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Revision 1
Class I
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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
HATCH NUCLEAR PLANT
UNIT 2, RELOAD 2

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GENERAL ELECTRIC

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CONTENTS OF THIS REPORT
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1. PLANT-UNIQUE ITEMS (1.0)*

Safety Relief Valve Capacity: Appendix A

Feedwater Temperature Reduction: Appendix B

Error in ODYN Code: Appendix C

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

	Fuel Type	Cycle Loaded	Number	Number Drilled
Irradiated	8DRB221(IC)	1	76	76
	8DRB221(IC)	1	200	200
	P8DRB284LA	2	164	164
New	P8DRB283	3	120	120
Total			560	560

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle: 12047 MWd/t

Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations: 12046 MWd/t

Assumed reload cycle core average exposure at end of cycle: 14,621 MWd/t

Core loading pattern: Figure 1

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

Minimum Shutdown Margin, BOC k_{eff}

Uncontrolled	1.112
Fully Controlled	0.955
Strongest Control Rod Out	0.986
R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, Δk	0.000

*( ) Refers to area of discussion in "Generic Reload Fuel Application," NEDE-24011-P-A-2 and NEDO-24011-A-2, July 1981.

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) ($20^{o}C$, Xenon Free)
660	0.048

6. RELOAD-UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 and 5.2)

(Loss of Feedwater Heating Event Only)

	EOC-3
Void Fraction (%)	41.4
Average Fuel Temperature (^{o}F)	1315.0
Void Coefficient N/A* (¢/% Rg)	-8.08/-10.10
Doppler Coefficient N/A (¢/^{o}F)	-0.228/-0.217
Scram Worth N/A ($)	-46.31/-37.05

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

Fuel Design	Peaking Factors Local	Radial	Axial	R-Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
BOC 3 to EOC 3							
P8X8R	1.20	1.47	1.40	1.051	6.258	114.0	1.29
8X8R	1.20	1.50	1.40	1.051	6.373	113.1	1.27

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Transient Recategorization:	No
Recirculation Pump Trip:	Yes
Rod Withdrawal Limiter:	No
Thermal Power Monitor:	Yes
Measured Scram Time:	No
Number of Exposure Points	1

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*N = Nuclear Input Data.
A = Used in Transient Analysis.

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Flux (% NBR)	Q/A (% NBR)	ΔCPR P8X8R	ΔCPR 8x8R	Figure
Exposure: BOC 3 to EOC 3 Load Rejection without Bypass	518	122	0.23	0.20	2
Exposure: BOC to EOC Loss of Feedwater Heater	126	121	0.14	0.14	3
Exposure: BOC 3 to EOC 3 Feedwater Controller Failure	316	122	0.20	0.18	4

10. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (5.2.1)

Limiting Rod Pattern: Figure 5
Includes 2.2% Power Spiking Penalty: Yes

Rod Block Reading	Rod Position (Feet Withdrawn)	ΔCPR P8x8R/8x8R	MLHGR (kW/ft) 8X8R/ 7X7 8X8 P8X8R
104	3.0	0.14/0.12	16.91
105	3.5	0.15/0.13	17.48
106	4.0	0.17/0.15	17.82
107	4.0	0.17/0.15	17.82
108	4.5	0.18/0.16	17.85
109	5.0	0.19/0.17	17.85
110	5.0	0.19/0.17	17.85

Setpoint Selected is: 107

11. CYCLE MCPR VALUES (5.2)

Non-Pressurization Events
Exposure Range: BOC to EOC

	P8X8R	8X8R
Loss of Feedwater Heater	1.21	1.21
Fuel Loading Error	1.24	
Rod Withdrawal Error	1.24	1.22

Pressurization Events
Exposure Range: BOC 3 to EOC 3

	Option A P8X8R	Option A 8X8R	Option B P8X8R	Option B 8X8R
Load Rejection without Bypass	1.36	1.33	1.26	1.23
Feedwater Controller Failure	1.33	1.30	1.29	1.27

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1205	1226	Figure 6

13. STABILITY ANALYSIS RESULTS (5.4)

Rod Line Analyzed: Extrapolated Rod Block Line
Decay Ratio: Figure 7

Reactor Core Stability Decay Ratio, x_2/x_0: 0.83

Channel Hydrodynamic Performance Decay Ratio, x_2/x_0
Channel Type
P8X8R/8X8R 0.64

14. LOADING ERROR RESULTS (5.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes
Includes 2.2% Power Spiking Penalty: Yes

Event	Initial MCPR	Resulting MCPR	Resulting* LHGR
Misoriented	1.22	1.07	17.66

15. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

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: None
- Resultant Peak Enthalpy, Cold:

- Parameter(s) not Bounded, HSB: None
- Resultant Peak Enthalpy, HSB:

*To be eliminated after approval of event classification.

16. LOSS-OF-COOLANT ACCIDENT RESULTS, NEW FUEL (5.5.2)

	FUEL TYPE: P8DRB283		
Exposure (MWd/T)	MAPLHGR (kW/ft)	PCT (^{o}F)	Local Oxidation (Fraction)
200	11.30	2133	0.029
1000	11.40	2134	0.028
5000	11.90	2185	0.033
10000	12.10	2195	0.033
15000	12.10	2199	0.033
20000	11.90	2184	0.032
25000	11.30	2112	0.025
30000	11.10	2061	0.021
35000	10.50	1981	0.030
40000	9.80	1878	0.017

