Y1003J01A09
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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

MILLSTONE UNIT 1

RELOAD 7

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

GENERAL ELECTRIC

8009300302

IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

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1. PLANT-UNIQUE ITEMS (1.0)*

Appendix A: New Bundle Loading Error Event Analysis Procedures

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

Fuel Type		Number	Number Drilled
Irradiated	8DB262	20	0
Irradiated	8DB274L	120	0
Irradiated	8DB274H	124	0
Irradiated	8DRB265H	48	48
Irradiated	8DRB265L	100	100
New	P8DRB265H	40	40
New	P8DRB282	128	128
Total		580	316

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 17,131 MWd/t. Assumed reload cycle exposure: 17,889 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	1.120
Fully Controlled	0.960
Strongest Control Rod Out	0.989
R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, Δk	0.0

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

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.029

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

	EOC8	EOC8-1800 MWd/t
Void Coefficient N/A,* (c/% Rg)	-6.18/-7.72	-7.25/-9.07
Void Fraction (%)	36.85	36.85
Doppler Coefficient N/A (c/%°F)	-0.217/-0.206	-0.210/-0.199
Average Fuel Temperature (°F)	1217	1217
Scram Worth N/A ($)	-37.79/-30.22	-35.72/-28.58
Scram Reactivity	Figure 2a	Figure 2b

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

	8x8/8x8R/P8x8R	8x8/8x8R/P8x8R
	EOC8	EOC8-1800 MWd/t
Peaking factors (local, radial and axial)	1.22/1.20/1.20 1.56/1.68/1.66 1.40/1.40/1.40	1.22/1.20/1.20 1.65/1.77/1.75 1.40/1.40/1.40
R-Factor	1.094/1.051/1.051	1.094/1.051/1.051
Bundle Power (MWt)	5.304/5.674/5.606	5.582/6.004/5.930
Bundle Flow (10^3 lb/hr)	101.8/99.9/100.2	100.0/97.8/98.2
Initial MCPR	1.37/1.37/1.39	1.29/1.29/1.31

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Exposure-dependent limits: BOC8 to EOC8-1800 MWd/t
EOC8-1800 MWd/t to EOC8

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

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Core Flow (%)	$\hat{\phi}$ (% NBR)	Q/A (% NBR)	P_{SL} (psig)	P_v (psig)	ΔCPR 8x8	ΔCPR 8x8R	ΔCPR P8x8R	Plant Response
Load Rejection w/o Bypass	EOC8	100	100	305	116	1193	1226	0.30	0.30	0.32	Figure 3a
	EOC8-1800 MWd/t	100	100	252	111	1183	1217	0.22	0.22	0.24	Figure 3b
Loss of 100°F Heater Feedwater	--	100	100	116	117	1033	1072	0.15	0.15	0.15	Figure 4
Feedwater Controller Failure	--	100	100	111	106	1033	1072	0.06	0.07	0.07	Figure 5

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

Rod Block Reading	Rod Position (Feet Withdrawn)	ΔCPR 8x8/8x8R/P8x8R	LHGR 8x8/8x8R and P8x8R	Limiting Rod Pattern
104	3.5	0.19/0.11/0.11	10.6/13.4	Figure 6
105	3.5	0.19/0.11/0.11	10.6/13.4	↓
106	4.0	0.22/0.12/0.12	10.9/13.6	↓
107*	4.5	0.24/0.14/0.14	11.0/13.7	↓
108	5.0	0.27/0.15/0.15	11.0/13.7	↓
109	5.0	0.27/0.15/0.15	11.0/13.7	↓
110	6.5	0.32/0.18/0.18	11.0/13.8	Figure 6

*RBM Setpoint

11. OPERATING MCPR LIMIT (5.2)

EOC8-1800 MWd/t to EOC8	BOC8 to EOC8-1800 MWd/t
1.37 (8x8 fuel)	1.31 (8x8 fuel)
1.37 (8x8R)	1.29 (8x8R)
1.39 (P8x8R)	1.31 (P8x8R)

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	Power (%)	Core Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	100	100	1244	1276	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio, x_2/x_o (Natural Circulation-Rod Block Power)	0.61

Channel Hydrodynamic Performance

	Decay Ratio (Natural Circulation - Rod Block Power)
P8x8R channel	0.20
8x8 channel	0.26

