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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

CLINTON POWER STATION UNIT 1

RELOAD 1, CYCLE 2

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CONTENTS OF THIS REPORT

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ACKNOWLEDGEMENTS

The engineering and reload licensing analyses, which form the technical basis of this Supplemental Reload Licensing Submittal, were performed in the Fuel Engineering Section by T. C. Hoang.

1. PLANT-UNIQUE ITEMS (1.0)*

Appendix A: GETAB and Transient Analysis Initial Conditions
Appendix B: Basis for Analysis of Loss-of-Feedwater Heating Event
Appendix C: Application of GEMINI Methods

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

Fuel Type	Cycle Loaded	Number
Irradiated		
P8SRB154	1	96
P8SRB200	1	360
New		
BP8SRB284LC**	2	88
BP8SRB284L	2	80
Total		624

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	9314 MWd/MT
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	8773 MWd/MT
Assumed reload cycle core average exposure at end of cycle (all rods out, rated power):	12,393 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 (dated May 1986); a letter "S" preceding the number refers to the United States Supplement.

**Bundle-specific information for this bundle design was submitted to the U.S. NRC in a letter, J.S. Charnley (GE) to H.N. Berkow (NRC), "Proposed Amendment 16 to GE Licensing Topical Report NEDE-24011-P-A," August 8, 1986. U.S. NRC approval of this Amendment was documented in a letter, A.C. Thadani (NRC) to J.S. Charnley (GE), "Acceptance for Referencing of Amendment 16 to GE Licensing Topical Report NEDE-24011-P-A, 'General Electric Standard Application for Reactor Fuel'," April 20, 1988.

4. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH- NO VOIDS, 20 DEG. C (3.3.2.1.1 AND 3.3.2.1.2)

Beginning of Cycle, k-effective

Uncontrolled	1.108
Fully Controlled	0.934
Strongest Control Rod Out	0.974
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, Delta k	0.000

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Delta k) (20 deg.C, Xenon Free)
660	0.065

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

Values normally reported in this section are REDY inputs. There were no transients aralyzed using REDY.

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (S.2 .2)

Exposure: BOC2 TO EOC2

Fuel Design	Peaking Factors Local	Radial	Axial	R-Factor	Bundle Power(MWT)	Bundle Flow (1000 lb/hr)	Initial MCPR
BP/P8x8R	1.20	1.55	1.40	1.051	7.054	107.1	1.12

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

Transient Recategorization:	No
Recirculation Pump Trip:	Yes
Rod Withdrawal Limiter:	Yes
Thermal Power Monitor:	Yes
Improved Scram Time:	No
Exposure Dependent Limits:	No
Exposure Points Analyzed:	1

9. OPERATING FLEXIBILITY OPTIONS (S.2.2.3)

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

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

Methods Used: GEMINI

Transient	Flux (% NBR)	Q/A (% NBR)	Delta CPR BP/P8x8R	Figure
Exposure Range: BOC2 to EOC2				
Pressure Regulator Failure Downscale	141	103	0.04	2
Loss of Feedwater Heating	*	*	0.11	*
Feedwater Controller Failure	111	105	0.05	3
Load Rejection Without Bypass	172	101	0.01	4

*See Appendix B

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

The generic bounding BWR/6 Rod Withdrawal Error analysis described in NEDE-24011-P-A-8-US is applied; the resulting delta CPR is 0.11.

12. CYCLE MCPR VALUES (S.2.2)*

Non-Pressurization Events

Exposure Range: BOC2 to EOC2

	BP/P8x8R
Loss of 100°F Feedwater Heating	1.18
Rod Withdrawal Error	1.18

Pressurization Events

Exposure Range: BOC2 to EOC2

	Option A BP/P8x8R
Pressure Regulator Failure Downscale	1.12
Feedwater Controller Failure	1.13
Load Rejection Without Bypass	1.09

*GEMINI ODYN adjustment factors are provided in the letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "GEMINI ODYN Adjustment Factors for BWR/6," dated July 6, 1987.

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	Steam Line Pressure (psig)	Vessel Pressure (psig)	Plant Response
MSIV Closure (Flux Scram)	1204	1247	Figure 5

14. LOADING ERROR RESULTS (S.2.5.4)

Loading Error Results are not applicable for BWR/6 plants. NRC approval of the non-applicability of Loading Errors to BWR/6 plants is documented in Section S.2.5.4 of NEDE-24011-P-A-8-US, entitled "Loading Error Accident Calculational Methods."

