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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR MONTICELLO NUCLEAR GENERATING PLANT RELOAD 10 (CYCLE 11)

GENERAL ELECTRIC

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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
MONTICELLO NUCLEAR GENERATING PLANT
RELOAD 10 (CYCLE 11)

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

Plant Parameter Differences: Appendix A

GETAB Transient Initial Condition Parameters

Fuel Channels

Feedwater Controller Failure Event: Appendix B

Control Rod Drop Analysis: Appendix C

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

Fuel Type	Cycle Loaded	Number	Number Drilled
Irradiated			
8DB250	4	24	0
8DB219L	5	48	0
8DB262	6	8	0
P8DRB282	8	56	56
P8DRB265L	8	44	44
P8DRB265L	9	40	40
P8DRB284LB	9	44	44
P8DRB265L	10	56	56
P8DRB284LB	10	48	48
New			
P8DRB284LB	11	116	116
Total		484	404

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	18,360 MWd/ST
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	18,061 MWd/ST
Assumed reload cycle core average exposure at end of cycle:	17,333 MWd/ST
Core loading pattern:	Figure 1

*( ) Refers to area of discussion in "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-6, April 1983; a letter "S" preceding the number refers to the appropriate section in the United States supplement, NEDE-24011-P-A-6-US, April 1983.

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

Beginning of cycle, k_{eff}	
Uncontrolled	1.120
Fully Controlled	0.963
Strongest Control Rod Out	0.990
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, Δk	0.000

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
645	0.035

6. RELOAD UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 AND S.2.2)

(REDY EVENTS ONLY)

Void Fraction (%)	37.2
Average Fuel Temperature (°F)	1145
Void Coefficient N/A* (¢/% Rg)	-6.56/-8.20
Doppler Coefficient N/A (¢/°F)	-0.191/-0.181
Scram Worth N/A ($)	**

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

**Generic exposure independent values are used as given in "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-6-US, April 1983.

7. RELOAD UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (S.2.2)

Fuel Design	Peaking Factors Local	Radial	Axial	R-Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
BOC11 to EOC11							
P8x8R	1.20	1.69	1.40	1.051	5.683	98.3	1.37
8x8	1.22	1.58	1.40	1.098	5.320	99.2	1.34

8. SELECTED MARGIN IMPROVEMENT OPTIONS(S.2.2.2)

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

9. OPERATING FLEXIBILITY OPTIONS (S.2.2.3)

Single-Loop Operation:	Yes
Load Line Limit:	Yes
Extended Load Line Limit:	No
Increased Core Flow:	No
Feedwater Temperature Reduction:	No

10. CORE-WIDE TRANSIENT ANALYSIS RESULTS (S.2.2.1)

Transient	Flux (% NBR)	Q/A (% NBR)	ΔCPR P8x8R	ΔCPR 8x8	Figure
Exposure: BOC11 to EOC11 Load Rejection w/o Bypass	613	120	0.30	0.27	2
Exposure: BOC11 to EOC11 Loss of Feedwater Heater	118	117	0.15	0.15	3
Exposure: BOC11 to EOC11 Feedwater Controller Failure*	459	124	0.32	0.30	4

*See Appendix B

11. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (S.2.2.1)

(Generic Bounding Analysis Results)

Rod Block Reading	ΔCPR (All Fuel Types)
104	0.13
105	0.16
106	0.19
107	0.22
108	0.28
109	0.32
110	0.36

Setpoint Selected: 108

12. CYCLE MCPR VALUES (S.2.2)

Nonpressurization Events

Exposure Range: BOC11 to EOC11

	P8x8R	8x8
Loss of Feedwater Heater	1.22*	1.22*
Fuel Loading Error	1.27	
Rod Withdrawal Error	1.35	1.35

Pressurization Events

Exposure Range: BOC11 to EOC11

	Option A		Option B	
	P8x8R	8x8	P8x8R	8x8
Load Rejection w/o Bypass	1.43	1.40	1.38	1.35
Feedwater Controller Failure	1.45	1.43	1.36	1.34

*Minimum MCPR required by ECCS is 1.24.

