

Y1003J01A56  
CLASS I  
MARCH 1983

**SUPPLEMENTAL RELOAD LICENSING  
SUBMITTAL FOR  
JAMES A. FITZPATRICK  
NUCLEAR POWER PLANT  
RELOAD 5**

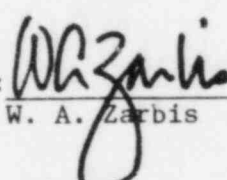
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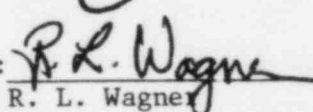
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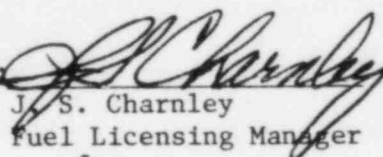
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NUCLEAR POWER SYSTEMS DIVISION • GENERAL ELECTRIC COMPANY  
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IMPORTANT NOTICE REGARDING  
CONTENTS OF THIS REPORT  
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## 1. PLANT UNIQUE ITEMS (1.0)\*

Appendix A: GETAB Analysis Initial Conditions

Appendix B: Verification of Operating Flexibility Options

## 2. RELOAD FUEL BUNDLES (1.0, 2.0, 3.3.1 and 4.0)

<u>Fuel Type</u>	<u>Cycle Loaded</u>	<u>Number</u>	<u>Number Drilled</u>
Irradiated			
8DRB283	3	12	12
P8DRB265L	4	24	24
P8DRB283	4	136	136
P8DRB284H	5	128	128
P8DRB299	5	60	60
New			
P8DRB299	6	<u>200</u>	<u>200</u>
Total		560	560

## 3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle: 17232 MWd/st

Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations: 17050 MWd/st

Assumed reload cycle core average exposure at end of cycle: 17509 MWd/st

Core loading pattern: Figure 1

\*( ) Refers to area of discussion in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-4, January 1982; a letter "S" preceding the number refers to the United States supplement.

4. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C (3.3.2.1.1 and 3.3.2.1.2)

Beginning of Cycle K-effective

Uncontrolled	1.112
Fully Controlled	0.957
Strongest Control Rod Out	0.989

R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle,  $\Delta K$

0.000

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

<u>ppm</u>	<u>Shutdown Margin (<math>\Delta k</math>) (20°C, Xenon Free)</u>
600	0.026

6. RELOAD UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 and S.2.2)  
(REDY EVENTS ONLY)

	<u>EOC</u>	<u>EOC-1000 MWd/st</u>	<u>EOC-2000 MWd/st</u>
Void Fraction (%)	41.7	41.7	41.7
Average Fuel Temperature (°F)	1271	1271	1271
Void Coefficient N/A* (¢/% Rg)	-9.19/-11.49	-9.99/-12.49	-10.37/-12.96
Doppler Coefficient N/A (¢/°F)	-0.234/-0.222	-0.229/-0.218	-0.224/-0.213
Scram Worth N/A (\$)	**	**	**

\*N = Nuclear Input Data

A = Used in Transient Analysis

\*\*Generic exposure independent values are used as given in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-4, January 1982.

## 7. RELOAD UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (S.2.2)

Fuel Design	Peaking Factors			R- Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
	Local	Radial	Axial				
Exposure: EOC							
8x8R	1.20	1.45	1.40	1.051	6.193	116.8	1.31
P8x8R	1.20	1.42	1.40	1.051	6.059	117.7	1.34
Exposure: EOC-1000 MWd/st							
8x8R	1.20	1.49	1.40	1.051	6.338	116.0	1.28
P8x8R	1.20	1.45	1.40	1.051	6.194	116.9	1.31
Exposure: EOC-2000 MWd/st							
8x8R	1.20	1.54	1.40	1.051	6.550	114.8	1.24
P8x8R	1.20	1.51	1.40	1.051	6.424	115.6	1.26

## 8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

Transient Recategorization: No

Recirculation Pump Trip : No

Rod Withdrawal Limiter : No

Thermal Power Monitor : Yes

ODYN Option B

Improved Scram Time : Yes

Exposure Dependent Limits : Yes

Exposure Points Analyzed : EOC, EOC-1000 MWd/st, EOC-2000 MWd/st

## 9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (S.2.2.1)

Transient	Exposure (MWd/st)	Flux (% NBR)	Q/A (% NBR)	$\Delta$ CPR		Figure
				8x8R	P8x8R	
Load Rejection Without Bypass	EOC	615	128	0.24	0.27	2a
	EOC-1000	550	125	0.21	0.24	2b
	EOC-2000	484	121	0.17	0.19	2c
Loss of 80°F Feedwater Heating	EOC	122	121	0.13	0.13	3
Feedwater Controller Failure	EOC	442	126	0.21	0.23	4a
	EOC-1000	309	122	0.17	0.19	4b
	EOC-2000	248	118	0.13	0.14	4c

# 10. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (S.2.2.1)

Limiting Rod Pattern: Figure 5

Includes 2.2% Power Spiking Penalty: Yes

<u>Rod Block Reading</u>	<u>Rod Position (ft withdrawn)</u>	<u><math>\Delta</math>CPR 8x8R/P8x8R</u>	<u>MLHGR (kW/ft) 8x8R/P8x8R</u>
104	3.5	0.12	14.17
105	4.0	0.14	14.72
106	4.5	0.15	14.84
107	5.0	0.17	14.84
108	5.5	0.18	14.84
109	6.0	0.20	14.84
110	9.0	0.26	16.23

Set Point Selected: 108

## 11. CYCLE MCPR VALUES (S.2.2)

### Non-Pressurization Events

<u>Exposure Range: BOC to EOC</u>	<u>P8x8R</u>	<u>8x8R</u>
Loss of 80°F Feedwater Heating	1.20	1.20
Fuel Loading Error	1.20	--
Rod Withdrawal Error	1.25	1.25

## Pressurization Events

	Option A		Option B	
	<u>P8x8R</u>	<u>8x8R</u>	<u>P8x8R</u>	<u>8x8R</u>
Exposure Range:				
BOC to EOC-2000 MWd/st				
Load Rejection Without Bypass	1.32	1.29	1.11	1.10
Feedwater Controller Failure	1.26	1.25	1.20	1.19
Exposure Range:				
EOC-2000 MWd/st to EOC-1000 MWd/st				
Load Rejection Without Bypass	1.37	1.34	1.15	1.13
Feedwater Controller Failure	1.32	1.29	1.25	1.23
Exposure Range:				
EOC-1000 MWd/st to EOC				
Load Rejection Without Bypass	1.40	1.37	1.28	1.25
Feedwater Controller Failure	1.36	1.34	1.29	1.27

## 12. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

<u>Transient</u>	<u>P<sub>sl</sub></u> (psig)	<u>P<sub>v</sub></u> (psig)	<u>Plant Response</u>
MSIV Closure (Flux Scram)	1218	1256	Figure 6

## 13. STABILITY ANALYSIS RESULT (S.2.4)

Rod Line Analyzed: Extrapolated Rod Block

Decay Ratio:

Figure 7

Reactor Core Stability Decay Ratio,  $x_2/x_0$ 

0.93

Channel Hydrodynamic Performance Decay Ratio,  $x_2/x_0$ 

Channel Type

8x8R/P8x8R

0.30



## 14. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

<u>Event</u>	<u>Initial CPR</u>	<u>Resulting CPR</u>
Misoriented	1.18	1.07

## 15. CONTROL ROD DROP ANALYSIS RESULTS (S.2.5.1)

## Bounding Analysis Results:

Doppler Reactivity Coefficient:	Figure 8
Accident Reactivity Shape Functions:	Figures 9 and 10
Scram Reactivity Functions:	Figures 11 and 12

## Plant Specific Analysis Results:

Parameter(s) not Bounded, Cold:	Accident Reactivity
Resultant Peak Enthalpy, Cold:	225.5 cal/gm
Parameter(s) not Bounded, HSB :	Accident Reactivity
Resultant Peak Enthalpy, HSB :	272.2 cal/gm

## 16. LOSS-OF-COOLANT ACCIDENT RESULT (S.2.5.2)

See "Loss-of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant (Lead Plant)," July 1977, NEDO-21662 (as amended).

