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ECCS LARGE BREAK SPECTRUM ANALYSIS FOR PRAIRIE ISLAND UNIT 1 USING ENC WREM-IIA PWR EVALUATION MODEL

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1.0 INTRODUCTION AND SUMMARY

This document presents results of a break spectrum analysis using the ENC WREM-IIA PWR ECCS evaluation model[1,2,3] for the Prairie Island Unit 1*Nuclear Power Plant operating at 1650 MWt. The results show that the criteria specified by 10 CFR 50.46 are satisfied with an analysis performed in conformance to Appendix K of 10 CFR 50. A calculation to evaluate the impact of injection of ECCS fluid above the core (upper plenum injection, UPI) rather than in the downcomer as modeled using the ENC WREM-IIA ECCS models is also presented. This analysis was done using an interim model which was developed by the NRC[20] staff and modified by Westinghouse.[21]

Guillotine break configurations were calculated for double-ended cold-leg pipe breaks (DECLG) with discharge coefficients of 1.0, 0.6 and 0.4. Split break configurations of the cold-leg pipe were calculated with a break area equal to twice the cross sectional pipe area (DECLS, 8.25 ft.^{2}), then with break areas reduced to 0.6 and 0.4 times this value. The break spectrum analysis was performed for a core composed of Exxon Nuclear Company (ENC) fuel at Beginning-of-Life (BOL) conditions.

The limiting break was calculated to be the 0.4 DECLG break, which resulted in a calculated peak clad temperature (PCT) of 2197°F and a calculated maximum local Zr/H_2O reaction of less than 13 percent. Addition of the interim UPI model calculated result +1F°, to the ENC WREM-IIA model result yields in a final PCT value of 2198°F. All of the calculations in the break spectrum were performed at a core power of 1683 MWt, which is 102 percent of rated power. The analysis

* This ECCS analysis applies also to Prairie Island Unit 2 with ENC fuel of the same fuel design as Unit 1, Cycle 5.

supports operation of the Prairie Island Unit 1 plant with a total Linear Heat Generation Rate (LHGR) of 14.03 kw/ft, at a total peaking factor (F_Q^T) of 2.21 at rated power. The peak clad temperature versus break discharge coefficient and break size are presented in Table 1.1. A detailed discussion of the break spectrum results is provided in Section 2.0.

The ENC WREM-IIA model was used for this analysis. The model includes the following computer codes: RELAP4-EM/ENC28C for blowdown and hot channel analyses; REFLEX for core reflood analysis; CONTEMPT LT/22, as modified in CSB 6-1[14] for containment backpressure analysis; TOODEE2/APR78 for heatup analysis; and the ENC modified interim NRC UPI model which accounts for UPI-core interaction. System models and nodalization used for the ENC WREM-IIA computer codes have been presented in the Two-Loop PWR Example Problem Document XN-NF-77-25(A)[16] and the generic PWR ECCS Evaluation Model Update ENC WREM-IIA Document XN-NF-78-30.[3,23]

TABLE 1.1

Peak Cladding Temperature Results

Prairie Island Unit 1 Reactor with ENC Fuel

$F_Q^T = 2.21$

	Maximum Power Node Clad Surface Temp. @ EOBY (°F)	PCT Results: PCT TIME (sec)	PCT Results: PCT (°F)
Guillotine Breaks			
DECLG $C_D = 1.0$	1150.	240.	2174.
DECLG $C_D = 0.6$	1176.	230.	2117.
DECLG* $C_D = 0.4$	1400.	206.	2197.
Split Breaks			
1.0 DECLS 8.25 ft^2	1139.	234.	2142.
0.6 DECLS 4.95 ft^2	1118.	232.	2114.
0.4 DECLS 3.30 ft^2	1065.	242.	2142.

* A value of +1F° must be added to the 0.4 DECLG PCT value to account for UPI effects.

2.0 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS-OF-COOLANT ACCIDENT)

2.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

The analysis for large breaks specified by 10 CFR 50.46[15], "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors", is presented in this section. The results of the loss of coolant accident analysis are shown in Tables 2.1 and 2.2, which indicate compliance with the Acceptance Criteria. The analytical techniques used are in compliance with Appendix K of 10 CFR 50, and are as described in XN-75-41, Volumes I and II, and supplements[1]; ENC-WREM-IIA model is described in XN-76-44[17], XN-76-36[18], XN-NF-78-30[3], XN-NF-78-25[23], and XN-76-27 plus supplements[2]. Except as noted below, the detailed system models are as given in the two-loop PWR example problem report: XN-NF-77-25(A)[16].

For the purpose of loss-of-coolant accident (LOCA) analyses, a LOCA is defined as a hypothetical rupture of the Reactor Primary Coolant System piping, up to and including the double-ended rupture of the largest pipe in the Reactor Coolant System or of any line connected to that system up to the first closed valve.

Should a major break occur, depressurization of the Reactor Coolant System results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. Reactor trip and scram were conservatively neglected for the large break analyses. A Safety Injection System signal is actuated when the appropriate setpoint (high containment pressure) is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complements void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.
2. Injection of borated water provides heat transfer from the reactor core and prevents excessive clad temperatures.

At the beginning of the blowdown phase, the entire Reactor Coolant System contains subcooled liquid which transfers heat from the core by forced convection cooling. After the break develops the time to departure from nucleate boiling (DNB) is calculated consistent with Appendix K of 10 CFR 50[14]. Post DNB core heat transfer (both transition and film boiling occurring) is also calculated in accordance with Appendix K of 10 CFR 50. As the core becomes uncovered, both turbulent and laminar forced convection to steam are considered as core heat transfer mechanisms.

When the Reactor Coolant System pressure falls below 715 psia, the accumulators begin to inject borated water. The conservative assumption is made that accumulator ECC water bypasses the core and goes out through the break until the termination of bypass. This conservatism is consistent with Appendix K of 10 CFR 50.

2.2 THERMAL ANALYSIS

2.2.1 Method of Analysis

For breaks greater than 1.0 ft^2, the RELAP4-EM code[1,2,17] is used to calculate the transient depressurization of the Reactor Coolant System as well as to describe the mass and enthalpy of flow out of the break. A specialized calculation (RELAP4-EM/HOT CHANNEL) is used to calculate cladding temperatures using time dependent boundary conditions in the upper and lower plenum

volumes from the basic blowdown analysis. Beyond the point of refill to the bottom of the core, a specialized calculation (REFLEX) is applied to determine the reflooding rate and system conditions. After end-of-bypass (EOBY), the program TOODEE2 is used to calculate peak clad temperatures.

2.2.2 Large Break LOCA Analysis Modeling

The Prairie Island Unit 1 nuclear power plant is a 2-loop Westinghouse pressurized water reactor with a dry containment. The reactor coolant system is nodalized into control volumes representing reasonably homogeneous regions, interconnected by flow-paths or "junctions" as described in XN-NF-77-25(A)[16]. The nodalized system blowdown model schematic is given in Figure 2.1.

For conservatism, the upper head temperature is taken to be that of the core outlet temperature. One percent of the steam generator tubes were assumed to be uniformly plugged. The unbroken loop was assumed symmetrical and modeled the same as the broken loop except for the break nodalization and the pressurizer. Pump performance curves characteristic of the Prairie Island pumps as supplied by the NSSS vendor were used in the analysis. System input parameters are given in Table 2.3. The evaluation of the EOBY has been updated to reflect available data from cold-leg steam-water mixing studies [5,6,7,8,9,&10] as shown in Reference 11.

The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and with decay heating as required by Appendix K of 10 CFR 50. The axial power profile used for the analysis is a chopped cosine curve with an axial peaking factor of 1.384 as given in Figure 2.2.

The values for the primary coolant system core inlet temperatures and the steam generator secondary side pressure were set based on Prairie Island Unit 1 plant operating data obtained from the utility. The values of the core inlet temperature and the steam generator secondary side pressure are 530°F and 710 psig, respectively.

The containment backpressure for the analysis 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]. 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"[14]. The containment parameters used in the containment analysis to determine the ECCS backpressure are presented in Table 2.4.

2.3 RESULTS

Using the ENC WREM-IIA codes, transient system behavior is determined by solving the governing conservation equations for mass, energy, and momentum. Energy transport, flow rates, and heat transfer are determined from appropriate correlations. Table 2.1 presents the timing and sequence of events as determined for the large break guillotine configuration with discharge coefficients of 1.0, 0.6 and 0.4 and the split break configuration with break areas of 8.25, 4.95, and 3.30 square feet.

In general, the transient events occur slower for smaller discharge coefficients or break sizes. Table 2.2 presents the peak clad temperatures and maximum metal-water reaction results for the above spectrum of break cases. This range of break sizes was determined to include the limiting case for peak clad temperature.

The analysis of the loss-of-coolant accident is performed at 102% of 1650 MWt (1683 MWt). The core power and other parameters used in the analyses are given in Table 2.3. Since there is usually margin between the value of the peak linear power density used in this analysis and the value expected in operation, a lower peak clad temperature would be obtained by using the peak linear power density expected during operation.

For the results discussed below, the hot spot is defined to be the location of maximum peak clad temperature. This location is given in Table 2.2 for each break size analyzed.

Figures 2.3 through 2.21 present the results of the analysis for the limiting break (0.4 DECLG). Unless otherwise noted, zero time corresponds to the time of break initiation.

The maximum peak cladding temperature of 2197°F (excluding the UPI model correction) was calculated for the double-ended cold-leg guillotine break configuration (C_D = 0.4) and a total linear heat generation rate of 14.31 kw/ft (F_Q^T = 2.21) for ENC fuel (102% of 14.03 kw/ft). The maximum PCT of 2198°F includes +1F° calculated with the UPI model. The maximum local metal-water reaction is less than 13% and the total core metal-water reaction reached will be much less than 1%, all well below the limits set by the criteria of 10 CFR 50.46.

Additional results from the break spectrum analysis are given in Table 2.5. Table 2.5 shows the maximum power node surface temperature for the limiting break (0.4 DECLG) to be more than 220F° higher than the other cases at end-of-bypass (EOBY). Based on these maximum power node temperatures at EOBY, the maximum temperature is clearly defined, and there is a well defined decreasing temperature trend with increasing C_D for the guillotine breaks and an increasing temperature trend with

increasing break size for the split breaks. Temperature differences between comparable fuel rod nodes are consistent between the limiting break case and the other break cases. Cladding temperature behavior remains consistent with EOBY temperature results until the reflood rate is calculated to be less than one inch per second. A reflood rate of less than one inch per second is predicted to occur 115 to 120 seconds from the initiation of the postulated LOCA.

The high cladding temperatures associated with the limiting break (0.4 DECLG) result in clad rupture (during abiabatic heatup) for the high power node prior to the initiation of reflood. For the other breaks which exhibit fuel cladding temperatures which are less than those of the 0.4 DECLG case, cladding rupture occurs 30 to 40 seconds after initiation of reflood and at higher axial elevations along the rod. The rupture node axial location information is also presented in Table 2.5. It is the lower initial cladding temperatures which delays cladding rupture and causes rupture to occur at higher elevations where the reflood heat transfer coefficients are lower. This accounts for the higher (but not limiting) PCT for the 1.0 DECLG case and the split breaks shown in Table 2.5. That is, since reflood rates are essentially the same for all cases, temperature behavior during abiabatic refill and during reflood would be expected to be consistent with the maximum power node temperatures as shown in Table 2.5. The limiting break is consistent with this expected behavior, but other cases deviate from the expected trend due to migration of the rupture node and associated steam cooling effects around the rupture node. When these phenomena are considered with the Table 2.2 events, the limiting break for Prairie Island Unit 1 is clearly defined as the 0.4 DECLG case, and the break spectrum shape can be

explained in terms of the model.

ENC has performed numerous analyses and sensitivity studies on PWR systems using the ENC ECCS evaluation model. These studies have demonstrated the adequacy of the system nodalization used. In addition, these studies have shown that for transient conditions similar to those calculated for the Prairie Island reactor during the LOCA, the reactor coolant inlet pipe or cold leg is the worst break location and that BOL fuel condition result in the maximum PCT.

Small break (1.0 ft^2 area or less) analyses have not been performed as a part of the Prairie Island ECCS analysis because the ECCS systems provide adequate protection and are capable of limiting the cladding thermal transient for small pipe breaks to temperatures well below those of the large breaks. This exclusion of small breaks is based on ENC analysis results from similar Westinghouse reactor designs(24). NSSS vendor results have also shown that small breaks are not limiting in all these cases including Prairie Island.

2.4 CONCLUSIONS

For breaks up to and including the double-ended severance of a reactor coolant pipe, the Prairie Island Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10 CFR 50.46 for the ENC reload fuel of similar design operating in accordance with the LHGR limits noted in Section 1.0. That is:

1. The calculated peak fuel element clad temperature does not exceed the 2200°F limit.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total zircaloy associated with the active fuel rod length in the reactor.

3. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The hot fuel rod cladding oxidation limits of 17% are not exceeded during or after quenching.
4. The system long-term cooling capabilities provided for previous cores remain applicable for ENC fuel.

