Y1003J01A39
CLASS I
MAY 1982

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR MONTICELLO NUCLEAR GENERATING PLANT RELOAD 9 (CYCLE 10)

Y1003J01A39
Revision 0
Class I
May 1982

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
MONTICELLO NUCLEAR GENERATING PLANT
RELOAD 9 (CYCLE 10)

Prepared: C L Hilf
C. L. Hilf
Reload Fuel Licensing

Verified:
J. L. Rash
Reload Fuel Licensing

Approved: R E Engel
R. E. Engel, Manager
Reload Fuel Licensing

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

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

This report was prepared by General Electric solely for Northern States Power Company (NSP) for NSP's use with the U. S. Nuclear Regulatory Commission (USNRC) for amending NSP's operating license of the Monticello Nuclear Generating Plant. The information contained in this report is believed by General Electric to be an accurate and true representation of the facts known, obtained or provided to General Electric at the time this report was prepared.

The only undertakings of the General Electric Company respecting information in this document are contained in the contract between Northern States Power Company and General Electric Company for nuclear fuel and related services for the nuclear system for Monticello Nuclear Generating Plant, dated December 4, 1967, and nothing contained in this document shall be construed as changing said contract. The use of this information except as defined by said contract, or for any purpose other than that for which it is intended, is not authorized; and with respect to any such unauthorized use, neither General Electric nor any of the contributors to this document makes any representation or warranty (express or implied) as to the completeness, accuracy or usefulness of the information contained in this document or that such use of such information may not infringe privately owned rights; nor do they assume any responsibility for liability or damage of any kind which may result from such use of such information.

1. PLANT-UNIQUE ITEMS (1.0)*

Plant Parameter Changes: Appendix A
    Safety/Relief Valves
    Channels
Feedwater Controller Failure Event: Appendix B
Generic, Rod Withdrawal Error Analysis - Section 10.
ATWS RPT assumed in Transient Analysis

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

Fuel Type	Cycle Loaded	Number	Number Drilled
Irradiated			
8DB250	4	16	0
8DB219L	5	52	0
8DB262	6	20	0
8DRB282	7	60	60
8DRB265L	7	48	48
P8DRB282	8	56	56
P8DRB265L	8	44	44
P8DRB265L	9	40	40
P8DRB284LB	9	44	44
New			
P8DRB265L	10	56	56
P8DRB284LB	10	48	48
Total		484	396

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	17784 MWd/ST
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	17505 MWd/ST
Assumed reload cycle core average exposure at end of cycle:	16953 MWd/ST
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)

Beginning of Cycle, k_{eff}	
Uncontrolled	1.113
Fully Controlled	0.958
Strongest Control Rod Out	0.990
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, ΔK	0.0

---

*( ) Refers to Area of Discussion in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-4, January 1982.

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20^{o}C, Xenon Free)
600	0.032

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

(REDY Events Only)

	EOC10
Void Fraction (%)	37.2
Average Fuel Temperature (^{o}F)	1158
Void Coefficient N/A* (¢/% RG)	-6.61/ -8.26
Doppler Coefficient N/A (¢/^{o}F)	-0.224/ -0.213
Scram Worth N/A ($)	-46.31/ -37.05

---

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

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
BOC10 to EOC10							
P8x8R	1.20	1.67	1.57	1.051	5.632	98.6	1.38
8x8R	1.20	1.70	1.57	1.051	5.722	97.7	1.36
8x8	1.22	1.56	1.57	1.098	5.272	98.0	1.35

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

Transient	Flux (%NBR)	Q/A (%NBR)	ΔCPR P8x8R	ΔCPR 8x8R	ΔCPR 8x8	Figure
Exposure: BOC10 to EOC10 Load Rejection W/O Bypass	596	122	0.31	0.29	0.28	2
Exposure: BOC10 to EOC10 Loss of Feedwater Heater	116	116	0.14	0.14	0.13	3
Exposure: BOC10 to EOC10 Feedwater Controller Failure*	449	120	0.29	0.27	0.27	4

