XN-NF-79-45

FORT CALHOUN LOCA ANALYSES AT 1500 MWT USING ENC WREM-IIA PWR ECCS EVALUATION MODEL

LARGE BREAK EXAMPLE PROBLEM

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1.0 INTRODUCTION AND SUMMARY

The purpose of this document is to show, by an example large break LOCA ECCS calculation, the system input model and example results that Exxon Nuclear Company (ENC) has obtained using the ENC WREM ECCS Evaluation model for the Fort Calhoun reactor operating at design power (1500 MWt). Calculated results are demonstrated to meet the criteria specified by 10 CFR 50.46 and to be performed by an approved ECCS evaluation model in conformance to 10 CFR 50, Appendix K.

This document presents the results of a large break analyses for the Fort Calhoun reactor using the ENC-WREM-IIA ECCS model at the proposed 1500 MWt power, 2250 psia pressurizer pressure, and 545°F core inlet temperature with ENC 14x14 reload fuel. The postulated loss-of-coolant accident (LOCA) evaluated for the example calculation was the double-ended guillotine configuration for one pump discharge line using a discharge coefficient of 0.6.

The analysis involved calculations using the following ENC-WREM-IIA code version[1,3,6,7] RELAP4-EM/ENC28F for blowdown and hot channel analyses, REFLEX[3,8] for core reflood analysis, CONTEMPT LT/22 for containment backpressure analysis, and TOODEE2/MAY79 (see Section 8.1 for latest update to TOODEE2) for heatup analysis. FLECHT/ENC2 multipliers,[1] developed for PWR's with containment backpressure similar to calculations for Fort Calhoun, were used in these analyses.

Principal results of these analyses for the 0.6 DECLG break are contained in Tables 1.1 and 1.2. These results include a Peak Cladding Temperature (PCT) of 2067°F and a maximum local Zr/H_2O reaction of less than 7.7%. These analyses were performed at 1530 MWT which is 102 percent of rated power as

indicated in Table 1.1. Including the Appendix K required two percent core power uncertainty factor, this analysis, if determined to be the limiting break, would support operation of the plant with a total Linear Heat Generation Rate (LHGR) of 15.53 kW/ft. Also included in Table 1.2 are summaries of the transient times calculated for major events. Additional results of this analysis are presented in subsequent sections of the report.

Table 1.1

Analysis Results of Example Problem (DECLS C_D = 0.6)

Analysis Results	DECLS
Peak Clad Temperature, °F	2067
Peak Clad Temperature Location, ft from Bottom	7.97
Local Zr/H_2O Reaction (max), %	<7.7%
Local Zr/H_2O Location, ft from Bottom	7.47
Total H_2 Generation, % of Total Zr Reacted	<<1.0
Hot Rod Burst Time, Sec	29.7
Hot Rod Burst Location, ft from Bottom	7.47
Peak Linear Heat Generation Rate, BOCREC, kW/ft	.820
Calculation	
License Core Power MWt	1500
Power Used for Analysis, MWt	1530
Peak Linear Power kW/ft	15.22*
Total Peaking Factor	2.53

---

* All power generated in fuel at 1500 MWt.

Table 1.2

Large Break Results
Time Sequence of Events

Event	Time of Event (seconds) DECLG ($C_D = 0.6$)
Start	0.0
Initiate Break	0.05
Safety Injection Signal	0.60
Pressurizer Empties	10.75
Accumulator Injection, Broken Loop	11.79
Accumulator Injection, Single Intact Loop	16.39
Accumulator Injection, Double Intact Loop	16.39
End-of-Bypass	19.78
Safety Injection Flow, SIS	20.5
Start of Reflood	34.17
Accumulators Empty, Single Intact Loop	58.49
Accumulators Empty, Double Intact Loop	58.06
Peak Clad Temperature Reached	166.6

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2.0 BLOWDOWN CALCULATION

2.1 MODEL DESCRIPTION

The input data for the example calculation was chosen to represent the Fort Calhoun (Combustion Engineering design) pressurized water reactor with a dry containment. Table 2.1 gives a general listing of some of the plant parameters used to develop the input for this model.

The system was modeled using 55 volumes, 70 junctions and 50 heat slabs (Figure 2.1). The model includes four accumulators (safety injection tanks), one pressurizer, and two vertical U-tube steam generators, with both primary and secondary sides of the steam generators modeled. The high and low pressure Safety Injection System (SIS) flows were modeled as fill junctions with typical flow rates given as a function of system backpressure. The reactor core power is calculated by the RELAP4-EM solution of the space independent core kinetics equations with radioactive decay energy (ANS + 20%) and actinide contributions. Mass and energy release rates from the primary coolant system to the containment were used as input to the CONTEMPT LT/22 code for containment backpressure analysis. The pump performance curves were for a Byron-Jackson pump, specific speed 5210, provided by the Omaha Public Power District (OPPD). The reactor core was modeled radially as an average core plus a single hot assembly, each with three axial nodes. Major hardware components were modeled using 38 heat conductors.

2.2 RESULTS

Results from the application of the RELAP4-EM program to the blowdown of the system are presented in Figures 2.2 through 2.6. The timing of various major blowdown events are listed in Table 1.2.

The results are typical of pressurized water reactor large break blowdown analyses. For large break analyses the system decompresses rapidly to the saturation point, and the pressure then continues to decrease smoothly to an ambient condition. After break initiation, the core inlet flow reverses, approaching a zero flow condition for several seconds before reversing again. The core flow approaches zero at the End-of-Bypass (EOBY).

Table 2.1

Fort Calhoun PWR Data

Primary Heat Output, MWt	1500*
Primary Coolant Flow, lbm/hr	7.223×10^7
Primary Coolant Volume, ft^3	12,062**
Operating Pressure, psia	2250
Inlet Coolant Temperature, °F	545
Reactor Vessel Volume, ft^3	2986
Pressurizer Volume, Total, ft^3	900
Pressurizer Volume, Liquid, ft^3	500
Accumulator Volume, Total, ft^3 (one of four)	1300
Accumulator Volume, Liquid, ft^3	825
Accumulator Pressure, psia	255
Steam Generator Heat Transfer Area, ft^2 (one of two)	47,184
Steam Generator Secondary Flow, lbm/hr	3.38×10^6
Steam Generator Secondary Pressure, psia	853
Reactor Coolant Pump Head, ft	201
Reactor Coolant Pump Speed, rpm	1192
Moment of Inertia, lbm ft^2/rad	71,000
Cold Leg Pipe, I.D., in	24.0
Hot Leg Pipe, I.D., in	32.0
Pump Suction Pipe, I.D., in	24.0

* Primary Heat Output used in RELAP4-EM Model - 1.02 x 1500 = 1530 MWt.
** Includes total accumulator and pressurizer volumes.

Table 2.1 (continued)

Fuel Assembly Rod Diameter, in*	.440
Fuel Assembly Rod Pitch, in*	.580
Fuel Assembly Pitch, in*	8.180
Fueled (Core) Height, in*	128.0
Fuel Heat Transfer Area, ft^2	28,762
Fuel Total Flow Area, ft^2	32.812
Steam Generator Tube Plugging (Assumed uniform)	1%

* ENC fuel parameters.

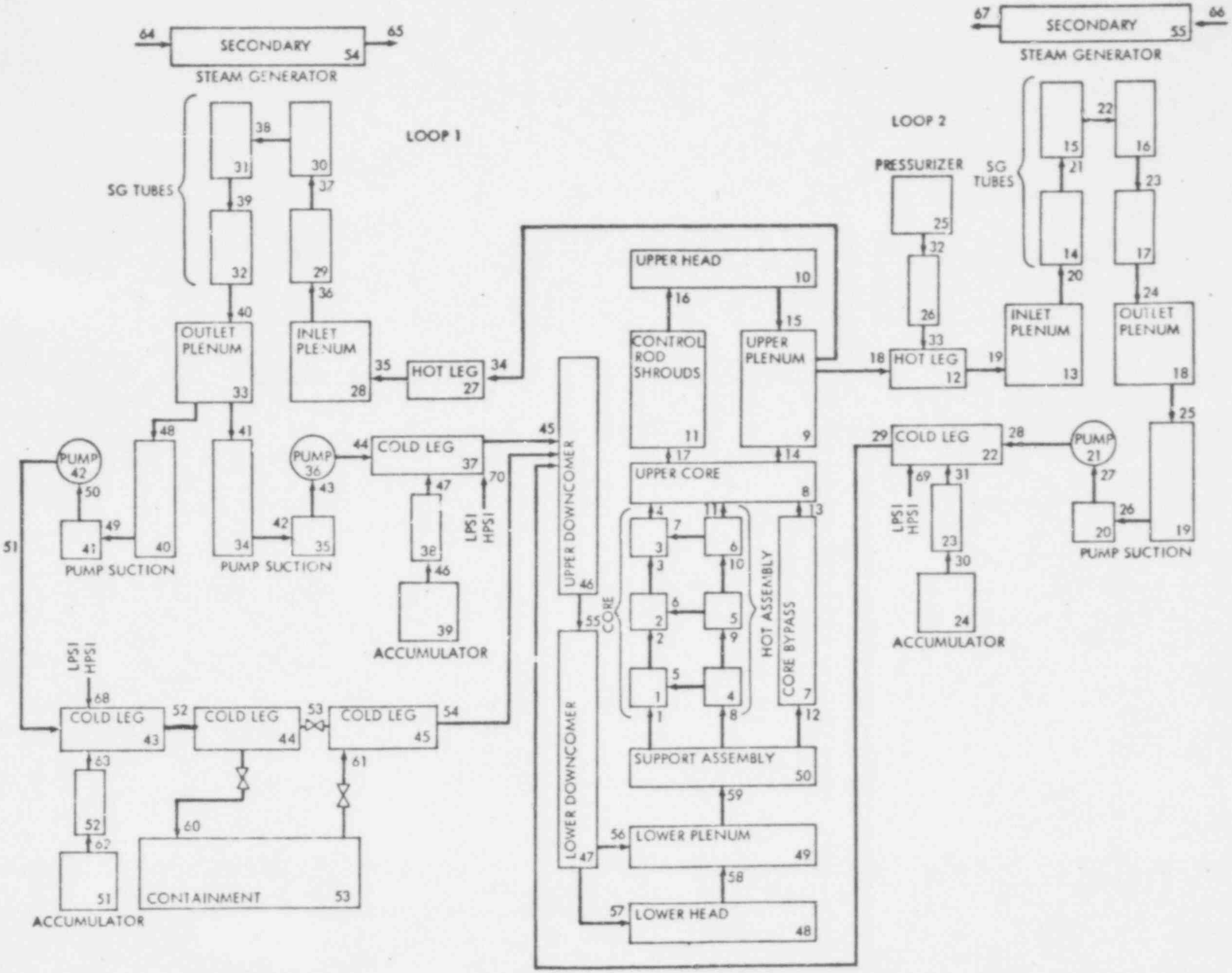

Figure 2.1 RELAP4-EM Blowdown System Nodalization for Fort Calhoun PWR

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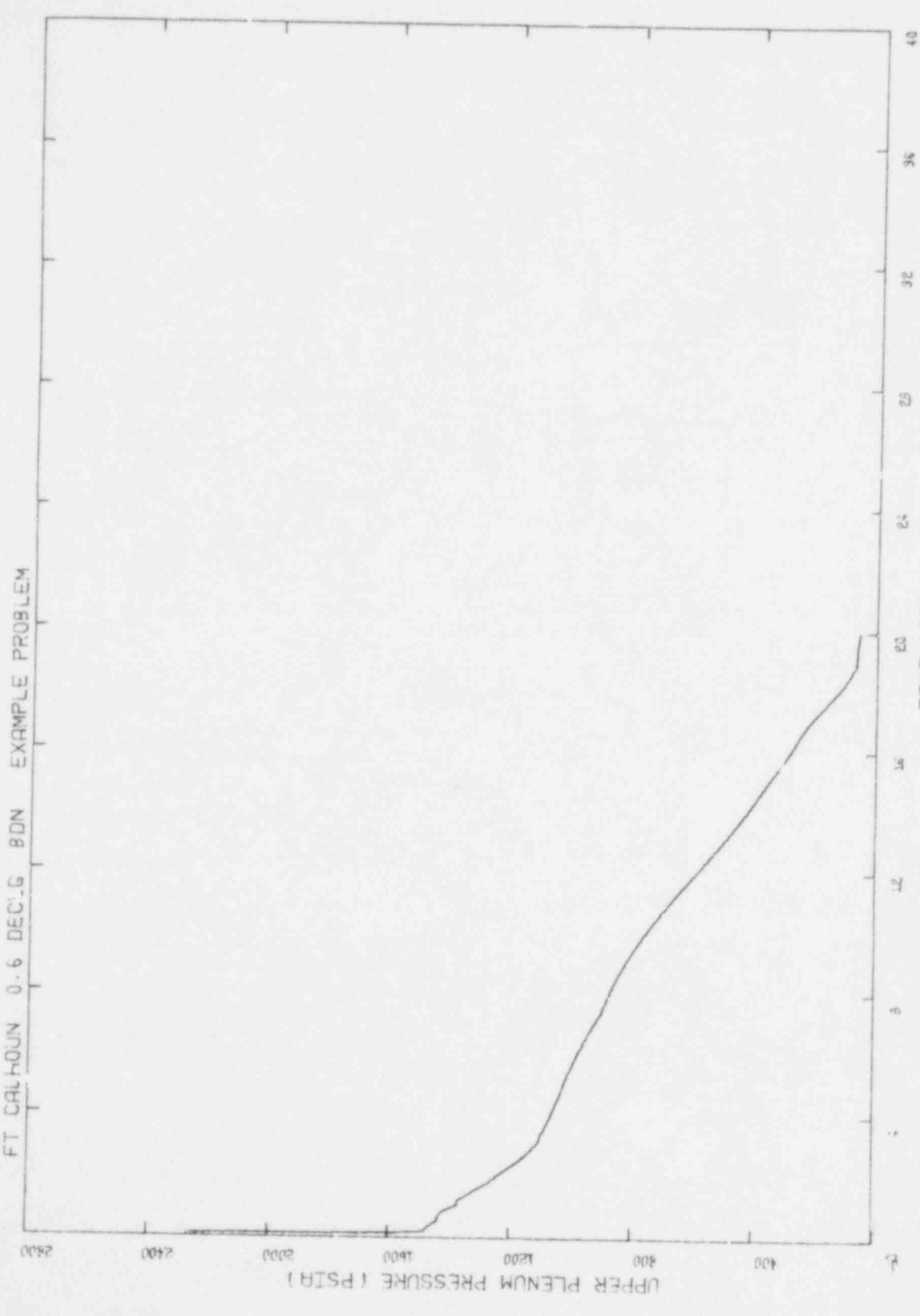

Figure 2.2 Upper Plenum Pressure, DECLG (C_D = 0.6)

