NEDO-24224
79NED408
CLASS I
NOVEMBER 1979

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR PILGRIM NUCLEAR POWER STATION UNIT 1 RELOAD 4

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GENERAL ELECTRIC

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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
PILGRIM NUCLEAR POWER STATION
UNIT 1 RELOAD 4

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NUCLEAR POWER SYSTEMS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

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IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

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1. PLANT-UNIQUE ITEMS (1.0)*

Margin to Opening of Unpiped Spring Safety Valves: Appendix A

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New Bundle Loading Error Analyses Procedures: Appendix D

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

	Fuel Type	Number	Number Drilled
Irradiated	8DB262	60	0
	8DB219H	124	124
	8DB219L	212	212
New	P8DRB265L	120	120
	P8DRB282	64	64
Total		580	520

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 11,700 MWd/t

Assumed reload cycle exposure: 13,910 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)

BOC k_{eff}

Uncontrolled	1.089
Fully Controlled	0.929
Strongest Control Rod Out	0.967

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R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, Δk 0.011

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

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
700	0.0682

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

	EOC5
Void Coefficient N/A* (¢/% Rg)	-6.08/-7.60
Void Fraction (%)	37.1
Doppler Coefficient N/A (¢/% °F)	-0.226/-0.217
Average Fuel Temperature (°F)	1197
Scram Worth N/A ($)	-38.53/-30.82
Scram Reactivity	Figure 2

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

	8x8 EOC5	P8x8R EOC5
Peaking Factors (local, radial and axial)	1.22/1.73/1.40	1.20/1.87/1.40
R-Factor	1.098	1.052
Bundle Power (MWt)	5.832	6.298
Bundle Flow (10^3 lb/hr)	97.0	97.37
Initial MCPR	1.21	1.22

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

New Bundle Loading Error Analyses Procedures

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

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9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Flow (%)	Φ (%)	Q/A (%)	P_{sl} (psig)	P_v (psig)	ΔCPR 8x8	ΔCPR P8x8R	Plant Response
Generator Load Rejection w/o Bypass	BOC-EOC	100	100	218	107	1259	1272	0.15	0.16	Figure 3
Loss of Feedwater Heating	BOC-EOC	100	100	118	117	<1100	<1100	0.15	0.15	Figure 4
Feedwater Controller Failure	BOC-EOC	100	100	138	109	1138	1171	0.12	0.12	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 P8x8R	MLHGR (kW/ft) 8x8	MLHGR (kW/ft) P8x8R	Limiting Rod Pattern
104	4.0	0.12	0.12	11.6	13.7	Figure 6
105	4.5	0.14	0.14	11.9	13.7	Figure 6
106	5.0	0.15	0.17	12.6	14.1	Figure 6
107*	5.5	0.17	0.19	13.4	14.8	Figure 6
108	6.0	0.19	0.21	13.8	15.2	Figure 6
109	6.5	0.20	0.23	13.8	15.4	Figure 6

11. OPERATING MCPR LIMIT (5.2)

BOC5 - EOC5	
1.29	(8x8 fuel)
1.29	(P8x8R fuel)

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	Power (%)	Core Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	100	100	1328	1341	Figure 7

*Indicates set point selected

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13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio, x_2/x_0 (Extrapolated Rod Block Line - Natural Circulation Power)	0.61

Channel Hydrodynamic Performance

	Decay Ratio, x_2/x_0 (Extrapolated Rod Block Line - Natural Circulation Power)
8x8/8x8R channel	0.22

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

8DB262

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Location Oxidation Fraction
200	11.1	2032	0.016
1,000	11.3	2028	0.015
5,000	11.9	2071	0.017
10,000	12.1	2061	0.016
15,000	12.2	2091	0.018
20,000	12.1	2104	0.019
25,000	11.6	2049	0.016
30,000	10.7	1928	0.010

8DB219H

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Location Oxidation Fraction
200	11.2	2038	0.018
1,000	11.3	2032	0.017
5,000	11.8	2056	0.017
10,000	12.2	2102	0.019
15,000	12.3	2131	0.021
20,000	12.1	2128	0.021
25,000	11.3	2015	0.015
30,000	10.2	1866	0.008

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

8DB219L

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Location Oxidation Fraction
200	11.4	2039	0.018
1,000	11.5	2039	0.017
5,000	11.9	2064	0.017
10,000	12.1	2098	0.019
15,000	12.3	2126	0.021
20,000	12.1	2126	0.021
25,000	11.3	2013	0.014
30,000	10.2	1866	0.008

P8DRB265L

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Location Oxidation Fraction
200	11.6	2125	0.023
1,000	11.6	2127	0.023
5,000	12.1	2136	0.022
10,000	12.1	2102	0.020
15,000	12.1	2108	0.020
20,000	11.9	2091	0.019
25,000	11.3	2012	0.015
30,000	10.7	1919	0.010

P8DRB282

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Location Oxidation Fraction
200	11.2	2087	0.020
1,000	11.2	2083	0.020
5,000	11.8	2110	0.021
10,000	12.0	2097	0.020
15,000	12.1	2108	0.020
20,000	11.8	2088	0.019
25,000	11.3	2011	0.015
30,000	11.1	1961	0.012

*A MAPLHGR multiplier of 0.95 is required for operation at flow less than 90% of rated.