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

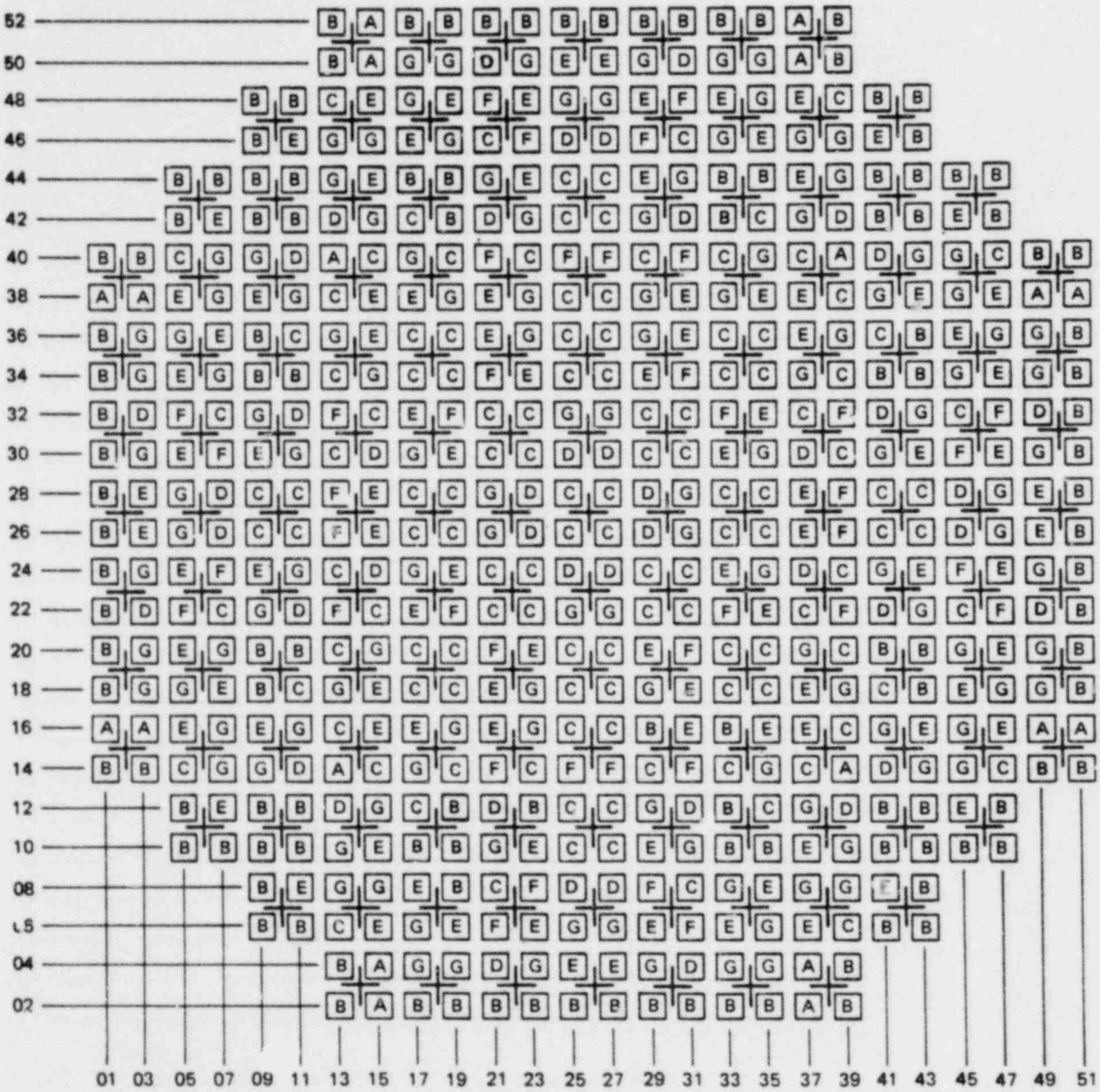
See NEDO-24085-1.

15. LOADING ERROR RESULTS (5.5.4)

See Appendix A.

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Maximum incremental control rod worth: 0.89% Δk