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

(Generic Bounding Analysis Results)

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

Set point selected is : 108

11. CYCLE MCPR VALUES (S.2.2)

Nonpressurization Events:
Exposure Range: BOC10 to EOC10

	P8x8R	8x8R	8x8
Loss of Feedwater Heater	1.21	1.21	1.20
Fuel Loading Error	1.32	----	----
Rod Withdrawal Error	1.35	1.35	1.35

Pressurization Events
Exposure Range: BOC10 to EOC10

	Option A			Option B		
	P8x8R	8x8R	8x8	P8x8R	8x8R	8x8
Load Rejection W/O Bypass	1.44	1.42	1.41	1.39	1.37	1.36
Feedwater Controller Failure	1.42	1.40	1.40	1.33	1.31	1.31

---

*See Appendix B.

12. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	P_{sl} (psig)	P_V (psig)	Plant Response
MSIV Closure (Flux Scram)	1200	1222	Figure 5

13. 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: 0.62
Channel Hydrodynamic Performance Decay Ratio, x_2/x_0

Channel Type	
8x8R/P8x8R	0.19
8x8	0.23

14. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: No

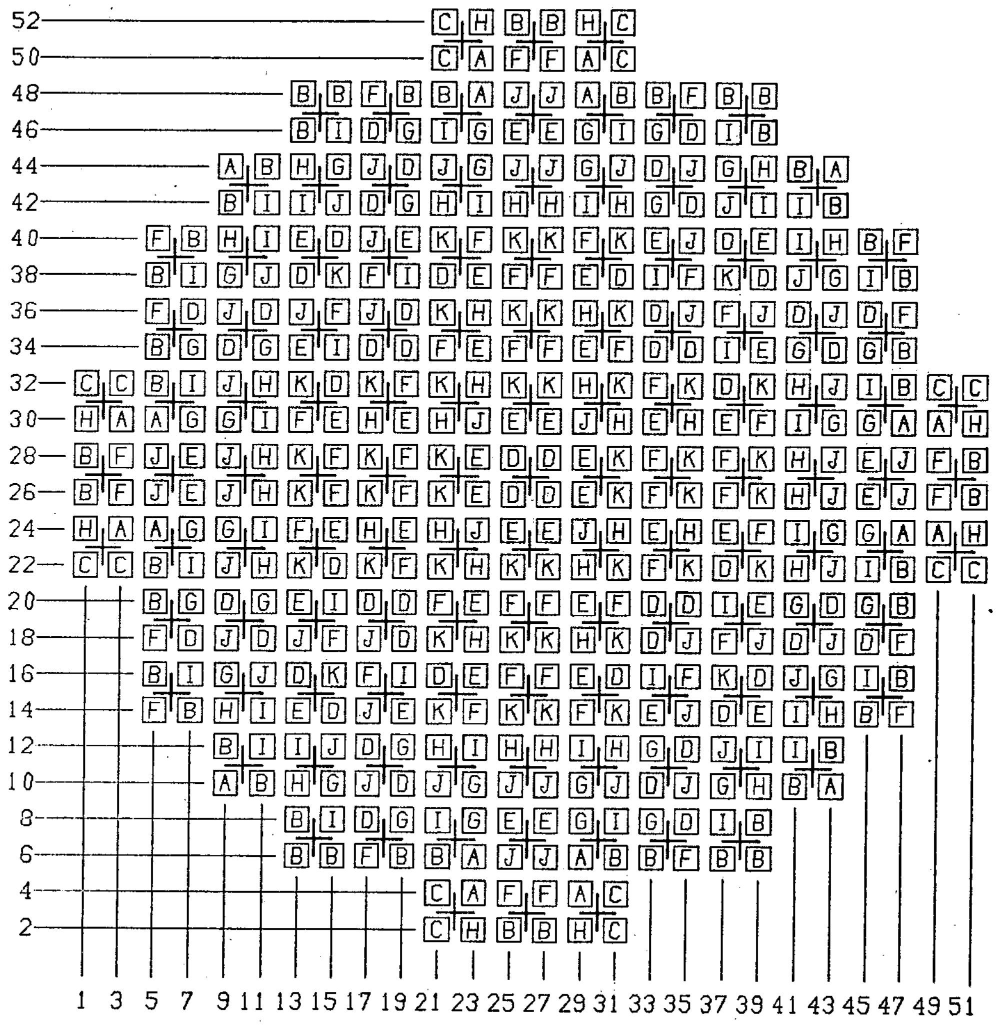
Event	Initial MCPR	Resulting MCPR
Misoriented	1.32	1.07

