery	36.15
Acc. Empty	44.67

TABLE 4

RESULTS FOR DECLG C_D=0.4

Peak Clad Temp, $^{\circ}F$	2172
Peak Clad Location, Ft.	7.75
Local Zr/H_2O RXN (max), %	7.81
Local Zr/H_2O Location, Ft.	7.75
Total Zr/H_2O RXN, %	$<$0.3
Hot Rod Burst Time, sec.	26.20
Hot Rod Burst Location, Ft.	6.0

TABLE 5

REFLOOD MASS AND ENERGY RELEASES
DECLG ($C_D = 0.4$)

TIME (SEC)	TOTAL MASS FLOWRATE (LB/SEC)	TOTAL ENERGY FLOWRATE (10^5BTU/SEC)
41.76	36.14	0.467
50.71	108.20	1.085
65.21	218.77	1.331
82.91	258.47	1.371
102.91	269.19	1.331
124.81	277.03	1.280
174.11	291.36	1.171
233.21	311.22	1.029
312.96	330.02	0.895

TABLE 6

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT
DECLG, C_D=0.4

TIME (SEC)	MASS FLOWRATE* (LBm/SEC)
0.0	0.0
1.0	4205
3.0	3590
5.0	3185
7.0	2892
10.0	2566
15.0	2189
20.0	1934
22.0	1857
22.86	0.0

---

*For energy mass flowrate multiply mass flowrate by a constant of 58.82 BTU/LBM.

Figure 1 Peaking Factor versus Core Height - FQ =2.05

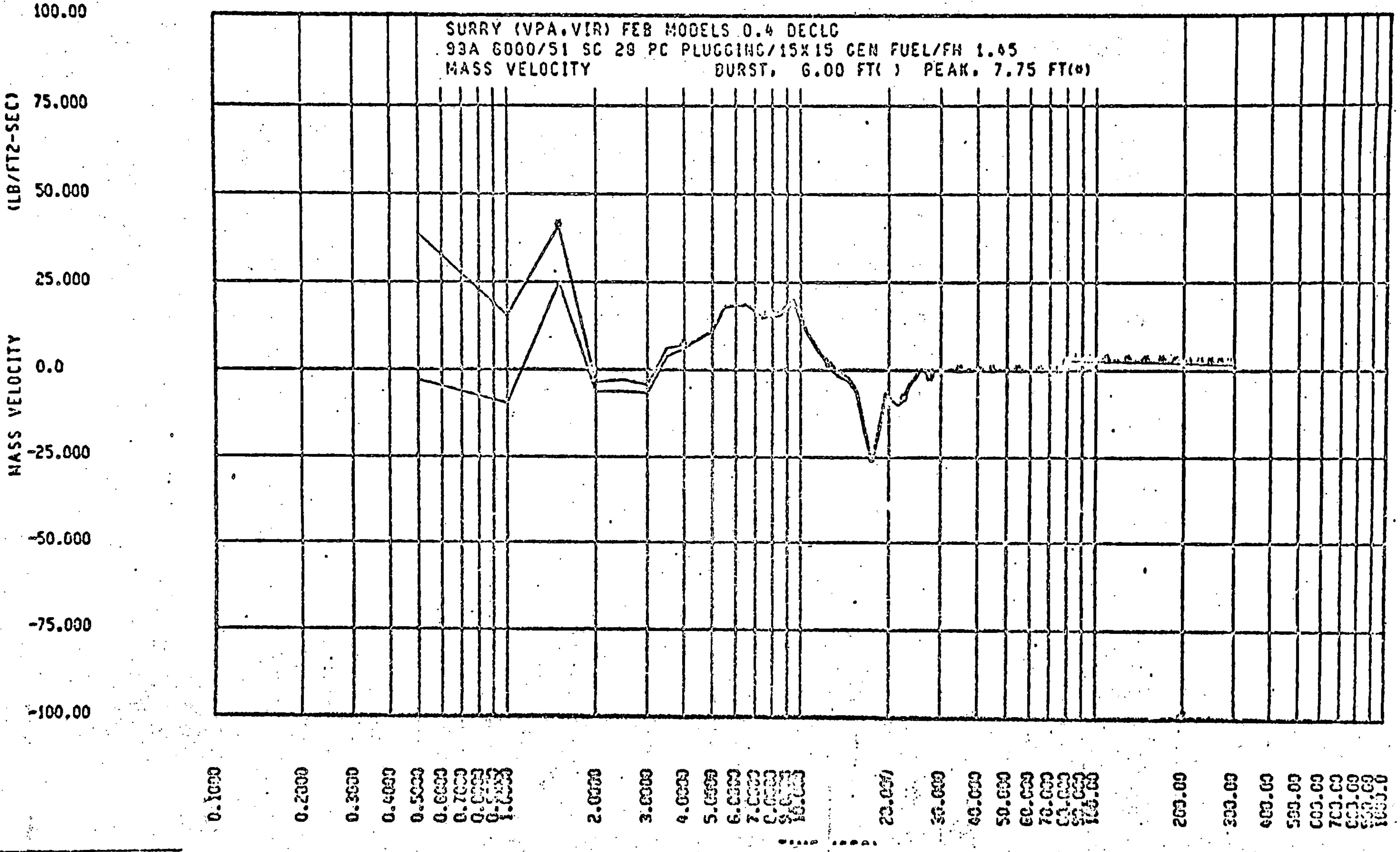


Figure 2 Mass Velocity - DECLG ($C_D = 0.4$)

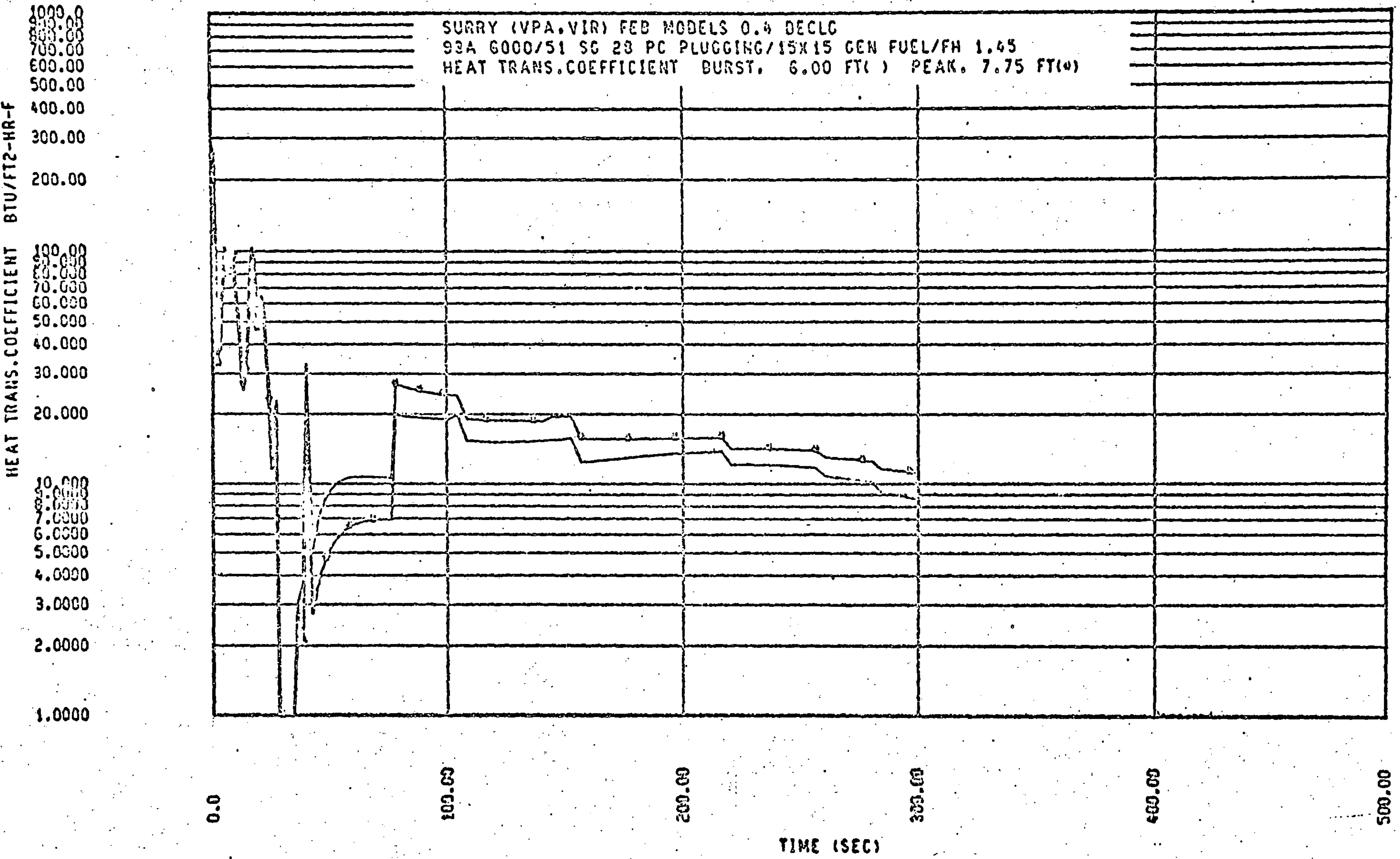


Figure 3 Heat Transfer Coefficient - DECLG ($C_D = 0.4$)

