ENCLOSURE 1

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR PNPP

RELOAD 2, CYCLE 3

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Supplemental Reload Licensing Submittal

for

Perry Nuclear Power Plant Unit 1

Reload 2 Cycle 3

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Important Notice Regarding Contents of This Report

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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 P. A. Hahn and J. L. Casillas of the Fuel Engineering Section. The Supplemental Reload Licensing Submittal was prepared by P. A. Lambert and verified by J. L. Embley of Regulatory and Analysis Services.

1. **Plant-unique Items (1.0)***

Appendix A: Analysis Conditions

Appendix B: Basis For Analysis of Loss-of-feedwater Heating Event

Appendix C: Analyzed Operating Domain

2. **Reload Fuel Bundles (1.0 and 2.0)**

Fuel Type	Cycle Loaded	Number
Irradiated:		
BP8SRB219	1	64 **
BP8SRB176	1	140 **
BS301E	2	136
BS301F	2	136
New:		
GE8B-P8SQB320-9GZ-120M-150-T	3	104
GE8B-P8SQB322-7GZ-120M-150-T	3	168
Total		748

3. **Reference Core Loading Pattern (3.2.1)**

Nominal previous cycle core average exposure at end of cycle:	16,436 MWd/MT
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	16,105 MWd/MT
Assumed reload cycle core average exposure at end of cycle:	18,357 MWd/MT
Core loading pattern:	Figure 1

*( ) refers to area of discussion in *General Electric Standard Application for Reactor Fuel*, NEDE-24011-P-A-9, September, 1988; a letter "S" preceding the number refers to the U.S. Supplement, NEDE-24011-P-A-9-US, September 1988.

**4 BP8SRB219 fuel bundles and 140 BP8SRB176 fuel bundles were removed for the second cycle; these fuel bundles are to be reinstalled for the third cycle.

4. **Calculated Core Effective Multiplication and Control System Worth - No Voids, 20°C (3.2.4.1 and 3.2.4.2)**

Beginning of Cycle, $K_{effective}$

Uncontrolled	1.127
Fully controlled	0.953
Strongest control rod out	0.985
R, Maximum increase in cold core reactivity with exposure into cycle, ΔK	0.003

5. **Standby Liquid Control System Shutdown Capability (3.2.4.3)**

Boron (ppm)	Shutdown Margin (ΔK) (20°C, Xenon Free)
660	0.029

6. **Reload Unique GETAB AOO Analysis Initial Condition Parameters (S.2.3)**

Exposure: **BOC 3 to EOC 3**

Fuel Design	Peaking Factors Local	Radial	Axial	R-Factor	Bundle Power (MWt)	Bundle Flow (1,000 lb/hr)	Initial MCPR
GE8x8EB	1.20	1.55	1.40	1.051	7.239	117.5	1.17
BP8x8R	1.20	1.54	1.40	1.051	7.166	117.1	1.18

7. **Selected Margin Improvement Options (S.5.1)**

Recirculation pump trip:	Yes
Rod withdrawal limiter:	Yes
Thermal power monitor:	Yes
Measured scram time:	No
Exposure dependent limits:	No
Exposure points analyzed:	1 (EOC)

8. **Operating Flexibility Options (S.5.2)**

Single-loop operation:	Yes
Load line limit:	No
Extended load line limit:	No
Increased core flow:	Yes
Flow point analyzed, %:	105
Feedwater temperature reduction:	Yes
ARTS Program:	No
Maximum extended operating domain:	Yes

9. **Core-wide AOO Analysis Results (S.2.2)***

Methods used: GEMINI and GEXL-PLUS

Event	Flux (% NBR)	Q/A (% NBR)	ΔCPR GE8x8EB	ΔCPR BP8x8R	Figure
Exposure range: BOC 3 to EOC 3					
Load rejection without bypass	280	108	0.06	0.06	2
Feedwater controller failure (143%)	203	112	0.10	0.10	3
Pressure regulator failure downscale	150	105	0.05	0.05	4
Loss of 100°F feedwater heating	**	**	0.11	0.11	**

*Limiting values listed; see Appendix C.
** See Appendix B.

