GE Nuclear Energy

Company Proprietary Information

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SUPPLEMENTAL RELOAD LICENSING REPORT

for

COOPER NUCLEAR STATION RELOAD 16 CYCLE 17

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Supplemental Reload Licensing Report

for

Cooper Nuclear Station

Reload 16 Cycle 17

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Important Notice Regarding

Contents of This Report

Please Read Carefully

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Acknowledgement

The engineering and reload licensing analyses, which form the technical basis of this Supplemental Reload Licensing Report, were performed by A. F. Alzaben. The Supplemental Reload Licensing Report was prepared by A. F. Alzaben. This document has been reviewed by W. E. Russell of Fuel Engineering and C. W. Smith of Fuel Licensing.

The basis for this report is *General Electric Standard Application for Reactor Fuel*, NEDE–24011–P–A–10, February 1991; and the U.S. Supplement, NEDE–24011–P–A–10–US, March 1991.

1. Plant–unique Items

Appendix A: Analysis Conditions
Appendix B: Decrease in Core Coolant Temperature Events
Appendix C: SRV Tolerance Analysis

2. Reload Fuel Bundles

Fuel Type	Cycle Loaded	Number
Irradiated:		
GE9B–P8DWB302–10GZ–80M–150–T (GE8x8NB)	14	48
GE9B–P8DWB320–10GZ1–80M–150–T (GE8x8NB)	15	164
GE9B–P8DWB348–11GZ–80M–150–T (GE8x8NB)	16	136
GE9B–P8DWB348–12GZ–80M–150–T (GE8x8NB)	16	48
New:		
GE9B–P8DWB348–11GZ–80M–150–T (GE8x8NB)	17	152
Total		548

3. Reference Core Loading Pattern[1]

Assumed previous cycle core average exposure at end of cycle:	24717 MWd/MT (22423 MWd/ST)
Assumed reload cycle core average exposure at beginning of cycle:	15417 MWd/MT (13986 MWd/ST)
Assumed reload cycle core average exposure at end of cycle:	25888 MWd/MT (23486 MWd/ST)
Reference core loading pattern:	Figure 1

1. The End of cycle core average exposure reflects the basis for the license work.

4. Calculated Core Effective Multiplication and Control System Worth – No Voids, 20°C

Beginning of Cycle, $k_{effective}$	
Uncontrolled	1.109
Fully controlled	0.965
Strongest control rod out	0.990
R, Maximum increase in cold core reactivity with exposure into cycle, Δk	0.000

5. Standby Liquid Control System Shutdown Capability

Boron (ppm)	Shutdown Margin (Δk) (20°C, Xenon Free)
660	0.037

6. Reload Unique GETAB Anticipated Operational Occurrences (AOO) Analysis Initial Condition Parameters

Exposure: BOC17 to EHFP17–2205 MWd/MT							
	Peaking Factors						
Fuel Design	**Local**	**Radial**	**Axial**	**R–Factor**	**Bundle Power (MWt)**	**Bundle Flow (1000 lb/hr)**	**Initial MCPR**
GE8x8NB	1.20	1.78	1.40	1.000	7.522	99.8	1.14

Exposure: EHFP17–2205 MWd/MT to EHFP17							
	Peaking Factors						
Fuel Design	**Local**	**Radial**	**Axial**	**R–Factor**	**Bundle Power (MWt)**	**Bundle Flow (1000 lb/hr)**	**Initial MCPR**
GE8x8NB	1.20	1.72	1.40	1.000	7.290	101.1	1.18

7. Selected Margin Improvement Options

Recirculation pump trip:	No
Rod withdrawal limiter:	No
Thermal power monitor:	No
Improved scram time:	Yes (ODYN Option B)
Exposure dependent limits:	Yes
Exposure points analyzed:	2 (EHFP–2205 MWd/MT, EHFP)

8. Operating Flexibility Options

Single–loop operation:	Yes
Load line limit:	Yes
Extended load line limit:	Yes
Increased core flow throughout cycle:	No
Increased core flow at EOC:	No
Feedwater temperature reduction throughout cycle:	No
Final feedwater temperature reduction:	No
ARTS Program:	Yes
Maximum extended operating domain:	No
Moisture separator reheater out of service:	No
Turbine bypass system out of service:	No
Safety/relief valves out of service:	No
Feetwater heaters out of service:	No
ADS out of service:	No