TABLE 2.1

PRAIRIE ISLAND UNIT 1

LARGE BREAK EVENTS

Event	Time (Seconds)					
	DECLG (C_D=1.0)	DECLG (C_D=0.6)	DECLG (C_D=0.4)	1.0 DECLS (8.25 ft^2)	0.6 DECLS (4.95 ft^2)	0.4 DECLS (3.30 ft^2)
Start	0.00	0.00	0.0	0.000	0.00	0.00
Initiate Break	0.05	0.05	0.05	0.05	0.05	0.05
Safety Injection Signal	0.50	0.55	0.65	0.50	0.55	0.60
Accumulator Injection, Intact Loop	6.80	7.40	8.80	6.85	7.00	8.55
Accumulator Injection, Broken Loop	0.15	2.50	4.80	0.85	3.10	6.95
Pressurizer Empties	8.80	8.80	8.80	8.80	8.80	8.80
End-of-Bypass	19.34	19.79	21.50	18.70	18.79	19.56
Safety Injection Flow, SIS	25.50	25.55	25.65	25.50	25.55	25.60
Start of Reflood	35.36	35.73	37.24	34.69	34.75	35.32
Accumulator Empty, Intact Loop	45.56	47.71	48.13	45.63	45.85	47.52
Peak Clad Temperature Reached	239.50	229.60	205.50	233.60	231.60	242.60

TABLE 2.2

PRAIRIE ISLAND UNIT 1

LARGE BREAK RESULTS

Event	DECLG (C_D=1.0)	DECLG (C_D=0.6)	DECLG (C_D 0.4)	1.0 DECLS (8.25 ft^2)	0.6 DECLS (4.95 ft^2)	0.4 DECLS (3.30 ft^2)
Peak Cladding Temperature,* ^oF	2174	2117	2197	2142	2114	2142
Peak Temperature Location, ft	7.00	6.75	7.81	7.00	7.00	7.25
Local Zr/H_2O Reaction (Max.), %	9.88	8.74	12.34	9.12	8.34	8.50
Local Zr/H_2O Location, ft	7.00	6.75	7.50	7.00	7.00	7.25
Total H_2 Generation, % of total Zr reacted	<1%	<1%	<1%	<1%	<1%	<1%
Hot Rod Burst Time, sec	67.24	63.69	36.80	67.60	72.09	87.06
Hot Rod Burst Location, ft	6.75	6.50	6.00	6.75	6.75	7.00
Linear Heat Generation Rate, kw/ft at BOCREC	0.7330	0.7311	0.7227	0.7363	0.7358	0.7329

* A value of +1°F must be added to the limiting PCT value t. account for upper plenum injection-core interaction effects.

TABLE 2.3

PRAIRIE ISLAND UNIT 1 2-LOOP PWR DATA

Primary Heat Output, MWt	1650*
Primary Coolant Flow, lbm/hr	6.82×10^7
Primary Coolant Volume, ft^3	10,309.**
Operating Pressure, psia	2,250.
Inlet Coolant Temperature, ^oF	530.
Reactor Vessel Volume, ft^3	2364.
Pressurizer Volume, Total, ft^3	1000.
Pressurizer Volume, Liquid, ft^3	600.
Accumulator Volume, Total, ft^3(each of two)	2000.
Accumulator Volume, Liquid, ft^3	1250.
Accumulator Trip Point Pressure, psia	714.7
Steam Generator Heat Transfer Area, ft^2	50,985.
Steam Generator Secondary Flow, lbm/hr	3.54×10^6
Steam Generator Secondary Pressure, psia	724.7
Reactor Coolant Pump Head, ft	277.
Reactor Coolant Pump Speed, rpm	1190.
Moment of Inertia, lbm-ft^2/rad	78,000.
Cold Leg Pipe, I.D., in	27.5
Hot Leg Pipe, I. D., in	29.0
Pump Suction Pipe, I. D., in	31.0

* Primary Heat Output used in RELAP4-EM Model = 1.02 x 1650 = 1683 MWt.
** Includes total accumulator and pressurizer volumes.

TABLE 2.3 (Continued)

Fuel Assembly Rod Diameter, in*	0.424
Fuel Assembly Rod Pitch, in*	0.556
Fuel Assembly Pitch, in*	7.803
Fueled (Core) Height, in*	144.
Fuel Heat Transfer Area, ft^2	28,851.
Fuel Total Flow Area, ft^2	26.71
Steam Generator Tube Plugging (Assumed uniform)	1%

* ENC fuel parameters

TABLE 2.4

PRAIRIE ISLAND UNIT 1 DRY CONTAINMENT DATA

Containment Physical and Thermal Parameters

Net Free Volume	$1.36 \times 10^6 ft^3$
Outside Air Temperature	-20 °F
Spray Flow	
Trip Setting	23 psig
Time Delay	5 sec
Fan Coolers	
Trip Setting	4 psig
Time Delay	15 sec
Containment Initial Conditions:	
Temperature	90^oF
Pressure	14.7 psia
Relative Humidity	100%
Containment Spray Water:	
Temperature	70^oF
Flow Rate (Total, 2 pumps)	3600 gpm
Fan Air Cooler Capacity (total 4 coolers)	

Vapor Temperature (^oF)	Capacity (Btu/hr)
150	3.10×10^7
170	3.65×10^7
190	4.60×10^7
210	5.55×10^7
230	6.75×10^7
250	8.20×10^7
270	9.60×10^7
290	1.11×10^8
300	1.18×10^8

TABLE 2.4 (Continued)

PRAIRIE ISLAND UNIT 1 CONTAINMENT DATA

Passive Heat Sink Thermal Conductivity and Volumetric Heat Capacity Data

Materials	Thermal Conductivity (Btu/hr-ft-^{o}F)	Volumetric Heat Capacity Btu/ft^{3}-^{o}F)
Steel	28.0	56.2
Structural Concrete	0.8	32.0
Metal Overcoat Paint	0.29	28.0
Metal Primer	1.50	28.0
Concrete Overcoat Paint	0.29	28.0
Concrete Primer	0.29	32.0

Containment Passive Heat Sinks

	DESCRIPTION	MATERIAL(s)	THICKNESS IN^{2}	SURFACE FT^{2}
1.	Contain cylinder	steel	1.5	41,300.
2.	Containment dome	steel	0.75	32,000.
3.	Reactor vessel and refueling canal	steel concrete	0.25 12.00	7,860. 7,860.
4.	HVAC ducting	steel	0.25	32,000.
5.	NSSS supports	steel	0.5	44,000.
6.	Exposed pipe	steel	0.375	6,800.
7.	Hand rails	steel	0.145	1,695.
8.	Grating	steel	0.09	12,400.
9.	Conduit and cable trays	steel	0.1	6,000.
10.	Accumulators	steel	1.44	2,200.
11.	Ductwork	steel	0.1875	35,125.
12.	Thick concrete walls	concrete	12.0	40,800.
13.	Thick concrete floors	concrete	6.0	25,070.
14.	Thin concrete walls	concrete	3.0	7,570.

TABLE 2.5

Heatup Calculated Results Summary

Prairie Island Unit 1 Reactor with ENC Fuel

Guillotine Breaks	Max. Power Node Clad Surf. Temp. @ EOBY (°F)	PCT Results: PCT Time (sec)	PCT Results: PCT (°F)	Heatup Rupture Node No.
DECLG CD = 1.0	1150.	240.	2174.	11
DECLG CD = 0.6	1176.	230.	2117.	10
DECLG* CD = 0.4	1400.	206.	2197.	8
Split Breaks				
1.0 DECLS 8.25 ft^2	1139.	234.	2142.	11
0.6 DECLS 4.95 ft^2	1118.	232.	2114.	11
0.4 DECLS 3.30 ft^2	1065.	242.	2142.	12

* A value of +1F° must be added to the 0.4 DECLG PCT to account for UPI effects.

