EMERGENCY CORE COOLING SYSTEM ANALYSIS
REQUIRED BY 10 CFR 50.46

SEPTEMBER 1, 1974
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SURRY POWER STATION
UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281
LICENSE NOS. DPR-32 AND DPR-37

VIRGINIA ELECTR... Y

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I. INTRODUCTION

The re-evaluation of ECCS cooling performance as required by the Order for Modification of License for Surry Power Station, Unit Nos. 1 and 2, issued by the Atomic Energy Commission on December 27, 1974, and as specified by 10 CFR 50.46,[1] "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors" is presented herein. This analysis was performed with the March 15, 1975 version of the Westinghouse evaluation model which includes modifications as specified by the Regulatory Staff in Reference 8. The implementation of these modifications in the evaluation model is described in Reference 13. The analytical techniques utilized in the analysis are in compliance with Appendix K to 10 CFR 50, and are described in the topical report, "Westinghouse ECCS Evaluation Model-Summary," WCAP 8339,[2] dated July 1974. The Nuclear Regulatory Commission Staff acceptance of the "Westinghouse ECCS Evaluation Model" is presented in Reference 14. The results of the analysis show that the calculated cooling performance of the emergency core cooling system (ECCS) following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph b of 10 CFR 50.46.

A loss-of-coolant accident may result from a rupture of the reactor coolant system or of any line connected to that system up to the first closed valve. Ruptures of very small cross section will cause explusion of coolant of a rate which can be accommodated by the charging pumps. Should such a small rupture occur, these pumps would maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown. A moderate quantity of coolant containing

such radioactive impurities as would normally be present in the coolant, would be released to the containment.

Should a larger break occur, subcooled fluid is expelled from the break rapidly reducing the pressure to saturation. Depressurization of the reactor coolant system causes fluid to flow to the reactor coolant system from the pressurizer, resulting in a pressure and level decrease in the pressurizer. For a postulated large break, reactor trip is initiated when the pressurizer low pressure setpoint is reached, while the safety injection system (SIS) is actuated by coincident pressurizer low pressure and low level. A high containment pressure signal serves as a backup by initiating a safety injection signal which in turn trips the reactor. The consequences of an accident are limited two ways:

1. Reactor trip and borated water injection supplement void formation in causing rapid reduction of the nuclear power to a residual level corresponding to delayed fissions and fission product decay.
2. Injection of borated water also ensures sufficient flooding of the core to prevent excessive clad temperatures.

The safety injection system, even when operating on emergency power, limits the cladding temperature to below the melting temperature of Zircaloy-4 and below the temperature at which gross core geometry distortion, including clad fragmentation, may be expected. In addition, the total core metal-water reaction is limited to less than one (1) per cent. This is valid for reactor coolant piping ruptures up to an including the double ended rupture of a reactor coolant loop. Consequences of these ruptures are well within those for the hypothetical

accident and are, therefore, well within the limits of 10 CFR 100.

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. The heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, secondary system 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. The safety injection signal stops normal feedwater flow by closing the main feedwater control valves, trips the main feedwater pumps and initiates emergency feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of reactor coolant system pressure. When the reactor coolant system depressurizes to 600 psia, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initialization of the accident and effects of pump coastdown are included in the blowdown analyses.

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. After the contents of the refueling water storage is emptied, 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 to subatmospheric pressure.

II. LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES WHICH ACTUATE THE EMERGENCY CORE COOLING SYSTEM

Methods of Analysis

For breaks less than 1 ft^2 the WFLASH[10] digital computer code is employed to calculate the transient depressurization of the reactor coolant system, as well as to describe the mass and enthalpy of flow through the break.

The WFLASH program used in the analysis of the small break loss of coolant accident is an extension of the FLASH-4 code developed at the Westinghouse Bettis Atomic Power Laboratory. The WFLASH program permits a detailed spatial representation of the reactor coolant system.

The reactor coolant system is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into the second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied throughout the system.

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume with the associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables an explicit and detailed spatial representation of various system components. In particular it enables a proper calculation of the behavior of the loop seal during a loss of coolant transient.

Safety injection flow rate to the reactor coolant system as a function of the system pressure is used as part of the input. The

safety injection system was assumed to be delivering borated water to the reactor coolant system 25 seconds after the generation of a safety injection signal.

For these analyses, the safety injection delivery considers pumped injection flow which is depicted in Figure II-1 as a function of reactor coolant system pressure. This figure represents injection flow from the safety injection pumps based on performance curves degraded five (5) per cent from the design head. The twenty-five (25) seconds delay includes time required for the emergency diesel generator to assume its load. The effect of low head safety injection pump flow is not considered since their shutoff head is lower than reactor coolant system pressure during the time portion of the transient considered. Minimum safeguards emergency core cooling system capability and operability has been assumed in these analyses.

Peak clad temperature analyses are performed with the LOCTA-IV[4] code which determines the reactor coolant system pressure, fuel rod power history, steam flow past the uncovered part of the core and mixture height history.

Results

The results of the limiting break size are given in Table II-2. The worst break size (small break) is a 4 inch diameter break. The depressurization transient for this break is shown in Figure II-2. The extent to which the core is uncovered is shown in Figure II-3.

During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The resultant heat transfer cools the fuel rod and clad to very near the coolant temperatures as long as the core remains covered by a two phase mixture.

The maximum hot spot clad temperature calculated during the transient is 1898 degrees Fahrenheit including the effects of fuel densification as described in Reference. 11. The peak clad temperature transient is shown in Figure II-4 for the worst break size, i.e., the break with the highest peak clad temperature. The steam flow rate for the worst break is shown on Figure II-5. When the mixture level drops below the top of the core, the steam flow computed in WFLASH provides cooling to the upper portion of the core. The rod film coefficients for this phase of the transient are given in Figure II-6. The hot spot fluid temperature for the worst break is shown in Figure II-7.

The core power (dimensionless) transient following the accident (relative to reactor scram time) is shown in Figure II-8.

The reactor shutdown time (3.4 sec) is equal to the reactor trip signal time (1.2 sec) plus 2.2 sec for rod insertion. During this rod insertion period the reactor is conservatively assumed to operate at rated power.

Additional Break Sizes

Additional break sizes were also analyzed. Figures II-9a and II-9b present the reactor coolant system pressure transient for the three (3) inch and six (6) inch breaks, respectively, and Figures II-10a and II-10b present the volume histroy (mixture height) for both breaks. The peak clad temperatures for both cases are less than the peak clad temperature of the four (4) inch break. The peak clad temperatures for both cases are given in Figures II-11a and II-11b.

The time sequence of events for all small breaks analyzed is shown in Table II-1 and Table II-2 presents the assumptions and results for these analyses.

The results of several sensitivity studies are reported in WCAP-8342.[12] These results are for conditions which are not limiting in nature and hence, are reported on a generic basis.

TABLE II-1

TIME SEQUENCE OF EVENTS FOR SMALL BREAKS

EVENT	BREAK SIZE (INCHES) 3	4	6
	TIME AFTER START OF LOCA (SECONDS)		
Start of LOCA	0.0	0.0	0.0
Reactor Trip Signal	21.2	13.4	8.0
Top of Core Uncovered	690	333	108
Accumulator Injection Begins	971	530	265
Peak Clad Temperature Occurs	992	574	276
Top of Core Covered	1011	853	315

TABLE II-2

ASSUMPTIONS AND RESULTS FOR SMALL BREAKS

	3 IN	4 IN	6 IN
Results			
Peak Clad Temp. (^{o}F)	1702	1898	1559
Peak Clad Location (Ft.)	11.0	11.0	10.5
Local Zr/H_2O Rxn (max) (%)	1.17	1.66	0.91
Local Zr/H_2O Location (Ft.)	11.0	10.5	10.5
Total Zr/H_2O Rxn (%)	<0.3	<0.3	<0.3
Hot Rod Burst Time (sec)	N/A	N/A	N/A
Hot Rod Burst Location (Ft.)	N/A	N/A	N/A

Calculation	
NSSS Power MWt 102% of	2542
Peak Linear Power Kw/ft 102% of	14.44
Peaking Factor	2.32

Fuel region analyzed (Most Limiting)	Region
Unit No. 1	3
Unit No. 2	3

FIGURE II-1

SAFETY INJECTION FLOWRATE

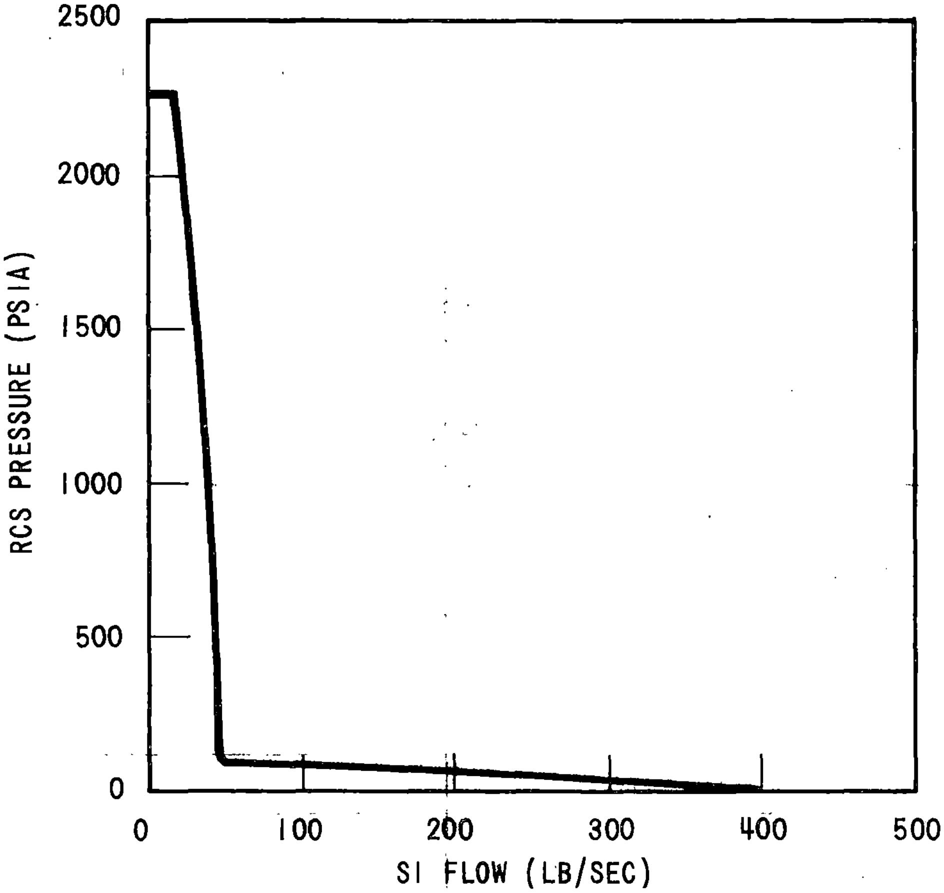


FIGURE II-2

RCS DEPRESSURIZATION TRANSIENT (4 inch)

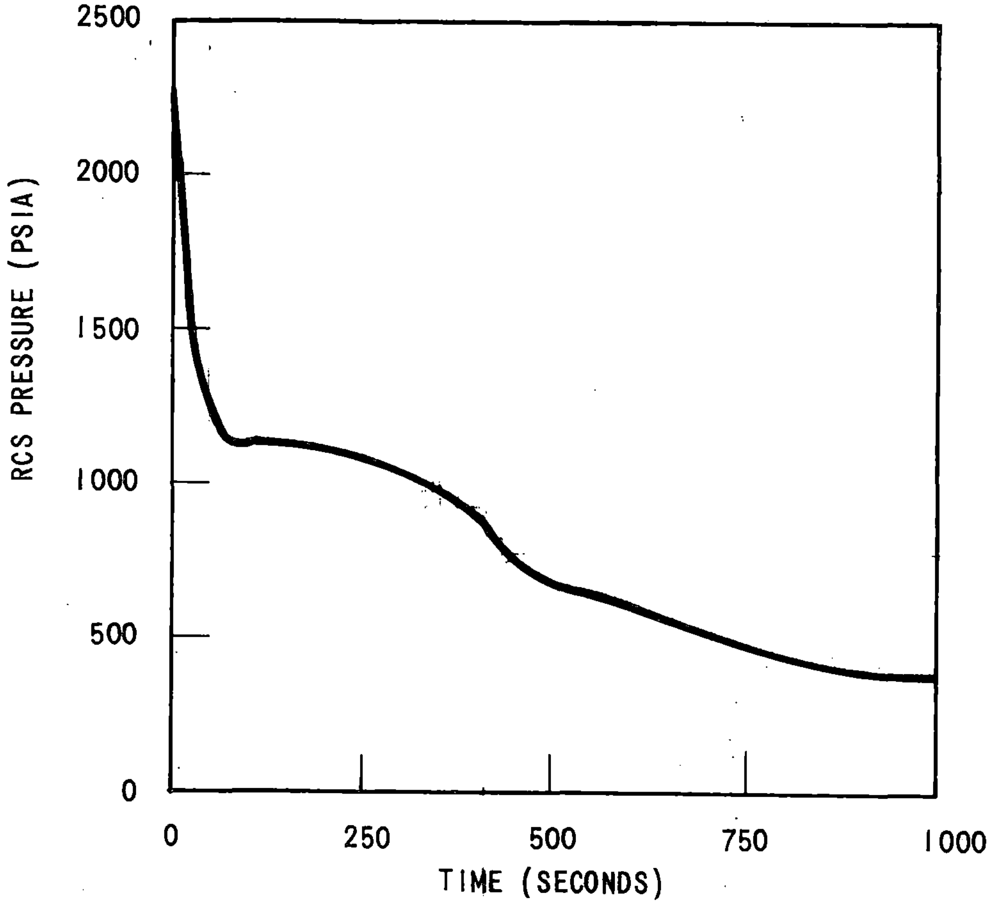


FIGURE II-3

CORE MIXTURE HEIGHT (4 inch)

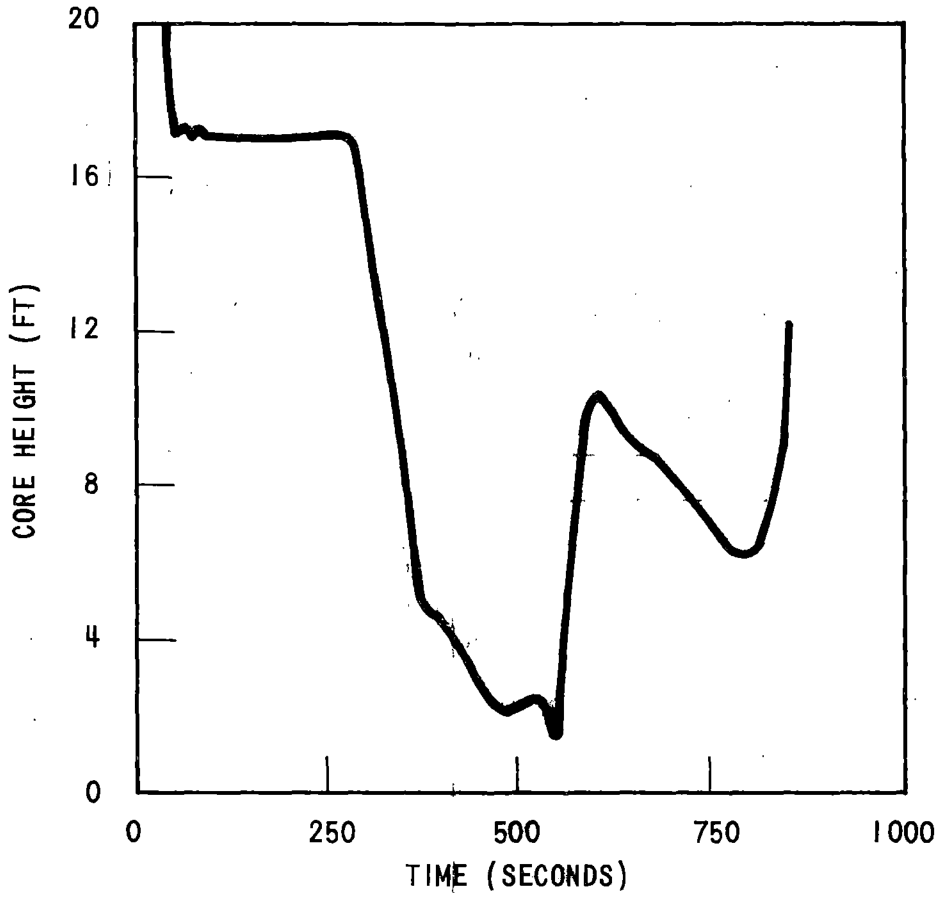


FIGURE II-4

CLAD TEMPERATURES TRANSIENT (4 inch)

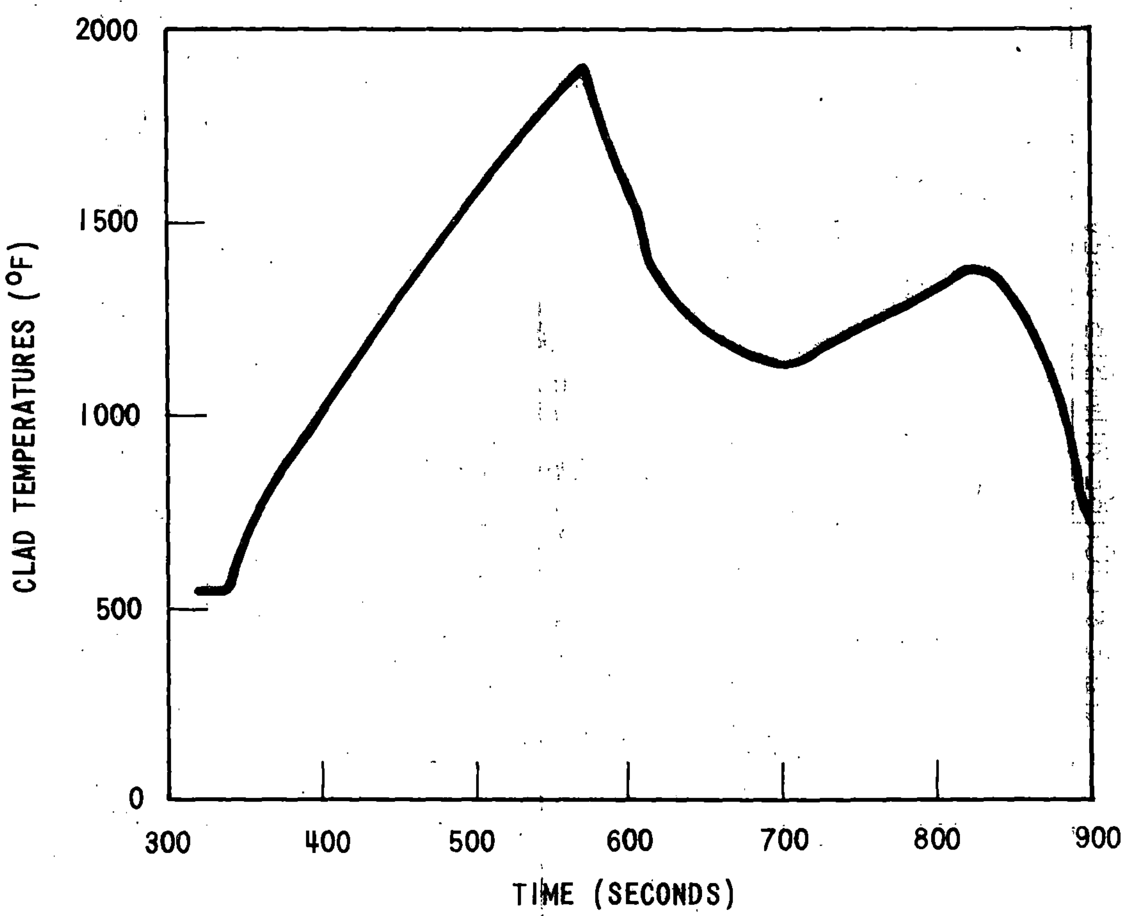
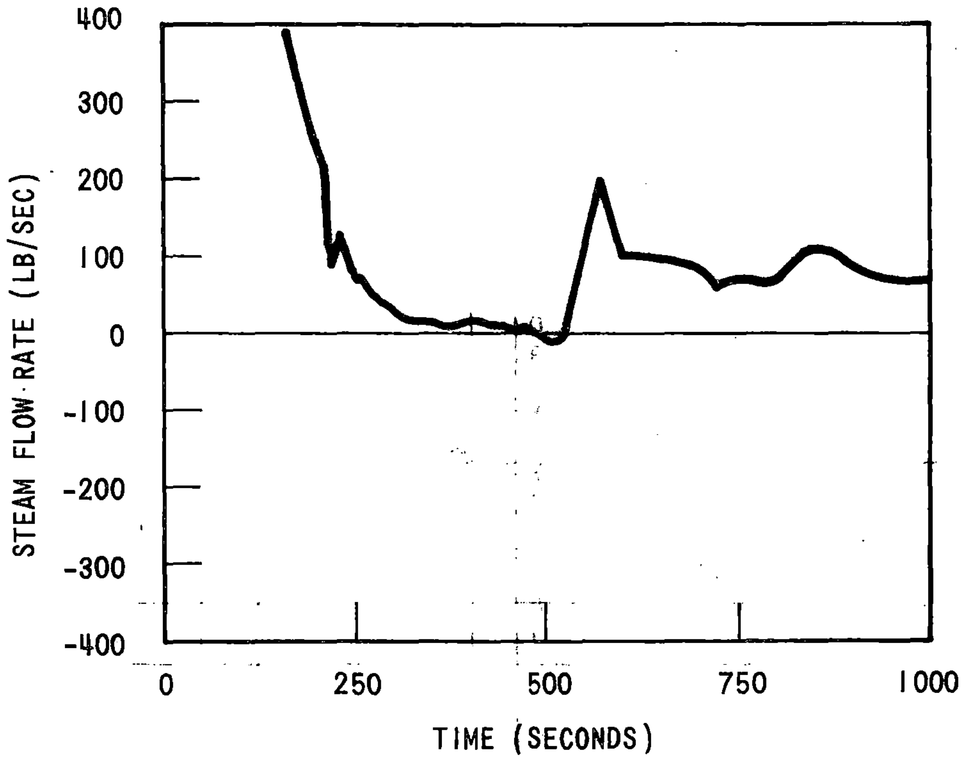


FIGURE II-5

STEAM FLOW (4 inch)



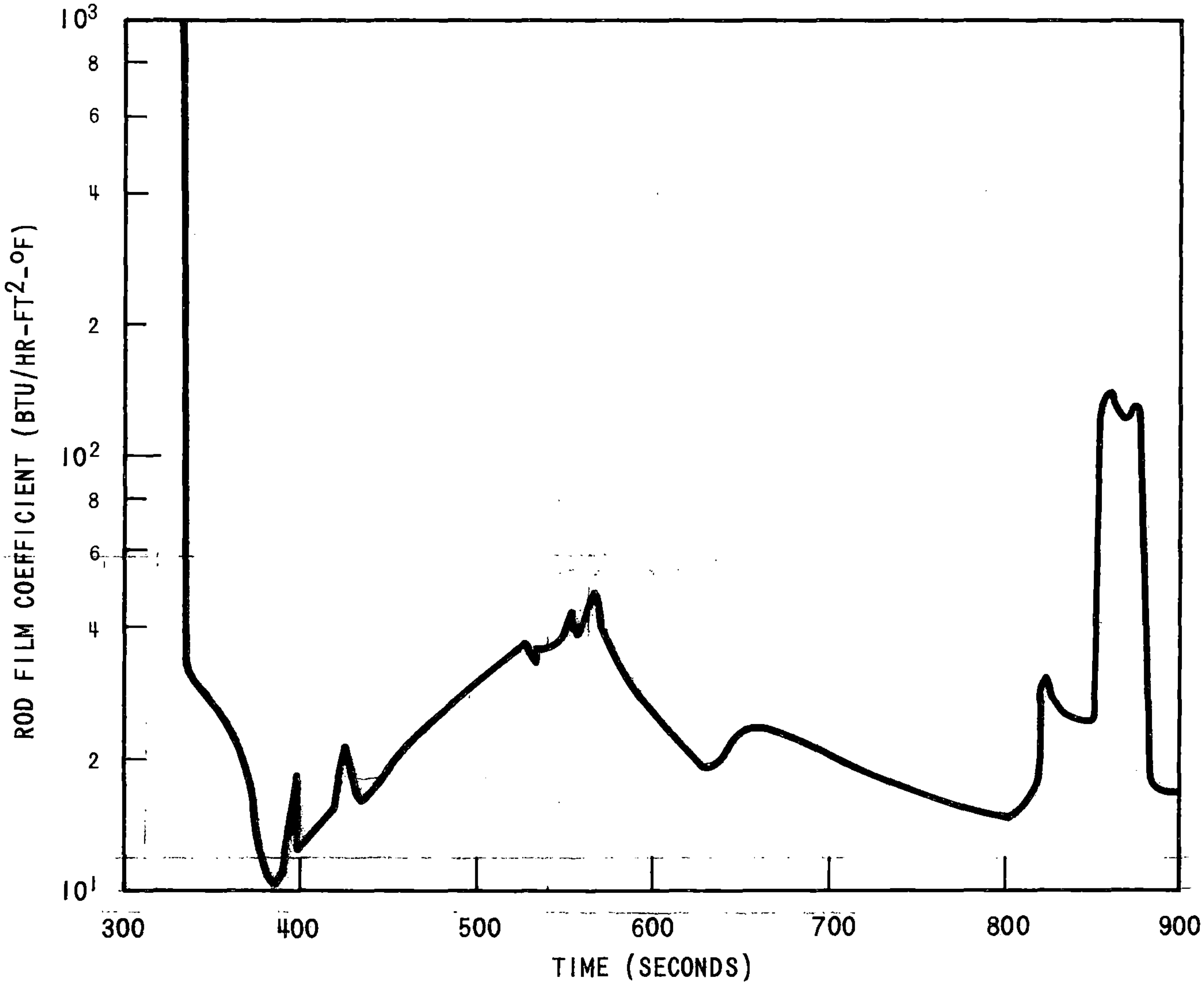
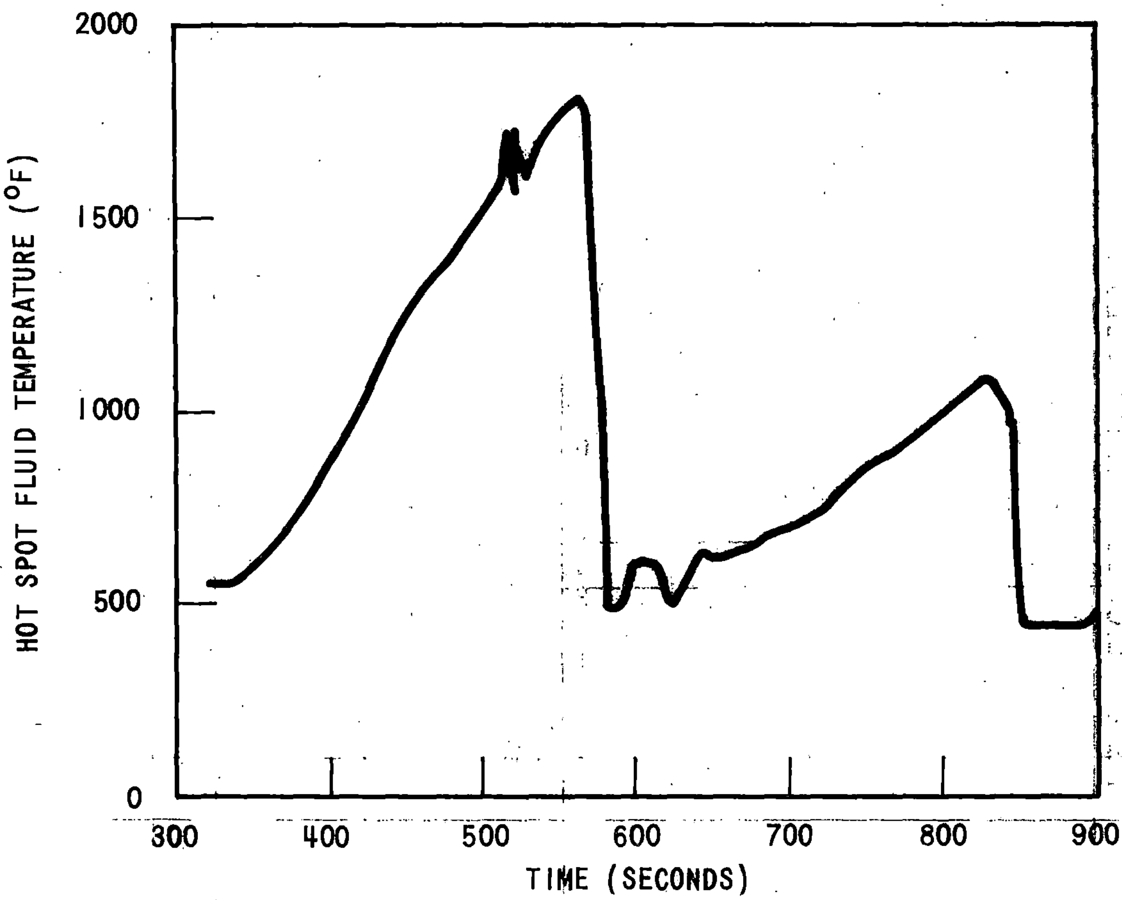


