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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
DRESDEN NUCLEAR POWER STATION
UNIT 2 RELOAD 5

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GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

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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	1.096
Fully Controlled	0.940
Strongest Control Rod Out	0.973
R, Max Increase in Cold Core Reactivity with Exposure Into Cycle, Δk	0.008

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.052

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

	EOC 8
Void Coefficient N/A*(¢/% Rg)	-5.45/-6.82
Void Fraction (%)	34.48
Doppler Coefficient N/A (¢/°F)	-0.231/-0.219
Average Fuel Temperature (°F)	1211
Scram Worth N/A ($)	-37.94/-30.35
Scram Reactivity vs Time	Figure 2

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

	EOC 8			
Exposure	7x7	8x8	8x8R	P8x8R
Peaking Factors (local, radial and axial)	1.24 1.54 1.40	1.22 1.73 1.40	1.20 1.87 1.40	1.20 1.85 1.40
R-Factor	1.100	1.098	1.052	1.052
Bundle Power (MWt)	5.247	5.888	6.379	6.299
Bundle Flow (10^3 lb/hr)	111.5	107.2	105.7	106.2
Initial MCPR	1.23	1.25	1.25	1.27

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

1. PLANT-UNIQUE ITEMS (1.0)*

Items different from or not included in Reference 1.

a.	Plant Parameter Changes	Appendix A
b.	Loading Error	Appendix A
c.	Densification Power Spiking	Appendix B
d.	Pressure Relief Setpoints	Reference 3
e.	GETAB Initial Conditions - Inlet Enthalpy	Reference 4
f.	ATWS RPT Setpoint of 1250 psig	Items 9 and 12

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

		Fuel Type	Number	Number Drilled
Irradiated	Reload 1	7DB230	4	0
	Reload 1	8DB250	48	0
	Reload 2	8DB250	28	0
	Reload 2	8DB262	68	0
	Reload 3	8DB250	192	0
	Reload 4	8DRB265L	160	160
New	Reload 5	P8DRB265L	88	88
	Reload 5	P8DRB265H	136	136
Total			724	384

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 18,751 MWd/t

Assumed reload cycle exposure: 17,481 MWd/t

Core loading pattern: Figure 1.

*( ) refers to areas of discussion in Reference 1.

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

None

9. CORE WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Flow (%)	ϕ (%)	Q/A (%)	$P_{s\ell}$ (psig)	P_v (psig)	ΔCPR 7x7	ΔCPR 8x8	ΔCPR 8x8R	ΔCPR P8x8R	Plant Response
Load Rejection without Bypass	EOC 8	100	100	233	111	1209	1244	0.13	0.18	0.18	0.20	Figure 3
Loss of 145°F Feedwater Heating	--	100	100	121	119	994	1045	0.16	0.17	0.17	0.17	Figure 4
Feedwater Controller Failure	EOC 8	100	100	138	108	1112	1148	0.06	0.09	0.09	0.09	Figure 5

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

Rod Block Reading (%)	Rod Position (Feet Withdrawn)	ΔCPR 8x8*	ΔCPR 8x8R/P8x8R	LHGR*** 8x8	LHGR*** 8x8R/P8x8R	Limiting Rod Pattern
104	4.0	0.12	0.12	11.32	13.70	Figure 6
105	4.5	0.14	0.14	11.49	13.70	Figure 6
106	5.0	0.16	0.15	11.49	13.77	Figure 6
107**	5.5	0.18	0.17	11.49	13.82	Figure 6
108	6.0	0.19	0.19	11.49	13.88	Figure 6
109	7.5	0.23	0.23	11.49	13.88	Figure 6
110	9.0	0.26	0.26	11.60	16.06	Figure 6

* 7x7 bundles are all located in peripheral regions.
**Indicates setpoint selected.
***Includes 2.2% penalty due to fuel densification.