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15. LOADING ERROR RESULTS* (5.5.4)

Limiting Event: Rotated P8DRB282

MCPR: 1.07

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Maximum Incremental Control Rod Worth: 0.95% Δk

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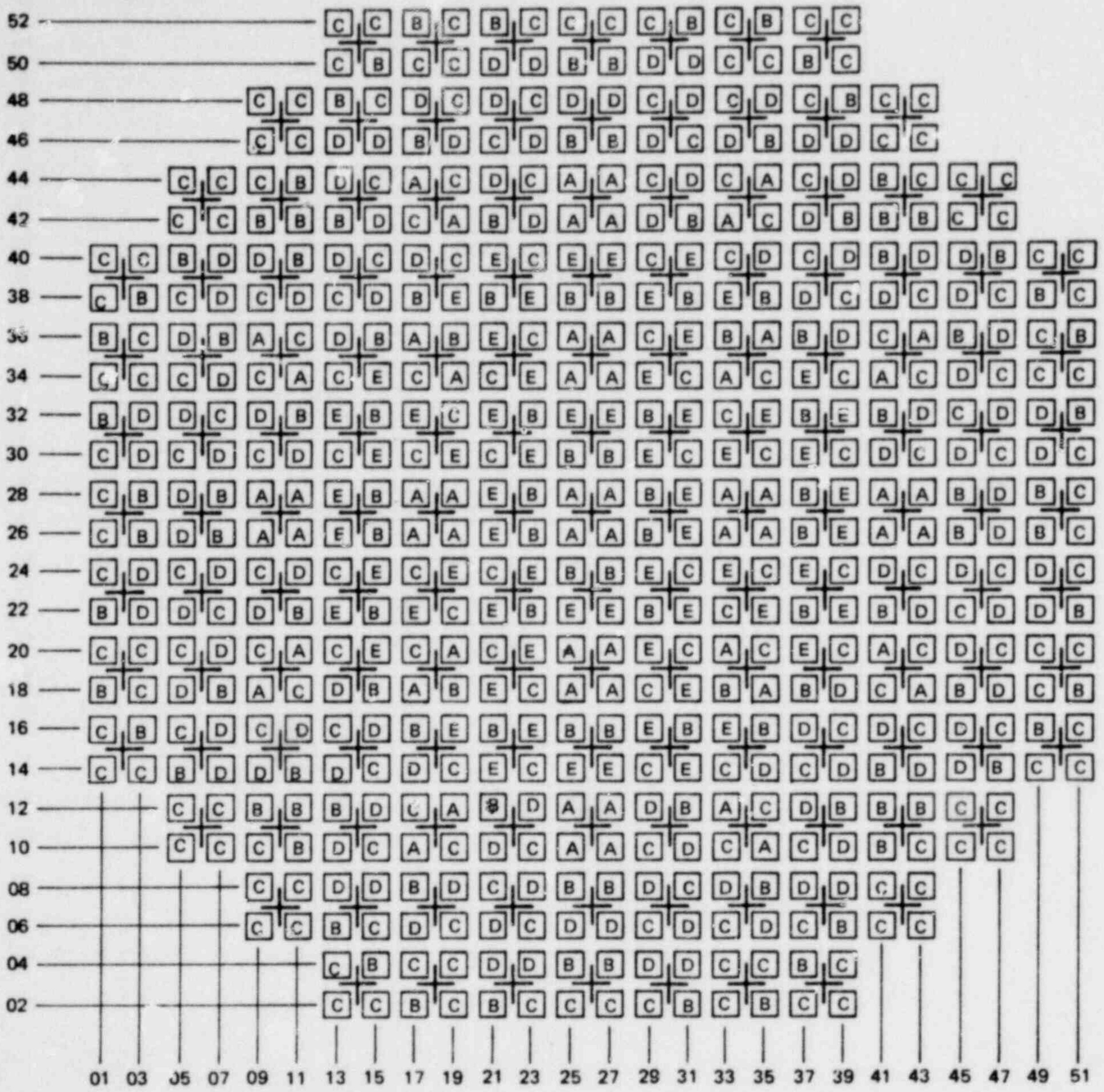

FUEL TYPE	
A = 8DB262	E = P8DRB282
B = 8DB219H	
C = 8DB219L	
D = P8DRB265L	

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Figure 1. Reference Core Loading