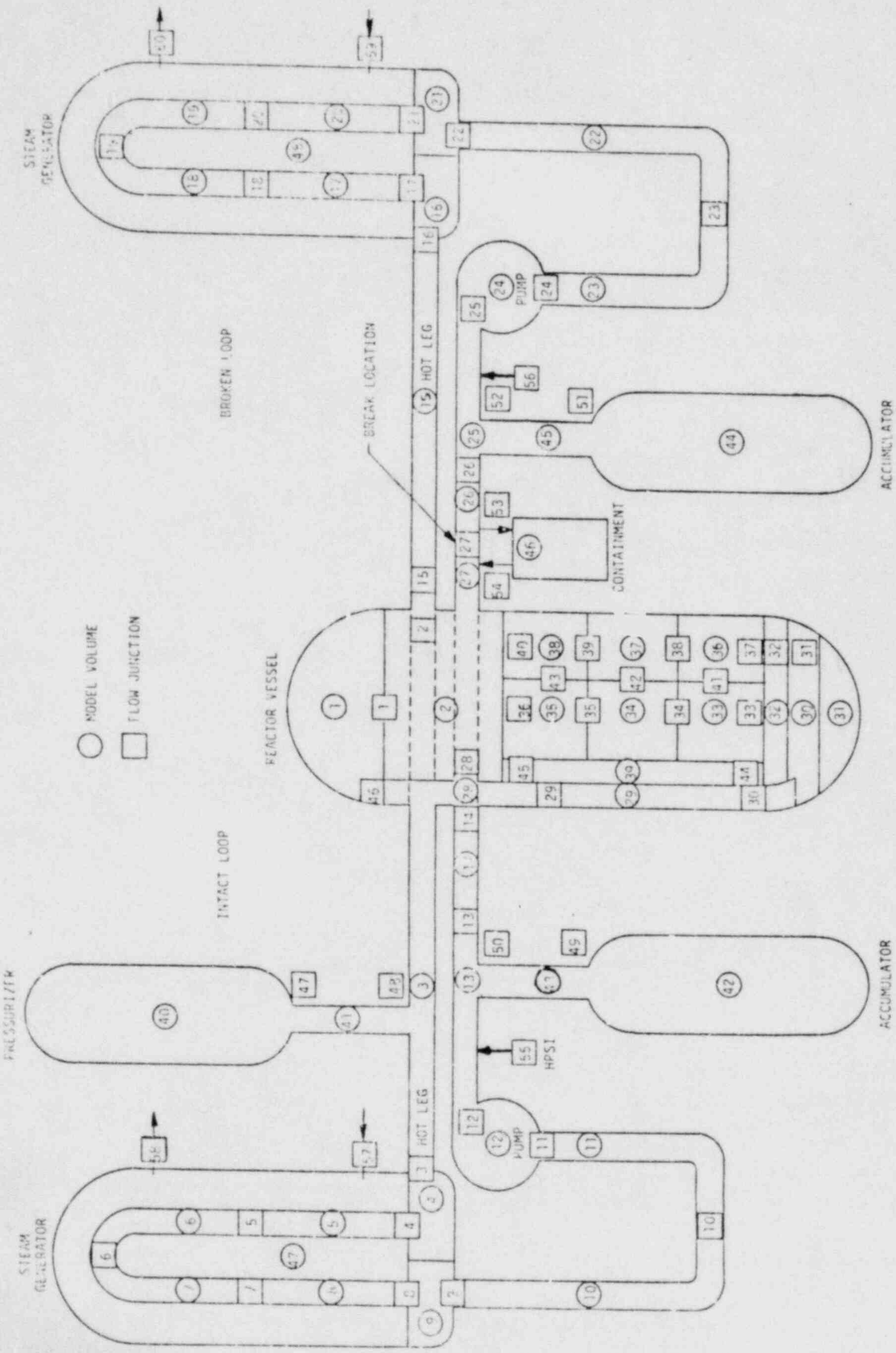


FIGURE 2.1 RELAP4/EM BLOWDOWN SYSTEM NODALIZATION FOR PRAIRIE ISLAND UNIT 1 2-LOOP PWR

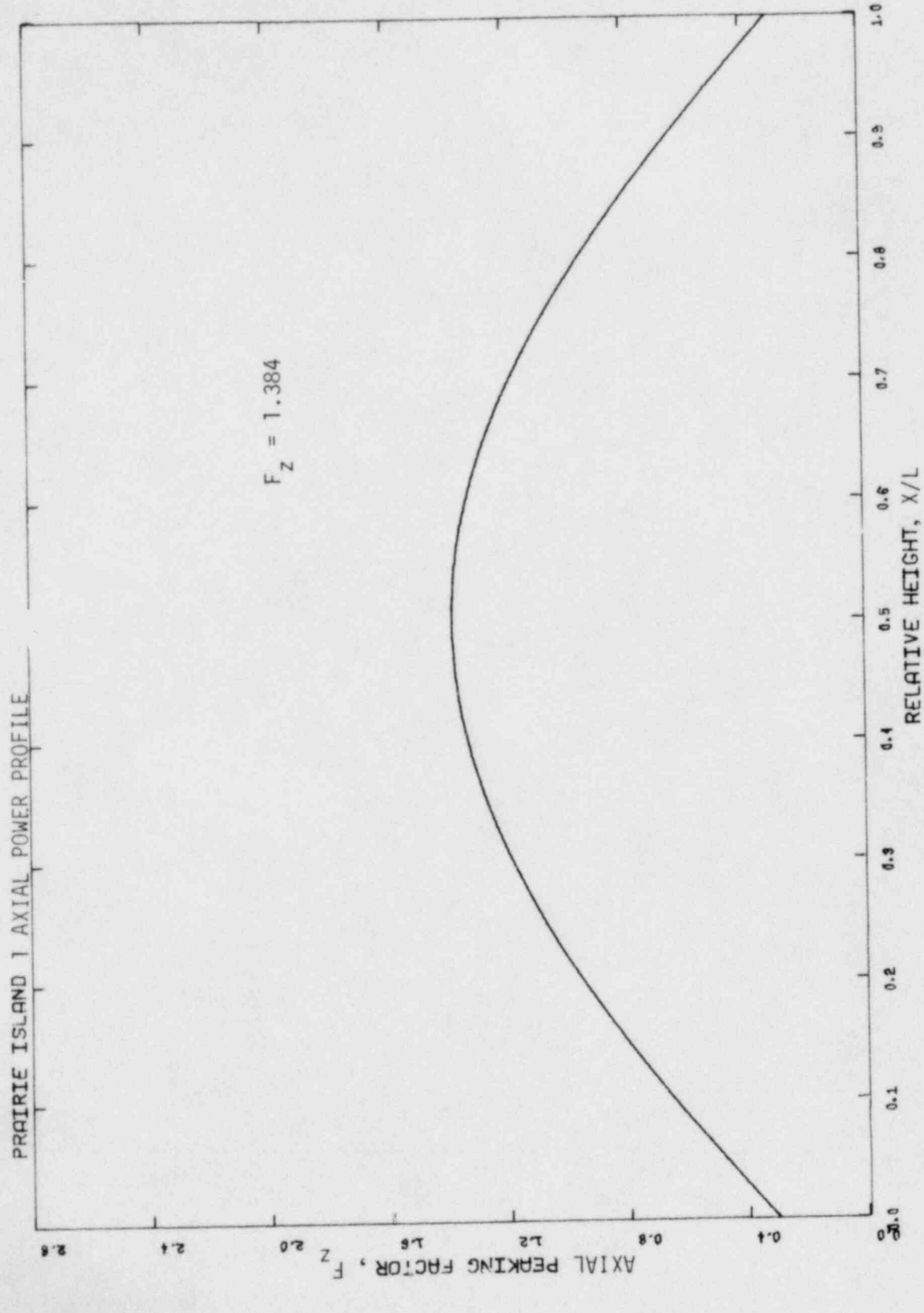


FIGURE 2.2 AXIAL PEAKING FACTOR VERSUS FUEL ROD LENGTH FOR PRAIRIE ISLAND UNIT 1 ECCS ANALYSIS

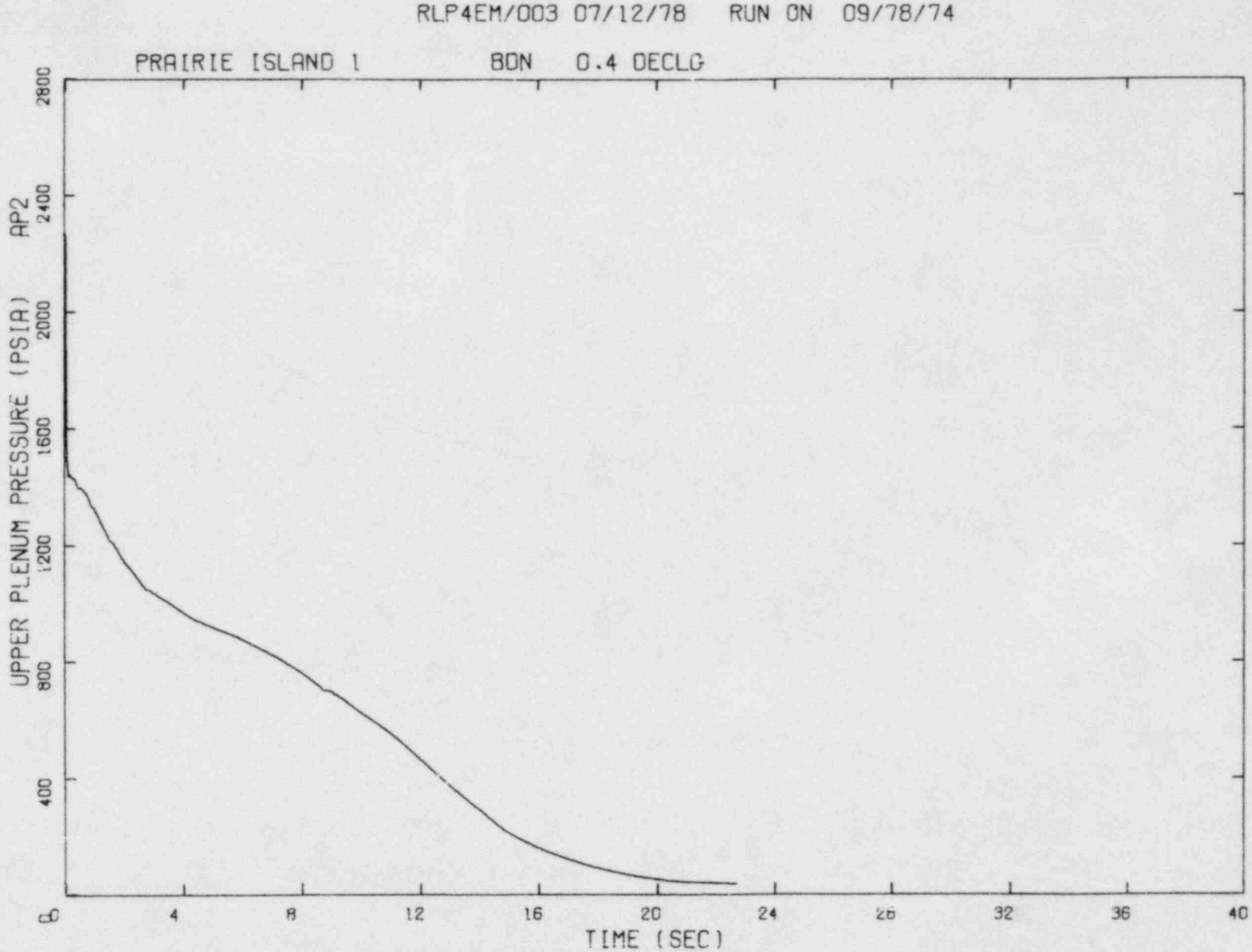


FIGURE 2.3 BLOWDOWN SYSTEM PRESSURE, 0.4 DECLG BREAK

PLOTTED ON 78/09/13.

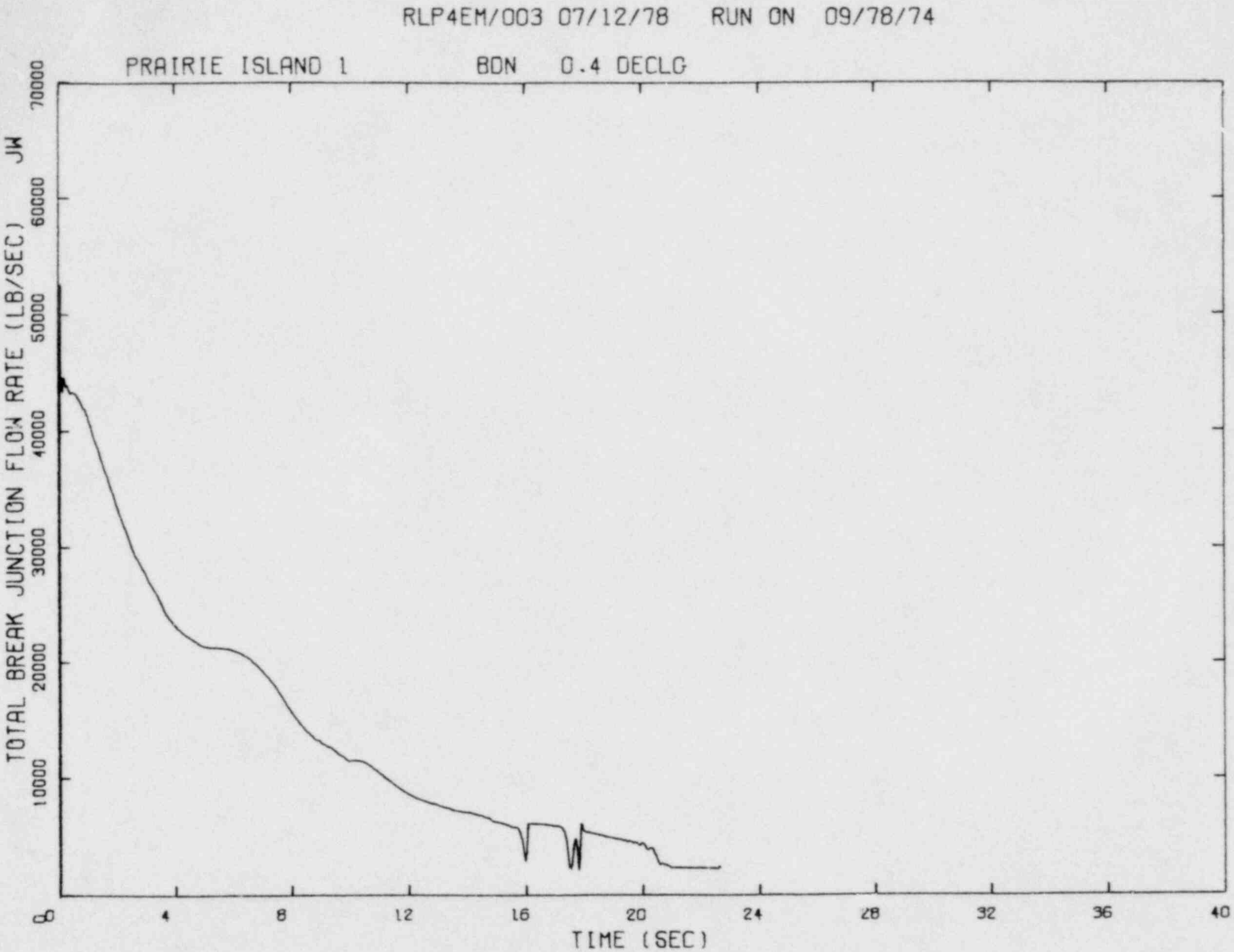


FIGURE 2.4 BLOWDOWN TOTAL BREAK FLOW, 0.4 DECLG BREAK

PLOTTED ON 78/09/15.

