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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR PILGRIM NUCLEAR POWER STATION UNIT 1, RELOAD 5

GENERAL ELECTRIC

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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
PILGRIM NUCLEAR POWER STATION
UNIT 1, RELOAD 5

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1. PLANT UNIQUE ITEMS (1.0)*

New Control Rod Withdrawal Error Analysis Procedure: Appendix A

2. RELOAD FUEL BUNDLES (1.0, 2.7, 3.3.1 AND 4.0)

	Fuel Designation	Cycle Loaded	Number	Number Drilled
Irradiated	8DB219H	4	68	68
	8DB219L	4	156	156
	P8DRB265L	5	120	120
	P8DRB282	5	64	64
New	P8DRB265H	6	60	60
	P8DRB282	6	112	112
Total			580	580

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	14.0 GWd/T
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	14.0 GWd/T
Assumed reload cycle core average exposure at end of cycle:	15.3 GWd/T
Core loading pattern:	Figure 1

*( ) refers to areas of discussion in "General Electric Boiling Water Reactor Generic Reload Fuel Application," NEDE-24011-P-A-1 and NEDO-24011-A, July 1979.

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

BOC k_{eff}	
Uncontrolled	1.113
Fully Controlled	0.952
Strongest Control Rod Out	0.985
R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, Δk	0.001

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
700	0.05

6. RELOAD UNIQUE TRANSIENT ANALYSIS INPUTS (3.3.2.1.5 AND 5.2)(1)

	EOC6
Void Coefficient N/A* (¢/% Rg)	-6.1/-7.6
Void Fraction (%)	36.9
Doppler Coefficient N/A (¢/°F)	-0.22/-0.21
Average Fuel Temperature (°F)	1205
Scram Worth N/A ($)(2)	
Scram Reactivity vs Time(2)	

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

(1) Applies to Loss of Feedwater Heating Event only.

(2) Generic, exposure independent values are used as given in "General Electric Boiling Water Reactor Generic Reload Fuel Application," NEDE-24011-P-A-1, Amendment 10, April 1981.

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

Fuel Design	Exposure (GWd/T)	Peaking Factors (Local, Radial, Axial)	R-Factor	Bundle Power (MWt)	Bundle Flow (10^3 lb/hr)	Initial MCPR
8x8	EOC6	1.22, 1.61 1.40	1.10	5.41	98	1.31
P8x8R	EOC6	1.20, 1.74 1.40	1.05	5.84	99	1.33

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Transient Recategorization: No

Recirculation Pump Trip: No

Rod Withdrawal Limiter: No

Thermal Power Monitor: No

Measured Scram Time: No

Exposure Dependent Limits: No

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

	Exposure Range (GWd/T)	$\hat{\phi}$ (% NBR)	Q/A (%)	ΔCPR 8x8	ΔCPR P8x8R	Figure
Load Rejection without Bypass	EOC6	589	119	0.24	0.27	3
Loss of 100°F Feedwater Heater	EOC6	117	115	0.14	0.15	4
Feedwater Controller Failure	EOC6	313	116	0.19	0.20	5

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

See Appendix A.

11. CYCLE MCPR VALUES (5.2)

Exposure Range (GWd/t)	Pressurization Events	Option A 8x8	Option A P8x8R	Option B 8x8	Option B P8x8R
BOC to EOC					
	Load Rejection w/o Bypass	0.30	0.33	0.25	0.28
	Feedwater Controller Failure	0.25	0.26	0.17	0.18

Nonpressurization Events	8x8	P8x8R
Loss of Feedwater Heating	0.14	0.15
Rotated Bundle Error	--	0.17
Rod Withdrawal Error	0.22	0.22

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1316	1330	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Rod Line Analyzed: Extrapolated Rod Block	Figure 8
Decay Ratio:	0.59
Reactor Core Stability Decay Ratio, x_2/x_0:	
Channel Hydrodynamic Performance Decay Ratio, x_2/x_0	
8x8 Channel:	0.22
P8x8R Channel:	0.18