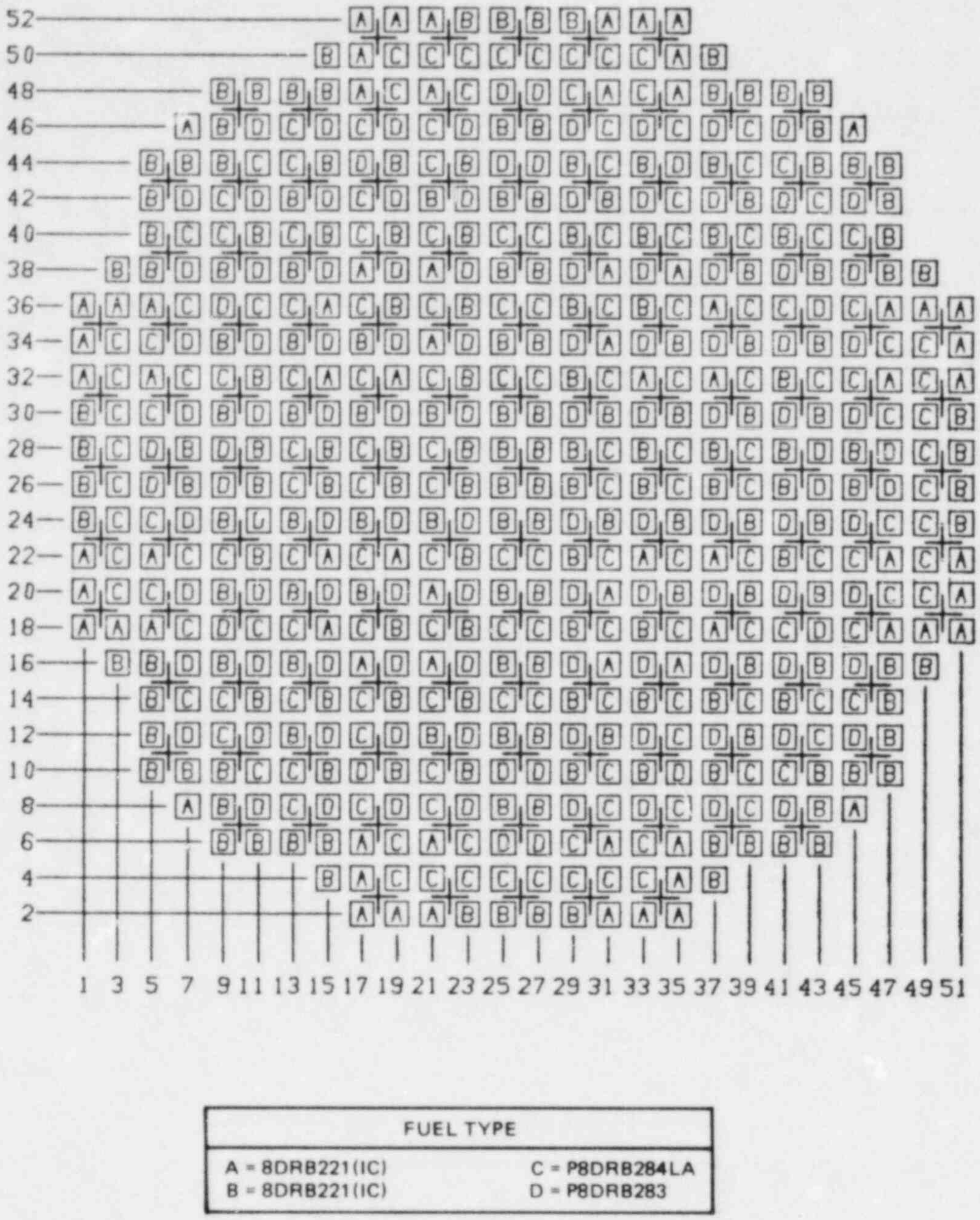

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Figure 1. Reference Core Loading Pattern