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1201	1223	Figure 5

14. STABILITY ANALYSIS RESULTS (S.2.4)

Rod Line Analyzed: Extrapolated Rod Block Line

Decay Ratio:	Figure 6
Reactor Core Stability Decay Ratio, x_2/x_0:	0.63
Channel Hydrodynamic Performance Decay Ratio, x_2/x_0	
Channel Type	
P8x8R	0.19
8x8	0.23

15. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

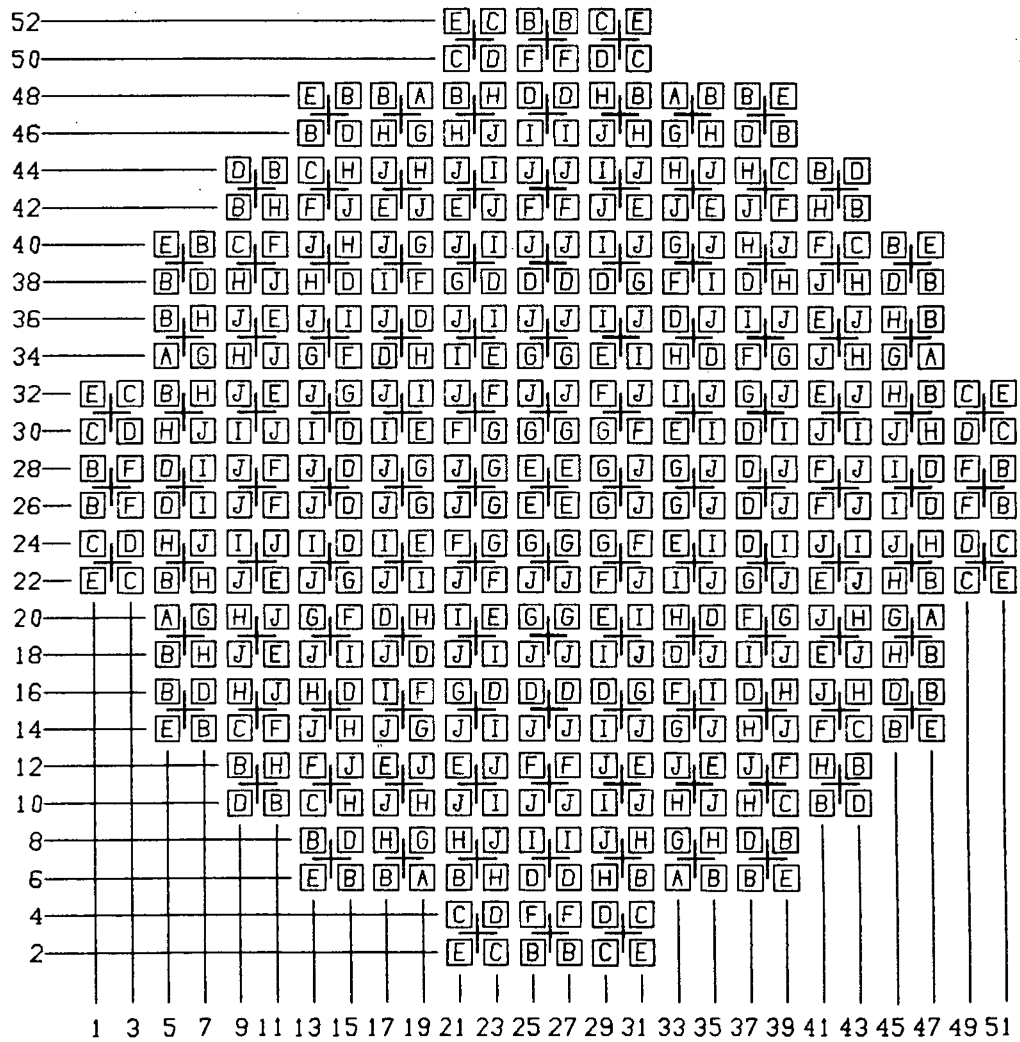
Event	Initial MCPR	Resulting MCPR
Misoriented	1.26	1.08

16. CONTROL ROD DROP ANALYSIS RESULTS (S.2.5.1)

See Appendix C

17. LOSS-OF-COOLANT ACCIDENT RESULT (S.2.5.2)

"Loss-of-Coolant Accident Analysis Report for Monticello Nuclear Generating Plant", December 1980 (as amended), NEDO-24050-1.



