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U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Small Break LOCA Analysis

We have completed a new Small Break Loss of Coolant Accident (LOCA) analysis. This analysis is being submitted per your request in Inspection Report 90015 and 90016 (dated October 26 1990). The results of the analysis are document in Attachment 1. The analysis is documented in the form of a draft Updated Safety Analysis Report section (this report section is a draft but the analysis results are final). This attachment will be incorporated into the next revision to the Updated Safety Analysis Report (June, 1992).

This accident was reanalyzed to correct an input error in auxiliary feedwater flow and several analysis concerns. The new peak cladding temperature is 1077°F.

Please contact us if you have questions concerning these reports.

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Manager
Nuclear Support Services

c: Regional Administrator Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
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Attachments:
1. Draft Updated Safety Analysis Report Section 14.7

A001

14.7 Loss of Reactor Coolant from Small Ruptured Pipes or From Cracks in Large Pipes which Actuate the Emergency Core Cooling System

14.7-1 Acceptance Criteria

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, an infrequent fault.

The Acceptance Criteria for the loss-of-coolant accident is described in 10 CFR 50.46 as follows:

(a) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(b) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(c) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(d) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(e) After any calculated sucessful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA.

14.7.2 Description of Small Break LOCA Transient

Ruptures of small cross-section will cause loss of the coolant at a rate which can be accomodated by the charging pumps. These pumps would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing at equilibrium.

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 through the postulated break against the charging pump makeup flow at normal Reactor Coolant System pressure; i.e., 2250 psia. A makeup flow rate from one charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 0.375 inch diameter hole. This break results in the loss of approximately 17.25 lbm/sec.

Should a larger break occur, depressurization of the Reactor Coolant System causes fluid into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the low pressurizer pressure trip setpoint is reached. During the early part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The Safety Injection System is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

1. Control rod insertion and void formation in the core cause a rapid reduction of the nuclear power to a residual level corresponding to the delayed fission and fission product decay.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the Reactor Coolant System. The heat transfer between the Reactor Coolant System and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater isolation valves and initiates auxiliary feedwater flow by starting auxiliary feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressures.

When the RCS depressurizes to the accumulator cover gas pressure, the cold leg accumulators begin to inject water into the reactor coolant loops. Due to the loss of offsite power assumption, the reactor coolant pumps are asssumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analyses.

14.7.3 Small Break LOCA Evaluation Model

The NOTRUMP computer code is used in the analysis of loss-of-coolant accidents due to small breaks in the reactor coolant system. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP small break, LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of Coolant Accidents in Westinghouse-Designed Operating Plants."

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of NOTRUMP is given in WCAP-10054-P-A and WCAP 10079-P-A.

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

Cladding thermal analyses are performed with the LOCTA-IV WCAP-8301-P code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations, as input.

A schematic representation of the computer code interfaces is given in Figure 14.7-1.

This model was developed to resolve TMI Action Item II.K.3.30. NRC acceptance of this model for Prairie Island was documented in an NRC staff letter dated June 6, 1985.

14.7-4 Small Break Input Parameters and Initial Conditions

Table 14.7-1 lists important input parameters and initial conditions used in the small break analysis.

The axial power distribution and core decay power assumed for the small break analysis are shown in Figures 14.7-2 and 14.7-3.

Safety injection flow rate to the Reactor Coolant System is a function of the system pressure is used as part of the input. The Safety Injection System (SI) was assumed to be delivering to the RCS 25 seconds after the generation of a safety injection signal.

For this analysis, the SI delivery considers injection flow which is depicted in Figure 14.7-4 as a function of RCS pressure. This figure represents injection flow from one degraded SI pump spilling to either RCS pressure (if break size is smaller than SI injection line diameter), or to 0 psig containment pressure (if break size is greater than or equal to the SI injection line diameter). The 25 second delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency buses. Flow from the RHR pumps does not affect the analysis since their shutoff head is lower than RCS pressure during the time portion of the transient considered here. Also, minimum safeguards Emergency Core Cooling System capability and operability has been asssumed in this analysis.

The hydraulic analyses are performed with the NOTRUMP code using 102% of the licensed core power. The core thermal transient analyses are performed with the LOCTA-IV code using 102% of the licensed core power.

14.7.5 Small Break Results

As noted previously, the calculated peak cladding temperature resulting from a small break LOCA is less than that calculated for a large break LOCA. A range of small break analyses are presented which establishes the limiting break size. The results of these analyses are summarized in Tables 14.7-2 and 14.7-3. Figures 14.7-5 through 14.7-11 present the principal parameters of interest for the small break ECCS analyses. For the cases analyzed, the following transient parameters are included except hot spot clad temperature when the core is not uncovered.

a. RCS Pressure
b. Core mixture height
c. Hot spot clad temperature

For the limiting break analyzed (6 inch), the following additional transient parameters are present (Figures 14.7-12 through 14.7-14):

a. Core steam flow rate
b. Core heat transfer coefficient
c. Hot spot fluid temperature

The maximum peak clad temperature for the breaks analyzed is 1077°F. These results are well below all Acceptance Criteria limits of 10 CFR 50.46, and in no case is limiting when compared to the results presented for large breaks.

TABLE 14.7-1

INPUT PARAMETERS USED IN THE SMALL BREAK LOCA ANALYSIS

Parameter	Input
Core Power	102% of 1650 MWt
Peak Linear Power (kW/ft) (includes 102% factor)	15.096 kW/ft
Total Peaking Factor	2.50
Power Shape	See Figure 14.7-2
Fuel Assembly Array	14X14 OFA
Accumulator Conditions:	
Cover Gas Pressure	710 psig
Water Volume	1250 ft^3
Total Volume	2000 ft^3
Pumped Safety Injection Flow	See Figure 14.7-4
Steam Generator Initial Pressure	664 psia
Steam Generator Tube Plugging Level	10%
Reactor Trip Signal	1700 psia
Safety Injection Signal	1700 psia
Rod Drop Time	2.4 seconds
Reactor Trip Signal Delay Time	2.0 seconds

TABLE 14.7-2

SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

EVENT	Break Size		
	4.0 Inch	6.0 Inch	8.0 Inch
Break Initiation, sec.	0.0	0.0	0.0
Reactor Trip Signal, sec.	8.0	5.6	5.2
Safety Injection Signal, sec.	8.0	5.6	5.2
Top of Core Uncovered, sec.	~160	~140	~78
Accumulator Injection Begins, sec.	~350	~150	~80
Peak Clad Temperature Occurs, sec.	~178	~192	~116
Top of Core Recovered, sec.	~180	~195	~122

TABLE 14.7-3

SMALL BREAK LOCA ANALYSIS RESULTS

RESULT	Break Size		
	4.0 Inch	6.0 Inch	8.0 Inch
Peak Clad Temperature, °F	834	1077	1053
Peak Clad Temperature Location, ft.	10.5	10.75	10.5
Local Zr/H_2O Reaction (max), %	0.0333	0.0339	0.0337
Local Zr/H_2O Reaction Location, ft.	10.5	10.75	10.5
Hot Rod Burst Time, seconds	NA	NA	NA
Hot Rod Burst Location, ft.	NA	NA	NA