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

Banked Position Withdrawal Sequence is utilized at the Clinton Power Station; therefore, the bounding Control Rod Drop Analysis (CRDA) described in NEDE-24011-P-A-8-US is applied. NRC approval of the bounding analysis is given in the letter to J.S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011, Revision 6, Amendment 9 'GESTAR-II General Electric Standard Application for Reactor Fuel'," January 25, 1985.

16. STABILITY ANALYSIS RESULTS (S.2.4)

GE SIL-380 recommendations have been included in the Clinton Power Station operating procedures and Technical Specifications; therefore, no stability analysis is required. Furthermore, Clinton Power Station operating procedures require an immediate manual scram if both recirculation pumps trip.

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

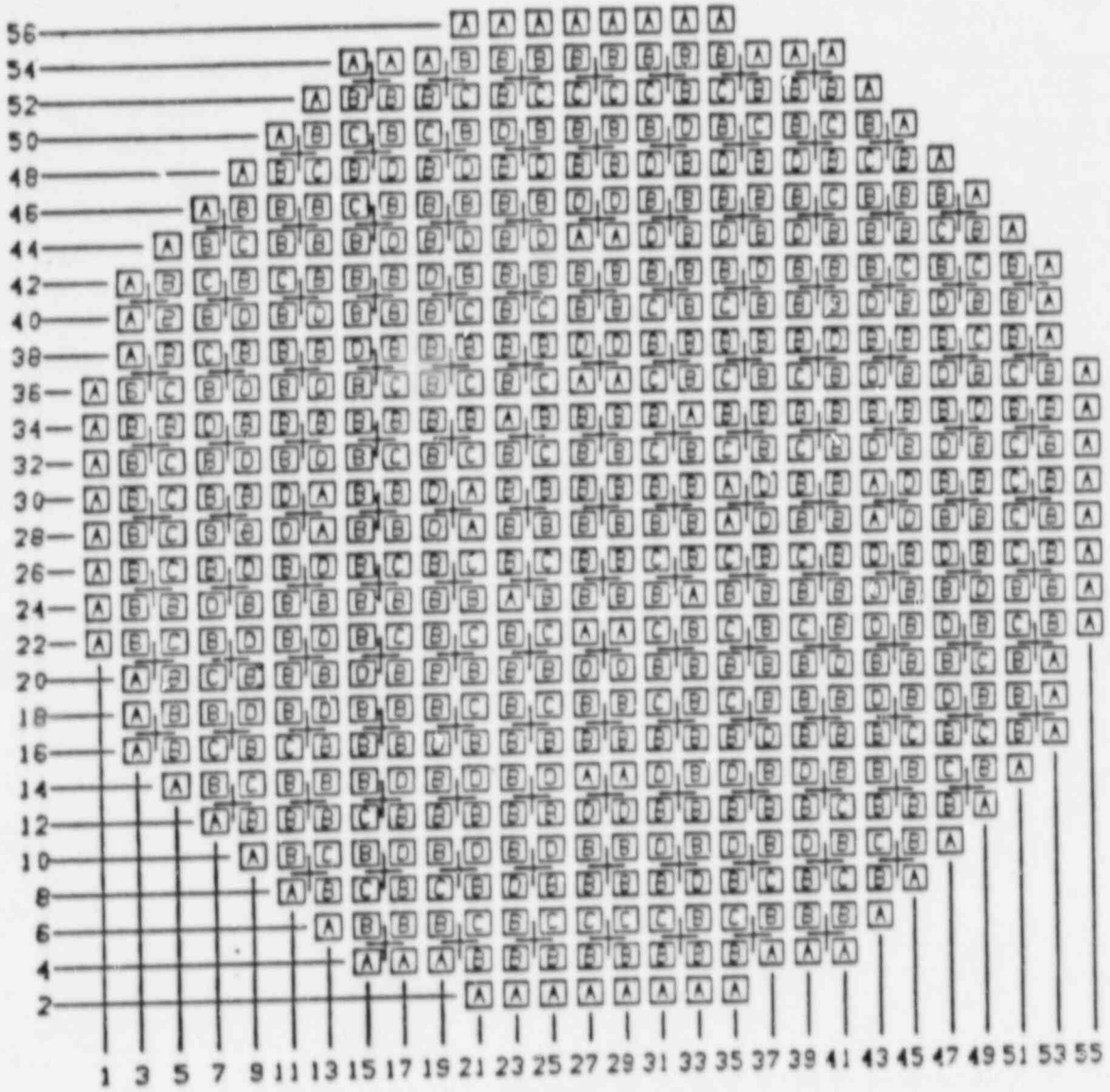
LOCA Method Used: SAFE/REFLOOD (see Clinton Power Station Final Safety Analysis Report)