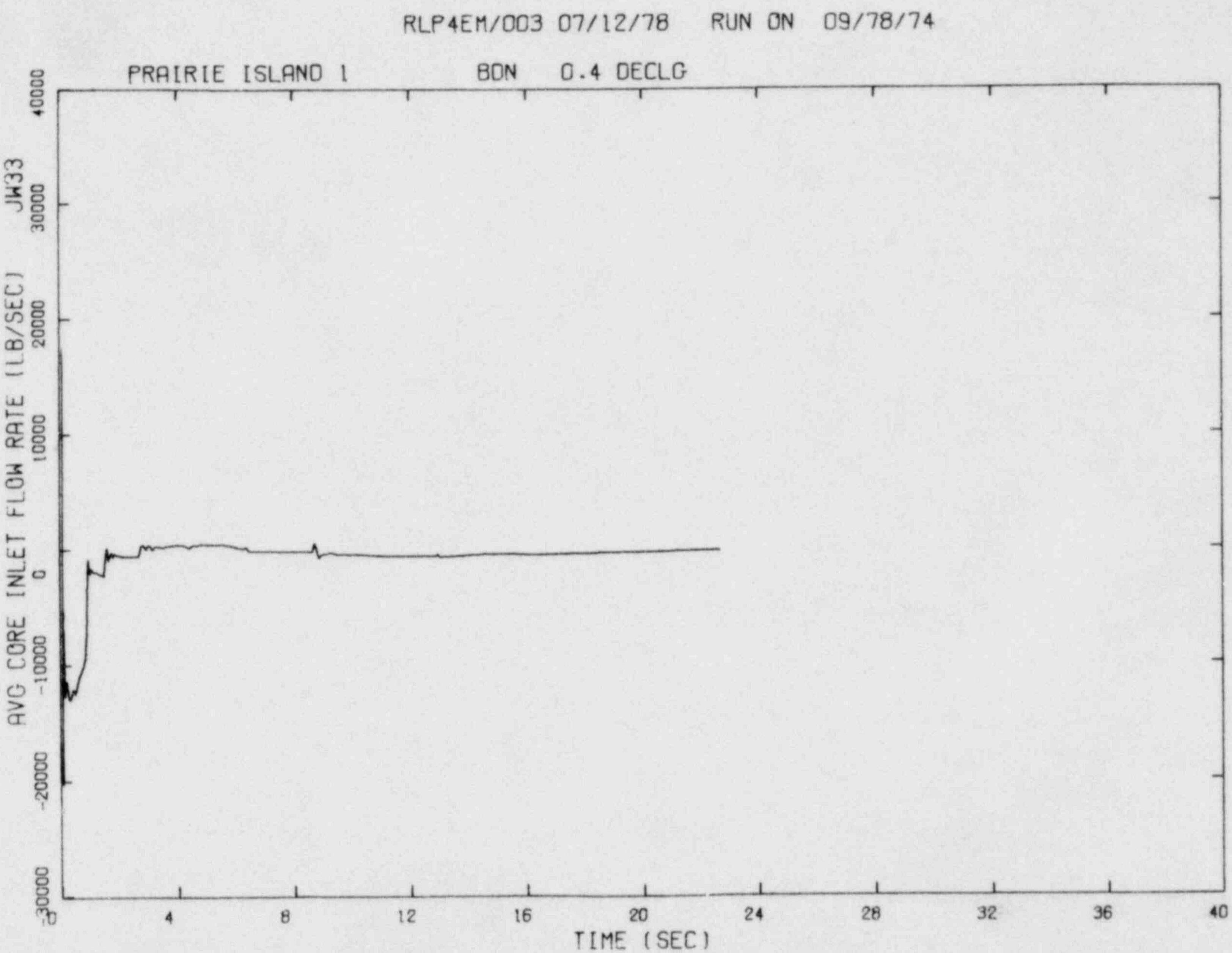


FIGURE 2.5 BLOWDOWN AVERAGE CORE INLET FLOW, 0.4 DECLG BREAK

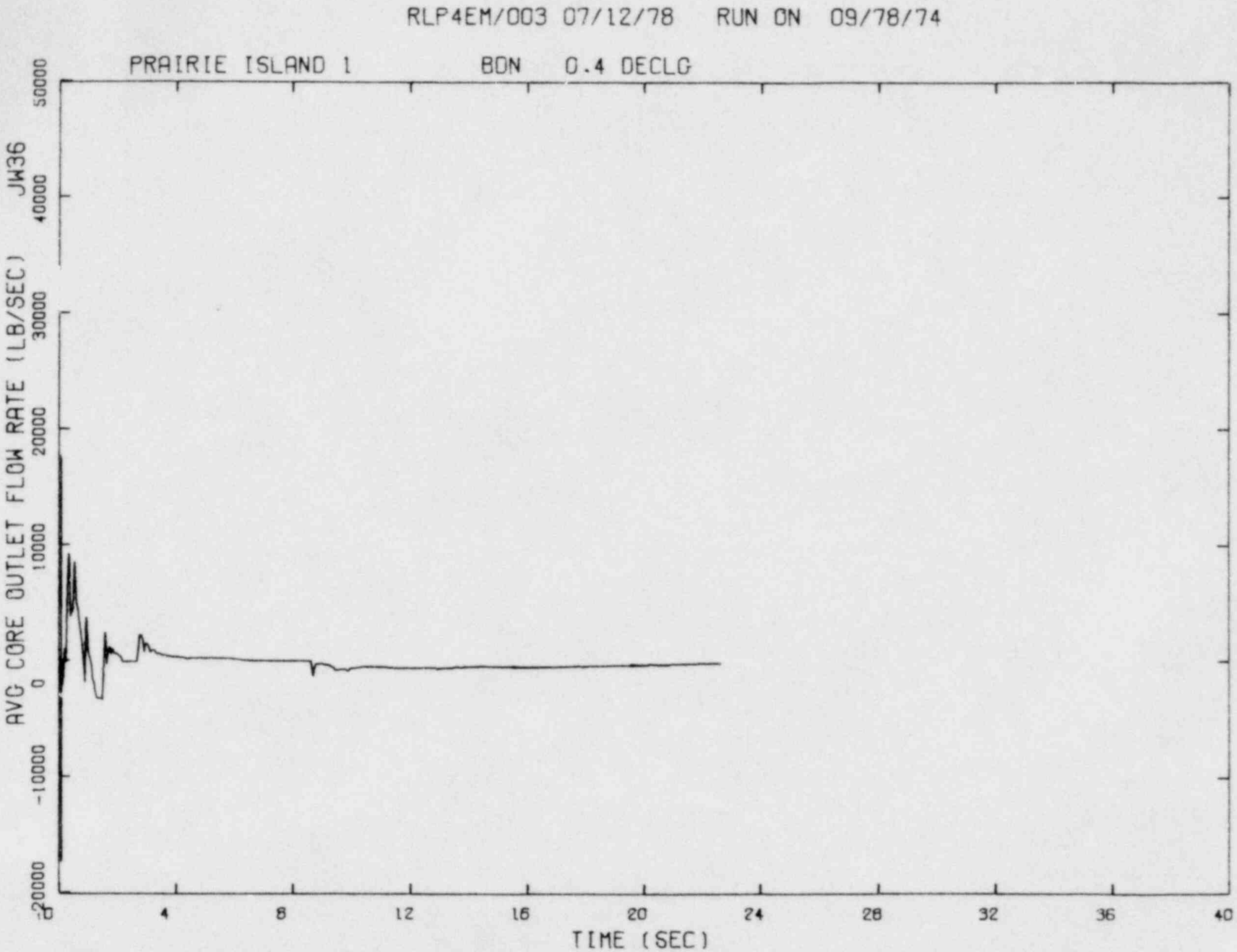


FIGURE 2.6 BLOWDOWN AVERAGE CORE OUTLET FLOW, 0.4 DECLG BREAK

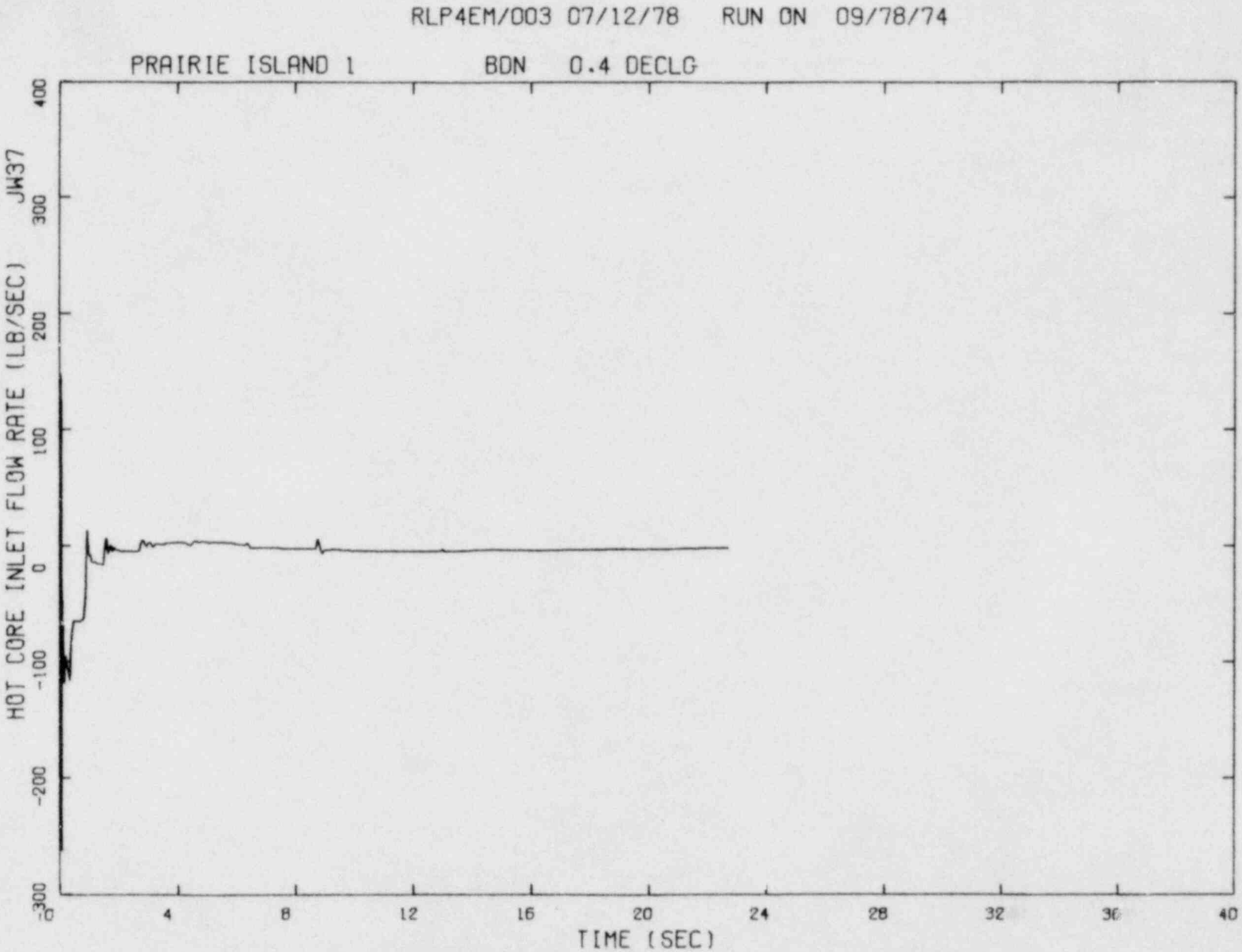


FIGURE 2.7 BLOWDOWN HOT CHANNEL INLET FLOW, 0.4 DECLG BREAK

