WCAP 9500

Amendment 1

Instruction Sheet

The following instructional information and check list is being furnished to help insert Amendment No. 1 into WCAP 9500.

Since in most cases the original WCAP contains information printed on both sides of a sheet of loose leaf paper, a new sheet is furnished to replace sheets containing superseded material. As a result, the front or back of a sheet may contain information that is merely reprinted rather than changed.

Discard the old sheets and insert the new sheets, as listed below. Keep these instruction sheets in the front of Volume I to serve as a record of changes.

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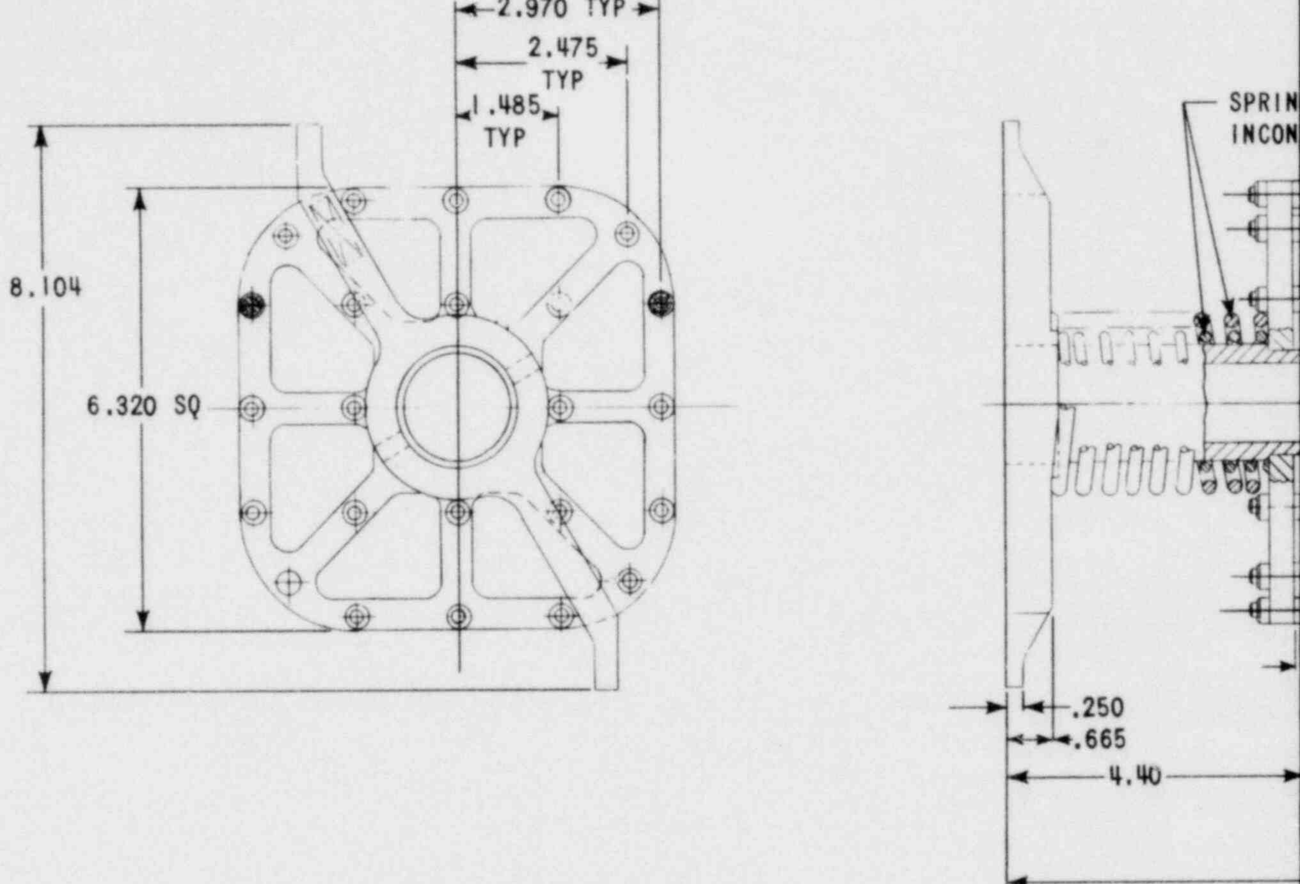
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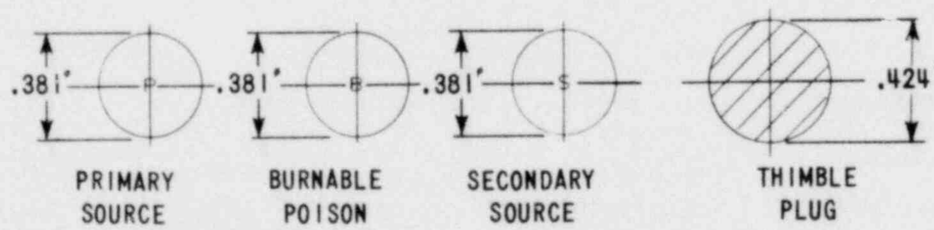
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PRIMARY SOURCE
BURNABLE POISON
SECONDARY SOURCE
THIMBLE PLUG

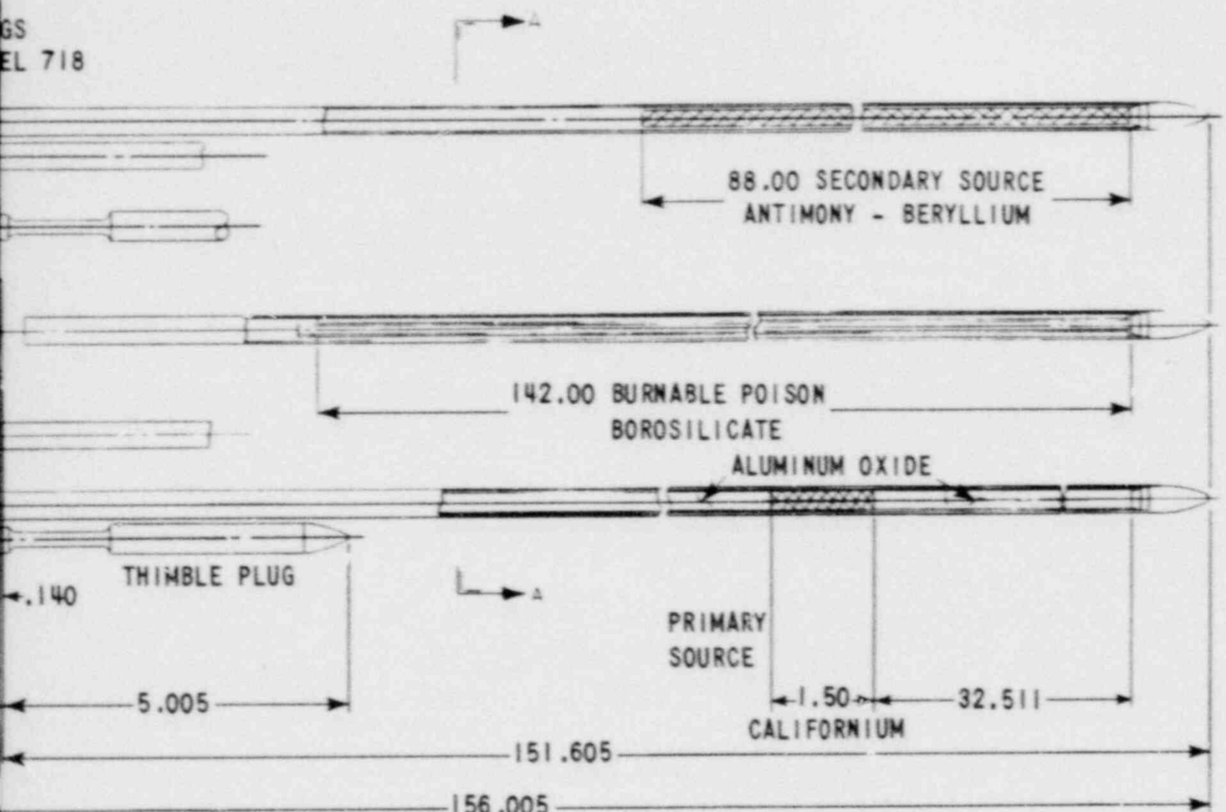

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Figure 4.2-12.

Composite Core Component Rods and Assembly Outline (Non-UHI Plants)

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6.2.1.5 Minimum Containment Pressure Analysis for Performance Capabililty Studies on Emergency Core Cooling System

The containment backpressure used for the limting case (C_D=0.6) double-ended cold leg guillotine break for the ECCS analysis presented in Section 15.6.5 is presented in Figure 6.2.1.5-1. The containment back-pressure is calculated using the methods and assumptions described in Appendix A of Reference 1. Input parameters including the containment initial conditions; net free containment volume; passive heat sink materials, thickness; and surface areas; and starting time and number of containment cooling systems used in the analysis are described in the following paragraphs.

6.2.1.5.1 Mass and Energy Release Data

The mass and energy releases to the containment during the blowdown and reflood portions of the limiting break transient are presented in Tables 6.2.1.5-1 and 6.2.1.5-2.

The mathematical models which calculate the mass and energy releases to the containment are described in Section 15.6.5. Since the requirements of 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," are very specific in regard to the modeling of the reactor coolant system during blowdown and the models used are in conformance with Appendix K, no alterations to those models have been made in regard to the mass and energy releases. A break spectrum analysis is performed (see references in Section 15.6.5) that considers various break sizes, break locations, and Moody discharge coefficients for the double-ended cold leg guillotines which affect the mass and energy released to the containment. This effect is considered for each case analyzed. During refill, the mass and energy released to the containment is assumed to be zero, which minimizes the containment pressure. During reflood, the effect of steam water mixing between the safety injection water and the steam flowing through the reactor coolant system intact loops reduces the available energy released to the containment vapor space and therefore tends to minimize containment pressure.

6.2.1.5.2 Initial Containment Internal Conditions

The following initial values were used in the analysis:

1. A containment pressure of 14.7 psia.
2. A containment temperature of 90^{o}F.
3. A refueling water storage tank temperature of 50^{o}F.
4. An outside temperature of -50^{o}F.
5. A relative humidity of 99 percent.

The containment initial conditions of 90^{o}F and 14.7 psia are representatively low values anticipated during normal full power operation.

6.2.1.5.3 Containment Volume

The volume used in the analysis was 3.05 x 10^6 ft^3.

6.2.1.5.4 Active Heat Sinks

The containment spray system operates to remove heat from the containment.

Pertinent data for this system which were used in the analysis are presented in Table 6.2.1.5-3.

The sump temperature was not used in the analysis because the maximum peak cladding temperature occurs prior to initiation of the recirculation phase for the containment spray system. In addition, heat transfer between the sump water and the containment vapor space was not considered in the analysis.

6.2.1.5.5 Steam-Water Mixing

Water spillage rates from the broken loop accumulator are determined as part of the core reflooding calculation and are included in the containment code (COCO) calculational model.

6.2.1.5.6 Passive Heat Sinks

The passive heat sinks used in the analysis, with their thermophysical properties, are given in Table 6.2.1.5-4. The passive heat sinks and thermophysical properties were derived in compliance with Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

6.2.1.5.7 Heat Transfer to Passive Heat Sinks

The condensing heat transfer coefficients used for heat transfer to the steel containment structures are given in Figure 6.2-2 for the limiting break. The containment pressure transient for the limiting break is shown in Figure 6.2.1.5-1.

6.2.1.5.8 Other Parameters

No other parameters have a substantial effect on the minimum containment pressure analysis.

6.2.2 REFERENCES

1. Bordelon, F. M., Massie, H. W., Jr., and Zordan, T. A., "Westinghouse Emergency Core Cooling System Evaluation Model-Summary," WCAP-8339, June, 1974.

TABLE 6.2.1.5-1

DOUBLE-ENDED COLD LEG GUILLOTINE BLOWDOWN MASS AND ENERGY RELEASES RATE (C_D=0.6)

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)
0.0	9.98×10^3	5.60×10^6
0.2	6.58×10^4	3.67×10^7
0.7	6.36×10^4	3.55×10^7
1.2	5.97×10^4	3.36×10^7
3.0	4.02×10^4	2.30×10^7
5.0	2.96×10^4	1.75×10^7
8.0	2.46×10^4	1.54×10^7
10.0	2.26×10^4	1.42×10^7
12.0	1.91×10^4	1.21×10^7
15.0	1.47×10^4	9.42×10^6
17.0	7.60×10^3	6.27×10^6
20.0	9.58×10^3	4.46×10^6
22.0	9.61×10^3	2.87×10^6
24.0	4.89×10^3	8.84×10^5
24.53	0.0	0.0

TABLE 6.2.1.5-2

DOUBLE-ENDED COLD LEG GUILLOTINE REFLOOD MASS AND ENERGY RELEASES (C_D=0.6)

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)
34.7	0.0	0.0
38.9	36.8	4.82×10^4
46.8	5680	5.68×10^5
61.1	360	2.00×10^5
73.9	378	1.98×10^5
98.8	388	1.94×10^5
120.4	397	1.88×10^5
168.8	411	1.75×10^5
225.1	425	1.60×10^5

Nitrogen Addition

Time (sec)	Mass Flow (lb/sec	Temperature ^oF
0.0	0.0	90.0
58.62	0.0	90.0
59.0	228.0	90.0
79.0	228.0	90.0
80.0	0.0	90.0
500.0	0.0	90.0

TABLE 6.2.1.5-3

ACTIVE HEAT SINK DATA FOR MINIMUM POST-LOCA CONTAINMENT PRESSURE

Containment Spray System Parameters

Number of pumps operating	2
Runout flow rate, total (gpm)	12,900
Actuation time, full flow (sec)	36

TABLE 6.2.1.5-4

PASSIVE HEAT SINK DATA FOR MINIMUM POST-LOCA CONTAINMENT PRESSURE

Heat Sink Description

Thickness (ft)	Surface Area (ft^2)
0.035 steel, 2.5 concrete	29,500
0.035 steel, 4.5 concrete	84,100
2.5 concrete, 0.021 steel, 12 concrete	11,800
1.0 concrete	160,000
0.2 steel	5,000
0.05 steel	65,000
0.03125 steel	90,000
0.030 steel	100,000
0.020 steel	70,000
0.0057 steel	45,000

Thermophysical Properties

	Density (lb/ft^3)	Volumetric Heat Capacity (Btu/ft^3-^oF)	Thermal Conductivity (Btu/hr-ft-^oF)
Concrete	145	0.156	0.92
Carbon steel	490	0.12	27.0

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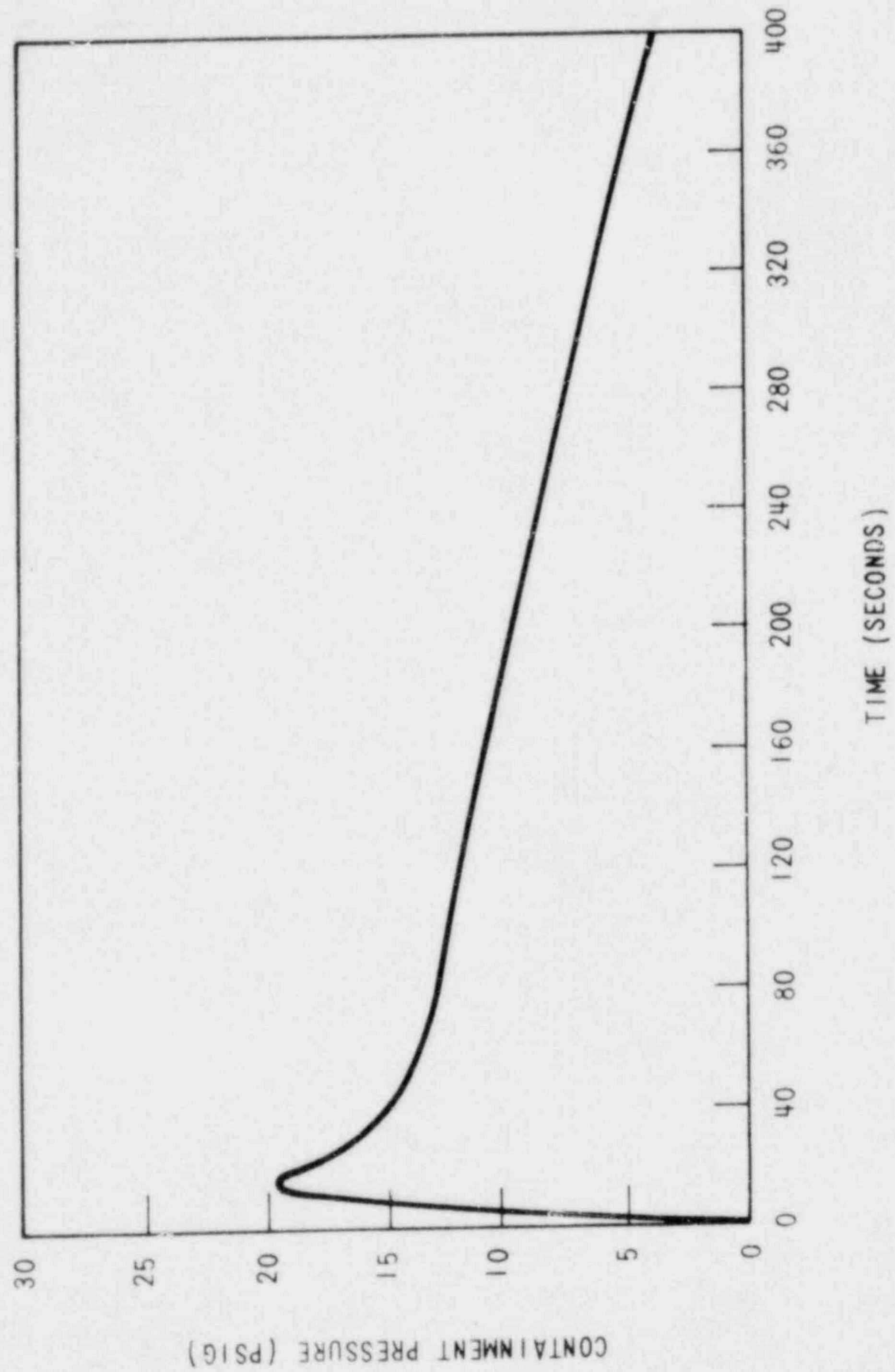

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Figure 6.2.1.5-1
Containment Pressure Transient
for ECCS Performance Capability
Analysis (DECLG C_D= 0.6)

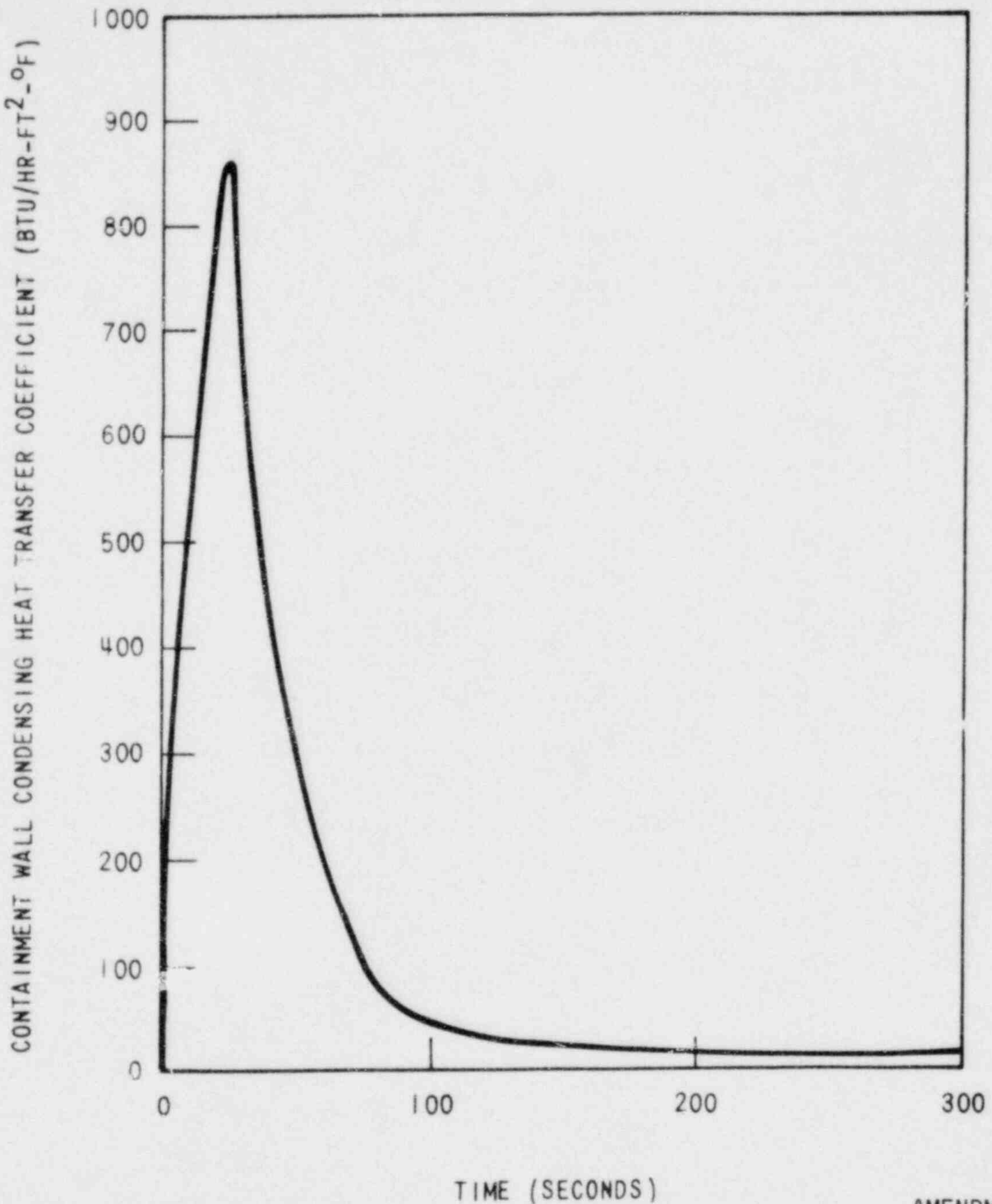

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Figure 6.2.1.5-2
Condenser Heat Transfer
Coefficients (DECLG C_D= 0.6)

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TABLE 15.0-7

IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE AND FUEL ROD GAPS*

Isotope	Core Activity (Curies)	Fraction of Activity in Gap** (%)	Gap Activity (Curies)
I-131	9.9×10^7	.10	9.9×10^6
I-132	1.4×10^8	.10	1.4×10^7
I-133	2.0×10^8	.10	2.0×10^8
I-134	2.2×10^8	.10	2.2×10^8
I-135	1.9×10^8	.10	1.9×10^8
Xe-131m	7.0×10^5	.10	7.0×10^4
XE-133	1.9×10^8	.10	1.9×10^7
Xe-133m	2.9×10^7	.10	2.9×10^6
Xe-135	4.2×10^7	.10	4.2×10^6
XE-135m	4.0×10^7	.10	4.0×10^6
XE-138	1.6×10^8	.10	1.6×10^7
Kr-83m	1.2×10^7	.10	1.2×10^6
Kr-85	6.6×10^5	.30	2.0×10^5
Kr-85m	2.7×10^7	.10	2.7×10^6
Kr-87	4.9×10^7	.10	4.9×10^6
Kr-88	7.0×10^7	.10	7.0×10^6
Kr-89	8.7×10^7	.10	8.7×10^6
	Core Kilograms		Gap Kilograms
I-127	2.8	.30	0.84
I-129	11.4	.30	3.4

* Based on a three-region equilibrium cycle core at end of life. The three regions have operated at a specific power of 40.03 MW/MTU for 300, 600 and 900 EFPD, respectively.

** NRC assumption in Regulatory Guide 1.25

(Subsection 15.6.5) because less primary coolant is released and the activity is lower as fuel damage is not predicted as a result of this event.

15.6.1.4 Conclusions

The results of the analysis show that the pressurizer low pressure and the low DNBR reactor protection system signals provide adequate protection against the RCS depressurization event. The DNBR remains above the limit valve throughout the transient; thus, the DNB design-basis as described in Section 4.4 is met. The radiological consequences of this event would be substantially less than that of the LOCA analyzed in Subsection 15.6.5.

15.6.2 FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

15.6.2.1 Identification of Causes and Accident Description

The accident results from a break in small lines such as a sample line connected to the primary coolant system and penetrating the containment. Ruptures of small cross-sectional lines will cause explusion of the coolant at a rate which can be accommodated by a charging pump which would maintain an operational water level in the pressurizer, permitting the operator to conduct an orderly shutdown. The release contains the radionuclide concentration of the primary coolant.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system (RCS) through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is adequate to sustain pressurizer level and a pressure of 2250 psia for a break through a 0.375 inch diameter hole. This break results in a loss of approximately 17.5 lb/sec, and, due to the use of a 0.245 inch restriction, is the maximum flow available for all reactor coolant sample line breaks outside of the containment. In addition, all such lines meet the requirements of General Design Criterion 55 of Appendix A 10 CFR 50. There are no instrument lines which pass through the containment and connect directly to the RCS. A failure of a small line carrying primary coolant outside containment is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.2 for a discussion of Condition II events.

15.6.2.2 Analysis of Effects and Consequences

Since this event does not result in a leakage rate greater than the capacity of a charging pump and pressurizer level does not decrease, normal shutdown procedures can be employed. There are no significant consequences to the reactor or its essential auxiliary systems.

15.6.2.3 Radiological Consequences

There could be moderate radioactive releases from the failure of a small line carrying primary coolant outside containment. This accident will be evaluated in the Applicant's SAR. The primary coolant activity that would be used in the small line break analysis is 60 μCi of dose equivalent I-131 resulting from a preexisting iodine spike.

15.6.3 STEAM GENERATOR TUBE RUPTURE

15.6.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube for a NON-UHI 4-Loop plant. This event is considered an ANS Condition IV event, a limiting fault (see Section 15.0.1). The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

In view of the fact that the steam generator tube material is Inconel-600 and is a highly ductile material, it is considered that the assumption of a complete severence is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the Steam and Power Conversion System is subject to continual surveillance and an accumullation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during the unit operation.

The operator is expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the faulty steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the accident diagnostics and isolation procedure can be completed within 30 minutes of accident initiation for the design basis event.

Note that break sizes smaller than complete severance of a tube, with less break flow from primary to seconday, exhibit a slower rise in steam generator water level, and an increased time interval for actuation of the blowdown line radiation monitor and the condenser air ejector radiation monitor. Therefore more time may be available to the operator to diagnose the accident and take steps to isolate the faulted steam generator.

If normal operation of the various plant control systems is assumed, the folowing sequence of events is initiated by a tube rupture.

1. Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side steam flow/feedwater flow mismatch occurs as feedwater flow to the affected steam generator is reduced as a result of primary coolant break flow to that unit.

2. Decrease in RCS pressure (Figure 15.6-84) due to continued loss of reactor coolant inventory leads to a reactor trip signal on low pressurizer pressure or overtemperature ΔT. Resultant plant cooldown (Figure 15.6-85) following reactor trip leads to a rapid decrease in pressurizer level (Figure 15.6-88), and a safety injection signal, initiated by low pressurizer pressure, follows soon after reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.

3. The steam generator blowdown liquid monitor and/or the condenser air ejector radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system and will auto- matically terminate steam generator blowdown.

4. The reactor trip automatically trips the turbine and if offsite power is available the steam dump valves open permitting steam dump to the condenser. In the event of a coincident station blackout (loss of offsite power), as assumed in the transients presented in this section, the steam dump valves automatically close to protect the condenser. The steam generator pressure (Figure 15.6-86) rapidly increases resulting in steam discharge to the atmosphere through the steam generator safety and/or power operated relief valves. In Figure 15.6-89 the steam flow is presented as a function of time. The flow is constant initially until reactor trip, followed by turbine trip, which results in a large decrease in flow, but a rapid increase in steam pressure to the safety valve setpoint.

5. Following reactor trip, the continued action of the auxiliary feedwater supply and borated safety injection flow (supplied from the RWST) provide a heat sink which absorbs the decay heat.

6. Safety injection flow results in increasing pressurizer water level (Figure 15.6-88), the rate of which depends upon the amount of auxiliary equipment operating.

15.6.3.2 Analysis of Effects and Consequences

Method of Analysis

Mass and energy balance calculations are performed to determine primary to secondary mass release and to determine amount of steam vented from each of the steam generators, using LOFTRAN (Reference 1).

In estimating the mass transfer from the RCS through the broken tube the following assumptions are made:

1. Reactor trip occurs automatically as a result of low pressurizer pressure or overtemperature ΔT. Loss of offsite power occurs at reactor trip.

2. Following the initiation of the safety injection signal, two centrifugal pumps are actuated and are assumed in the analyses to continue to deliver flow for 30 minutes.

3. After reactor trip the break flow reaches equilibrium when incoming safety injection flow is balanced by outgoing break flow as shown in Figure 15.6-83. The resultant break flow is assumed to persist from plant trip until 30 minutes after the accident. No operator actions are assumed.

4. The steam generators are controlled at the safety valve setting rather than the power operated relief valve setting.

5. The operator identifies accident type (using procedure guidlines in Reference 27) and terminates break flow to the faulted steam generator within 30 minutes of accident initiation.

The above assumptions, suitably conservative for the design basis tube rupture, are made to maximize doses and do not explicitly model operator actions for recovery.

Recovery Procedure

Immediately apparent symptoms of a tube rupture accident such as falling pressurizer pressure and level and increased charging pump flow are also symptoms of small steam line breaks and loss of coolant accidents. It is therefore important for the operator to determine that the accident is a rupture of a steam generator tube in order that he may carry out the correct recovery procedure. The accident under discussion can be identified by the following method. In the event of a complete tube ruture the reactor coolant system pressure decreases, Figure 15.6-84, and the condenser air ejector radiation and/or steam generator blowdown radiation monitors exhibit abnormally high readings. If the containment pressure, containment radiation and containment recirculation sump level exhibit normal readings, then a steam generator tube rupture is diagnosed to have occurred.

The recovery procedures for the double ended rupture of a steam generator tube can be found in Reference 27. These procedures are presented for a plant with a high pressure safety injection system and a low pressure safety injection system.

Results

Figure 15.6-83 illustrates the flow rate that would result through the ruptured steam generator tube. The previous assumptions lead to an estimate of 68296 pounds for the total amount of reactor coolant transferred to the secondary side of the faulted steam generator as a result of a tube rupture accident. The integrated steam flow is 59290 pounds released through the safety valves.

The following is a list of figures of pertinent time dependent parameters.

Figure 15.6-84 Reactor Coolant System Pressure
Figure 15.6-85 Reactor Coolant System Temperature
Figure 15.6-86 Steam Generator Pressure (For Faulted Steam Generator)
Figure 15.6-87 Steam Generator Temperature (For Faulted Steam Generator)
Figure 15.6-88 Pressurizer Water Volume
Figure 15.6-89 Steam Generator Flow

The DNB calculations performed with LOFTRAN (Reference 1) indicate that DNB limits are met.

In Table 15.6-1 the sequence of events are presented. These events are the normal plant response to the normal plant setpoints. Loss of off-site power at reactor trip and no operator actions were assumed.

10CFR50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

Small Break Results

As noted previously, the calculated peak clad temperature resulting from a small break LOCA is less than that calculated for a large break. Based on the results of the LOCA sensitivity studies Reference [21] the limiting small break was found to be less than a 10 inch diameter rupture of the RCS cold leg. Therefore, a range of small break analyses are presented which establishes the limiting break size. The results of these analyses are summarized in Tables 15.6-1 and 15.6-4.

Figures 15.6-34 through 15.6-47 present the principal parameters of interest for the small break ECCS analyses. For all cases analyzed the following transient parameters are presented:

1. RCS pressure.

2. Core mixture height.

3. Hot spot clad temperature.

4. Core power after reactor trip.

5. Pumped safety injection.

For the limiting break analyzed, the following additional transient parameters are presented:

1. Core steam flow rate.

2. Core heat transfer coefficient.

3. Hot spot fluid temperature.

The maximum calculated peak clad temperature for all small breaks analyzed is 1792°F. These results are well below all Acceptance Criteria limits of 10CFR50.46 and in all cases are not limiting when compared to the results presented for large breaks.

15.6.5.3.4 Analysis for N-1 Loop Operation

During the lifetime of a nuclear power plant the owner-utility may desire, under certain circumstances, to operate with one reactor coolant loop out of service. Analyses were performed for the operation of a four loop plant without loop isolation valves and with one idle loop. All calculations were performed utilizing the February 1978 Evaluation Model with changes as described in WCAP-9167[32].

A spectrum of analyses were performed for large breaks in an active loop and the limiting break from this spectrum was analyzed in the idle loop. The analyses were performed at 70 percent of full reactor power. Sensitivity studies in WCAP-9167 show that the worst break location is in an active loop. Also, a comparison of WCAP-9167 and WCAP-9281[33] indicates that small breaks are not limiting for operation with one loop out of service. Therefore, only large breaks are reported herein.

Table 15.6-1 presents the occurrence time for various events throughout the accident transient for breaks in an active loop and for the idle loop break analyzed.

Tables 15.6-2 and 15.6-3 present various input values and results from the hot fuel rod thermal transient calculation for the breaks in an active loop. There Tables also give similar information for the idle loop break. For these results, the hot spot defined as the location of maximum peak clad temperatures. That location is specified in Tables 15.6-3 for each break analyzed. The location is indicated in feet which represents elevation above the bottom of the active fuel stack.

Figures 15.6-49 through 15.6-82 present the parameters of principal interest. For all cases analyzed transients of the following parameters are presented:

1. Hot spot clad temperature.
2. Coolant pressure in the reactor core.
3. Water level in the core and downcomer during reflood.
4. Core reflooding rate.
5. Thermal power during blowdown.
6. Containment pressure.

For the limiting break analyzed, the following additional transient parameters are presented:

1. Core flow during blowdown (inlet and outlet).
2. Core heat transfer coefficients.
3. Hot spot fluid temperature.
4. Mass released to Containment during blowdown.
5. Energy released to Containment during blowdown.
6. Fluid quality in the hot assembly during blowdown.
7. Mass velocity during blowdown.
8. Accumulator water flow rate during blowdown (intact loop).
9. Accumulator water flow rate during blowdown (idle loop).
10. Pumped safety injection during reflood.
11. The containment wall condensing heat transfer coefficient.

The highest peak clad temperature calculated is 1742oF which meets the Acceptance Criteria limit of 2200oF of 10CFR50.46. The maximum local water reaction reached is 1.21 percent, which is well below the embrittlement limit of 17 percent as required by 10CFR50.46. The total core metal water reaction is less than 0.3 percent for all breaks, as compared with the 1 percent criterion of 10CFR50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The core temperature is reduced and the decay heat is removed for an extended period of time.

15.6.5.4 Radiological Consequences of a Postulated Loss-of-Coolant Accident

Two analyses will be performed: 1) a realistic analysis, and 2) an analysis based on Regulatory Guide 1.4, Revision 2. The parameters to be used for each of these analyses are listed in Table 15.6-5. The radiological consequences of a LOCA will be evaluated on a plant specific basis.

Fission Product Release to the Containment

The radiological assessment will be based on the conservative fission product release given in Regulatory Guide 1.4.

Thus, a total of 100 percent of the noble gas core inventory and 25 percent of the core iodine inventory is assumed to be immediately available for leakage from the primary containment. Of the halogen activity available for release, will be assumed that 91 percent is in elemental form, 4 percent in methyl form and 5 percent in particulate form. The total core noble gas and iodine inventories are given in Table 15.0-7.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable.

19. Letter from T. M. Anderson of Westinghouse Electric Corporation to R. L. Tedesco of the Nuclear Regulatory Commission, letter number NS-TMA-2014, December 11, 1978.

20. Johnson, W. J. and Thompson, C. M., "Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version," WCAP-9168 (Proprietary) and WCAP-9169 (Non-Proprietary), September 1977.

21. "Westinghouse ECCS Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-Proprietary), July 1974.

22. Salvatori, R., "Westinghouse ECCS - Plan Sensitivity Studies, WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), July 1974.

23. Johnson, W. J., Massie, H. W. and Thompson, C. M., "Westinghouse ECCS-Four Loop Plant (17x17) Sensitivity Studies", WCAP-8565-P-A (Proprietary) and WCAP-8566-A (Non-Proprietary), July 1975.

24. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-2030, January 1979.

25. Thompson, C. M., et. al., "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing (N-1) Loop Operation of Plants Without Loop Isolation Valves for Large Break LOCA's", WCAP-9166 (Proprietary), WCAP-9167 (Non-Proprietary), March, 1978.

26. Monty, B. S., et. al., "Westinghouse Emergency Core Cooling System Evaluation Model For Analyzing Small LOCA. During Operation With One Loop Out of Service For Plants Without Loop Isolation Valves, WCAP-9280 (Proprietary), WCAP-9281 (Non Proprietary, February, 1978.

27. Letter, C. Reed to D. Ross, OG-13, October 16, 1979.

TABLE 15.6-1 (Sheet 1 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec) N Loop	TIME (sec) N-1 Loop
Inadvertent opening of a pressurizer safety valve	Safety valve opens fully	0.0	0.0
	Low Pressurizer Pressure reactor trip setpoint reached	41.3	35.4
	Rods begin to drop	43.3	37.4
	Minimum DNBR occurs	43.6	38.2
Large break LOCA (N Loop)			
1. DECLG $C_D = 0.8$	Start	0.0	
	Reactor trip signal	0.83	
	Safety injection signal	1.1	
	Accumulator injection begins	14.1	
	End-of-bypass	23.8	
	End-of-blowdown	26.7	

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec)
	Pump injection begins	26.1
	Bottom of core recovery	34.4
	Accumulator empty	51.3
2. DECLG $C_D = 0.6$	Start	0.0
	Reactor trip signal	0.84
	Safety injection signal	1.3
	Accumulator injection begins	16.4
	End-of-bypass	24.51
	End-of-blowdown	24.53
	Pump injection begins	26.3
	Bottom of core recovery	34.7
	Accumulator empty	52.6

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec)
3. DECLG $C_D = 0.4$	Start	0.0
	Reactor trip signal	0.88
	Safety injection signal	1.6
	Accumulator injection begins	21.2
	End-of-bypass	33.2
	End-of-blowdown	37.0
	Pump injection begins	26.6
	Bottom of core recovery	44.3
	Accumulator empty	60.7
Large Break LOCA (N-1 Loop)		
1. DECLG $C_D = 1.0$ (break in active loop)	Start	0.0
	Reactor trip signal	0.76

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec)
	Safety injection signal	1.1
	Accumulator injection begins	14.0
	End-of-bypass	27.6
	End-of-blowdown	28.9
	Pump injection begins	26.1
	Bottom of core recovery	39.8
	Accumulator empty	51.6
2. DECLG $C_{D=0.8}$ (break in active loop)	Start	0.0
	Reactor trip signal	0.77
	Safety injection signal	1.2
	Accumulator injection begins	14.7
	End-of-bypass	32.4

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec)
	End-of-blowdown	35.2
	Pump injection begins	26.2
	Bottom of core recovery	45.6
	Accumulator empty	52.8
3. DECLG $C_{D=0.6}$ (break in active loop)	Start	0.0
	Reactor trip signal	0.79
	Safety injection signal	1.4
	Accumulator injection begins	17.7
	End-of-bypass	27.9
	End-of-blowdown	29.3
	Pump injection begins	26.4
	Bottom of core recovery	39.1
	Accumulator empty	55.1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec)
4. DECLG $C_{D-0.8}$ (break in idle loop)	Start	0.0
	Reactor trip signal	0.77
	Safety injection signal	1.3
	Accumulator injection begins	14.0
	End-of-bypass	23.7
	End-of-blowdown	27.1
	Pump injection begins	26.3
	Bottom of core recovery	36.9
	Accumulator empty	51.0
Small break LOCA		
1. 3 inch	Start	0.0
	Reactor trip signal	27.3
	Top of core uncovered	695

TABLE 15.6-1 (Sheet 7 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec)
	Accumulator injection begins	N/A
	Peak clad temperature occurs	1426
	Top of core covered	2295
2. 4 inch	Start	0.0
	Reactor trip signal	16.9
	Top of core uncovered	330
	Accumulator injection begins	802
	Peak clad temperature occurs	790
	Top of core covered	1190
3. 6 inch	Start	0.0
	Reactor trip signal	11.2
	Top of core uncovered	126

TABLE 15.6-1 (Sheet 8 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec)
	Accumulator injection begins	349
	Peak clad temperature occurs	235
	Top of core covered	365
	Tube rupture occurs	0.0
	Reactor trip signal	470
	Rod motion	472
	Feedwater terminated	472
	Steam generator safety valves opened (assumed to stay open to maximize release)	480
	S.I. signal	851
	S.I. injection	875
	Auxiliary feedwater injection	910
	Assumed that operator completes actions to isolate and equilibrate	1800

TABLE 15.6-2

INPUT PARAMETERS USED IN THE ECCS ANALYSIS

	N Loop	N-1 Loop
Licensed core power[a], (MWt)	3411	2388
Peak linear power, includes 102% factor (kW/ft)	12.88	10.57
Total peaking factor, F_Q^T	2.32	2.72
Axial peaking factor, F_Z	1.451	1.561
Power shape		
Large break	Chopped cosine	Chopped cosine
Small break	See Figure 15.6-48	N/A
Fuel assembly array	Optimized 17x17	Optimized 17x17
Accumulator water volume, nominal (ft^3/accumulator)	1200	1200
Accumulator tank volume, nominal (ft^3/accumulator)	1650	1650
Accumulator gas pressure, minimum (psia)	600	600
Safety injection pumped flow	See Figures 15.6-21 and 15.6-47	See Figure 15.6-21a
Containment parameters	See Sec. 6.2	See Sec. 6.2
Initial loop flow (lb/sec)	9984	10,745[b]
Vessel inlet temperature (^oF)	560.7	552.9[b]
Vessel outlet temperature (^oF)	643.3	597.1[b]
Reactor coolant pressure (psia)	2250	2250
Steam pressure (psia)	988	946[b]
Steam generator tube plugging level (%)	0	0

[a]Two percent is added to this power to account for calorimetric error.