10. **Local Rod Withdrawal Error (With Limiting Instrument Failure) AOO Summary (S.2.2.1.5)**

The generic bounding BWR/6 rod withdrawal error is analyzed in NEDE-24011-P-A-9-US and GESSAR-II Appendix 15B is applied; the resulting ΔCPR is 0.11. The original generic analysis in GESSAR-II was not applicable for control cell core operation; however, it was subsequently shown to be applicable for control cell core operation and GESSAR-II was revised to reflect this application in Revision 21.

11. **Cycle MCPR Values* (4.3.1 and S.2.2)**

Safety limit: 1.07

Exposure range: BOC 3 to EOC 3

Non-pressurization events:

	GE8x8EB	BP8x8R
Rod withdrawal error	1.18	1.18
Loss of 100°F feedwater heating	1.18	1.18

Pressurization events:

	Option A	
	GE8x8EB	BP8x8R
Load rejection without bypass	1.13	1.13
Feedwater controller failure	1.18	1.18
Pressure regulator failure downscale	1.13	1.13

* 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. The MCPR limit does not change because of channel bow. Channel bow is reflected in the monitoring of the core.

12. **Overpressurization Analysis Summary (S.3)**

Event	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV closure (flux scram)*	1226	1258	Figure 5

13. **Loading Error Results (S.2.2.3.7)**

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.2.2.3.7 of NEDE-24011-P-A-9-US.

14. **Control Rod Drop Analysis Results (S.2.2.3.1)**

Banked Position Withdrawal Sequence is utilized at the Perry Nuclear Power Plant Unit 1; therefore, the bounding control rod drop analysis (CRDA) described in NEDE-24011-P-A-9-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.

15. **Stability Analysis Results (S.4)**

GE SIL-380 recommendations have been included in the Perry Nuclear Power Plant Unit 1 operating procedures and/or Technical Specifications and, therefore, the stability analysis is not required. NRC approval for deletion of a cycle-specific stability analysis is documented in Amendment 8 to NEDE-24011-P-A-9-US. In addition, the Perry Nuclear Power Plant Unit 1 recognizes the issuance of NRC Bulletin No. 88-07, Supplement 1, *Power Oscillations in Boiling Water Reactors (BWRs)*, and will continue to comply with the recommendations contained herein.

*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. The analysis was performed with 13 highest setpoint safety valves operational.

16. **Loss-of-coolant Accident Results (S.2.2.3.2)**

LOCA method used: SAFE/REFLOOD (see the Perry Nuclear Power Plant Unit 1 Updated Safety Analysis Report, as amended)

Bundle Type: GE8B-P8SQB320-9GZ-120M-150-T (GE8X8EB)*

Average Planar Exposure		MAPLHGR (kw/ft)	
(GWd/ST)	(GWd/MT)	Most Limiting	Least Limiting
0.0	0.0	11.75	11.76
0.2	0.2	11.78	11.79
1.0	1.1	11.83	11.90
2.0	2.2	11.91	12.00
3.0	3.3	12.02	12.13
4.0	4.4	12.17	12.27
5.0	5.5	12.32	12.40
6.0	6.6	12.44	12.53
7.0	7.7	12.56	12.67
8.0	8.8	12.70	12.82
9.0	9.9	12.84	12.96
10.0	11.0	12.97	13.07
12.5	13.8	13.00	13.03
15.0	16.5	12.73	12.74
20.0	22.0	12.10	12.12
25.0	27.6	11.48	11.49
35.0	38.6	10.23	10.24
45.0	49.6	8.66	8.68
50.0	55.1	6.16	6.18

The Peak Clad Temperature (PCT) is ≤2105°F at all exposures; the Local Oxidation (Fraction) is ≤0.049 at all exposures.

*MAPLHGR multiplier for single-loop operation (SLO) is 0.80.