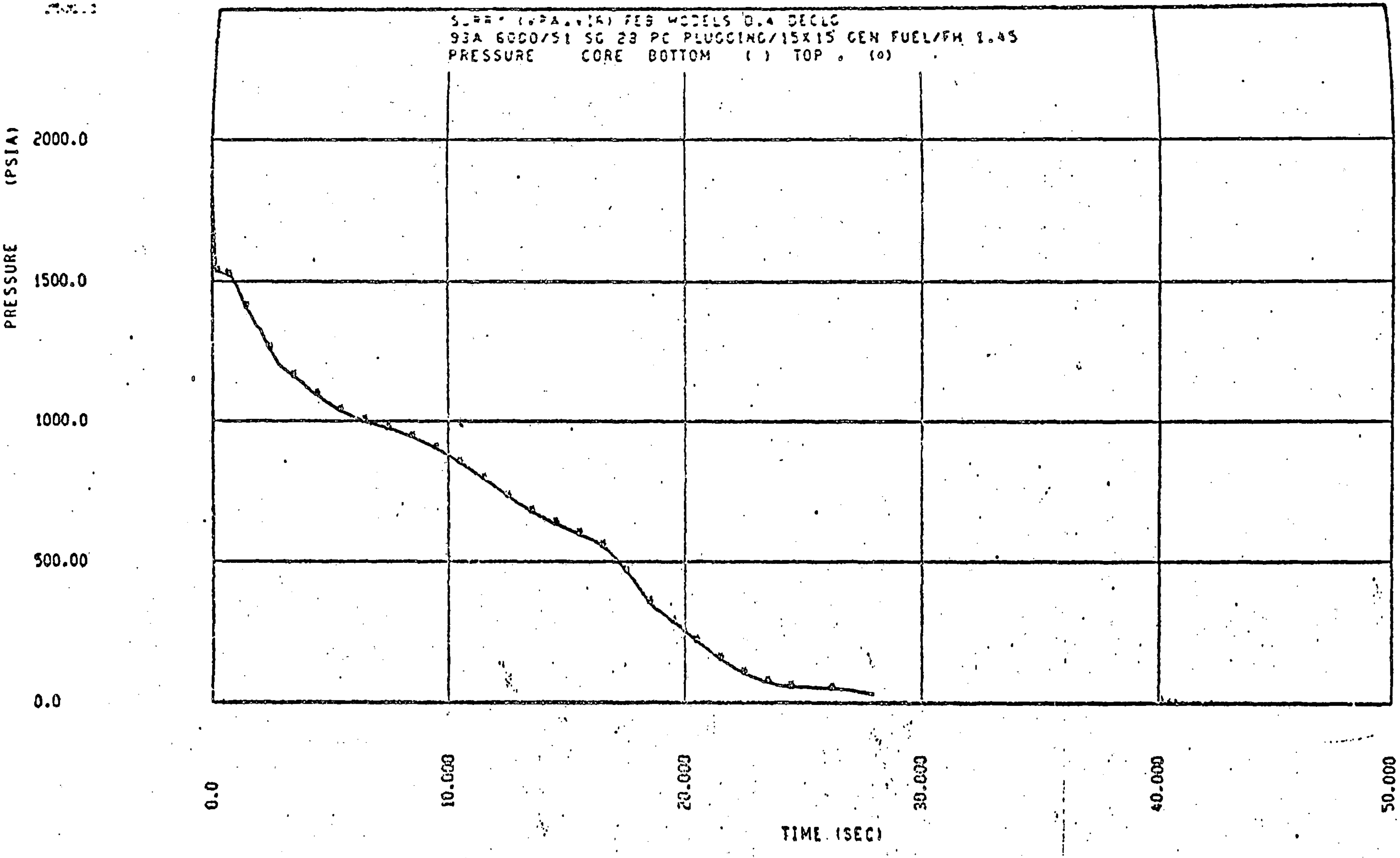


Figure 4 Core Pressure - DECLG (C_D = 0.4)

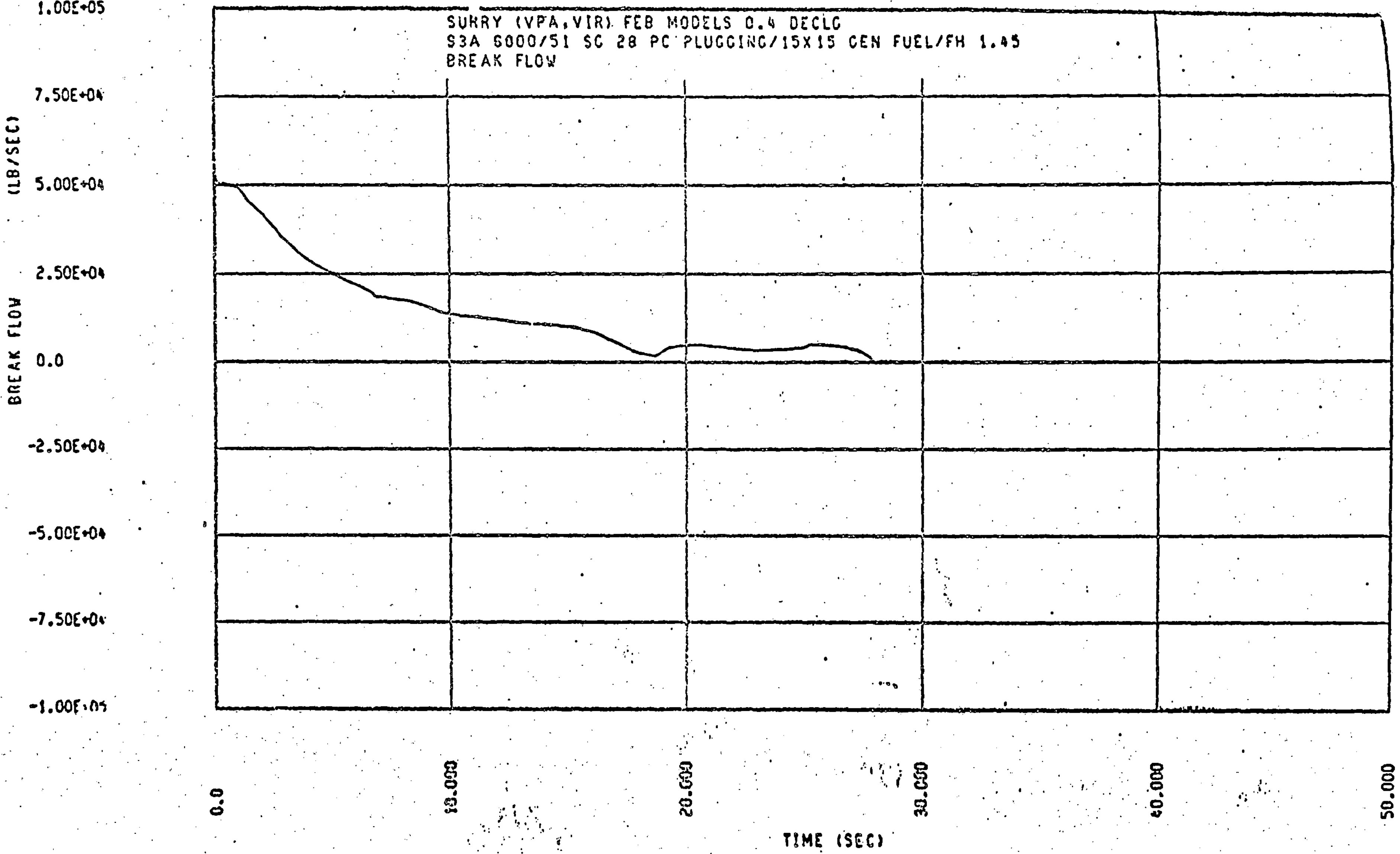


Figure 5 Break Flow Rate - DECLG (C_D = 0.4)

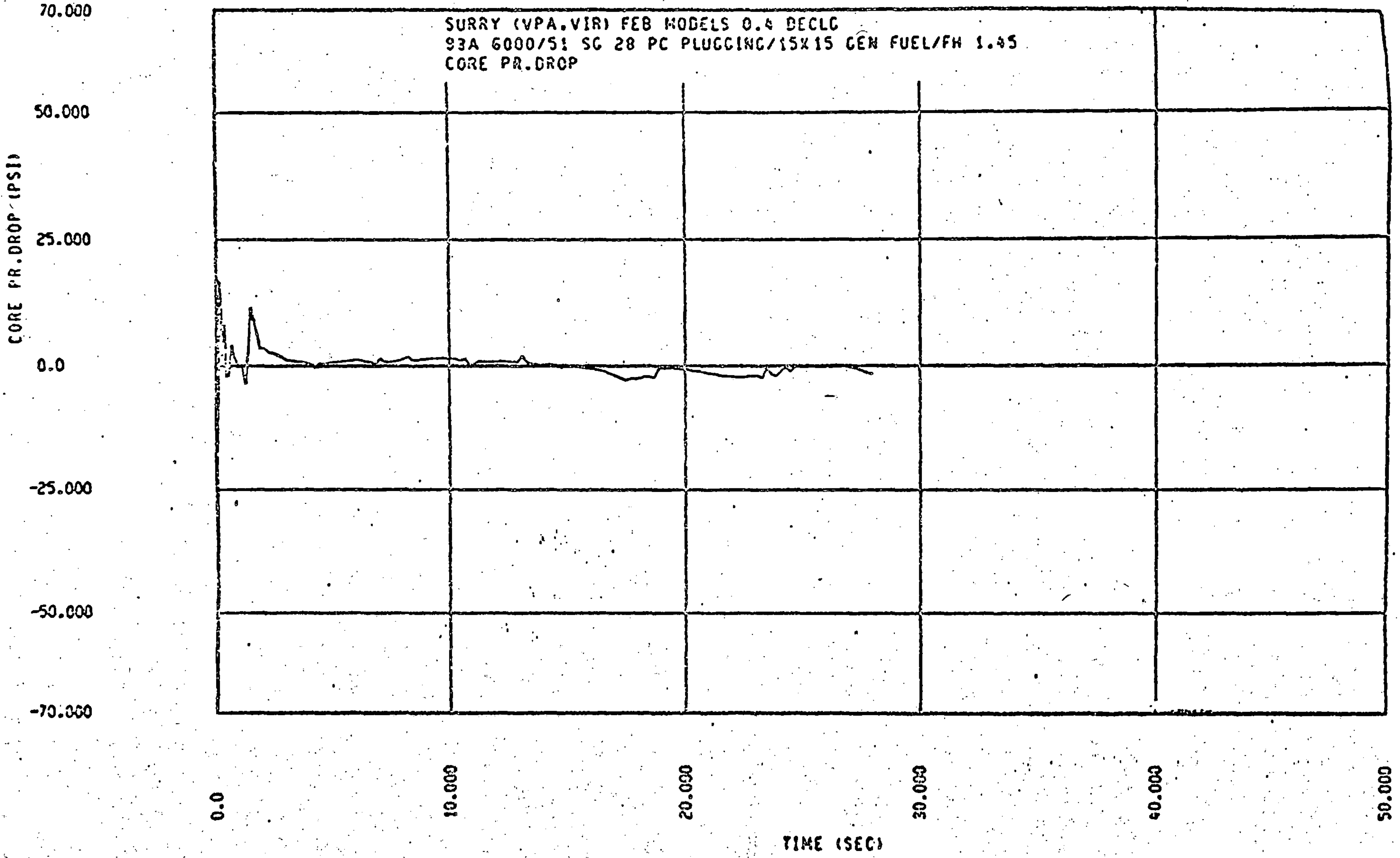


Figure 6 Core Pressure Drop - DECLG (C_D = 0.4)