15. CONTROL ROD DROP ANALYSIS RESULT (S.2.5.1)

Maximum Incremental Control Rod Worth: 0.38% Δk

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

See "Loss-of-Coolant Analysis," NEDO-24050-1.



	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51
52											C	H	B	B	H	C										
50											C	A	F	F	A	C										
48							B	B	F	B	B	A	J	J	A	B	B	F	B	B						
46							B	I	D	G	I	G	E	E	G	I	G	D	I	B						
44					A	B	H	G	J	D	J	G	J	J	G	J	D	J	G	H	B	A				
42					B	I	I	J	D	G	H	I	H	H	I	H	G	D	J	I	I	B				
40			F	B	H	I	E	D	J	E	K	F	K	K	F	K	E	J	D	E	I	H	B	F		
38			B	I	G	J	D	K	F	I	D	E	F	F	E	D	I	F	K	D	J	G	I	B		
36			F	D	J	D	J	F	J	D	K	H	K	K	H	K	D	J	F	J	D	J	D	F		
34			B	G	D	G	E	I	D	D	F	E	F	F	E	F	D	D	I	E	G	D	G	B		
32	C	C	B	I	J	H	K	D	K	F	K	H	K	K	H	K	F	K	D	K	H	J	I	B	C	C
30	H	A	A	G	G	I	F	E	H	E	H	J	E	E	J	H	E	H	E	F	I	G	G	A	A	H
28	B	F	J	E	J	H	K	F	K	F	K	E	D	D	E	K	F	K	F	K	H	J	E	J	F	B
26	B	F	J	E	J	H	K	F	K	F	K	E	D	D	E	K	F	K	F	K	H	J	E	J	F	B
24	H	A	A	G	G	I	F	E	H	E	H	J	E	E	J	H	E	H	E	F	I	G	G	A	A	H
22	C	C	B	I	J	H	K	D	K	F	K	H	K	K	H	K	F	K	D	K	H	J	I	B	C	C
20			B	G	D	G	E	I	D	D	F	E	F	F	E	F	D	D	I	E	G	D	G	B		
18			F	D	J	D	J	F	J	D	K	H	K	K	H	K	D	J	F	J	D	J	D	F		
16			B	I	G	J	D	K	F	I	D	E	F	F	E	D	I	F	K	D	J	G	I	B		
14			F	B	H	I	E	D	J	E	K	F	K	K	F	K	E	J	D	E	I	H	B	F		
12					B	I	I	J	D	G	H	I	H	H	I	H	G	D	J	I	I	B				
10					A	B	H	G	J	D	J	G	J	J	G	J	D	J	G	H	B	A				
8							B	I	D	G	I	G	E	E	G	I	G	D	I	B						
6							B	B	F	B	B	A	J	J	A	B	B	F	B	B						
4											C	A	F	F	A	C										
2											C	H	B	B	H	C										

FUEL TYPE

A = 8DB262
B = 8DB219L
C = 8DB250
D = P8DRB282
E = P8DRB265L
F = 8DRB282
G = P8DRB265L
H = 8DRB265L
I = P8DRB284LB
J = P8DRB265L
K = P8DRB284LB