[b]Active loop.

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TABLE 15.6-3

LARGE BREAK LOCA RESULTS FUEL CLADDING DATA

Results for N Loop	C_D = 0.8 DECLG	C_D = 0.6 DECLG	C_D = 0.4 DECLG
Peak clad temperature (^oF)	1964	1991	1707
Peak clad temperature location (ft)	7.5	7.5	7.5
Local Zr/H_2O reaction, maximum (%)	3.45	3.81	1.07
Local Zr/H_2O location (ft)	7.5	7.5	7.5
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3
Hot rod burst time (sec)	52.8	52.4	N/A
Hot rod burst location (ft)	6.0	6.25	N/A

Results for N-1 Loop	C_D = 1.0 DECLG(1)	C_D = 0.8 DECLG(1)	C_D = 0.4 DECLG(1)	C_D = 0.8 DECLG(2)
Peak clad temperature (^oF)	1693	1742	1713	1610
Peak clad temperature location (ft)	7.5	7.35	7.25	7.25
Local Zr/H_2O reaction, maximum (%)	1.00	1.21	1.12	0.70
Local Zr/H_2O location (ft)	7.25	7.25	7.25	7.25
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3	<0.3
Hot rod burst time (sec)	N/A	N/A	N/A	N/A
Hot rod burst location (ft)	N/A	N/A	N/A	N/A

(1) Break in active loop
(2) Break in idle loop

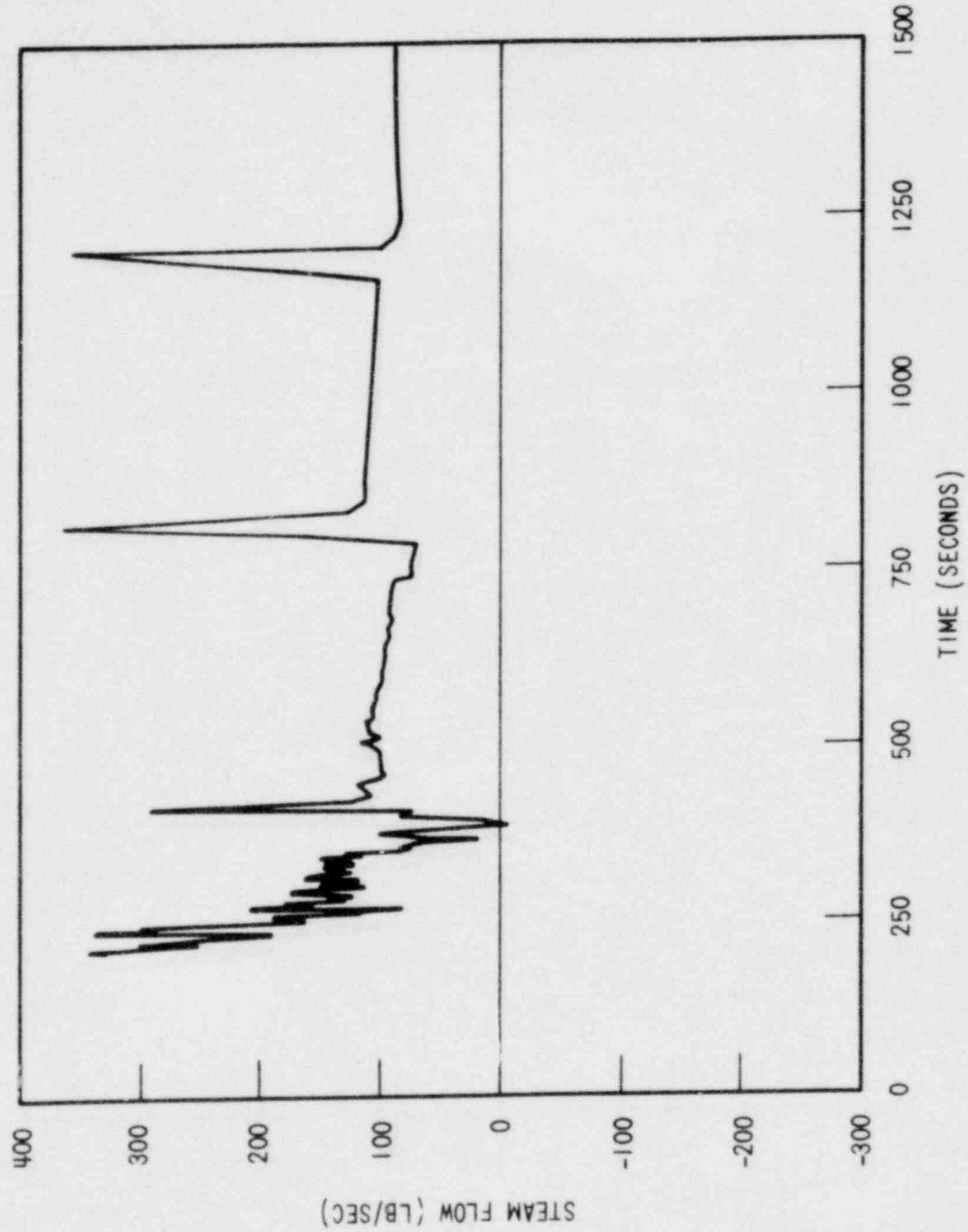

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Figure 15.6-38.

Steam Flow (4 Inch Break)

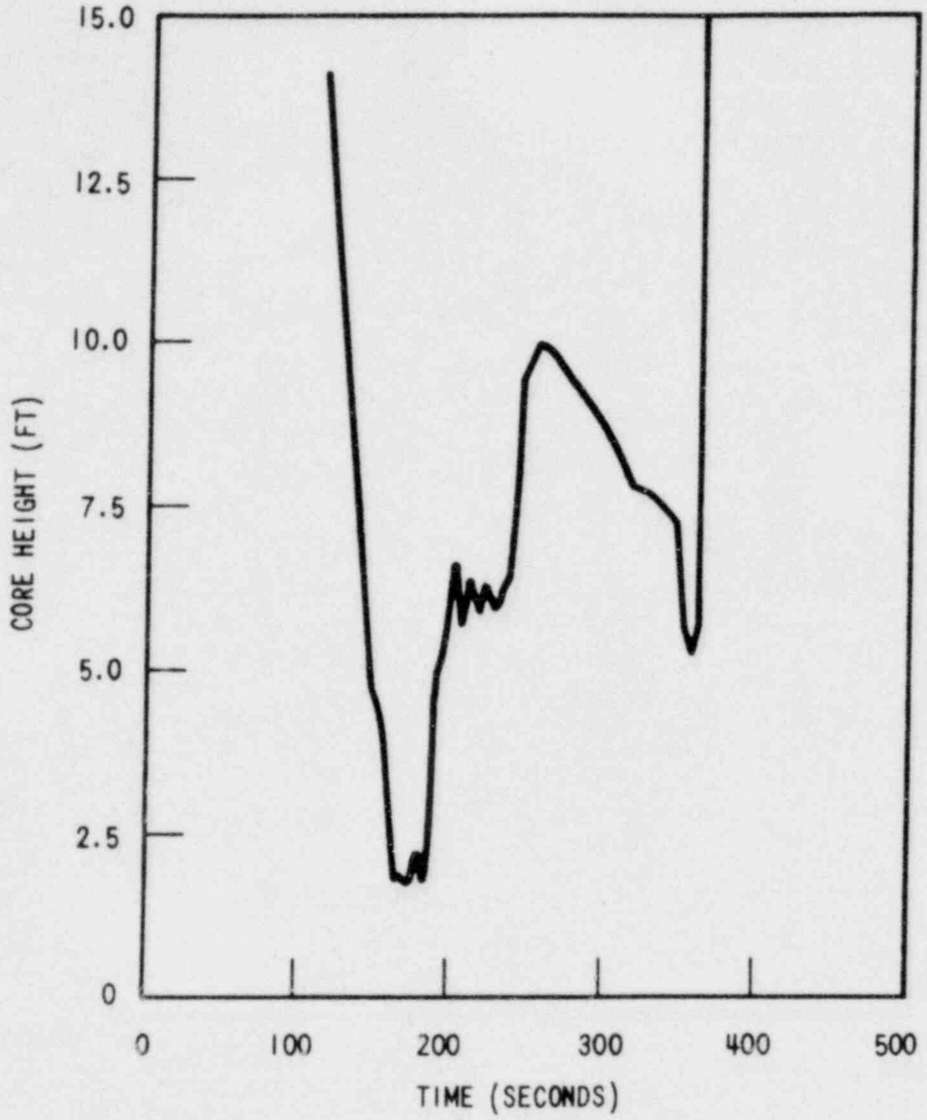

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Figure 15.6-44.

Core Mixture Height (6 Inch Break)

16132-1

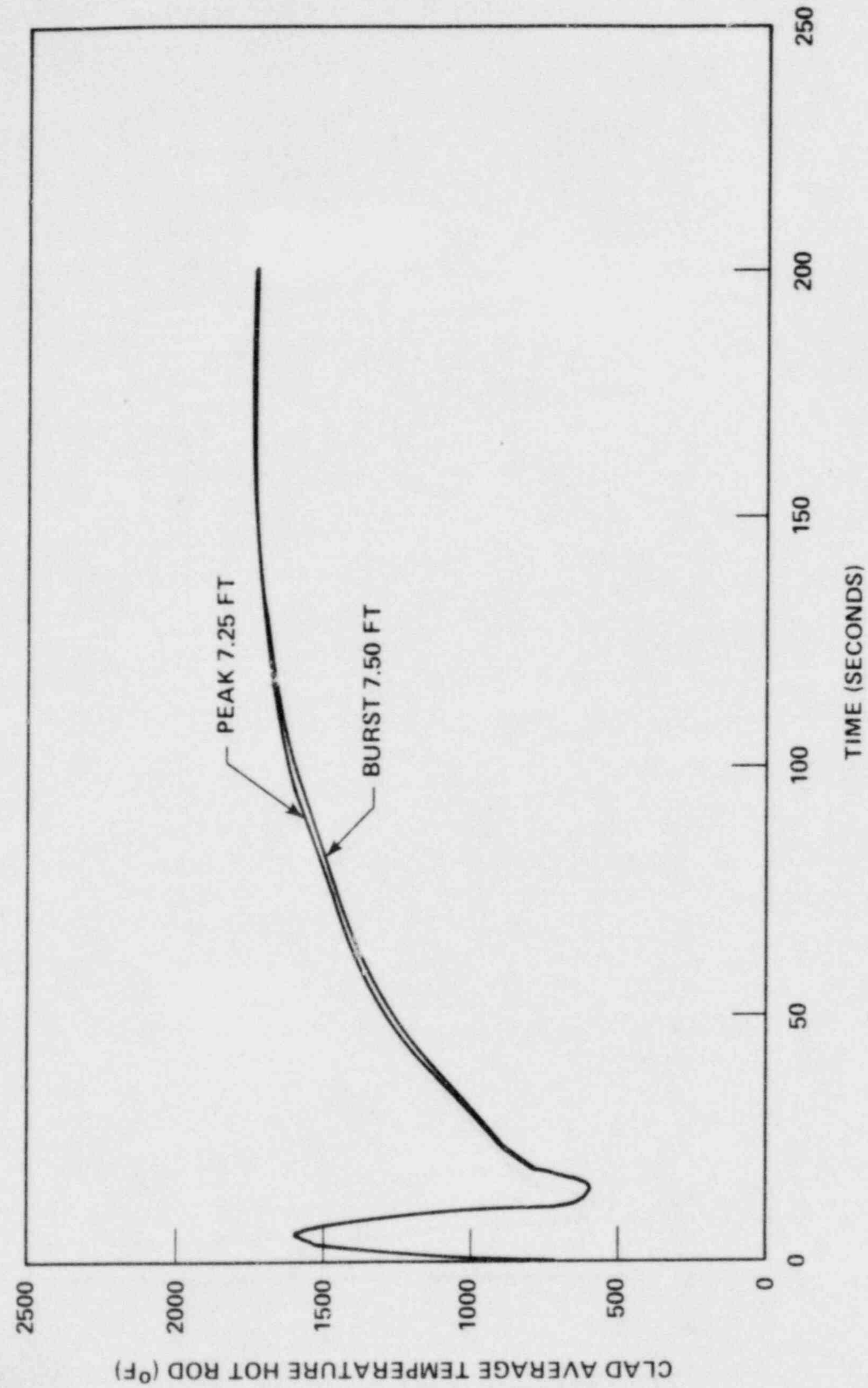

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Figure 15.6-49

Peak Clad Temperature - DECLG (C_D = 0.8)

16132-2

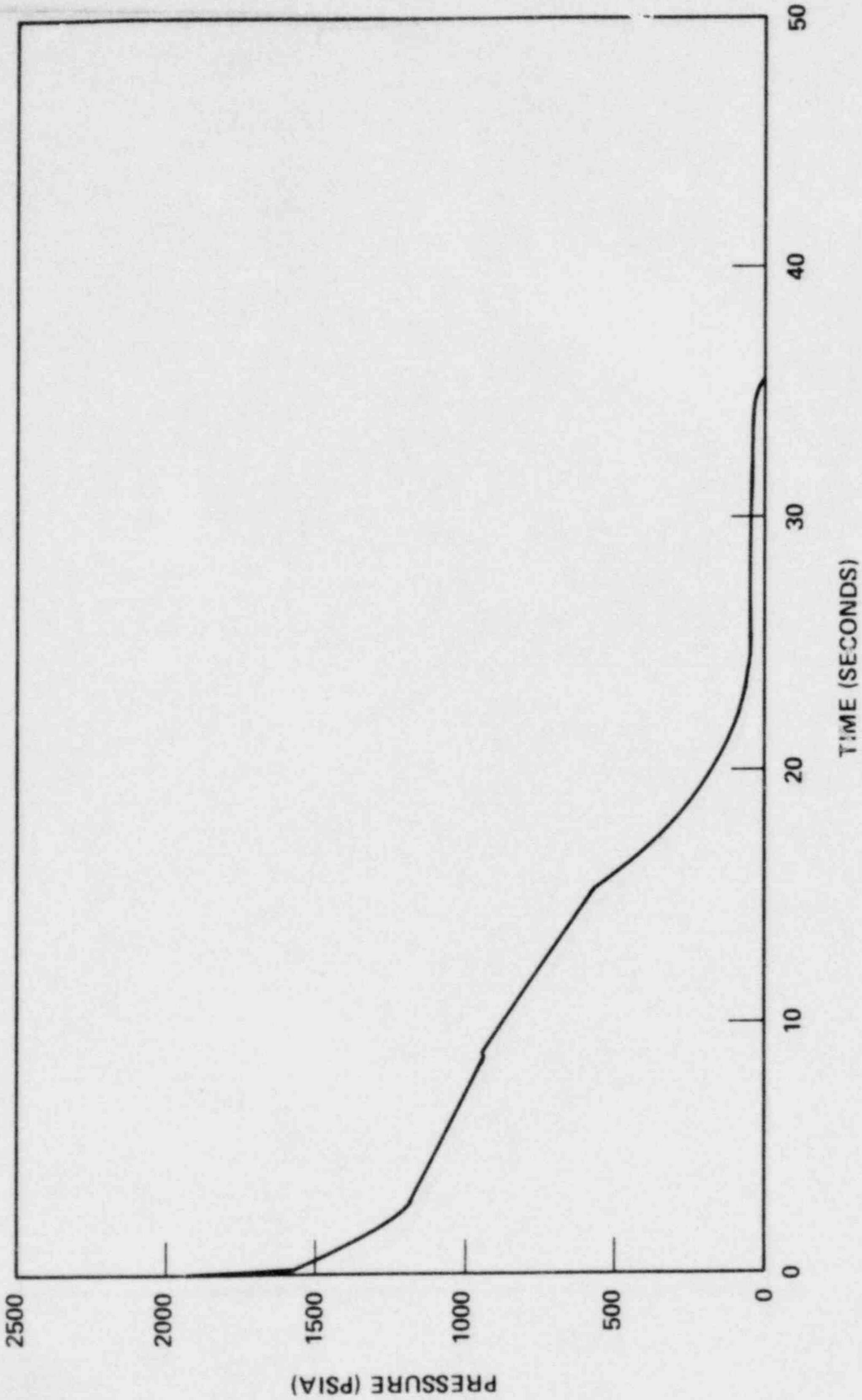

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Figure 15.6-50

Core Pressure - DECLG (C_D = 0.8)

16132-3

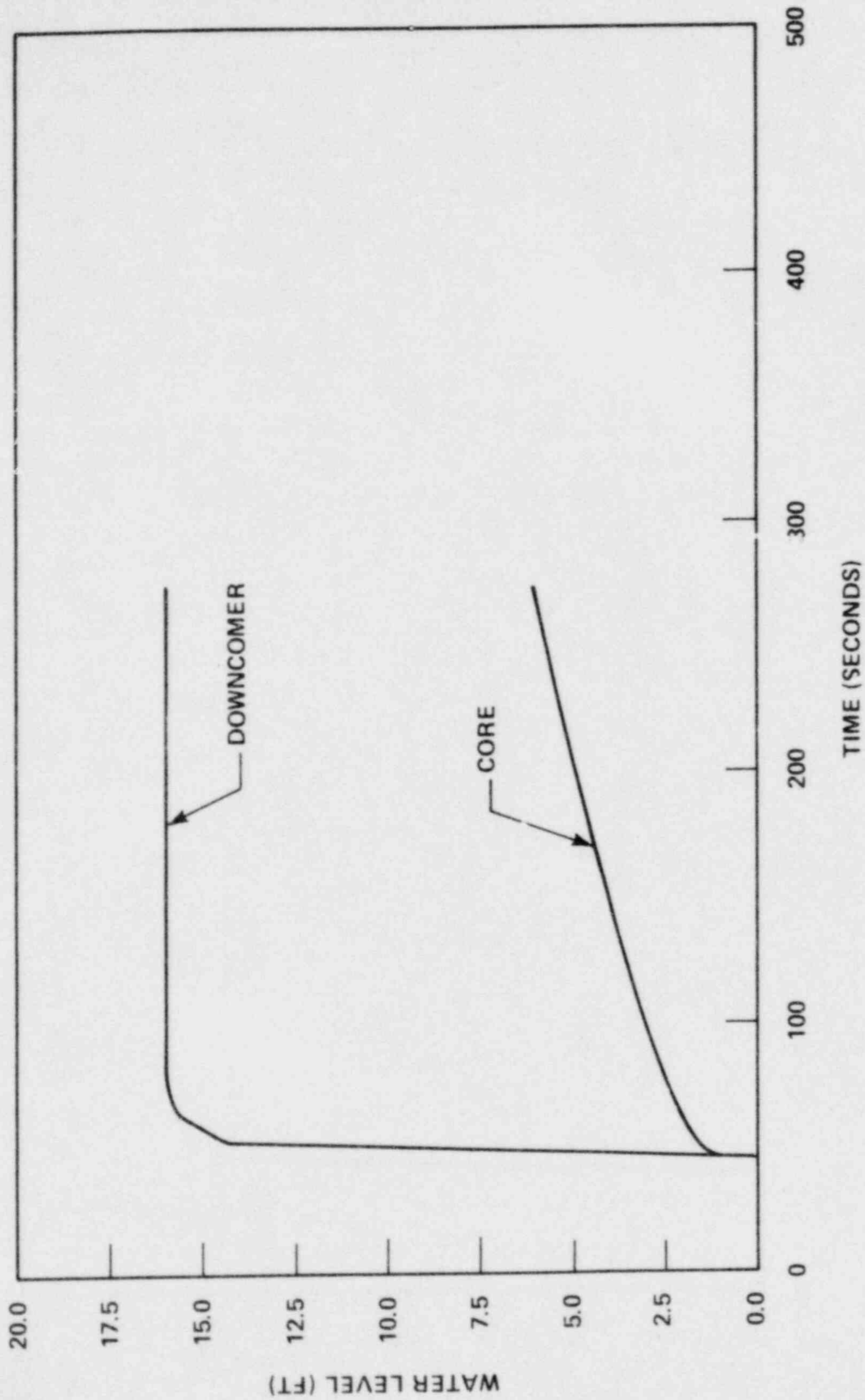

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Figure 15.6-51

DECLG (C_D = 0.8) Downcomer and Core Water Levels During Reflood

16132-4

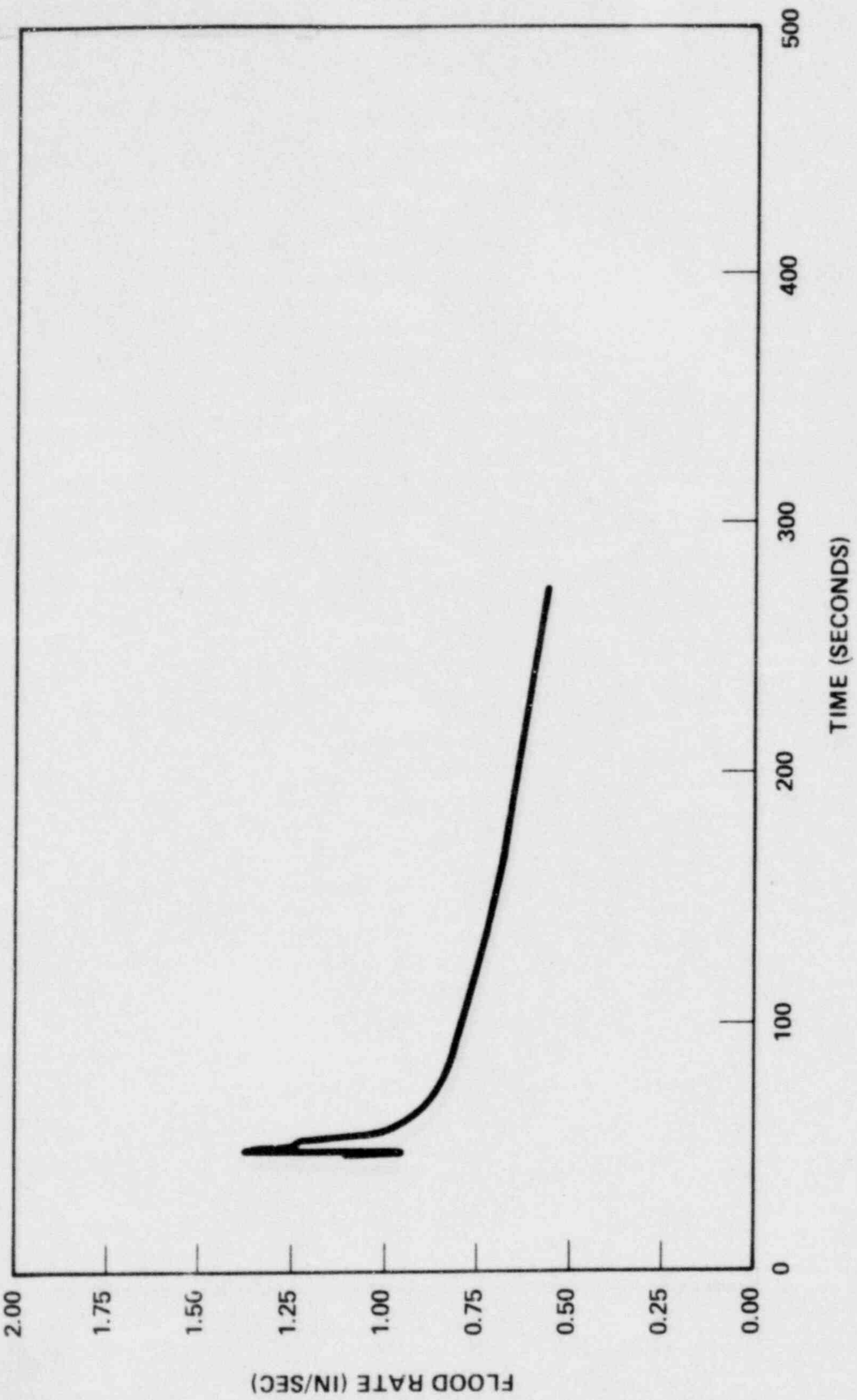

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Figure 15.6-52

DECLG (C_D = 0.8) Core Inlet Velocity During Reflood

16132-5

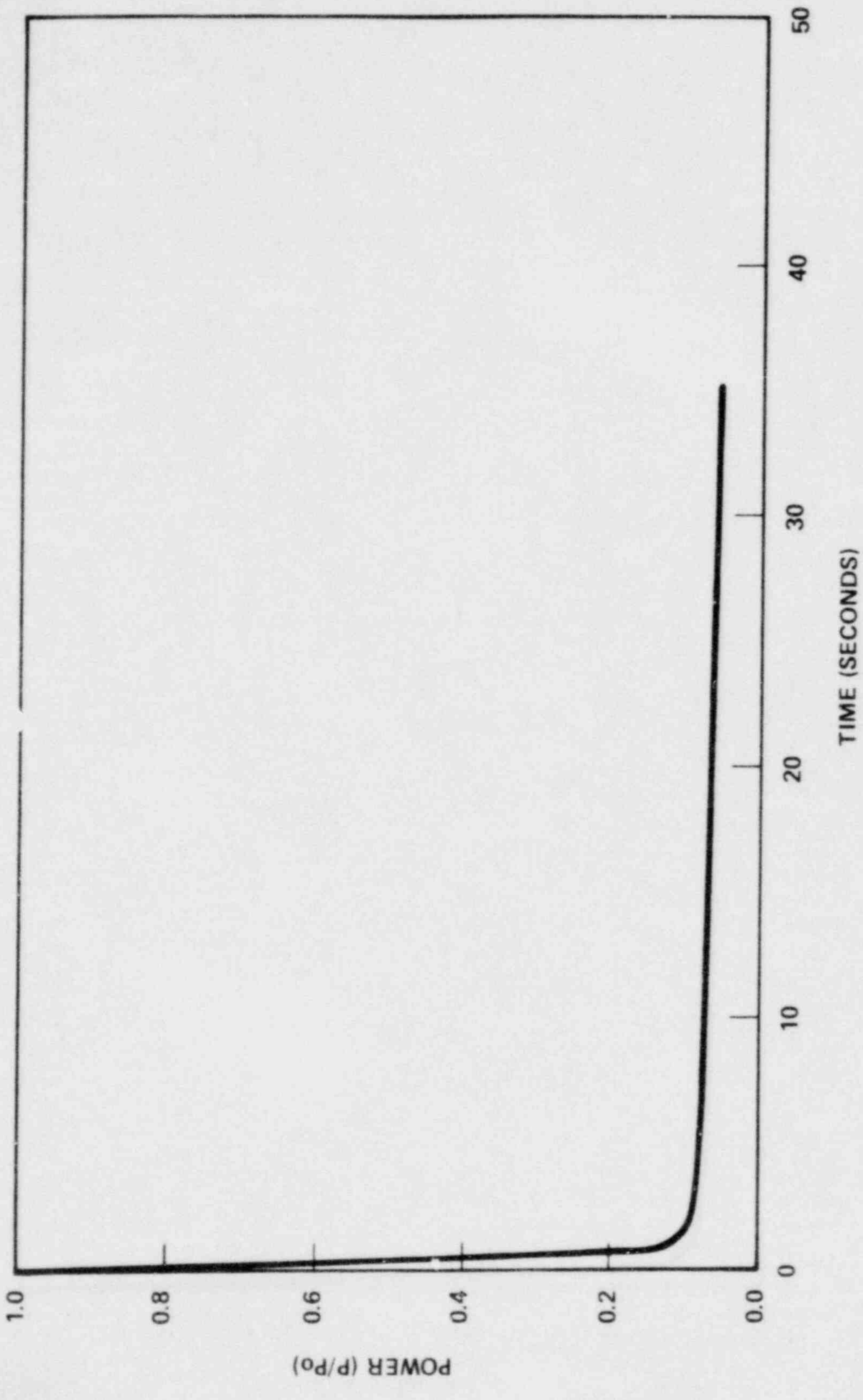

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Figure 15.6-53

Core Power Transient – DECLG (C_D = 0.8)

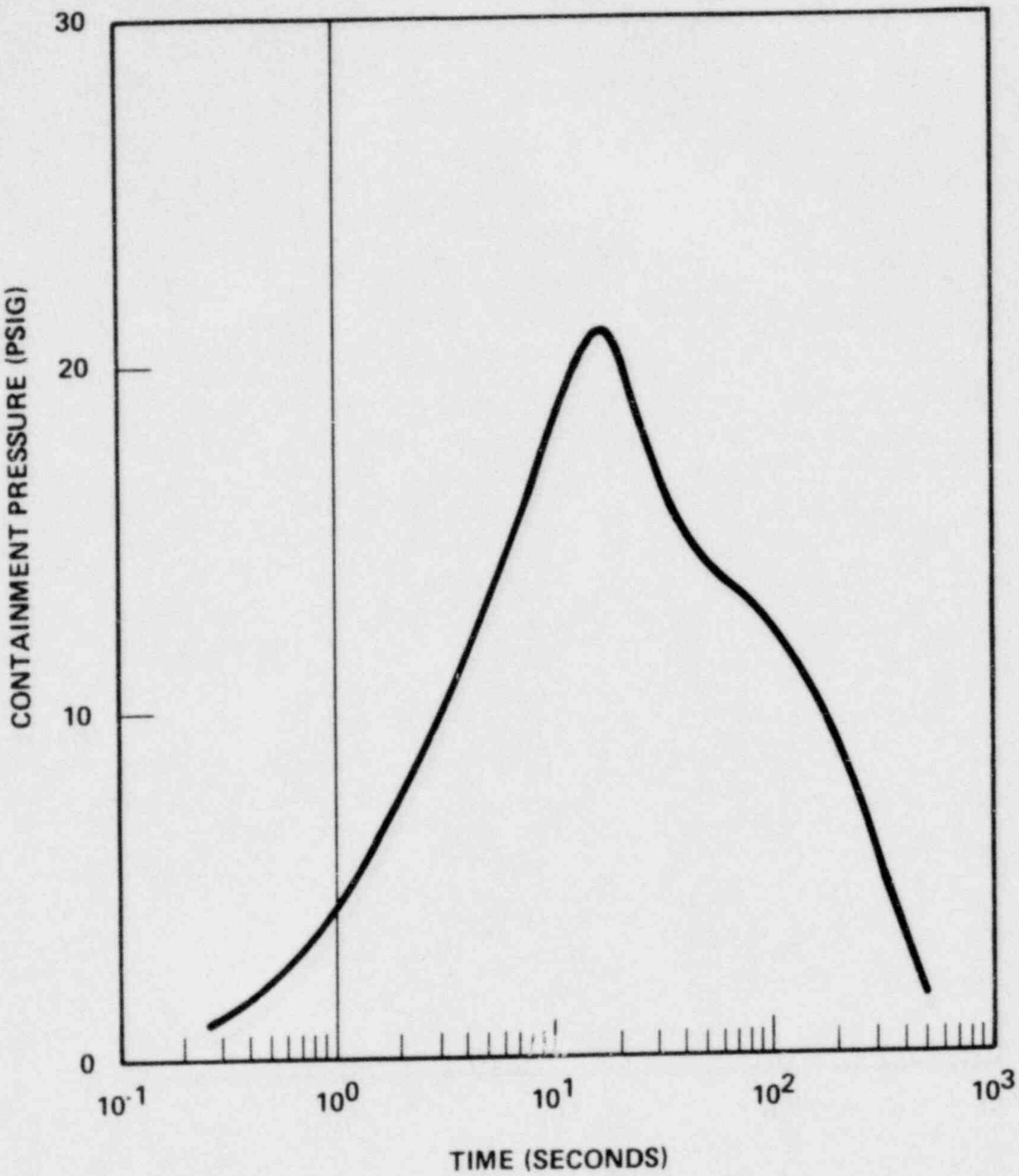

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Figure 15.6-54

Containment Pressure – DECLG (C_D = 0.8)

16132-7

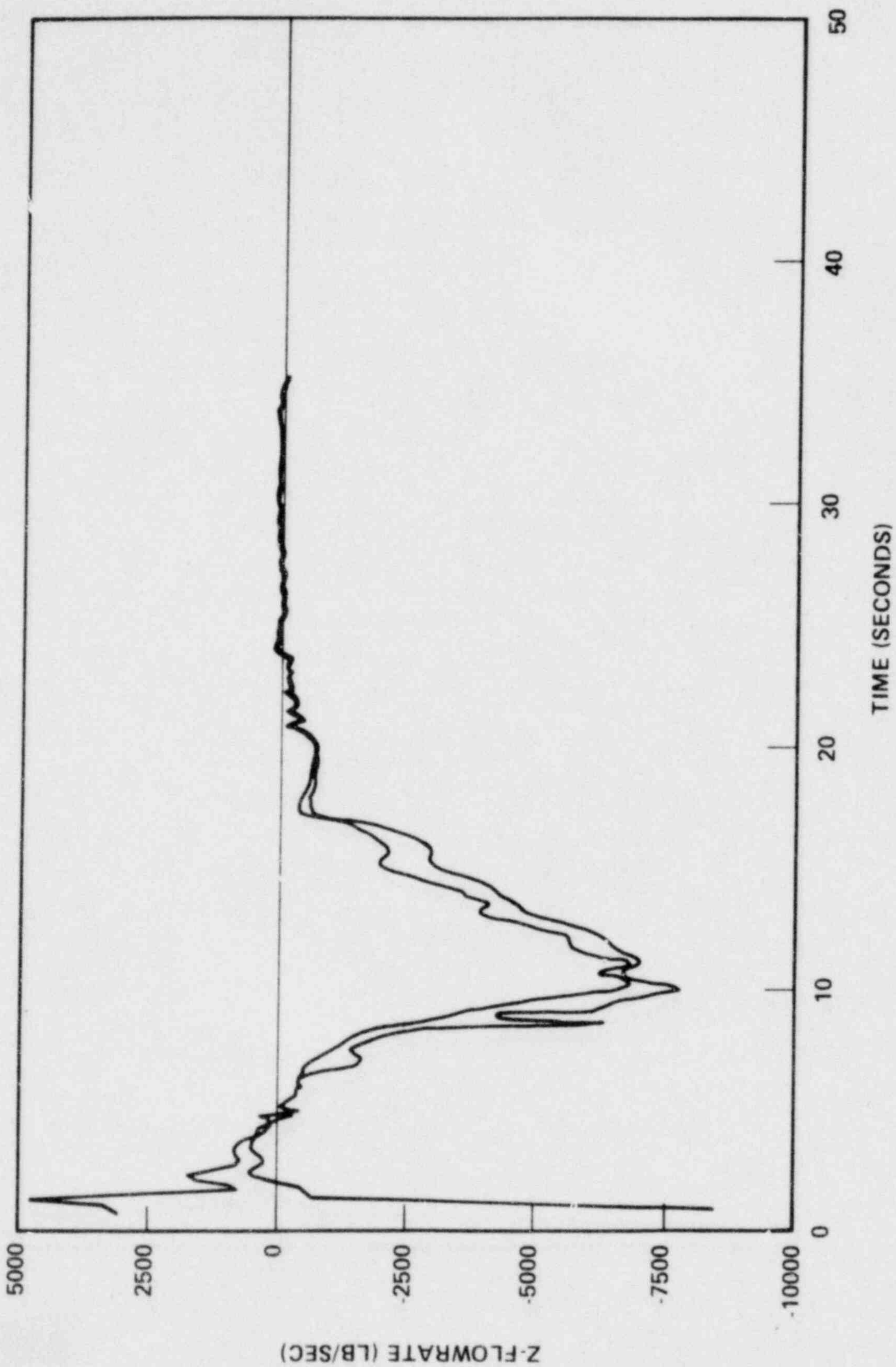

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Figure 15.6-55

Core Flow – Top and Bottom – DECLG (C_D = 0.8)

16132-8

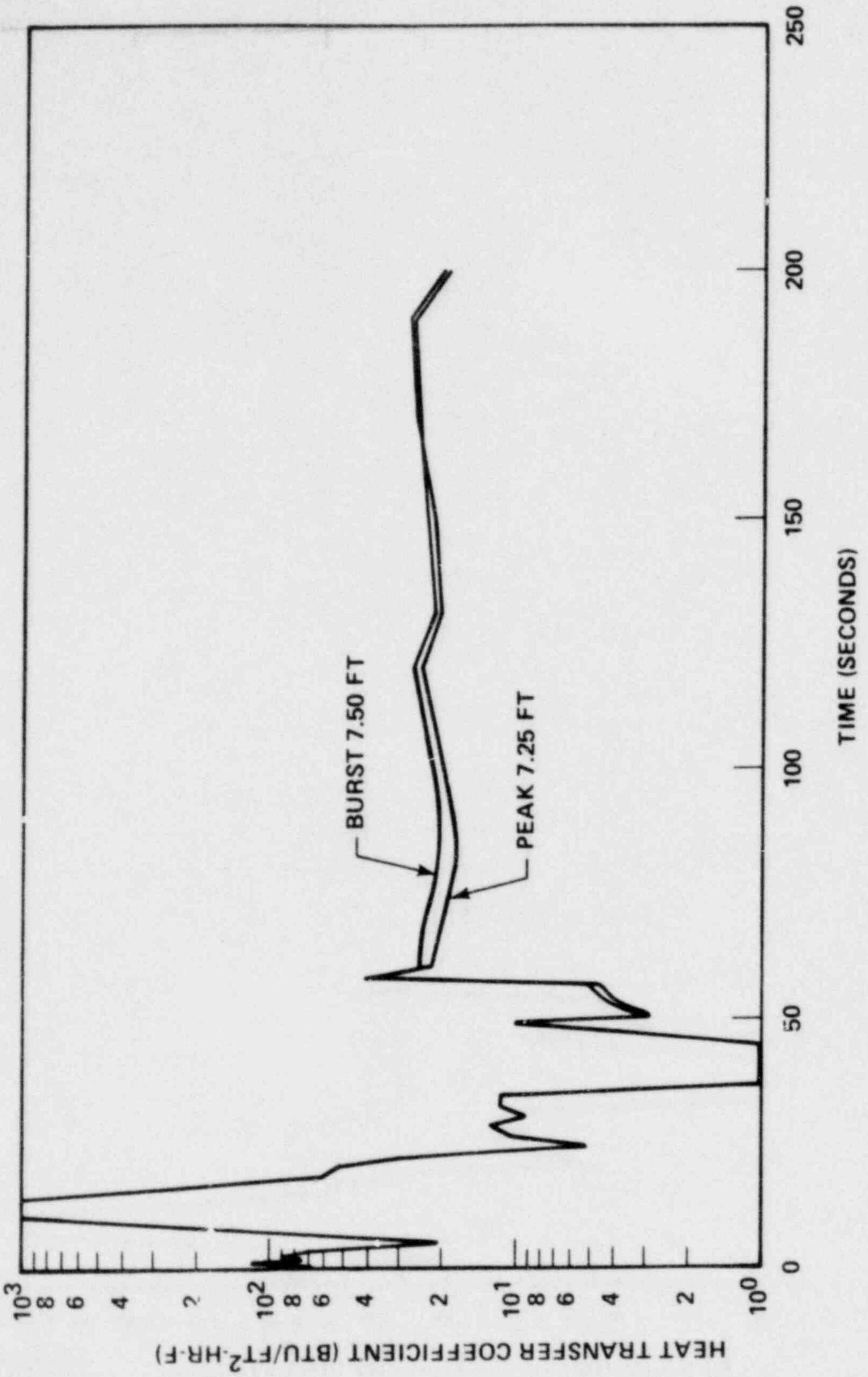

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Figure 15.6-56

Core Heat Transfer Coefficient – DECLG (C_D = 0.8)