FIGURE II-6

ROD FILM COEFFICIENT (4 Inch)

FIGURE II-7

HOT SPOT FLUID TEMPERATURE (4 inch)



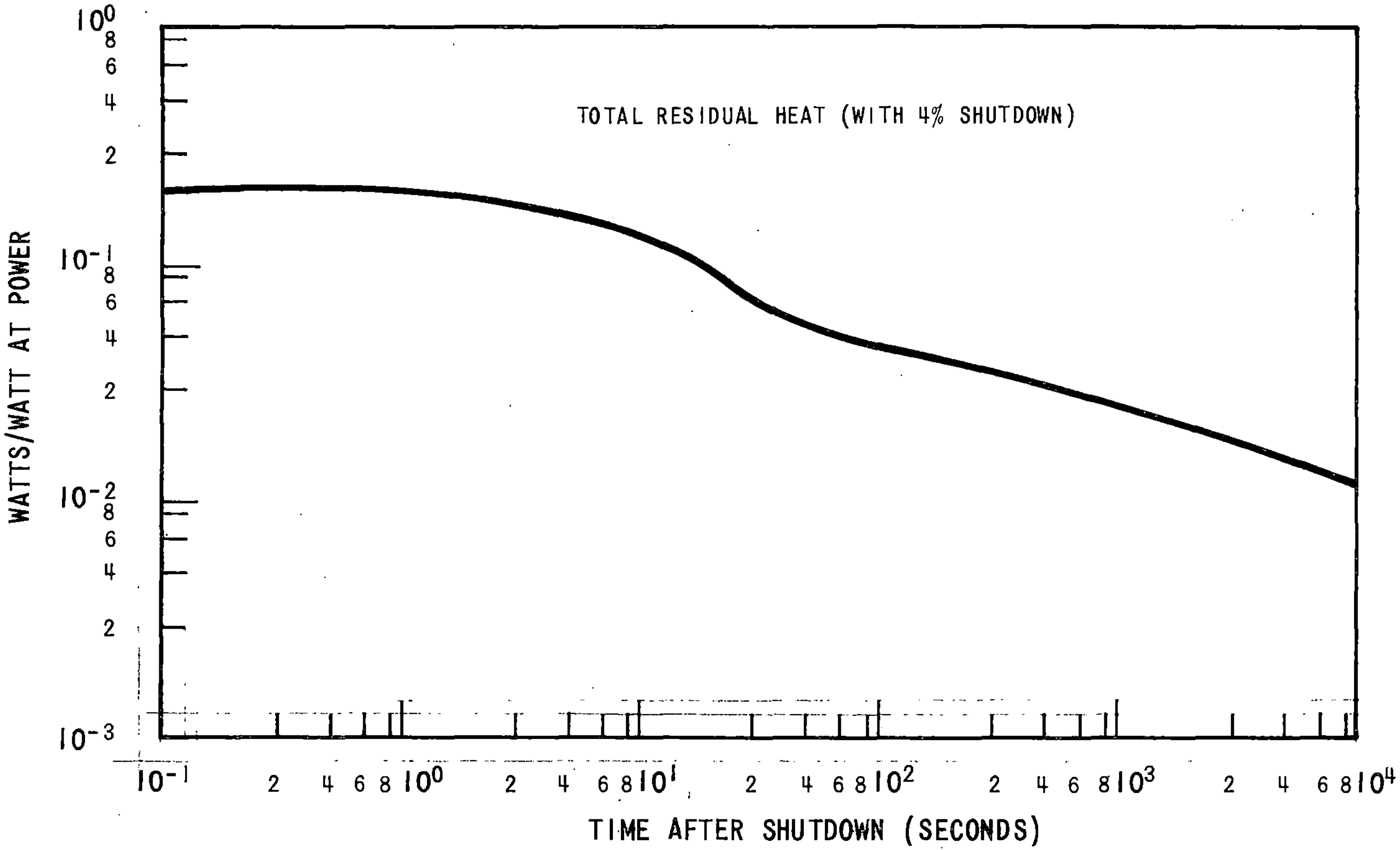


FIGURE II-8

CORE POWER

FIGURE II-9a

RCS DEPRESSURIZATION TRANSIENT (3 inch)

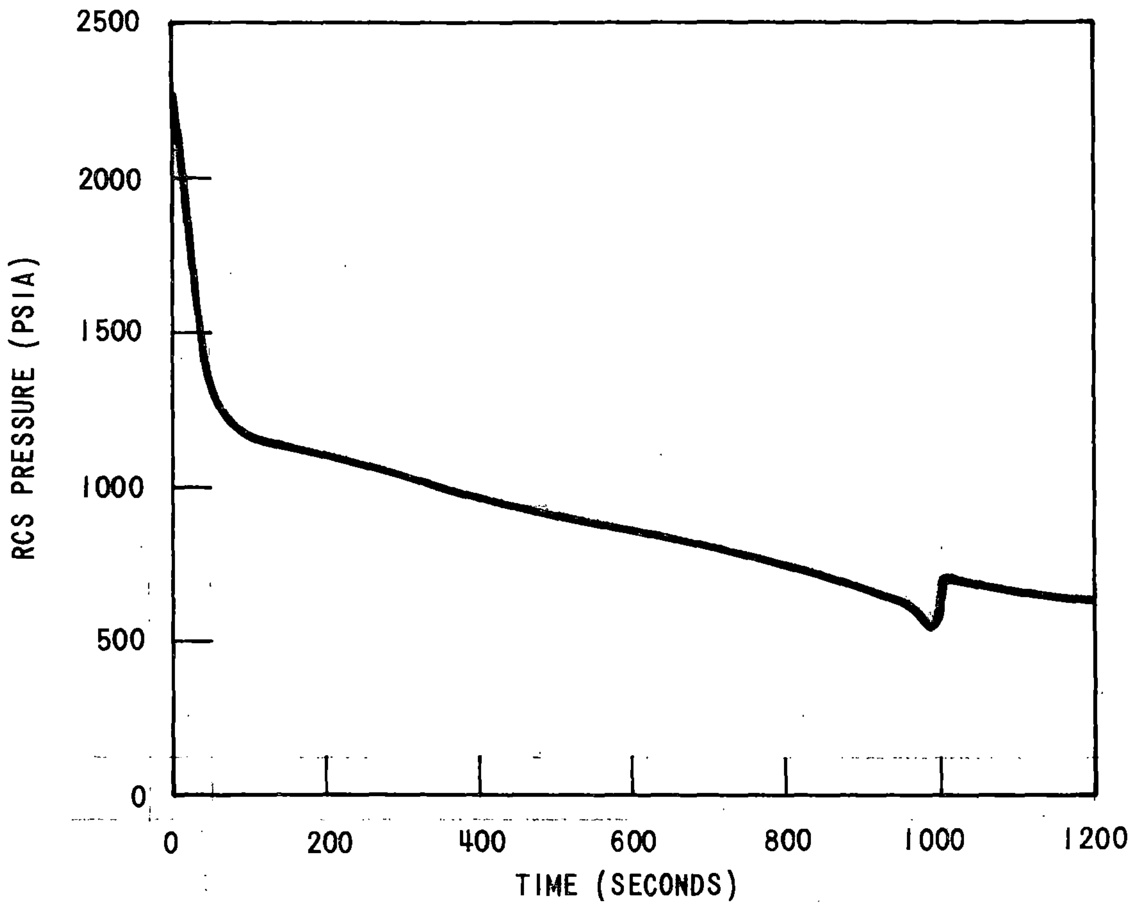


FIGURE II-9b

RCS DEPRESSURIZATION TRANSIENT (6 inch)

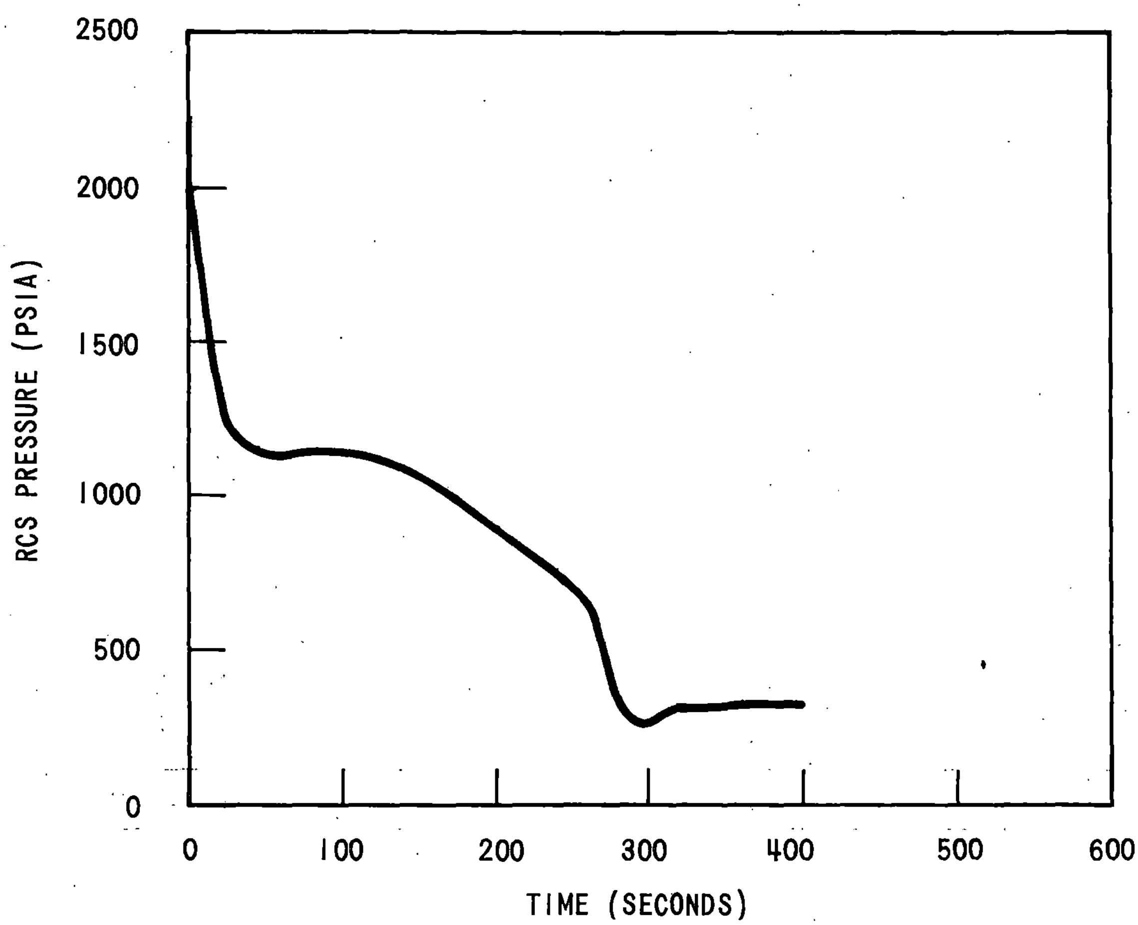


FIGURE II-10a

CORE MIXTURE HEIGHT (3 inch)

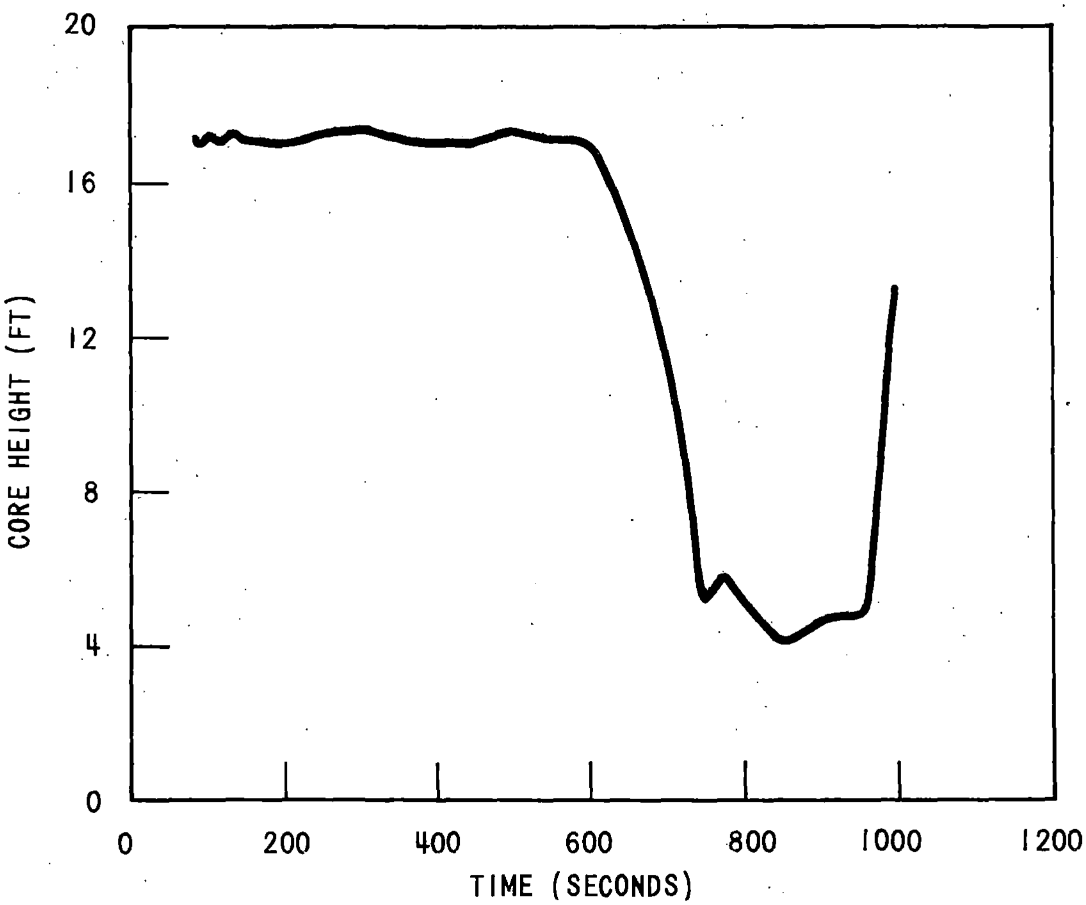


FIGURE II-10b

CORE MIXTURE HEIGHT (6 inch)

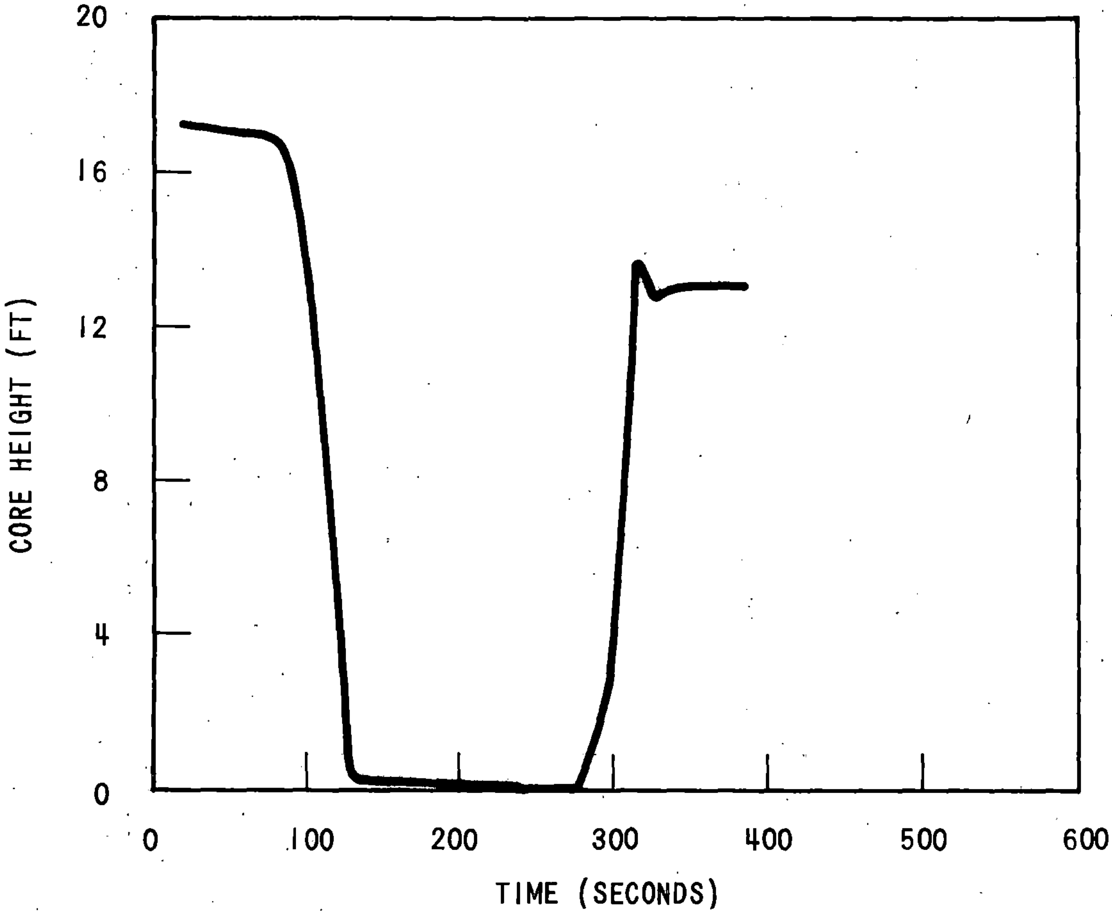


FIGURE II-11a

CLAD TEMPERATURE TRANSIENT (3 inch)

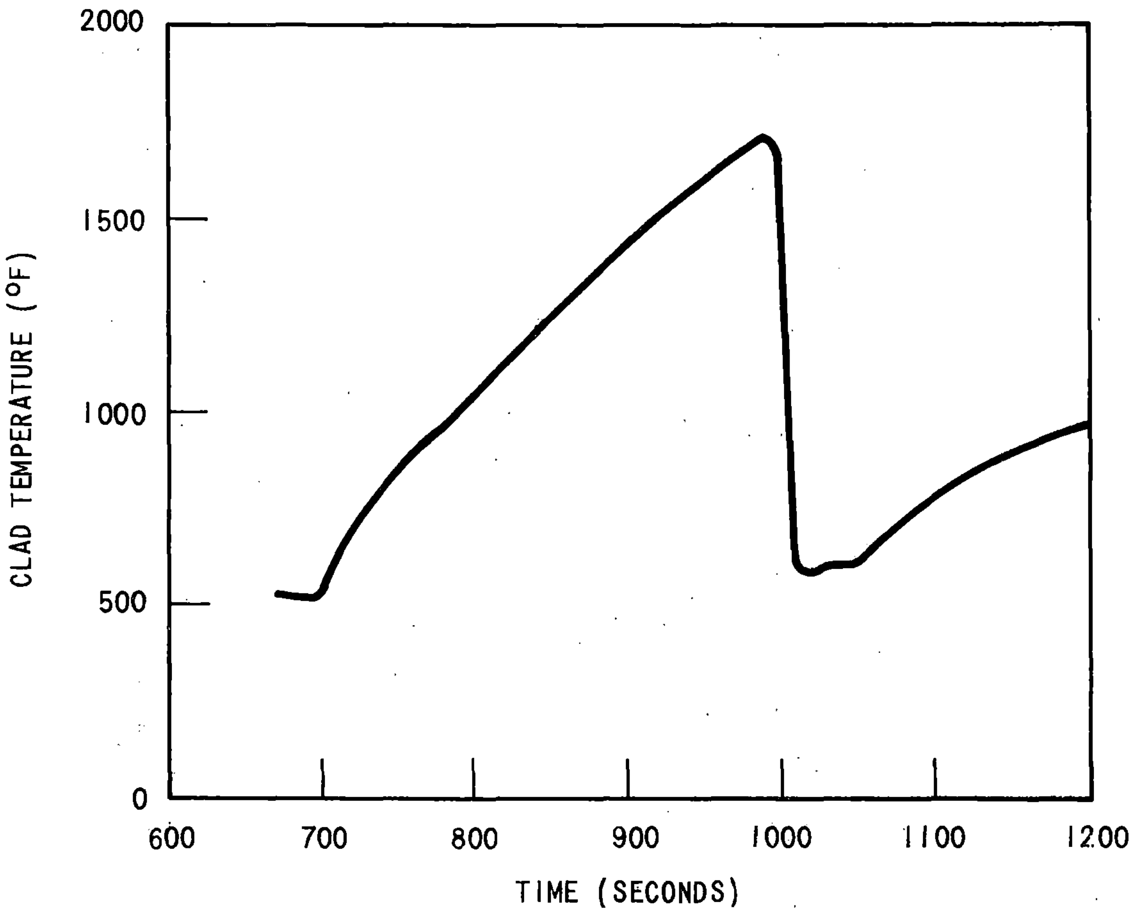
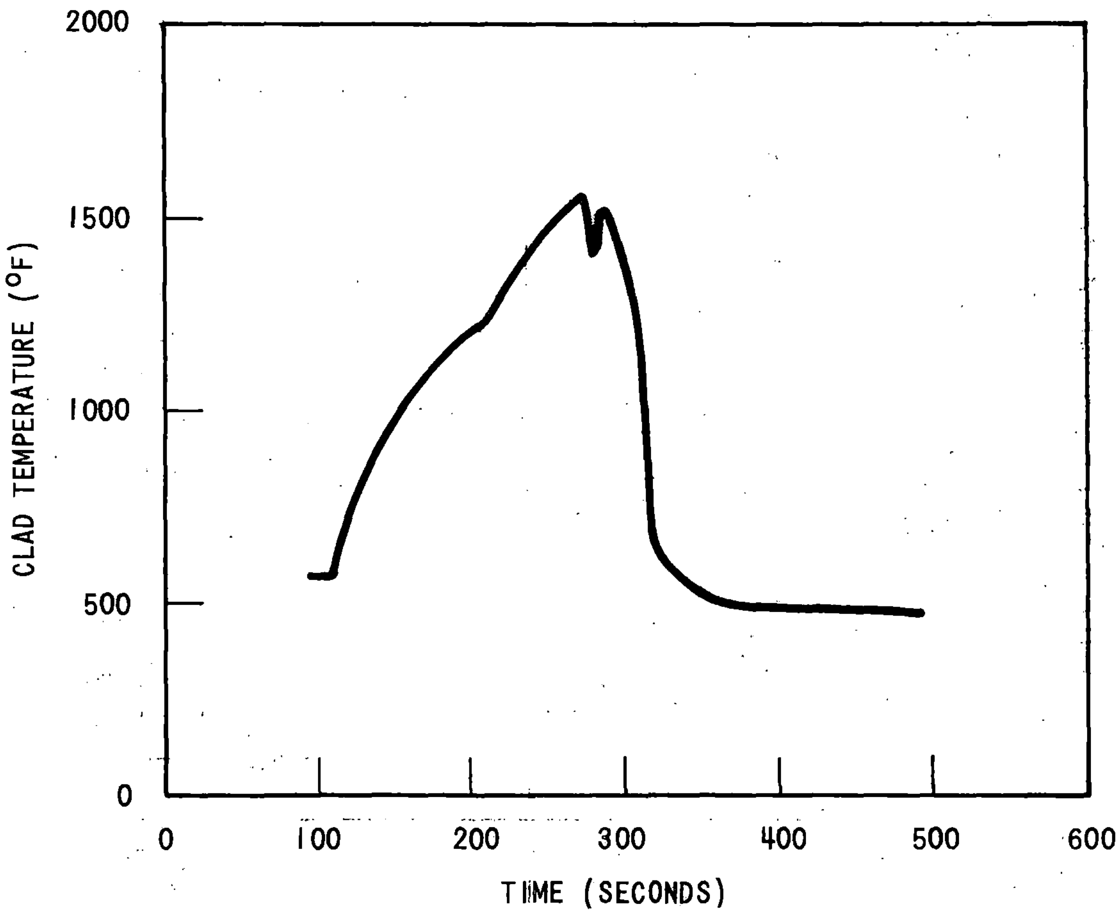


FIGURE II-11b

CLAD TEMPERATURE TRANSIENT (6 inch)



III. MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS OF COOLANT ACCIDENT)

Method of Thermal Analysis

The description of the various aspects of the LOCA analysis is given in WCAP-8339.[2] This document describes the major phenomena modeled, the interfaces among the computer codes and features of the codes which maintain compliance with the Acceptance Criteria. The SATAN-IV, WREFLOOD, COCO, and LOCTA-IV codes used in their analysis are described in detail in WCAP-8306,[3] WCAP-8171,[5] WCAP-8326[6] and WCAP-8305[4], respectively. The containment parameters used in the containment analysis code to determine the ECCS backpressure are presented in Table III-3.

Results

Table III-2 presents results for the double ended cold leg break for three (3) large break sizes (discharge coefficients (C_D)). This range of discharge coefficients was determined to include the limiting case for peak clad temperature from sensitivity studies reported in WCAP-8356.[7]

The analysis of the loss of coolant accident was performed at 102 per cent of 2542 megawatts thermal. The peak linear power and core power used in the analyses are given in Table III-2. The equivalent core peaking factor at the license application power level (2441 MWth) is also shown in Table III-2.

The limiting power peaking envelope applicable to this analysis, along with the actual power peaking values expected during operation of

each Surry unit, are shown in Figures III-14 and III-15. The actual power peaking values were determined using the procedures described in WCAP-8385[9] assuming a +6, -9 per cent delta flux (ΔI) band. The limiting envelope was determined on the basis that:

1. Extensive sensitivity studies presented in WCAP-8356[7] and WCAP-8472[15] have shown that the cosine is the worst power shape as long as the change of F_Q with elevation is maintained as shown on Figures III-14 and III-15 in the region from 0.0 feet to 11.0 feet.
2. The computed peak clad temperature does not exceed 2200°F for the case of a cosine power shape with $F_Q = 2.10$.
3. The location of the line segment between 11.0 feet and 12.0 feet is the same as that presented in WCAP-8356.[7] This line segment was confirmed by the small break results presented in Section II of this report.

For results discussed below, the hot spot is defined to be the location of maximum peak clad temperature. This location is given in Table III-2 for each of the three (3) discharge coefficients used for the double-ended break.

The time sequence of events for the three (3) discharge coefficients used in the large break analysis is shown in Table III-1.

Figures III-1 through III-13 present the transients for the principal parameters for the double ended cold break for the three (3) discharge coefficients used. Each of these parameters is enumerated below:

1. Fluid Quality - Figures III-1a, b and c show the fluid quality at the clad burst and hot spot locations (location of maximum clad temperature) on the hottest fuel rod (hot rod) for the three (3) discharge coefficients used.

2. Mass Velocity - Figures III-2a, b and c show the mass velocity at the clad burst and hot spot locations on the hottest fuel rod for the three (3) discharge coefficients used.