9. Core–wide AOO Analysis Results

Methods used: GEMINI; GEXL–PLUS

Exposure range: BOC17 to EHFP17–2205 MWd/MT				
			Uncorrected ΔCPR	
Event	Flux (%NBR)	Q/A (%NBR)	GE8x8NB	Fig.
FW Controller Failure	187	112	0.08	2
Turbine Trip w/o Bypass	276	111	0.07	3
Load Reject w/o Bypass	285	111	0.07	4

Exposure range: EHFP17–2205 MWd/MT to EHFP17				
			Uncorrected ΔCPR	
Event	Flux (%NBR)	Q/A (%NBR)	GE8x8NB	Fig.
FW Controller Failure	209	116	0.12	5
Turbine Trip w/o Bypass	288	115	0.12	6
Load Reject w/o Bypass	300	115	0.12	7

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

Rod withdrawal error (RWE) is analyzed in GE Licensing Report, *Extended Load Line Limit and ARTS Improvement Program Analyses for Cooper Nuclear Station Cycle 14*, NEDC–31892P, January 1991. A cycle–specific analysis was performed for this cycle to verify that the ARTS RWE generic limits in NEDC–31892P remain valid with the use of the Cycle 17 fuel design. The results obtained verified that the existing ARTS limits are still valid for this cycle.

11. Cycle MCPR Values[2]

Safety limit:	1.06
Single loop operation safety limit:	1.07

Non–pressurization events:

Exposure range: BOC17 to EHFP17	
Event	**GE8x8NB**
Loss of 100 °F feedwater heating	1.15
Inadvertent HPCI	1.22
Fuel loading error (Misoriented)	1.22
Fuel loading error (Mislocated)	1.22
Rod withdrawal error (for RBM setpoint to 108%)	1.19

Pressurization events:

Exposure range: BOC17 to EHFP17–2205 MWd/MT		
	Option A	**Option B**
	GE8x8NB	**GE8x8NB**
FW Controller Failure	1.20	1.18
Turbine Trip w/o Bypass	1.22	1.15
Load Reject w/o Bypass	1.22	1.15

Exposure range: EHFP17–2205 MWD/MT to EHFP17		
	Option A	**Option B**
	GE8x8NB	**GE8x8NB**
FW Controller Failure	1.22	1.19
Turbine Trip w/o Bypass	1.23	1.19
Load Reject w/o Bypass	1.23	1.19

2. For single–loop operation, the MCPR operating limit is 0.01 greater than the two–loop value.

12. Overpressurization Analysis Summary

Event	Psl (psig)	Pv (psig)	Plant Response
MSIV Closure (Flux Scram)	1217	1242	Figure 8

13. Loading Error Results

Variable water gap misoriented bundle analysis: Yes

Event	ΔCPR
Fuel loading error (Misoriented)	0.16
Fuel loading error (Mislocated)	0.16

14. Control Rod Drop Analysis Results

Cooper Nuclear Station operates in the banked position withdrawal sequence (BPWS), so the control rod drop accident analysis is not required. NRC approval to use the generic analysis is documented in NEDE–24011–P–A–US, March 1991. CNS implemented the BPWS into the Rod Worth Minimizer (RWM) as documented in License Amendment No. 117. Removal of the Rod Sequence Control System (RSCS) at CNS has been approved by the NRC in License Amendment No. 156.

15. Stability Analysis Results

GE SIL–380 recommendations have been included in the Cooper Nuclear Station Technical Specifications; therefore, no stability analysis is required as documented in the 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.

Cooper Nuclear Station recognizes the issuance of NRC Bulletin No. 88–07, Supplement 1, *Power Oscillations in Boiling Water Reactors (BWRs)*, and has taken appropriate actions to address the identified concerns.

16. Loss–of–Coolant Accident Results

LOCA method used: SAFE/REFLOOD/CHASTE

Reference the *Loss–of–Coolant Accident Analysis Report for Cooper Nuclear Power Station*, NEDO–24045, August 1977, as amended.