FUEL TYPE	
A = 8DB262	F = P8DRB265L
B = 8DB219L	G = P8DRB284LB
C = 8DB250	H = P8DRB265L
D = P8DRB282	I = P8DRB284LB
E = P8DRB265L	J = P8DRB284LB

Figure 1. Reference Core Loading Pattern

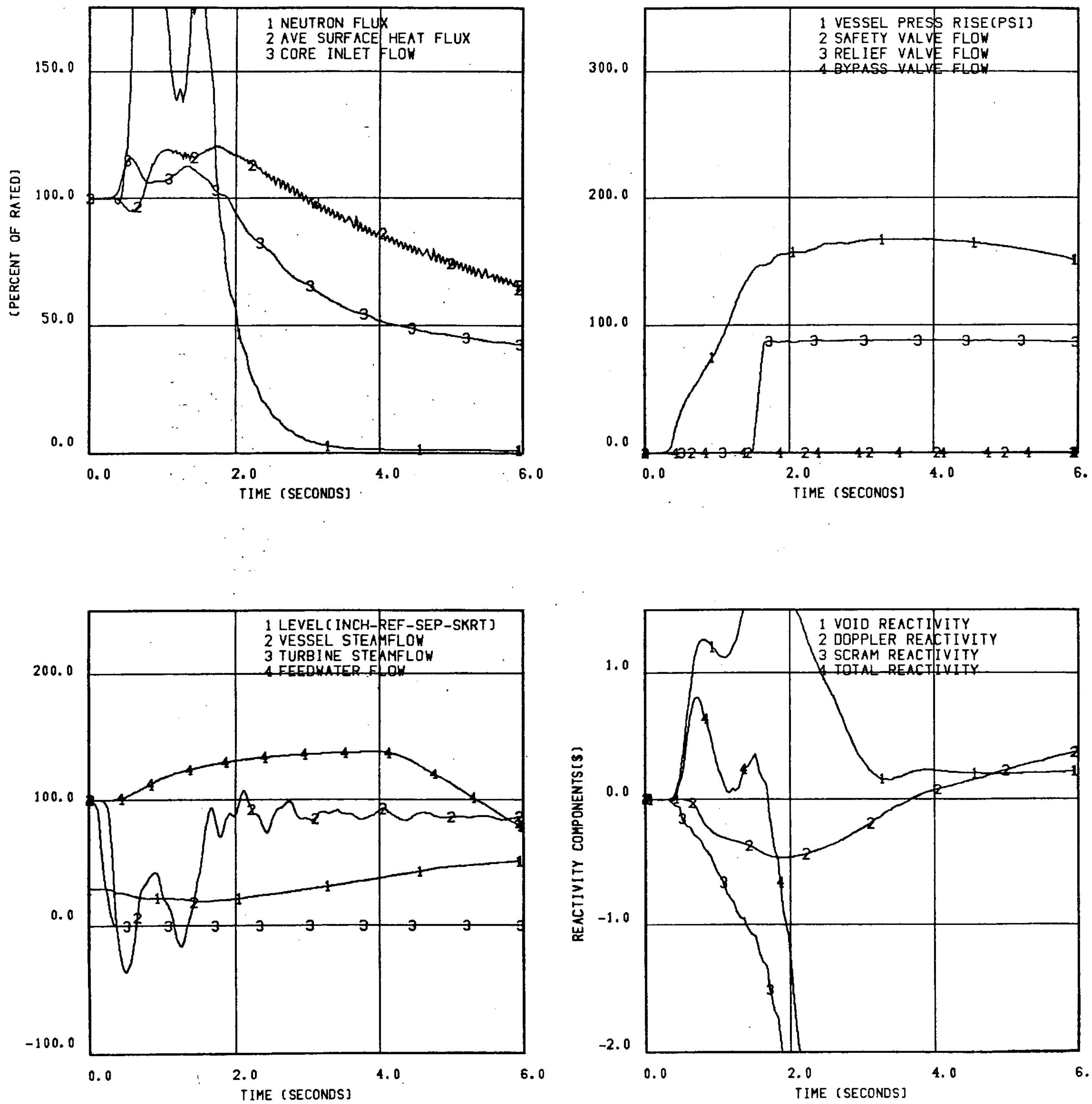


Figure 2. Plant Response to Generator Load Rejection Without Bypass

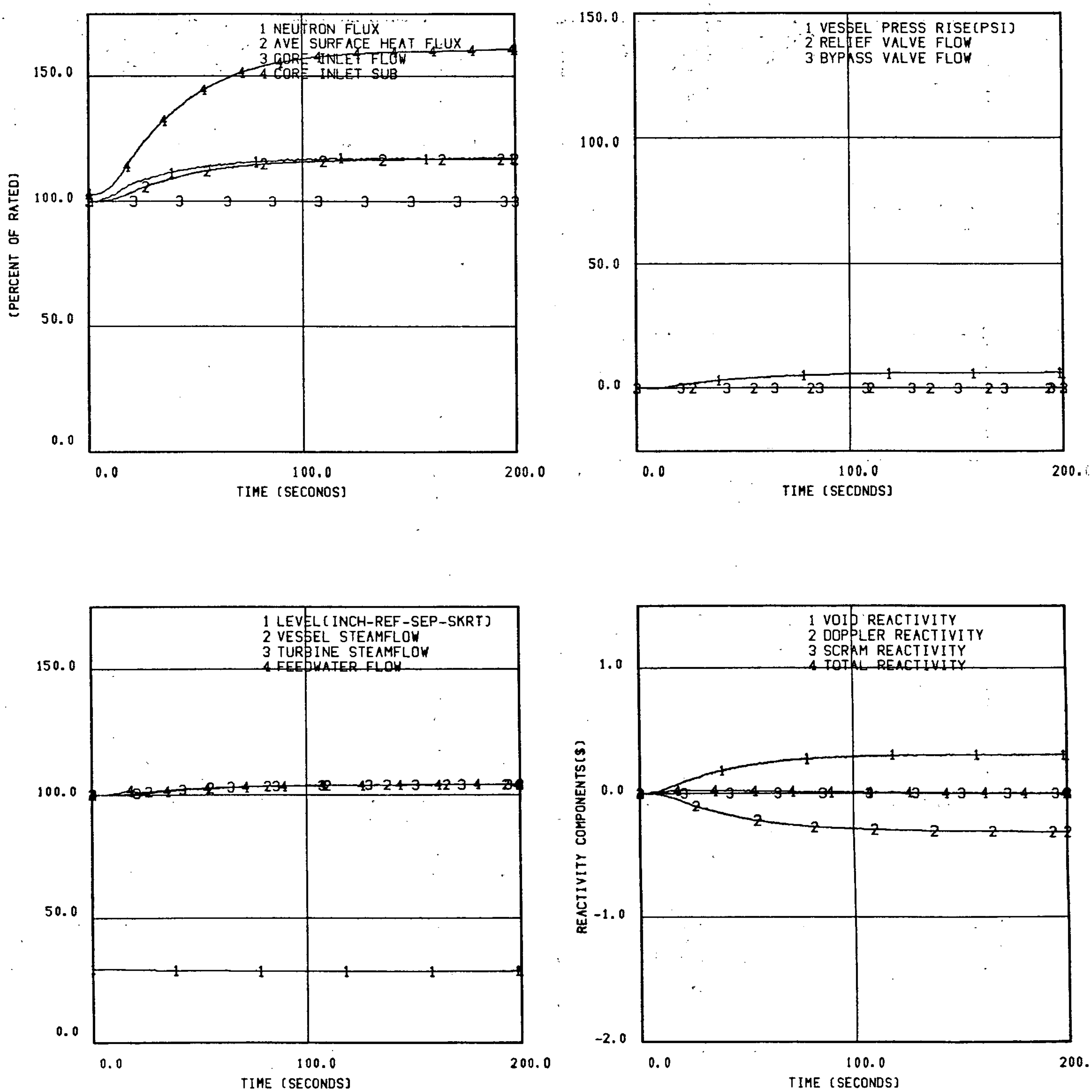


