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## ACKNOWLEDGMENT

The engineering and reload licensing analyses, which form the technical basis of this Supplemental Reload Licensing Submittal, were performed by M. I. Khan of the Nuclear Fuel and Engineering Services Department.

1. PLANT-UNIQUE ITEMS (1.0)\*

Appendix A: Plant Parameter Differences

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

<u>Fuel Type</u>	<u>Cycle Loaded</u>	<u>Number</u>
Irradiated		
P8DRB265L	9	16
P8DRB284LB	9	28
P8DRB265L	10	32
P8DRB284LB	10	48
P8DRB284LB	11	116
BP8DRB299L	12	124
New		
BD319B	13	<u>120</u>
Total		484

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle: 21,766 MWd/MT

Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations: 14,929 MWd/MT

Assumed reload cycle core average exposure at end of cycle: 22,289 MWd/MT

Core loading pattern: Figure 1

\* ( ) Refers to area of discussion in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-8, May 1986; a letter "S" preceding the number refers to the appropriate section in the U.S. Supplement, NEDE-24011-P-A-8-US, May 1986.

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	0.960
Strongest Control Rod Out	0.984
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, $\Delta K$	0.006

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

<u>ppm</u>	<u>Shutdown Margin (<math>\Delta K</math>) (20°C, Xenon Free)</u>
660	0.043

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

(Cold Water Injection Events Only)

Void Fraction (%)	37.6
Average Fuel Temperature (°F)	974
Void Coefficient N/A* ( $\Delta/\%$ Rg)	-6.47/-8.09
Doppler Coefficient N/A* ( $\Delta/^\circ\text{F}$ )	-0.277/-0.263
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-8-US, May 1986.

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

<u>Fuel Design</u>	<u>Peaking Factors</u>			<u>R-Factor</u>	<u>Bundle Power (MWt)</u>	<u>Bundle Flow (1000 lb/hr)</u>	<u>Initial MCPR</u>
	<u>Local</u>	<u>Radial</u>	<u>Axial</u>				
BOC13 to EOC13							
BP/P8x8R	1.20	1.85	1.40	1.051	6.232	95.3	1.24
GE8x8EB	1.20	1.85	1.40	1.051	6.230	96.1	1.24

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 Point Analyzed:	EOC

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:	No
ARTS Program:	Yes
Maximum Extended Operating Domain:	No

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

Methods Used: GEMINI

<u>Transient</u>	<u>Flux</u> <u>(% NBR)</u>	<u>Q/A</u> <u>(% NBR)</u>	<u>Δ CPR</u>		<u>Figure</u>
			<u>BP/P8x8R</u>	<u>GE8x8EB</u>	
Exposure: BOC13 to EOC13					
Turbine Trip w/o Bypass	455	120	0.17	0.17	2
Loss of Feedwater Heating	116	115	0.14	0.14	3
Feedwater Controller Failure	313	119	0.16	0.16	4

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

Rod Withdrawal Error is analyzed in "General Electric BWR Licensing Report: Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant," NEDC-30492-P, April 1984. The MCPR Limit for Rod Withdrawal Error is 1.20.

12. CYCLE MCPR VALUES (S.2.2)\*

Minimum MCPR required by ECCS is 1.24.

## Non-Pressurization Events

Exposure Range: BOC13 to EOC13

	<u>BP/P8x8R</u>	<u>GE8x8EB</u>
Loss of Feedwater Heating	1.21	1.21
Fuel Loading Error	--	1.15
Rod Withdrawal Error	1.20	1.20

## Pressurization Events

Exposure Range: BOC13 to EOC13

	<u>Option A</u>		<u>Option B</u>	
	<u>BP/P8x8R</u>	<u>GE8x8EB</u>	<u>BP/P8x8R</u>	<u>GE8x8EB</u>
Turbine Trip w/o Bypass	1.30	1.30	1.25	1.25
Feedwater Controller Failure	1.29	1.29	1.24	1.24

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

<u>Transient</u>	<u>P<sub>sl</sub></u> <u>(psig)</u>	<u>P<sub>v</sub></u> <u>(psig)</u>	<u>Plant Response</u>
MSIV Closure (Flux Scram)	1183	1212	Figure 5

\*GEMINI ODYN statistical adjustment factors for BWR/3 plants are given in the letter from J.S. Charnley (GE) to G.C. Lainas (NRC), "GEMINI/ODYN Statistical Adders for BWR/2,3 Plants (Without RPT)-MOC/EOC," March 13, 1987.