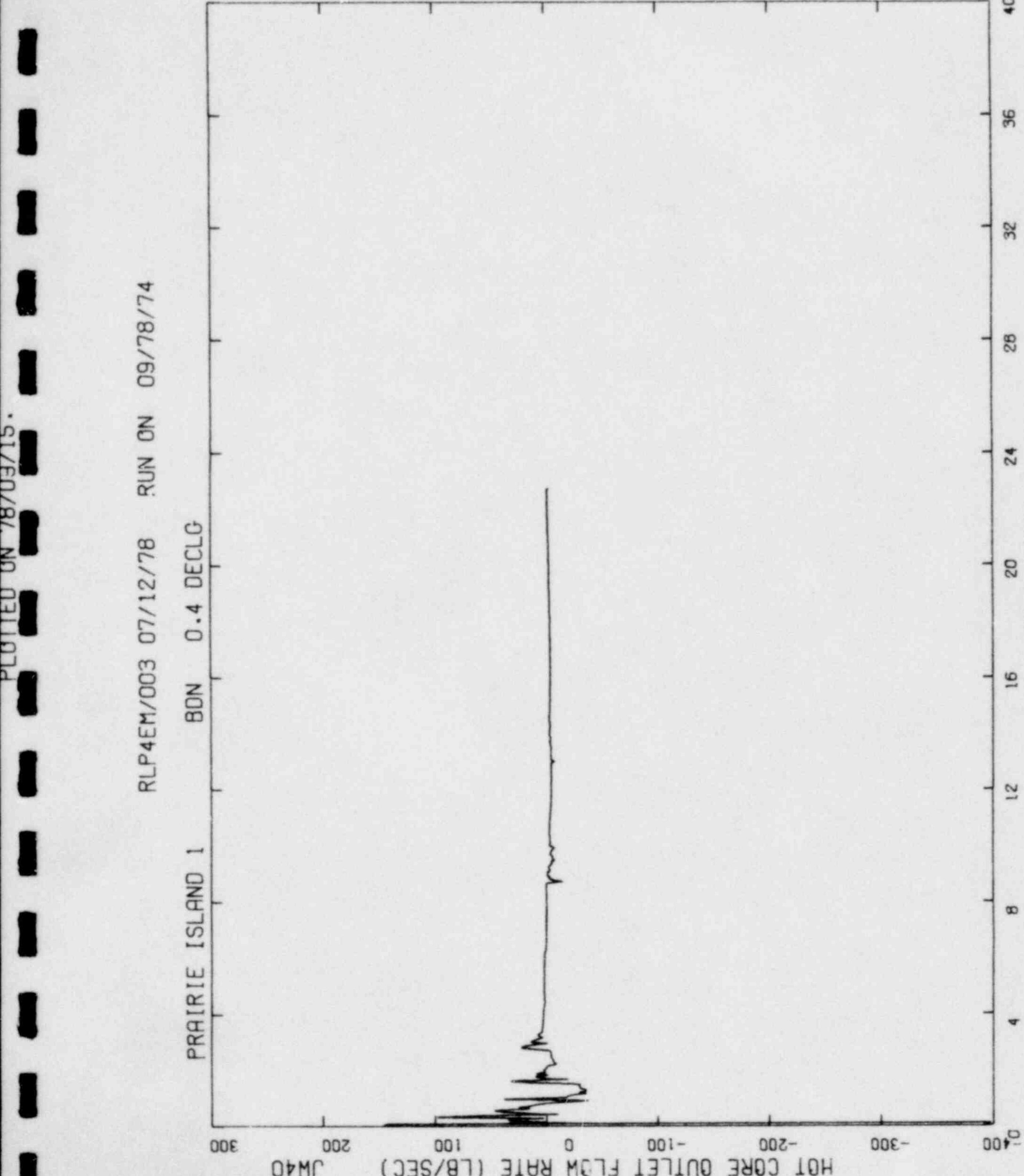


FIGURE 2.8 BLOWDOWN HOT CHANNEL OUTLET FLOW, 0.4 DECLG BREAK

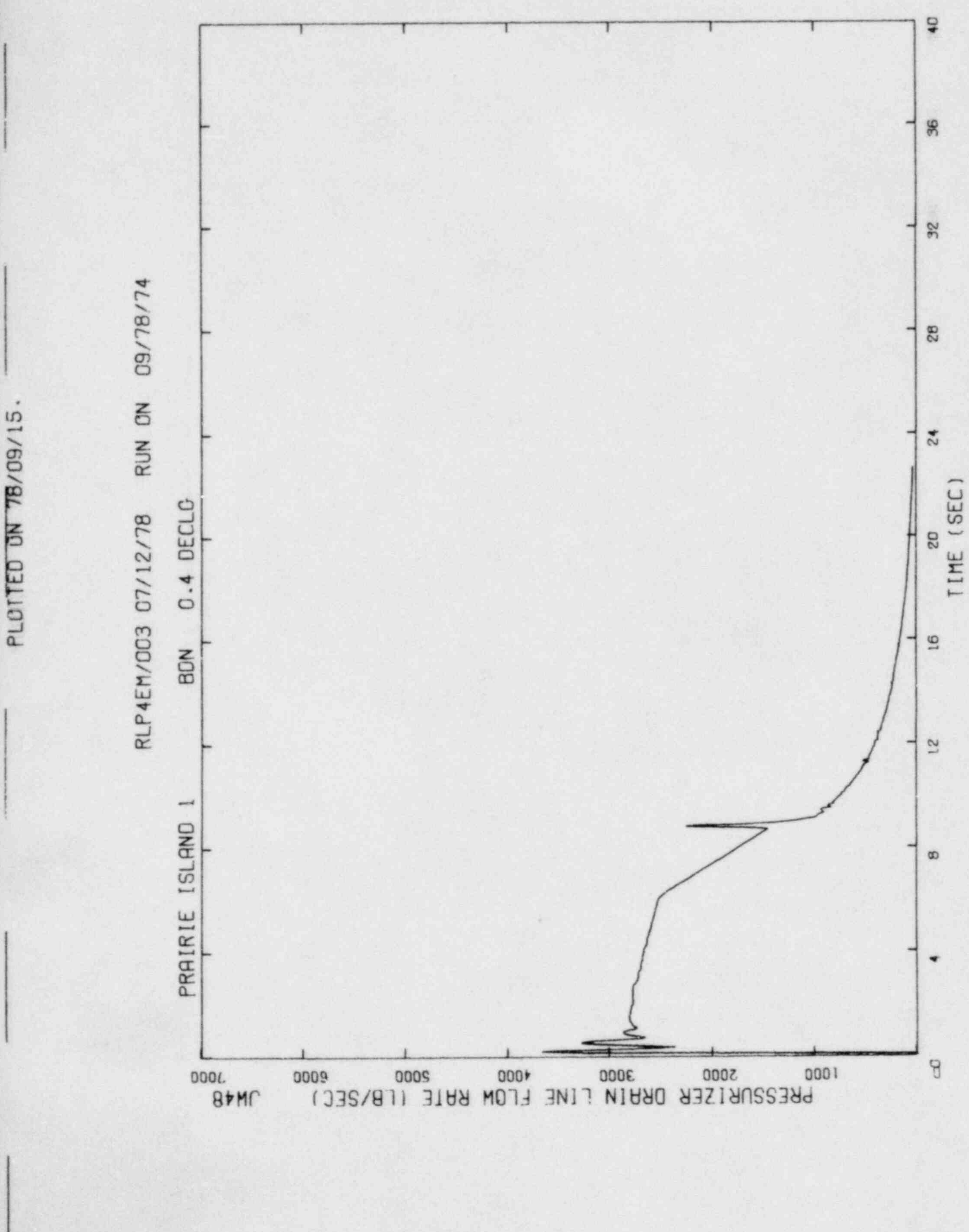


FIGURE 2.9 BLOWDOWN PRESSURIZER SURGE LINE FLOW, 0.4 DECLG BREAK

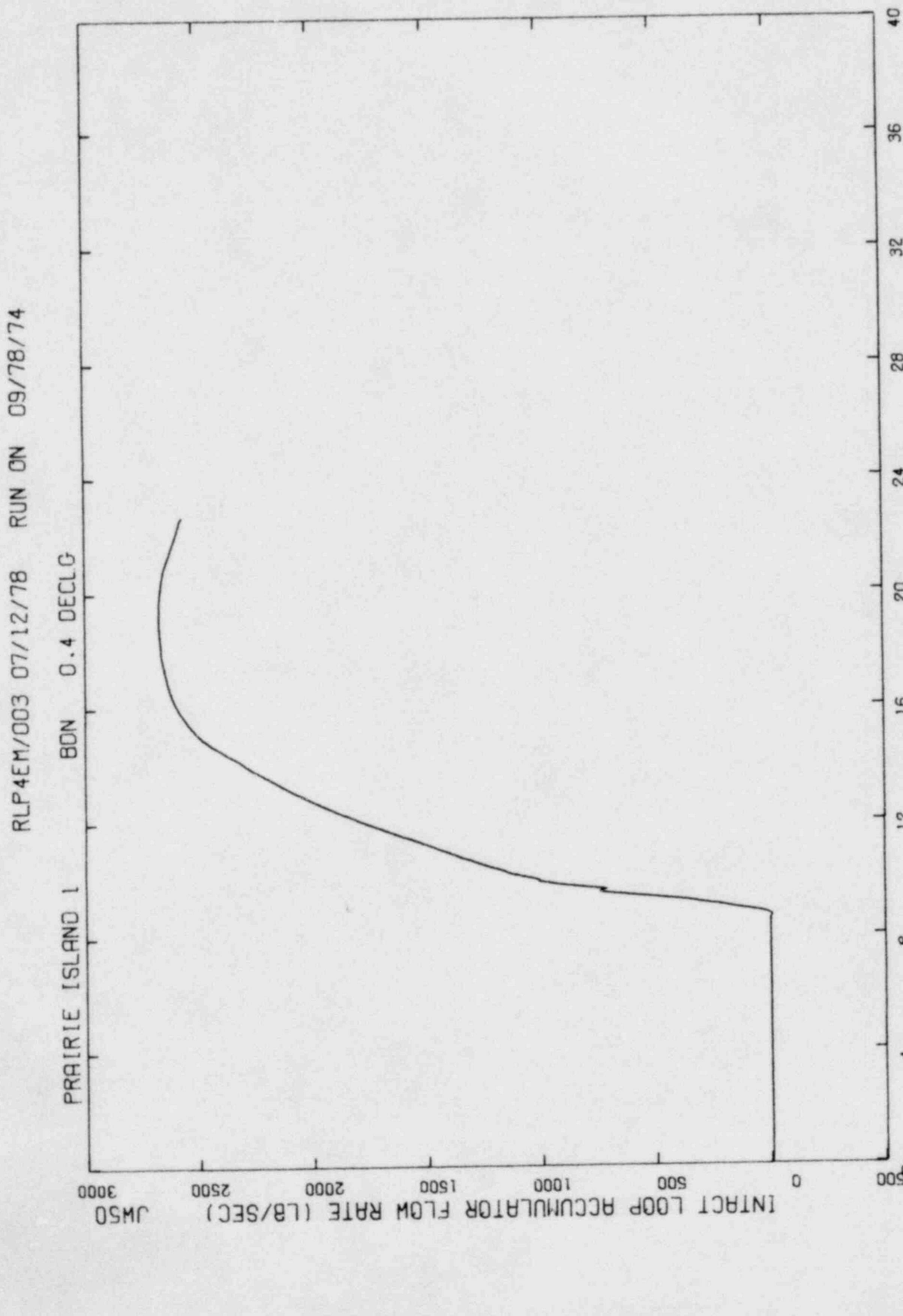
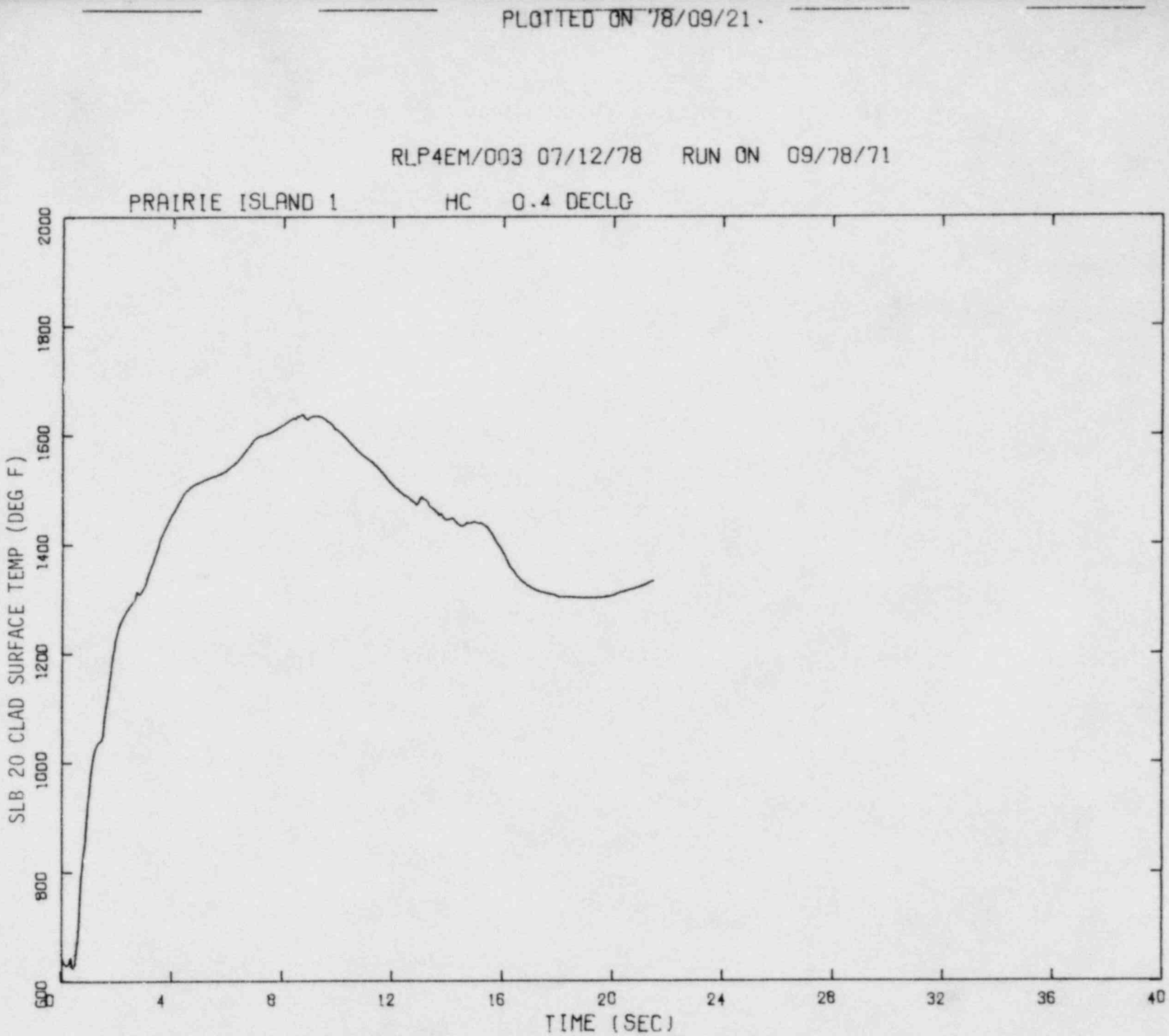