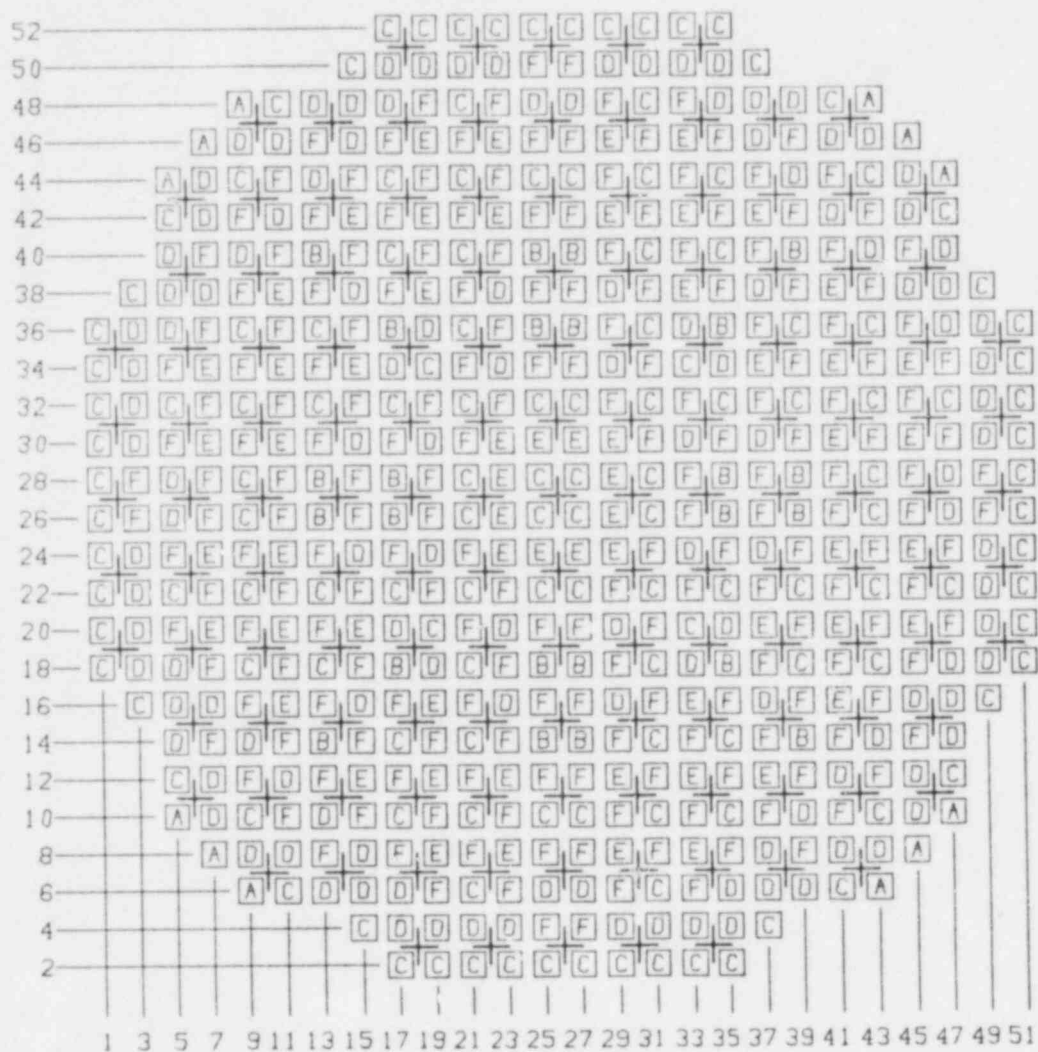


Figure 1. Reference Core Loading Pattern

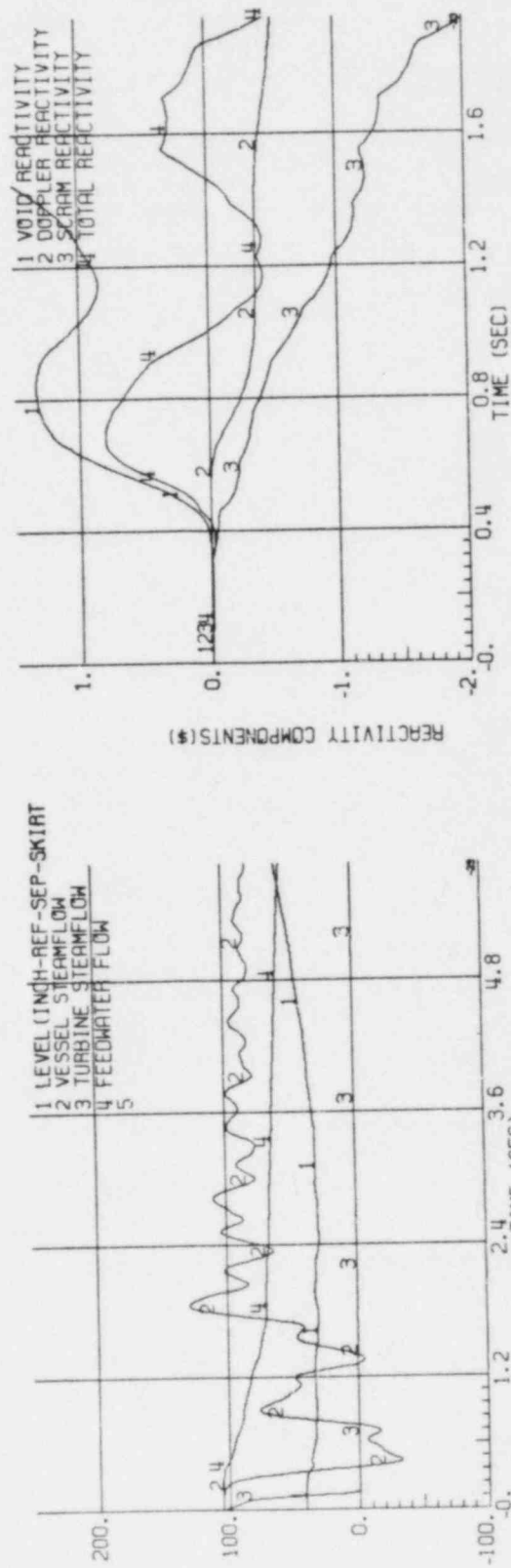
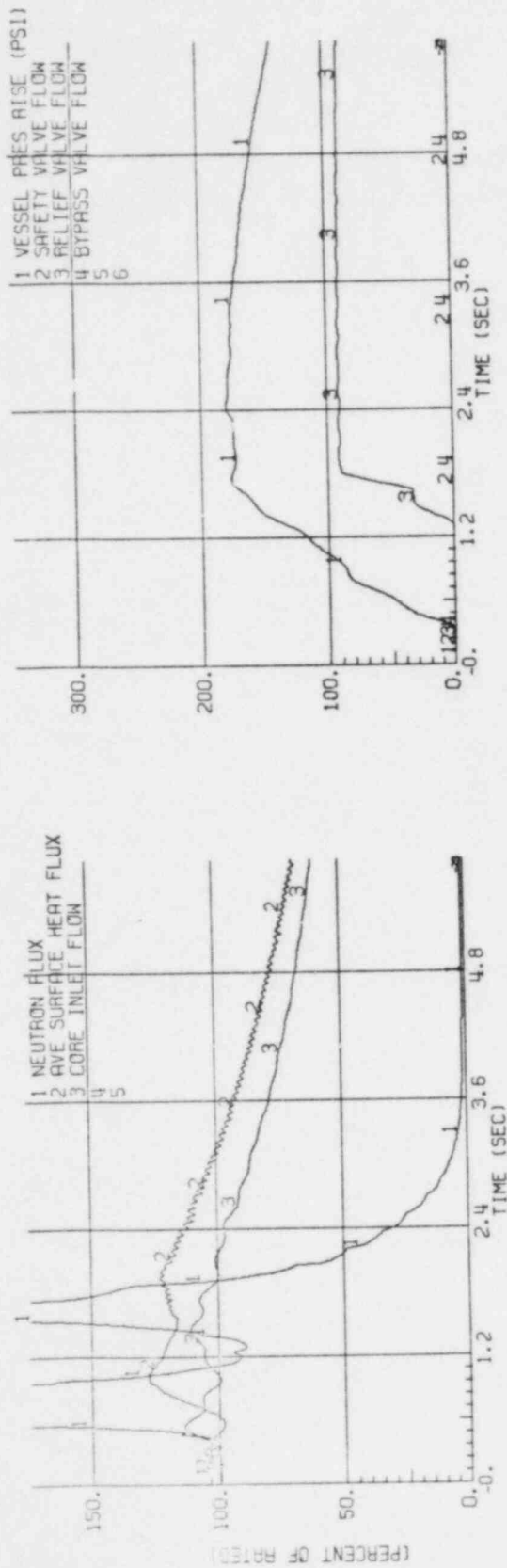


Figure 2a. Plant Response to Load Rejection Without Bypass (EOC)

