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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR MILLSTONE UNIT 1 RELOAD 10

GENERAL ELECTRIC

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FOR

MILLSTONE UNIT 1

RELOAD 10

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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
GETAB and Transient Analysis Initial Conditions	Appendix B
Stability Analysis	Appendix C
Feedwater Temperature Reduction Analysis	Appendix D
Fuel Bundle Description	Appendix E

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

Fuel Type	Cycle Loaded	Number	Number Drilled
Irradiated			
P8DRB282	9	72	72
P8DRB283H	9	108	108
BP8DRB300**	10	200	200
New			
BP8DRB300**	11	200	200
		580	580

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	17,533 MWd/ST
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	17,533 MWd/ST
Assumed reload cycle core average exposure at end of cycle:	18,256 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, dated April 1983. A letter "S" preceding the number refers to the appropriate country-specific supplement.
**See Appendix E.

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	1.106
Fully Controlled	0.954
Strongest Control Rod Out	0.979
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, Δk	0.005

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
660	0.046

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

(Cold Water Injection Events Only)

Void Fraction (%)	36.8
Average Fuel Temperature (°F)	1151
Void Coefficient N/A* (¢/% Rg)	-5.76/-7.20
Doppler Coefficient N/A (¢/°F)	-0.183/-0.174
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, dated 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
Exposure: BOC11 to EOC11							
BP8x8R/ P8x8R	1.20	1.62	1.40	1.051	5.482	100.2	1.41

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

Transient Recategorization:	No
Recirculation Pump Trip:	No
Rod Withdrawal Limiter:	No
Thermal Power Monitor:	No
Improved Scram Time:	Yes (ODYN Option B)
Exposure-Dependent Limits:	No
Exposure Points Analyzed:	1

9. OPERATING FLEXIBILITY OPTIONS (S.2.2.3)

Single-Loop Operation:	Yes
Load Line Limit:	Yes
Extended Load Line Limit:	Yes
Increased Core Flow:	No
Flow Point Analyzed:	N/A
Feedwater Temperature Reduction:	Yes

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

Exposure: BOC11 to EOC11

Transient	Flux (% NBR)	Q/A (% NBR)	ΔCPR BP8x8R/ P8x8R	Figure
Load Rejection Without Bypass	571	127	0.34	2
Loss of Feedwater Heater	116	115	0.14	3
Feedwater Controller Failure	109	108	0.07	4

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

	BP8x8R	P8x8R
Loss of Feedwater Heater	1.21	1.21
Fuel Loading Error	1.26	-
Rod Withdrawal Error	1.35	1.35

Pressurization Events

Exposure Range: BOC11 to EOC11

	Option A	Option B
	BP8x8R/P8x8R	BP8x8R/P8x8R
Load Rejection Without Bypass	1.48	1.42
Feedwater Controller Failure	1.19	1.12