16132-9

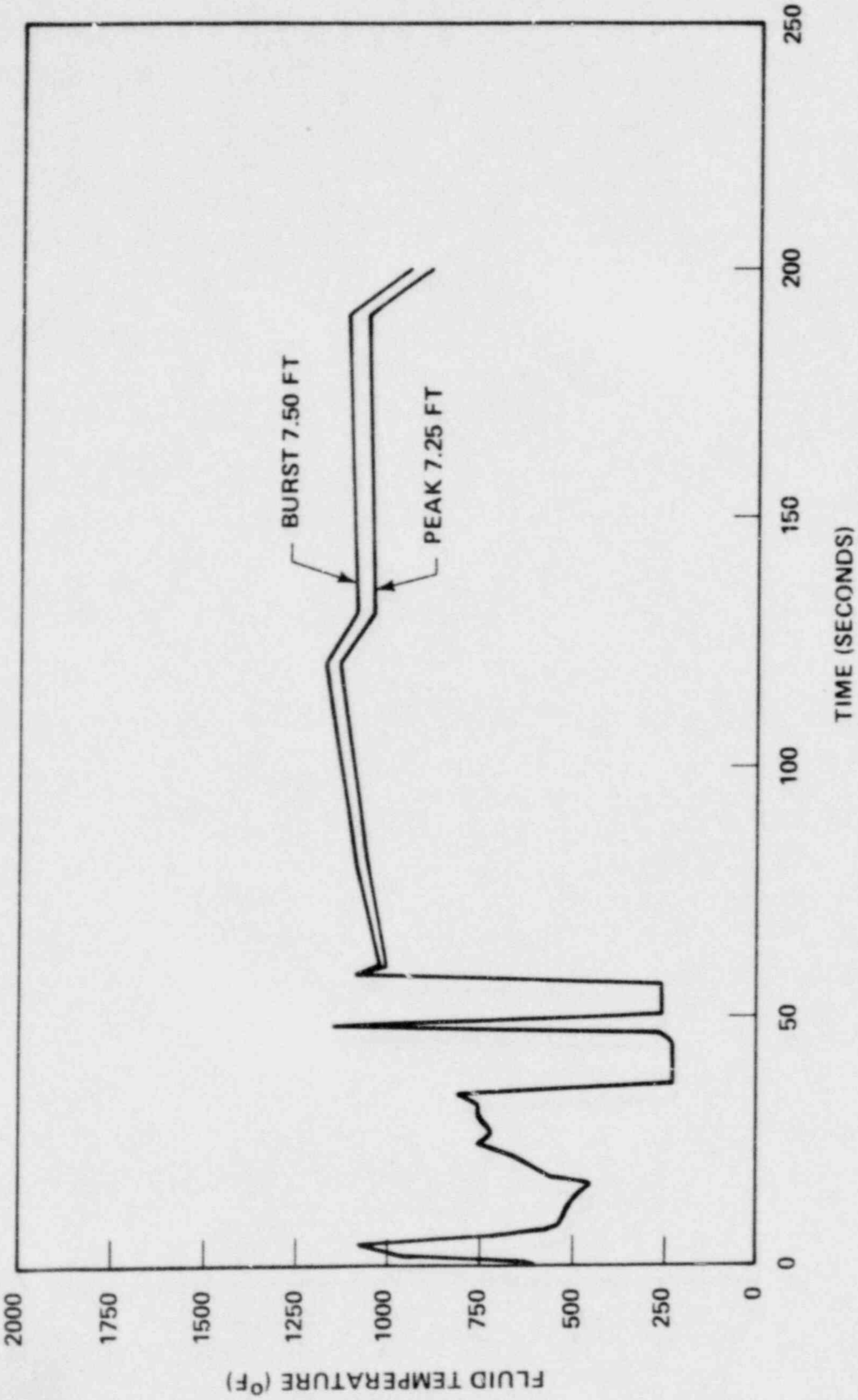

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Figure 15.6-57

Fluid Temperature - DECLG (C_D = 0.8)

16132-10

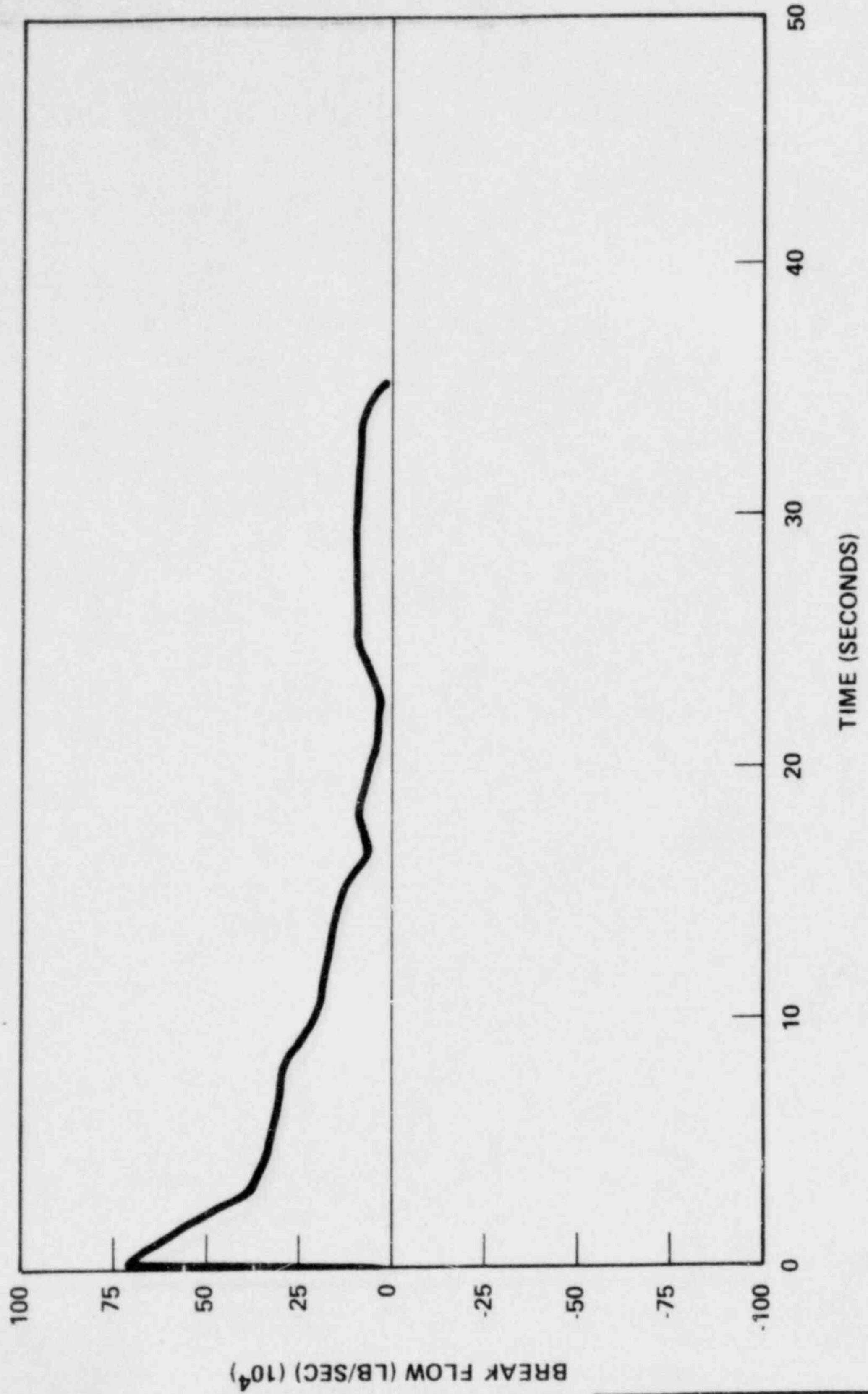

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Figure 15.6-58

Break Mass Flow Rate – DECLG (C_D = 0.8)

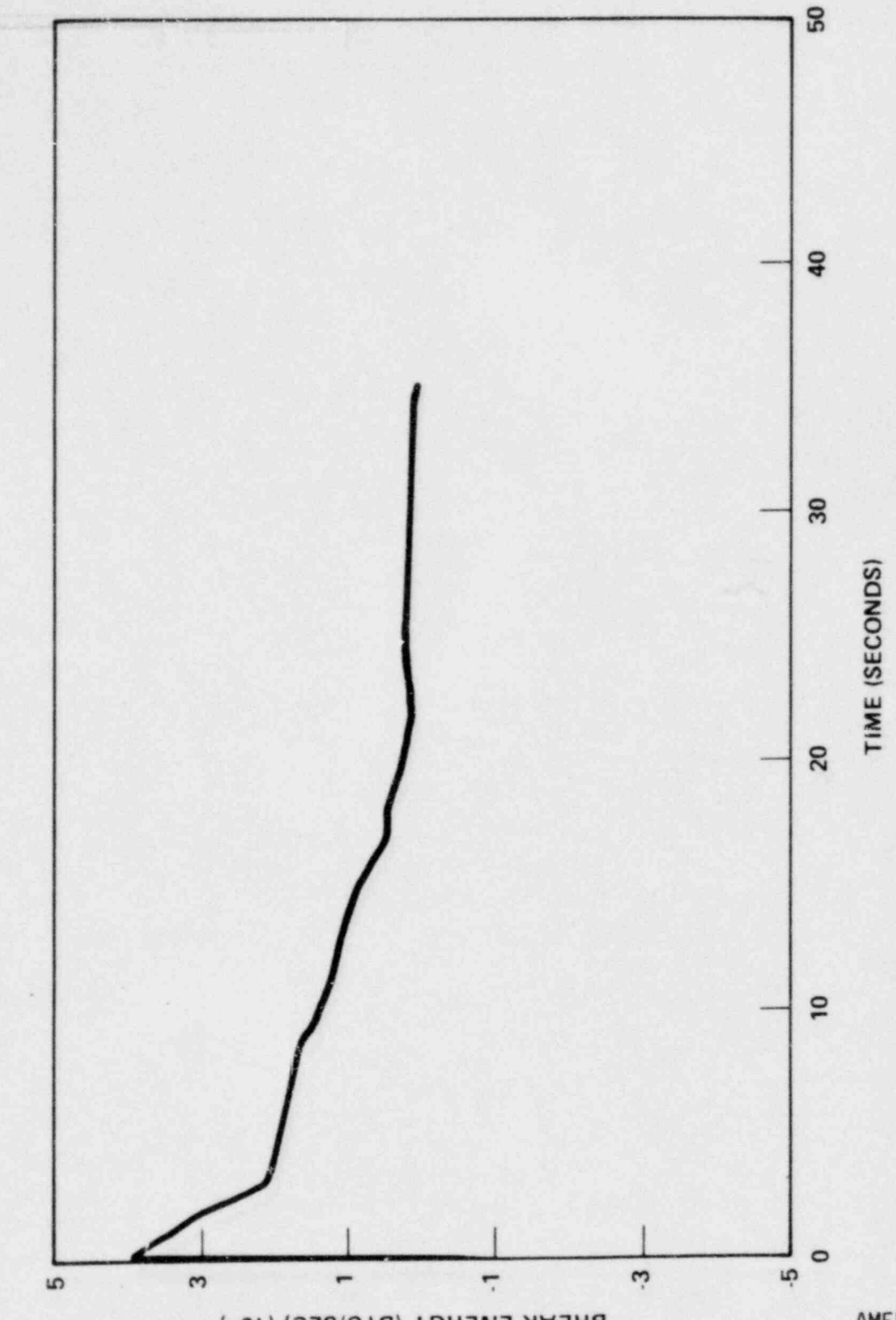

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Figure 15.6-59

Break Energy Release Rate
– DECLG (C_D = 0.8)

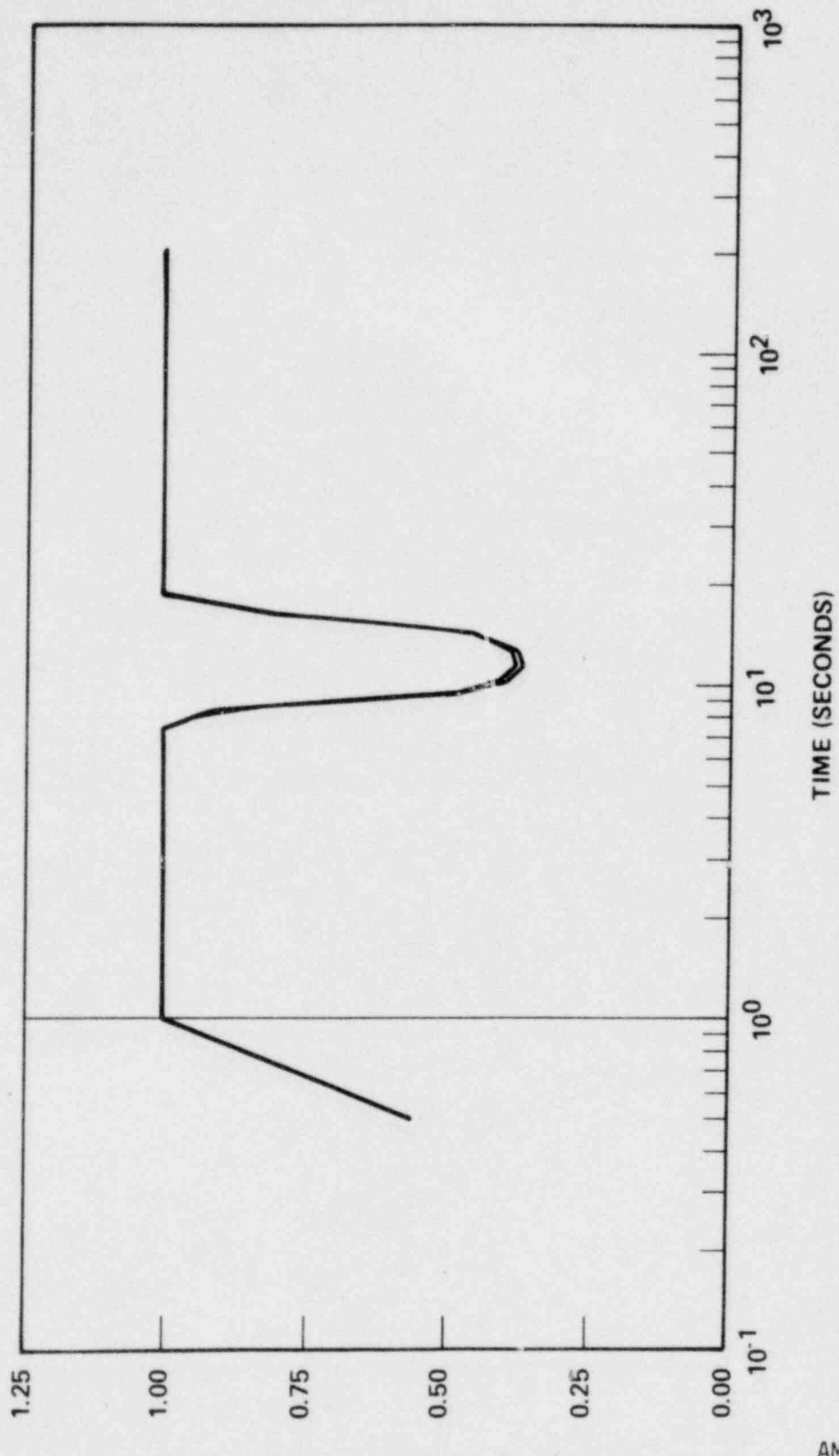

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Figure 15.6-60

Fluid Quality – DECLG (C_D = 0.8)

16132-13

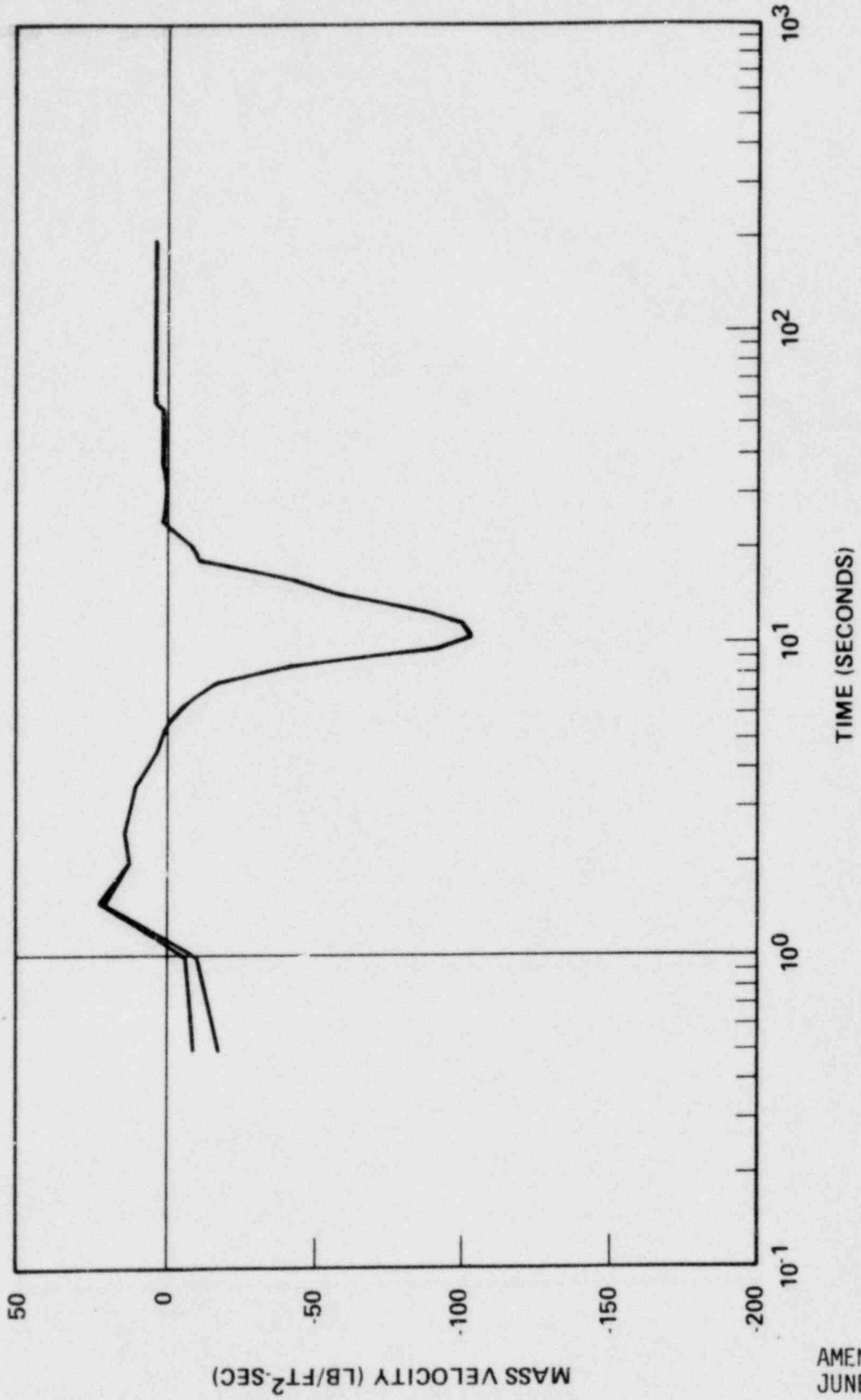

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Figure 15.6-61

Mass Velocity DECLG (C_d = 0.8)

16132-14

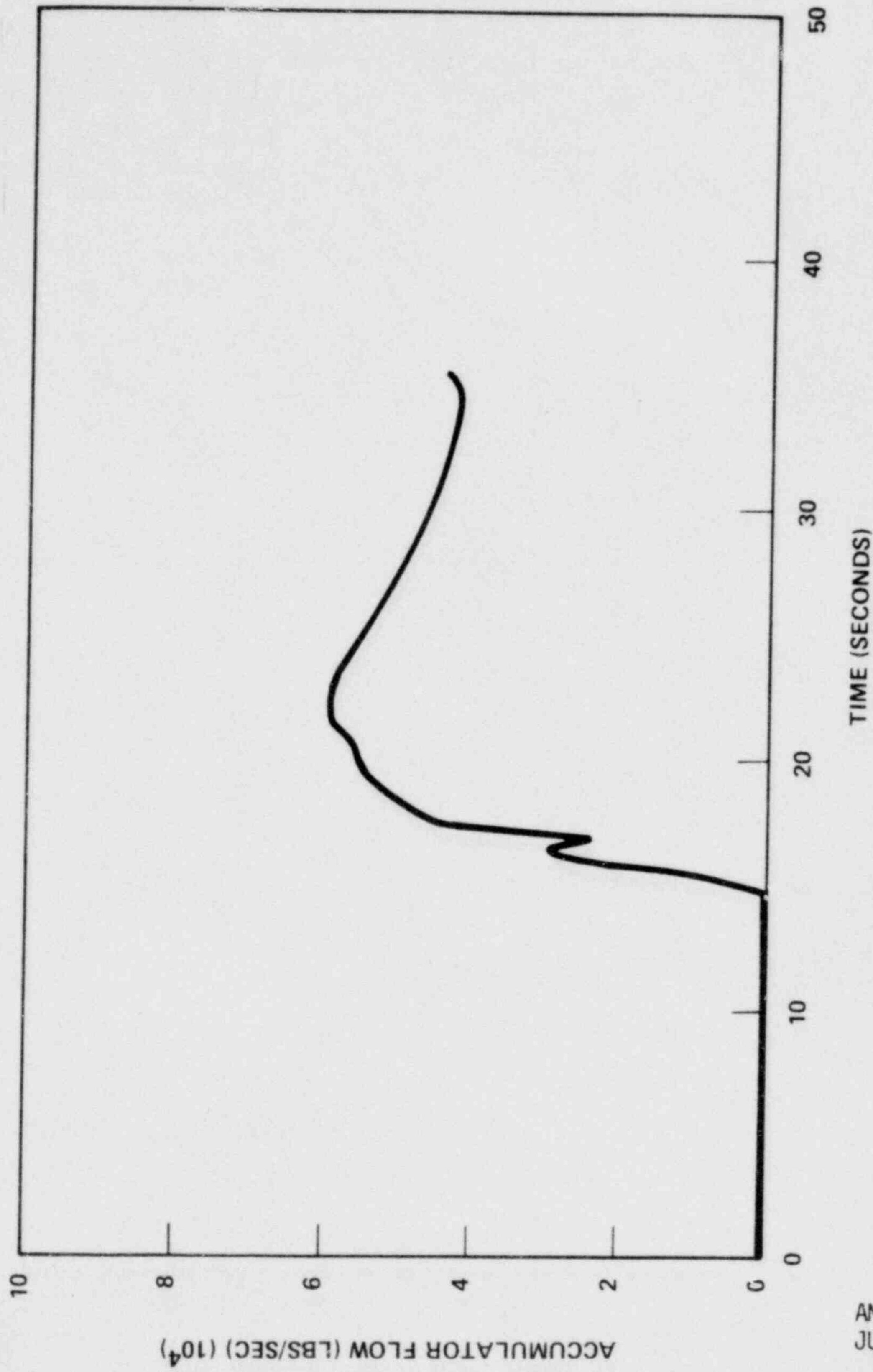

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Figure 15.6-62

Intact Loop Accumulator Flow During Blowdown — DECLG (C_D = 0.8)

16132-15

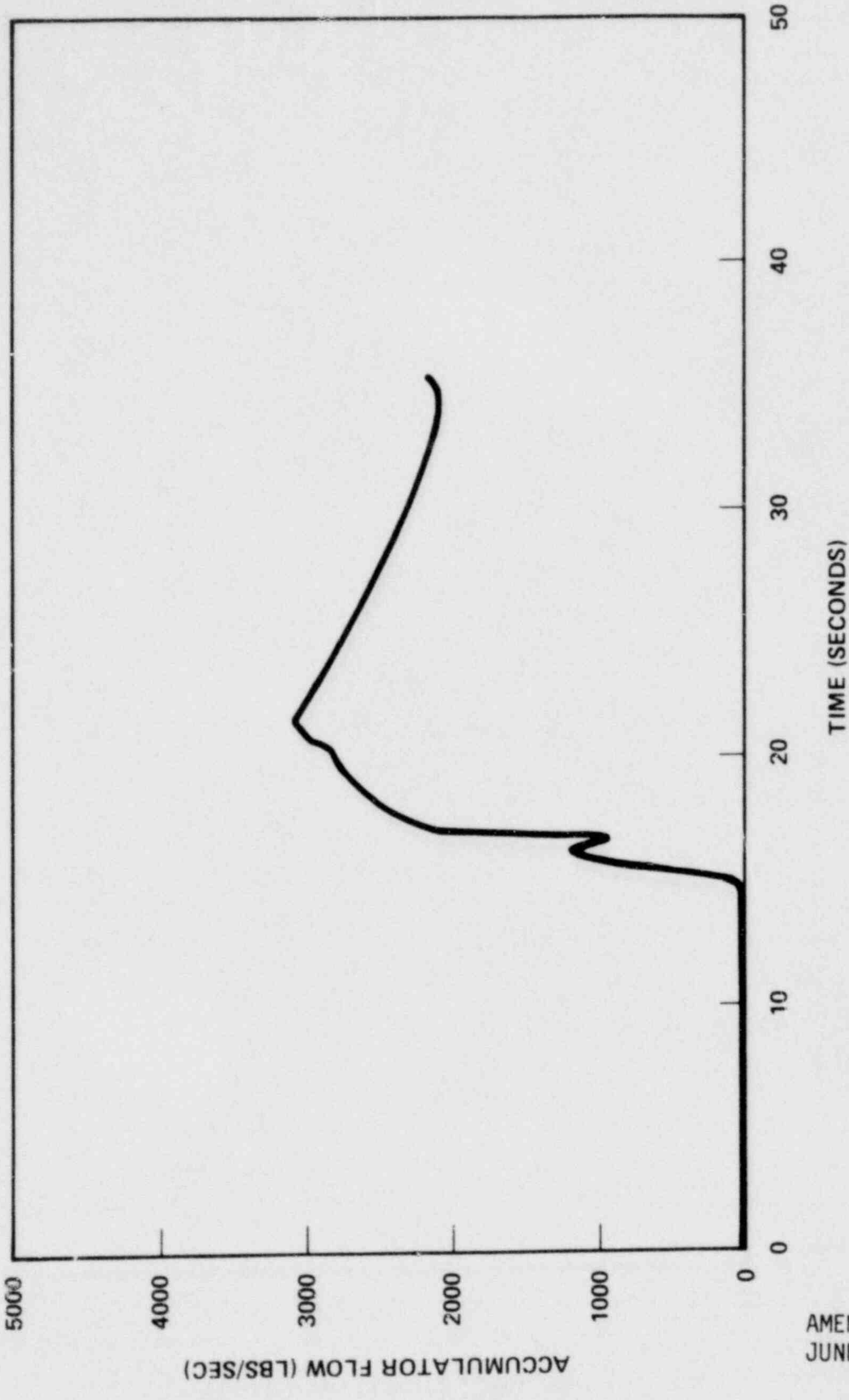

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Figure 15.6-63

Idle Loop Accumulator Flow During Blowdown – DECLG (C_D = 0.8)

16132-16

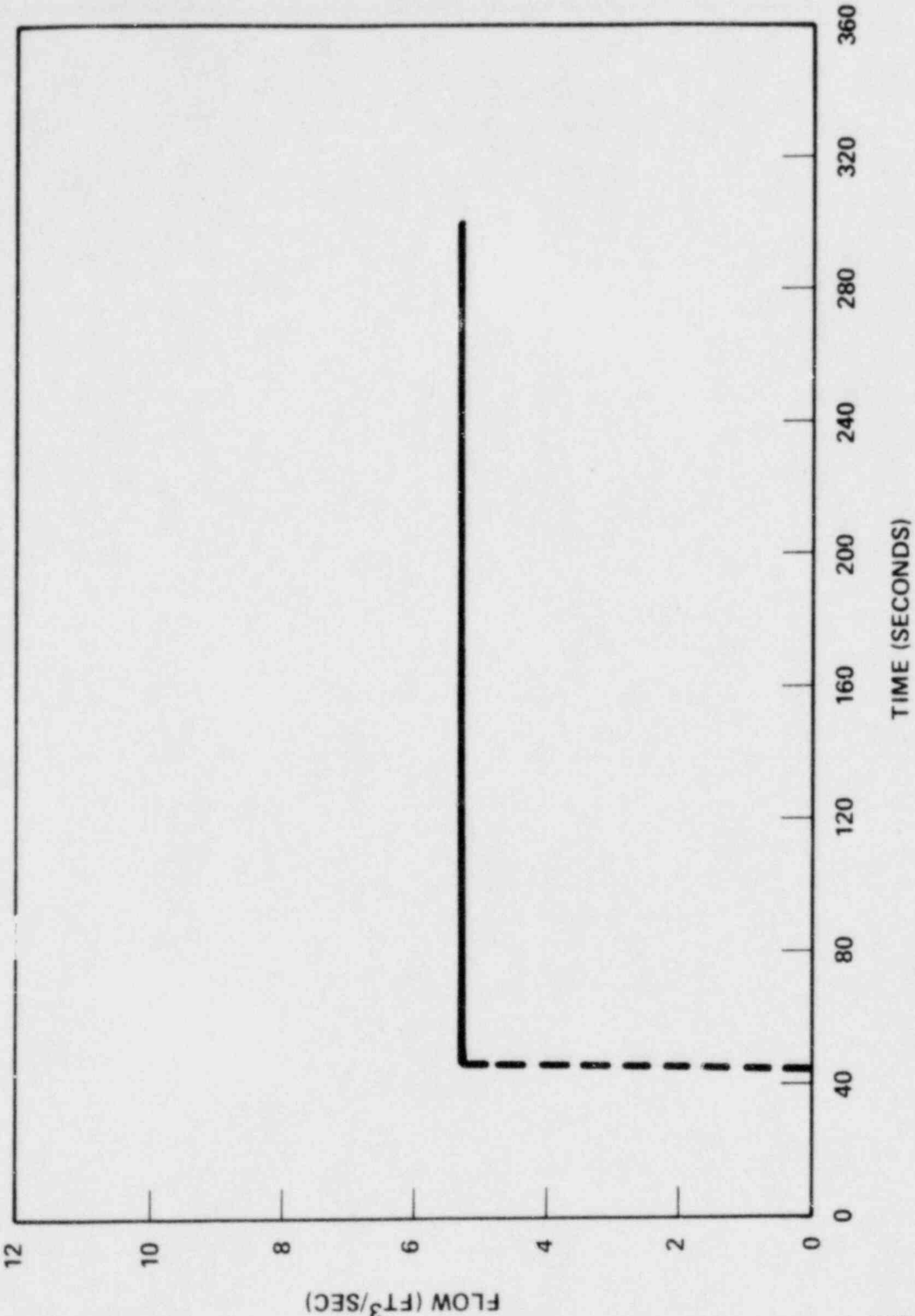

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Figure 15.6-64

Pumped ECCS Flow During Reflood – DECLG (C_D = 0.8)

16132-17

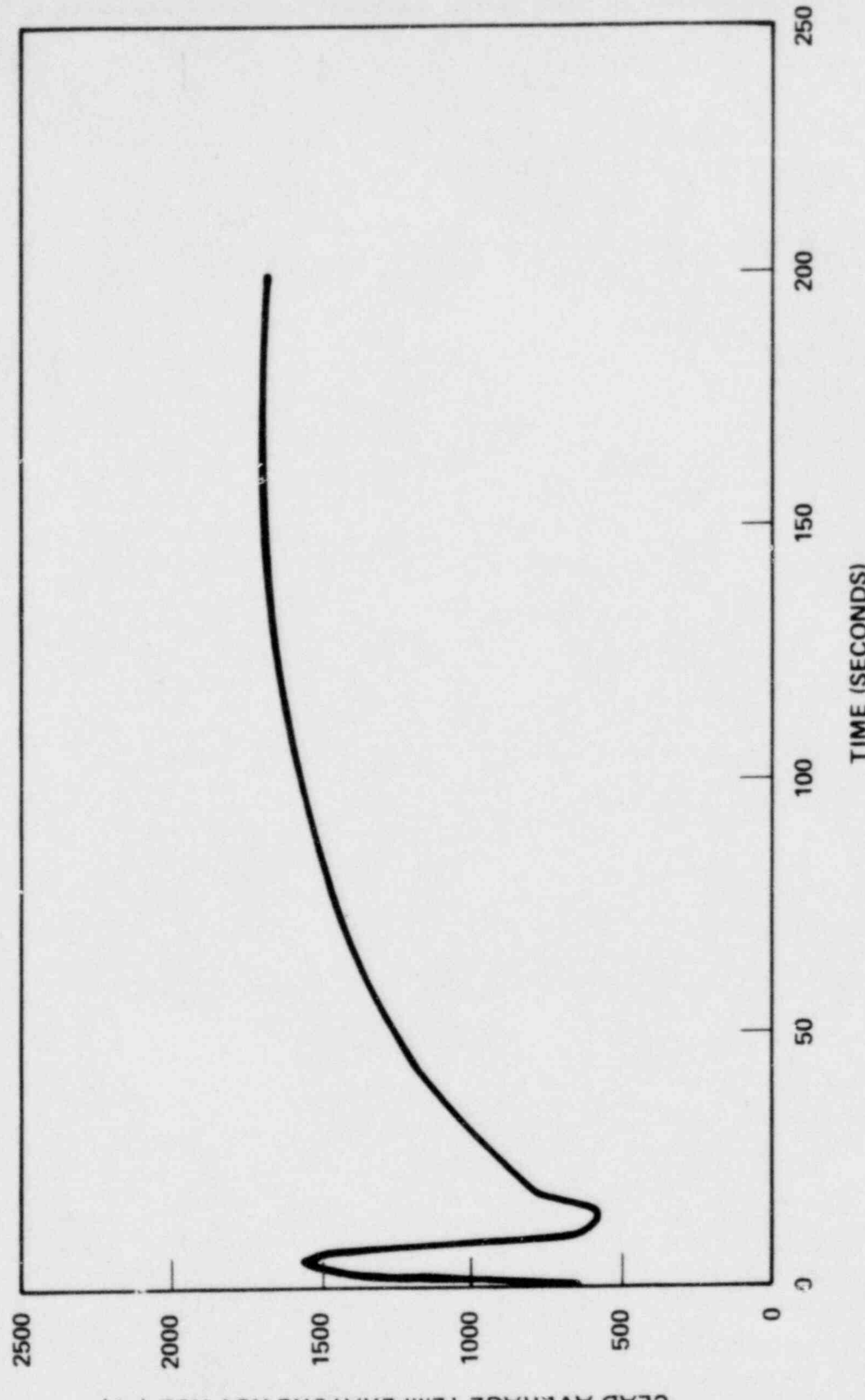

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Figure 15.6-65

Peak Clad Temperature
– DECLG (C_D = 1.0)

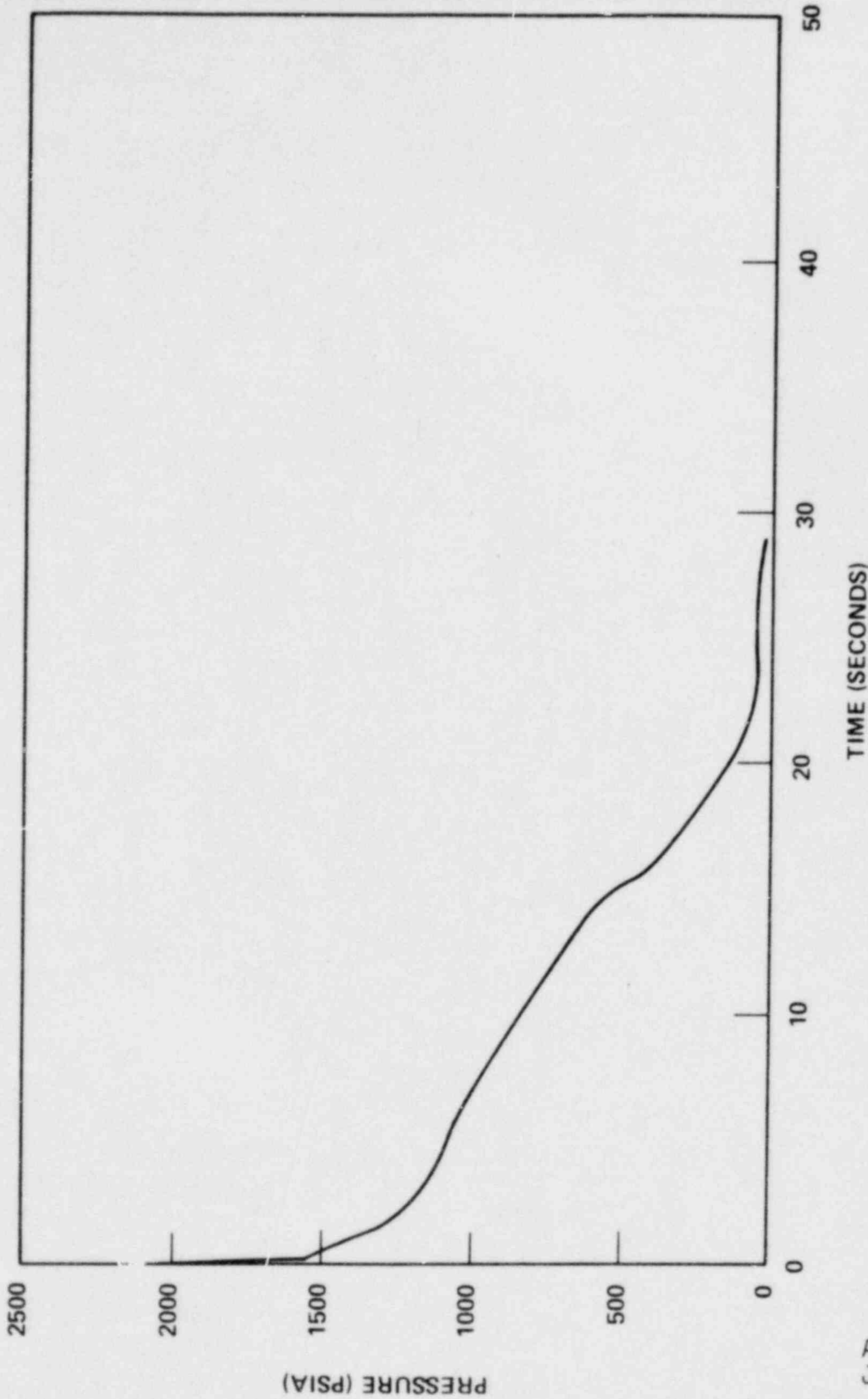

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Figure 15.6-66

Core Pressure -
DECLG (C_D = 1.0)