3. Heat Transfer Coefficient - Figures III-3a, b and c show the heat transfer coefficient at the clad burst and hot spot locations on the hottest rod for the three (3) discharge coefficients used. The heat transfer coefficient shown was calculated by the LOCTA IV code.

4. Core Pressure - Figures III-4a, b and c show the calculated pressure in the core for the three (3) discharge coefficients used.

5. Break Flow Rate - Figures III-5a, b and c show the calculated flow rate out of the break for the three (3) discharge coefficients used. The flow rate out the break is plotted as the sum of both ends for the guillotine break cases.

6. Core Pressure Drop - Figures III-6a, b and c show the calculated core pressure drop for the three (3) discharge coefficients used. The core pressure drop

shown is from the lower plenum, near the core, to the upper lenum at the core outlet.

7. Peak Clad Temperature - Figures III-7a, b and c show the calculated hot spot clad temperature transient and the clad temperature transient at the burst location for the three (3) discharge coefficients used.
8. Fluid Temperature - Figures III-8a, b and c show the calculated fluid temperature for the hot spot and burst locations for the three (3) discharge coefficients used.
9. Core Flow - Figures III-9a, b and c show the calculated core flow, both top and bottom, for the three (3) discharge coefficients used.
10. Reflood Transient - Figures III-10a, b and c show the calculated reflood transient for the three (3) discharge coefficients used.
11. Accumulator Flow - Figures III-11a, b and c show the accumulator flow for the three (3) discharge coefficients used. The accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in refill reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.
12. Pumped ECCS Flow (Reflood) - Figures III-12a, b and c show the calculated flow of the emergency core cooling system for the three (3) discharge coefficients used.

13. Containment Pressure - Figures III-13a, b and c show the calculated pressure transient for the three (3) discharge coefficients used. These pressure transients are based on the data given in Table III-3.

The clad temperature analysis is based on a total peaking factor of 2.10. The hot spot metal reaction reached is 5.6 per cent, which is will below the embrittlement limit of 17 per cent, as required by 10 CFR 50.46. In addition, the total core metal-water reaction is less than 0.3 per cent for all breaks as compared with the 1 per cent criterion of 10 CFR 50.46.

The results of several sensitivity studies are reported in WCAP-8356.[7] These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

TABLE III-1

TIME SEQUENCE OF EVENTS FOR DOUBLE ENDED COLD LEG GUILLOTINE BREAKS (DECLG)

EVENT	DISCHARGE COEFFICIENT (C_D)		
	(C_D=1.0)	(C_D=0.6)	(C_D=0.4)
	TIME AFTER START OF LOCA (SECONDS)		
Start of LOCA	0.0	0.0	0.0
Reactor Trip Signal	0.63	0.65	0.66
Safety Injection Signal	1.43	1.78	2.2
Accumulator Injection	11.5	14.4	18.0
End of Blowdown	23.9	23.3	30.6
Pump Injection	26.5	26.8	27.2
Bottom of Core Recovery	37.3	36.9	40.99
Accumulator Empty	43.4	45.8	49.6

TABLE III-2

ASSUMPTIONS AND RESULTS FOR
DOUBLE ENDED COLD LEG GUILLOTINE BREAKS (DECLG)

Results	DECLG (C_D=1.0)	DECLG (C_D=0.6)	DECLG (C_D=0.4)
Peak Clad Temp. (^{o}F)	1986	1995	2090
Peak Clad Location (Ft.)	7.5	7.00	6.75
Local Zr/H_2O Rxn (max) (%)	3.90	4.19	5.60
Local Zr/H_2O Location (Ft.)	7.5	7.25	7.0
Total Zr/H_2O Rxn (%)	<0.3	<0.3	<0.3
Hot Rod Burst Time (sec)	48.2	32.4	27.2
Hot Rod Burst Location (Ft.)	6.0	6.00	5.75

Assumptions	
Power (MWt) 102% of	2542
Peak Linear Power (Kw/ft) 102% of	13.12
Peaking Factor Used	2.10
Accumulator Water Volume (Ft^3)	975.0

Fuel region analyzed (Most Limiting)	Cycle	Region
Unit No. 1	2	4
Unit No. 2	2	4

TABLE III-3

CONTAINMENT DATA

NET FREE VOLUME	1.863×10^6 ft^3
INITIAL CONDITIONS	
Pressure	9.35 psi
Temperature	75 °F
RWST Temperature	40 °F
Service Water Temperature	35.0 °F
Outside Temperature	9.0 °F
SPRAY SYSTEM I	
Number of Pumps Operating	2
Runout Flow Rate (each)	3200 gpm
Actuation Time	20.0 secs
SPRAY SYSTEM II	
RECIRCULATION SPRAY FROM SUMP	
Number of Pumps Operating	2
Runout Flow Rate (each)	3500 gpm
Actuation Time	125 secs
Heat Exchanger UA (per pump)	3.5×10^6 BTU/hr-°F
Service Water Flow (per exchanger)	6100 gpm

TABLE III-3

CONTAINMENT DATA

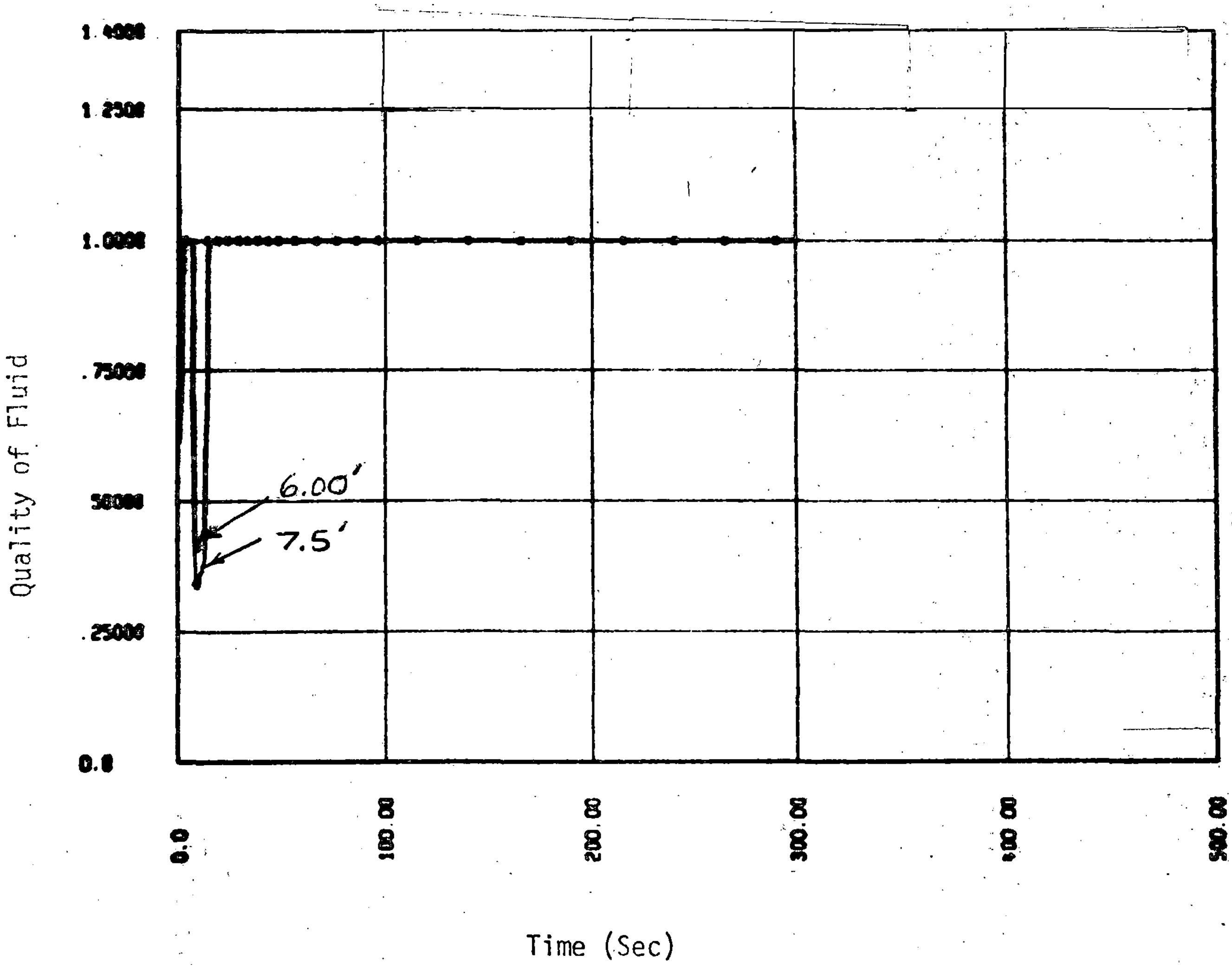
STRUCTURAL HEAT SINKS

Thickness (In)	Area (Ft^2), including uncertainty
Concrete 6.	6972
Concrete 12.	57,960
Concrete 18.	40,470
Concrete 24.	10,500
Concrete 36.	4410
Carbon Steel 0.375 Concrete 54	46,887
Carbon Steel 0.50 Concrete 30.	25,075
Concrete 24.	11,284
Carbon Steel 0.366	167,165
Stainless Steel 0.426	3399

FIGURE III-1a

FLUID QUALITY

(At hottest fuel rod)

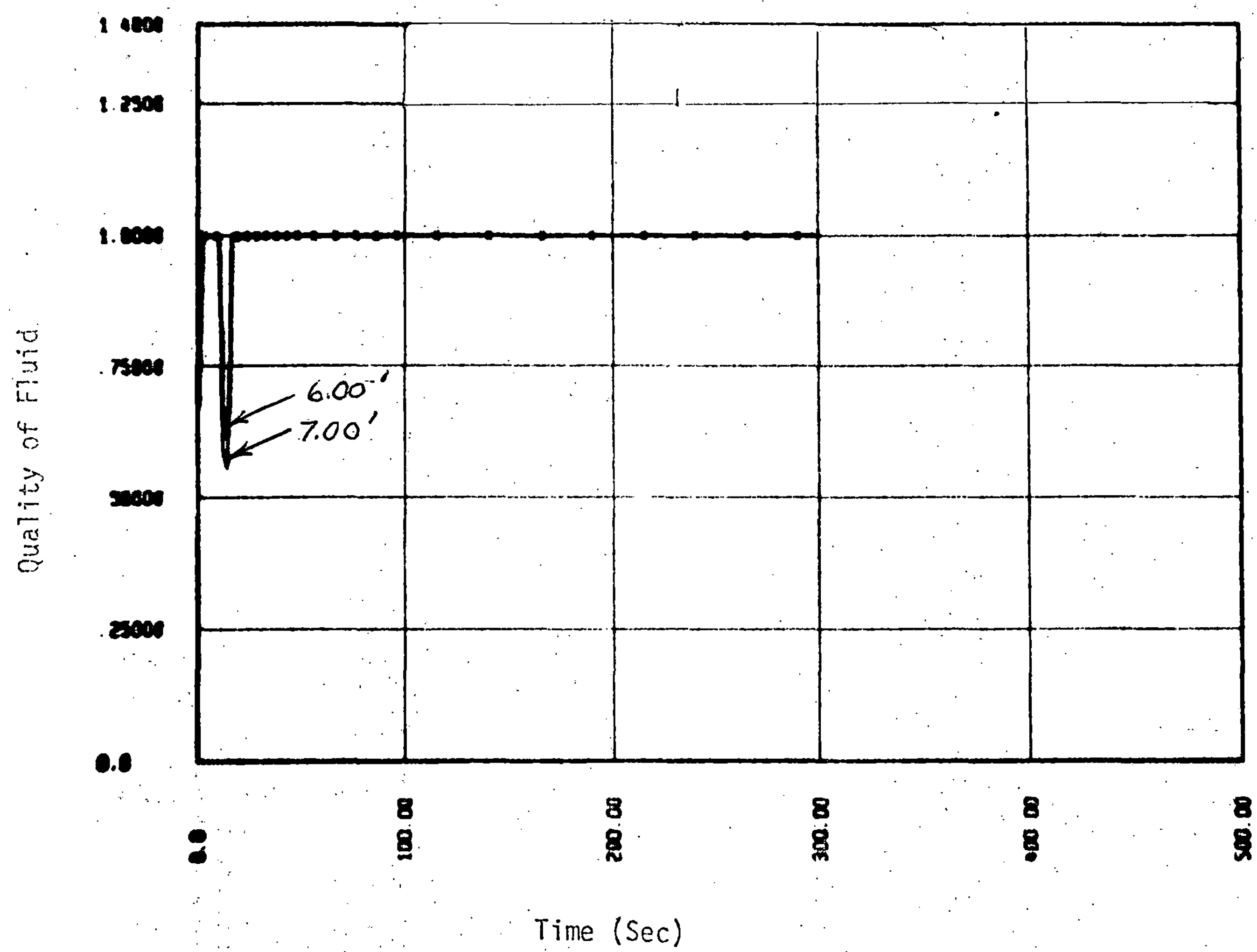


DECLG (C_D=1.0)

FIGURE III-1b

FLUID QUALITY

(At hottest fuel rod)

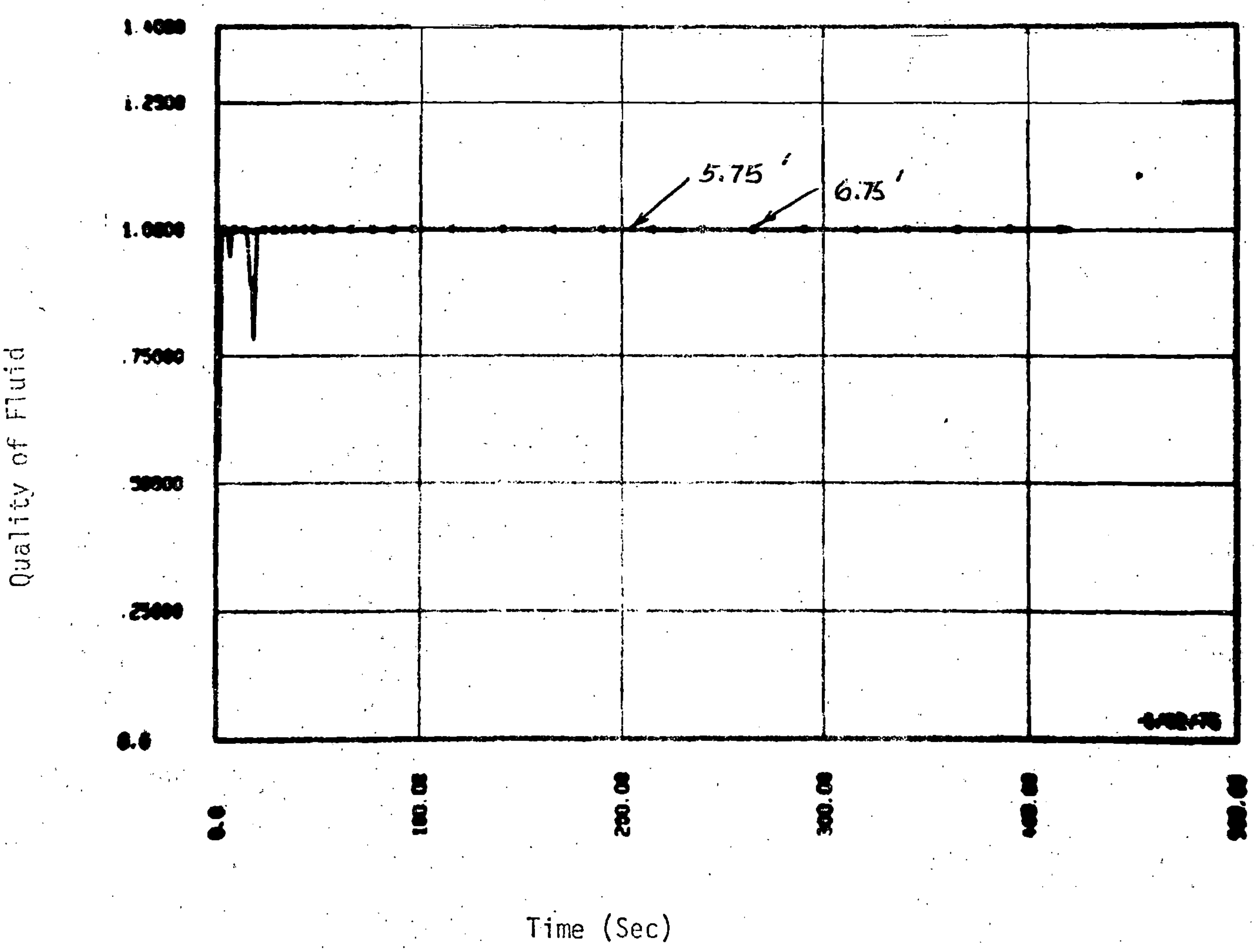


DECLG (C_D=0.6)

FIGURE III-1c

FLUID QUALITY

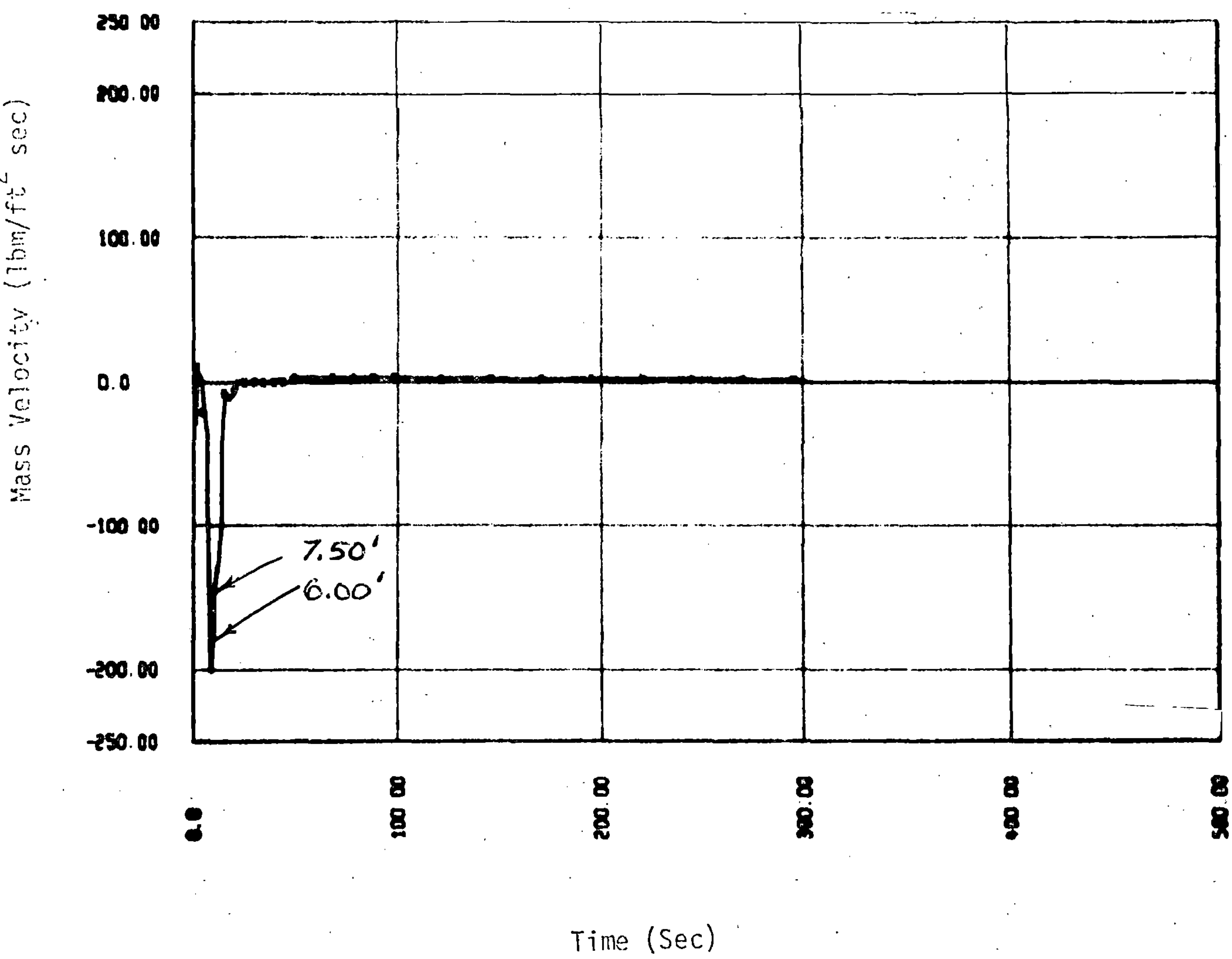
(At hottest fuel rod)



DECLG (C_D=0.4)

FIGURE III-2a

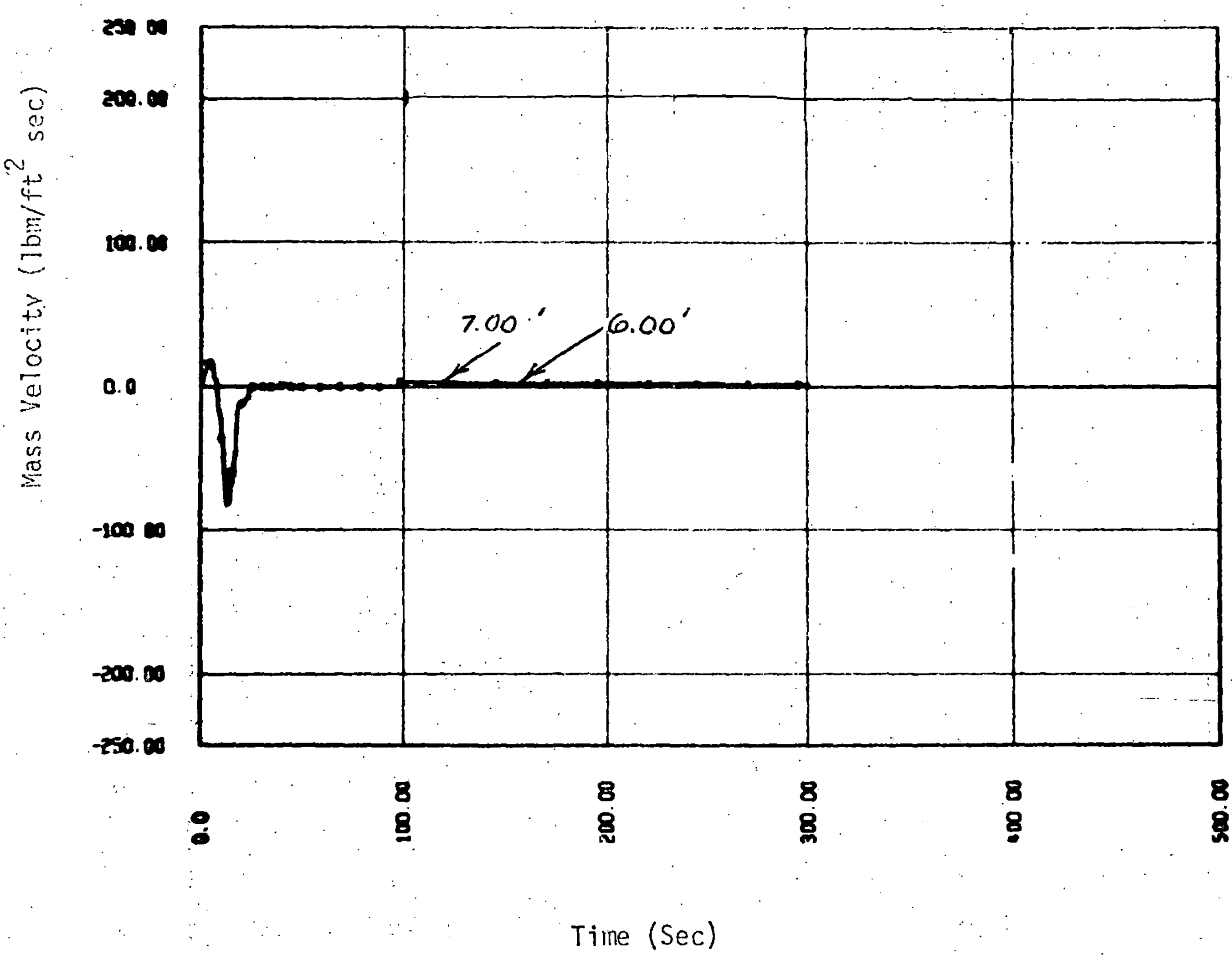
MASS VELOCITY



DECLG (C_D=1.0)

FIGURE III-2b

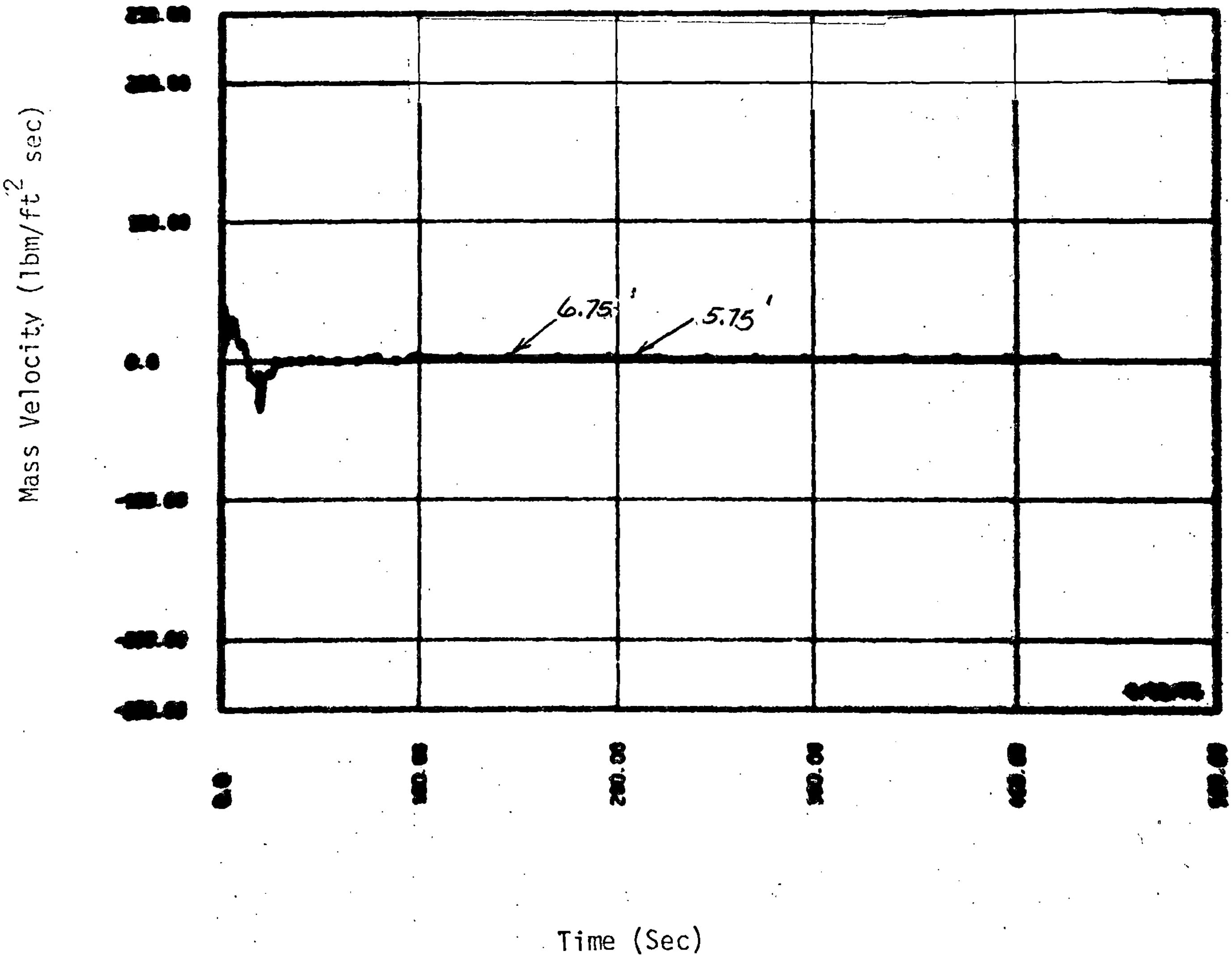
MASS VELOCITY



DECLG (C_D=0.6)