11. OPERATING MCPR LIMIT (5.2)

BOC8 to EOC8

1.27	(P8x8R fuel)
1.25	(8x8R fuel)
1.25	(8x8 fuel)
1.23	(7x7 fuel)

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	Power (%)	Core Flow (%)	$P_{s\ell}$ (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	100	100	1293	1310	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability: Decay Ratio, x_2/x_0 (100% Rod Line - Natural Circulation Power) 0.52

Channel Hydrodynamic Performance

	Decay Ratio, x_2/x_0 (100% Rod Line - Natural Circulation Power)
P8x8R channel	0.17
8x8R channel	0.17
8x8 channel	0.24
7x7 channel	0.12

14. LOSS-OF-COOLANT ACCIDENT RESULTS (5.5.2)

Reference 2

15. LOADING ERROR RESULTS (5.5.4)

Limiting Event: Rotated Bundle (8x8R)

MCPR: 1.08

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Maximum incremental control rod worth: 0.465% ΔK

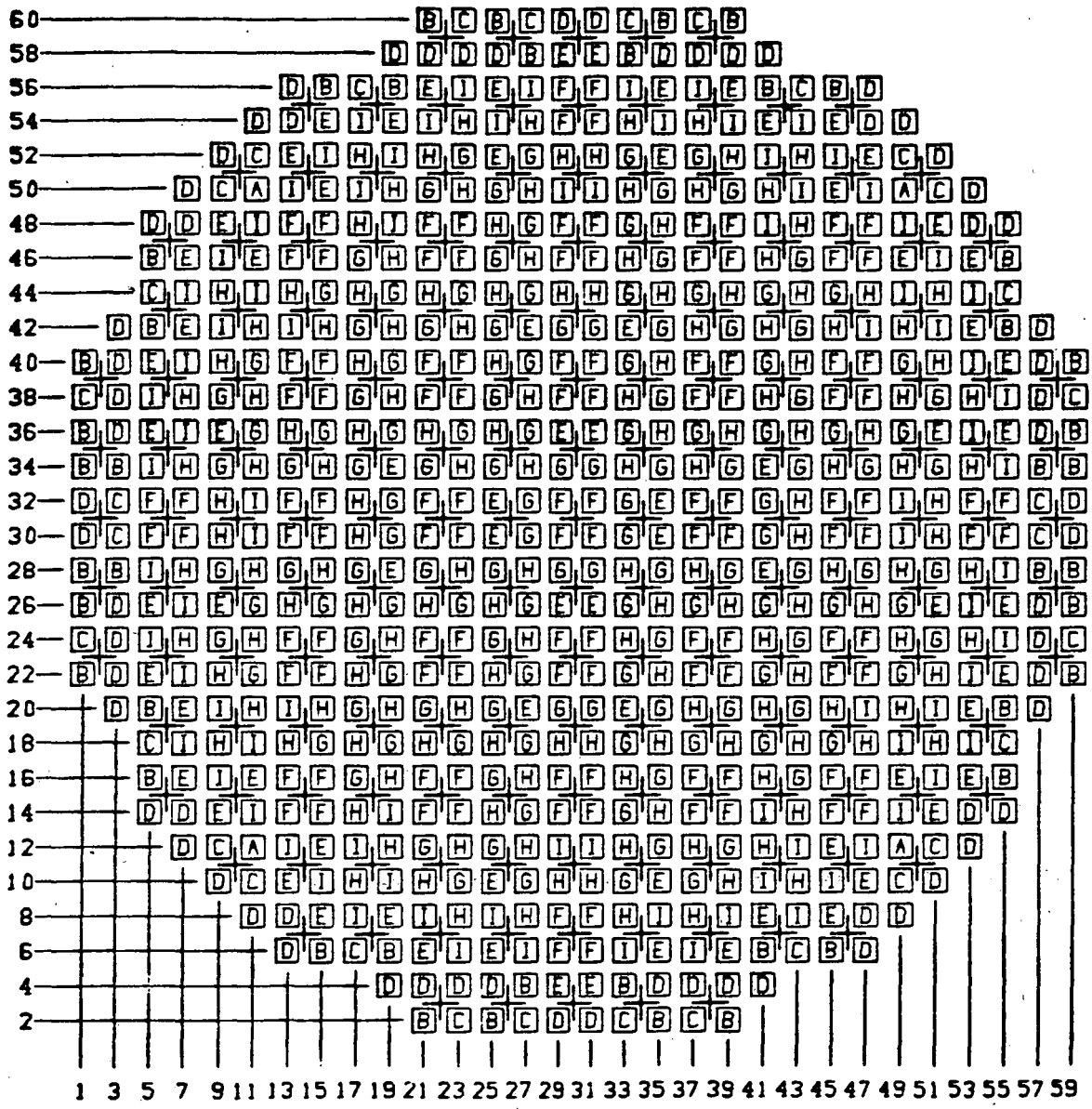
REFERENCES

1. "General Electric Boiling Water Reactor Generic Reload Fuel Application," July 1979 (NEDE-24011-P-1).

2. "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, April 1979.