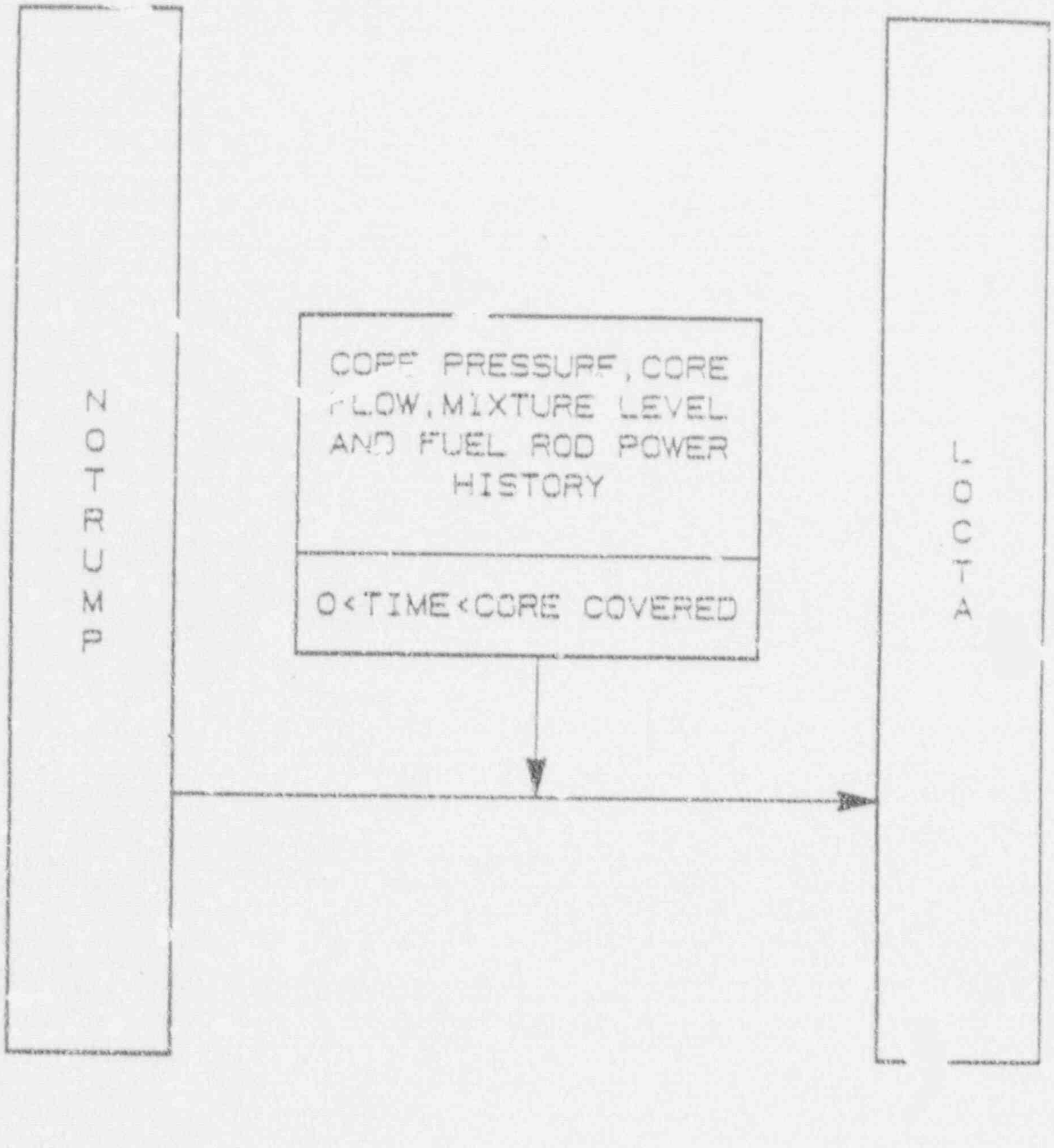
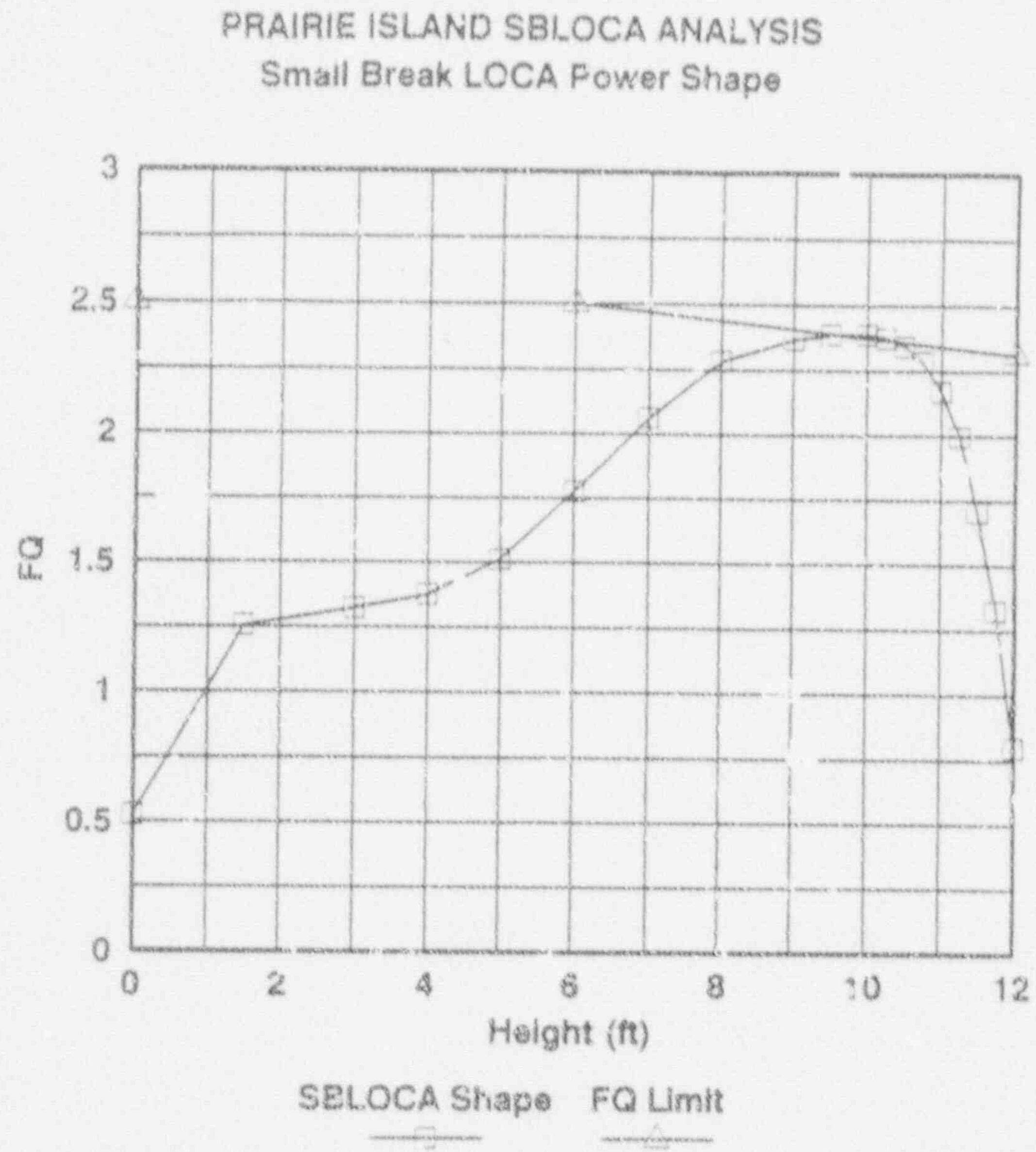