FIGURE III-2c

MASS VELOCITY

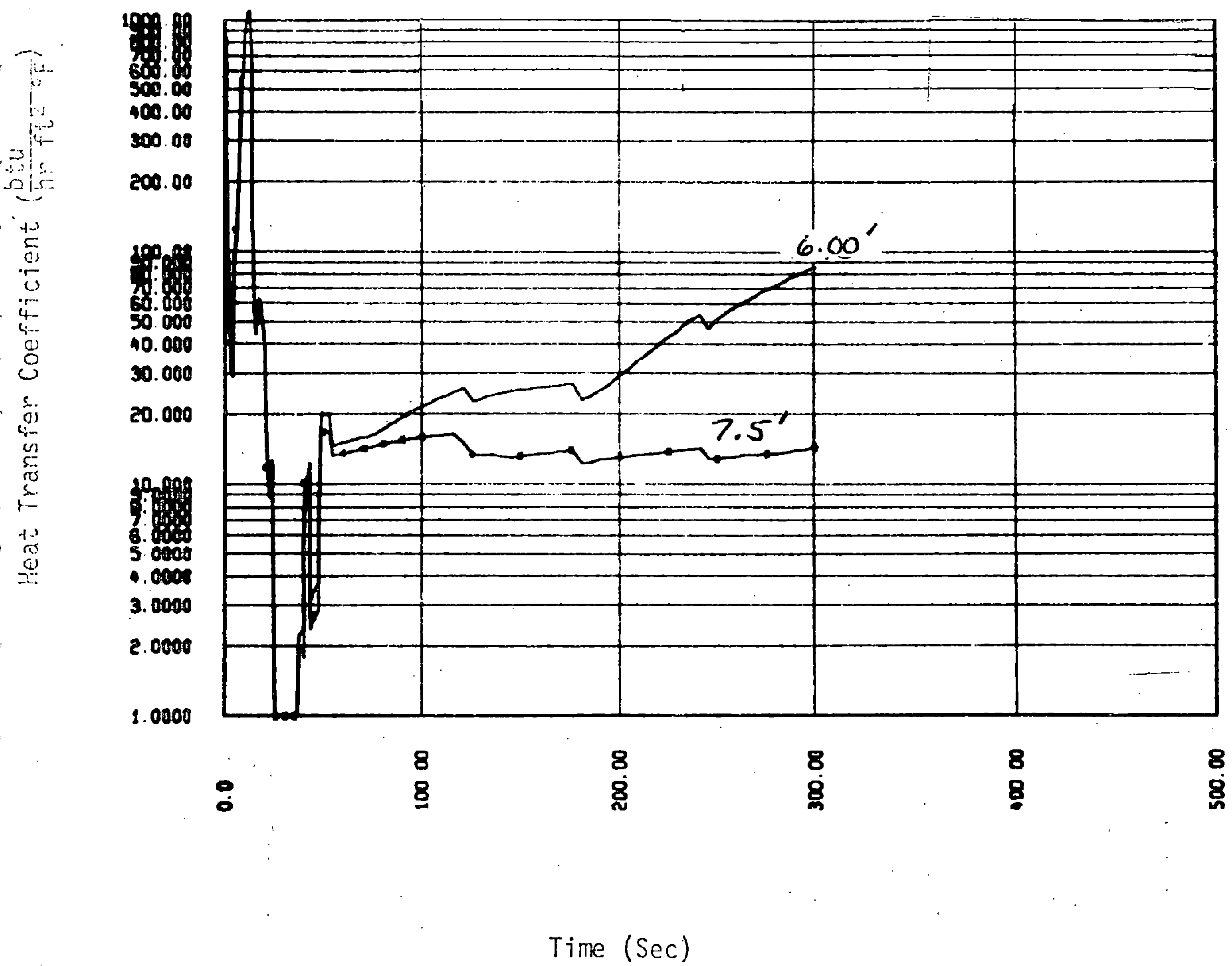


DECLG (C_D=0.4)

FIGURE III-3a

HEAT TRANSFER COEFFICIENT

(At hottest fuel rod)

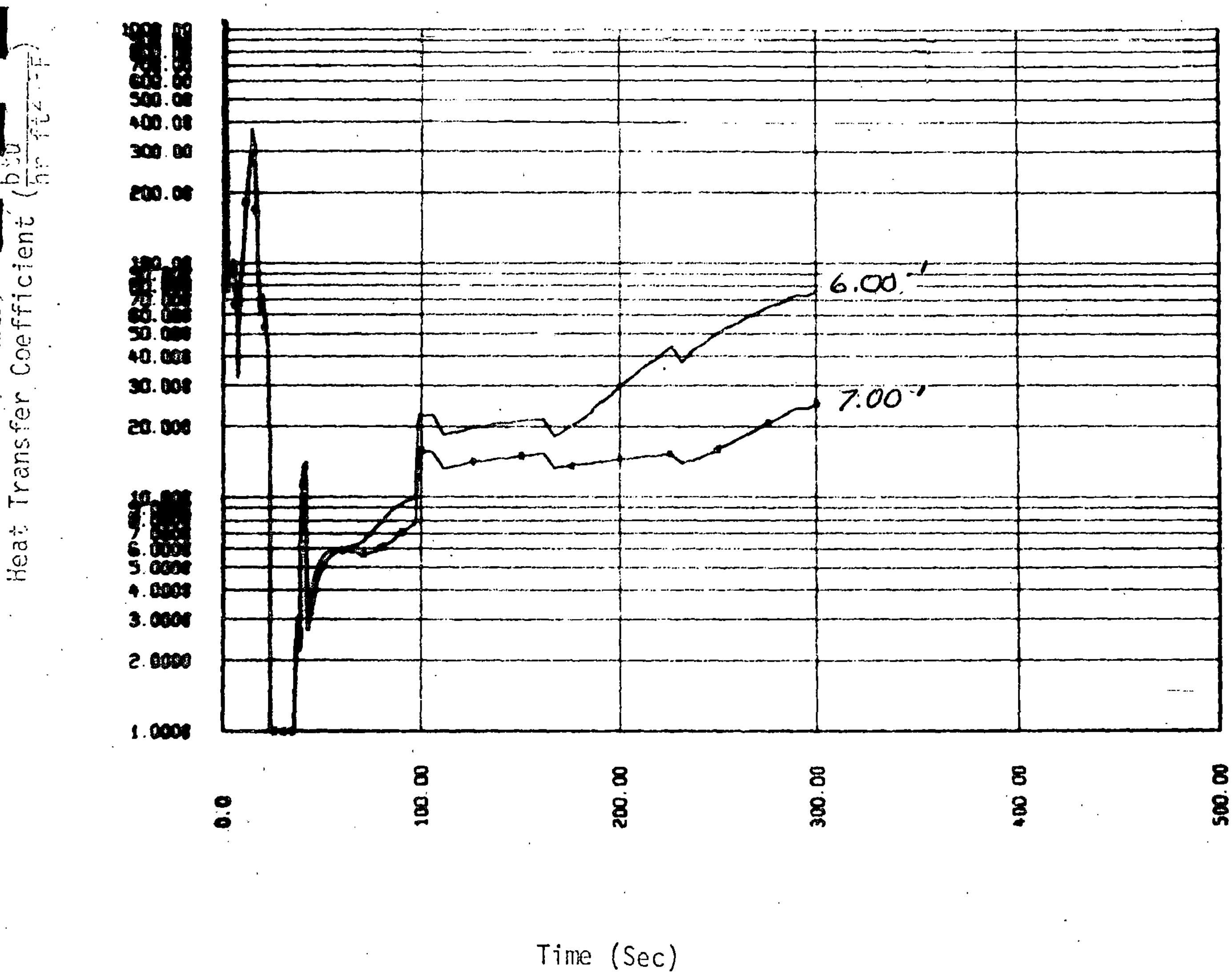


BFGLC (C_D=1.0)

FIGURE III-3b

HEAT TRANSFER COEFFICIENT

(At hottest fuel rod)

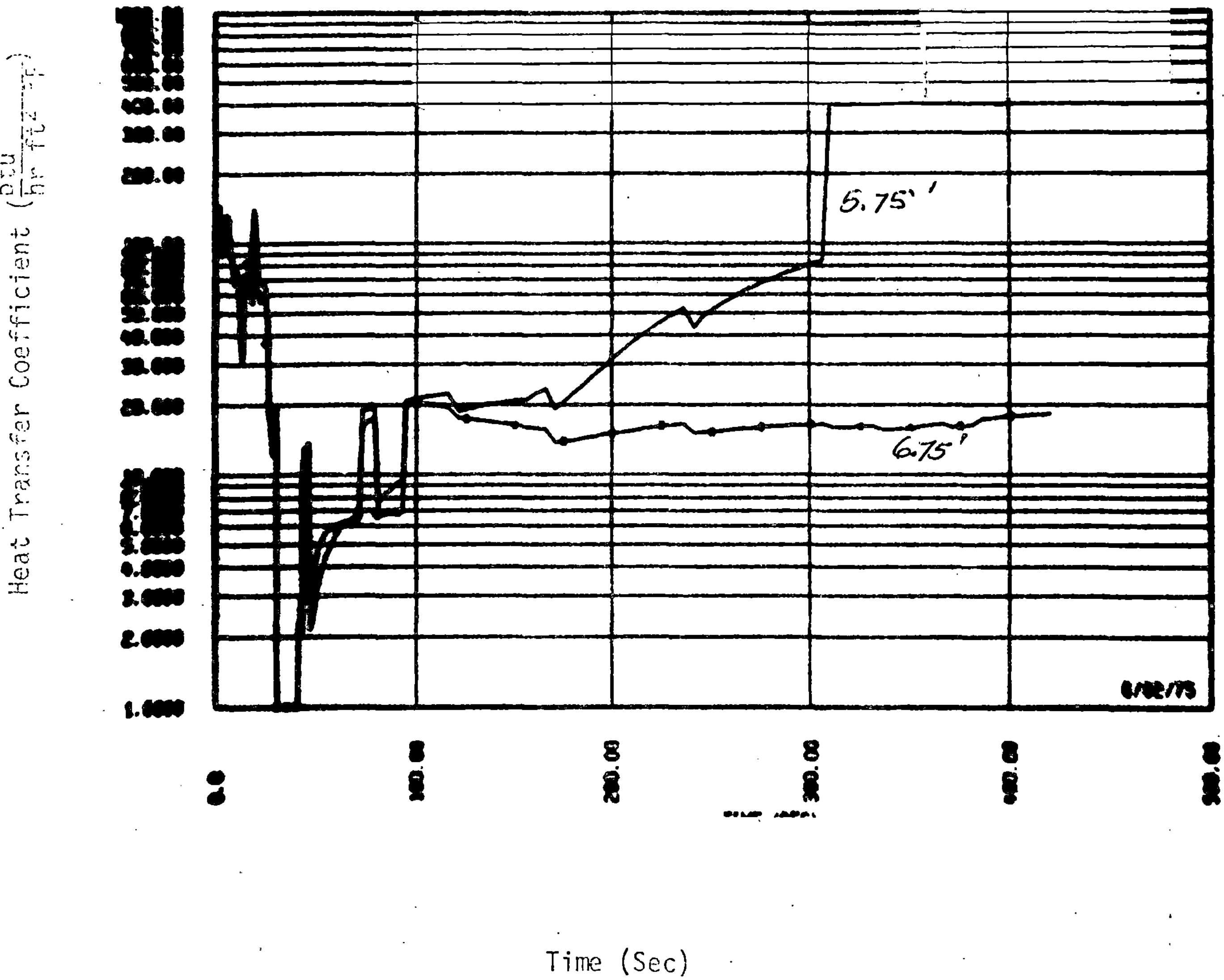


DECLG (C_D=0.6)

FIGURE III-3c

HEAT TRANSFER COEFFICIENT

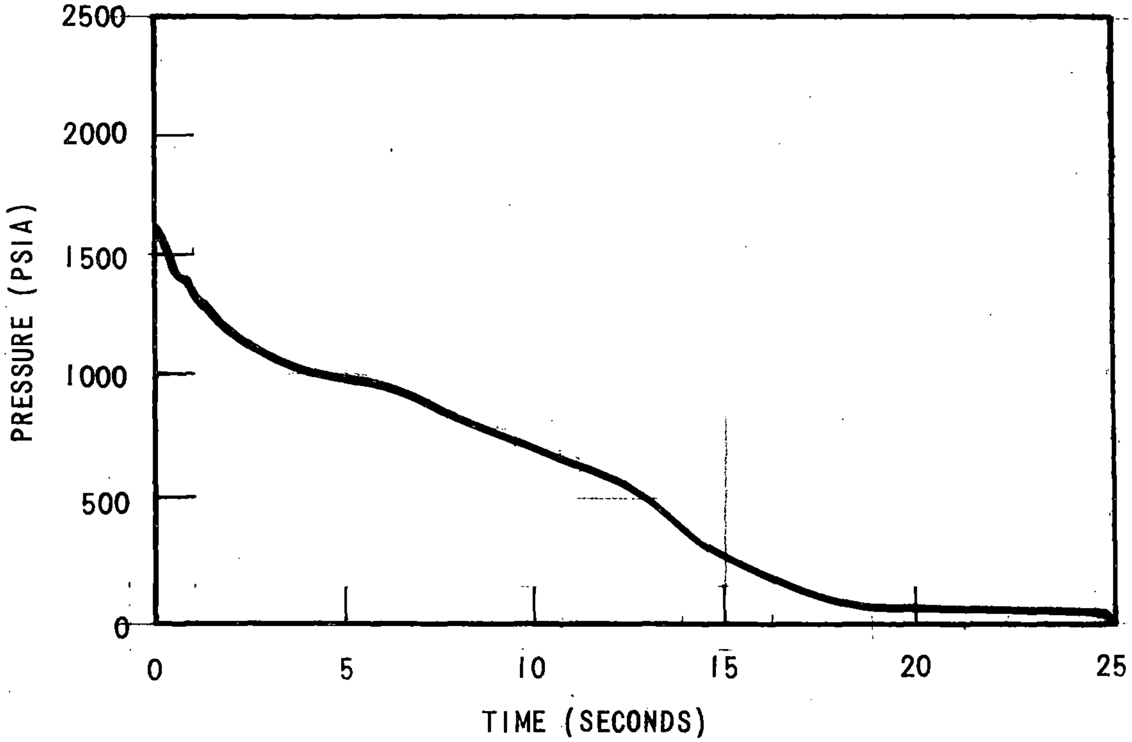
(At hottest fuel rod)



DECLG (C_D=0.4)

FIGURE III-4a

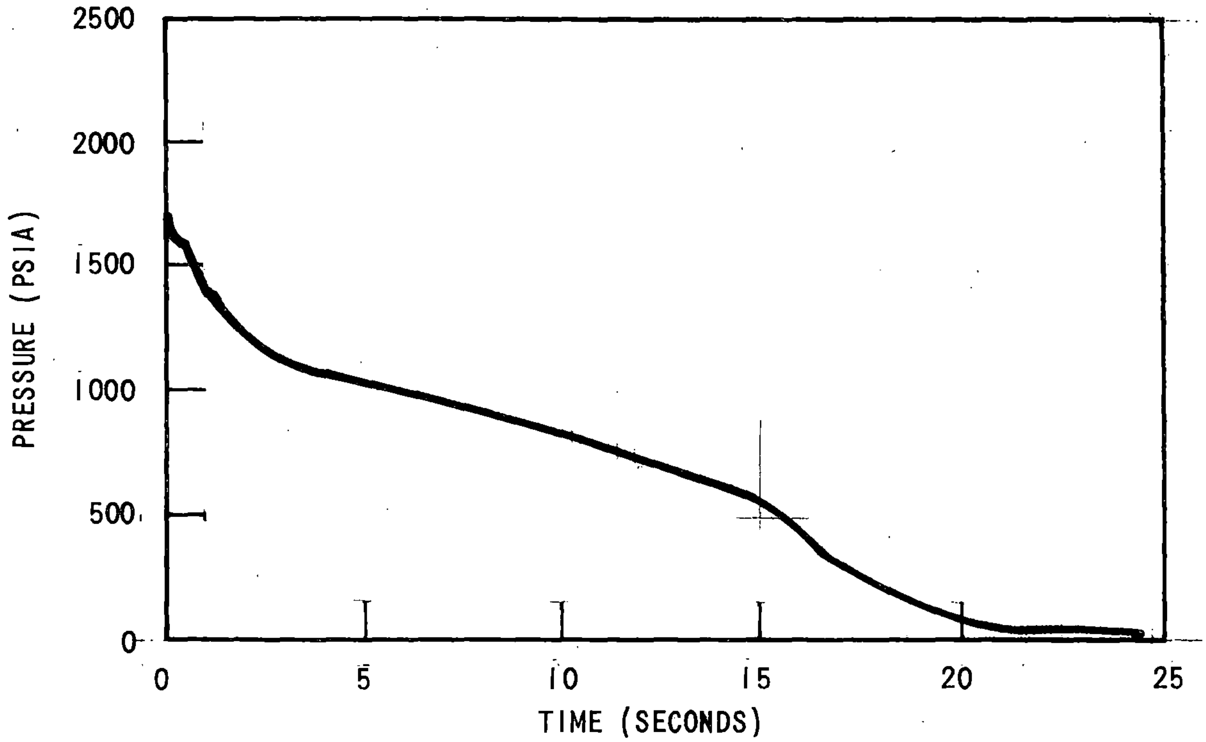
CORE PRESSURE



DECLG (C_D=1.0)

FIGURE III-4b

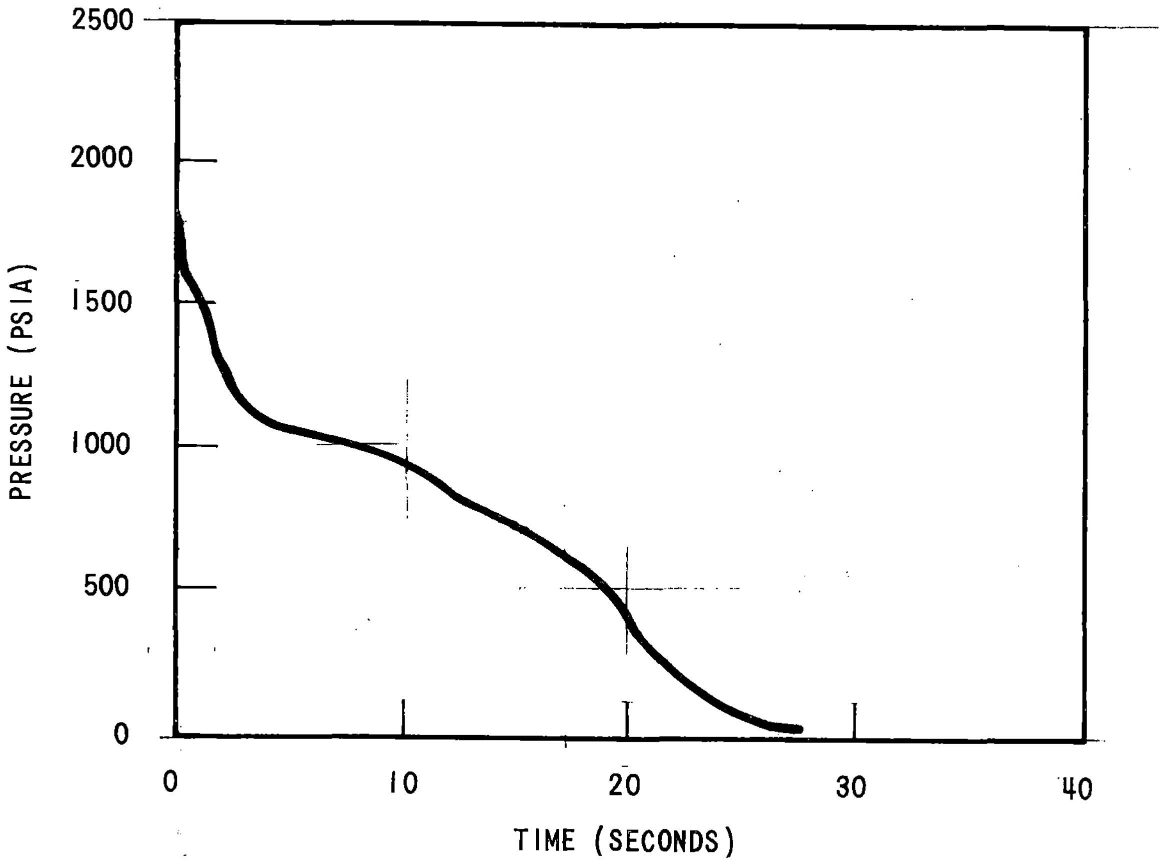
CORE PRESSURE



DECLG (C_D=0.6)

FIGURE III-4c

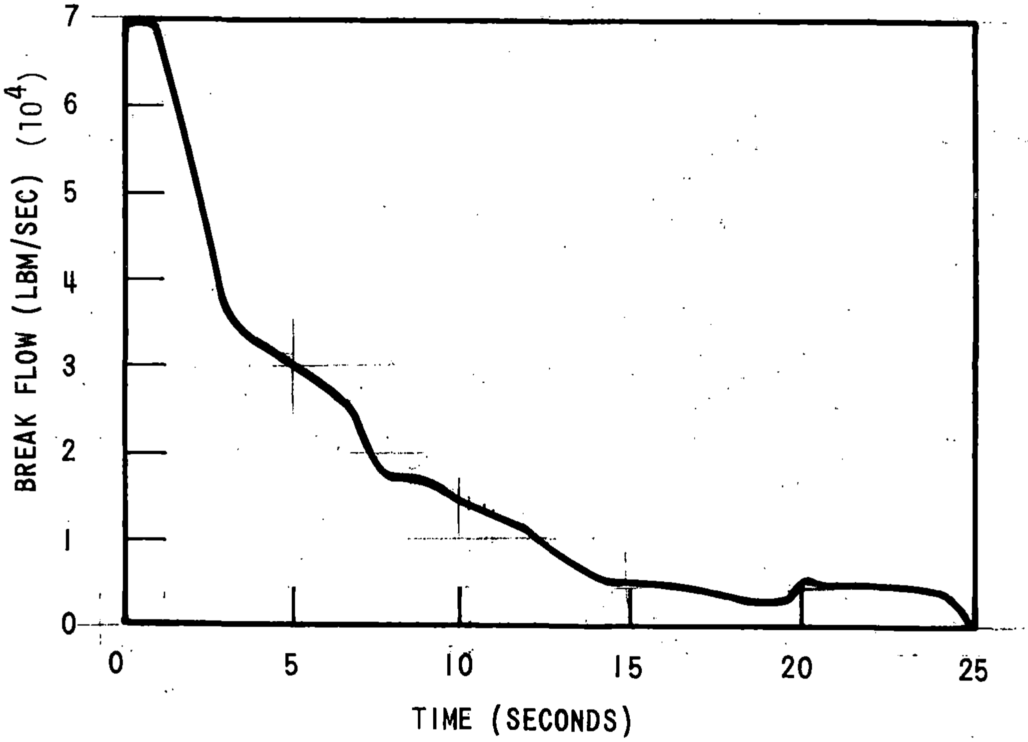
CORE PRESSURE



DECLG (C_D=0.4)

FIGURE III-5a

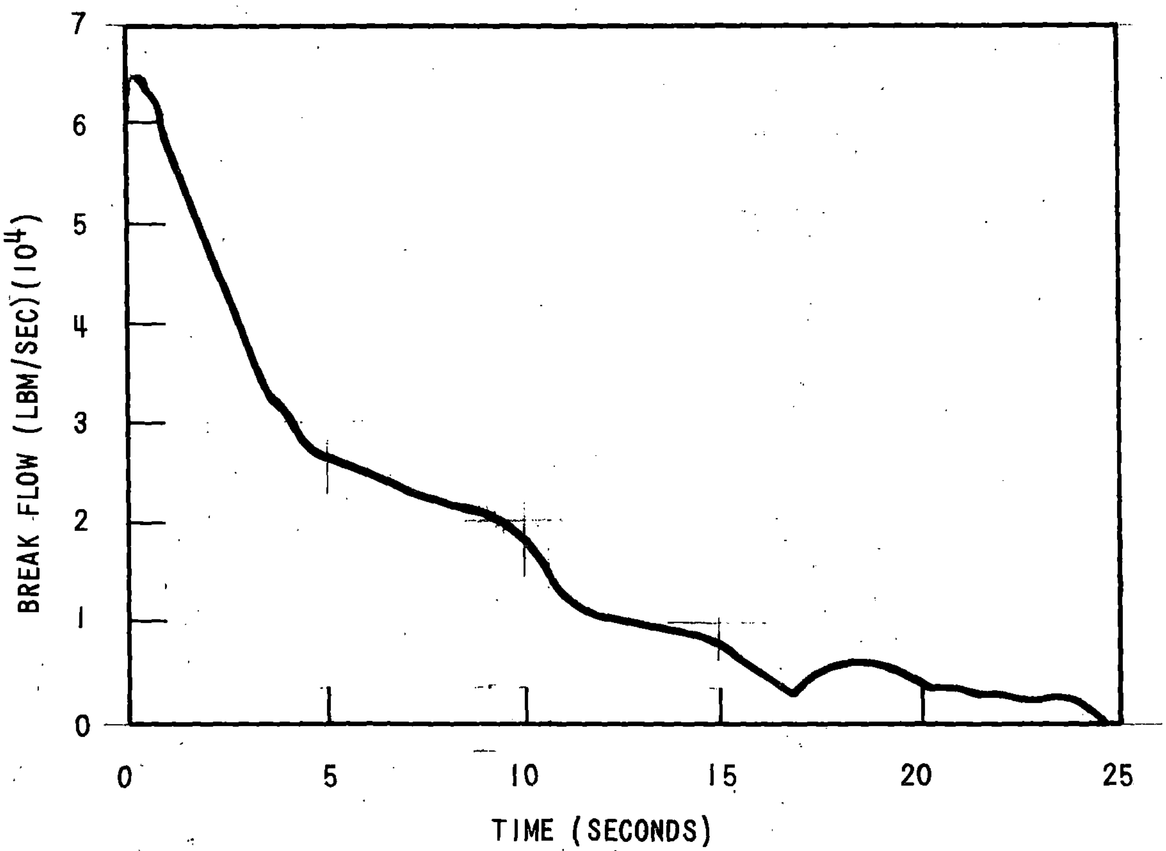
BREAK FLOW RATE



DECLG (C_D=1.0)