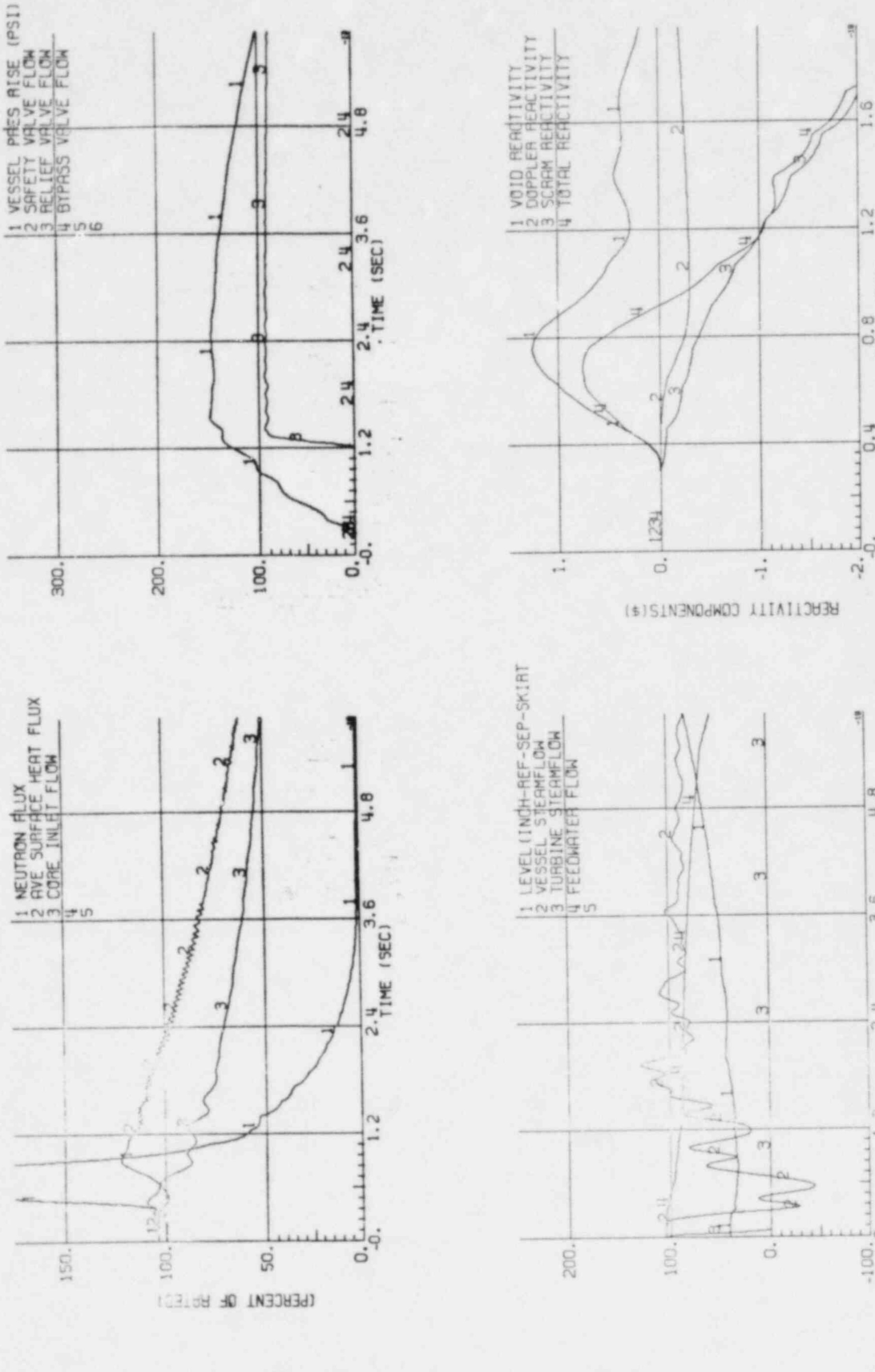

Figure 2. Plant Response to Generator Load Rejection, without Bypass

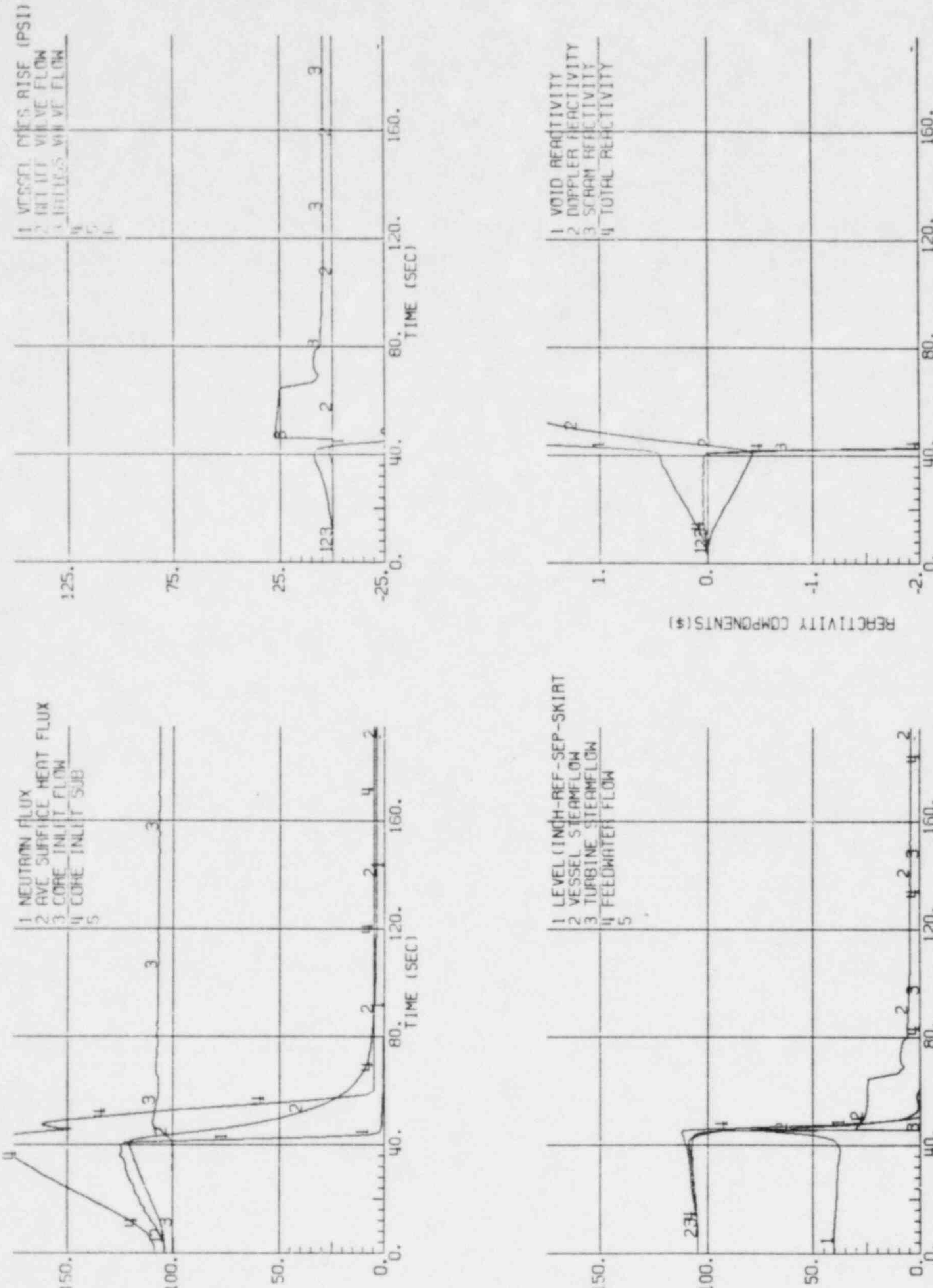

Figure 3. Plant Response to Loss of 140°F Feedwater Heating

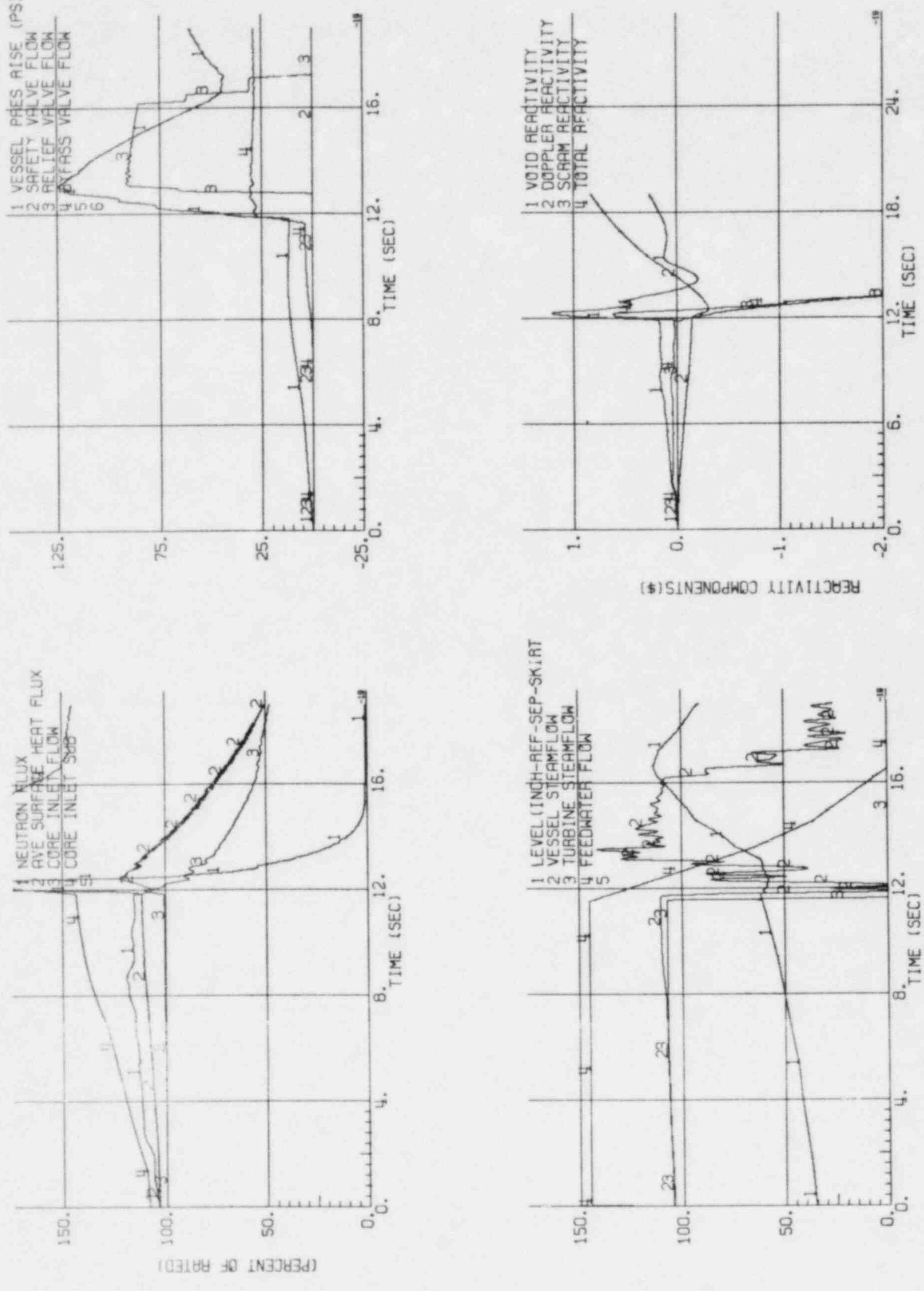

Figure 4. Plant Response to Feedwater Controller Failure

NOTES: 1. ROD PATTERN IS 1/4 CORE MIRROR SYMMETRIC.
2. NO. INDICATES NUMBER OF NOTCHES WITHDRAWN OUT OF 48. BLANK IS A WITHDRAWN ROD.
3. ERROR ROD IS (26, 31).

	2	6	10	14	18	22	26
51						40	
47			6		10		10
43		36		40		40	
39			10		10		10
35		40		40		40	
31	6		10		10		0
27		40		40		40	

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Figure 5. Limiting RWE Rod Pattern