Figure 14.7-1 Code Interface Description for Small Break Model



PRAIRIE ISLAND UNITS 1 AND 2	SBLOCA POWER SHAPE
	FIGURE 14.7-2

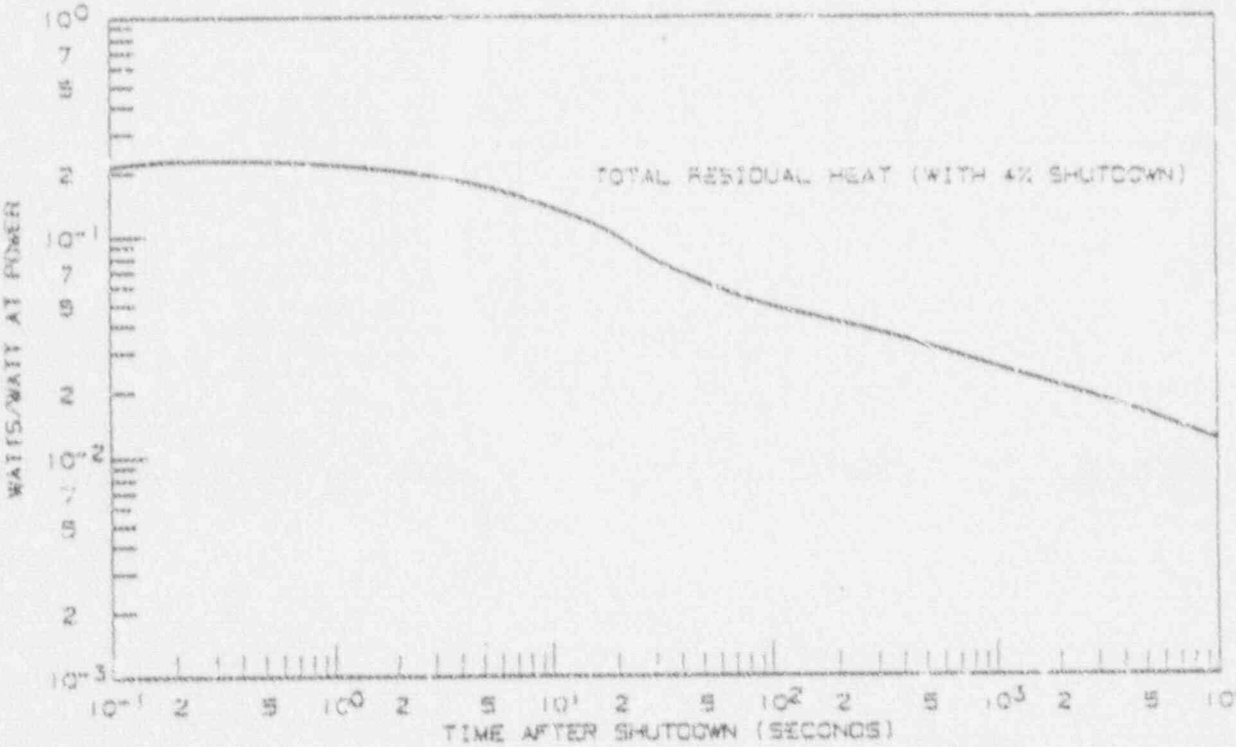
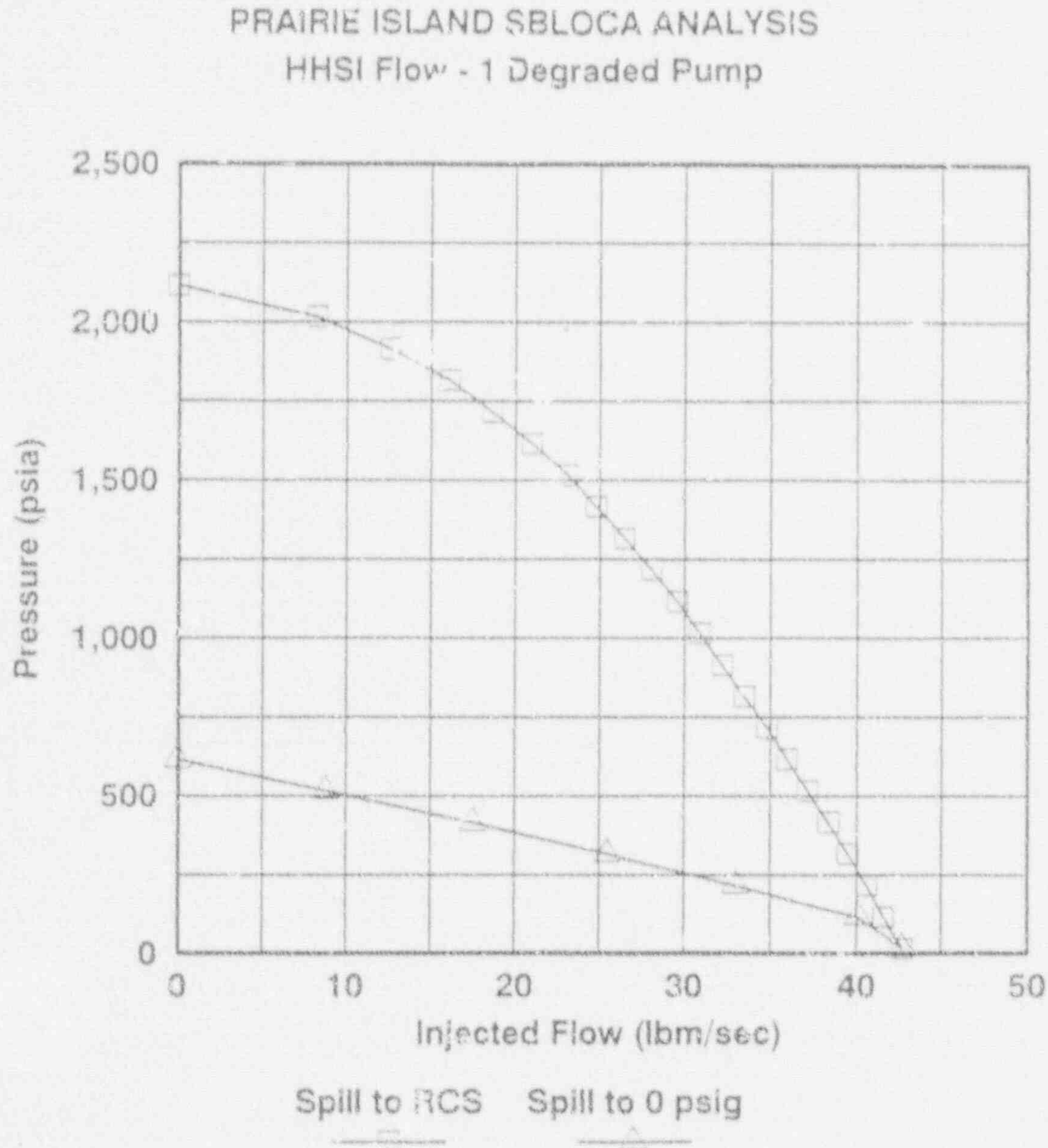
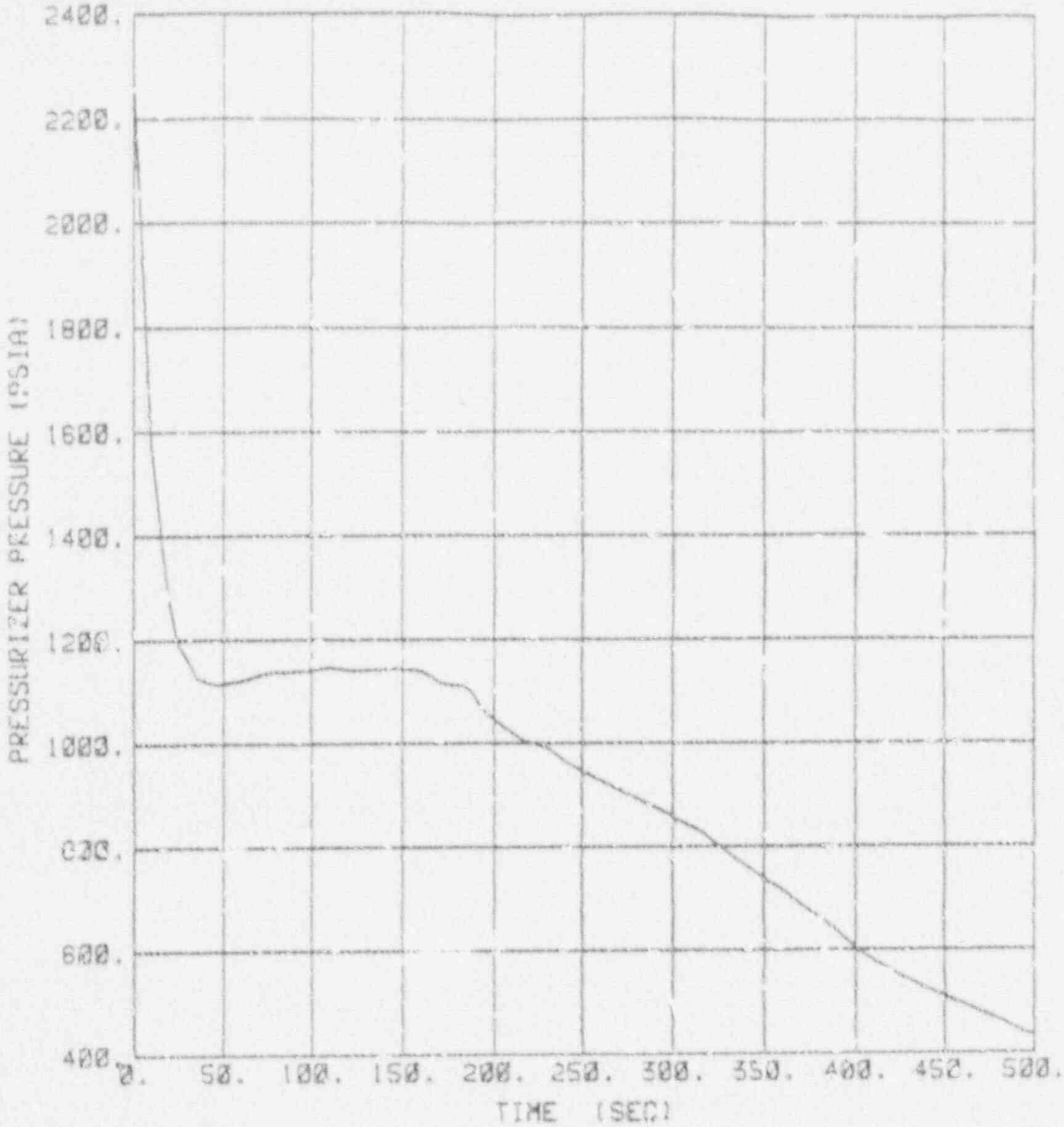