3. Attachment to letter No. MFN-048-80, R. E. Engel to T. A. Ippolito, "Categorization of Requested Reload Fuel Licensing Topical Report Revisions," February 27, 1980.

4. Attachment to letter, J. F. Quirk to Olan D. Parr, "General Electric Licensing Topical Report NEDE-24011-P-A-1, Generic Reload Fuel Application, Appendix D, Second Submittal," February 28, 1979.



FUEL TYPE					
A =	7DB230-GEN/B	BOC4	F =	8DB250	BOC6
B =	8DB250	BOC4	G =	P8DRB265H	BOC8
C =	8DB250	BOC5	H =	8DRB265L	BOC7
D =	8DB262	BOC5	I =	P8DRB265L	BOC8
E =	8DB250	BOC6			

Figure 1. Reference Core Loading Pattern

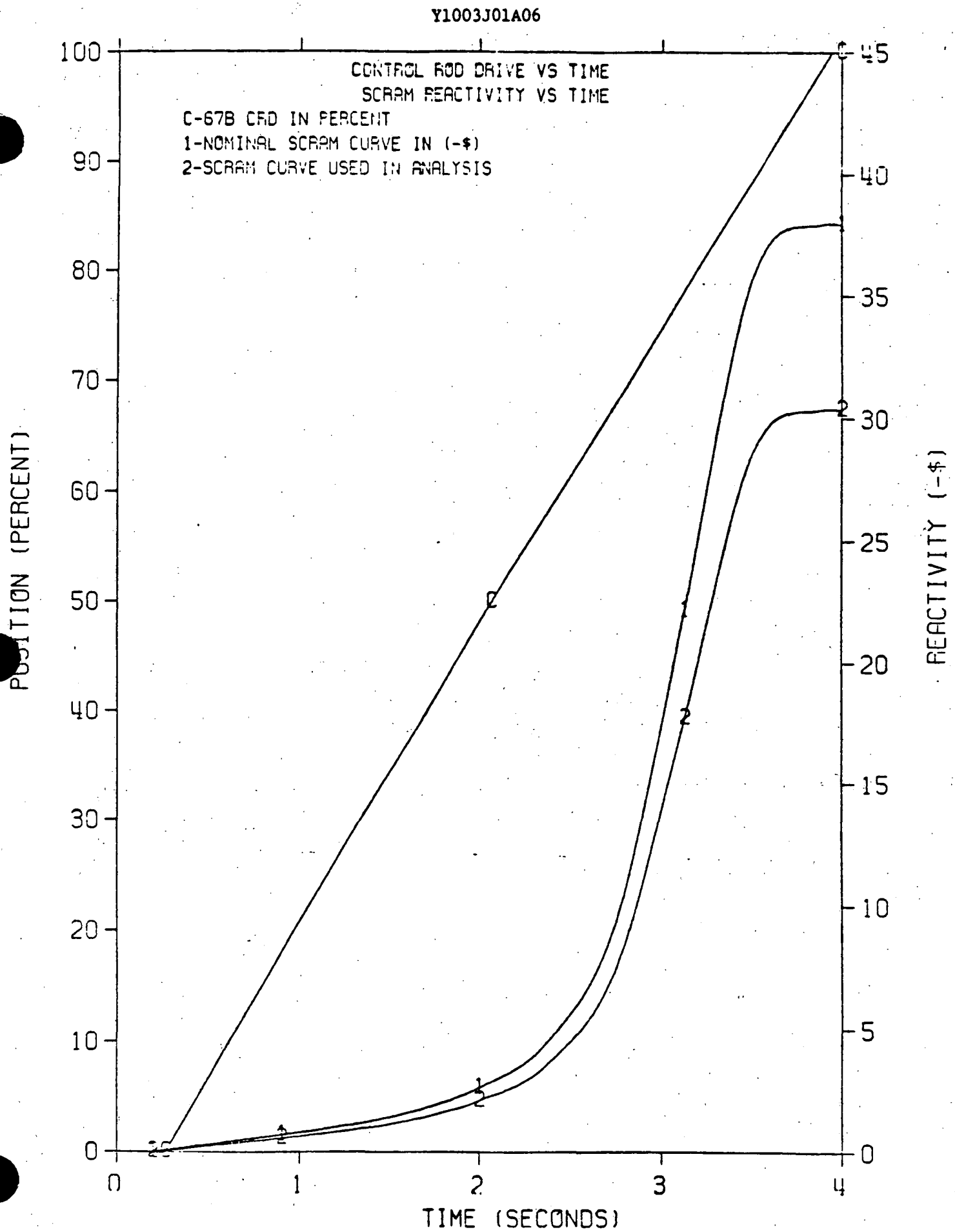


Figure 2. Scram Reactivity and Control Rod Drive Specification

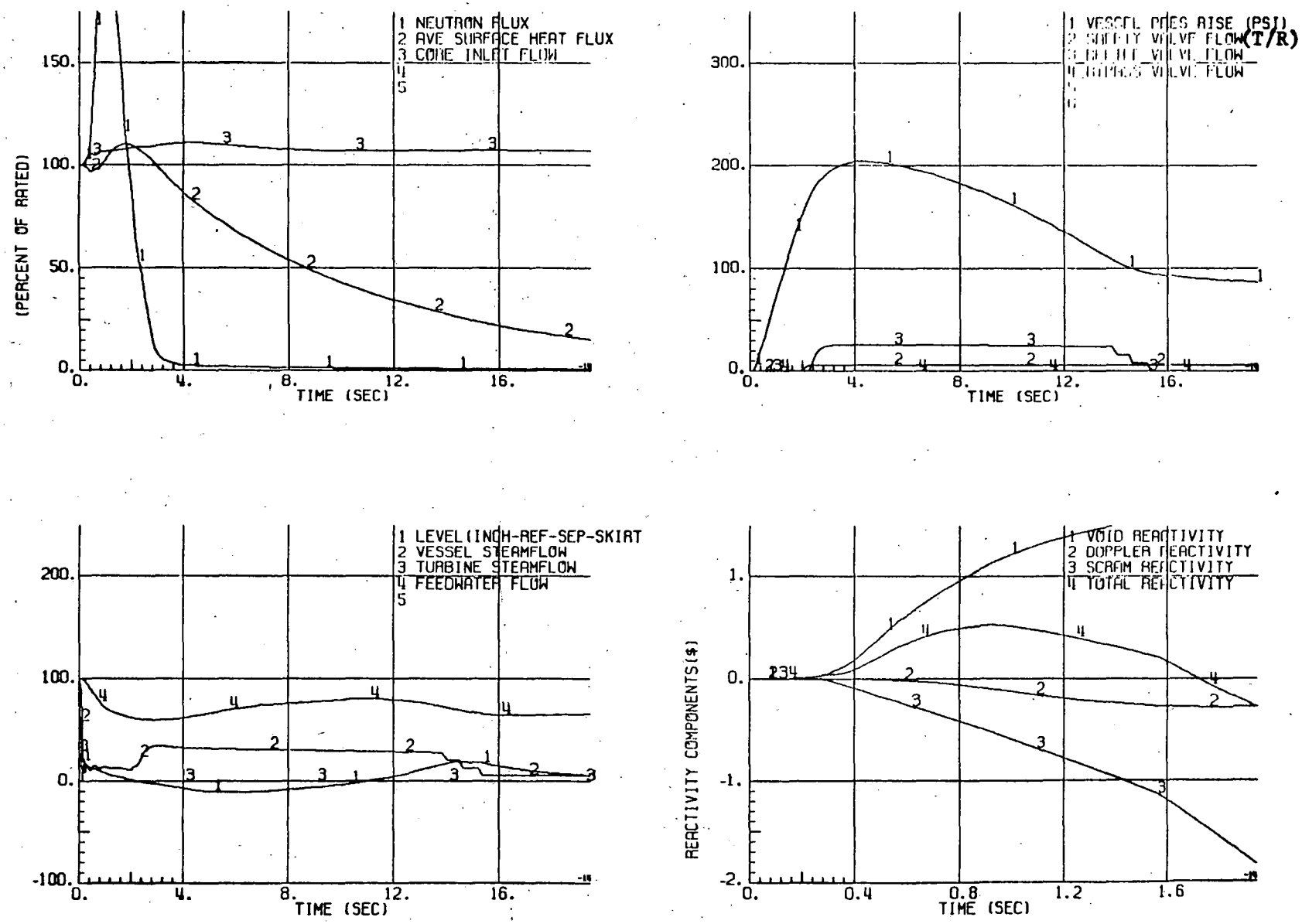