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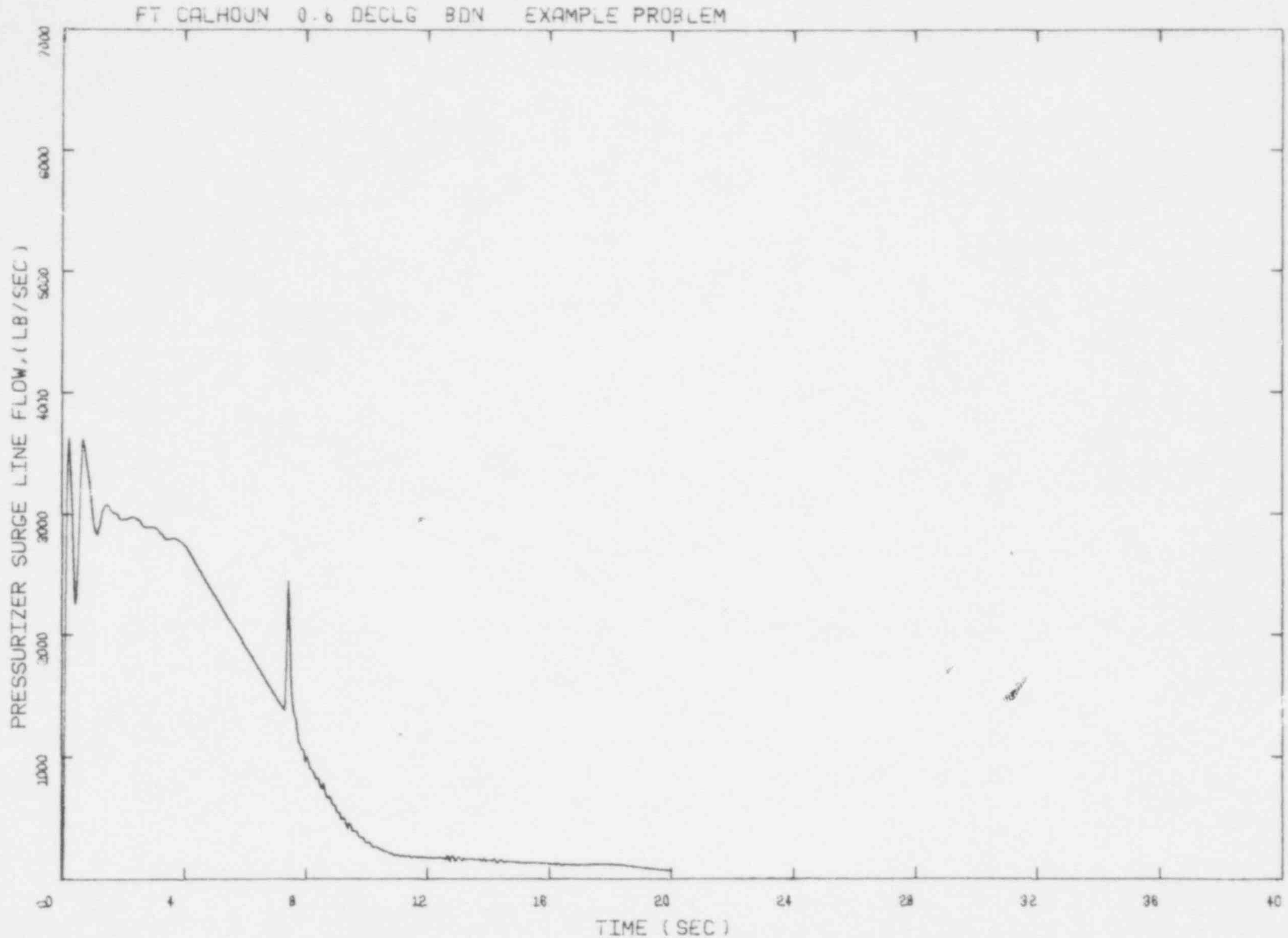

Figure 2.3 Pressurizer Surge Line Flow, DECLG (C_D = 0.6)

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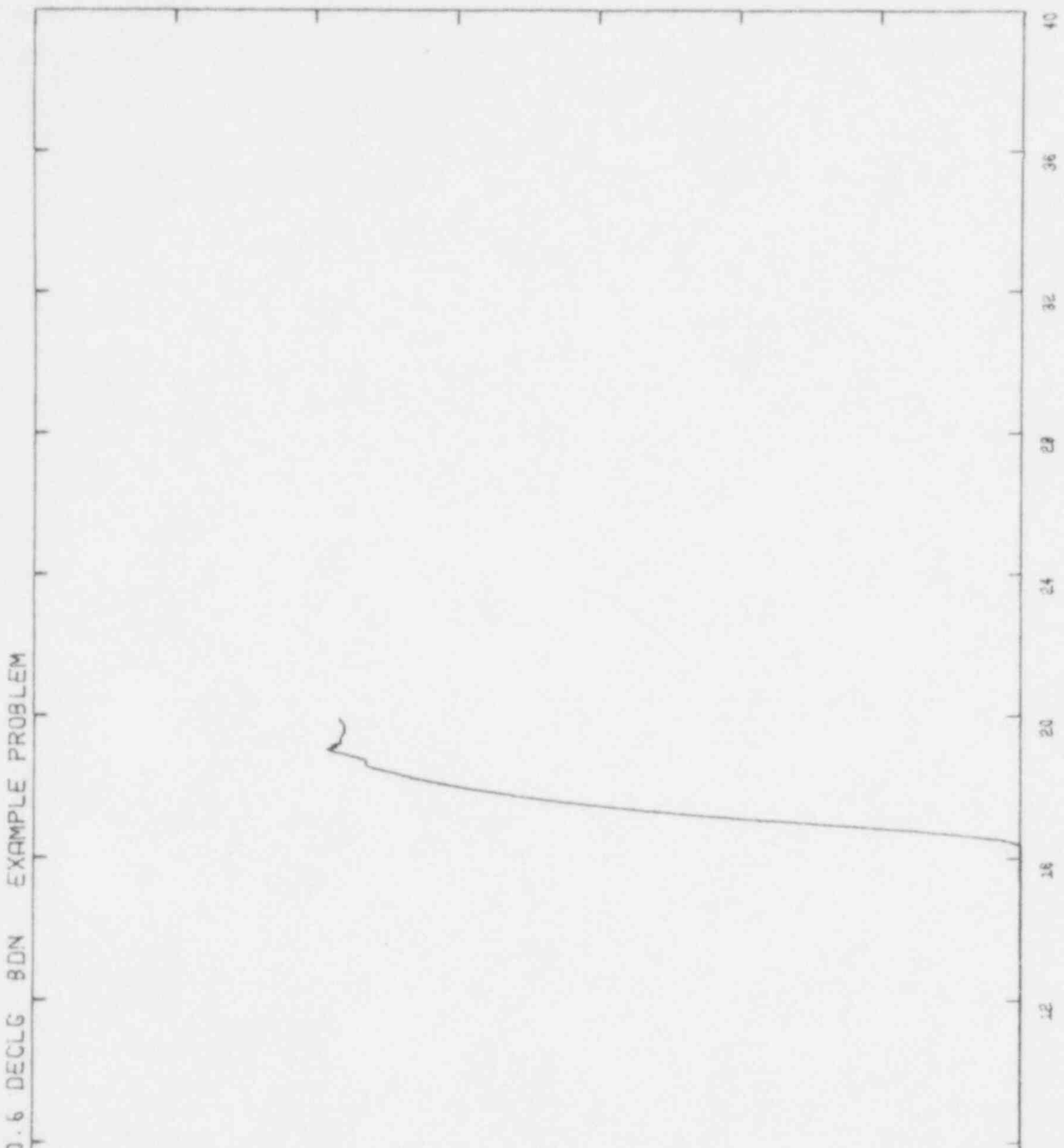

Figure 2.4 Single Intact Loop Accumulator Flow, DECLG (C_D = 0.6)

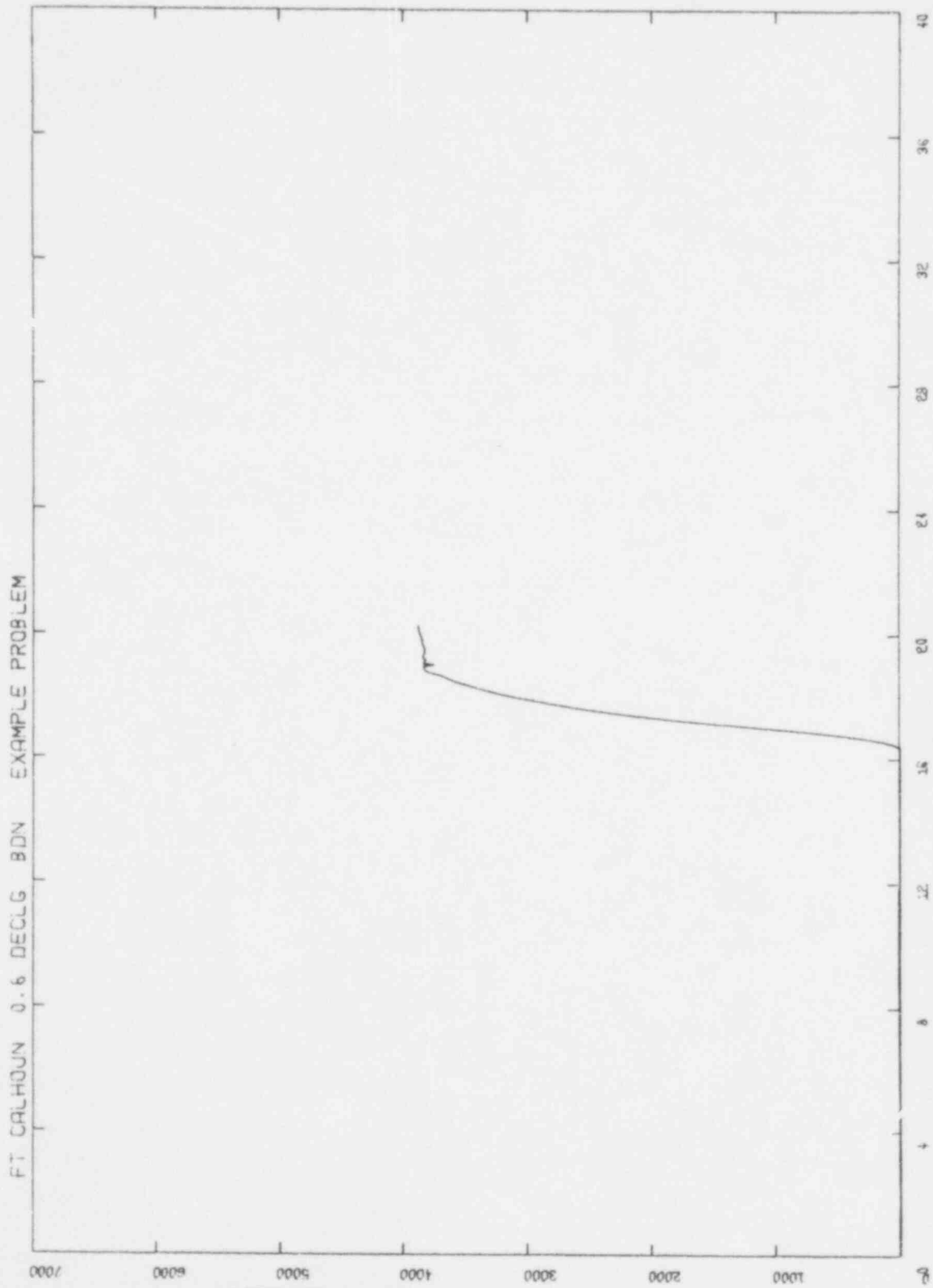

Figure 2.5 Double Intact Loop Accumulator Flow, DECLG (C_D = 0.6)

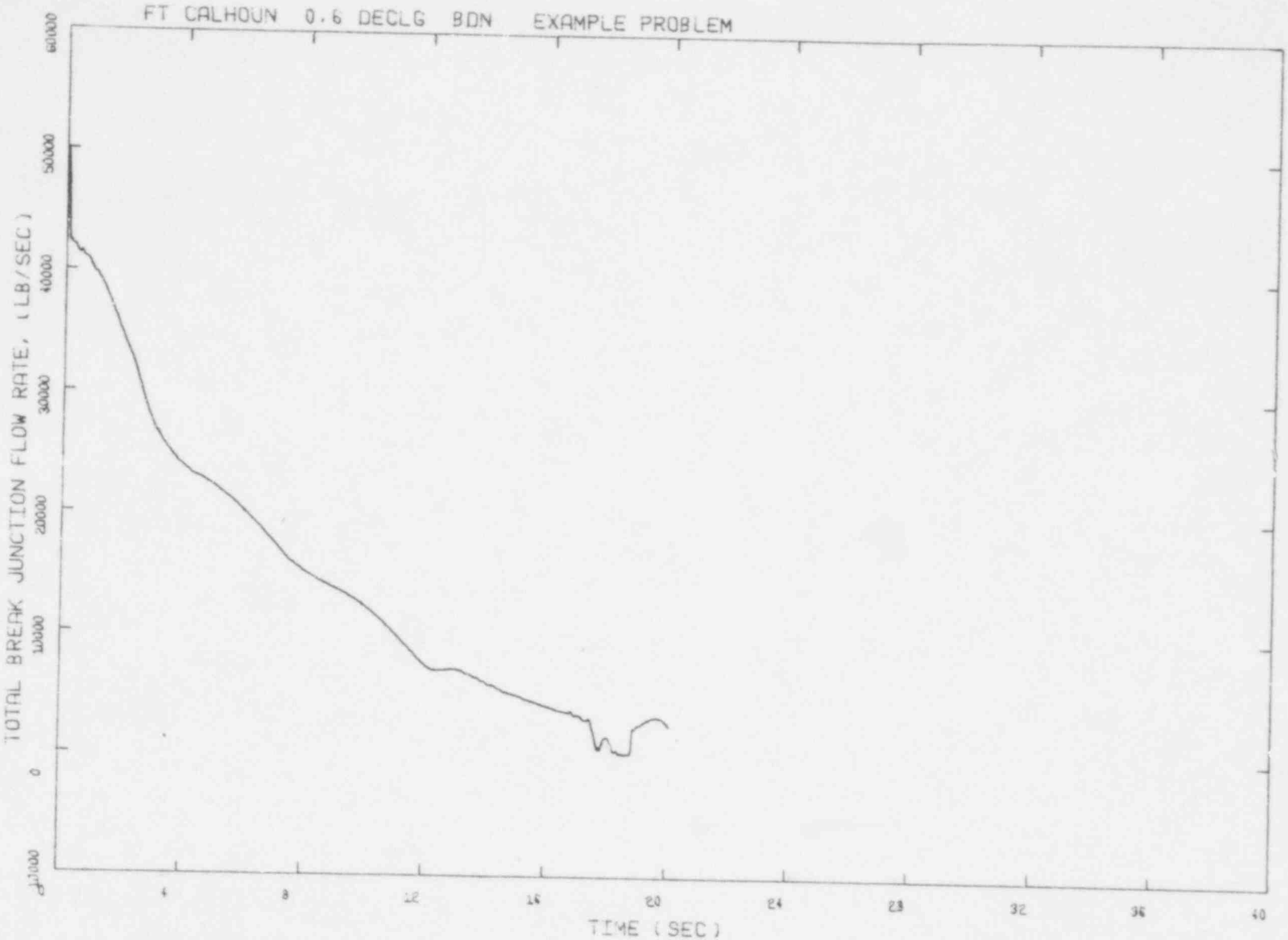

Figure 2.6 Total Break Junction Flow Rate, DECLG (C_D = 0.6)

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3.0 HOT CHANNEL CALCULATION

3.1 MODEL DESCRIPTION

The RELAP4-EM/HOT CHANNEL model is used 1) calculate the maximum power fuel rod heatup occurring during the blowdown phase, and 2) to establish the temperature profile and extent of the metal-water reaction at the EOBY for input into the fuel rod heatup code TOODEE2.

The hot channel model employed was nodalized to be compatible with both RELAP4-EM/BLOWDOWN and TOODEE2. The model contains eight volumes and eleven junctions as depicted in Figure 3.1. Volumes 1 and 8 (lower plenum and upper core) correspond to RELAP4-EM/BLOWDOWN, Volumes 50 and 8, respectively. RELAP4-EM/HOT CHANNEL uses the time dependent volume conditions from RELAP4-EM/BLOWDOWN for these volumes.