Fuel Type: BP8SRB284L

Average Planar Exposure (GWd/MT)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
0.2	11.7	2018	0.016
1.1	11.8	2016	0.016
5.5	12.4	2053	0.017
11.0	12.6	2067	0.017
16.5	12.6	2075	0.018
22.0	12.5	2078	0.018
27.6	11.9	2007	0.014
38.6	10.8	1850	0.008
49.6	9.4	1681	0.004

Fuel Type: BP8SRB284LC

Average Planar Exposure (GWd/MT)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
0.2	11.6	2007	0.016
1.1	11.8	2014	0.016
5.5	12.6	2071	0.018
11.0	12.6	2064	0.017
16.5	12.6	2073	0.018
22.0	12.6	2078	0.018
27.6	11.9	2008	0.014
38.6	10.8	1852	0.008
49.6	9.5	1680	0.004



FUEL TYPE	
A = P8SRB154	C = BP8SRB284LC
B = P8SRB200	D = BP8SRB284L

Figure 1. Reference Core Loading Pattern

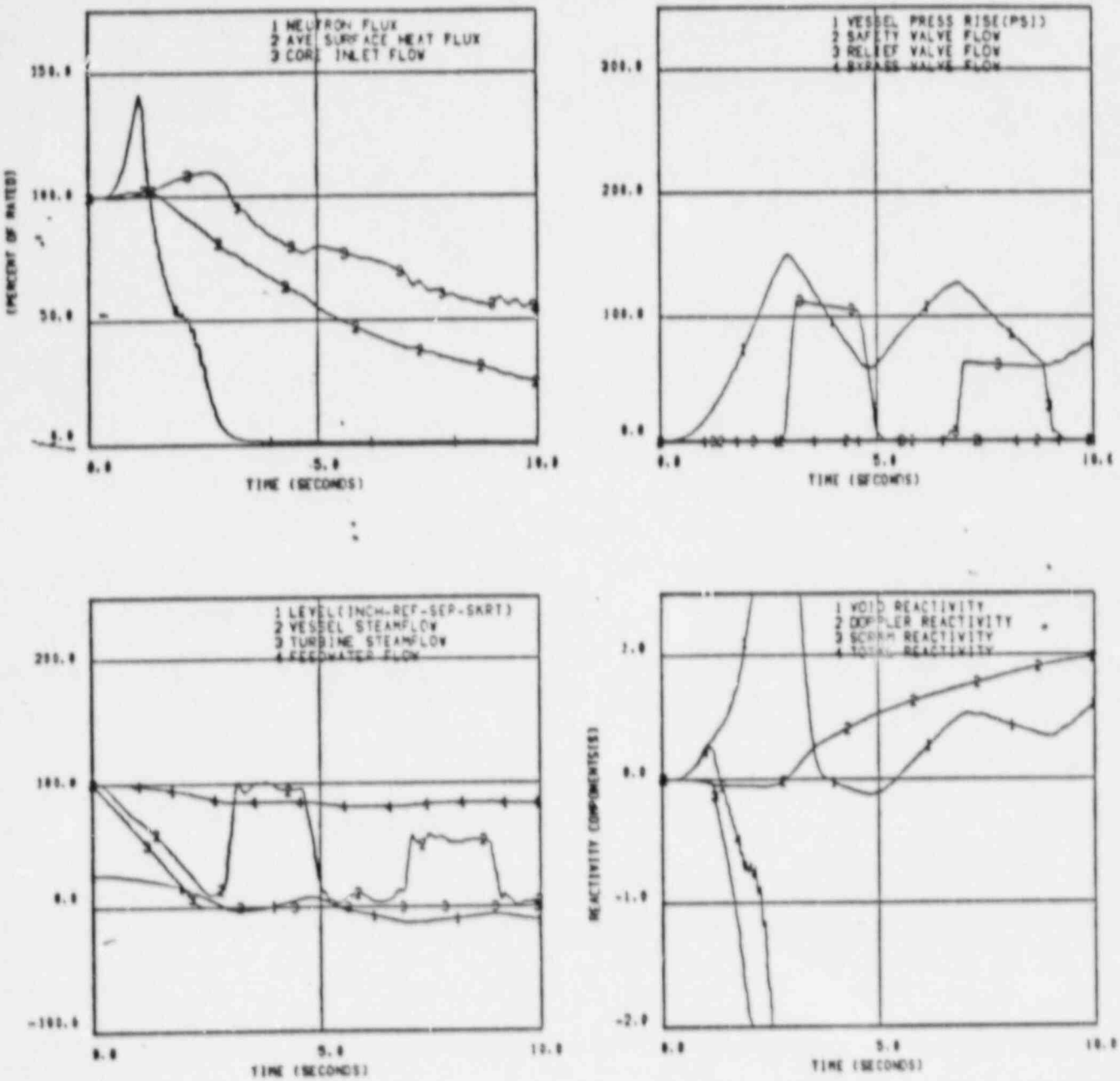


Figure 2. Plant Response to Pressure Regulator Failure Downscale (BOC2 to EOC2)