Figure 1. Reference Core Loading Pattern

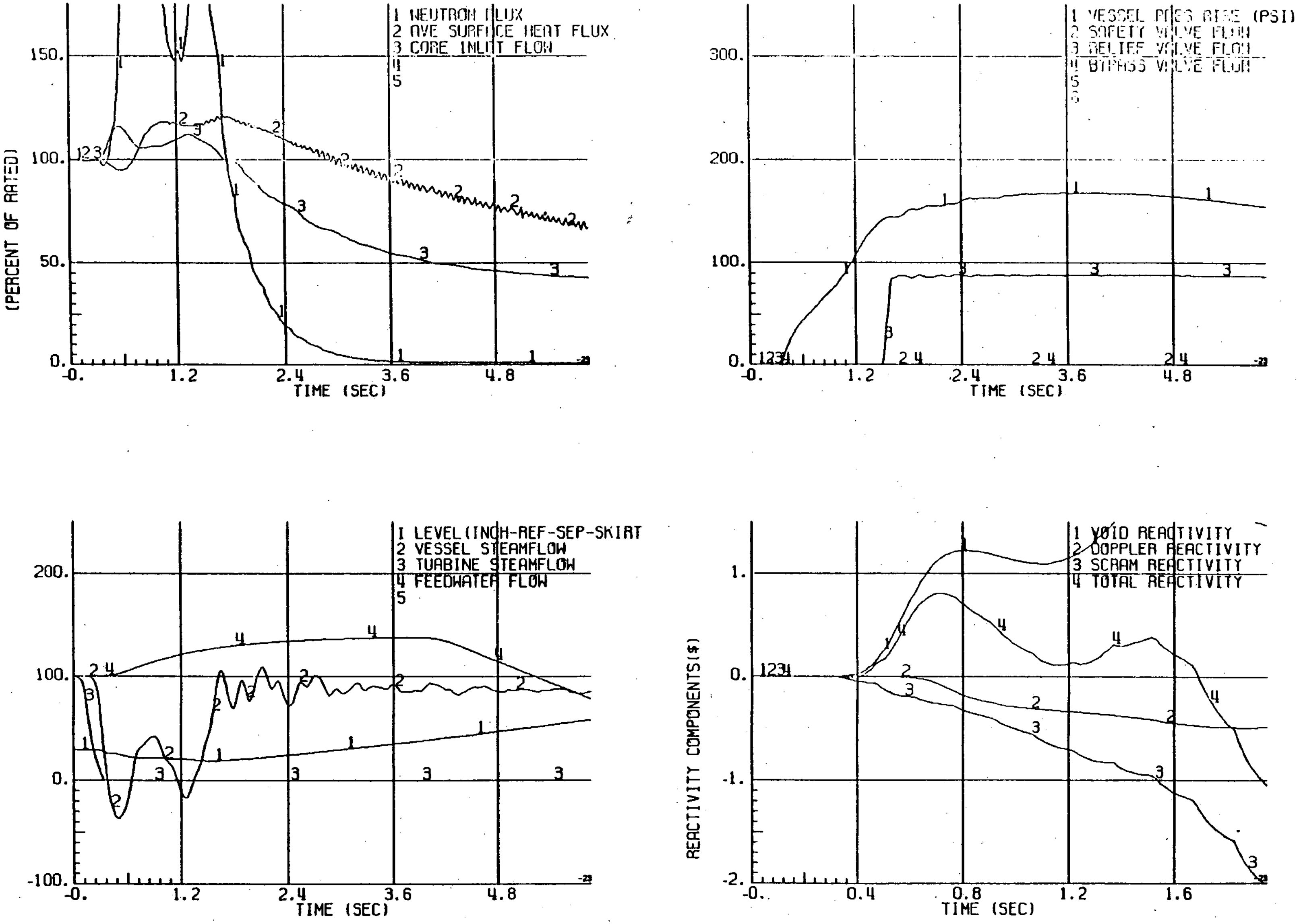


Figure 2. Plant Response to Generator Load