14. ROTATED BUNDLE ERROR RESULTS (5.5.4)

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

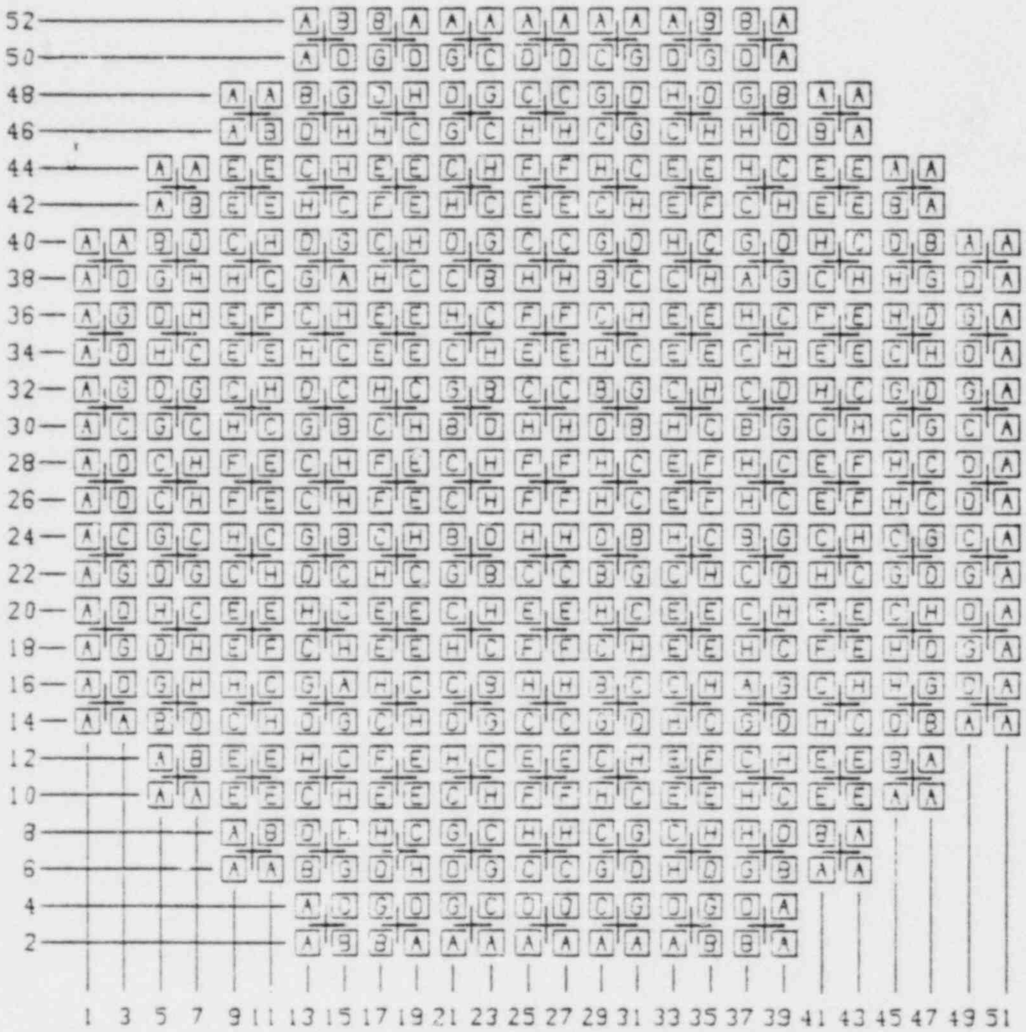
Initial MCPR	Resulting MCPR	Resulting LHGR (kW/ft)
1.22	1.07	17.67

15. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Maximum incremental control rod worth: 0.70% Δk

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

See "Loss-of-Coolant Accident Analysis Report for Pilgrim Nuclear Power Station," August 1977, NEDO-21696, as amended.