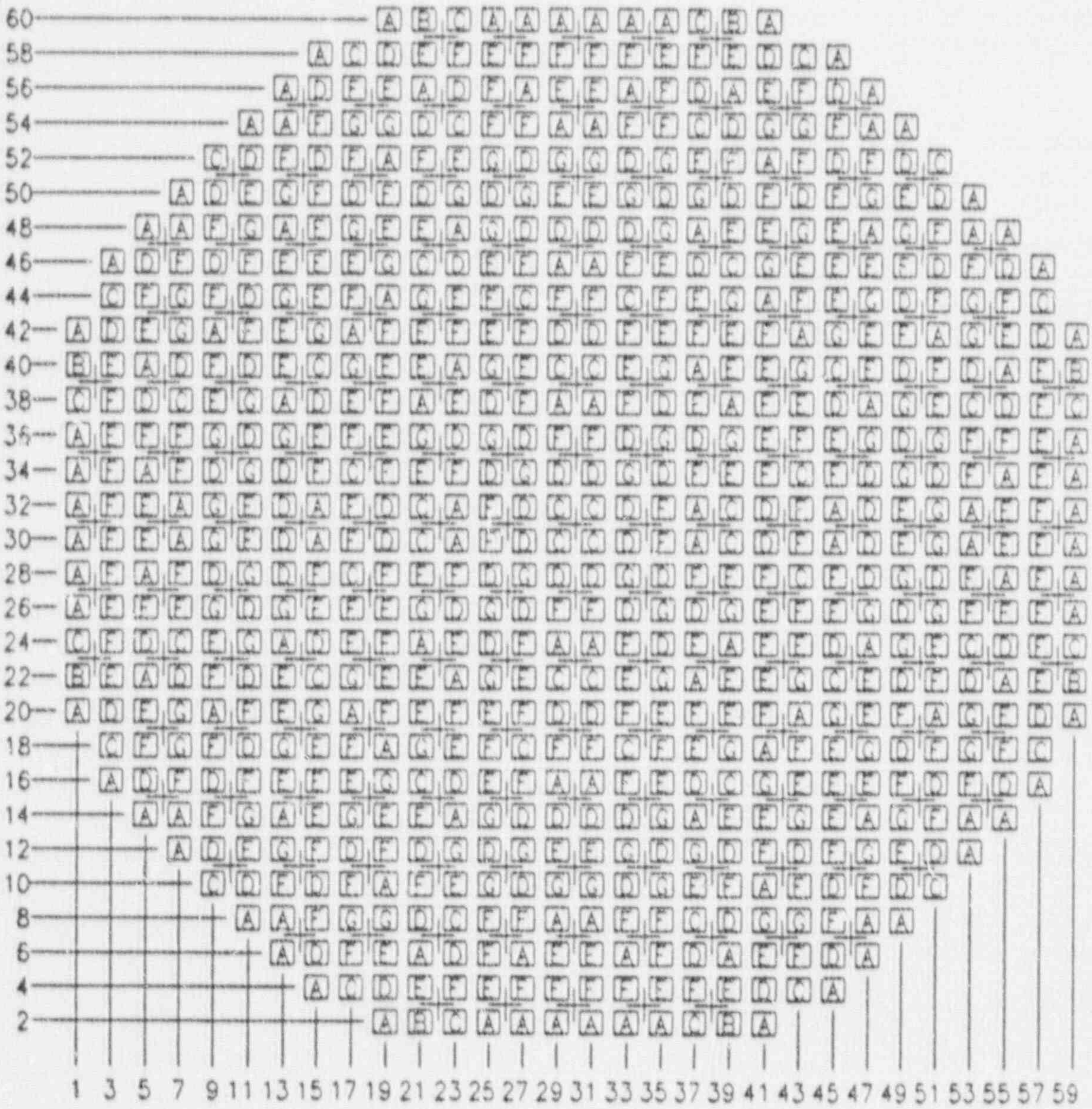
16. **Loss-of-coolant Accident Results (S.2.2.3.2) (continued)**