Rejection Without Bypass

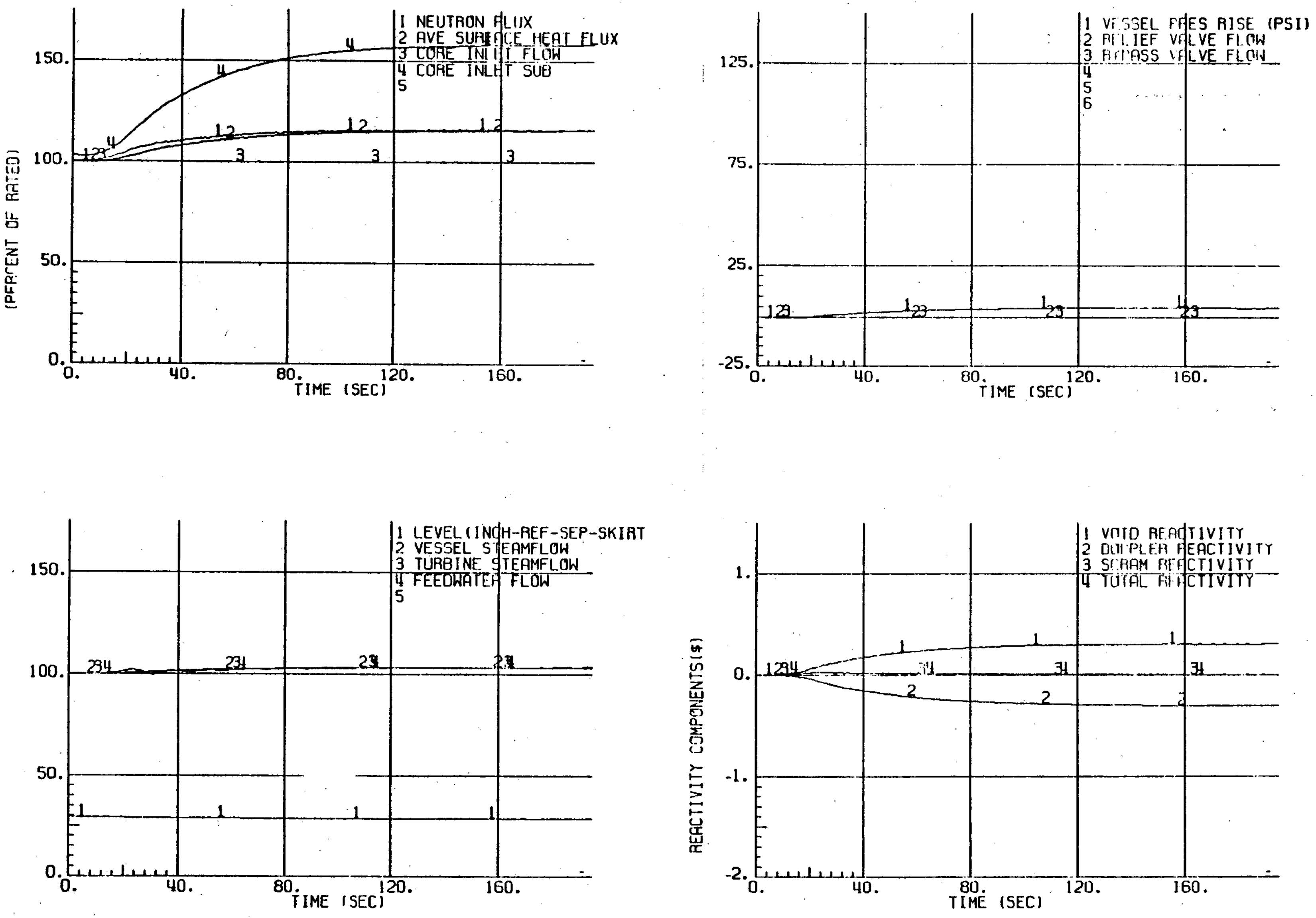


Figure 3. Plant Response to Loss of 100°F Feedwater Heating