FUEL TYPE	
A = 8DB262	E = 8DRB26 L
B = 8DB274L	F = P8DRB265H
C = 8DB274H	G = P8DRB262
D = 8DRB265H	

Figure 1. Core Loading Pattern

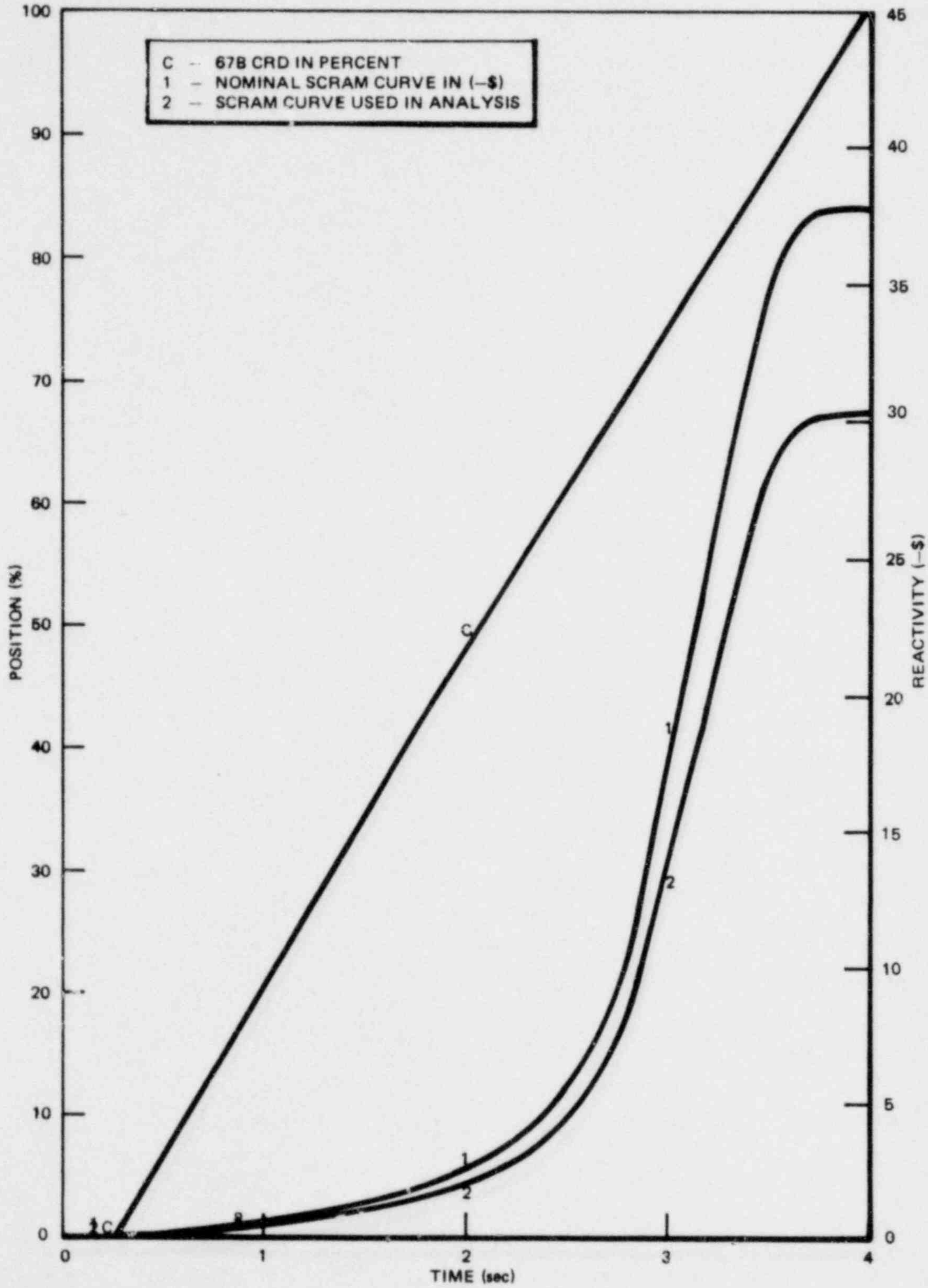