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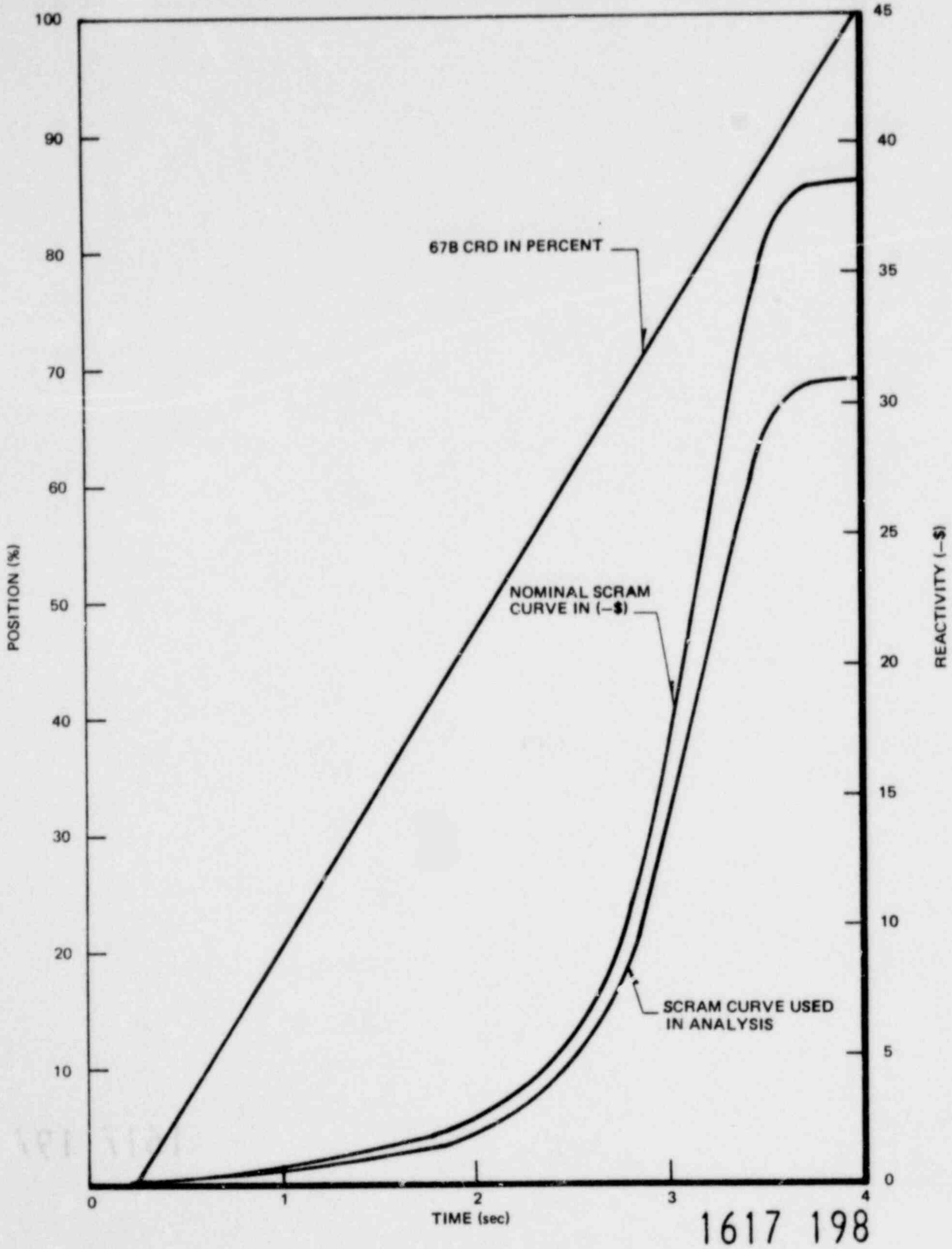

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Figure 2. Scram Reactivity and Control Rod Drive Specifications