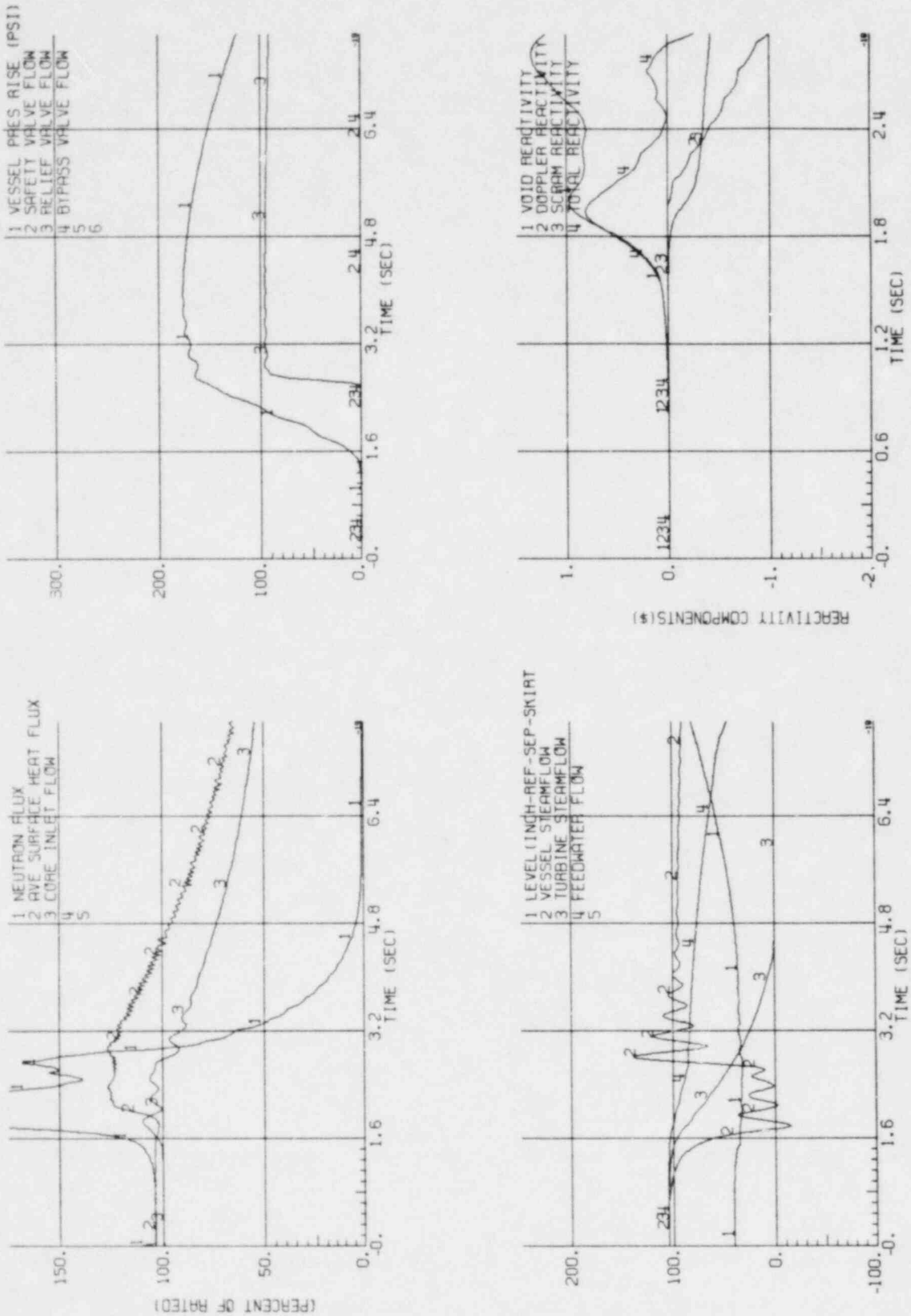

Figure 6. Plant Response to MSIV Closure, Flux Scram

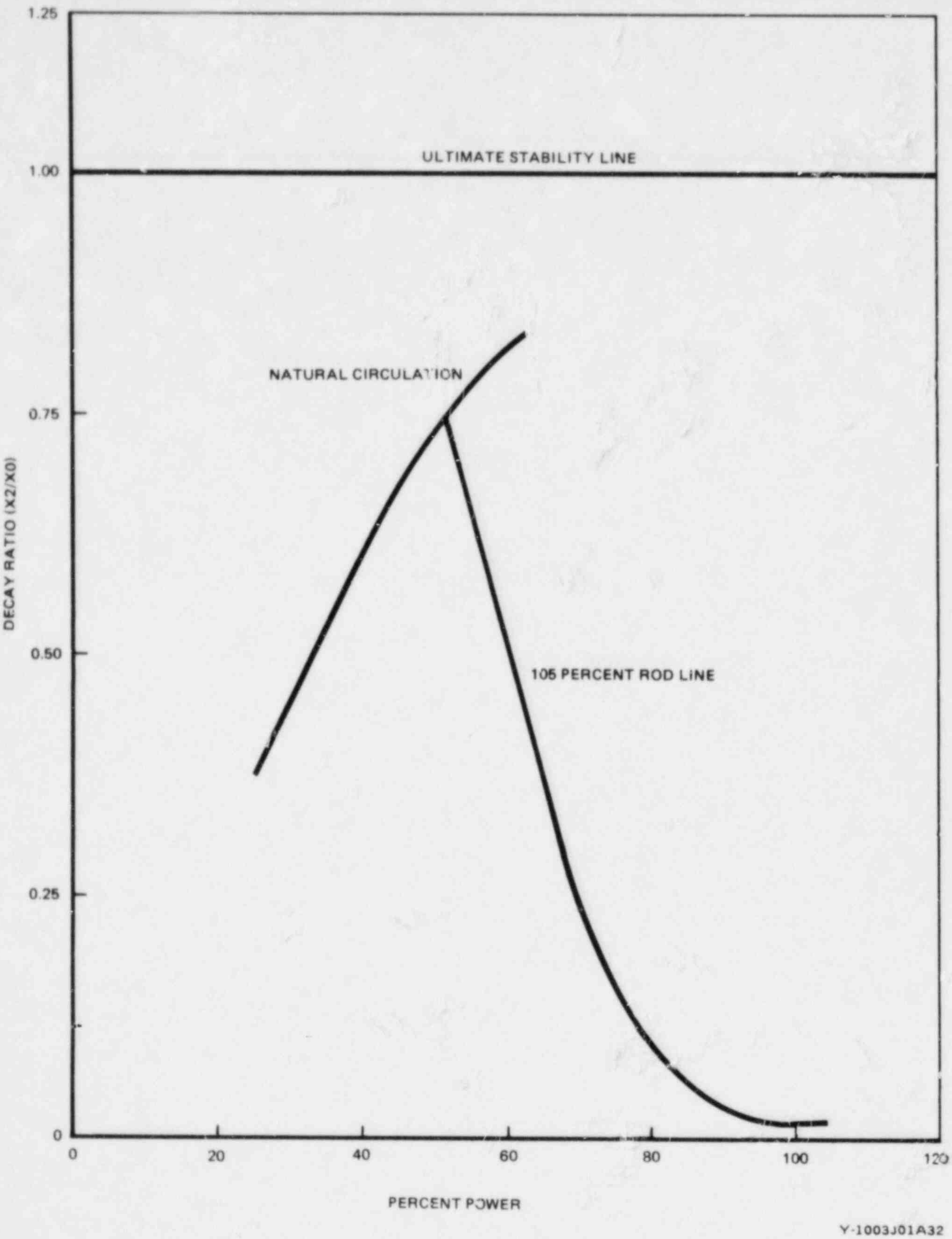

Figure 7. Reactor Core Decay Ratio

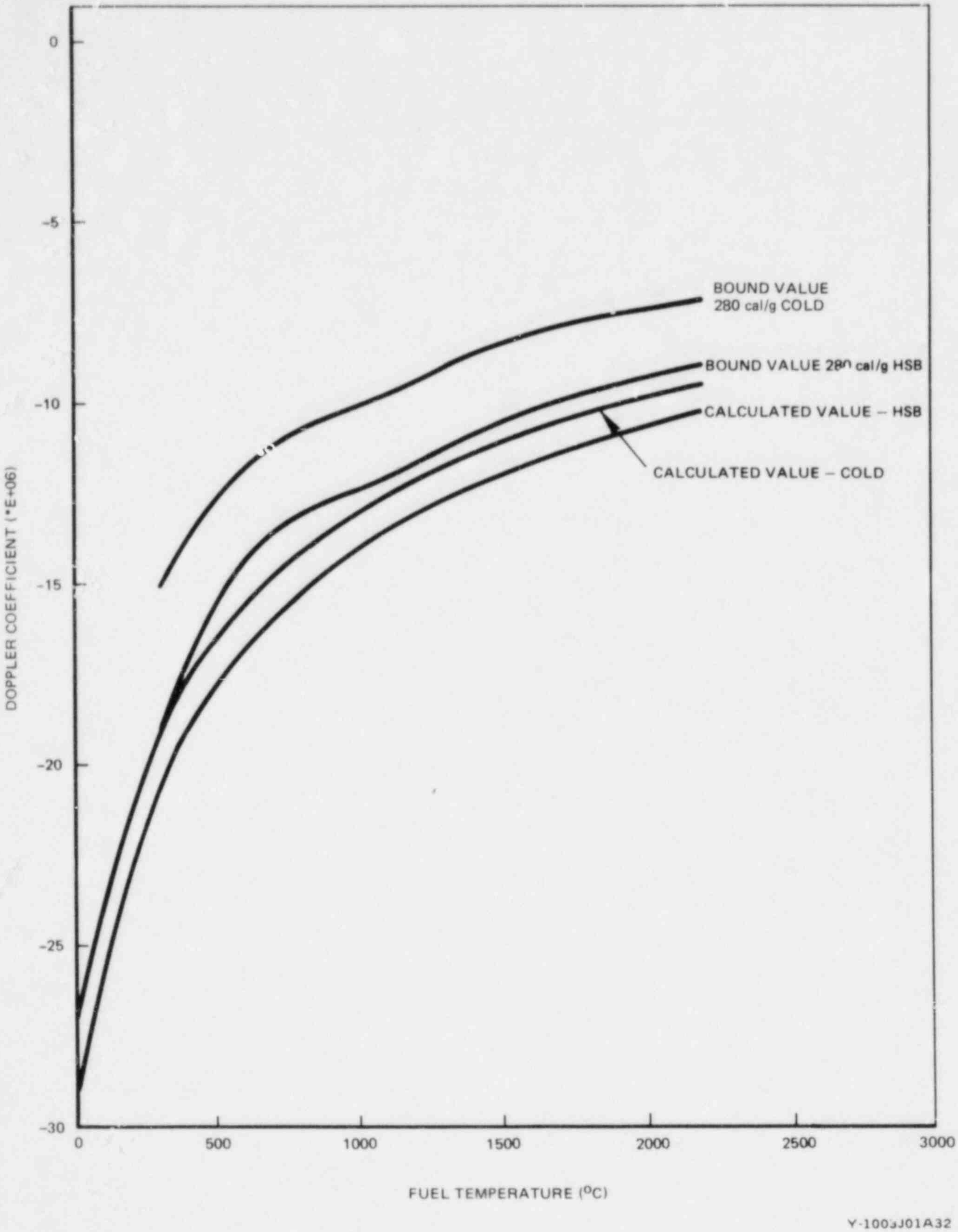

Figure 8. Fuel Doppler Coefficient in 1/Δ °C