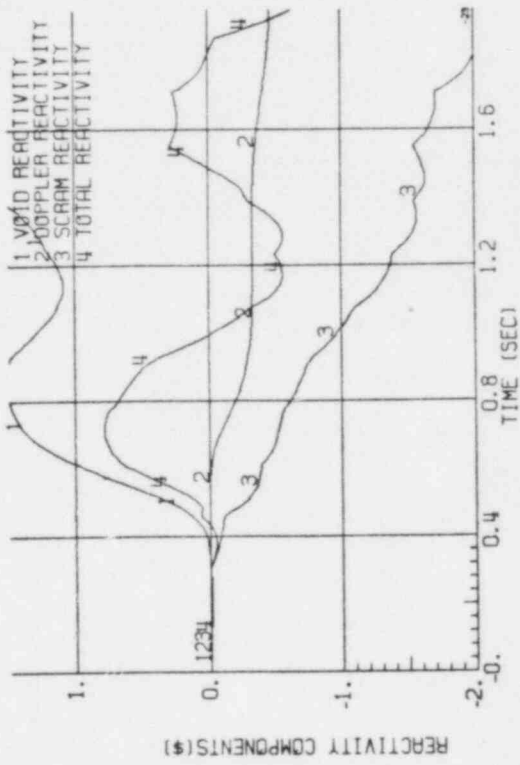
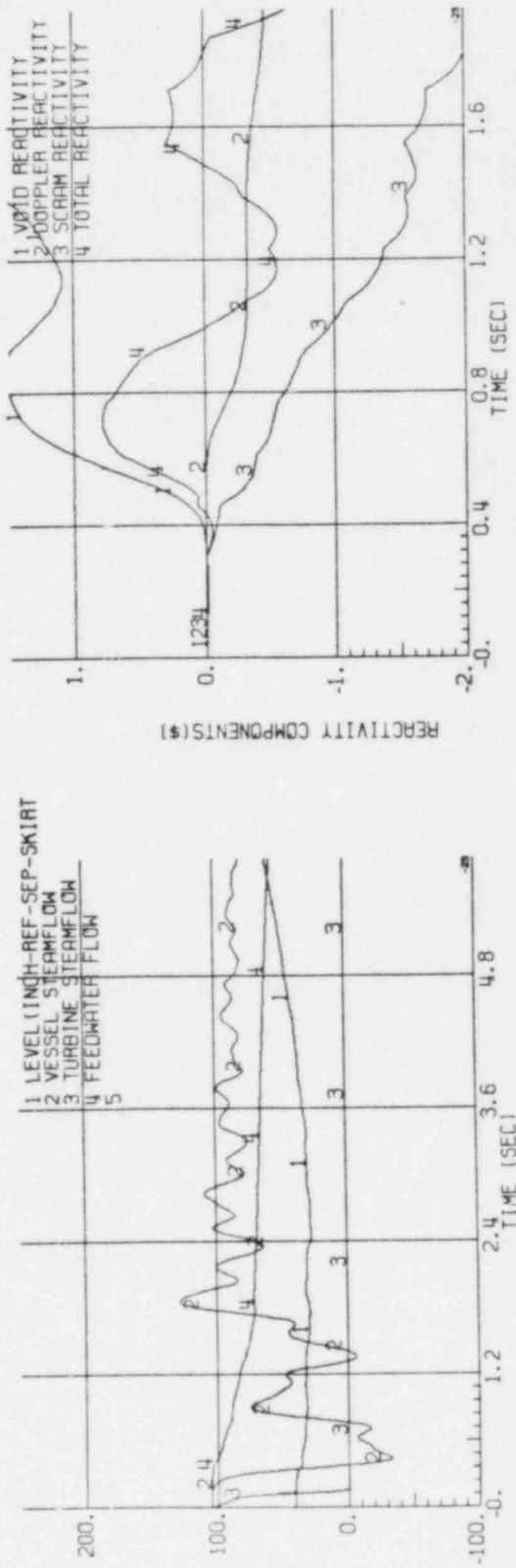
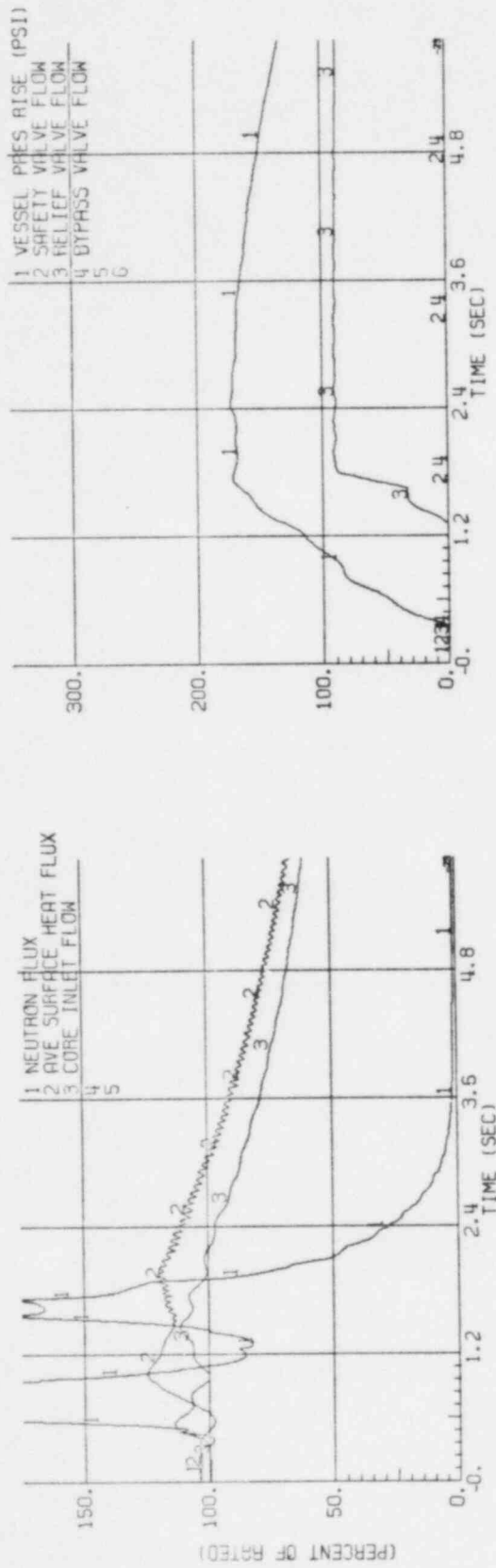


Figure 2b. Plant Response to Load Rejection Without Bypass (EOC-1000 MWd/st)

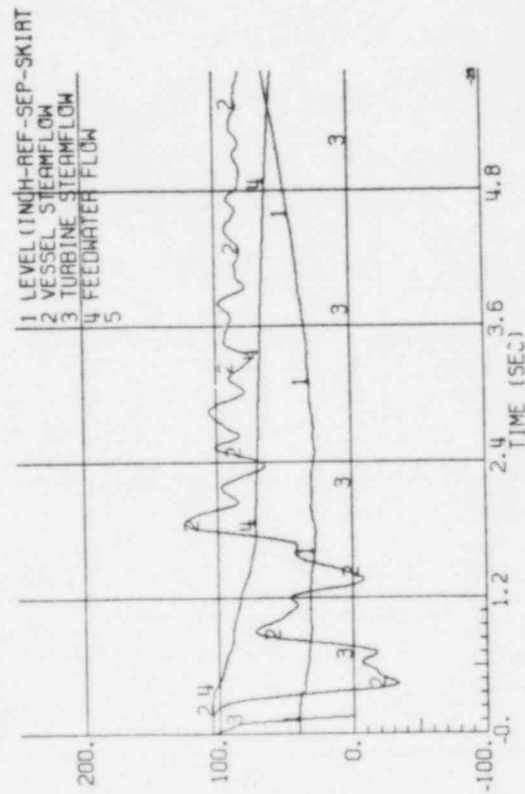
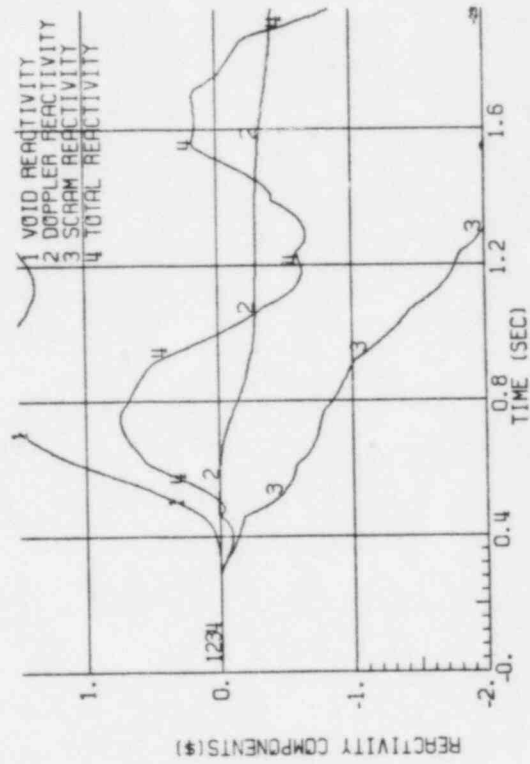
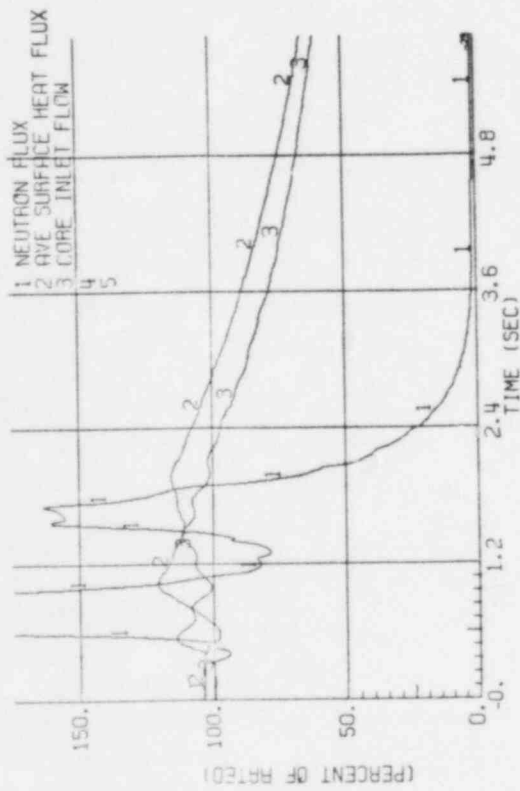
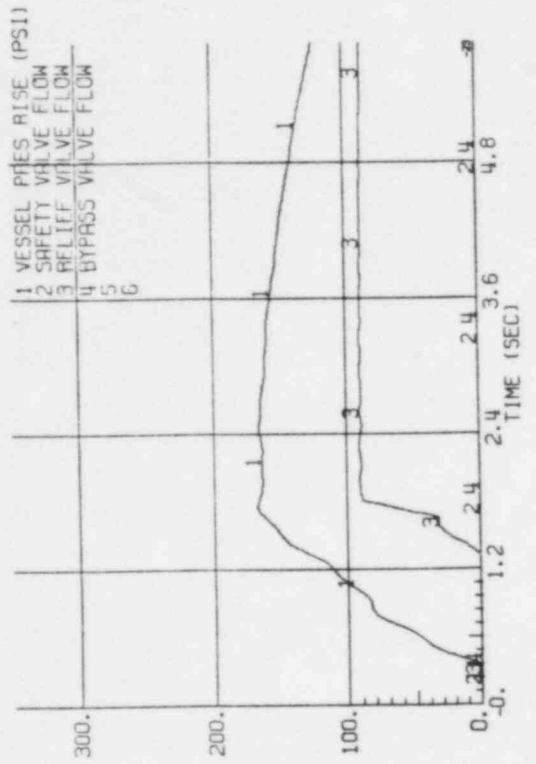


