THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
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Dalwyn R. Davidson
VICE PRESIDENT
SYSTEM ENGINEERING AND CONSTRUCTION

April 21, 1982

Mr. A. Schwencer
Chief, Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Response to Draft SER
Reactor Systems

Dear Mr. Schwencer:

The purpose of this letter is to forward Perry Plant specific overpressure protection analyses using the ODYN code and Perry Plant specific ECCS analyses.

These analyses will be incorporated in the next amendment to our Final Safety Analysis Report.

Very Truly Yours,

Dalwyn R. Davidson

Dalwyn R. Davidson
Vice President
System Engineering and Construction

DRD: mlb

cc: Jay Silberg
John Stefano
Max Gildner

B001
S/1

a. Feedwater line break

b. Steam system piping break outside of containment

c. Loss-of-coolant accidents

Chapter 15 provides the radiological consequences of the above listed events.

6.3.3.1 ECCS Bases for Technical Specifications

The maximum average planar linear heat generation rates calculated in this performance analysis provide the basis for technical specifications designed to ensure conformance with the acceptance criteria of 10 CFR 50.46. Minimum ECCS functional requirements are specified in Sections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed in Section 6.3.4. Limits on minimum suppression pool water level are discussed in Section 6.2.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors", are listed and for each criterion, applicable parts of Section 6.3.3 where conformance is demonstrated are indicated. A detailed description of the methods used to show compliance are shown in Reference 1.

Criterion 1, Peak Cladding Temperature

"The calculated maximum fuel element cladding temperature shall not exceed 2200°F Conformance to Criterion 1", is shown in Sections 6.3.3.7.3, 6.3.3.7.4, 6.3.3.7.5, 6.3.3.7.6 and specifically in Table 6.3-4 (maximum average planar linear heat generation rate, oxidation fraction, and peak cladding temperature versus exposure).

Criterion 2, Maximum Cladding Oxidation

"The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformance to Criterion 2 is shown in Figure 6.3-9 (break spectrum plot), in Table 6.3-4 (local oxidation versus exposure) and in Table 6.3-5 (break spectrum summary).

Criterion 3, Maximum Hydrogen Generation

"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 the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is shown in Table 6.3-5.

Criterion 4, Coolable Geometry

"Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 1, Section III.A, conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2.

Criterion 5, Long-Term Cooling

"After any calculated successful 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." Conformance to Criterion 5 is demonstrated generically for General Electric BWRs in Reference 1, Section III.A. Briefly summarized, the core remains covered to at least the jet pump suction elevation and the uncovered region is cooled by spray cooling and/or by steam generated in the covered part of the core.

6.3.3.3 Single Failure Considerations

The functional consequences of potential operator errors and single failures, (including those which might cause any manually controlled electrically operated valve in the ECCS to move to a position which could adversely affect the ECCS) and the potential for submergence of valve motors in the ECCS are discussed in Section 6.3.2. There it was shown that all potential single failures are no more severe than one of the single failures identified in Table 6.3-3.

It is therefore only necessary to consider each of these single failures in the emergency core cooling system performance analyses. For large breaks, failure of one of the diesel standby generators is in general the most severe failure. For small breaks, the HPCS is the most severe failure.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as:

a. receiving an initiation signal,

b. a small lag time (to open all valves and have the pumps up to rated speed), and

c. the ECCS flow entering the vessel.

Key ECCS initiating signals and time delays for all the ECC systems are provided in Table 6.3-1. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on accelerating the diesel-generators and pumps. The delay time due to valve motion in the case of high pressure system provides a suitably conservative allowance for valves available for this application. In the case of the low pressure system, the time delay for valve motion is such that the pumps are at rated speed prior to the time the vessel pressure reaches the pump shutoff pressure.

slower, being governed by decay heat and core spray heat transfer. Finally the heatup is terminated when the core is recovered by the accumulation of ECCS water.

6.3.3.7.3 Break Spectrum Calculations

A complete spectrum of postulated break sizes and locations is considered in the evaluation of ECCS performance. The general analytical procedures for conducting break spectrum calculations are discussed in Section III.B of Reference 1. For ease of reference, a summary of all figures presented in Section 6.3.3 is shown in Table 6.3-6.

A summary of the results of the break spectrum calculations is shown in tabular form in Table 6.3-5 and graphically in Figure 6.3-9. Conformance to the acceptance criteria (PCT of 2200°F, local oxidation of 17 percent and core wide metal-water reaction of 1 percent) is demonstrated. Details of calculations for specific breaks are included in subsequent paragraphs.

6.3.3.7.4 Large Recirculation Line Break Calculations

The characteristics that determine which is the most limiting large break are:

a. The calculated hot node reflooding time,
b. the calculated hot node uncovery time, and
c. the time of calculated boiling transition.

The time of calculated boiling transition increases with decreasing break size, since jet pump suction uncovery (which leads to boiling transition) is determined primarily by the break size for a particular plant. The calculated

hot node uncovery time also generally increases with decreasing break size, as it is primarily determined by the inventory loss during the blowdown.

The hot node reflooding time is determined by a number of interacting phenomena such as depressurization rate, counter current flow limiting and a combination of available ECCS.

The period between hot node uncovery and reflooding is the period when the hot node has the lowest heat transfer. Hence, the break that results in the longest period during which the hot node remains uncovered results in the highest calculated PCT. If two breaks have similar times during which the hot node remains uncovered, then the larger of the two breaks will be limiting as it would have an earlier boiling transition time (i.e., the larger break would have a more severe LAMB/SCAT blowdown heat transfer analysis).

Figure 6.3-73 shows the variation with break size of the calculated time the hot node remains uncovered. Based on these calculations, the DBA was determined to be the break that results in the highest calculated PCT in the large break region. Confirmation that this is the most limiting break over the entire break spectrum is shown in Figure 6.3-9.

Important variables from the analyses of the DBA are shown in Figures 6.3-11 through 6.3-20. These variables are:

a. Core average pressure as a function of time from LAMB.

b. Core flow as a function of time from LAMB.

c. Core inlet enthalpy as a function of time from LAMB.

d. Minimum critical power ratio as a function of time from SCAT.

e. Water level as a function of time from SAFE/REFLOOD.

f. Pressure as a function of time from SAFE/REFLOOD.

g. Fuel rod convective heat transfer coefficient as a function of time from CHASTE.

h. Peak cladding temperature as a function of time from CHASTE.

i. Average fuel temperature as a function of time from CHASTE.

j. PCT rod internal pressure as a function of time from CHASTE.

The maximum average planar linear heat generation rate, maximum local oxidation, and peak cladding temperature as a function of exposure from the CHASTE analysis of the DBA are shown in Table 6.3-3.

6.3.3.7.5 Transition Recirculation Line Break Calculations

Important variables from the analysis of the transition (1.0 ft^2) break are shown in Figures 6.3-37 through 6.3-48. These variables are:

a. Core average pressure (large break methods) as a function of time from LAMB.

b. Core flow (large break methods) as a function of time from LAMB.

c. Core inlet enthalpy (large break methods) as a function of time from LAMB.

c. Convective heat transfer coefficients as a function of time from REFLOOD.

d. Peak cladding temperature as a function of time from REFLOOD.

The same variables resulting from the analysis of a less limiting small break are shown in Figures 6.3-53 through 6.3-56.

6.3.3.7.7 Calculations for Other Break Locations

Reactor water level and vessel pressure from SAFE/REFLOOD and peak cladding temperature and fuel rod convective heat transfer coefficients from REFLOOD are shown in Figures 6.3-57 through 6.3-60 for the HPCS line break, Figures 6.3-61 through 6.3-64 for the feedwater line break, and in Figures 6.3-65 and 6.3-68 for the main steam line break inside the containment.

An analysis was also done for the main steam line break outside the containment. Reactor water level and vessel pressure from SAFE/REFLOOD and peak cladding temperature and fuel rod convective heat transfer coefficients from REFLOOD are shown in Figures 6.3-69 through 6.3-72.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Section 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10 CFR 50.46 acceptance criteria, given operation at or below the maximum average planar linear heat generation rates in Table 6.3-4.

6.3.4 TESTS AND INSPECTIONS

6.3.4.1 ECCS Performance Tests

All systems of the ECCS are tested for their operational ECCS function during the pre-operational and/or startup test program. Each component is tested for power source, range, direction of rotation, set point, limit switch setting,

HPCS injection begins as soon as the HPCS pump is up to speed and the injection valve is open since the HPCS is capable of injecting water into the RPV over a pressure range from 1160 psid (psid - differential pressure between RPV and pump suction source) to 0 psid.

6.3.6 REFERENCES FOR SECTION 6.3

1. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K", NEDO-20566P, November 1975.

2. H. M. Hirsch, "Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems," January 1973 (NEDO-10739).

TABLE 6.3-1

SIGNIFICANT INPUT VARIABLES USED IN THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Variable	Units	Value
A. PLANT PARAMETERS		
Core thermal power	MW_t	3729
Vessel steam output	lb_m/hr	$16.2x10^6$
Corresponding percent of rated steam flow	%	105
Vessel steam dome pressure	psia	1060
Maximum recirculation line break area	ft^2	2.7
B. EMERGENCY CORE COOLING SYSTEM PARAMETERS		
B.1 Low Pressure Coolant Injection System		
Vessel pressure at which flow may commence	psid (vessel to drywell)	225
Minimum rated flow at vessel pressure	GPM psid (vessel to drywell)	19500 20
Initiating Signals		
low water level	ft. above top of active fuel	$\geq$1.0
or		
high drywell pressure	psig	$\leq$2.0
Maximum allowable time delay from initiating signal to pumps at rated speed	sec	27
Injection valve fully open	sec. after DBA	$\leq$40

TABLE 6.3-1 (Continued)

Variable		Units	Value
B.2	Low Pressure Core Spray System		
	Vessel pressure at which flow may commence	psid (vessel to drywell)	289
	Minimum rated flow at vessel pressure	gpm psid (vessel to drywell)	6000 122
	Initiating Signals		
	low water level or high drywell pressure	ft. above top of active fuel psig	$\geq$1.0 $\leq$2.0
	Maximum allowed (runout) flow	gpm	7800
	Maximum allowed delay time from initiating signal to pump at rated speed	sec	27.0
	Injection valve fully open	sec. after DBA	$\leq$40
B.3	High Pressure Core Spray		
	Vessel pressure at which flow may commence	psid	1177
	Minimum rated flow available at vessel pressure	gpm psid (vessel to pump suction)	517 1550 6000 1177 1147 200
	Initiating Signals		
	low water level or high drywell pressure	ft. above top of active fuel psig	$\geq$10.9 $\leq$2.0
	Maximum allowed (runout) flow	gpm	7800
	Maximum allowed delay time from initiating signal to rated flow available and injection valve wide open	sec	27.0

TABLE 6.3-1 (continued)

Variable	Units	Value
B.4 Automatic Depressurization System		
Total number of relief valves with ADS function		8
Total minimum flow capacity at vessel pressure	lb/hr psig	6.4×10^6 1125
Initiating Signals		
low water level and	ft. above top of active fuel	≥ 1.0
high drywell pressure	psig	≤ 2.0
Delay time from all initiating signals completed to the time valves are open	sec	≤ 120
C. FUEL PARAMETERS		
Fuel type		Initial core
Fuel bundle geometry		8 x 8
Lattice		C
Number of fueled rods per bundle		62
Peak technical specification linear heat generation rate	kW/ft	13.4
Initial minimum critical power ratio		1.17
Design axial peaking factor		1.4

TABLE 6.3-2

OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS FOR DESIGN BASIS ACCIDENT[1]

Time (sec)	Events
0	Design basis loss-of-coolant accident assumed to start; normal auxiliary power assumed to be lost.
~0	Drywell high pressure and reactor low water level reached. All diesel generators signaled to start; scram; HPCS, LPCS, LPCI signaled to start on high drywell pressure.
~3	Reactor low-low water level reached. HPCS receives second signal to start.
~7	Reactor low-low-low water level reached. Second signal to start LPCI and LPCS; auto-depressurization sequence begins; main steam isolation valve signaled to close.
<10	All diesel generators ready to load; energize HPCS pump motor; open HPCS injection valve: begin energizing LPCI and LPCS pump motors.
≤27	HPCS injection valve open and pump at design flow, which completes HPCS startup.
≤40	LPCI and LPCS pumps at rated flow, LPCI and LPCS injection valves open, which completes the LPCI and LPCS startups.
See Figure 6.3-14	Core effectively reflooded assuming worst single failure; heatup terminated.
≥10 min	Operator shifts to containment cooling.

NOTE:

1. For the purpose of all but the next to last entry on this table, all ECCS equipment is assumed to function as designed. Performance analysis calculations consider the effects of single equipment failures. (see Sections 6.3.2.5 and 6.3.3.3).

TABLE 6.3-3

SINGLE FAILURE EVALUATION[1]

Assumed Failure	Suction Break[2] Systems Remaining
LPCI Emergency Diesel Generator (D/G)	All ADS, HPCS, LPCS, 1 LPCI
LPCS Emergency D/G	All ADS, HPCS, 2 LPCI
HPCS System	All ADS, LPCS, 3 LPCI
One ADS Valve	All ADS minus one, LPCS, HPCS, 3 LPCI

NOTE:

1. Single, active failures are considered in the ECCS performance evaluation. Other postulated failures are not specially considered because they all result in at least as much ECCS capacity as one of the above designed failures.

2. Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed, less the ECCS system in which the break is assumed.

TABLE 6.3-4

MAPLHGR, MAXIMUM LOCAL OXIDATION, AND PEAK CLAD TEMPERATURE VERSUS EXPOSURE

Exposure MWD/T	MAPLHGR KW/FT	P.C.T. DEG-F	Oxide Frac.
High Enrichment IC Fuel			
200.0	11.9	2045.	0.018
1000.0	12.0	2045.	0.018
5000.0	12.1	2038.	0.016
10000.0	12.2	2036.	0.016
15000.0	12.3	2051.	0.017
20000.0	12.1	2042.	0.016
25000.0	11.6	1977.	0.013
30000.0	11.2	1900.	0.010
35000.0	10.6	1809.	0.007
40000.0	9.9	1730.	0.005
Medium Enrichment IC Fuel			
200.0	12.0	2051.	0.017
1000.0	12.2	2059.	0.018
5000.0	12.7	2089.	0.019
10000.0	12.9	2100.	0.020
15000.0	12.9	2115.	0.021
20000.0	12.6	2082.	0.019
25000.0	11.7	1953.	0.012
30000.0	10.8	1826.	0.008
35000.0	10.2	1747.	0.006
40000.0	9.6	1680.	0.004

TABLE 6.3-4 (continued)

Exposure MWD/T	MAPLHGR KW/FT	P.C.T. DEG-F	Oxide Frac.
Natural Uranium IC Fuel			
200.0	11.5	1965.	0.013
1000.0	11.4	1933.	0.011
5000.0	11.3	1892.	0.010
10000.0	11.5	1886.	0.009
15000.0	11.5	1884.	0.009
20000.0	11.0	1824.	0.008
25000.0	10.4	1747.	0.006
30000.0	9.7	1670.	0.004
35000.0	9.0	1601.	0.003

TABLE 6.3-5

SUMMARY OF RESULTS OF LOCA ANALYSIS

Break Spectrum Analysis		PCT (^oF)	PEAK LOCAL OXIDATION (% of Initial Cladding Thickness)
Break Size Location Single Failure			
2.7 ft^2 (DBA) Recirc. Suction LPCI D/G		2115 (1)	2.05
1.0 ft^2 Recirc. Suction LPCI D/G	Large Break Method	1855 (1)	0.78
	Small Break Method	1407 (2)	<1.0
0.09 ft^2 Recirc. Suction HPCS System		1345 (2)	<1.0

NOTES:

1. CHASTE - large break method

2. Non-DBA reflood

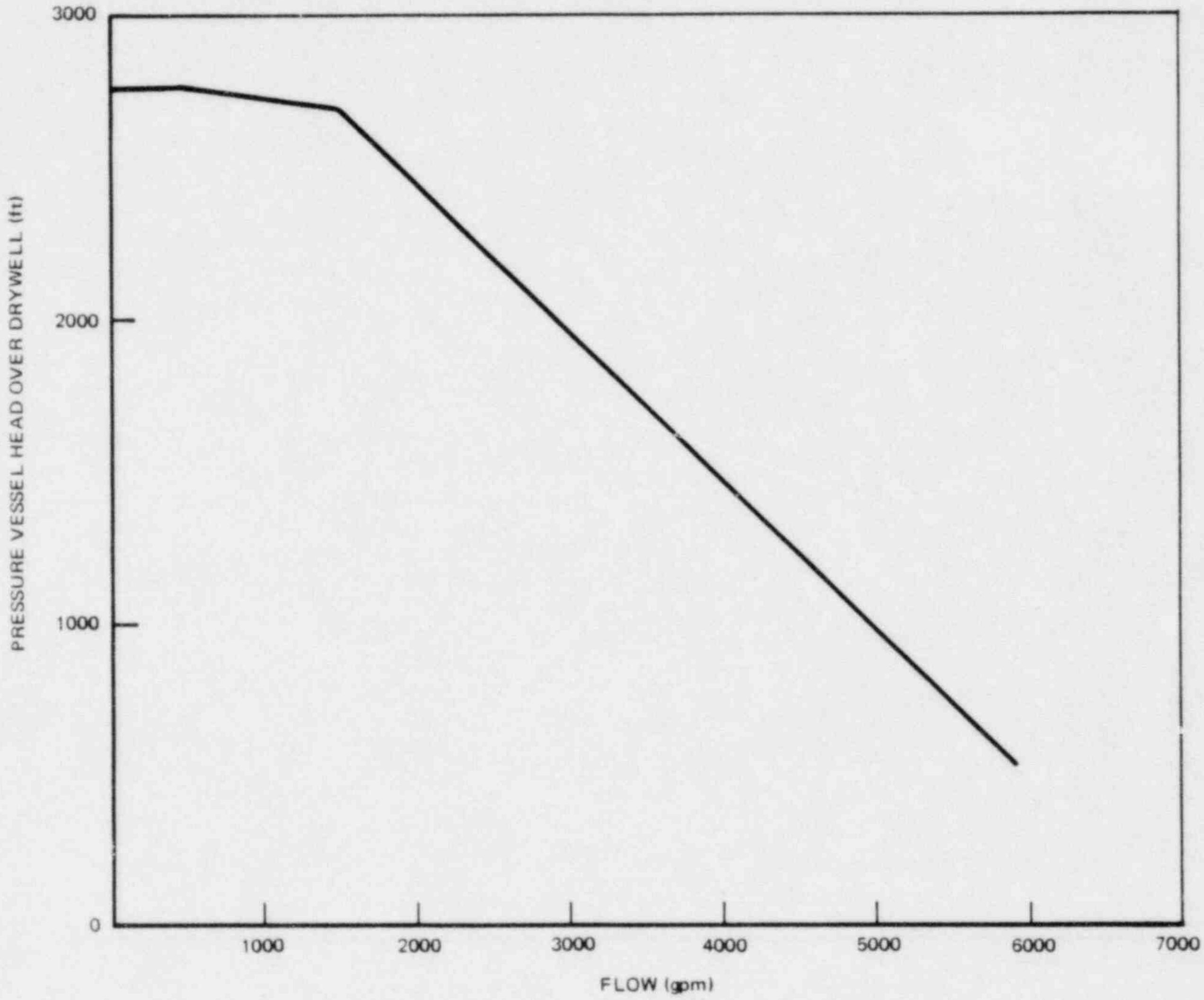
The corewide metal-water reaction for the subject plant has been calculated using method 1 described in reference 2.

The value is as follows: Corewide Metal-Water Reaction % = 0.14

TABLE 6.3-6

KEY TO FIGURES

	LARGE BREAK METHOD		SMALL BREAK METHOD						
	DBA	$1.0\ ft^2$ Large Break Methods	$1.0\ ft^2$ Small Break Methods	Worst Small Break $0.09\ ft^2$	Additional Small Break $0.9\ ft^2$	Core Spray Line (CSLN)	Feedwater Line (FDWR)	Main Steam Line Inside Containment (STML)	Main Steam Line Outside Containment (STMO)
Core Average Pressure	6.3-11	6.3-37	NA	NA	NA	NA	NA	NA	NA
Core Average Inlet Flow	6.3-12	6.3.38	NA	NA	NA	NA	NA	NA	NA
Core Inlet Enthalpy	6.3-16	6.3-39	NA	NA	NA	NA	NA	NA	NA
Minimum Critical Power Ratio	6.3-17	6.3-40	NA	NA	NA	NA	NA	NA	NA
Water Level Inside Shroud	6.3-14	6.3-41	6.3-45	6.3-49	6.3-53	6.3-57	6.3-61	6.3-65	6.3-69
Reactor Vessel Pressure	6.3-18	6.3-42	6.3-46	6.3-50	6.3-54	6.3-58	6.3-62	6.3-66	6.3-70
Convective Heat Transfer Coefficient	6.3-13	6.3-43	6.3-47	6.3-51	6.3-55	6.3-59	6.3-63	6.3-67	6.3-71
Peak Cladding Temperature	6.3-15	6.3-44	6.3-48	6.3-52	6.3-56	6.3-60	6.3-64	6.3-68	6.3-72
Average Fuel Temperature	6.3-19	NA	NA	NA	NA	NA	NA	NA	NA
PCT Rod Internal Pressure	6.3-20	NA	NA	NA	NA	NA	NA	NA	NA
Peak Cladding Temperature and Peak Local Oxidation Versus Break Area	6.3-9	6.3-9	6.3-9	6.3-9	6.3-9	6.3-9	6.3-9	6.3-9	6.3-9
Normalized Power Versus Time	6.3-10	6.3-10	6.3-10	6.3-10	6.3-10	6.3-10	6.3-10	6.3-10	6.3-10
Total Time for Which Highest Powered Node Remains Uncovered Versus Break Area	6.3-73	6.3-73	NA	NA	NA	NA	NA	NA	NA



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PERRY NUCLEAR POWER PLANT
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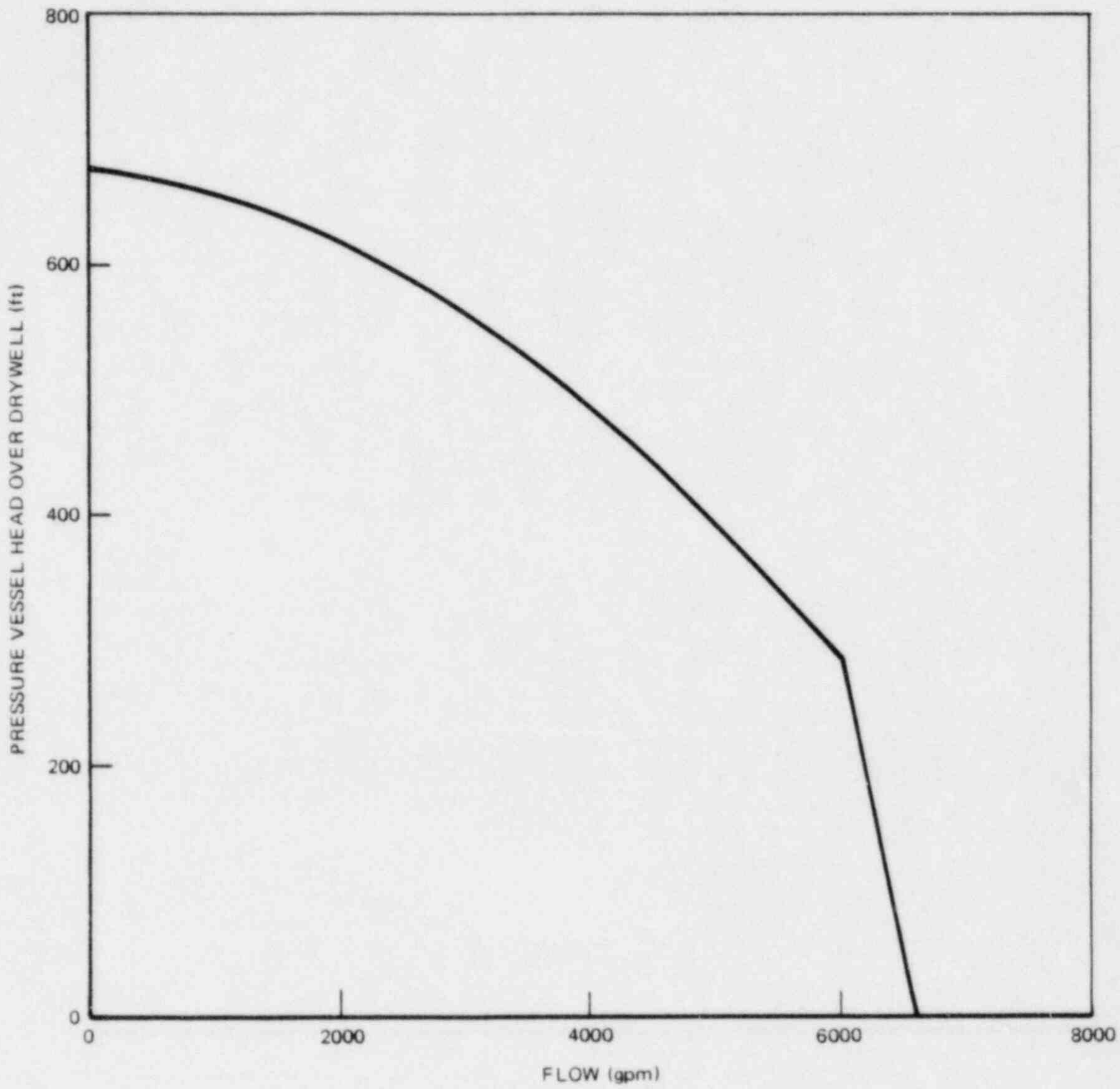
Head Versus High Pressure Core Spray Flow Used in LOCA Analysis

Figure 6.3-4



PERRY NUCLEAR POWER PLANT
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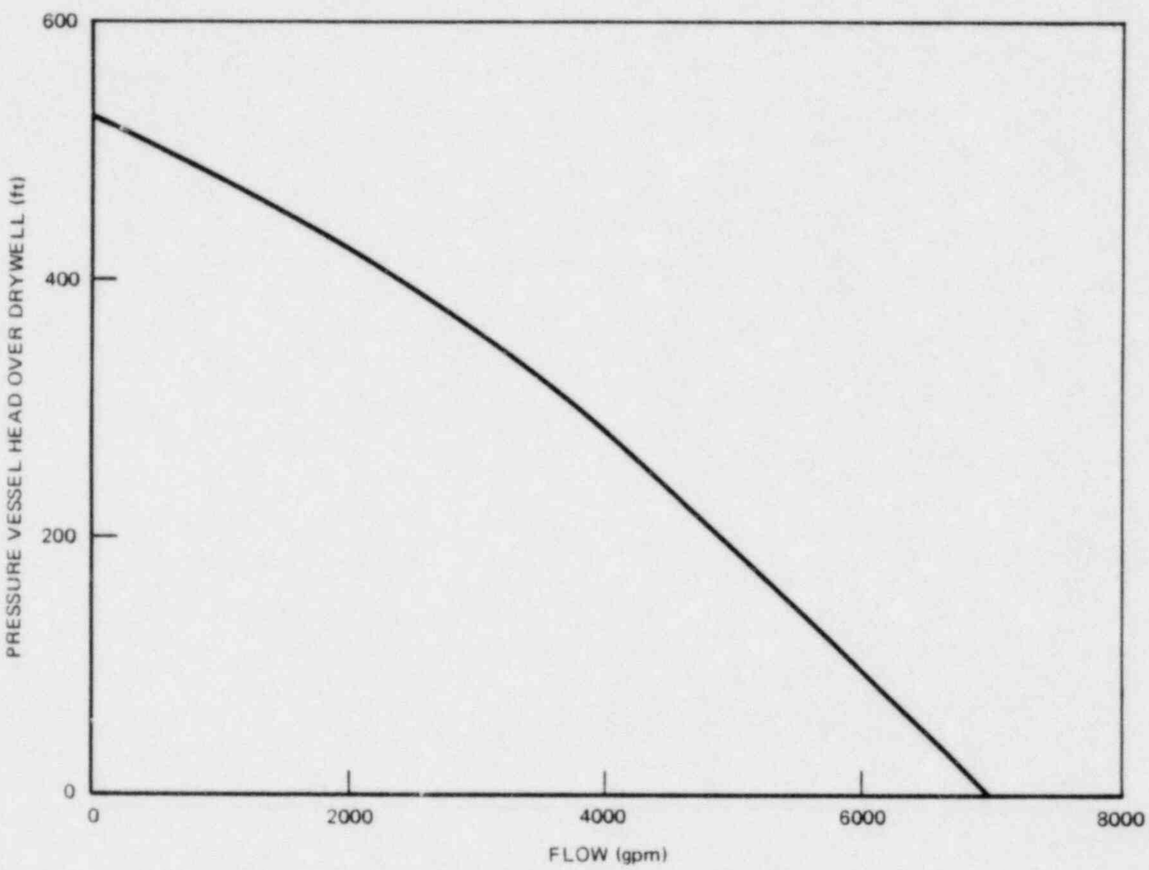
Head Versus Low Pressure Core Spray Flow Used in LOCA Analysis

Figure 6.3-5



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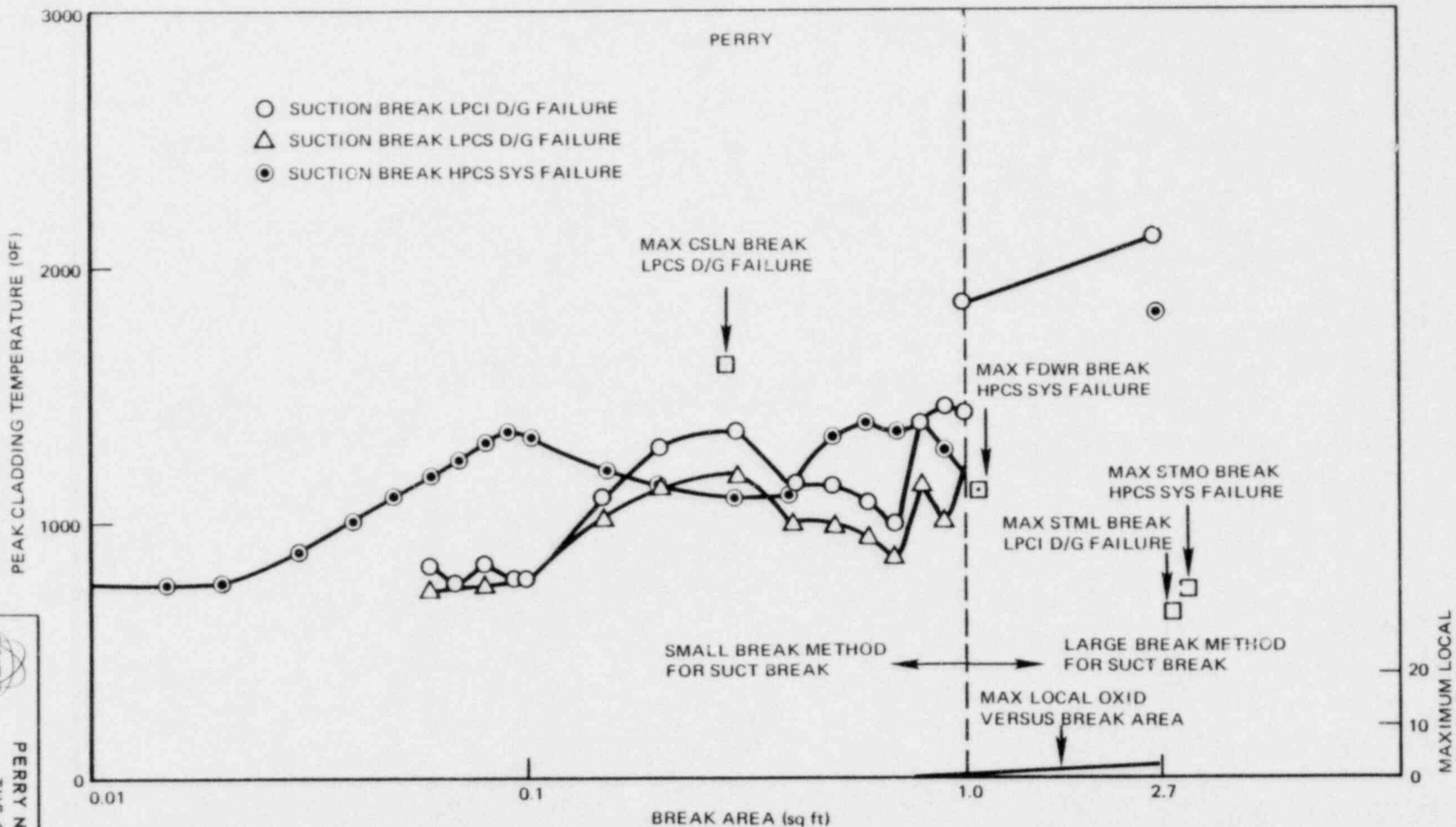
Head Versus Low Pressure Coolant
Injection Flow Used in LOCA Analysis
for 1 Pump Only

Figure 6.3-6



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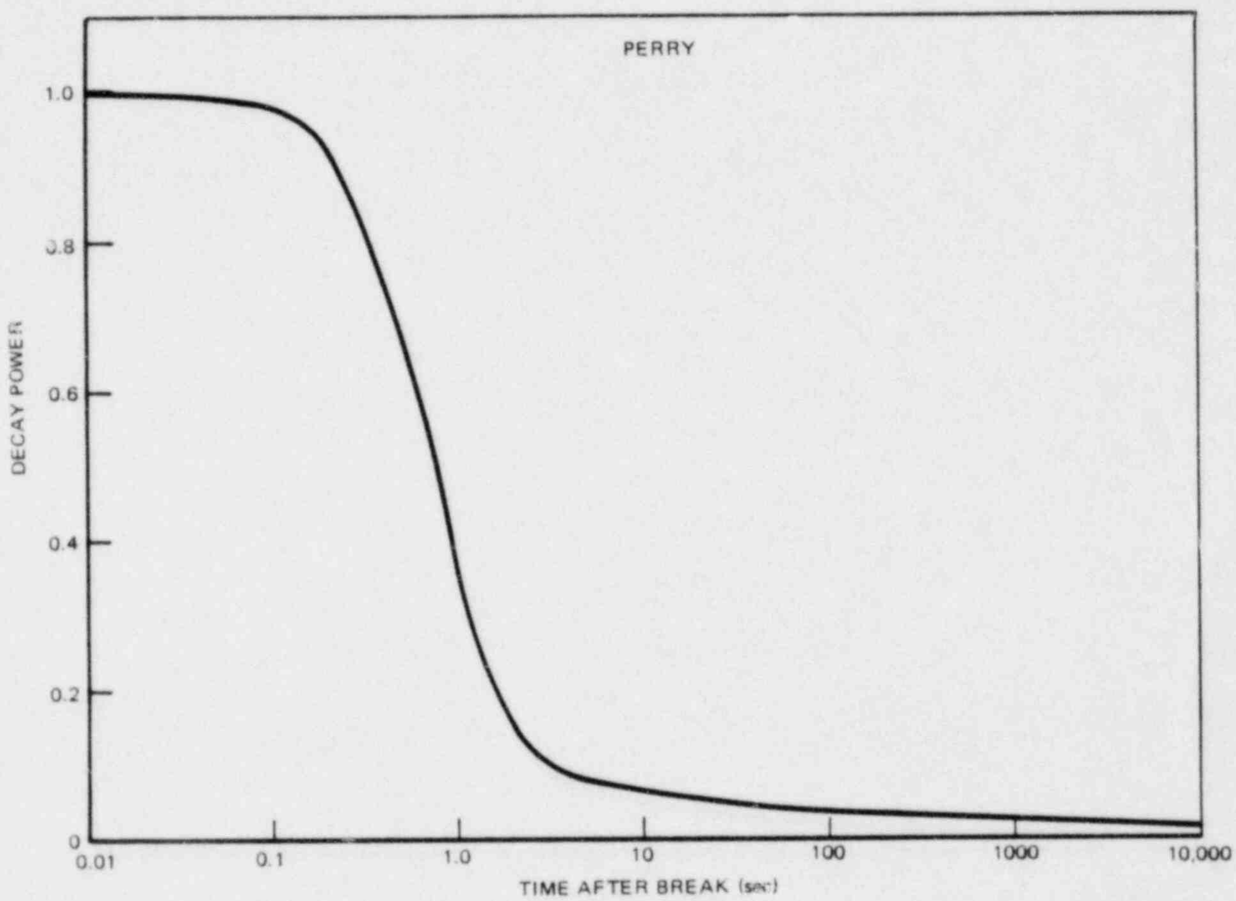
Peak Cladding Temperature and
Maximum Local Oxidation Versus
Break Area

Figure 6.3-9



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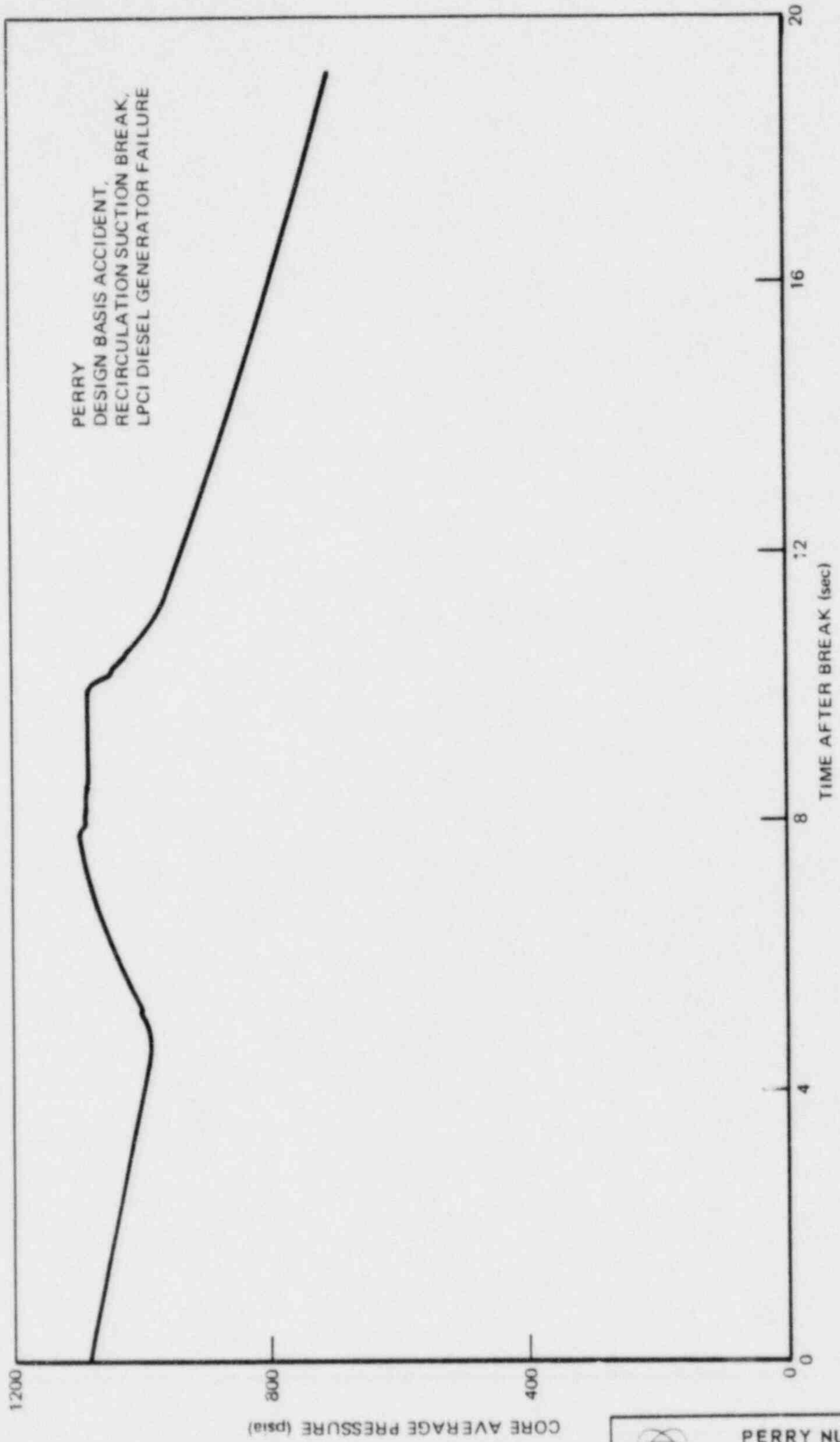
Normalized Power Versus Time