16132-19

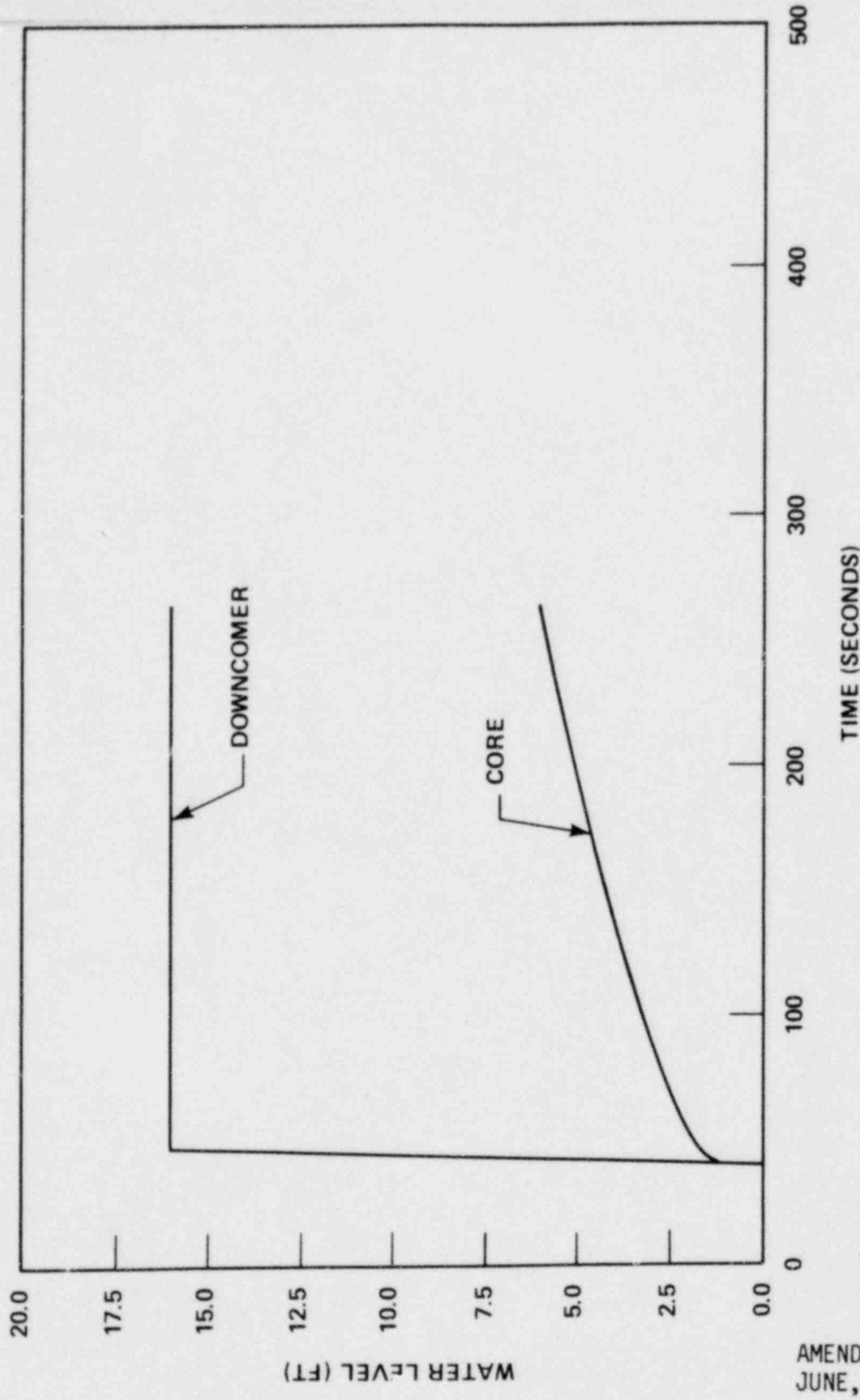

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Figure 15.6-67

DECLG (C_D = 1.0) Downcomer and Core Water Levels During Reflood

16132-20

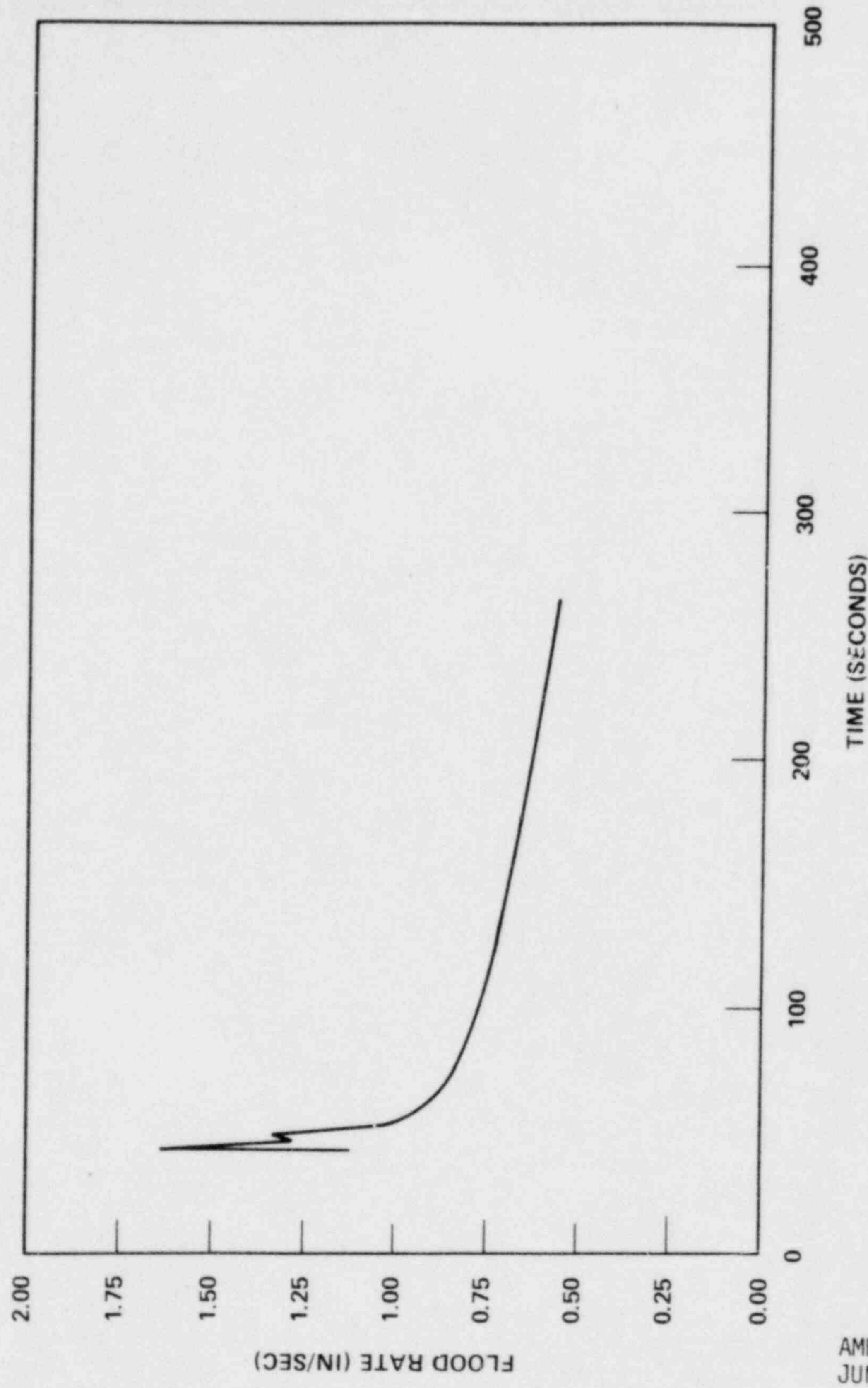

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Figure 15.6-68

DECLG (C_D = 1.0) Core Inlet Velocity During Reflood

16132-21

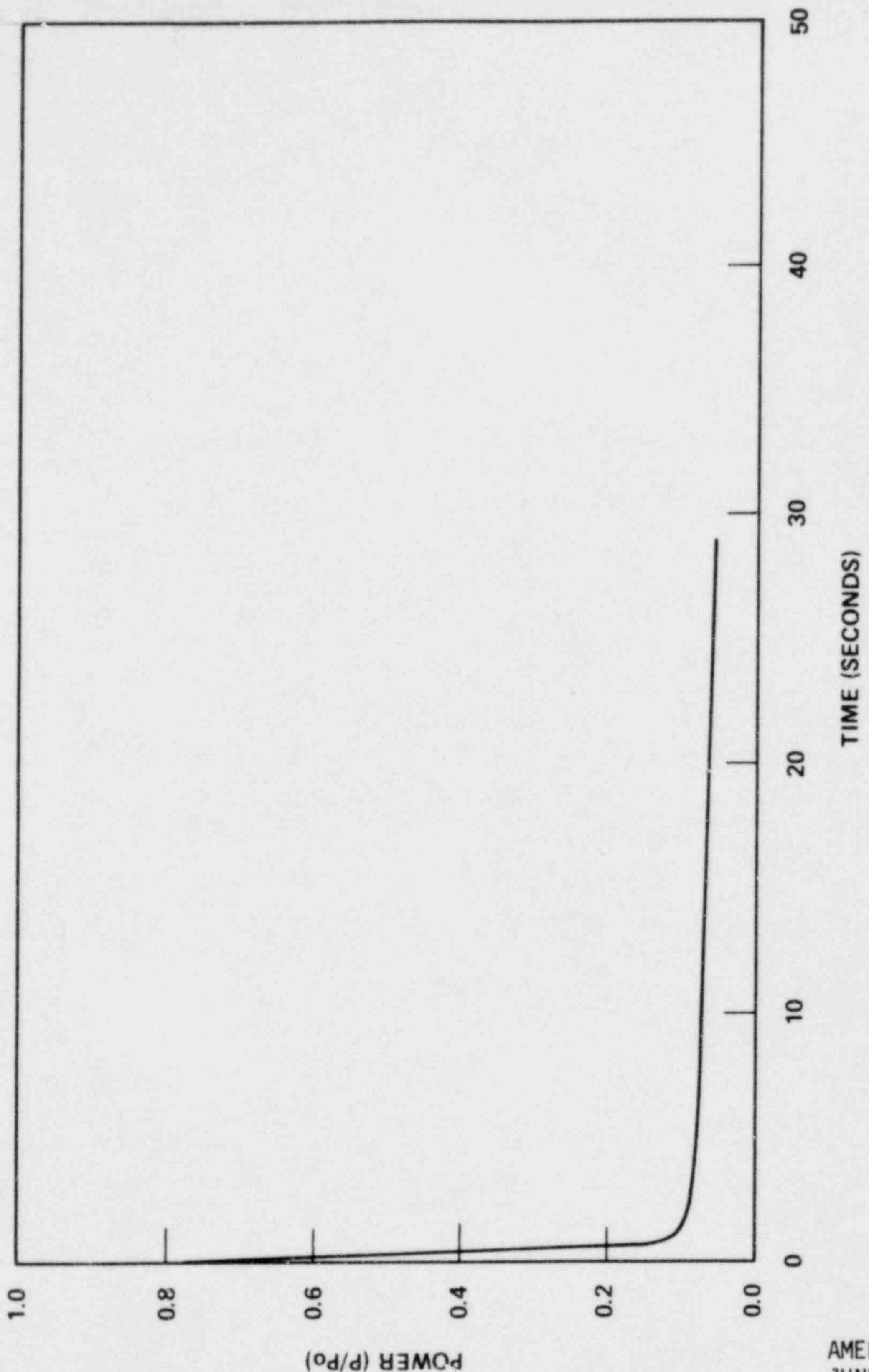

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Figure 15.6-69

Core Power Transient -
DECLG (C_D = 1.0)

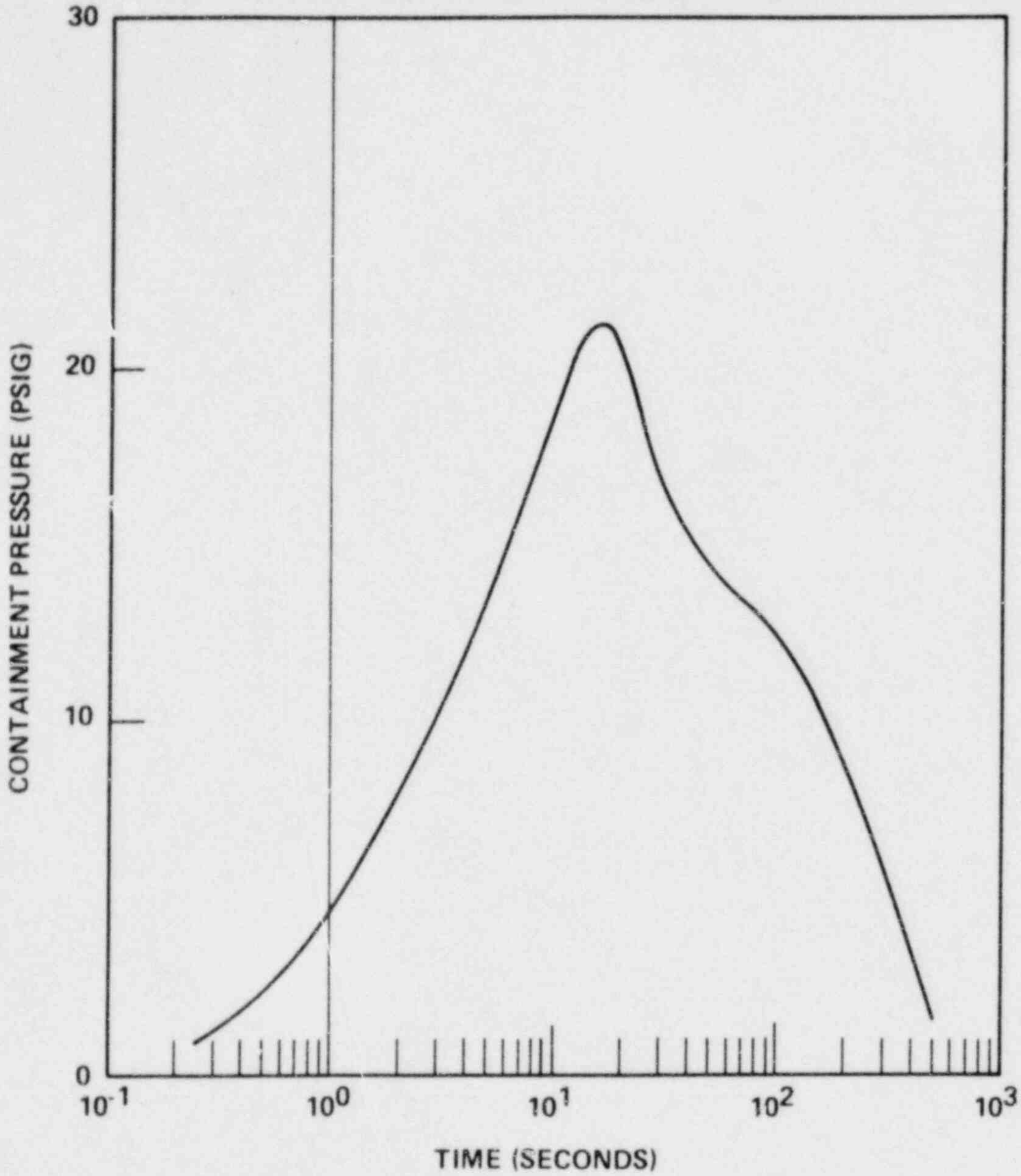

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Figure 15.6-70

Containment Pressure - DECLG (C_D = 1.0)

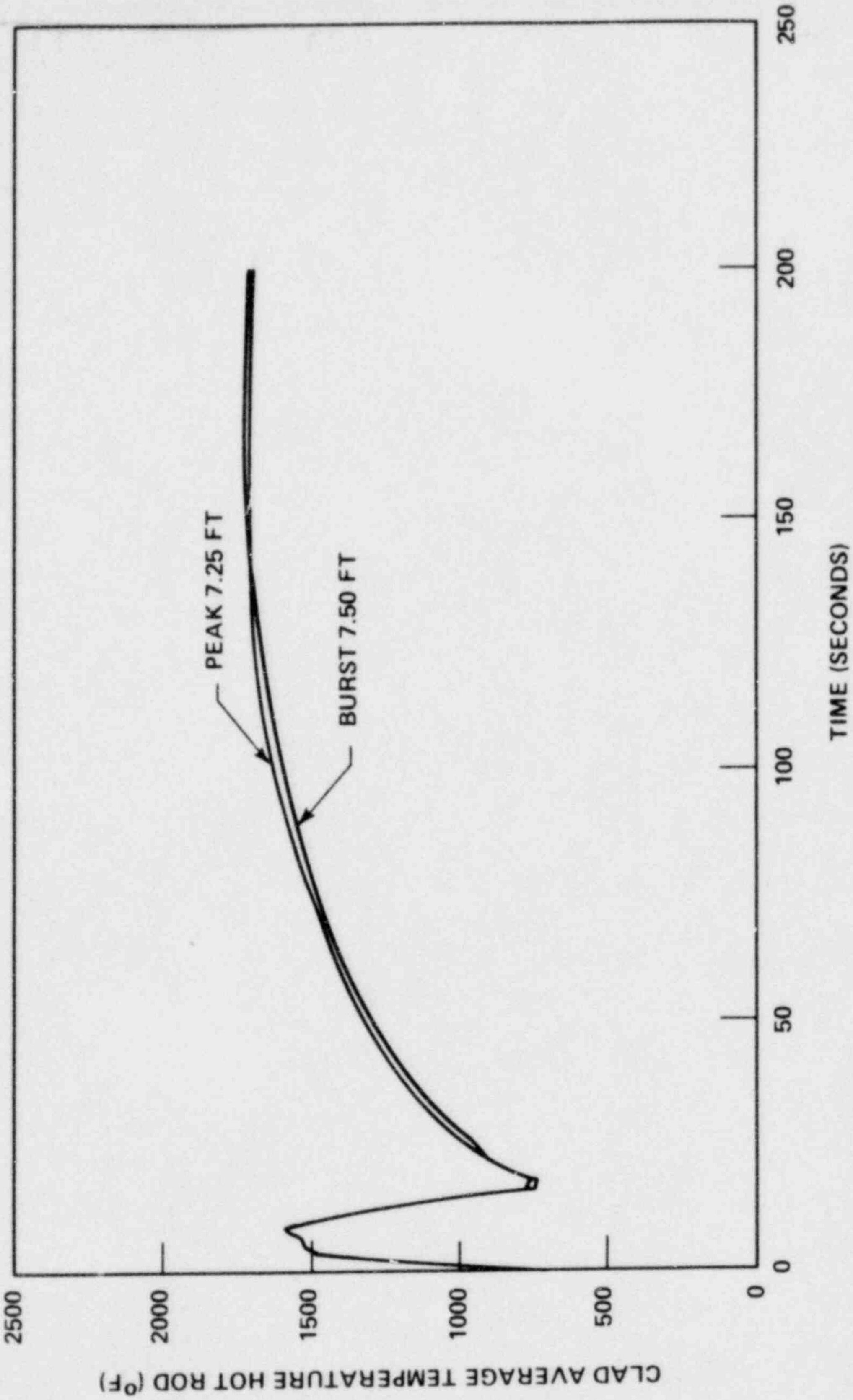

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Figure 15.6-71

Peak Clad Temperature - DECLG (C_D = 0.6)

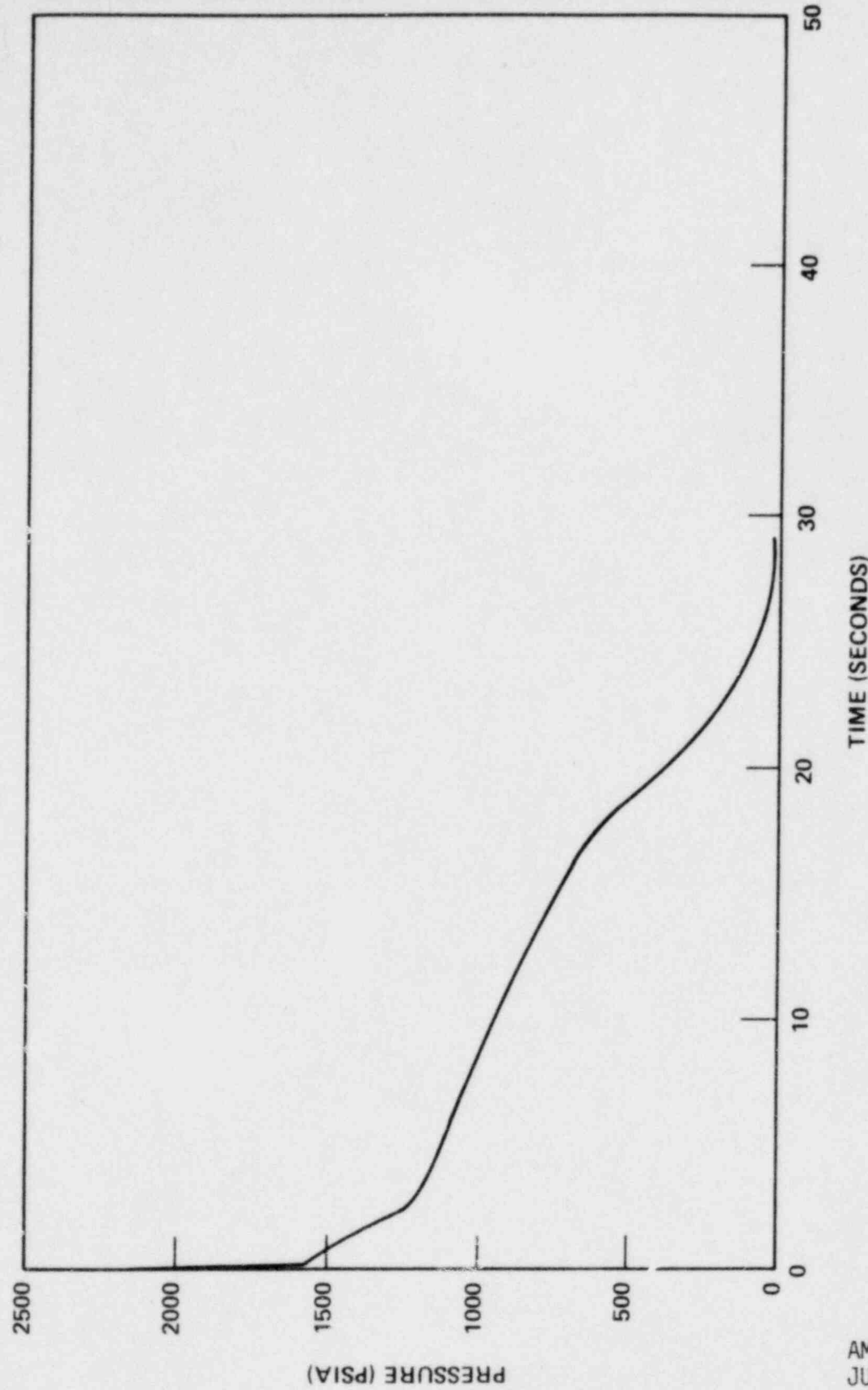

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Figure 15.6-72

Core Pressure -
DECLG (C_D = 0.6)

16132-25

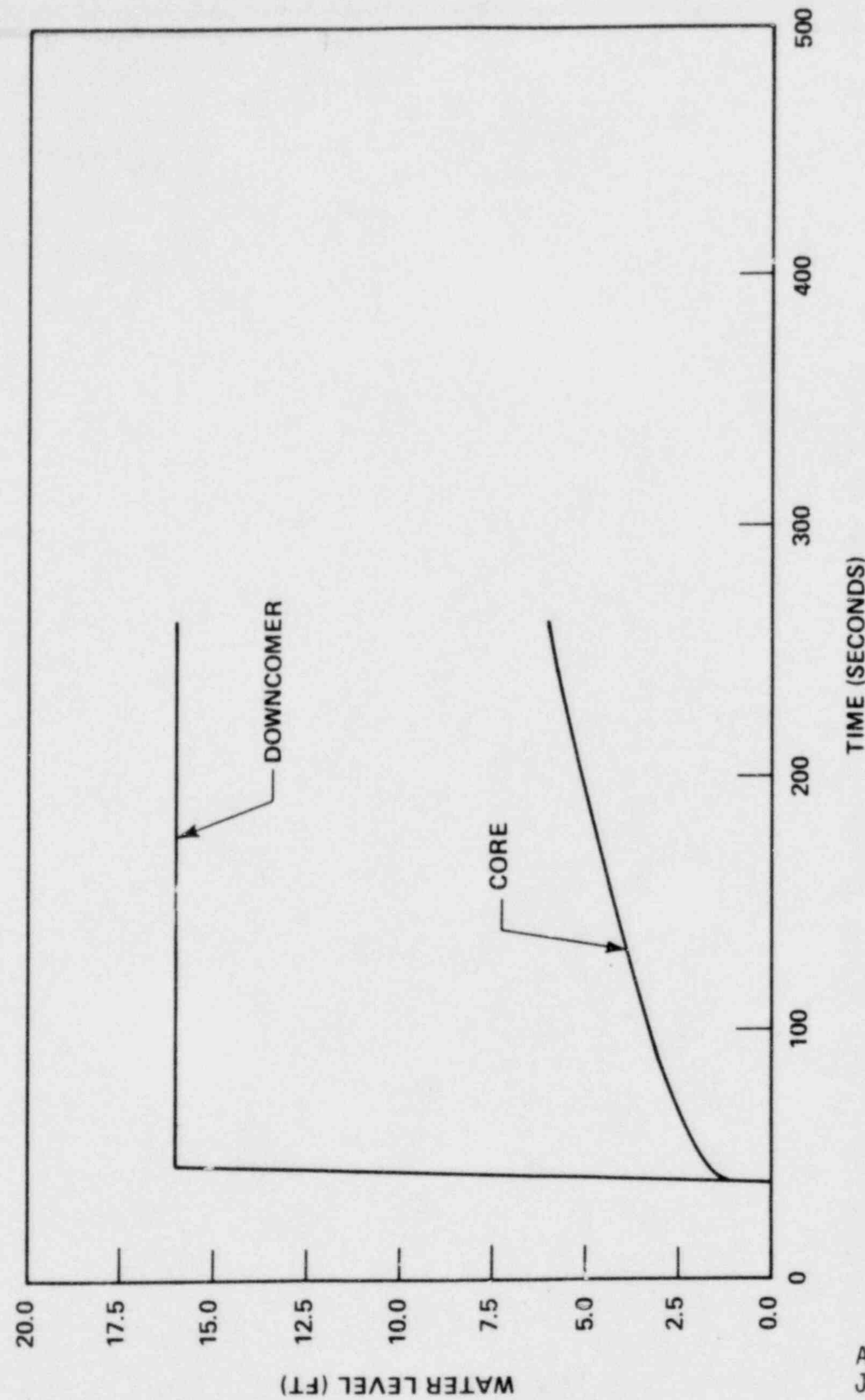

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Figure 15.6-73

DECLG (C_D = 0.6) Downcomer and Core Water Levels During Reflood

16132-26

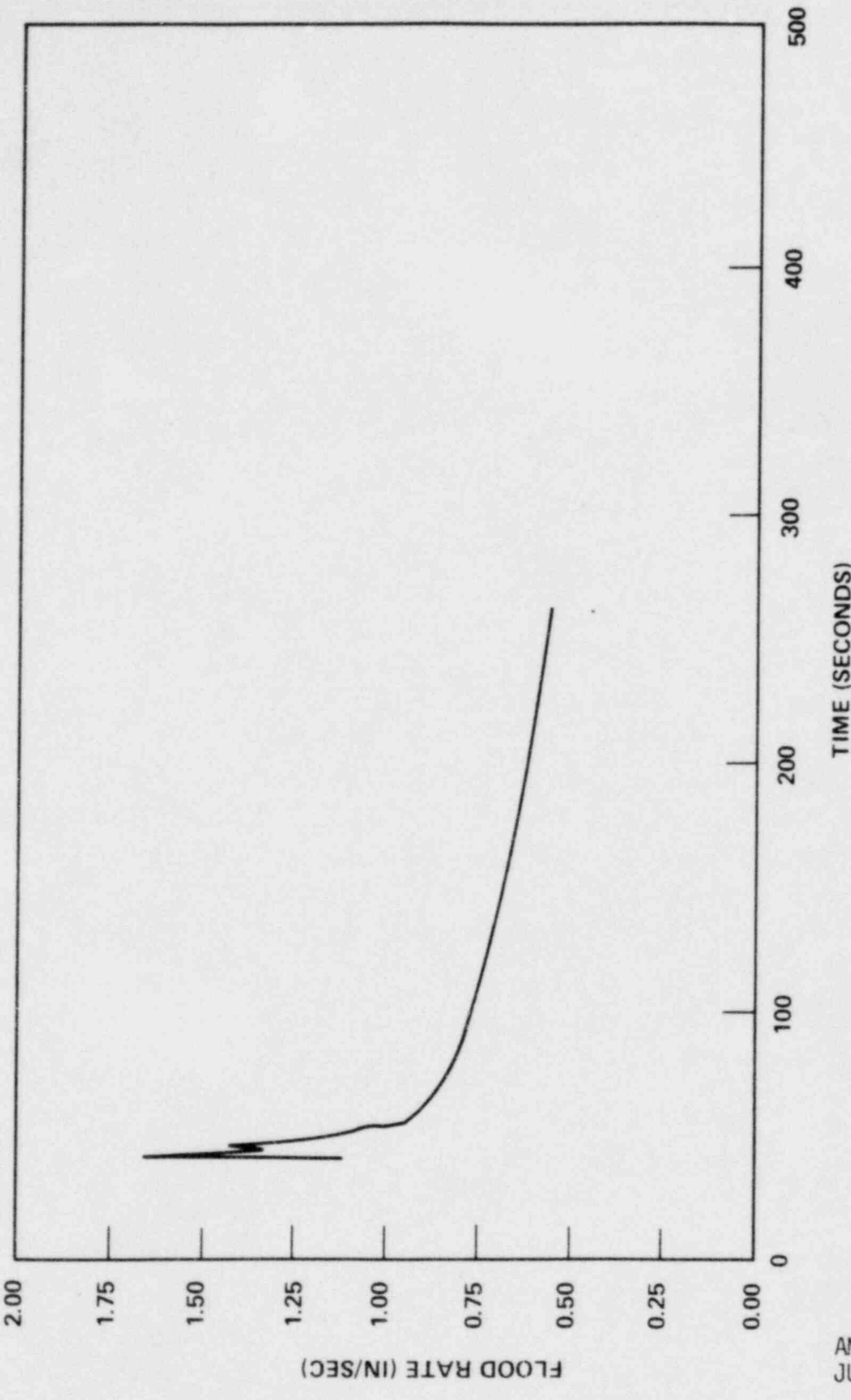

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Figure 15.6-74

DECLG (C_D = 0.6) Core inlet Velocity During Reflood

16132-27

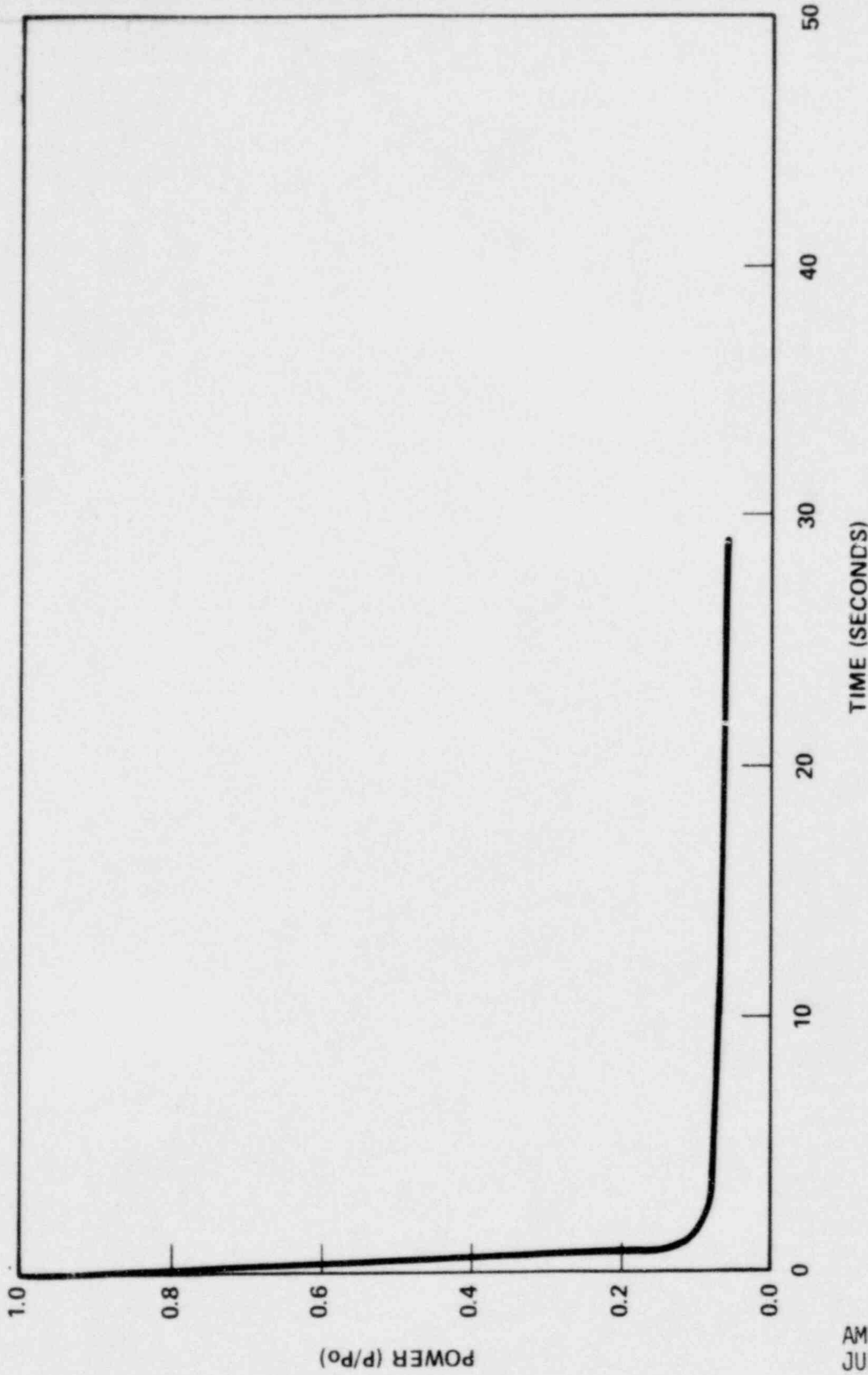

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Figure 15.6-75

Core Power Transient – DECLG (C_D = 0.6)

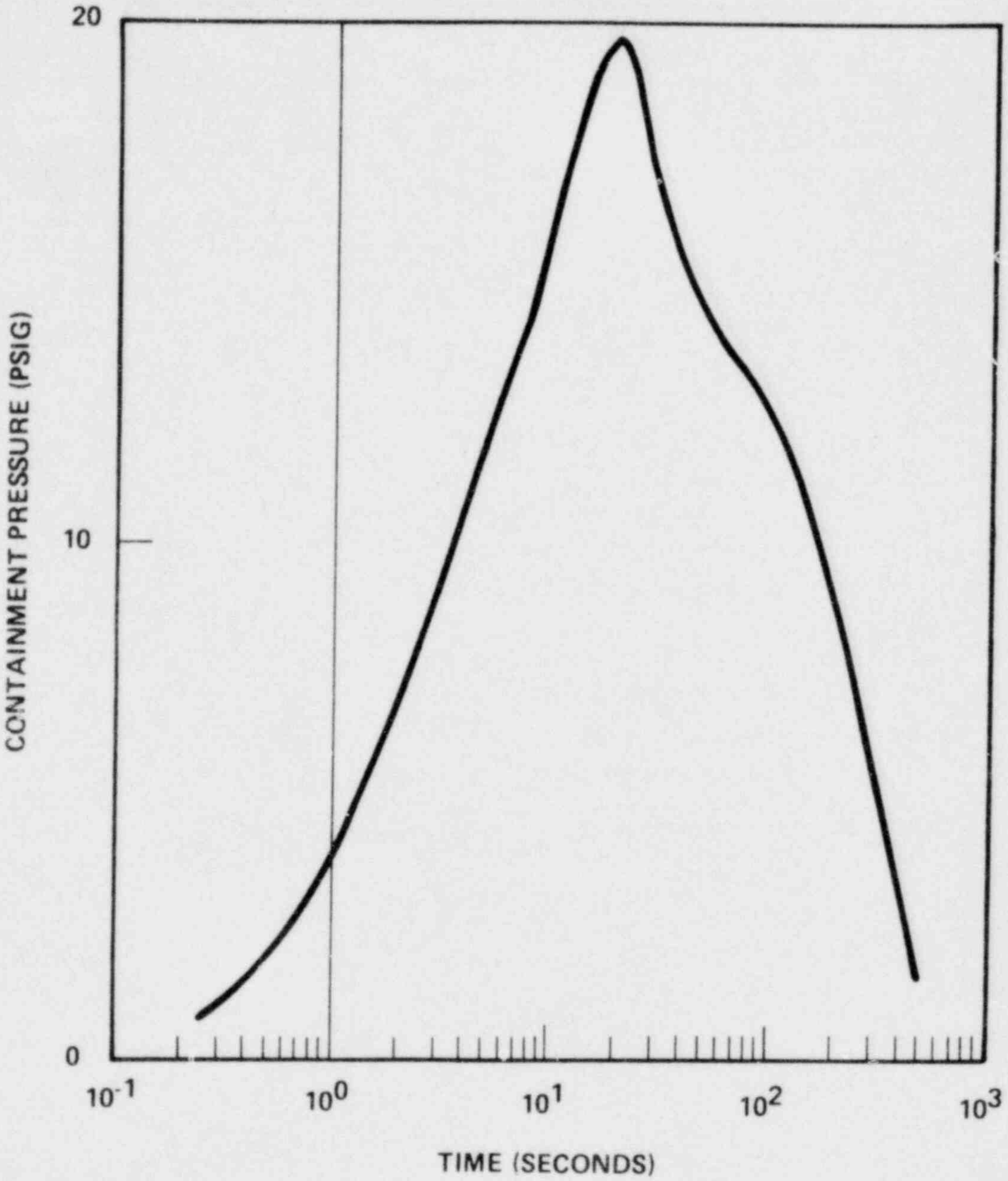

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Figure 15.6-76

Containment Pressure — DECLG (C_D = 0.6)