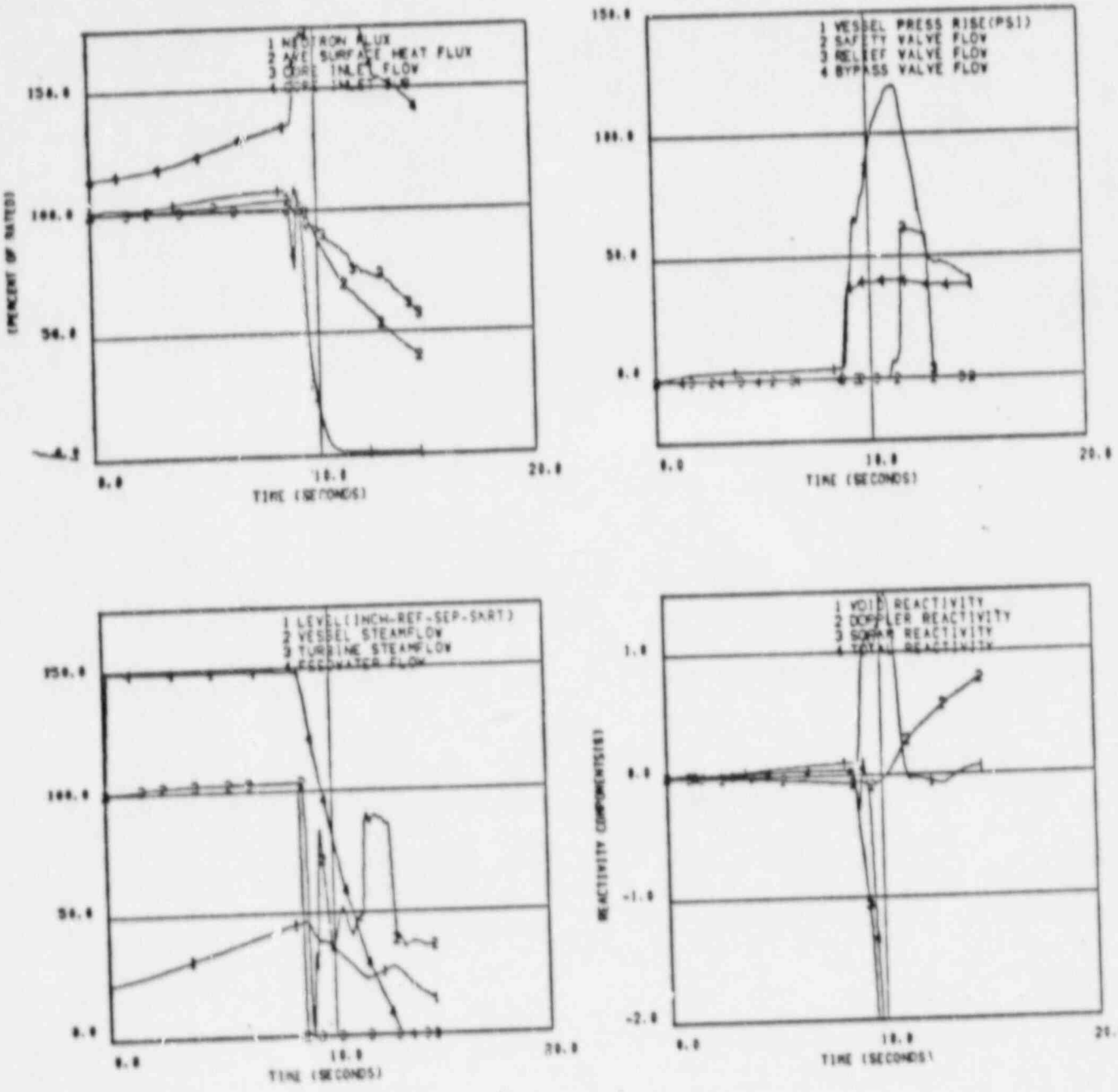


Figure 3. Plant Response to Feedwater Controller Failure (BOC2 to EOC2)