Bundle Type: GE8B-P8SQB322-7GZ-120M-150-T (GE8x8EB)*

Average Planar Exposure		MAPLHGR (kw/ft)	
(GWd/ST)	(GWd/MT)	Most Limiting	Least Limiting
0.0	0.0	12.11	12.13
0.2	0.2	12.10	12.13
1.0	1.1	12.09	12.14
2.0	2.2	12.16	12.22
3.0	3.3	12.28	12.35
4.0	4.4	12.42	12.51
5.0	5.5	12.58	12.67
6.0	6.6	12.67	12.77
7.0	7.7	12.75	12.86
8.0	8.8	12.83	12.95
9.0	9.9	12.92	13.04
10.0	11.0	13.02	13.11
12.5	13.8	13.07	13.09
15.0	16.5	12.79	12.79
20.0	22.0	12.19	12.19
25.0	27.6	11.56	11.56
35.0	38.6	10.29	10.30
45.0	49.6	8.77	8.80
50.0	55.1	6.27	6.30

The Peak Clad Temperature (PCT) is ≤2100°F at all exposures; the Local Oxidation (Fraction) is ≤0.048 at all exposures.

*MAPLHGR multiplier for single-loop operation (SLO) is 0.80.



FUEL TYPE	
A = BP8SRB176	E = BS301E
B = BP8SRB219	F = GE8B-P8SQB322-7GZ-120M-150-T
C = BP8SRB219	G = GE8B-P8SQB320-9GZ-120M-150-T
D = BS301F	