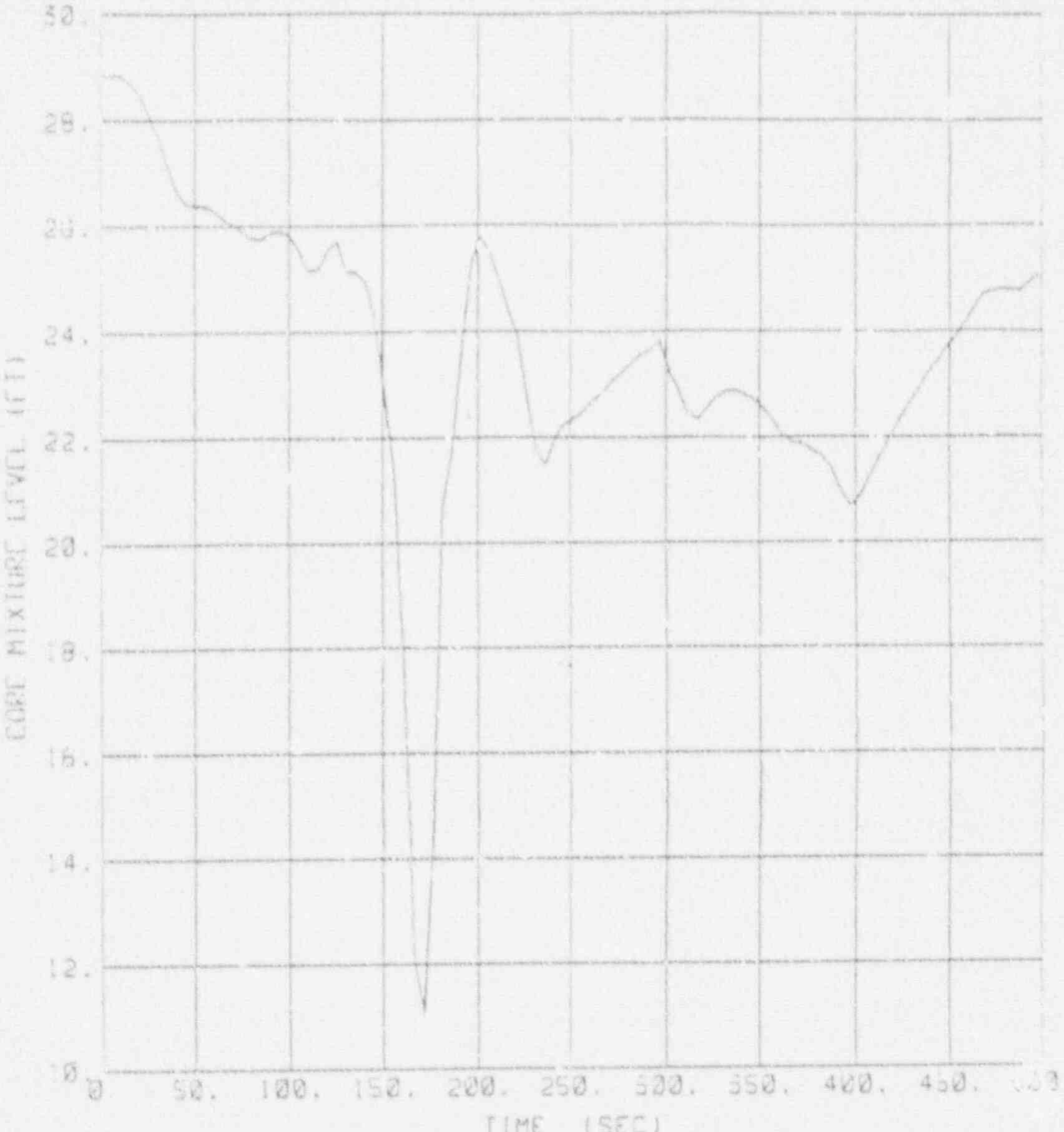
Figure 14.7-3 Core Power After Reactor Trip (Applies to all Small Breaks)