Figure 2a. Scram Reactivity and CRD Specifications, EOC8

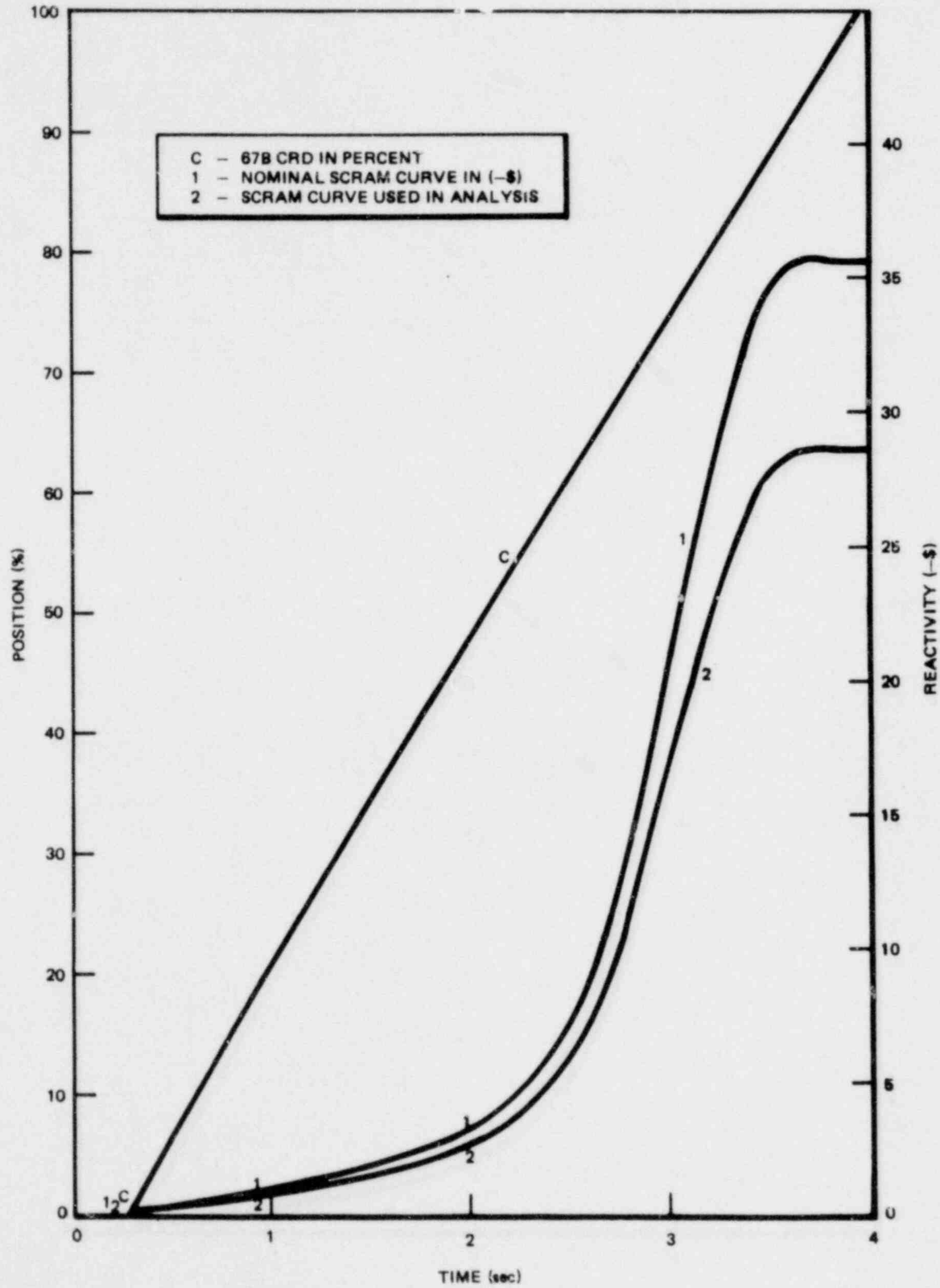


Figure 2b. Scram Reactivity and CRD Specifications, EOC8-1800 MWd/t