Figure 1 Reference Core Loading Pattern

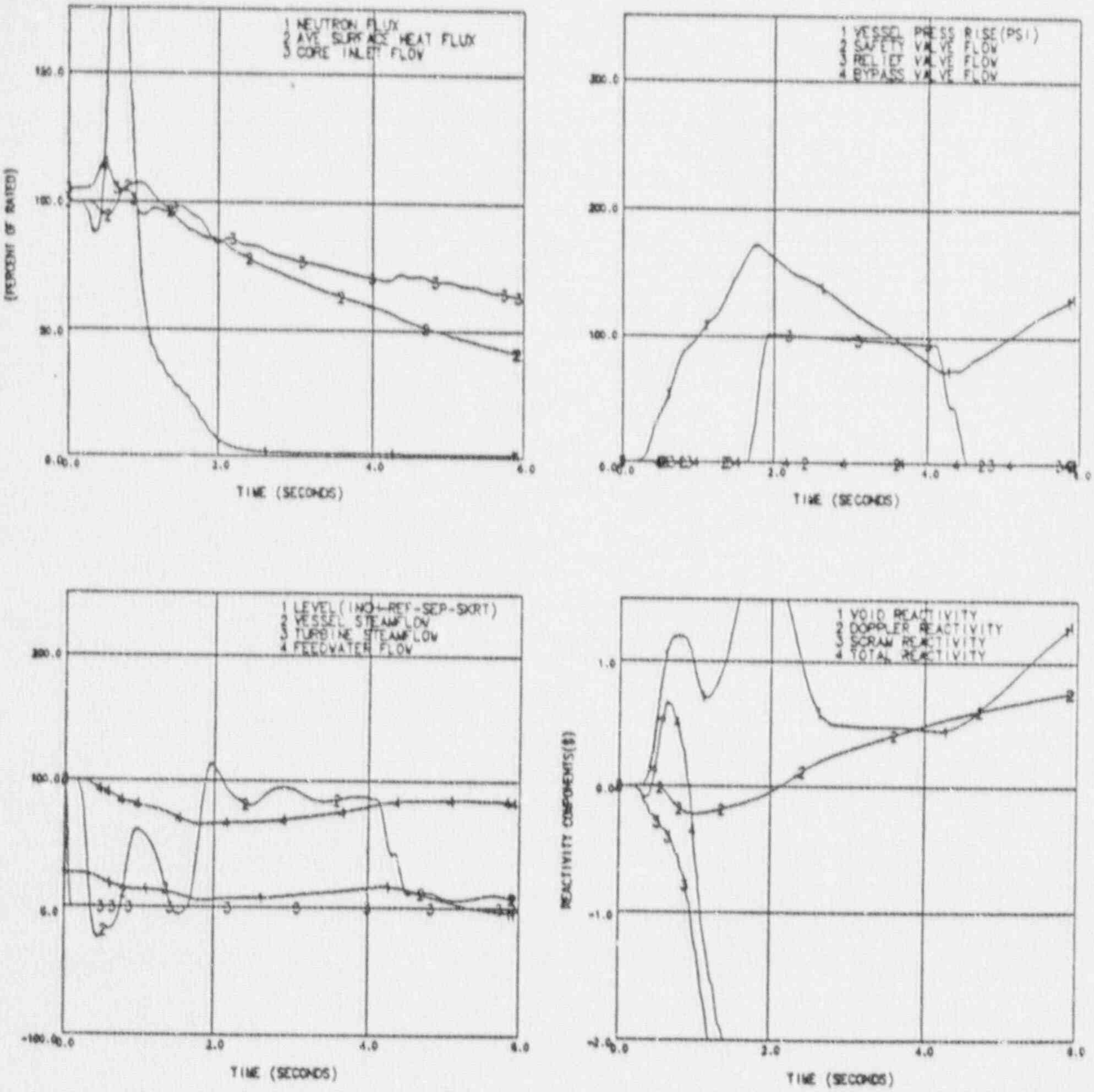


Figure 2. Plant Response to Load Rejection without Bypass

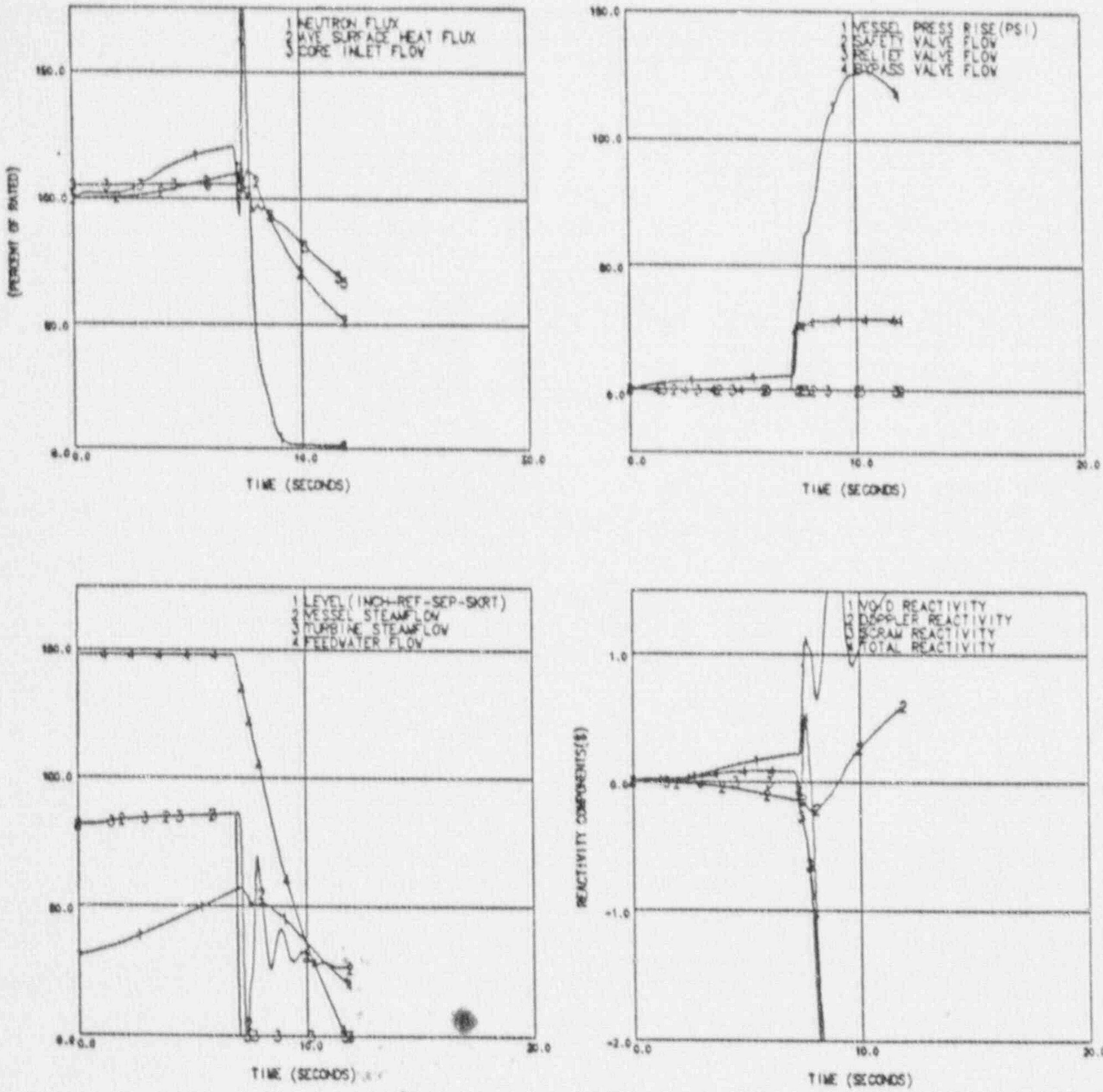


Figure 3. Plant Response to Feedwater Controller Failure

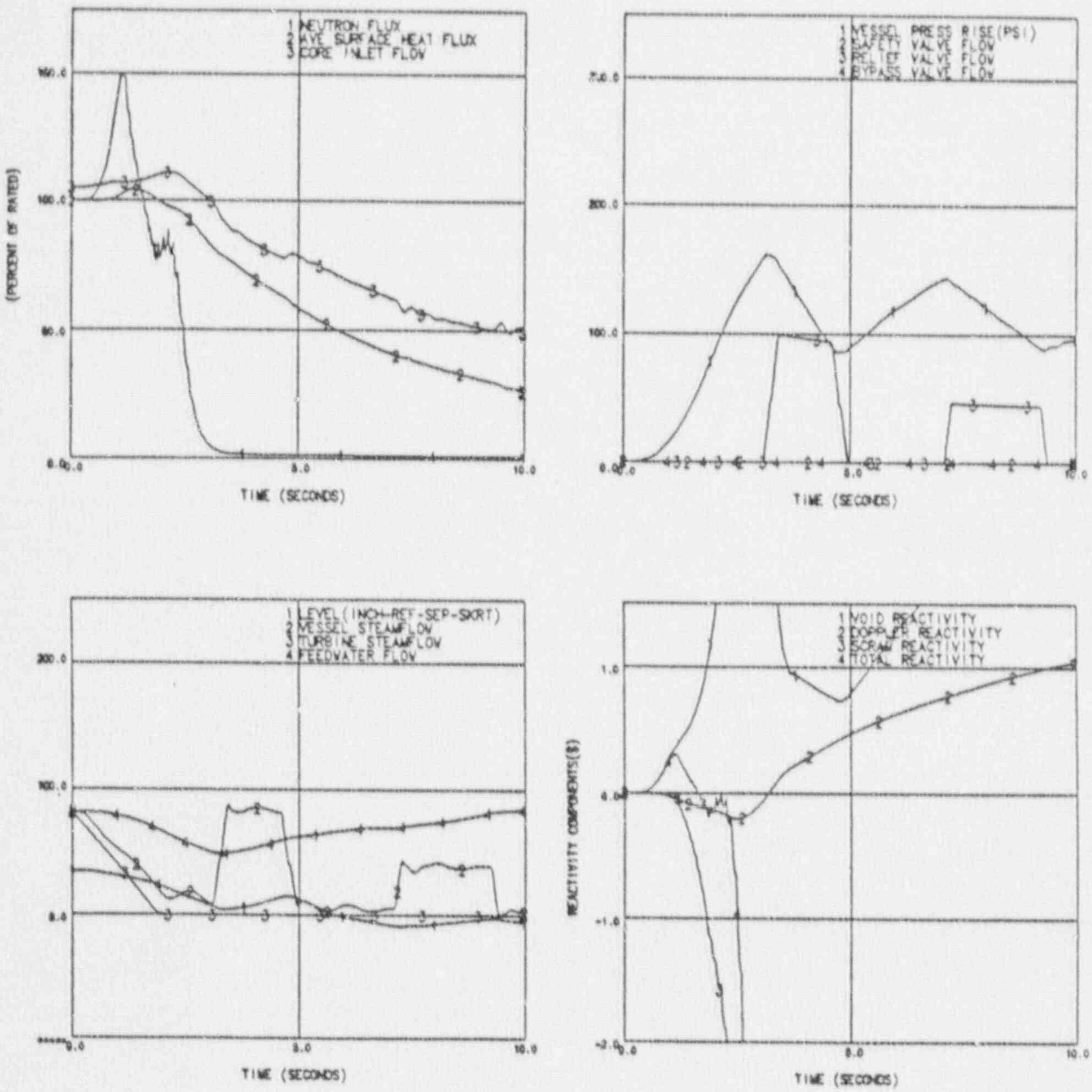


Figure 4. Plant Response to Pressure Regulator Failure Downscale

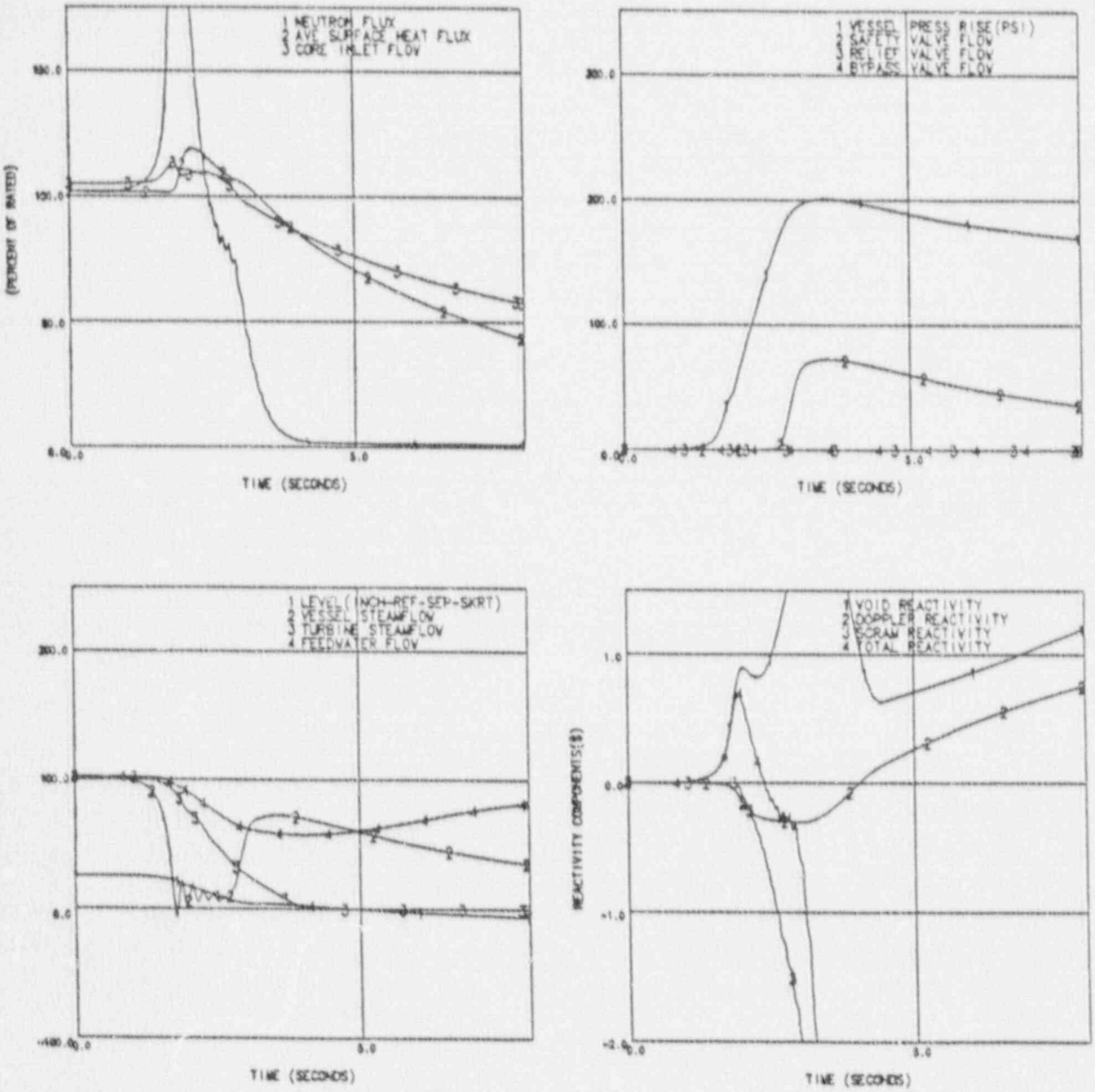


Figure 5. Plant Response to MSIV Closure, Flux Scram

Appendix A

Analysis Conditions

To reflect actual plant parameters accurately, the values shown in Table A-1 were used this cycle to reflect the bounding conditions.

Table A-1

Parameter	Analysis Value	
	250°F FW Temp.	420°F FW Temp.
Thermal power, MWt	3579	3579
Dome pressure, psig	1008	1026
Rated steam flow, Mlb/hr	12.58	15.40