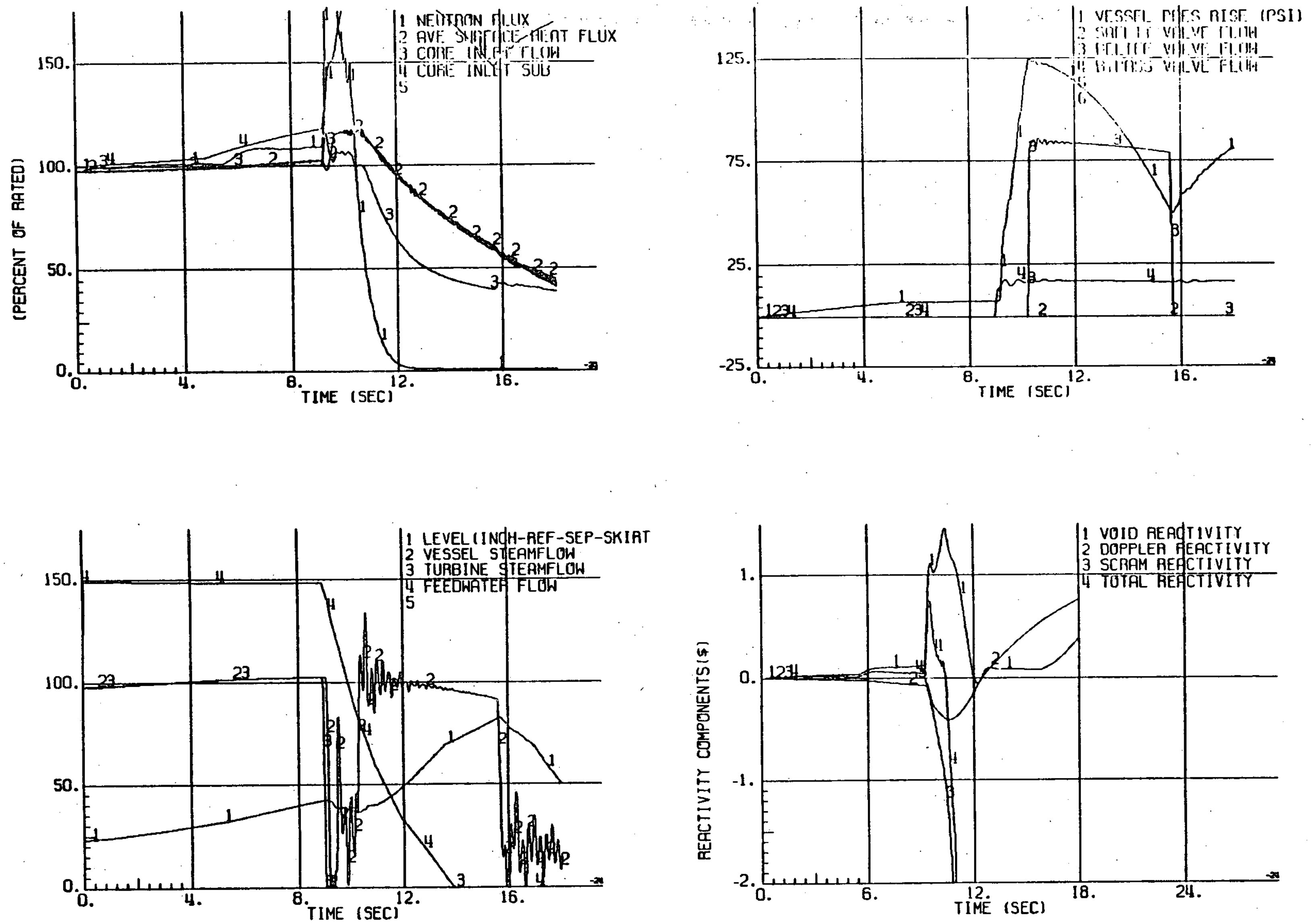


Figure 4. Plant Response to Feedwater Controller Failure
(98% Power, 100% Flow)