16132-29

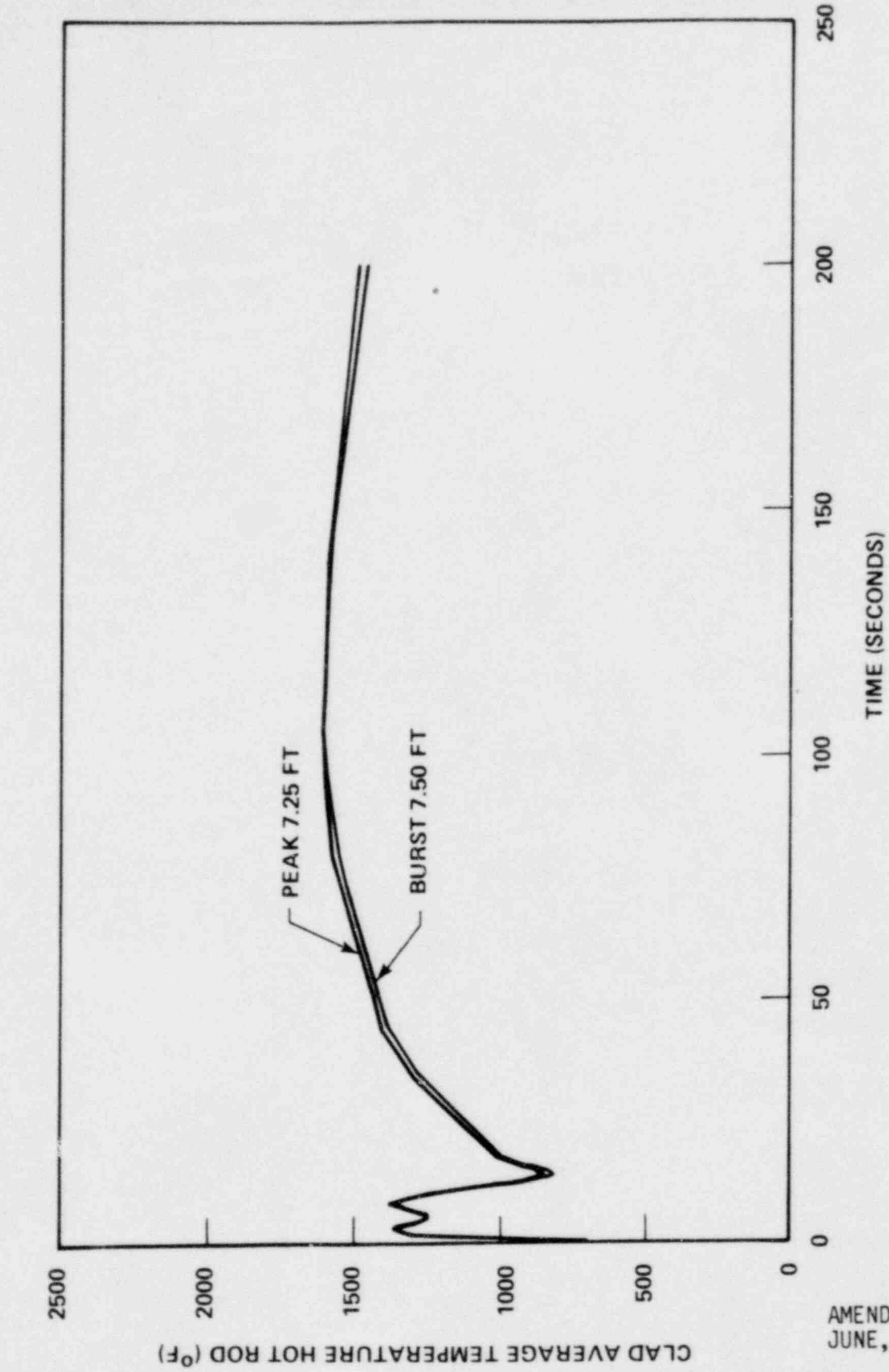

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Figure 15.6-77

Peak Clad Temperature –
DECLG (C_D = 0.8) in Idle Loop

16132-30

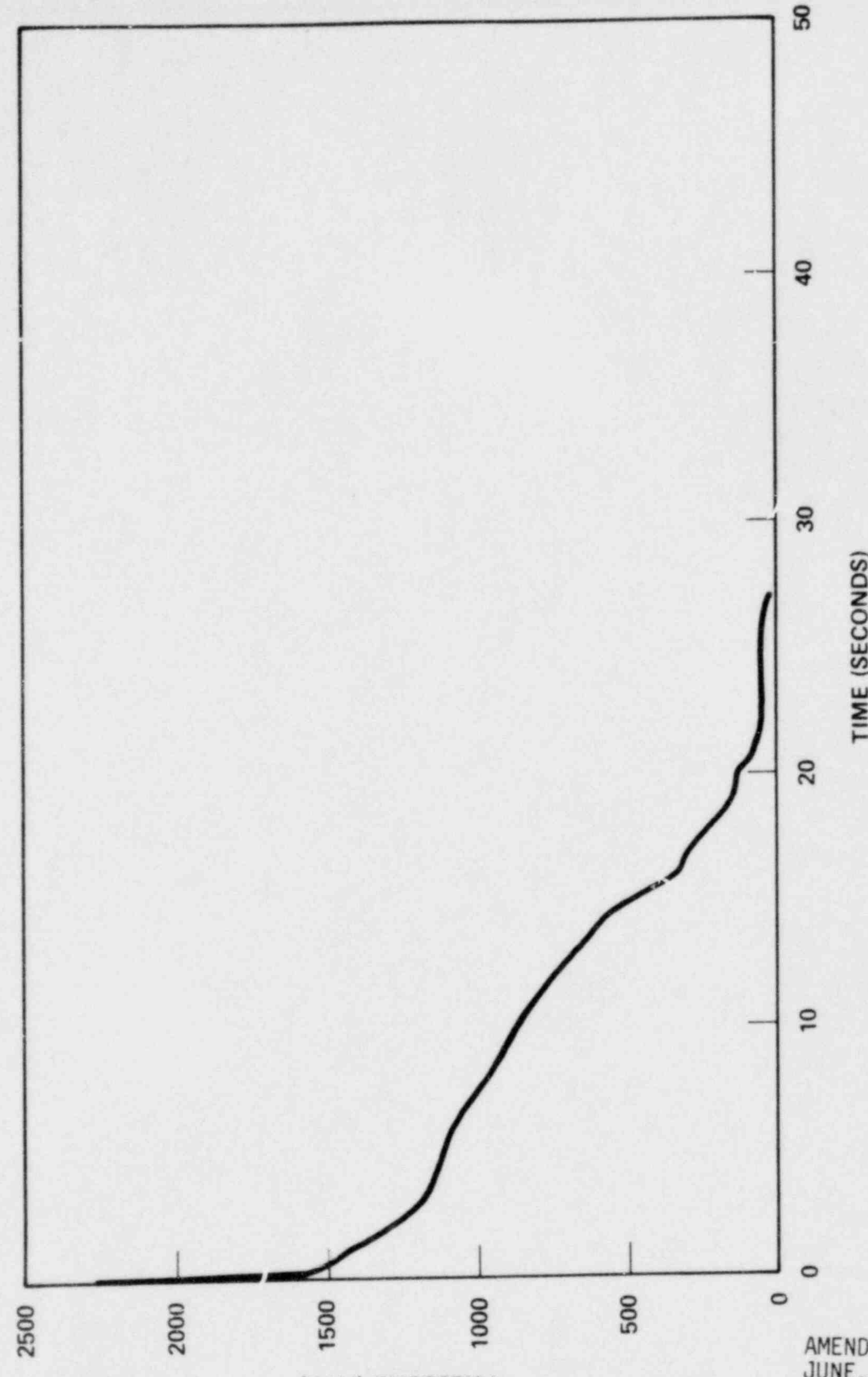

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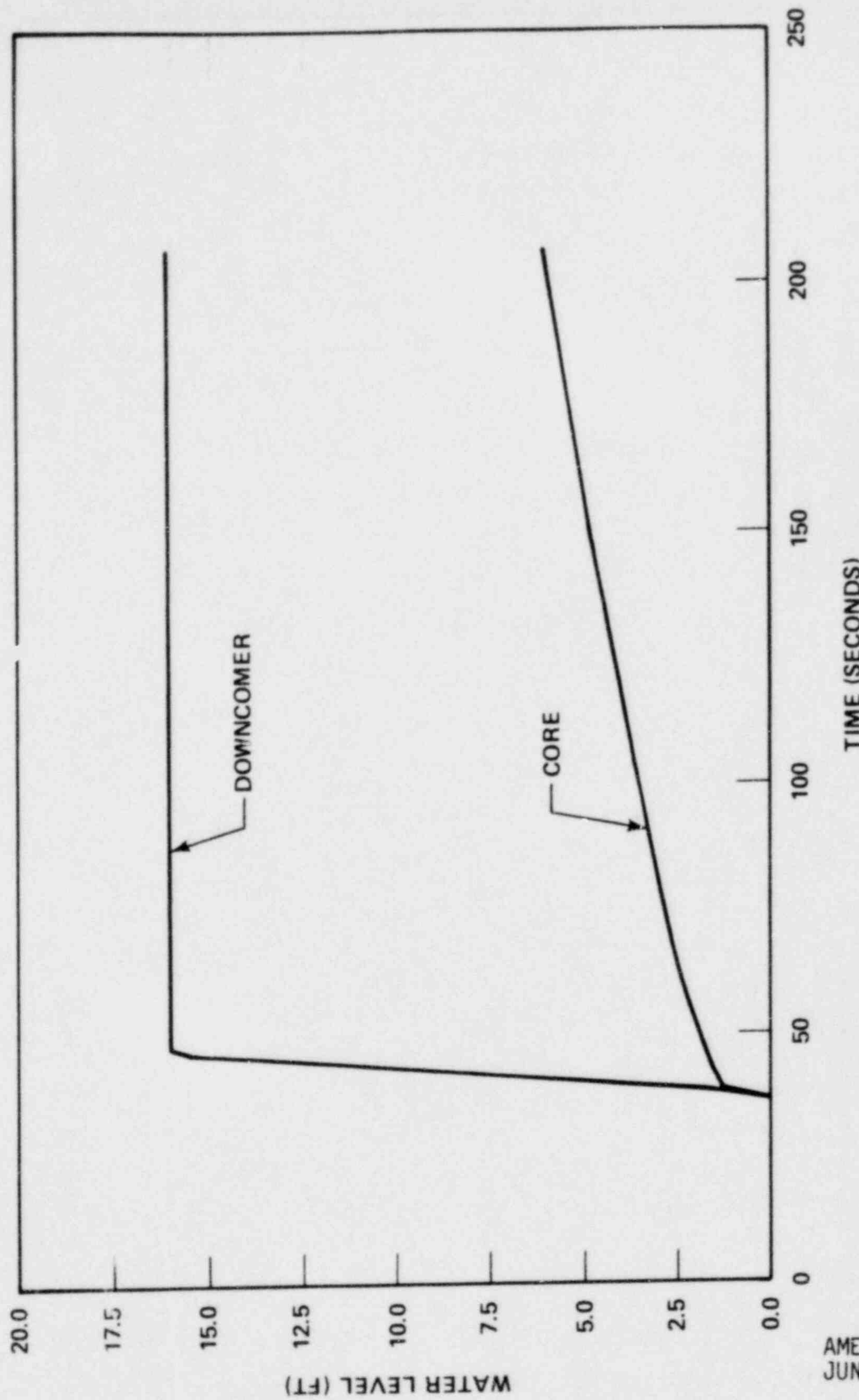
Figure 15.6-78

Core Pressure – DECLG
(C_D = 0.8) in Idle Loop

16132-31



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WCAP—95[illegible]

Figure 15.6-79

DECLG (C_D = 0.8) Downcomer and Core Water Levels During Reflood

16132-32

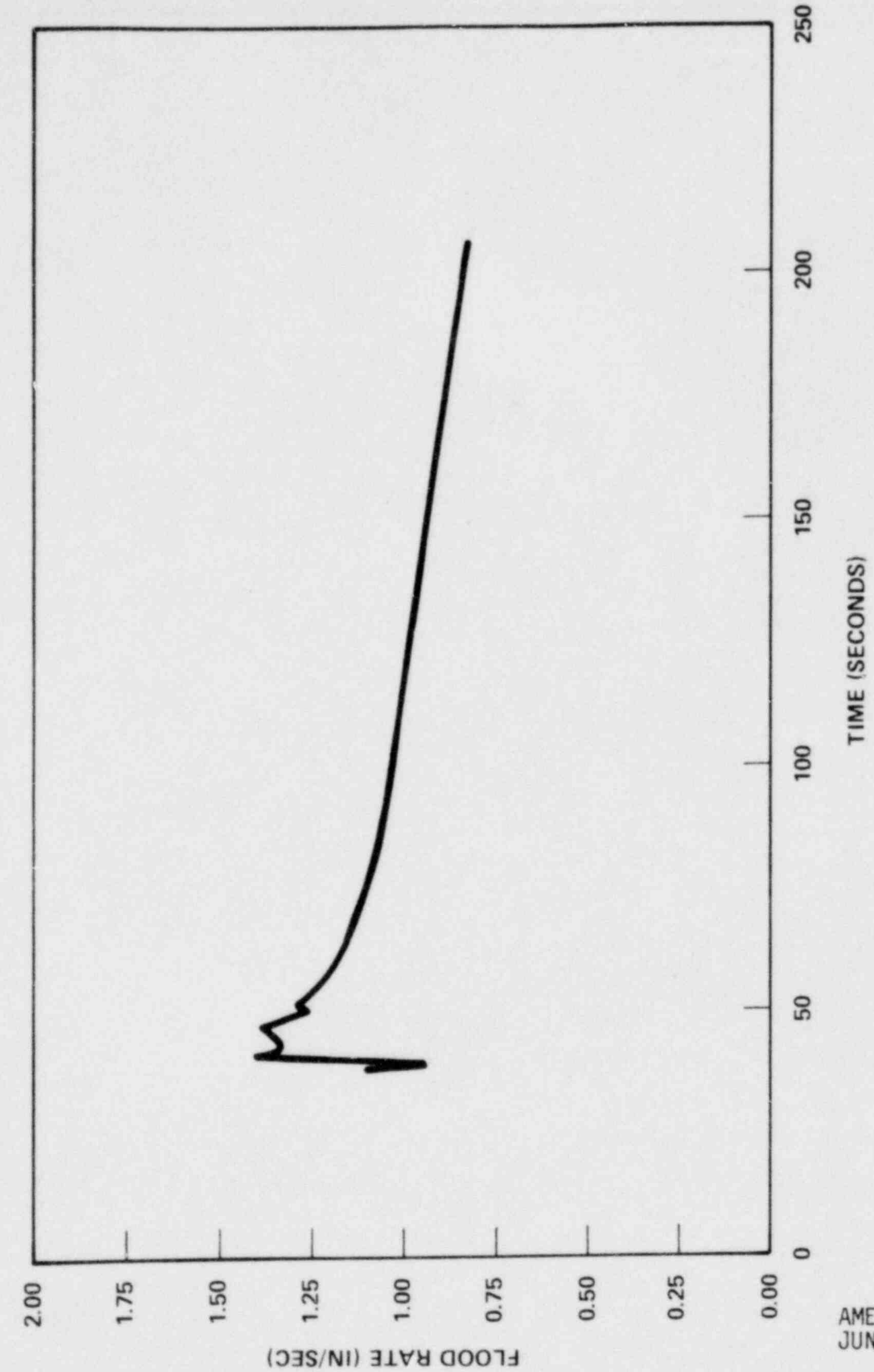

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Figure 15.6-80

Core Inlet Velocity During Reflood – DECLG (C_D = 0.8) in Idle Loop

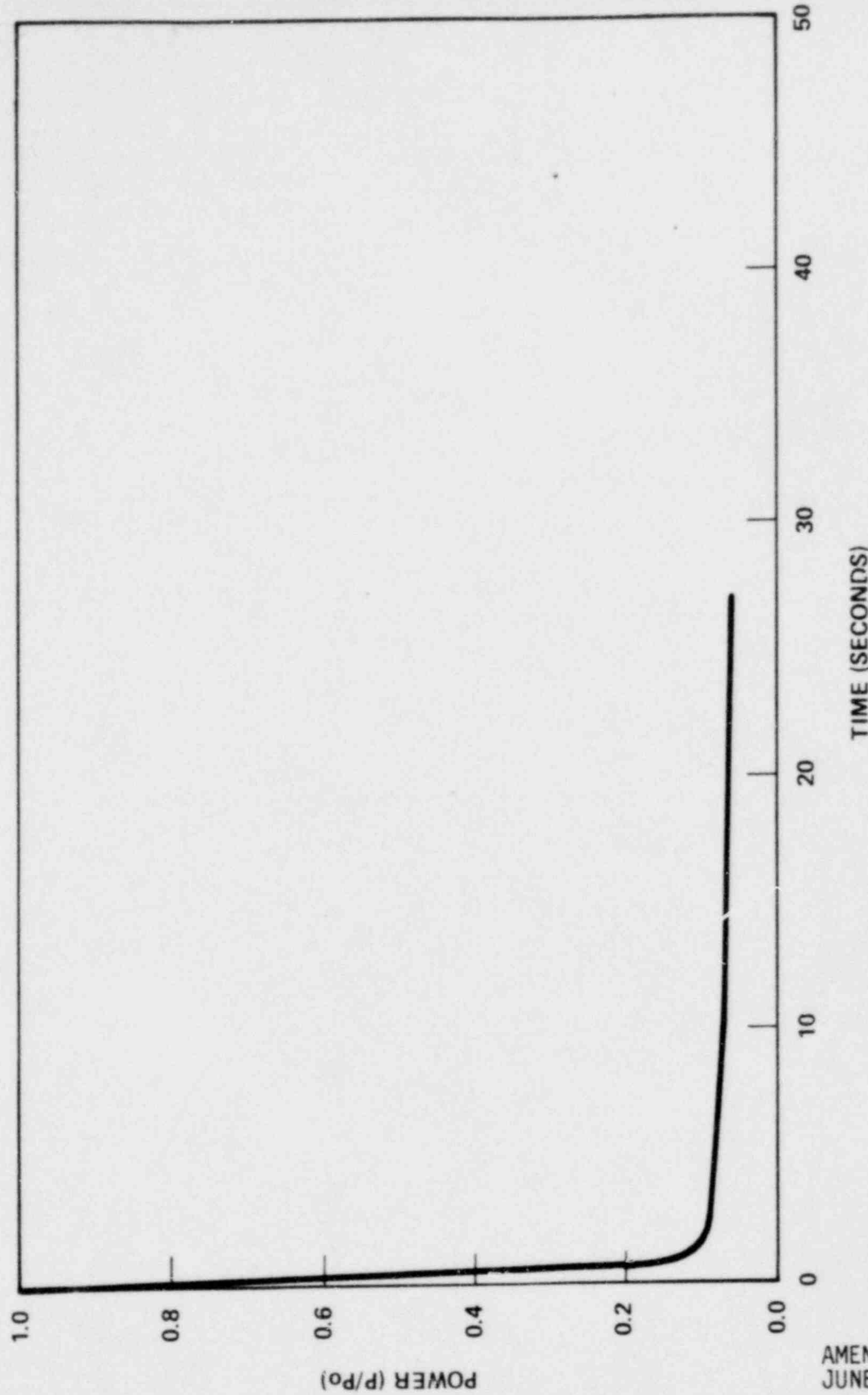

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Figure 15.6-81

Core Power Transient —
DECLG (C_D = 0.8) in Idle Loop

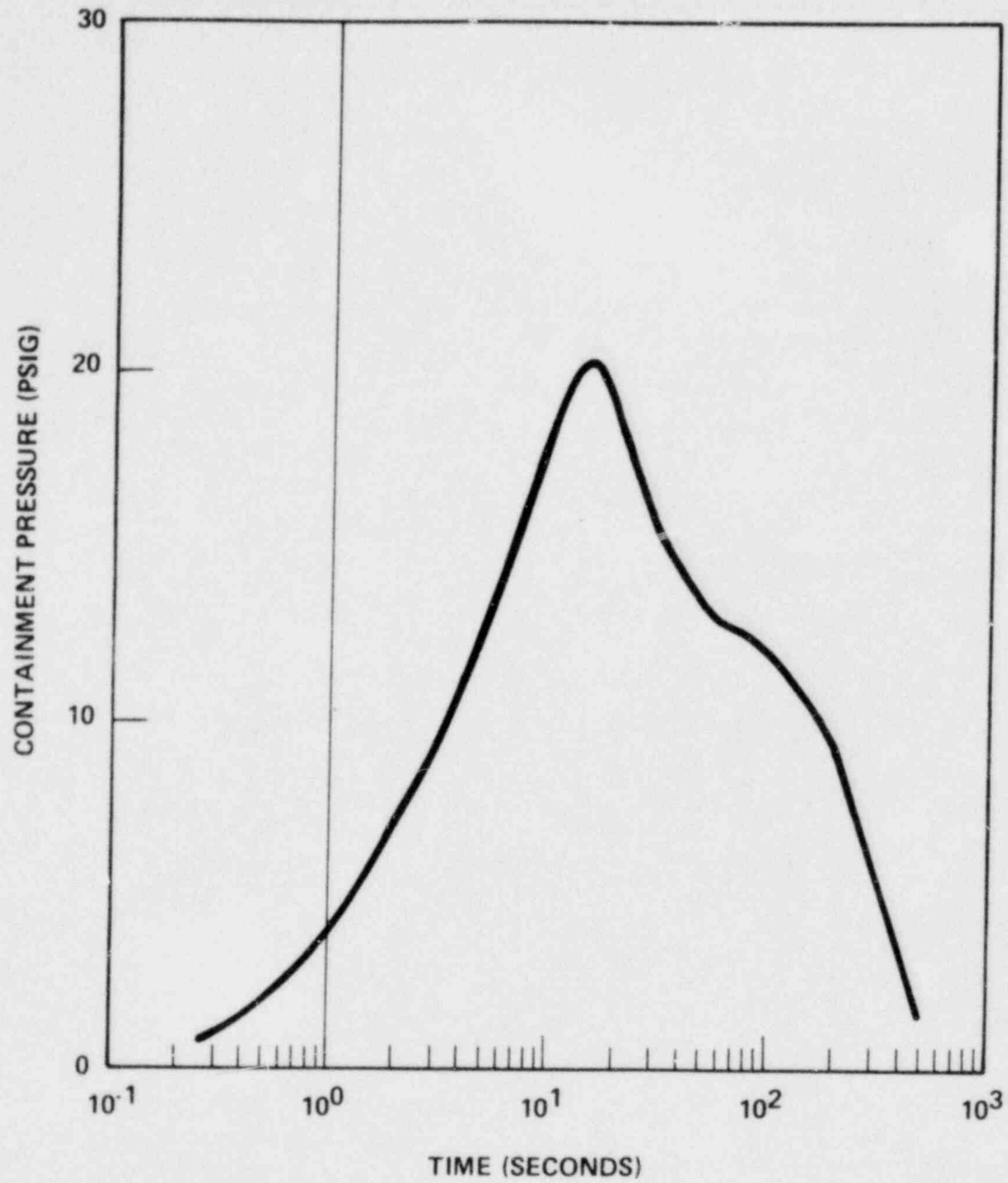

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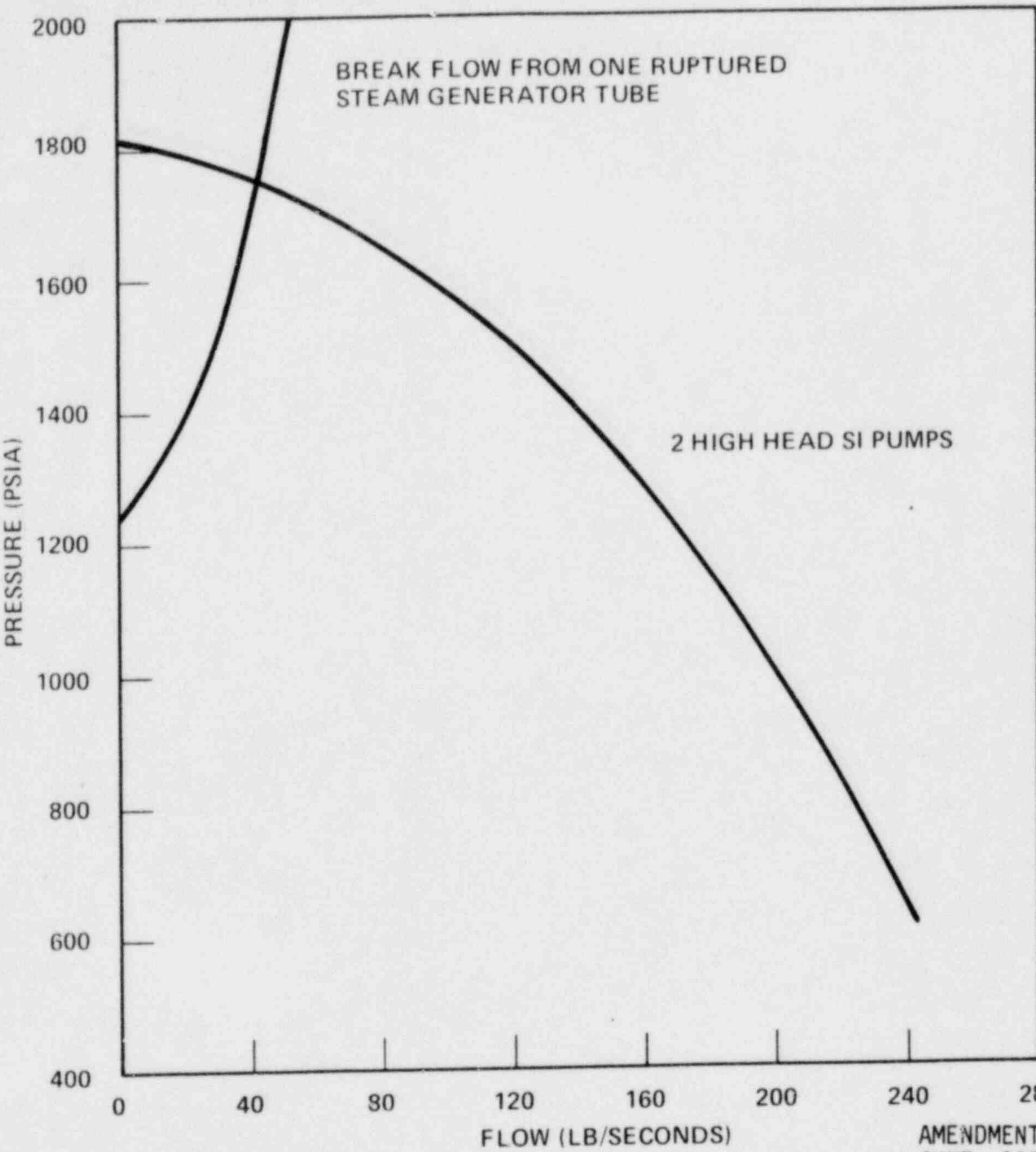
Figure 15.6-82

Containment Pressure – DECLG (C_D = 0.8) in Idle Loop



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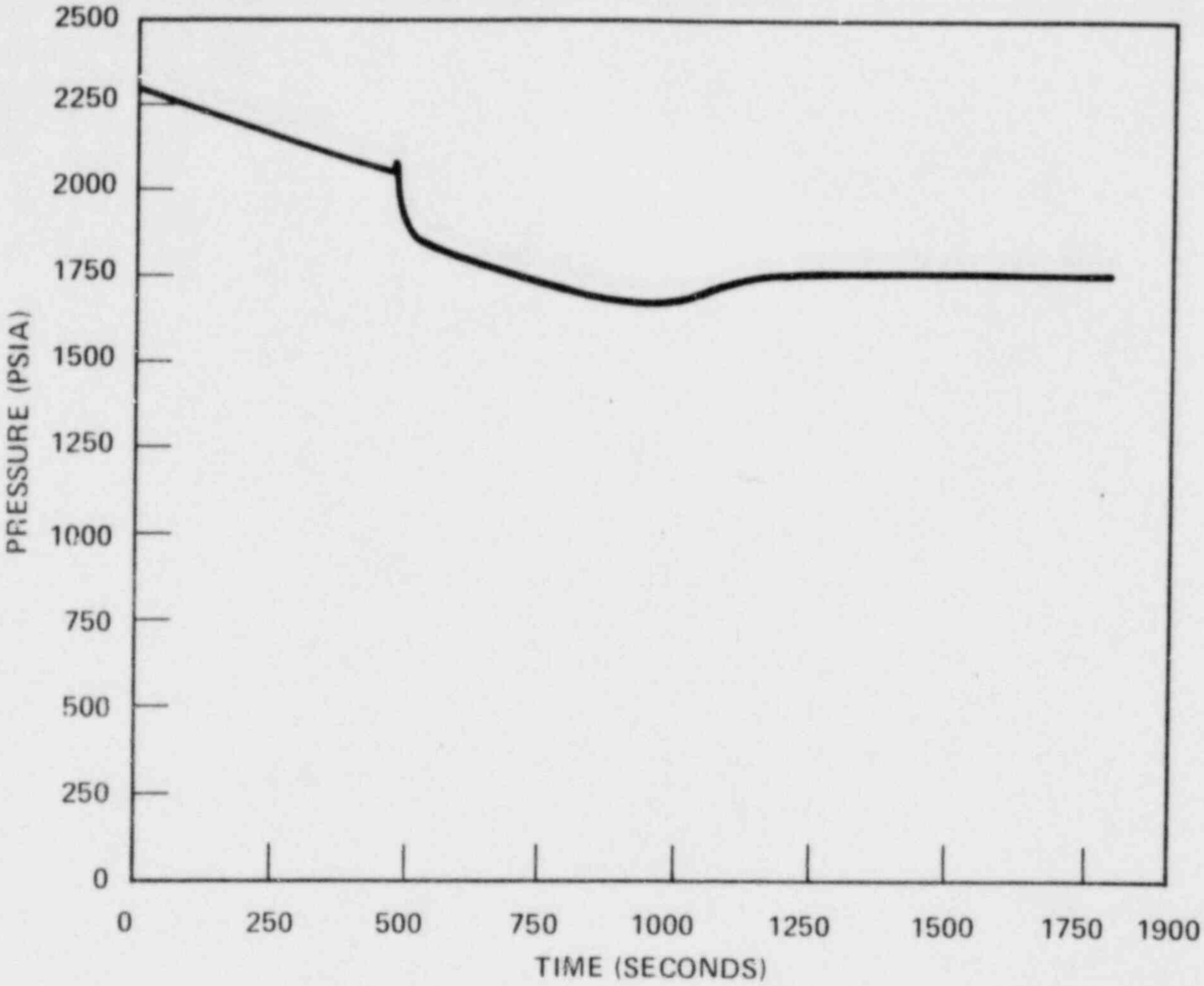
Steam Generator Tube Rupture
Figure 15.6-83
Break Flow and Injection Flow



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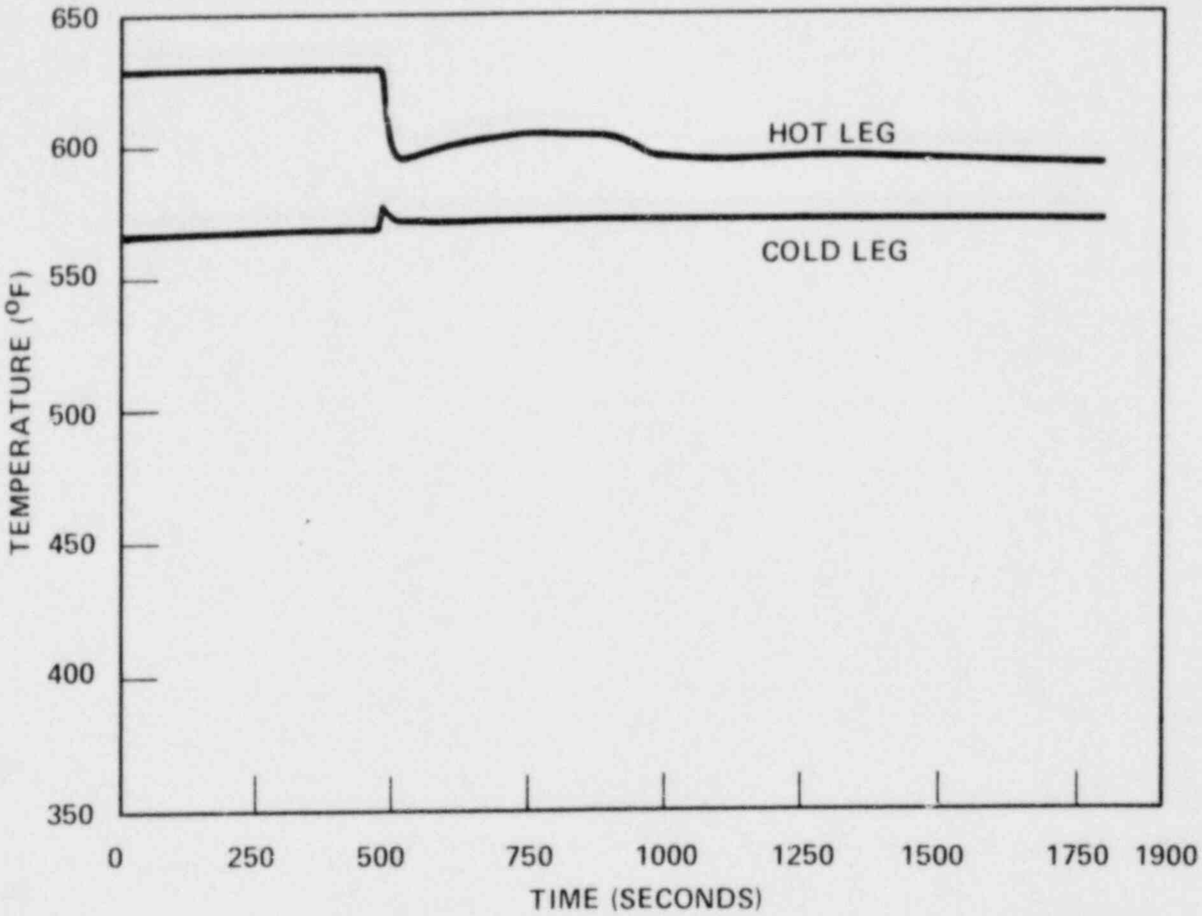
Steam Generator Tube Rupture

Figure 15.6-84

Reactor Coolant System Pressure

16497-10



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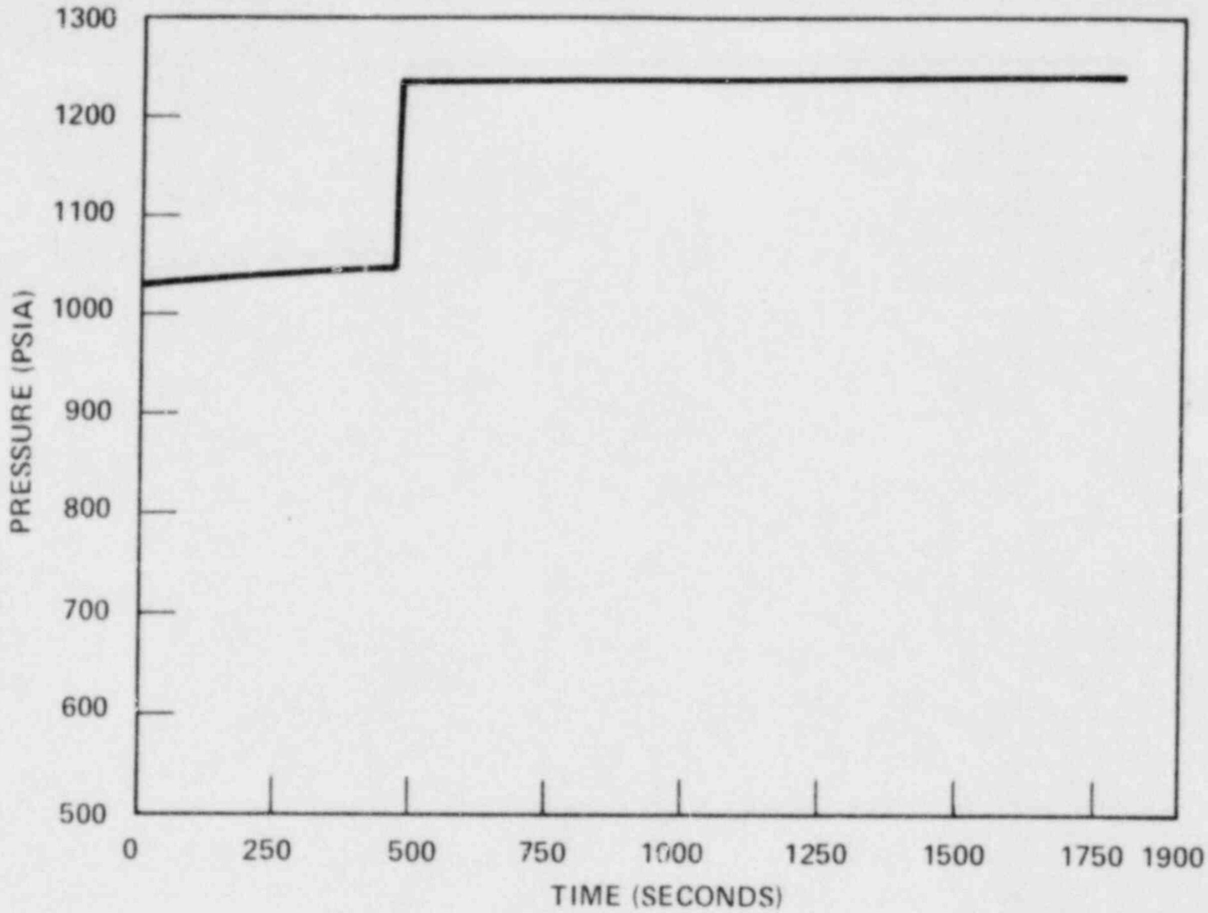
Steam Generator Tube Rupture

Figure 15.6-85

Reactor Coolant System Temperature



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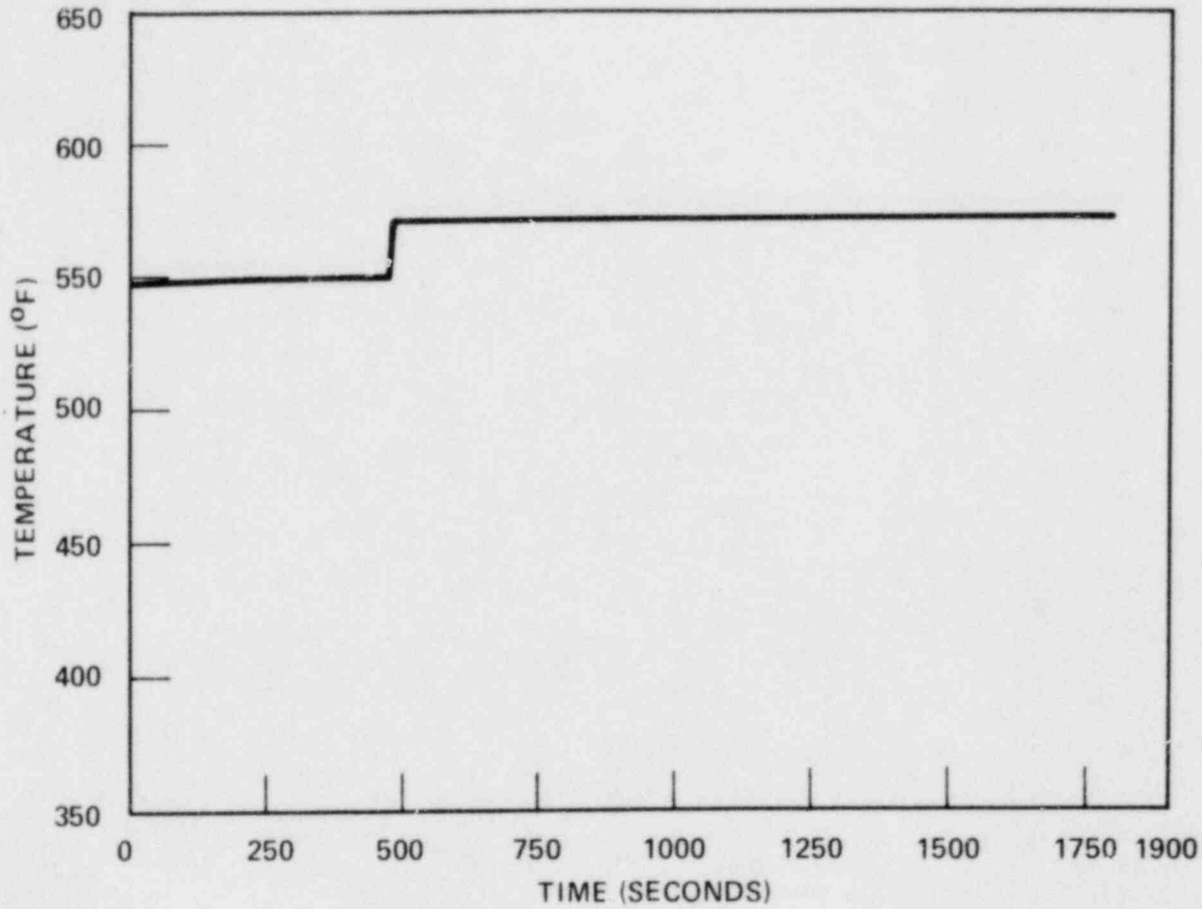
Steam Generator Tube Rupture