Turbine pressure, psig	974	976
Core flow, Mlb/hr	109.2	109.2
Reactor pressure, psia	1056	1056
Inlet enthalpy, Btu/lb	512.4	528.8
Non-fuel power fraction	0.038	0.038
No. of dual mode Safety/Relief Valves	17*	17*
Relief mode lowest setpoint, psig	1143*	1143*
Safety mode lowest setpoint, psig	1177	1177

*There are a total of 19 valves; the 2 lowest setpoint safety/relief valves are assumed to be out-of-service in the transient analyses.

Appendix B

Basis for Analysis of Loss-of-feedwater Heating Event

The loss-of-feedwater heating event was analyzed using the BWR Simulator Code (Reference B-1). The use of this code is permitted in GESTAR II (Reference B-2). The transient plots, neutron flux and heat flux values normally reported in Section 10 are not an output of the BWR Simulator code; therefore, these items are not included in this document.

The transient analysis inputs normally reported in Section 6 of the licensing submittal are internally calculated in the BWR Simulator Code and in ODYN.

References

B-1 *Steady-State Nuclear Methods,* NEDE-30130-P-A and NEDO-30130-A, April 1985.

B-2 *General Electric Standard Application for Reactor Fuel,* NEDE-24011-P-A-9, September 1988.

Appendix C

Analyzed Operating Domain

The core-wide abnormal operational occurrence (AOO) analysis results reported in Section 9 are the most limiting values over the entire allowable operating range. This range covers the following operating options:

1. Standard 100% power/flow map;

2. End-of-cycle power coastdown;

3. MEOD with 100% power flow range from 75% to 105% of rated; and

4. Partial feedwater heating to 320°F during the cycle with final feedwater temperature reduction to 250°F after *All Rods Out* at end of cycle.

Limiting events and conditions analyzed are based on Reference C-1 and the USAR analytical results. The Reload 2/Cycle 3 analyses were performed assuming all four turbine control valves in a full arc mode of operation. This is conservative for partial arc configuration.

The single-loop operation (SLO) analysis was reverified for the standard power/flow map with normal feedwater temperature.

<u>References</u>

C-1 *General Electric Standard Application for Reactor Fuel,* NEDE-24011-P-A-9-US, September 1988.

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