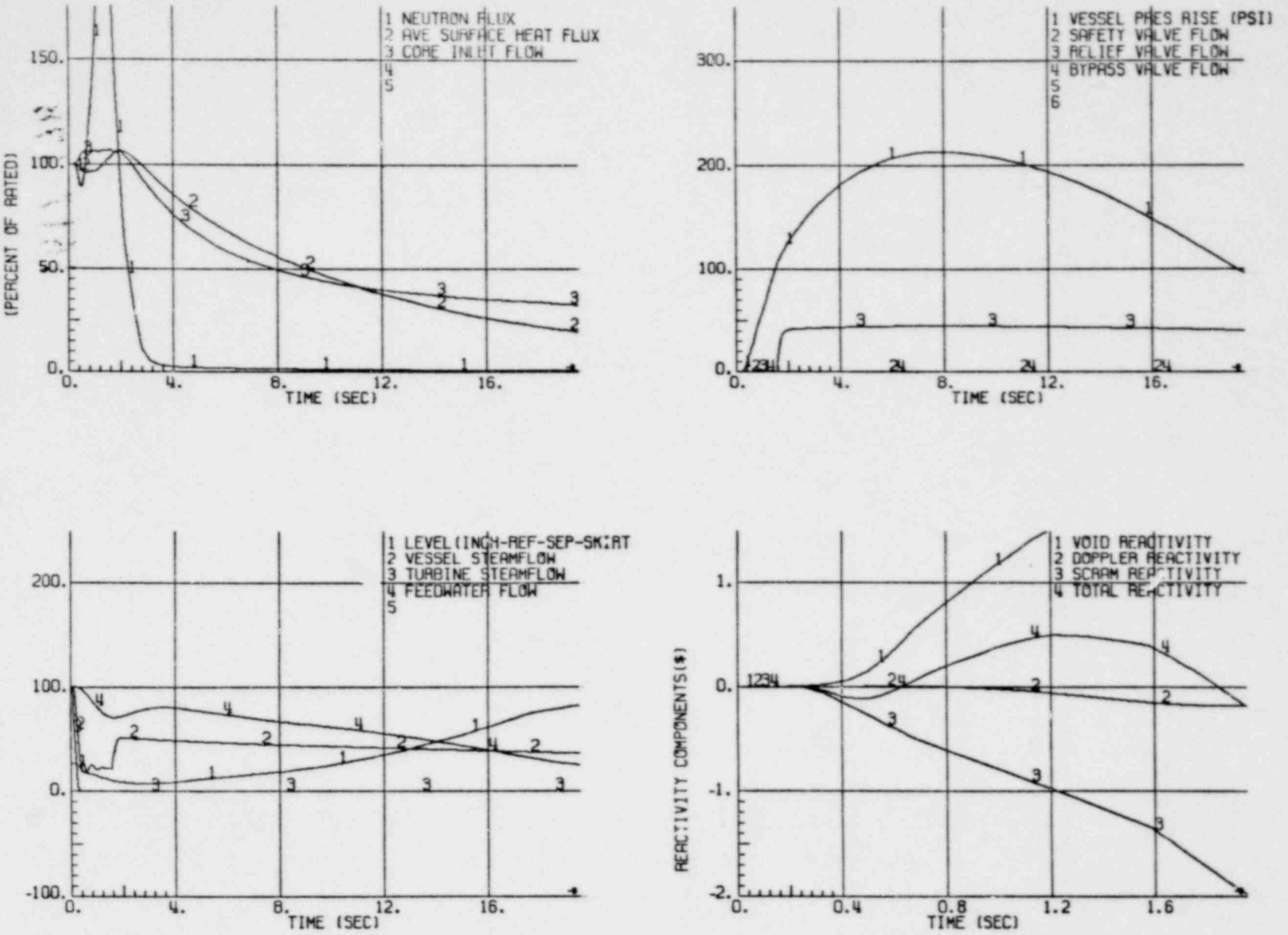

Figure 3. Plant Response to Generator Load Rejection without Bypass

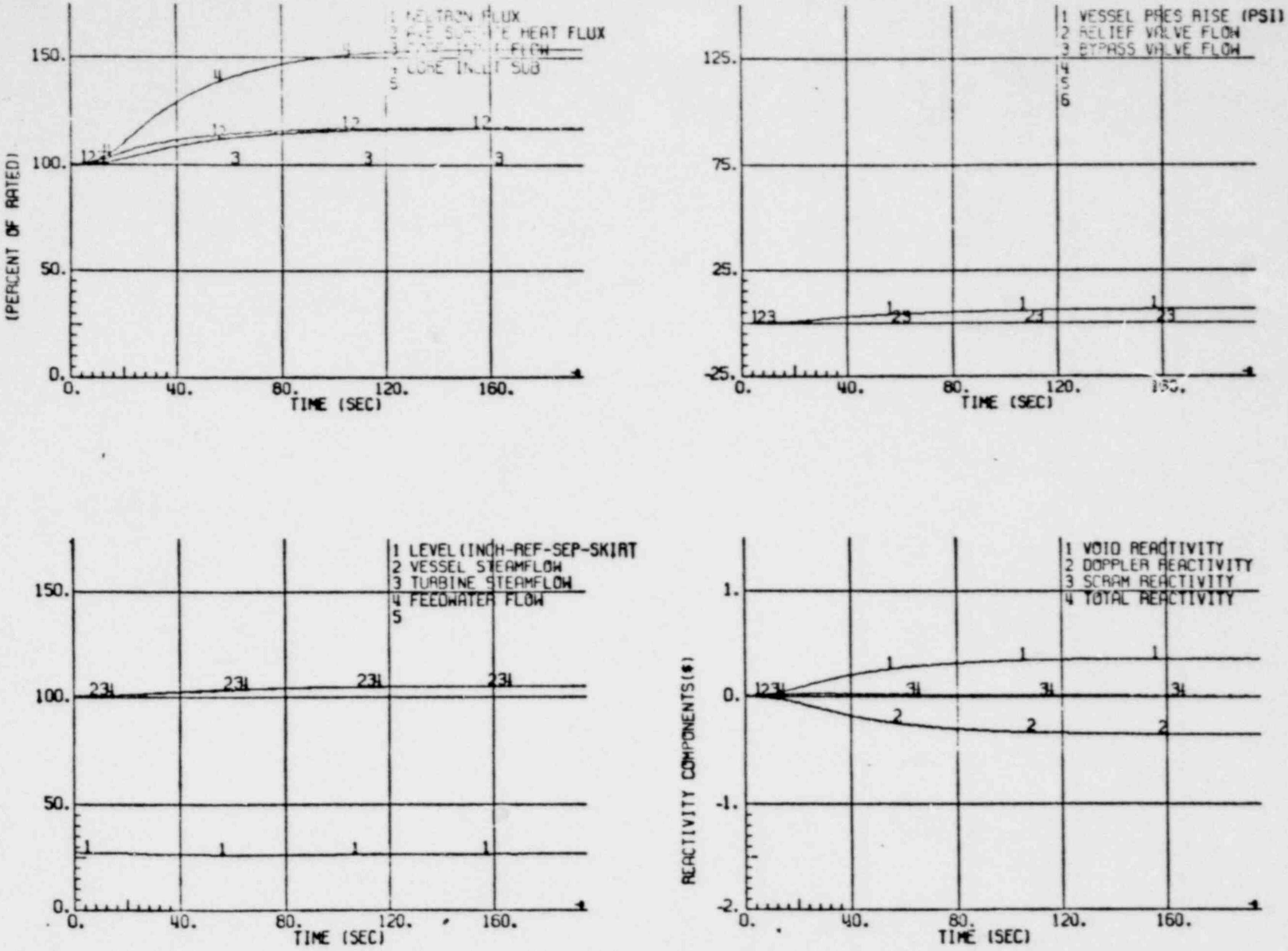

Figure 4. Plant Response to Loss of 100°F Feedwater Heating

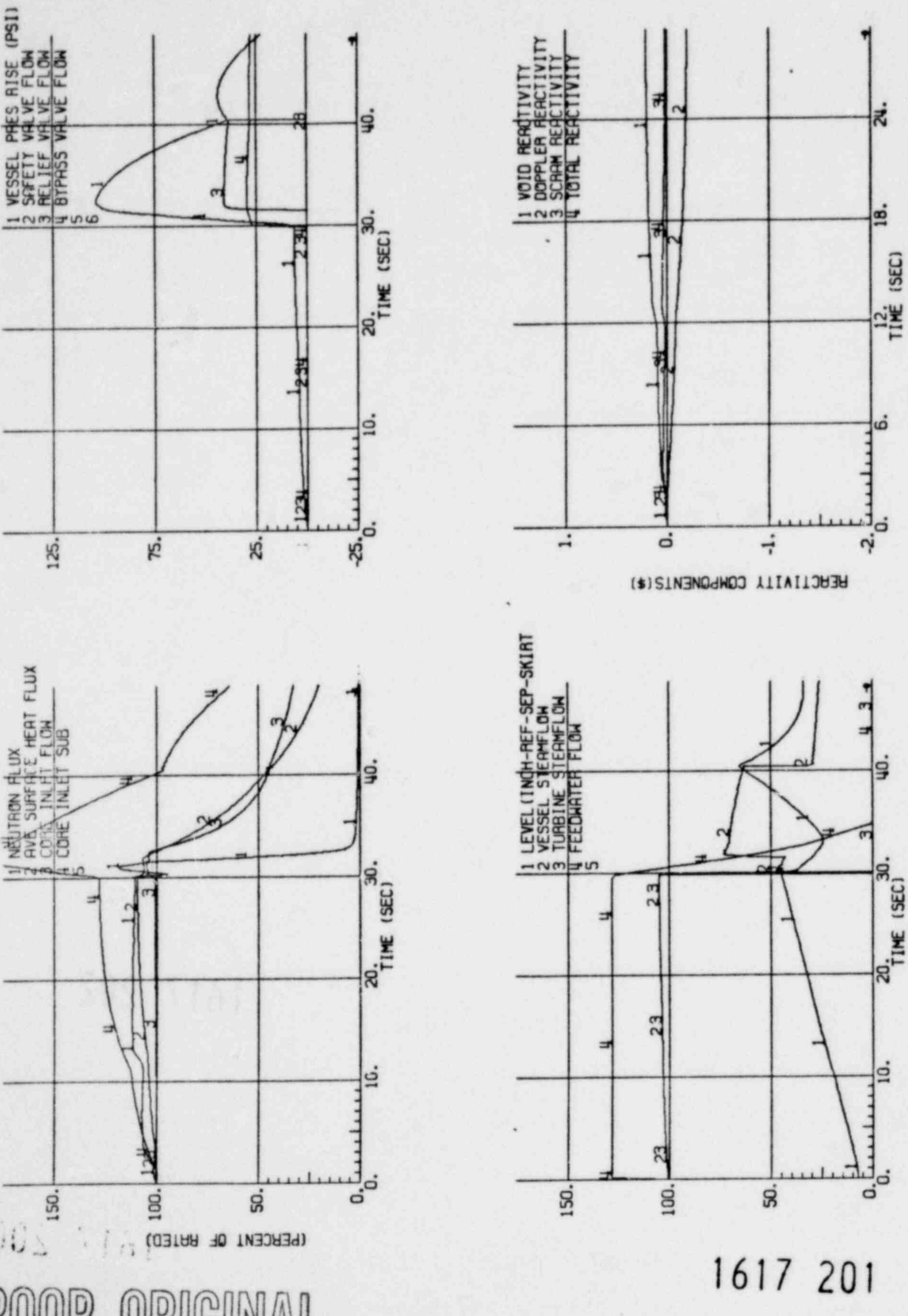

Figure 5. Plant Response to Feedwater Controller Failure, Maximum Demand, with High Level Turbine Trip

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	02	06	10	14	18	22	26
51					4		26
47				26		32	
43			2				
39		2		36		0	
35	2		6				
31		2		10		36	
27	2		2		2		2

NOTES:

1. Rod pattern is 1/4 core mirror symmetric upper left quadrant shown on map.
2. No. indicate number of notches withdrawn out of 48. Blank is a withdrawn rod.
3. Error rod is 22-39.