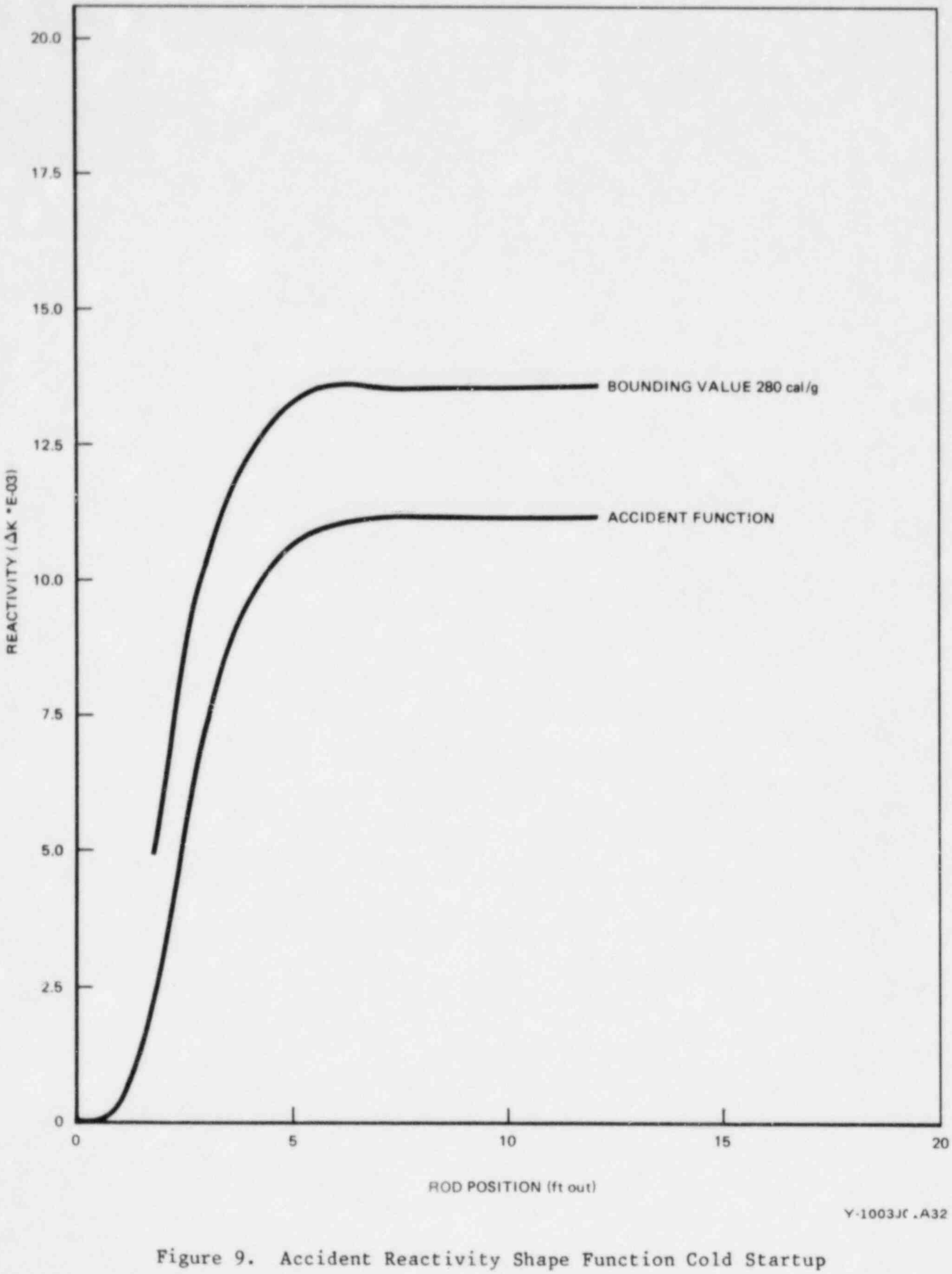

Figure 9. Accident Reactivity Shape Function Cold Startup

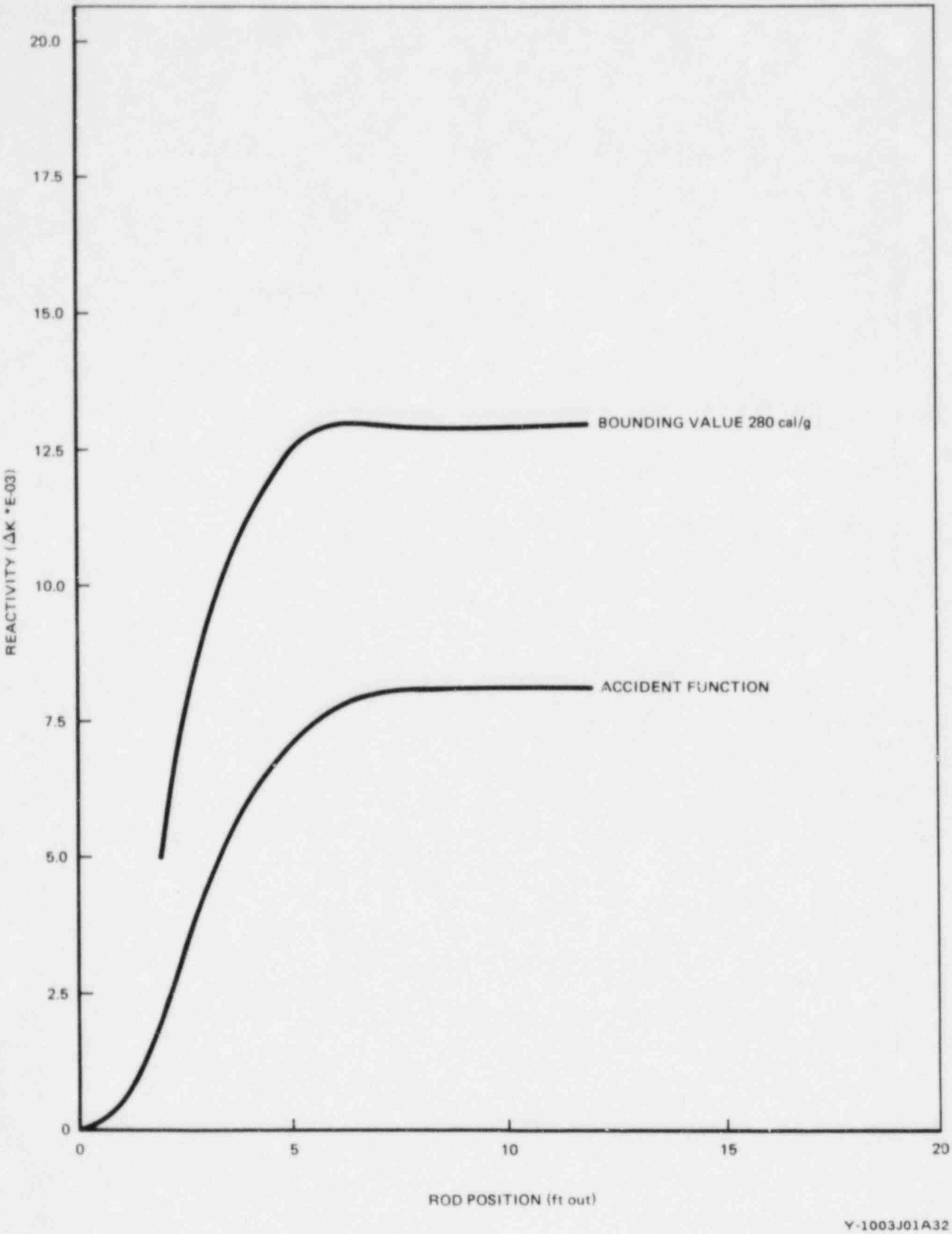

Figure 10. Accident Reactivity Shape Function Hot Startup

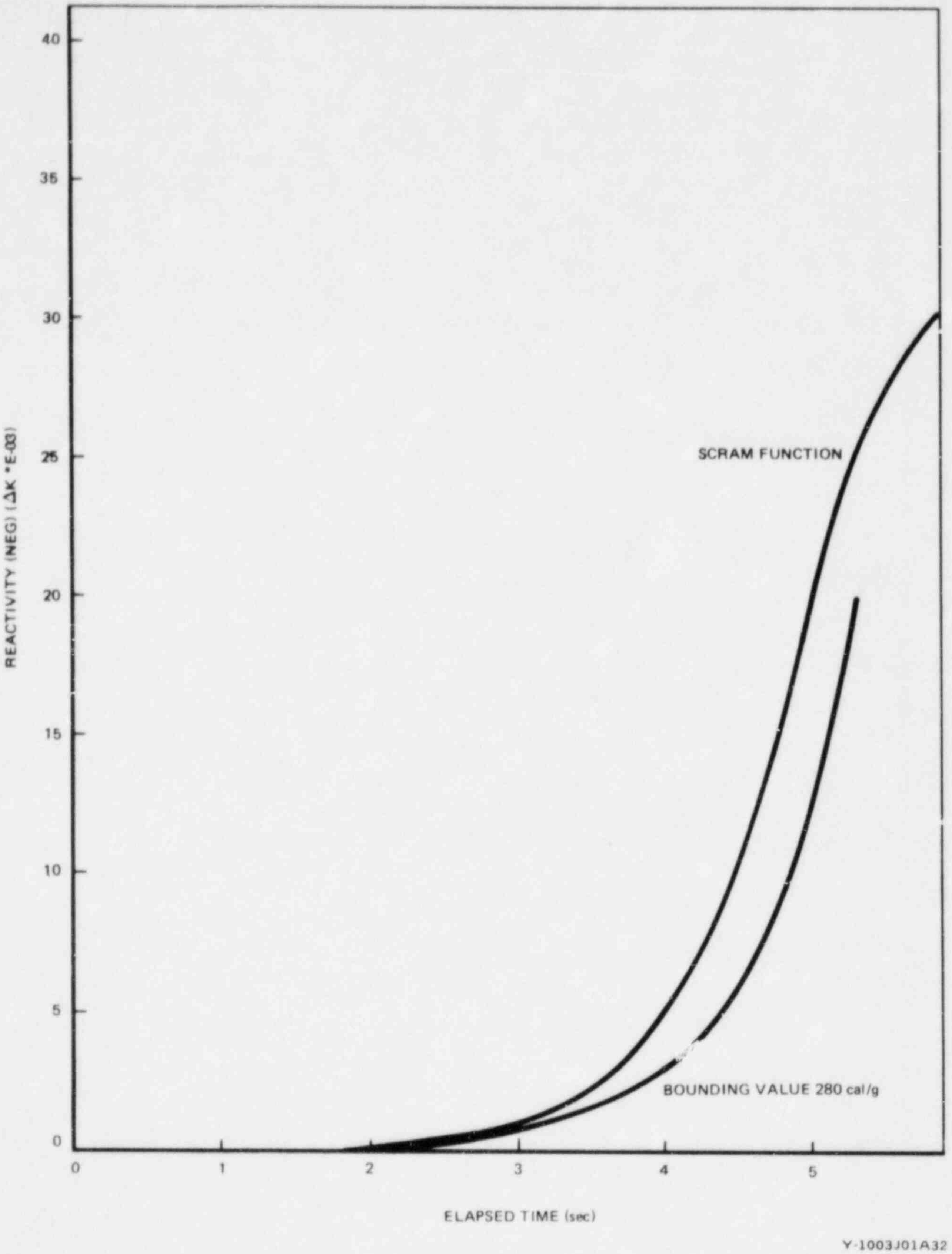