FIGURE III-5b

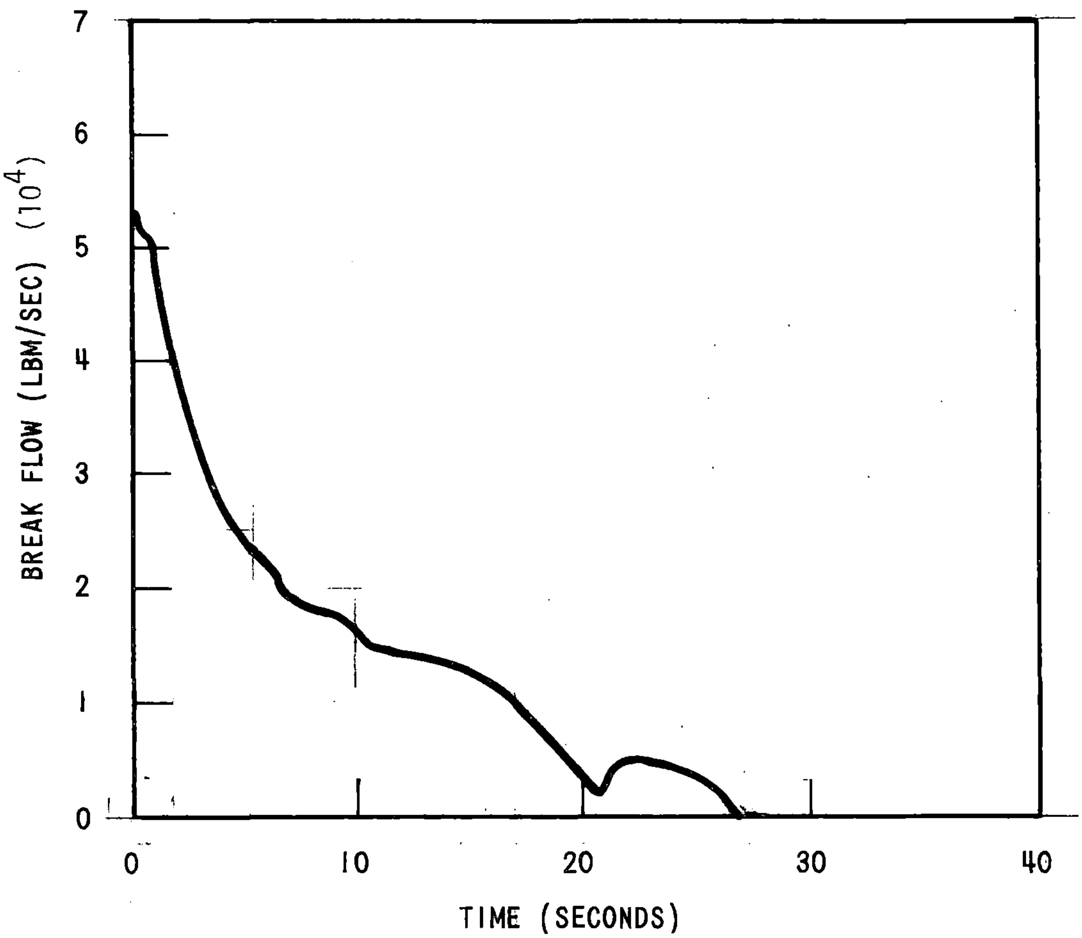
BREAK FLOW RATE



DECLG (C_D=0.6)

FIGURE III-5c

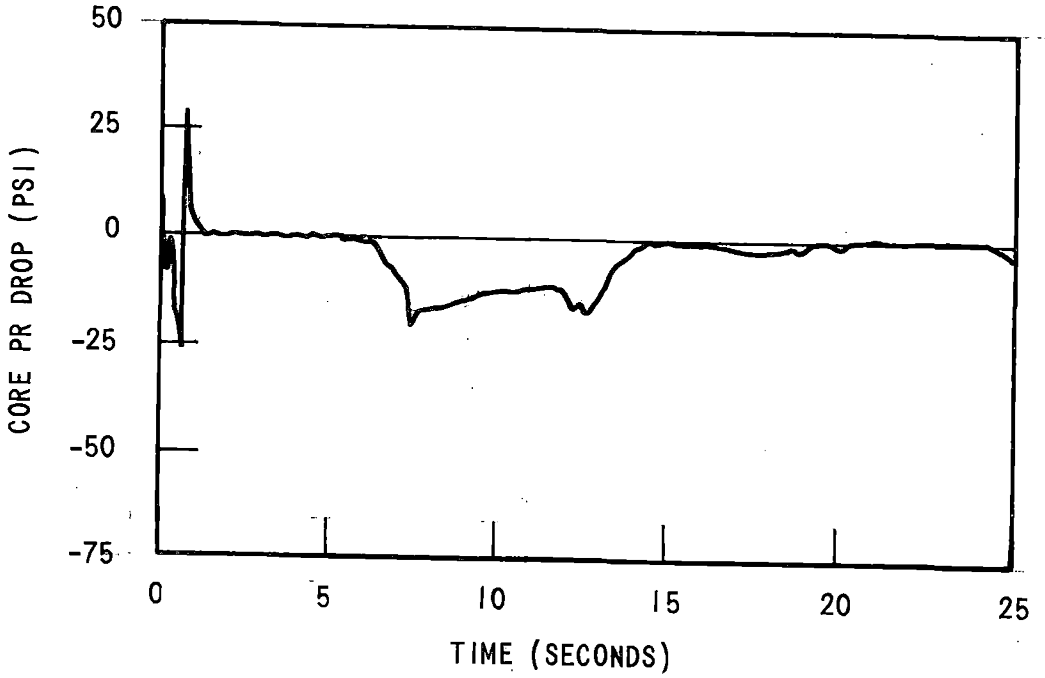
BREAK FLOW RATE



DECLG (C_D=0.4)

FIGURE III-6a

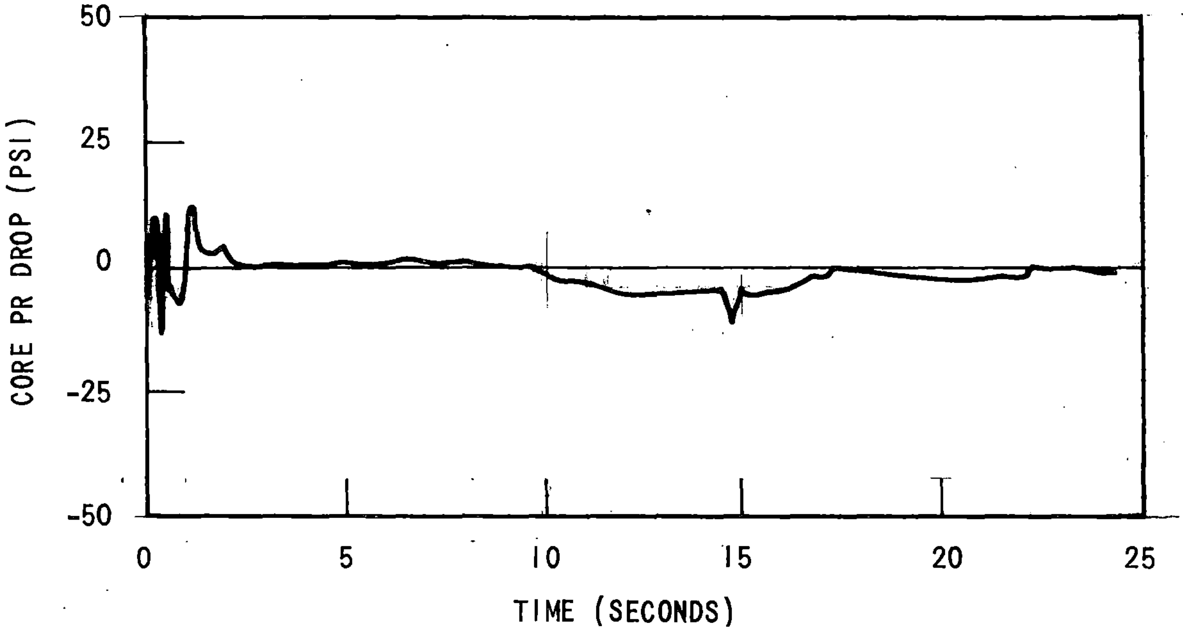
CORE PRESSURE DROP



DECLG (C_D=1.0)

FIGURE III-6b

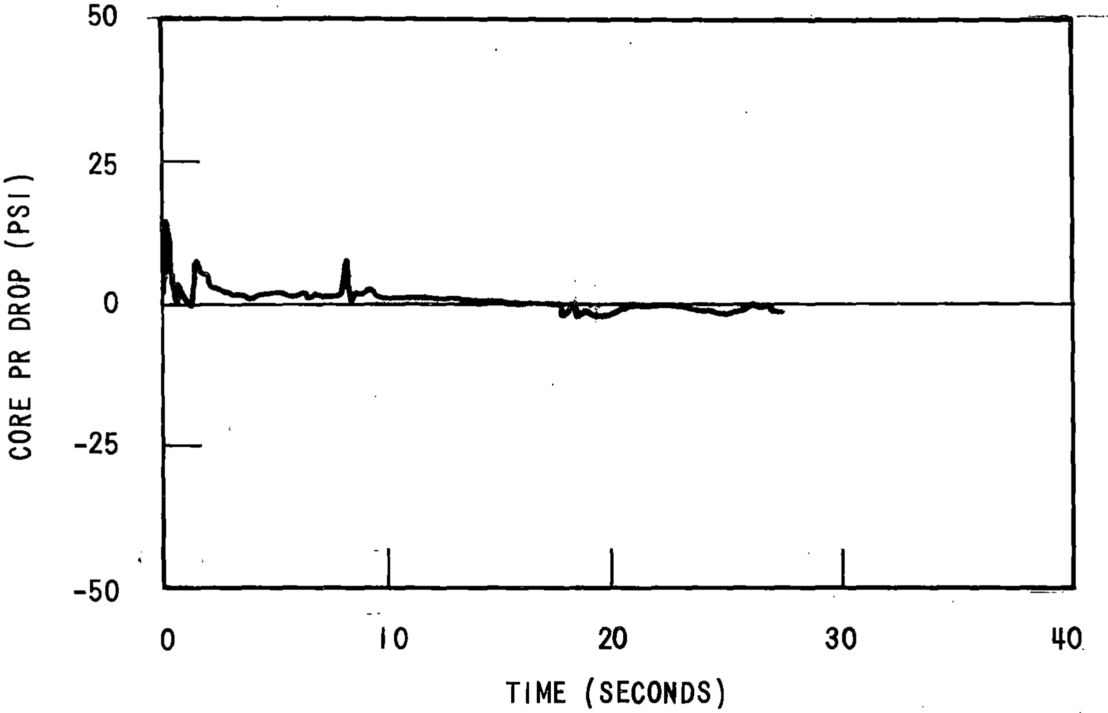
CORE PRESSURE DROP



DECLG (C_D=0.6)

FIGURE III-6c

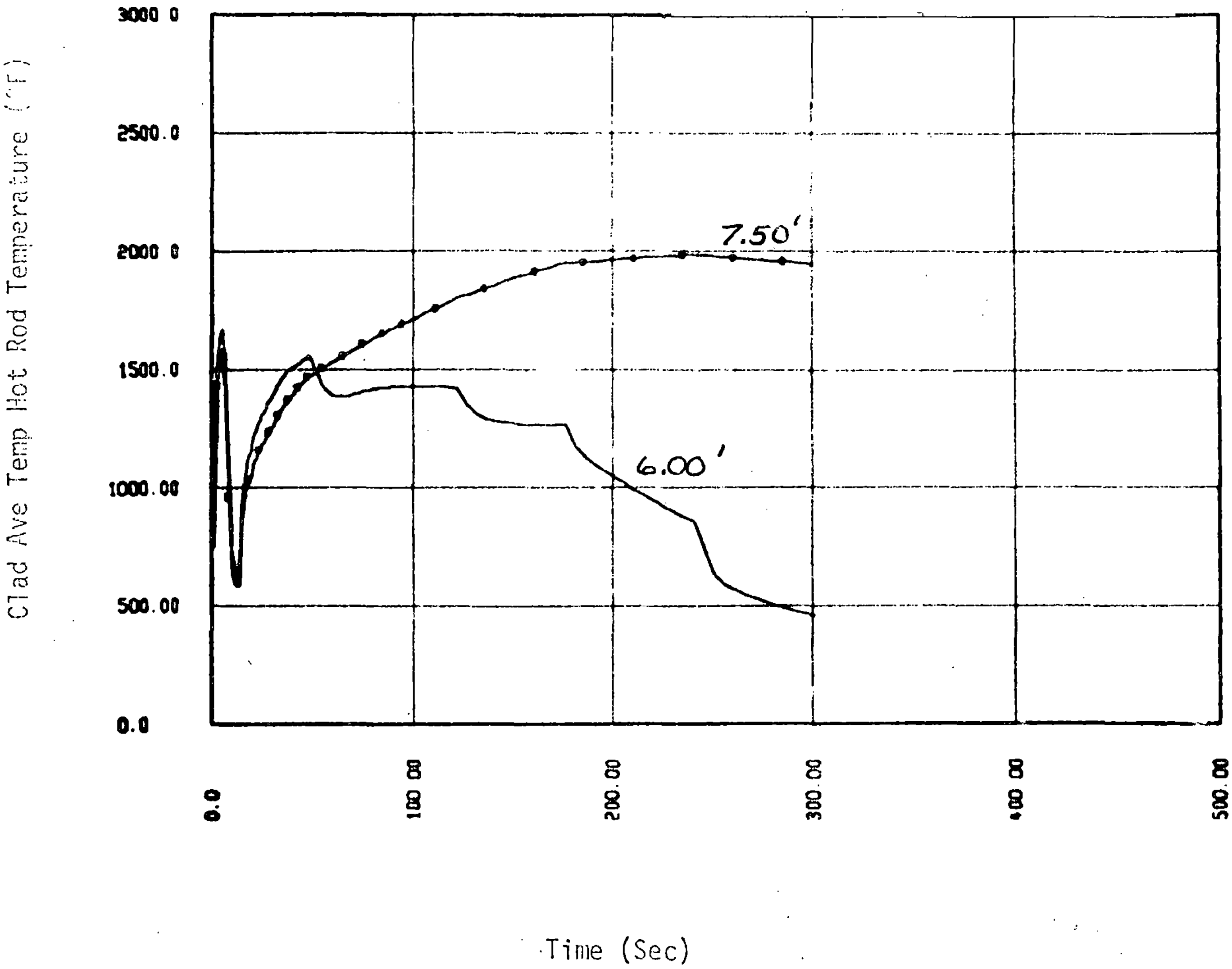
CORE PRESSURE DROP



DECLG (C_D=0.4)

FIGURE III-7a

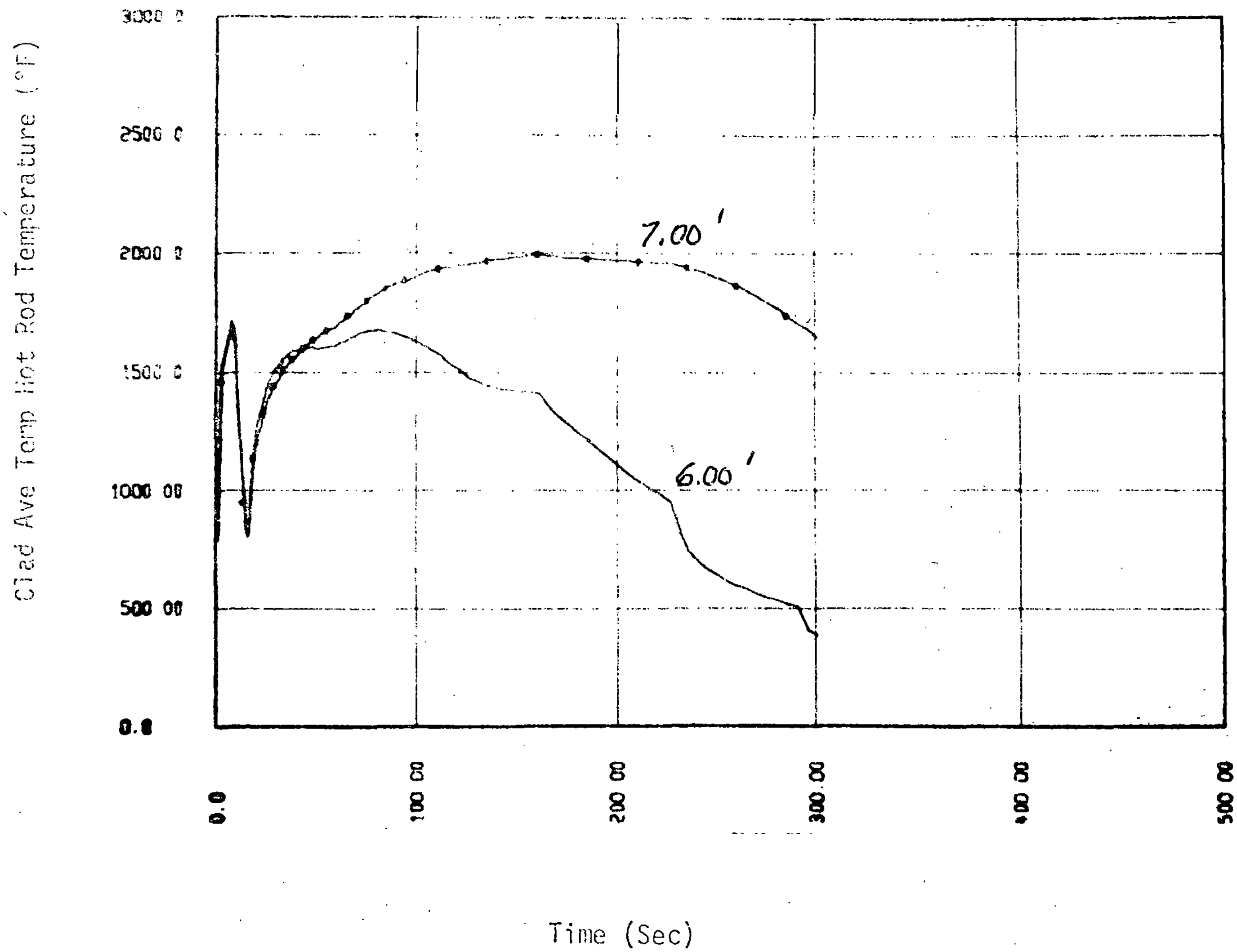
PEAK CLAD TEMPERATURE



DECLG (C_D=1.0)

FIGURE III-7b

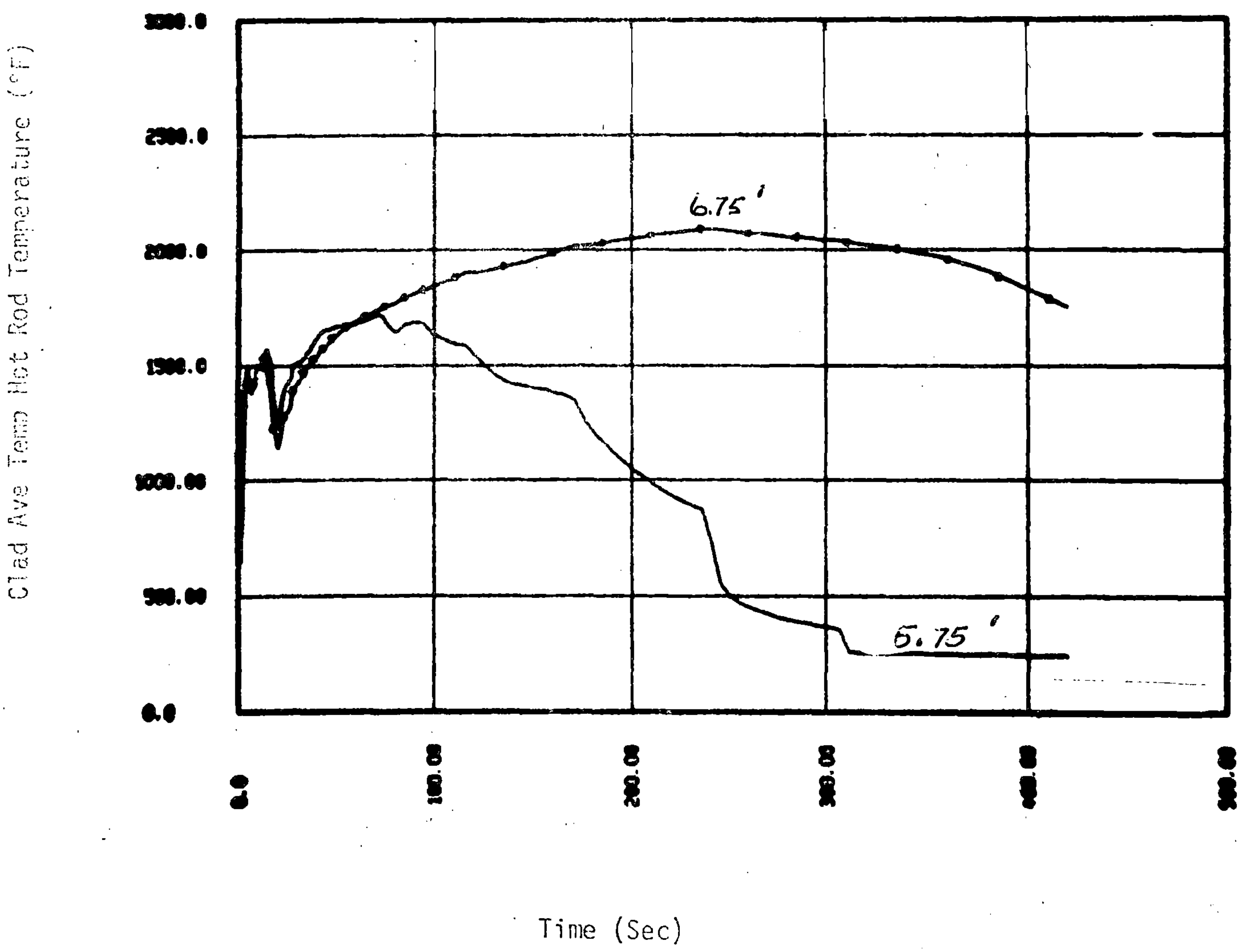
PEAK CLAD TEMPERATURE



DECLG (C_D=0.6)

FIGURE III-7c

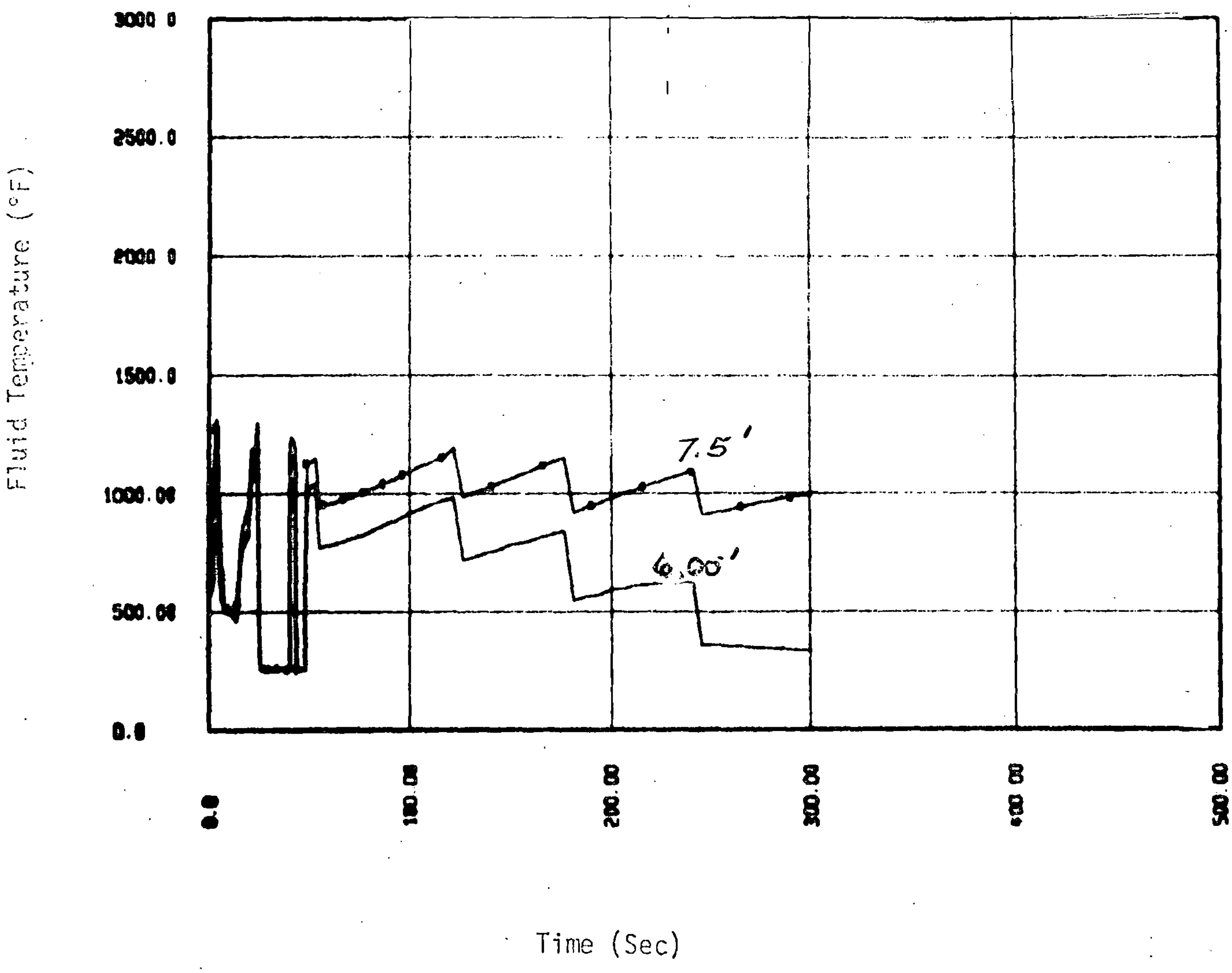
PEAK CLAD TEMPERATURE



DECLG (C_D=0.4)

FIGURE III-8a

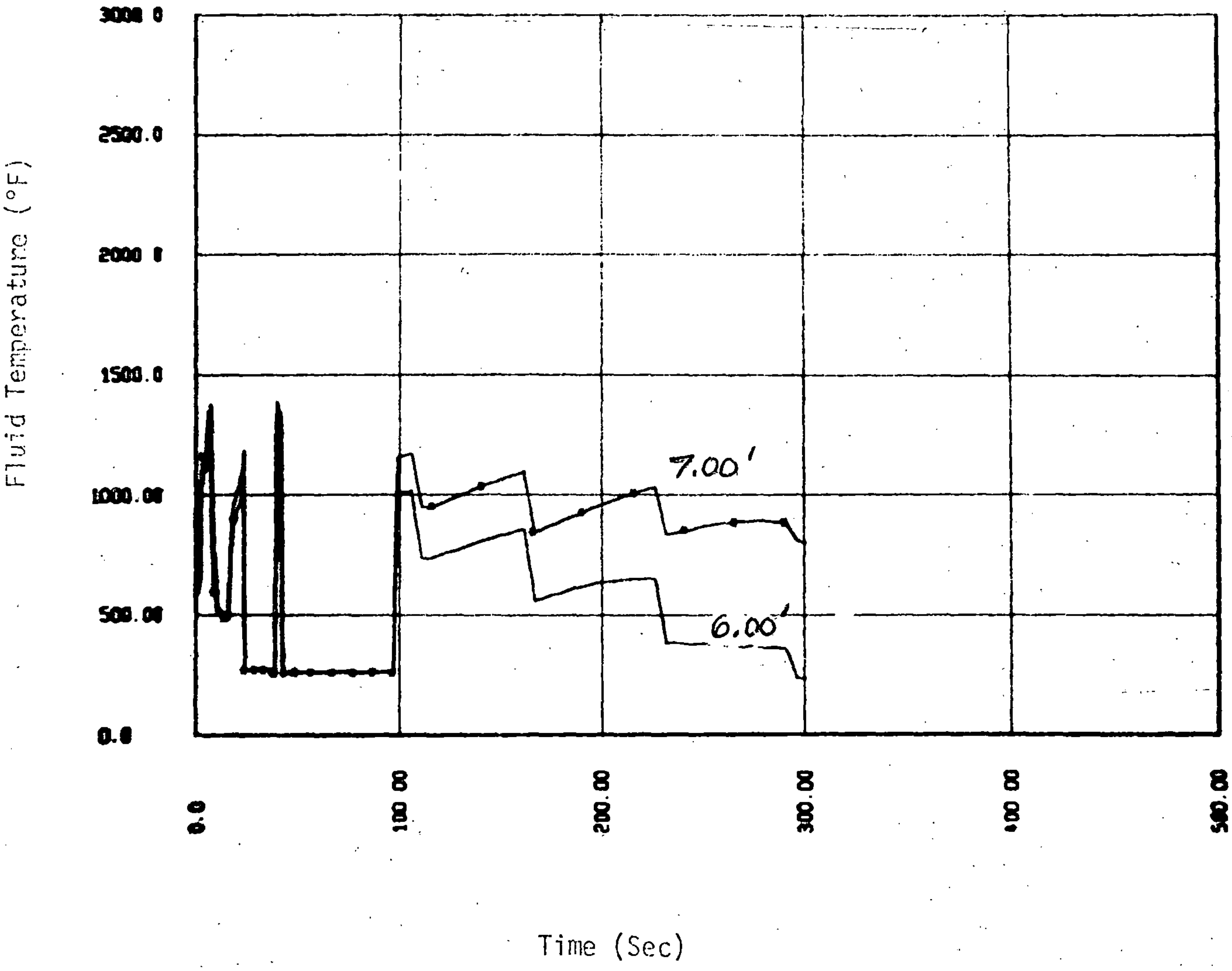
FLUID TEMPERATURE



DECLG (C_D=1.0)