FUEL TYPE	
A = 8DB219L, C4	E = 8DB219L, C4
B = 8DB219H, C4	F = 8DB219H, C4
C = P8DRB265L, C5	G = P8DRB265H, C6
D = P8DRB282, C5	H = P8DRB282, C6

Figure 1. Reference Core Loading Pattern

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See Section 6

Figure 2. Scram Reactivity and Control Rod Drive Specifications

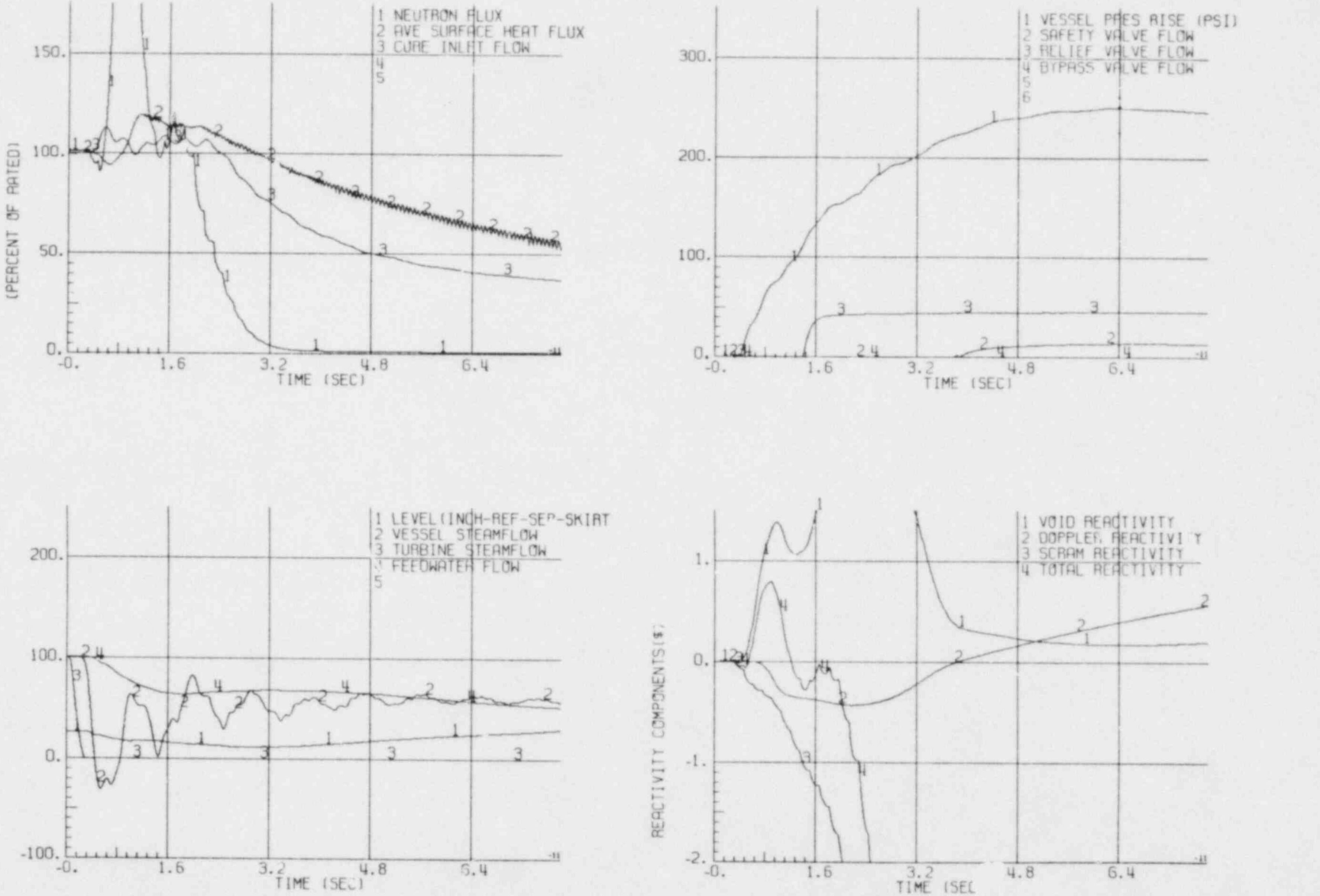

Figure 3. Plant Response to Generator Load Rejection Without Bypass