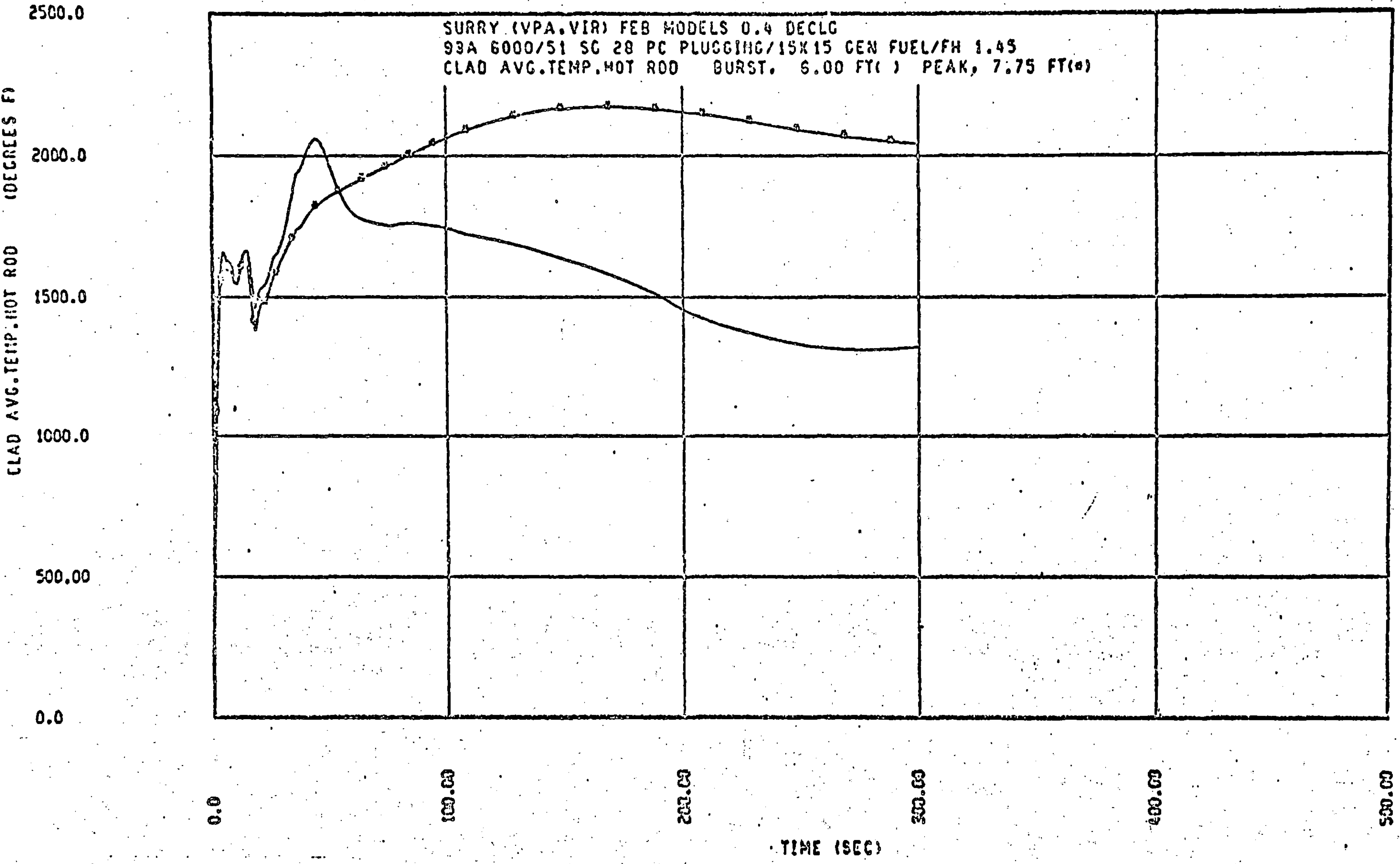


Figure 7 Peak Clad Temperature - DECLG ($C_D = 0.4$)

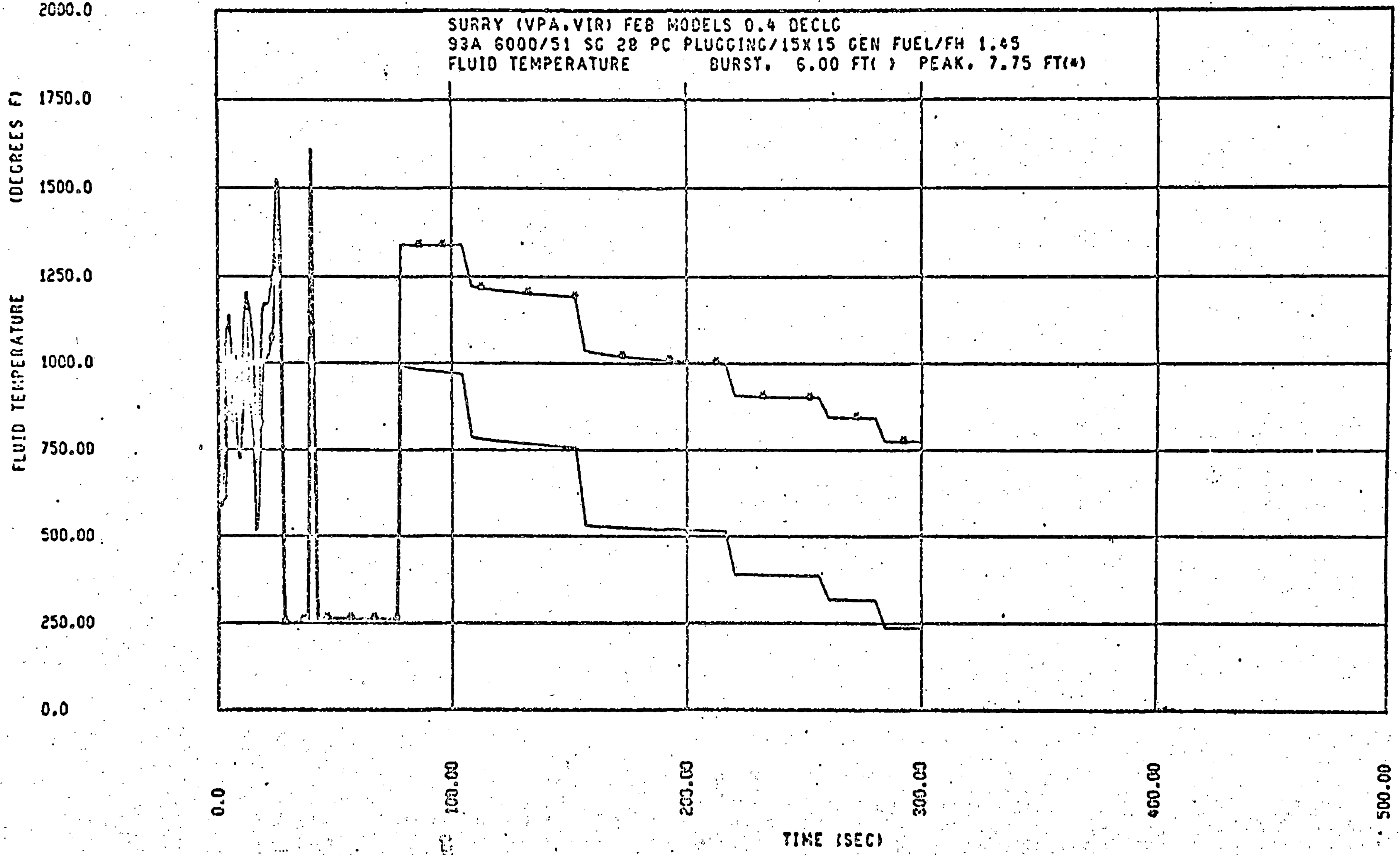


Figure 8 Fluid Temperature - DECLG (C_D = 0.4)