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1254	1270	Figure 5

14. STABILITY ANALYSIS RESULTS (S.2.4)

See Appendix C

15. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

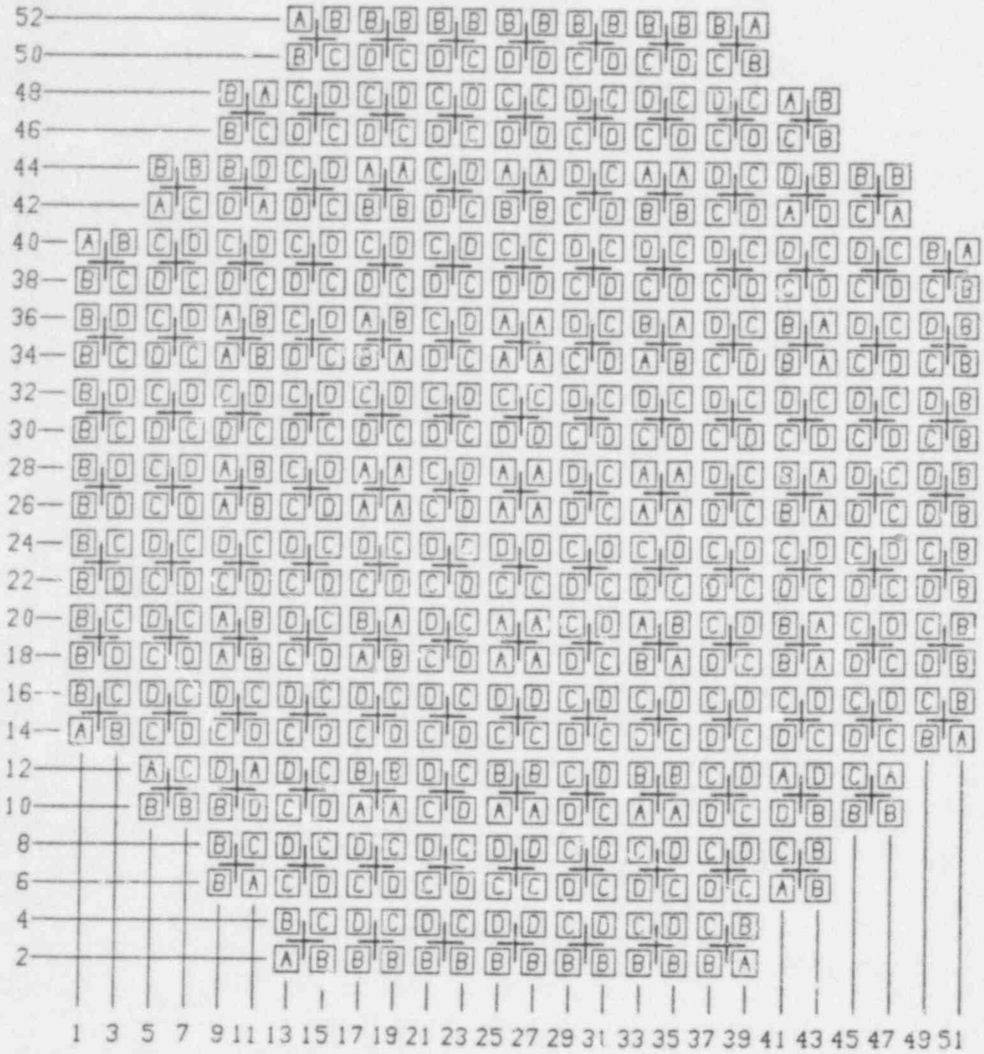
Event	Initial MCPR	Resulting MCPR
Misoriented	1.24	1.07

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

See Appendix A.