Figure 3. Plant Response to Load Rejection without Bypass

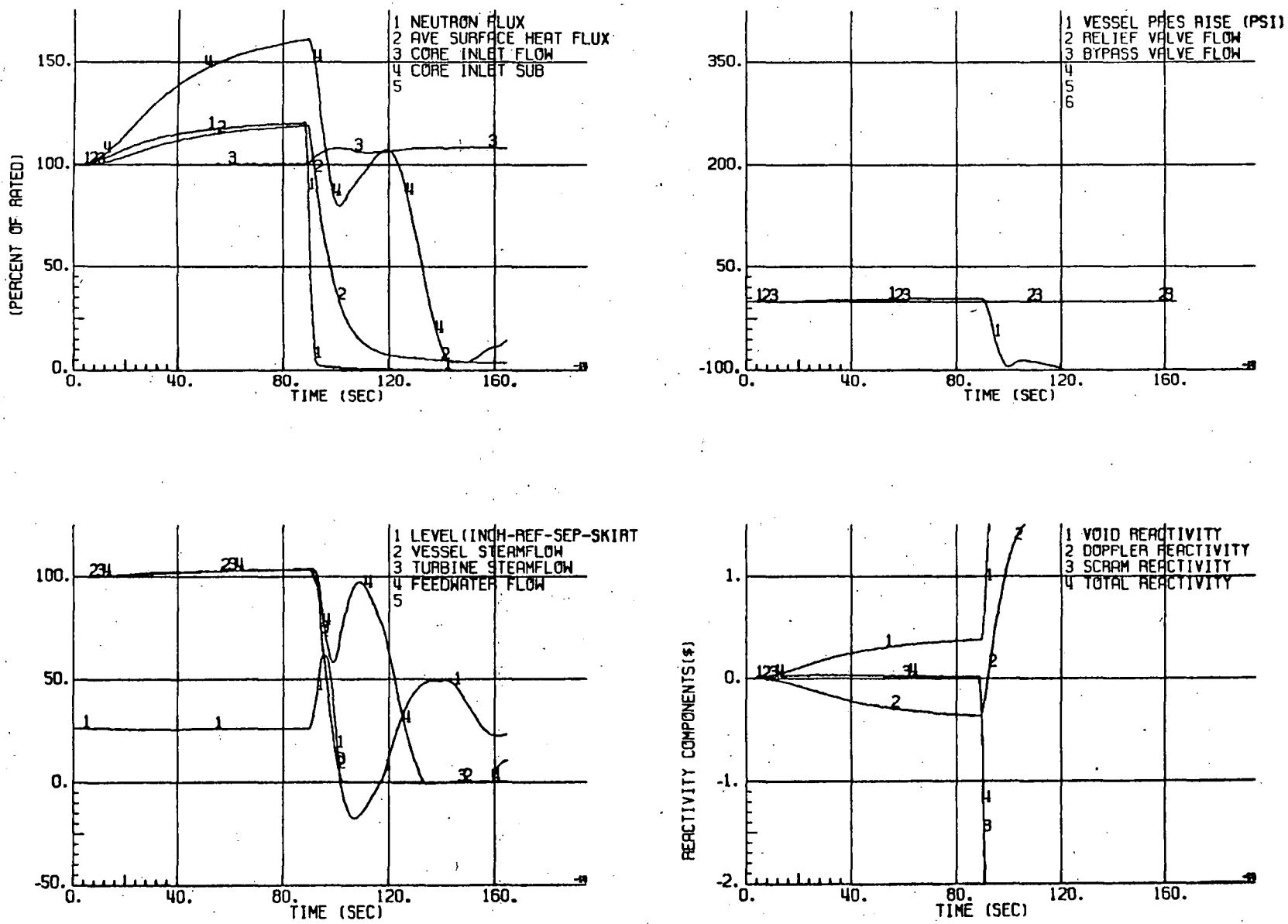


Figure 4. Plant Response to Loss of 145°F Feedwater Heating

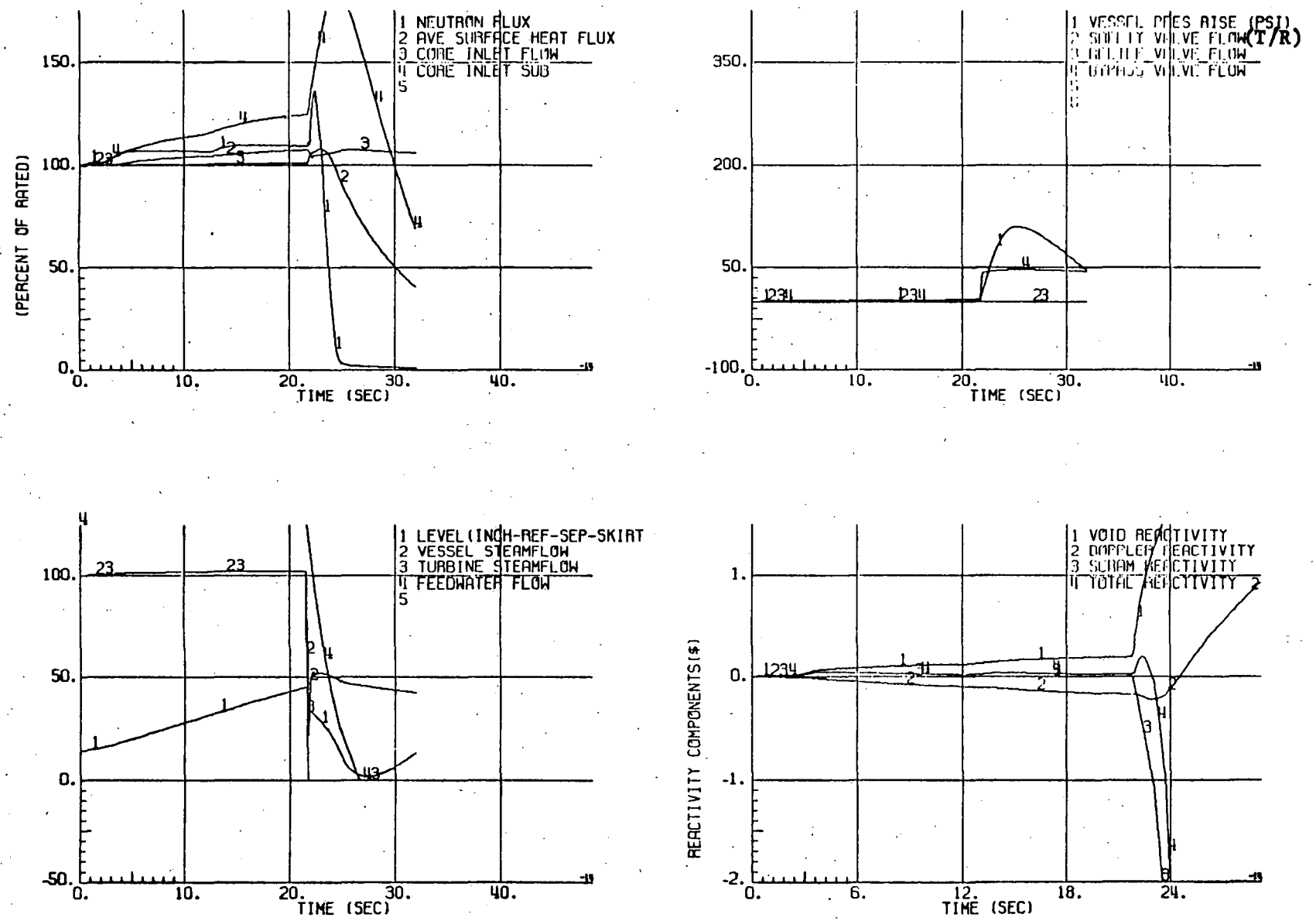


Figure 5. Plant Response to Feedwater Controller Failure

59							4		4						
55				36		38		36		38		36			
51			2		6		8		8		6		2		
47				36		44		44		44		36			
43			4		8		0		0		8		4		
39		36		38		46		46		46		38		36	
35	6		2		8		16		16		8		2		6
31		36		36		40		40		40		36		36	
27	6		2		8		16		16		8		2		6
23		36		38		46		46		46		38		36	
19			4		8		0		0		8		4		
15				36		44		44		44		36			
11			2		6		8		8		6		2		
7				36		38		36		38		36			
3							4		4						
	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58

NOTES:

1. No. indicates number of notches withdrawn out of 48. Blank is a withdrawn rod.
2. Error rod is (26, 43).

Figure 6. Limiting RWE Rod Pattern

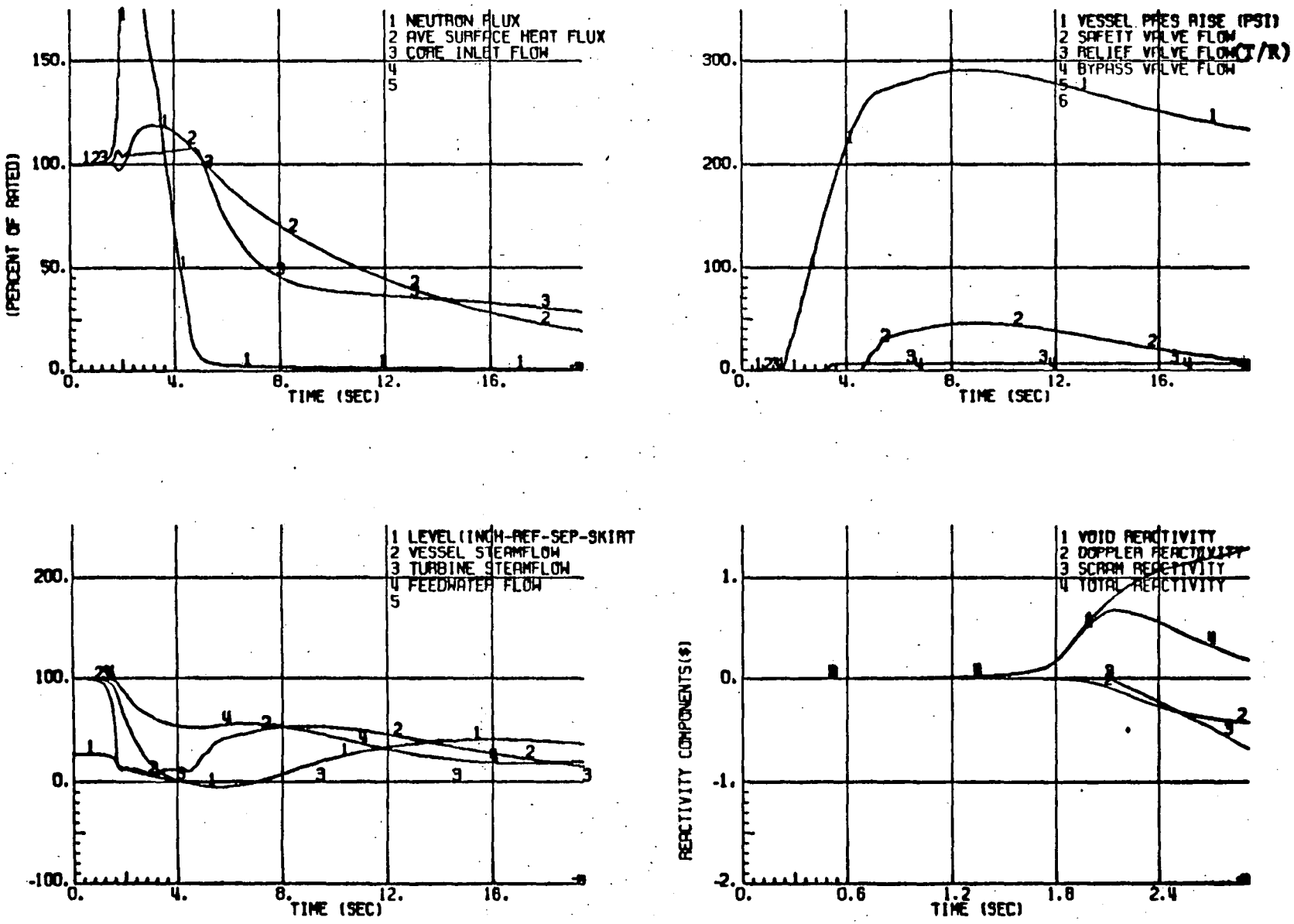


Figure 7. Plant Response to MSIV Closure

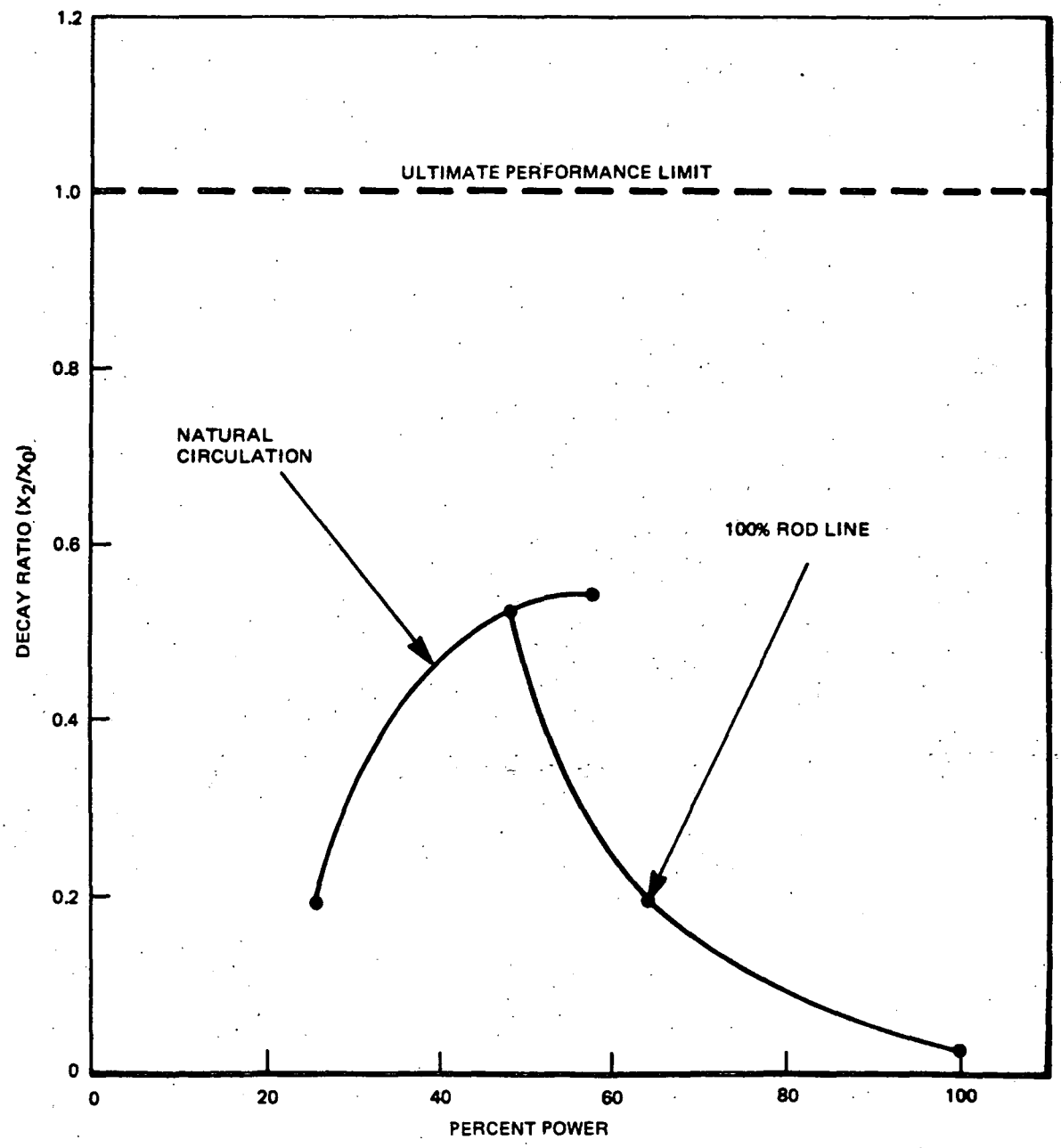


Figure 8. Decay Ratio

APPENDIX A

PLANT PARAMETER CHANGES

Pressure Relief Systems (Table 5-4, pg 5-62, NEDO-24011)	
Safety/Relief Valve setpoint (psig)	1115 + 1%
Safety/Relief Valve capacity (% rated steam flow)	5/31.03
Safety Valve Capacity (% rated steam flow)	8/51.56
Transient Operating Parameters (Table 5-6, pg 5-64, NEDO-24011)	
Turbine Pressure (psig)	950
GETAB Initial Conditions	
Reactor Core Pressure (psia)	1030
Stability Analysis	
Natural Circulation/Extrapolated Rod Block Decay Ratio	0.54
Loading Error	
Linear Heat Generation Rate (kW/ft):*	
Rotated Bundle	16.84
Mislocated Bundle	16.85

*Includes 2.2% penalty due to fuel densification.

APPENDIX B

DENSIFICATION POWER SPIKING

Reference B-1 documents the NRC staff position that ". . . it (is) acceptable to remove the 8x8 and 8x8R spiking penalty factor from the plant Technical Specification for those operating BWR's for which it can be shown that the predicted worst case maximum transient LHGR's, when augmented by the power spike penalty, do not violate the exposure-dependent safety limit LHGR's".

The Dresden 2 Reload 5 submittal contains the required information to demonstrate that the stated criterion is met for Dresden 2 Reload 5, Section 10 (Rod Withdrawal Error) and Appendix A (Linear Heat Generation Rate for Bundle Loading Error) include the densification effect in the calculated LHGR of the 8x8 fuels.

REFERENCE

B-1 "Safety Evaluation of the General Electric Methods for the Consideration of Power Spiking Due to Densification Effects in BWR 8x8 Fuel Design and Performance," Reactor Safety Branch, DOR, May 1978.