Figure 6.3-10



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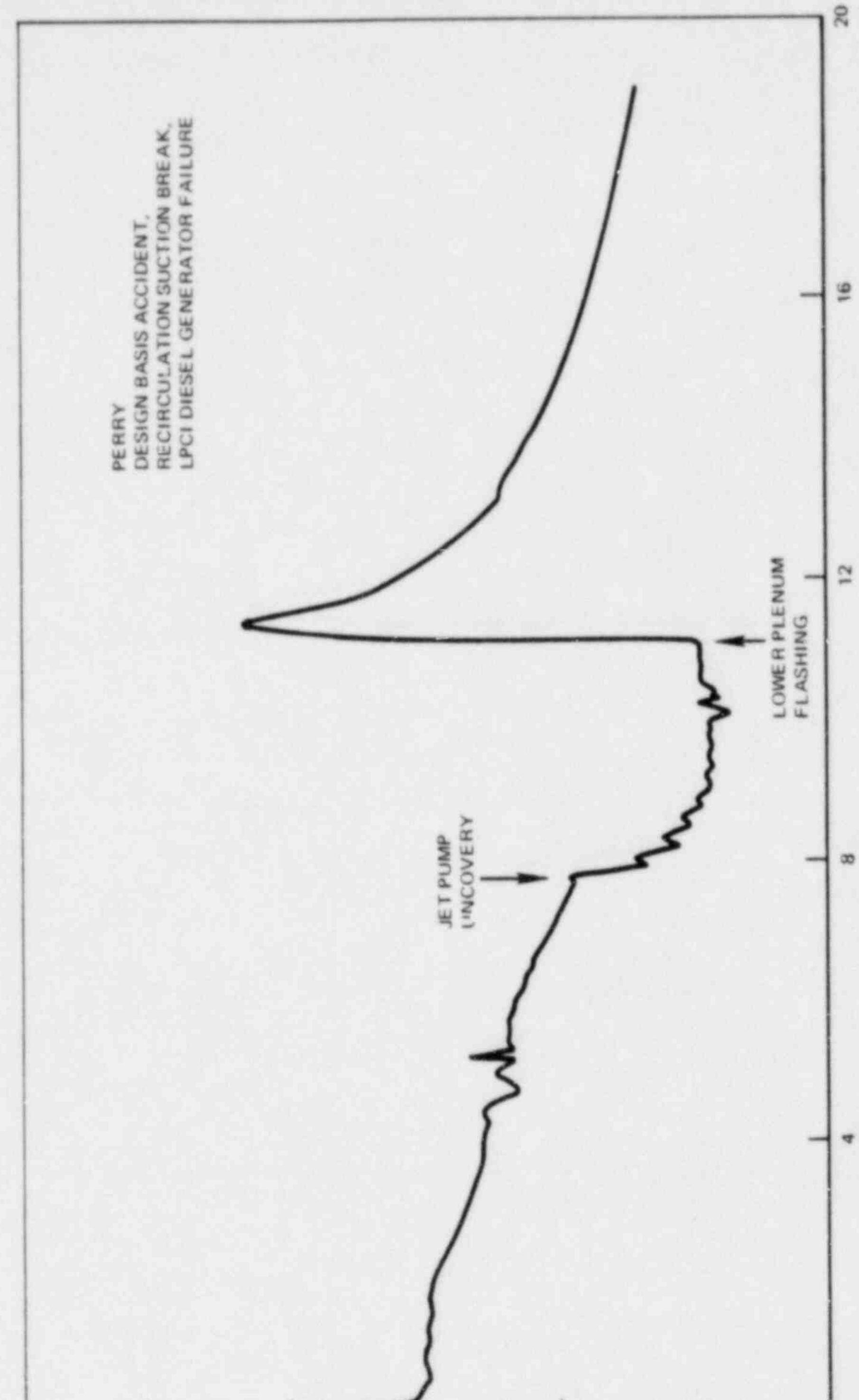
Core Average Pressure Versus
Time After Break

Figure 6.3-11



PERRY NUCLEAR POWER PLANT
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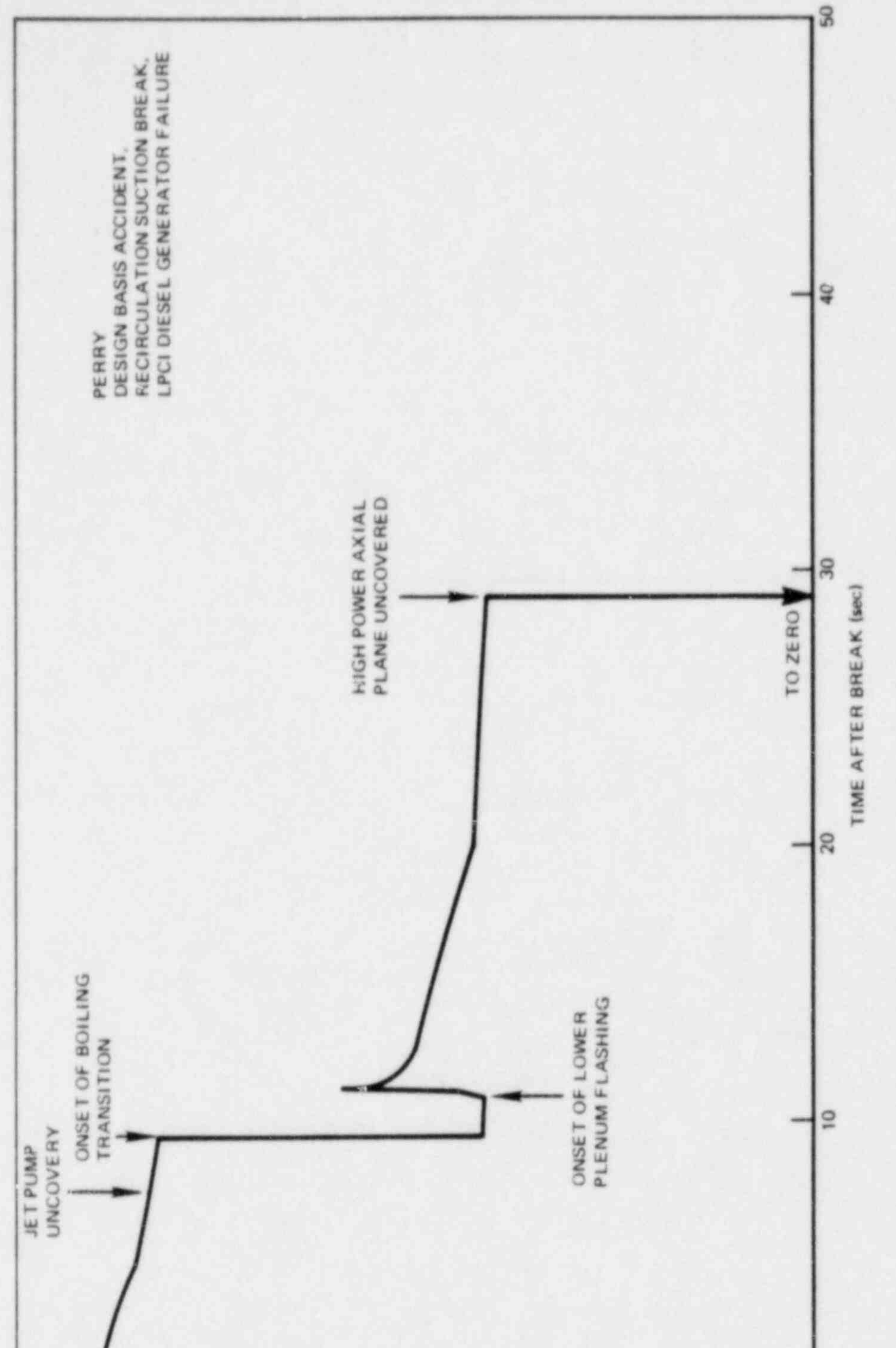
Normalized Core Average Inlet Flow
Versus Time After Break

Figure 6.3-12



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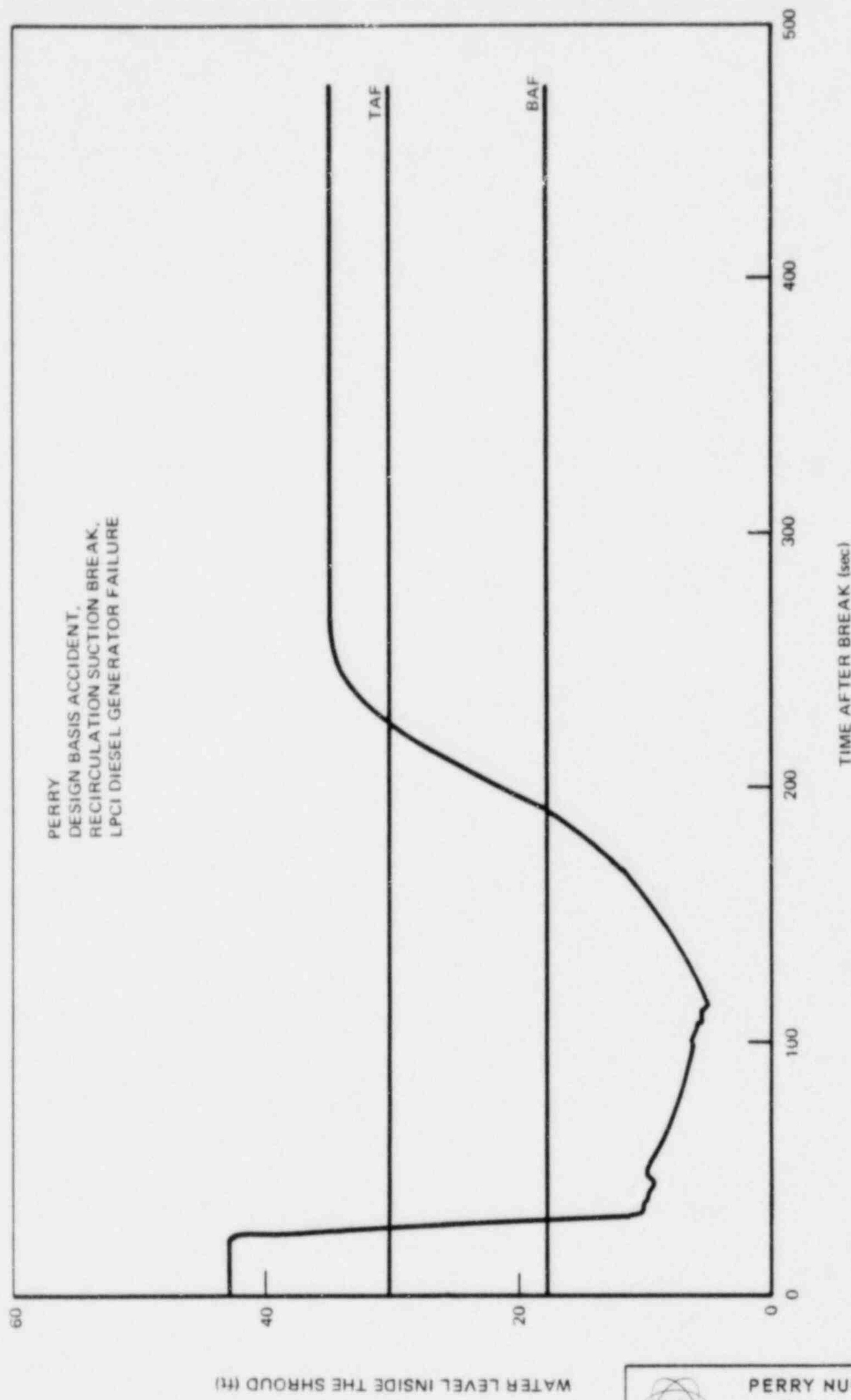
Convective Heat Transfer
Coefficient Versus
Time After Break

Figure 6.3-13



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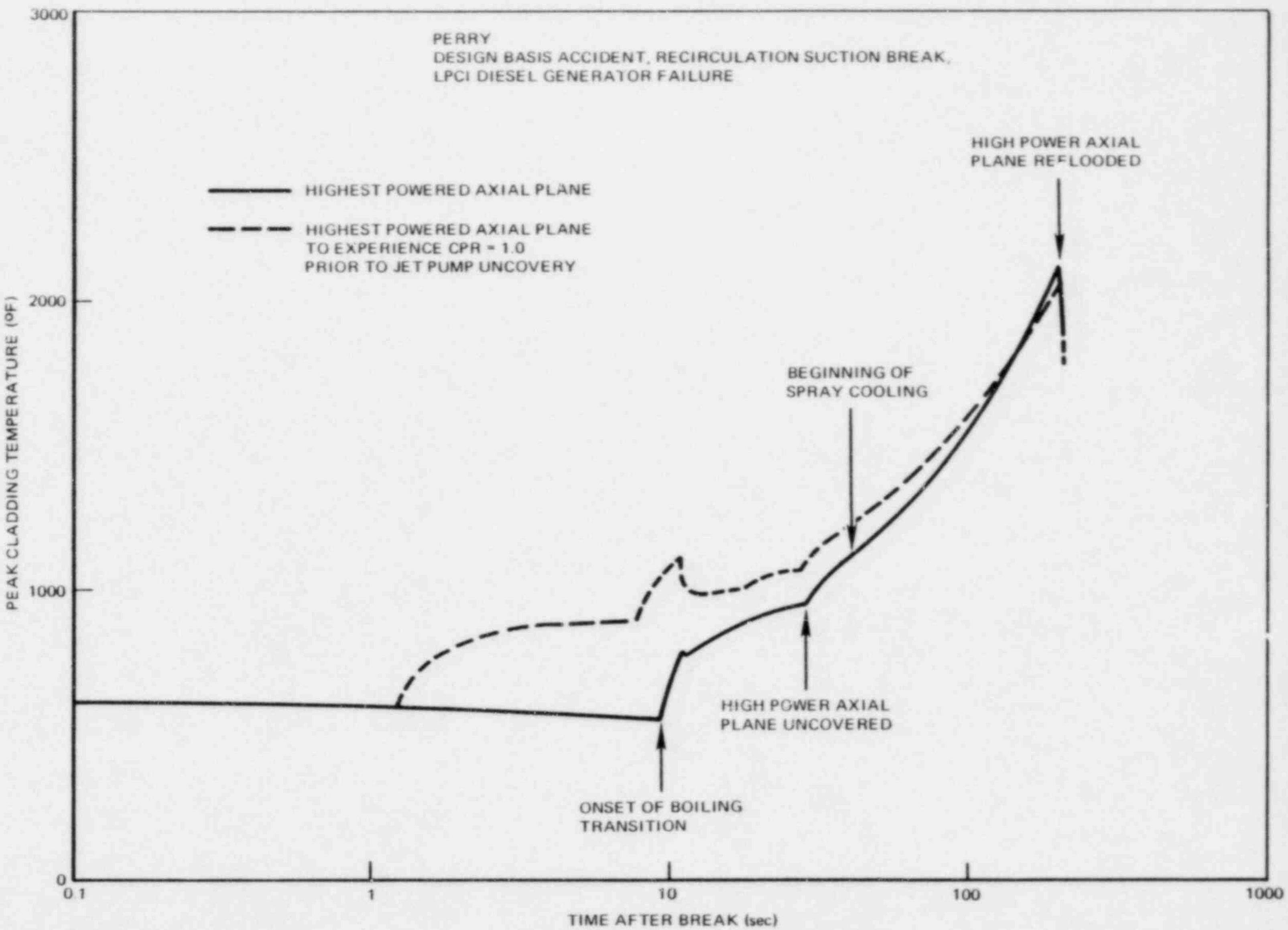
Water Level Inside the
Shroud Versus Time After Break

Figure 6.3-14



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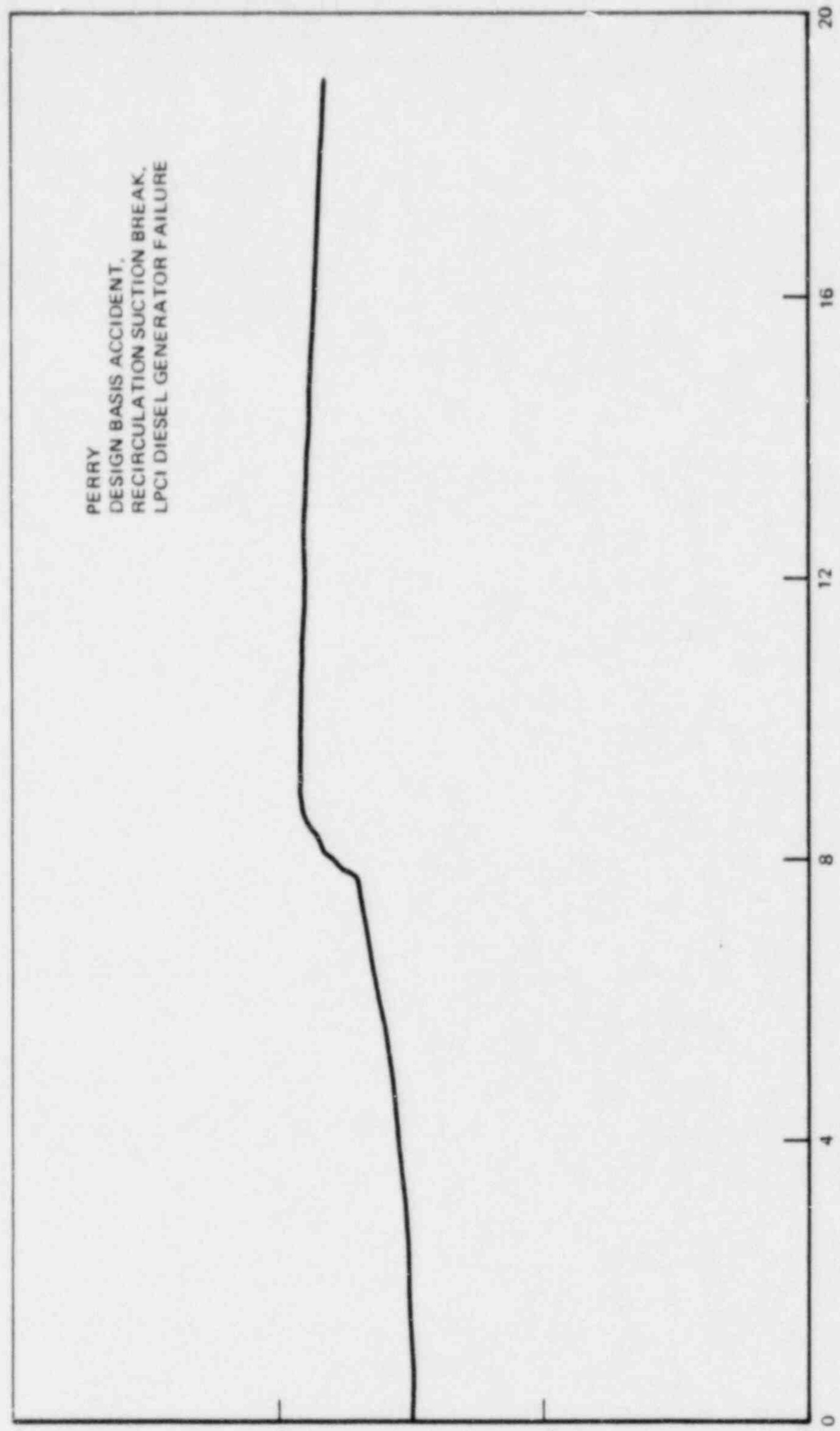
Peak Cladding Temperature
Versus Time After Break

Figure 6.3-15



PERRY NUCLEAR POWER PLANT
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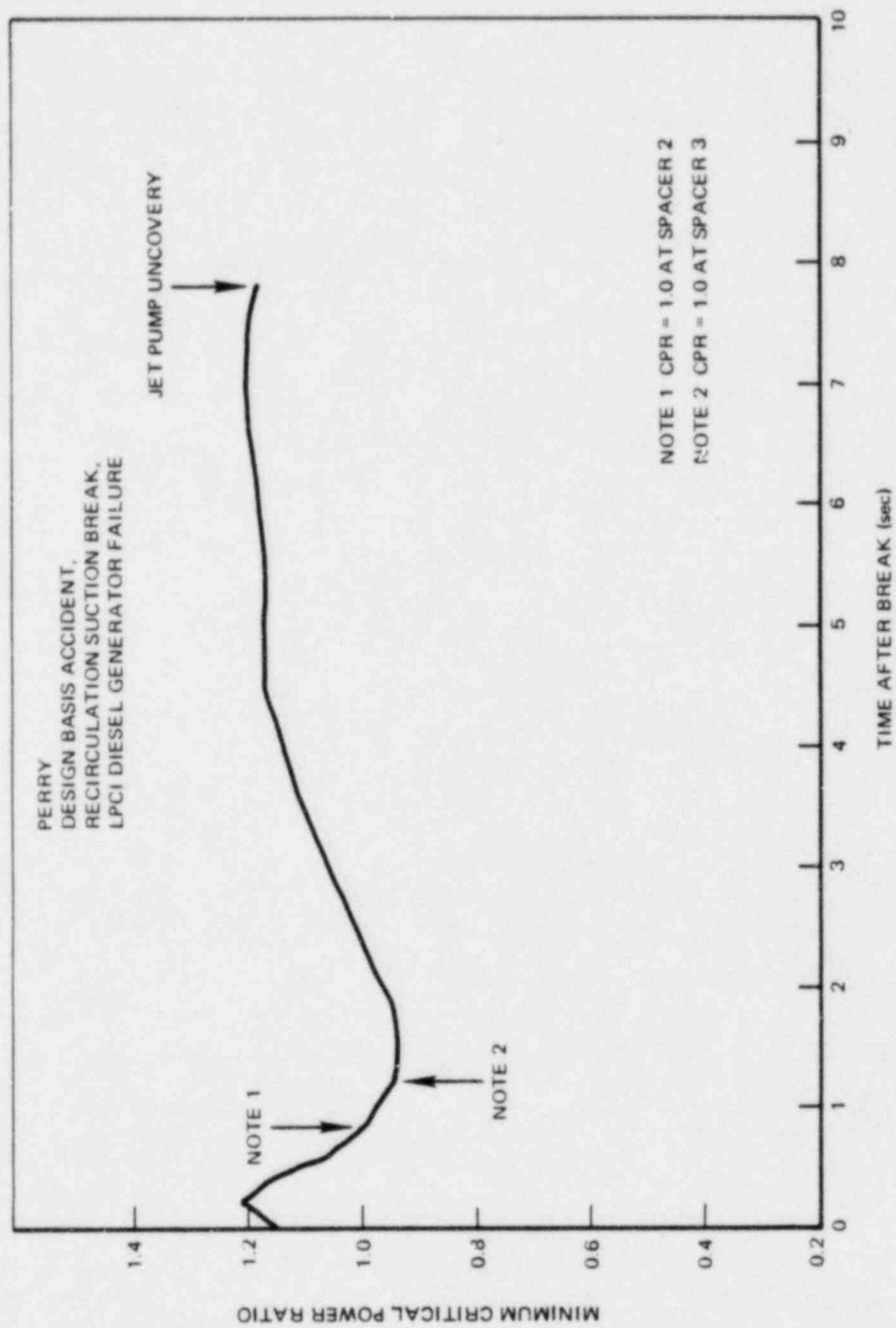
Core Inlet Enthalpy Versus
Time After Break

Figure 6.3-16



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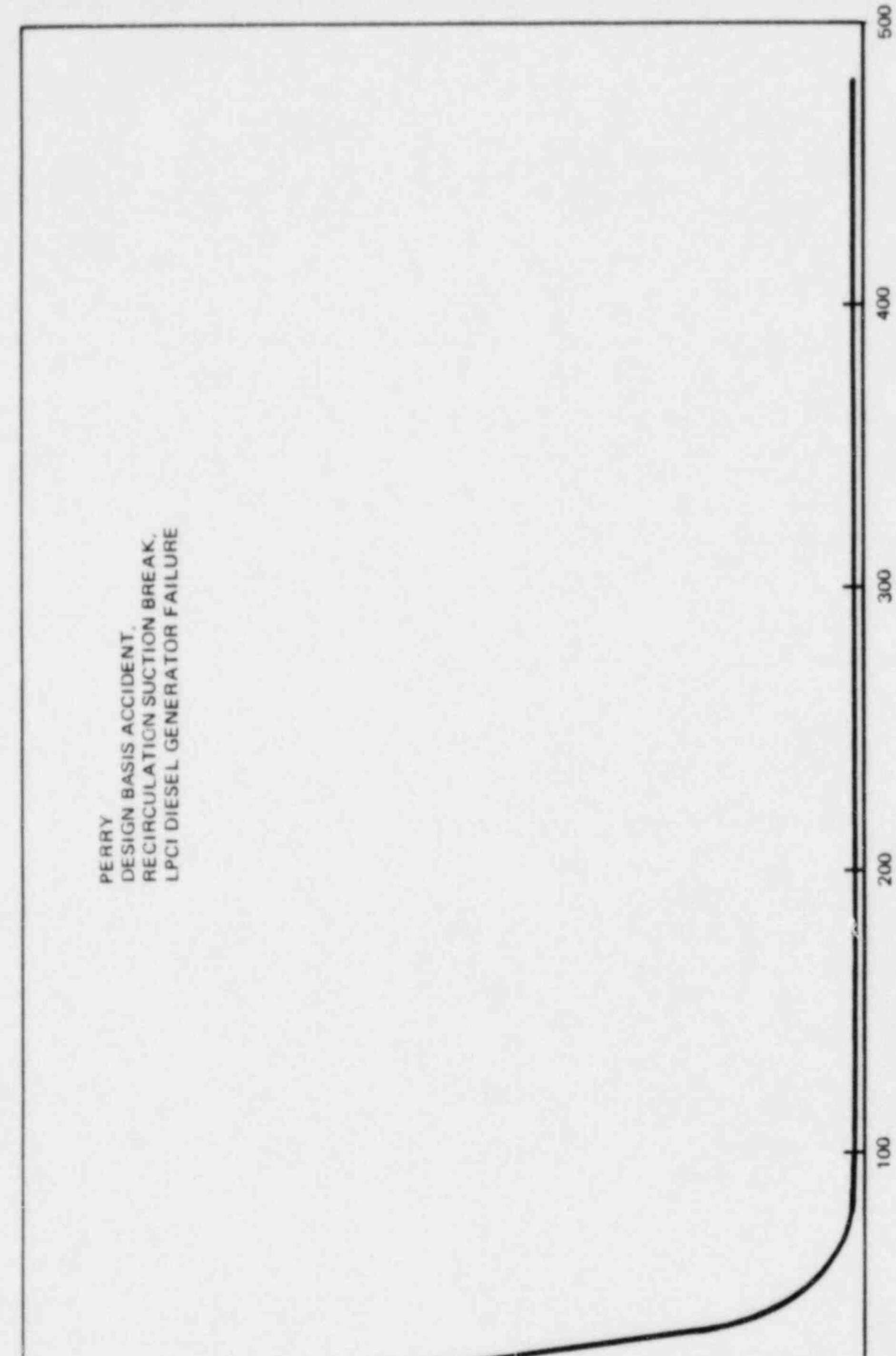
Minimum Critical Power Ratio
Versus Time After Break

Figure 6.3-17



PERRY NUCLEAR POWER PLANT
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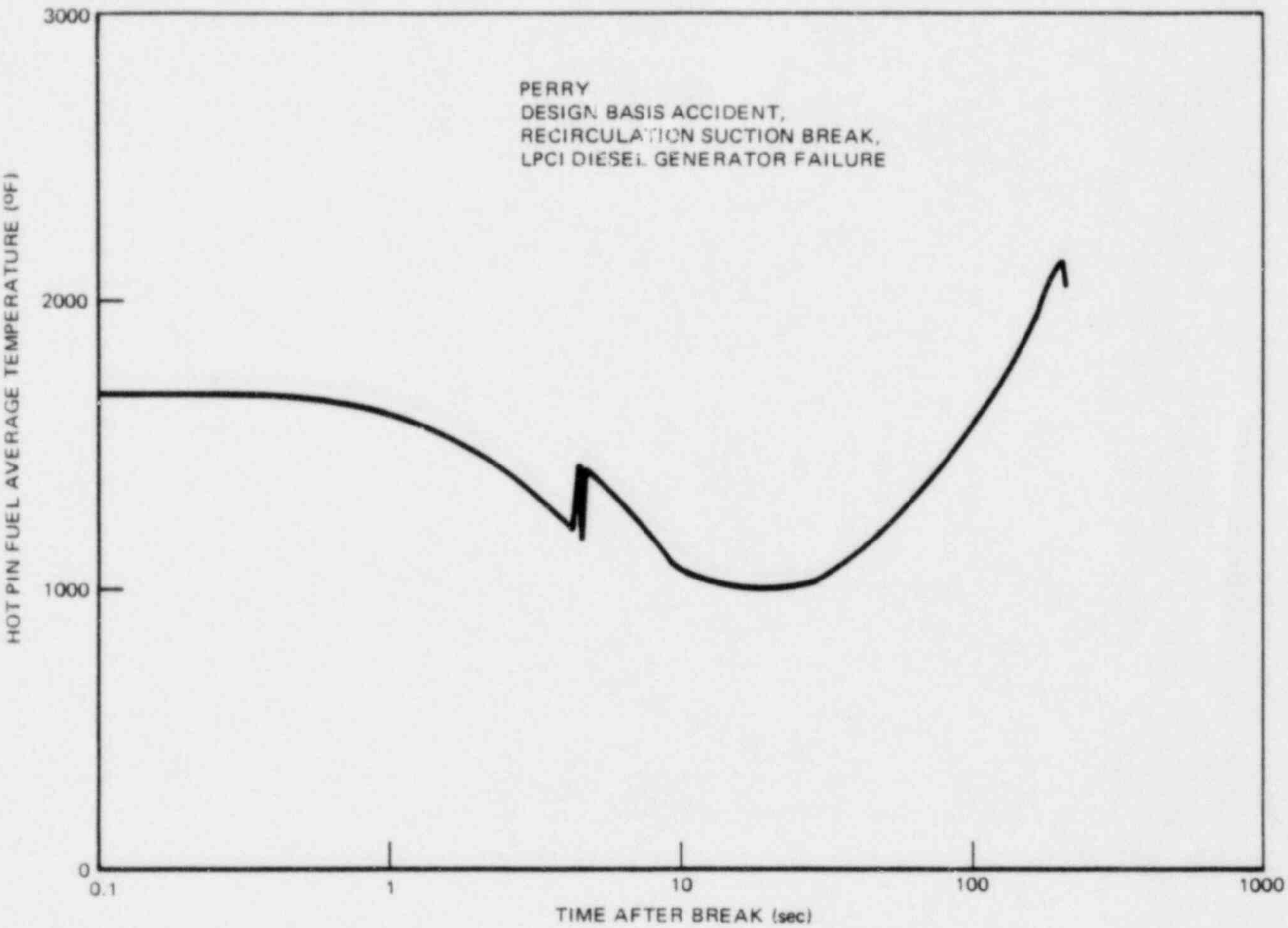
Reactor Vessel Pressure
Versus Time After Break

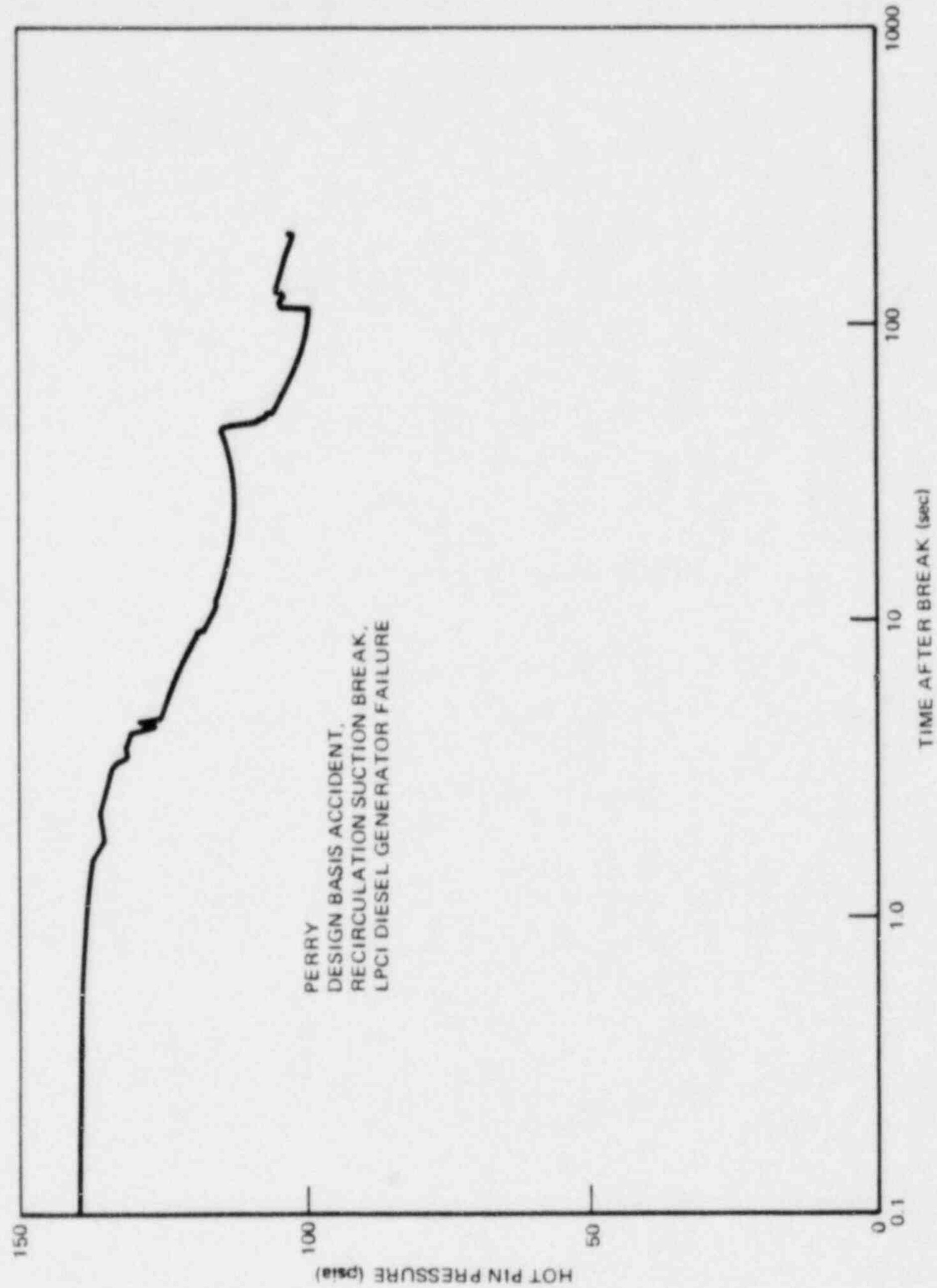
Figure 6.3-18



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Hot Pin Fuel Average Temperatures
Versus Time After Break

Figure 6.3-19

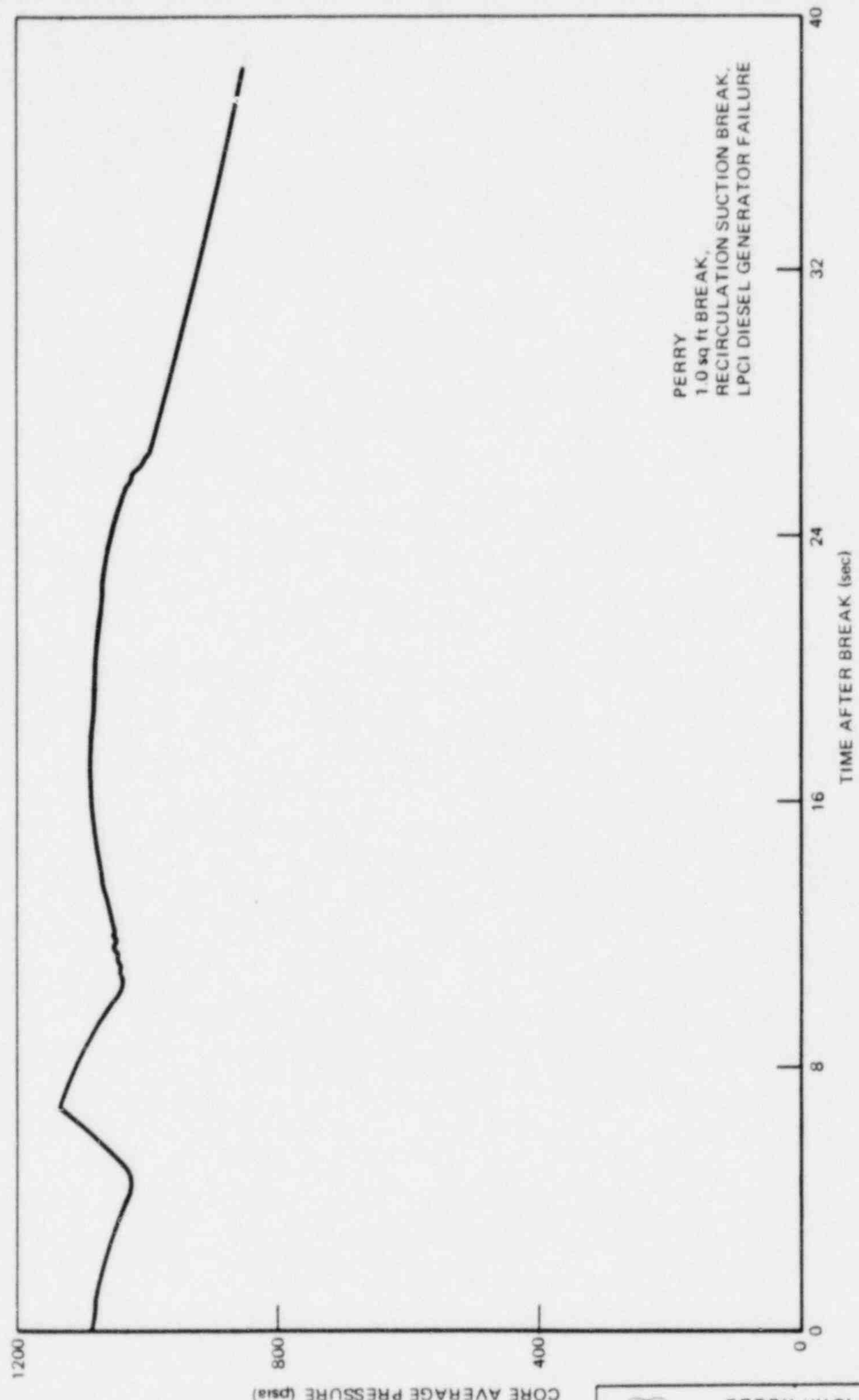


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Hot Pin Pressure
Versus Time After Break