Figure 15.6-86

Steam Generator Pressure

16497-8



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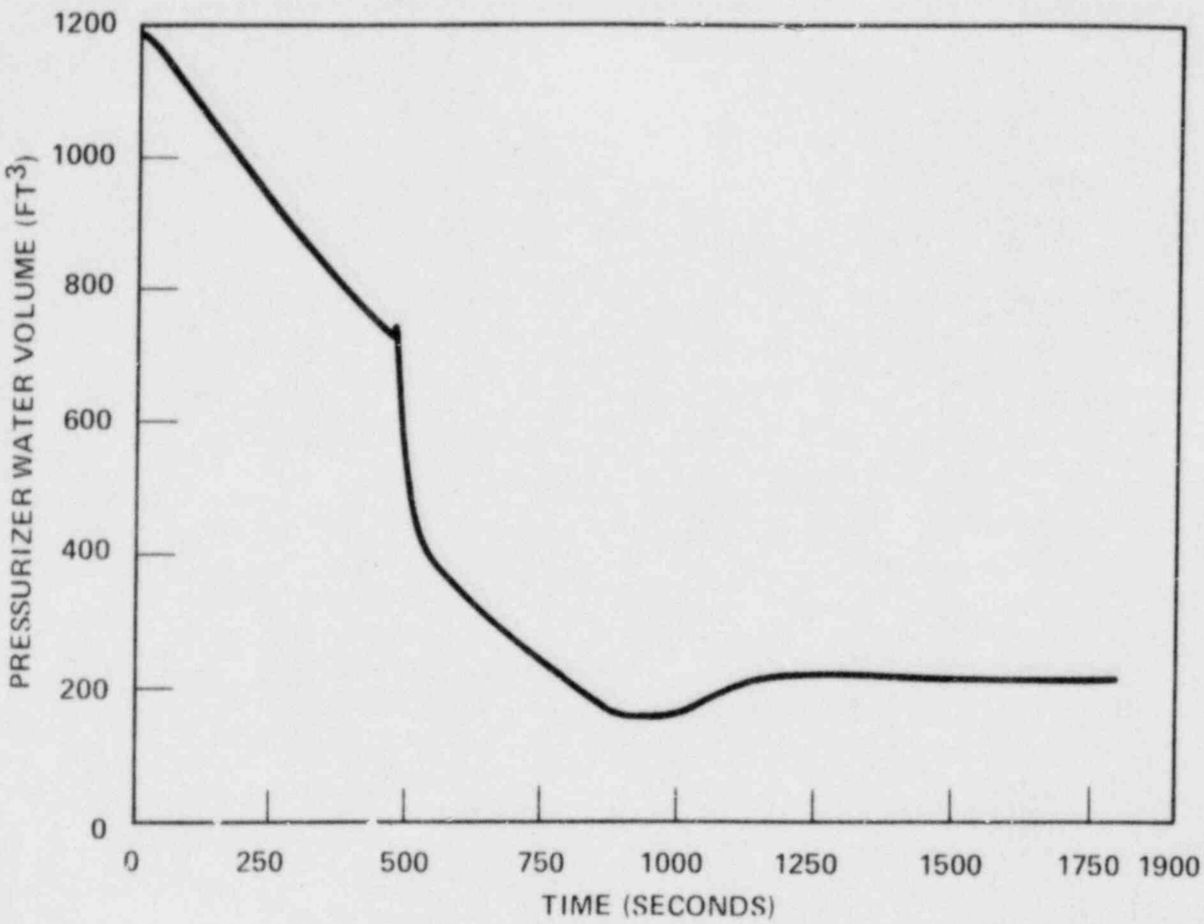
Steam Generator Tube Rupture
Figure 15.6-87
Steam Generator Temperature



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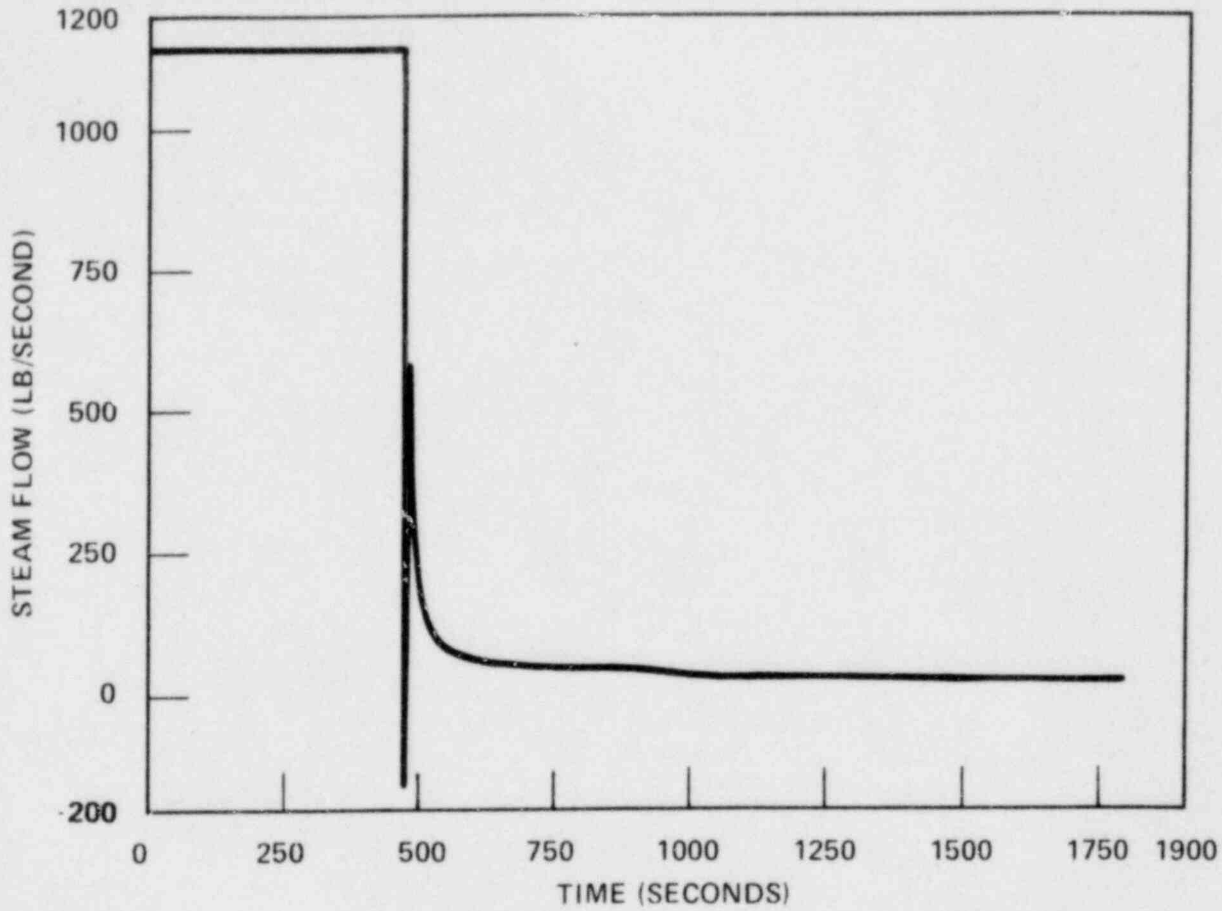
Steam Generator Tube Rupture
Figure 15.6-88
Pressurizer Water Volume



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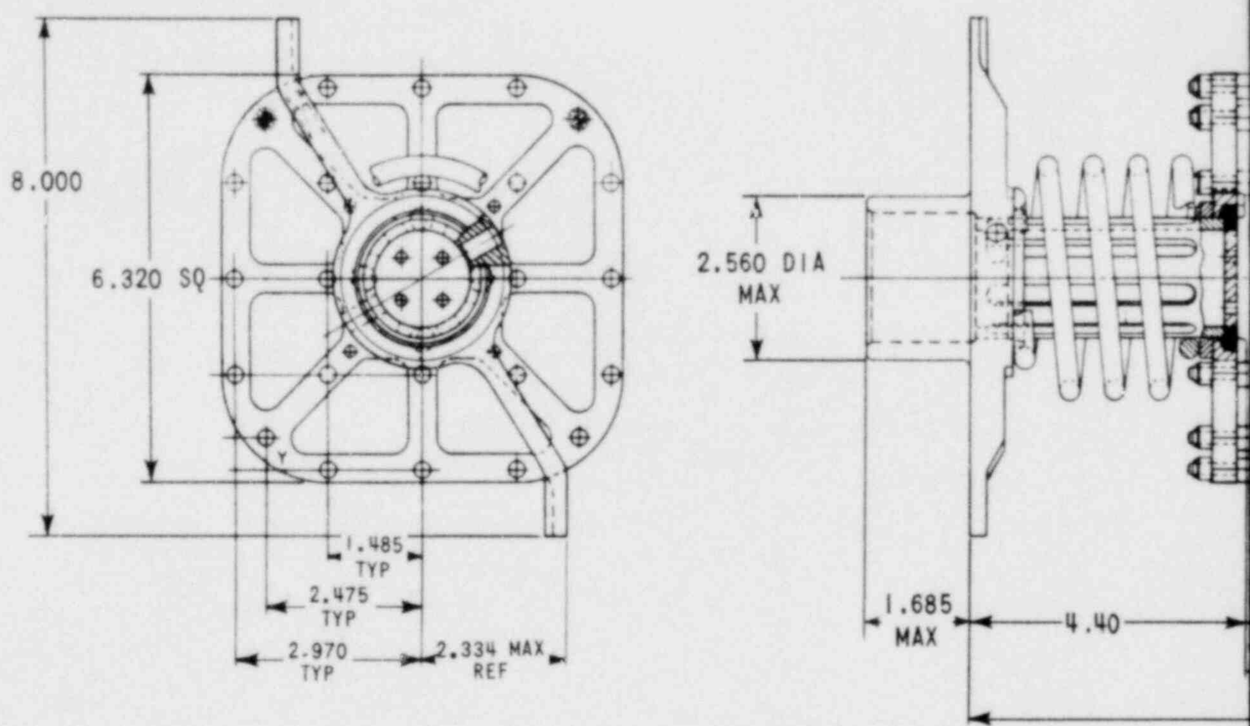
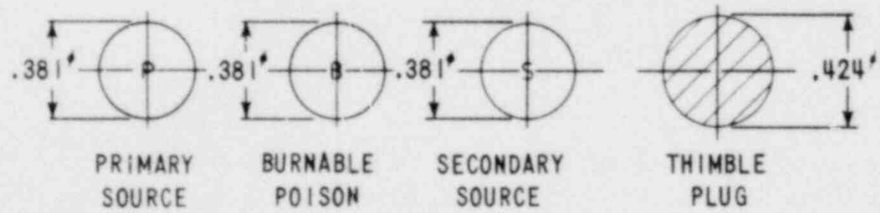
Steam Generator Tube Rupture

Figure 15.6-89

Steam Generator Flow


8.000
6.320 SQ
1.485 TYP
2.475 TYP
2.970 TYP
2.334 MAX REF
2.560 DIA MAX
1.685 MAX
4.40

F


.381"
P
.381"
B
.381"
S
.424"
PRIMARY SOURCE
BURNABLE POISON
SECONDARY SOURCE
THIMBLE PLUG

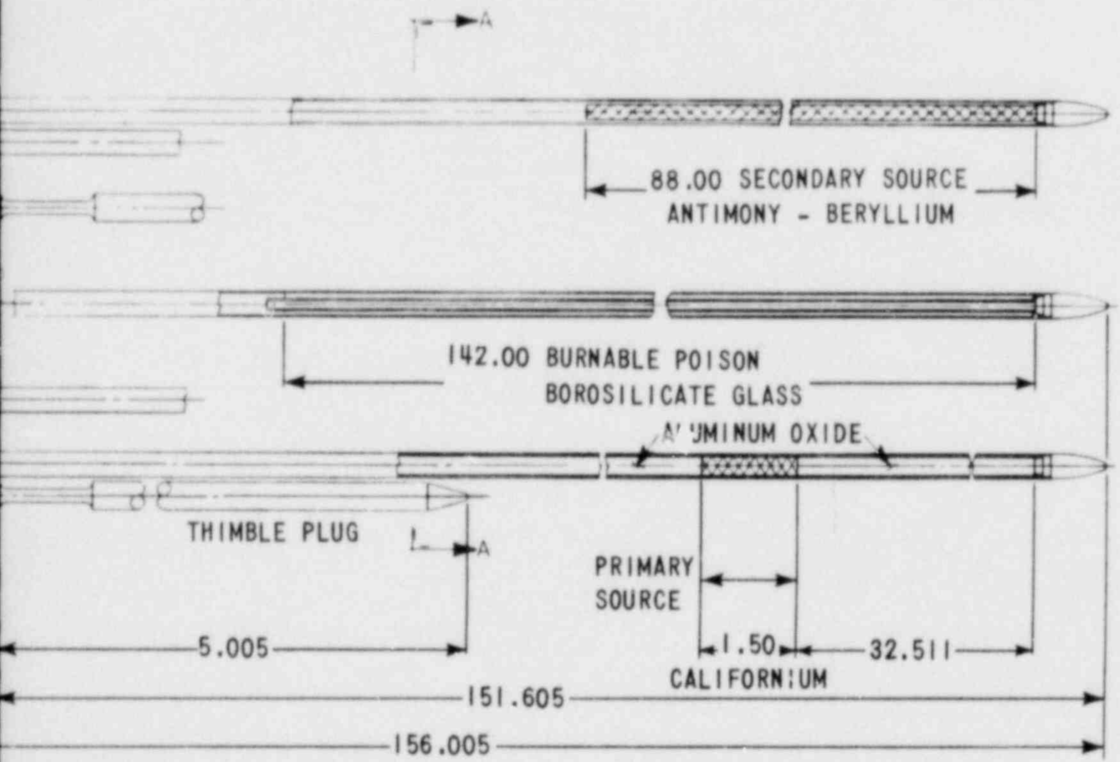


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Figure 4.2-12.

Composite Core Component Rods and Assembly Outline (UHI Plants)

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system (RCS) through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is adequate to sustain pressurizer level and a pressure of 2250 psia for a break through a 0.375 inch dimater hole. This break results in a loss of approximately 17.5 lb/sec, and, due to the use of a 0.245 inch restriction, is the maximum flow available for all reactor coolant sample line breaks outside of the containment. In addition, all such lines meet the requirements of General Design Criterion 55 of Appendix A 10 CFR 50. There are no instrument lines which pass through the containment and connect directly to the RCS. A failure of a small line carrying primary coolant outside containment is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.2 for a discussion of Condition II events.

15.6.2.2 Analysis of Effects and Consequences

Since this event does not result in a leakage rate greater than the capacity of a charging pump and pressurizer level does not decrease, normal shutdown procedures can be employed. There are no significant consequences to the reactor or its essential auxiliary systems.

15.6.2.3 Radiological Consequences

There could be moderate radioactive releases from the failure of a small line carrying primary coolant outside containment. This accident will be evaluated in the Applicant's SAR. The primary coolant activity that would be used in the small line break analysis is 60 μCi of dose equivalent I-131 resulting from a preexisting iodine spike.

Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the accident diagnostics and isolation procedure can be completed within 30 minutes of accident initiation for the design basis event.

Note that break sizes smaller than complete severance of a tube, with less break flow from primary to secondary, exhibit a slower rise in steam generator water level, and an increased time interval for actuation of the blowdown line radiation monitor and the condenser air ejector radiation monitor. Therefore more time may be available to the operator to diagnose the accident and take steps to isolate the faulted steam generator.

If normal operation of the various plant control systems is assumed, the following sequence of events is initiated by a tube rupture.

1. Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side steam flow/feedwater flow mismatch occurs as feedwater flow to the affected steam generator is reduced as a result of primary coolant break flow to that unit.

2. Decrease in RCS pressure (Figure 15.6.3-2) due to continued loss of reactor coolant inventory leads to a reactor trip signal on low pressurizer pressure or overtemperature ΔT. Resultant plant cooldown (Figure 15.6.3-3) following reactor trip leads to a rapid decrease in pressurizer level (Figure 15.6.3-6), and a safety injection signal, initiated by low pressurizer pressure, follows soon after reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.

3. The steam generator blowdown liquid monitor and/or the condenser air ejector radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system and will automatically terminate steam generator blowdown.

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2. Following the initiation of the safety injection signal, two centrifugal pumps are actuated and are assumed in the analyses to continue to deliver flow for 30 minutes.

3. After reactor trip the break flow reaches equilibrium when incoming safety injection flow is balanced by outgoing break flow as shown in Figure 15.6.3-1. The resultant break flow is assumed to persist from plant trip until 30 minutes after the accident. No operator actions are assumed.

4. The steam generators are controlled at the safety valve setting rather than the power operated relief valve setting.

5. The operator identifies accident type (using procedure guidelines in Reference 11) and terminates break flow to the faulted steam generator within 30 minutes of accident initiation.

   The above assumptions, suitably conservative for the design basis tube rupture, are made to maximize doses and do not explicitly model operator actions for recovery.

Recovery Procedure

Immediately apparent symptoms of a tube rupture accident such as falling pressurizer pressure and level and increased charging pump flow are also symptoms of small steam line breaks and loss of coolant accidents. It is therefore important for the operator to determine that the accident is a rupture of a steam generator tube in order that he may carry out the correct recovery procedure. The accident under discussion can be identified by the following method. In the event of a complete tube rupture the reactor coolant system pressure decreases, Figure 15.6.3-2, and the condenser air ejector radiation and/or steam generator blowdown radiation monitors exhibit abnormally high readings. If the containment pressure, containment radiation and containment recirculation sump level exhibit normal readings, then a steam generator tube rupture is diagnosed to have occurred. The recovery procedures for the double ended

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 square foot (ft^2). This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis.

A minor pipe break (small break), as considered in this section, is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event, in that it is an infrequent fault which may occur during the life of the plant.

The Acceptance Criteria for the LOCA is described in 10CFR50.46 as follows:

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Fission Product Release to the Containment

The radiological assessment will be based on the conservative fission product release given in Regulatory Guide 1.4.

Thus, a total of 100 percent of the noble gas core inventory and 25 percent of the core iodine inventory is assumed to be immediately available for leakage from the primary containment. Of the halogen activity available for release, will be assumed that 91 percent is in elemental form, 4 percent in methyl form and 5 percent in particulate form. The total core noble gas and iodine inventories are given in Table 15.0.9-1.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable.

15.6.7 REFERENCES

1. Burnett, T. W. T., et. al., "LOFTRAN Code Description," WCAP-7907 June 1972.

2. Young, M. Y., Westinghouse Emergency Core Cooling System Evaluation Model Application to Plants Equipped with Upper Head Injection, WCAP-8479, (Westinghouse Proprietary), and WCAP-8480, January, 1975.

3. Bordelon, F. M., H. W. Massie, and T. A. Borden, "Westinghouse ECCS Evaluation Model-Summary," WCAP-8339, (Non-Proprietary) July 1974.

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

5. Kelly, R. D., et. al., "Calculated Model for Core Reflooding (Proprietary) June 1974, and (Non-Proprietary) June 1974.

TABLE 15.6.3-1

STEAM GENERATOR TUBE RUPTURE SEQUENCE OF EVENTS

Event	Time (seconds)
Tube Rupture Occurs	0.0
Reactor Trip Signal	590.0
Rod Motion	592.0
Feedwater Terminated	592.0
Steam Generator Safety Valves Opened (assumed to stay open to maximize release)	599.0
S.I. Signal	773.0
S.I. Injection	798.0
Auxiliary Feedwater Injection	833.0
Assumed that operator completes actions to isolate and equilibrate	1800.0

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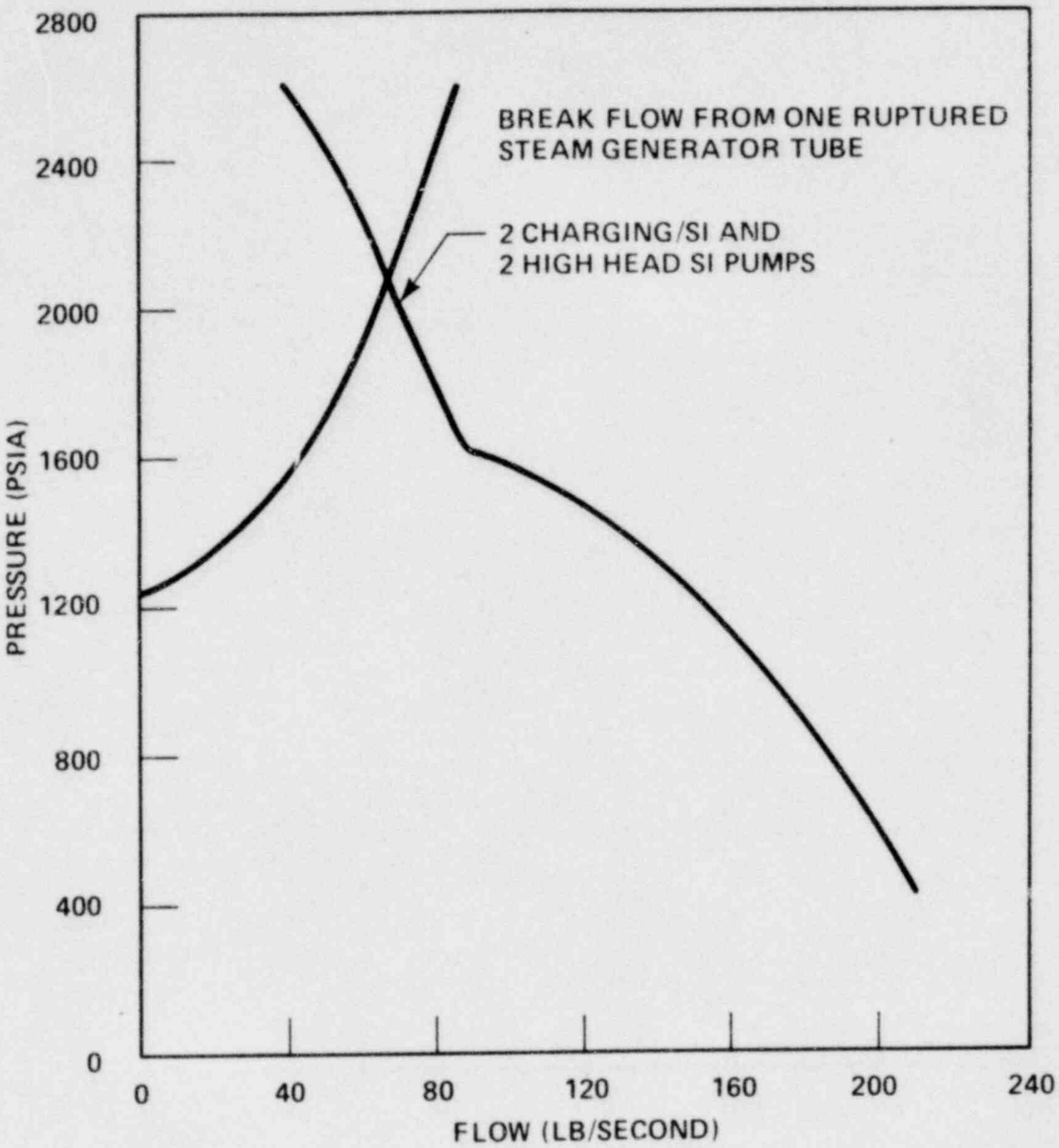

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Steam Generator Tube Rupture

Figure 15.6.3-1

Break Flow and Injection Flow

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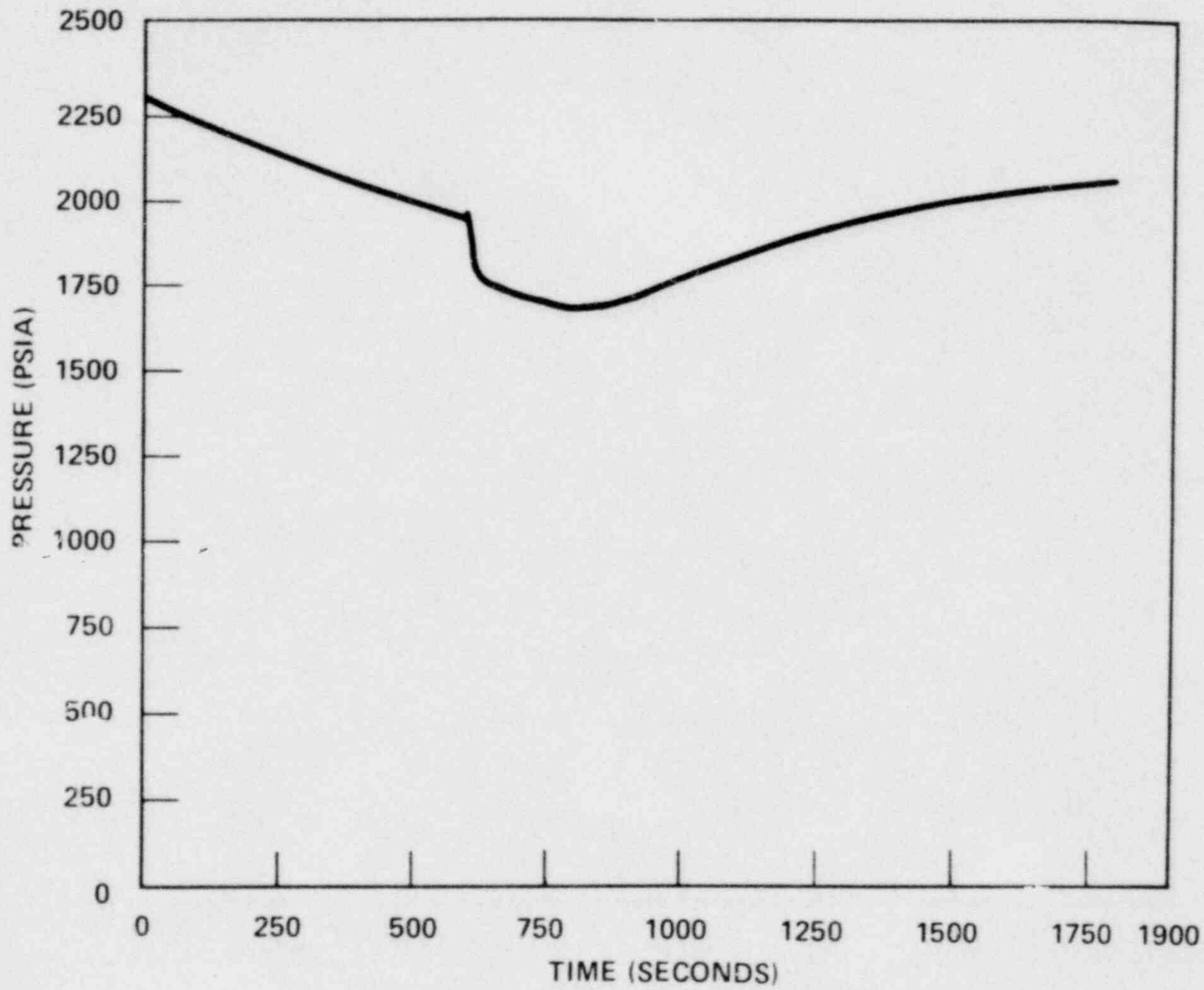
16497-5



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Steam Generator Tube Rupture

Figure 15.6.3-2

Reactor Coolant System Pressure

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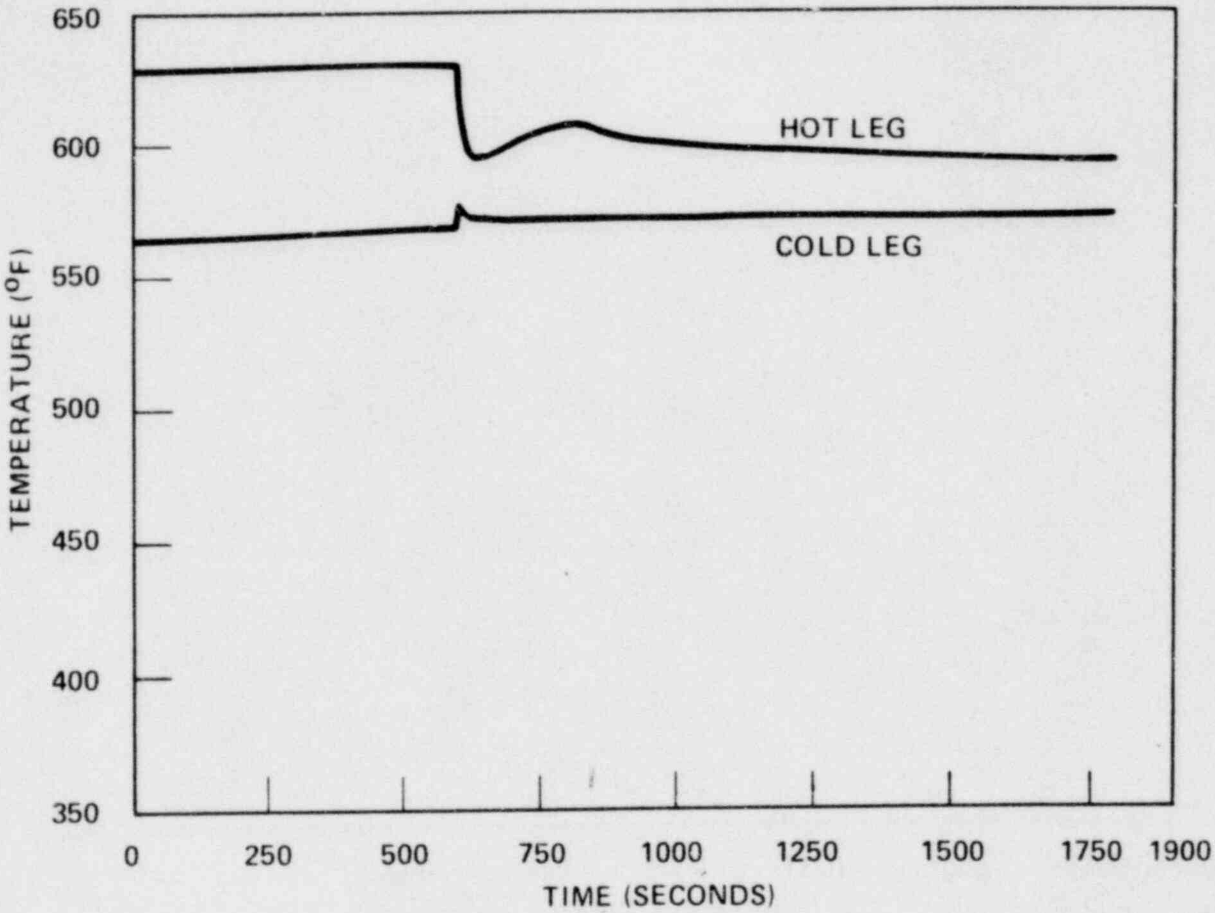

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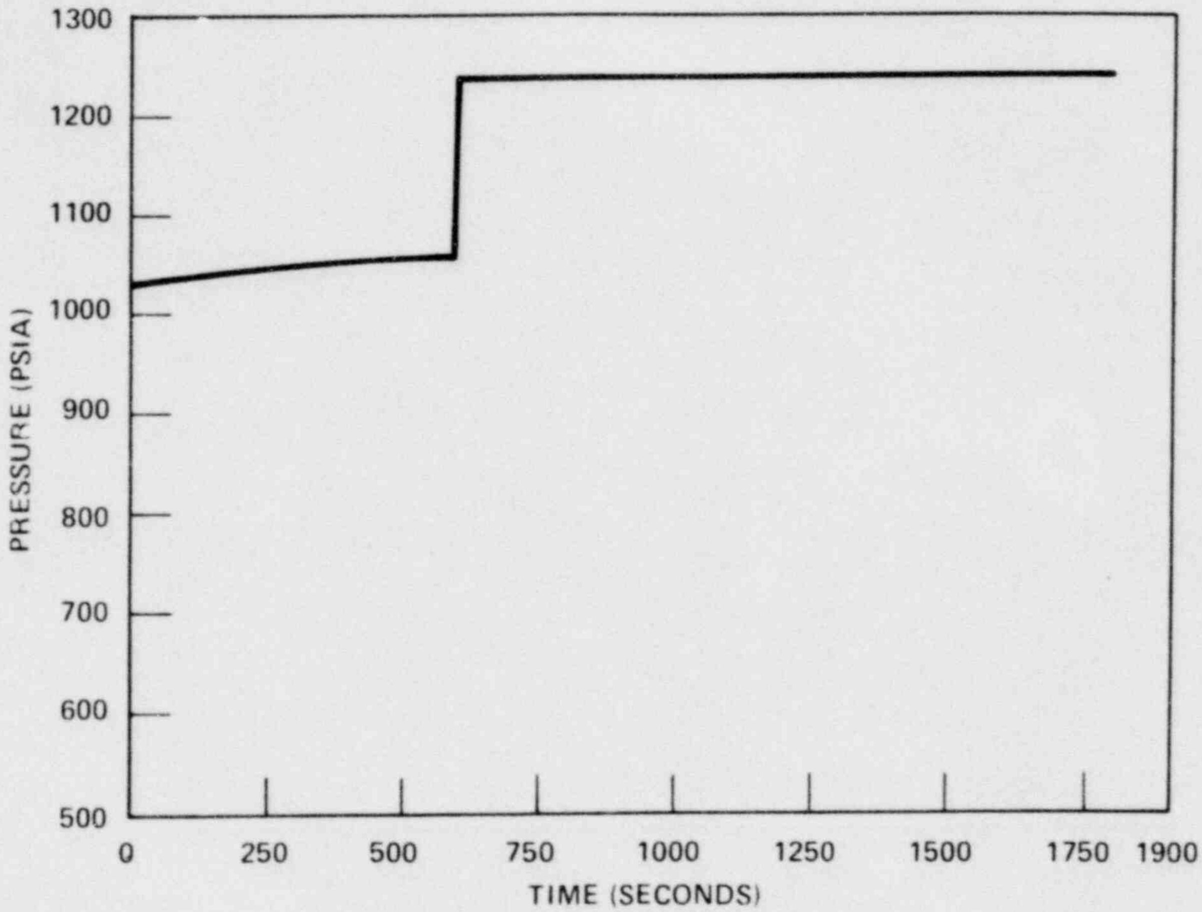
Steam Generator Tube Rupture

Figure 15.6.3-3

Reactor Coolant System Temperatures

Blue

16497-3



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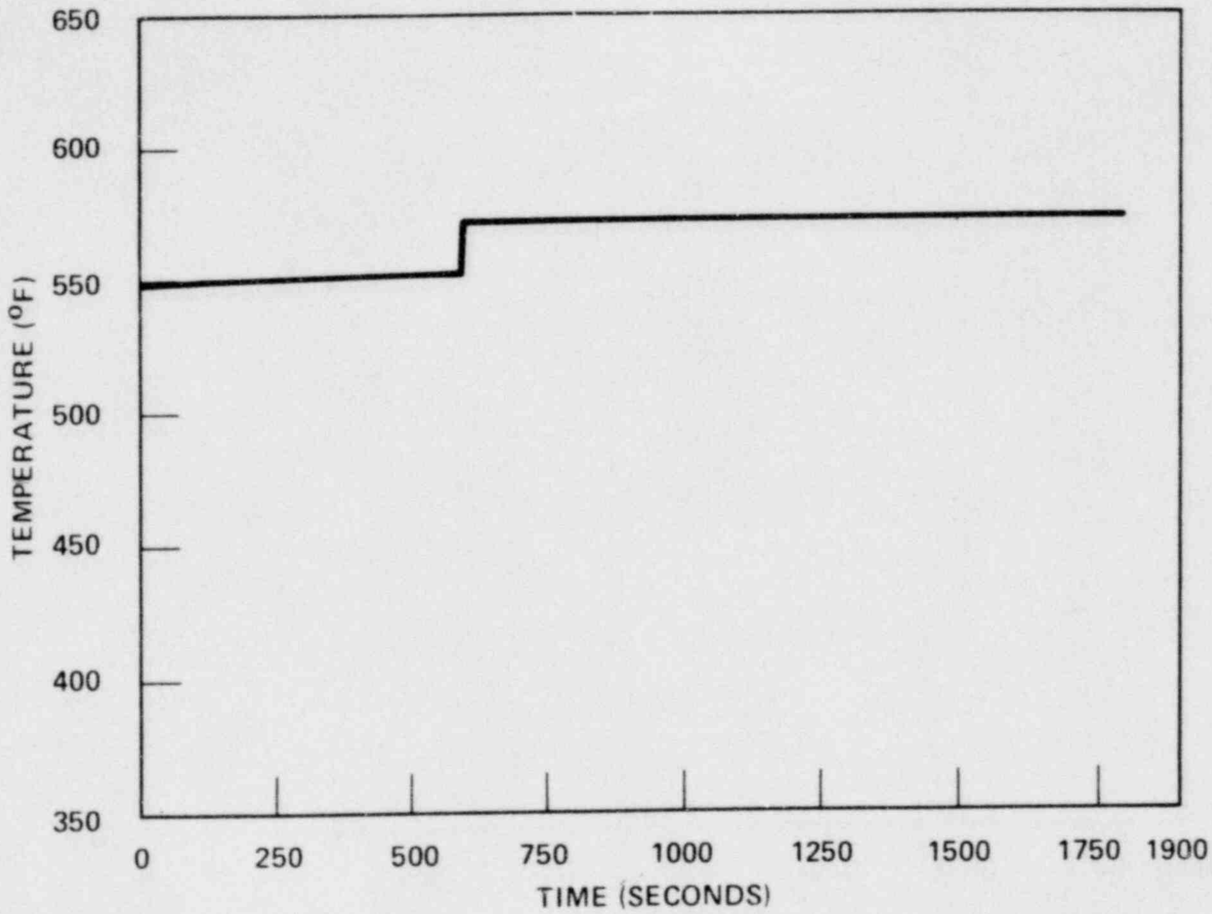
Steam Generator Tube Rupture

Figure 15.6.3-4

Steam Generator Pressure

Blue

16497-2



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Steam Generator Tube Rupture

Figure 15.6.3-5

Steam Generator Temperature

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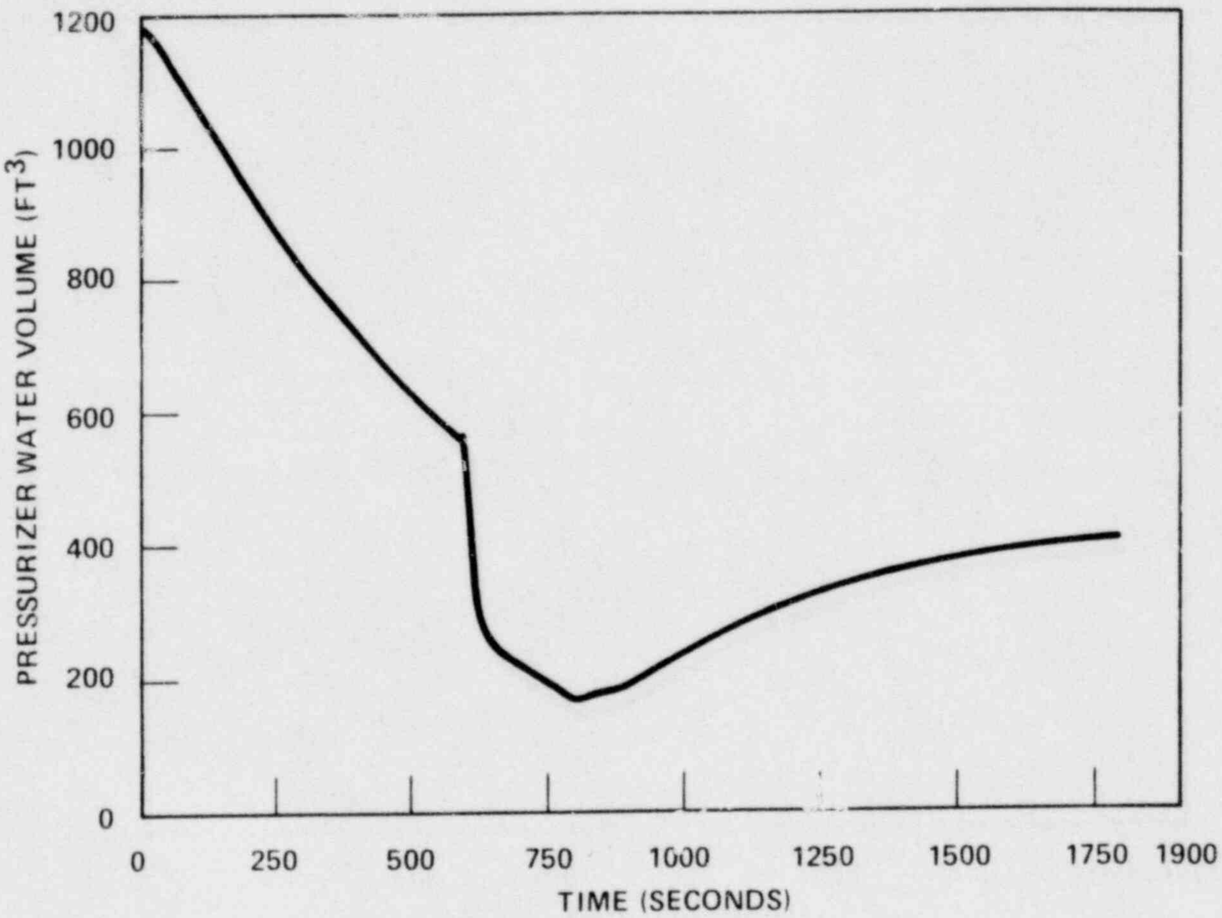