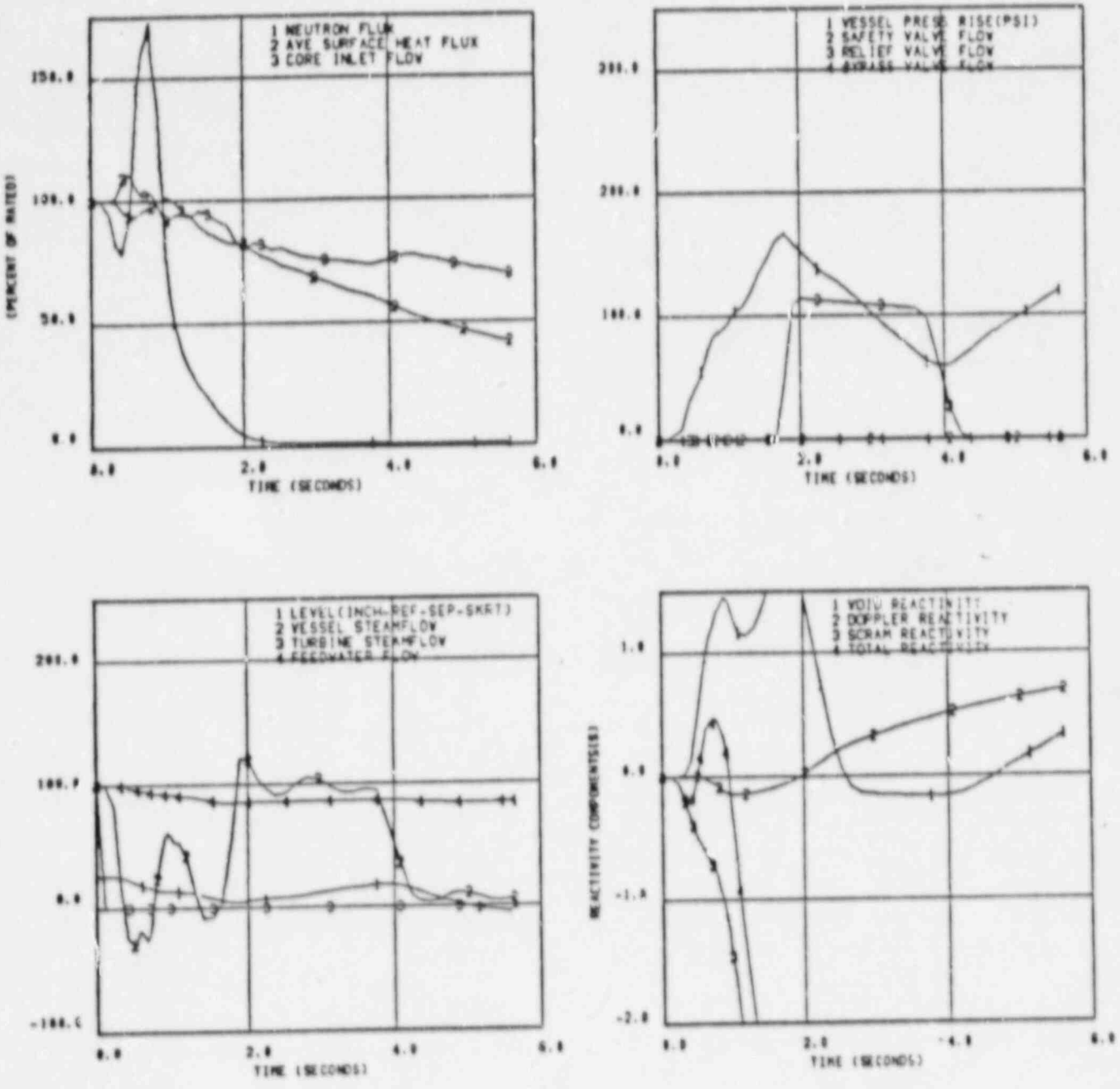


Figure 4. Plant Response to Generator Load Rejection Without Bypass (BOC2 to EOC2)