Figure 2c. Plant Response to Load Rejection Without Bypass (EOC-2000 MWd/st)

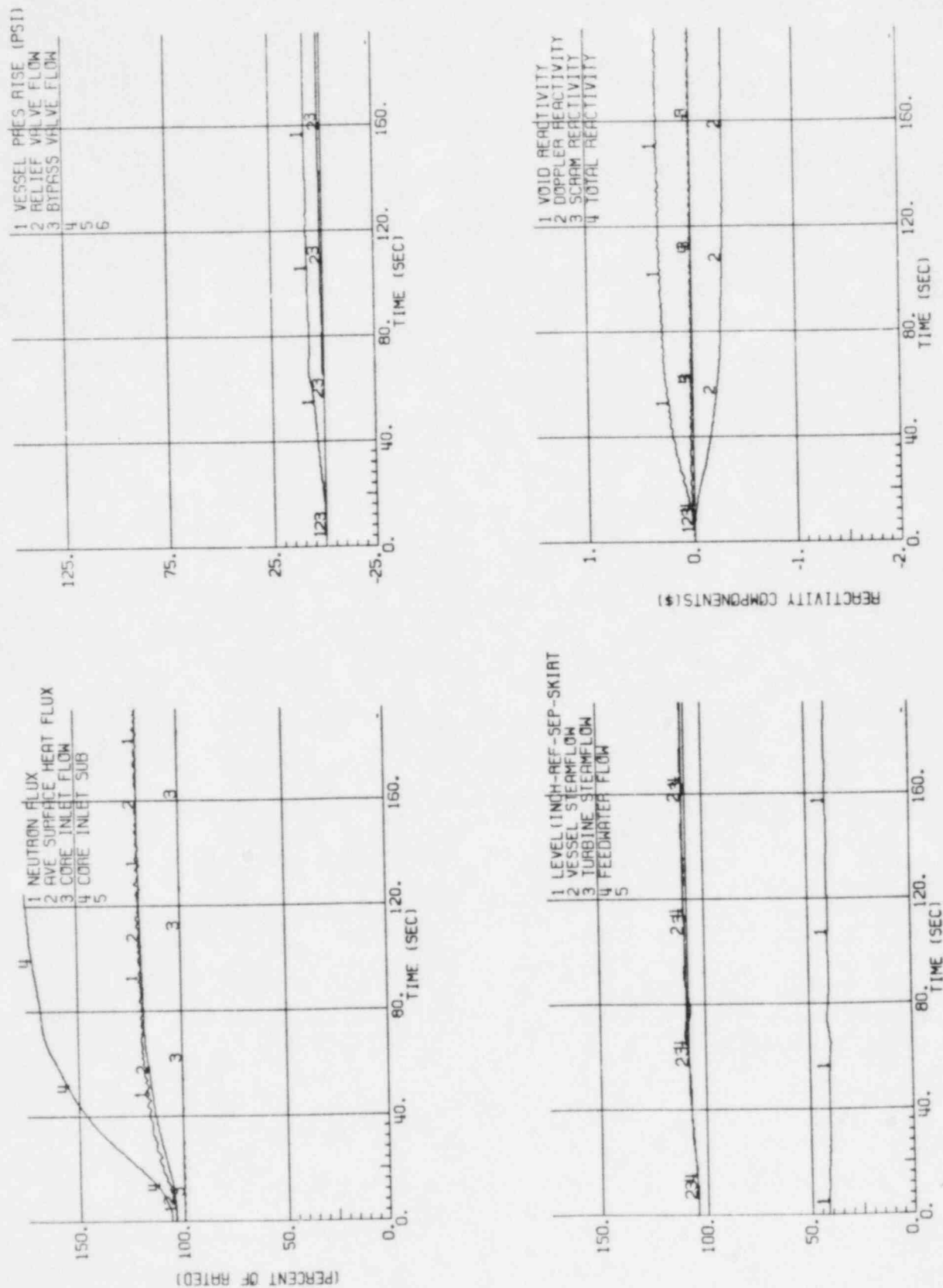


Figure 3. Plant Response to Loss of 80°F Feedwater Heating

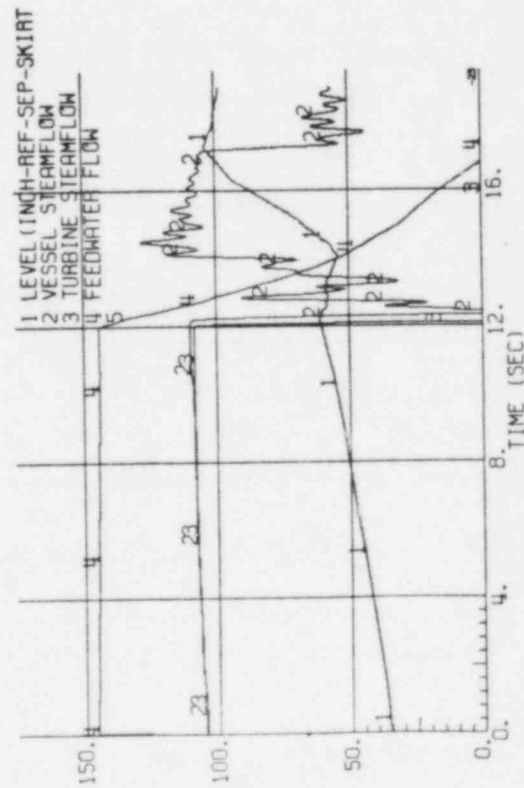
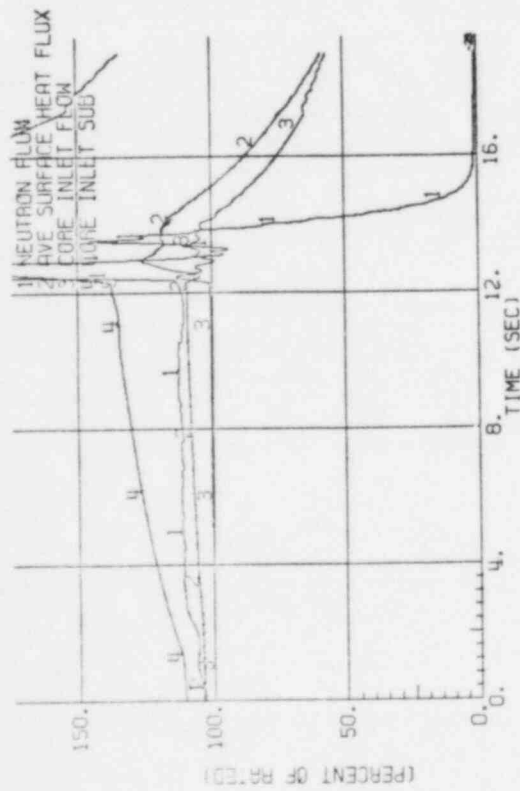
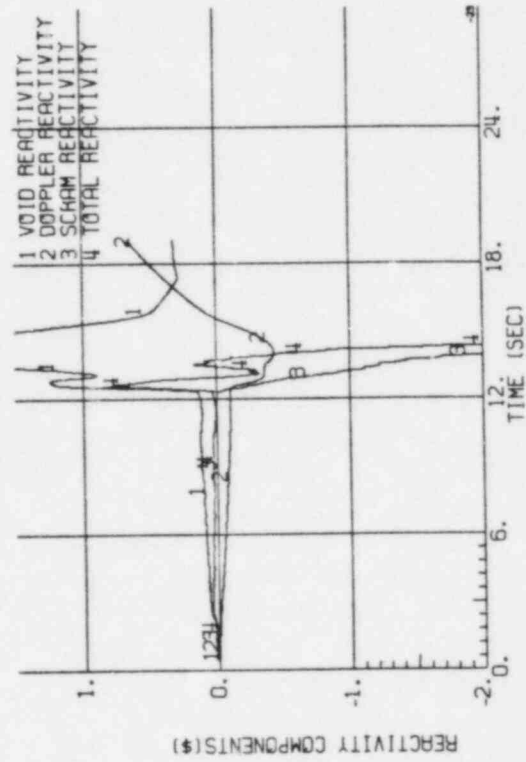
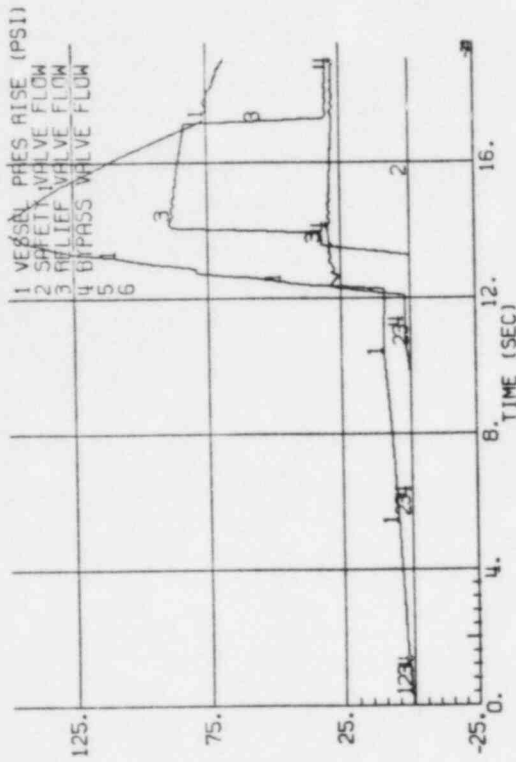


Figure 4a. Plant Response to Feedwater Controller Failure (EOC)