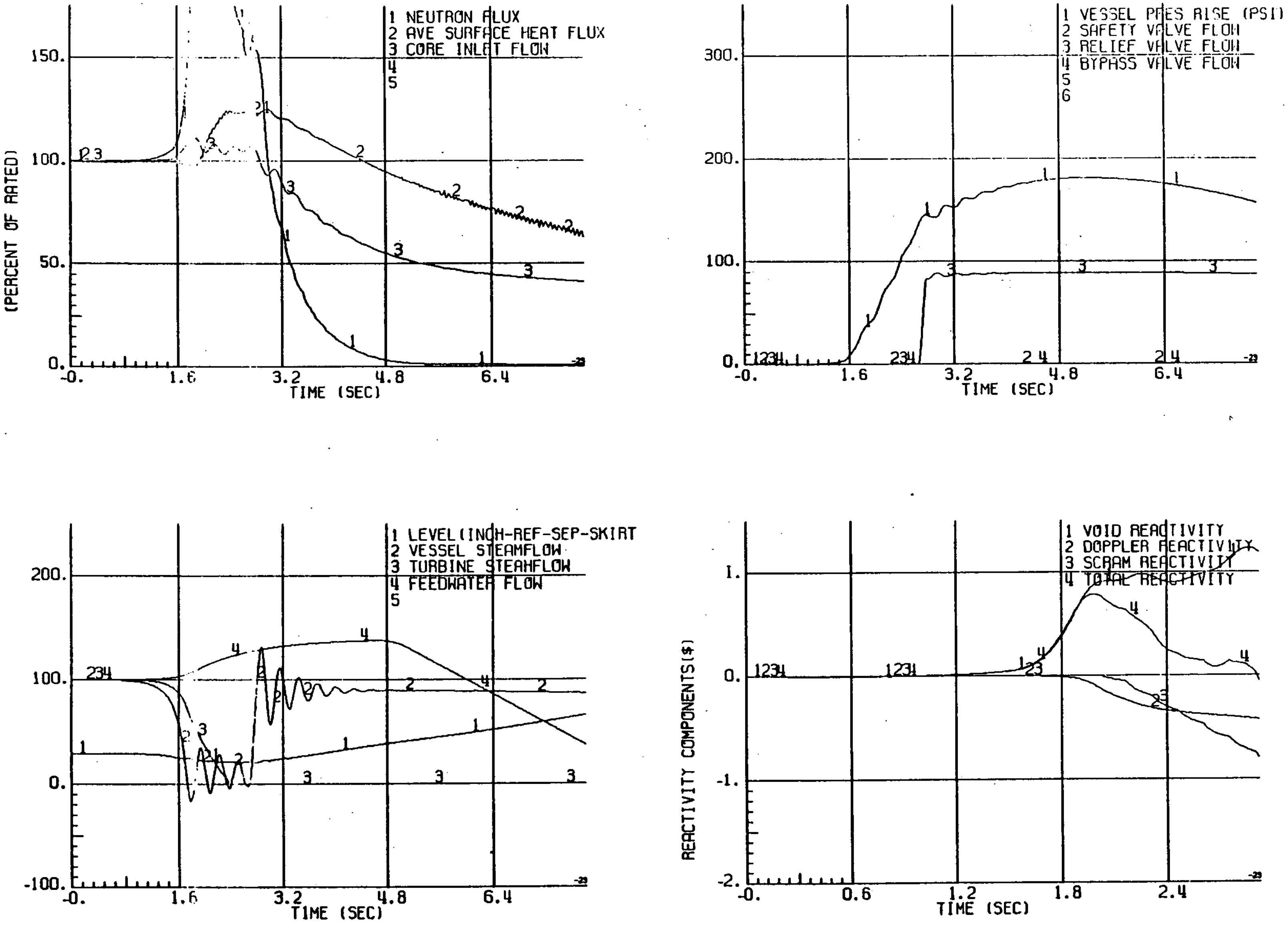


Figure 5. Plant Response to MSIV Closure (Flux Scram)