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Steam Generator Tube Rupture

Figure 15.6.3-6

Pressurizer Water Volume

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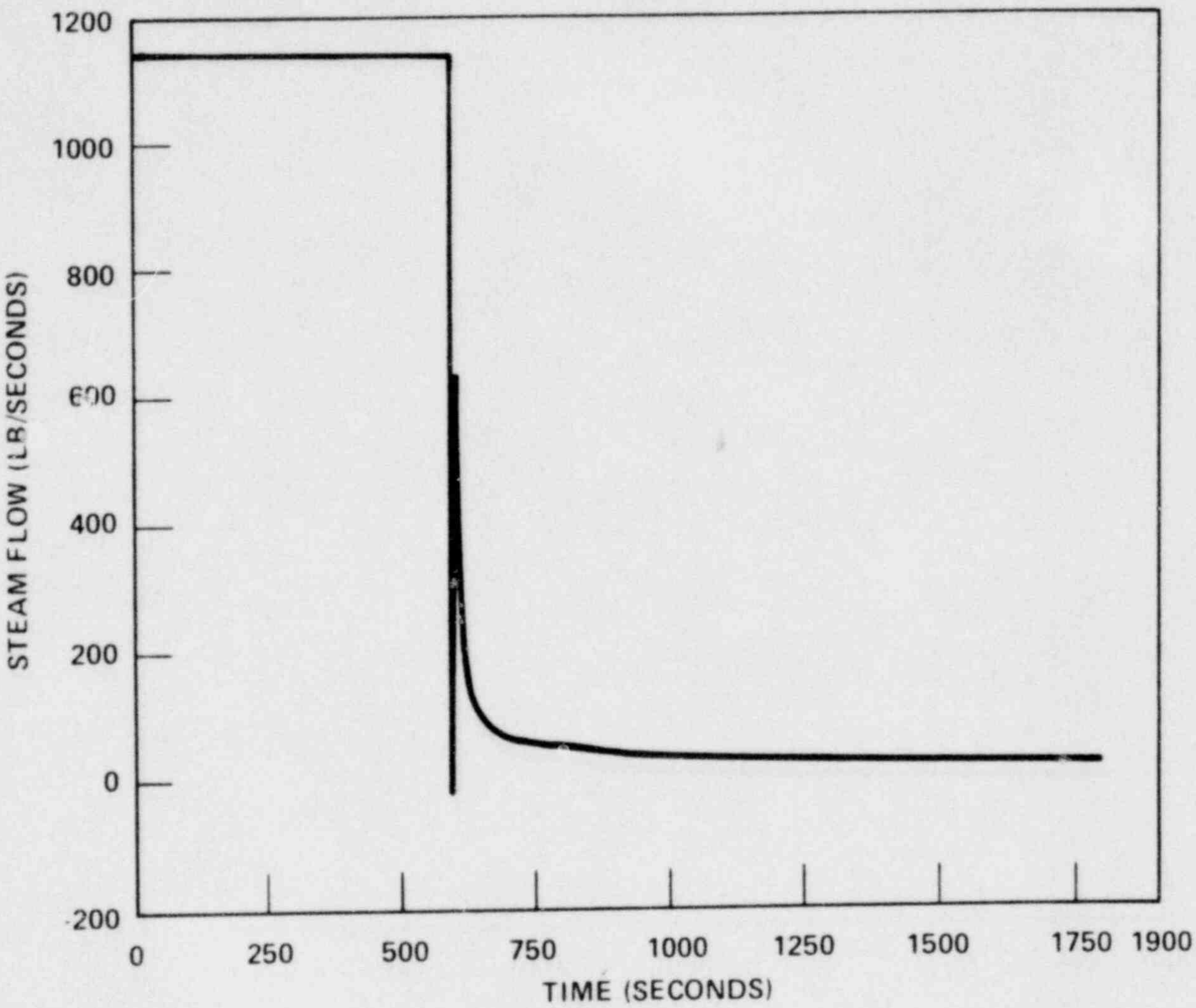

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Steam Generator Tube Rupture

Figure 15.6.3-7

Steam Generator Flow

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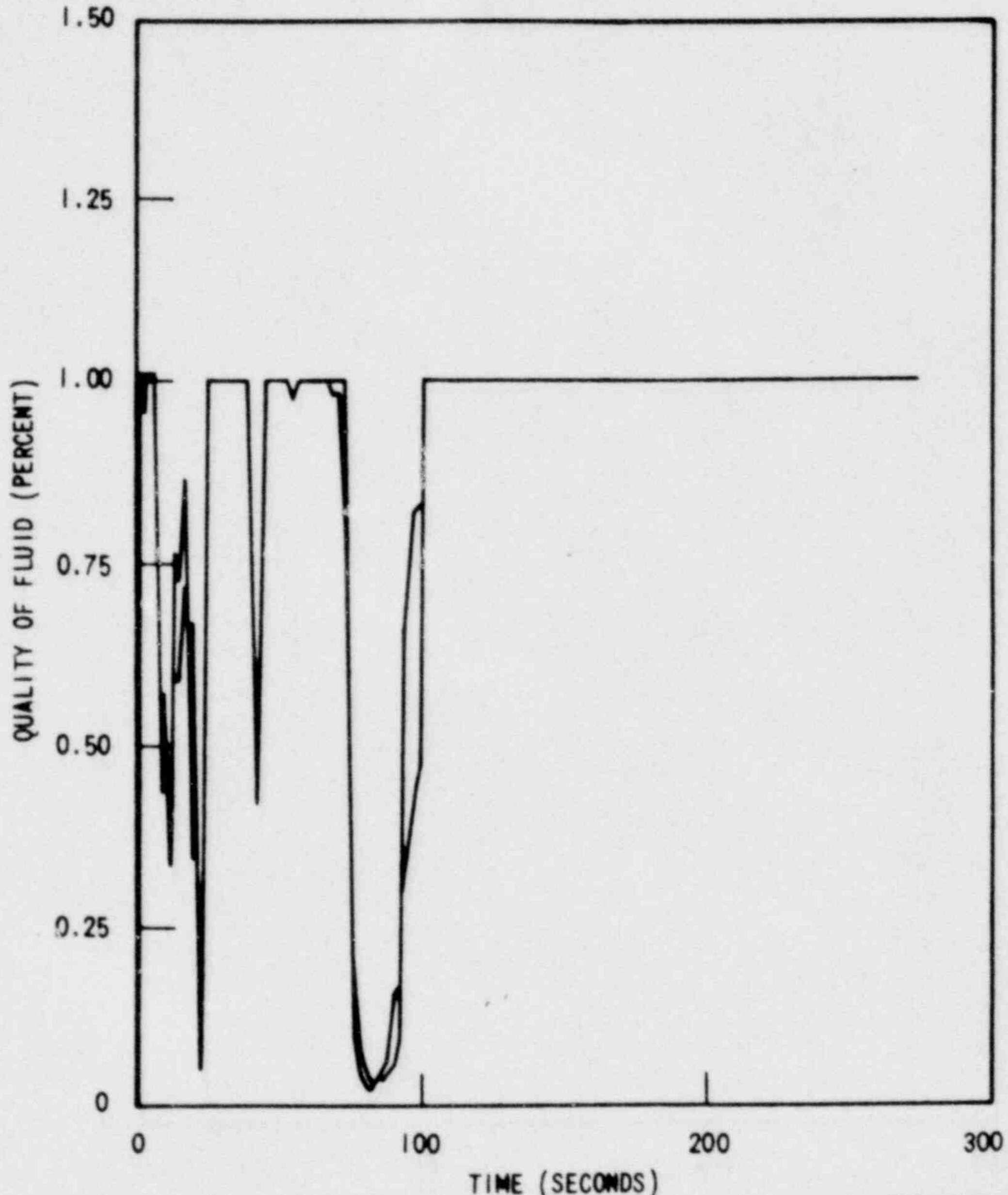

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Figure 15.6.5-3.
Fluid Quality – DECLG (C_D = 1.0)
Perfect Mixing

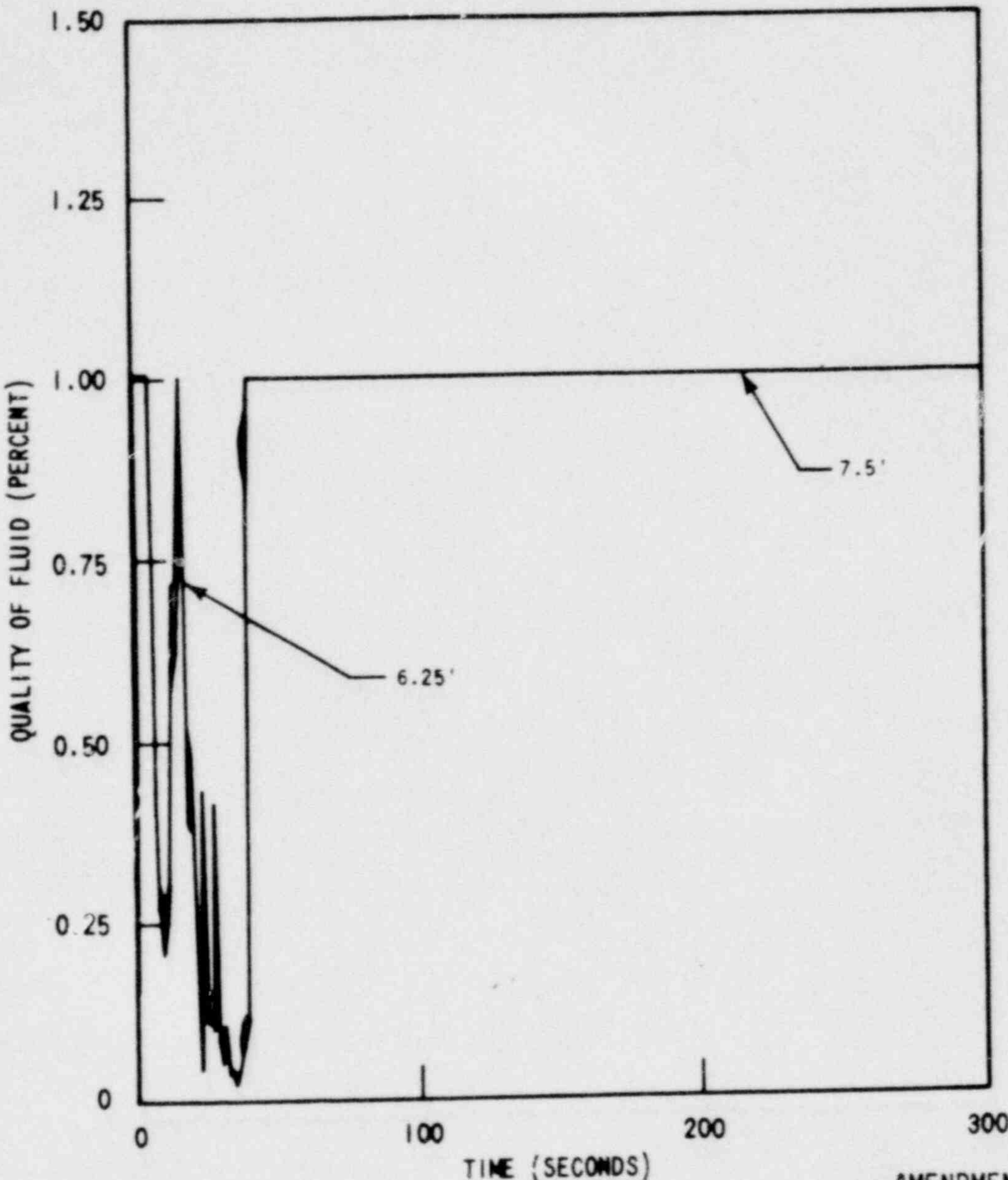

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Figure 15.6.5-4.
Fluid Quality – DECLG (C_D = 1.0)
Imperfect Mixing

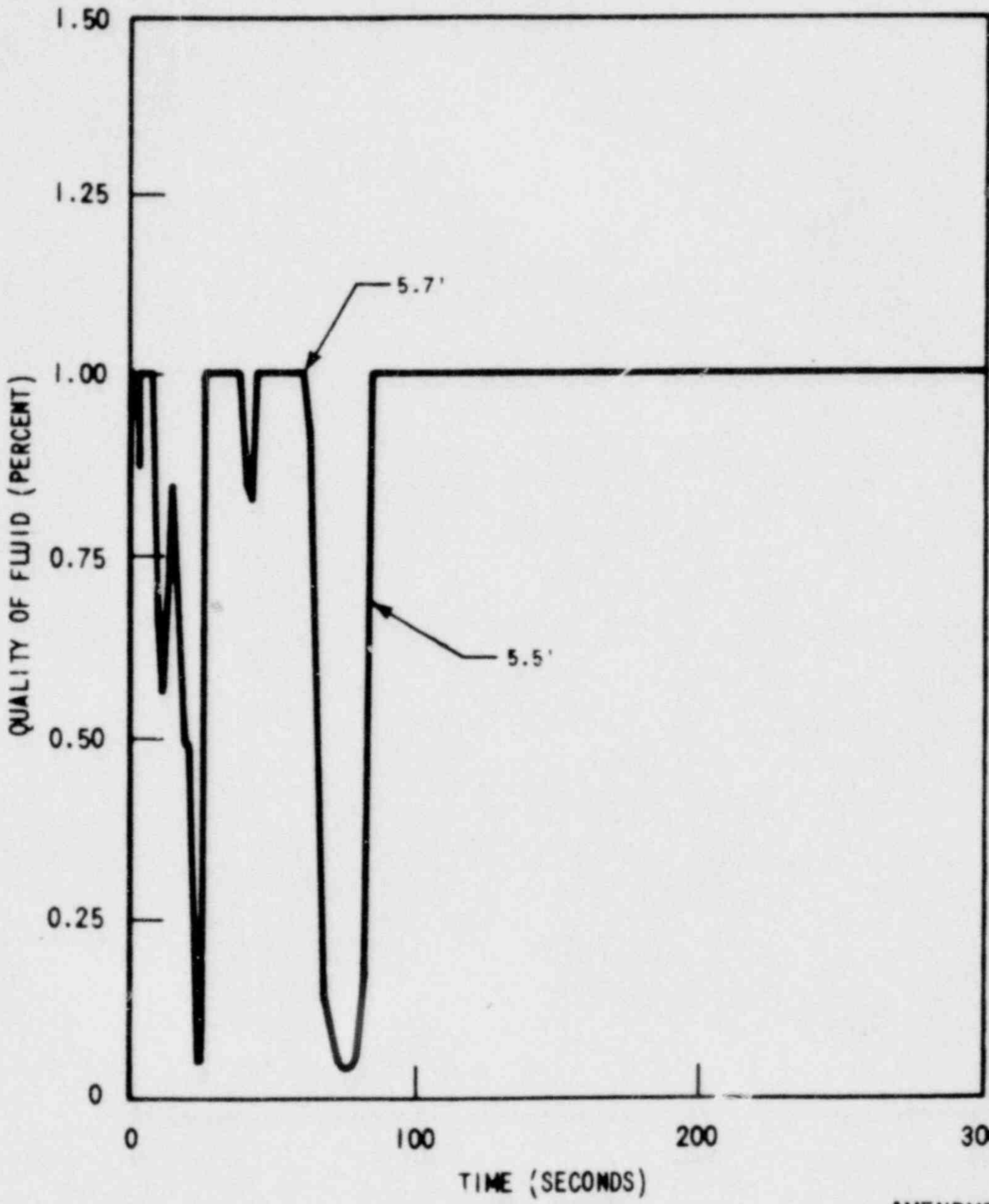

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Figure 15.6.5-5.
Fluid Quality – DECLG (C_D = 0.8)
Perfect Mixing

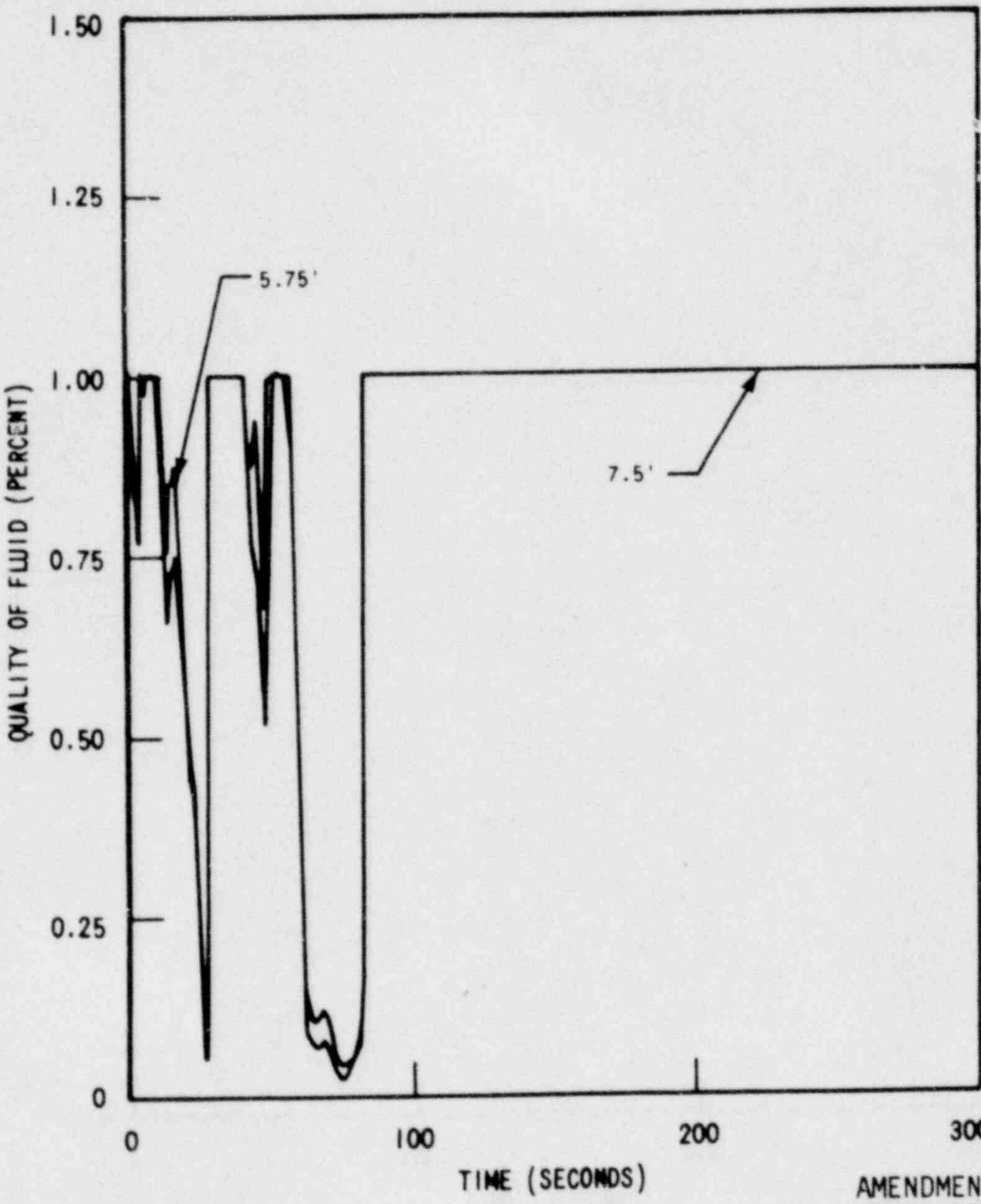

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Figure 15.6.5-6.
Fluid Quality – DECLG (C_D = 0.6)
Perfect Mixing

14.395-359

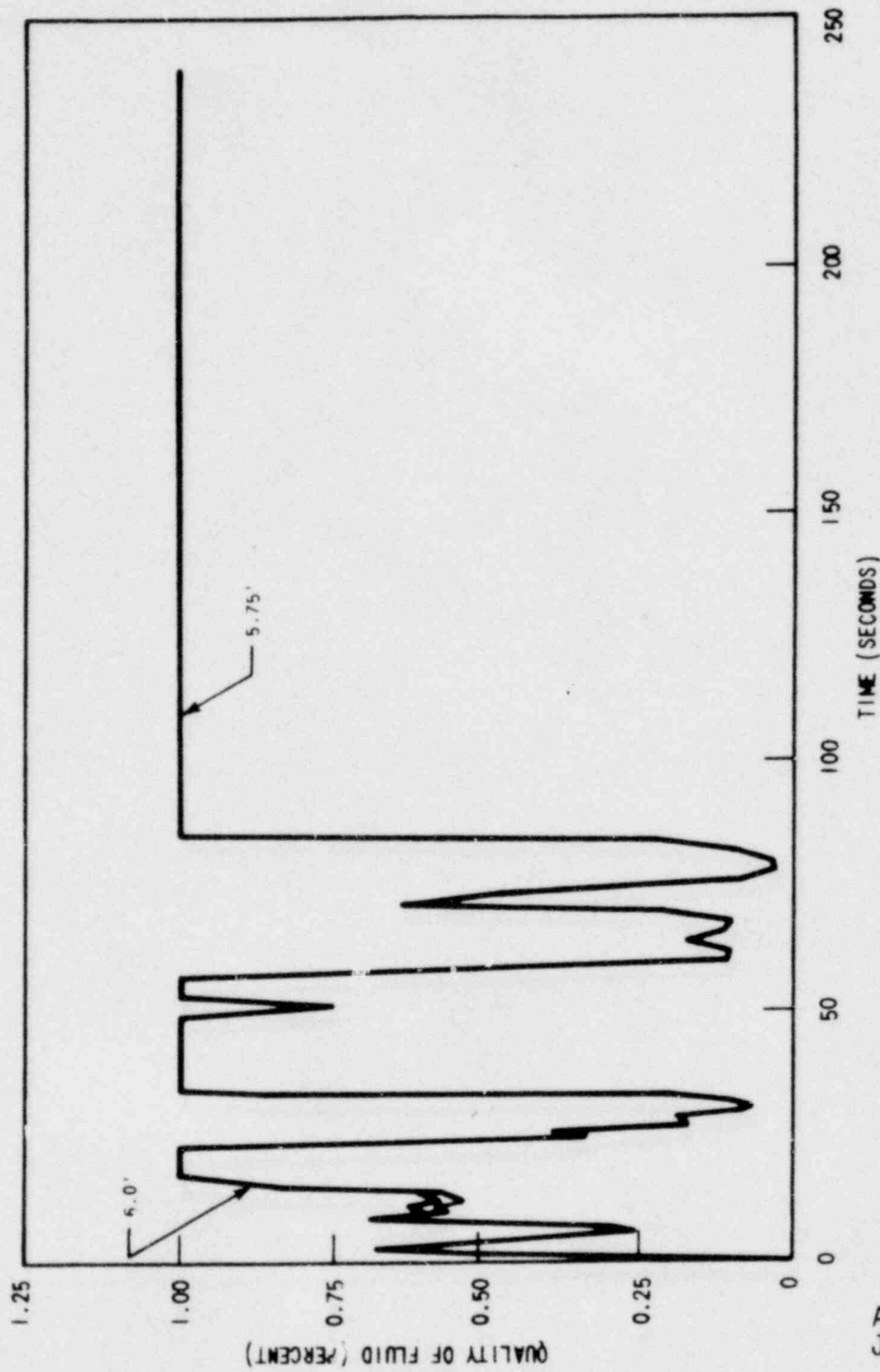

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Figure 15.6.5-7.
Fluid Quality – DECLG (C_D = 0.4)
Perfect Mixing

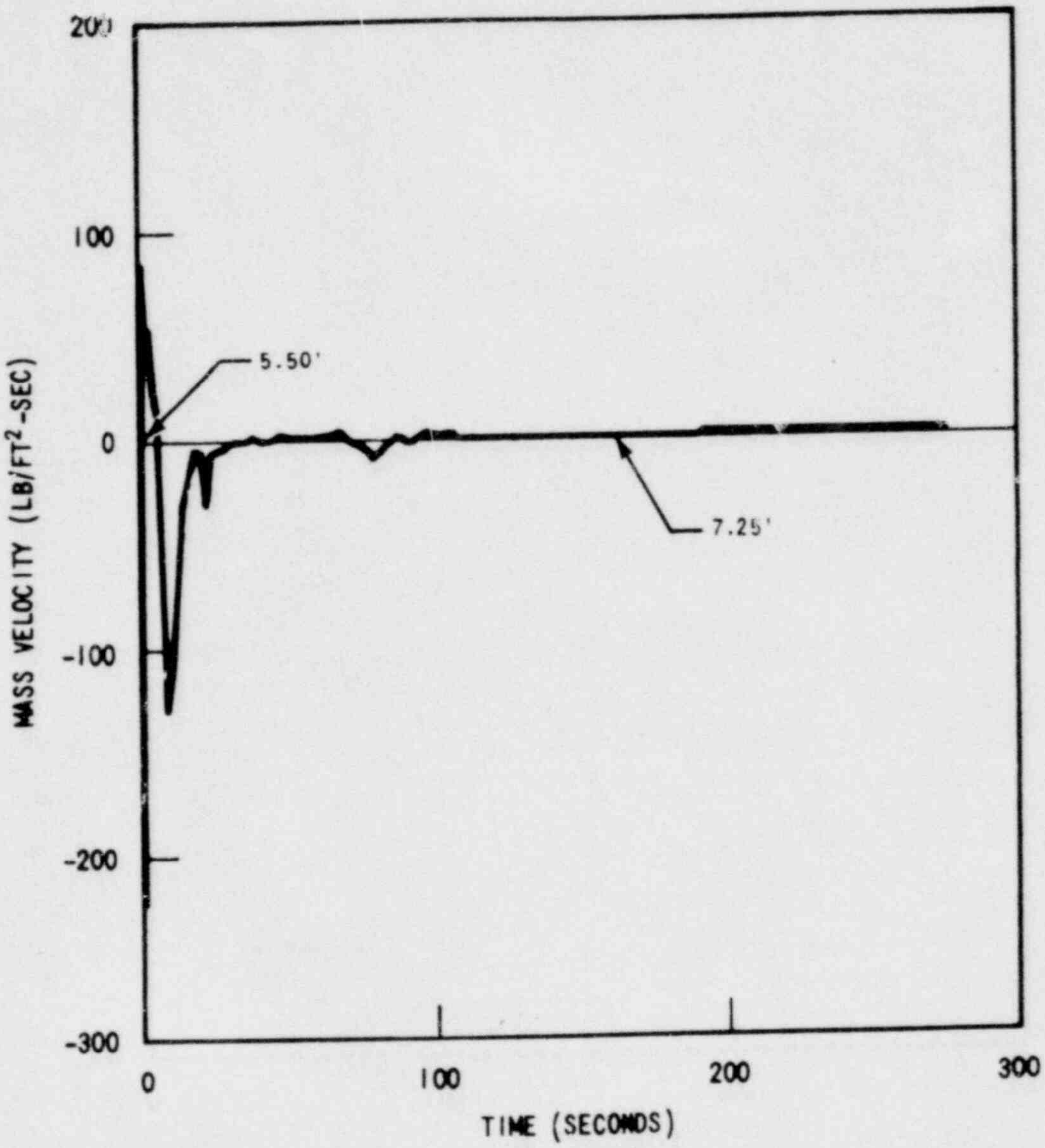

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Figure 15.6.5-8.
Mass Velocity – DECLG (C_D = 1.0)
Perfect Mixing

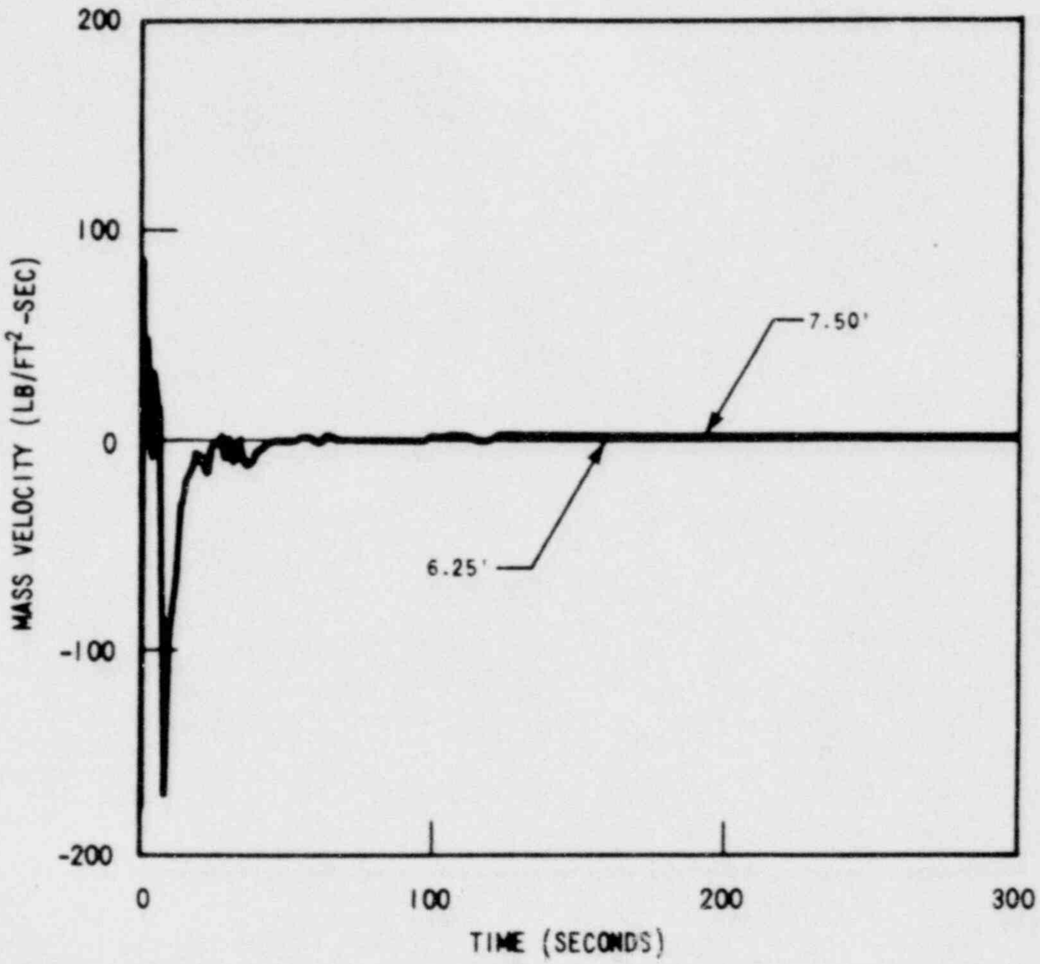

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Figure 15.6.5-9.
Mass Velocity -- DECLG (C_D = 1.0)
Imperfect Mixing

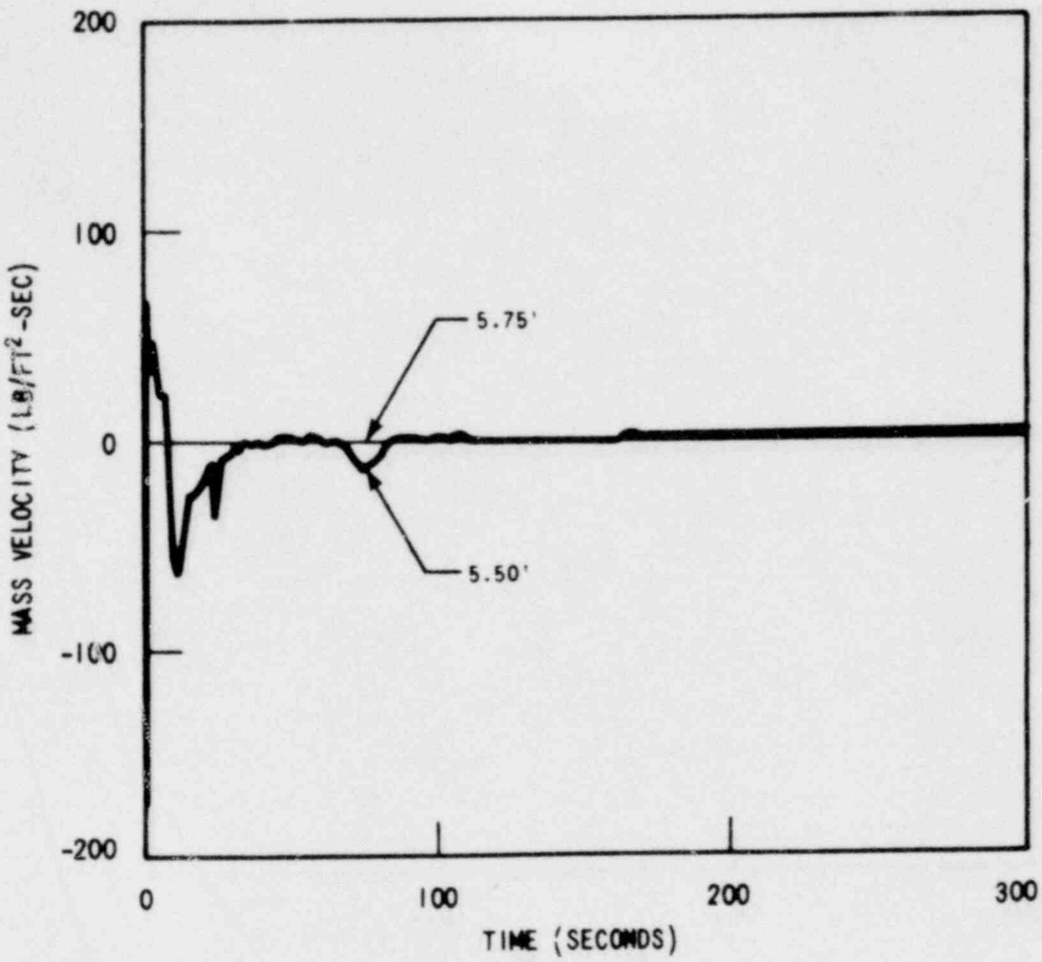

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Figure 15.6.5-10.
Mass Velocity – DECLG (C_D = 0.8)
Perfect Mixing

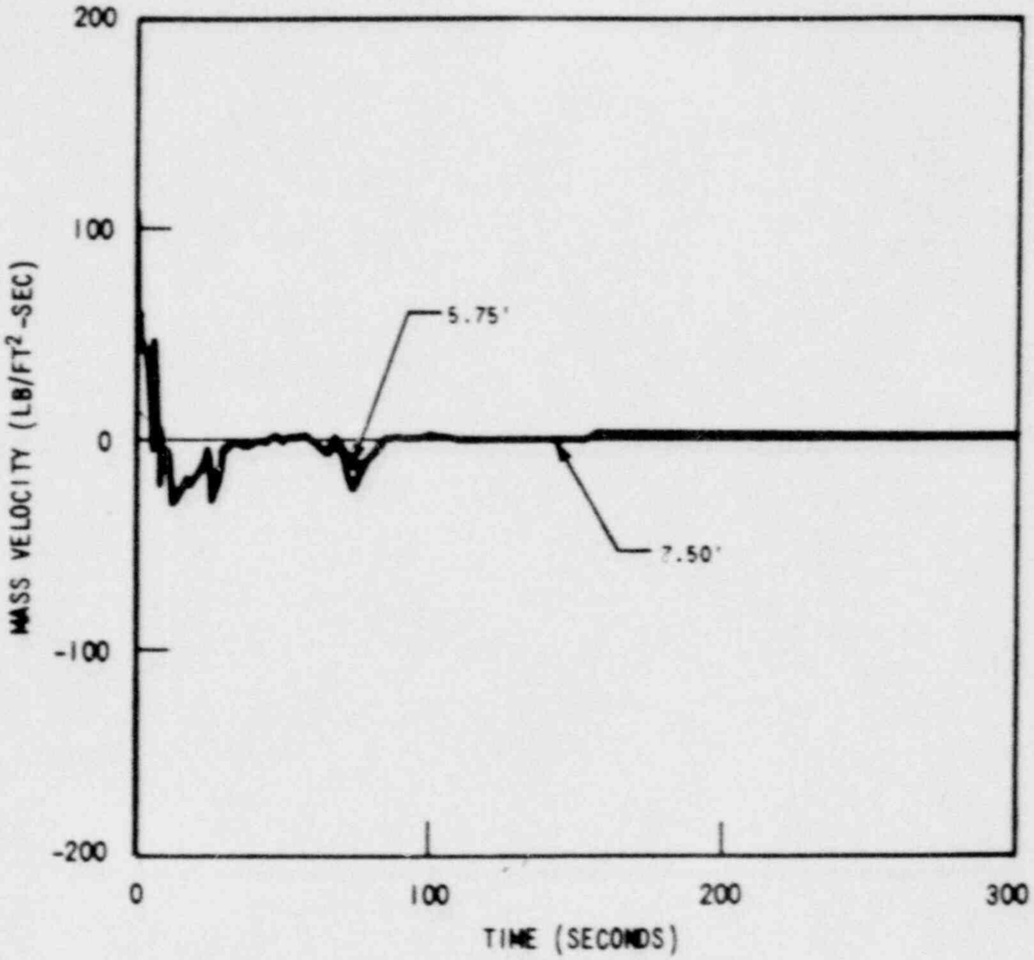

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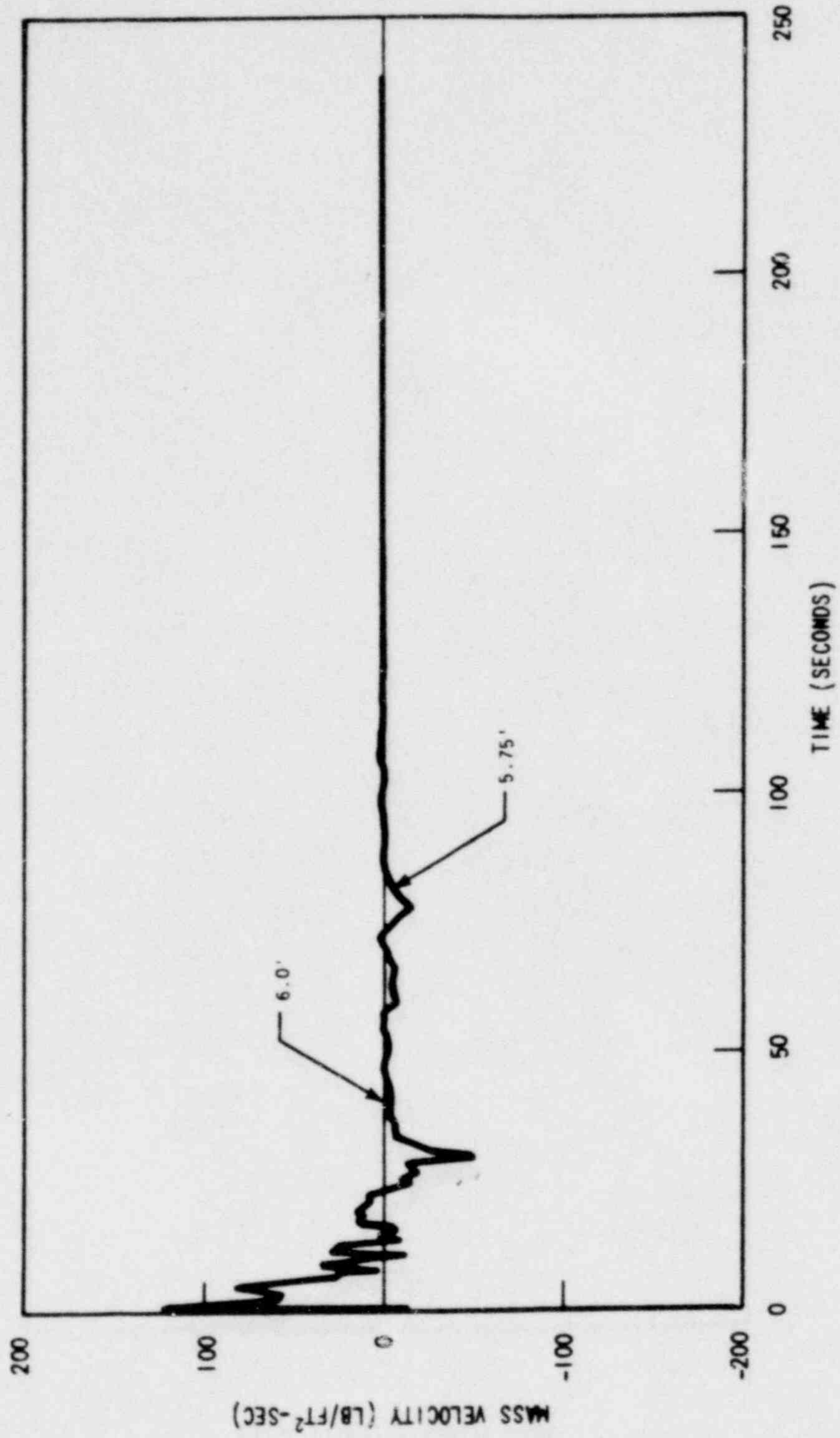
Figure 15.6.5-11.
Mass Velocity – DECLG (C_D = 0.6)
Perfect Mixing