Figure 6.3-20

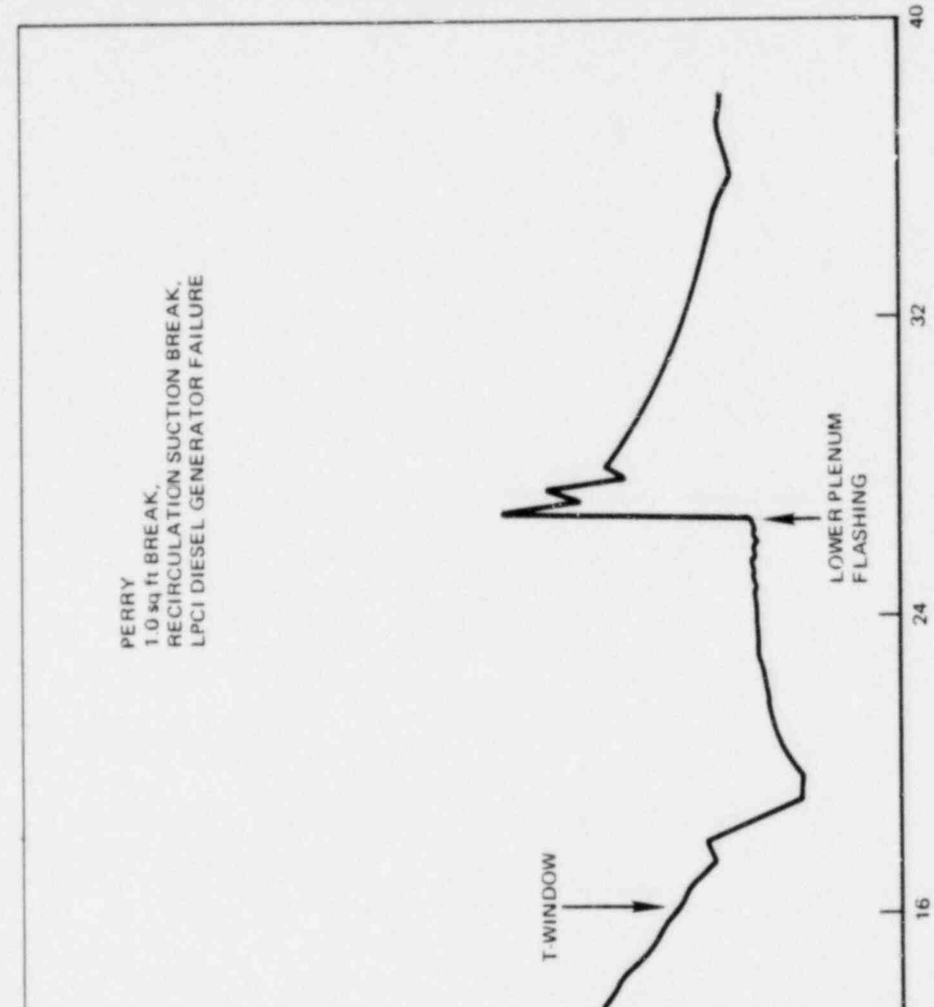
FIGURES 6.3-21 THROUGH 6.3-36 HAVE BEEN INTENTIONALLY DELETED



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Core Average Pressure
Versus Time After Break

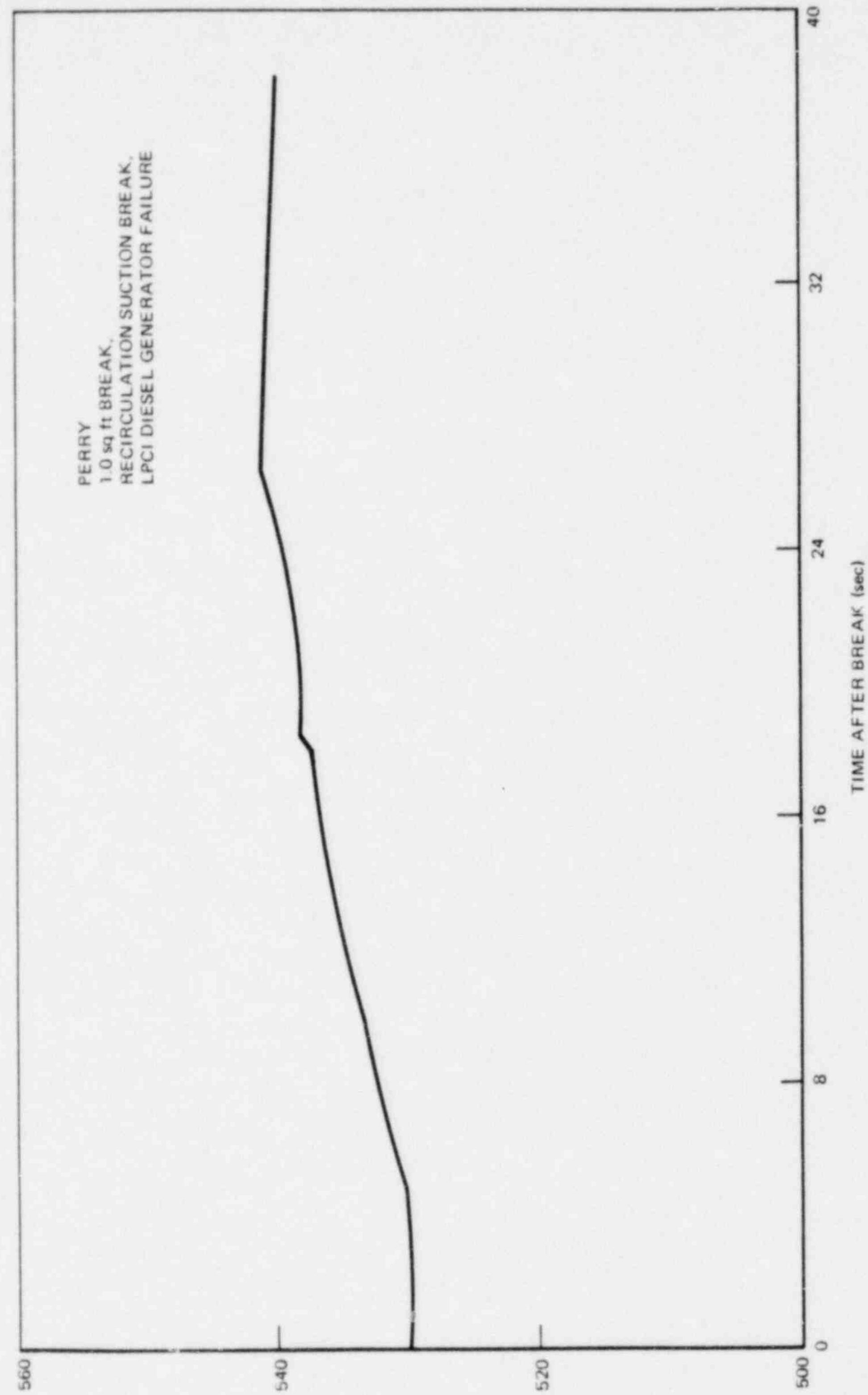
Figure 6.3-37



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Normalized Core Average Inlet Flow Versus Time After Break

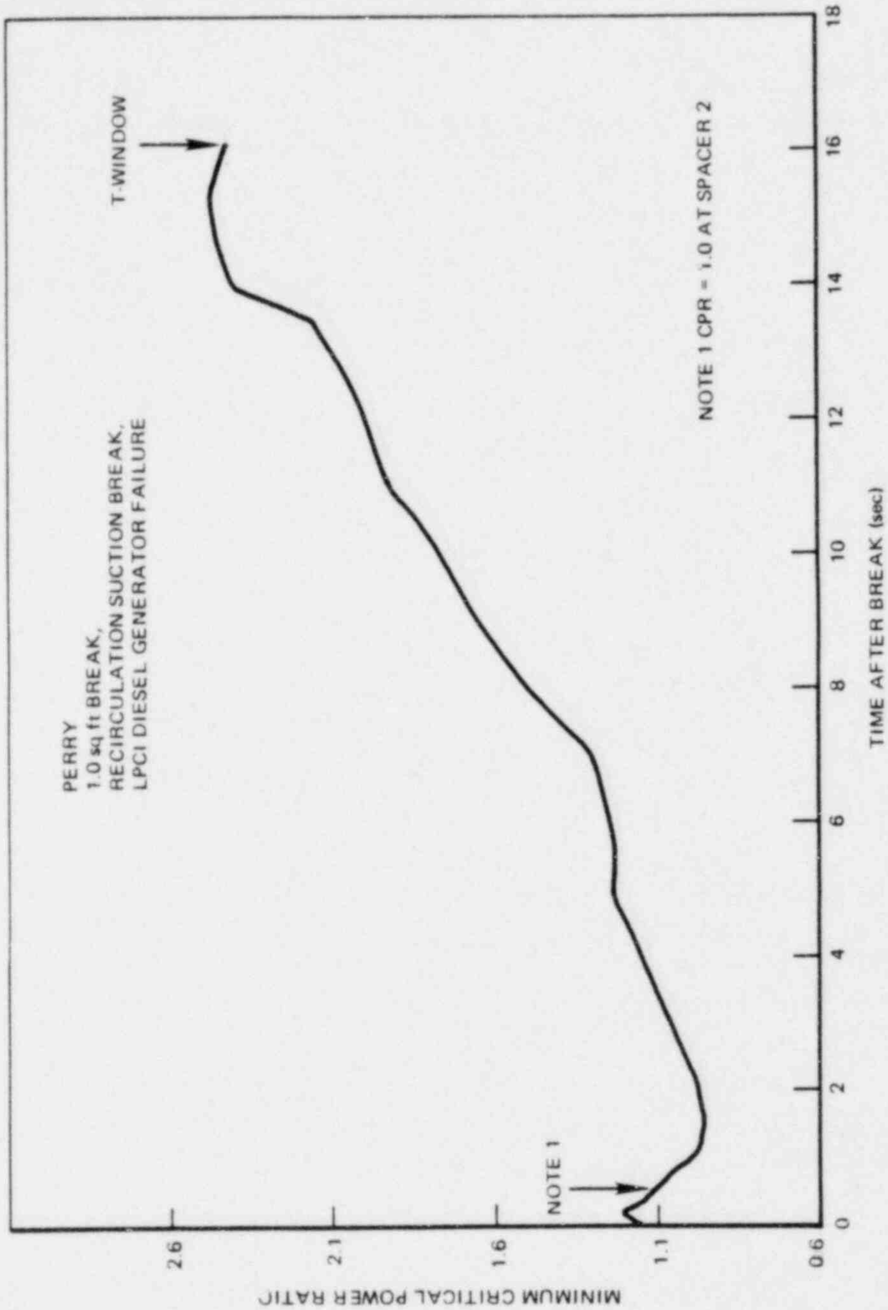
Figure 6.3-38



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Core Inlet Enthalpy
Versus Time After Break

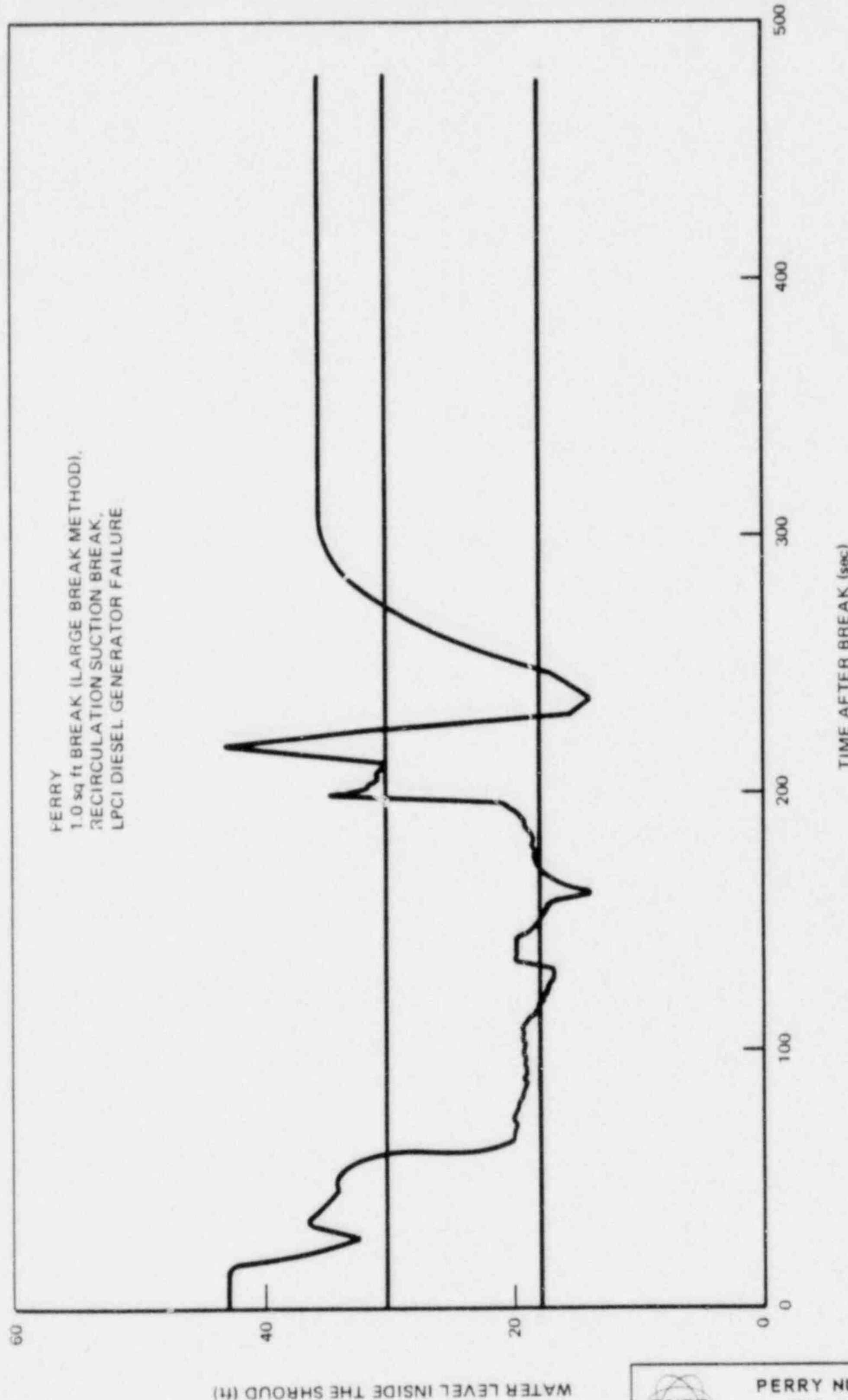
Figure 6.3-39



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Minimum Critical Power
Ratio Versus Time After Break

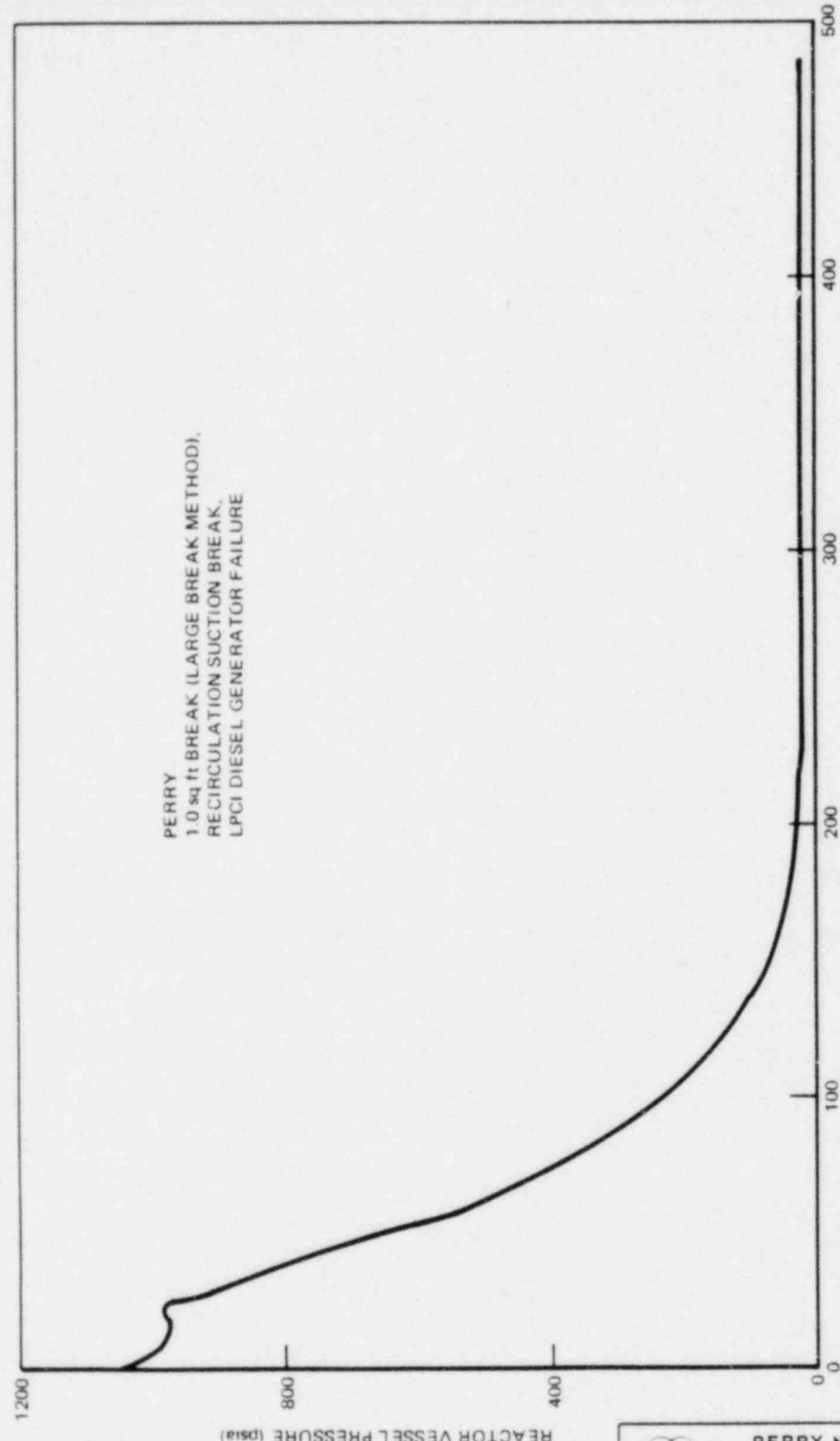
Figure 6.3-40



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Water Level Inside the Shroud
Versus Time After Break

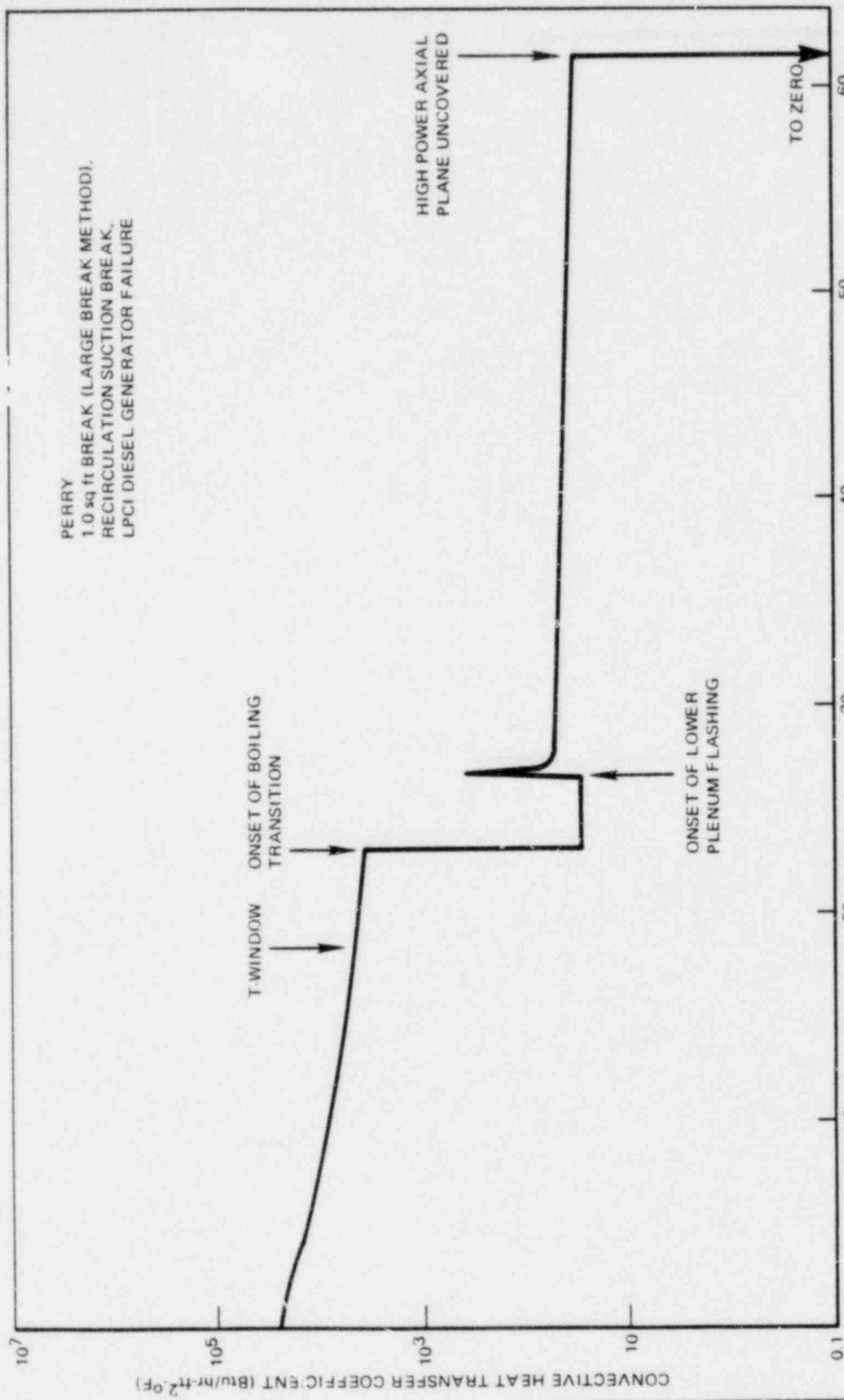
Figure 6.3-41



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Pressure
Versus Time After Break

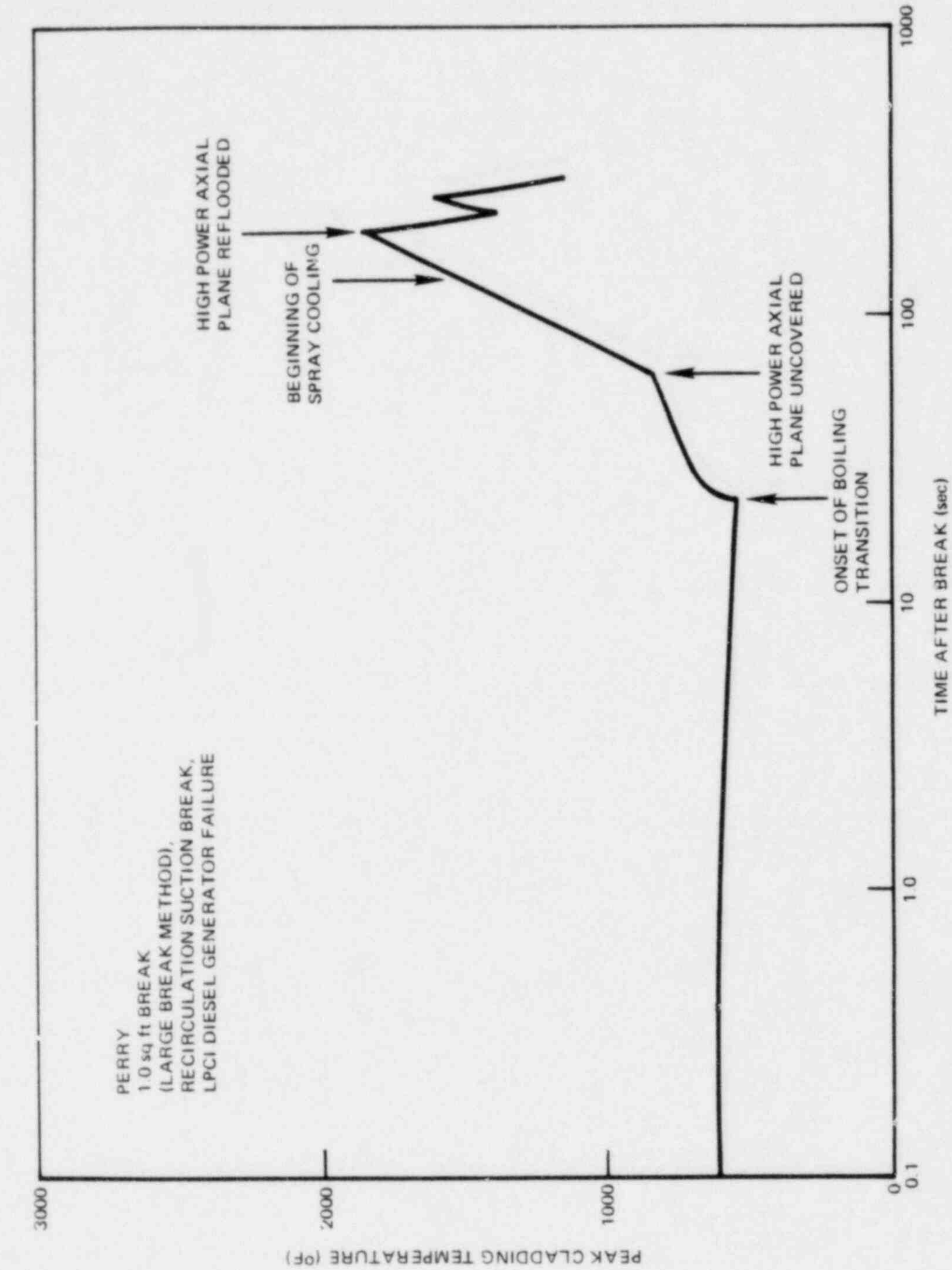
Figure 6.3-42



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Convective Heat Transfer Coefficient Versus Time After Break

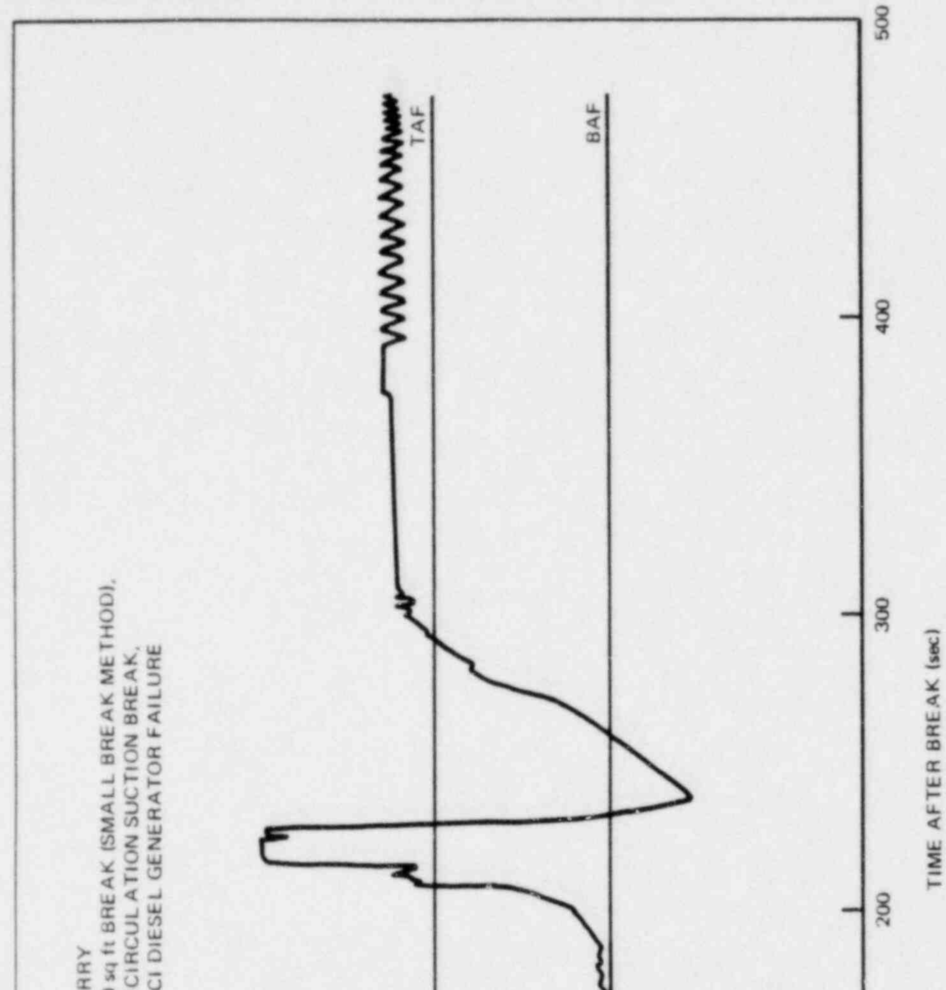
Figure 6.3-43



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Peak Cladding Temperature
Versus Time After Break

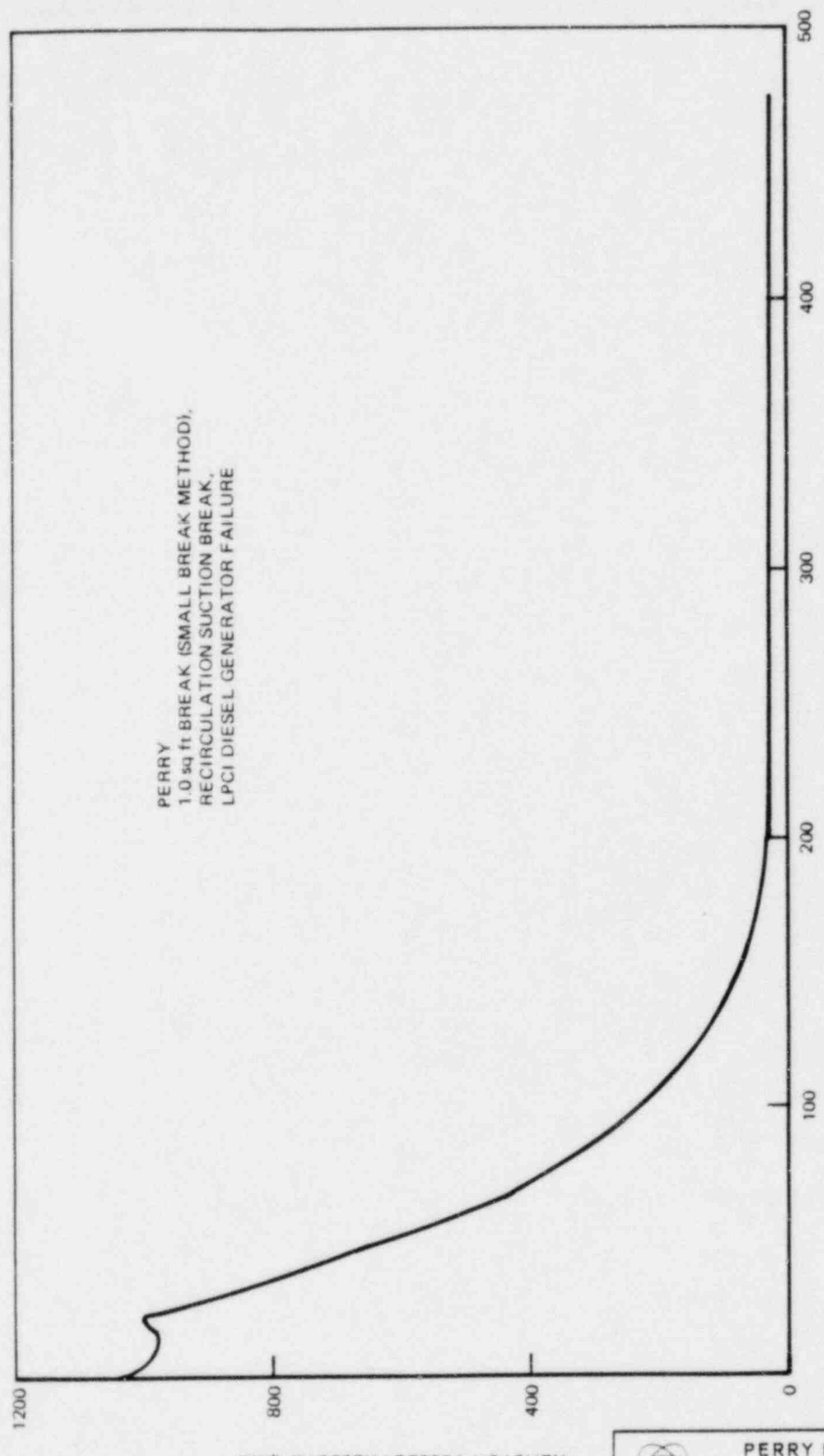
Figure 6.3-44



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Water Level Inside the Shroud
Versus Time After Break

Figure 6.3-45

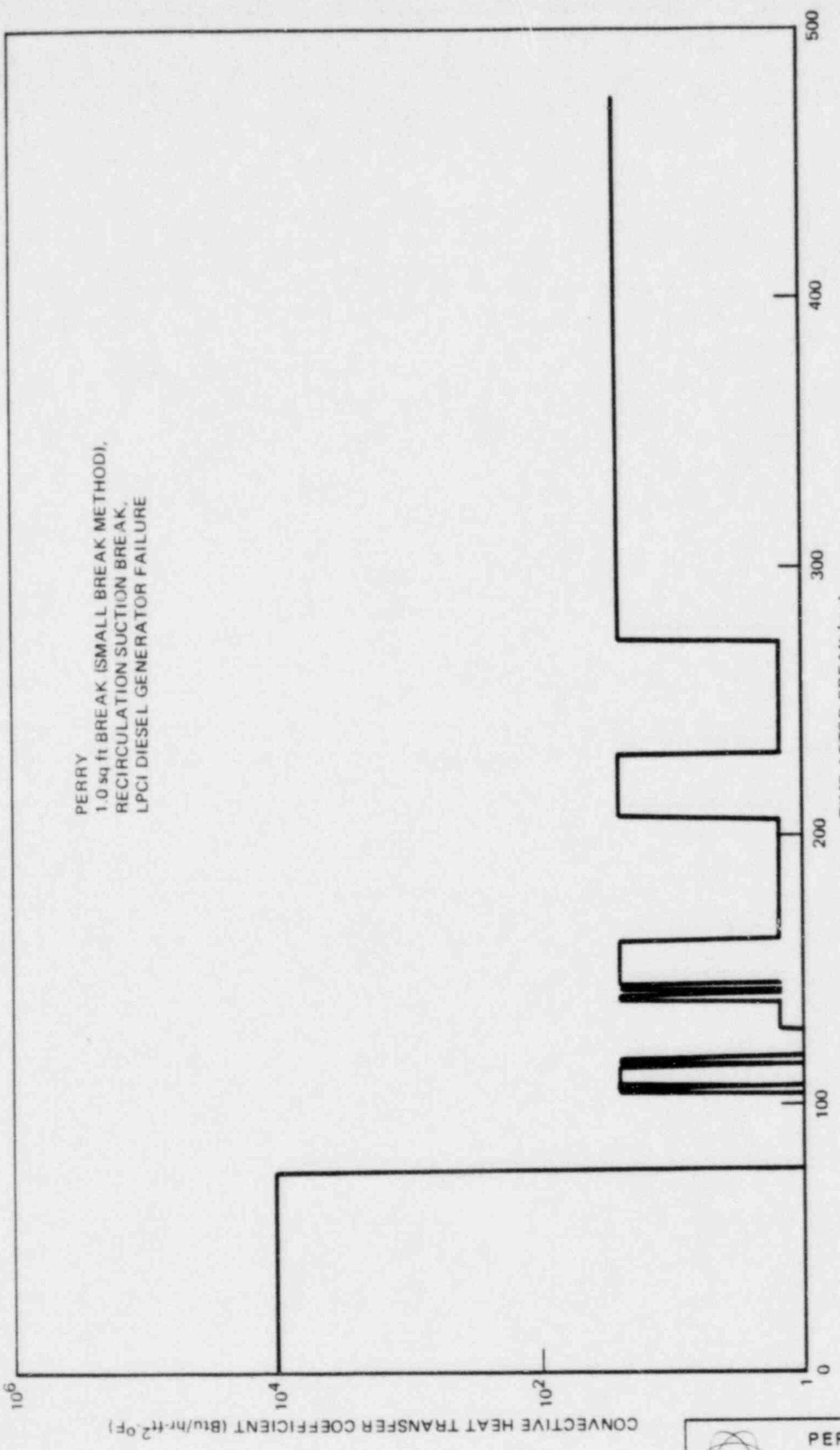


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Pressure
Versus Time After Break

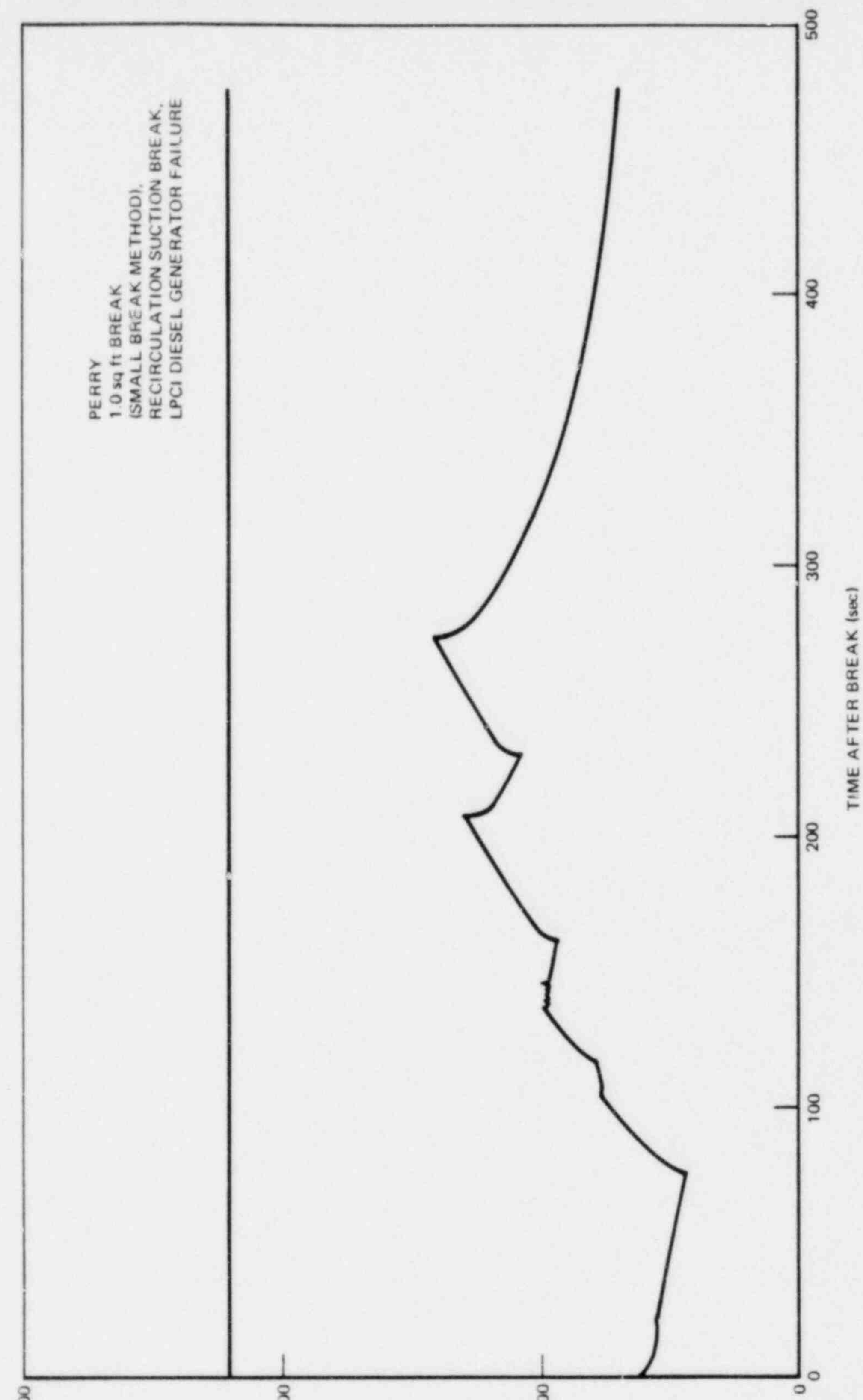
Figure 6.3-46



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Convective Heat Transfer Coefficient Versus Time After Break