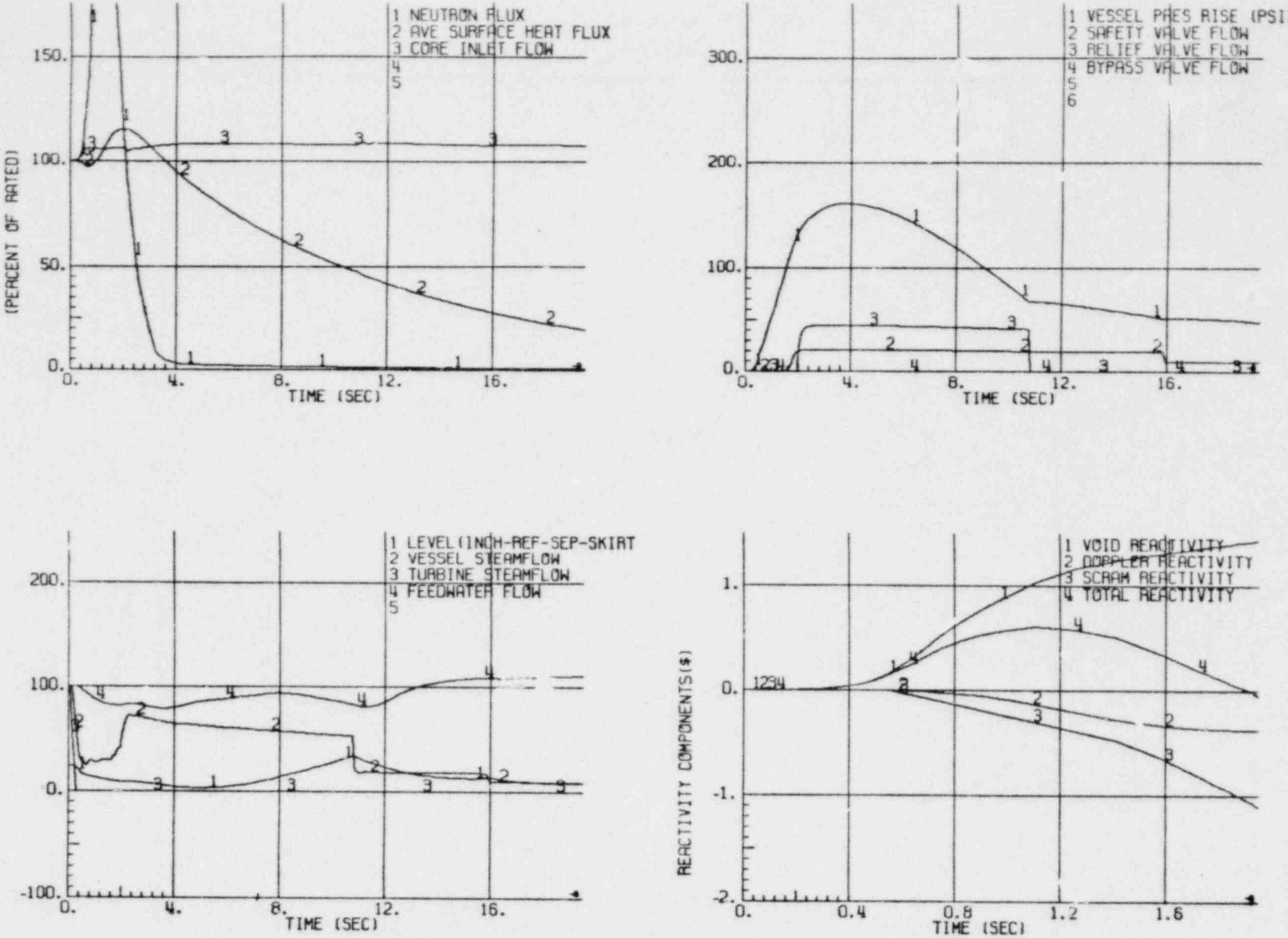


Figure 3a. Generator Load Rejection, Without Bypass, EOC8

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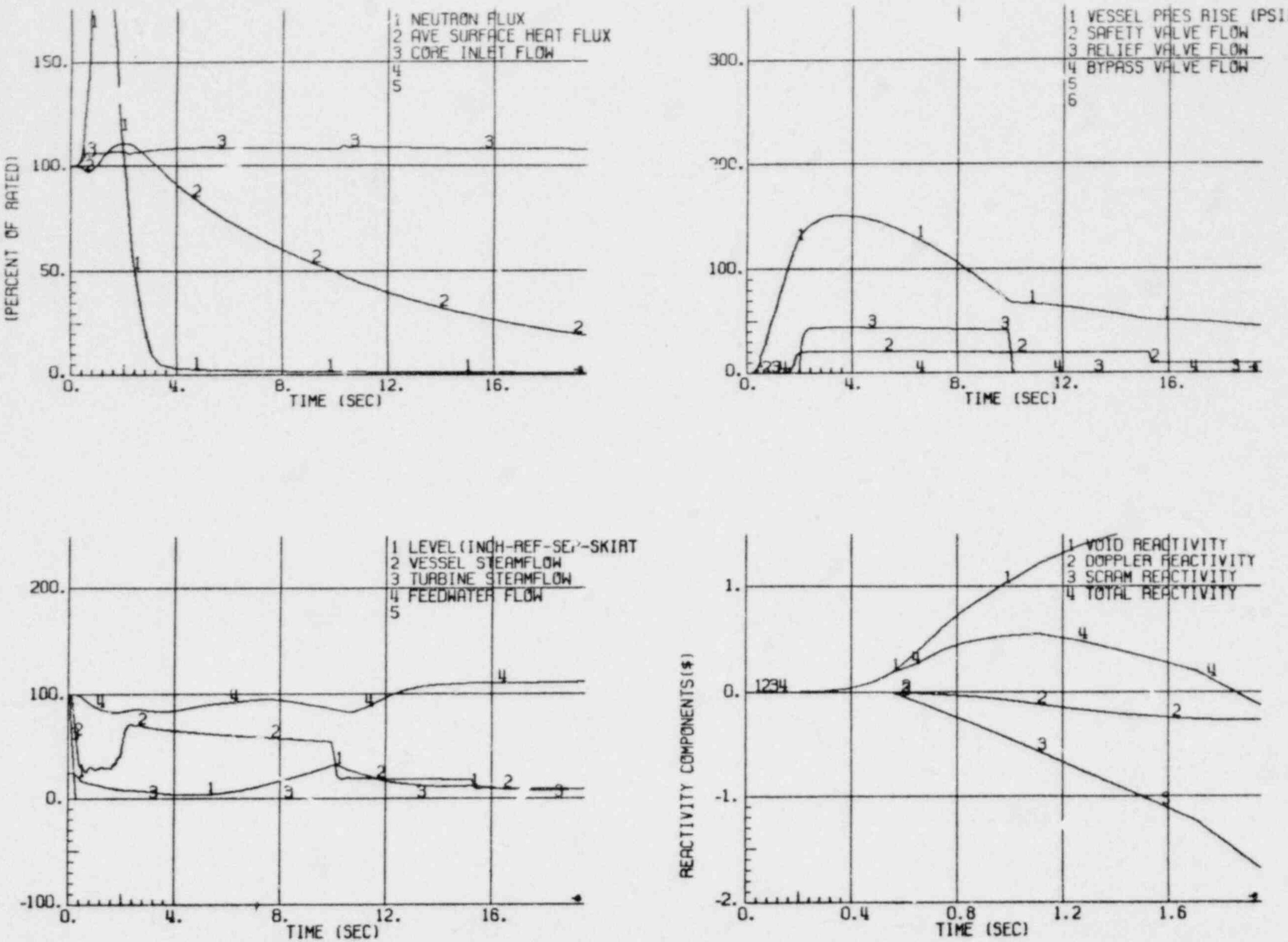