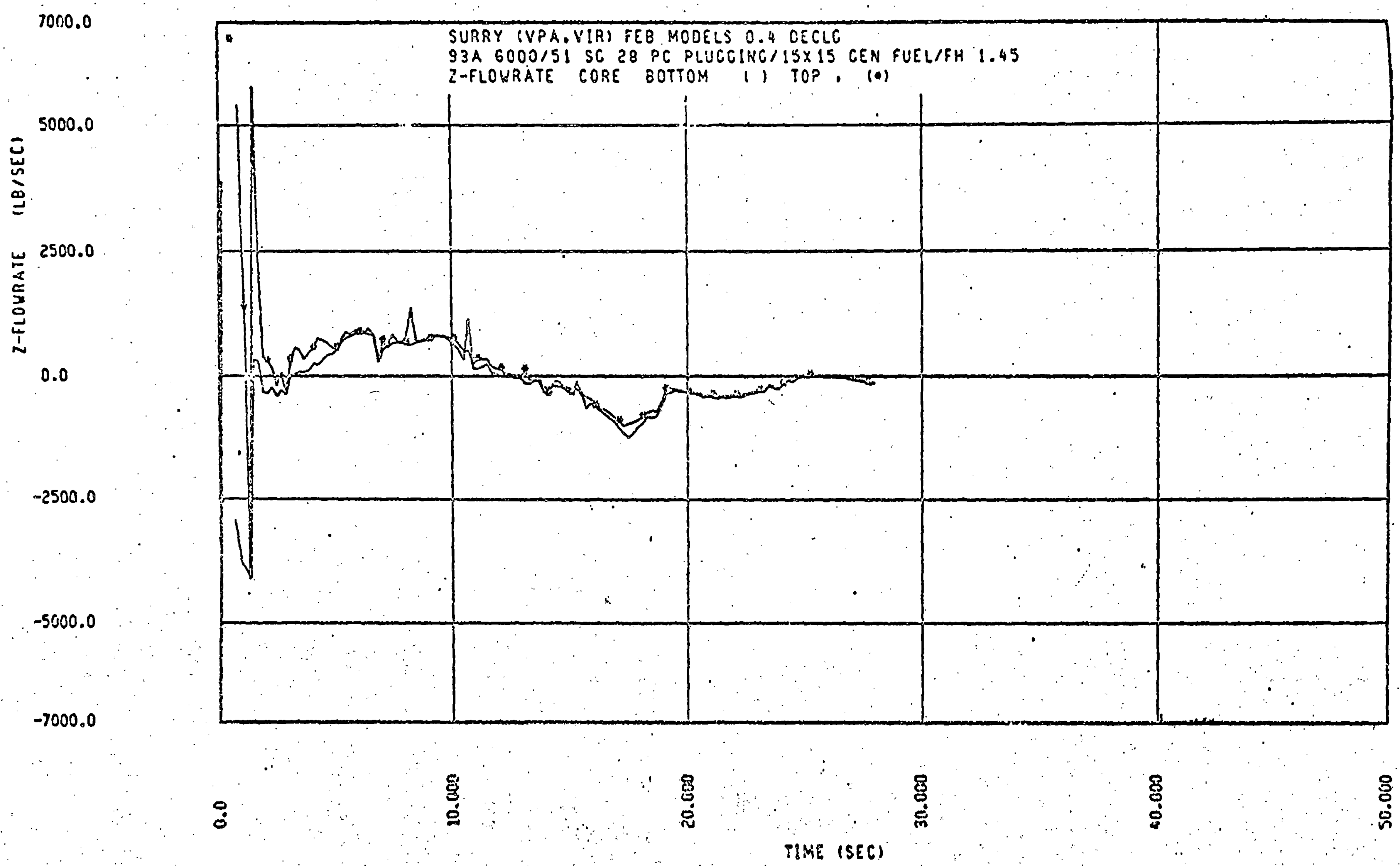


Figure 9 Core Flow-Top and Bottom - DECLG ($C_D = 0.4$)

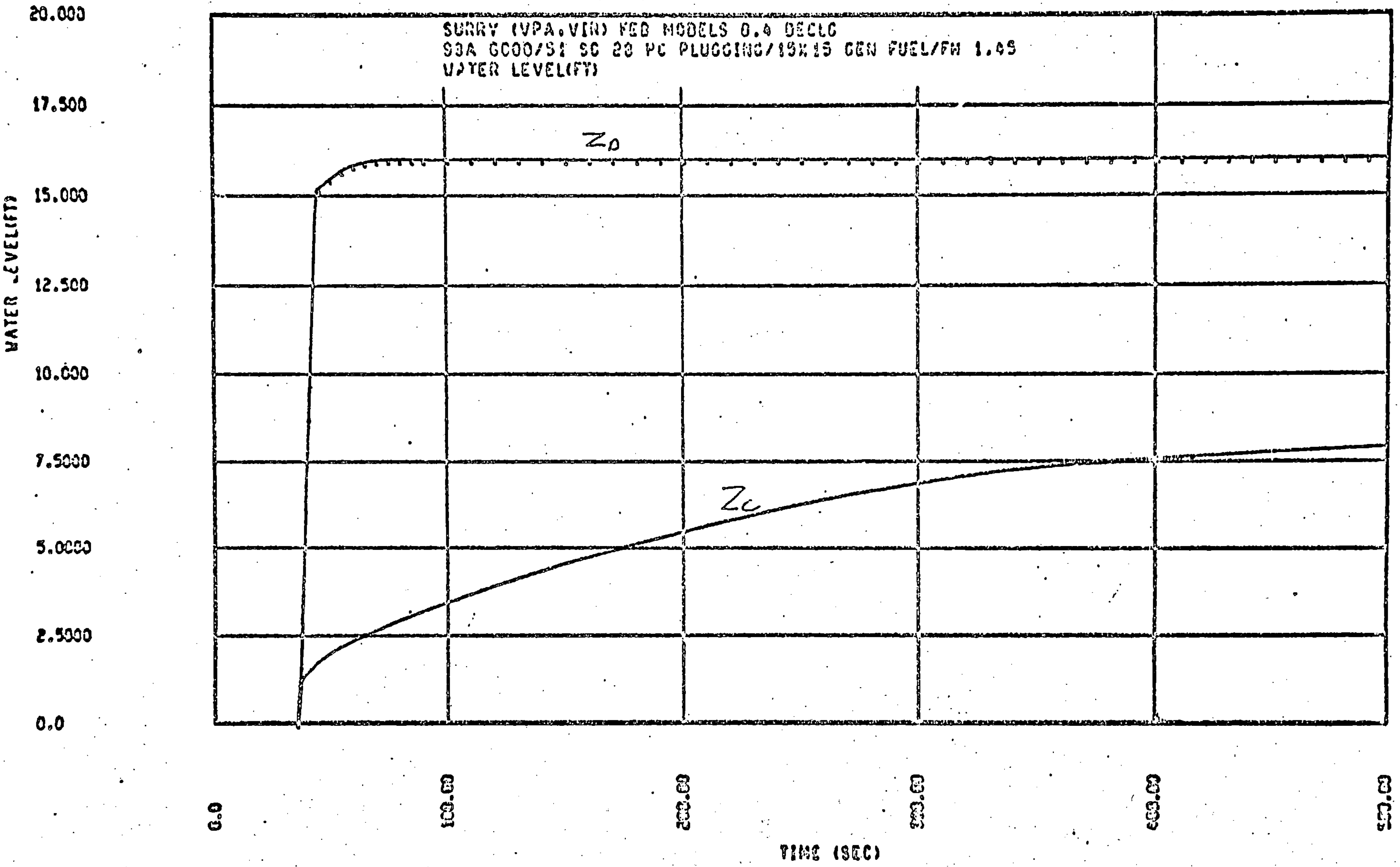


Figure 10 Reflood Transient - DECLG (C_D = 0.4)

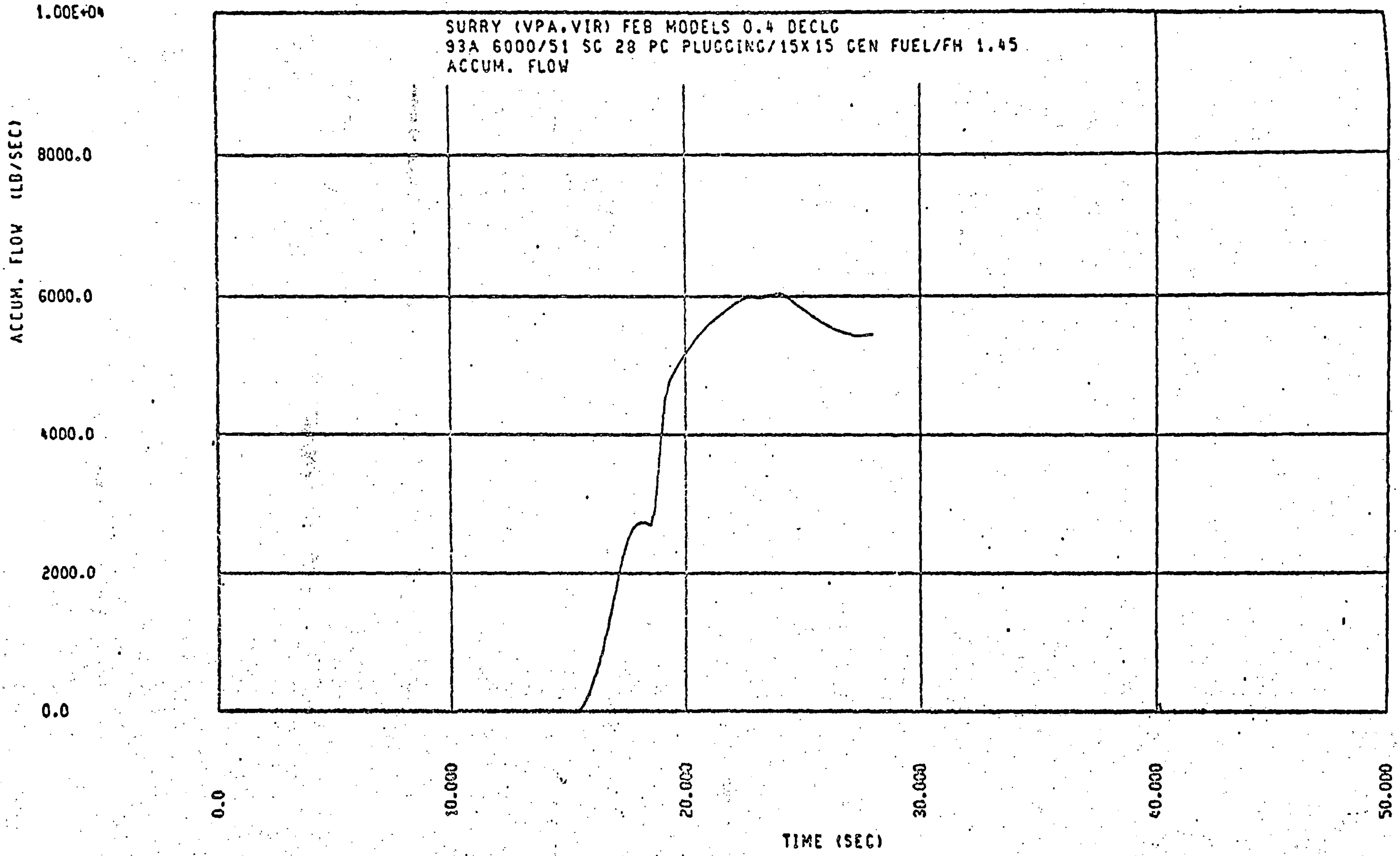


Figure 11 Accumulator Flow (Blowdown) - DECLG ($C_D = 0.4$)