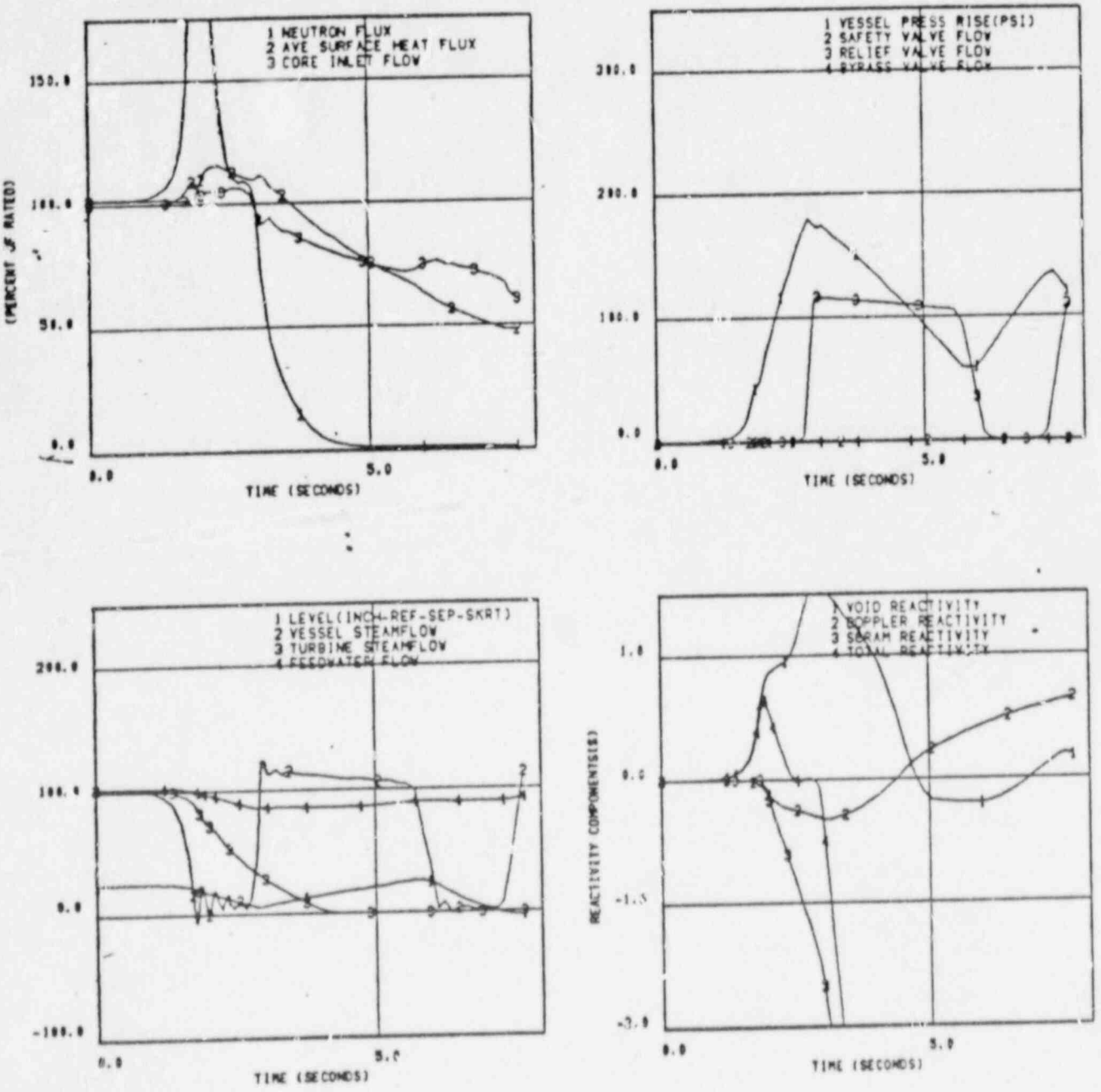


Figure 5. Plant Response to MSIV Closure (Flux Scram)

APPENDIX A

GETAB AND TRANSIENT ANALYSIS INITIAL CONDITIONS

To accurately reflect actual plant parameters, the values shown in Table A-1 were used instead of the values reported in NEDE-24011-P-A-8-US, May 1986.

Table A-1

PLANT PARAMETER

Parameter	Analysis Value
Thermal Power, MWt	2894
Rated Steamflow, 10^6 lb/hr	12.45
Dome Pressure, psig	1025
Turbine Pressure, psig	980
Non-Fuel Power Fraction	0.038
Dual Mode Safety/Relief Valves	
Number of Valves	16
Relief Mode Low Setpoint, psig	1133
Safety Mode Low Setpoint, psig	1180
Capacity, lb/hr	924,933
(Ref. Pressure psig)	(1190)

APPENDIX B

BASIS FOR ANALYSIS OF LOSS-OF-FEEDWATER HEATER EVENT

The Loss-of-Feedwater Heater event was analyzed with the 3D BWR Simulator code described in NEDE-24011-P-A-8-US. The transient analysis inputs normally reported in Section 6 of the licensing submittal are internally calculated in the 3D BWR Simulator code and in ODYN. The transient plots, flux, and Q/A normally reported in Section 10 are not outputs of the 3D BWR Simulator code; therefore, these items are not included in this document.

APPENDIX C
APPLICATION OF GEMINI METHODS

The GEMINI system of methods are used to perform the licensing analyses of Clinton Power Station (CPS) Reload 1. The GEMINI system of methods is described in Reference 1; NRC approval of these methods is documented in Reference 2. In Reference 3, the application of GEMINI methods in licensing analyses is described. Pressurization events that could establish the Operating Limit MCPR are analyzed at the 100% power level. Power level uncertainties specified in Regulatory Guide 1.49 are accounted for by adding adjustment factors to the calculated delta CPR. NRC approval of this procedure is provided in Reference 4.

