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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR MILLSTONE UNIT 1 RELOAD 8

GENERAL ELECTRIC

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FOR
MILLSTONE UNIT 1
RELOAD 8

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GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

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1. PLANT-UNIQUE ITEMS (1.0)*

Control Rod Drop Analysis	Appendix A
Transient Analysis Initial Conditions:	Appendix B
GETAB Analysis Initial Conditions:	Appendix C

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

Fuel Type	Cycle Loaded	Number	Number Drilled
Irradiated			
8DB262	3	1	0
8DB274H	6	71	0
8DRB265H	7	48	48
8DRB265L	7	100	100
P8DRB265H	8	40	40
P8DRB282	8	128	128
New			
P8DRB282	9	72	72
P8DRB283H	9	120	120
		580	508

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	17580 MWd/ST
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	17580 MWd/ST
Assumed reload cycle core average exposure at end of cycle:	18036 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	0.955
Strongest Control Rod Out	0.982
R, Maximum Increase on Cold Core Reactivity with Exposure into Cycle, ΔK	0.008

*( ) 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°C, Xenon Free)
660	0.043

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

(REDY Events Only)

	EOC9
Void Fraction (%)	37.0
Average Fuel Temperature (°F)	1171
Void Coefficient N/A* (¢/% RG)	-6.10/ -7.62
Doppler Coefficient N/A (¢/°F)	-0.229/ -0.218
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
BOC9 to EOC9							
P8x8R	1.20	1.62	1.40	1.051	5.469	100.5	1.42
8x8R	1.20	1.65	1.40	1.051	5.590	99.2	1.38
8x8	1.22	1.52	1.40	1.098	5.150	102.3	1.38

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: BOC9 to EOC9 Load Rejection w/o Bypass	613	127	0.35	0.32	0.32	2
Exposure: BOC9 to EOC9 Loss of Feedwater Heater	115	114	0.12	0.12	0.12	3
Exposure: BOC9 to EOC9 Feedwater Controller Failure	109	108	0.07	0.07	0.07	4

10. 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

Set point selected is : 107

11. CYCLE MCPR VALUES (S.2.2)

Nonpressurization Events:
Exposure Range: BOC9 to EOC9

	P8x8R	8x8R	8x8
Loss of Feedwater Heater	1.19	1.19	1.19
Fuel Loading Error	1.25	----	----
Rod Withdrawal Error	1.29	1.29	1.29

Pressurization Events
Exposure Range: BOC9 to EOC9

	Option A			Option B		
	P8x8R	8x8R	8x8	P8x8R	8x8R	8x8
Load Rejection w/o Bypass	1.48	1.45	1.45	1.43	1.40	1.40
Feedwater Controller Failure	1.19	1.19	1.19	1.12	1.12	1.12

12. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1270	1285	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: 0.60
Channel Hydrodynamic Performance Decay Ratio, x_2/x_0

Channel Type

8x8R/P8x8R	0.21
8x8	0.26

14. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes
Includes 2.2% Power Spiking Penalty: No