FIGURE III-8b

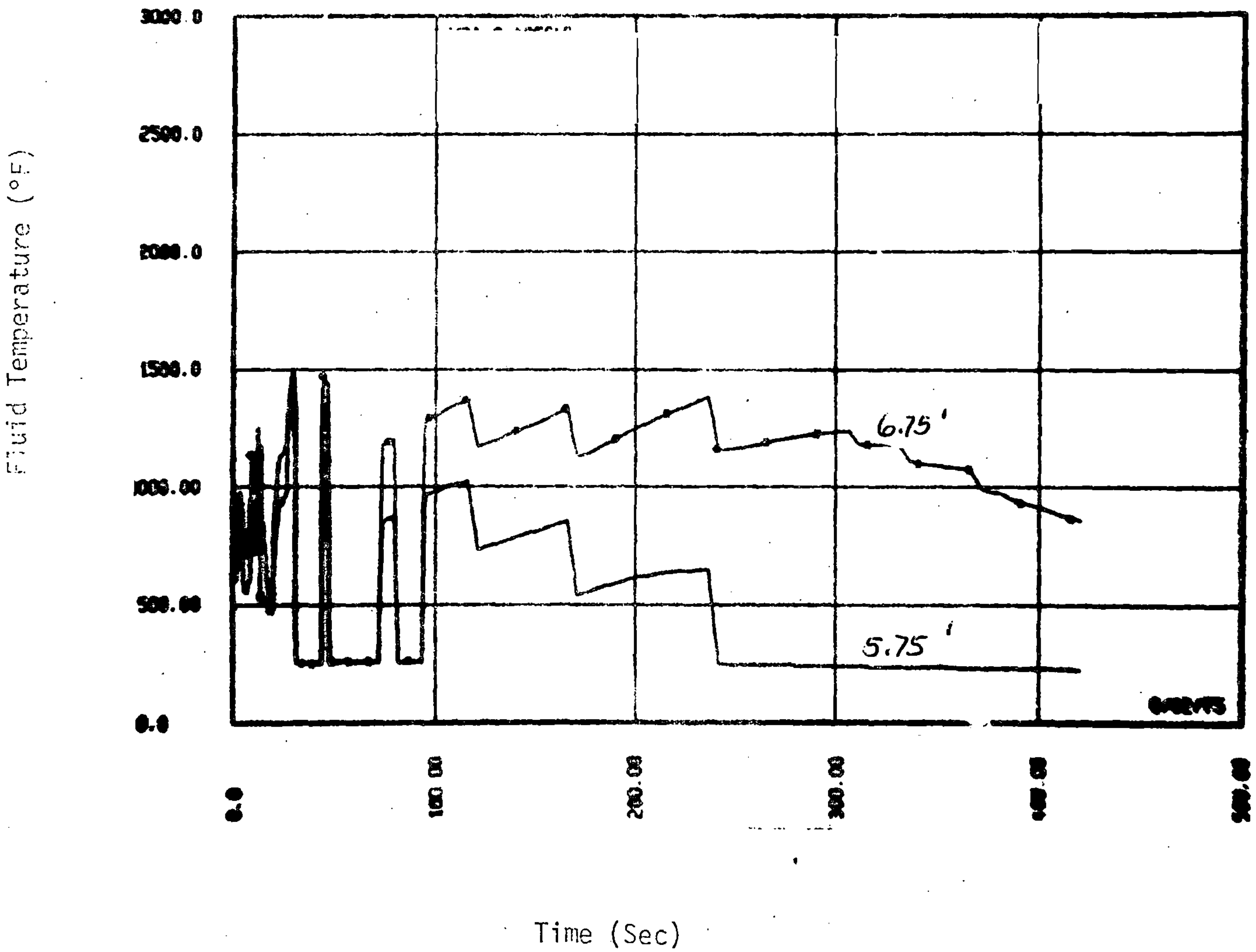
FLUID TEMPERATURE



DECLG (C_D=0.6)

FIGURE III-8c

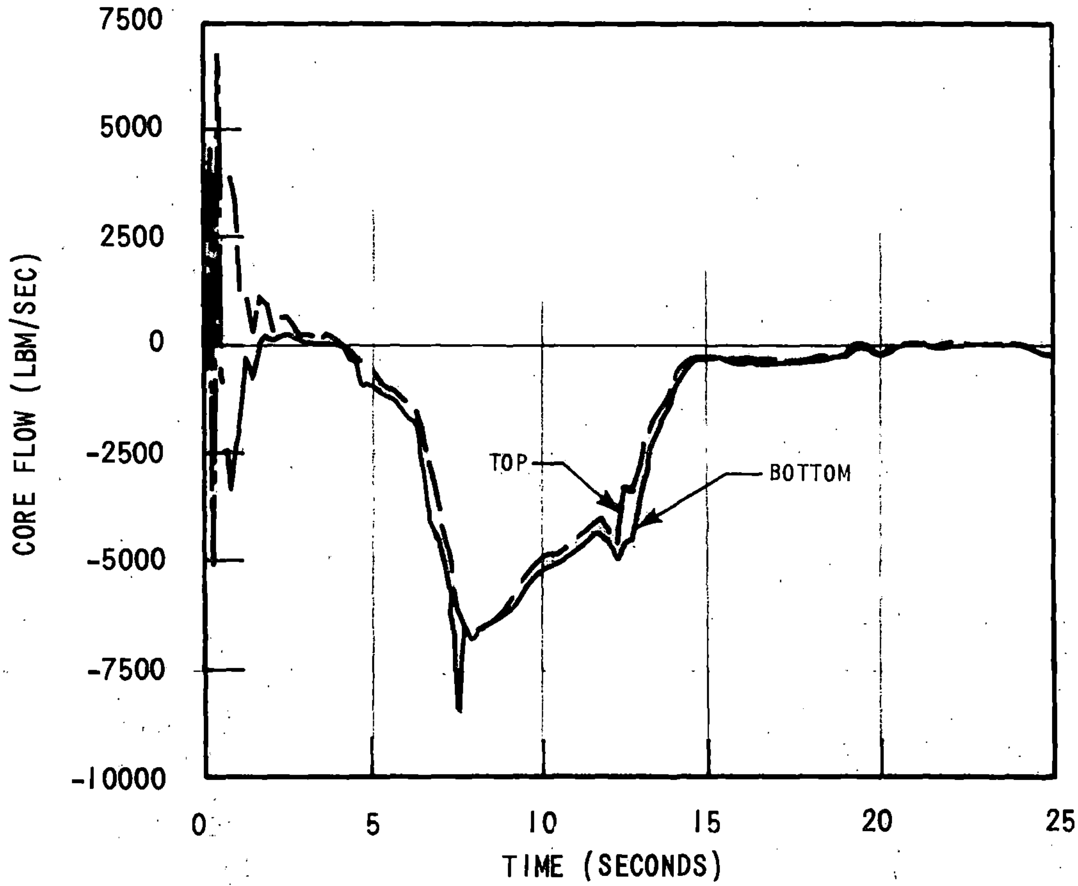
FLUID TEMPERATURE



DECLG (C_D=0.4)

FIGURE III-9a

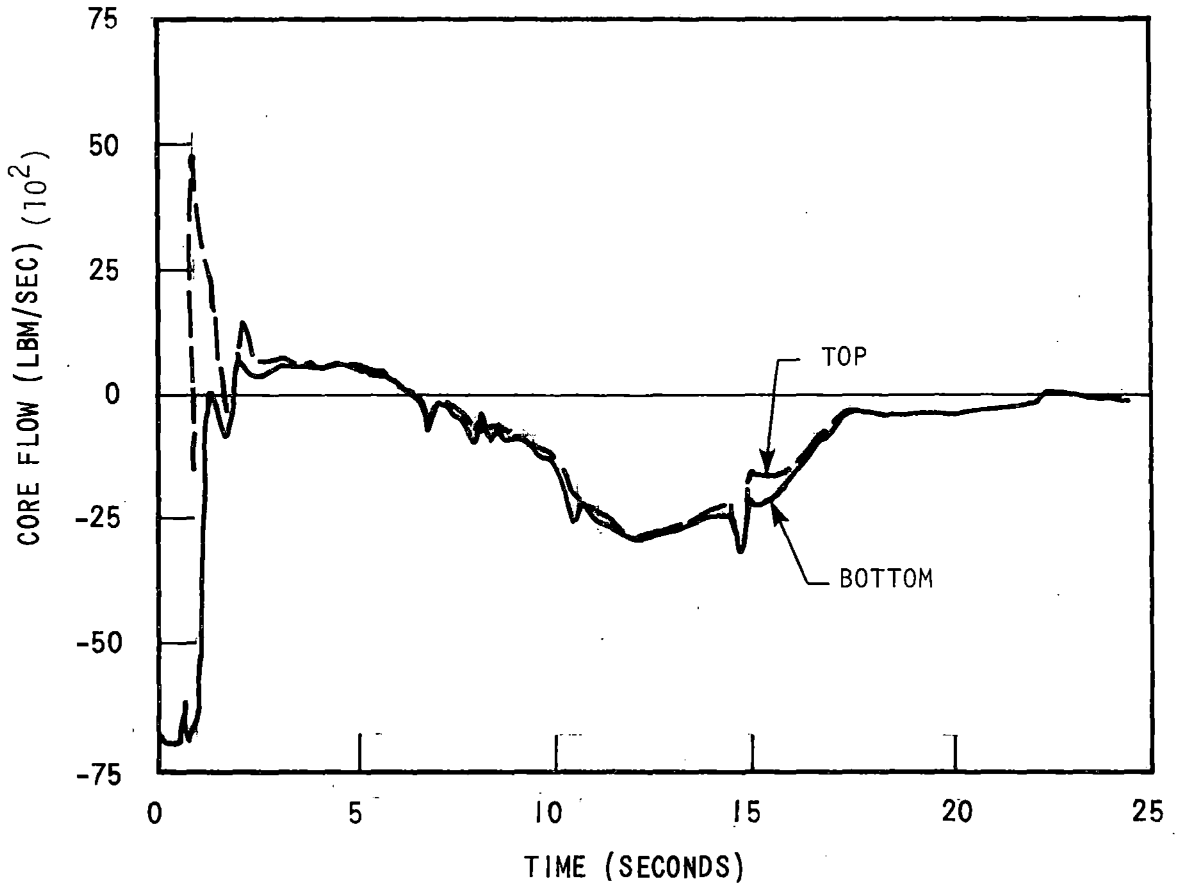
CORE FLOW - TOP AND BOTTOM



DECLG (C_D=1.0)

FIGURE III-9b

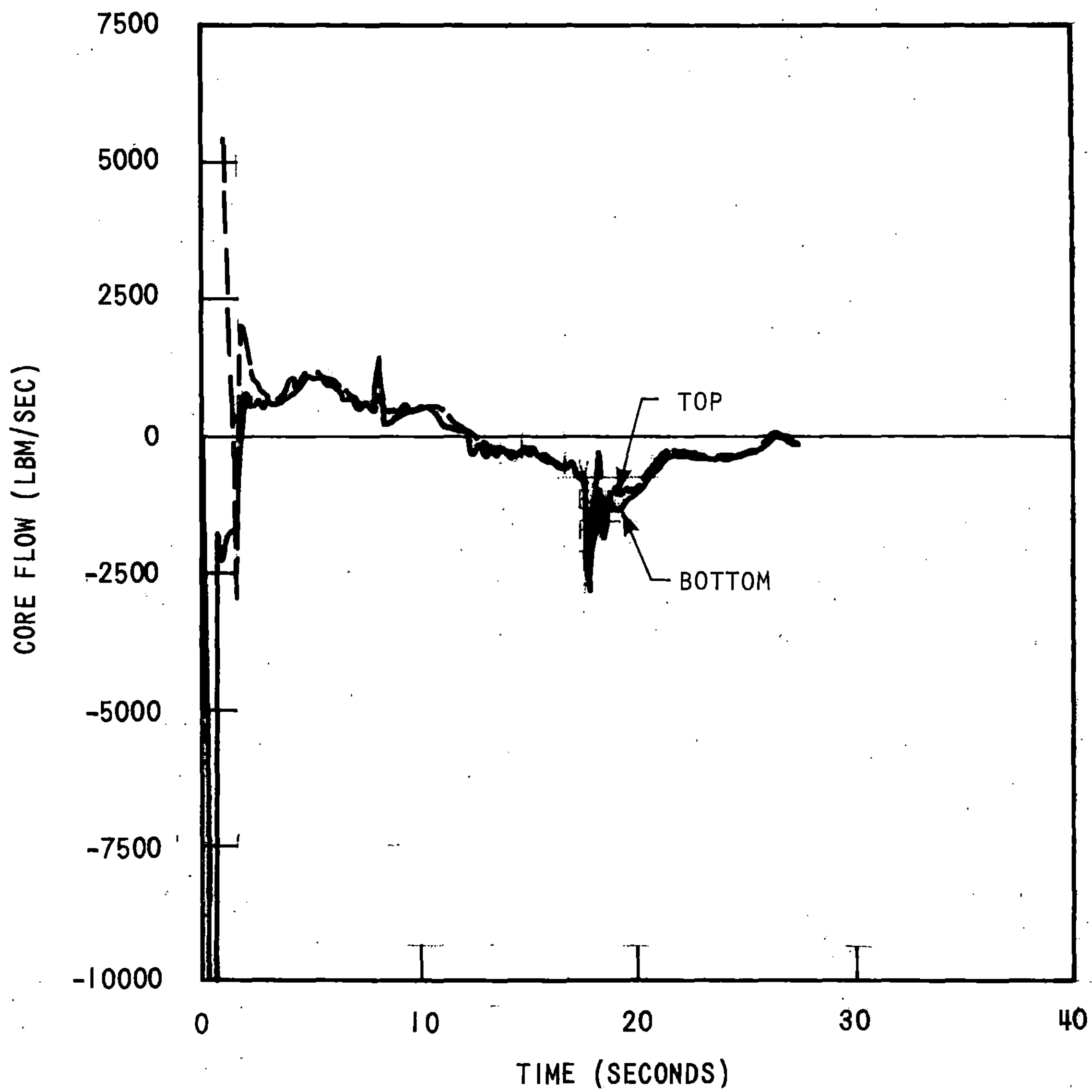
CORE FLOW - TOP AND BOTTOM



DECLG (C_D=0.6)

FIGURE III-9c

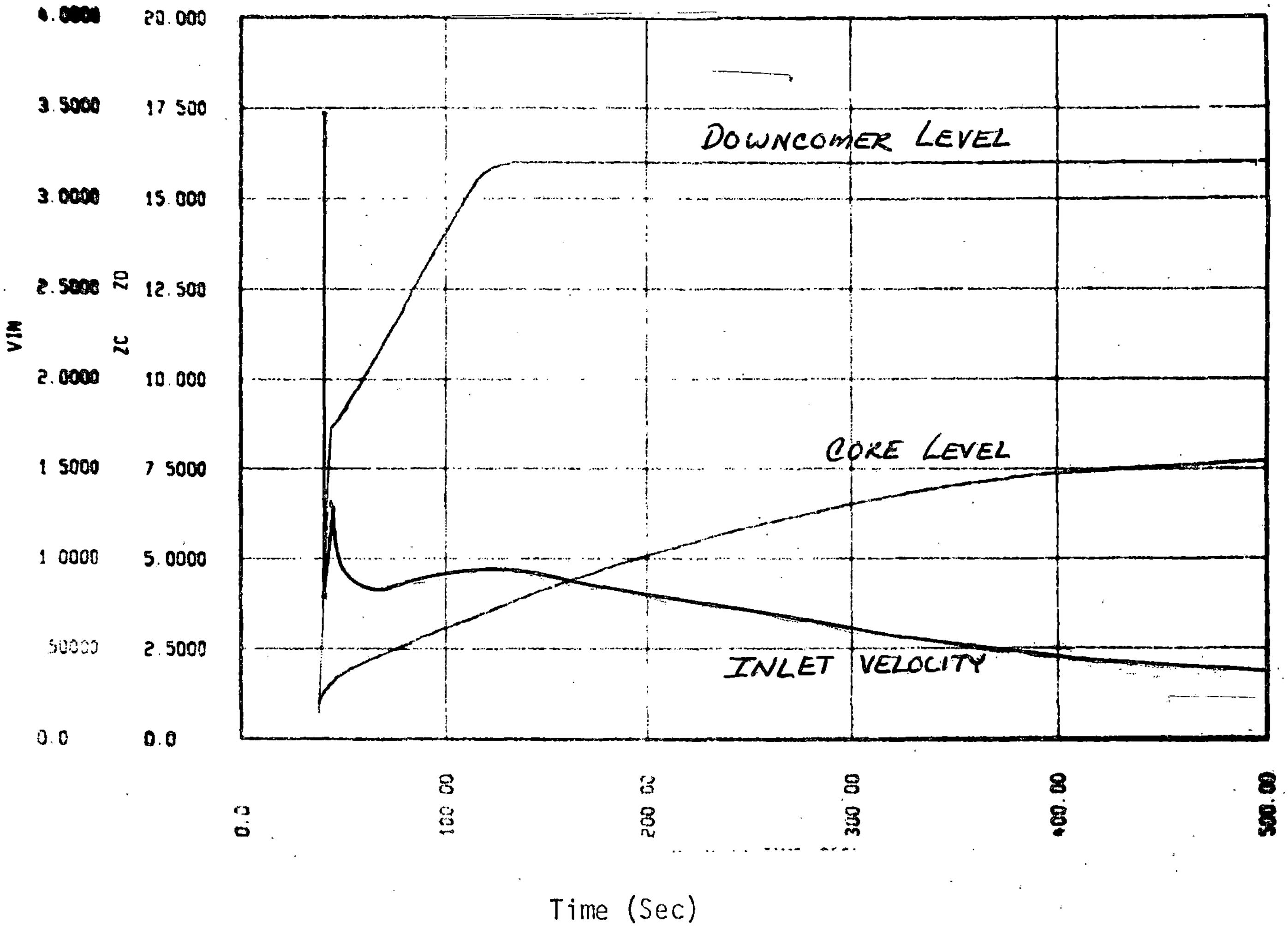
<u>CORE FLOW - TOP AND BOTTOM</u>



DECLG (C_D=0.4)

FIGURE III-10a

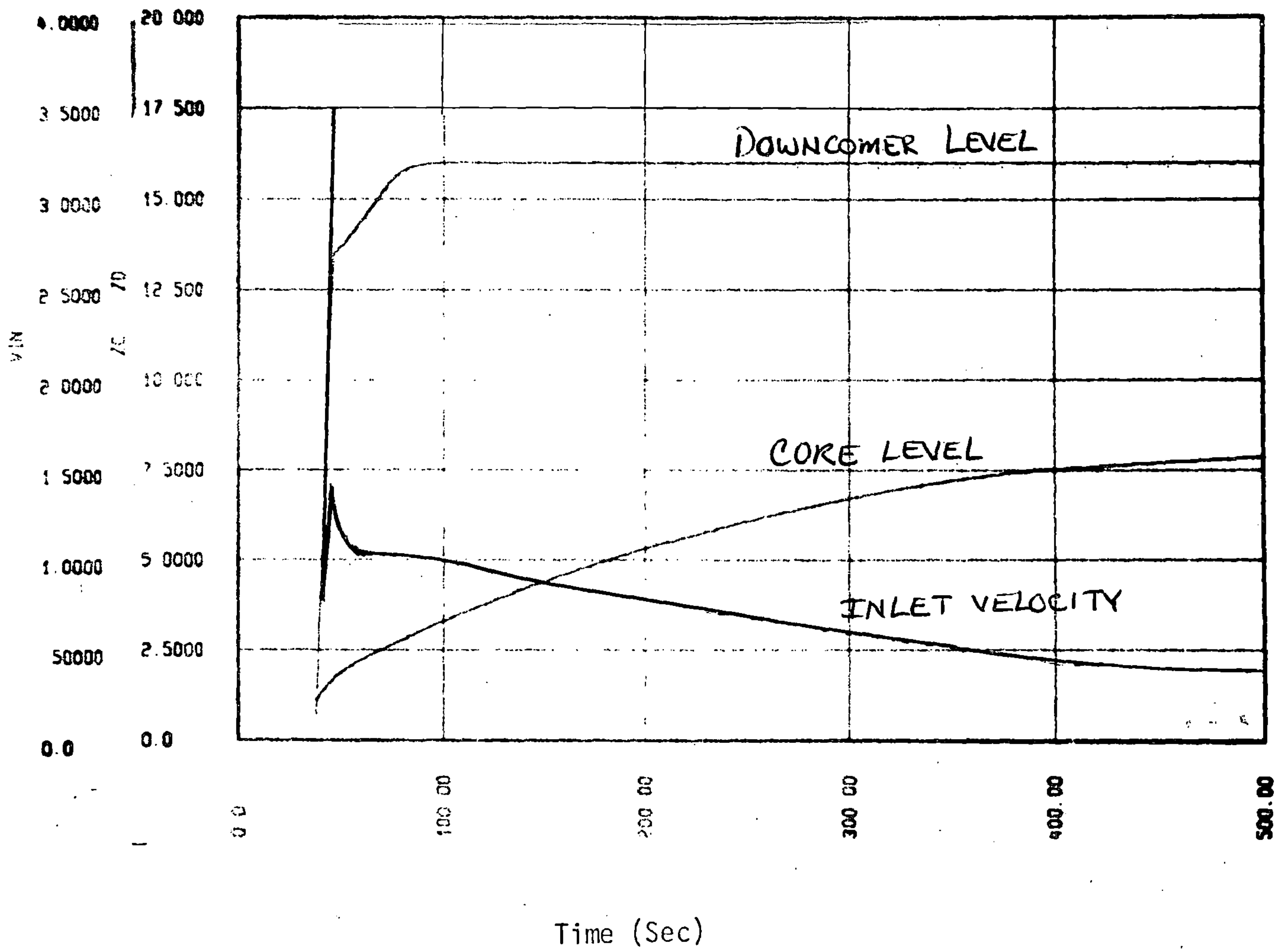
REFLOOD TRANSIENT



DECLG (C_D=1.0)

FIGURE III-10b

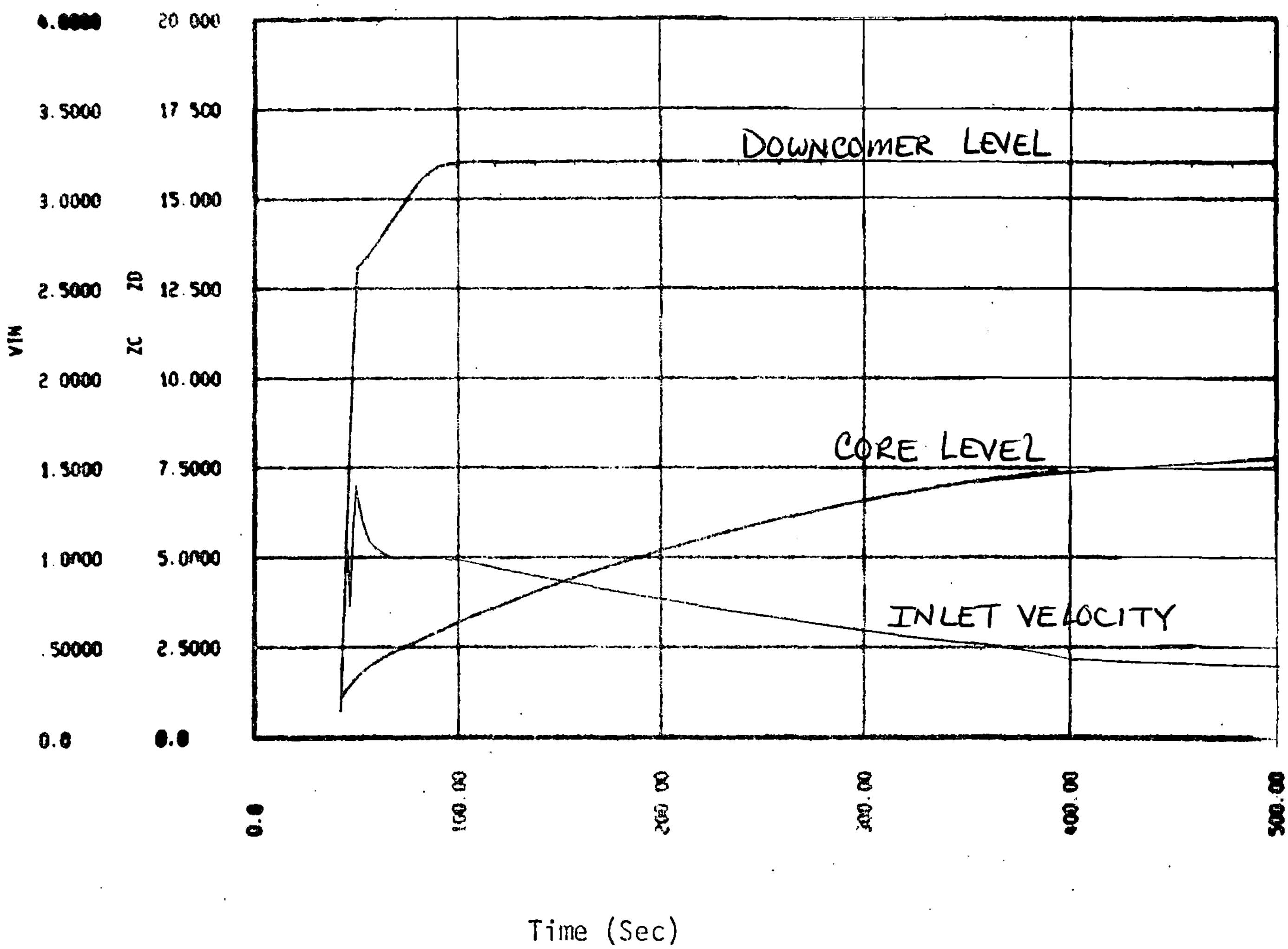
REFLOOD TRANSIENT



DECLG (C_D=0.6)

FIGURE III-10c

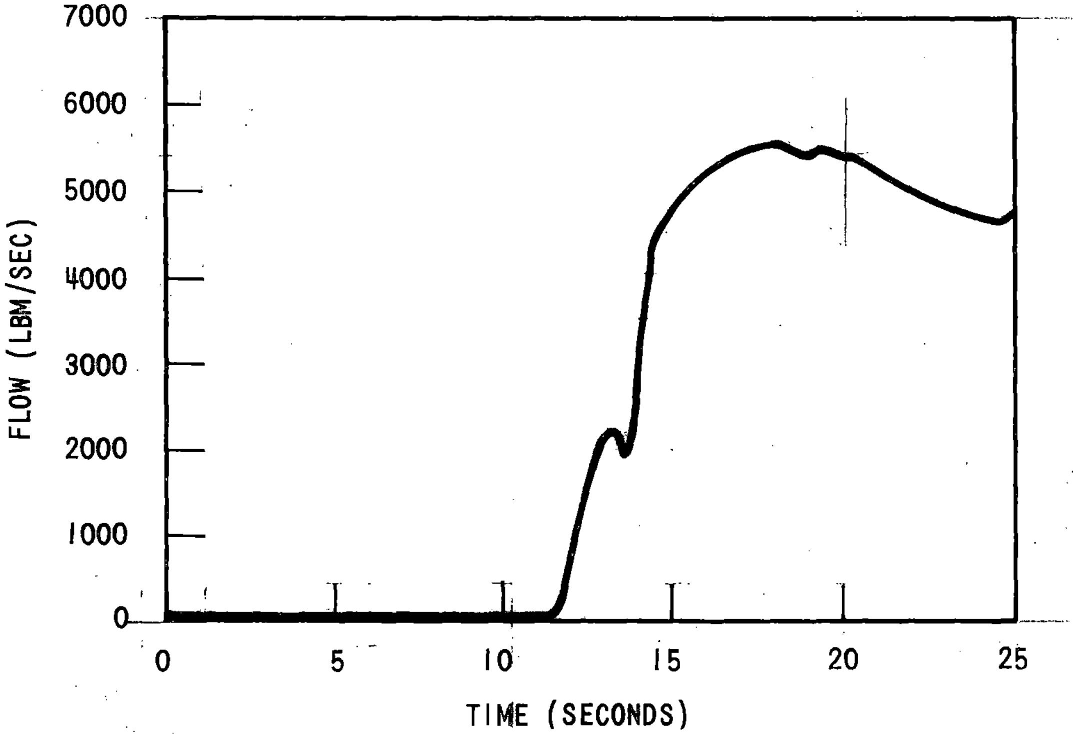
REFLOOD TRANSIENT



DECLG (C_D=0.4)