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Figure 6. Limiting RWE Rod Pattern

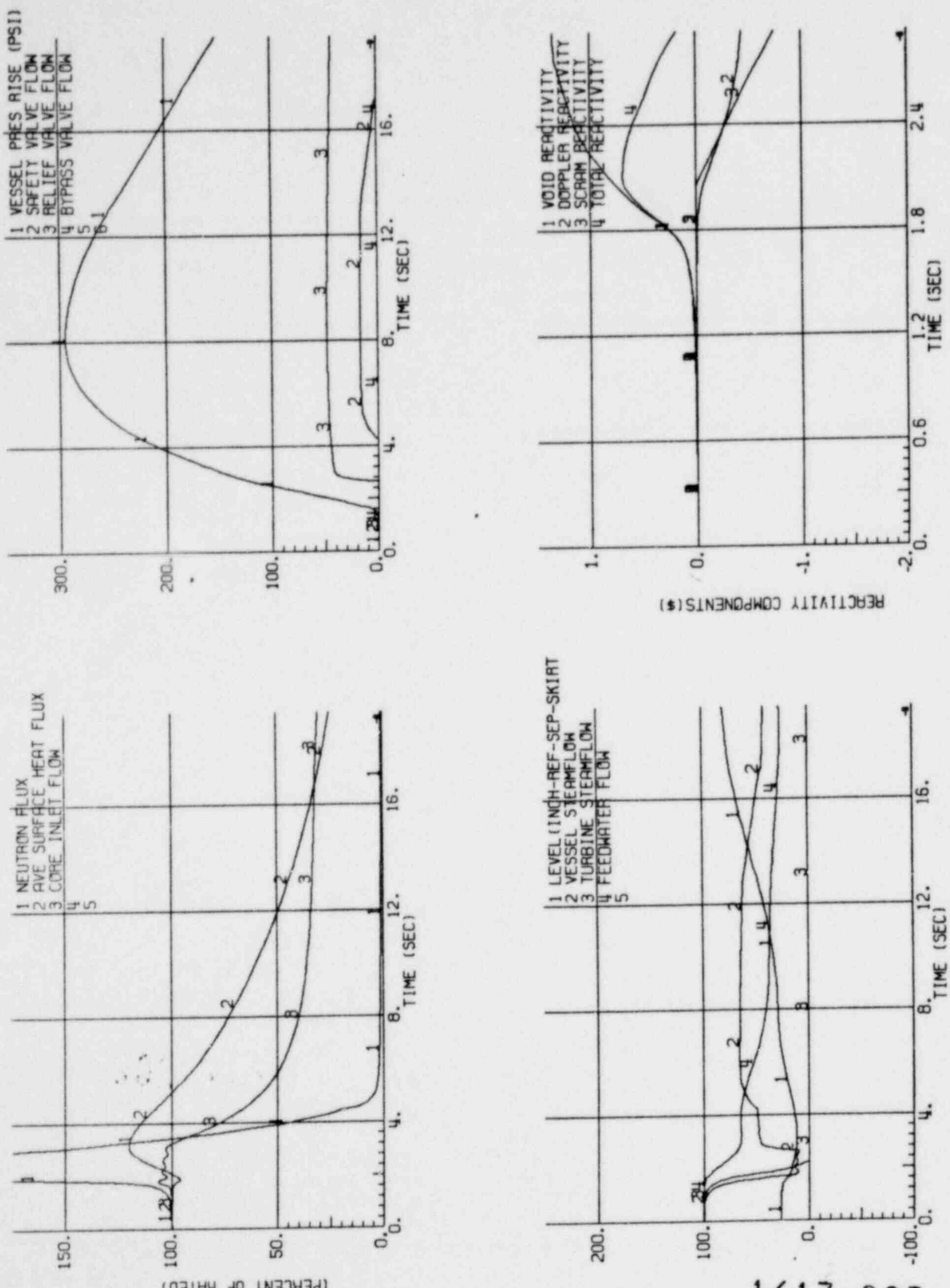

Figure 7. Plant Response to MSIV Closure, Flux Scram

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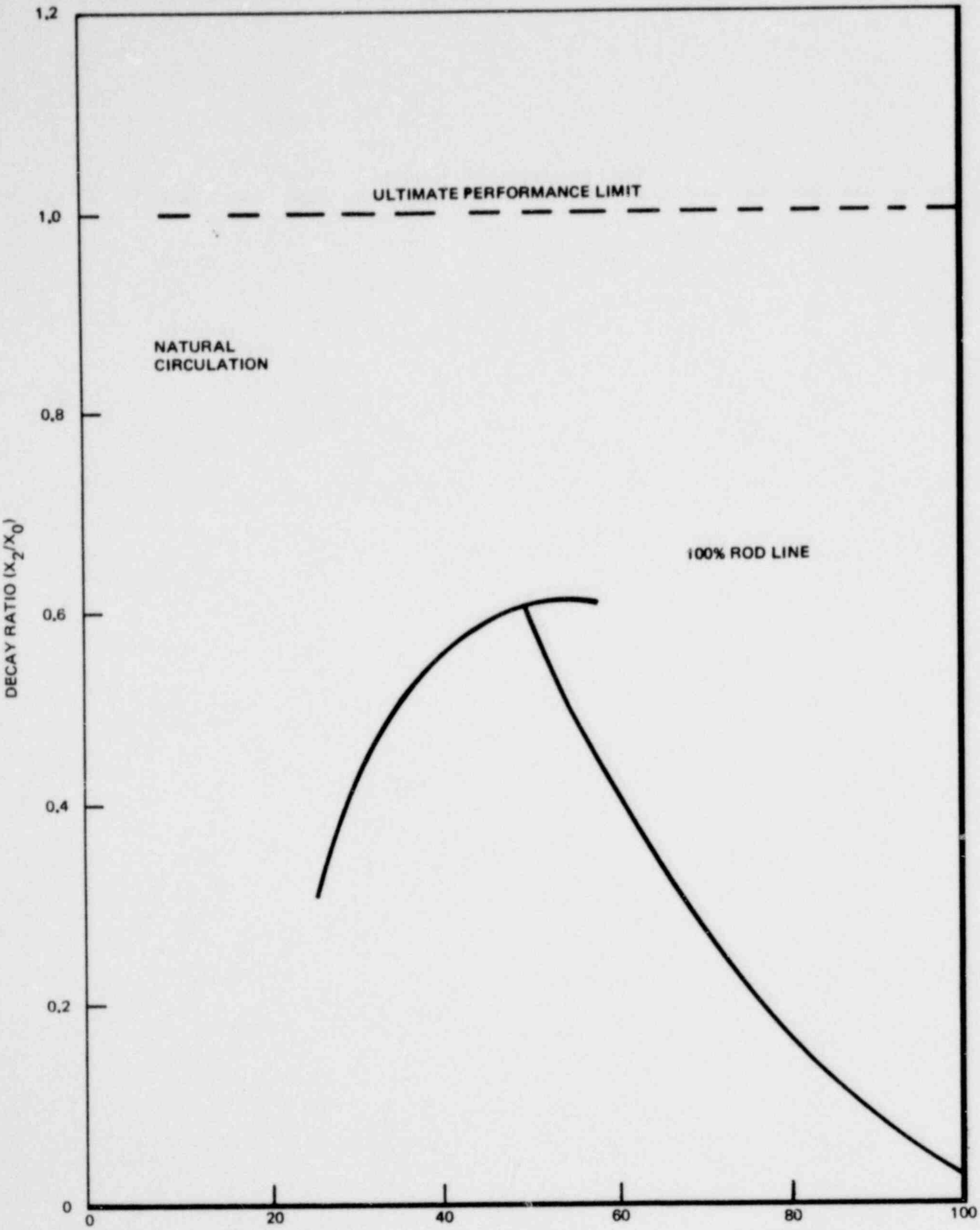

Figure 8. Decay Ratio

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REFERENCES

1. "General Electric Boiling Water Generic Fuel Application," NEDE-24011-P, Revision 3, March 1978.

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APPENDIX A
MARGIN-TO-SPRING SAFETY VALVES

The rationale for changing the basis for providing pressure margin to the spring safety valves is presented in:

- J. F. Quirk (GE) letter to Olan D. Parr (NRC), "General Electric Licensing Topical Report NEDE-24011-P-A, 'Generic Reload Fuel Appication,' Appendix D, Second Submittal," dated February 28, 1979.

On this basis the plant can operate at full power throughout the cycle.

The core response to the limiting infrequent event is given in Table A-1 and Figure A-1.

Table A-1
CORE-WIDE TRANSIENT ANALYSIS RESULTS

Transient	Exposure	Power (%)	Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure Trip Scram	BOC-EOC	100	100	1158	1188	Figure A-1

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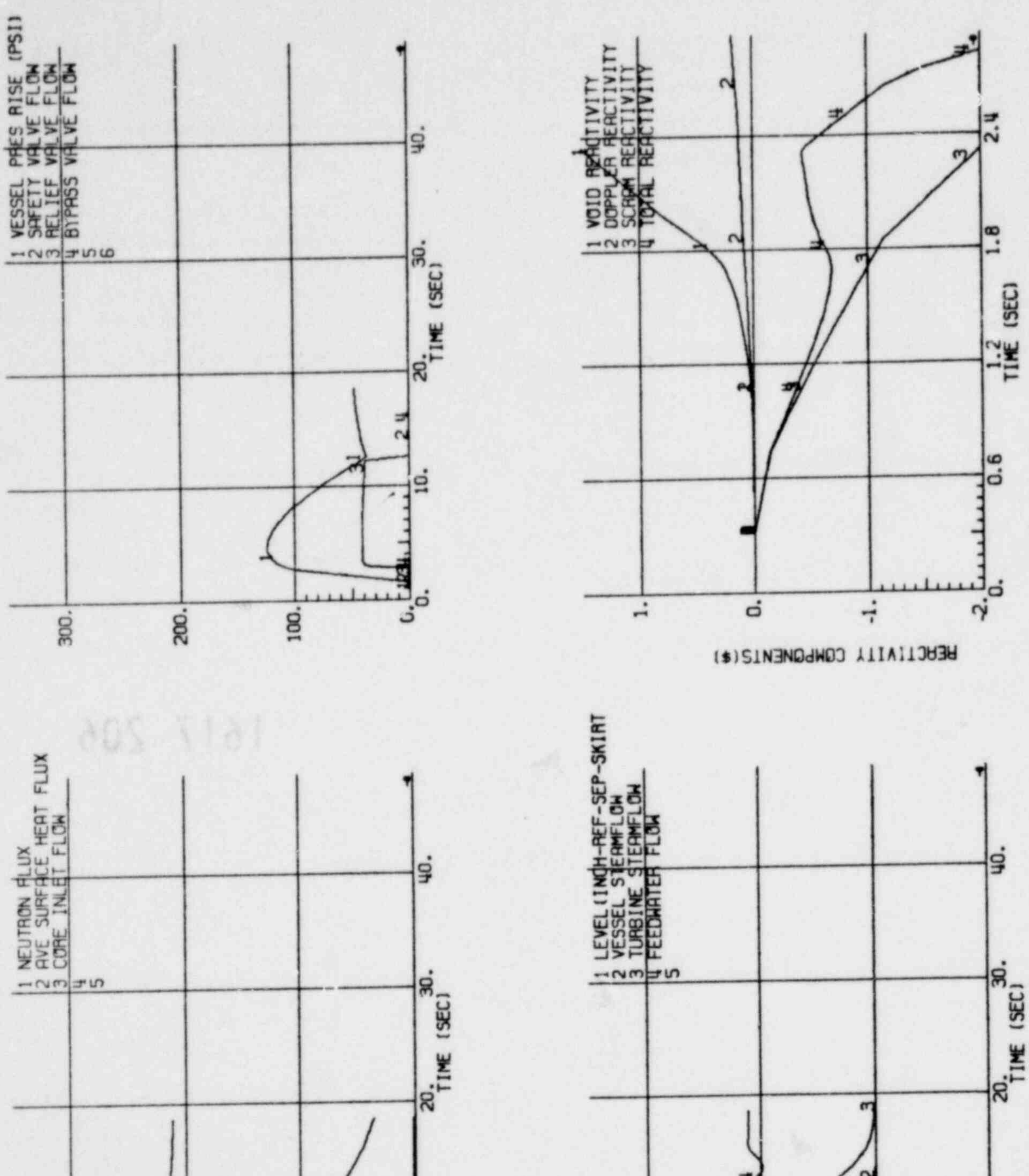

