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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR MONTICELLO NUCLEAR GENERATING PLANT RELOAD 8 (CYCLE 9)

GENERAL ELECTRIC

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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

MONTICELLO NUCLEAR GENERATING PLANT

RELOAD 8 (CYCLE 9)

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GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT
PLEASE READ CAREFULLY

1. PLANT-UNIQUE ITEMS (1.0)*

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GETAB Initial Conditions-	- see Appendix A
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2. RELOAD FUEL BUNDLES (1.0, 2.0, 3.3.1 AND 4.0)

	Fuel Type	Number	Number Drilled
Irradiated, Reload 2	8DB262	12	None
Irradiated, Reload 3	8DB250	28	None
Irradiated, Reload 4	8DB219L	40	None
Irradiated, Reload 5	8DB262	108	None
Irradiated, Reload 6	8DRB265L	52	52
Irradiated, Reload 6	8DRB282	60	60
Irradiated, Reload 7	P8DRB282	56	56
Irradiated, Reload 7	P8DRB265L	44	44
New	P8DRB265L	40	40
New	P8DRB284LB**	44	44
Total		484	296

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle: 16.57 GWd/st.

Assumed reload cycle core average exposure at end of cycle: 16.82 GWd/st.

Core loading pattern: Figure 1.

*( )refers to areas of discussion in "Generic Reload Fuel Application," NEDE-24011-P-A-1, August 1979.

**Letter MFN-196-80/REE-077-80, R.E. Engel (GE) to J.S. Berggren (NRC), "General Electric Co. Licensing Topical Report NEDE-24011-P-A-1, Generic Reload Fuel Application, Amendment 9", November 17, 1980.

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	1.1105
Fully Controlled	0.9547
Strongest Control Rod Out	0.9899
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)
600*	0.037

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

	EOC9
Void Coefficient N/A** (¢/% Rg)	-6.57/-8.22
Void Fraction (%)	37.14
Doppler Coefficient N/A (¢/°F)	-0.223/-0.212
Average Fuel Temperature (°F)	1171
Scram Worth N/A ($)	-37.05/-29.64
Scram Reactivity versus Time	See "Reactivity Components" on Figures 3, 4, and 5

*Previous documents have indicated a boron concentration of 900 ppm, but that concentration is the concentration injected into the core to produce a conservatively calculated concentration of 600 ppm due to imperfect mixing and dilution from water in the cooldown circuit. The 600 ppm boron concentration is sufficient to provide the 3% Δk subcritical condition.

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

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

	EOC9		
Exposure	8x8	8x8R	P8x8R
Peaking factor (local, radial, axial)	1.22, 1.52, 1.40	1.20, 1.67, 1.40	1.20, 1.63, 1.40
R-Factor	1.098	1.052	1.052
Bundle Power (MWt)	5.131	5.618	5.496
Bundle Flow (10^3 lb/hr)	104.2	97.2	97.9
Initial MCPR	1.42	1.42	1.46

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

None

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Core Flow (%)	ϕ (% NBR)	Q/A (% NBR)	P_{SL} (psig)	P_v (psig)	Nominal ΔCPR 8x8/8x8R/P8x8R	Plant Response
Generator Load Rejection without Bypass	EOC9	100	100	578.6	124.8	1207	1231	0.35/0.35/0.39	Figure 3
Loss of 100°F Feedwater Heater	--	100	100	118.1	116.5	1023	1067	0.15/0.16/0.16	Figure 4
Feedwater Controller Failure	EOC9	100	100	518.2	124.3	1169	1201	0.33/0.34/0.37	Figure 5

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

		ΔCPR	LHGR	
Rod Block Reading	Rod Position (Feet Withdrawn)	8x8/8x8R and P8x8R	8x8/8x8R and P8x8R	Limiting Rod Pattern
104	3.0	0.15/0.15	13.5/15.8	
105	3.0	0.15/0.15	13.5/15.8	Figure 6
106	3.5	0.20/0.19	14.6/17.2	
107	3.5	0.20/0.19	14.6/17.2	
108*	4.0	0.27/0.23	15.3/18.2	
109	4.5	0.32/0.25	15.6/18.6	
110	4.5	0.32/0.25	15.6/18.6	

*Indicates setpoint selected.