FIGURE III-11a

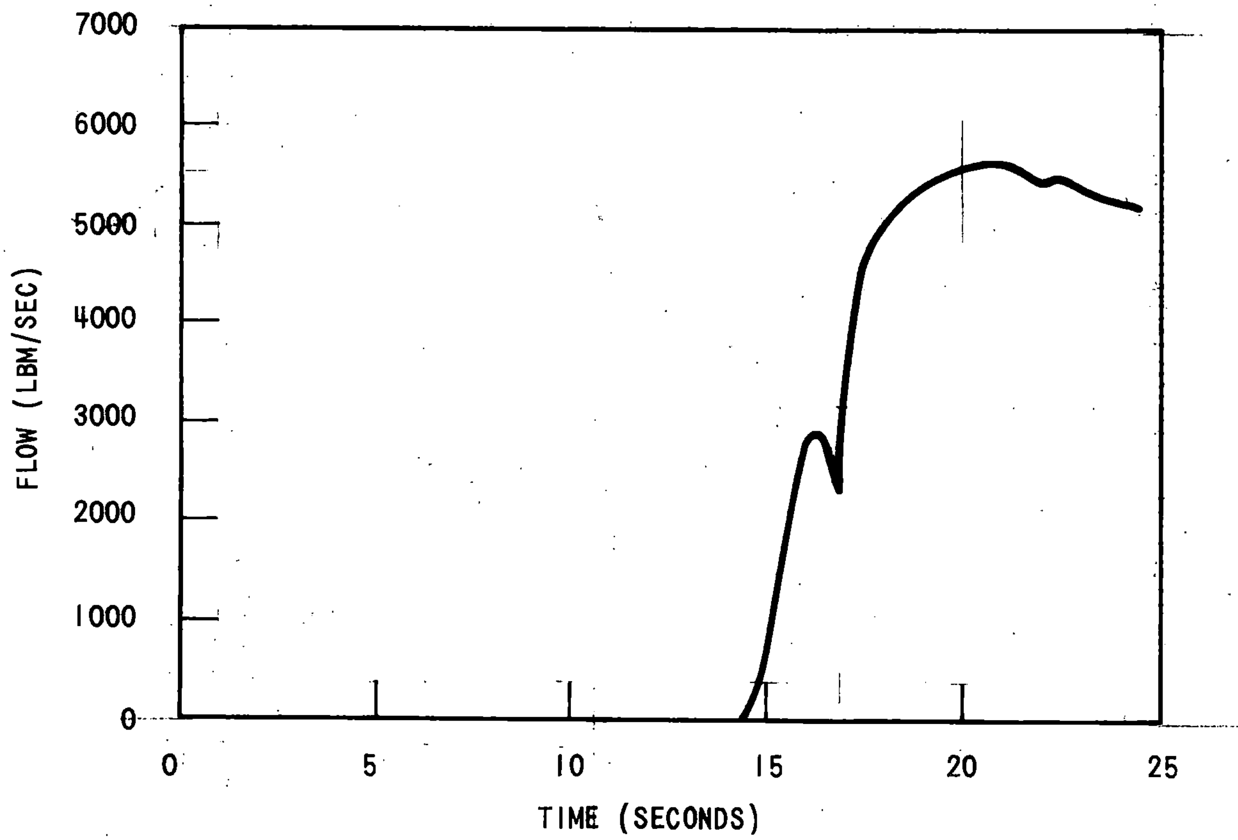
ACCUMULATOR FLOW (BLOWDOWN)



DECLG (C_D=1.0)

FIGURE III-11b

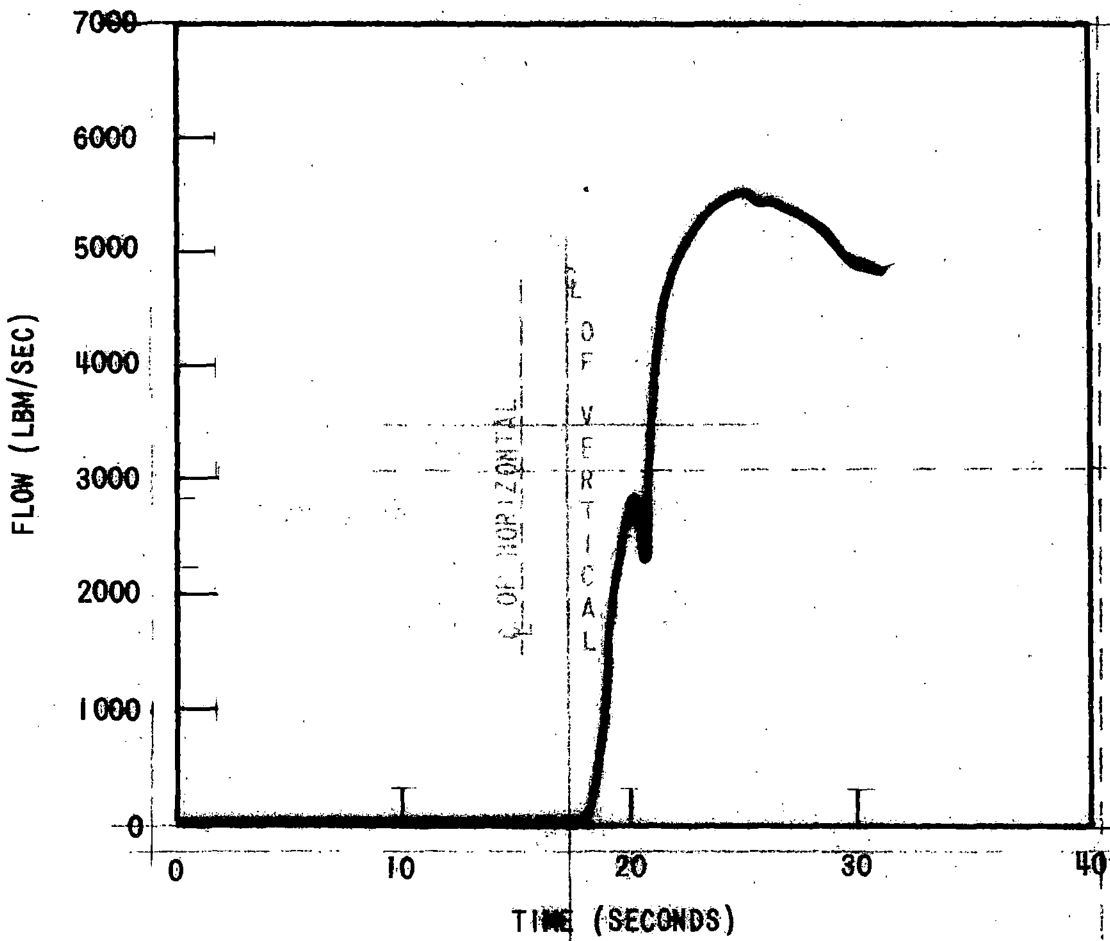
ACCUMULATOR FLOW (BLOWDOWN)



DECLG (C_D=0.6)

FIGURE III-11c

ACCUMULATOR FLOW (BLOWDOWN)



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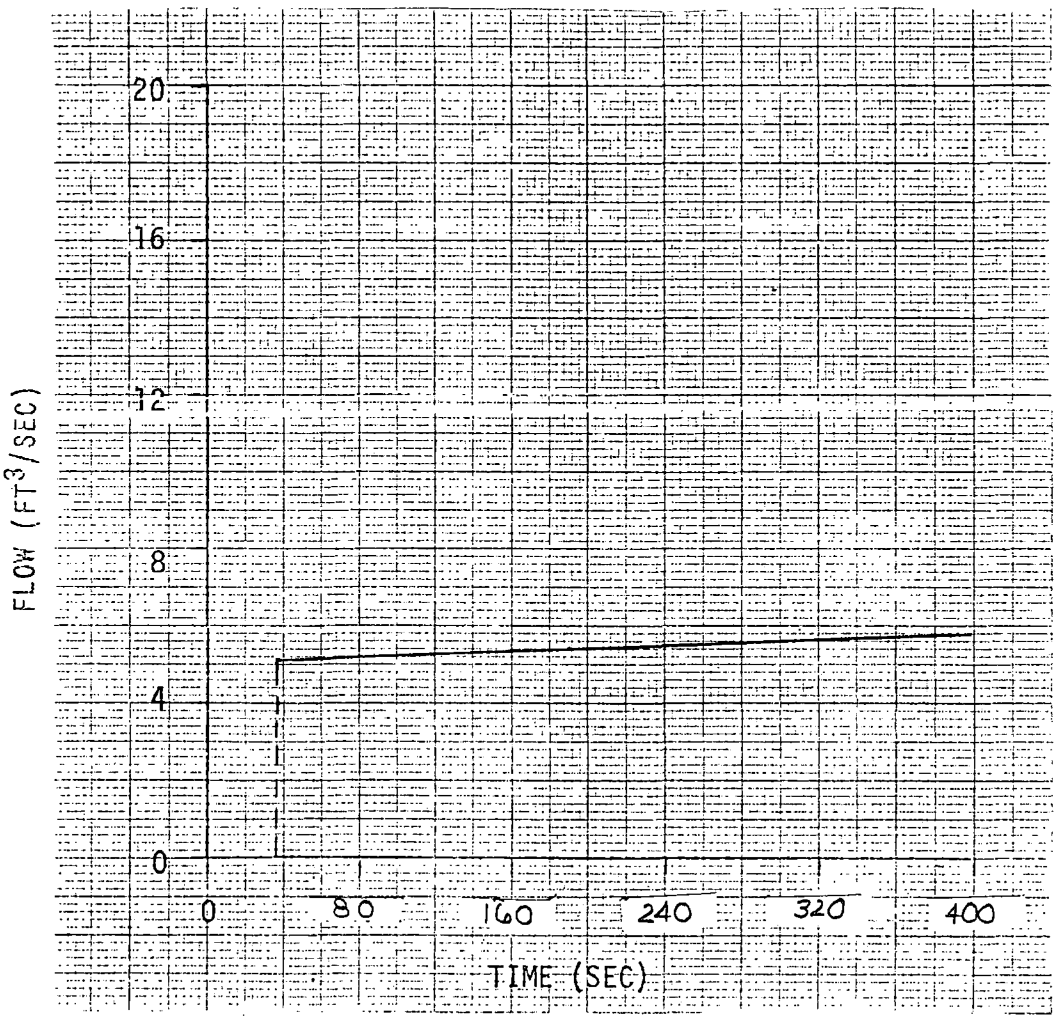
DECLG (C_D=0.4)

FIGURE III-12a

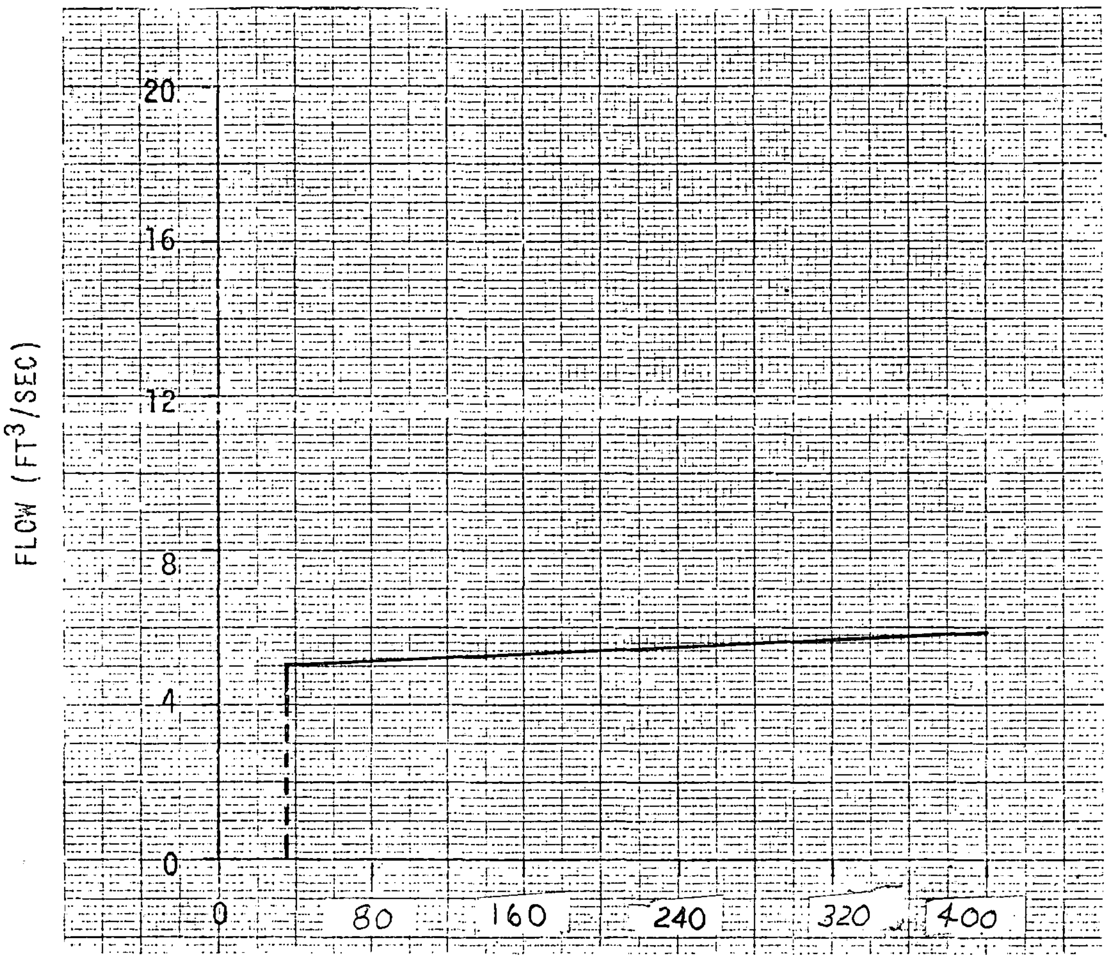
PUMPED ECCS FLOW (REFLOOD)



DECLG (C_D=1.0)

FIGURE III-12b

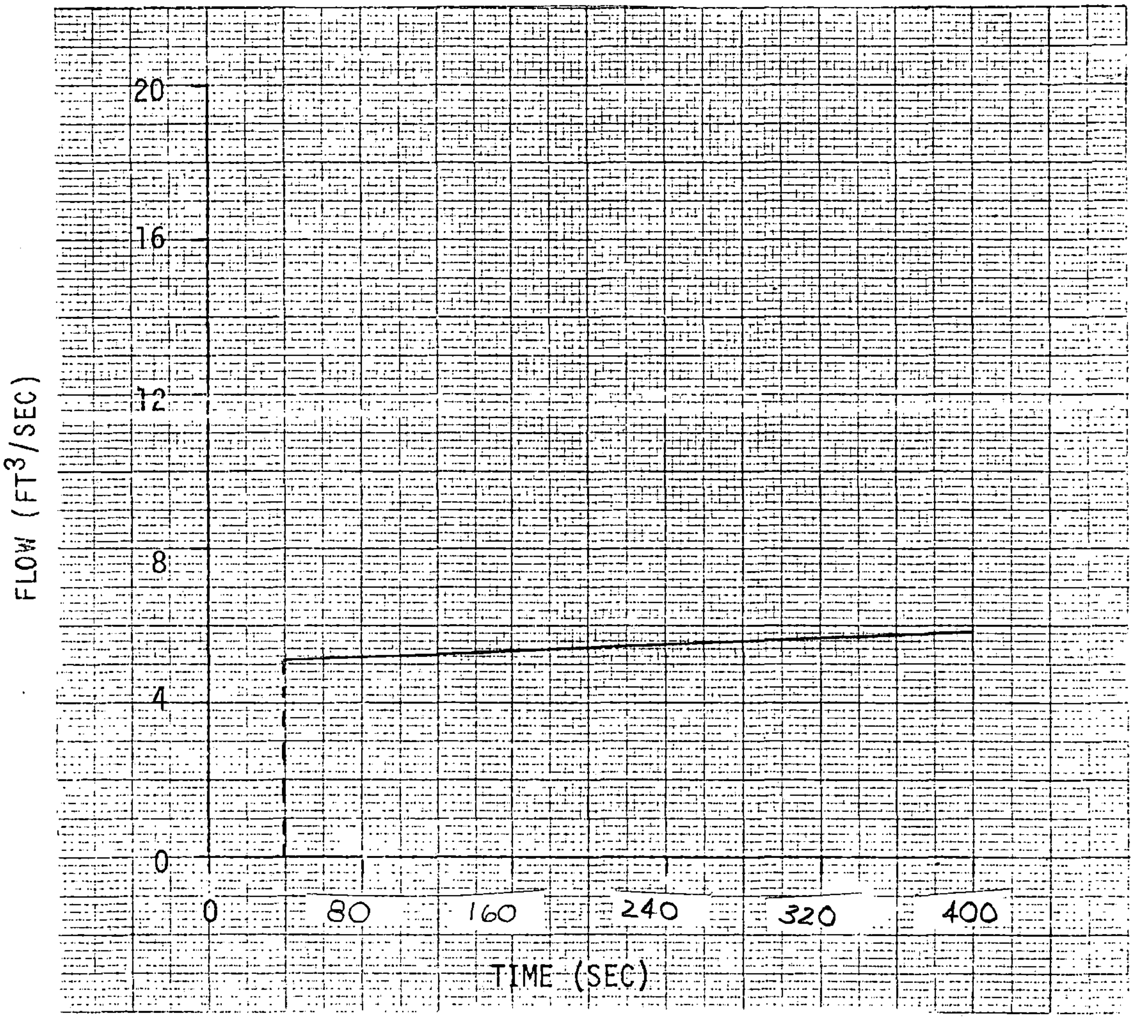
PUMPED ECCS FLOW (REFLOOD)



DECLG (C_D=0.6)

FIGURE III-12c

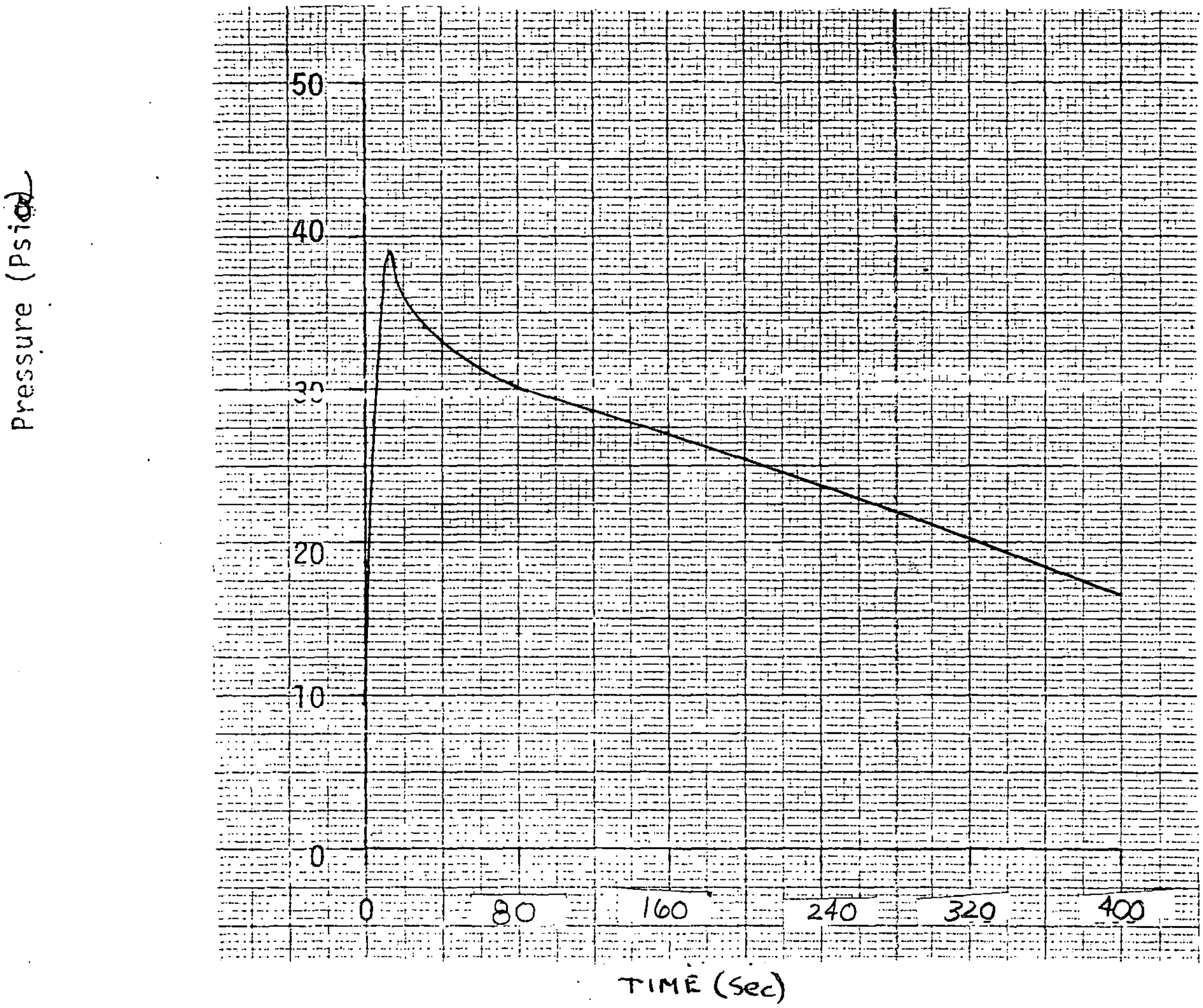
PUMPED ECCS FLOW (REFLOOD)



DECLG (C_D=0.4)

FIGURE III-13a

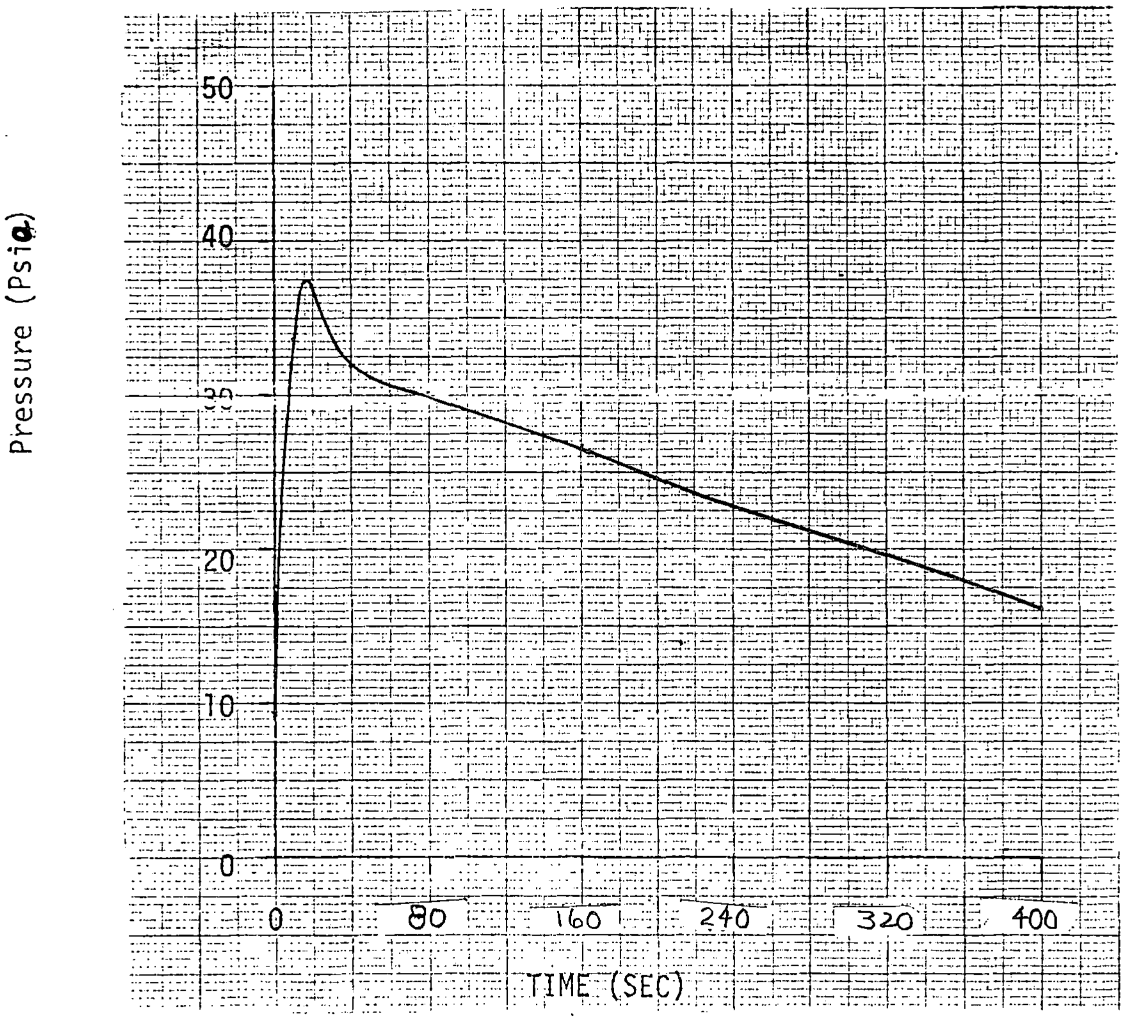
CONTAINMENT PRESSURE



DECLG (C_D=1.0)