The model uses 24 heat slabs. Six of these represent the average core and hot assembly, and the remaining 18 heat slabs are allocated to the hot fuel rod. The hot fuel rod slabs are of varying height, with a concentration of 3-inch slabs around the point where the peak temperatures are expected. The heat slab nodalization of the hot fuel rod is detailed in Figure 3.2. This axial nodalization of the hot fuel rod is identical to the nodalization used in TOODEE2.

A skewed axial power profile was chosen to represent the conditions for LOCA. From this profile the axial power profile input used in the blowdown and hot channel models was developed. This axial profile was applied to both average and hot fuel assemblies. Axial peaking was 1.52 and total peaking was 2.53.

3.2 RESULTS

Results for the node which ultimately becomes the PCT node during the LOCA transient are given in Table 3.1. The PCT clad node temperature history and depth of metal-water reaction are shown in Figures 3.3 and 3.4, respectively. Heat transfer coefficients are presented in Figure 3.5 and the volume average temperature for node 18 is shown in Figure 3.6. The core inlet and outlet flows for the average and hot channels are shown in Figures 3.7 through 3.10.

The PCT node in the hot channel analysis, which is node 18, corresponds to node 13 in the TOODEE2 analysis.

TABLE 3.1

RELAP4-EM/HOT CHANNEL RESULTS FOR NODE 18

Node 18 (Blowdown Conditions)	0.6 DECLG
Blowdown Peak Temperature, ^{o}F	1593
Elevation of Peak, Feet	7.97
Time of Blowdown Peak, Seconds	10.70
Clad Temperature at End-of-Bypass, ^{o}F	1440

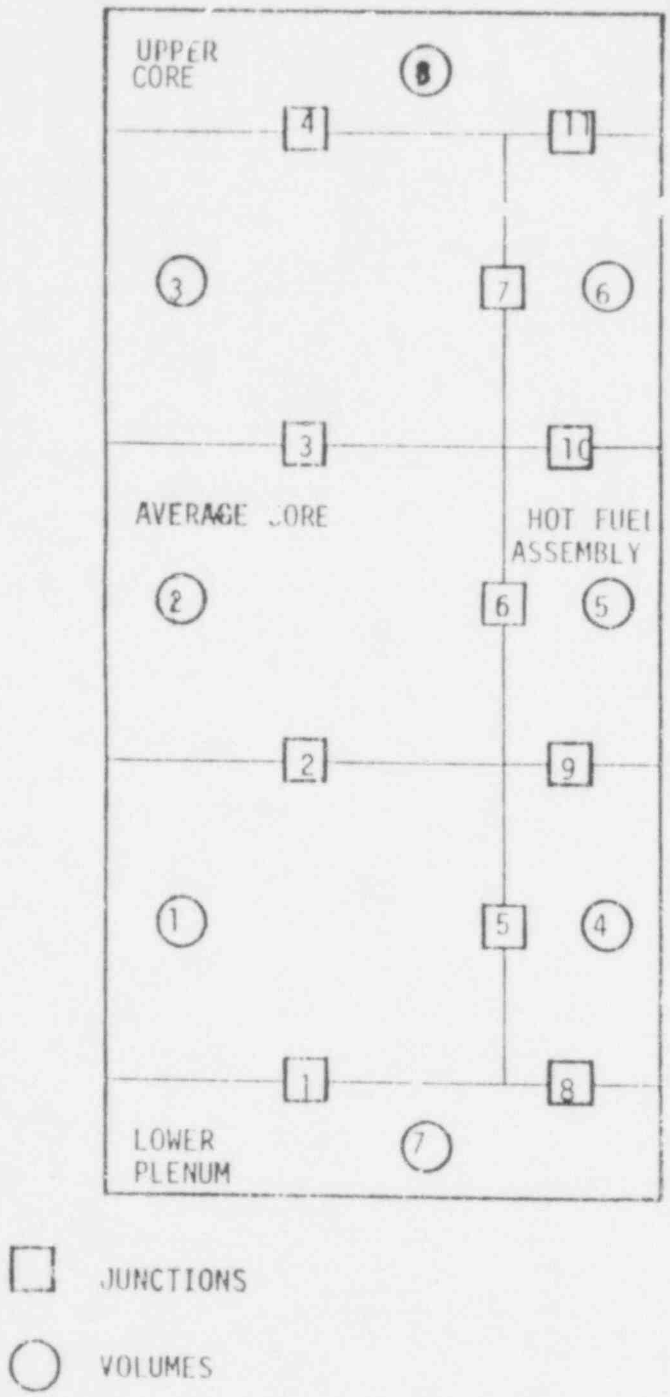

Figure 3.1 RELAP4-EM/Hot Channel Core Volumes and Junctions

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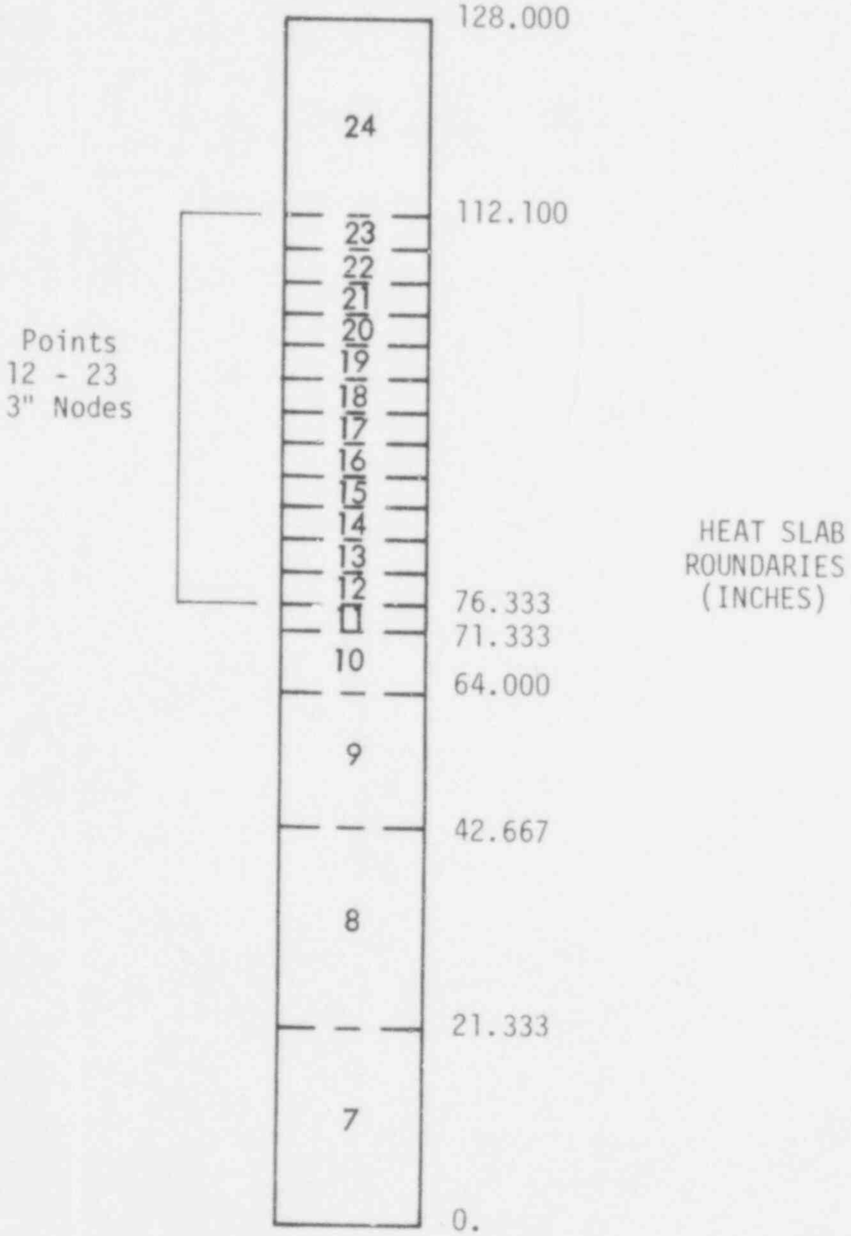

Figure 3.2 Hot Channel Hot Rod Heat Slab Nodalization

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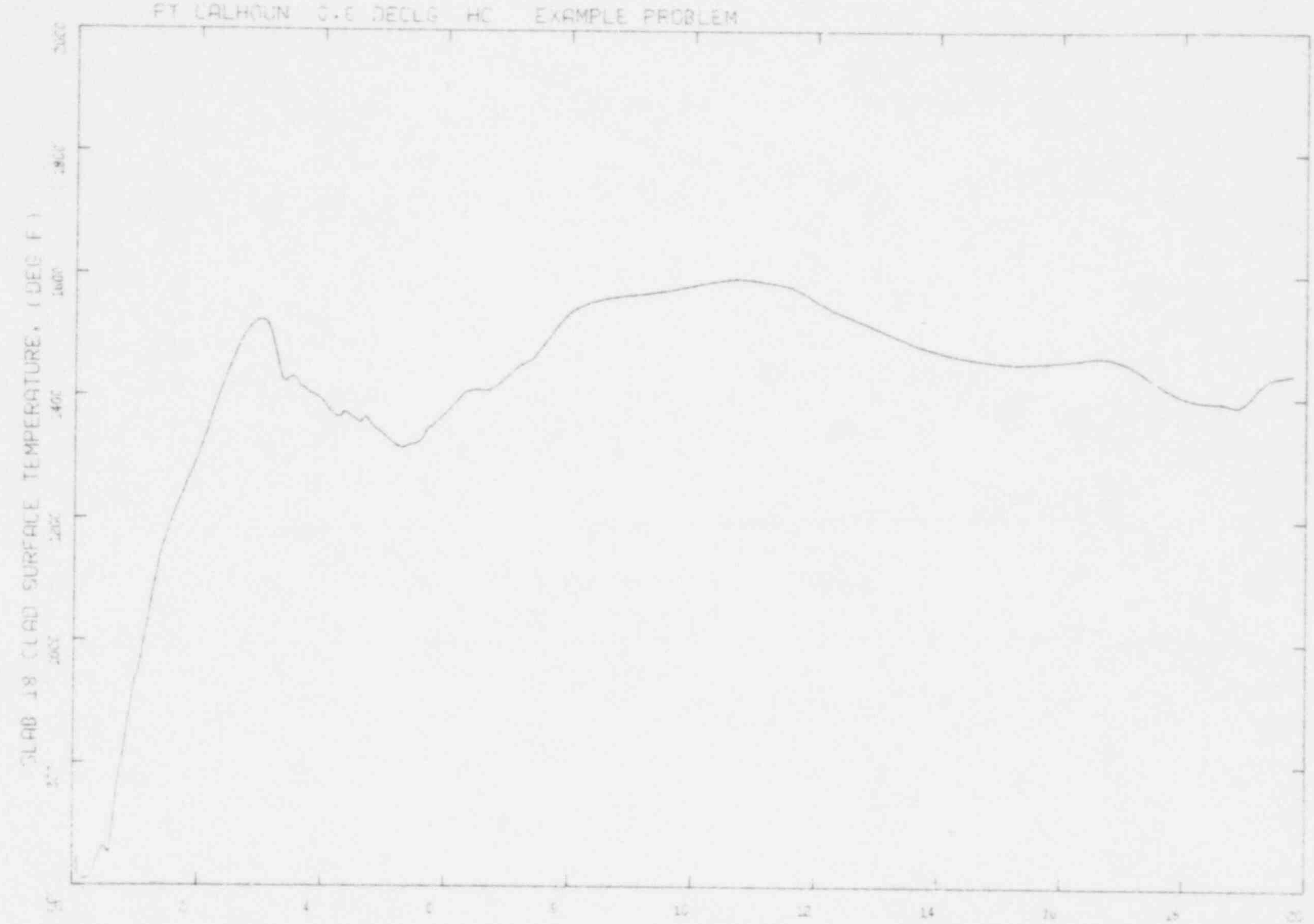

Figure 3.3 Cladding Temperature @ Node 18 During Blowdown. DECLG (C_D = 0.6)

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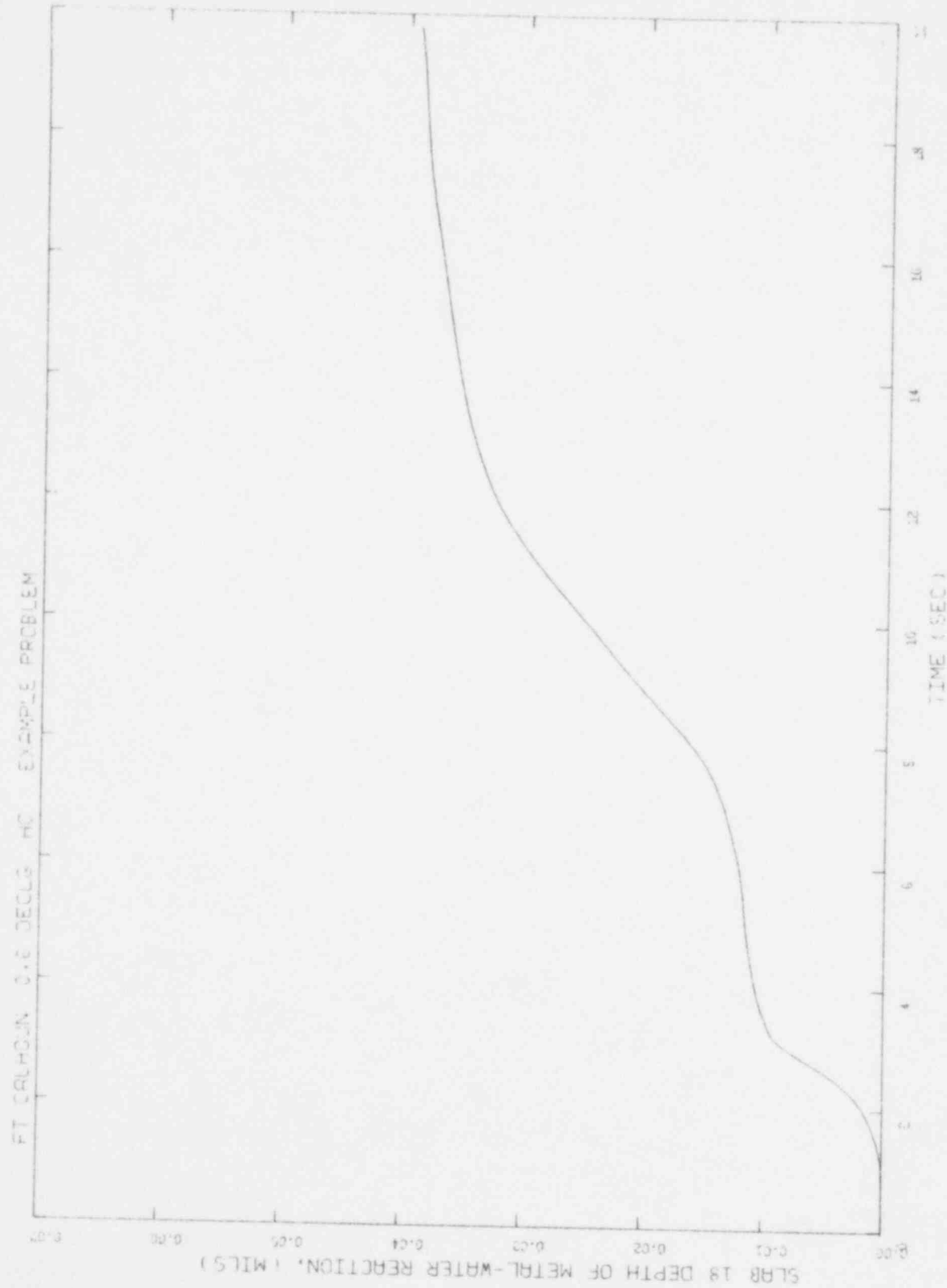