14. STABILITY ANALYSIS RESULTS (S.2.4)

The NRC has approved a bounding stability approach that exempts Monticello Nuclear Generating Plant (MNGP) from the requirement to perform a stability analysis. NRC approval is documented in the letter from 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'," dated April 24, 1985.

MNGP Cycle 13 differs from previous cycles with the introduction of GE8x8EB fuel. During NRC review of the bounding stability approach, it was demonstrated that GE8x8EB fuel has only a small impact on stability performance. The NRC approval states that the bounding stability approach is applicable to cores with GE8x8EB fuel.

15. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

<u>Event</u>	<u>Δ CPR</u>
Misoriented	0.08

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

Not required for Banked Position Withdrawal Sequence plants.

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

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

Fuel Type: BD319B

<u>Exposure</u>		<u>MAPLHGR (kW/ft)</u>	
<u>(Gwd/ST)</u>	<u>(Gwd/MT)</u>	<u>Most Limiting</u>	<u>Least Limiting</u>
0.2	0.22	11.19	11.50
1.0	1.1	11.31	11.60
5.0	5.5	11.99	12.20
10.0	11.0	12.60	12.60
15.0	16.5	12.34	12.35
20.0	22.0	11.95	11.96
25.0	27.6	11.56	11.57
30.0	33.1	10.54	10.56
35.0	38.6	9.53	9.55
50.0	55.1	6.28	6.32