14,395-364



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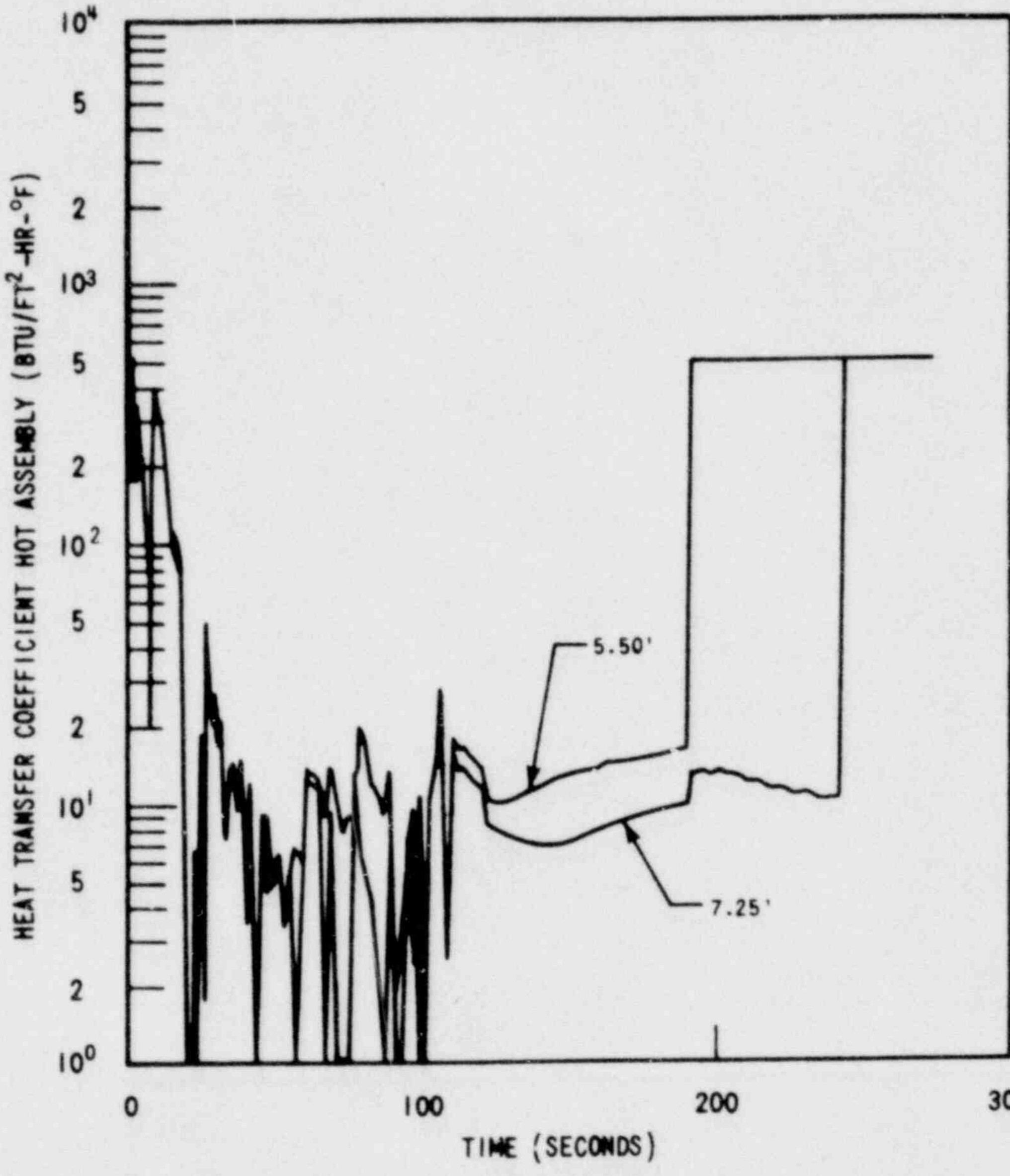
BLUE

Figure 15.6.5-12.
Mass Velocity – DECLG (C_D = 0.4)
Perfect Mixing



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BLUE

Figure 15.6.5-13.
Heat Transfer Coefficient
DECLG (C_D = 1.0)
Perfect Mixing

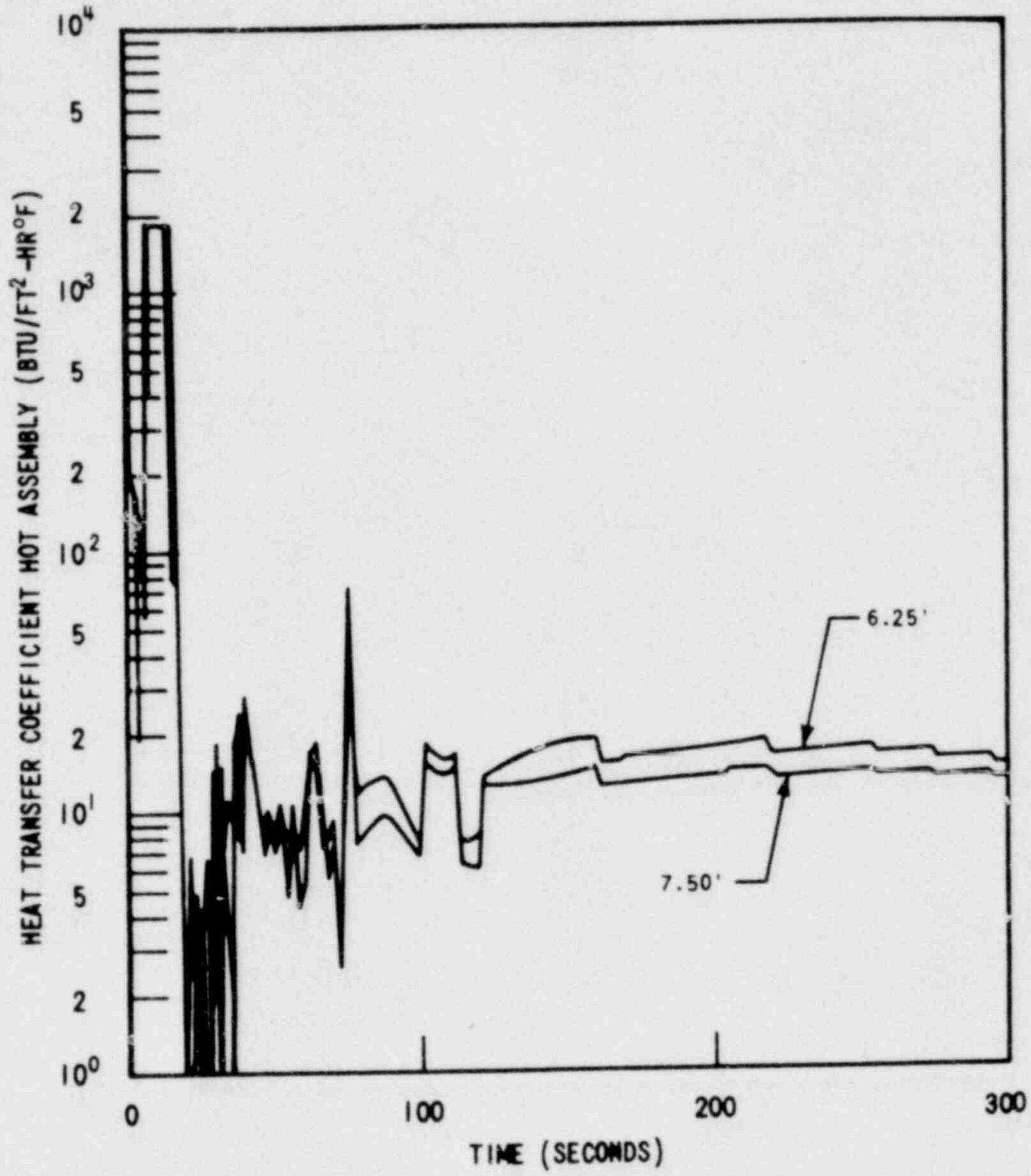

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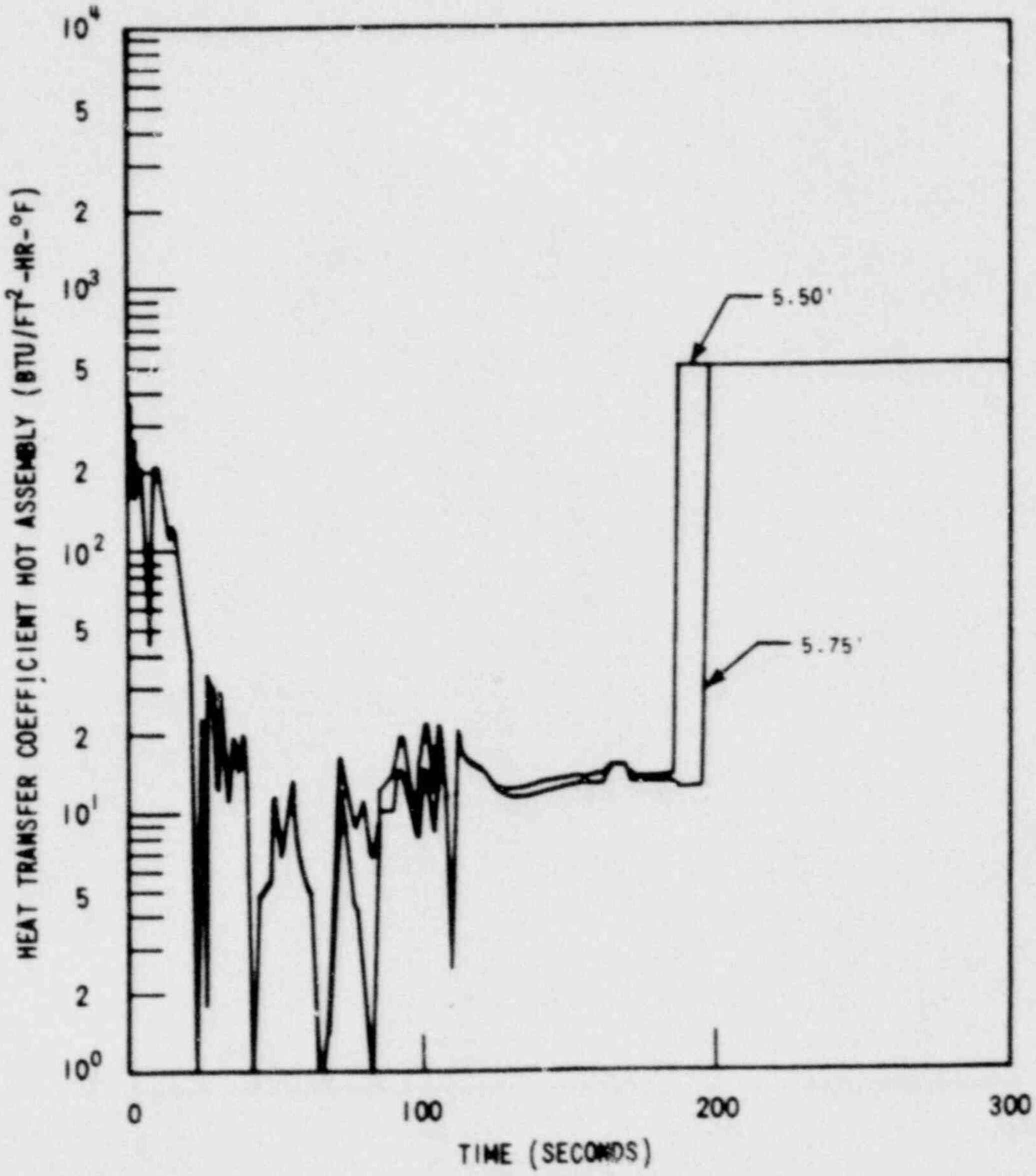
Figure 15.6.5-14.
Heat Transfer Coefficient
DECLG (C_D = 1.0)
Imperfect Mixing



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BLUE

Figure 15.6.5-15.

Heat Transfer Coefficient
DECLG (C_D = 0.8)
Perfect Mixing

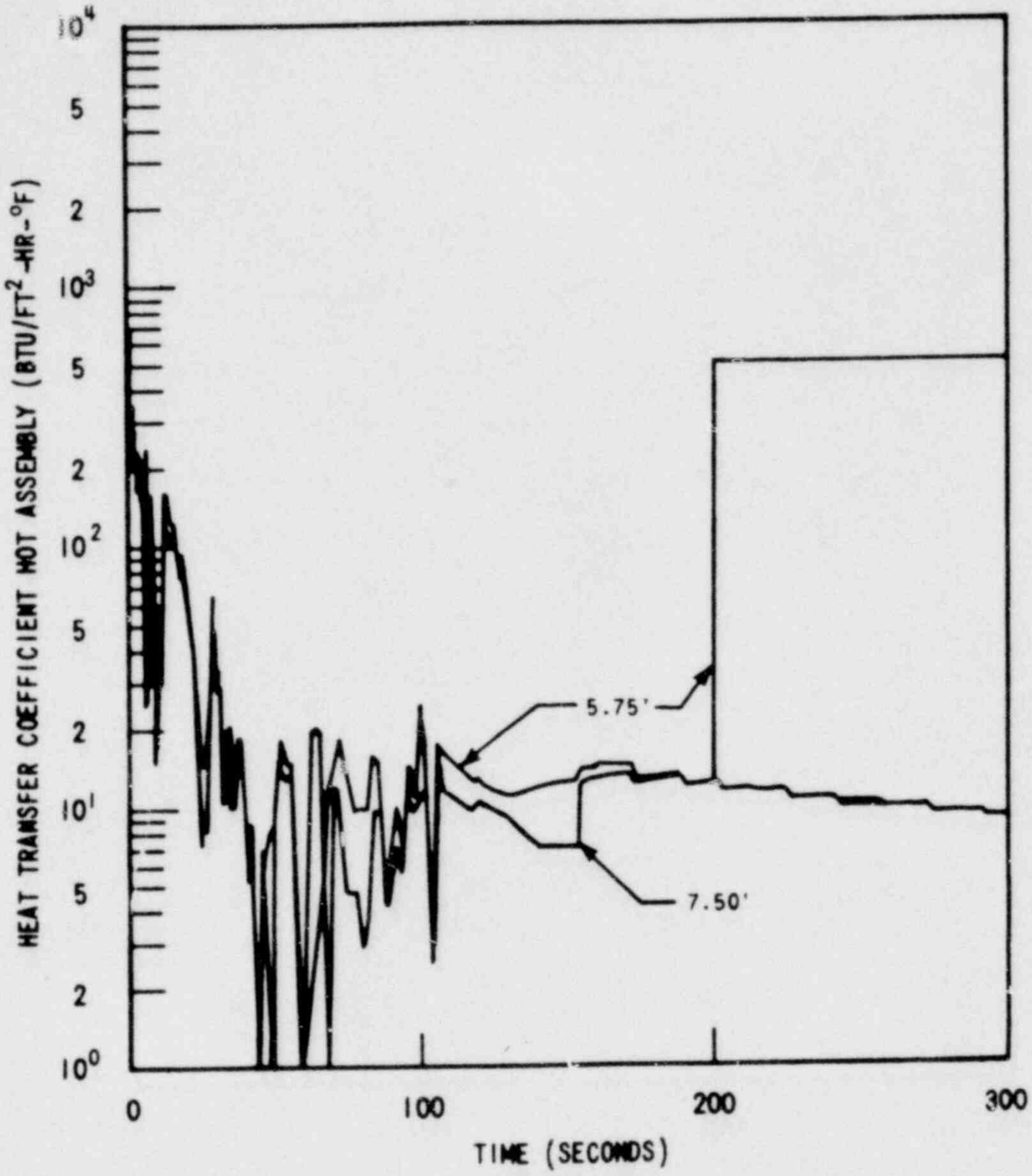

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Figure 15.6.5-16.
Heat Transfer Coefficient
DECLG (C_D = 0.6)
Perfect Mixing

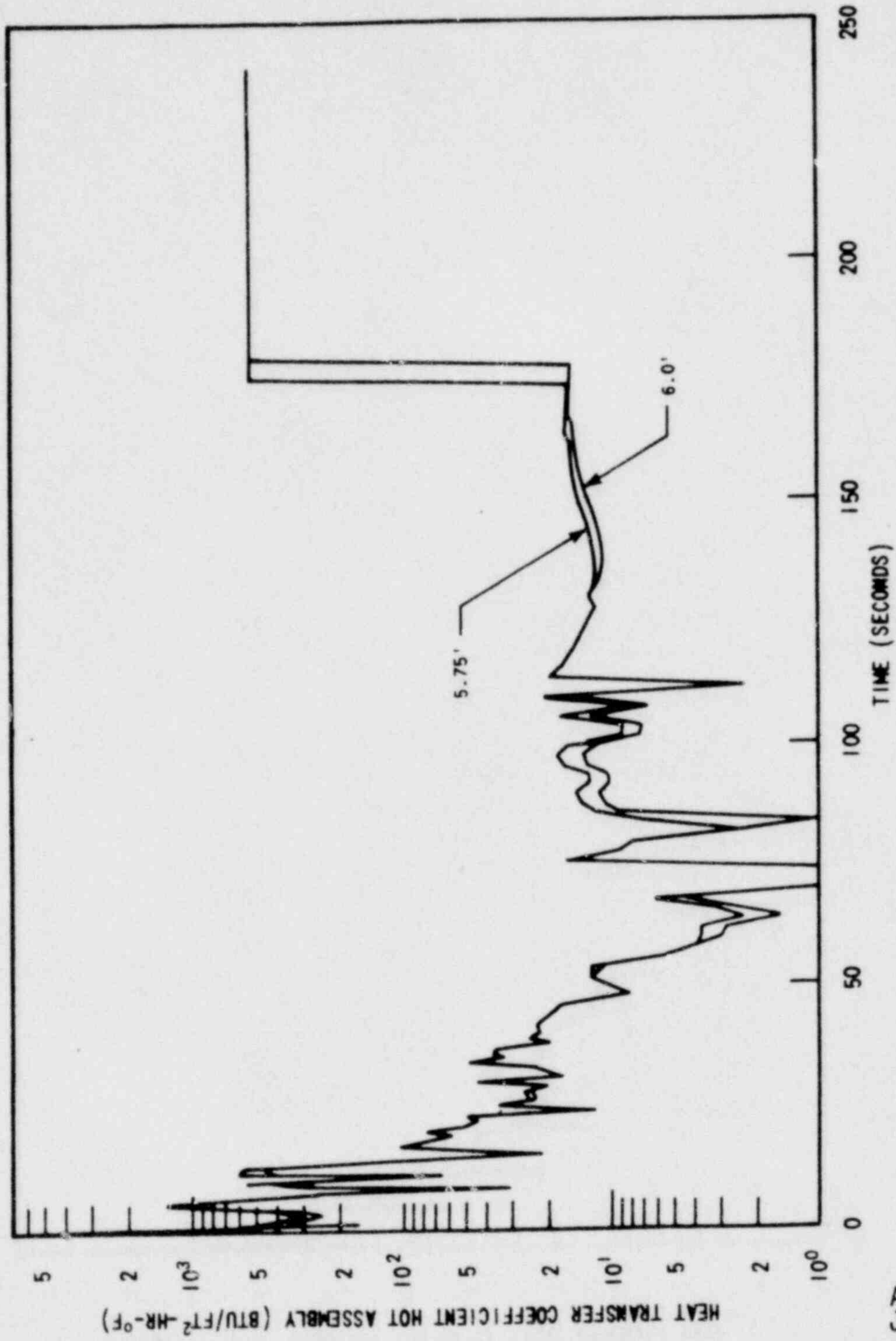

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Figure 15.6.5-17.

Heat Transfer Coefficient
DECLG (C_D = 0.4)
Perfect Mixing

14,395-370

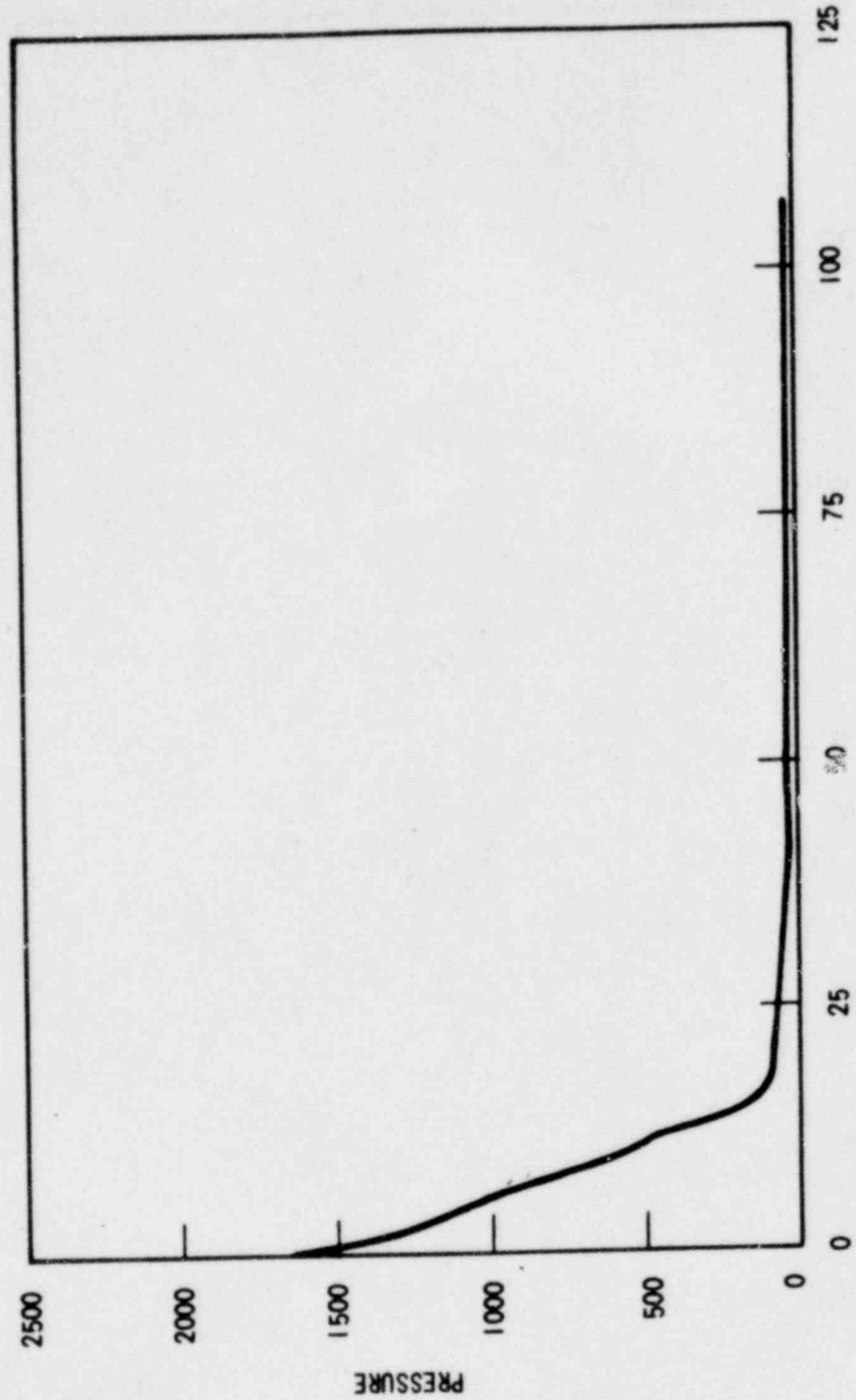

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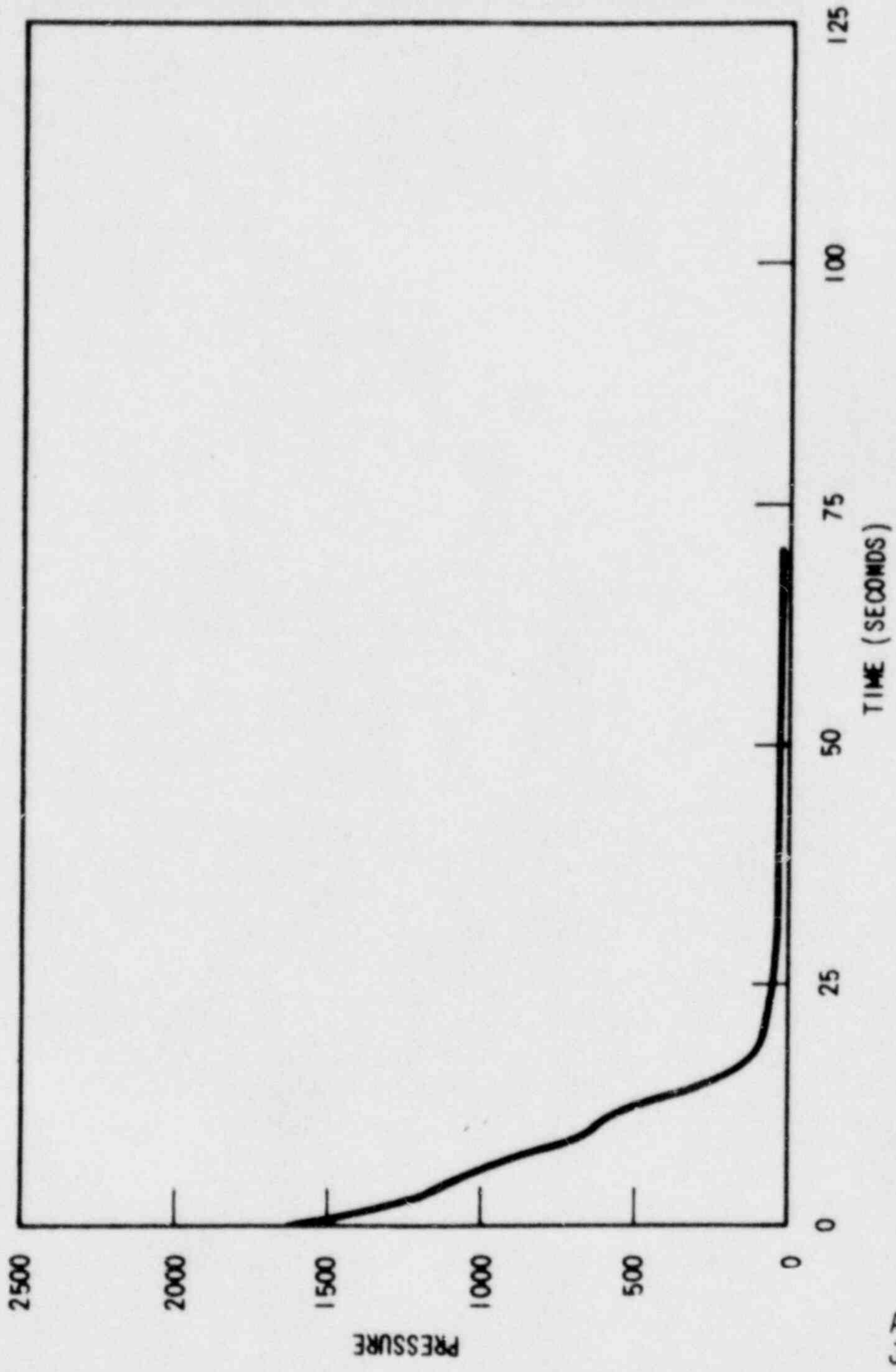
Figure 15.6.5-18.
Core Pressure – DECLG (C_D = 1.0)
Perfect Mixing



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BLUE

Figure 15.6.5-19.
Core Pressure – DECLG (C_D = 1.0)
Imperfect Mixing

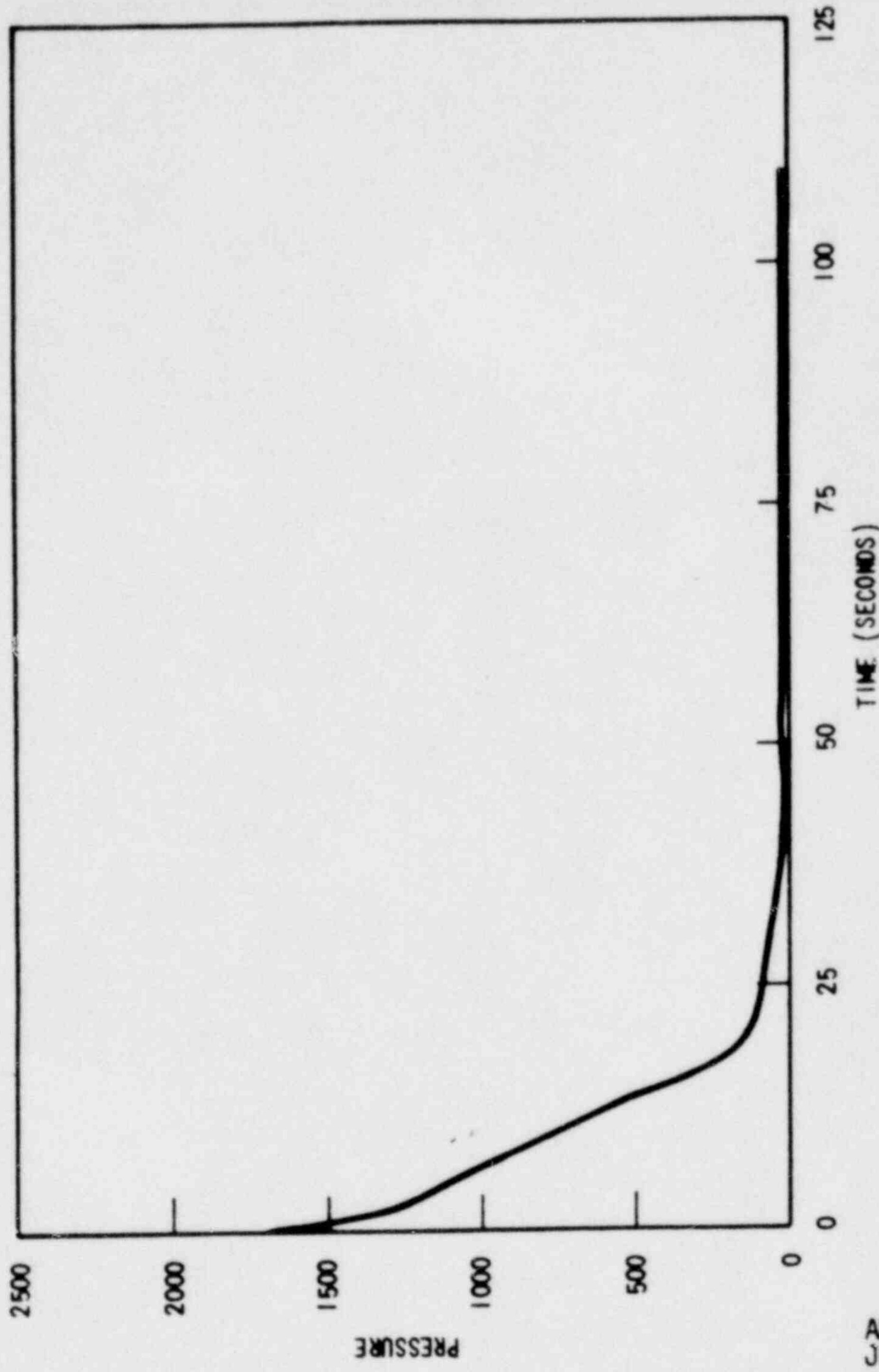

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Figure 15.6.5-20.
Core Pressure – DECLG (C_D = 0.8)
Perfect Mixing

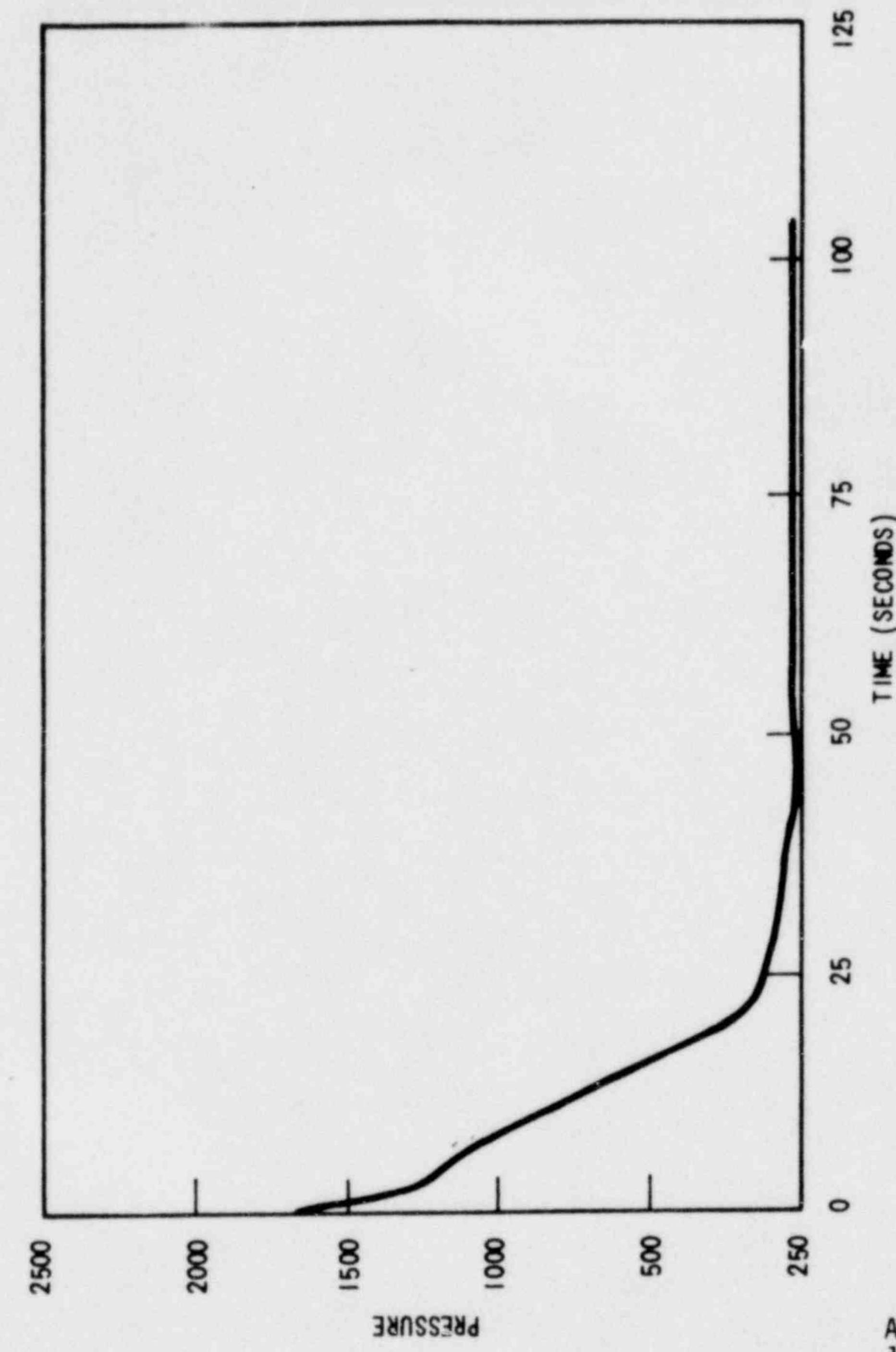

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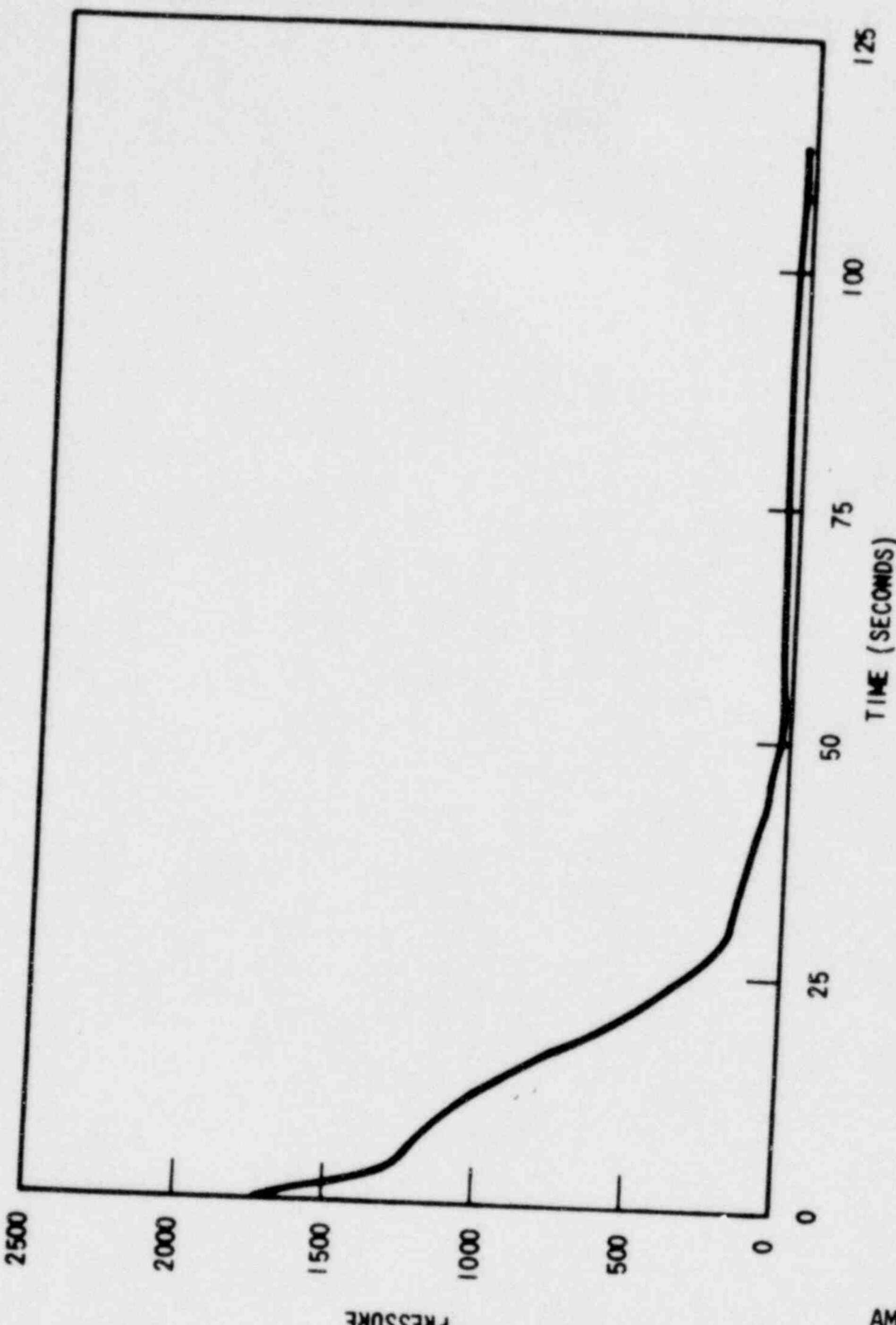
Figure 15.6.5-21.
Core Pressure — DECLG (C_D = 0.6)
Perfect Mixing

14.395-374



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BLUE

Figure 15.6.5-22.
Core Pressure – DECLG (C_D = 0.4)
Perfect Mixing