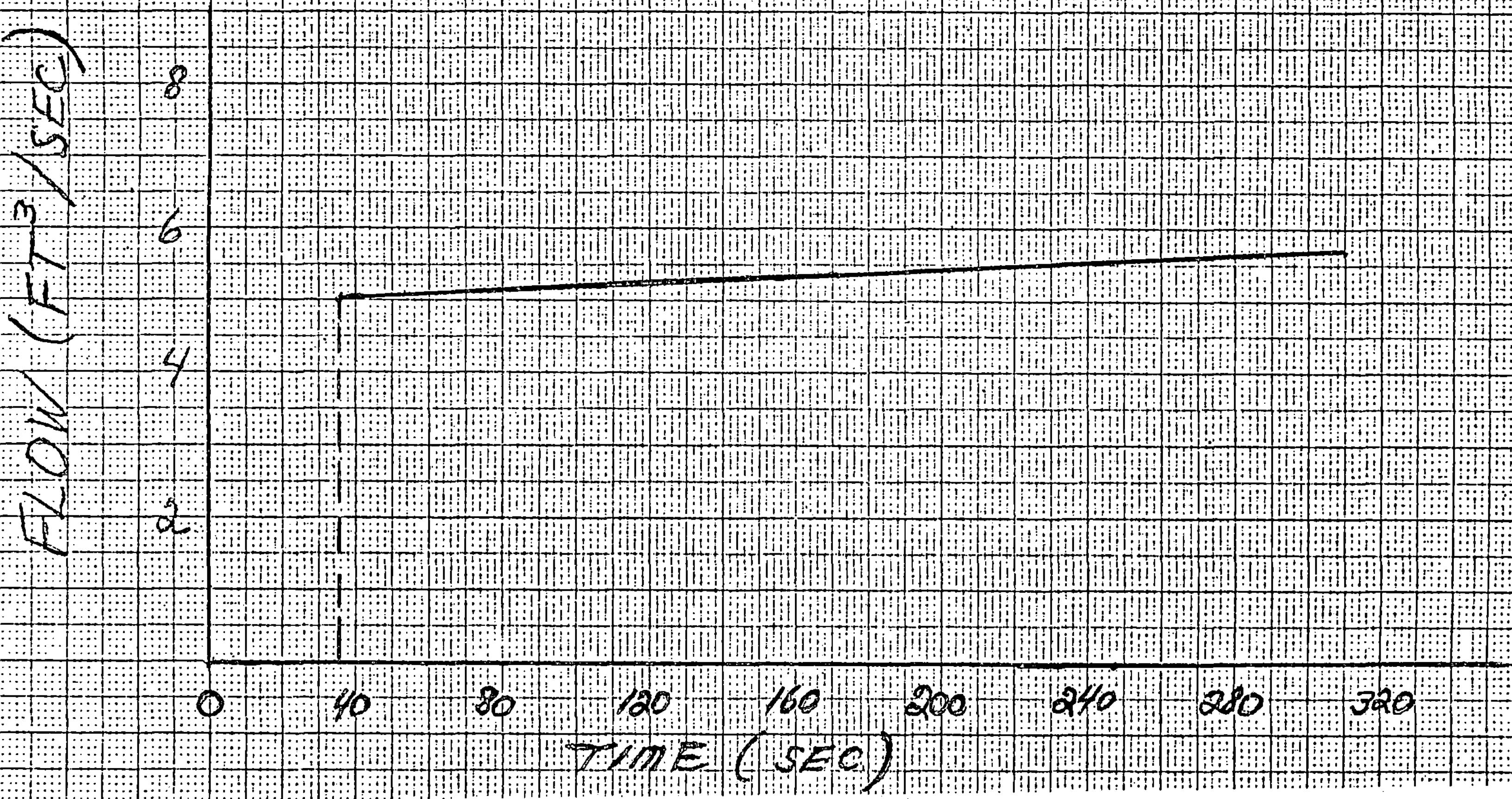


Figure 12 Pumped ECCS Flow (Reflood) - DECLG (C_D = 0.4)

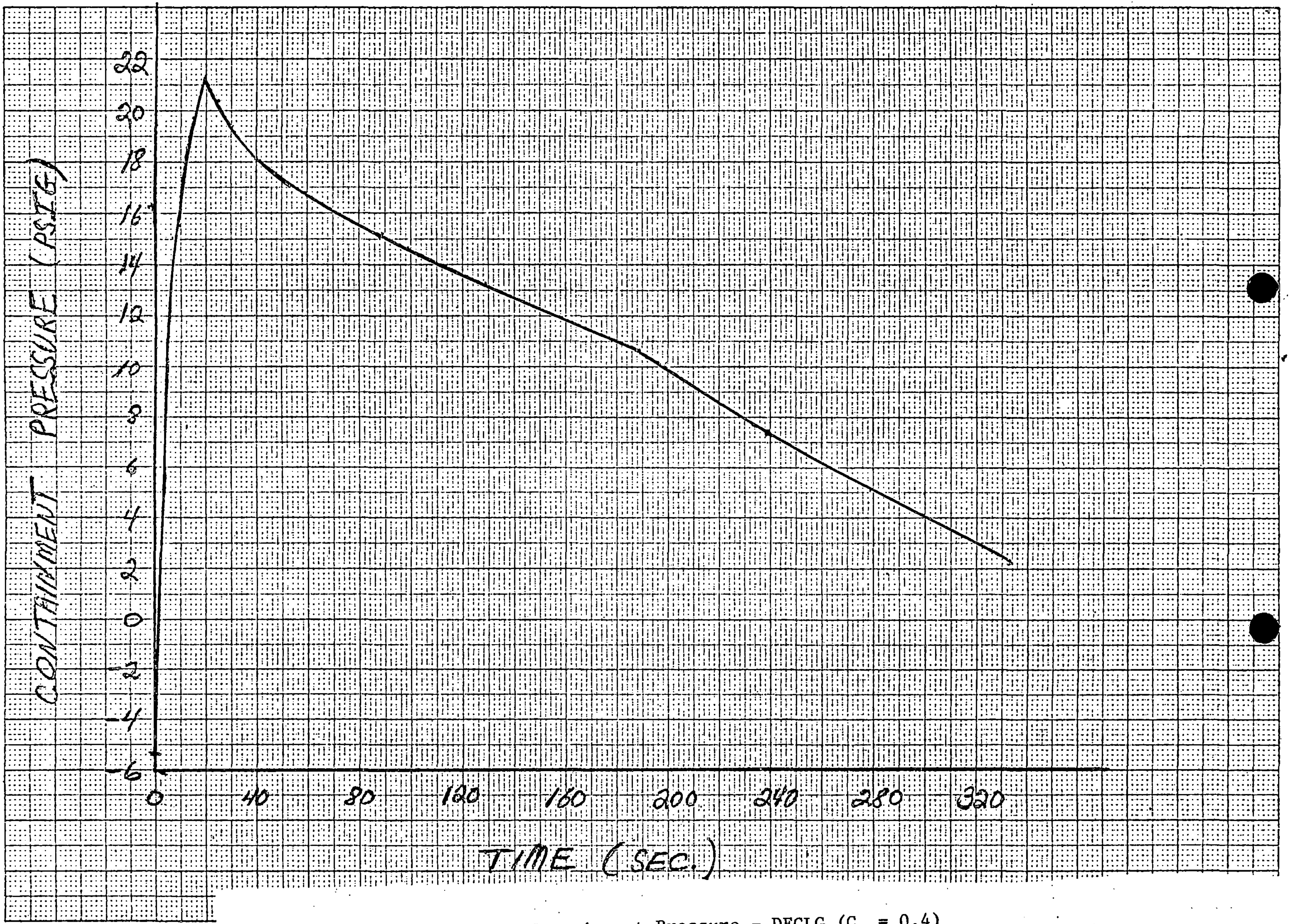


Figure 13 Containment Pressure - DECLG (C_D = 0.4)