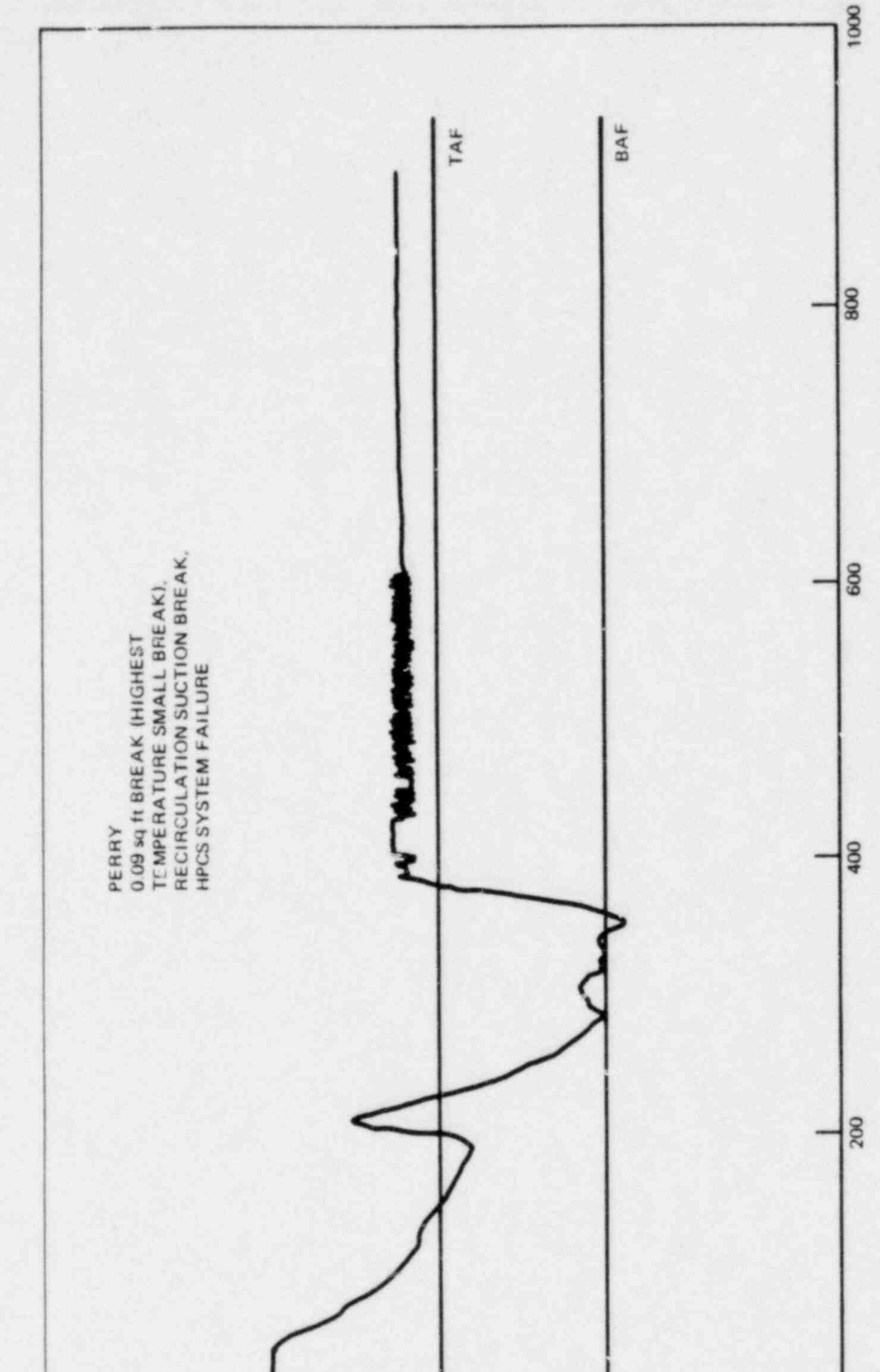
Figure 6.3-47



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Peak Cladding Temperature
Versus Time After Break

Figure 6.3-48

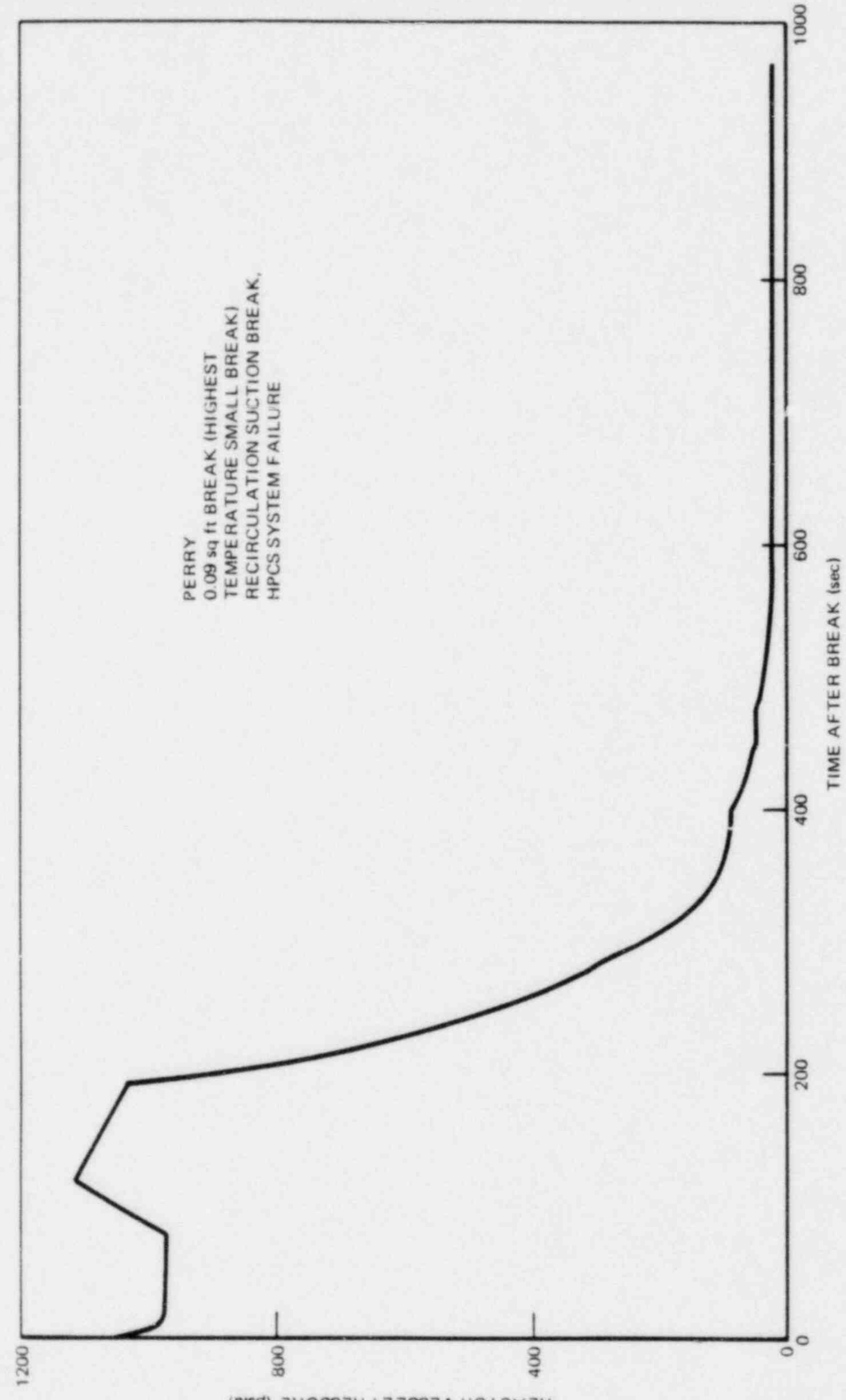


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Water Level Inside the Shroud
Versus Time After Break

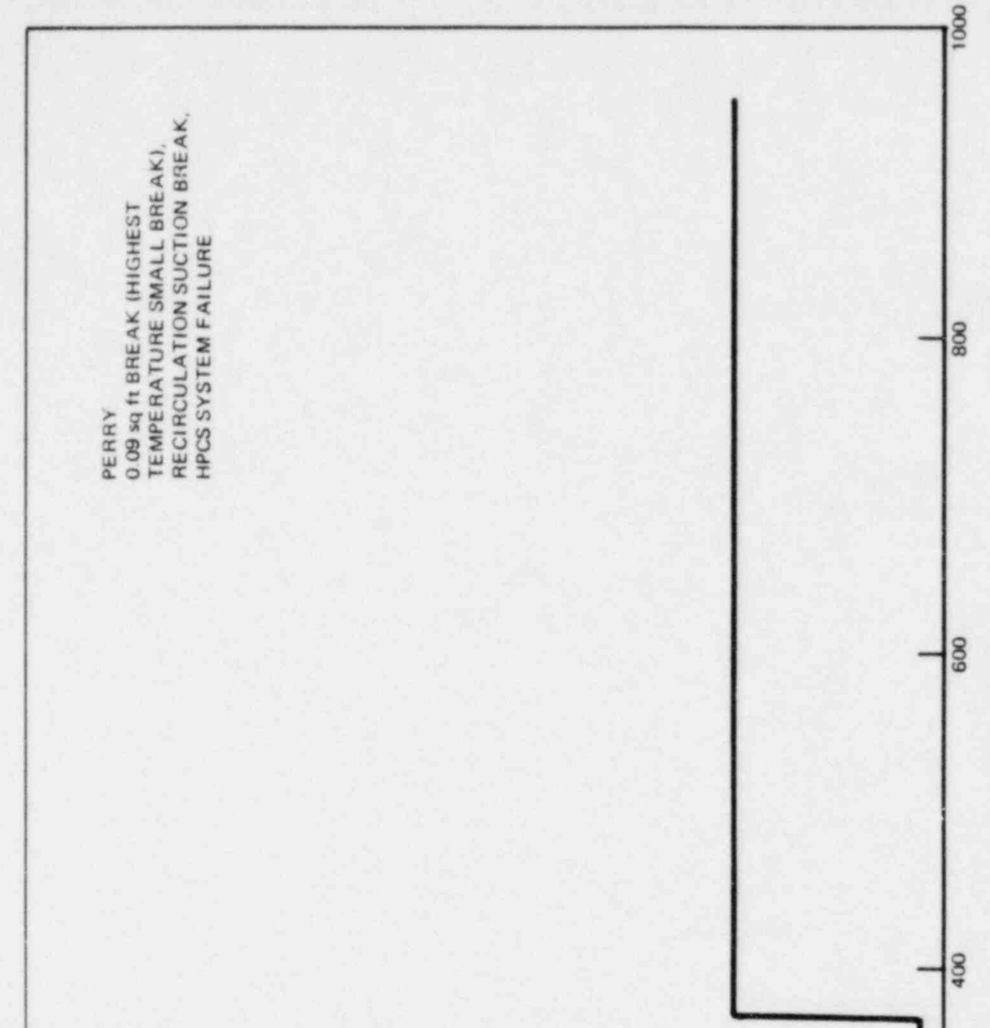
Figure 6.3-49



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Pressure
Versus Time After Break

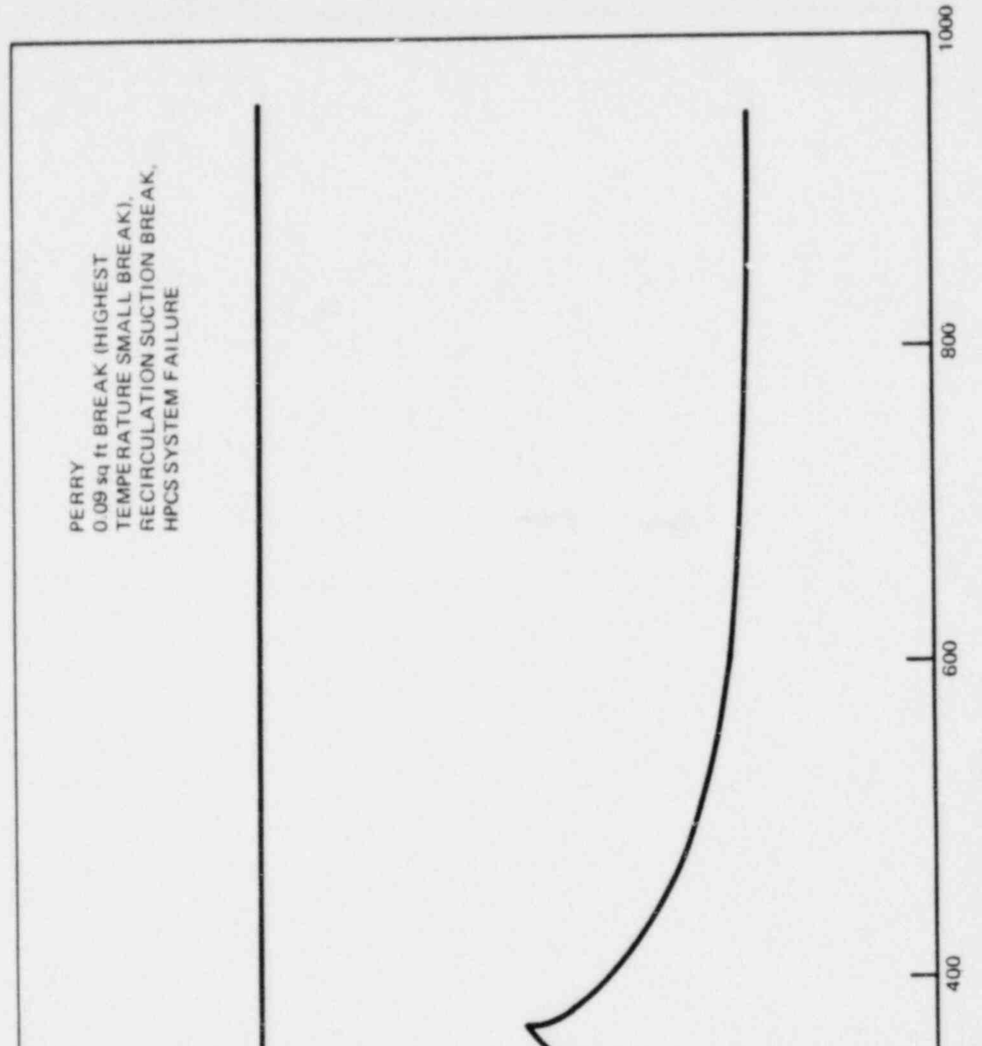
Figure 6.3-50



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Convective Heat Transfer
Coefficient Versus Time
After Break

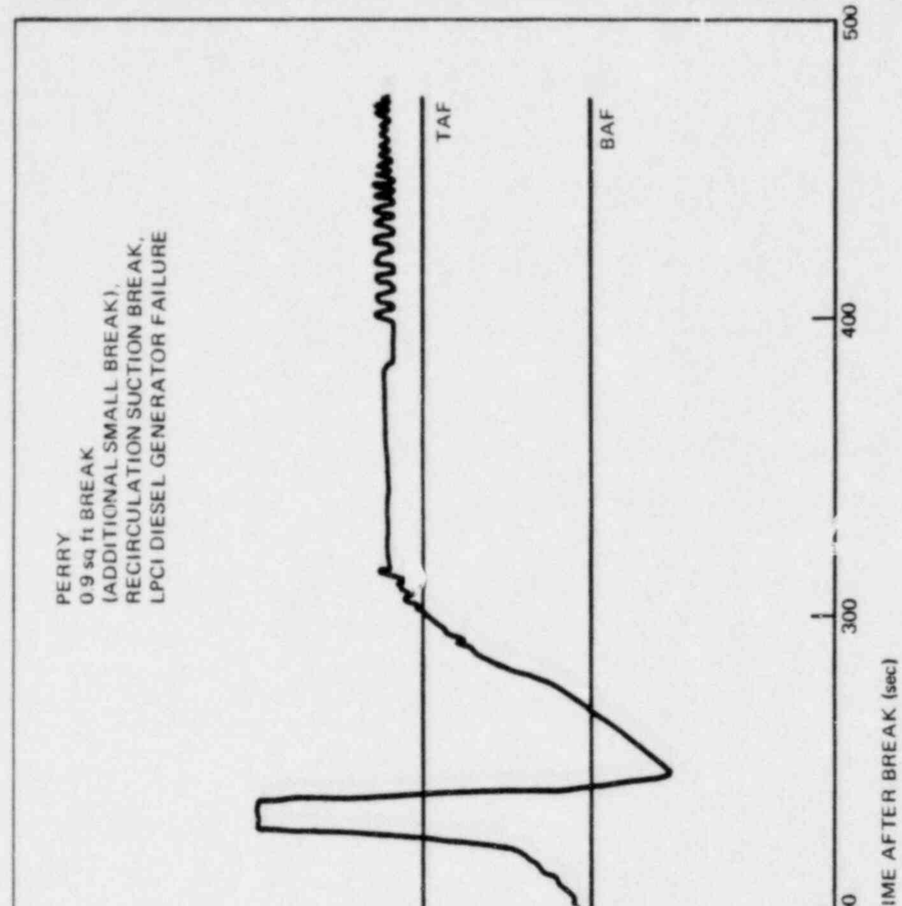
Figure 6.3-51



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Peak Cladding Temperature
Versus Time After Break

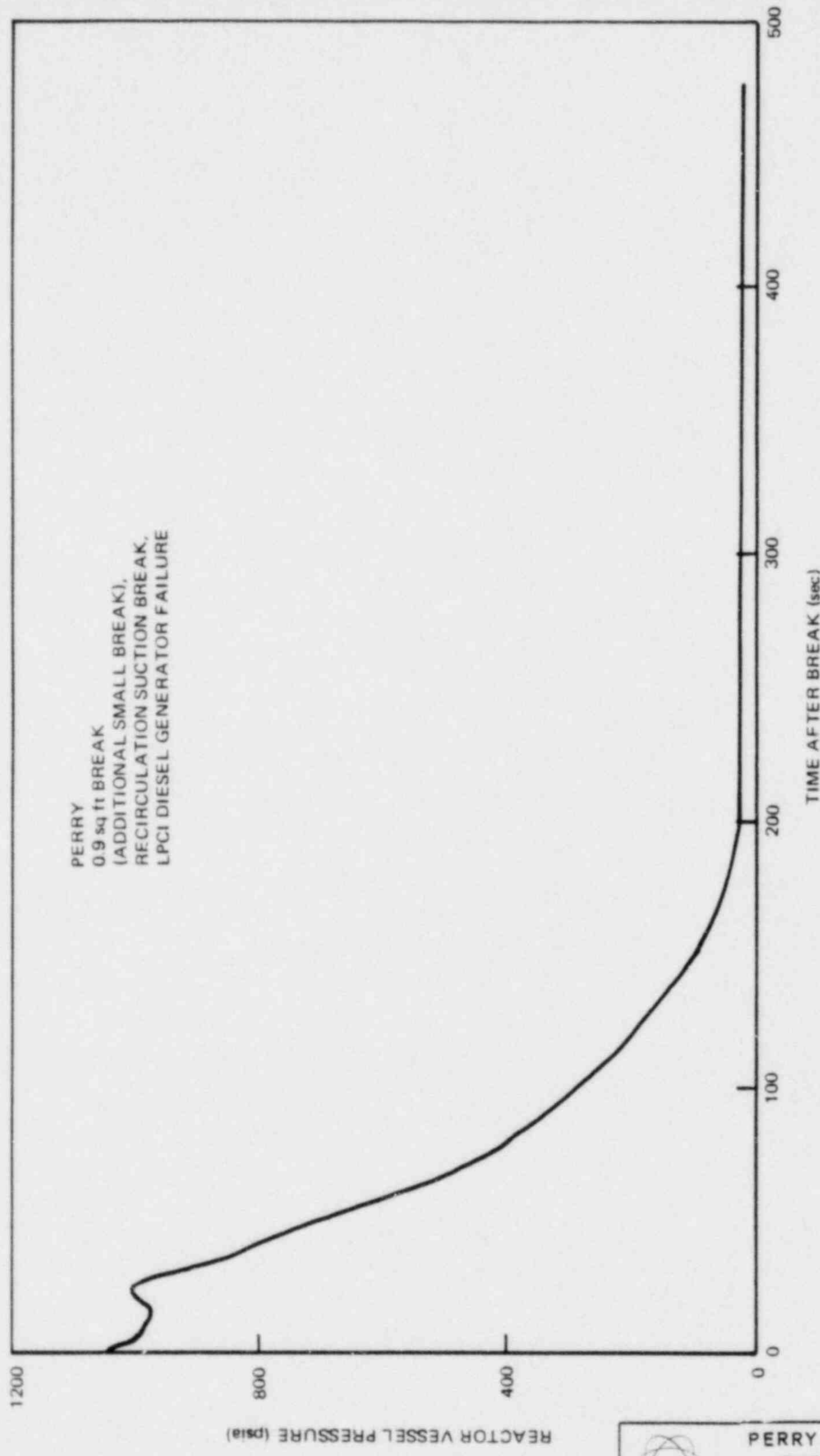
Figure 6.3-52



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Water Level Inside the Shroud
Versus Time After Break

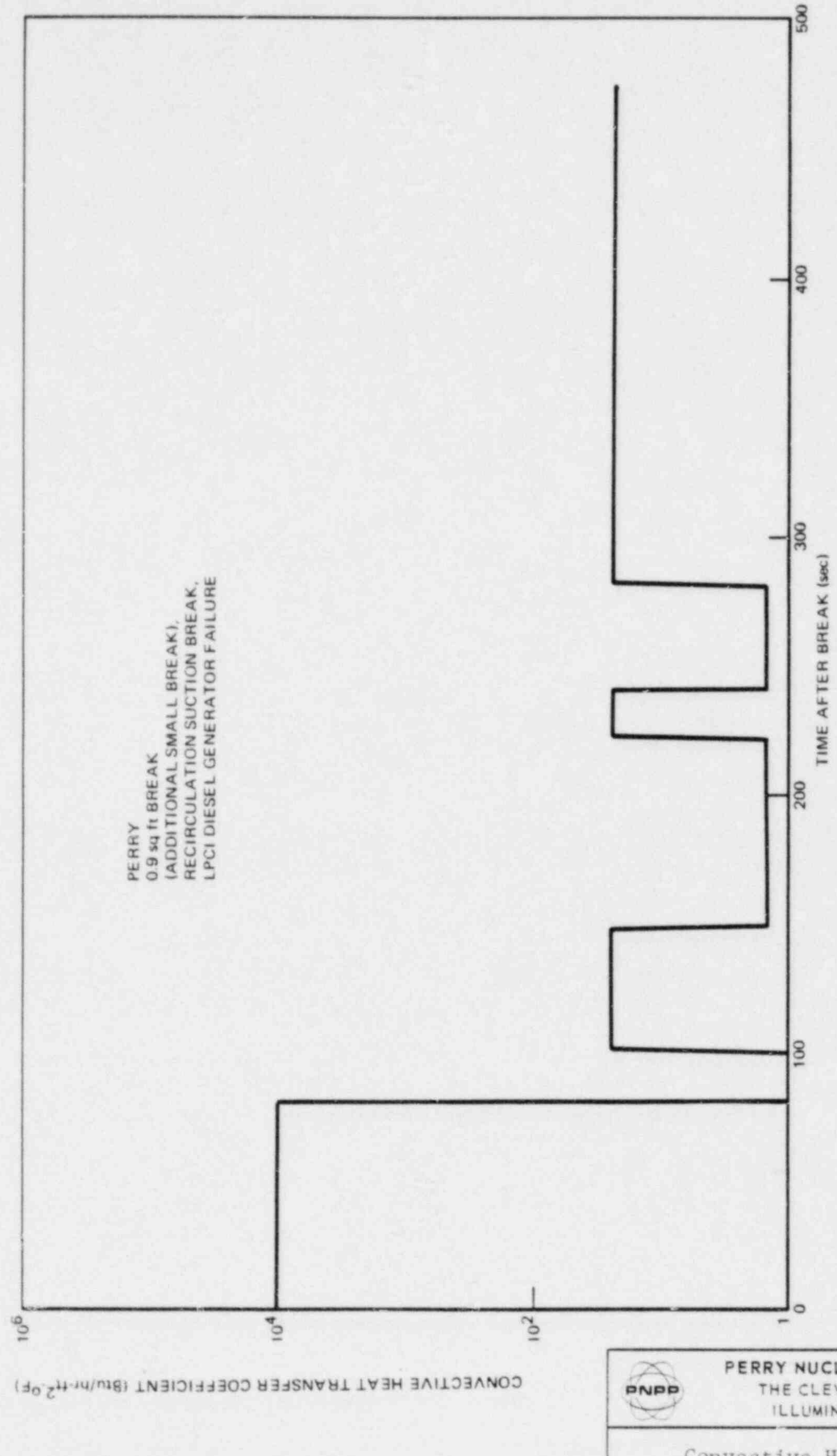
Figure 6.3-53



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Pressure
Versus Time After Break

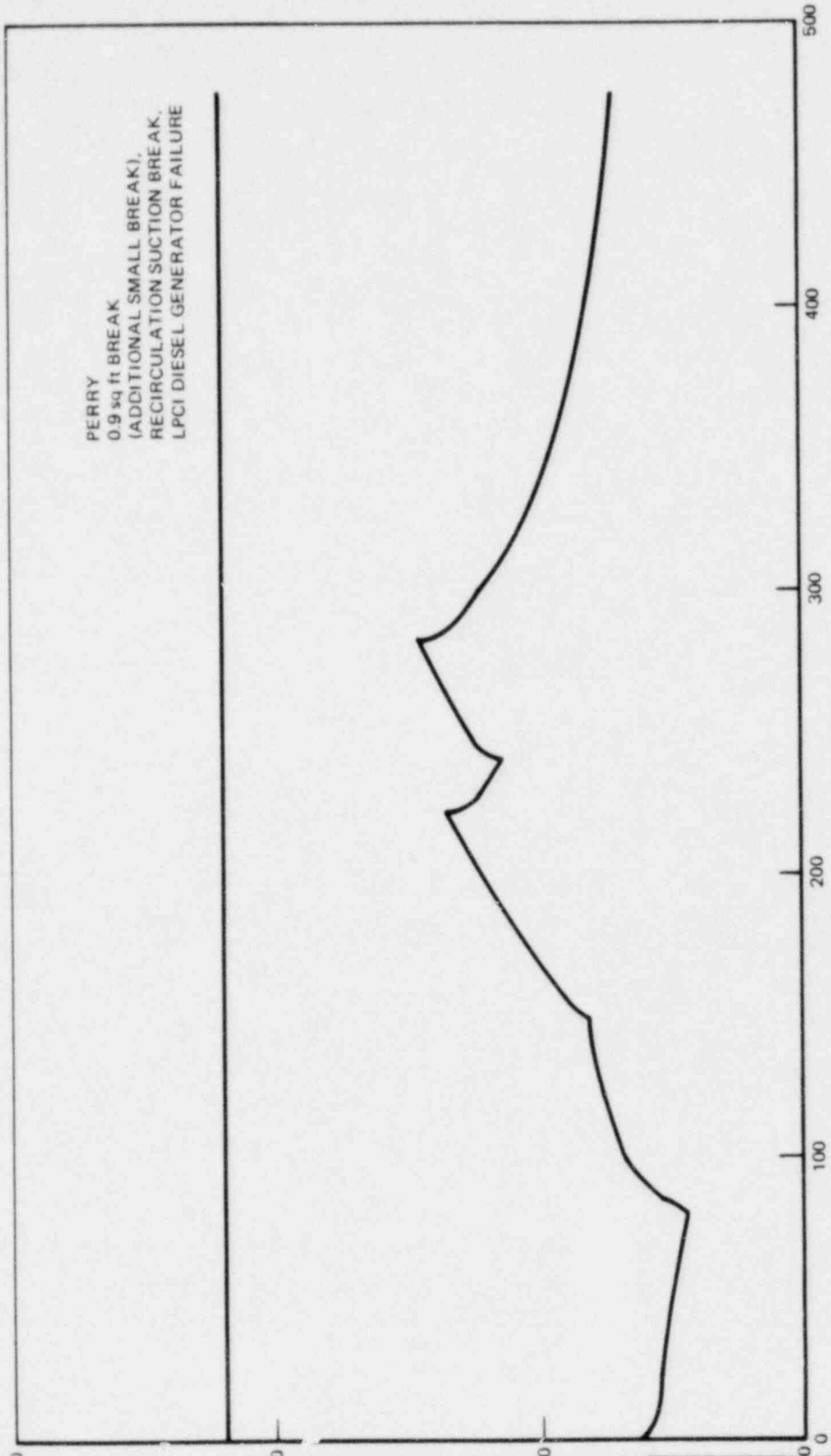
Figure 6.3-54



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Convective Heat Transfer
Coefficient Versus Time
After Break

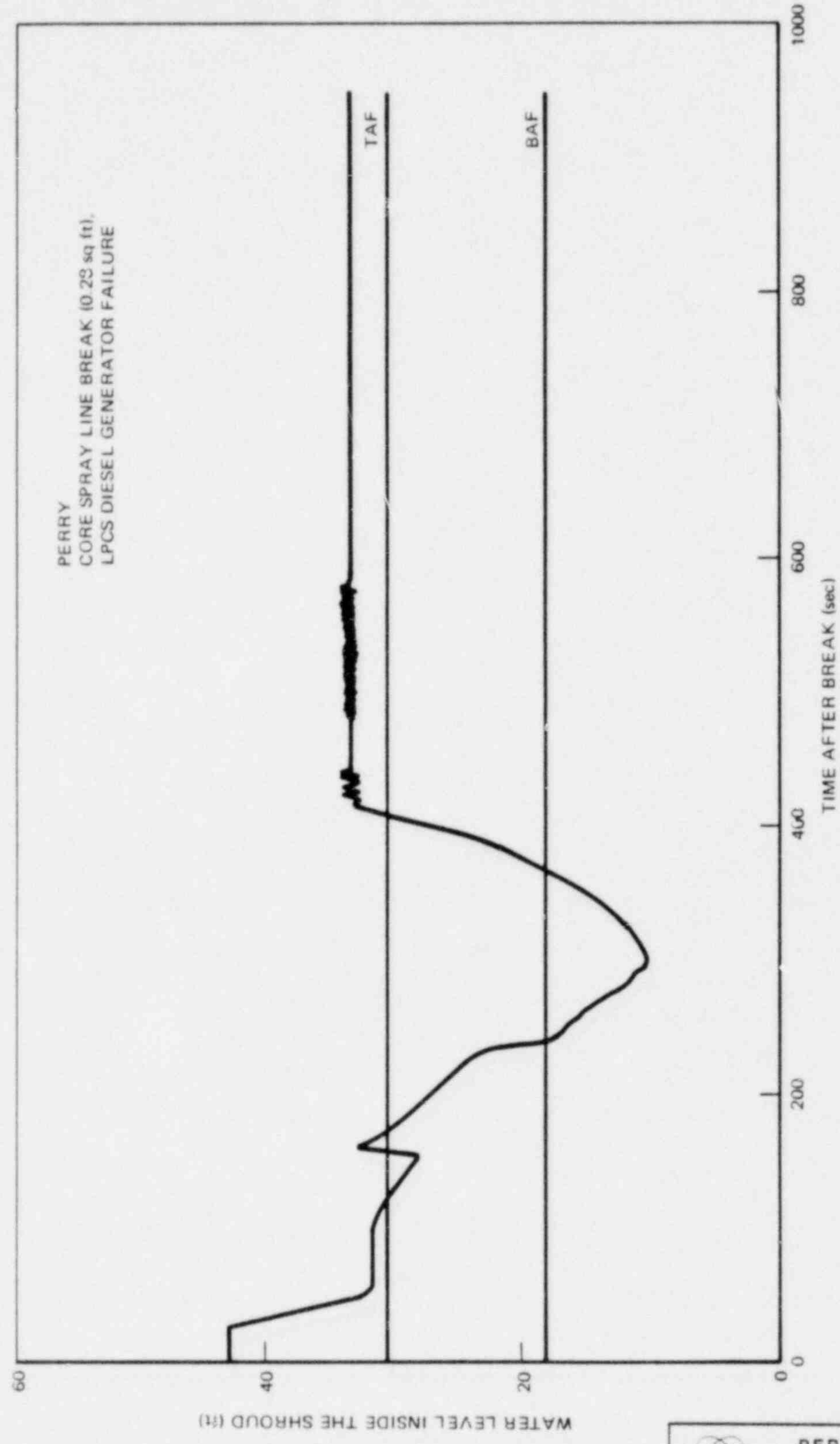
Figure 6.3-55



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Peak Cladding Temperature
Versus Time After Break

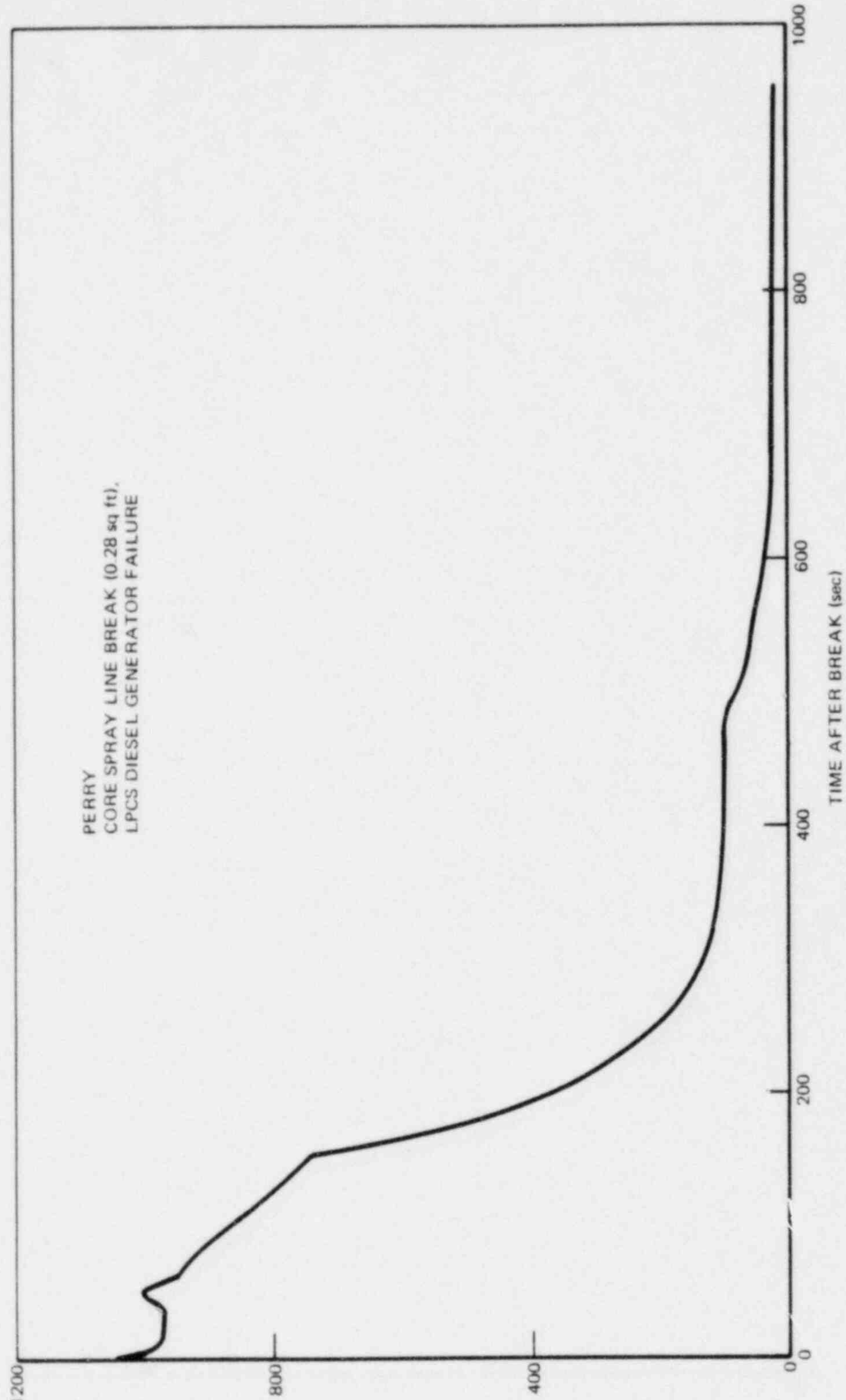
Figure 6.3-56



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Water Level Inside the Shroud
Versus Time After Break

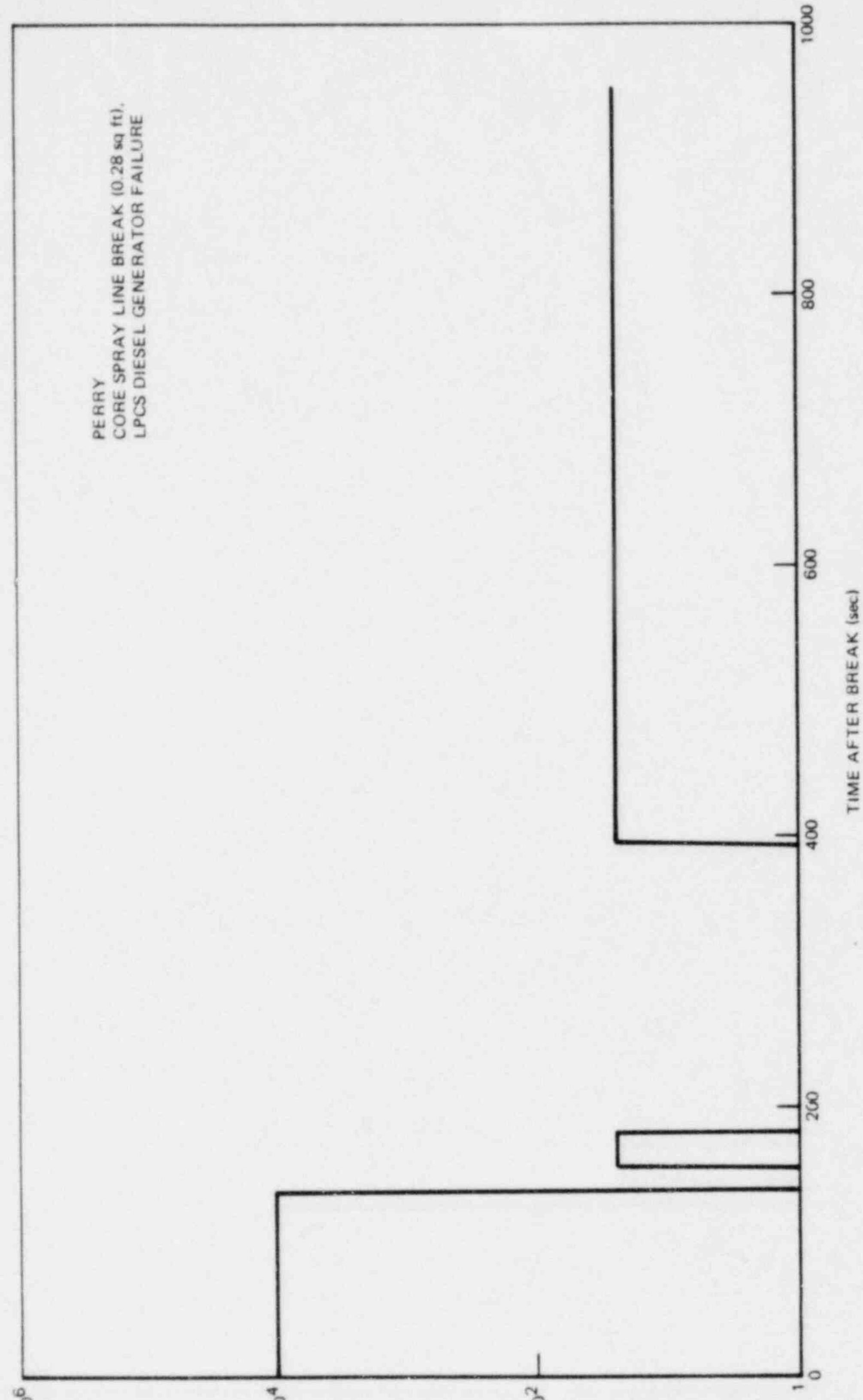
Figure 6.3-57



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Pressure
Versus Time After Break