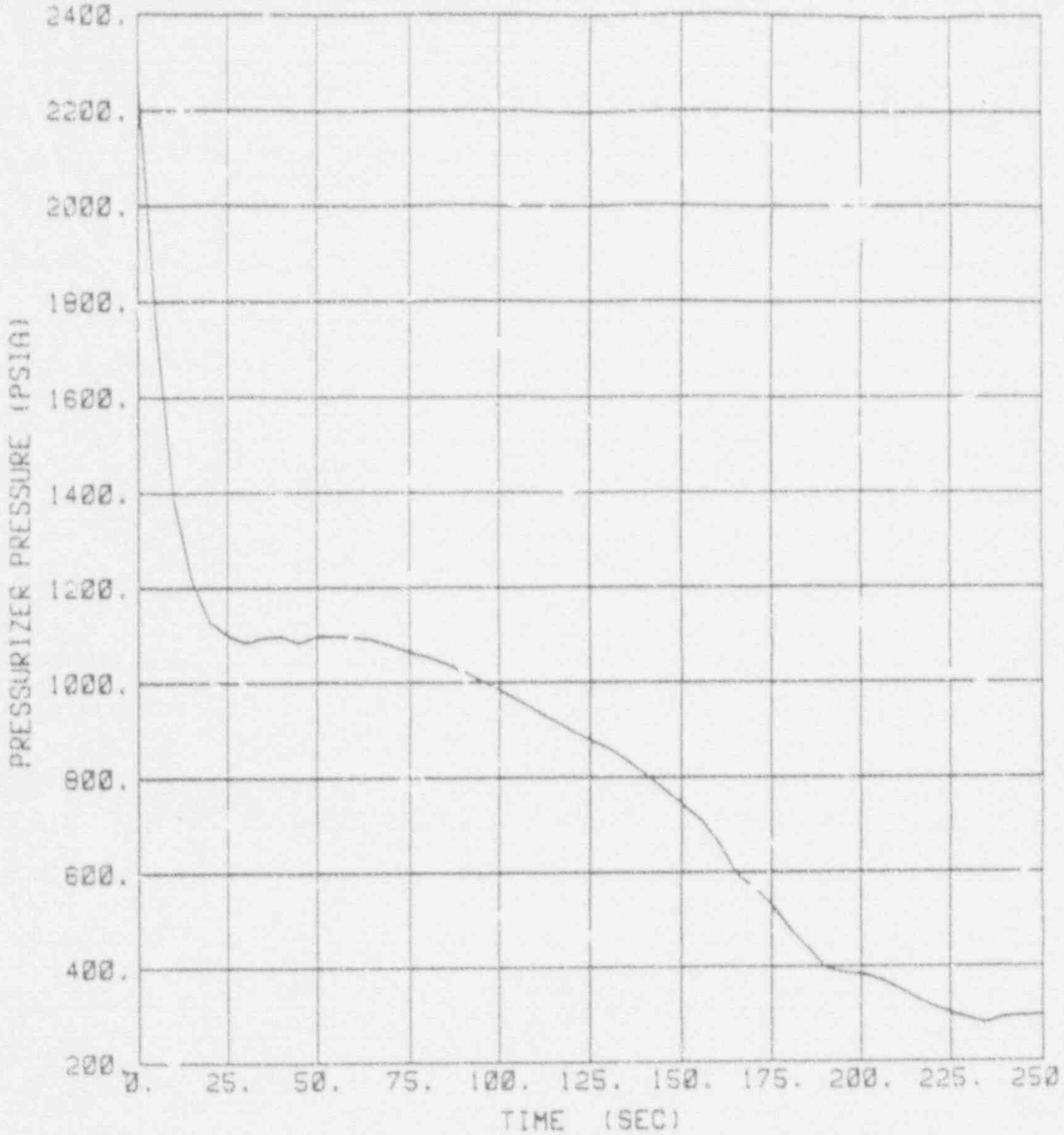
PRAIRIE ISLAND UNITS 1 AND 2	HHSI FLOW RATE
	FIGURE 14.7-4