11. OPERATING MCPR LIMIT (5.2, APPENDIX C)

	8x8	8x8R	P8x8R
Load Rejection without Bypass (EOC9)			
Option A	1.48	1.48	1.52
Option B	1.43	1.43	1.47
Feedwater Controller Failure (EOC9)			
Option A	1.46	1.47	1.50
Option B	1.37	1.38	1.41
Loss of Feedwater Heating (BOC-EOC9)	1.22	1.23	1.23
Rod Withdrawal Error (BOC-EOC9)	1.34	1.30	1.30
Fuel Loading Error (BOC-EOC9)	1.48	1.48	1.48

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	Power (%)	Core Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	100	100	1223	1244	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio, x_2/x_o (Natural Circulation-100% Rod Line) 0.55

Channel Hydrodynamic Performance

	Decay Ratio (Natural Circulation-100% Rod Line)
8x8 Channel	0.12
P8x8R/8x8R Channel	0.07

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

(See Loss-of-Coolant Analysis, NEDO-24050-1)

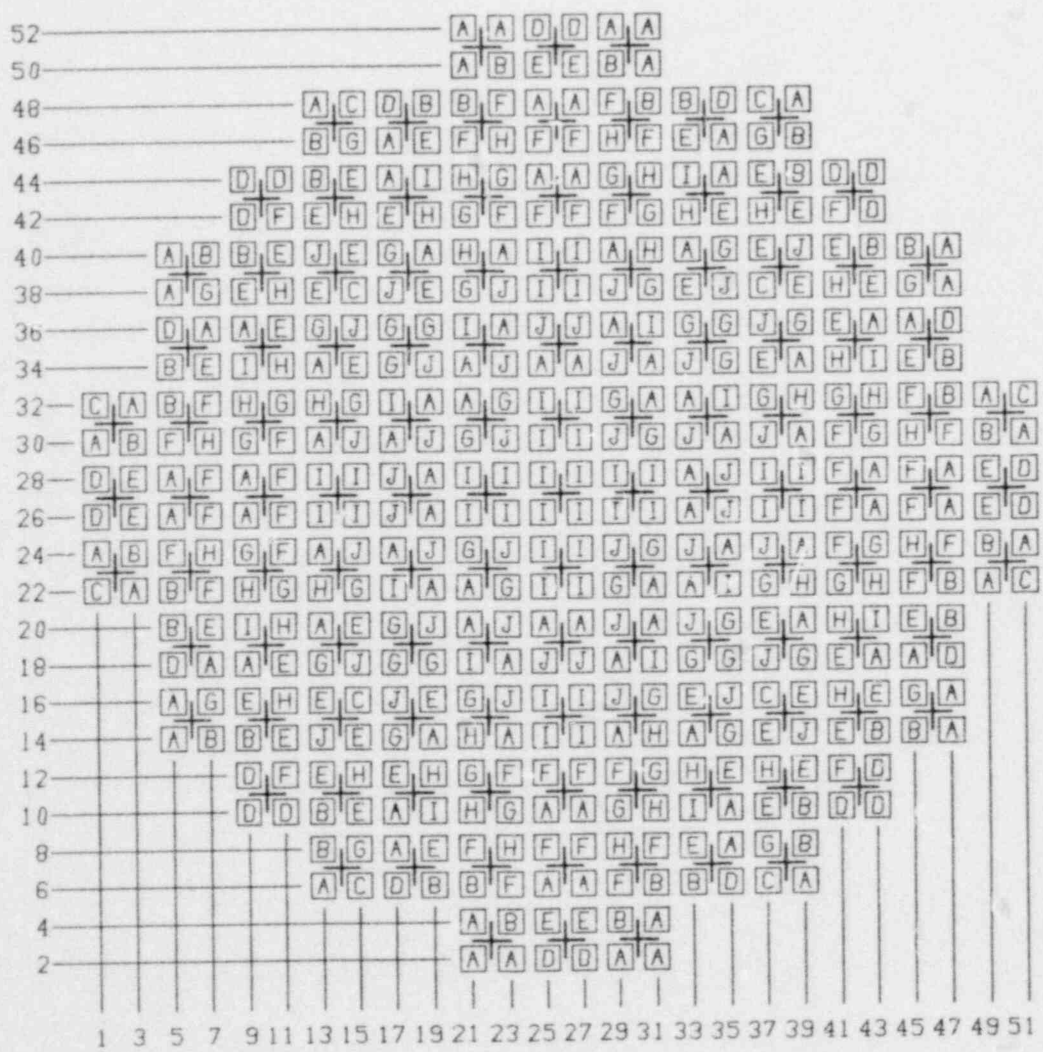
15. LOADING ERROR RESULTS (5.5.4)