Figure 3.4 Depth of Metal-Water Reaction for Node 18 During Blowdown

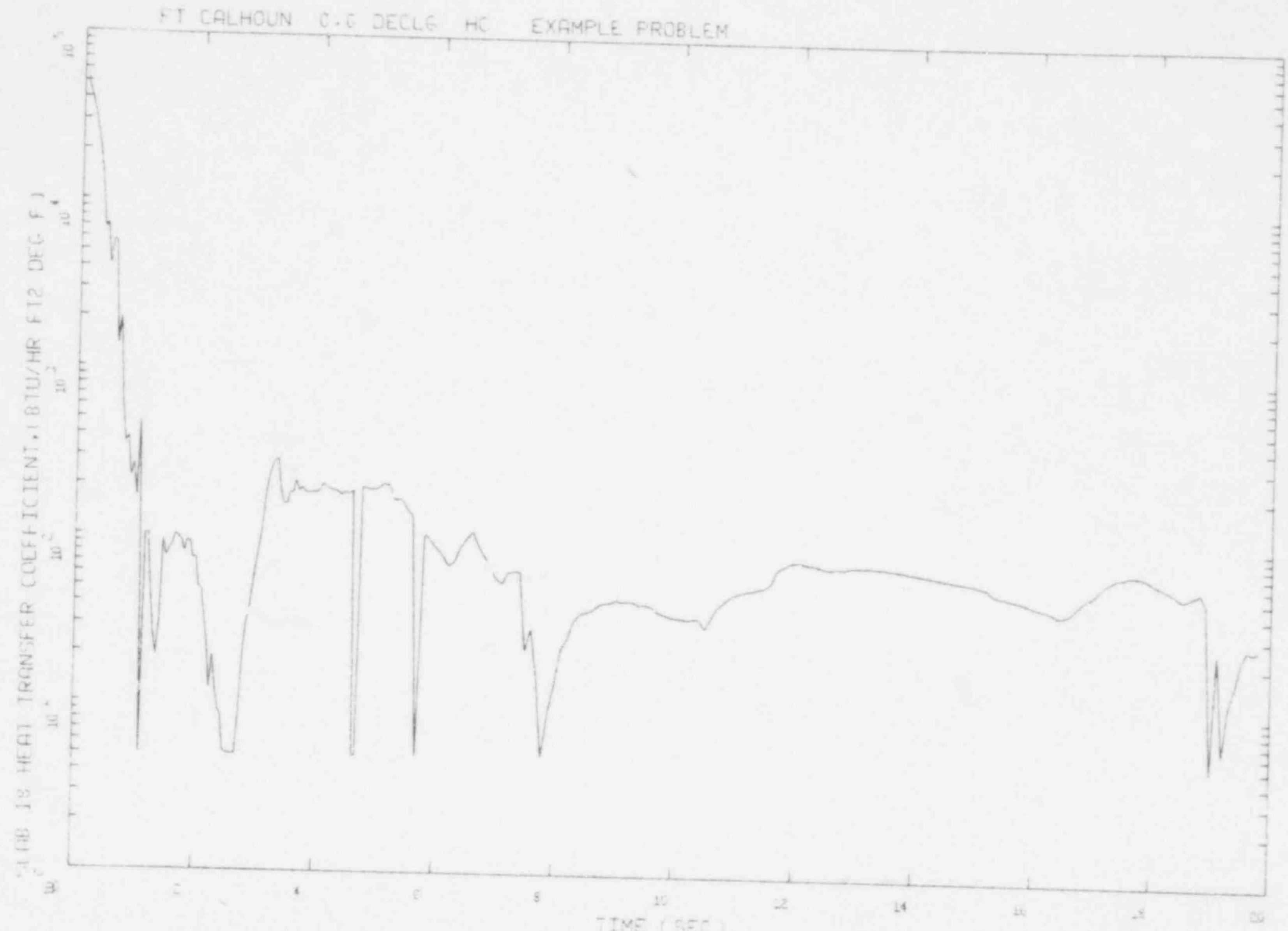

Figure 3.5 Heat Transfer Coefficient for Node 18 During Blowdown

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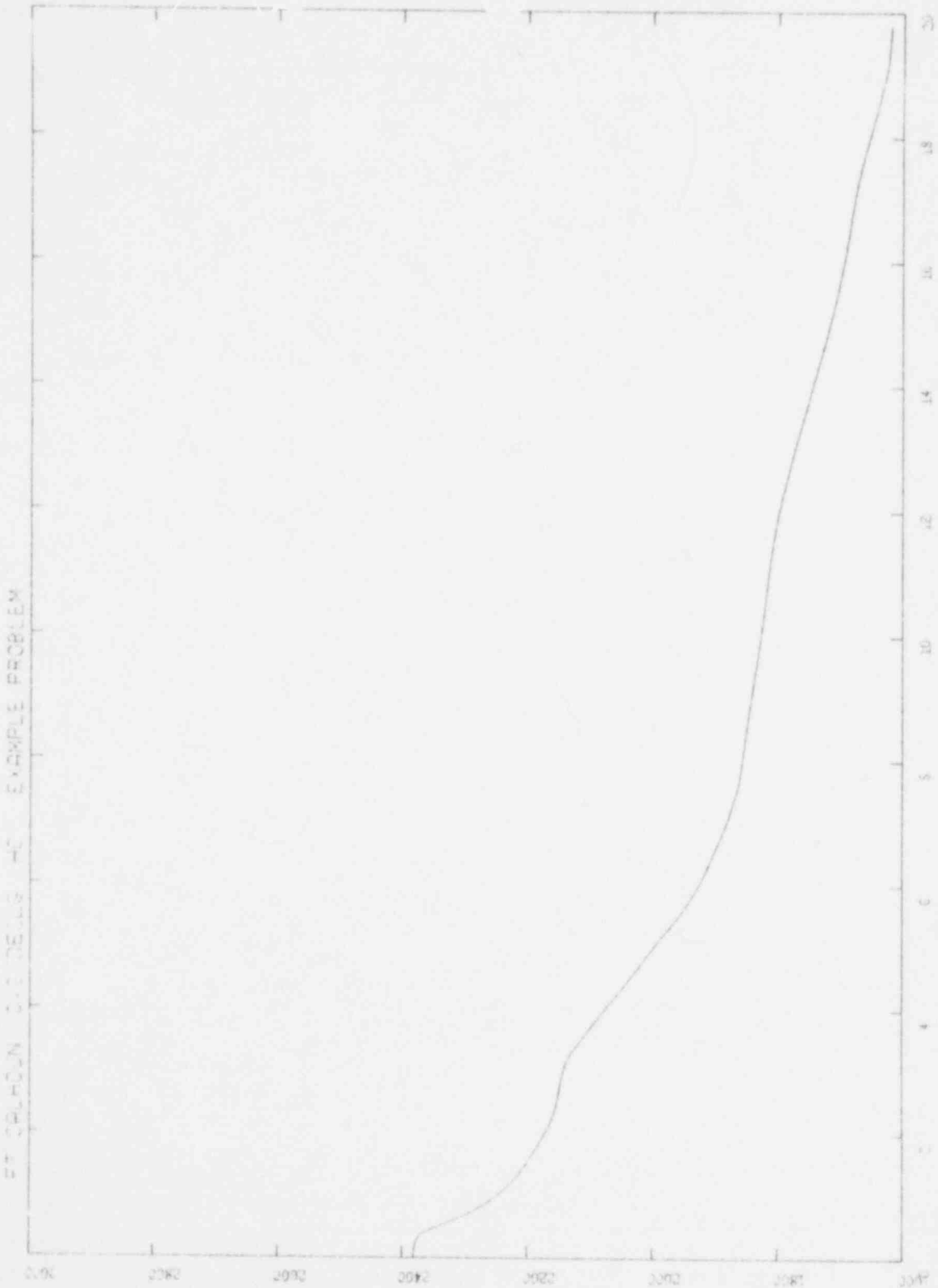

Figure 3.6 Volume Average Temperature at Node 18 During Blowdown, DECLG (C_D = 0.6)

Figure 3.7 Average Channel Inlet Flow, DECLG (C_d = 0.6)

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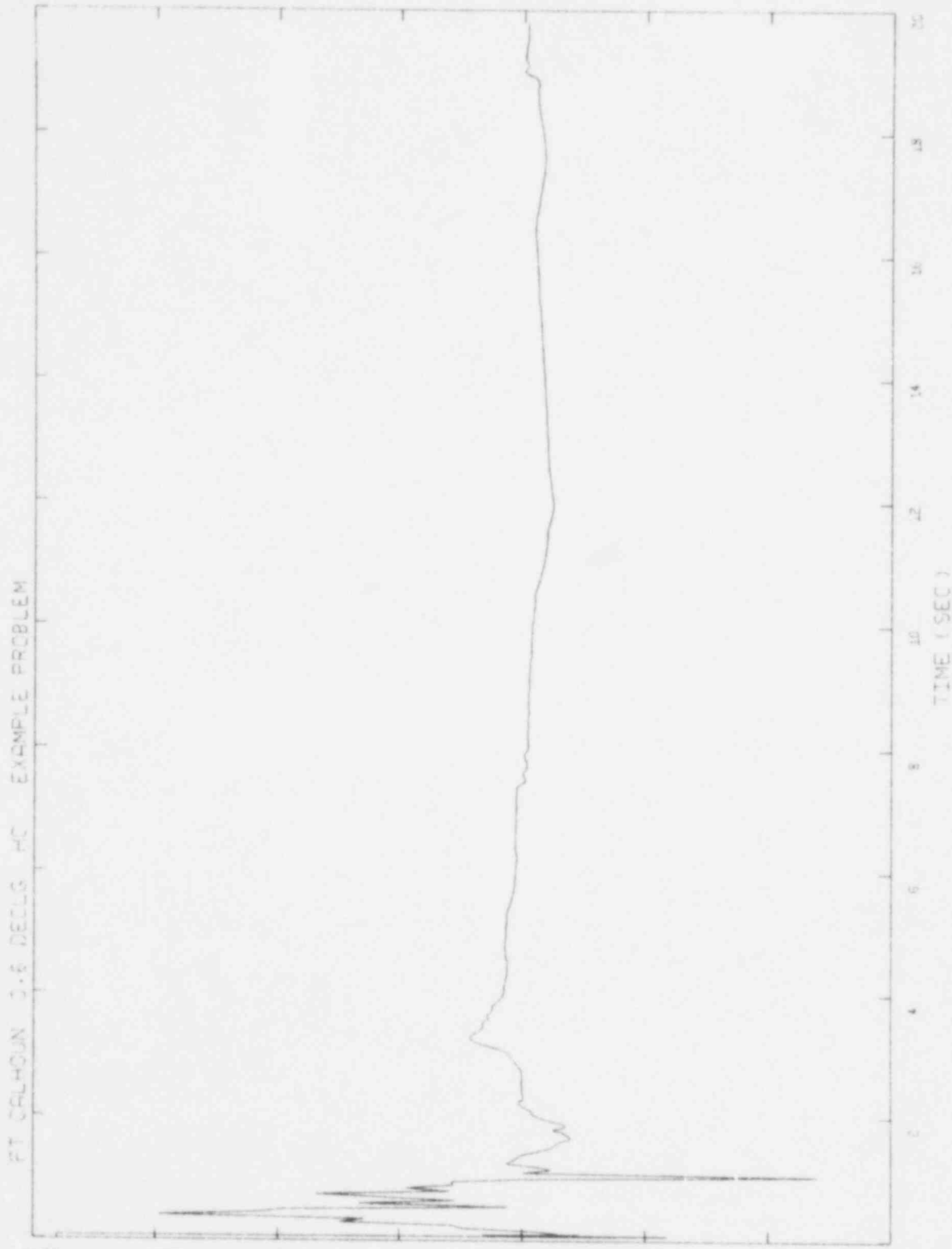

Figure 3.8 Average Channel Outlet Flow, DECLG (C_D = 0.6)

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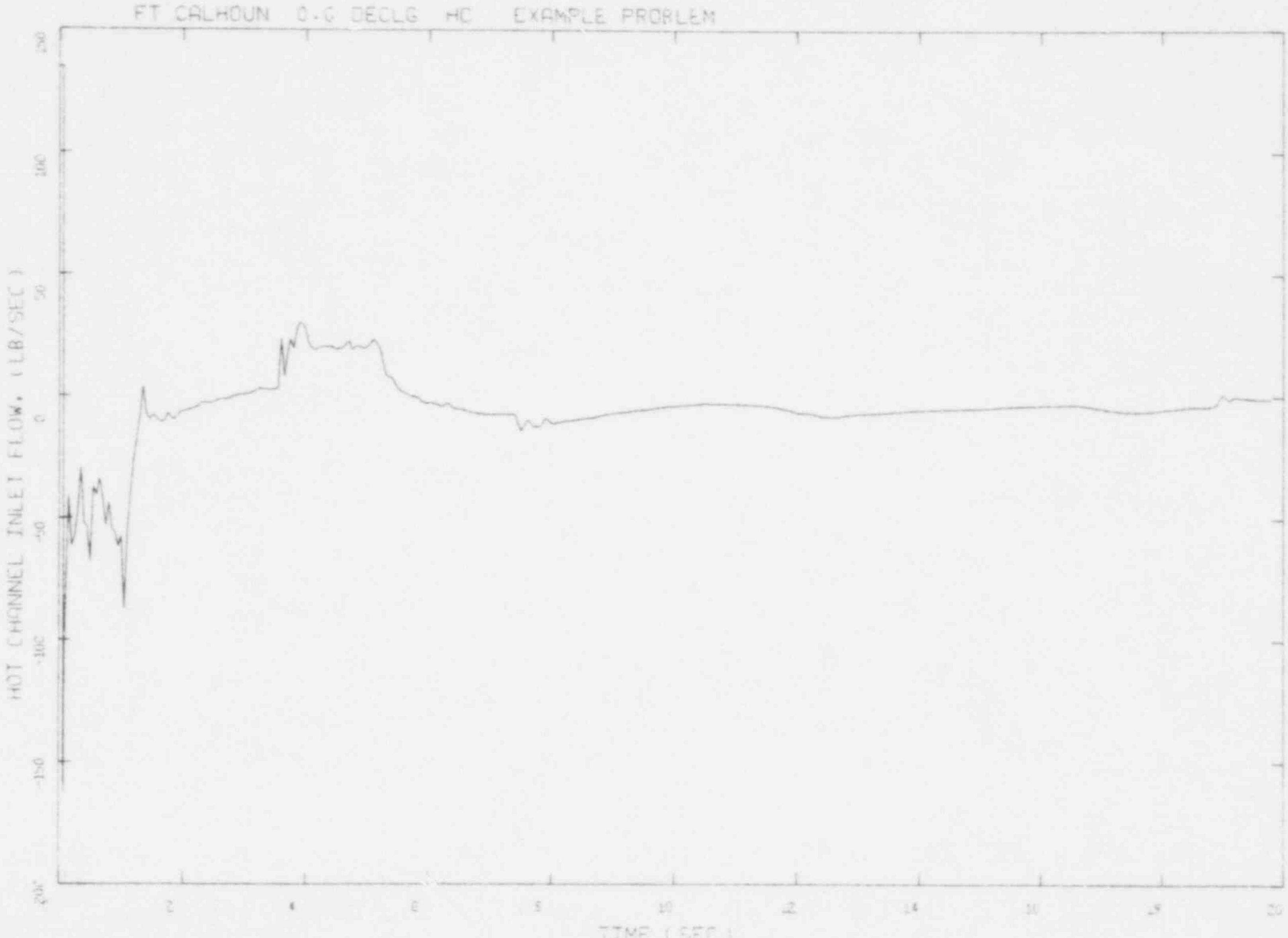