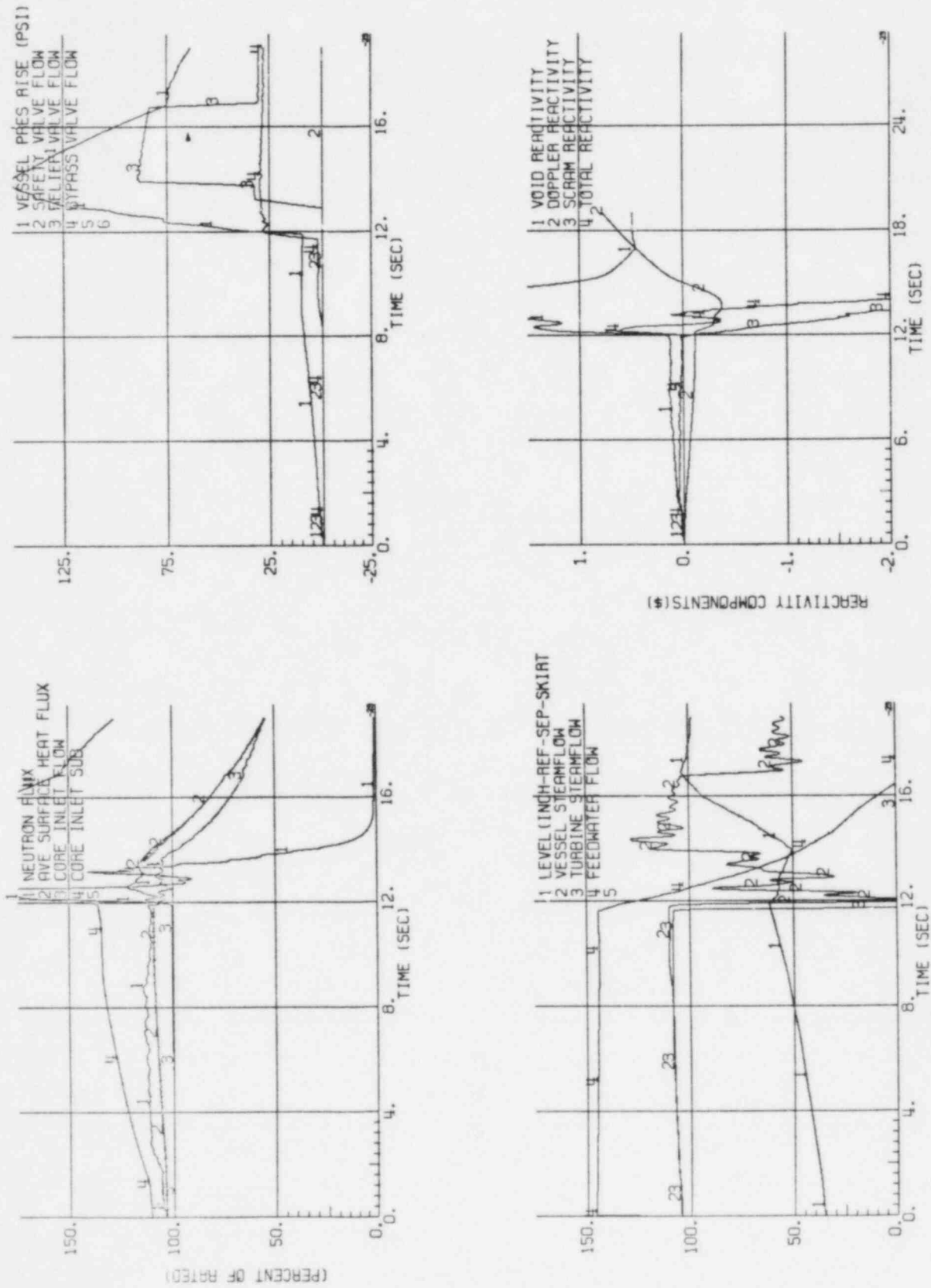


Figure 4b. Plant Response to Feedwater Controller Failure (EOC-1000 MWd/st)



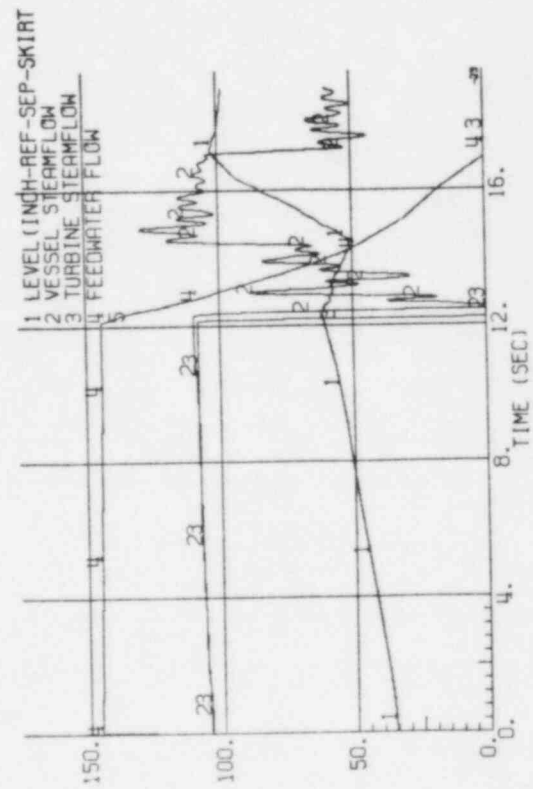
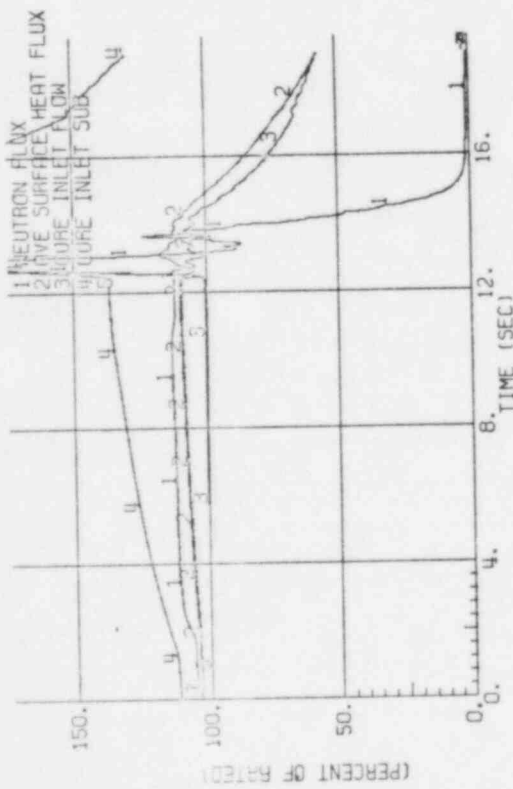
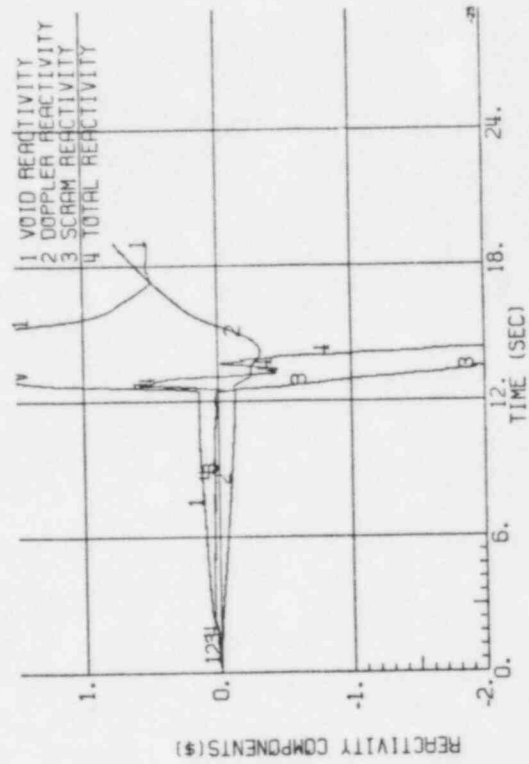
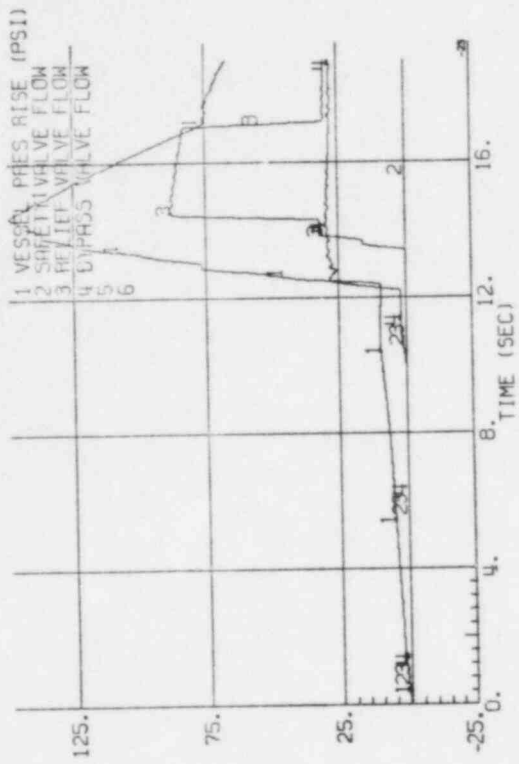


Figure 4c. Plant Response to Feedwater Controller Failure (EOC-2000 MWd/st)

	2	6	10	14	18	22	26	30	34	38	42	46	50
51					36		36		36				
47				6		6		6		6			
43			36		36		36		36		36		
39		6		6		12		12		6		6	
35	36		36		40		44		40		36		36
31		6		12		0		0		12		6	
27	36		36		44		40		44		36		36
23		0		12		0		0		12		6	
19	36		36		40		44		40		36		36
15		6		6		12		12		6		6	
11			36		36		36		36		36		
7				6		6		6		6			
3					36		36		36				

- NOTES: 1. NUMBER INDICATES NUMBER OF NOTCHES WITHDRAWN  
OUT OF 48. BLANK IS A WITHDRAWN ROD  
2. ERROR ROD IS (22,31).