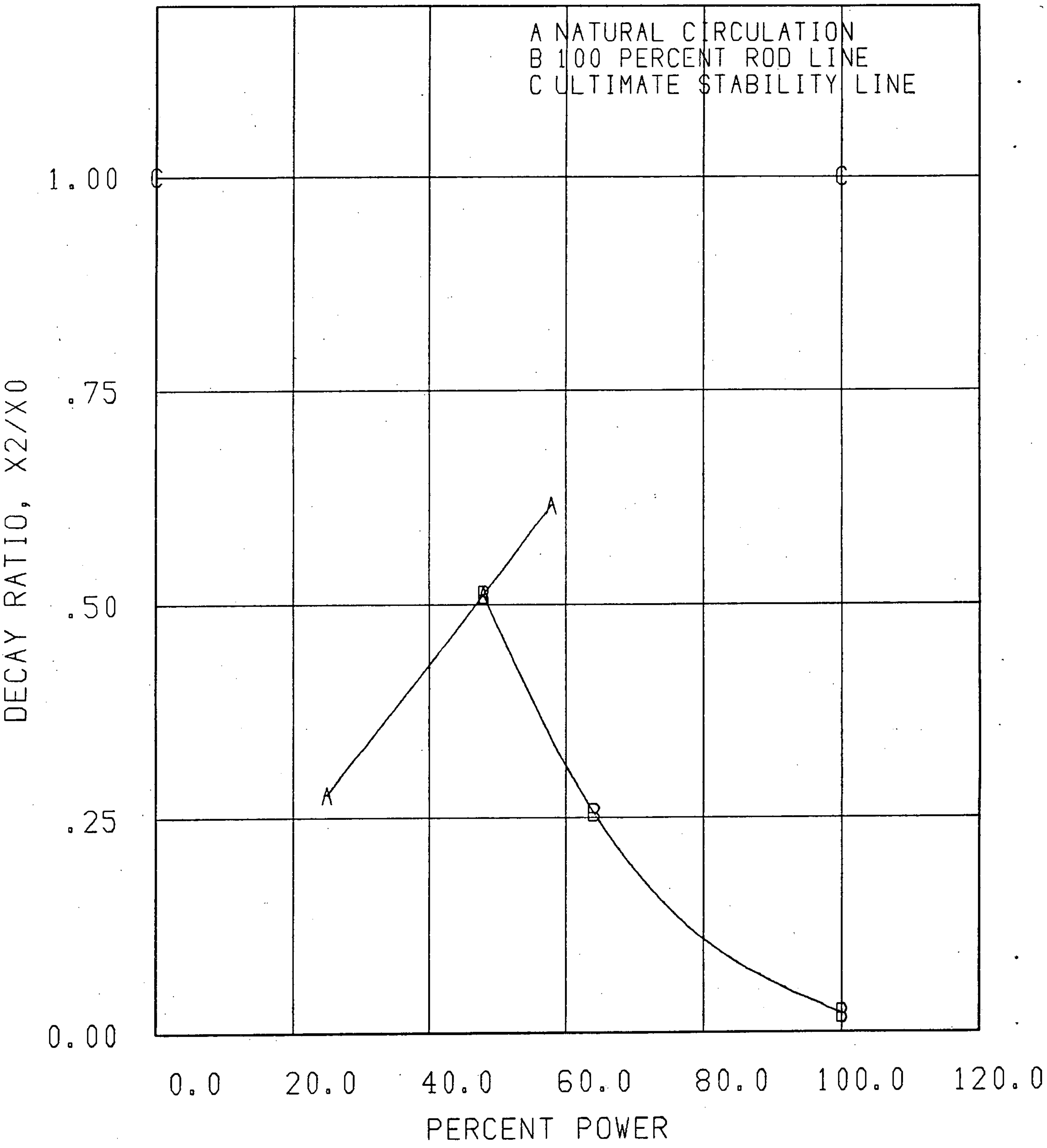


Figure 6. Reactor Core Decay Ratio

APPENDIX A

PLANT PARAMETER DIFFERENCES

Safety/Relief Valves

Analysis used seven S/R valves
Lowest set point = 1108 + 1% psig
Capacity at set point = 82.8%

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. Northern States Power, therefore, assumes all responsibility for justifying this conclusion.

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) set point 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.

GENERAL ELECTRIC