Figure 3. Plant Response to Loss of 100°F Feedwater Heating

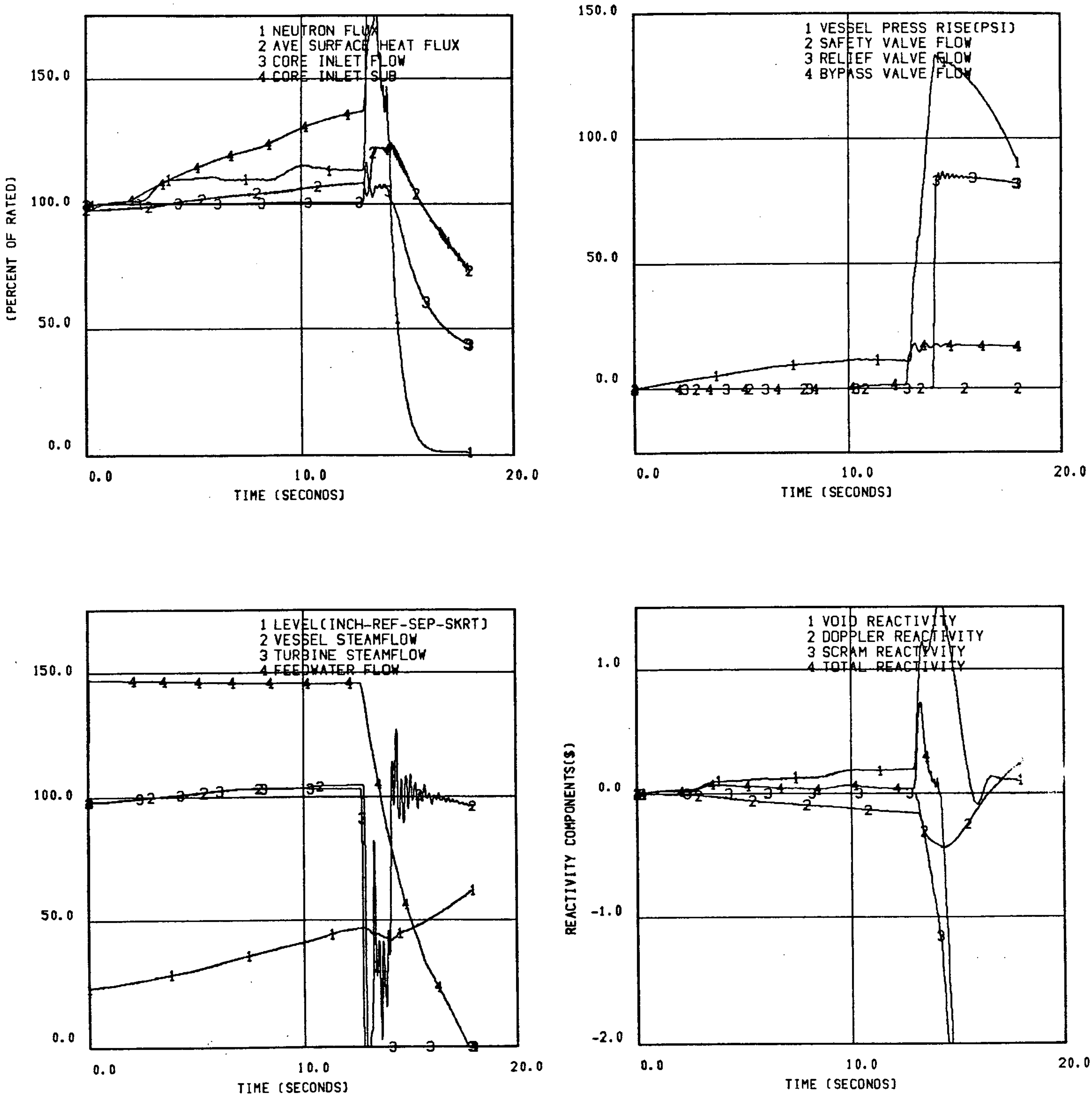


Figure 4. Plant Response to Feedwater Controller Failure (98% Power, 100% Flow)