Figure 5. Limiting RWE Rod Pattern

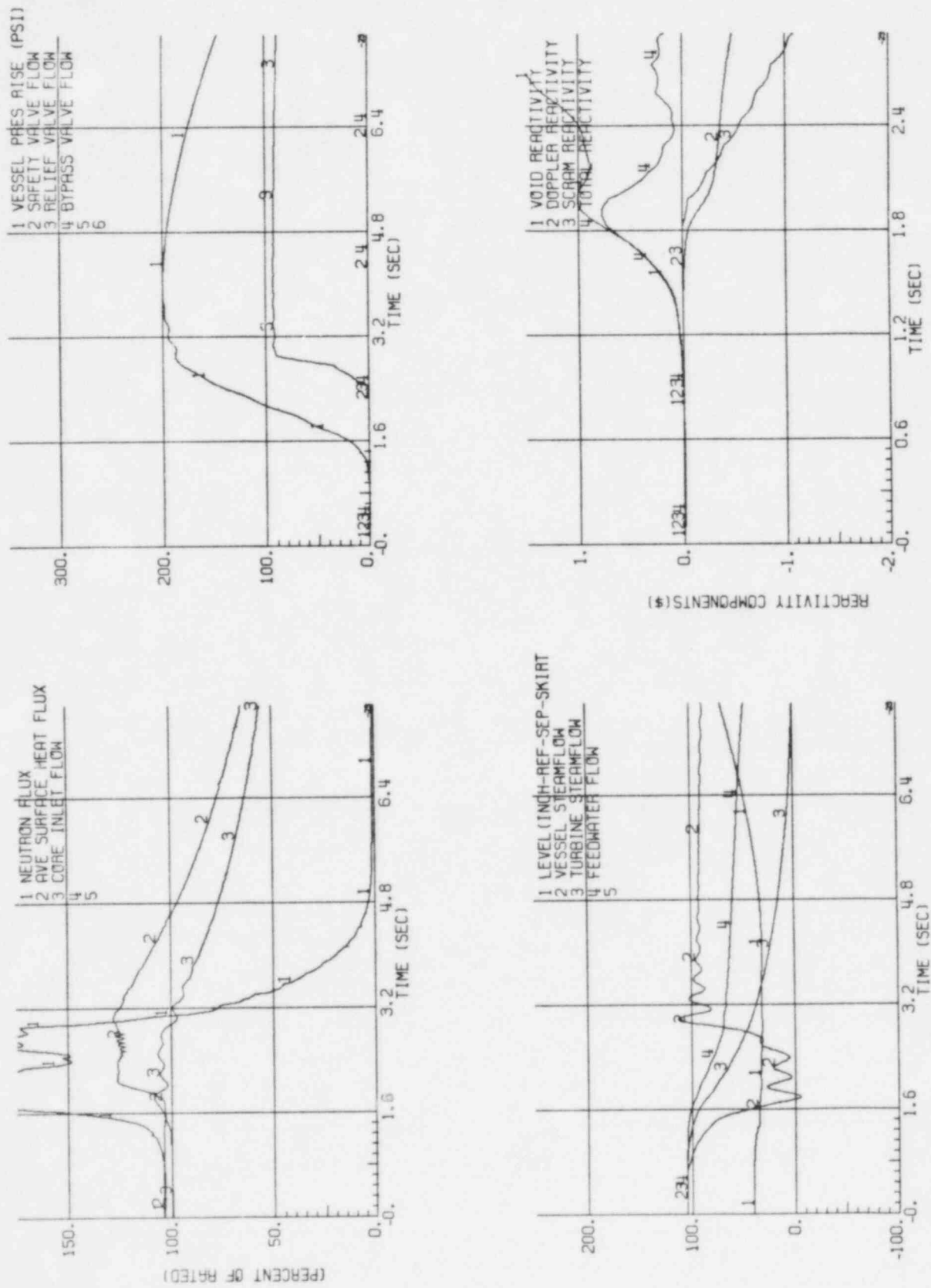


Figure 6. Plant Response to MSIV Closure (Flux Scram)