Rod Withdrawal Error

The NRC approved generic Rod Withdrawal Error analysis for BWR/6's described in Reference 5 is applied to CPS Reload 1. An evaluation of the impact of GEMINI methods on the generic analysis indicates that the results of the generic analysis continue to be conservative and bounding.

Overpressurization Analysis

The MSIV Closure (Flux Scram) analysis is performed using GEMINI methods at the 102% power level to account for the power level uncertainties specified in Regulatory Guide 1.49.

Control Rod Drop Accident

The NRC approved bounding Control Rod Drop Accident analysis for Banked Position Withdrawal Sequence plants (such as CPS) described in Reference 1 is applied to CPS Reload 1. The impact of GEMINI methods on the results of the generic analysis is negligible.

Stability

The NRC approved generic stability approach described in Reference 1 is applied to CPS Reload 1. The use of GEMINI methods does not impact the generic analysis.

References

1. Letter, J.S. Charnley (GE) to C.O. Thomas (NRC), "Amendment 11 to GE LTR NEDE-24011-P-A," February 27, 1985.

2. Letter, C.O. Thomas (NRC) to J.S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Rev. 6, Amendment 11, 'General Electric Standard Application for Reactor Fuel'," November 5, 1985.

3. Letter, J.S. Charnley (GE) to H.N. Berkow (NRC), "Revised Supplementary Information Regarding Amendment 11 to GE Licensing Topical Report NEDE-24011-P-A," January 16, 1986.

4. Letter, G.C. Lainas (NRC) to J.S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, 'GE Generic Licensing Reload Report', Supplement to Amendment 11," March 22, 1986.

5. "GESSAR-II 238 BWR/6 Nuclear Island Design," NRC Docket No. STN50-447.