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

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



FUEL TYPE

A = P8DRB282
B = P8DRB283H
C = BP8DRB300
D = BP8DRB300

Figure 1. Reference Core Loading Pattern

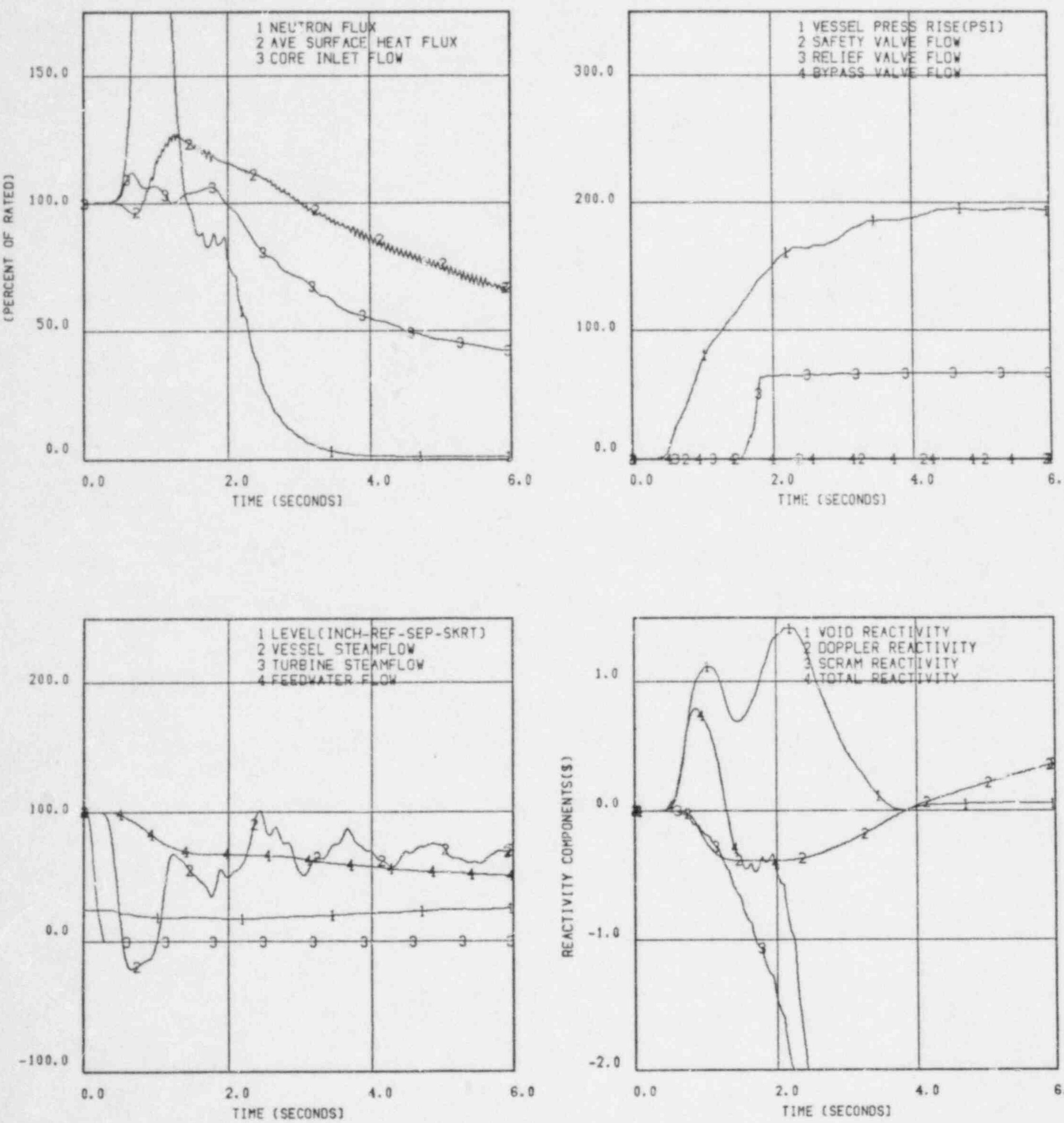


Figure 2. Plant Response to Generator Load Rejection Without Bypass, EOC11

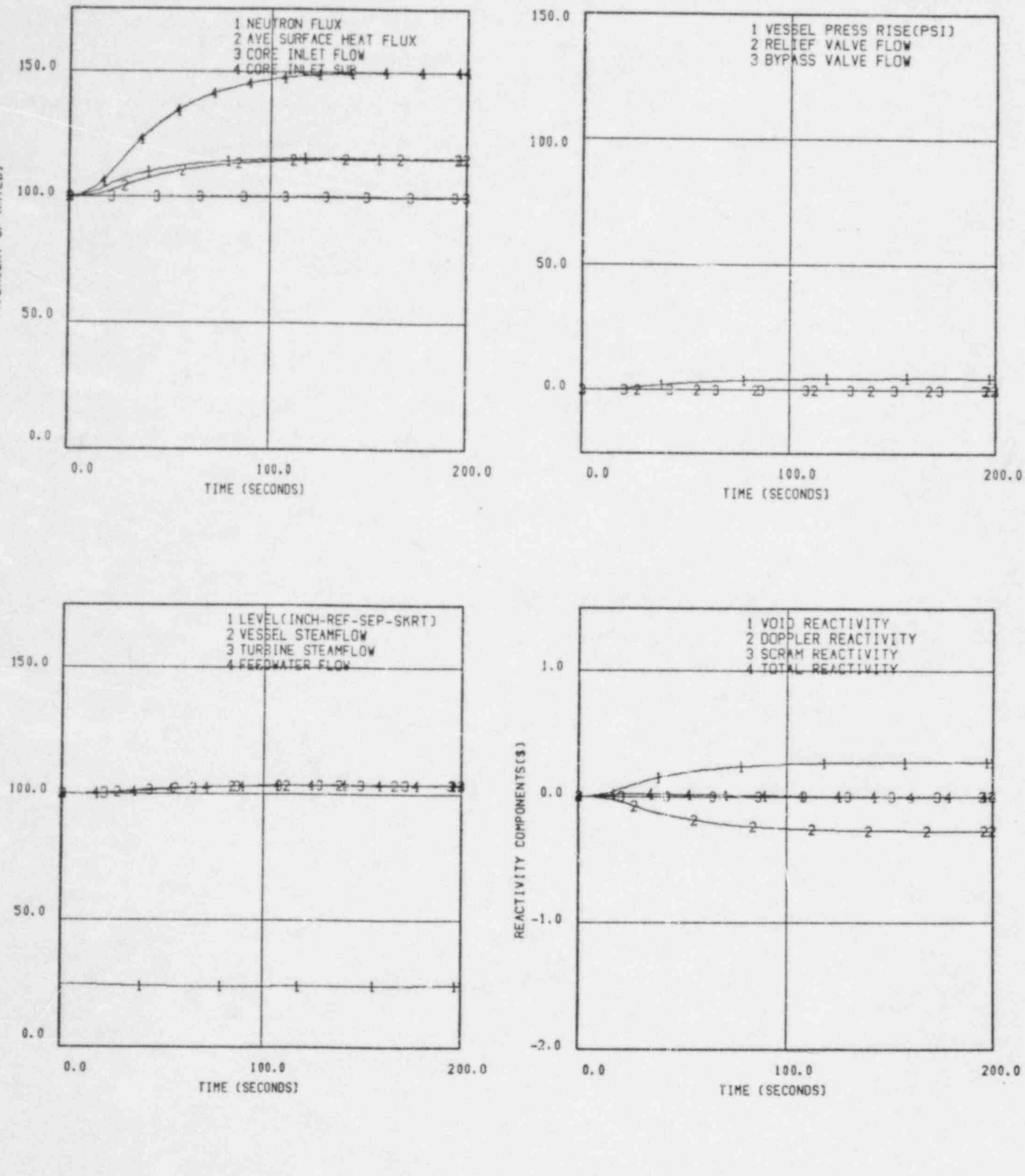


Figure 3. Plant Response to Loss of 100°F Feedwater Heating

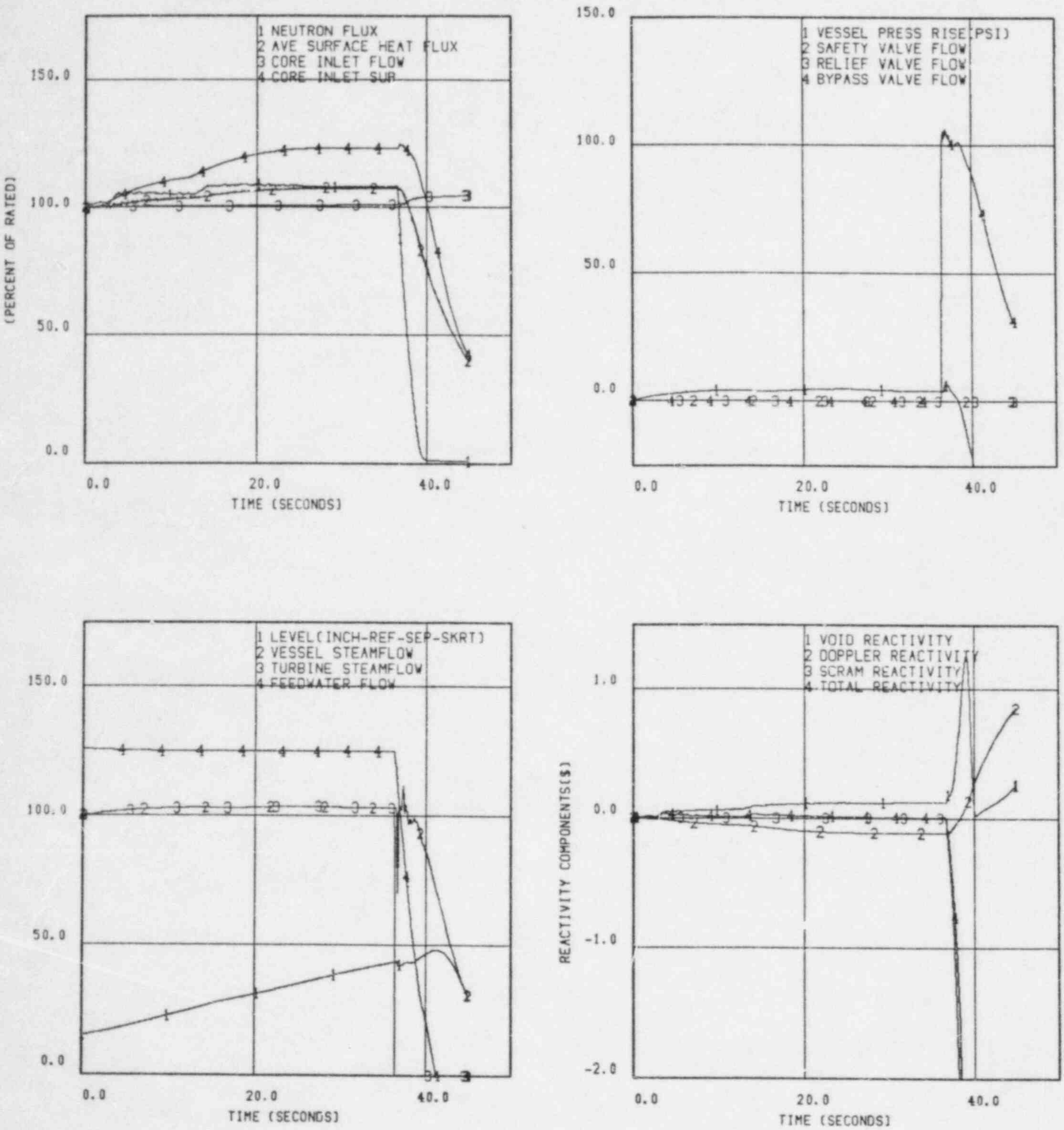


Figure 4. Plant Response to Feedwater Controller Failure, EOC11

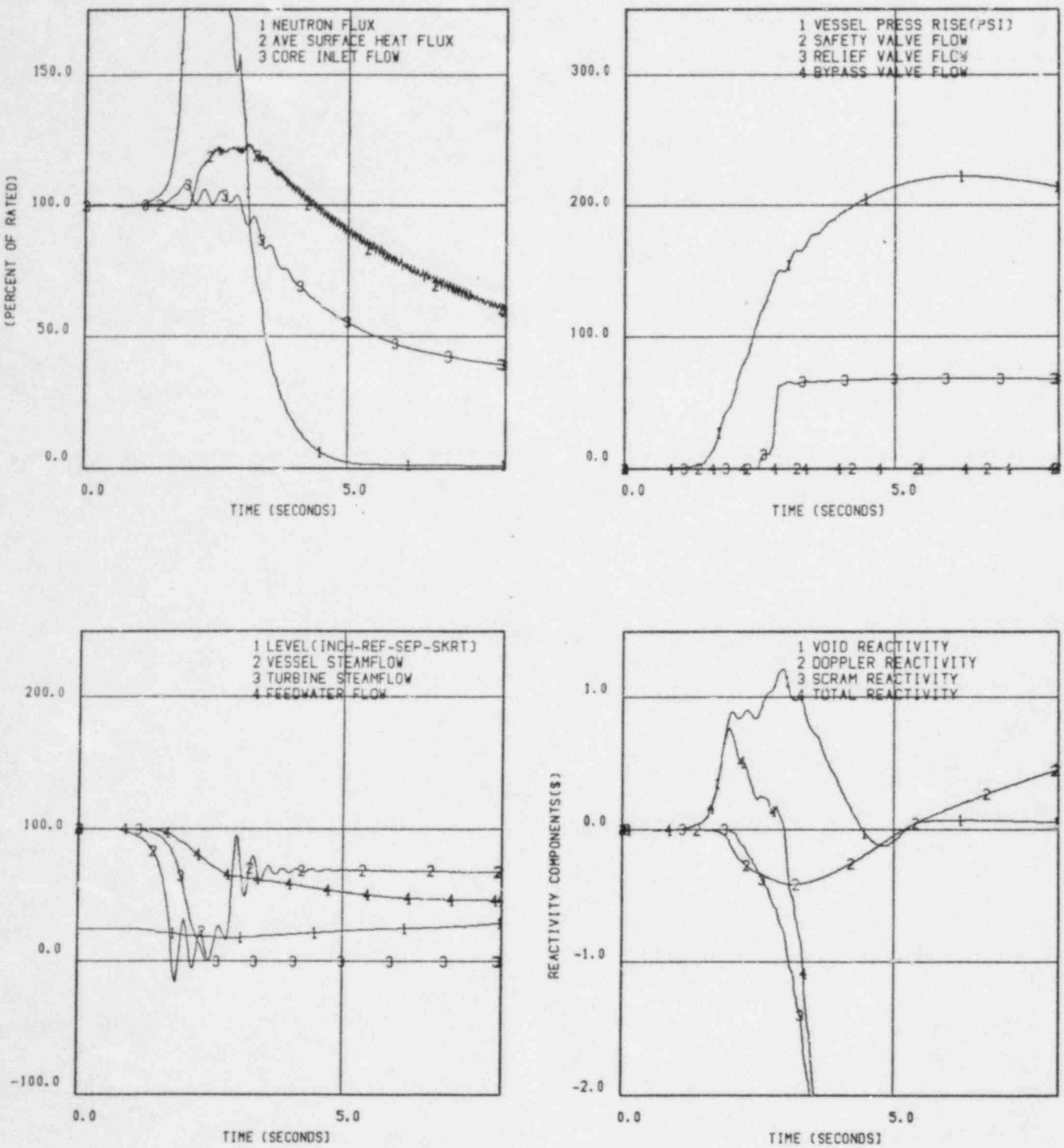


Figure 5. Plant Response to MSIV Closure (Flux Scram), EOC11

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-2 indicates that this change is acceptable to the NRC.

REFERENCES

A-1. Letter, J. S. Charnley (GE) to C. O. Thomas (NRC), "Proposed Administrative Amendment to GE Licensing Topical Report NEDE-24011-P-A", January 25, 1984.