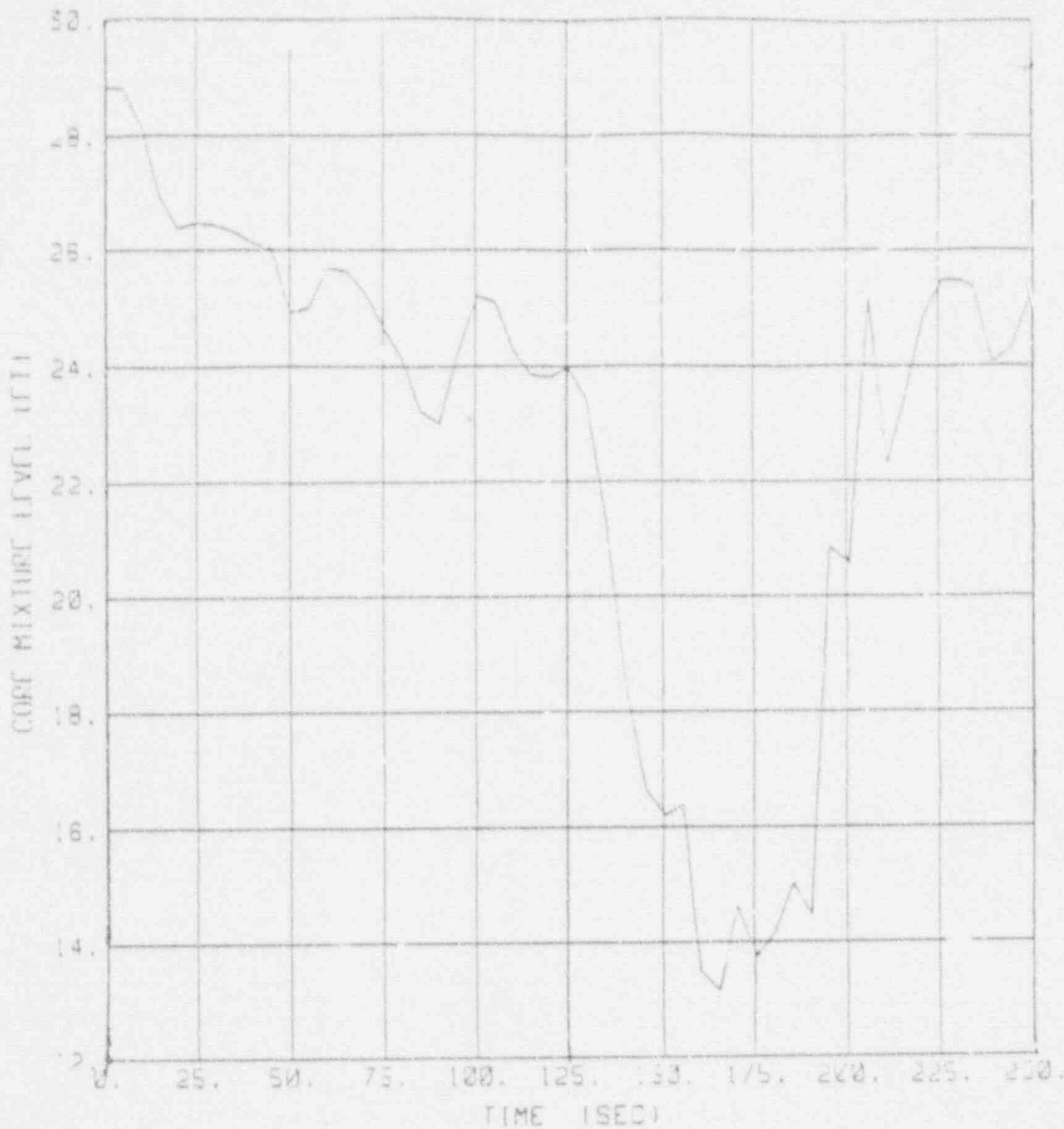
PRAIRIE ISLAND UNITS 1 AND 2	4.0" SMALL BREAK LOCA RCS PRESSURE
	FIGURE 14.7-5



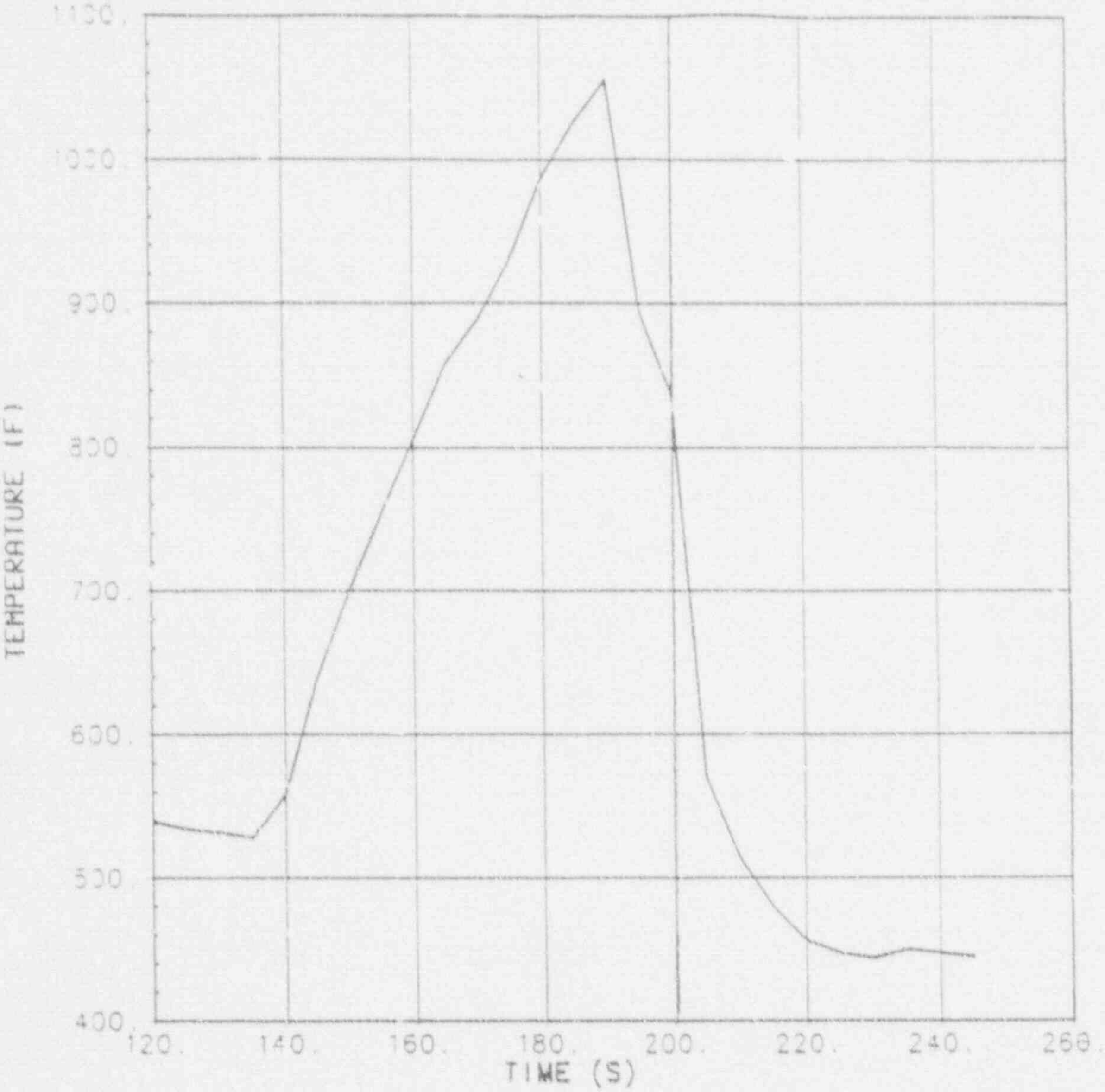
PRAIRIE ISLAND UNITS 1 AND 2	4.0" SMALL BREAK LOCA CORE MIXTURE LEVEL
	FIGURE 14.7-6