Figure 11. Scram Reactivity Function Cold Startup

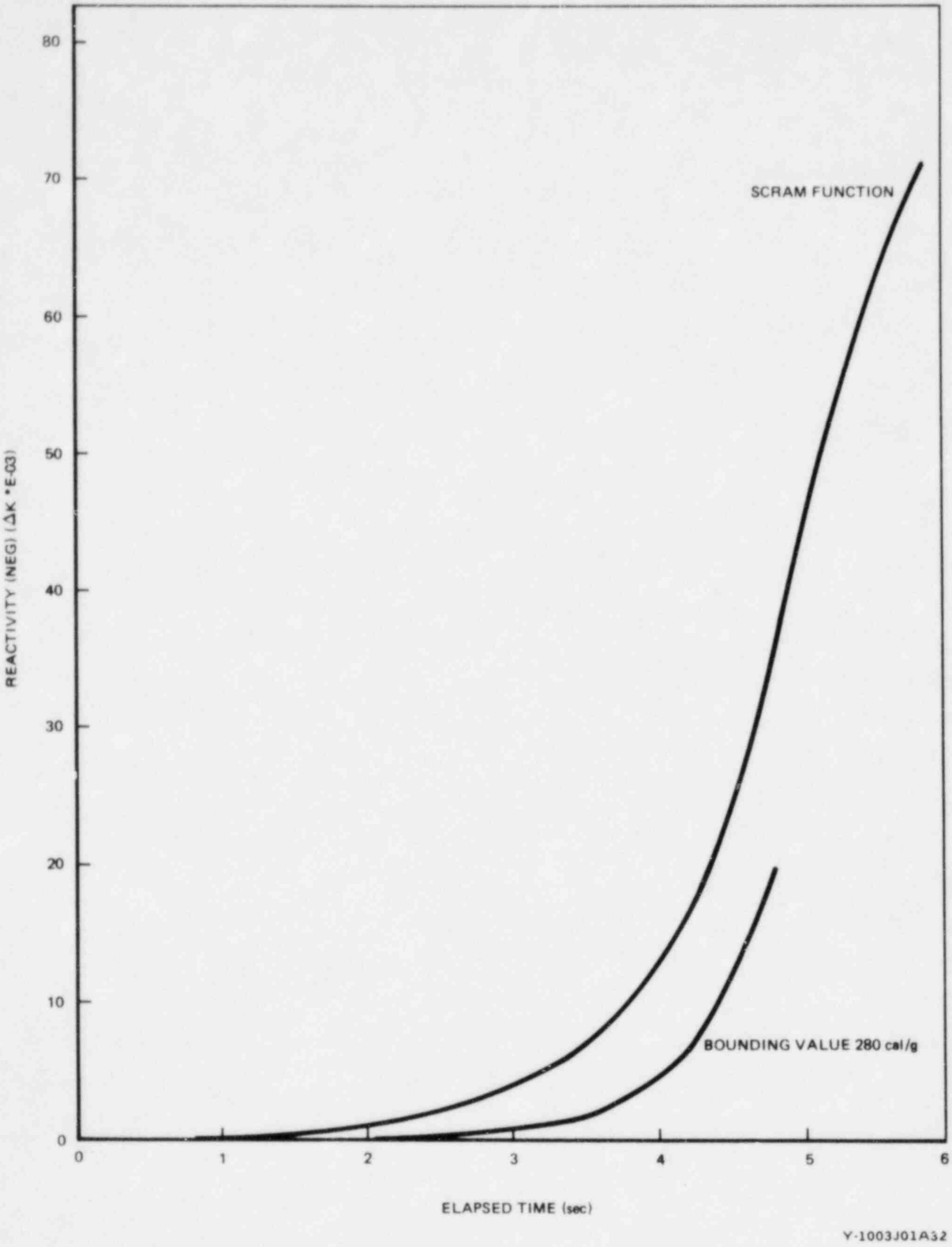

Figure 12. Scram Reactivity Function Hot Startup

APPENDIX A

Safety Relief Valve Capacity	91.4%

APPENDIX B
FEEDWATER TEMPERATURE REDUCTION (FWTR)