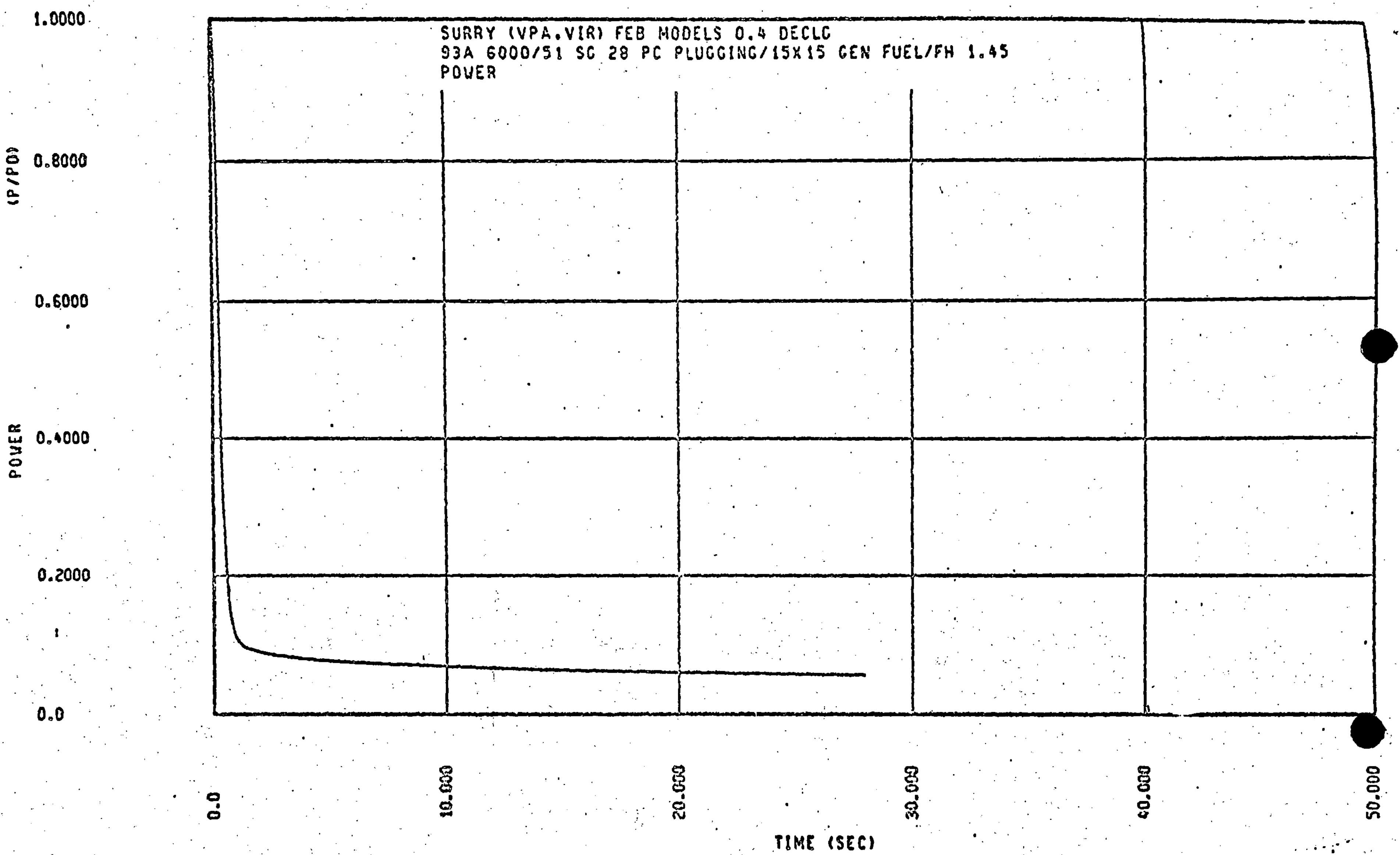


Figure 14 Core Power Transient - DECLG (C_D = 0.4)

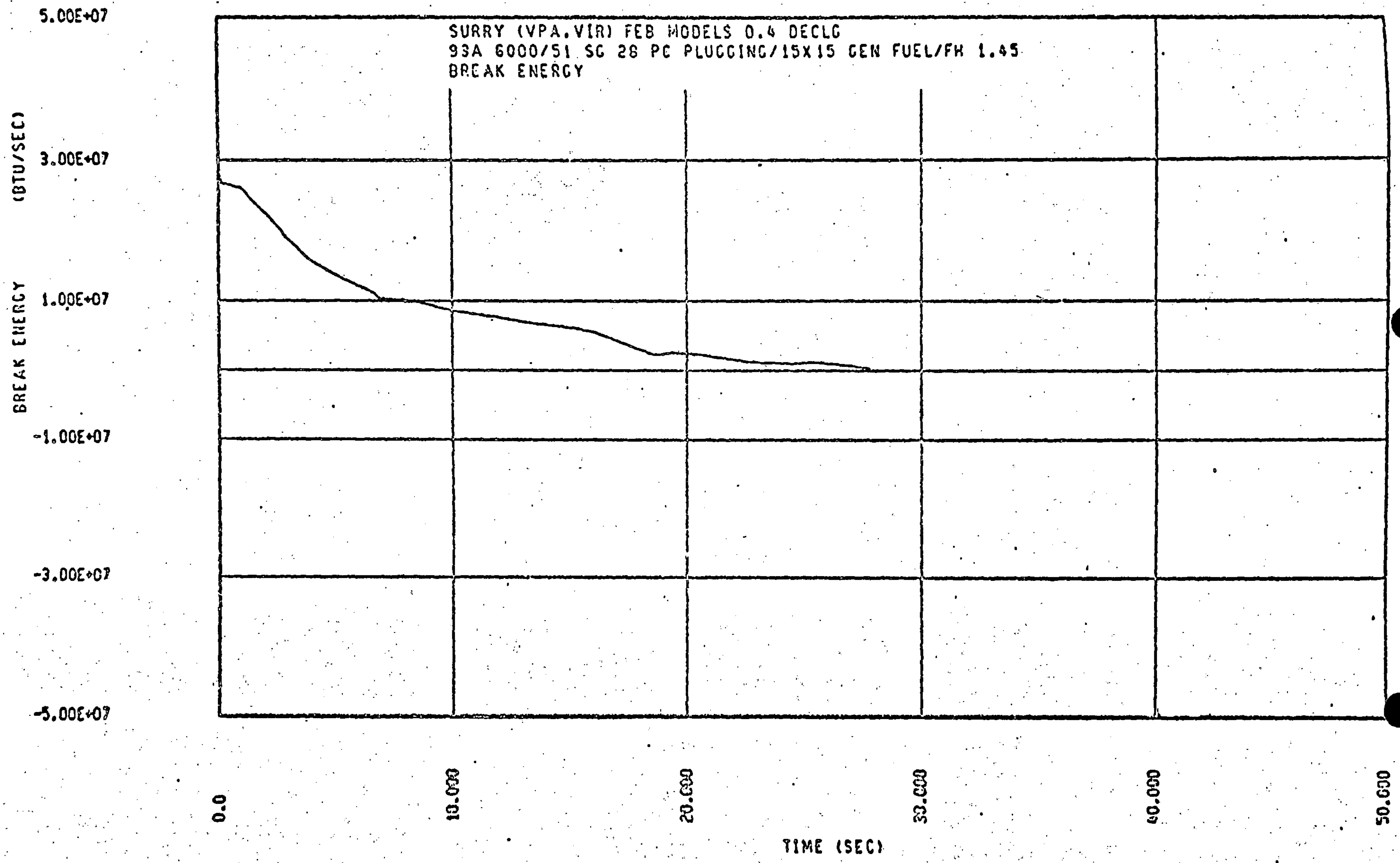


Figure 15 Break Energy Released to Containment - DECLG ($C_D = 0.4$)

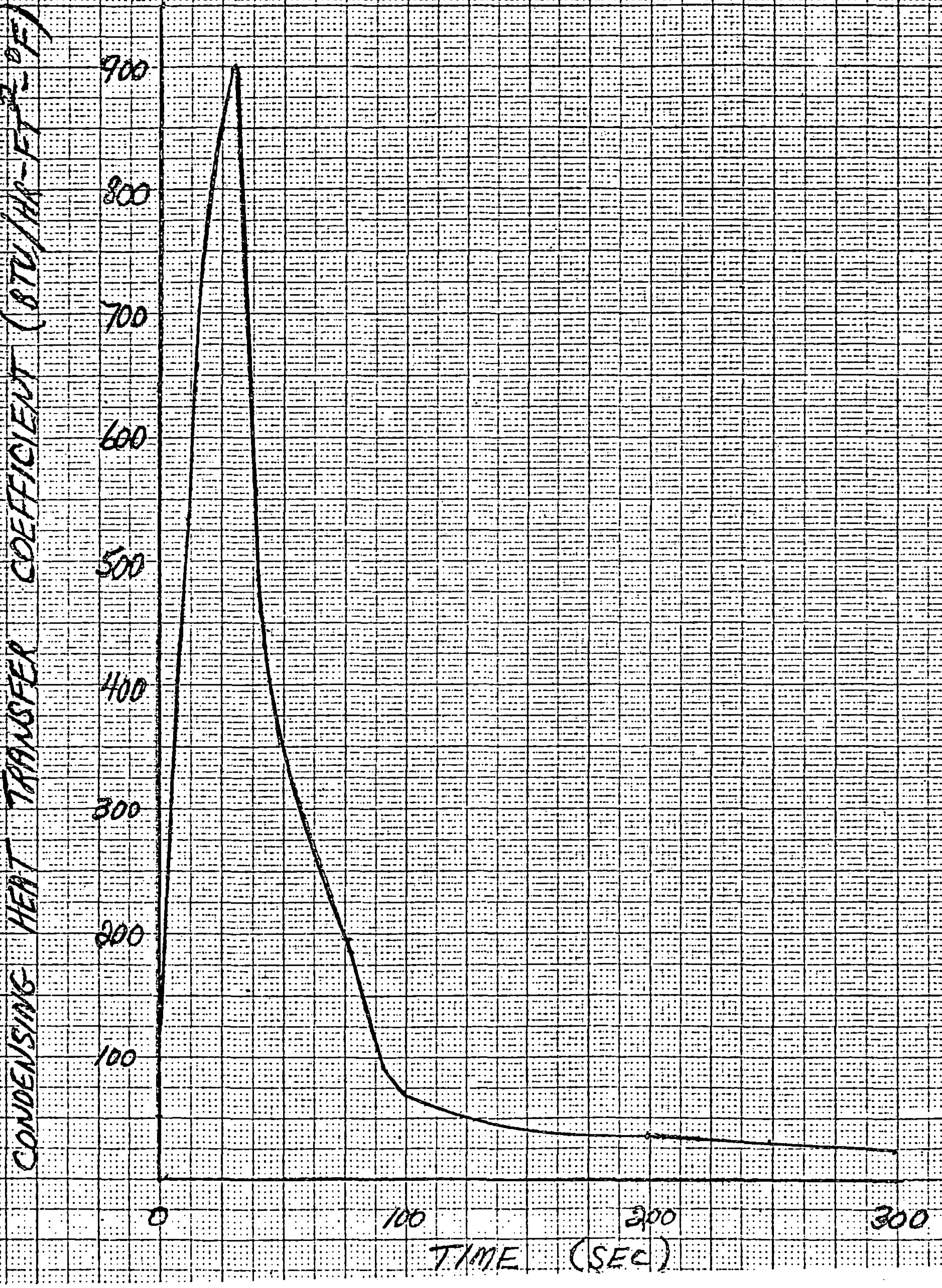


Figure 16 Containment Wall Heat Transfer Coefficient - DECLG (C_D = 0.4)