Figure A-1. Plant Response to MSIV Closure, Position Scram

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APPENDIX B
GETAB INITIAL CONDITIONS

Table 5-8 of Reference 1 states the "Nonvarying Plant GETAB Analysis Initial Conditions." The PNPS parameters, core pressure inlet enthalpy, and nonfuel power fraction are given as 1045 psia, 526.1 Btu/lb, and 0.035, respectively. Values of 1065 psia, 526.6 Btu/lb, and 0.030 which more nearly reflect actual plant data, were assumed for this submittal.

Reference 1 will be revised to eliminate these discrepancies.

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APPENDIX C
ATWS RECIRCULATION PUMP TRIP

Reference 1 states that PNPS has no ATWS-RPT. BECo will install ATWS-RPT during the fourth refueling outage. The transient analyses described in this document assume the ATWS-RPT is installed and functioning. Reference 1 will be revised to reflect this plant modification.

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APPENDIX D
NEW BUNDLE LOADING ERROR EVENT ANALYSES PROCEDURES

The bundle loading error analyses results presented in Section 15 in this supplement are based on new analyses procedures for both the rotated bundle and the mislocated bundle loading error events. The use of these new analysis procedures is discussed below.

NEW ANALYSIS PROCEDURE FOR THE ROTATED BUNDLE LOADING ERROR EVENT

The rotated bundle loading error analysis results presented in this supplement are based on the new analysis procedure described and approved in Reference D-1. This new method of performing the analysis is based on a more accurate detailed analytical model.

The principle difference between the previous analysis procedure and the new analysis procedure is the modeling of the water gap along the axial length of the bundle. The previous analysis used a uniform water gap, whereas the new analysis utilizes a variable water gap which is more representative of the actual condition, since the interfacing between the top guide and the fuel spacer buttons, caused by misorientation, causes the bundle to lean. The effect of the variable water gap is to reduce the power peaking and the R-factor in the upper regions of the limiting fuel rod. This results in the calculation of a reduced CPR for the rotated bundle. The calculation was performed using the same analytical models as were previously used. The only change is in the simulation of the water gap, which more accurately represents the actual geometry.

NEW ANALYSIS PROCEDURE FOR THE MISLOCATED BUNDLE LOADING ERROR EVENT

The mislocated bundle loading error event analyses results presented in this supplement are based on the new analysis procedure described in Reference D-1. This new method of performing the analysis employs a statistically corrected Haling procedure and analyzes every bundle in the core.

The use of the statistically corrected Haling analyses procedure indicates that the minimum CPR for mislocated bundles is greater than the safety limit (1.07) for all exposures throughout Cycle 7.

REFERENCES

D-1. Safety Evaluation Report (letter), D.G. Eisenhut (NRC) to R. E. Engel (GE), MFN-200-78, dated May 8, 1978.

APPENDIX E
LINEAR HEAT GENERATION RATE FOR BUNDLE LOADING ERROR

17.7 kW/ft

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APPENDIX F
DENSIFICATION POWER SPIKING

Reference F-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 PNPS-1 Reload-4 submittal contains the required information to remove the power spiking penalty from the PNPS-1 Technical Specifications. Section 10, Rod Withdrawal Error, and Appendix E (Linear Heat Generation Rate for Bundle Loading Error) include the densification effect in the calculated LHGR of the 8x8 fuels.

REFERENCES

F-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.

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