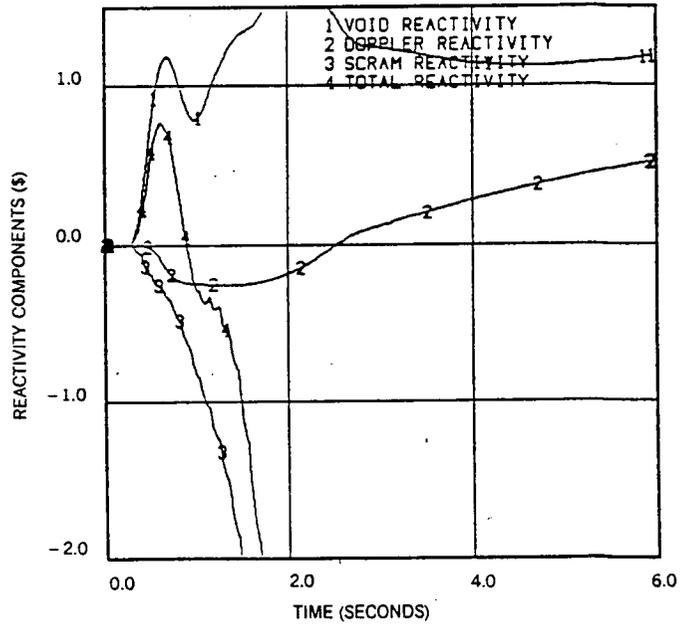
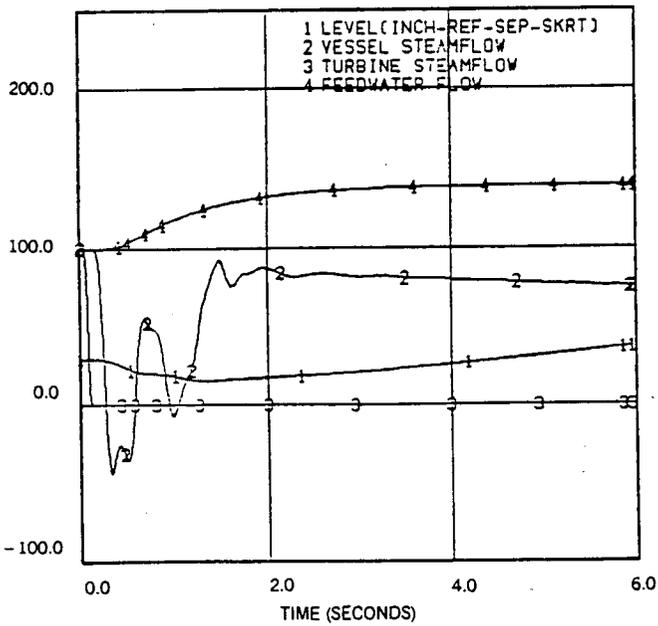
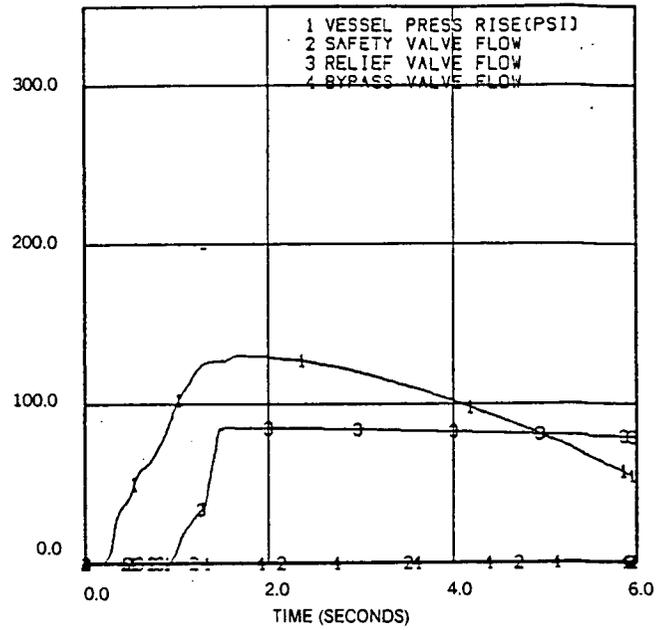
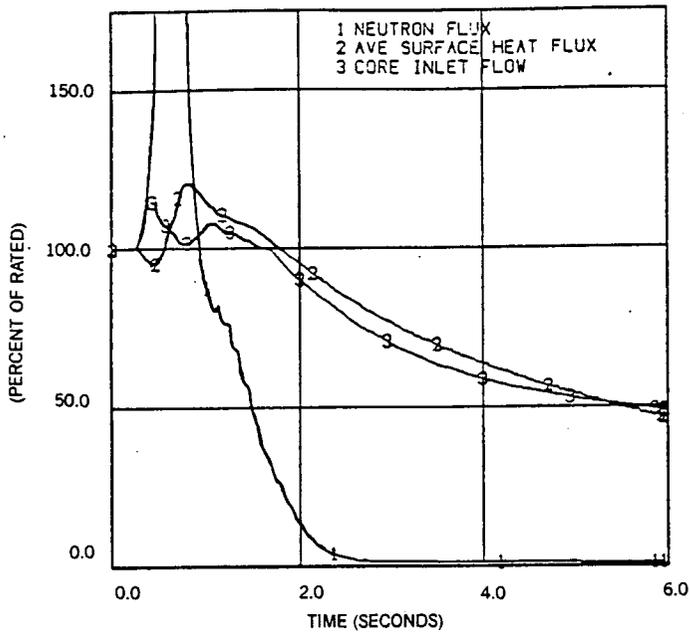