FIGURE III-13b

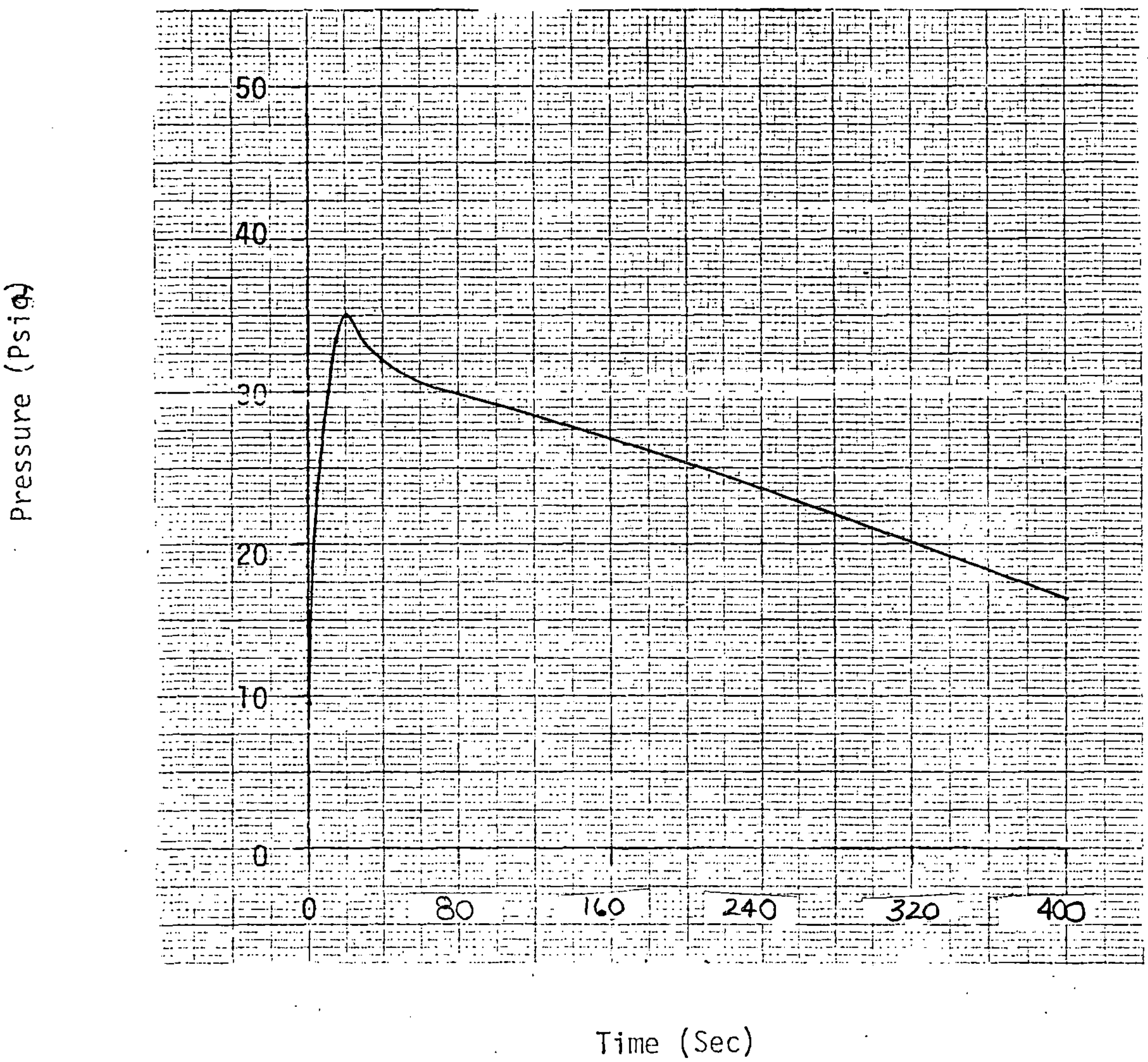
CONTAINMENT PRESSURE



DECLG (C_D=0.6)

FIGURE III-13c

CONTAINMENT PRESSURE



DECLG (C_D=0.4)

FIGURE III-14

SURRY 1, CYCLE 2 MAXIMUM PEAKING FACTOR vs AXIAL CORE HEIGHT

(+6, -9% Delta Flux Band)

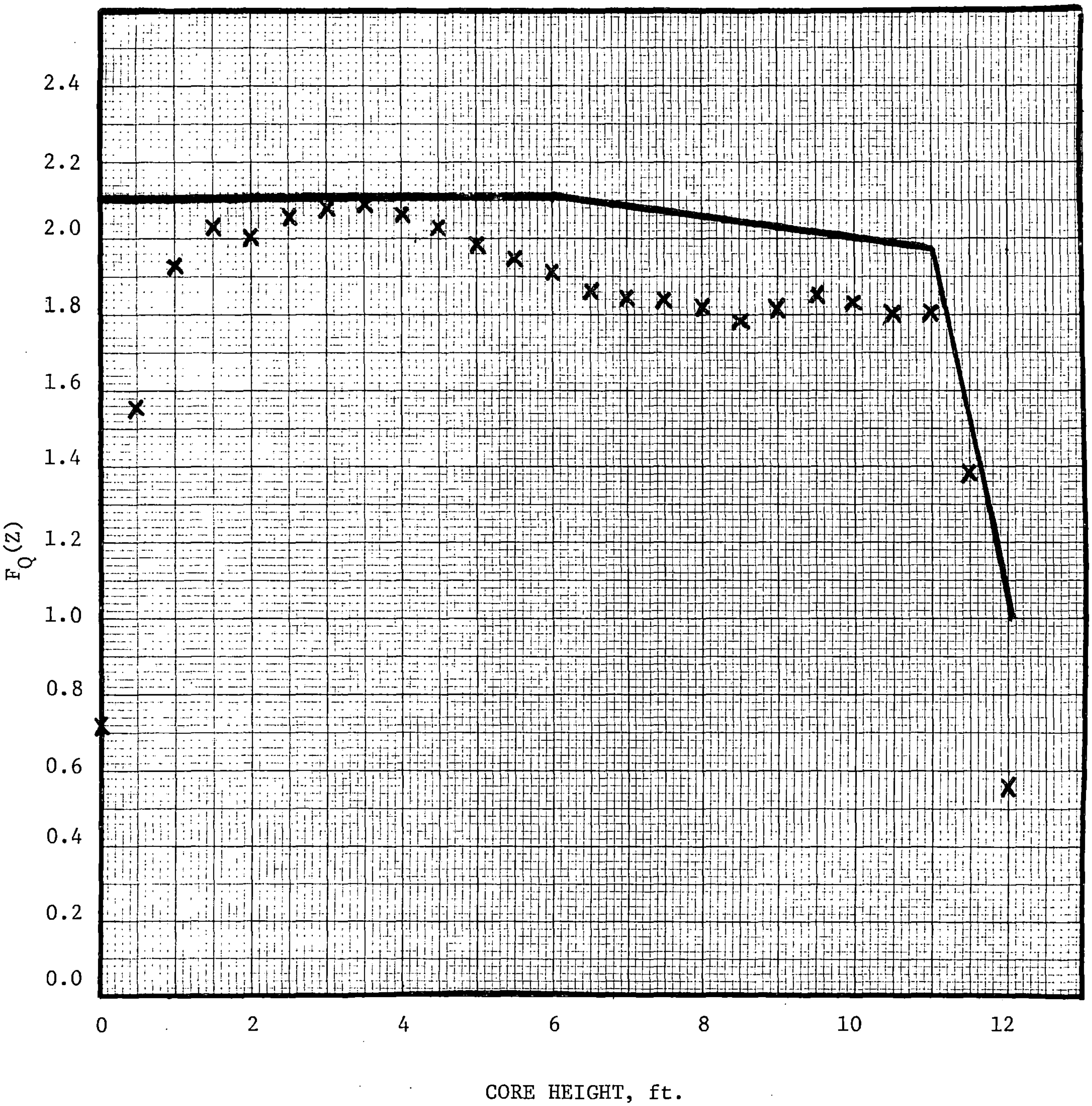
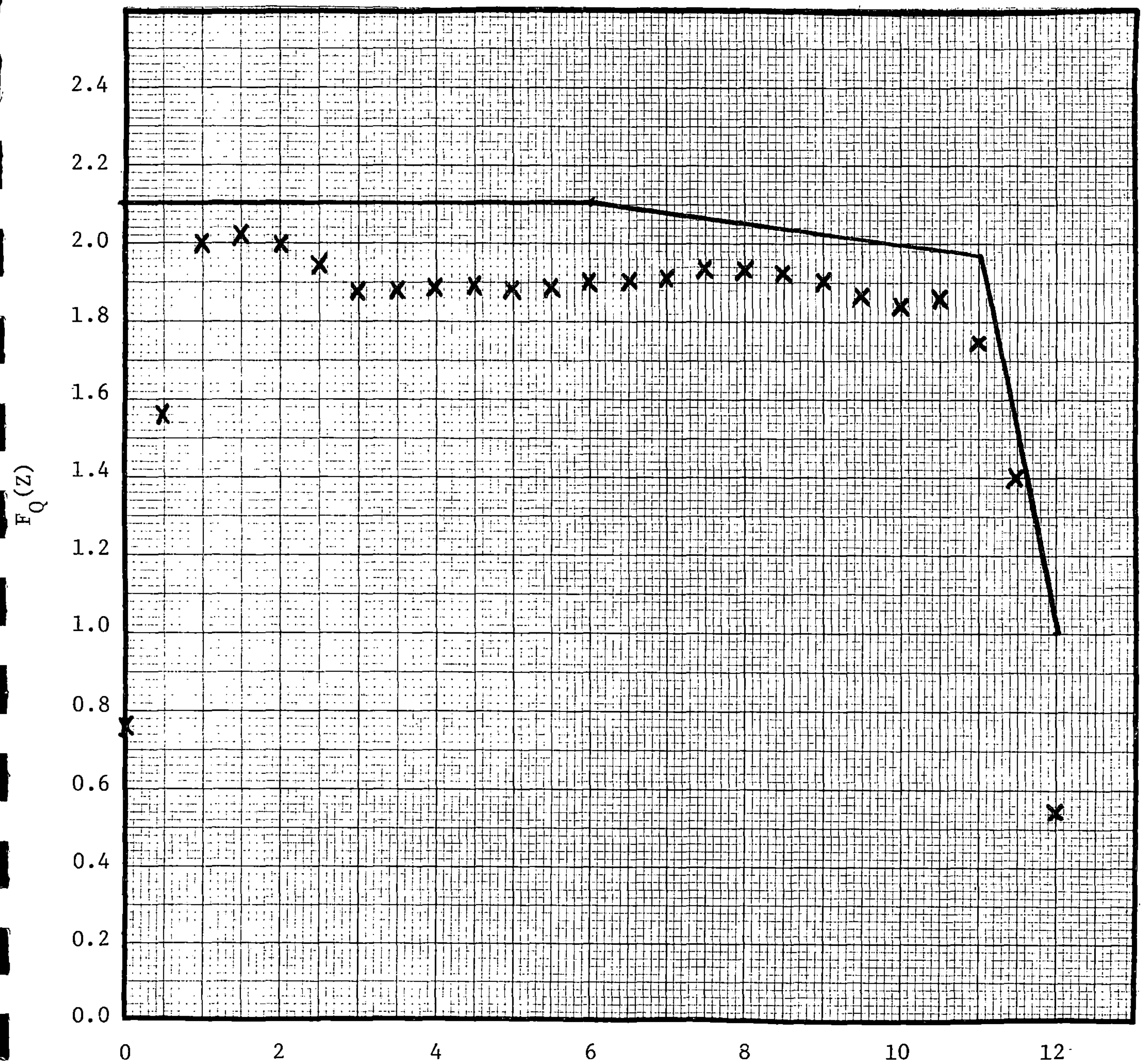


FIGURE III-15

SURRY 2, CYCLE 2 MAXIMUM PEAKING FACTOR vs AXIAL CORE HEIGHT

(+6, -9% Delta Flux Band)



IV. CONCLUSIONS

For breaks up to and including the double ended severance of a reactor coolant pipe, the emergency core cooling system meets the Acceptance Criteria as presented in 10 CFR 50.46. Specifically,

1. Peak Cladding Temperature

   The calculated peak fuel element clad temperature provides margin to the requirement of 2200 degrees F.

2. Cladding Oxidation

   The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17 per cent are not exceeded during or after quenching.

3. Hydrogen Generation

   The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 per cent of the total amount of Zircaloy in the reactor.

4. Coolable Geometry

   The core remains amenable to cooling during and after the break.

5. Long-term Cooling

   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 by the safety injection system.

V. REFERENCES

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors" 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3 January 4, 1974.

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

3. Bordelon, F.M., et al., "SATAN-IV Program: Comprehensive Space-Time Department Analysis of Loss-of-Coolant," WCAP-8306, June 1974.

4. Bordelon, F.M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974.

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

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

7. Buterbaugh, T.L., Johnson, W.J. and Kopelic, S.D., "Westinghouse ECCS-Plant Sensitivity Studies," WCAP-8356, July 1974.

8. Federal Register, "Supplement to the Status Report by the Directorate of Licensing in the matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR 50, 'Appendix K,'" November 1974.

9. Morita, T., et al., "Power Distribution Control and Load Following Procedures," WCAP-8385, September 1974.

10. Esposito, V.J., et al., "WFLASH-Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8261, July 1974.

11. WCAP-8219, "Fuel Densification Experimental Results and Model for Reactor Applications," October 1973.

12. WCAP-8342, "Westinghouse Emergency Core Cooling System Evaluation Model-Sensitivity Studies," July 1974.

13. Bordelon, F.M., et al., "Westinghouse ECCS Evaluation Model-Supplemental Information," WCAP-8472, April 1975.

14. Letter from D.B. Vassallo of the Nuclear Regulatory Commission to C. Eicheldinger of Westinghouse Electric Corporation dated May 30, 1975.

15. WCAP-8472, "Westinghouse ECCS Evaluation Model, Supplementary Information."

VI. PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS

The proposed Technical Specification changes contained herein are designated as Change No. 29. The proposed changes are denoted by a heavy black line in the righthand margin. The operation of Unit Nos. 1 and 2, Surry Power Station, in accordance with these specifications will assure the validity of the ECCS analysis presented herein.

L. Low Power Physics Tests

Low power physics tests are tests conducted below 5% of rated power which measure fundamental characteristics of the reactor core and related instrumentation.

(Deleted)

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 psig 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 portions 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

3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rod assemblies and power distribution limits.

Objective

To ensure core subcriticality after a reactor trip, a limit on potential reactivity insertions from hypothetical control rod assembly ejection, and an acceptable core power distribution during power operation.

Specification

A. Control Bank Insertion Limits

1. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the shutdown control rods shall be fully withdrawn.

2. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the full length control rod banks shall be inserted no further than the appropriate limit determined by core burnup shown on TS Figures 3.12-1A, 3.12-1B, 3.12-2, or 3.12-3 for three-loop operation and TS Figures 3.12-4A, 3.12-4B, 3.12-5, or 3.12-6 for two-loop operation.

3. The limits shown on TS Figures 3.12-1A through 3.12-6 may be revised on the basis of physics calculations and physics data obtained during unit startup and subsequent operation, in accordance with the following:

   a. The sequence of withdrawal of the controlling banks, when going from zero to 100% power, is A, B, C, D.

   b. An overlap of control banks, consistent with physics calculations and physics data obtained during unit startup and subsequent operation, will be permitted.

   c. The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown on TS Figure 3.12-7 under all steady-state operation conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions ($T_{avg} \geq 547^{o}F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

4. Whenever the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod position at which criticality would be achieved if the control rod assemblies were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

5. Operation with part length rods shall be restricted such that except during physics tests, the part length rod banks are withdrawn from the core at all times.

6. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in TS Figure 3.12-7 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod, expected to have the highest worth, inserted and part length rods fully withdrawn.

B. Power Distribution Limits

1. At all times except during physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.10/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.20) \times K(Z)$ 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 3.12-8, and Z is the core height location of F_Q.

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^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

   b. The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error.

If either 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 $F_Q \leq 2.10 \times K(Z)$ and $F_{\Delta H}^N \leq 1.55$ within 24 hours, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

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 during physics tests, during excore detector calibration and except as modified by 3.12.B.4.a., b., or c. below, the indicated axial flux difference shall be maintained within a +6 to -9% band about the target flux difference (defines the target band on axial flux difference).

   a. At a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band, or the reactor power shall immediately be reduced to a level no greater than 90 percent of rated power.

b. At a power level no greater than 90 percent of rated power,

(1) The indicated axial flux difference may deviate from its +6 to -9% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -18 percent and +11.5 percent at 90% power. For every 4 percent below 90% power, the permissible positive flux difference boundary is extended by 1 percent. For every 5 percent below 90% power, the permissible negative flux difference boundary is extended by 2 percent.

(2) If 3.12.B.4.b.(1) is violated then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint shall be reduced to no greater than 55% power.

(3) A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

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 two hours (cumulative) out of the preceding 24 hour period. One half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributed to the one hour cumulative maximum the flux difference maximum deviate from its target band at a power level less than or equal 90 percent of rated power.

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. 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.

5. The allowable quadrant to average power tilt is

$$T = 2.0 + 50\ (1.435/F_{xy} - 1) \leq 10\%$$

where F_{xy} is 1.435, or the value of the unrodded horizontal plane peaking factor appropriate to F_Q as determined by a movable incore detector map taken on at least a monthly basis; and T is the percentage operating quadrant tilt limit, having a value of 2% if F_{xy} is 1.435 or a value up to 10% if the option to measured F_{xy} is in effect.

6. If the quadrant to average power tilt exceeds a value T% as selected in 3.12.B.5, except for physics and rod exercise testing, 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 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 $\pm$10% except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

7. If after a further period of 24 hours, the power tilt in 3.12.B.6 above is not corrected to less than +T%:

   a. If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as an abnormal occurrence to the Nuclear Regulatory Commission.

b. If the design hot channel factors for rated power are exceeded and the power is greater than 10%, the Nuclear Regulatory Commission shall be notified and the nuclear overpower, overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factor exceeds the rated power design values.

c. If the hot channel factors are not determined the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

C. Inoperable Control Rods

1. A control rod assembly shall be considered inoperable if the assembly cannot be moved by the drive mechanism, or the assembly remains misaligned from its bank by more than 15 inches. A full-length control rod shall be considered inoperable if its rod drop time is greater than 1.8 seconds to dashpot entry.

2. No more than one inoperable control rod assembly shall be permitted when the reactor is critical.

3. If more than one control rod assembly in a given bank is out of service because of a single failure external to the individual rod

drive mechanisms, i.e. programming circuitry, the provisions of 3.12.C.1 and 3.12.C.2 shall not apply and the reactor may remain critical for a period not to exceed two hours provided immediate attention is directed toward making the necessary repairs. In the event the affected assemblies cannot be returned to service within this specified period the reactor will be brought to hot shutdown conditions.

4. The provisions of 3.12.C.1 and 3.12.C.2 shall not apply during physics test in which the assemblies are intentionally misaligned.

5. If an inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in TS Figure 3.12-2 apply.

6. If an inoperable full-length rod cannot be located, or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in TS Figure 3.12-3 apply.

7. No insertion limit changes are required by an inoperable part-length rod.

8. If a full-length rod becomes inoperable and reactor operation is continued the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the unit power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

D. If the reactor is operating above 75% of rated power with one excore nuclear channel out of service, the core quadrant power balance shall be determined.

1. Once per day, and

2. After a change in power level greater than 10% or more than 30 inches of control rod motion.

The core quadrant power balance shall be determined by one of the following methods:

1. Movable detectors (at least two per quadrant)

2. Core exit thermocouples (at least four per quadrant).

E. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service then:

   a. For operation between 50% and 100% of rated power, the position of the RCC shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to motion, of the non-indicating rod, exceeding 24 steps, whichever occurs first.

   b. During operation below 50% of rated power no special monitoring is required.

2. Not more than one rod position indicator (RPI) channel per group nor two RPI channels per bank shall be permitted to be inoperable at any time.

F. Misaligned or Dropped Control Rod

1. If the Rod Position Indicator Channel is functional and the associated part length or full length control rod is more than 15 inches out of alignment with its bank and cannot be realigned, then unless the hot channel factors are shown to be within design limits as specified in Section 3.12.B.1 within 8 hours, power shall be reduced so as not to exceed 75% of permitted power.

2. To increase power above 75% of rated power with a part-length or full length control rod more than 15 inches out of alignment with its bank an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 3.12.B.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated for by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition.

The control rod assembly insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical assembly ejection, and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement. The maximum shutdown margin require-

ment occurs at end of core life and is based on the value used in the analysis of the hypothetical steam break accident. The rod insertion limits are based on end of core life conditions. Early in core life, less shutdown margin is required, and TS Figure 3.12-7 shows the shutdown margin equivalent to 1.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

Relative positions of control rod banks are determined by a specified control rod bank overlap. This overlap is based on the consideration of axial power shape control.

The specified control rod insertion limits have been revised to limit the potential ejected rod worth in order to account for the effects of fuel densification.

The various control rod assemblies (shutdown banks, control banks A, B, C, and D and part-length rods) are each to be moved as a bank, that is, with all assemblies in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks and a linear position indicator, Linear Variable Differential Transformer, which indicates the actual assembly position. The position indication accuracy of the Linear Differential Transformer is approximately ±5% of span (±7.5 inches) under steady state conditions. The relative accuracy of the linear position indicator is such that, with the

most adverse errors, an alarm is actuated if any two assemblies within a bank deviate by more than 14 inches. In the event that the linear position indicator is not in service, the effects of malpositioned control rod assemblies are observable from nuclear and process information displayed in the Main Control Room and by core thermocouples and in-core movable detectors. Below 50% power, no special monitoring is required for malpositioned control rod assemblies with inoperable rod position indicators because, even with an unnoticed complete assembly misalignment (part-length of full length control rod assembly 12 feet out of alignment with its bank) operation at 50% steady state power does not result in exceeding core limits.

The specified control rod assembly drop time is consistent with safety analyses that have been performed.

An inoperable control rod assembly imposes additional demands on the operators. The permissible number of inoperable control rod assemblies is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable control rod assemblies upon reactor trip.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature and cladding mechanical properties. First, the peak value of linear power density must not exceed 21.1 kw/ft for Unit No. 1 and 20.4 kw/ft for Unit No. 2. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200^{o}F on peak clad temperature. This is required to meet the initial conditions assumed for the loss of coolant accident. To aid in specifying the limits on power distribution the following hot channel factors are defined.

$F_Q(Z)$, 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.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

An upper bound envelope of 2.10 times the normalized peaking factor axial dependent of TS Figure 3.12-8 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control given in Section 3.12.B.4. 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.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerances.

In the specified limit of $F^N_{\Delta H}$ there is an eight 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 affect 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 the F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system.

Measurement of the hot channel factors are required as part of startup physics tests, during each effective full 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 core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as shown in Figures 3.12-1A, 3.12-1B and 3.12-2.

3. The full length and part length control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference referes to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ with decreasing power level allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In specification 3.12.B.1 F_Q is arbitrarily limited for $P \leq .5$ (except for physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically control of flux difference is required to limit the difference between the current value of flux difference (ΔI) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = ΔI/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 given in 3.12.B.4 assure that the F_Q upper bound envelope of 2.10 times Figure 3.12-8 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 the full length rod control bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full 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 +6 to -9% 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 month. For this reason, the specification provides two methods for updating the target flux difference.

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 required, periodic, excore detector calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or 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 90% power, in the range +14.5 to -21 percent (+11.5 percent to -18 percent indicated) where for every 4 percent below rated power, the permissible positive flux difference boundary is extended

by 1 percent, and for every 5 percent below rated power, the permissible negative flux difference boundary is extended by 2 percent.

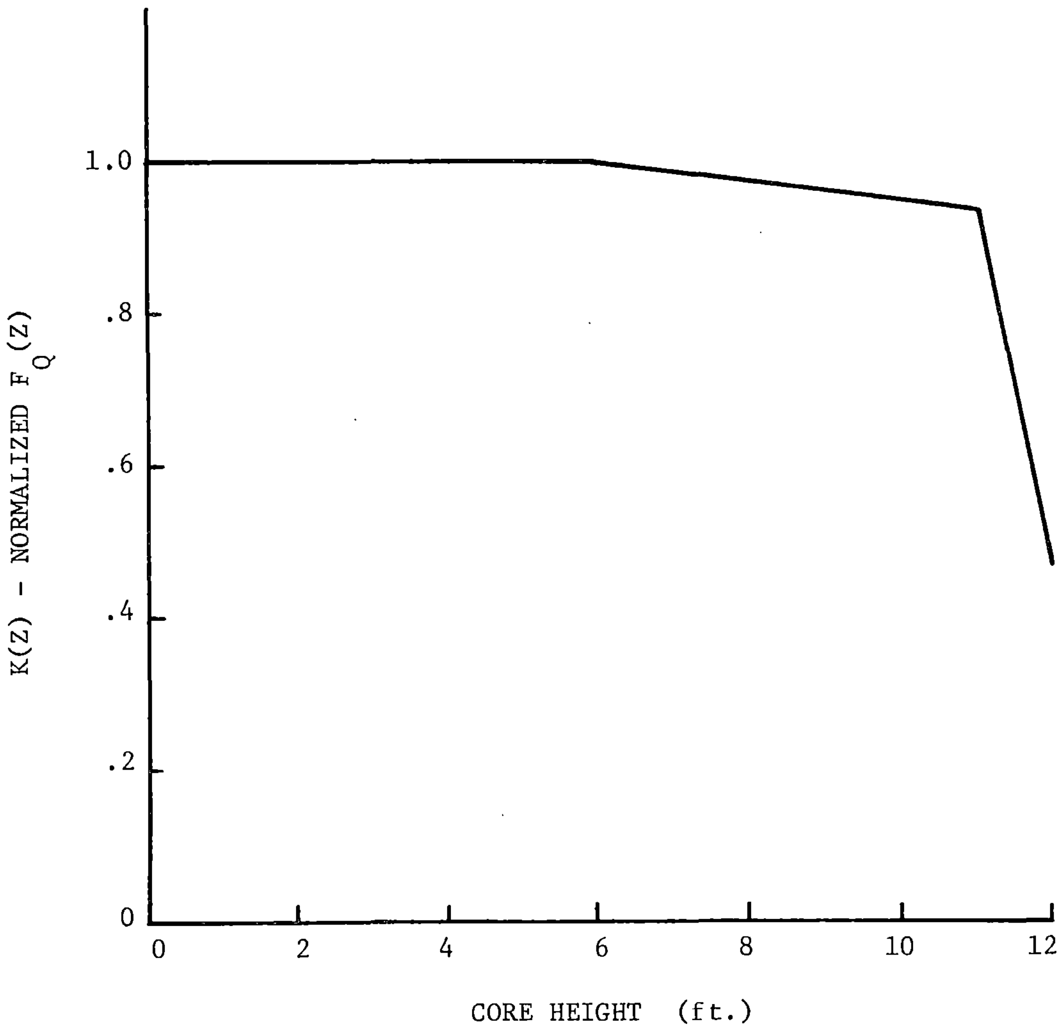
As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference.

At the option of the operator, credit may be taken for measured decreases in the unrodded horizontal plane peaking factor, F_{xy}. This credit may take the form of an expansion of permissible quadrant tilt limits over tilt limits over the 2% value, up to a value of 10%, at which point specified power reductions are prudent. Monthly surveillance of F_{xy} bounds the quantity because it decreases with burnup. (WCAP-7912 L).

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rods and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

HOT CHANNEL FACTOR NORMALIZED
OPERATING ENVELOPE

SURRY POWER STATION
UNIT NOS. 1 AND 2



4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of power, the hot channel factors, F_Q and $F^N_{\Delta H}$, shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed values of

$$F_Q(Z) \leq (2.10/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.20) \times K(Z) \text{ for } P \leq .5$$

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

(where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, and Z is the core height location of F_Q), an evaluation as to the cause of the anomaly shall be made.

Basis

BORON CONCENTRATION

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod assembly groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration, and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive control rod assembly in the fully withdrawn position is always maintained.

PEAKING FACTORS

A thermal criterion in the reactor core design specifies that "no fuel melting during any anticipated normal operating condition" should occur. To meet the above criterion during a thermal overpower of 118% with additional margin for design uncertainties, a steady state maximum linear power is selected. This then is an upper linear power limit determined by the maximum central temperature of the hot pellet.

The peaking factor is a ratio taken between the maximum allowed linear power density in the reactor to the average value over the whole reactor. It is of course the average value that determines the operating power level. The peaking factor is a constraint which must be met to assure that the peak linear power density does not exceed the maximum allowed value.

During normal reactor operation, measured peaking factors should be significantly lower than design limits. As core burnup progresses, measured designed peaking factors are expected to decrease. A determination of F_Q and $F^N_{\Delta H}$ during each effective full power month of operation is adequate to ensure that core reactivity changes with burnup have not significantly altered peaking factors in an adverse direction.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight per cent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 816 poison rods in the form of 12 rod clusters, which are located in vacant control rod assembly guide thimbles. The burnable poison rods consist of pyrex glass clade with stainless steel.

5. There are 48 full-length control rod assemblies and 5 part-length control rod assemblies in the reactor core. The full-length control rod assemblies contain a 144 inch-length of silver-indium-cadmium alloy clad with stainless steel. The part-length control rod assemblies contain a 36 inch-length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.

6. The initial core and subsequent cores will meet the following performance criteria at all times during the operating lifetime.

   a. Hot channel factors:

   $$F_Q(Z) \leq (2.10/P) \times K(Z) \text{ for } P > .5$$

   $$F_Q(Z) \leq (4.20) \times K(Z) \text{ for } P \leq .5$$

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

   where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, and Z is the core height location of F_Q.

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