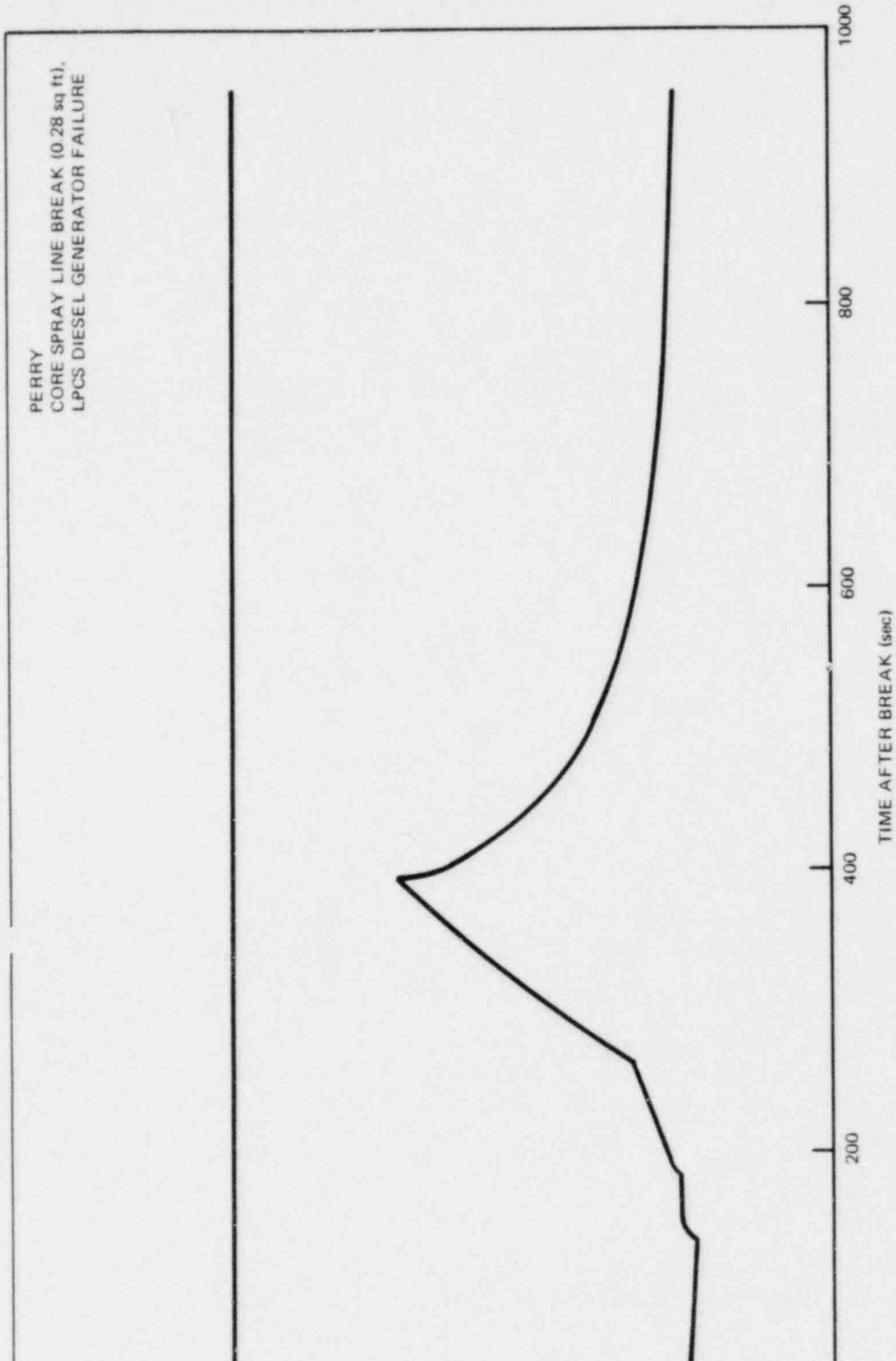
Figure 6.3-58



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Convective Heat Transfer
Coefficient Versus Time
After Break

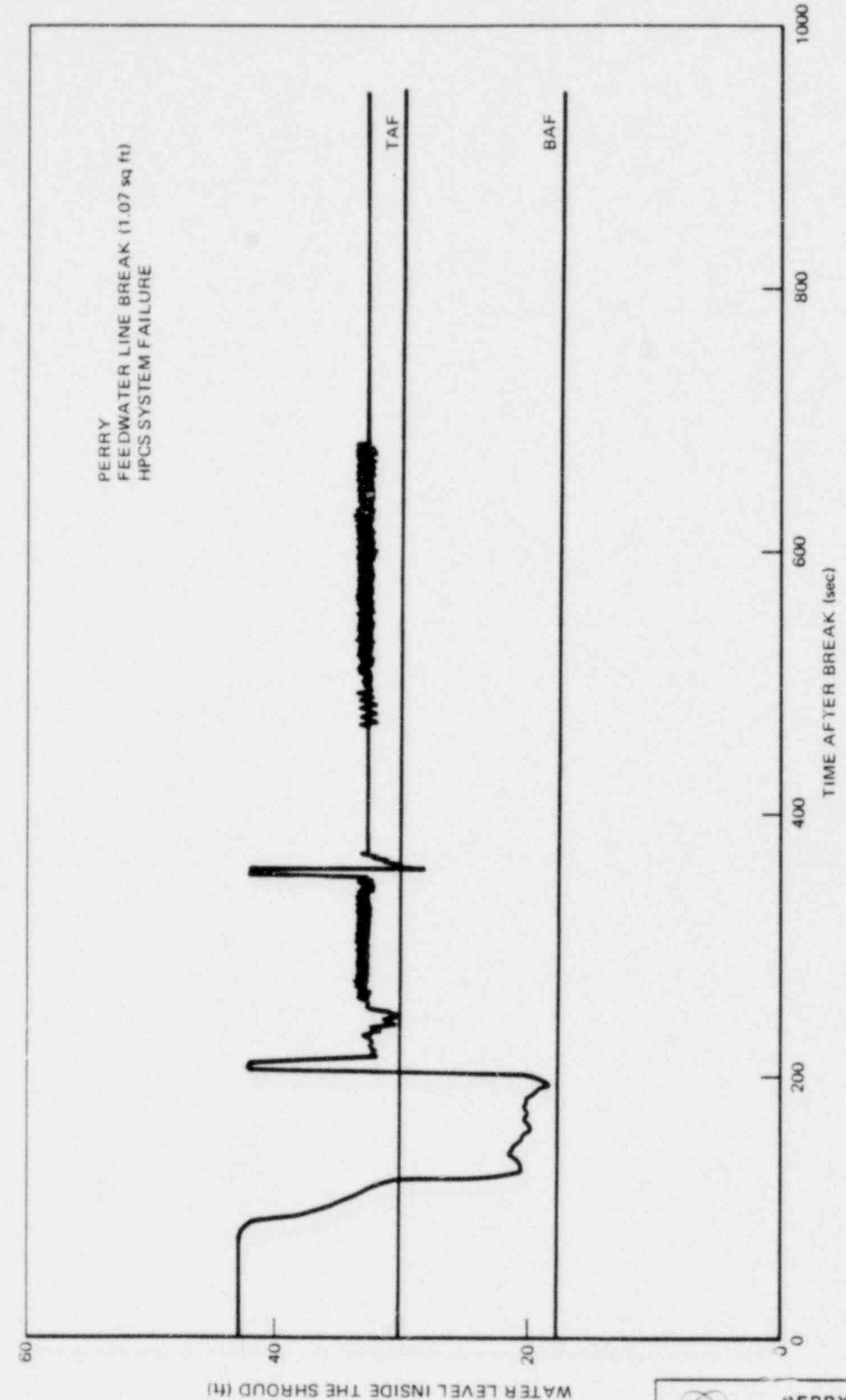
Figure 6.3-59



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Peak Cladding Temperature
Versus Time After Break

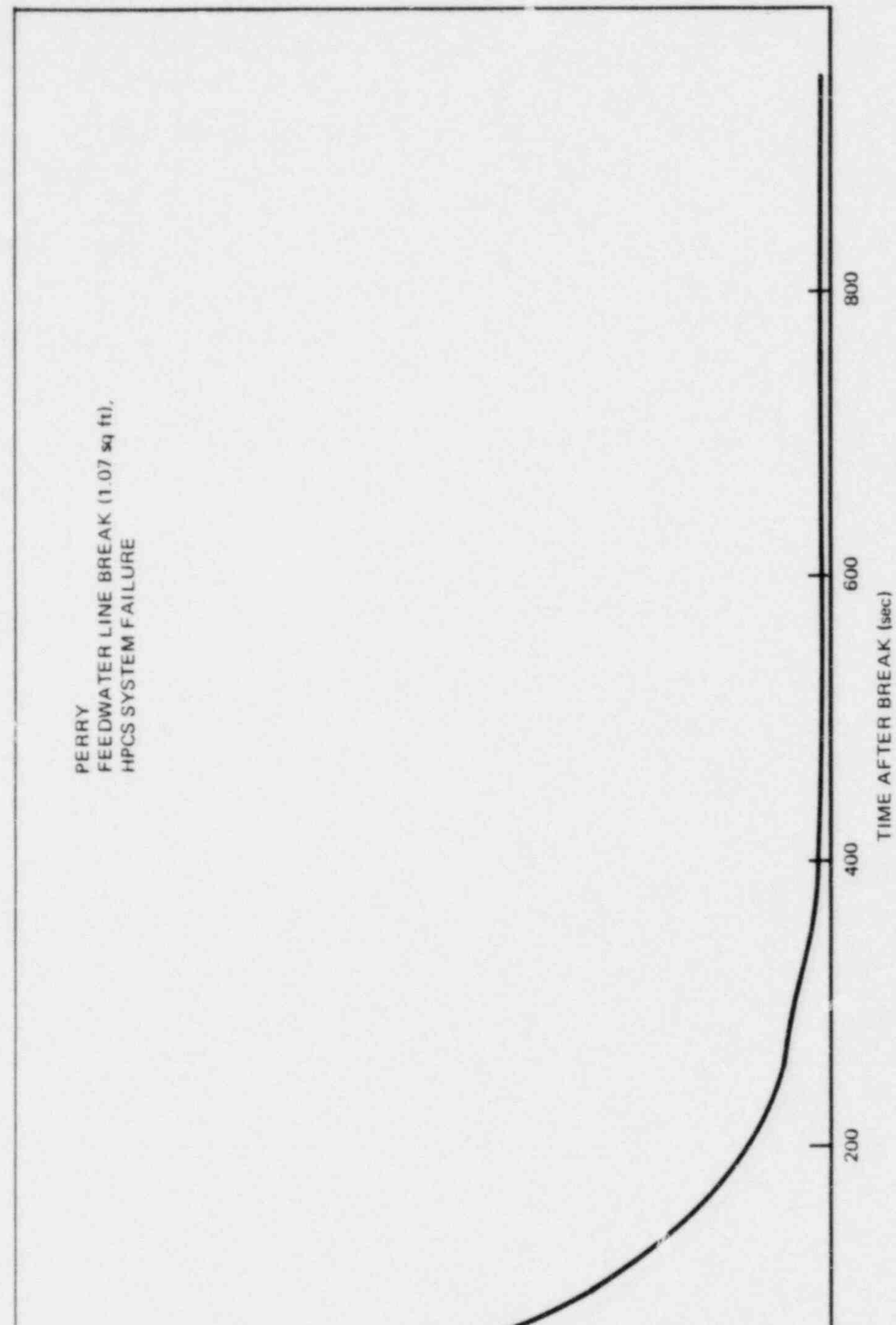
Figure 6.3-60



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Water Level Inside the Shroud
Versus Time After Break

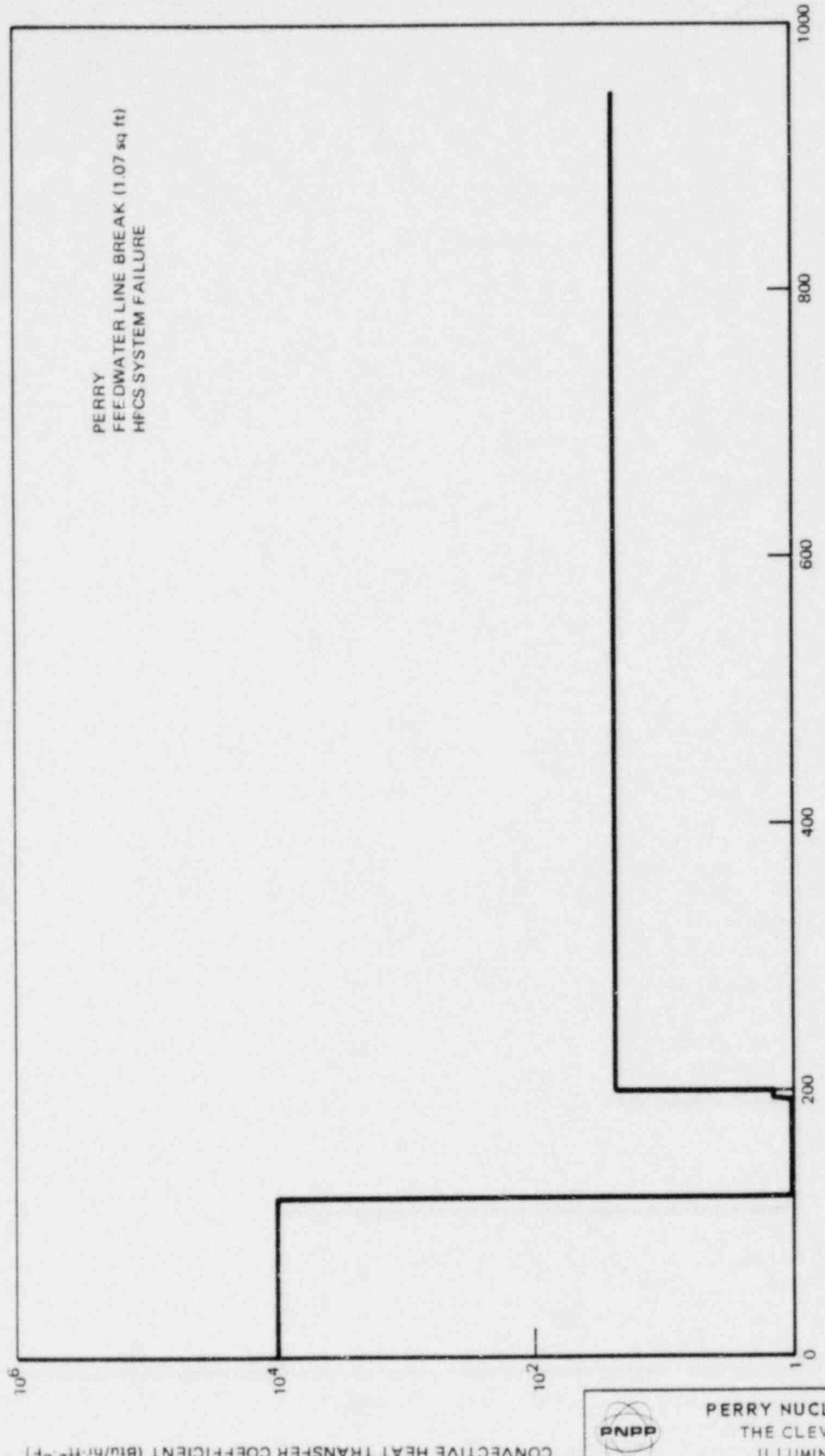
Figure 6.3-61



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Pressure Versus
Time After Break

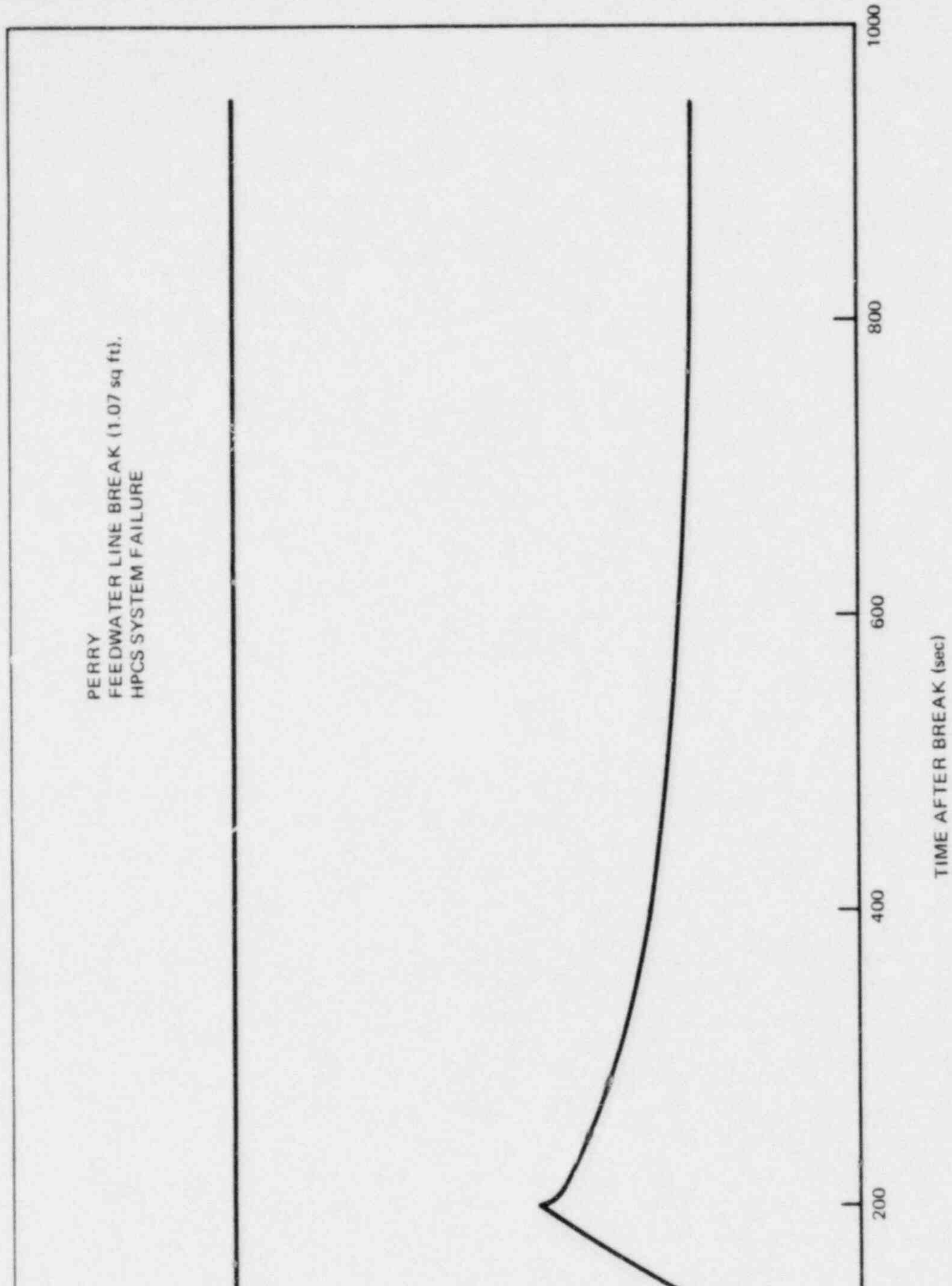
Figure 6.3-62



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Convective Heat Transfer
Coefficient Versus Time
After Break

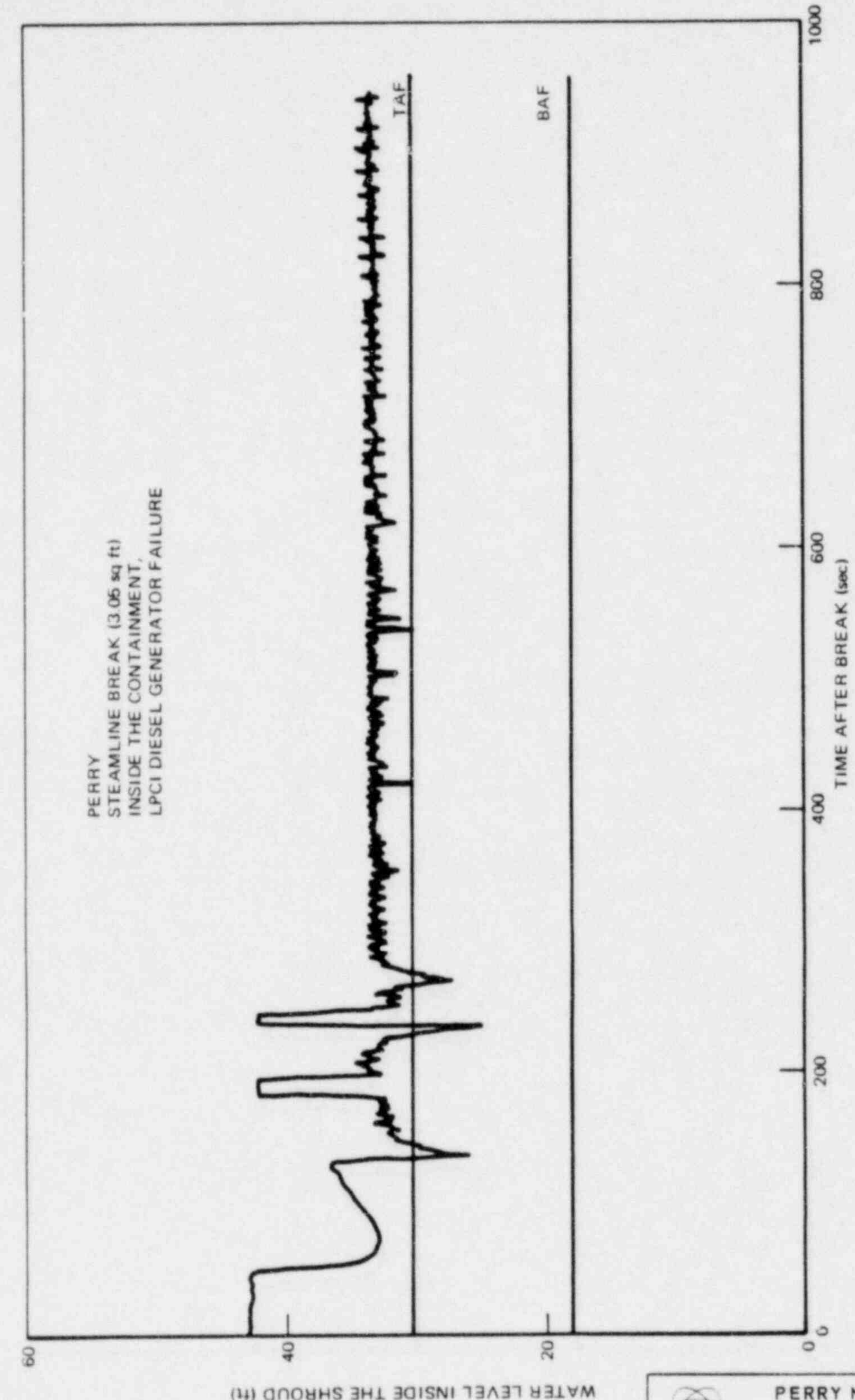
Figure 6.3-63



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Peak Cladding Temperature
Versus Time After Break

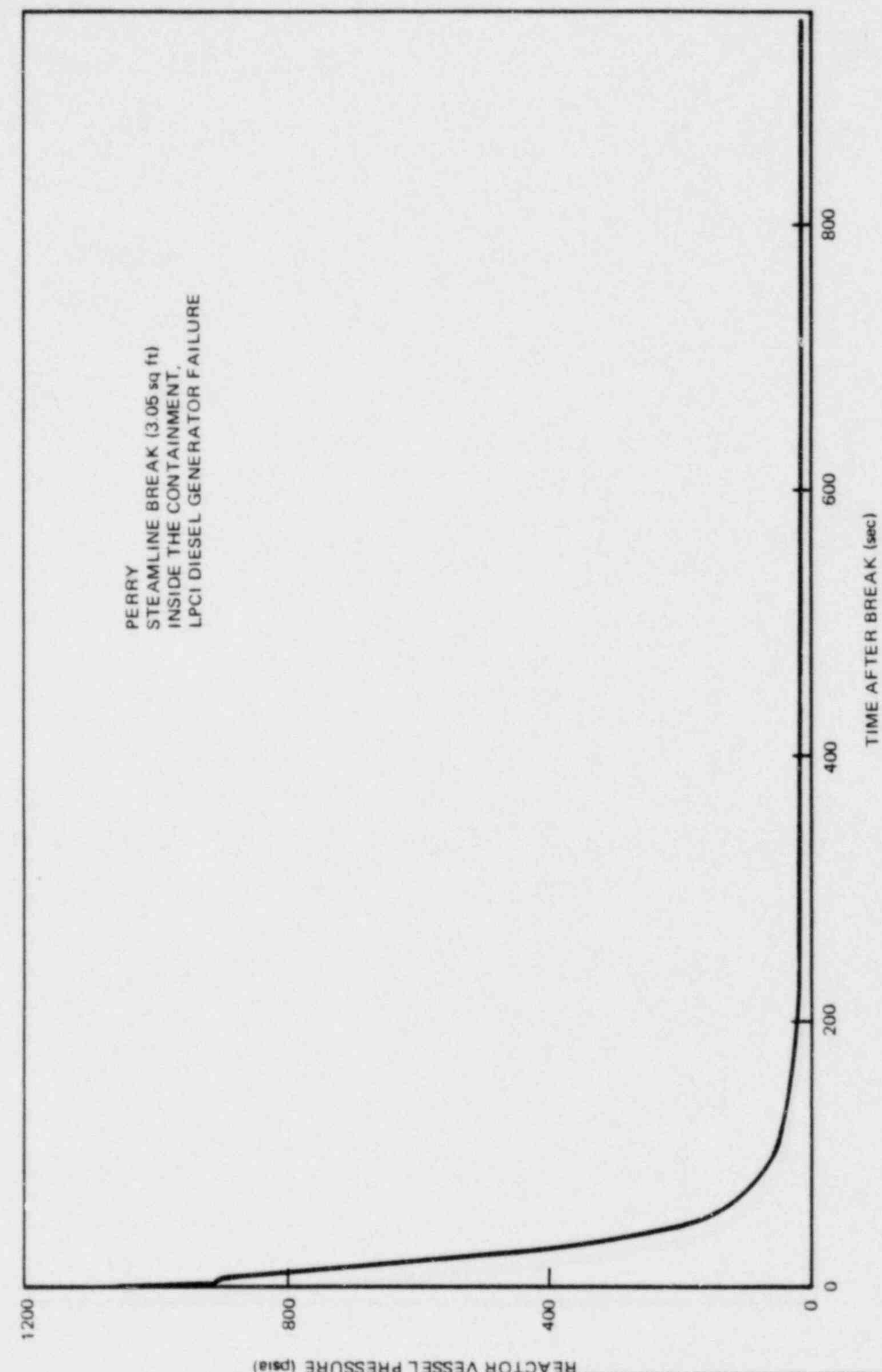
Figure 6.3-64



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Water Level Inside the Shroud
Versus Time After Break

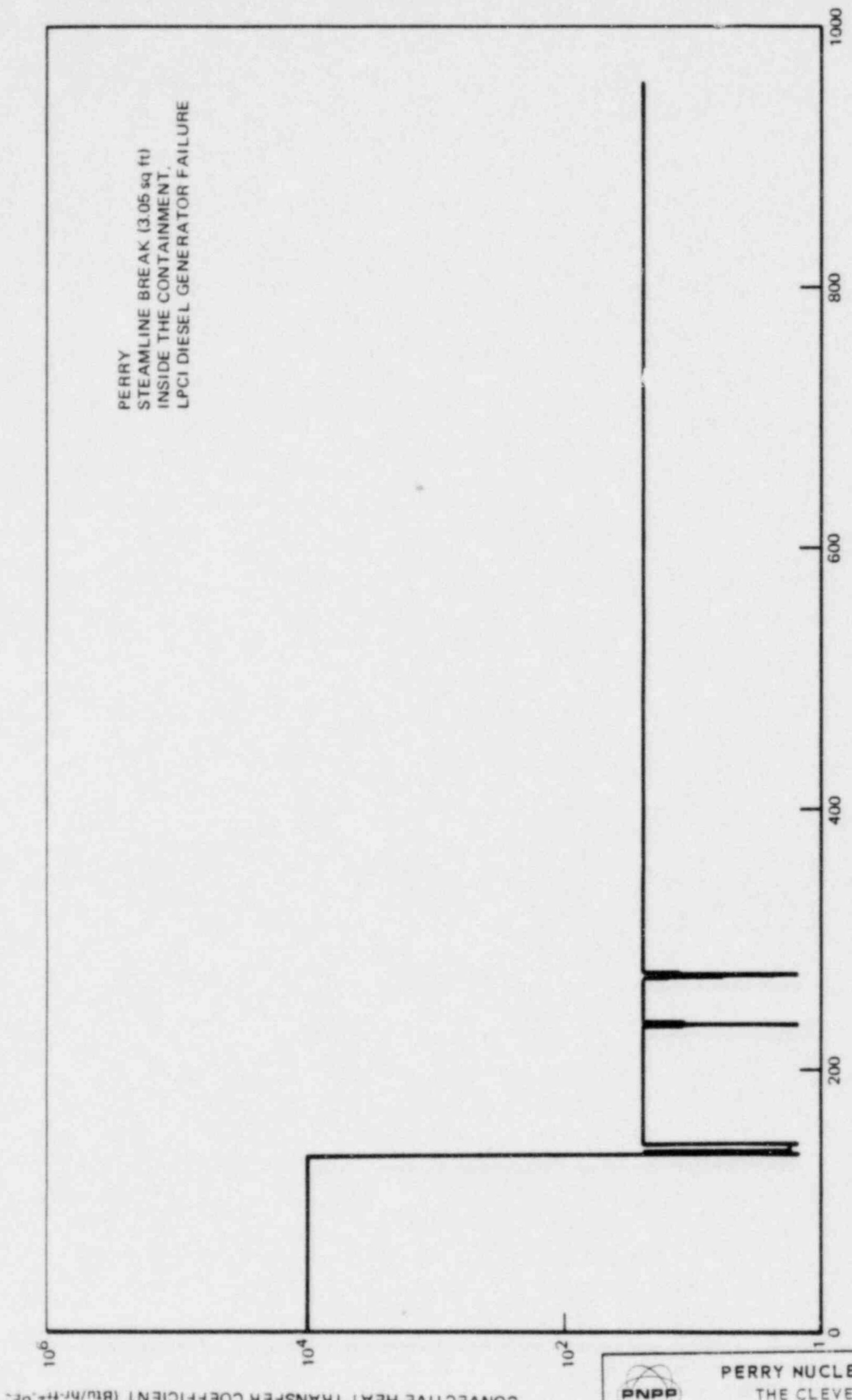
Figure 6.3-65



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Pressure Versus
Time After Break

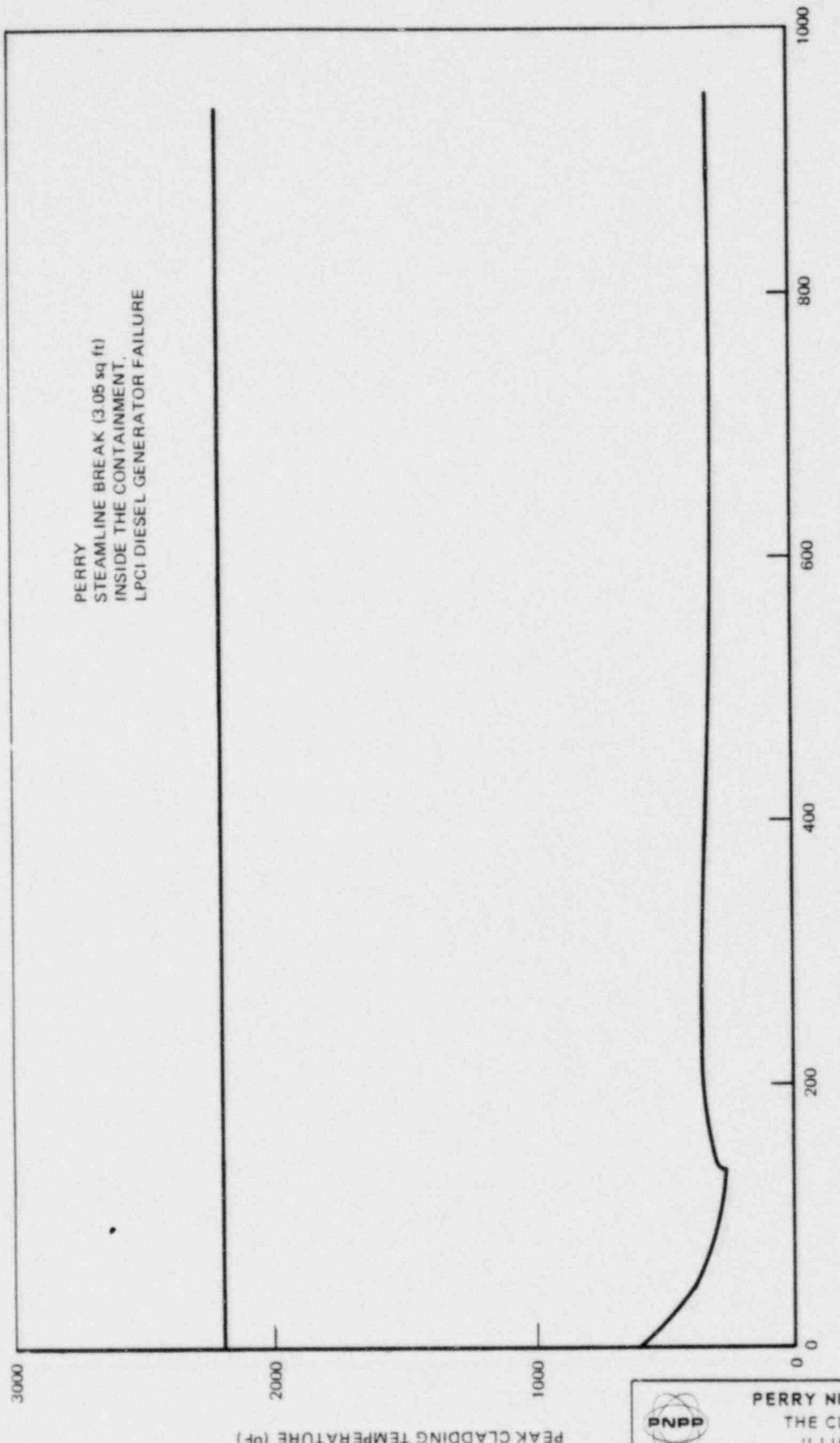
Figure 6.3-66



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Convective Heat Transfer
Coefficient Versus Time
After Break

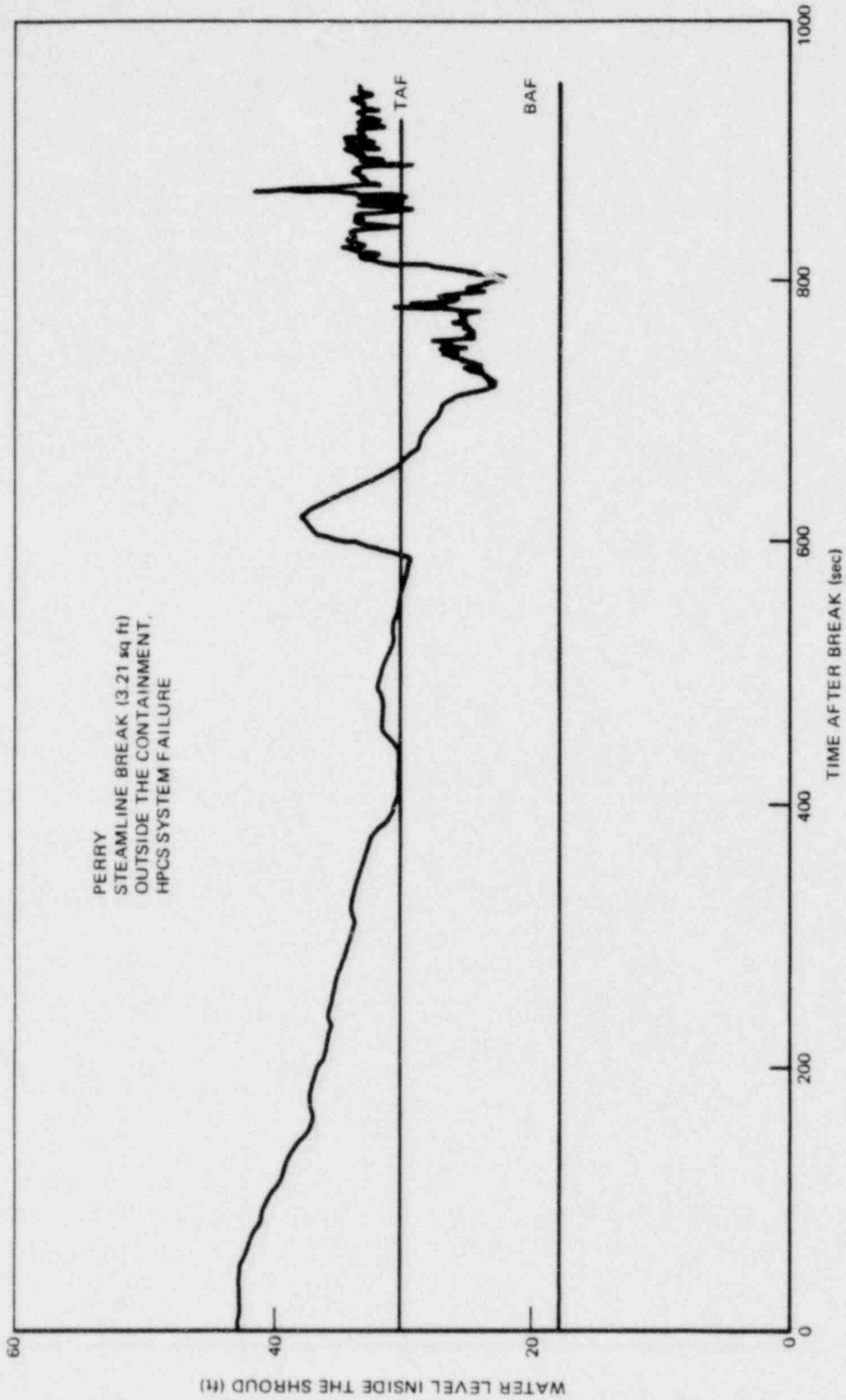
Figure 6.3-67



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Peak Cladding Temperature
Versus Time After Break