Figure 2. Plant Response to Turbine Trip Without Bypass

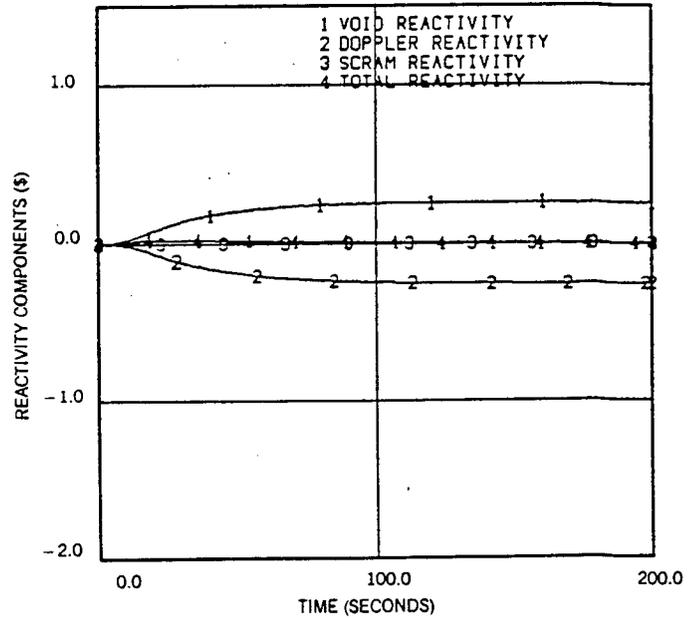
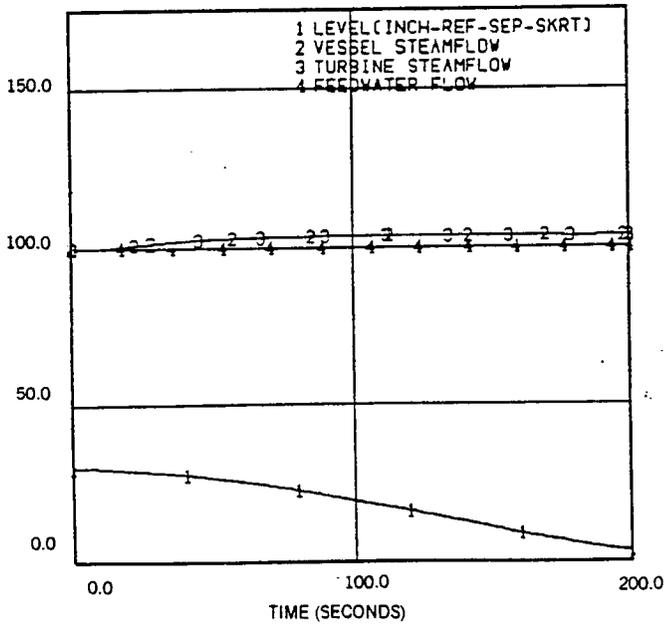
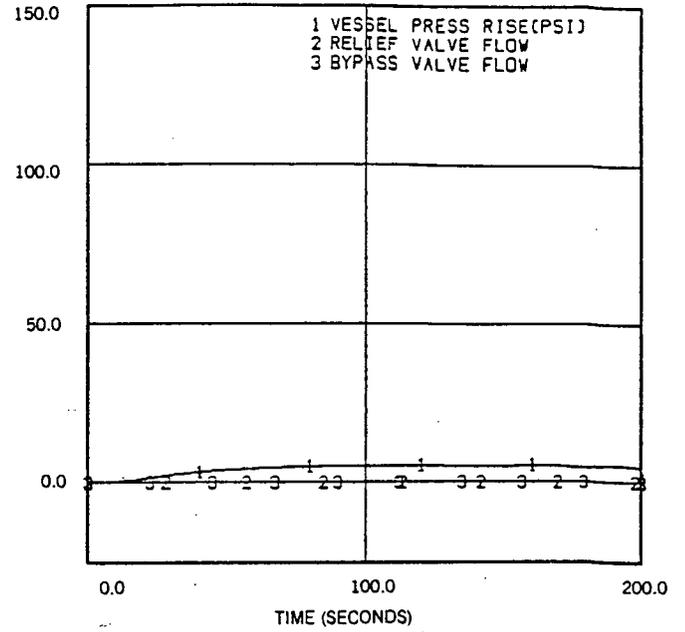
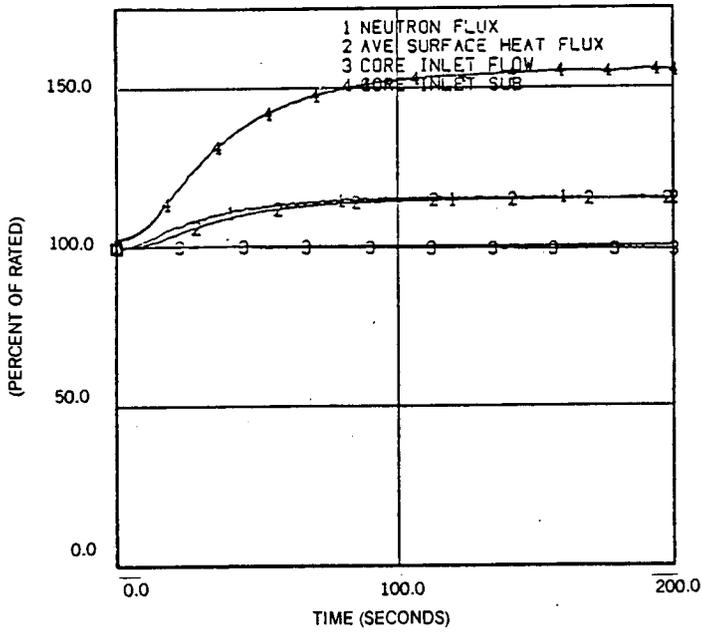