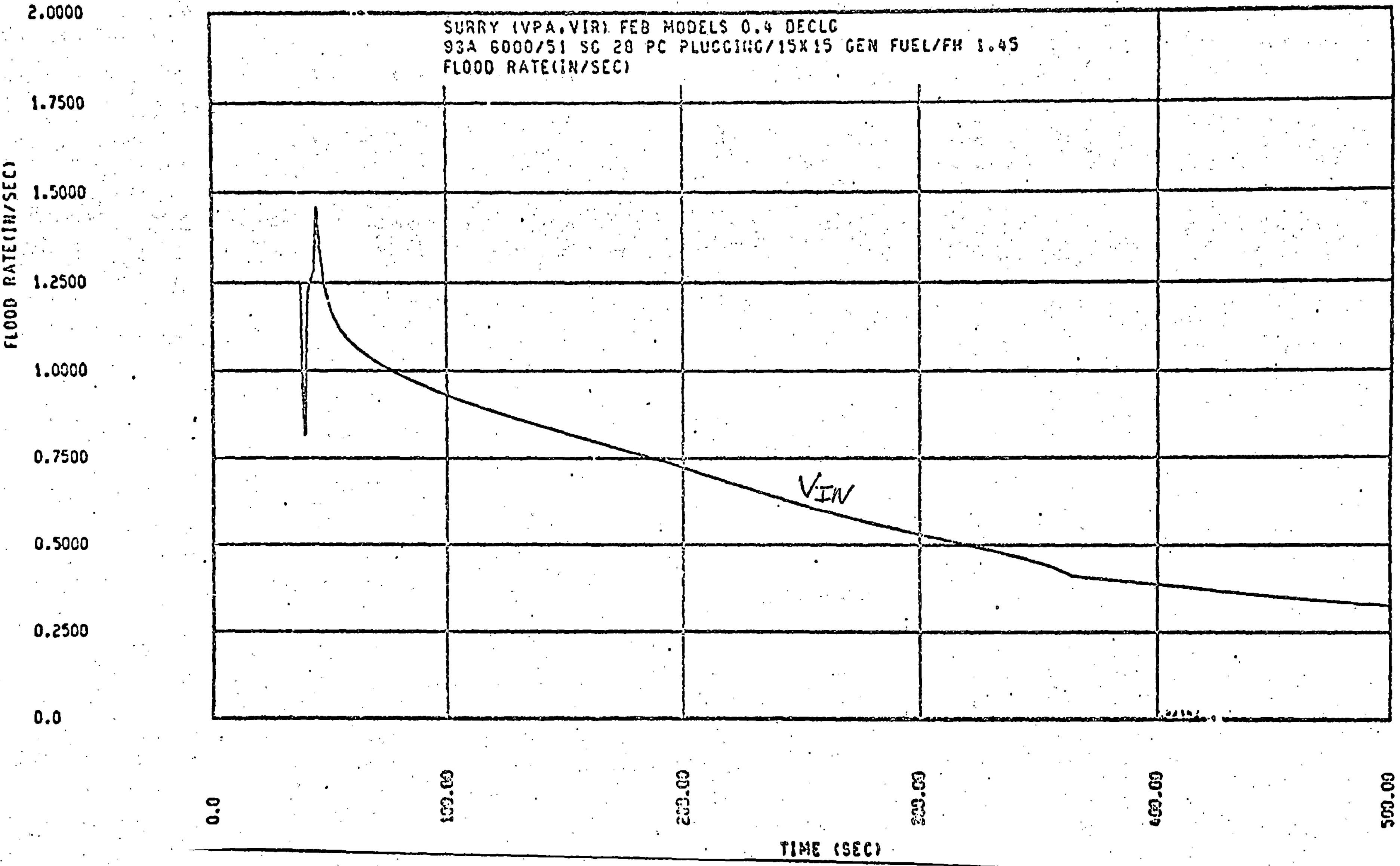


Figure 17 Reflood Transient - DECLG (C_D = 0.4)

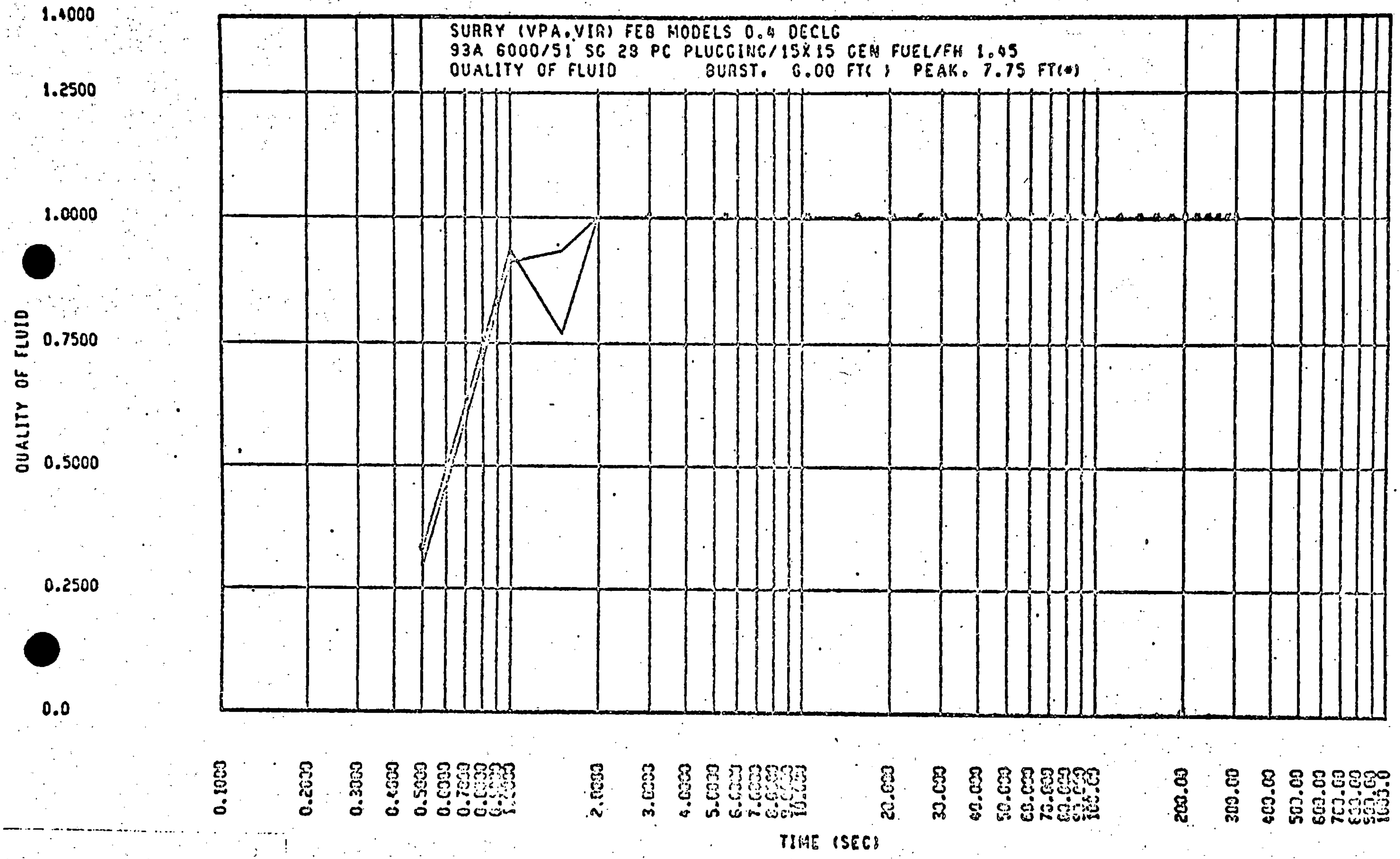


Figure 18 Fluid Quality - DECLG ($C_D = 0.4$)

ATTACHMENT 2

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated cooling water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless the following conditions are met:

1. The refueling water tank contains not less than 350,000 gal. of borated water with a boron concentration of at least 2000 ppm.

2. Each accumulator system is pressurized to at least 600 psia and contains a minimum of 975 ft^3 and a maximum of 989 ft^3 of borated water with a boron concentration of at least 1950 ppm.

3. The boron injection tank and isolated portion of the inlet and outlet piping contains no less than 900 gallons of water with a boron concentration equivalent to at least 11.5% to 13% weight boric acid solution at a temperature of at least 145°F. Additionally, recirculation between a unit's Boron Injection Tank and the Boric Acid Tank(s) assigned to the unit shall be maintained.

$F_Q(Z) \leq 2.05/P \times K(Z)$ for $P > .5$

$F_Q(Z) \leq 4.10 \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1-P)) \times T(BU)$

$F^N_{\Delta H}\Big|^{\text{LOCA}}_{\text{Assm.}} \leq 1.38/P$

$F^N_{\Delta H}\Big|^{\text{LOCA}}_{\text{Rod}} \leq 1.45/P$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, Z is the core height location of F_Q, and T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9.