Figure 3.9 Hot Channel Inlet Flow, DECLG (C_D = 0.6)

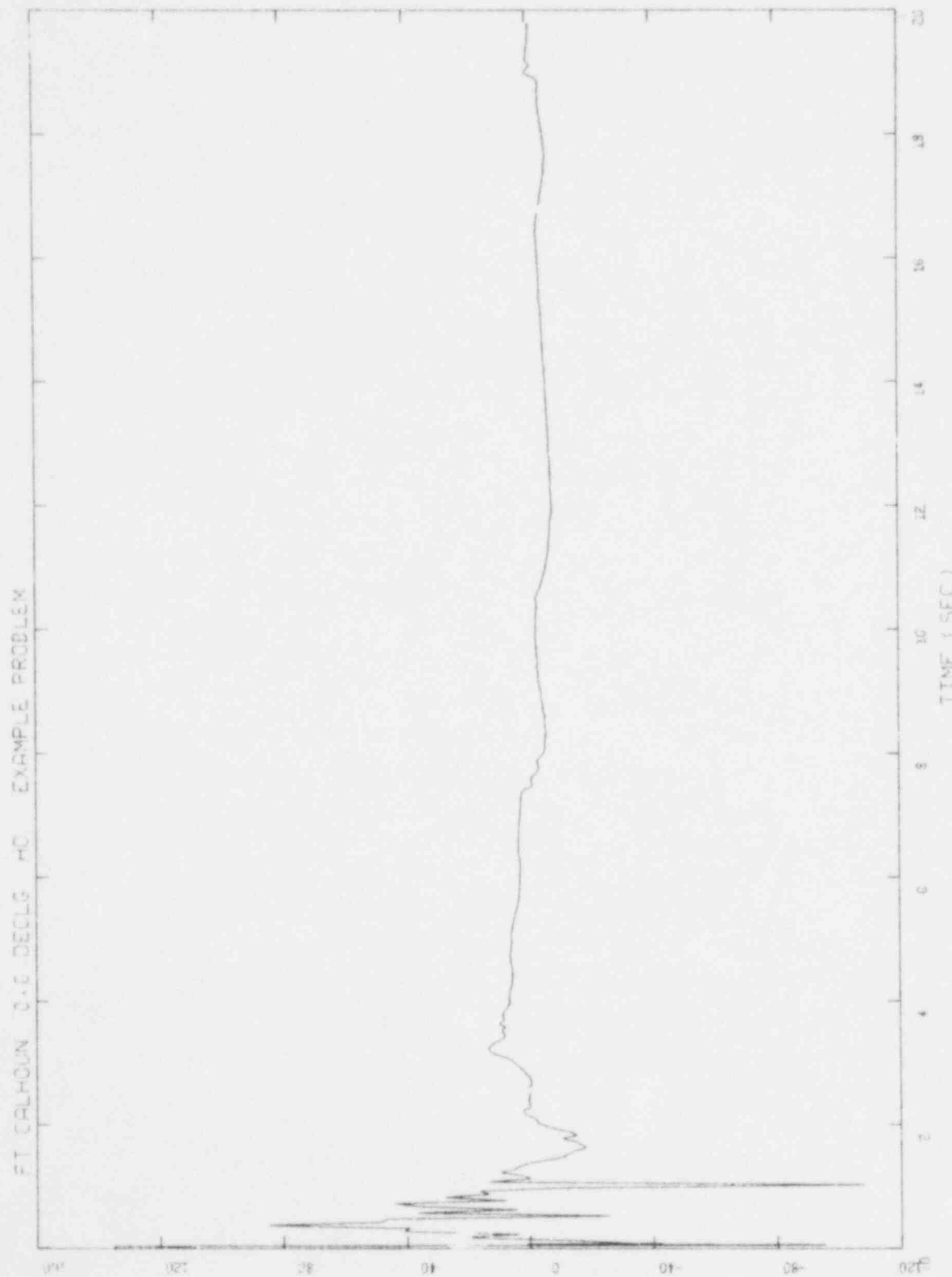

Figure 3.10 Hot Channel Outlet Flow, DECLG (C_D = 0.6)

4.0 REFILL CALCULATIONS

4.1 RELAP4 POWER

The power generated in the core during the refill and reflood portions of the transient is calculated using a one-volume RELAP4 model as described in XN-75-41, Appendix A, Section A6.0.[1] Input parameters include the shutdown reactivity from the RELAP4-EM blowdown calculation (voiding). The long term reactivity is input assuming the core becomes entirely voided. The fission product decay is expected to dominate the power calculation during the refill and reflood portions of the transient, the resulting power calculations include fission, decay energy (ANS + 20%) and actinide contributions.[4] The calculated power versus time for the example problem is shown in Figure 4.1.

4.2 RELAP4 REFILL

As described in XN-75-41, Supplements 5 and 6,[1] a five-volume, five-junction RELAP4 model was set up to determine the rate at which ECCS fluid is injected into the primary system intact recirculation lines during the refill and reflood portions of the transients. The model consists of three accumulators, accumulator lines, and cold leg volumes. High and low pressure safety injection systems are modeled as fill junctions to the cold leg in the RELAP4-EM BLOWDOWN model. Initial conditions in the five volumes are set at (EOBY) conditions. The pressure transients in the cold leg are input as time dependent conditions with the cold leg pressure equal to the containment backpressure. The ECCS fluid therefore flows against the containment backpressure. The flow rates calculated by five-volume RELAP4 program are input to the calculation to determine the time of beginning of reflood.

4.3 BOCREC CALCULATION

Following the EOBY as determined during the RELAP4-EM/BLOWDOWN calculation, downflow is calculated in the downcomer region of the reactor vessel. Emergency Core Cooling (ECC) water injected into the intact loops of the reactor will flow to the lower plenum under the influence of gravitational force. When the water level reaches the Bottom of the Core (BOCREC), the reflood portion of the transient can begin. ECCS flow rates are obtained from RELAP4 refill model.

The time to begin reflood is computed in accordance with ENC's generic PWR model as given in XN-75-41, Supplement 5, Revision 1.[1] The hot wall delay computation is based on results from the CREARE reports TN-188[10] and TN-202.[11] This hot wall delay is detailed in XN-76-27[2] which is the base document for the approved WREM-II model. Output from the BOCREC calculation defines the time to begin reflood and specifies the ECC injection rates to the lower plenum following beginning of reflood. The start of reflood is given by the BOCREC time plus the transport delay.

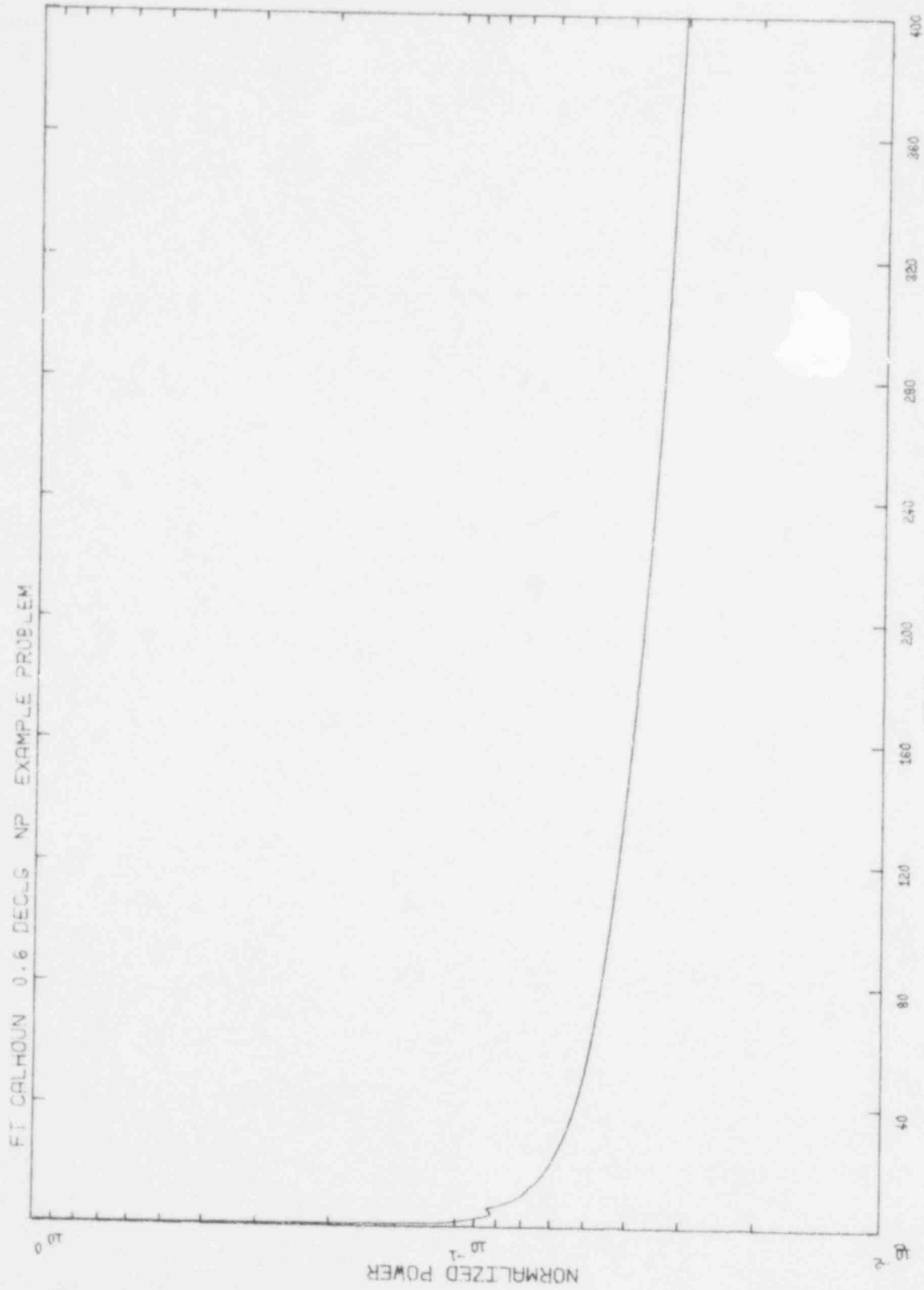

Figure 4.1 Normalized Power Refill and Reflood

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5.0 REFLOOD CALCULATION

The REFLEX[3,8] computer program was used to perform a reflooding analysis. This calculation considers refilling of the reactor vessel and the rate of reflooding of the reactor core. In the model, the primary system coolant pumps were assumed to have locked impellers, the ENC carryover fraction model was used, and effects of compressed gas were conservatively ignored. The reflood model is consistent with the plant data for the Fort Calhoun plant and with the BLOWDOWN, HOT CHANNEL, and TOODEE2 models.

5.1 REFLEX MODEL DESCRIPTION

The RELFEX calculation used the 25 volume, 22 junction model shown in Figure 5.1. Geometrical data for the system were input to the model and checked for consistency with other portions of the WREM-IIA analysis. The REFLEX calculation begins at BOCREC plus the free fall delay time which is 34.17 sec from the beginning of the LOCA analysis. The method of computing the start of reflood time is described in Revision 1 of Supplement 5, XN-75-41.[1] Injection pressure penalties of .4 and .15 psi, respectively, were input for 60° injection angles.[1,3]

Decay power was input to the REFLEX calculation from the results given in Figure 4.1.

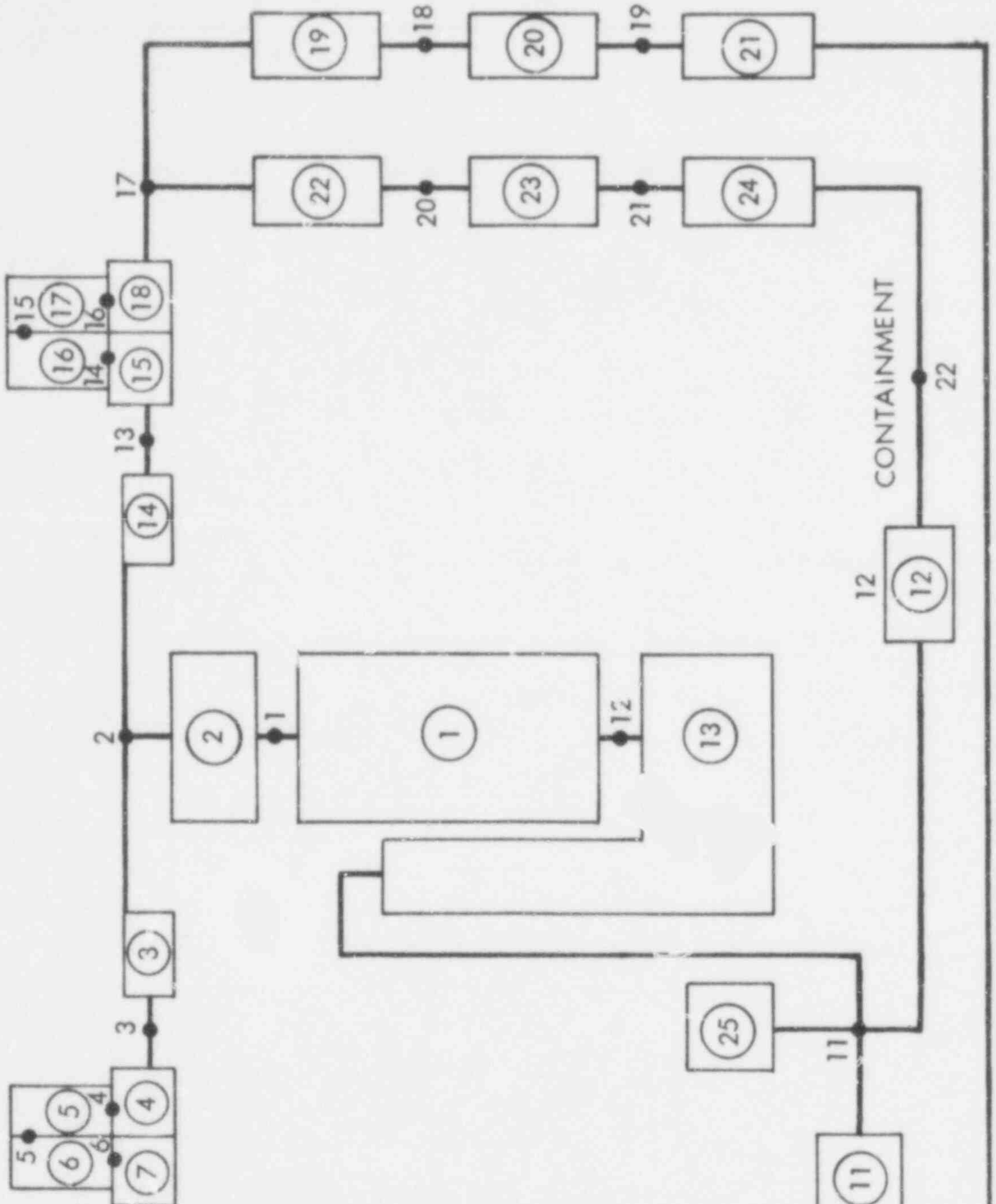
In REFLEX the core was represented by a single volume with a constant enthalpy release from the core of 1298.6 Btu/lbm.