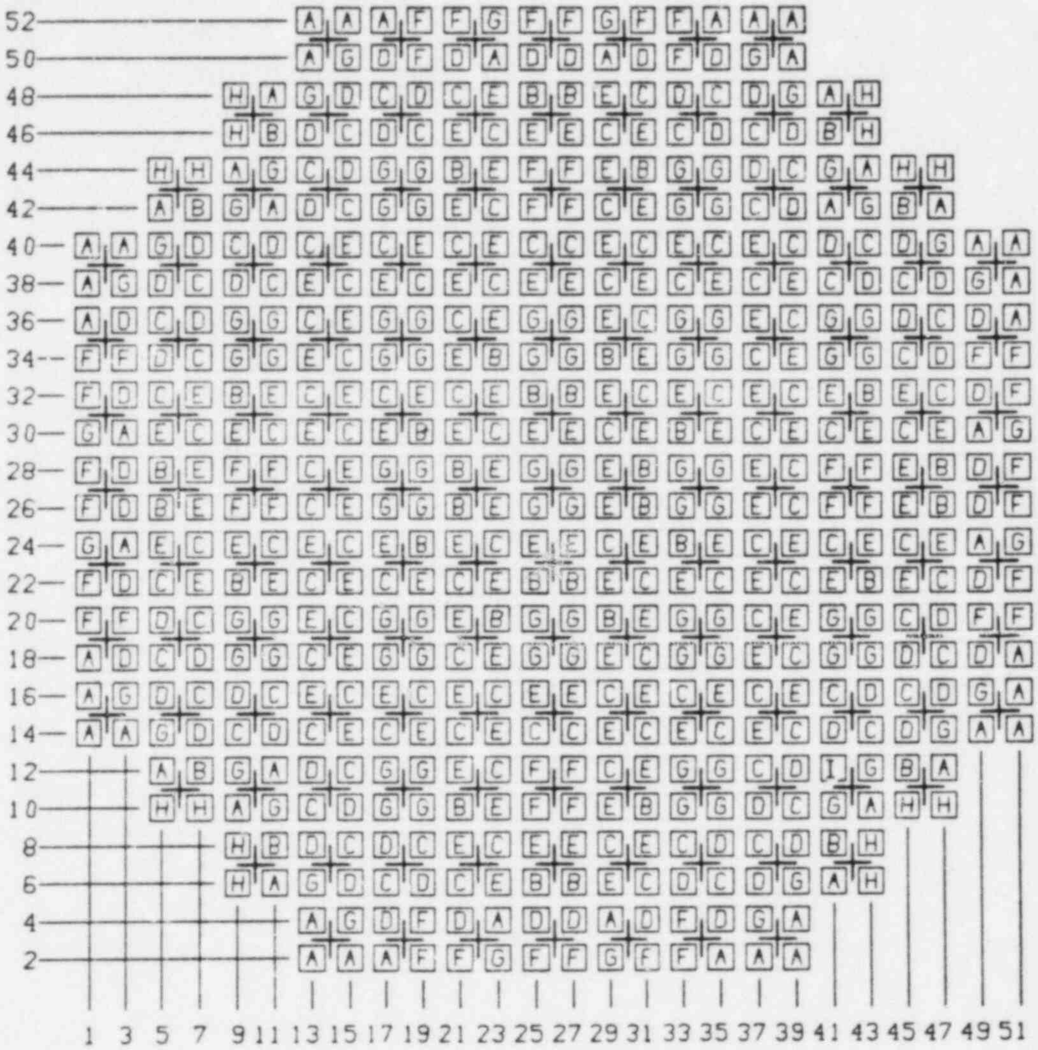
Event	Initial MCPR	Resulting MCPR	Resulting LHGR (kW/ft)
Misoriented	1.23	1.07	15.2

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

Deleted: See Appendix A

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

See "Loss-of-Coolant Analysis Report for Millstone Unit 1 Nuclear Power Station," General Electric Company, July 1980, (NEDO-24085-1, as amended).



FUEL TYPE	
A = 8DB274H	E = P8DRB283H
B = P8DRB265H	F = 8DRB265H
C = P8DRB282	G = 8DRB265L
D = P8DRB282	H = 8DB274H
	I = 8DB262