Limiting event: Mislocated Bundle

MCPR: 1.07

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Maximum incremental control rod worth: 0.52% Δk



FUEL TYPE	
A = 8DB262, RELOAD 5	F = P8DRB265L, RELOAD 7
B = 8DB219L, RELOAD 4	G = 8DRB282, RELOAD 6
C = 8DB262, RELOAD 2	H = P8DRB265L, NEW
D = 8DB250, RELOAD 3	I = 8DRB265L, RELOAD 6
E = P8DRB282, RELOAD 7	J = P8DRB284LB, NEW

Figure 1. Reference Core Loading Pattern

This figure has been deleted because of the use of the ODYN code.

Scram reactivity versus time is indicated on the figures provided for the transients considered (Figures 3, 4, and 5).

Figure 2. Scram Reactivity and Control Rod Drive Specifications

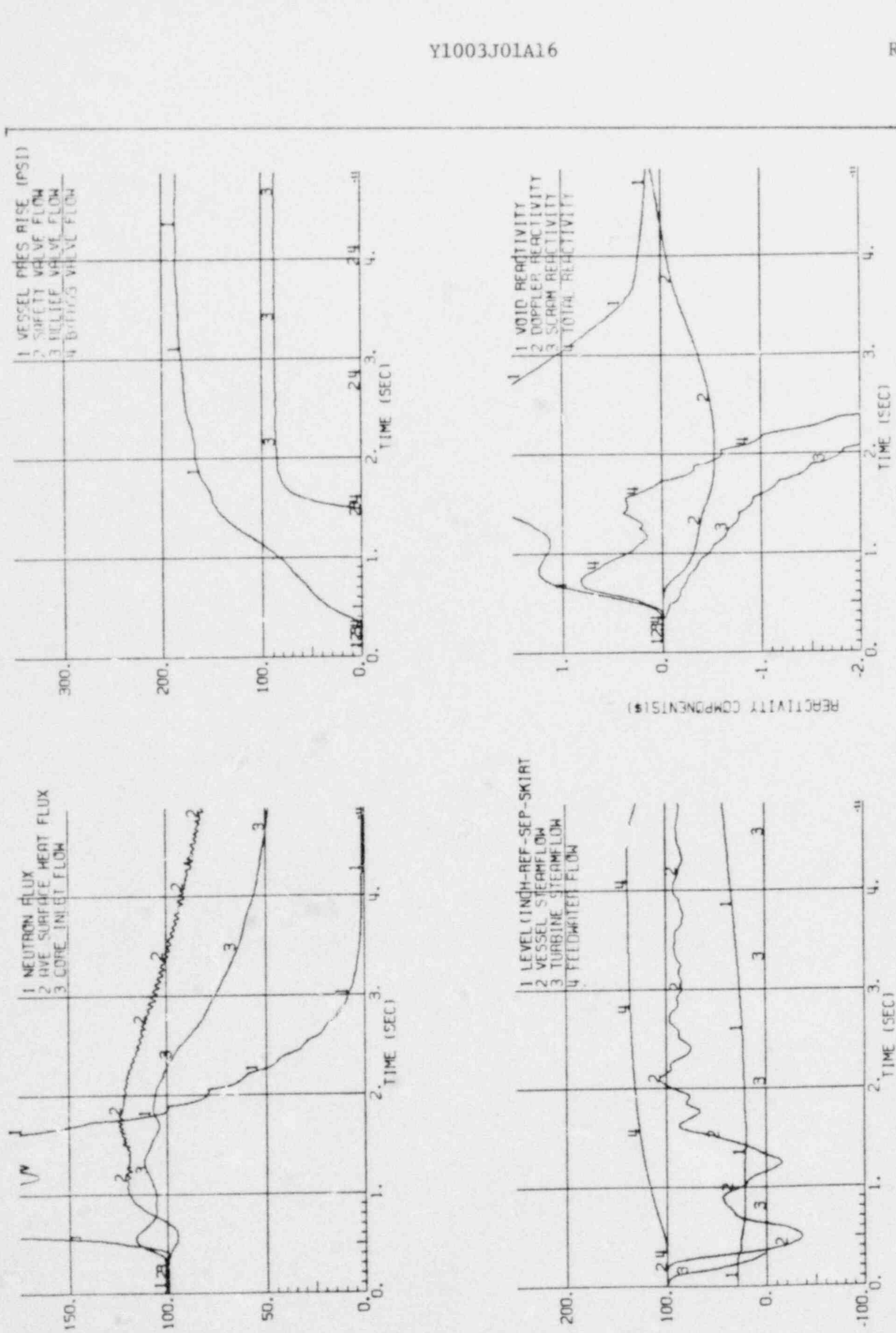


Figure 3. Plant Response to Generator Load Rejection Without Bypass

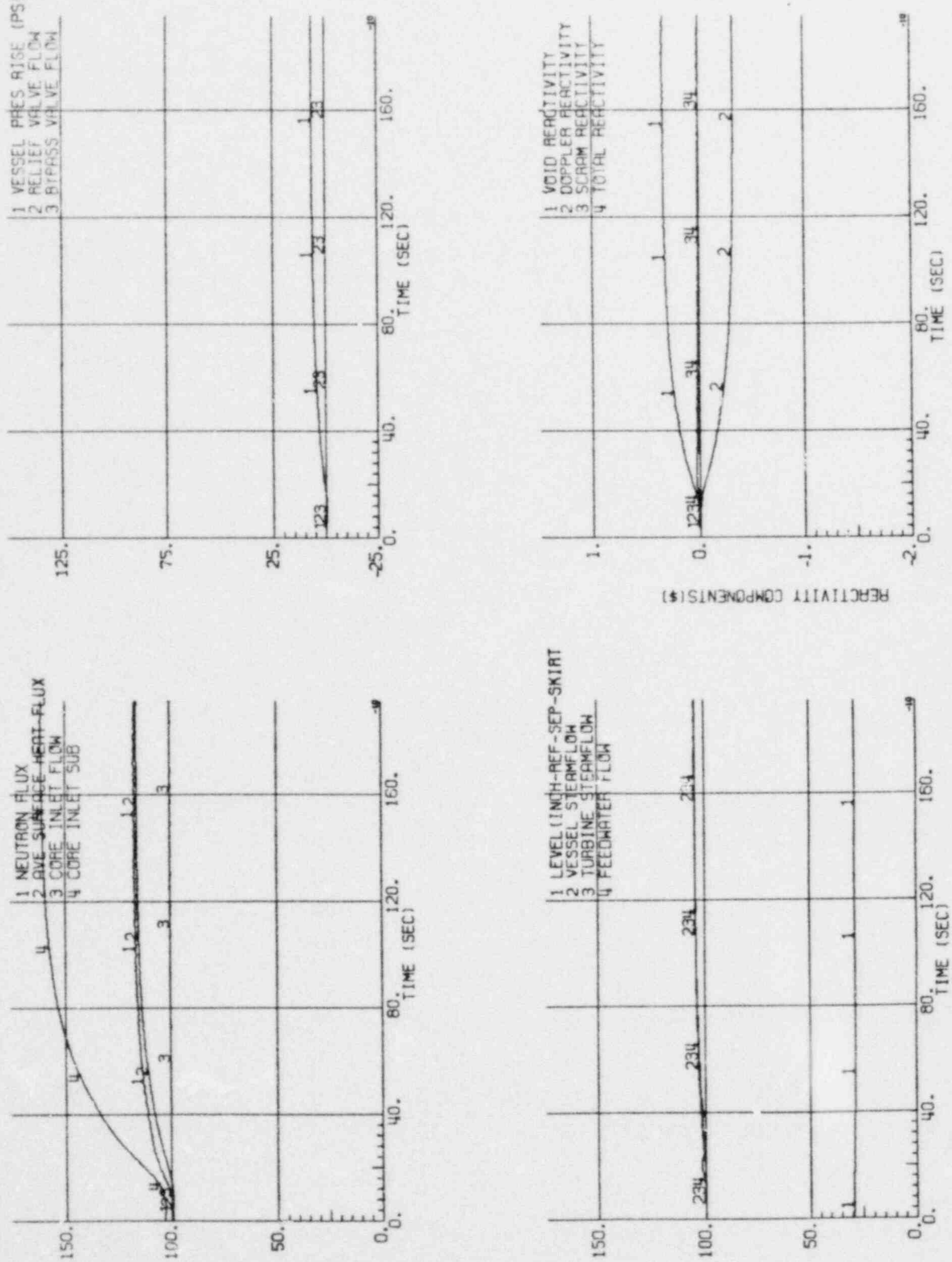


Figure 4. Plant Response to Loss of 100°F Feedwater Heating