This value of core outlet enthalpy could be achieved only for a 0.7 - 0.8 in/sec flooding rate if all the decay energy were removed as generated and the stored energy were transferred over a 400 second transient. The assumption of a conservative core outlet enthalpy is part of the approved REFLEX WREM IIA model.(3)

5.2 REFLOOD CALCULATION

The purpose of the reflood calculation is to supply the reflooding rate and fluid conditions for the TOODEE2 reflood cladding temperature calculation. The quantities obtained from REFLEX including reflood rates, core mixture level, downcomer mixture level, upper plenum pressure and core inlet flow rate are shown in Figures 5.2 through 5.6 for the 0.6 DECLG case.

Containment pressure during reflood was calculated using CONTEMPT-LT modified to conform to Branch Technical Position Paper CSB 6-1.(12) Containment backpressure calculations are discussed in Section 7.0.



1 - 22 JUNCTIONS

(1)(25) VOLUMES

Figure 5.1 REFLEX Nodalization Diagram for Fort Calhoun PWR

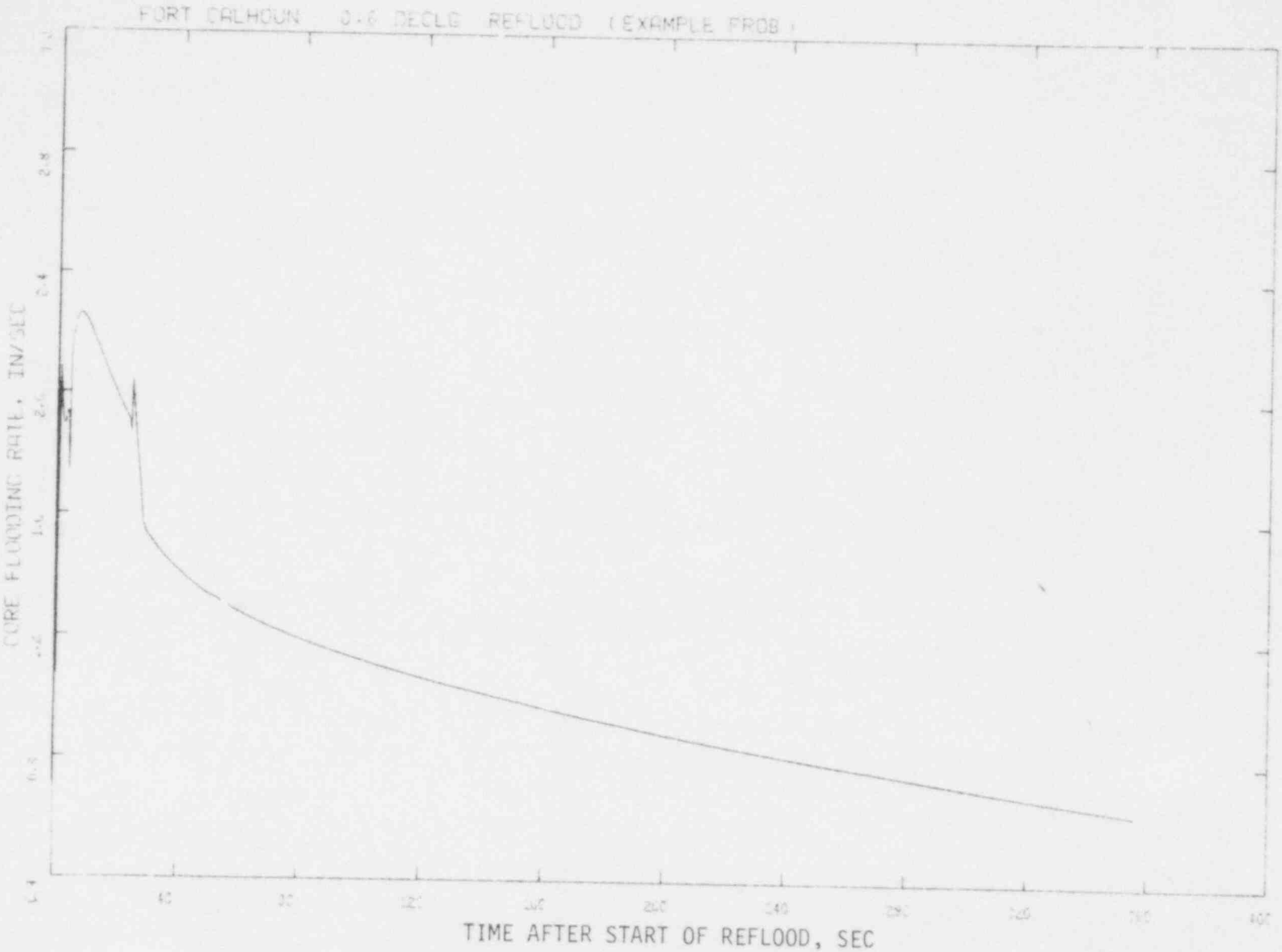


Figure 5.2 Core Flooding Rate, DECLG (C_D = 0.6)

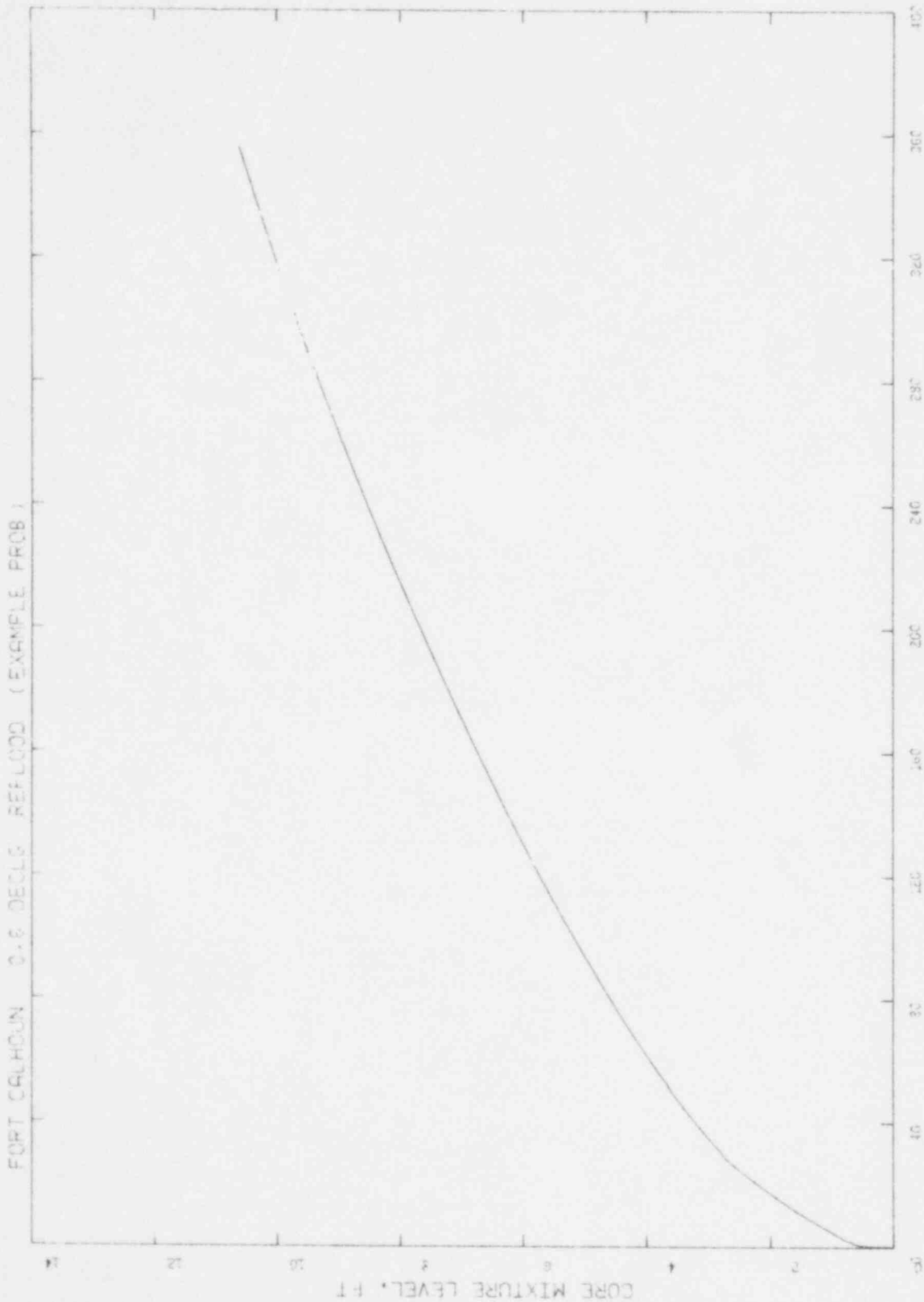


Figure 5.3 Core Mixture Level, DECLG (C_D = 0.6)

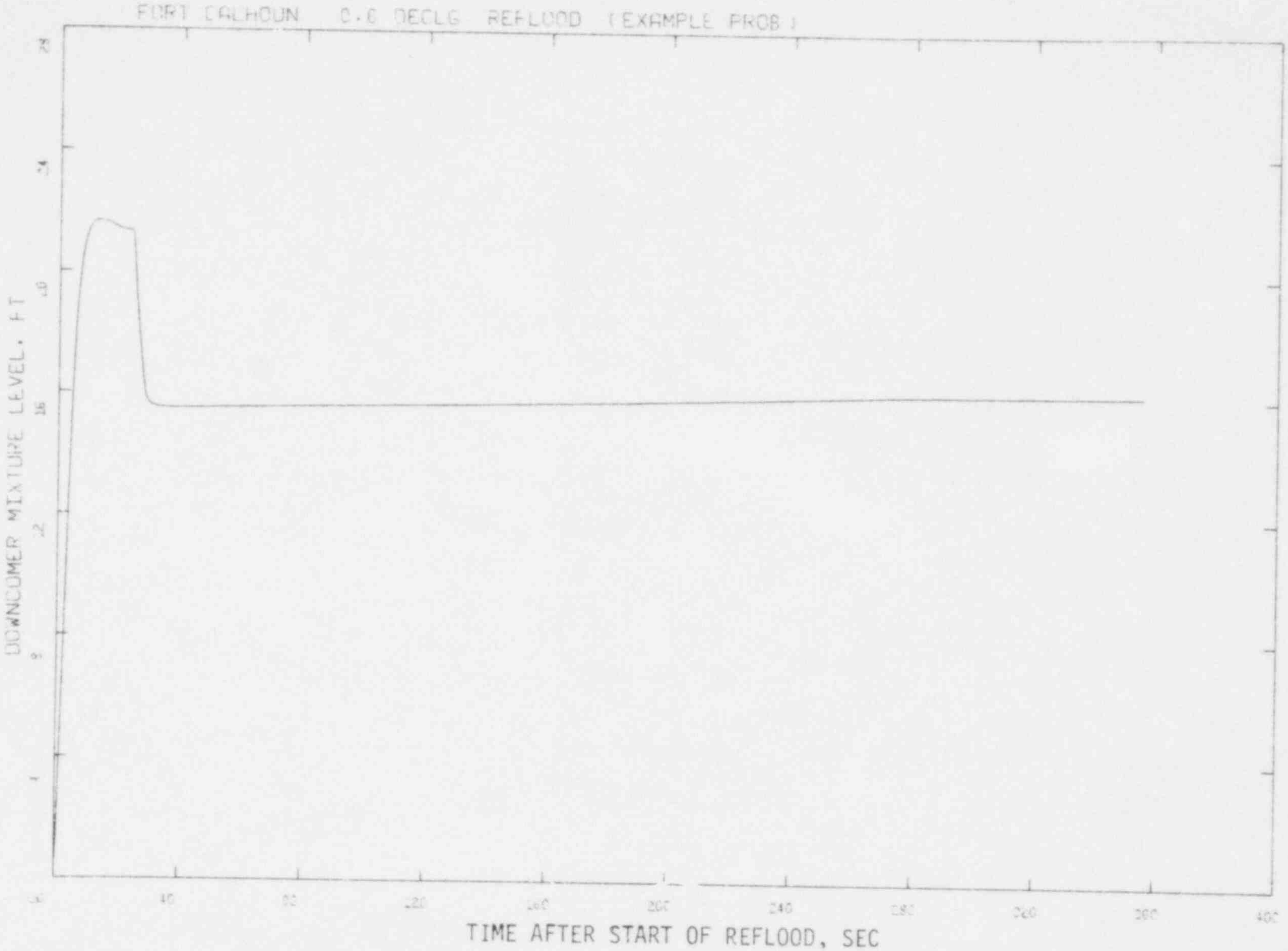


Figure 5.4 Downcomer Mixture Level, DECLG (C_D = 0.6)

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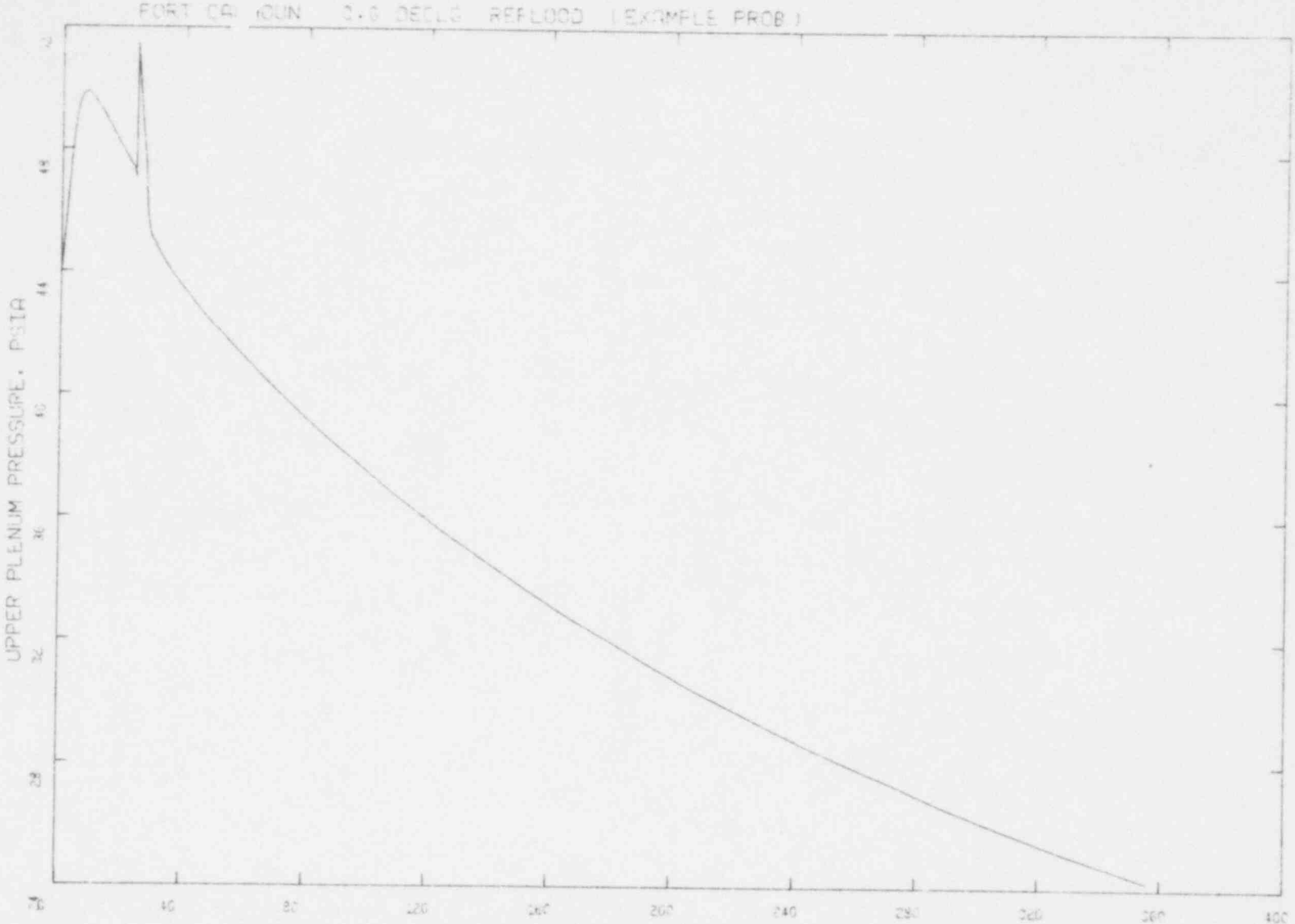