A-2. Letter, C. O. Thomas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report Amendment 9 to NEDE-24011, Revision 6, 'GESTAR-II General Electric Standard Application for Reactor Fuel'", January 25, 1985.

APPENDIX B

GETAB AND TRANSIENT ANALYSIS INITIAL CONDITIONS

The values used in the GETAB analysis for reactor core pressure and inlet enthalpy and in the transient analysis for rated steam flow are given in Table B-1. The following values are different from those reported in NEDE-24011-P-A-6-US, dated April 1983.

Table B-1

PLANT PARAMETER

Parameter		Analysis Value	NEDE-24011 Value
Reactor Core Pressure		1065 psia	1057 psia
Inlet Enthalpy		526.0 Btu/lb	525.2 Btu/lb
Rated Steam Flow		7.99×10^6 lb/hr	$7.94 \times 10^6 \pm 0.2\%$ lb/hr
Safety/Relief Valve (SRV)			
Number of SRVs at:			
Lowest Setpoint (psig)	Capacity (lb/hr)		
1095	791,000	0	4
1095	829,000	3	2
1125	791,000	3	0

APPENDIX C
STABILITY ANALYSIS

According to Reference C-1, Millstone Unit 1 is exempt from the current requirement to submit a cycle specific stability analysis to the NRC.

REFERENCES

C-1. Letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, 'Thermal Hydraulic Stability Amendment to GESTAR II'", April 24, 1985.

APPENDIX D
FEEDWATER TEMPERATURE REDUCTION AT EOC11

Analyses were performed for end-of-cycle (EOC) 11 operation with the last-stage feedwater heaters valved out-of-service, in order to justify operation with feedwater temperature reduced by 75°F. The pressurization events of Section 12 were reanalyzed for operation at the reduced feedwater temperature. This appendix presents the results of these transient analyses.

The balance of the safety analysis required to justify operation at a reduced feedwater temperature (as defined in Reference D-1) will be provided by NUSCO.

REFERENCES:

D-1. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-6-US, dated April 1983.

D.1 CORE AVERAGE EXPOSURE

Assumed reload core average exposure for Feedwater Temperature Reduction (FWTR) analysis (Extended EOC11)	18976 MWd/ST

D.2 RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

Fuel Design	Peaking Factors: Local	Radial	Axial	R-Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
Exposure: EOC11 to Extended EOC11							
BP8x8R/ P8x8R	1.20	1.67	1.40	1.051	5.640	99.3	1.39

D.3 CORE-WIDE TRANSIENT ANALYSIS RESULTS

Exposure: EOC11 to Extended EOC11

Transient	Flux (% NBR)	Q/A (% NBR)	ΔCPR BP8x8R/ P8x8R	Figure
Load Rejection Without Bypass	515	125	0.32	D-1
Feedwater Controller Failure	121	114	0.12	D-2