Additional analyses were performed assuming a reduction of 63°F in feedwater temperature in order to determine operating limits in the event of a loss of feedwater heating.

Table B-1 provides the results of the transient analysis performed, and Table B-2 provides the resulting Minimum Critical Power Ratio (MCPR) values. The feedwater controller failure transient is the limiting event with MCPR operating limits higher than the limits without Feedwater Temperature Reduction (FWTR).

Results of the stability analysis are presented in Table B-3 and in Figure B-3.

The analysis of the core response to a control rod drop accident showed all parameters to be within the bounding limits.

The ΔCPR of the rod withdrawal error is increased under FWTR conditions due to the higher initial CPR required by the pressurization transients with FWTR. However, in no case is RWE the limiting event.

As indicated in Section 5 of Reference B-1, the additional fatigue stresses are small compared to the allowable fatigue damage.

The loss-of-coolant accident analysis is relatively insensitive to feedwater temperature changes and is affected only slightly by a 63°F decrease. No change in MAPLHGR limits is required.

The operating limits for operation with FWTR at 100% power and 100% flow also bound operation within the region of the power/flow map described in Reference B-2.

REFERENCES

B-1. "Safety Review of Hatch Nuclear Power Station Unit No. 2 at Core Flow Conditions Above Rated Flow Throughout Cycle 2," General Electric Company, October 1981 (NEDO-24292, Revision 2).

B-2. "General Electric Boiling Water Reactor Load Line Limit Analysis for Edwin I. Hatch Nuclear Plant Unit 2," General Electric Company, October 1980 (NEDO-24295).

Table B-1

TRANSIENT ANALYSIS RESULTS WITH FWTR

Transient	Flux (% NBR)	Q/A (% NBR)	ΔCPR P8X8R	ΔCPR 8X8R	Figure
Exposure: BOC 3 to EOC 3					
Load Rejection without Bypass	483	123	0.21	0.18	B-1
Feedwater Controller Failure	363	129	0.24	0.22	B-2

Table B-2

MCPR VALUES WITH FWTR

Exposure: BOC 3 to EOC 3

	Option A		Option B	
	P8X8R	8X8R	P8X8R	8X8R
Load Rejection without Bypass	1.34	1.30	1.24	1.22
Feedwater Controller Failure	1.37	1.35	1.34	1.32

Table B-3

STABILITY ANALYSIS RESULTS WITH FWTR

Rod Line Analyzed: Extrapolated Rod Block Line

Decay Ratio:	Figure B-3
Reactor Core Stability Decay Ratio, X_2/X_o:	0.91
Channel Hydrodynamic Performance Decay Ratio, X_2/X_o	
Channel Type	
P8X8R/8X8R	0.63