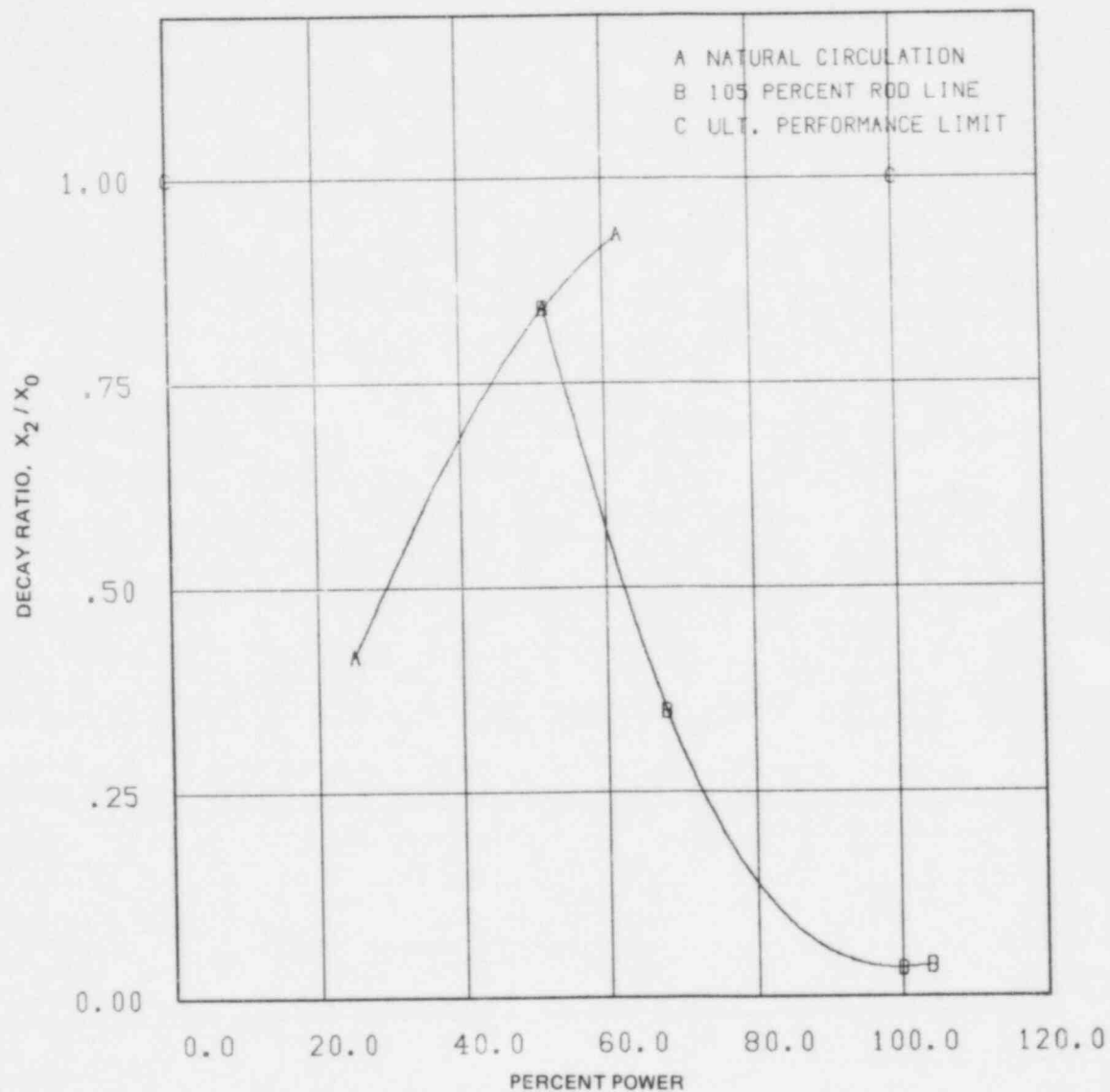


Figure 7. Reactor Core Decay Ratio versus Power

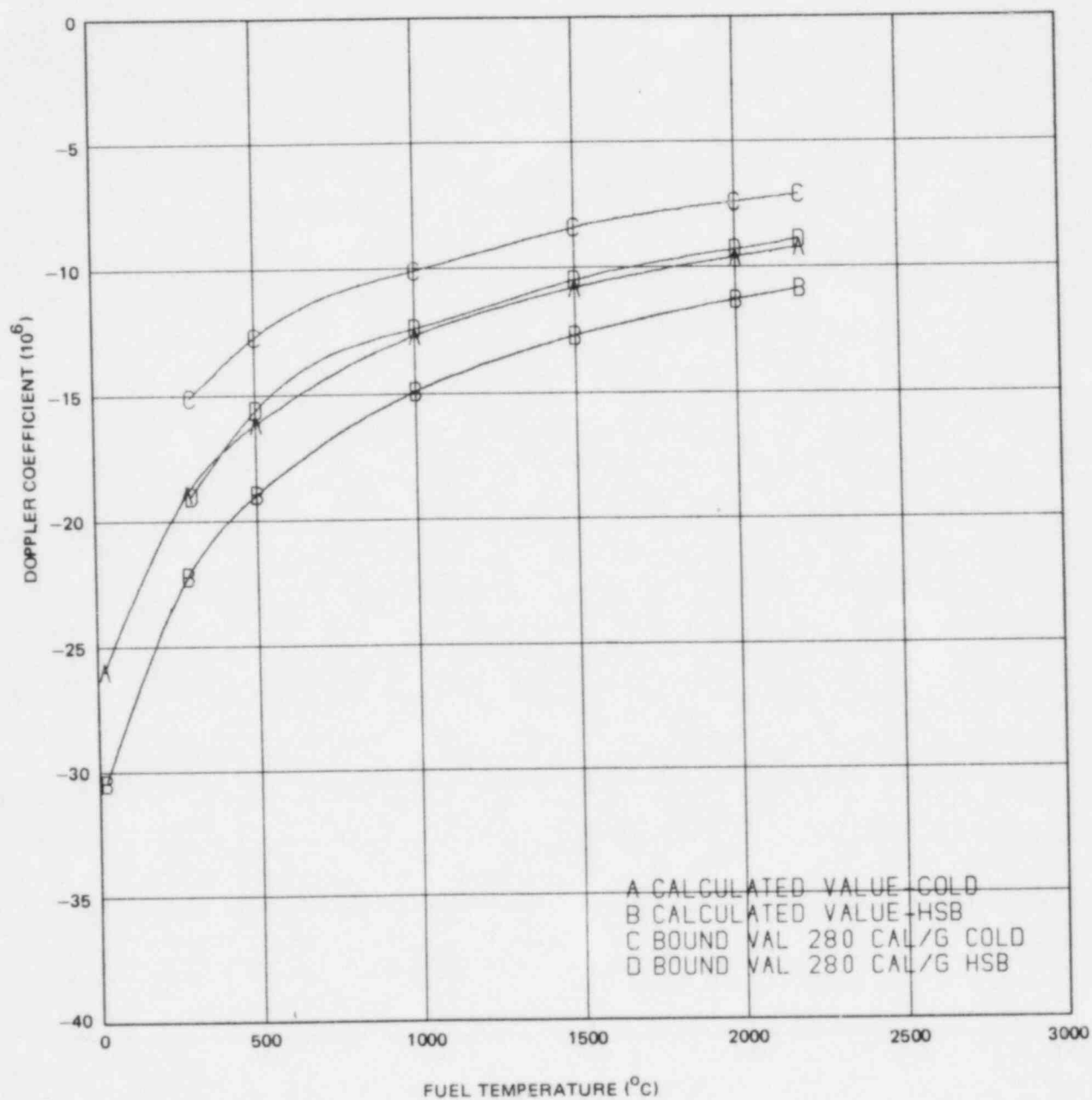


Figure 8. Doppler Reactivity Coefficient in  $1/\Delta^{\circ}\text{C}$

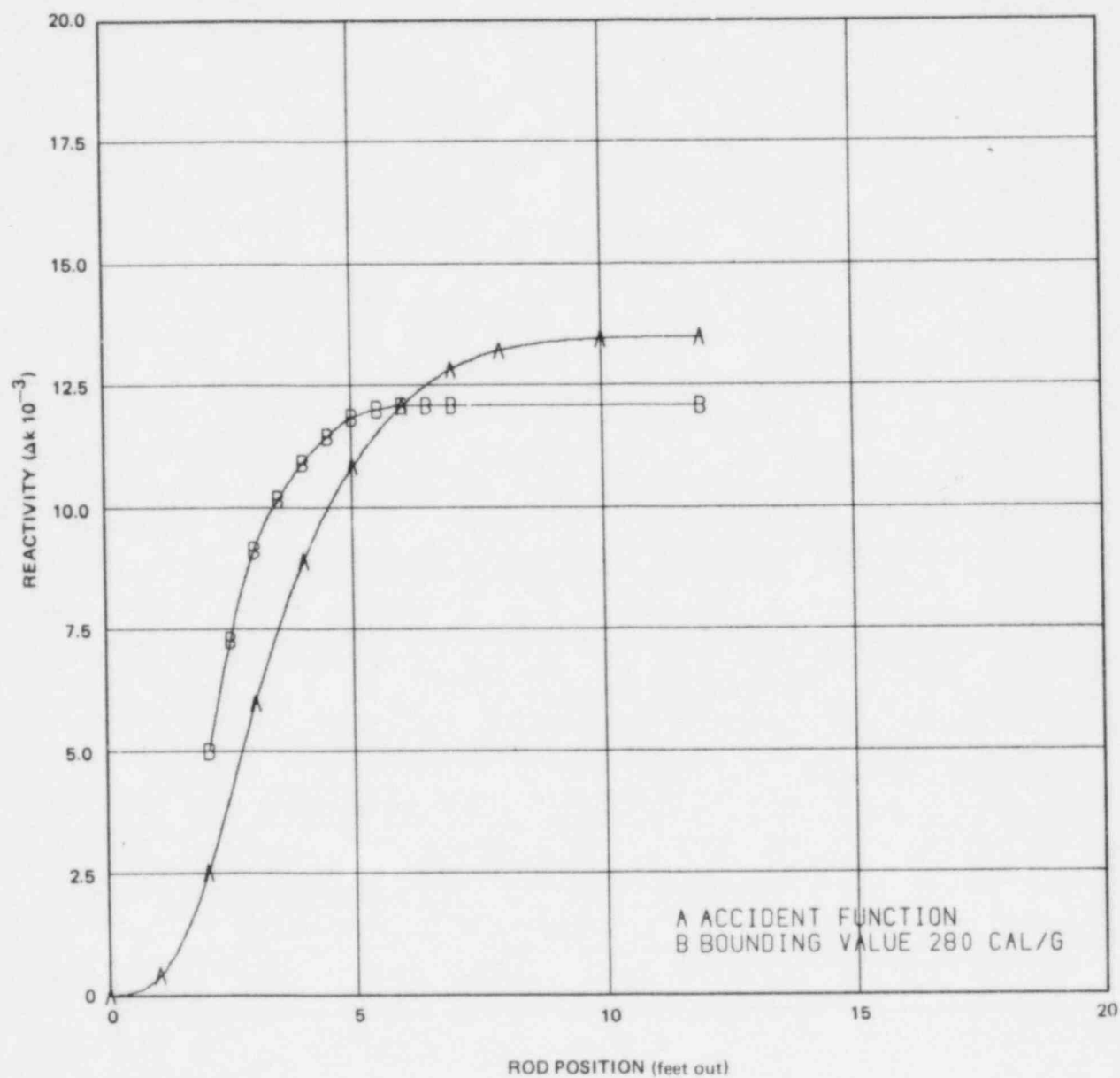


Figure 9. Accident Reactivity Shape Function, Cold Startup

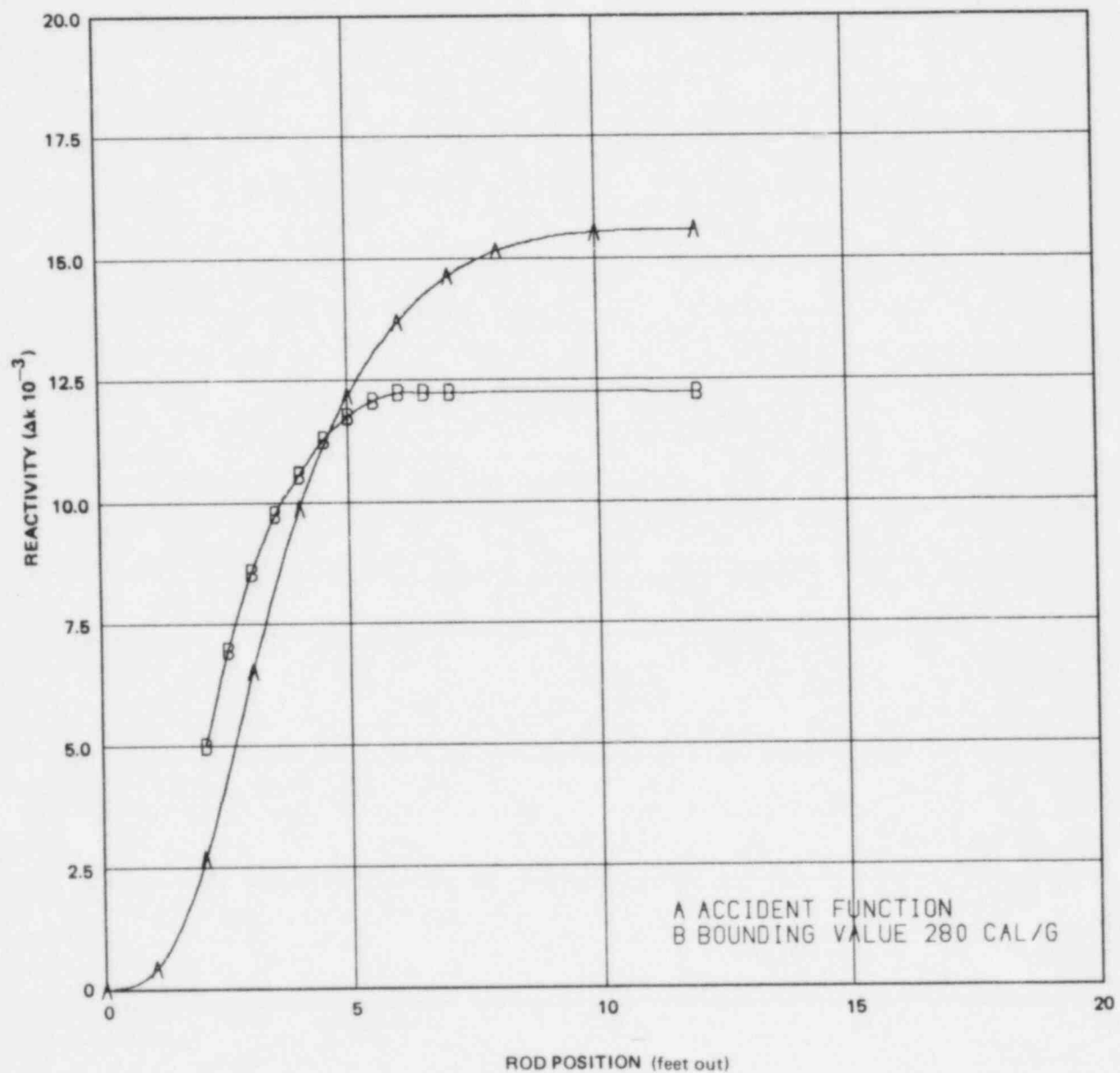


Figure 10. Accident Reactivity Shape Function, Hot Standby

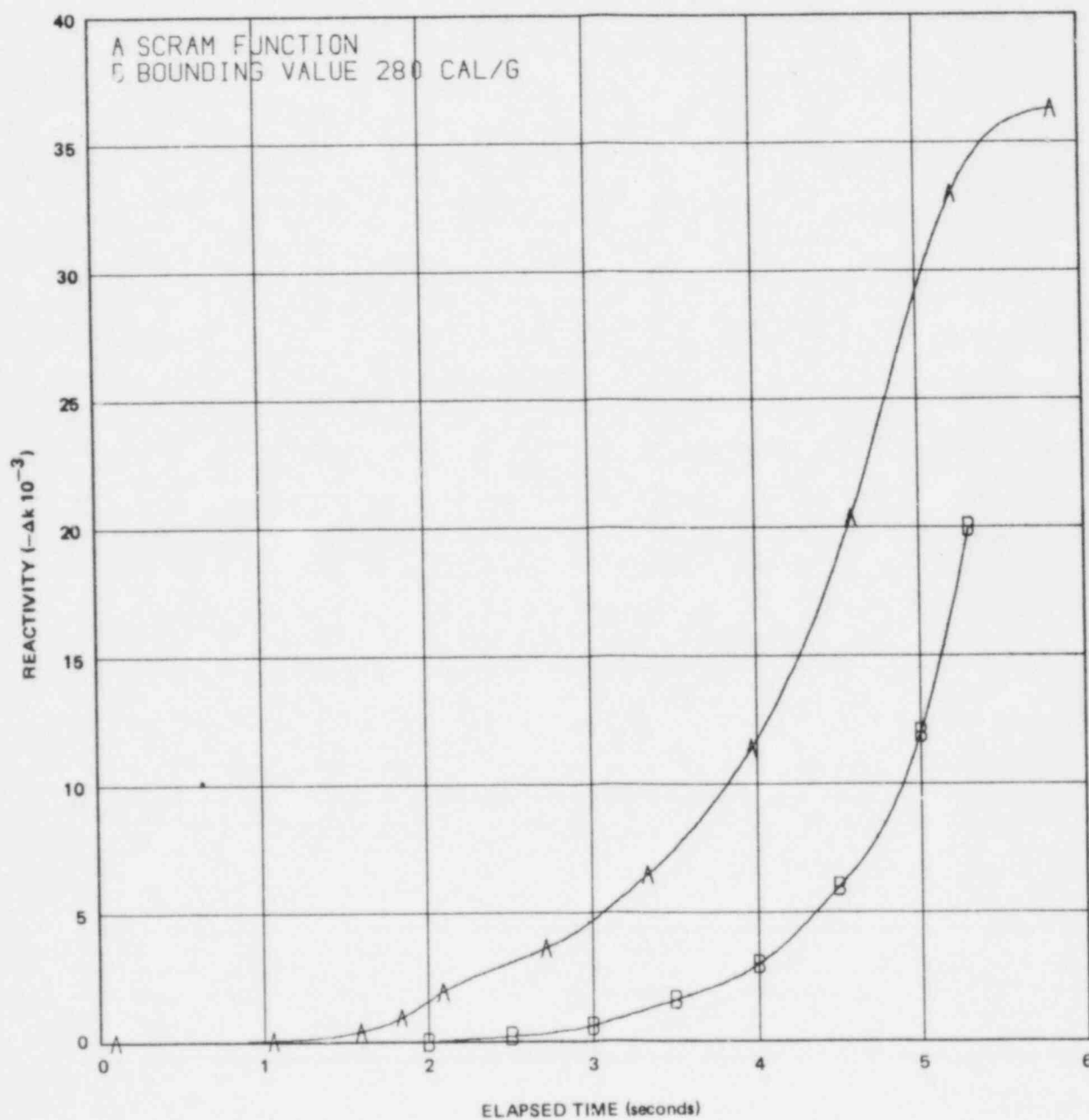


Figure 11. Scram Reactivity Function, Cold Startup



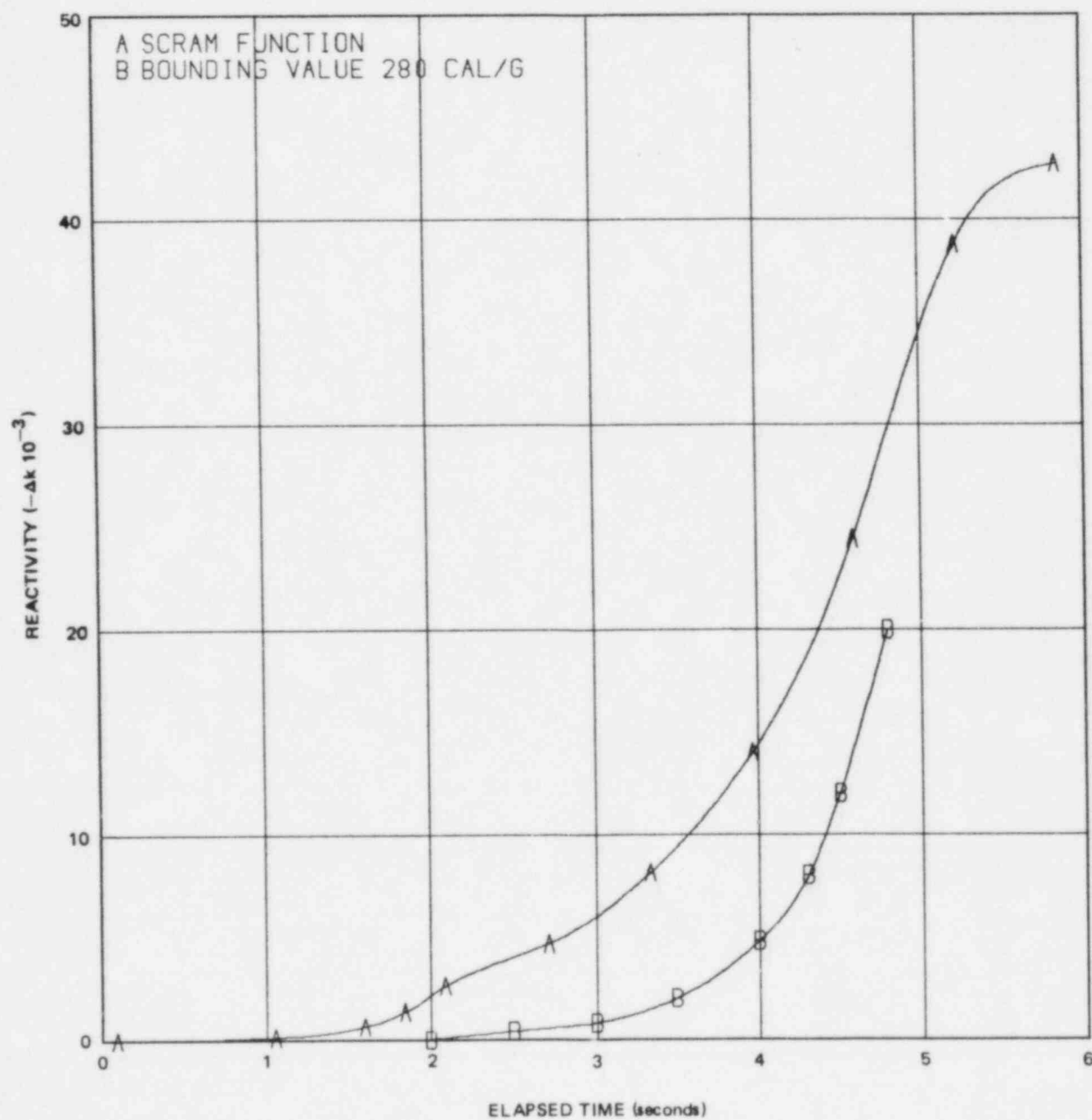


Figure 12. Scram Reactivity Function, Hot Standby

APPENDIX A  
GETAB ANALYSIS INITIAL CONDITIONS

The values listed below were used in the GETAB analysis for this reload rather than the values given in Reference A-1, to more nearly reflect actual plant data.

Reactor Pressure (psia)	1035
Inlet Enthalpy (Btu/lb)	527.0

REFERENCE

- A-1. "General Electric Standard Application for Reactor Fuel,"  
NEDE-24011-P-A-4, January 1982.

APPENDIX B  
VERIFICATION OF OPERATING FLEXIBILITY OPTIONS

The following operating flexibility options have been developed for BWRs.  
A "Yes" indicates that the option has been verified as being applicable to  
Cycle 6.

1. Single Loop Operation: Yes
2. Load Line Limit: Yes
3. Extended Load Line Limit: No
4. Increased Core Flow: No
5. Feedwater Temperature Reduction: No

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