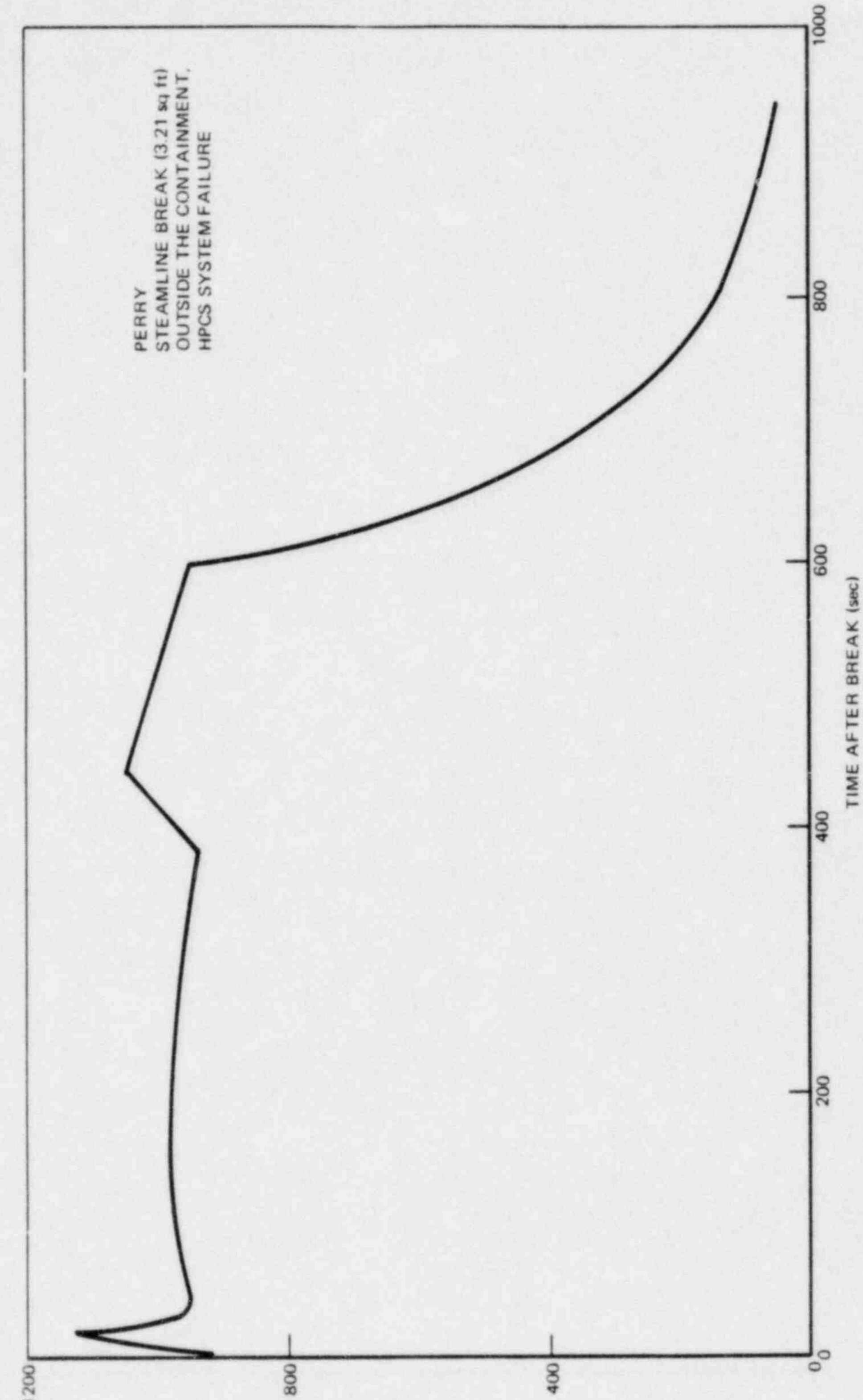
Figure 6.3-68



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Water Level Inside the Shroud
Versus Time After Break

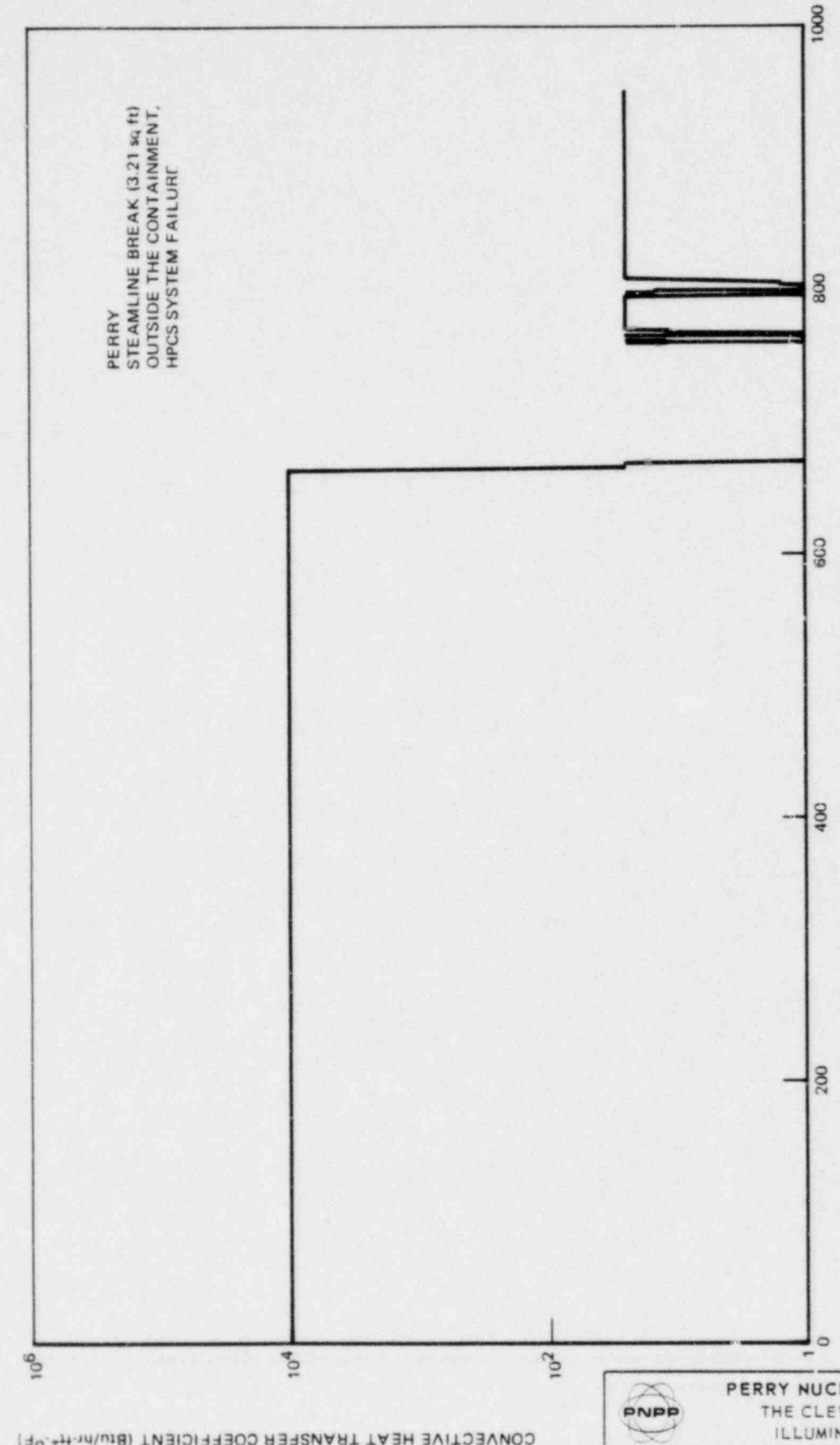
Figure 6.3-69



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Pressure
Versus Time after Break

Figure 6.3-70



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

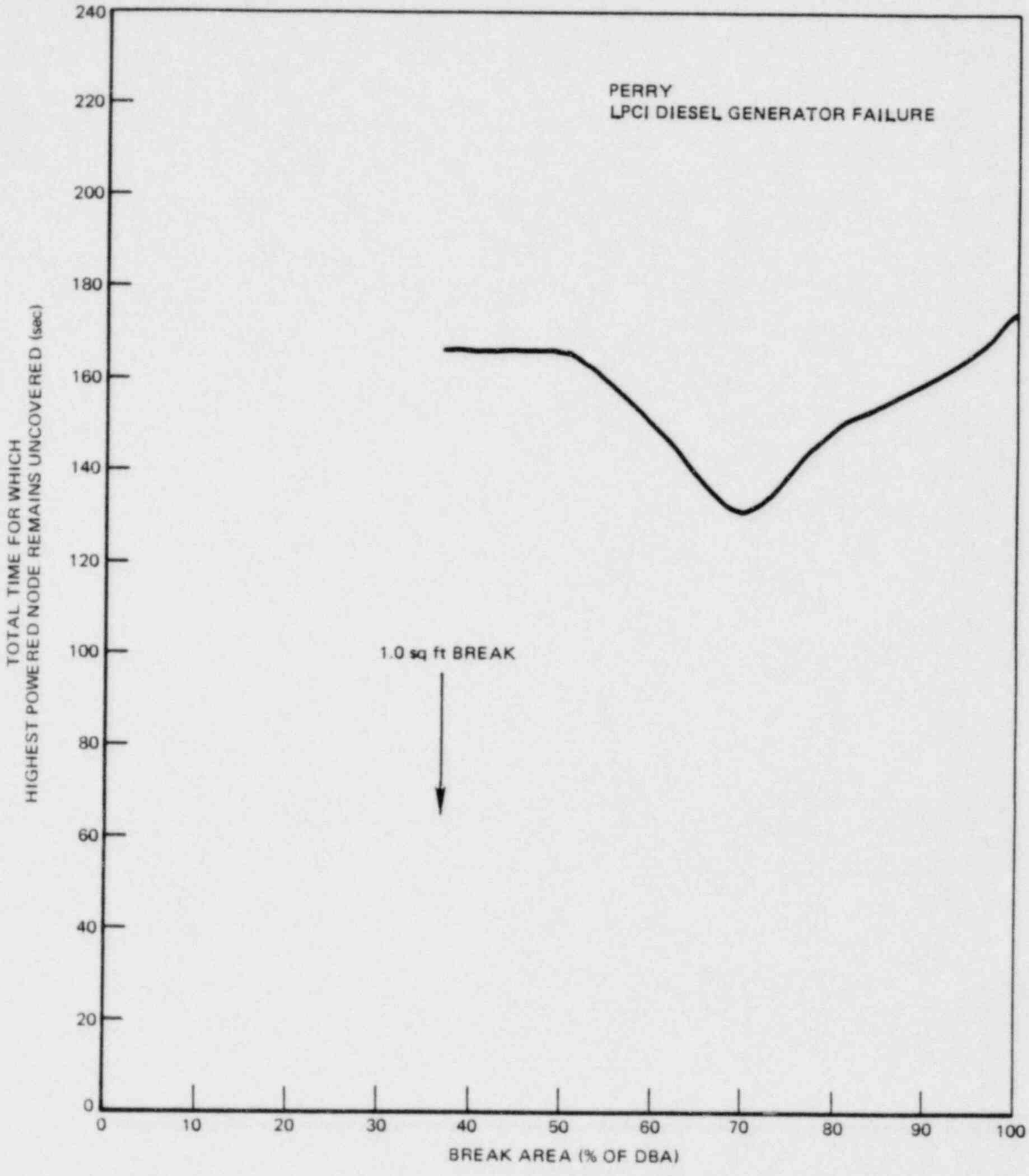
Convective Heat Transfer
Coefficient Versus Time
After Break

Figure 6.3-71

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Peak Cladding Temperature
Versus Time After Break

Figure 6.3-72



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Total Time for Which Highest
Powered Node Remains Uncovered
Versus Break Area

Figure 6.3-73

The rated capacity of the pressure relieving devices are sufficient to prevent a rise in pressure within the protected vessel of more than 110 percent of the design pressure (1.10 x 1250 psig = 1,375 psig) for events defined in Section 15.2.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

Table 5.2-3 lists the systems which could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

To design the pressure protection for the nuclear boiler system, extensive analytical models representing all essential dynamic characteristics of the system are simulated on a large computing facility. These models include the hydrodynamics of the flow loop, the reactor kinetics, the thermal characteristics of the fuel and its transfer of heat to the coolant, and all the principal controller features, such as feedwater flow, recirculation flow, reactor water level, pressure, and load demand. These are represented with all their principal nonlinear features in models that have evolved through extensive experience and favorable comparison of analysis with actual BWR test data.

A detailed description of this model is documented in licensing topical report NEDO-24154, "Qualification of the One Dimensional Core Transient Model for BWR."[1] Safety/relief valves are simulated in a nonlinear representation, and the model thereby allows full investigation of the various valve response times, valve capacities and acutuation setpoints that are available in applicable hardware systems.

Typical valve characteristics as modeled are shown in Figures 5.2-1 and 5.2-2 for the power activated relief and spring action safety modes of the dual purpose safety/relief valves. The associated bypass, turbine control valve, main steam isolation valve and pump trip due to high reactor pressure characteristics are also simulated in the model.

5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the safety/relief valves based on the assumptions that follow.

5.2.2.2.2.1 Operating Conditions

Operating conditions are as follows:

a. Operating power is 3,729 MW_t (104.2 percent of nuclear boiler rated power).

b. Vessel dome pressure $\leq$ 1,045 psig.

c. Steam flow is 16.71 x 10^6 lb/hr (105 percent of nuclear boiler rated steam flow).

d. Nuclear characteristics: End of Cycle.

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

5.2.2.2.2.2 Transients

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients, the closure of all main steam line isolation valves and turbine/generator trip with a coincident closure of the turbine steam bypass system valves that represent the most

severe abnormal operational transients resulting in a nuclear system pressure rise. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is

taken only for indirect derived scrams; therefore, it is used as the overpressure protection basis event and is shown in Figure 5.2-3. Table 5.2-4 lists the sequence of events for the main steam line isolation valve closure event with flux scram and with the installed safety/relief valve capacity.

5.2.2.2.2.3 Scram

The scram reactivity curve and control rod drive scram motion are illustrated by Figures 5.2-4 and 5.2-5, respectively.

5.2.2.2.2.4 Safety/Relief Valve Transient Analysis Specification

These assumptions are:

a. Simulated valve groups

Power actuated relief mode - 4 groups
Spring action safety mode - 5 groups

b. Opening pressure setpoint (maximum safety limit)

Power actuated relief mode -	Group 1:	1,145 psig
	Group 2:	1,155 psig
	Group 3:	1,165 psig
	Group 4:	1,175 psig

Spring action safety mode -	Group 1:	1,175 psig
	Group 2:	1,185 psig
	Group 3:	1,195 psig
	Group 4:	1,205 psig
	Group 5:	1,215 psig

The above analyses input set points are assumed at a conservatively high level above the nominal set points. This is to account for initial set point errors and any instrument set point drift that might occur during operation. Typically the

assumed setpoints in the analysis are 2 to 4 percent above the actual nominal set points. Highly conservative safety/relief valve response characteristics are also assumed. Therefore, the analysis conservatively bounds all safety/relief operating conditions.

5.2.2.2.2.5 Safety/Relief Valve Capacity

Sizing of the safety/relief valve capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1,375 psig) in response to the reference transients.

Whenever system pressure increases to the relief pressure set point of a group of valves having the same set point, half of those valves are assumed to operate in the relief mode, opened by the pneumatic power actuation. When the system pressure increases to the valve spring set pressure of a group of valves, those valves not already considered open are assumed to begin opening and to reach full open at 103 percent of the valve spring set pressure. By this method, the total valve capacity can be determined.

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 Safety/Relief Valve Capacity

The required safety/relief valve capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient. The plant is assumed to be operating at the turbine-generator design conditions at a maximum vessel dome pressure of 1,045 psig which is the maximum steady state operating pressure allowed by the Technical Specification. The analysis hypothetically assumes the failure of the direct isolation valve position scram. The reactor is shut down by the backup, indirect, high neutron flux scram. For the analysis, the power actuated relief set points of the safety/relief valve are assumed to be in the range of 1,145 to 1,175 psig and the spring action set points to be in the range of 1,175 to 1,215 psig. The resulting peak pressure at the bottom of the vessel is 1,276 psig. Therefore, the analysis indicates that the design valve capacity is capable of maintaining adequate margin below the peak ASME code allowable pressure in the nuclear system (1,375 psig). Figure 5.2-3 shows curves produced by this analysis. The sequence of events in Table 5.2-4 assumed in this analysis was investigated to meet code requirements and to evaluate the pressure relief system exclusively.

Under the General Requirements for Protection Against Overpressure as given in Section III of the ASME Boiler and Pressure Vessel Code, credit can be allowed for a scram from the reactor protection system. In addition, credit is also taken for the protective circuits which are indirectly derived when determining the required safety/relief valve capacity. The backup reactor high neutron flux scram is conservatively applied as a design basis in determining the required capacity of the pressure relieving dual purpose safety/relief valves. Application of the direct position scrams in the design basis could be used since they qualify as acceptable pressure protection devices when determining the required safety/relief valve capacity of nuclear vessels under the provisions of the ASME code. The safety/relief valves are operated in a relief mode (pneumatically) at set points lower than those specified for the safety function. This ensures sufficient margin between anticipated relief mode closing pressures and valve spring forces for proper seating of the valves.

The time response of the vessel pressure to the MSIV transient with flux scram is illustrated in Figure 5.2-8. This shows that the pressure at the vessel bottom exceeds 1,250 psig for less than five seconds. This is not long enough to transfer any appreciable amount of heat into the vessel metal which was at a temperature well below 550°F at the start of the transient.

The peak pressure results in this overpressure analysis bound all moderate frequency transients in Chapter 15.

5.2.2.2.3.2 Low-Low Set Relief Function

To assure that no more than one relief valve reopens following a reactor isolation event, two automatic depressurization system (ADS) safety/relief valves and four non-ADS valves are provided with lower opening and closing setpoints. These setpoints override the normal setpoints following the initial opening of the relief valves and act to hold open these valves longer, thus preventing more than a single valve to reopen subsequently. This system logic is referred to as the low-low set relief logic and functions to ensure that the containment design basis of one safety/relief valve operating on subsequent actuations is met.

The low-low set relief function is armed whenever any safety/relief valves are called upon to open in the relief mode by pressure instruments. Thus, the low-low set valves will not actuate during normal plant operation even though the reopening setpoints of one of the valves is in the normal operating pressure range. This arming method results in the low-low set safety/relief valves opening initially during an overpressure transient at the normal relief opening setpoint.

The lowest setpoint low-low set valve will cycle to remove decay heat. Since this valve will have a larger differential between its opening and closing set pressures than assumed for the normal relief function, the number of single safety relief valve actuations during isolation events will be reduced. Table 5.2-2 shows the opening and closing setpoints for the low-low set safety/relief valves.

The assumptions used in the calculation of the pressure transient after the initial opening of the relief valves are:

a. The transient event is a three-second closure of all MSIV's with position scram.

b. Nominal relief valve setpoints are used.

3. Operability checked by comparing one method versus another (sump fill up versus pump out and particulate monitoring, air cooler condensate flow versus sump fill up rate)

4. Continuous monitoring of floor drain sump level is provided.

These satisfy position c.8.

i. Limiting unidentified leakage to 5 gpm and identified to 25 gpm satisfies position c.9.

5.2.6 REFERENCES FOR SECTION 5.2

1. Qualification of One-Dimensionsal Core Transient Model for BWR, NEDO-24154, October, 1978.

2. J.M. Skarpelos and J.W. Bagg, "Chloride Control in BWR Coolants," June 1973, NEDO-10899.

3. W.L. Williams, Corrosion, Vol 13, 1957, p. 539t

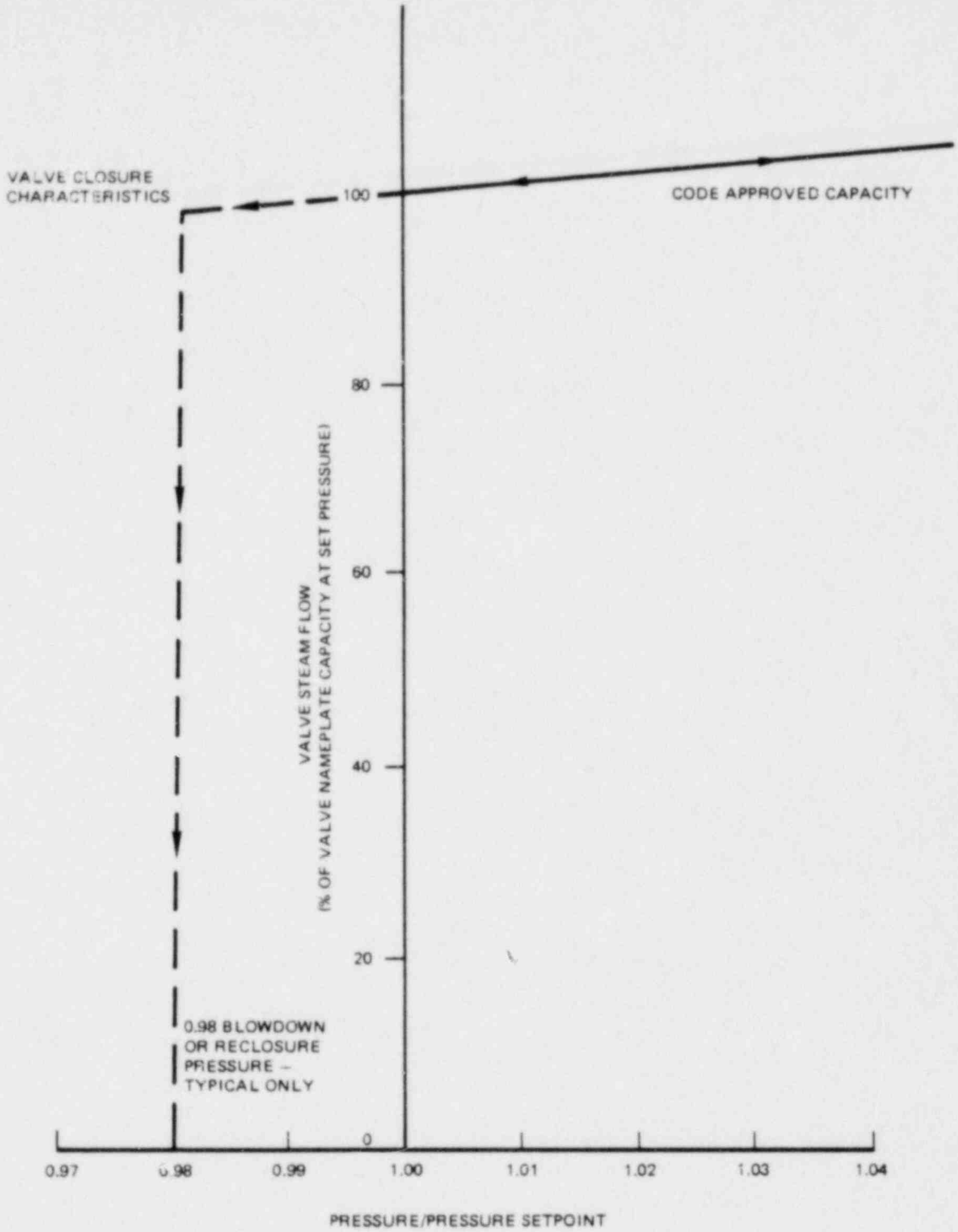
4. GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws, by M.B. Reynolds, April, 1968.

5. "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," NUREG-76/067, NRC/PCSG, dated October 1975.

TABLE 5.2-4

SEQUENCE OF EVENTS FOR FIGURE 5.2-3

Time-Sec	Events
0	Closure of all main steam isolation valves (MSIV) was initiated.
0.3	MSIVs reached 90% open. Failure of direct position scram was assumed.
1.6	Neutron flux reached the high APRM flux scram setpoint and initiated reactor scram.
2.1	Reactor dome pressure reached the setpoint of recirculation pump trip.
2.1	Reactor dome pressure reached the group 1 safety/relief valves pressure setpoint (power-actuated mode). Only one half of valves in this group was assumed functioning.
2.3	Steamline pressure reached the group 1 safety/relief valves pressure setpoint (spring-action mode). Valves which were not opened in this power-actuated mode were opened.
2.4	Recirculation pump initiated to coastdown.
2.8	All safety/relief valves opened in either power-actuated mode or spring action mode due to high pressure.
2.9	Vessel bottom pressure reached its peak value.
3.0	MSIVs completely closed.
>10 (est)	Safety/relief valves opened in their spring-action mode closed.
>20 (est)	Wide-range sensed water level reached L2 setpoint. HPCS and RCIC flow entered reactor vessel. Safety valves closed and reopen cyclicly.



PERRY NUCLEAR POWER PLANT
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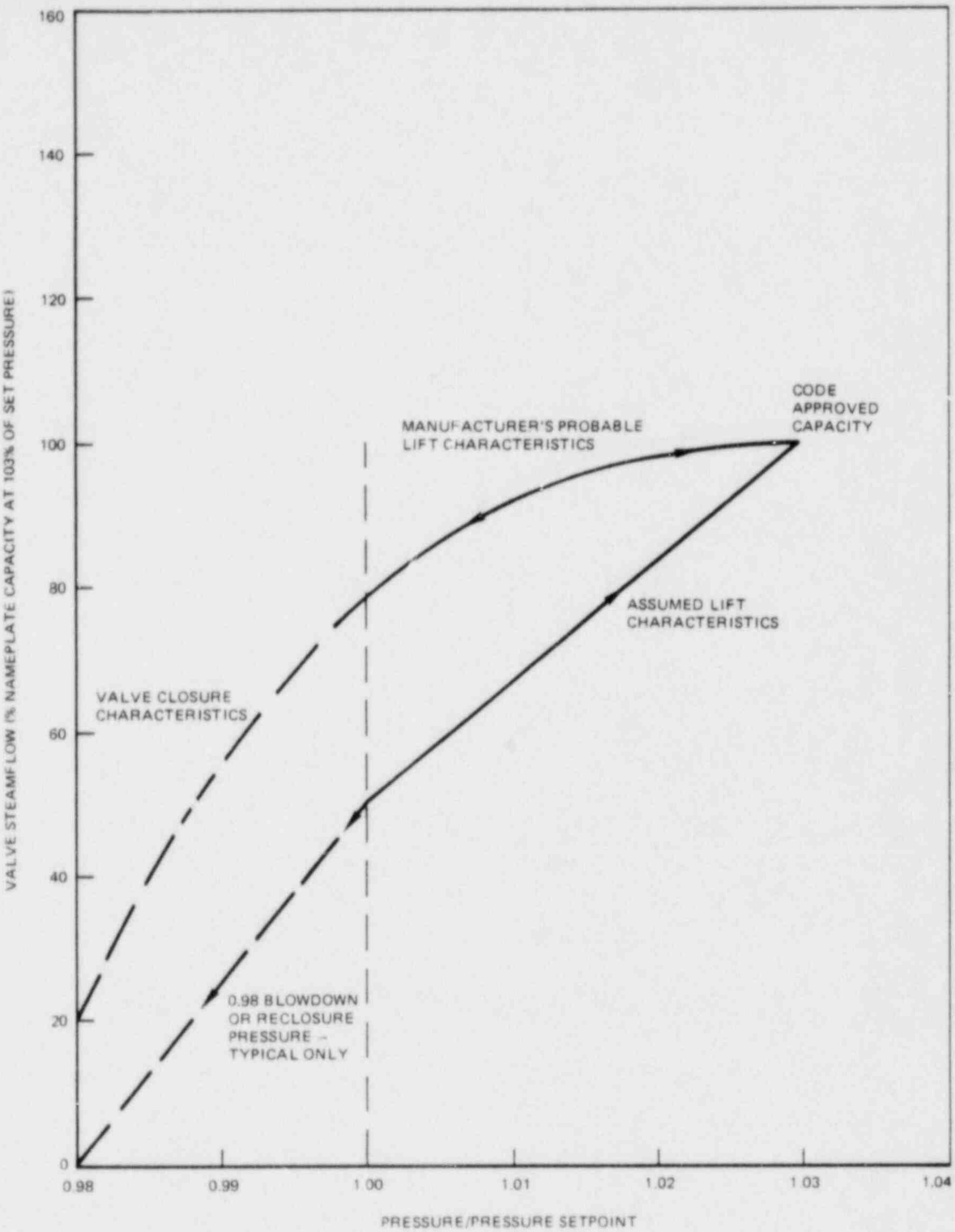
Typical Dual Safety/Relief Valve
Capacity Characteristics Power -
Actuated Relief Mode
Figure 5.2-1



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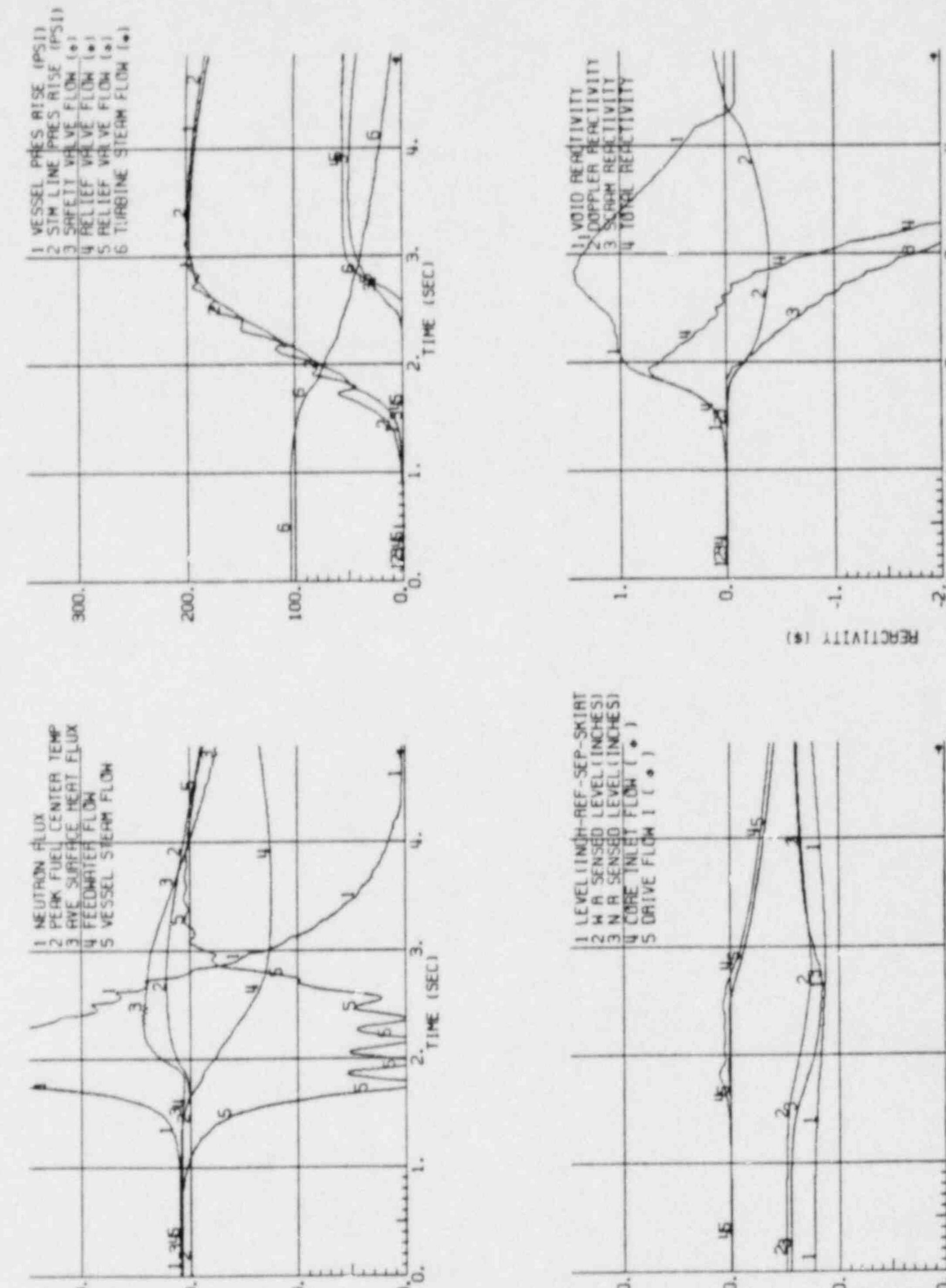
Typical Dual Safety Relief Valve Capacity Characteristics - Spring Action Safety Mode

Figure 5.2-2



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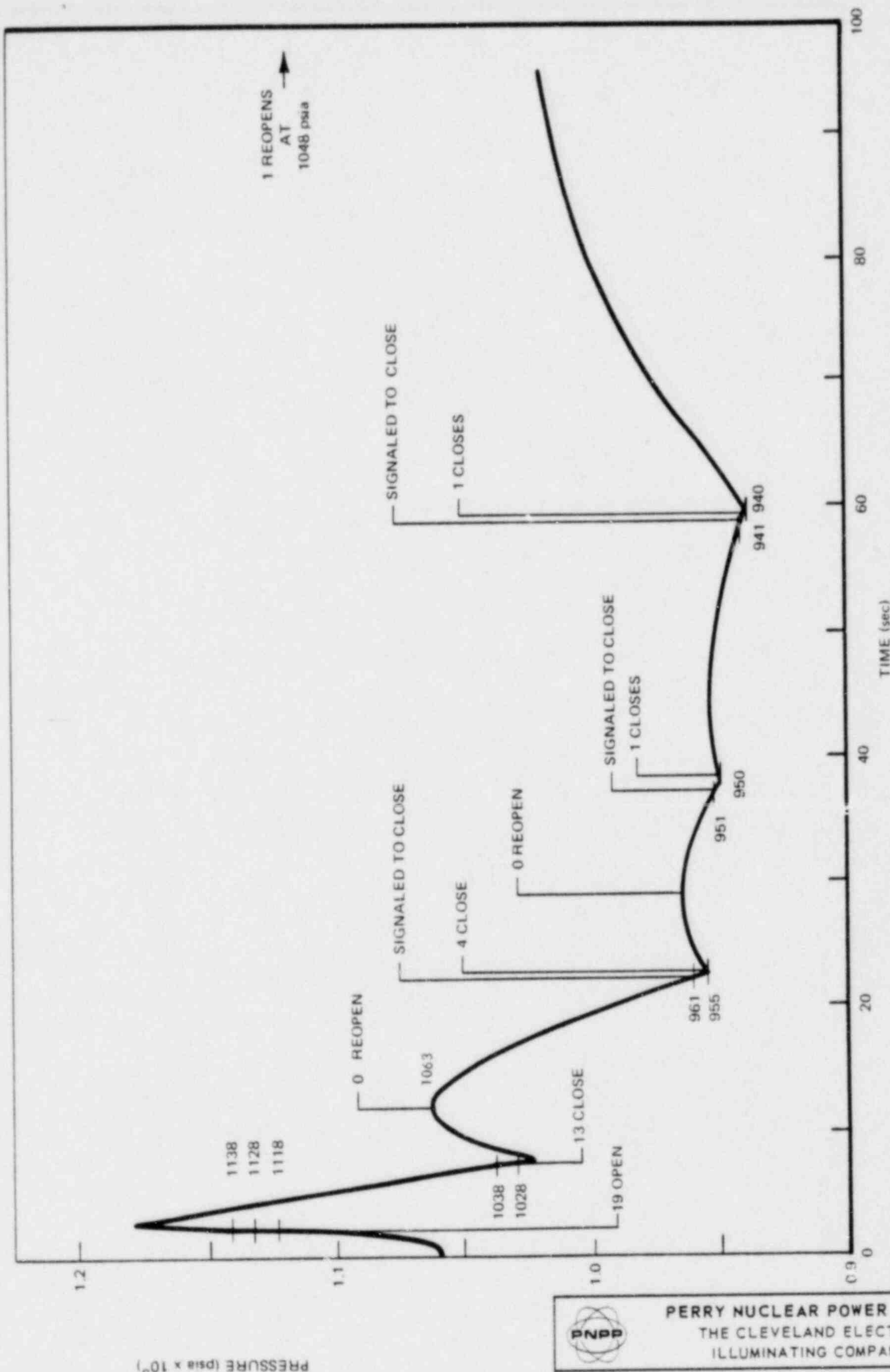
MSIV Closure with Flux Scram
and Installed Safety/Relief
Valve Capacity

Figure 5.2-3



PERRY NUCLEAR POWER PLANT
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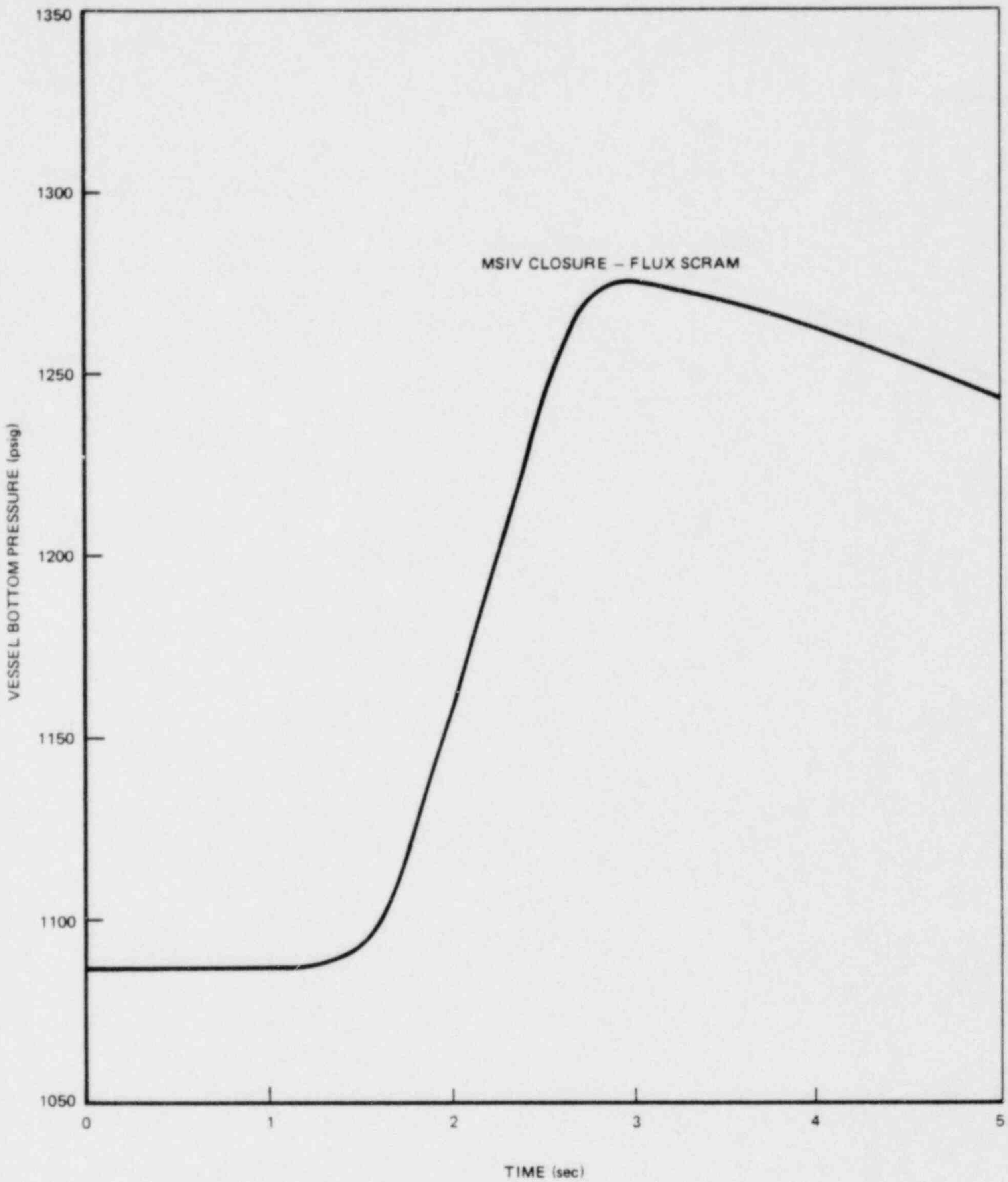
Reactor Vessel Pressure Following
Transient Isolation Event

Figure 5.2-6b



PERRY NUCLEAR POWER PLANT
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Time Response for Pressurization

Figure 5.2-8

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

Section 4.4, "Thermal and Hydraulic Design," describes the various fuel failure mechanisms. Avoidance of unacceptable results 1 and 2 (Section 4.4.1.4) for incidents of moderate frequency is verified statistically with consideration given to date, calculation, manufacturing, and operating uncertainties. An acceptable criterion was determined to be that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition[1]. This criterion is met by demonstrating that incidents of moderate frequency do not result in a minimum critical power ratio (MCPR) less than 1.06 for the initial core and 1.07 for subsequent reload cores. The reactor steady-state CPR operating limit is derived by determining the decrease in MCPR for the most limiting event. All other events result in smaller MCPR decreases and are not reviewed in depth in this chapter. The MCPR during significant abnormal events is calculated using a transient core heat transfer analysis computer program. The computer program is based on a multinode, single channel thermal-hydraulic model which requires simultaneous solution of the partial differential equations for the conservation of mass, energy, and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time.