Figure 3. Plant Response to Loss of Feedwater Heating

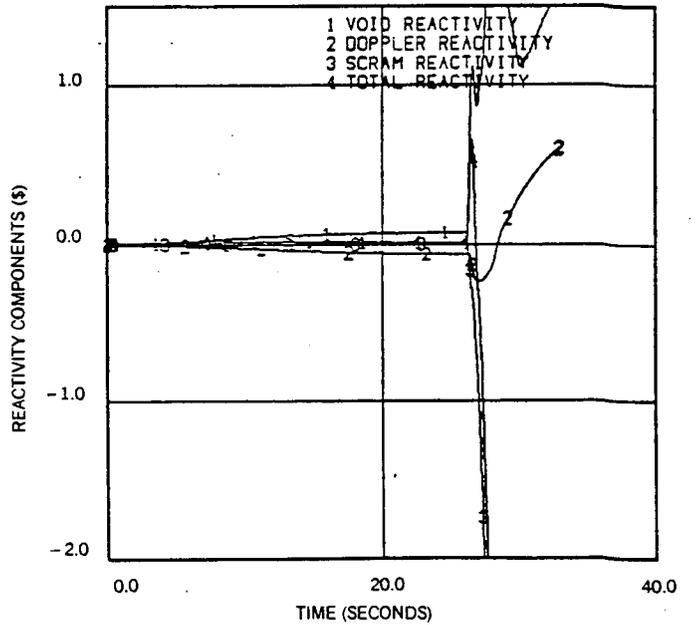
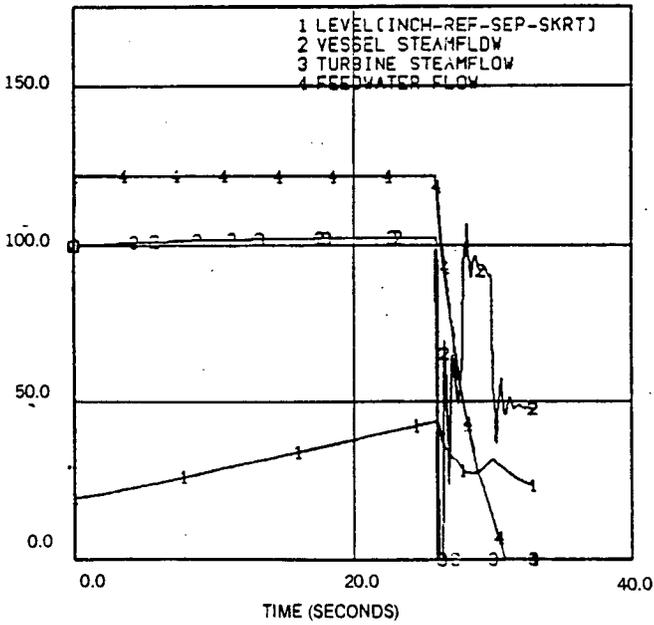
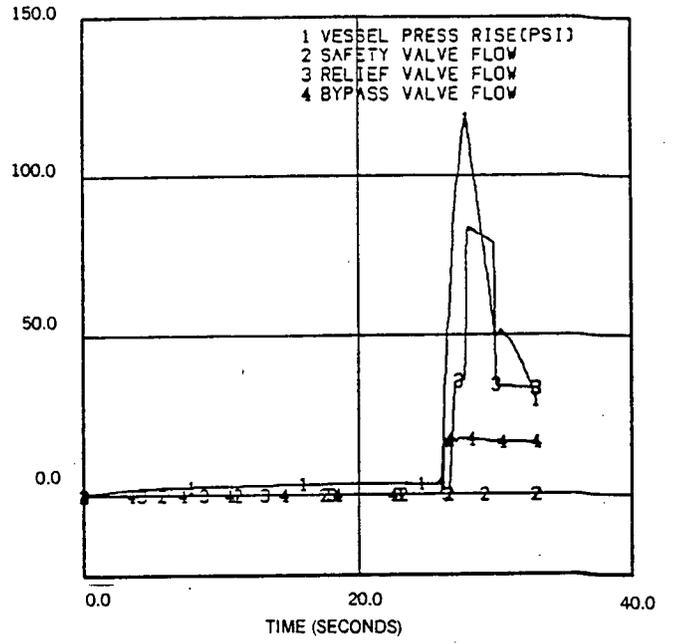
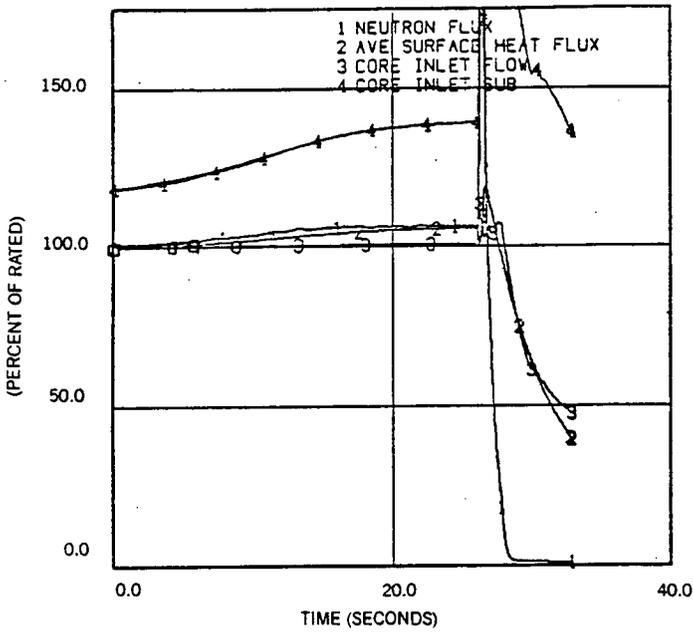