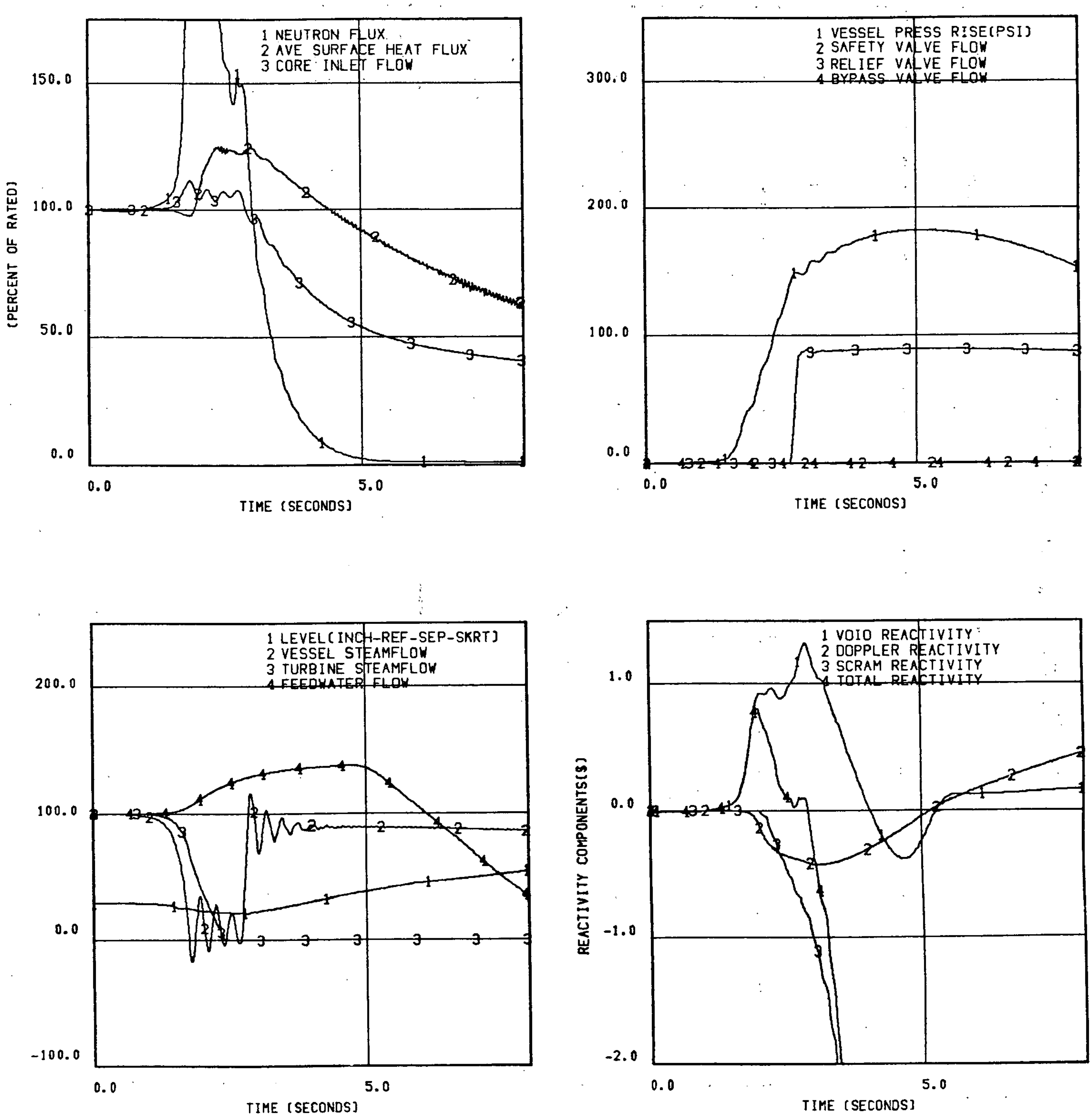


Figure 5. Plant Response to MSIV Closure (Flux Scram)