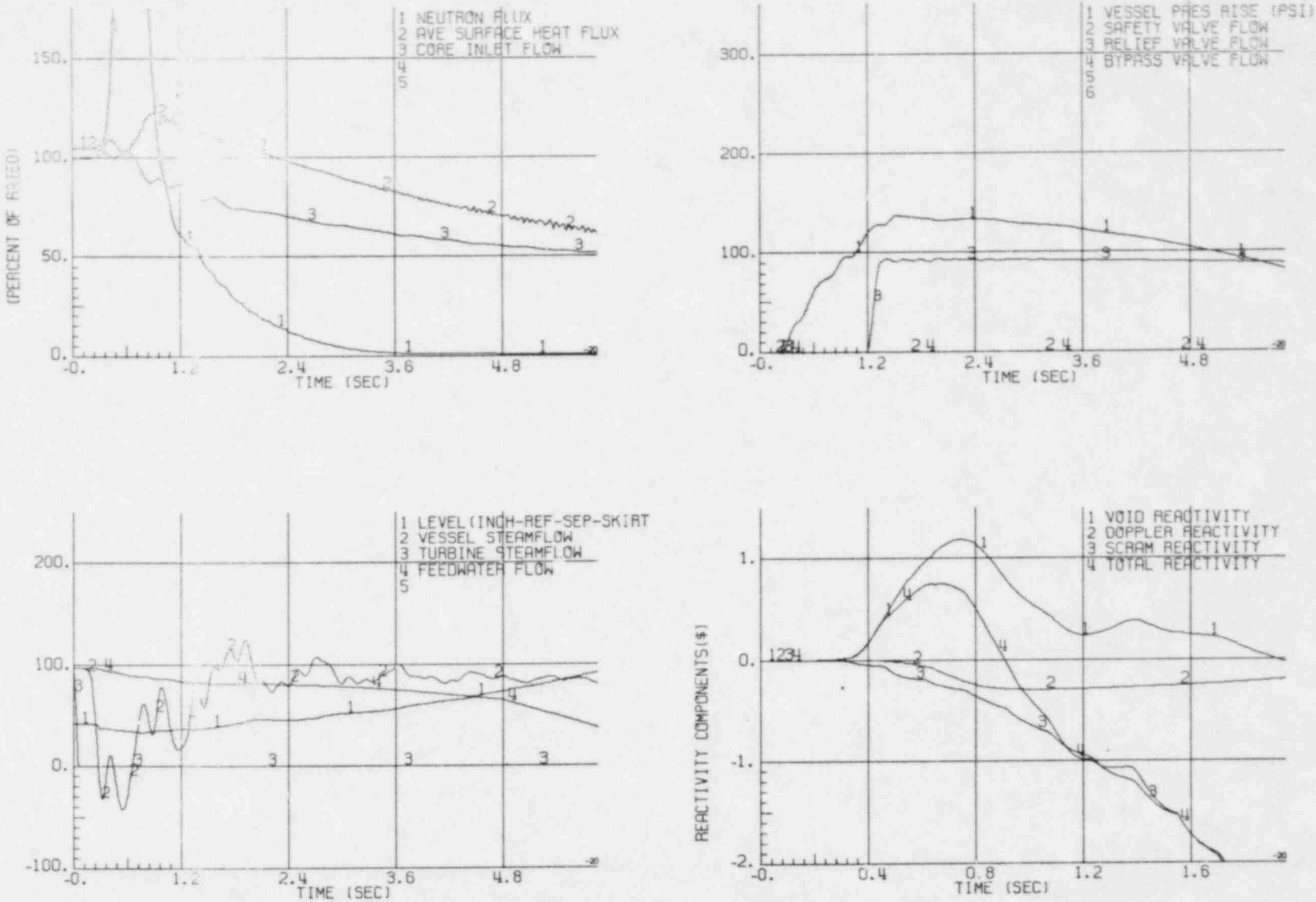

Figure B-1. Plant Response to Generator Load Rejection without Bypass with FWTR

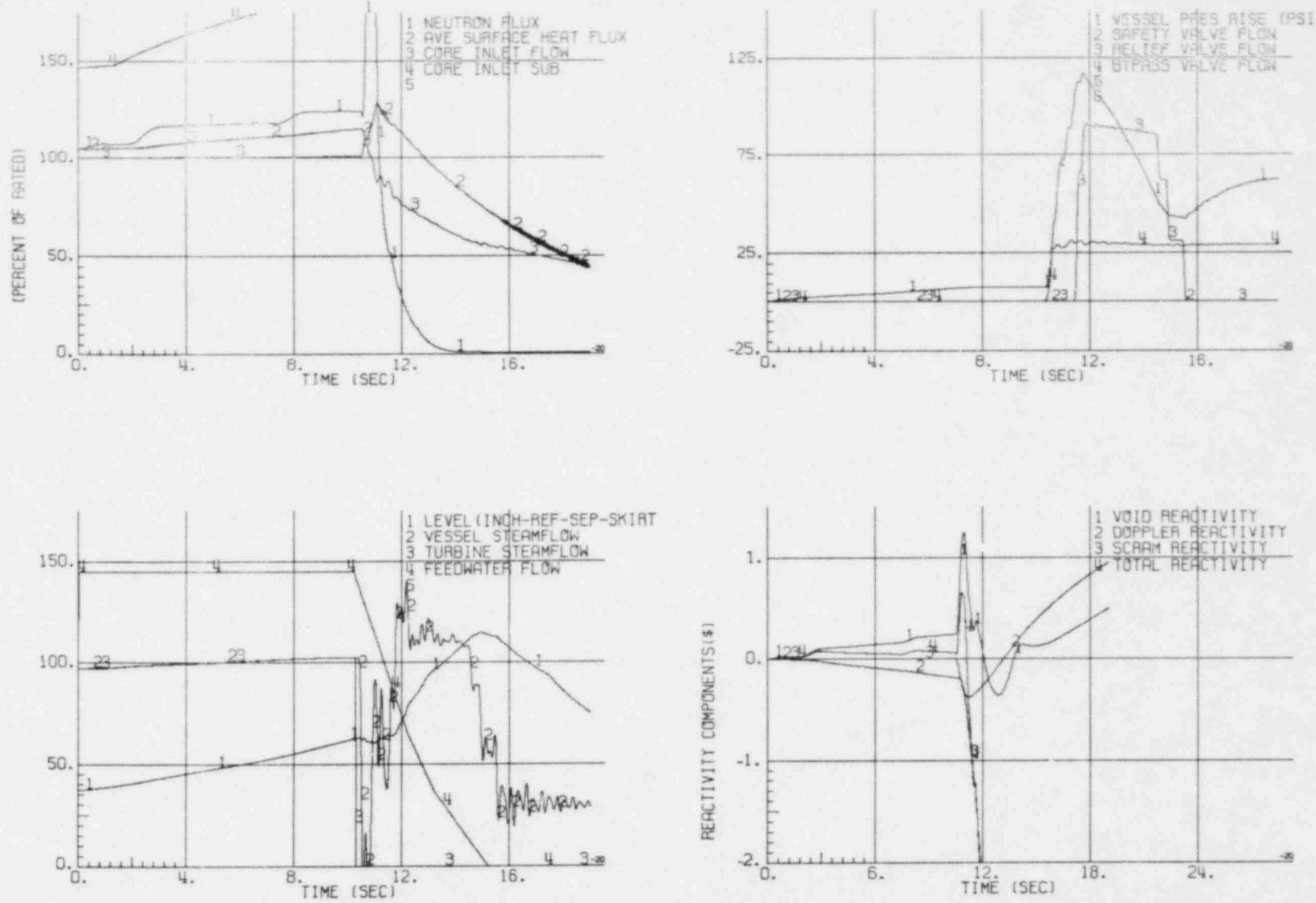

Figure B-2. Plant Response to Feedwater Controller Failure with FWTR

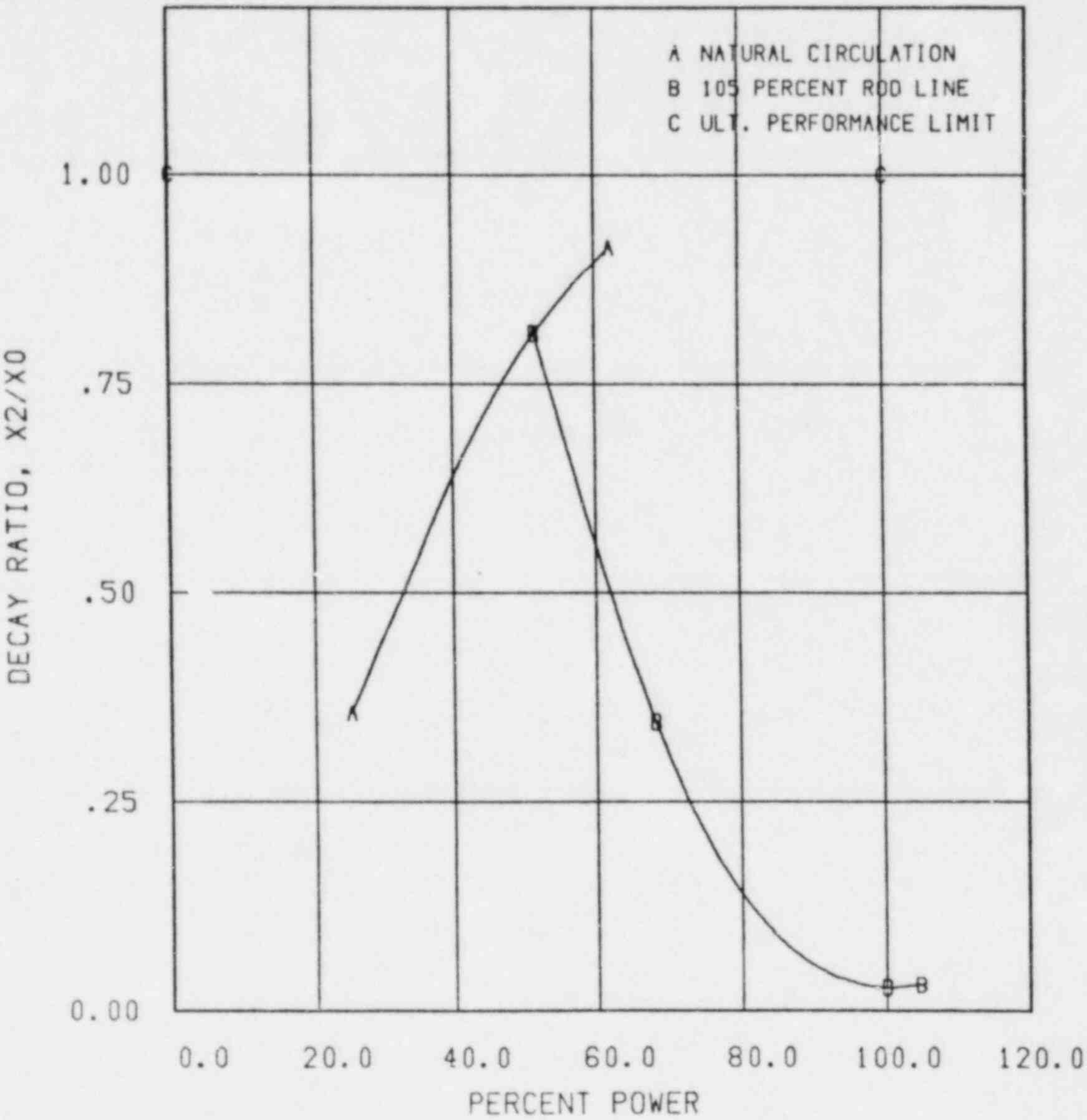

Figure B-3. Reactor Core Decay Ratio with FWTR

APPENDIX C

An error was discovered in the ODYN code used to analyze the pressurization transients, and the analyses were redone (C-1).

REFERENCE

C-1. Letter, H. C. Pfefferlen (GE) to D. G. Eisenhut (NRC), "Correction of ODYN Errors," June 8, 1982.

LIST OF FIGURES

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