Figure 4. Plant Response to Feedwater Controller Failure

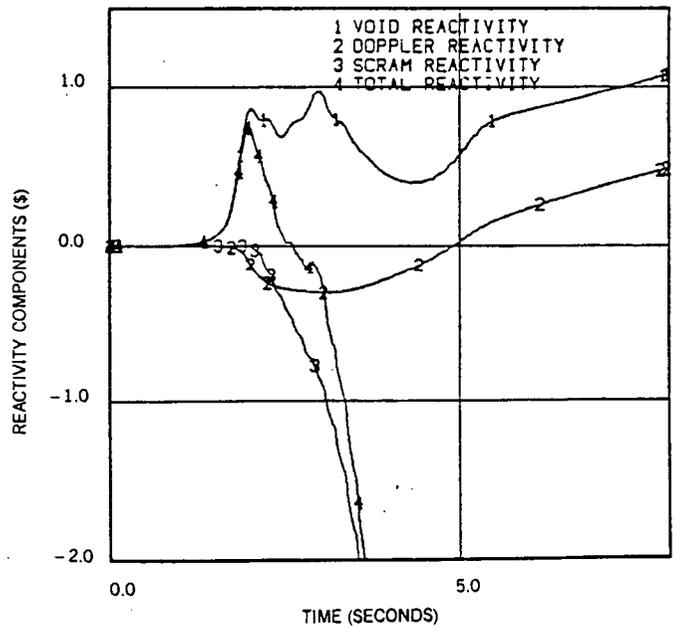
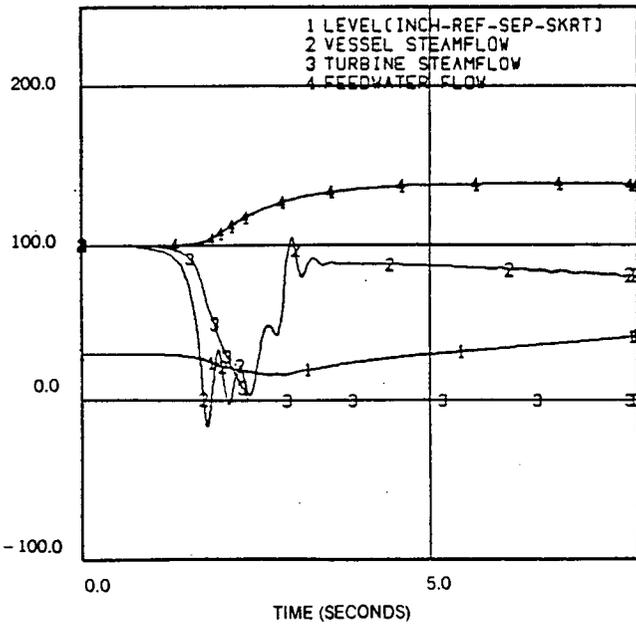
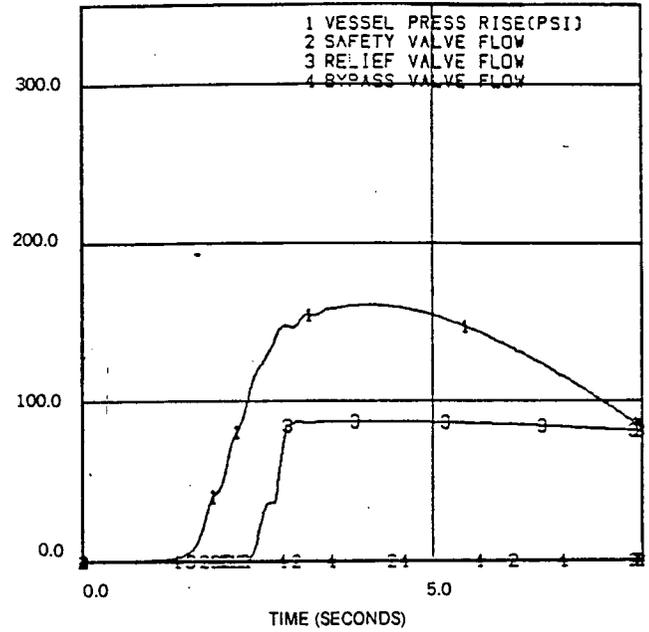
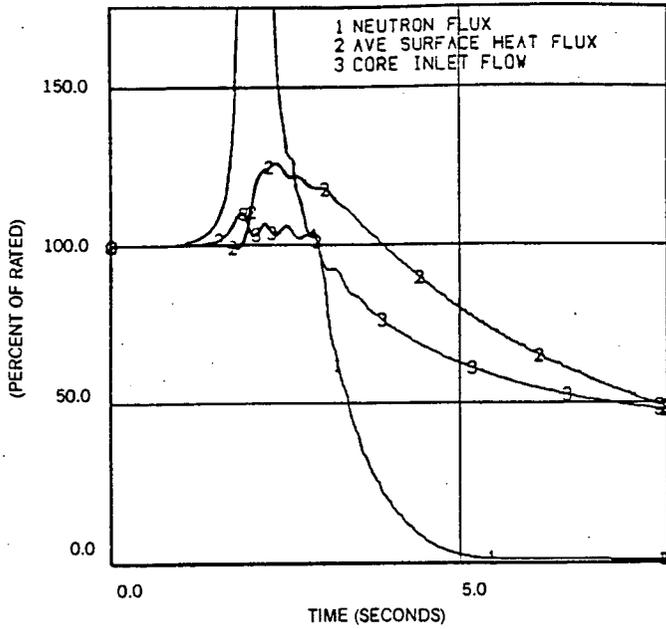


Figure 5. Plant Response to MSIV Closure, Flux Scram

APPENDIX A  
PLANT PARAMETER DIFFERENCES

GETAB and Transient Analysis Initial Conditions

The values used in the GETAB and Transient Analysis for non-fuel power fraction and safety/relief valve capacities are given in Table A-1. The following values differ from the values reported in NEDE-24011-P-A-8-US, May 1986.

Table A-1  
PLANT PARAMETER

<u>Parameter</u>	<u>Analysis Value</u>	<u>NEDE-24011 Value</u>
Non-Fuel Power Fraction	0.038	0.035
Number of Safety/Relief Valves		
791,000 lb/hr capacity	7	5
829,000 lb/hr capacity	0	2

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.