Figure 3b. Generator Load Rejection, Without Bypass, EOC8-1800 MWd/t

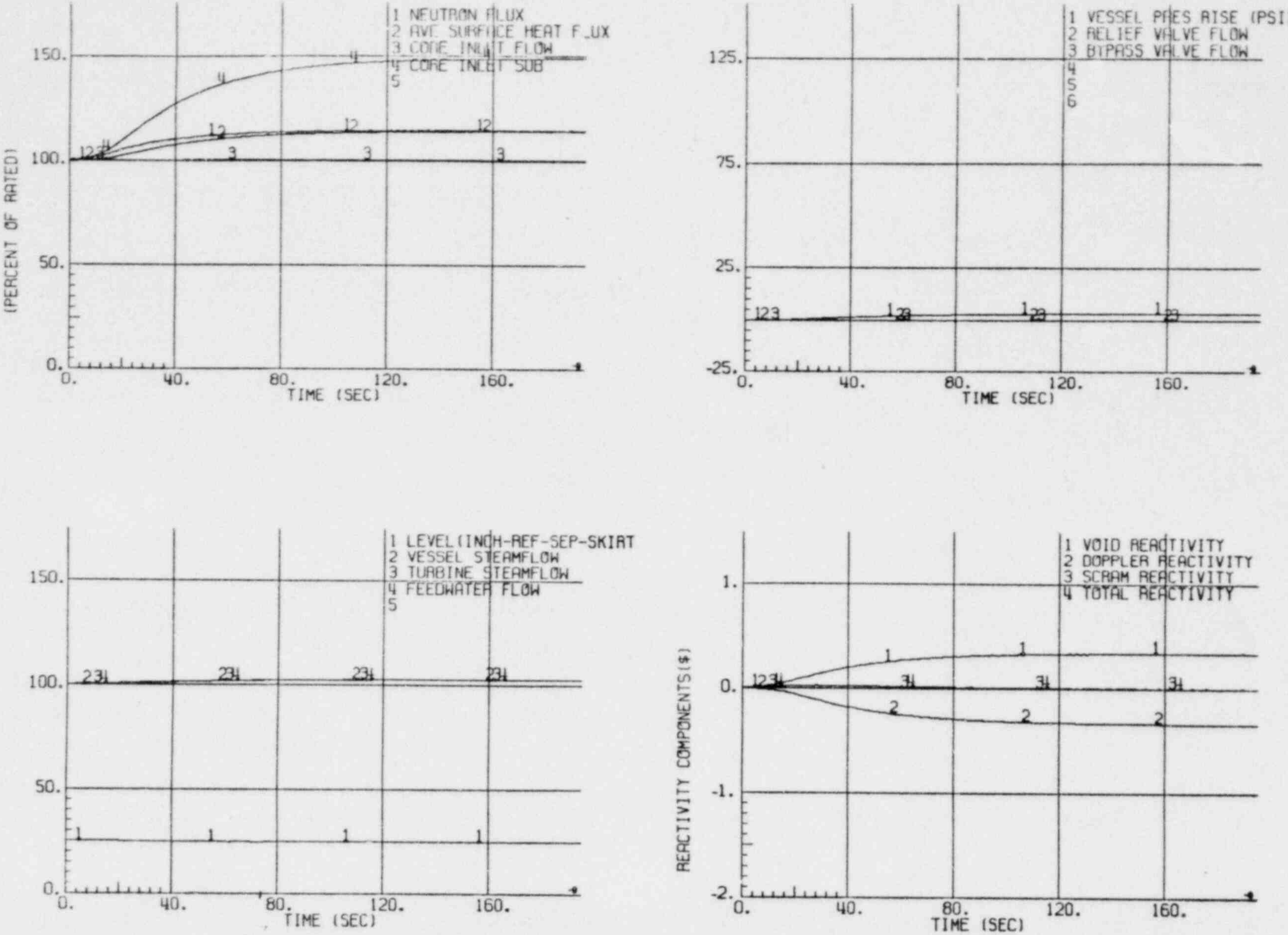

Figure 4. Loss of 100°F Feedwater Heating

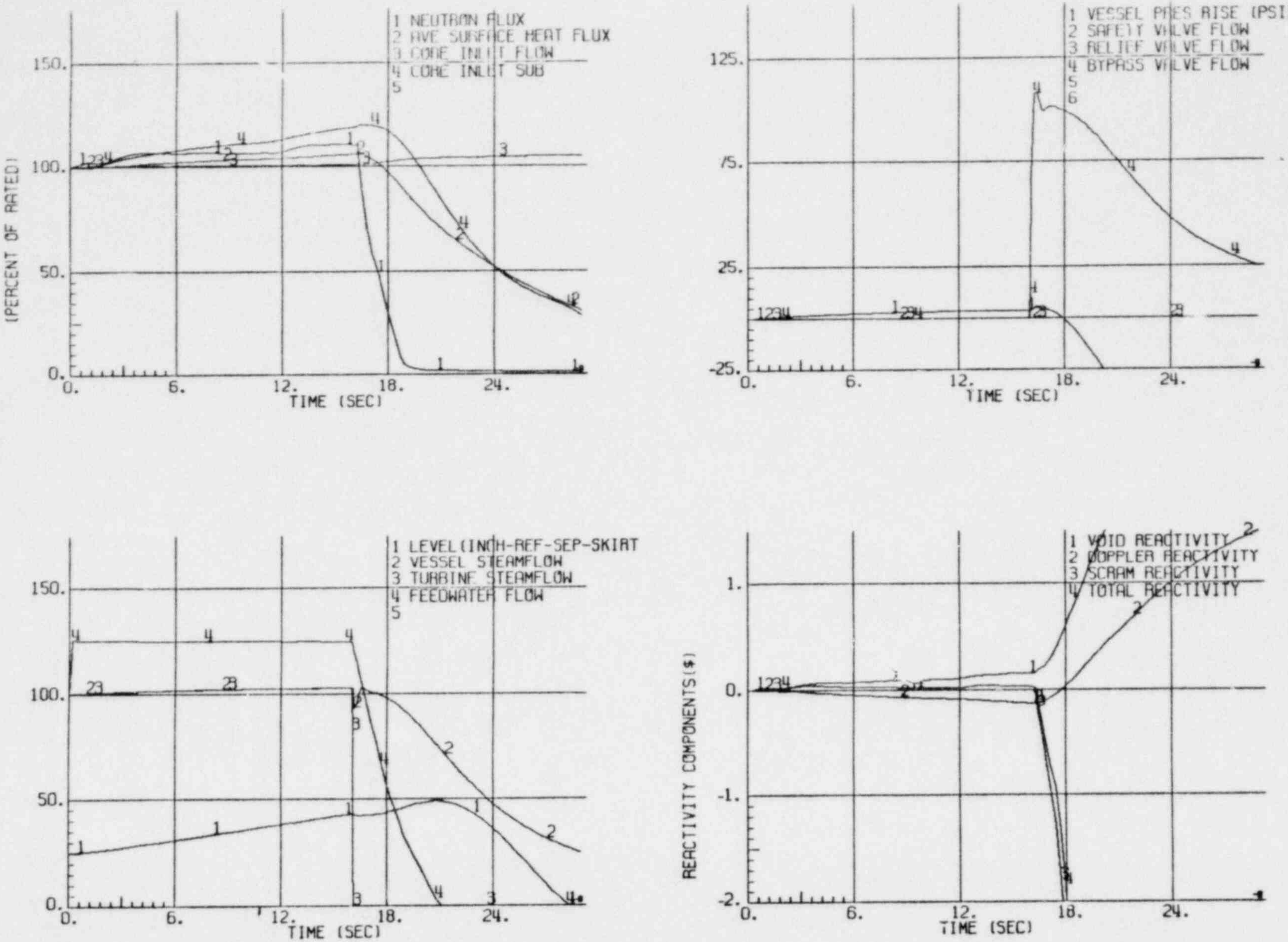

Figure 5. Feedwater Controller Failure

	01	03	05	07	09	11	13
01				06		06	
03			36		38		38
05		06		06		08	
07	36		38		38		38
09		06		08		10	
11	38		38		40		40
13		08		10		00	

NOTES: 1. Rod pattern is 1/4 core mirror symmetric.