FIGURE 2.10 BLOWDOWN INTACT LOOP ACCUMULATOR FLOW, 0.4 DECLG BREAK



<u>FIGURE 2.11</u> BLOWDOWN HOT ROD CLADDING SURFACE TEMPERATURE, NODE 20, 0.4 DECLG BREAK

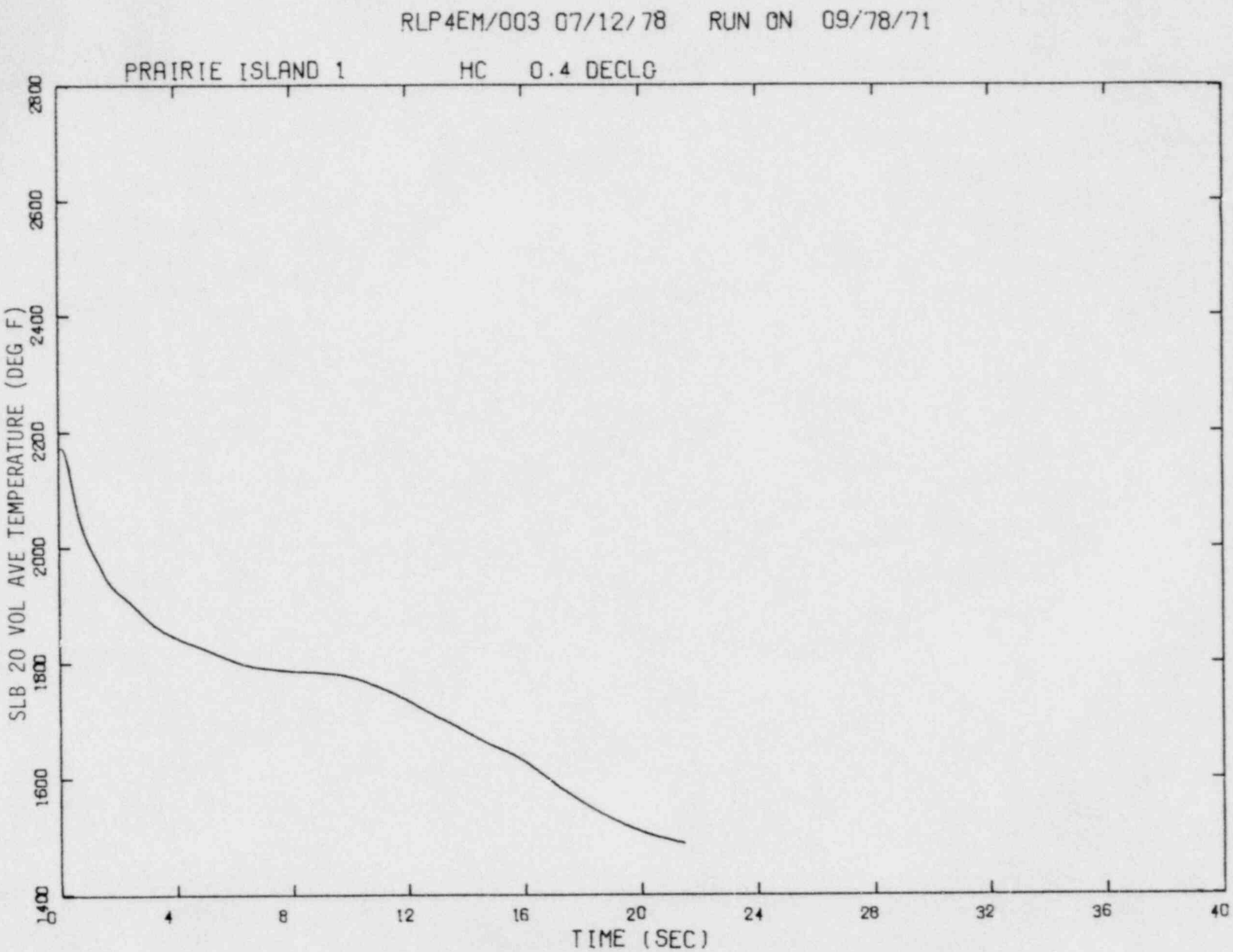


FIGURE 2.12 BLOWDOWN HOT ROD VOLUMETRIC AVERAGE FUEL TEMPERATURE, NODE 20, 0.4 DECLG BREAK

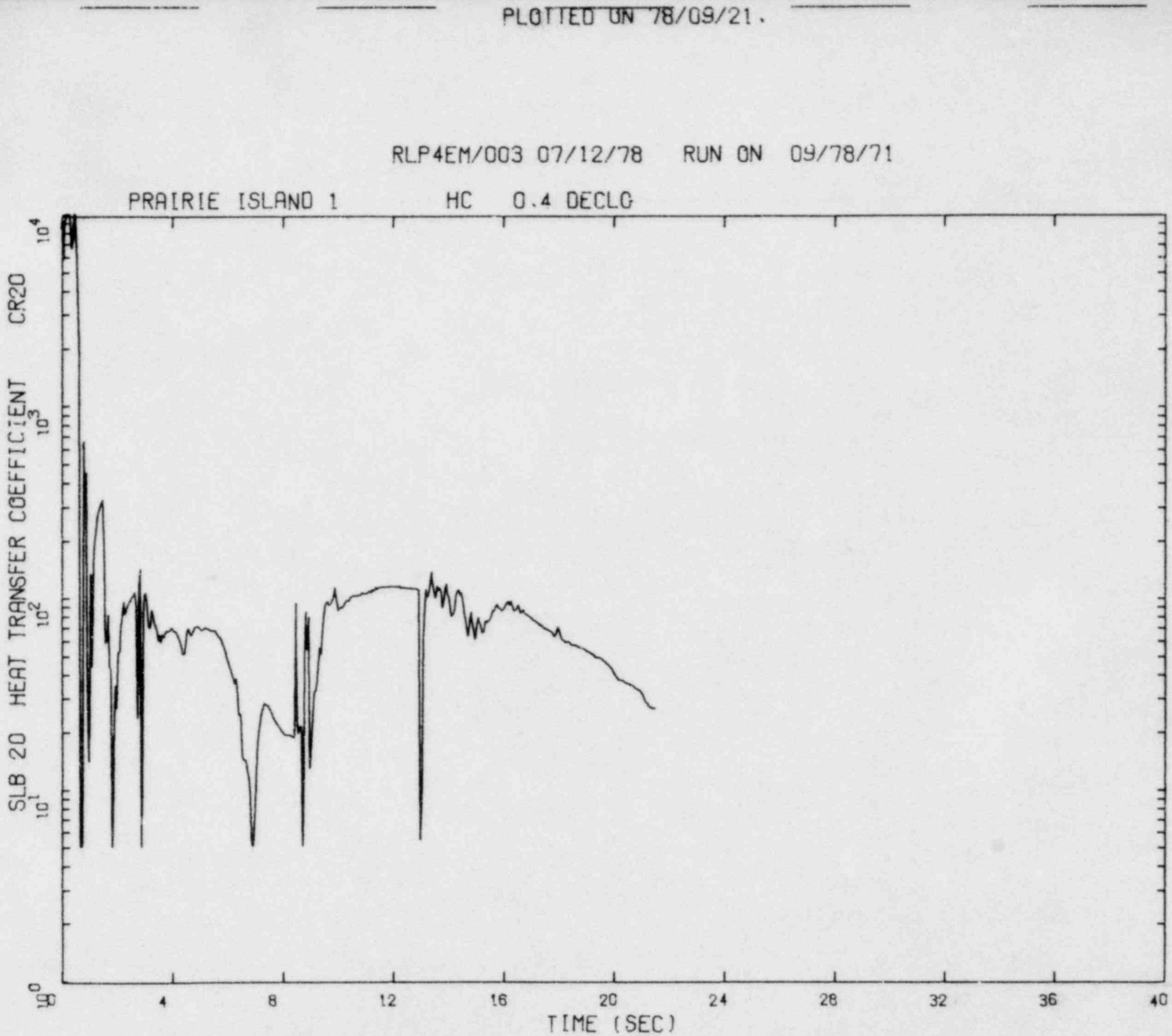


FIGURE 2.13 HOT ROD BLOWDOWN HEAT TRANSFER COEFFICIENT, NODE 20, 0.4 DECLG BREAK

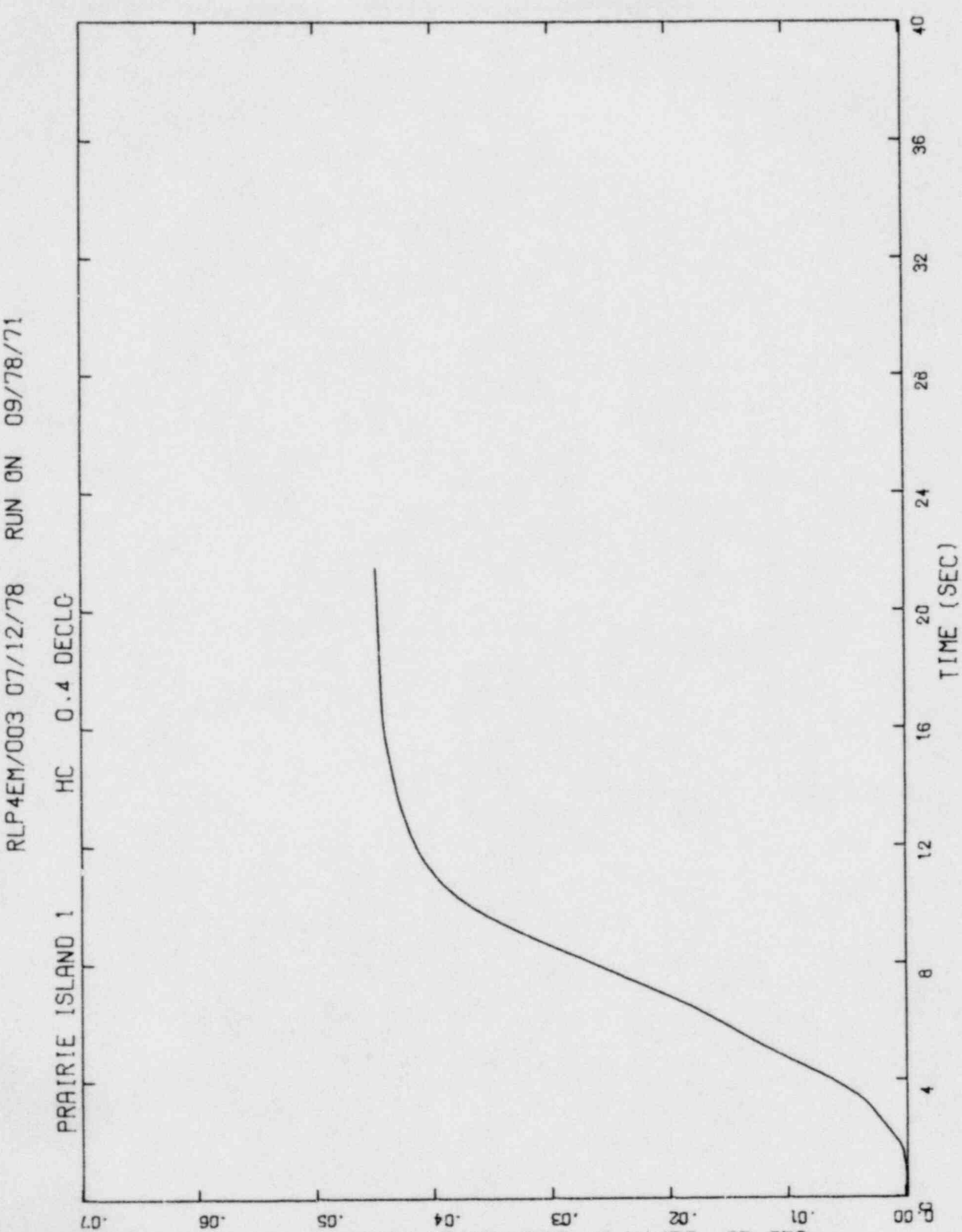


FIGURE 2.14 HOT ROD BLOWDOWN DEPTH OF ZIRCONIUM - WATER REACTION, NODE 20, 0.4 DECLG BREAK

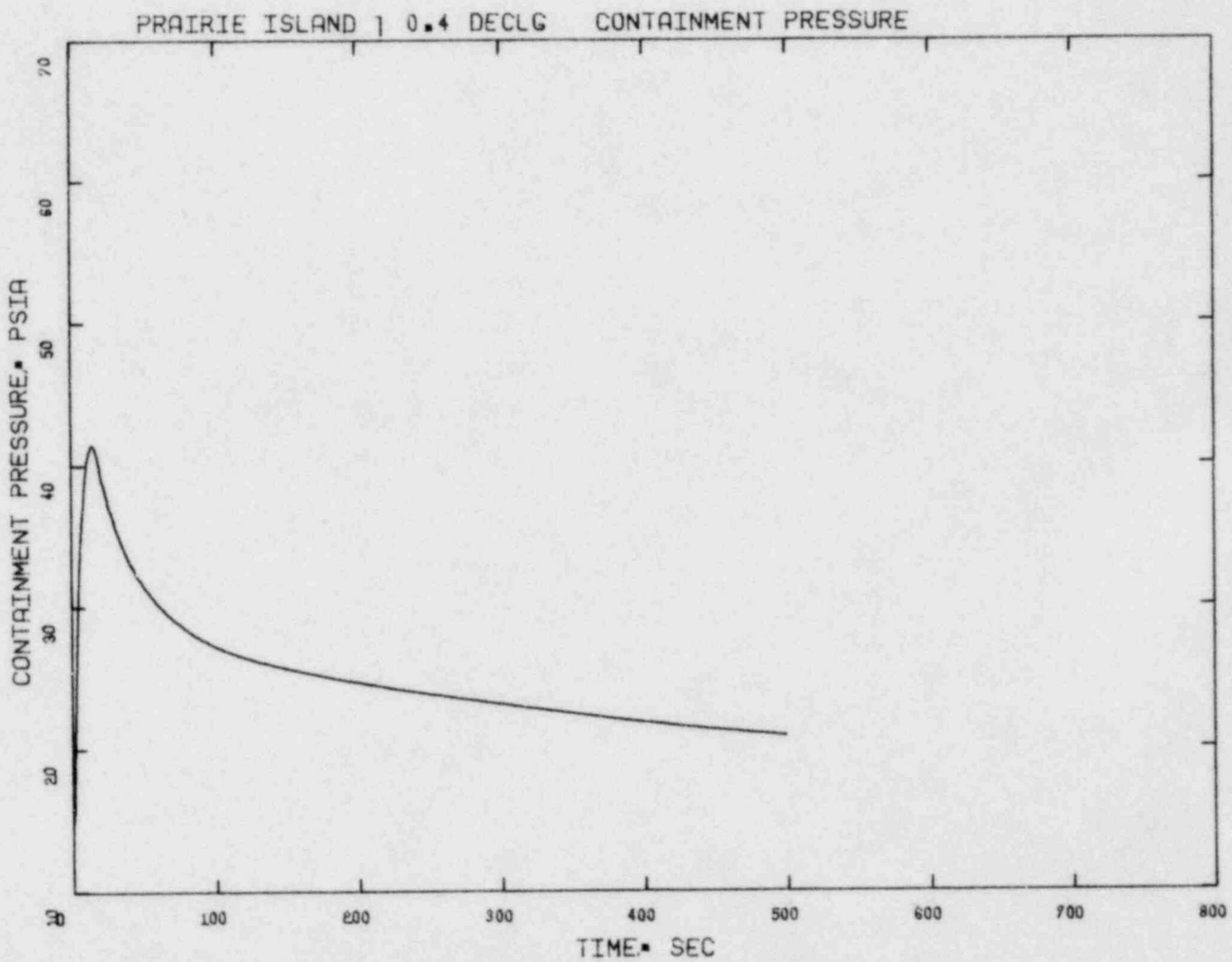


FIGURE 2.15 CONTAINMENT BACKPRESSURE, 0.4 DECLG BREAK

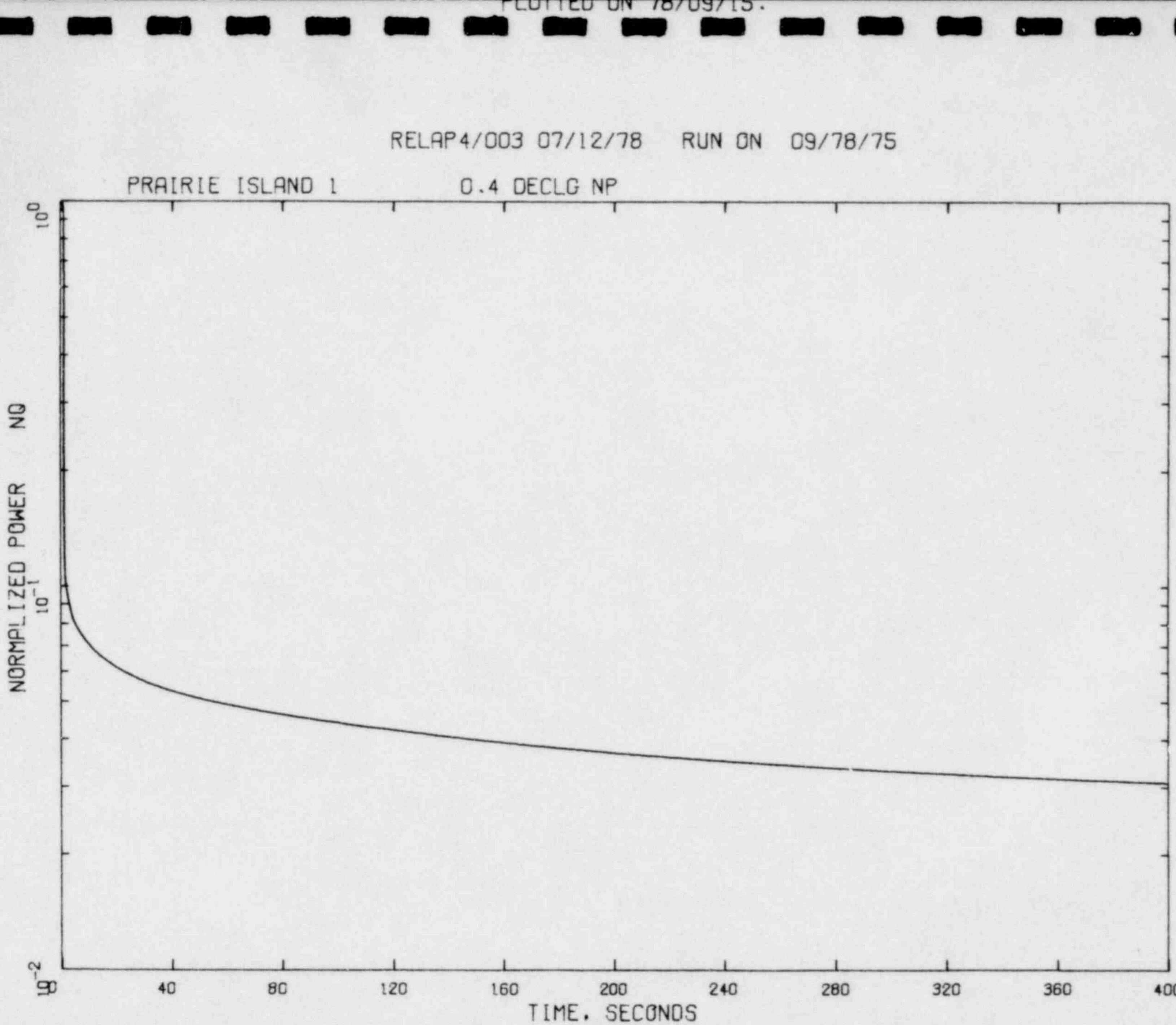


FIGURE 2.16 NORMALIZED POWER, 0.4 DECLG BREAK

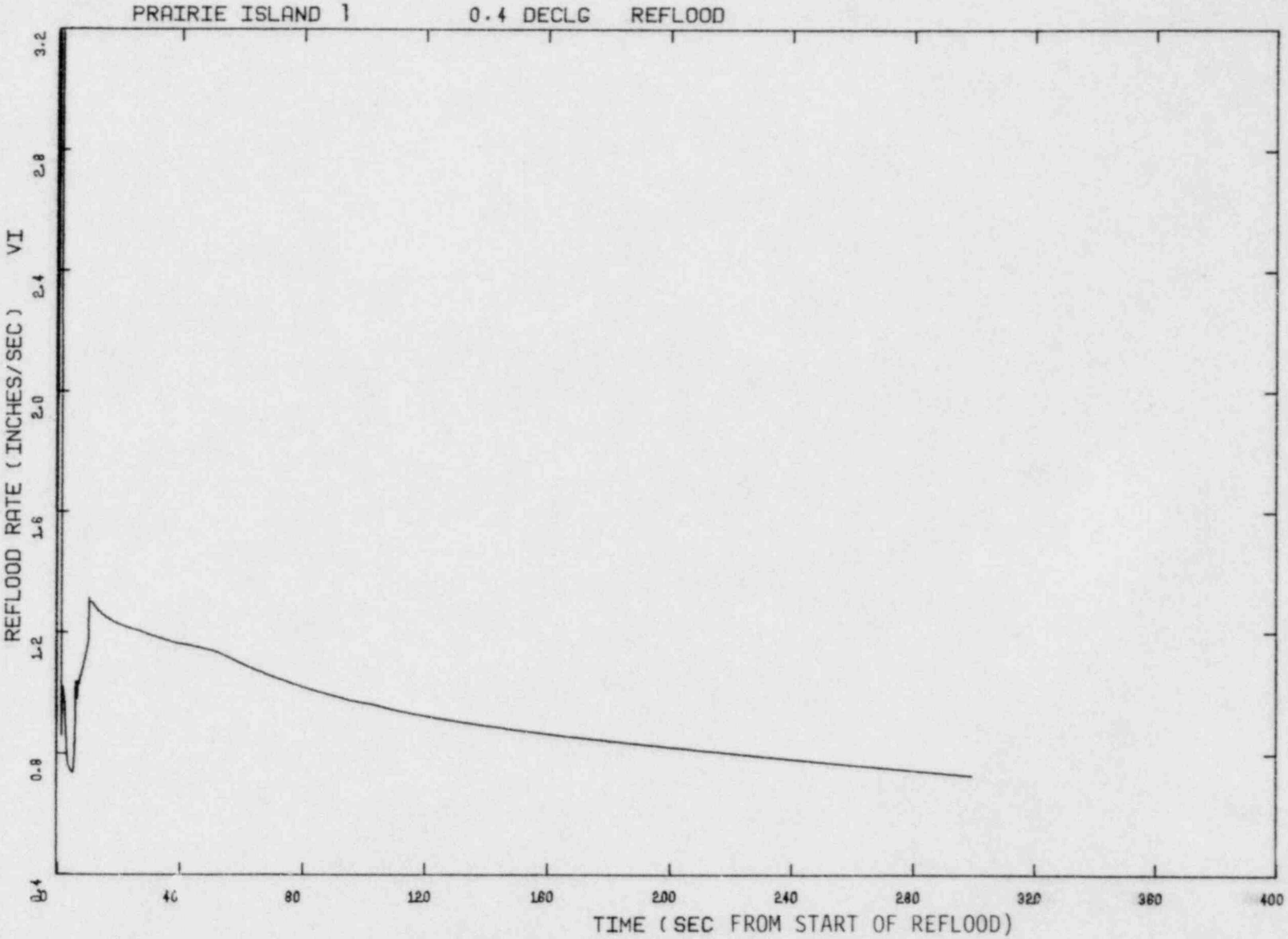


FIGURE 2.17 REFLOOD CORE FLOODING RATE, 0.4 DECLG BREAK

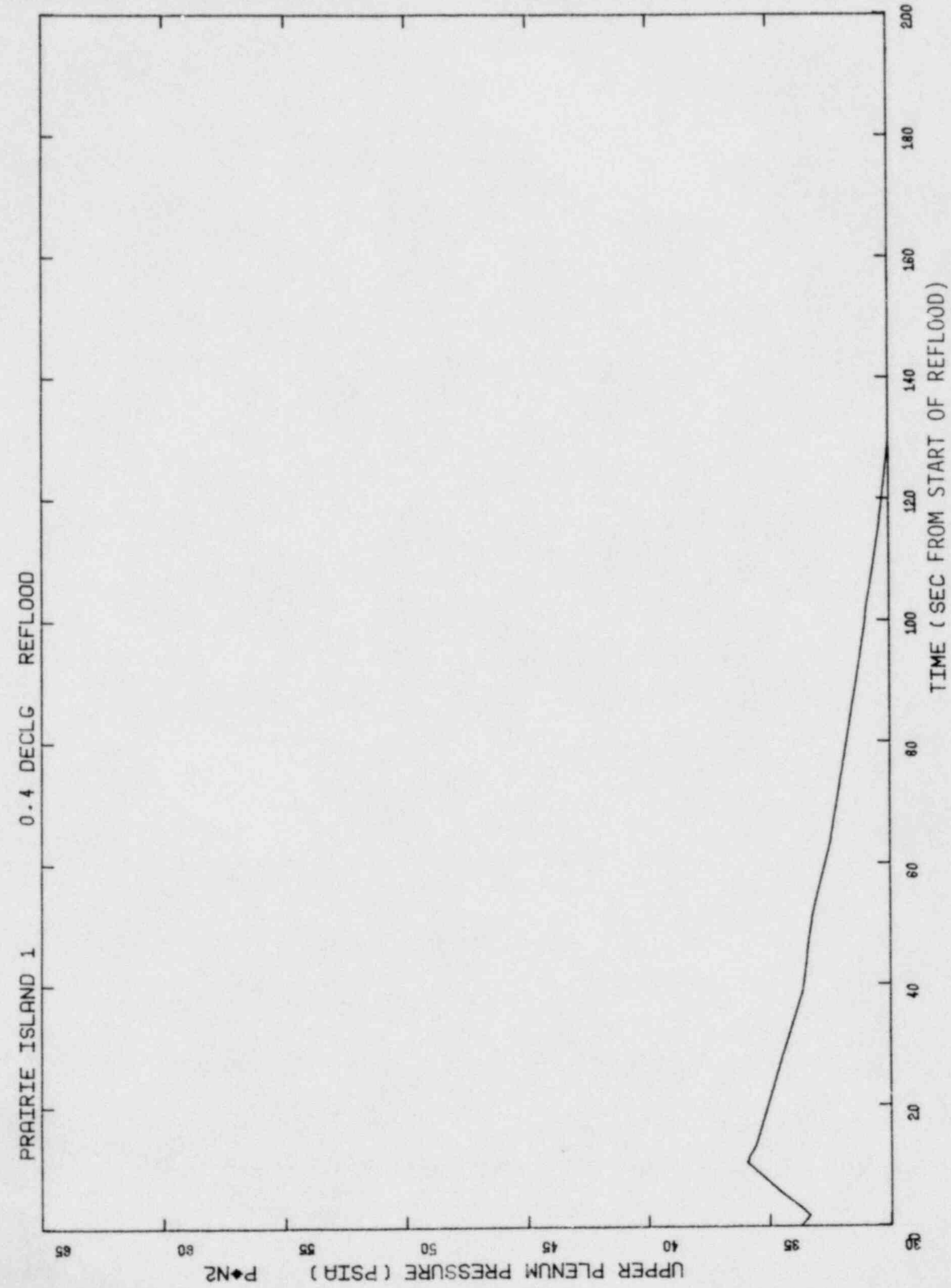


FIGURE 2.18 REFLOOD SYSTEM PRESSURE, 0.4 DECLG BREAK

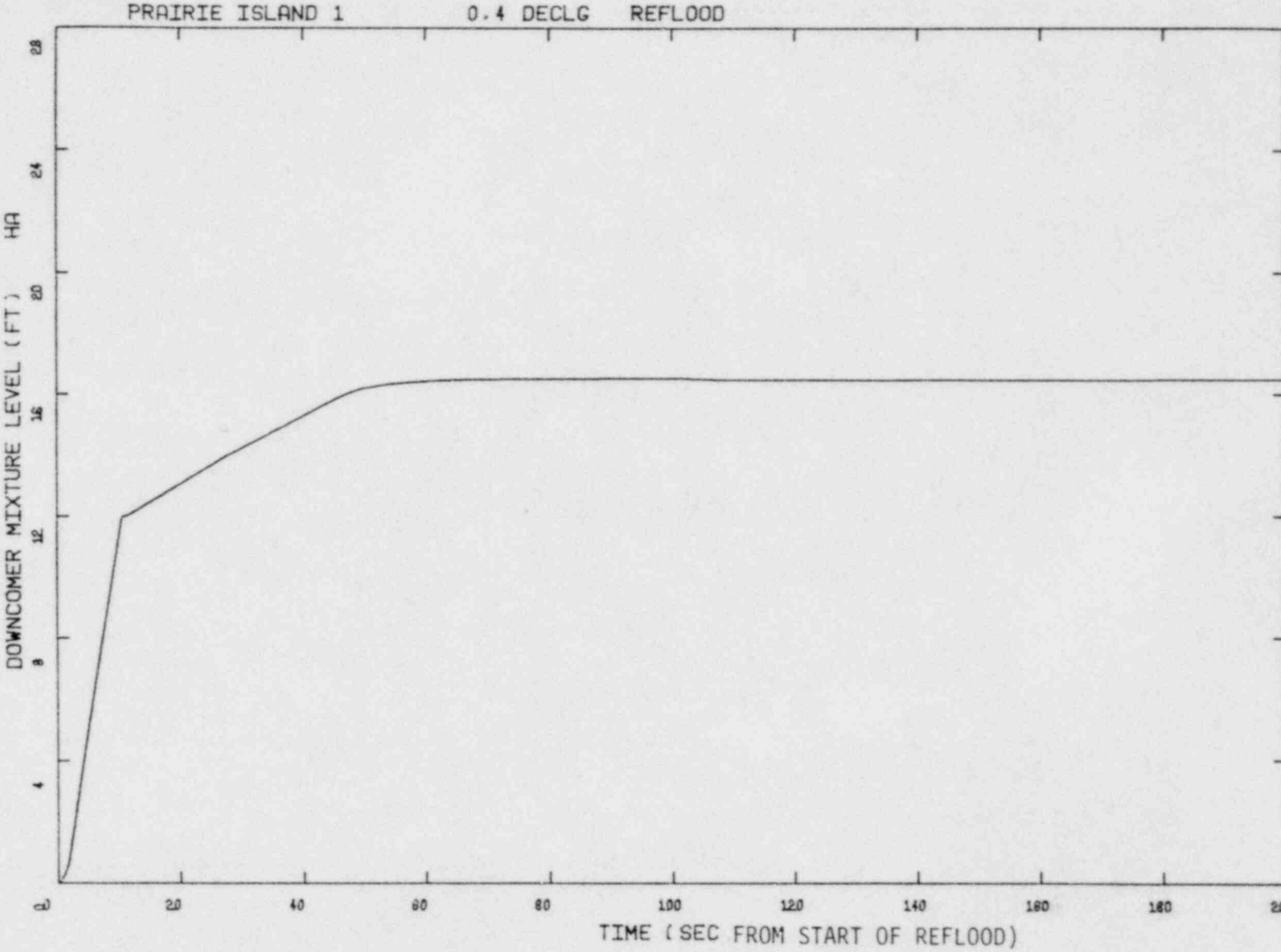


FIGURE 2.19 REFLOOD DOWNCOMER MIXTURE LEVEL, 0.4 DECLG BREAK

PLOTTED ON 18/09/78

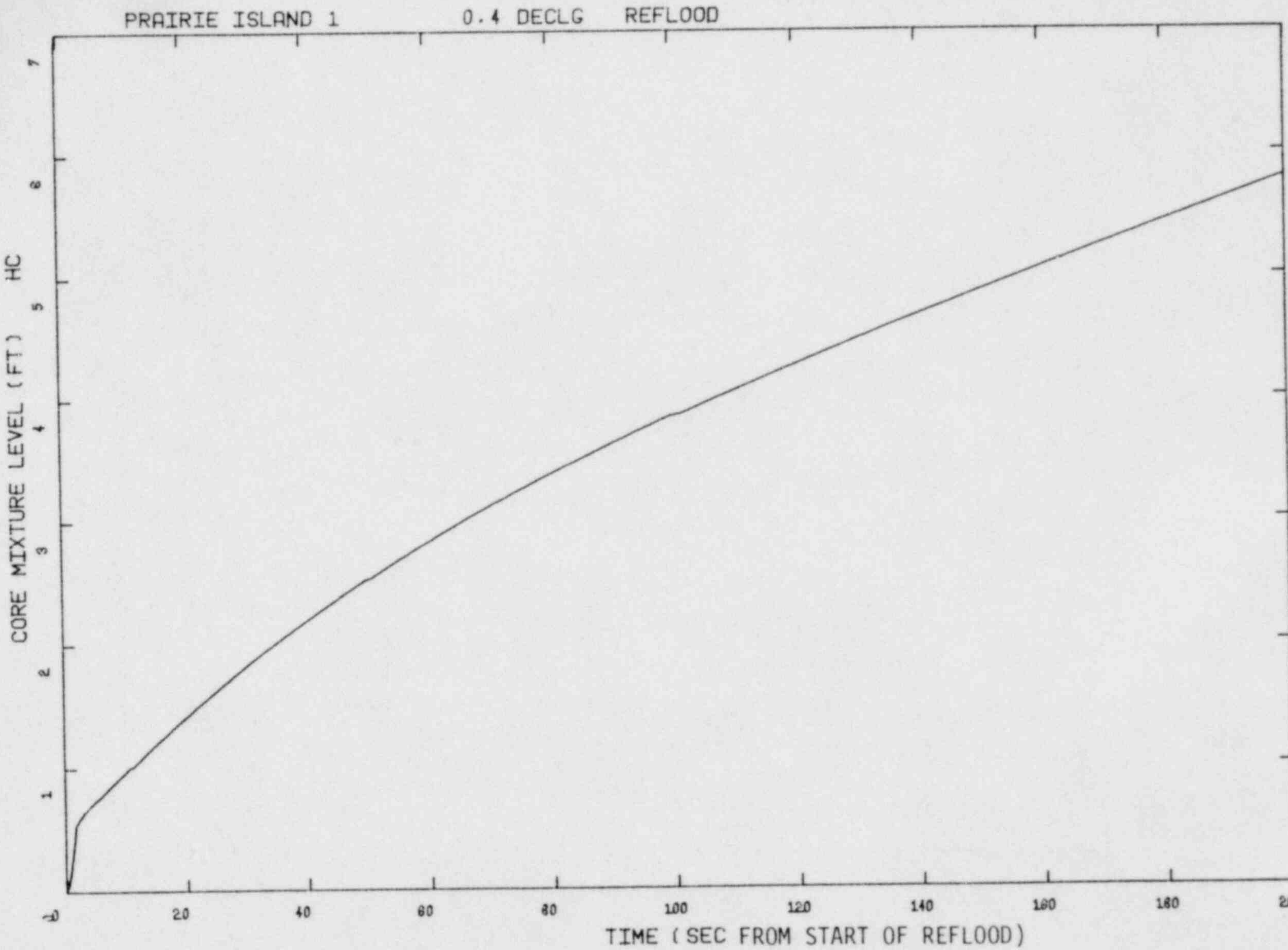


FIGURE 2.20 REFLOOD CORE MIXTURE LEVEL, 0.4 DECLG BREAK

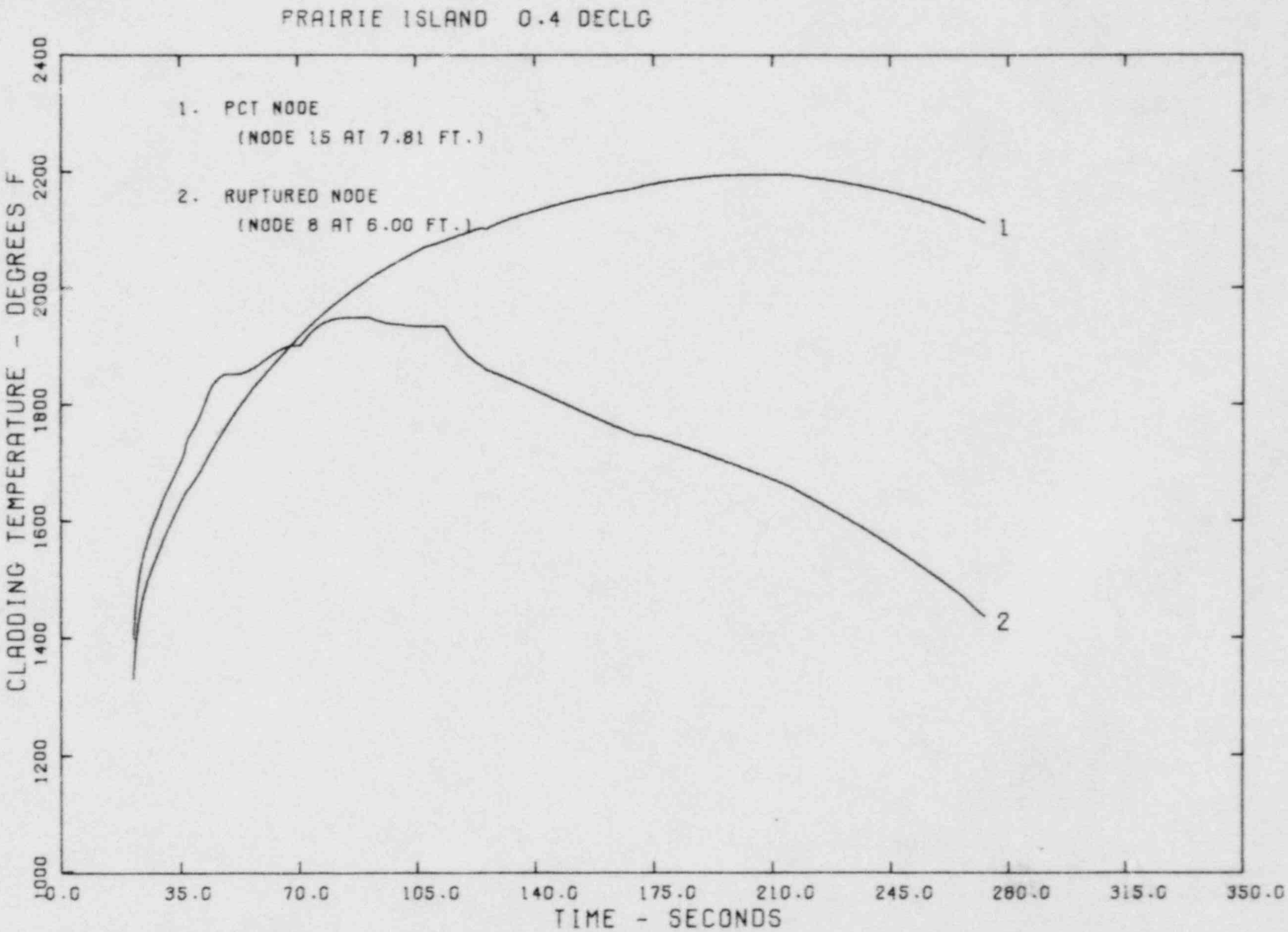


FIGURE 2.21 TOODEE2 CALCULATED CLADDING SURFACE TEMPERATURE, 0.4 DECLG BREAK

3.0 INTERIM UPPER PLENUM INJECTION MODEL AND RESULTS

3.1 INTERIM UPI MODEL CHANGES

The interim UPI analysis applied to the Prairie Island Unit 1 reactor follows the approach presented in the DSS SER "Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two-Loop Plants"[20]. The computer program representing the NRC Staff model was obtained and verified against the NRC Staff sample results. The computer program was then modified as required to represent the Prairie Island reactor and incorporate the base ENC ECCS analysis results. Modifications to the model include the following:

(1) Capability to use time-varying data from the base ENC WREM-IIA ECCS analysis was added for the following parameters: Containment pressure, saturation temperature, decay power, reflood rate, and low pressure safety injection (LPSI) flow and subcooling.

(2) The carry over rate fraction value used was 0.7 [21, 22].

(3) An upper plenum structures conduction heat transfer conduction model was incorporated in the program.