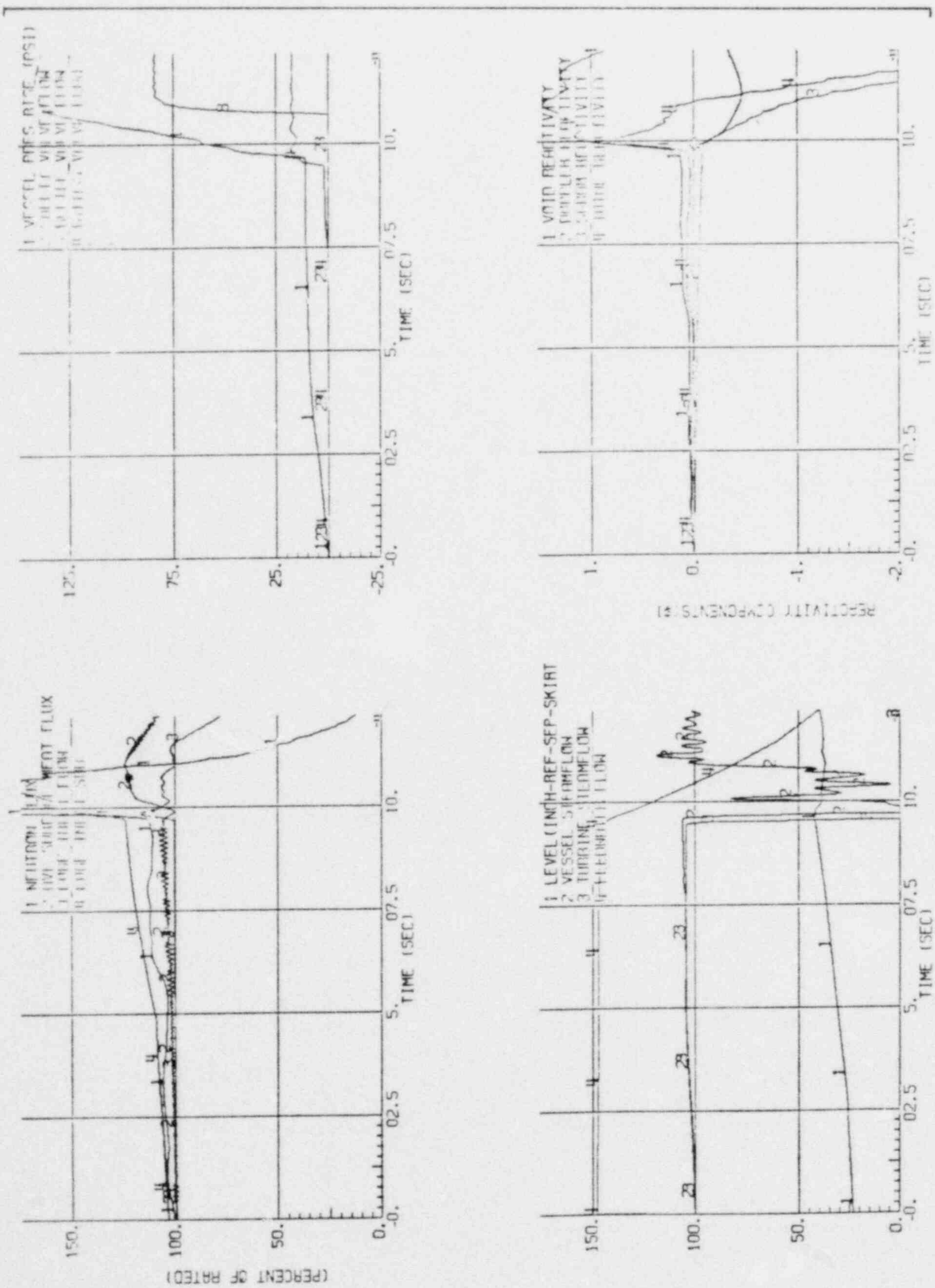


Figure 5. Plant Response to Feedwater Controller Failure

	2	6	10	14	18	22	26	30	34	38	42	46	50
51							4						
47				10		0		0		10			
43			10		28		10		28		10		
39		6		22		6		6		22		6	
35			22		20		26		20		22		
31		32				8		8				32	
27	18		0		20		0		20		0		18
23		32				8		8				32	
19			22		20		26		20		22		
15		6		22		6		6		22		6	
11			10		28		10		28		10		
7				10		0		0		10			
3							4						

NOTES:

1. NUMBER INDICATES NUMBER OF NOTCHES WITHDRAWN OUT OF 48. BLANK IS A WITHDRAWN ROD.
2. ERROR ROD IS (10,27).

Figure 6. Limiting RWE Rod Pattern

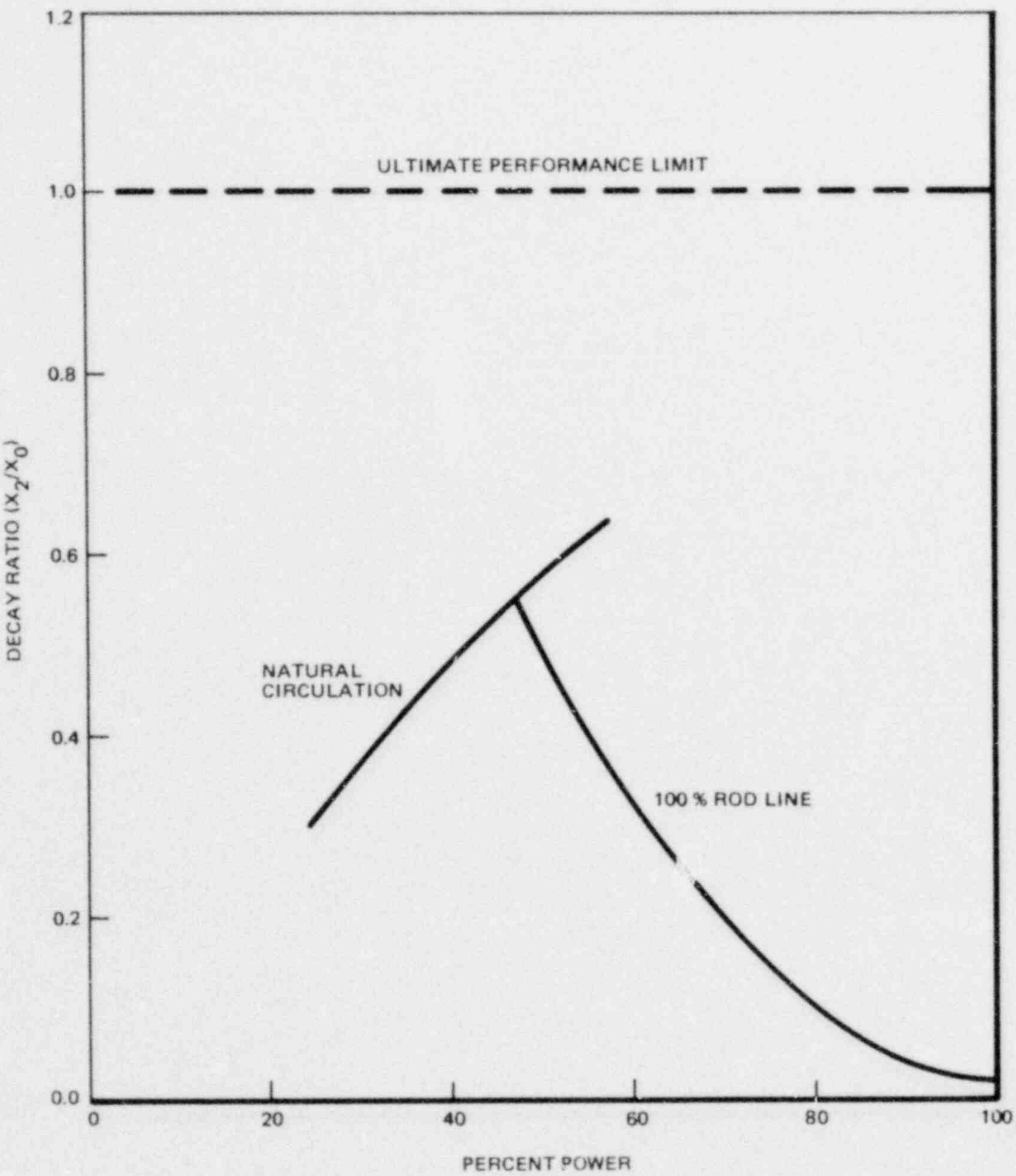


Figure 8. Decay Ratio

Appendix A
PLANT PARAMETER CHANGES

Safety/Relief Valve - (Tables 5-4, page 5-62, Operating Plants Pressure Relief Systems) 8 S/R valves installed

7 S/R valves assumed in analysis. Capacity at setpoint = 83.0%

Lowest setpoint = 1108 + 1% psig

GETAB Initial Conditions (Table 5-8, page 5-66)

See the revision to Table 5-8 enclosed with the letter, J. F. Quirk (GE) to Olan D. Parr (NRC), "General Electric Co. Licensing Topical Report NEDE-24011-P-A, Generic Reload Fuel Application, Appendix D, Second Submittal," February 28, 1979.

Initial MCPR

The initial MCPR for the 8x8R and P8x8R fuel was less than the operating limit MCPR. This is discussed on pp. B-114 and B-115 of the "Generic Reload Fuel Application," NEDE-24011-P-A-1.

Loading Error Results (5.5.4, Table 5-8, page 5-66)

LHGR: 21.2 kW/ft

All Channels not supplied by GE - At the direction of Northerm States Power Company, the analyses have assumed that performance characteristics of channels not supplied by GE are identical to the characteristics of channels supplied by GE.

RWM Operability Limit (Item 16) - 10%

Appendix B
ODYN CODE FOR TRANSIENT ANALYSES

All rapid pressurization and overpressure protection events have been analyzed using the ODYN transient code as specified in Reference B-1. The ΔCPR values given for the pressurization events in Section 9 are the plant-specific deterministic values calculated by ODYN based on the initial MCPR given in Item 7 of this submittal. These ΔCPRs may be adjusted to reflect either Option A or Option B ΔCPRs by employing the conversion method described in Reference B-2. These adjustments are based on conservatism factors applied to the ratio ΔCPR/ICPR. The resulting MCPR is calculated by adding the ΔCPR to the safety limit. The MCPRs resulting from the adjusted ΔCPRs from pressurization events as well as the MCPRs for nonpressurization events are presented in Section 11. Code overpressure protection analysis results are deterministic as discussed in Reference B-2.