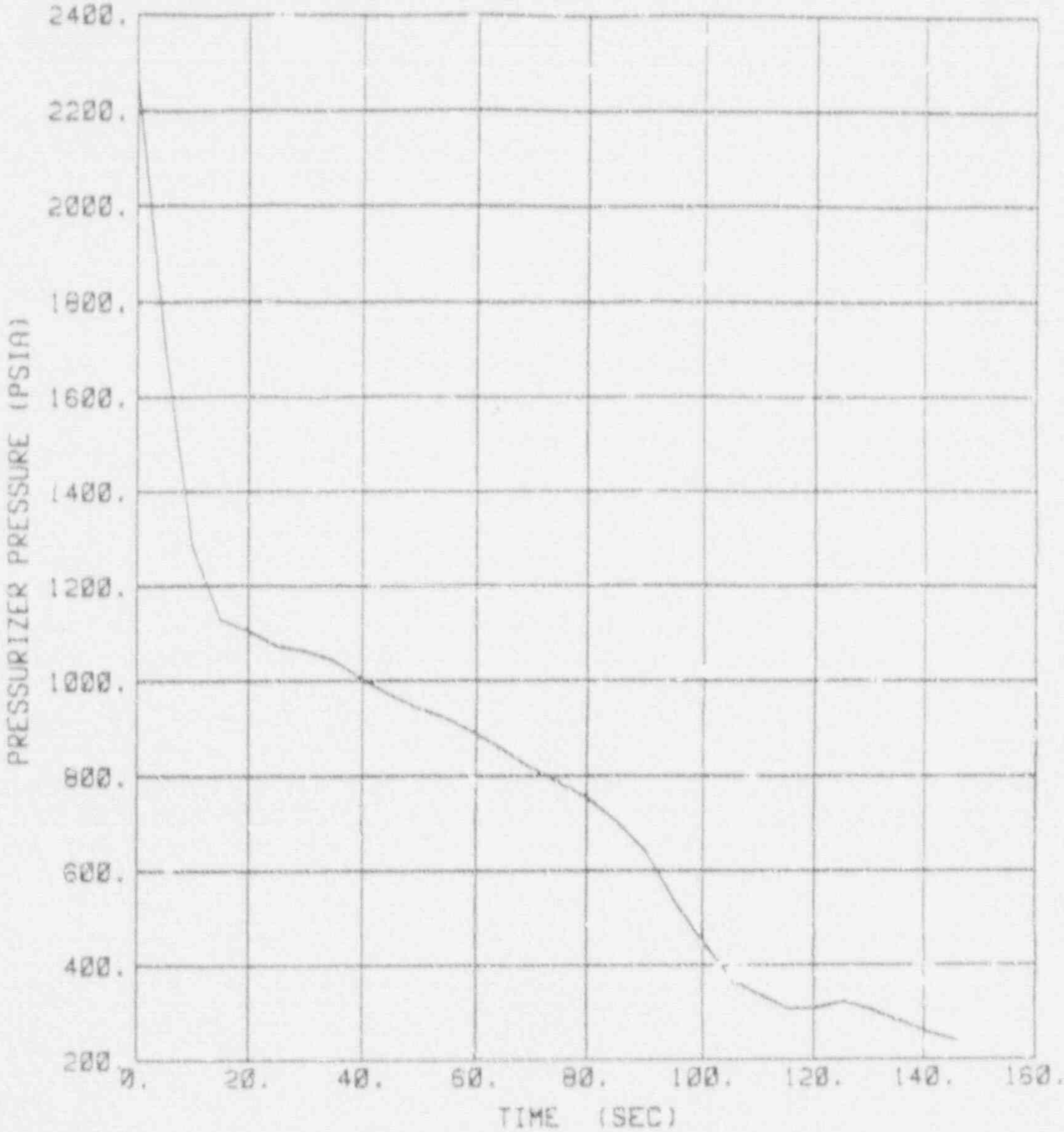
PRAIRIE ISLAND UNITS 1 AND 2	6.0" SMALL BREAK LOCA RCS PRESSURE
	FIGURE 14.7-7



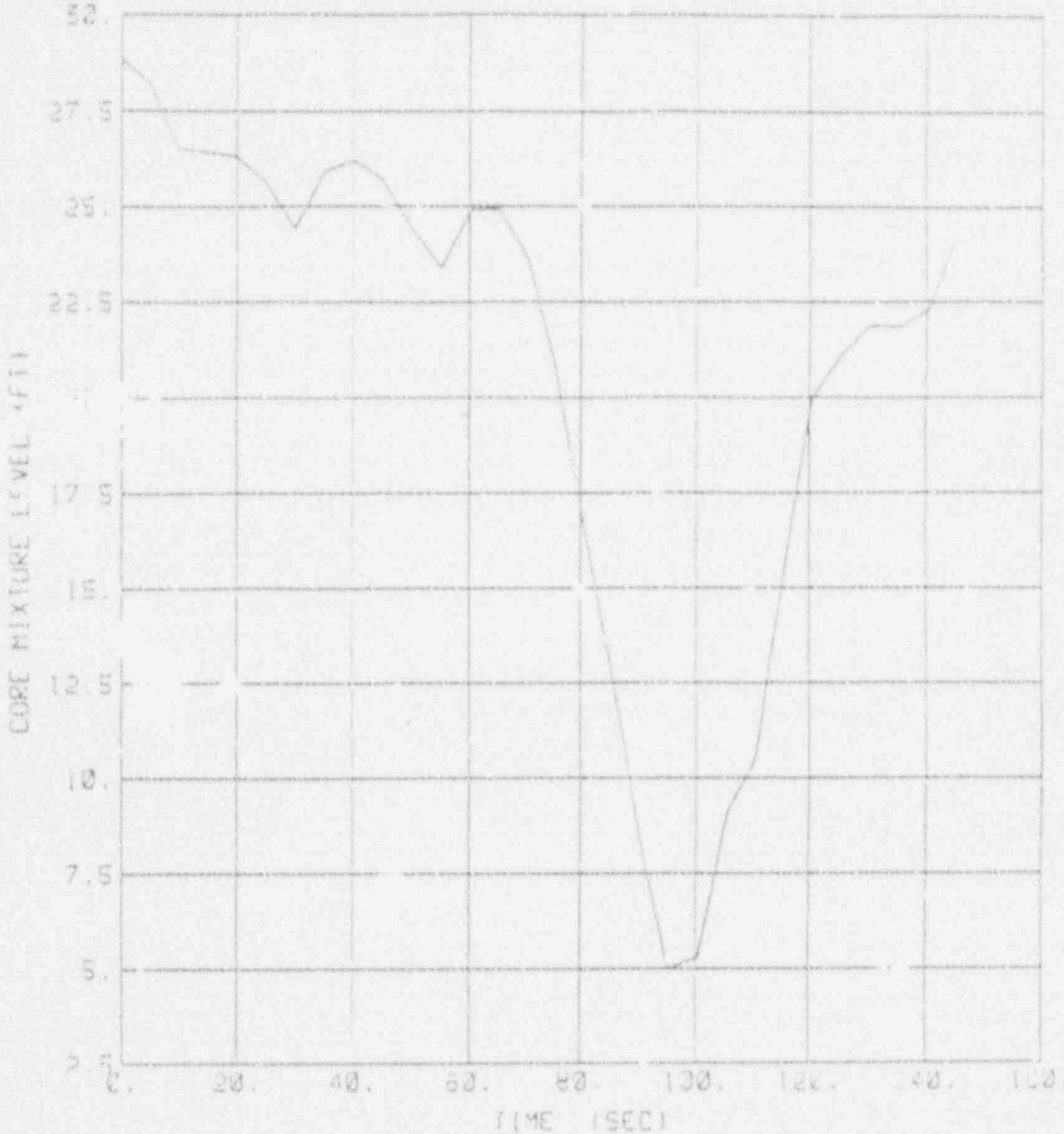
PRAIRIE ISLAND UNITS 1 AND 2	6.0" SMALL BREAK LOCA CORE MIXTURE LEVEL
	FIGURE 14.7-8



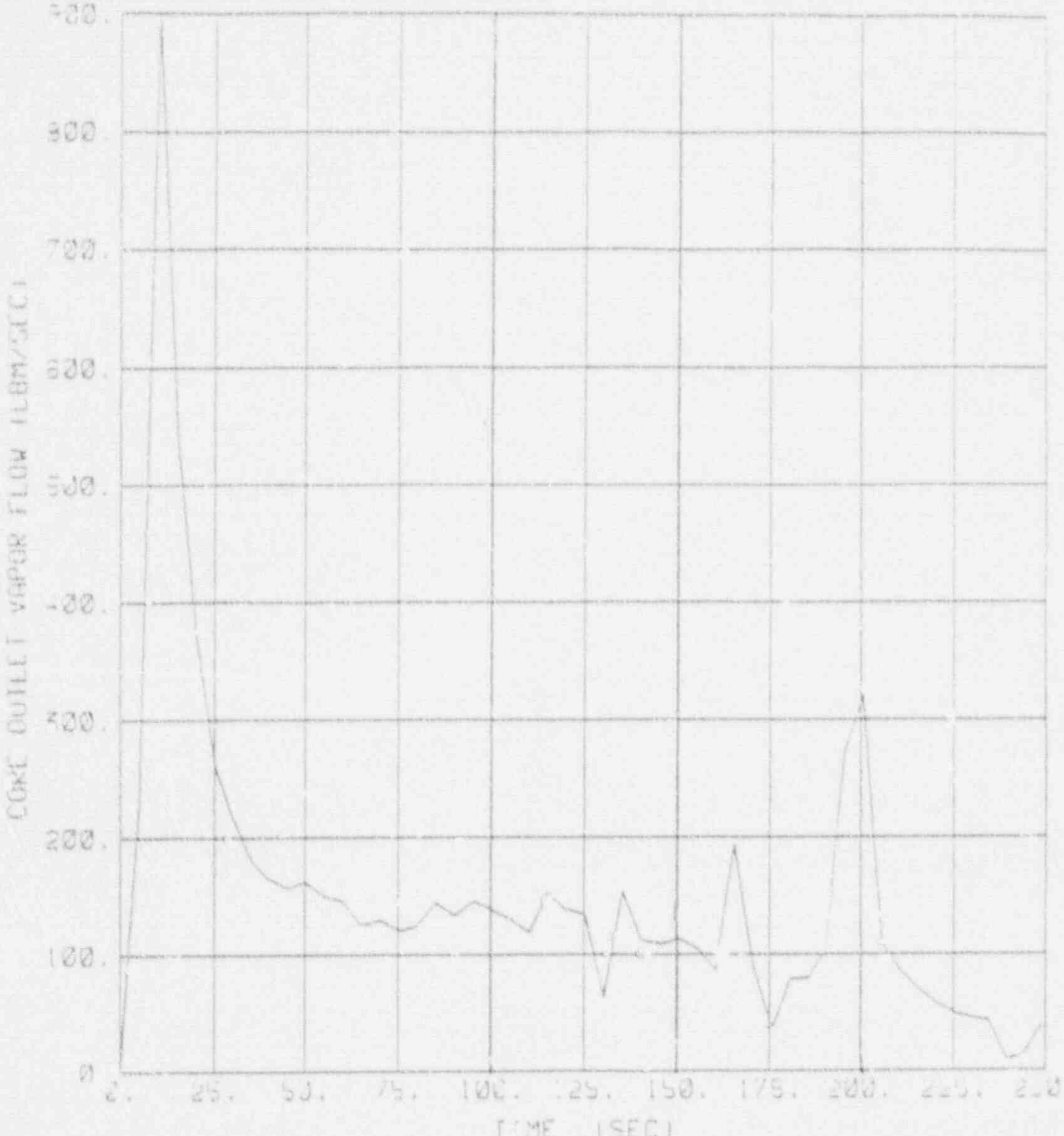
PRAIRIE ISLAND UNITS 1 AND 2	6.0" SMALL BREAK LOCA PEAK CLAD TEMPERATURE
	FIGURE 14.7-9