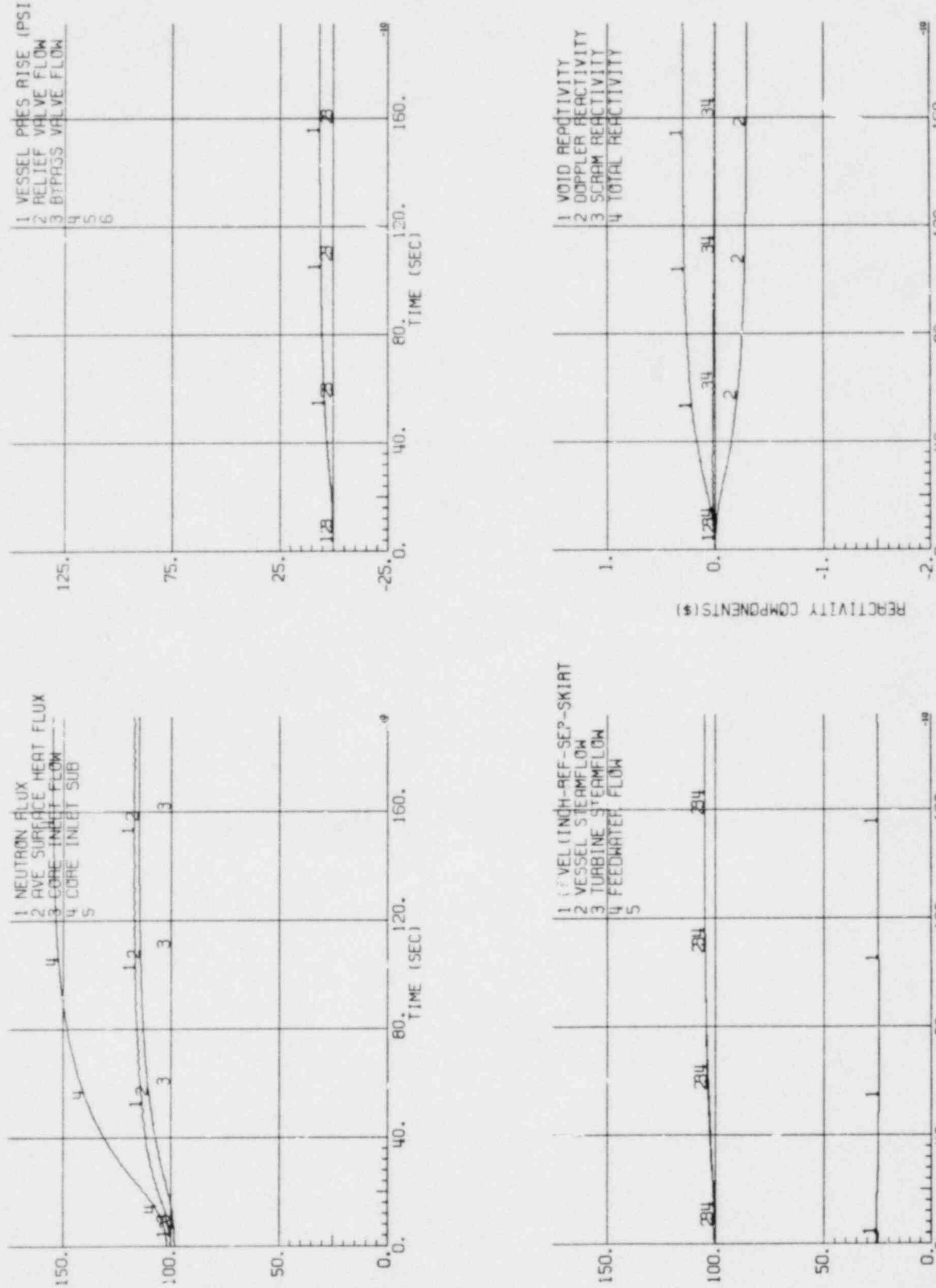

Figure 4. Plant Response to Loss of 100°F Feedwater Heating

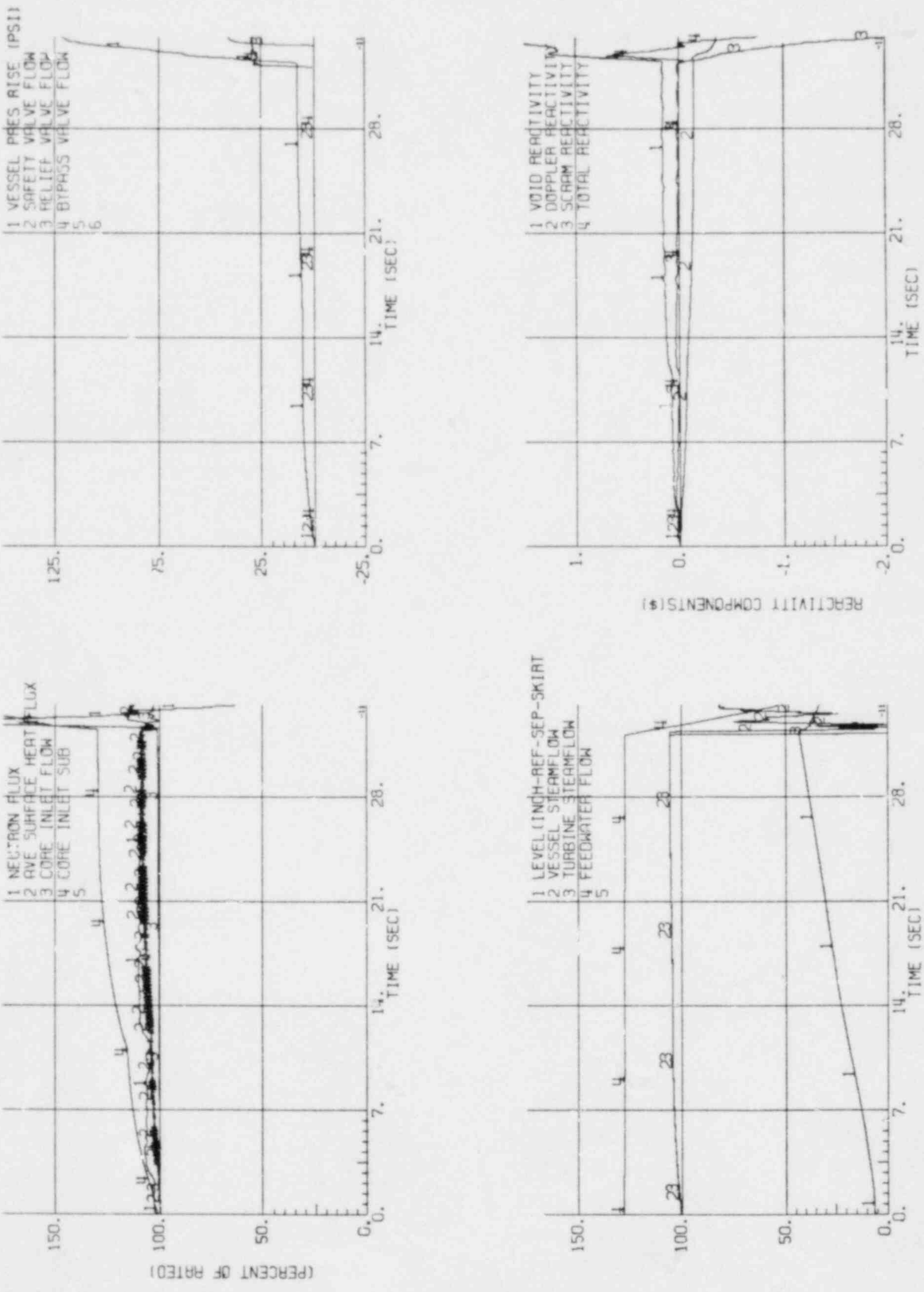

Figure 5. Plant Response to Feedwater Controller Failure