D.4 CYCLE MCPR VALUES

Exposure Range: EOC11 to Extended EOC11

	Option A	Option B
	BP8x8R/P8x8R	BP8x8R/P8x8R
Load Rejection Without Bypass	1.45	1.40
Feedwater Controller Failure	1.24	1.17

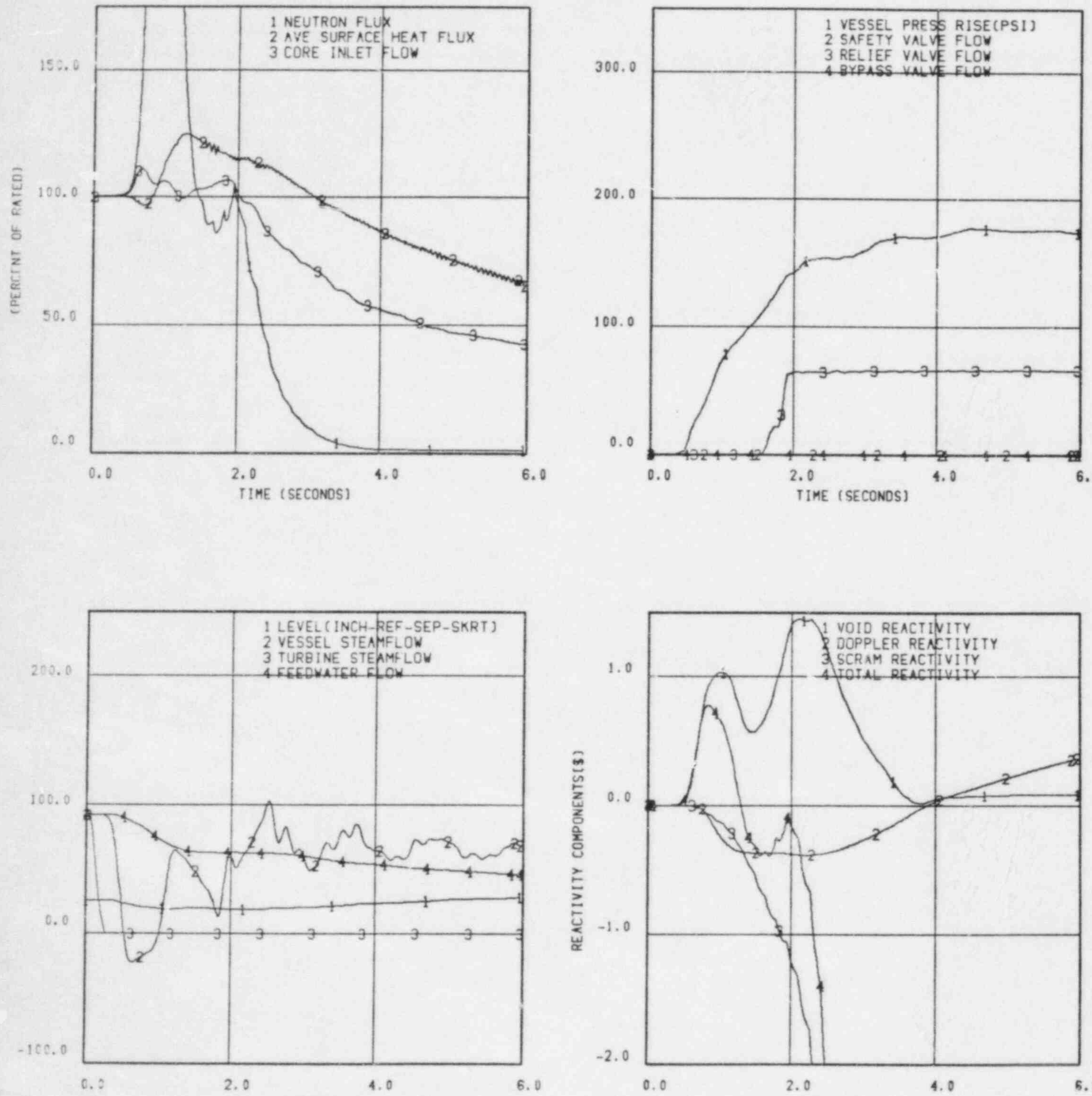


Figure D-1. Plant Response to Generator Load Rejection Without Bypass, FWTR

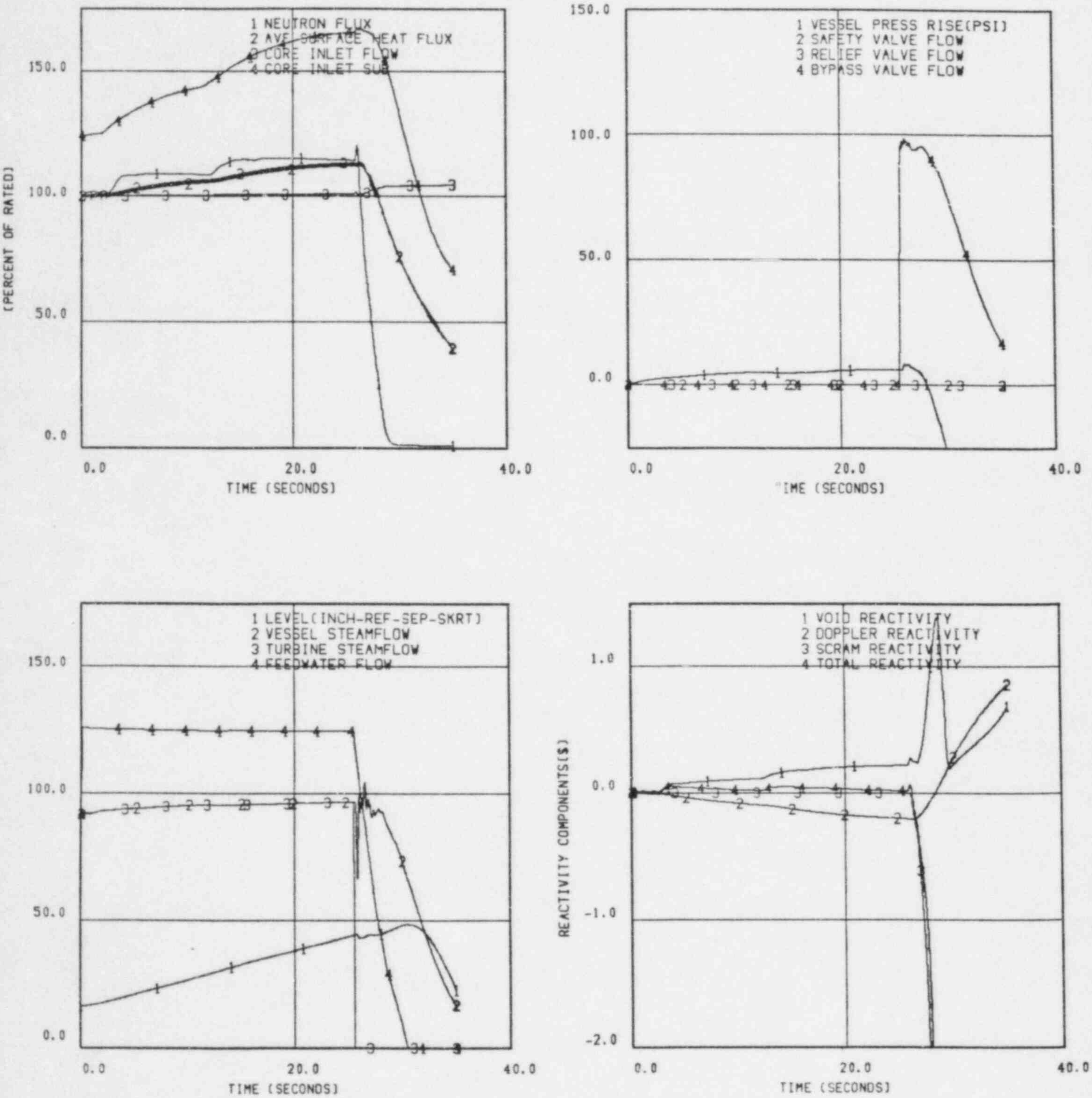


Figure D-2. Plant Response to Feedwater Controller Failure, FWTR

APPENDIX E
FUEL BUNDLE DESCRIPTION

The BP8DRB300 fuel bundle description will be provided in Amendment 13 of "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-6, dated April 1983. This information was previously provided in Reference E-1.

REFERENCE:

E-1. Letter, W. G. Counsil (NUSCO) to D. M. Crutchfield (NRC), "Millstone Nuclear Power Station, Unit 1, Fuel Bundle Proprietary Information," March 27, 1984.

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