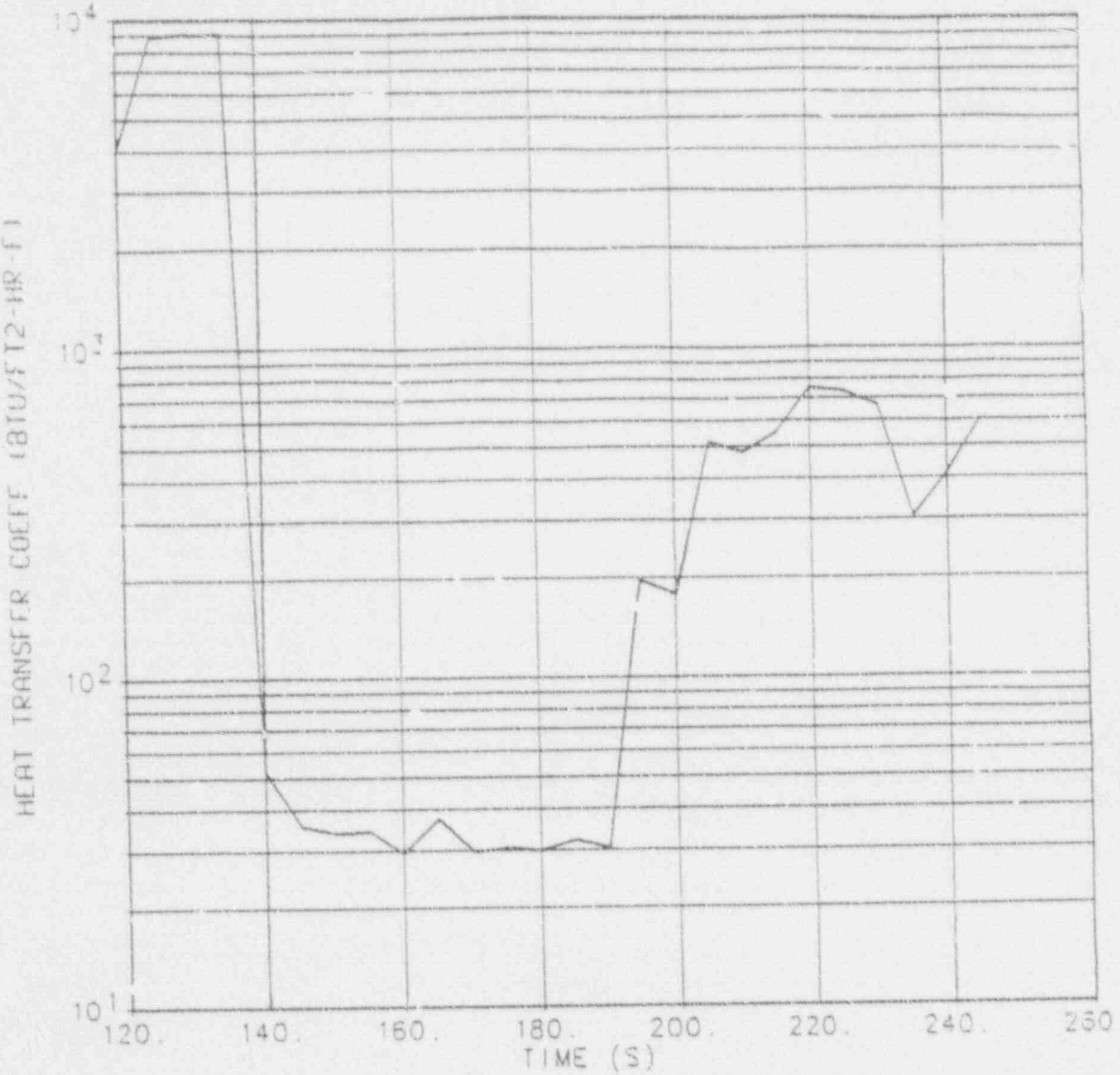
PRAIRIE ISLAND UNITS 1 AND 2	8.0" SMALL BREAK LOCA RCS PRESSURE
	FIGURE 14.7-10



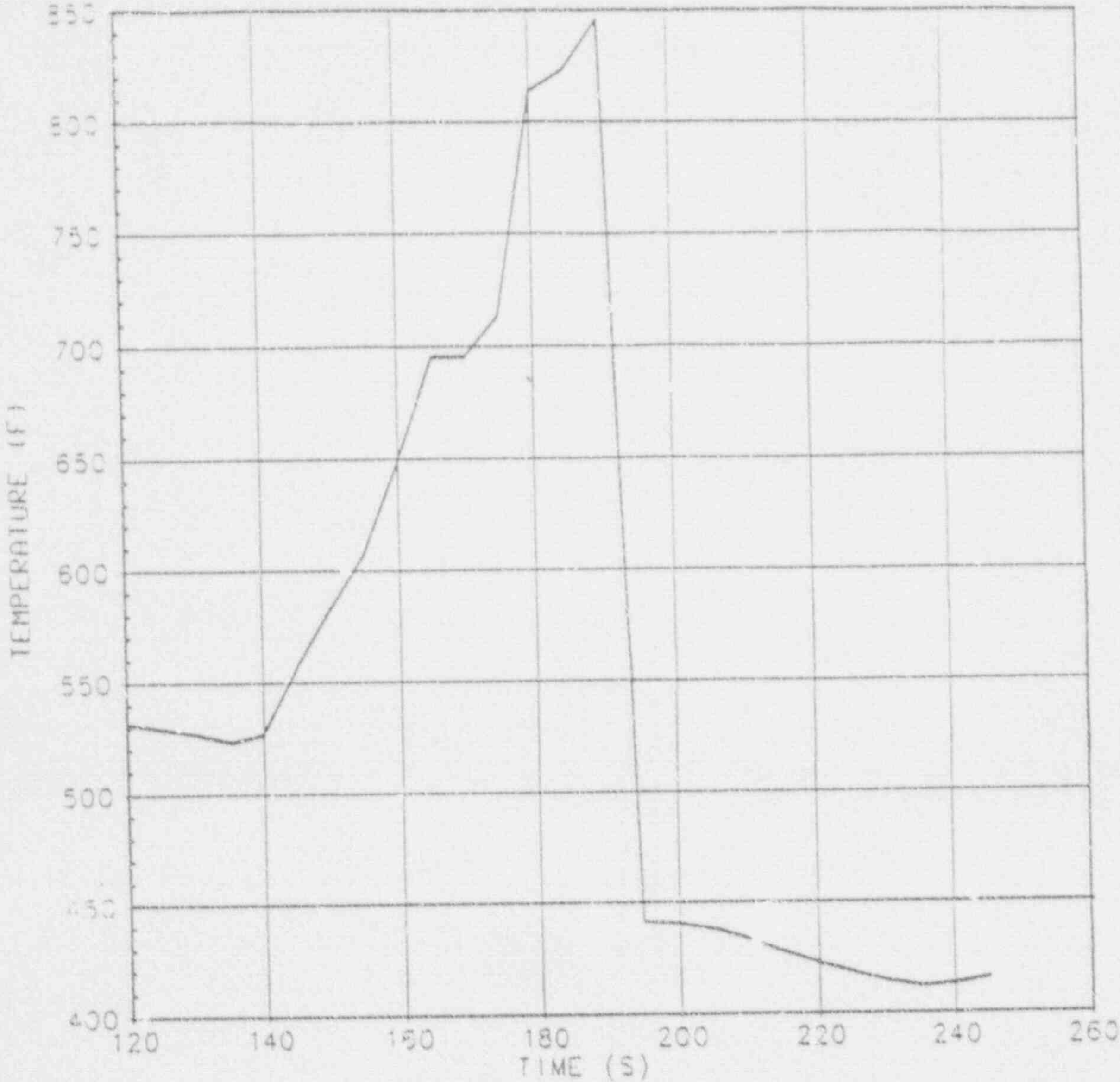
PRAIRIE ISLAND UNITS 1 AND 2	8.0" SMALL BREAK LOCA CORE MIXTURE LEVEL
	FIGURE 14.7-11



PRAIRIE ISLAND UNITS 1 AND 2	6.0" SMALL BREAK LOCA CORE OUTLET VAPOR FLOW FIGURE 14.7-12



PRAIRIE ISLAND UNITS 1 AND 2	6.0" SMALL BREAK LOCA HOT SPOT ROD SURFACE HEAT TRANSFER COEFFICIENT
	FIGURE 14.7-13



PRAIRIE ISLAND UNITS 1 AND 2	6.0" SMALL BREAK LOCA HOT SPOT FLUID TEMPERATURE
	FIGURE 14.7-14