(4) The clad temperature rise versus flooding rate curve. Figure 24 in Reference 20, was replaced with the updated data provided in Figure 2 from Reference 21. The updated curve is more representative of the peak rod power, initial rod temperature, reflood subcooling, and system pressure for a two-loop PWR.

(5) The horizontal entrainment was taken as a constant fraction of the LPSI injection flow (1.6%) [20].

(6) The core flow area was changed as appropriate to reflect all ENC fuel in the Prairie Island Unit 1 reactor.

(7) The core heat capacity was revised to be consistent with the fraction of the core being quenched by UPI downflow.

3.2 INTERIM UPI MODEL RESULTS

Applying the above model to the Prairie Island reactor with Exxon Nuclear fuel gives the following results:

ENC WREM-IIA Analysis for Prairie Island 0.4 DECLG Break		Revised Analysis with UPI
Peak Cladding Temperature	F_Q^T	Peak Cladding Temperature
2197°F	2.21	2198°F

These results show continued compliance with 10CFR 50.46 and Appendix K to 10CFR Part 50, and are considered conservative since the adverse effects of UPI steam generation have been considered in reducing reflood rates, while the benefits of UPI steam generation in reducing core temperatures have been neglected.

4.0 MODEL HISTORY

The following section presents the genealogy of the models used in the subject analysis.

4.1 GENEALOGY OF MODELS

The Prairie Island Unit 1 ECCS analysis was performed with the following code versions.

Blowdown	∿ RELAP4-EM/ENC28C
Hot Channel	∿ RELAP4-EM/ENC28C
Reflood	∿ REFLEX
Heatup	∿ TOODEE2/APR78
Containment	∿ CONTEMPT-LT/VERSION 22

RELAP4-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.

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.

REFLEX - The REFLEX code is described in XN-NF-78-30 [3] which is the ENC WREM-IIA document.

TOODEE2/APR78 - This version incorporated the ENC WREM-II models into ENC master code version, TOODEE2/JAN77. These code modifications are described in XN-NF-77-27 and in XN-NF-77-58. These code modifications included logic to facilitate the calculations required in conjunction with the sensitivity

studies required by the NRC for the rupture pressure uncertainty analysis. The following AVAIL groups were modified or added to TOODEE2:

AVAIL (78)		FLECHT multipliers option
	0.0	use FLECHT correlation multipliers
	1.0	use FLECHT/ENC2-WREM I correlation multipliers
	2.0	use FLECHT/ENC3-WREM-II correlation multipliers
AVAIL (82)		Gas volume in fuel (dishes and cracks)