Figure 5.5 Upper Plenum Pressure, DECLG (C_D = 0.6)

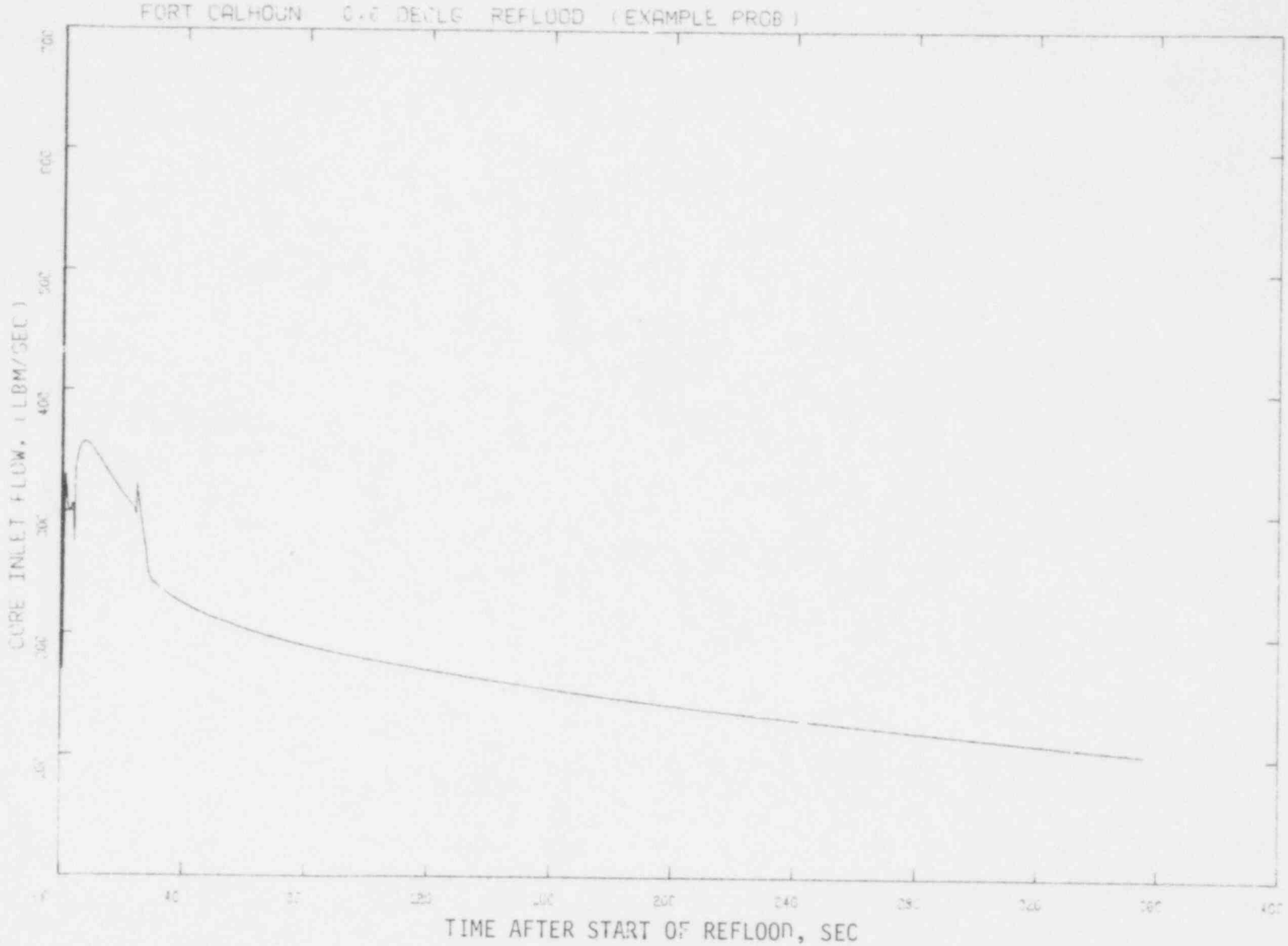

Figure 5.6 Core Inlet Flow, DECLG (C_D = 0.6)

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6.0 HEATUP CALCULATION

6.1 MODEL DESCRIPTION

The time dependent fuel rod thermal analysis program TOODEE2 is used to determine both the PCT and extent of metal-water reaction during the refill and reflood periods of a PWR LOCA. The hot fuel rod from the hot assembly is modeled with a total peaking factor of 2.53. The hot fuel rod is divided into 18 axial nodes as shown in Figure 6.1. As in the hot fuel rod nodalization used in the hot channel analysis, the nodes are of varying heights with the smaller nodes (3-inch) concentrated in the region of the expected maximum temperature. The fuel rod is divided into ten radial nodes, comprised of two cladding nodes, seven equally spaced fuel nodes, and one fuel/gap node. The radial nodalization is shown in Figure 6.2. The axial power distribution corresponds to that used in RELAP4-EM/HOT CHANNEL.

The code requires input from two sources, the initial fuel rod temperature distributions and depths of metal-water reaction from RELAP4-EM/HOT CHANNEL calculated values at the end-of-bypass. The time dependent fluid conditions (flooding rate, inlet enthalpy, etc.) are taken from REFLEX results. During the period from end-of-bypass to beginning of reflood, the ENC radiation model (Section 7.0 in Volume I, XN-75-41)[1] is conservatively neglected. After reflood, heat transfer coefficients are determined using the heat transfer model from the WREM-IIA model.

The heatup portion of the transient has been calculated by the method as reported in XN-75-41[1], XN-76-27[2], and XN-NF-79-18[9]. The results of the calculation with this model are shown in Figure 6.3.

A minor change was made to TOODEE2 to make the mathematical modeling within TOODEE2 internally consistent. This change resulted in less than a 1°F change in PCT and is discussed in detail in Section 8.1

The peak cladding temperature occurs at the 7.97 foot elevation. The temperature at this location turns around at 166.6 seconds and continues to decrease throughout the remainder of the transient.

6.2 RESULTS

Peak clad temperatures and corresponding times are presented in Tables 1.1 and 1.2. Also included in this table are clad rupture time, peak clad temperature, and peak linear heat generation rate at initiation of reflood.

The temperature history for the node of peak cladding temperature from the end-of-bypass through temperature turn-around is plotted in Figure 6.3.

The PCT Node in the TOODEE2 analysis, which is node 13, corresponds to node 18 in the hot channel analysis.

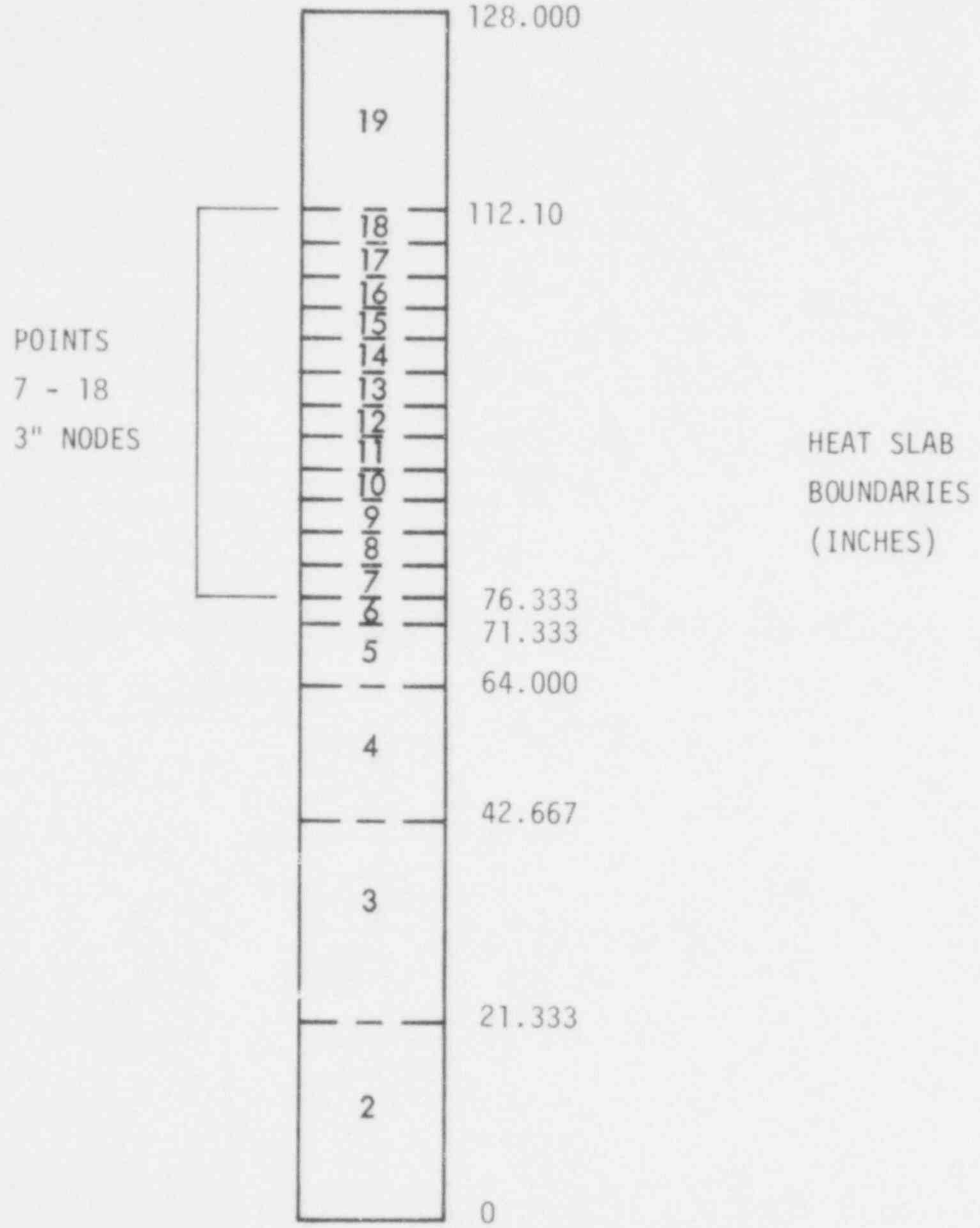

Figure 6.1 TOODEE2 Hot Rod Heat Slab Nodalization

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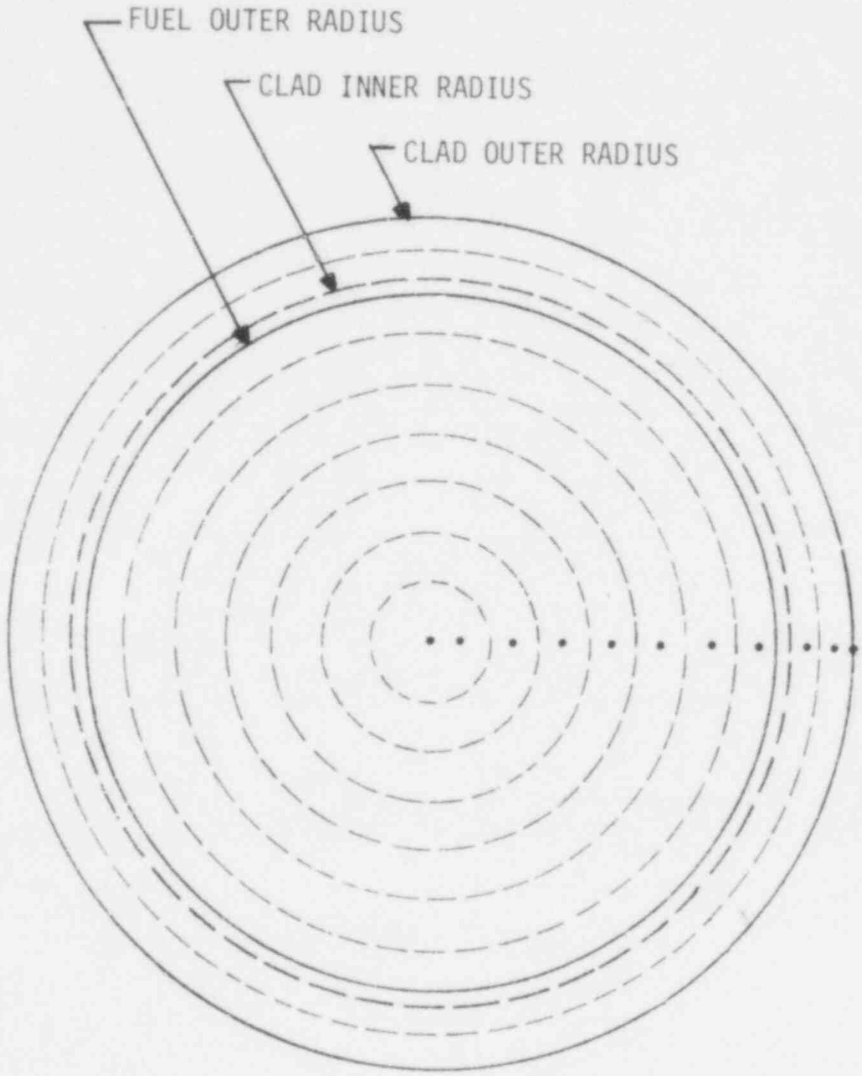

10 Grid line Locations

11 Point Locations

Figure 6.2 TOODF-2 Radial Point and Boundary Assignments for Fort Calhoun PWR Hot Channel Analysis