Figure 1. Reference Core Loading Pattern

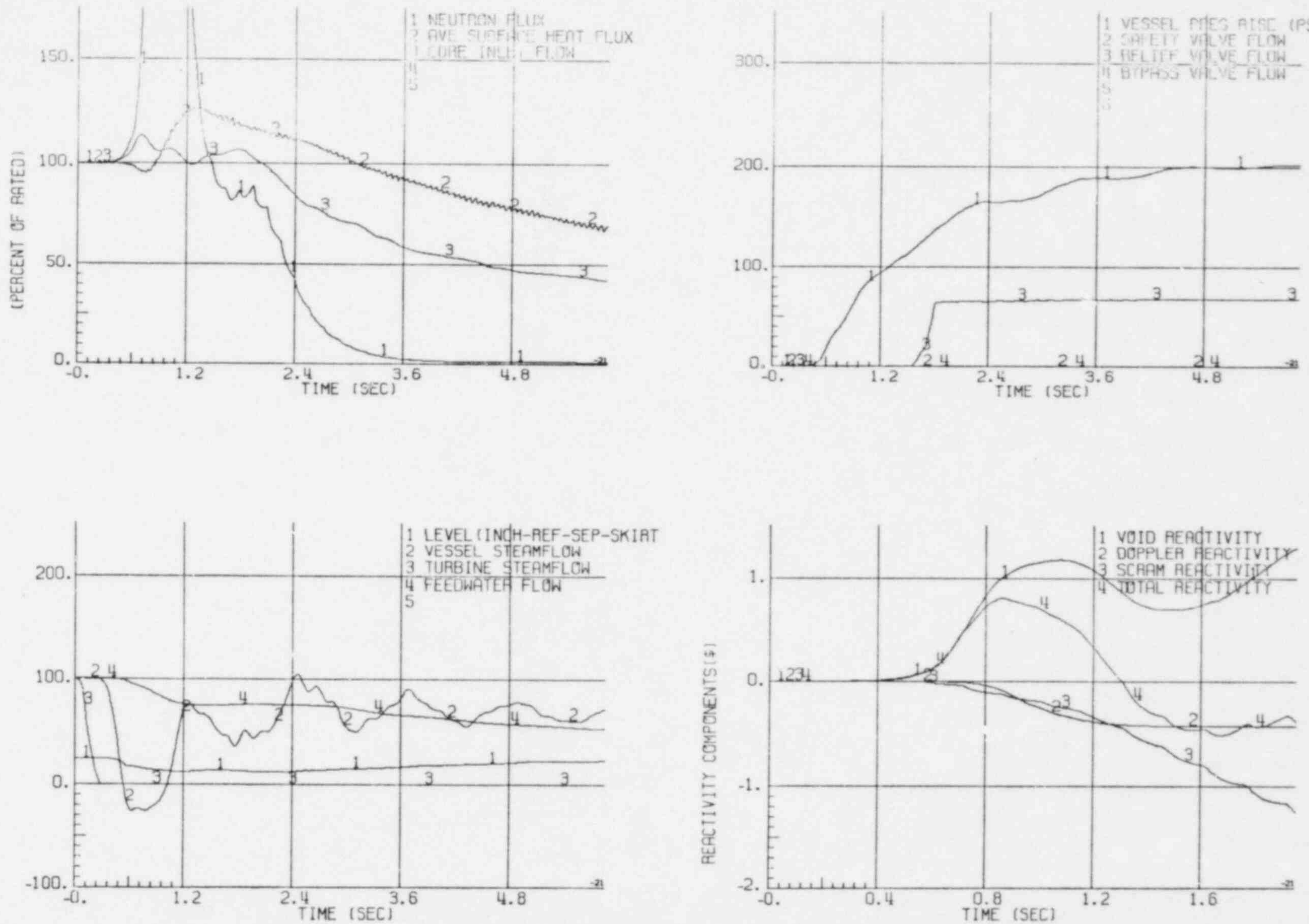


Figure 2. Plant Response to Generator Load Rejection Without Bypass, EOC9

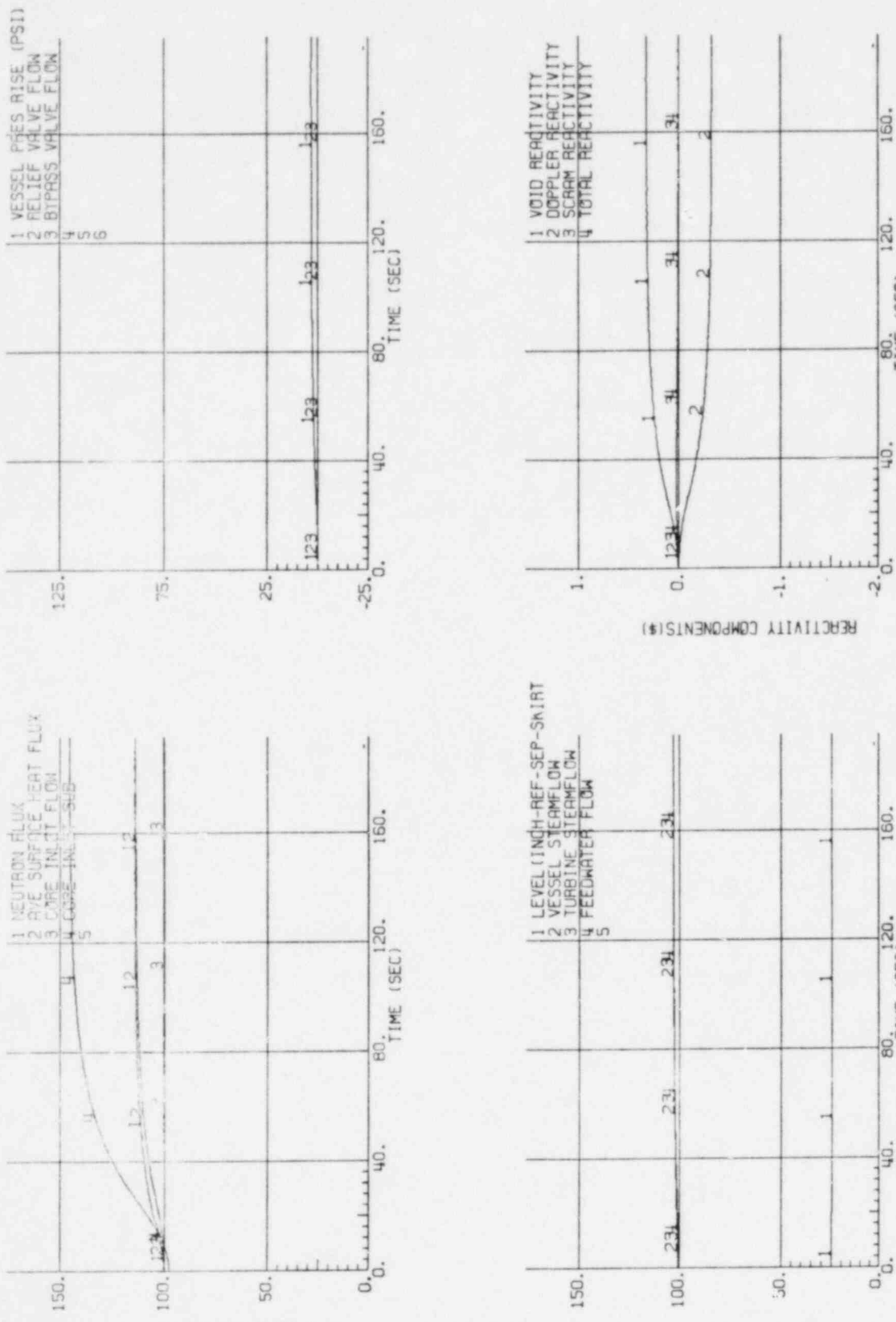