A detailed description of the analytical model may be found in Appendix C of Reference 3. The initial condition assumed for all full power transient MCPR calculations is that the bundle is operating at or above the MCPR limit 1.18 for the initial core and 1.19 for the subsequent reload cores. Maintaining MCPR greater than 1.06 for the initial core and 1.07 for subsequent reload cores is a sufficient, but not necessary condition to assure that no fuel damage occurs. This is discussed in Section 4.4, "Thermal and Hydraulic Design."

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics.

These correlations are substantiated by fuel rod failure tests and are discussed in Section 4.4, "Thermal and Hydraulic Design," and Section 6.3, "Emergency Core Cooling System."

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed within this section have values for input parameters and initial conditions as specified in Table 15.0-1. Analyses which assume data inputs different than these values are designated accordingly in the appropriate event discussion.

15.0.3.3.3 Initial Power/Flow Operating Constraints

The analyses basis for most of the transient safety analyses is the thermal power at rated core flow (100 percent) corresponding to 105 percent Nuclear Boiler Rated steam flow. This operating point is the apex of a bounded operating power/flow map which, in response to any classified abnormal operational transients, will yield the minimum pressure and thermal margins of any operating point within the bounded map. Referring to Figure 15.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (104.2 percent rod line A-D), the lower bound is the zero power line H-J, the right bound is the rated core flow line A-H, and the left bound is the natural circulation line D-J.

The power/flow map, A-D-J-H-A, represents the acceptable operational constraints for abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the pump cavitation regions, the licensed power limit, and other restrictions based on pressure and thermal margin criteria. For instance, if the licensed power is 100 percent nuclear boiler rated (NBR), the power/flow map is truncated by the line B- C and reactor operation must be confined within the boundary B- C- D- J- L- K- B.

15.0.3.4 Barrier Performance

This section primarily evaluates the performance of the reactor coolant pressure boundary (RCPB) and the containment system during transients and accidents.

During transients that occur with no release of coolant to the containment, only RCPB performance is considered. If release to the containment occurs as in the case of limiting faults, then challenges to the containment are evaluated as well.

15.0.3.4.1 Reactor Coolant Pressure Boundary Performance

The significant areas of interest for internal pressure damage are the high pressure portions of the reactor coolant pressure boundary (the reactor vessel and the high pressure pipelines attached to the reactor vessel). The overpressure below which no damage can occur is defined as the pressure increase over design pressure allowed by the applicable ASME Boiler and Pressure Vessel Code 4 for the reactor vessel and the high pressure nuclear system piping. Because this ASME Code permits pressure transients up to 10 percent over design pressure for upset events, the design pressure portion of the reactor coolant pressure boundary meets the design requirement if peak nuclear system pressure remains below 1375 psig (110 percent x 1250 psig). Comparing the events considered in this section with those used in the mechanical design of equipment reveals that either the accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

The Low-Low Set (LLS) Relief Function, armed upon relief actuation of any S/R valve; will cause a greater magnitude blowdown, in the relief mode, for certain specified S/R valves and a subsequent cycling of a single low set valve. The effect of the LLS design on reactor coolant pressure is demonstrated, in Chapter 5, on the MSIV closure event. This is considered bounding for all other pressurization events and, therefore, is not simulated in the analysis presented in this chapter.

A sensitivity study has also been performed to support higher analytical limits for relief valve setpoints. The study shows an increase of 20 psi in the relief valve setpoint causes less than 20 psi increase in reactor peak pressures. However, these reactor peak pressures are still well below the ASME code limit (1375 psig). Also, the increase of 20 psi in the relief setpoints does not have any effect on the peak surface heat flux of ΔCPR, since all safety/relief valves open after the occurrence of MPCR during transients. Therefore, the analytical limits for relief valve setpoints in Technical Specification are 20 psi higher than those listed in Table 15.0-1.

15.0.3.5 Radiological Consequences

In this chapter, the consequences of radioactivity release during the three types of events: incidents of moderate frequency (anticipated operational transients), infrequent incidents (abnormal operational transients), and limiting faults (design basis accidents) are considered. For all events whose consequences are limiting a detailed quantitative evaluation is presented. For non-limiting events a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

For limiting faults (design basis accidents) two quantitative analyses are considered:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of worst case bounding the event and determining the adequacy of the plant design to meet 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis".

b. The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis".

Results for both are shown to be within NRC guidelines.

Doses resulting from the events in Chapter 15 are determined either manually or by computer code. Time dependent releases are evaluated with the Tact 3S computer code[2]. Instantaneous or "puff" type releases are evaluated by methods based on those presented in Regulatory Guide 1.3. Dose conversion factors and breathing rates are presented in Table 15.0-4.

15.0.4 NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA) RELATIONSHIP

Appendix 15A is a comprehensive, total plant, system-level, qualitative failure modes and effects analysis, relative to all the Chapter 15 events considered, the protective sequences utilized to accommodate the events and their effects, and the systems involved in the protective actions.

Interdependency of analysis and cross-referral of protective actions is an integral part of this chapter and the appendix.

Contained in Appendix 15A is a summary table which classifies events by frequency only (i.e., not just within a given category such as decrease in core coolant temperature).

15.0.5 REFERENCES FOR SECTION 15.0

1. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," November 1973 (NEDO-10959 and NEDE-10958).

2. U.S. Nuclear Regulatory Commission Computer Code Tact 3S, Computer Code for Calculating Radiological Consequences of Time Varying Radioactive Releases, Feb. 1975, Accident Analysis Branch, personal communication.

3. "General Electric Company Model for Loss of Coolant Analysis in Accordance with 10 CFR 50, Appendix K," December 1975 (NEDO-20566).

4. ASME Boiler and Pressure Vessel Code, Section III, Class 1, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure".

TABLE 15.0-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

1.	Thermal Power Level, MWt	
	Warranted Value	3,579
	Analysis Value	3,729
2.	Steam Flow, lbs per hr	
	Warranted Value	15.40×10^6
	Analysis Value (nominal)(1)(3)	16.71×10^6
	Analysis Value (nominal)(1)(4)	16.17×10^6
3.	Core Flow, lbs per hr	104×10^6
4.	Feedwater Flow Rate, lb per sec	
	Warranted Value (NBR)	4,269
	Analysis Value (nominal)(1)	4,483
5.	Feedwater Temperature, °F	425
6.	Vessel Dome Pressure, psig	1,045
7.	Vessel Core Pressure, psig	1,056
8.	Turbine Bypass Capacity, % NBR	35
9.	Core Coolant Inlet Enthalpy,	
	Btu per lb(4)	529.9
	Btu per lb(3)	528.9
10.	Turbine Inlet Pressure, psig	960
11.	Fuel Lattice	P8 x 8R
12.	Core Average Gap Conductance,	
	$Btu/sec\text{-}ft^2\text{-}°F$(4)	0.1546
	$Btu/sec\text{-}ft^2\text{-}°F$(3)	0.1892
13.	Core Leakage Flow, %(4)	12.9
	%(3)	11.0
14.	Required MCPR Operating Limit	
	First Core	1.18
	Reload Cores (Estimated)	1.19
15.	MCPR Safety Limit for Incidents of Moderate Frequency	
	First Core	1.06
	Reload Core	1.07
16.	Doppler Coefficient (-)¢/°F	
	Analysis Data(4)(5)	0.132

TABLE 15.0-1 (Continued)

17. Void Coefficient (-)¢/% Rated Voids	
Analysis Data for Power Increase Events[4][5]	14.0
Analysis Data for Power Decrease Events[4][5]	4.0
18. Core Average Rated Void Fraction, %[4][5]	42.54
19. Scram Reactivity, $ΔK Analysis Data[4][5]	Figure 15.0-2
20. Control Rod Drive Speed, Position versus time	Figure 15.0-3
21. Nuclear Characteristics used in ODYN Simulations	EOEC[6]
22. Jet Pump Ratio, M	2.257
23. Safety/Relief Valve Capacity, % NBR	
@ 1,210 psig[4]	111.4
@ 1,210 psig[3]	110.8
Manufacturer	Dikker
Quantity Installed	19
24. Relief Function Delay, seconds	0.4
25. Relief Function Response Time Constant, seconds	0.1
26. Safety Function Delay, seconds	0.0
27. Safety Function Response Time Constant, seconds	0.2
28. Set Points for Safety/Relief Valves	
Safety Function, psig	1,175, 1,185, 1,195, 1,205, 1,215
Relief Function, psig	1,125, 1,135, 1,145, 1,155
29. Number of Valve Groupings Simulated	
Safety Function, No.	5
Relief Function, No.	4
30. S/R Valve Reclosure Set point both Modes (% of setpoint)	
Maximum Safety Limit (used in analysis)	98
Minimum Operational Limit	89

TABLE 15.0-1 (Continued)

31.	High Flux Trip, % NBR	
	Analysis set point (122 x 1.042), % NBR	127.2
32.	High Pressure Scram Set Point, psig	1,095
33.	Vessel Level Trips, Feet Above Separator Skirt Bottom	
	Level 8 - (L8), feet	5.89
	Level 4 - (L4), feet	4.04
	Level 3 - (L3), feet	2.165
	Level 2 - (L2), feet	(-) 1.739

34. APRM Simulated Thermal Power Trip Scram, % NBR	
Set Point % NBR	118.8
Time Constant, sec.	7
35. RPT Delay, seconds	0.14
36. RPT Inertia Time Constant for Analysis, (2)	
Maximum - sec	5
Minimum - sec	3
37. Total Steamline Volume, ft^3	3850
38. Set Pressure of Anticipated Transient Pump Trip - psig (nominal)	1135

NOTES:

1. Actual analysis value is within ±0.2%.
2. The inertia time constant is defined by the expression:

$$t = \frac{2 \pi J_o n}{g T_o}$$

where t = inertia time constant (sec).

J_o = pump motor inertial ($lb\text{-}ft^2$).

n = rated pump speed (rps).

g = gravitational constant (ft/sec^2).

T_o = pump shaft torque (lb-ft).

3. Used only for ODYN.
4. Used only for REDY.
5. For transients simulated on the ODYN computer model, this input is calculated by ODYN and shown in the plot for each simulated transient.
6. EOEC - End of equilibrium cycle.

TABLE 15.0-2

SUMMARY OF EVENT RESULTS

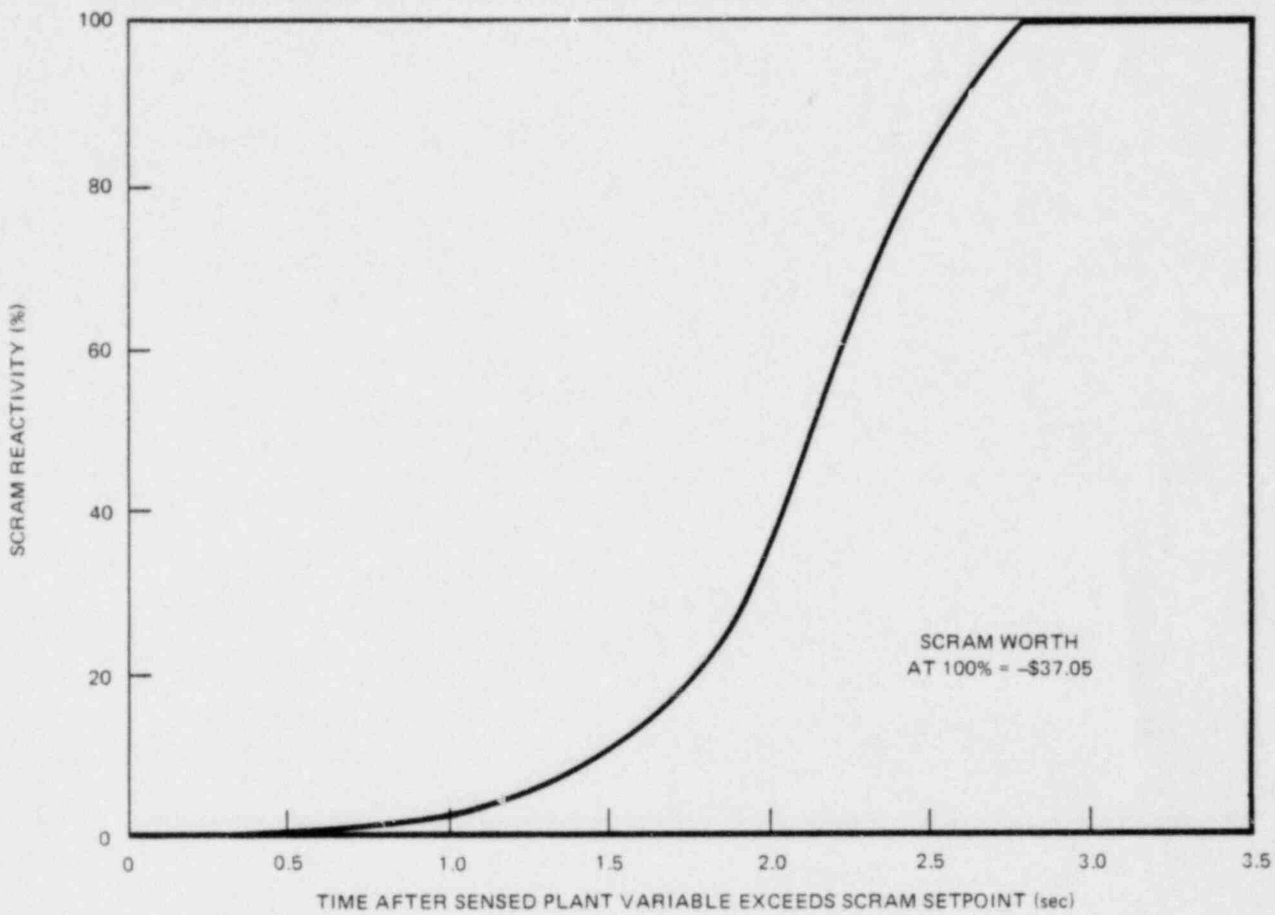
Section No.	Figure No.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR[4] -	Frequency Category[1]	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.1		DECREASE IN REACTOR COOLANT TEMPERATURE									
15.1.1	15.1-1	Loss of Feedwater Heater, AFC	111.5	1,045	1,087	1,034	105.8	(2)	a	0	0
15.1.1	15.1-2	Loss of Feedwater Heater, MFC	124.2	1,060	1,102	1,047	113.7	0.12	a	0	0
15.1.2	15.1-3	Feedwater Control Failure, Max Demand	124.3	1,163	1,193	1,159	105	0.10	a	19	5
15.1.3	15.1-4	Pressure Regulator Failure - Open, 130% Flow	104.2	1,138	1,161	1,136	100	(2)	a	10	5
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1	15.2-1	Pressure Regulation Downscale Failure	156.8	1,187	1,221	1,181	102.6	0.09	a	19	7
15.2.2	15.2-2	Generator Load Rejection, Bypass-On	128.2	1,160	1,189	1,157	100	<0.05	a	19	5
15.2.2	15.2-3	Generator Load Rejection, Bypass-Off	198.7	1,203	1,233	1,202	102.7	0.08	b	19	7
15.2.3	15.2-4	Turbine Trip, Bypass-On	114.5	1,158	1,188	1,155	100	<0.05	a	19	5
15.2.3	15.2-5	Turbine Trip, Bypass-Off	179.4	1,202	1,231	1,201	101.3	0.05	b	19	7
15.2.4	15.2-6	Main Steam Line Isolation, Position Scram	105.3	1,177	1,207	1,174	100	(2)	a	19	5
15.2.5	15.2-7	Loss of Condenser Vacuum at 2 inches per sec	113.7	1,157	1,186	1,153	100	(2)	a	19	5
15.2.6	15.2-8	Loss of Auxiliary Power Transformer	104.2	1,100	1,112	1,098	100	(2)	a	1	5
15.2.6	15.2-9	Loss of All Grid Connections	105.3	1,159	1,184	1,156	100	(2)	a	19	7
15.2.7	15.2-10	Loss of All Feedwater Flow	104.2	1,045	1,086	1,034	100	(2)	a	0	0

TABLE 15.0-2 (Continued)

Section No.	Figure No.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR(4) -	Frequency Category(1)	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	104.3	1,046	1,087	1,035	100	(2)	a	0	0
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	104.2	1,141	1,155	1,139	100	(2)	a	10	5
15.3.2	15.3-3	Fast Closure of One Recirc. Valve - 60%/sec	104.2	1,135	1,149	1,133	100	(2)	a	10	5
15.3.2	15.3-4	Fast Closure of Two Main Recirc. Valves 11%/sec	104.2	1,142	1,153	1,139	100	(2)	a	10	5
15.3.3	15.3-5	Seizure of One Recirculation Pump	104.2	1,139	1,153	1,137	100	(2)	c	10	5
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.4	15.4-1	Startup of Idle Recirculation Loop	109.3	988	1,002	983	148.7	(3)	a	0	0
15.4.5	15.4-2	Fast Opening of One Recirculation Valve	235.3	978	974[7]	974	135	(3)	a	0	0
15.4.5	15.4-3	Fast Opening of Two Recirc. Valves - 11%/sec	162.2	974	990	971	123.4	(3)	a	0	0
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS Pump Start	104.2	1,045	1,087	1,034	100	(2)	a	0	0

NOTES:

1. a = incidents of moderate freq; b = infrequent incidents; c = limiting faults
2. no significant change in CPR
3. Not start from full power
4. Option A ΔCPR adjustment factor is included as specified in the NRC staff safety evaluation for the General Electric Topical Report - Qualification of the One-Dimensional Core Transient Model for BWR, NEDO-24154 and NEDE-24154-P is applicable.



PERRY NUCLEAR POWER PLANT
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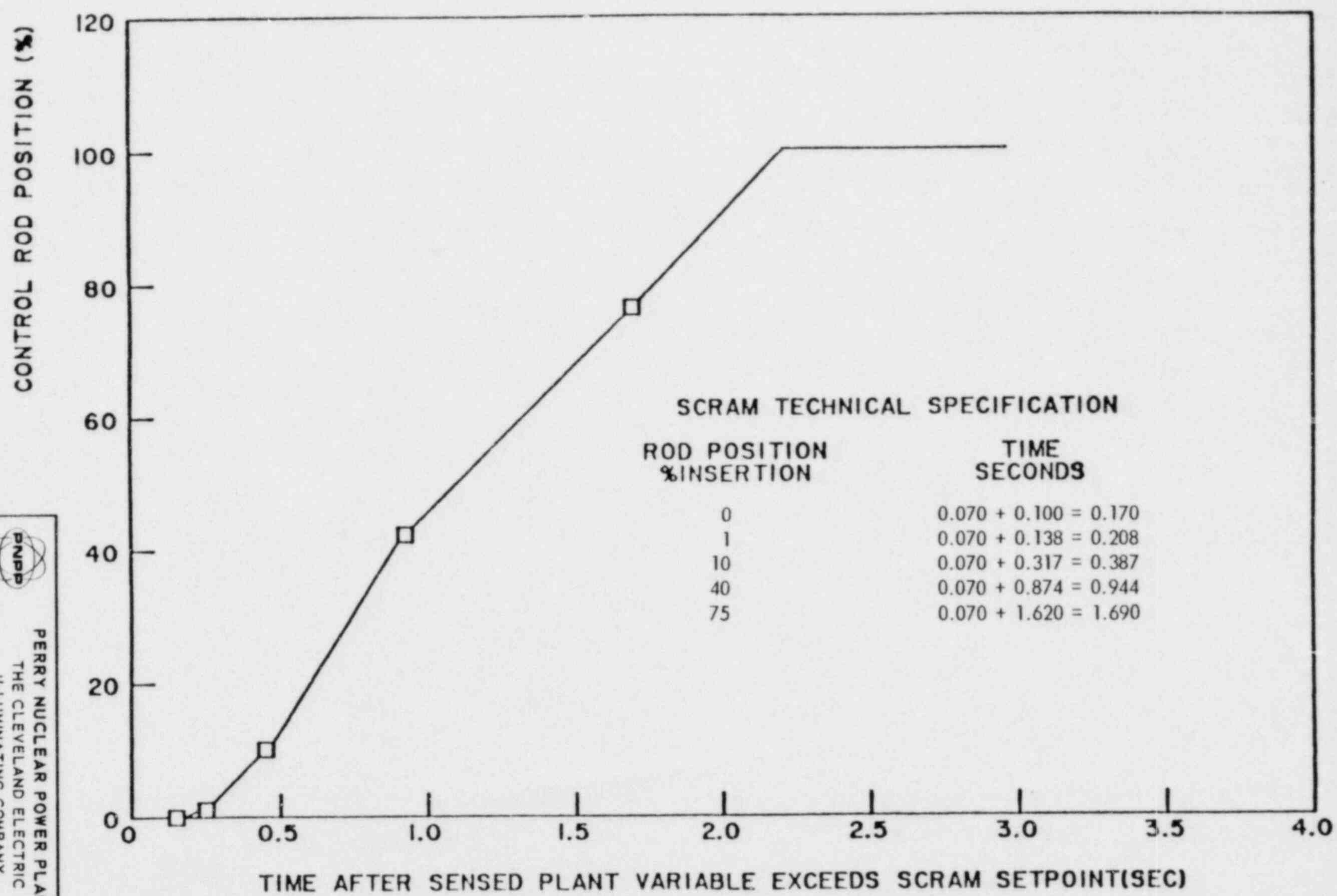
Scram Reactivity Characteristics
(REDY)

Figure 15.0-2



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Scram Time Characteristics

Figure 15.0-3

15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

15.1.1 LOSS OF FEEDWATER HEATING

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

A feedwater heater can be lost in at least two ways:

a. Steam extraction line to heater is closed,

b. Steam is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the steam bypasses the heater and no heating of that feedwater occurs. In either case the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. This event has been conservatively estimated to incur a loss of up to 100°F of the feedwater heating capability of the plant and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. The event can occur with the reactor in either the automatic or manual control mode. In automatic control, some compensation of core power is realized by modulation of core flow, so the event is less severe than in manual control.

15.1.1.1.2 Frequency Classification

The probability of this event is considered low enough to warrant it being categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this transient disturbance is analyzed as an incident of moderate frequency.

c. Four primary system pressure nodes are simulated. The nodes represent the core exit pressure, vessel dome pressure, steam line pressure (at a point representative of the safety/relief valve location) and turbine inlet pressure.

d. The active core void fraction is calculated from a relationship between core exit quality, inlet subcooling, and pressure. This relationship is generated from multinode core steadystate calculations. A second-order void dynamic model with the void boiling sweep time calculated as a function of core flow and void conditions is also utilized.

e. Principal controller functions such as feedwater flow, recirculation flow, reactor water level, pressure and load demand are represented together with their dominant nonlinear characteristics.

f. The ability to simulate necessary reactor protection system functions is provided.

15.1.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-1.

The plant is assumed to be operating at 105 percent of NB rated power and at thermally limited conditions. Both automatic and manual modes of flow control are considered.

The same void reactivity coefficient conservatism used for pressurization transients is applied since a more negative value conservatively increases the severity of the power increase. The values for both the feedwater heater time constant and the feedwater time volume between the heaters and the spargers are adjusted to reduce the time delays since they are not critical to the calculation of this transient. The transient is simulated by programming a change in feedwater enthalpy corresponding to a 100°F loss in feedwater heating.

15.1.1.3.3 Results

In the automatic flux/flow control mode, the recirculation flow control system responds to the power increase by reducing core flow so that steam flow from the reactor vessel to the turbine remains essentially constant. In order to maintain the initial steam flow with the reduced inlet temperature, reactor thermal power increases above the initial value and settles at about 110 percent NBR (106 percent of initial power), below the flow-referenced APRM thermal power scram setting and core flow is reduced to approximately 80 percent of rated flow. The MCPR reached in the automatic control mode is greater than for the more limiting manual flow control mode.

The increased core inlet subcooling aids thermal margins, and smaller power increase makes this event less severe than the manual flow control case given below. Nuclear system pressure does not change and consequently the reactor coolant pressure boundary is not threatened. If scram occurs, the results become very similar to the manual flow control case. This transient is illustrated in Figure 15.1-1.

In manual mode, no compensation is provided by core flow and thus the power increase is greater than in the automatic mode. A scram on high APRM thermal power occurs. Vessel steam flow increases and the initial system pressure increase is slightly larger. Peak heat flux is 114 percent of its initial value and the average fuel temperature increases 120°F. The increased core inlet subcooling aids core thermal margins and minimum CPR is 1.08. Therefore, the design basis is satisfied. The transient responses of the key plant variables for this mode of operation are shown in Figure 15.1-2.

If the reactor scrams, water level drops to the low level trip point (L2). This initiates RPT as shown in Table 15.1-2.

This transient is less severe from lower initial power levels for two main reasons: lower initial power levels will have initial values greater than the limiting initial value assumed, and the magnitude of the power rise

initiation of the reactor core isolation cooling system and the high pressure core spray system to maintain long term water level control following tripping of feedwater pumps.

15.1.2.2.3 The Effect of Single Failures and Operator Errors

In Table 15.1-3 the first sensed event to initiate corrective action to the transient is the vessel high water level (L8) scram. Scram trip signals from Level 8 are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation. Therefore, single failures are not expected to result in a more severe event than analyzed. See Appendix 15A for a detailed discussion of this subject.

15.1.2.3 <u>Core and System Performance</u>

15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in Reference 2. This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

The nonlinear computer simulated analytical model is designed to predict associated transient behavior of this reactor. Some of the significant features of the model are:

a. An integrated one-dimensional core model is assumed which includes a detailed description of hydraulic feedback effects, axial power shape changes, and reactivity feedbacks.

b. The fuel is represented by an average cylindrical fuel and cladding model for each axial location in the core.

c. The steam lines are modeled by eight pressure nodes incorporating mass and momentum balances which will predict any wave phenomena present in the steam line during pressurization transient.

d. The core average axial water density and pressure distribution is calculated using a single channel to represent the heat active flow and a single channel to represent the bypass flow. A model, representing liquid and vapor mass and energy conservation and mixture momentum conservation, is issued to describe the thermal-hydraulic behavior. Changes in the flow split between the bypass and active channel flow are accounted for during transient events.

e. Principal controller functions such as feedwater flow, recirculation flow, reactor water level, pressure and load demand, are represented together with their dominant nonlinear characteristics.

f. The ability to simulate necessary reactor protection system functions is provided.

g. The control systems and reactor protection system models are, for the most part, identical to those employed in the point reactor model, which is described in detail in Reference 1 and used in analysis for other transients.

15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-1.

End of equilibrium cycle (all rods out) scram characteristics are assumed. The safety-relief valve action is conservatively assumed to occur with higher than nominal set points. The transient is simulated by programming an upper limit failure in the feedwater system such that 130 percent NBR feedwater flow occurs at a system design pressure of 1,065 psig.

15.1.2.3.3 Results

The simulated feedwater controller transient is shown in Figure 15.1-3. The high water level turbine trip and feedwater pump trip are initiated at approximately 12 seconds. Scram occurs almost simultaneously, and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. MCPR remains above safety limit. The turbine bypass system opens to limit peak pressure in the steam line near the safety/relief valves to 1,159 psig and the pressure at the bottom of the vessel to about 1,193 psig.

The level will gradually drop to the low level reference point (Level 2), activating the RCIC/HPCS systems for long term level control.

It is true that a drop in the feedwater temperature with an increase in feedwater flow will occur. However, the feedwater heater usually has a large time constant (minutes, not seconds) so the feedwater temperature change is very slow. In addition, there is a long transport delay time before the lower temperature feedwater will reach the vessel. Thus, it is expected that this feedwater temperature change during the first part of the feedwater controller failure (maximum demand) transient is insignificant, and its effect on transient severity is minimal.

15.1.2.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the most conservative allowable response (e.g., relief set points, scram stroke time and reactivity characteristics). Expected plant behavior is, therefore, expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

As noted above the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

A 5-second isolation valve closure instead of a 3-second closure is assumed when the turbine pressure decreases below the turbine inlet low pressure set point for main steam line isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

Reactor scram is initiated when the isolation valves reach the 10 percent closed position. This is the maximum travel from the full open position allowed by specification.

This analysis has been performed, unless otherwise noted, with the plant conditions listed in Table 15.0-1.

15.1.3.3.3 Results

Figure 15.1-4 shows graphically how the high water level trip and isolation valve closure stops vessel depressurization and produces a normal shutdown of the isolated reactor.

The main steam line isolation valves automatically close at approximately 28 seconds when pressure at the turbine decreases below 825 psig. Depressurization results in formation of voids in the reactor coolant and causes a rapid decrease in reactor power almost immediately. Reactor vessel isolation limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary. After the rapid portion of the transient is complete and the isolation effective, the nuclear system safety/relief valves operate intermittently to relieve the pressure rise that results from decay heat generation. No significant reductions in fuel thermal margins occur. Because the rapid portion of the transient results in only momentary depressurization of the nuclear system and because the safety/relief valves need operate only to relieve the pressure increase caused by decay heat, the reactor coolant pressure boundary is not threatened by high internal pressure.

15.1.6.5 Radiological Consequences

Since this event does not result in any fuel failures, no analysis of radiological consequences is required for this event.

15.1.7 REFERENCES FOR SECTION 15.1

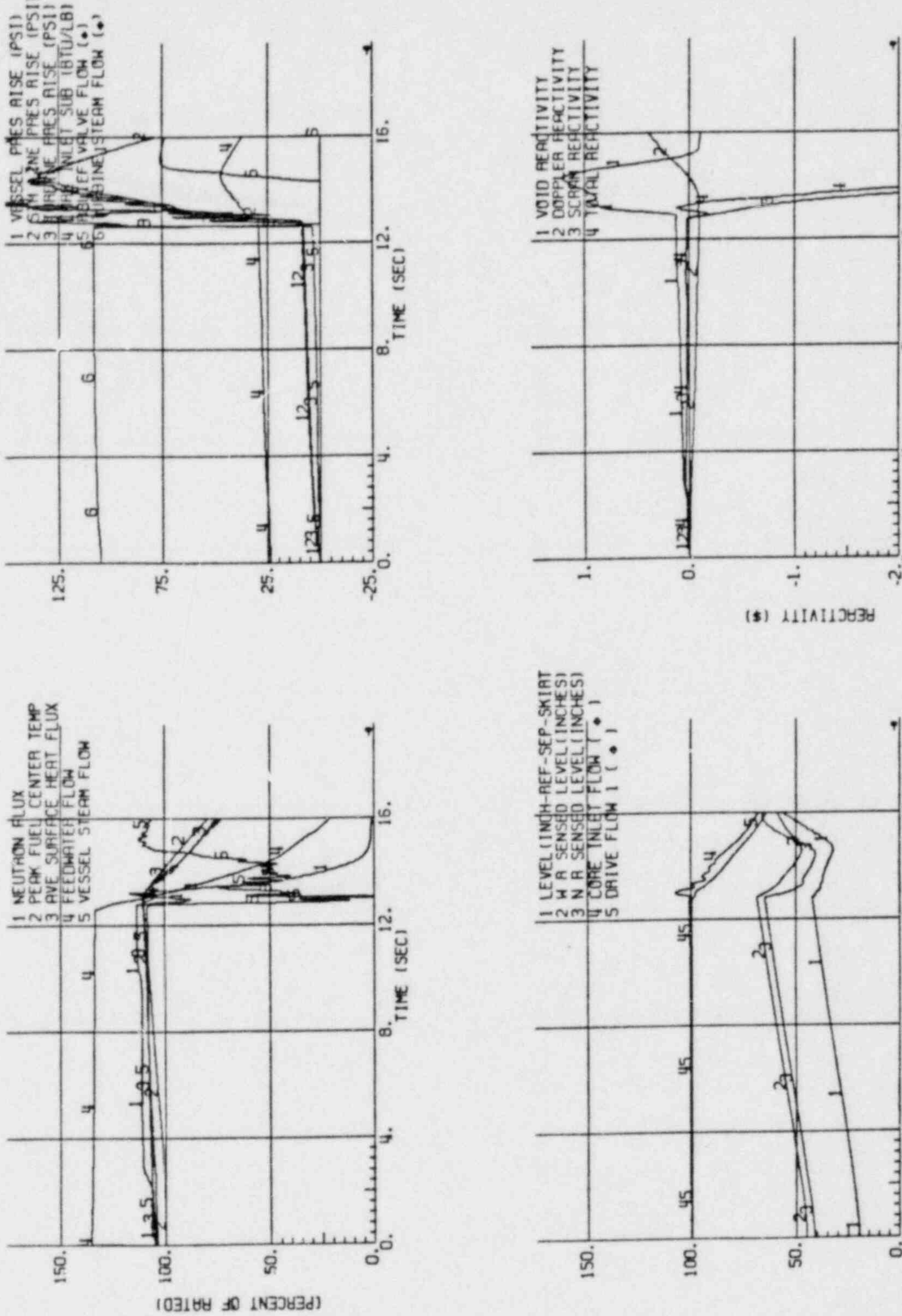
1. R. B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," April 1973 (NEDO-10802).

2. "Qualification of the One-Dimensional Core Transient Model for BWR," October, 1978, NEDO-24154.

TABLE 15.1-3

SEQUENCE OF EVENTS FOR FIGURE 15.1-3

Time-sec	Event
0	Initiate simulated failure of 130% upper limit at system design pressure of 1,065 psig on feedwater flow.
11.8	L8 vessel level set point initiates reactor scram and trips main turbine and feedwater pumps.
11.9	Recirculation pump trip (RPT) actuated by stop valve position switches.
11.9	Main turbine bypass valves opened due to turbine trip.
13.2	Safety/relief valves open due to high pressure.
18.2	Safety/relief valves close.
>20 (est.)	Water level dropped to low water level setpoint (L2).
>50 (est.)	RCIC and HPCS flow into vessel (not simulated).



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Feedwater Controller Failure,
Maximum Demand, With
HWL Trips

Figure 15.1-3

15.2.1.2.3 The Effect of Single Failures and Operator Errors

15.2.1.2.3.1 One Pressure Regulation Failure - Closed

The nature of the first assumed failure produces a slight pressure increase in the reactor until the backup regulator gains control, since no other action is significant in restoring normal operation. If we fail the backup regulator at this time, the control valves will start to close causing reactor pressure to increase, a flux scram trip would be initiated to shut down the reactor. This event is similar to that described in Section 15.2.1.2.1.1. Detailed discussions on this subject can be found in Appendix 15A.

15.2.1.2.3.2 Pressure Regulation Downscale Failure

This transient leads to a loss of pressure control such that the zero steam flow demand causes a pressurization. The high neutron flux scram is the mitigating system and is designed to be single failure proof. Therefore, single failures are not expected to result in a more severe event than analyzed. Detailed discussions on this subject can be found in Appendix 15A.

15.2.1.3 <u>Core and System Performance</u>

15.2.1.3.1 Mathematical Model

The nonlinear, dynamic model (ODYN) described briefly in Section 15.1.2.3.1 is used to simulate this event.

15.2.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.[illegible]

15.2.1.3.3 Results

15.2.1.3.3.1 One Pressure Regulator Failure - Closed

Qualitative evaluation provided only.

Response of the reactor during this regulator failure is such that pressure at the turbine inlet increases quickly, less than 2 seconds or so, due to the sharp closing action of the turbine control valves which reopen when the backup regulator gains control. This pressure disturbance in the vessel is not expected to exceed flux or pressure scram trip set points.

15.2.1.3.3.2 Pressure Regulation Downscale Failure

A pressure regulation downscale failure is simulated at 105 percent NB rated steam flow condition in Figure 15.2-1.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. When the sensed neutron flux reaches the high neutron flux scram set point, a reactor scram is initiated. The neutron flux increase is limited to 157 percent NB rated by the reactor scram. Peak fuel surface heat flux does not exceed 102.6 percent of its initial value. MCPR for this transient is still above the safety MCPR limit. Therefore, the design basis is satisfied.

15.2.1.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the most conservative allowable response (e.g., relief set points, scram stroke time, and worth characteristics). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.1.4 Barrier Performance

15.2.1.4.1 One Pressure Regulator Failure - Closed

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.1.4.2 Pressure Regulation Downscale Failure

Peak pressure at the safety/relief valves reaches 1,181 psig. The peak nuclear system pressure reaches 1,221 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1,375 psig.

15.2.1.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Section 15.2.4.5 (for a Type 2 event). Therefore, the radiological exposures noted in Section 15.2.4.5 cover the consequences of this event.

15.2.2 GENERATOR LOAD REJECTION

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation set points is assumed to function normally during the time period analyzed.

15.2.2.2.2.2 Generator Load Rejection with Failure of Bypass

The sequence of events for this failure is the same as in Section 15.2.2.2.2.1 except that failure of the main turbine bypass valves is assumed for the entire transient.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the reactor protection system functions. Turbine control valve trip scram and RPT are designed to satisfy the single failure criterion. An evaluation of the most limiting single failure (i.e., failure of the bypass system) was considered in this event. Details of single failure analysis can be found in Appendix 15A.

15.2.2.3 Core and System Performance

15.2.2.3.1 Mathematical Model

The computer model described in Section 15.1.2.3.1 was used to simulate this event.

15.2.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-1.

The turbine electrohydraulic control system (EHC) detects load rejection before a measurable speed change takes place.

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 seconds (the specification in Figure 10.2-2 requires the closure time to be bounded by 0.0008xP, where P is steam flow in percent NBR. Sensitivity studies have shown that the effect of valve closure time on ΔCPR to be small.)

Auxiliary power is independent of any T-G overspeed effects and is continuously supplied at rated frequency, assuming automatic fast transfer to auxiliary power supplies. However, overspeed effects on recirculation pumps are included in the analysis.