2. Number indicates number of notches withdrawn out of 48. Blank is a withdrawn rod.

3. Error rod is (11,13).

Figure 6. Limiting Rod Pattern

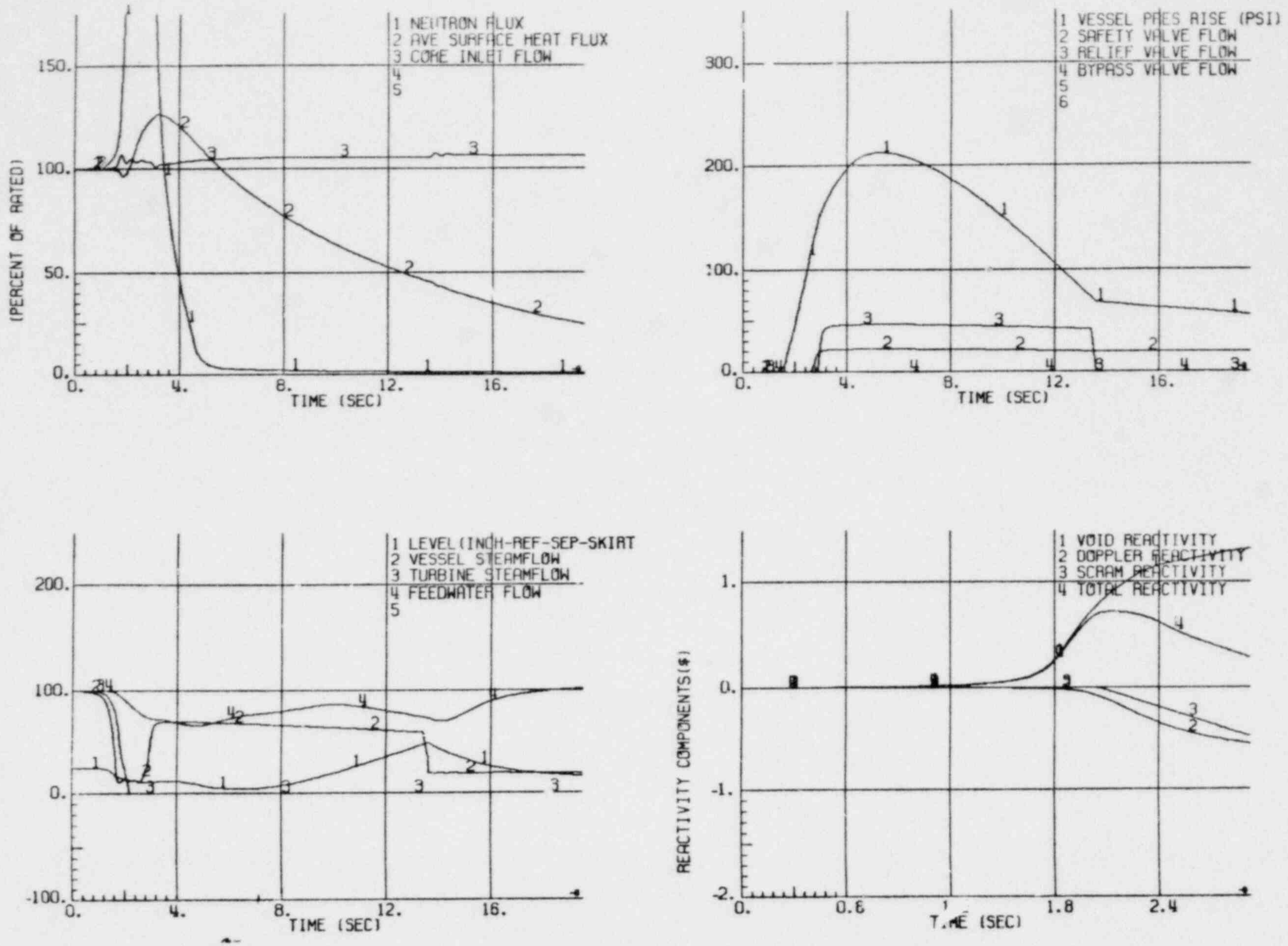

Figure 7. MSIV Closure, Flux Scram

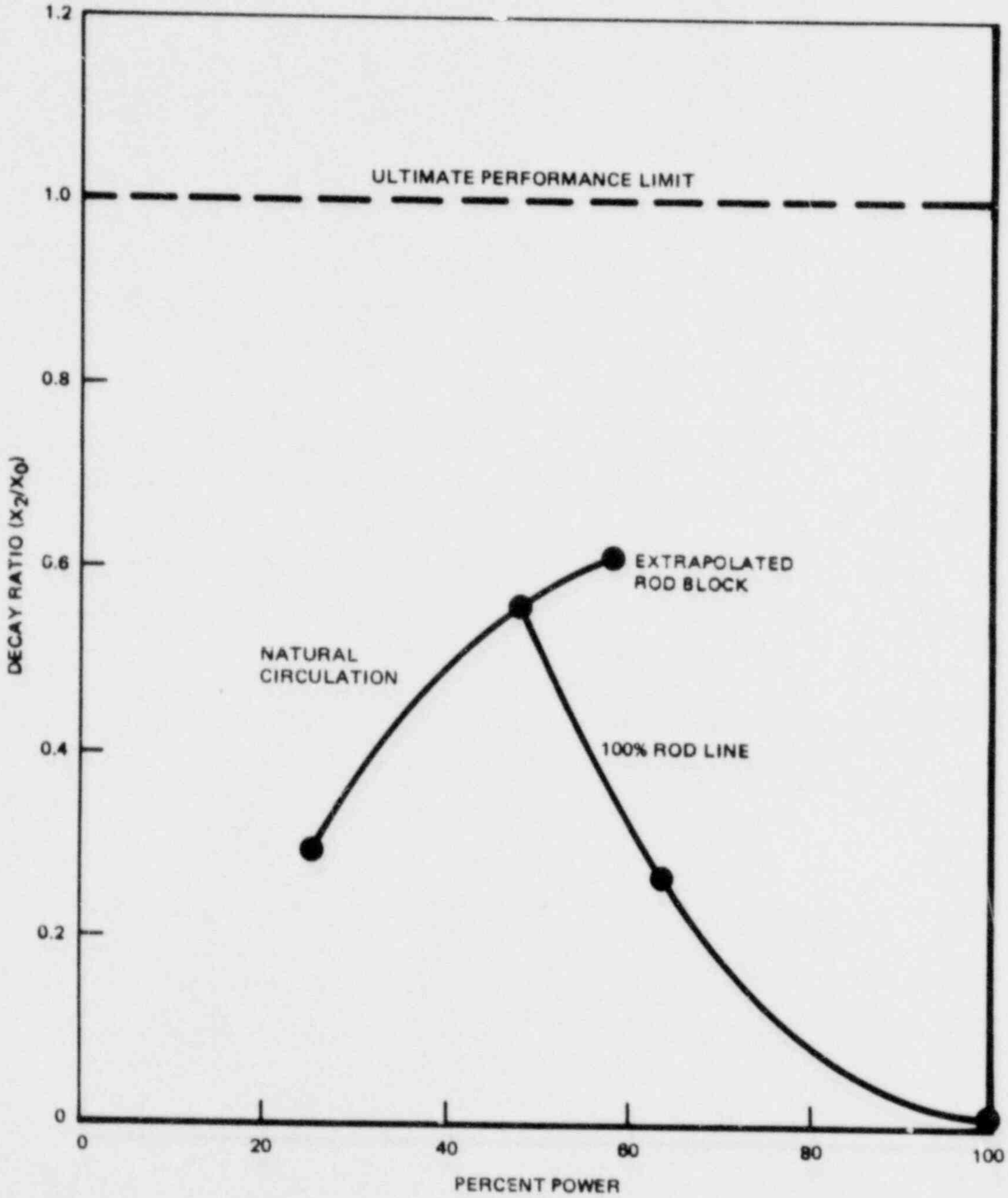

Figure 8. Reactor Core Decay Ratio vs Power

APPENDIX

NEW BUNDLE LOADING ERROR EVENT ANALYSES PROCEDURES

The bundle loading error analyses results are based on new analyses procedures for both the rotated bundle and the mislocated bundle loading error events. The use of these new analyses procedures is discussed below.

A.1 NEW ANALYSIS PROCEDURE FOR THE ROTATED BUNDLE LOADING ERROR EVENT

The rotated bundle loading error event analysis results presented in this supplement are based on the new analysis procedure described and approved in Reference A-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.

The results of the analysis indicate for the P8x8R bundle a 17.3 kW/ft LHGR and 0.19 ΔCPR (includes a 0.02 penalty due to variable water gap R-factor uncertainty) with a minimum CPR of 1.07.

A.2 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 A-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 LHGR is 17.2 kW/ft and that the minimum CPR for mislocated bundles is greater than the safety limit (1.07) for all exposures throughout Cycle 5.

A.3 REFERENCES

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

DOCKET NO. 50-245

ATTACHMENT 2

"LOSS-OF-COOLANT ACCIDENT ANALYSIS REPORT

FOR MILLSTONE UNIT 1 NUCLEAR POWER STATION",

NEDO-24085-1, JULY, 1980

SEPTEMBER, 1980