The operating limit MCPR is the maximum MCPR of the following events:

1. turbine trip or load rejection without bypass based on ODYN;

2. feedwater controller failure event based on ODYN;

3. loss of feedwater heating event;

4. rod withdrawal error event; and

5. bundle loading error accident;

where the loss of feedwater heating, rod withdrawal error, and loading error MCPRs are calculated as described in Reference B-3 but the MCPRs for the pressurization events analyzed with ODYN have been adjusted as follows:

1. MCPRs adjusted for Option A (adding 0.044 to ΔCPR/ICPR) for all plants choosing to operate under Option A.

2. MCPRs adjusted for Option B for all plants choosing to operate under Option B which meet all scram specifications given in Reference B-2.

3. MCPRs are determined by a linear interpolation between the Option A MCPR and the Option B MCPR for all plants choosing to operate under Option B which do not meet the scram time specification. This interpolation is based on the tested measured scram time and is described in Reference B-2.

REFERENCES

B-1 Letter, R. P. Denise (NRC) to G. G. Sherwood (GE), January 23, 1980

B-2 Letter (with attachment), R. H. Buchholz (GE) to P.S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

B-3 "Generic Reload Fuel Application," NEDE-24011-P-A-1, August 1979.

APPENDIX C
EXTENDED EXPOSURE FUEL

This extended exposure program description and initial safety evaluation is presented in Reference C-1. The information contained in this appendix updates the safety evaluation for continued irradiation of the selected bundles for Cycle 9. The information contained herein is presented in the same format as Reference C-1.

1.0 Prop[illegible] Program

The peak pellet exposure is calculated to be 52,300 MWd/ST at EOC9.

2.1 Fuel Rod Thermal Analysis

Analyses performed for the extended exposure fuel bundles resulted in values (for 1% plastic strain limit) of 14.7 kW/ft at a peak pellet exposure of 52,300 MWd/ST for UO_2 fuel rods and 15.5 kW/ft at 44,760 MWd/ST for urania-gadolinia rods.

2.1.1 Fuel Cladding Temperatures

The inside, average, and outside cladding temperature during normal operation at the end of Cycle 9 are calculated not to exceed 815°F, 782°F, and 750°F, respectively.

2.2.1 Cladding Creep Collapse

The calculation demonstrated that cladding creep collapse is not expected to occur in the event of a maximum overpressure transient for an exposure of 52,300 MWd/ST.

2.2.2 Stress Evaluations

Fuel rod stress analyses of the extended exposure bundles were performed with the model documented in Reference 1 (to NEDO-24202) and nominal input values (Reference C-2) for operation through Cycle 9. Fuel design ratios were shown to be well below 1.0

2.2.4 Fatigue Evaluation

The cumulative fatigue damage is calculated to be less than the allowable fatigue limit for Cycle 9.

2.3.3 Fretting Wear and Corrosion

The fuel bundles which will operate to higher exposures will again be visually examined before loading in Cycle 9.

3.1.1 Reactivity

Hot reactivity of the extended exposure fuel bundles decreases by approximately 0.003 Δk_{∞} from 45,000 MWd/ST to 55,000 MWd/ST.

3.1.2 Local Peaking Factors

Calculated maximum local peaking factor for the extended exposure fuel bundles increases by approximately 0.04 from 45,000 MWd/ST to 55,000 MWd/ST.]

3.3.5 Accident Evaluations

The new MAPLHGR values and associated peak cladding temperatures and oxidation fractions are given in Reference C-3.

4. References

C-1 "Monticello Nuclear Generating Plant Extended Exposure Fuel Program", NEDO-24202, dated July 1979.

C-2 R. L. Tedesco (NRC) letter to R. E. Engle (GE), "Acceptance for Licensing Reference of Changes to Topical Report Number NEDE-24011-P-A-1, 'Generic Reload Fuel Application', dated August 1979", dated November 7, 1980.

C-3 "Loss-of-Coolant Accident Analysis Report for Monticello Nuclear Generating Plant", NEDO-24050-1, dated December 1980.