Figure 3. Plant Response to Loss of 100°F Feedwater Heating

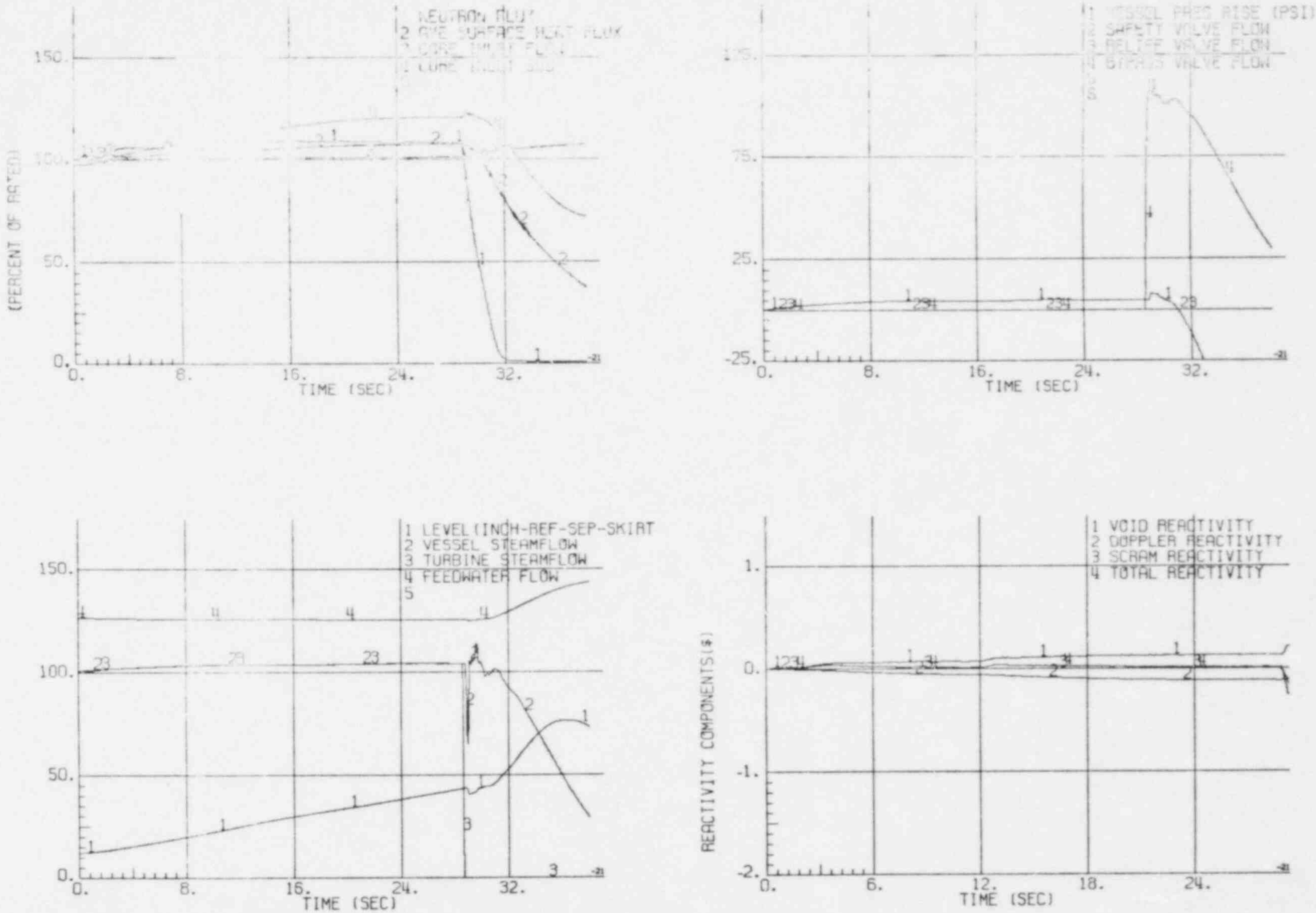


Figure 4. Plant Response to Feedwater Controller Failure, EOC9

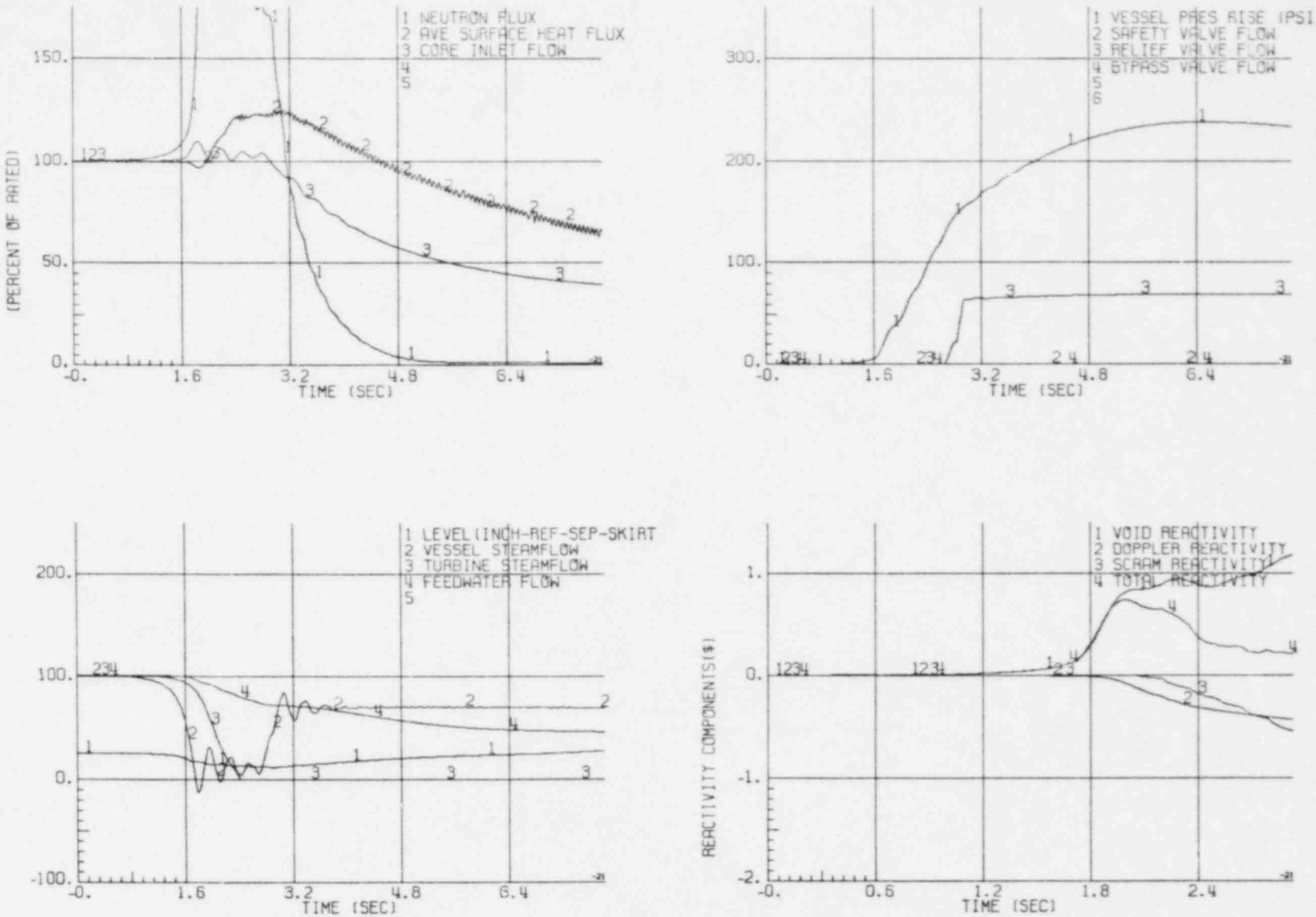