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See Appendix A

Figure 6. Limiting RWE Rod Pattern

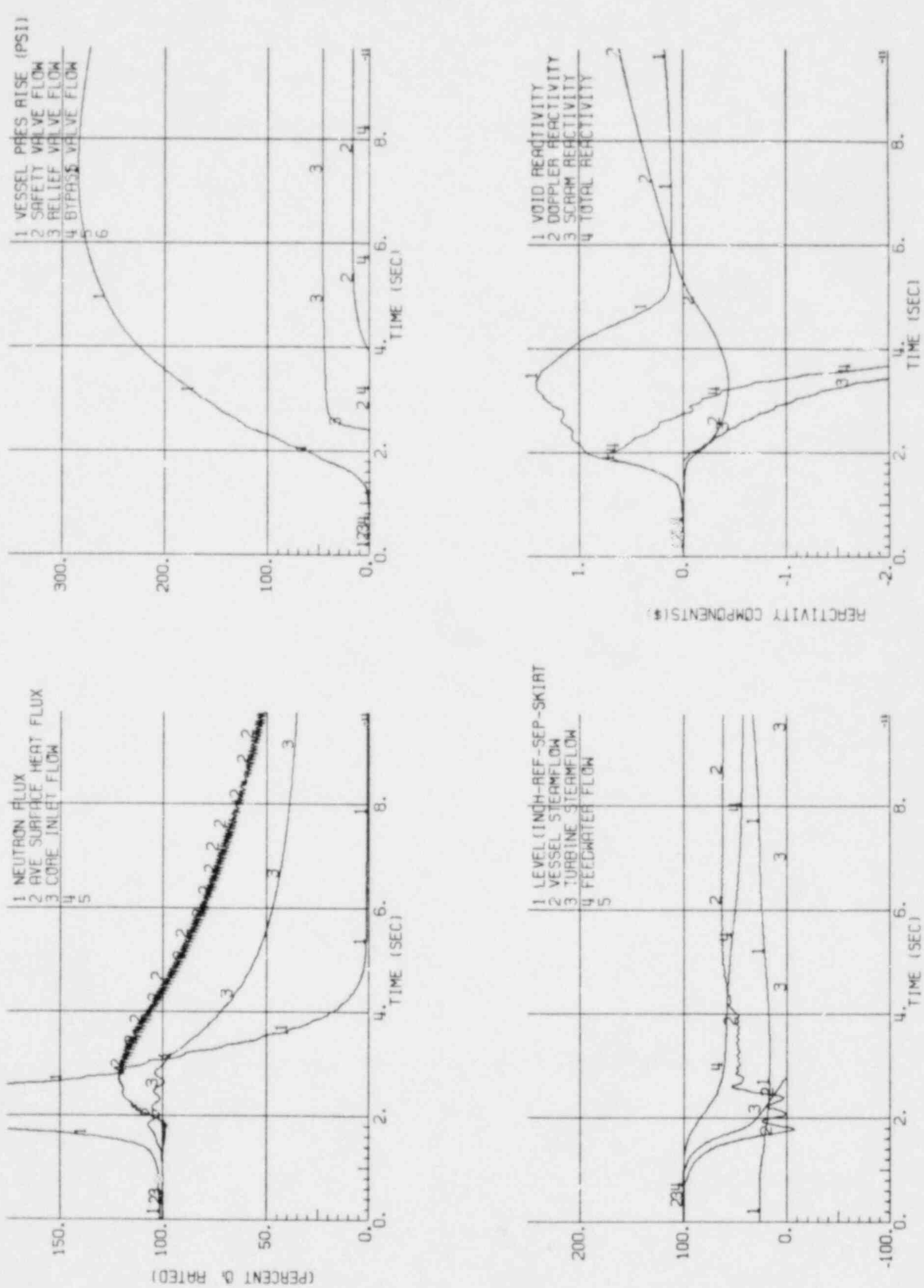

Figure 7. Plant Response to MSIV Closure (Flux Scram)