2. Prior to exceeding 75% power following each core loading, and during each effective full power month of operation thereafter, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

   a. The measurement of total peaking factor, $F_Q^{\text{Meas.}}$, shall be increased by eight percent to account for manufacturing tolerances, measurement error, and the effects of rod bow. The measurement of enthalpy rise hot channel factor, the hot assembly enthalpy rise factor, $F^N_{\Delta H}\Big|^{\text{LOCA}}_{\text{Assm.}}$, and the hot rod enthalpy rise factor, $F^N_{\Delta H}\Big|^{\text{LOCA}}_{\text{Rod}}$, shall be increased by four percent to account for measurement error. If any measured hot channel factor exceeds its limit specified under 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits $F_Q \leq 2.05 \times K(Z)$, $F^N_{\Delta H} \leq 1.55 \times T(BU)$, $F^N_{\Delta H}\Big|^{\text{LOCA}}_{\text{Rod}} \leq 1.45$, and $F^N_{\Delta H}\Big|^{\text{LOCA}}_{\text{Assm.}} \leq 1.38$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

Deleted

b. $F_Q(Z)$ shall be evaluated for normal (Condition I) operation of Unit 2 by combining the measured values of $F_{xy}(Z)$ with the design Condition I axial peaking factor values, $F_Z(Z)$, as listed in TS Table 3.12-1B. For the purpose of this specification $F_{xy}(Z)$ shall be determined between 1.5 feet and 10.5 feet elevations of the core exclusive of grid plane regions located at 25.9 ±3.2 inches, 52.1 ±3.2 inches, 78.3 ±3.2 inches, and 104.5 ±3.2 inches. The measured values of $F_{xy}(Z)$ shall be increased by nine percent to account for manufacturing tolerances, measurement error, rod bow, xenon redistribution, and any burnup dependent peaking factor increases. If the results of this evaluation predict that $F_Q(Z)$ could potentially violate its limiting values as established in Specification 3.12.B.1, either:

(1) the thermal power and high neutron flux trip setpoint shall be reduced at least 1% for each 1% of the potential violation (for the purpose of this specification, this power level shall be called $P_{THRESHOLD}$), or

(2) movable detector surveillance shall be required for operation when the reactor thermal power exceeds $P_{THRESHOLD}$. This surveillance shall be performed in accordance with the following:

(a) The normalized power distribution, $F_Q(Z)\Big|^{j}_{APDM}$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux system for

3. The reference equilibrium indicated axial flux difference (called the target flux difference) at a given power level P_o, is that indicated axial flux difference with the core in equilibrium xenon conditions (small or no oscillation) and the control rods more than 190 steps withdrawn. The target flux difference at any other power level, P, is equal to the target value of P multiplied by the ratio, P/P_o. The target flux difference shall be measured at least once per equivalent full power quarter. The target flux difference must be updated during each effective full power month of operation either by actual measurement, or by linear interpolation using the most recent value and the value predicted for the end of the cycle life.

4. Except as modified by 3.12.B.4.a, b, c, or d below, the indicated axial flux difference shall be maintained within a ±5% band about the target flux difference (defines the target band on axial flux difference).

   a. At a power level greater than 88 percent of rated power, if the indicated axial flux difference deviates from its target band, within 15 minutes either restore the indicated axial flux difference to within the target band, or reduce the reactor power to less than 88 percent of rated power.

   b. At a power level no greater than 88 percent of rated power,

      (1) The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference is within the limits shown on Figure 3.12-10.

One minute penalty is accumulated for each one minute of operation outside of the target band at power levels equal to or above 50% of rated power.

(2) If 3.12.B.4.b(1) is violated, then the reactor power shall be reduced to less than 50% power within 30 minutes and the high neutron flux setpoint shall be reduced to no greater than 55% power within the next four hours.

(3) A power increase to a level greater than 88 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

(4) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Table 4.1-1 provided the indicated AFD is maintained within the limits of Figure 3.12-10. A total of 16 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

c. At a power level no greater than 50 percent of rated power,

(1) The indicated axial flux difference may deviate from its target band.

(2) A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than one hour accumulated penalty during the preceding 24-hour period. One half minute penalty is accumulated for each one minute of operation outside of the target band at power levels between 15% and 50% of rated power.

d. The axial flux difference limits of Specifications 3.12.B.4.a, b, and c may be suspended during the performance of physics tests provided:

(1) The power level is maintained at or below 85% of rated power, and

(2) The limits of Specification 3.12.B.1 are maintained.

The power level shall be determined to be $\leq$ 85% of rated power at least once per hour during physics tests. Verification that the limits of Specification 3.12.B.1 are being met shall be demonstrated through in-core flux mapping at least once per 12 hours.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in 3.12.B.4.a and the flux difference time limits in 3.12.B.4.b and c. If the alarms are out of service temporarily, the axial flux difference shall be logged, and conformance to the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

The indicated axial flux difference for each excore channel shall be monitored at least once per 7 days when the alarm is operable and at least once per hour for the first 24 hours after restoring the alarm to operable status.

5. The allowable quadrant to average power tilt is 2.0%.

6. If, except for physics and rod exercise testing, the quadrant to average power tilt exceeds 2%, then:
    a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the specification of 3.12.B.1, or
    b. If the hot channel factors are not determined within two hours, the power level and high neutron flux trip setpoint shall be reduced from rated power, 2% for each percent of quadrant tilt.
    c. If the quadrant to average power tilt exceeds ±10%, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power for both LOCA and non-LOCA considerations.

$F_{\Delta H}^N\Big|_{\text{Assm.}}^{\text{LOCA}}$, Hot Assembly Nuclear Enthalpy Rise Factor, is defined as the ratio of the integral of linear power along the assembly with the highest integrated power to the average assembly power.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and LOCA calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46 using an upper bound envelope of 2.05 times the hot channel factor normalized operating envelope given by TS Figure 3.12-8.

DELETED

For normal (Condition I) operation of Unit 2, it may be necessary to perform surveillance to insure that the heat flux hot channel factor, $F_Q(Z)$, limit is met. To determine whether and at what power level surveillance is required, the potential (Condition I) values of $F_Q(Z)$ shall be evaluated monthly by combining the measured values of $F_{xy}(Z)$ obtained from the analysis of the monthly incore flux map with the values of the design Condition I axial peaking factors, $F_Z(Z)$. The product of these shall be increased by nine percent to account for measurement uncertainty, manufacturing tolerances, rod bow, radial redistribution of xenon during normal (Condition I) operation, and any burnup dependent peaking factor increases. $P_{THRESHOLD}$ is defined as the value of rated power minus one percent power for each percent of potential $F_Q(Z)$ violation. If the potential values of $F_Q(Z)$ for normal (Condition I) operation are greater than the $F_Q(Z)$ limit, then surveillance shall be performed at all power levels above $P_{THRESHOLD}$.

Movable incore instrumentation thimbles for surveillance are selected so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor $F_Q(Z)$ can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor, $\overline{R}$, can be determined as the ratio of the total peaking 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 always possible during certain physics tests or during excore detector calibrations. Therefore, the specifications on power distribution control are less restrictive during physics tests and excore detector calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 88% power, in the range ±13.5 percent (±10.5 percent indicated) where for every 2 percent below rated power, the permissible flux difference boundary is extended by 1 percent.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition

DELETED

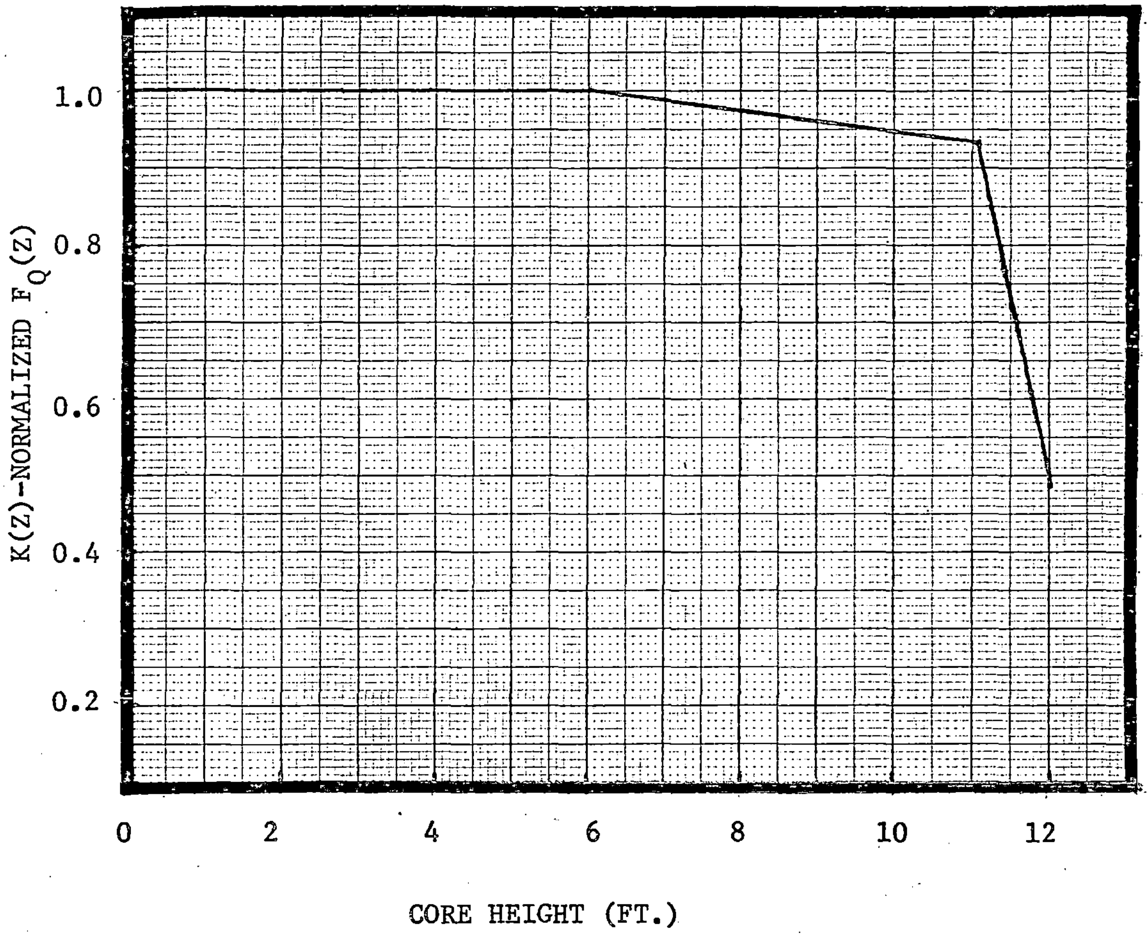
DELETED

HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

SURRY POWER STATION

UNIT NOS. 1 AND 2



TS FIGURE 3.12-10

AXIAL FLUX DIFFERENCE LIMITS

AS A FUNCTION OF RATED POWER

SURRY POWER STATION