Figure 5. Plant Response to MSIV Closure (Flux Scram), EOC6

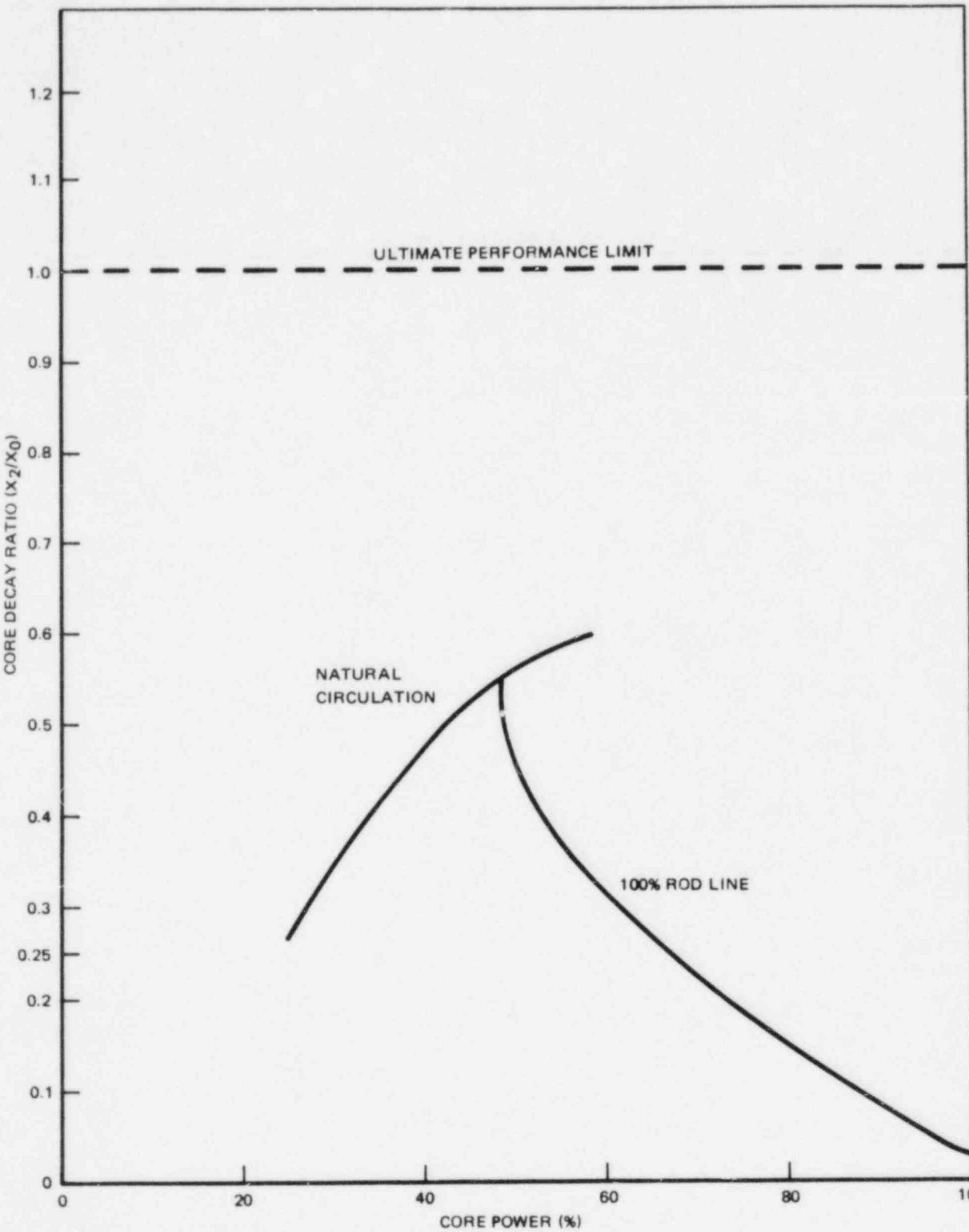


Figure 6. Reactor Core Decay Ratio versus Power

APPENDIX A

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 A-1.

Reference

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

APPENDIX B
TRANSIENT ANALYSIS INITIAL CONDITIONS

Rated Steam Flow	7.99×10^6 lb/hr
Turbine Pressure	979 psig

APPENDIX C
GETAB ANALYSIS INITIAL CONDITIONS

Reactor Core Pressure	1065 psia
Inlet Enthalpy	526.0 Btu/lb

Docket No. 50-245

Attachment No. 2

Changes to the

"Loss of Coolant Accident Analysis

Report for Millstone Unit 1 Nuclear Power Station"

NEDO-24085-1, dated July 1980

October 1982