AVAIL (83)		Heat transfer coefficient multiplier (default 1.0)
AVAIL (84)		Prerupture strain constant (default 0.20)
AVAIL (85)		Blockage model switch
	1.0	ENC WREM-I model (must also set AVAIL (78) = 1.0)
	2.0	ENC WREM-II model (default valve)

In addition, the code I/O was modified such that additional data at the time of fuel rupture is printed (fraction of flow area blocked, rupture pressure, rupture temperature, temperature and volume of the fuel plenum, and average temperature and volume for the gas in the cracks, dishes, and gap.) These changes do not affect calculated results.

CONTEMPT-LT/VERSION22 - The RELAP4-EM environmental package was added to allow input data to be submitted as free format as a convenience to the user. None of the code models were changed, and no changes in the calculated results occurred.

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   c. Volume III, Revision 2, August 1975
   d. Supplement 1, August 1975
   e. Supplement 2, August 1975
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   g. Supplement 4, August 1975
   h. Supplement 5, Revision 5, Octover 1975
   i. Supplement 6, October 1975
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2. Exxon Nuclear Company, Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II, XN-76-27:

   a. July 1976
   b. Supplement 1, September 1976
   c. Supplement 2, November 1976

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5. Lilly, G. P., Mixing of Emergency Core Cooling Water with Steam: 1/3 - Scale Test and Summary, EPRI 294-2 Final Report, June 1975.

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14. U.S. Nuclear Regulatory Commission, Minimum Containment Pressure Model for PWR ECCS Performance Evaluation, Branch Technical Position CSB 6-1.

15. 10 CFR 50.46 and Appendix K of 10 CRF 50, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors, Federal Register, Volume 39, Number 3, January 4, 1974.

16. Exxon Nuclear Company, Exxon Nuclear Company ECCS Evaluation of a 2-Loop Westinghouse PWR With Dry Containment Using the ENC WREM-II ECCS Model - Large Break Example Problem, XN-NF-77-25(A), September 1978.

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21. Letter from L. O. Mayer to Director of Nuclear Reactor Regulation, February 24, 1978 (Docket No. 50-282 and 50-306).

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