The reactor is operating in the manual flow-control mode when load rejection occurs. Results do not significantly differ if the plant had been operating in the automatic flow-control mode.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

Events caused by low water level trips, including initiation of HPCS and RCIC core cooling system functions are not included in the simulation. Should these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.2.2.3.3 Results

15.2.2.3.3.1 Generator Load Rejection with Bypass

Figure 15.2-2 shows the results of the generator trip from 105 percent NB rated power. Peak neutron flux rises 35 percent above initial conditions.

The average surface heat flux shows no increase from its initial value and MCPR does not significantly decrease below its initial value.

15.2.2.3.3.2 Generator Load Rejection with Failure of Bypass

Figure 15.2-3 shows that, for the case of bypass failure, peak neutron flux reaches about 199 percent of rated, average surface heat flux reaches 102.7 percent of its initial value. Since this event is classified as an infrequent incident, it is not limited by the GETAB criteria and the MCPR limit is permitted to fall below the safety limit for the incidents of moderate frequency. MCPR stays above 1.10 for this event.

15.2.2.3.4 Consideration of Uncertainties

The full stroke closure time of the turbine control valve of 0.15 seconds is conservative. Typically, the actual closure time is more like 0.2 seconds. Clearly the less time it takes to close, the more severe the pressurization effect.

All systems utilized for protection in this event were assumed to have the most conservative allowable response (e.g., relief set points, scram stroke time and worth characteristics). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection

Peak pressure remains within normal operating range and no threat to the barrier exists.

15.2.2.4.2 Generator Load Rejection with Failure of Bypass

Peak pressure at the safety/relief valves reaches 1,202 psig. The peak nuclear system pressure reaches 1,233 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1,375 psig.

15.2.2.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Section 15.2.4.5. Therefore, the radiological exposures noted in Section 15.2.4.5 for Type 2 exposure cover these consequences of this event.

15.2.3 TURBINE TRIP

15.2.3.1 Identification of Causes and Frequency Classification

15.2.3.1.1 Identification of Causes

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are moisture separator high level and first stage reheater drain tank high levels, high vibrations, operator lock out, loss of control fluid pressure, low condenser vacuum and reactor high water level.

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip

This transient is categorized as an incident of moderate frequency. In defining the frequency of this event, turbine trips which occur as a byproduct of other transients such as loss of condenser vacuum or reactor high level trip events are not included. However, spurious low vacuum or high level trip signals which cause an unnecessary turbine trip are included in defining the frequency. In order to get an accurate event-by-event frequency breakdown, this type of division of initiating causes is required.

15.2.3.1.2.2 Turbine Trip with Failure of the Bypass

This transient disturbance is categorized as an infrequent incident. Frequency is expected to be as follows:

Frequency: 0.0064/plant year

MTBE: 156 years

Frequency Basis: As discussed in Section 15.2.2.1.2.2, the failure rate of the bypass is 0.0048. Combining this with the turbine trip frequency of 1.33 events/plant year yields the frequency of 0.0064/plant year.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

15.2.3.2.1.1 Turbine Trip

Turbine trip at high power produces the sequence of events listed in Table 15.2-4.

15.2.3.2.1.2 Turbine Trip with Failure of the Bypass

Turbine trip at high power with bypass failure produces the sequence of events listed in Table 15.2-5.

15.2.3.2.1.3 Identification of Operator Actions

The operator should:

a. Verify auto transfer of buses supplied by generator to incoming power if automatic transfer does not occur, manual transfer must be made.

b. Monitor and maintain reactor water level at required level.

c. Check turbine for proper operation of all auxiliaries during coastdown.

d. Depending on conditions, initiate normal operating procedures for cool-down, or maintain pressure for restart purposes.

e. Put the mode switch in the startup position before the reactor pressure decays to <850 psig.

f. Secure the RCIC operation if auto initiation occurred due to low water level.

g. Not allow the reactor vessel water level to drop to a point of isolating MSIVs.

h. Monitor control rod drive positions and insert both the IRMs and SRMs.

i. Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

j. Cool down the reactor per standard procedure if a restart is not intended.

15.2.3.2.2 Systems Operation

15.2.3.2.2.1 Turbine Trip

All plant control systems maintain normal operation unless specifically designated to the contrary.

Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the reactor protection system.

Turbine stop valve closure initiates recirculation pump trip (RPT) thereby terminating the jet pump drive flow.

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation set points is assumed to function normally during the time period analyzed.

It should be noted that below 40 percent NB rated power level, a main stop valve scram trip inhibit signal derived from the first stage pressure of the turbine is activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of shutting down the reactor. All other protection system functions remain functional as before and credit is taken for those protection system trips.

15.2.3.2.2.2 Turbine Trip with Failure of the Bypass

This sequence of events is the same as in Section 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed.

It should be noted that below 40 percent NB rated power level, a main stop valve scram trip inhibit signal derived from the first stage pressure of the turbine is activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of shutting down the reactor. All other protection system functions remain functional as before and credit is taken for those protection system trips.

15.2.3.2.3 The Effect of Single Failures and Operator Errors

15.2.3.2.3.1 Turbine Trips at Power Levels Greater Than 40 Percent NBR

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the reactor protection system functions. Main stop valve closure scram trip and RPT are designed to satisfy single failure criterion.

15.2.3.2.3.2 Turbine Trips at Power Levels Less Than 40 Percent NBR

This sequence is the same as in Section 15.2.3.2.3.1 except RPT and stop valve closure scram trip is normally inoperative. Since protection is still provided by high flux, high pressure, etc., these will also continue to function and scram the reactor should a single failure occur.

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

The computer model described in Section 15.1.2.3.1 was used to simulate these events.

15.2.3.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-1.

Turbine stop valves full stroke closure time is 0.1 second. This is consistent with the design specification limit given in Figure 10.2-1, and has been upheld by all plant operating experience to date. Sensitivities of key parameters to this closure time are not great, and the potential uncertainties are conservatively bounded by the generator load rejection event analysis.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90 percent open. This stop valve scram trip signal is automatically bypassed when the reactor is below 40 percent NB rated power level.

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry which trips the recirculation pumps.

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip

A turbine trip with the bypass system operating normally is simulated at 105 percent NB rated steam flow conditions in Figure 15.2-4.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 114.5 percent of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed its initial value.

15.2.3.3.3.2 Turbine Trip with Failure of Bypass

A turbine trip with failure of the bypass system is simulated at 105 percent NB rated steam flow conditions in Figure 15.2-5.

Peak neutron flux reaches 180 percent of its rated value, and average surface heat flux reaches 101 percent of initial value. Therefore, this transient is less severe than the generator load rejection with failure of bypass transient as described in Section 15.2.2.3.3.2.

15.2.3.3.3.3 Turbine Trip with Bypass Valve Failure, Low Power

This transient is less severe than a similar one at high power. Below 40 percent of rated power, the turbine stop valve closure and turbine control valve closure scrams and the end of cycle recirculation pump trip are automatically bypassed. At these lower power levels, turbine first stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high neutron flux or high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief set points are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve set points and will be significantly below the RCPB transient limit of 1,375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR remains well above the GETAB safety limit.

15.2.3.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rods-out end-of-equilibrium cycle conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves include errors (high) for all valves.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1,188 psig, which is below the ASME code limit of 1,375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1,158 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip with Failure of the Bypass

The safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the set points of the valves. Peak nuclear system pressure reaches 1,231 psig at the vessel bottom, therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1,375 psig. Peak dome pressure does not exceed 1,202 psig.

conditions. If this occurs, it is also included in this category. During the main steam line isolation valve closure, position switches on the valves provide a reactor scram if the valves in three or more main steam lines are less than 90 percent open (except for interlocks which permit proper plant startup.). Protection system logic, however, permits the test closure of one valve without initiating scram from the position switches.

15.2.4.1.2.2 Closure of One Main Steam Line Isolation Valve

This event is categorized as an incident of moderate frequency. One MSLIV may be closed at a time for testing purposes, this is done manually. Operator error or equipment malfunction may cause a single MSLIV to be closed inadvertently. If reactor power is greater than about 80 percent when this occurs, a high flux scram may result, (if all MSLIVs close as a result of the single closure, the event is considered as a closure of all MSLIVs).

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

Table 15.2-6 lists the sequence of events for Figure 15.2-6.

15.2.4.2.1.1 Identification of Operator Actions

The following is the sequence of operator actions expected during the course of the event assuming no restart of the reactor. The operator should:

a. Observe that all rods have inserted.

b. Observe that the relief valves have opened for reactor pressure control.

c. Check that RCIC/HPCS auto starts on the impending low reactor water level condition.

15.2.4.2.2.2 Closure of One Main Steam Line Isolation Valve

A closure of a single MSLIV at any given time will not initiate a reactor scram. This is because the valve position scram trip logic is designed to accommodate single valve operation and testability during normal reactor operation at limited power levels. Credit is taken for the operation of the pressure and flux signals to initiate a reactor scram.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase is accomplished by initiation of the reactor scram via MSIV position switches and the protection system. Relief valves also operate to limit system pressure. All of these aspects are designed to single failure criterion and additional single failures would not alter the results of this analysis.

Failure of a single relief valve to open is not expected to have any significant effect. Such a failure is expected to result in less than a 5 psi increase in the maximum vessel pressure rise. The peak pressure will still remain considerably below 1,375 psig. The design basis and performance of the pressure relief system is discussed in Section 5.0.

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The computer model described in Section 15.1.2.3.1 was used to simulate these transient events.

15.2.4.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-1.

The main steam isolation valves close in 3 to 5 seconds. The worst case, the 3 second closure time, is assumed in this analysis.

Position switches on the valves initiate a reactor scram when the valves are less than 90 percent open. Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow.

Because of the loss of feedwater flow, water level within the vessel decreases sufficiently to initiate runback, trip of the recirculation pump and initiate the HPCS and RCIC systems.

15.2.4.3.3 Results

15.2.4.3.3.1 Closure of All Main Steam Line Isolation Valves

Figure 15.2-6 shows the changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at 105 percent of NB rated steam flow. Peak neutron flux and fuel surface heat flux show no increase.

Water level decreases sufficiently to cause a recirculation system trip and initiation of the HPCS and RCIC system at some time greater than 10 seconds. However, there is a delay up to 30 seconds before the water supply enters the vessel. Nevertheless, there is no change in the thermal margins.

15.2.4.3.3.2 Closure of One Main Steam Line Isolation Valve

Only one isolation valve is permitted to be closed at a time for testing purposes to prevent scram. Normal test procedure requires an initial power reduction to approximately 75 to 80 percent of design conditions in order to

avoid high flux scram, high pressure scram, or full isolation from high steam flow in the "live" lines. With a 3 second closure of one main steam isolation valve during 105 percent rated power conditions, the steam flow disturbance raises vessel pressure and reactor power enough to initiate a high neutron flux scram. This transient is considerably milder than closure of all MSIV's at full power. No quantitative analysis is furnished for this event. However, no significant change in thermal margins is experienced and no fuel damage occurs. Peak pressure remains below SRV set points.

Inadvertent closure of one or all of the isolation valves while the reactor is shut down (such as operating state C, as defined in Appendix 15A) will produce no significant transient. Closures during plant heatup (operating state D) will be less severe than the maximum power cases (maximum stored and decay heat) discussed in Section 15.2.4.3.3.1.

15.2.4.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For examples:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rod-out end-of-equilibrium cycle conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves are assumed to be 1 to 2 percent higher than the valve's nominal set point.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All Main Steam Line Isolation Valves

The nuclear system relief valves begin to open at approximately 2.7 seconds after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1,207 psig, clearly below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steam line is 1,174 psig.

15.2.4.4.2 Closure of One Main Steam Line Isolation Valve

No significant effect is imposed on the RCPB, since if closure of the valve occurs at an unacceptably high operating power level, a flux or pressure scram will result. The main turbine bypass system will continue to regulate system pressure via the other three "live" steam lines.

15.2.4.5 Radiological Consequences

15.2.4.5.1 General Observations

The radiological impact of many transients and accidents involves the consequences: a) which do not lead to fuel rod damage as a direct result of the event itself. b) Additionally, many events do not lead to the depressurization of the primary system but only the venting of sensible heat and energy via fluids at coolant loop activity through relief valves to the suppression pool. c) In the case of previously defective fuel rods, a depressurization transient will result in considerably more fission product carry-over to the suppression pool than hot-standby transients; and, d) the time duration of the transient varies from several minutes to four hours plus.

TABLE 15.2-1

SEQUENCE OF EVENTS FOR FIGURE 15.2-1
PRESSURE REGULATION DOWNSCALE FAILURE

Time-sec	Event
0	Simulate zero steam flow demand to main turbine and bypass valves.
0	Turbine control valves start to close.
1.0	Neutron flux reaches high flux scram set point and initiates a reactor scram.
2.3	Recirculation pump drive motors are tripped due to high dome pressure.
2.4	Safety/relief valves open due to high pressure.
6.1	Vessel water level (L8) trip initiates main turbine and feedwater turbine trips.
6.3	Main turbine stop valves closed.
9.3	Safety/relief valves close.
9.65	Group 1 safety/relief valves open again to relieve decay heat.
>15 (est.)	Group 1 safety/relief valves close.

TABLE 15.2-2

SEQUENCE OF EVENTS FOR FIGURE 15.2-2
GENERATOR LOAD REJECTION, TRIP SCRAM, BYPASS-OFF

Time-sec	Event
(-)0.015 (approx.)	Turbine-generator detection of loss of electrical load.
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Fast control valve closure (FCV) initiates scram trip.
0	Fast control valve closure (FCV) initiates a recirculation pump trip (RPT).
0.07	Turbine control valves closed.
0.1	Turbine bypass valves start to open.
1.5	Safety/relief valves open due to high pressure.
4.0	Vessel water level (L8) trip initiates trip of the feedwater turbines.
6.9	Safety/relief valves close.

TABLE 15.2-3

SEQUENCE OF EVENTS FOR FIGURE 15.2-3
GENERATOR LOAD REJECTION, TRIP SCRAM, BYPASS-OFF

Time-sec	Event
(-)0.015 (approx.)	Turbine-generator detection of loss of electrical load.
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0	Fast control valve closure (FCV) initiates scram trip.
0	Fast control valve closure (FCV) initiates a recirculation pump trip (RPT).
0.07	Turbine control valves closed.
1.2	Safety/relief valves open due to high pressure.
5.1	Vessel water level (L8) trip initiates trip of the feedwater turbines.
8.4	Safety/relief valves close.
9.3	Group 1 safety/relief valves open again to relieve decay heat.
>10 (est.)	Group 1 safety/relief valves close again.

TABLE 15.2-4

SEQUENCE OF EVENTS FOR FIGURE 15.2-4
TURBINE TRIP, TRIP SCRAM, BYPASS AND RPT-ON

Time-sec	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine trip initiates bypass operation.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.01	Main turbine stop valves reach 90% open position and initiate a recirculation pump trip (RPT).
0.1	Turbine stop valves close.
0.1	Turbine bypass valves start to open to regulate pressure.
1.6	Safety/relief valves open due to high pressure.
4.0	Vessel water level (L8) trip initiates trip of the feedwater turbines.
6.9	Safety/relief valves close.

TABLE 15.2-5

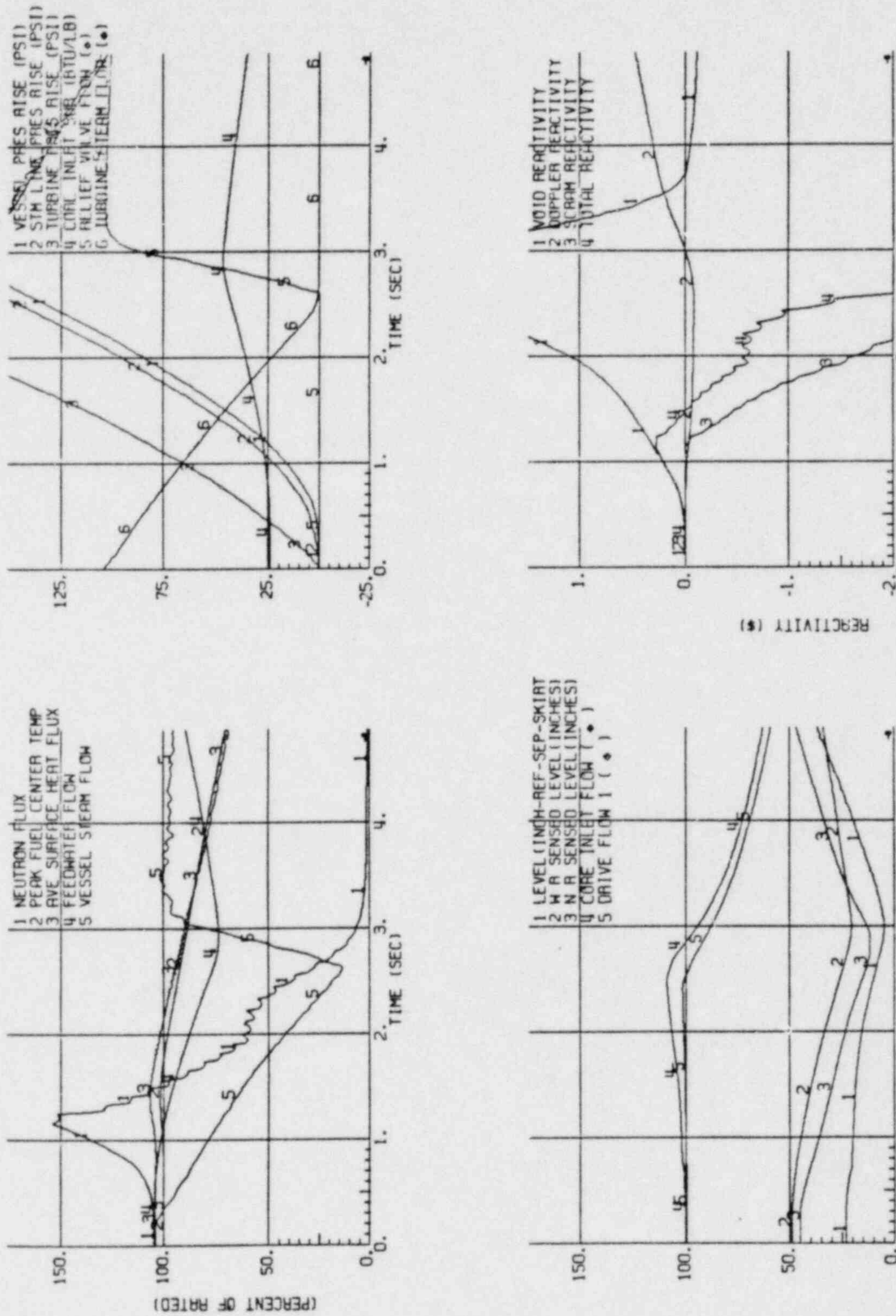
SEQUENCE OF EVENTS FOR FIGURE 15.2-5
TURBINE TRIP, TRIP SCRAM, BYPASS-OFF, RPT-ON

Time-sec	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.01	Main turbine stop valves reach 90% open position and initiate a recirculation pump (RPT) trip.
0.1	Turbine stop valves close.
1.2	Safety/relief valves open due to high pressure.
5.1	Vessel water level (L8) trip initiates trip of the feedwater turbines.
8.4	Safety/relief valves close.
9.2	Group 1 safety/relief valves open again to relieve decay heat.
>10 (est.)	Group 1 safety/relief valves close again.

TABLE 15.2-6

SEQUENCE OF EVENTS FOR FIGURE 15.2-6
THREE SECOND CLOSURE OF ALL MAIN STEAM LINE
ISOLATION VALVES WITH POSITION SWITCH SCRAM TRIP

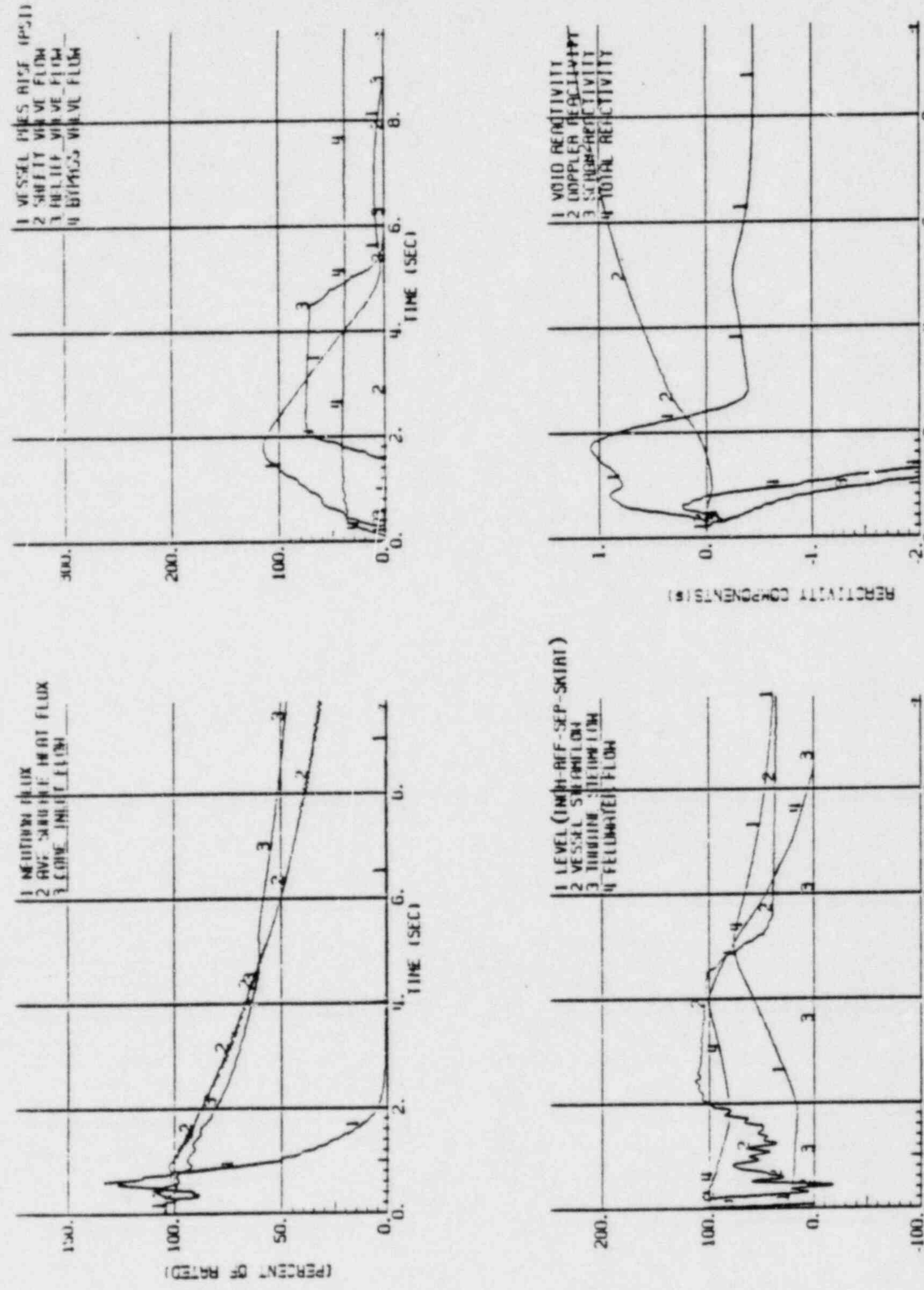
Time-sec	Event
0	Initiate closure of all main steam line isolation valves (MSIV).
0.3	MSIVs reach 90% open.
0.3	MSIV position trip scram initiated.
1.9	Recirculation pump drive motors are tripped due to low water level 3 (L3) trip.
2.7	Safety/relief valves open due to high pressure.
8.1	Safety/relief valves close.
9.1	Group 1 safety/relief valves open again to relieve decay heat.
>10 (est.)	Group 1 safety/relief valves close again.
>10 (est.)	Vessel water level reaches L2 setpoint.
>40 (est.)	HPCS and RCIC flow into vessel (not included in simulation).



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Pressure Regulator Downscale Failure

Figure 15.2-1



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PERRY NUCLEAR POWER PLANT
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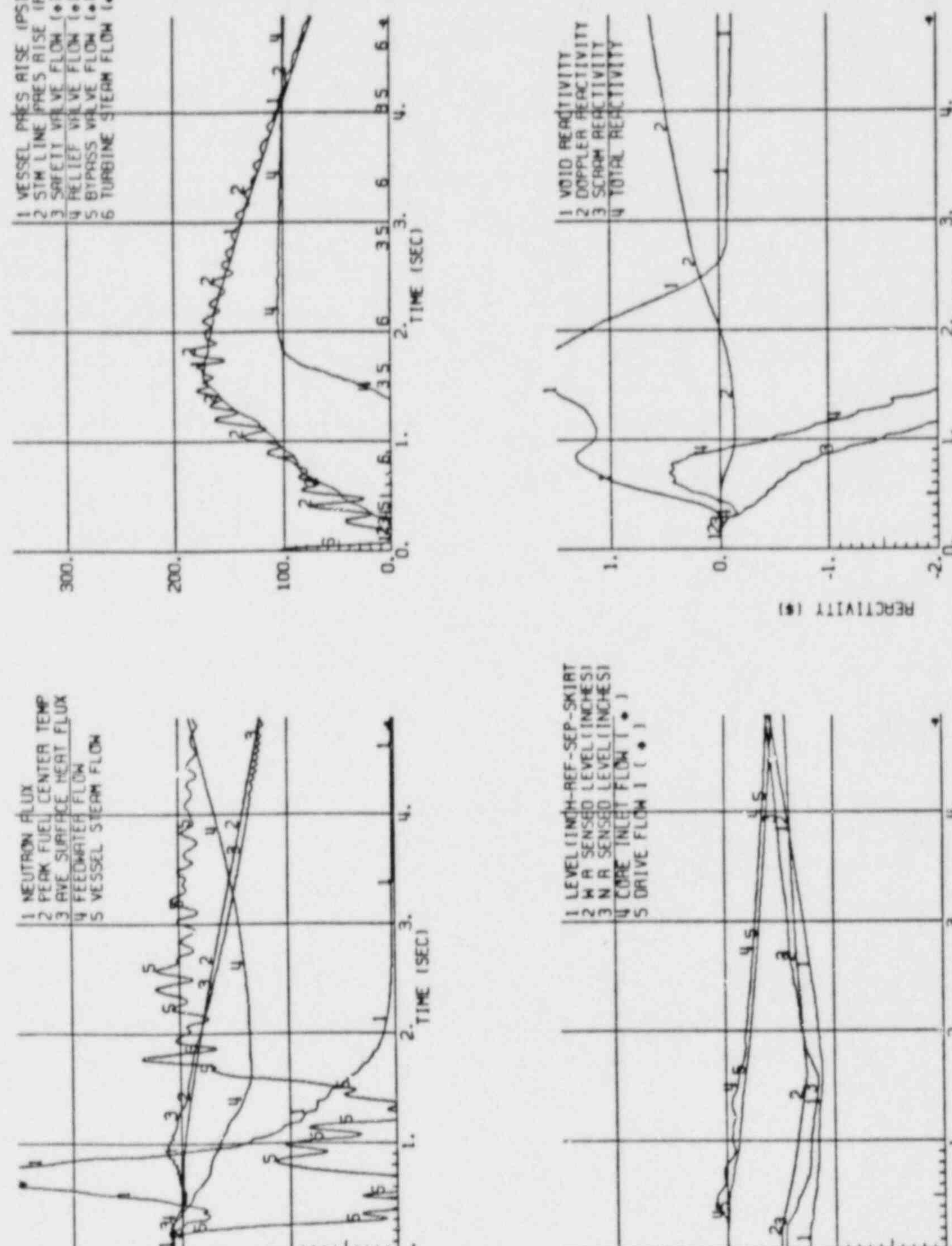
Generator Load Rejection
Trip Scram, Bypass - On

Figure 15.2-2



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PERRY NUCLEAR POWER PLANT
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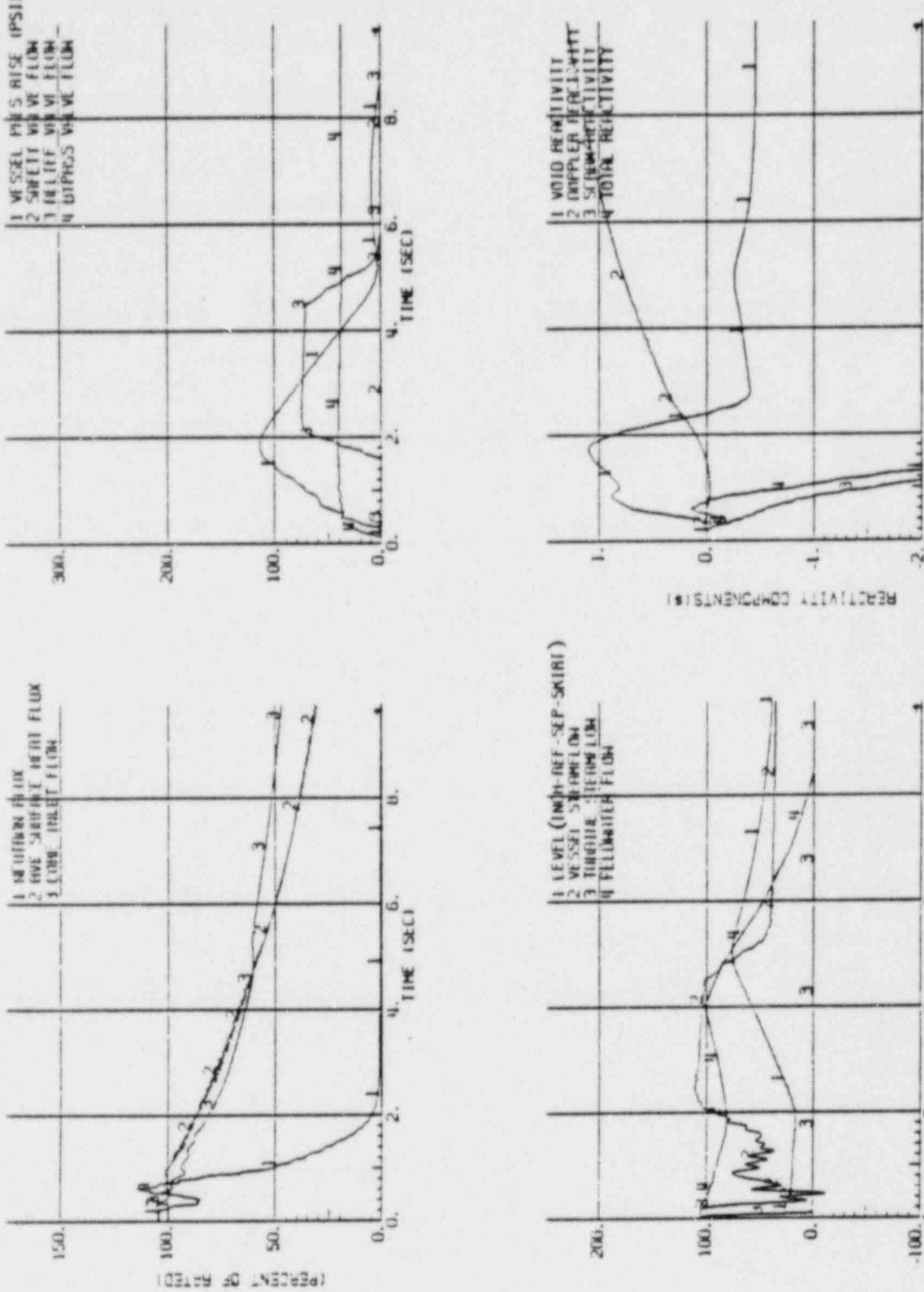
Generator Load Rejection
Trip Scram, Bypass - Off

Figure 15.2-3



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PERRY NUCLEAR POWER PLANT
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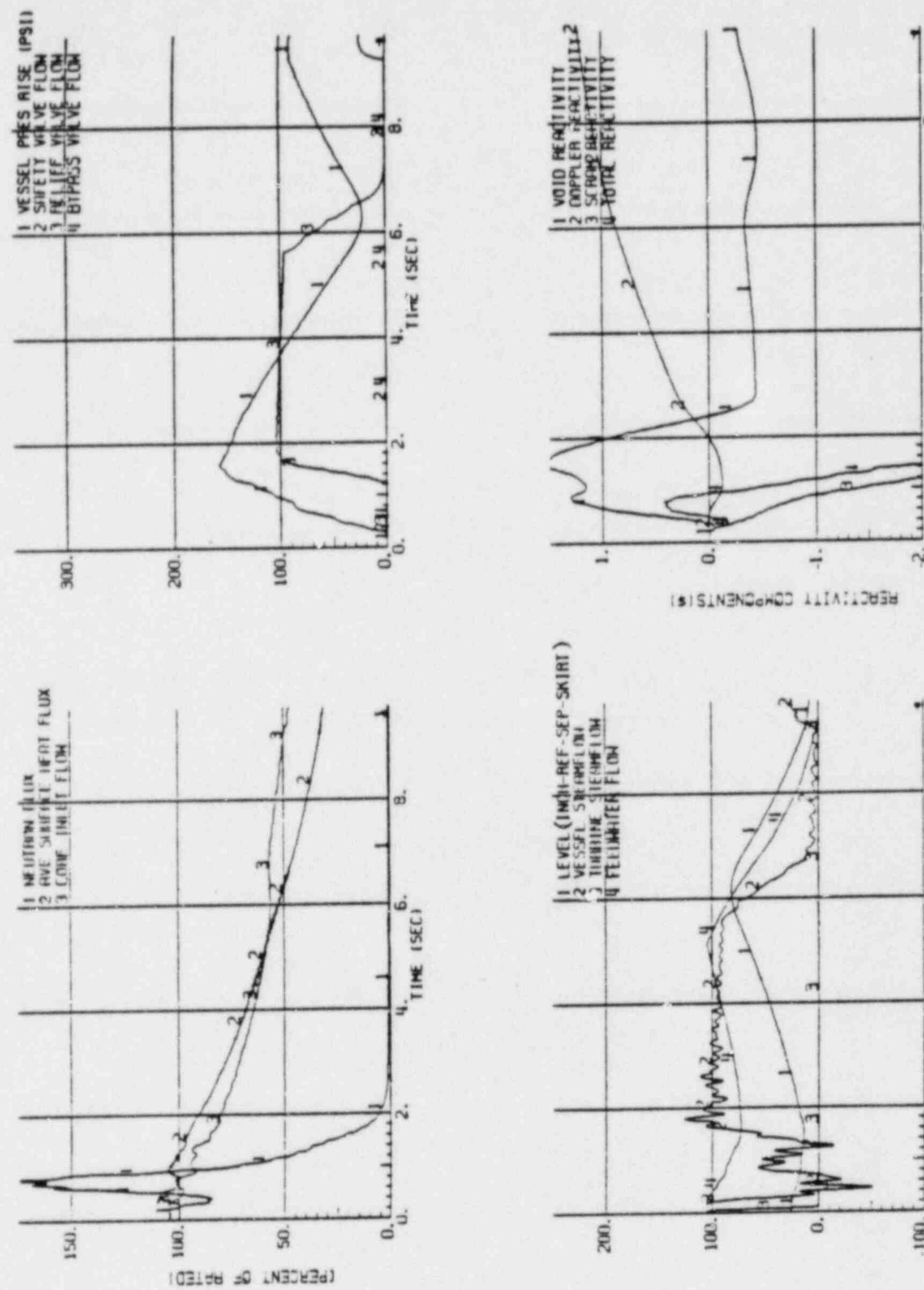
Turbine Trip, Trip Scram,
Bypass and RPT - On

Figure 15.2-4



PERRY NUCLEAR POWER PLANT
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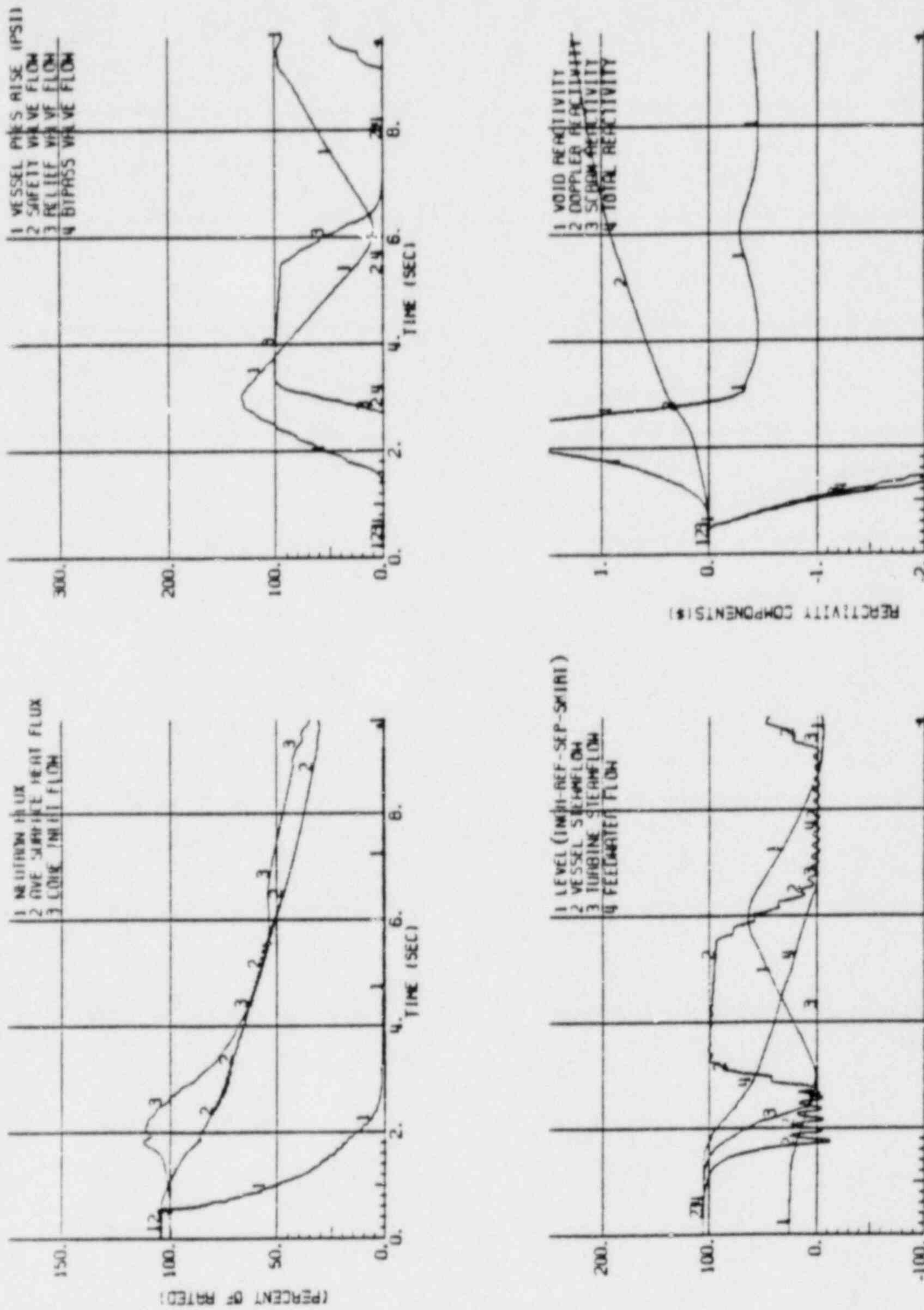
Turbine Trip, Trip Scram,
Bypass - Off, RPT - On

Figure 15.2-5



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
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Three Second Closure of all Main
Steam Line Isolation Valves
With Position Switch Scram Trip

Figure 15.2-6