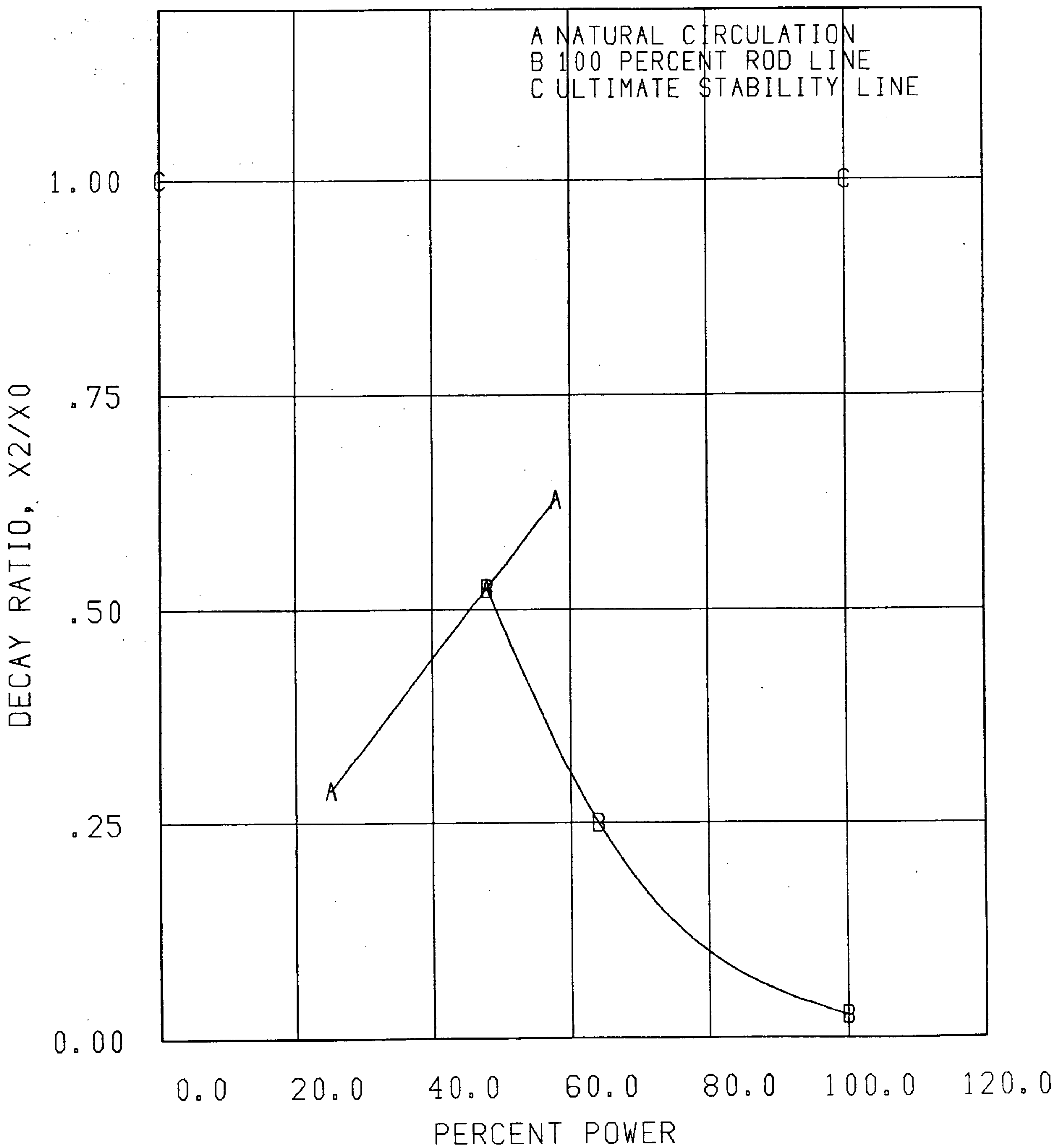


Figure 6. Reactor Core Decay Ratio

APPENDIX A
PLANT PARAMETER DIFFERENCES

GETAB Transient Analysis Initial Condition Parameters

Reactor Core Pressure	1038 psia
Inlet Enthalpy	524.2 Btu/lb

Fuel Channels

Not all channels were supplied by GE. At the direction of Northern States Power Company, the analyses were performed assuming that the performance characteristics of channels not supplied by GE are identical to the characteristics of channels supplied by GE.

APPENDIX B

FEEDWATER CONTROLLER FAILURE EVENT

The Feedwater Controller Failure (FWCF) event was analyzed at the 98% power/100% flow point. This point was found to be more conservative than the 100% power/100% flow point.

At the 100% power/100% flow initial condition, the safety/relief valve (S/RV) setpoint is exceeded by the initial pressurization wave after the turbine trip on high water level. This is unique to Monticello because the increased steam flow during the FWCF coupled with Monticello's small turbine bypass capacity (15%) results in an initial pressurization of the steam line higher than that typically calculated for other plants for a turbine trip initiated from rated conditions. This actuation of the S/RVs occurs early enough to reduce the severity of the FWCF event. However, when the transient is initiated at 98% power, the S/RVs are not actuated until much later in the transient, thus yielding more severe results.

APPENDIX C
CONTROL ROD DROP ANALYSIS

The cycle-specific control rod drop accident analysis has been discontinued for banked position withdrawal sequence (BPWS) plants based on the fact that in all cases the peak fuel enthalpy from a control rod drop accident would be much less than the 280 cal/gm limit. This change in procedures was reported and justified in Reference C-1. Reference C-2 indicates this change is acceptable to the NRC.

REFERENCES

C-1. Letter, R. E. Engel (GE) to D. B. Vassallo (NRC), "Control Rod Drop Accident," February 24, 1982.

C-2. NRC Memo, L. S. Rubenstein to G. C. Lainas, "Changes in GE Analysis of the Control Rod Drop Accident for Plant Reloads (TACS-48058)," February 15, 1983.

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