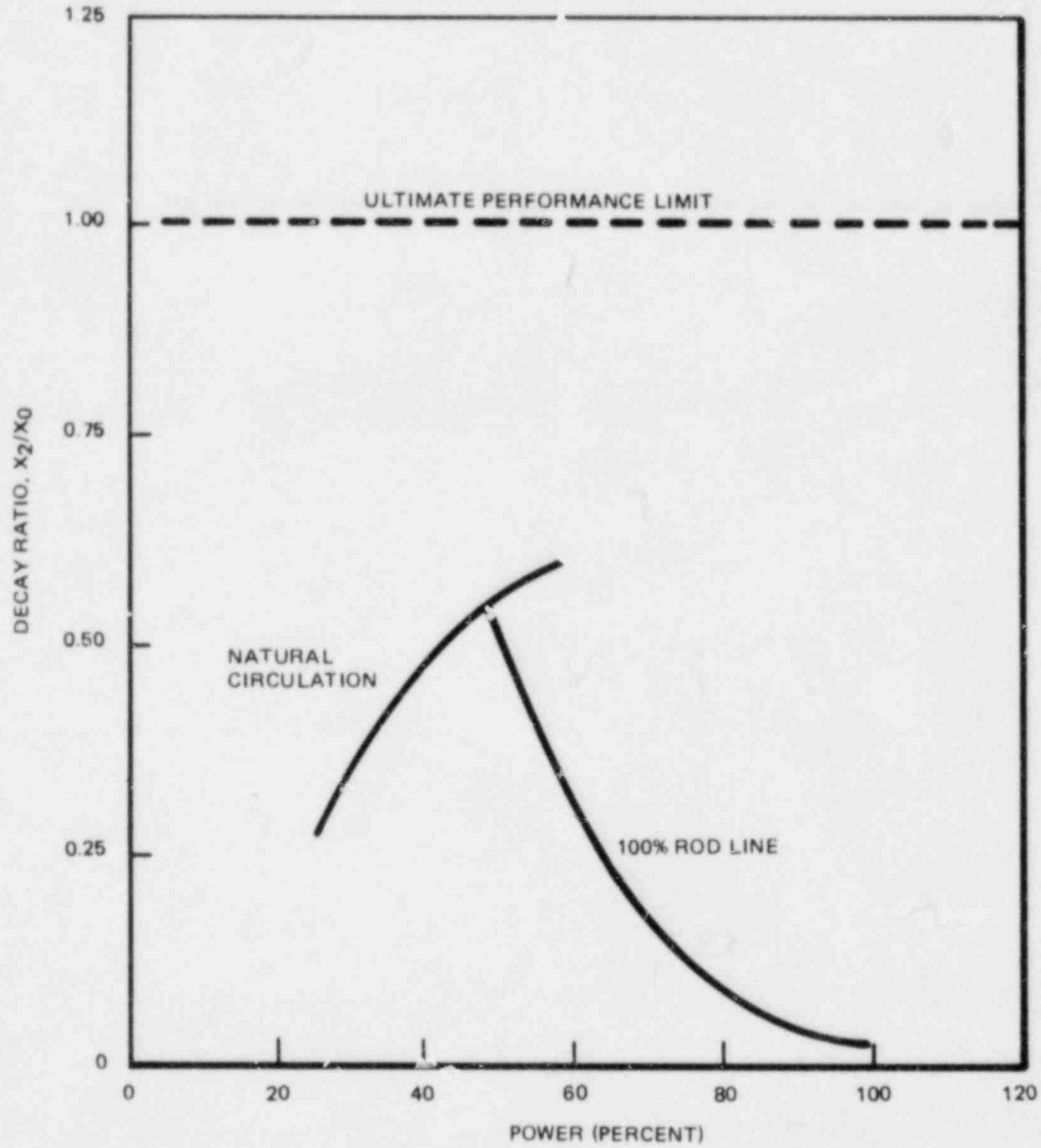

Figure 8. Reactor Core Decay Ratio versus Power

APPENDIX A

LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (NEW PROCEDURE)

The Local Rod Withdrawal Error results are reported below in accordance with Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "Change in General Electric Methods for Analysis of Control Rod Withdrawal Error," May 18, 1981.

Rod Block Reading*	ΔCPR 8x8/P8x8R
104	0.13
105	0.16
106	0.19
107*	0.22
108	0.28
109	0.32
110	0.36

*Indicates set point selected.

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