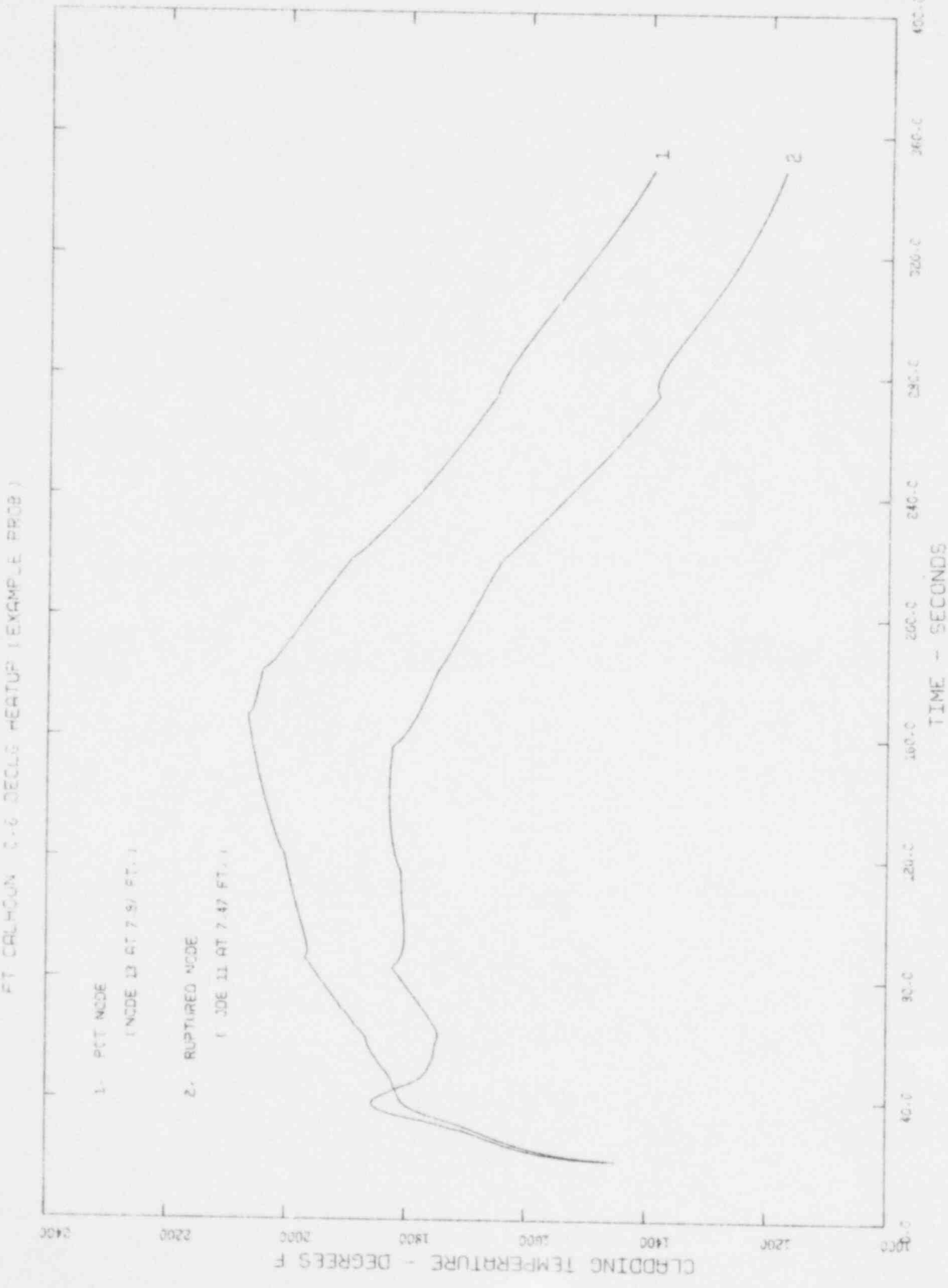

Figure 6.3 Cladding Temperature Versus Time, DECLG (C_D = 0.6)

7.0 CONTAINMENT BACKPRESSURE CALCULATION

The containment backpressure for the reflood period of the postulated LOCA was evaluated in accordance with the discussion presented in XN-75-41, Supplement 5, Section 4.6. A containment analysis was performed using the computer code CONTEMPT-LT, Version 22 modified as described in Supplement 5, Revision 1, of XN-75-41(1) and XN-NF-78-46(13).

The containment analysis considered the equivalent double-ended cold leg guillotine break using the mass and energy release from the RELAP4-EM blowdown analyses. Table 7.1 summarizes some of the pertinent input data such as containment volume, initial pressure and temperature, heat sink dimensions and properties, and capacity and initiation times for safety features.

The condensing heat transfer coefficient is modeled in accordance with Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."(12) Twelve passive heat sinks were modeled.

The mass and energy from the blowdown analysis is input through the end-of-blowdown, then assumed zero for the remainder of the transient. This is conservative as it neglects energy released during reflood to containment which would result in higher containment pressure.

The predicted containment pressure history for the reflood period is shown in Figure 7.1.

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Table 7.1

Dry Containment Data

Containment Physical and Thermal Parameters

Net Free Volume	$1.05 \times 10^6 ft^3$
Outside Air Temperature	-17°F
Initiation Time for:	
Spray Flow	55.0 sec
Fan Coolers	25.0 sec
Containment Initial Conditions:	
Temperature	85°F
Pressure	14.7 psia
Relative Humidity	80%
Containment Spray Water:	
Temperature	40°F
Flow Rate (Total, 3 pumps)	5100 gpm

Fan Air Cooler Capacity
(total 4 coolers)

Vapor Temperature (°F)	Capacity (Btu/hr)
150	$.50 \times 10^8$
185	1.38×10^8
244	3.00×10^8
288	4.37×10^8
320	5.40×10^8

Thermal Conductivity and Volumetric Heat Capacity

Materials	Thermal Conductivity (Btu/hr-ft-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)
Steel	26.0	59.0
Structural Concrete	0.85	32.0
Paint for Steel Surfaces	1.5	57.6
Paint for Concrete Surfaces	0.3	43.2

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Table 7.1 (Continued)

Dry Containment Data

Containment Passive Heat Sinks

DESCRIPTION	MATERIAL	THICKNESS	SURFACE AREA FT^2
1. Containment Cylindrical Wall	paint	3 mil	44,090
	steel	.25 in	
	concrete	3.875 in	
2. Containment Dome	paint	3 mil	6,850
	steel	.25 in	
	concrete	3 ft	
3. Foundation Slab	paint	3 mil	8,650
	steel	.25 in	
	concrete	15 ft	
4. Refueling Cavity Walls	stainless steel	0.06 in	14,160*
	concrete	4 ft	
5. Refueling Cavity Floor	stainless steel	0.06 ft	400
	concrete	16.5 ft	
6. Misc. Concrete	paint	6 mil	8,000*
	concrete	.75 ft	
	paint	6 mil	
7. Misc. Concrete	paint	6 mil	11,150*
	concrete	1 ft	
	paint	6 mil	
8. Misc. Concrete	paint	6 mil	53,600*
	concrete	2.0 ft	
	paint	6 mil	
9. Misc. Concrete	paint	6 mil	9,260*
	concrete	5 ft	
	paint	6 mil	
10. Misc. Steel	paint	3 mil	10,960*
	steel	1.0 in	
	paint	3 mil	

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Table 7.1 (Continued)

Dry Containment Data

Containment Passive Heat Sinks

DESCRIPTION	MATERIAL	THICKNESS	SURFACE AREA FT^2
11. Misc. Steel	paint steel paint	3 mil 0.25 in 3 mil	5,700
12. Ventilation Ducts	galvanized steel	0.125 in	72,000*

* Tabulated surface area includes areas of both sides of the slab.

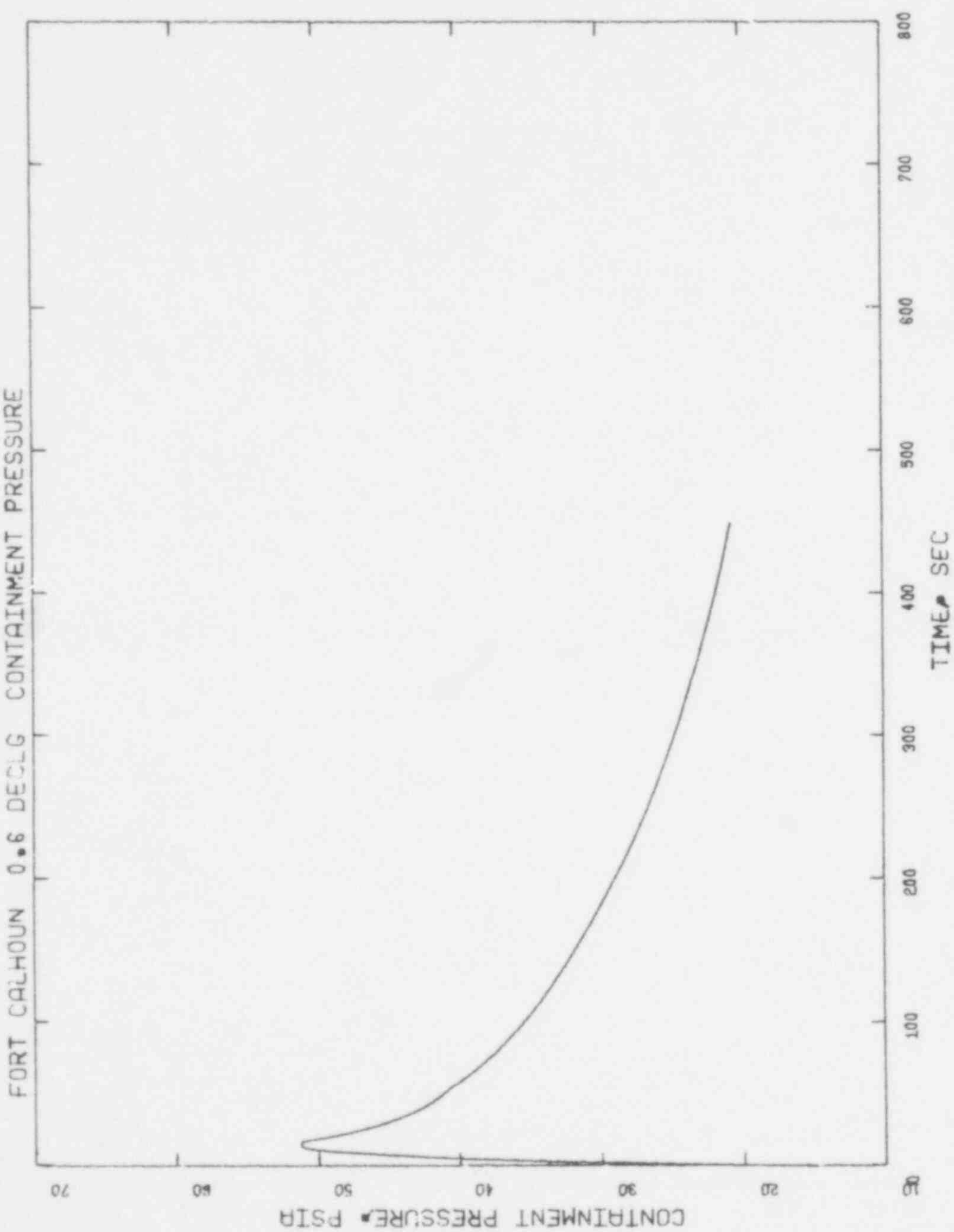

Figure 7.1 Containment Backpressure for DECLG (C_D = 0.6)

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8.0 MODEL HISTORY

The following section presents the genealogy of the RELAP4-EM and TOODEE2 models used in the subject analysis.

8.1 RELAP4-EM GENEALOGY

RELAP-EM/ENC28B - The code changes to RELAP4-EM/ENC26A to produce RELAP4-EM/ENC28B are described in the attachment to a letter to D. F. Ross from G. F. Owsley dated October 1978. These code changes have since been approved as noted in Reference 7.

RELAP4/EM/ENC28C - Three plot variables were added to RELAP4-EM/ENC28B to permit plotting of fuel related heat slab internal temperatures, i.e., pellet surface temperature, clad inside surface temperature, etc. A causal heat slab variable (time step control) was initialized to permit execution of a RELAP4 case without a core and with zero heat slabs. These changes do not affect calculated results.

RELAP4-EM/ENC28D - A change was made to output tape edit subroutine to allow output tape (TAPE4) re-editing. This change does not affect calculated results.

RELAP4-EM/ENC28E - The environmental package was modified to allow input data to be entered on the data cards in column 80. This change does not affect calculated results.

RELAP4-EM/ENC28F - The maximum number of words allowed on the input data cards was increased to permit larger problems. This change does not affect calculated results.

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8.2 TOODEE2 MODEL UPDATE (TOODEE2/MAY79)

The TOODEE2 model was updated in Reference 9 to include the updates recommended by the Swedish memorandum(14). Further review of one of these Swedish updates indicated that this update was inconsistent with the mathematical model in TOODEE2. This update has been removed from the TOODEE2 version of reference 9 to create the new TOODEE2 version labeled TOODEE2/MAY79. This update change was found to make an insignificant change in PCT (<1F°) and is discussed below. The governing differential form of the coolant energy equation solved in TOODEE2 is:

$$\frac{\partial T}{\partial t} = \frac{q/s}{\rho C(Vol)} - \dot{U}\frac{\partial T}{\partial \delta} \qquad (1)$$

where T = temperature

q/s = heat flow

ρ = fluid density

C = fluid heat capacity

Vol = fluid volume

$\dot{U}$ = volumetric flow rate

δ = space variable in direction of flow

The Peaceman-Rockford type of numerical method utilized in TOODEE2 uses two difference equations for each time step each of which is applicable for one-half of the interval. To assure stability, the Swedish(14) recommended that the following implicit backward difference algorithm be used in the second half of the time step (n+1/2 to n+1):

$$(T^{n+1}_{i,j-1} - T^{n+1/2}_{i,j-1}) = -Q_{i,j}\,(T^{n+1}_{i,j} - T^{n+1}_{i-1,j}) - U_{i,j}\,(T^{n+1}_{i,j} - T^{n+1}_{i,j-1}) \qquad (2)$$

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Solving for $T^{n+1}_{i,j}$ gives:

$$T^{n+1}_{i,j}\left(1 + \frac{Q_{i,j}}{U_{i,j}}\right) = T^{n+1/2}_{i,j-1} + \left(1 - \frac{1}{U_{i,j}}\right)\left(T^{n+1}_{i,j-1} - T^{n+1/2}_{i,j-1}\right) + \frac{Q_{i,j}}{U_{i,j}} T^{n+1}_{i-1,j} \quad (3)$$

where the dimensionless velocity ($U_{i,j}$) is the same in all terms as defined by:

$$U_{(i,j)} = \frac{V\ (\Delta t)}{A(J)-A(J-1)} \quad (4)$$

where V is the velocity of the fluid entering node j and (A(J)-A(J-1)) is the length between node j and j-1 centers.

The Swedish recommended (apparently based on intuition) that the U in the Q/U terms be based the fluid node velocity and length of the heat transfer node as shown below

$$U(i,j)_{Swedish} = \frac{V_n\ (\Delta t)}{[AX(J)-AX(J-1)]} \quad (5)$$

where V_n is the fluid in the center of node j and (AX(J)-AX(J-1)) is the length of node j.

It can be seen from the above derivation of algorithm (3) that the original dimensionless velocity (Equation (4)) is mathematically consistent with algorithm (3) and should be used. The only change in TOODEE2 from that described in Reference 9 to make version TOODEE2/MAY79 is that the expression for $U_{(i,j)}$ was changed back to Equation (4).

9.0 REFERENCES

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    d. Supplement 1, August 1975
    e. Supplement 2, August 1975
    f. Supplement 3, August 1975
    g. Supplement 4, August 1975
    h. Supplement 5, Revision 5, October 1975
    i. Supplement 6, October 1975
    j. Supplement 7, November 1975

2. Exxon Nuclear Company, Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II, XN-76-27:

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    b. Supplement 1, September 1976
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10. Block, J. A., and Wallis, G. B., Effect of Hot Walls on Flow in a Simulated PWR Downcomer During a LOCA, CREARE-TN-188, May 1974.

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12. U.S. Nuclear Regulatory Commission, Minimum Containment Pressure Model for PWR ECCS Performance Evaluation, Branch Technical Position CSB 601.

13. Exxon Nuclear Company, ECCS Large Break Spectrum Analysis for Prairie Island Unit 1 Using ENC WREM-IIA PWR Evaluation Model, XN-NF-78-46, November 1978.

14. NRC memo, N. Lauben to Z. Rosztocey, dated December 12, 1978

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