16. Loss-of-Coolant Accident Results (cont)

Bundle Type: GE9B-P8DWB348-11GZ-80M-150-T (GE8x8NB)

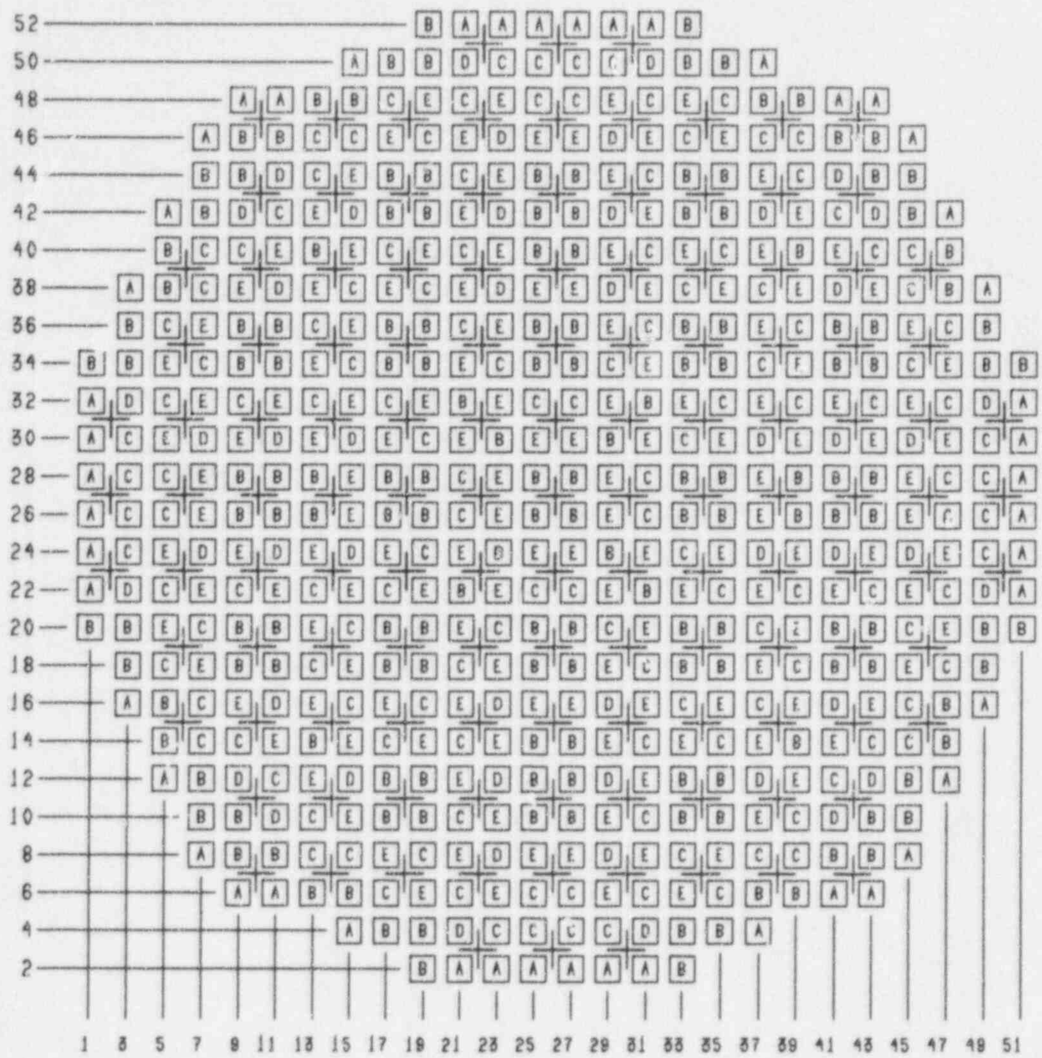
Average Planar Exposure		MAPLHGR(kW/ft)	
(GWd/ST)	(GWd/MT)	Most Limiting	Least Limiting
0.00	0.00	10.85	11.82
0.20	0.22	10.90	11.87
1.00	1.10	11.01	11.96
2.00	2.20	11.17	12.08
3.00	3.31	11.36	12.19
4.00	4.41	11.56	12.32
5.00	5.51	11.76	12.44
6.00	6.61	11.91	12.55
7.00	7.72	12.07	12.65
8.00	8.82	12.23	12.68
9.00	9.92	12.38	12.67
10.00	11.02	12.48	12.80
12.50	13.78	12.61	12.93
15.00	16.53	12.47	12.60
20.00	22.05	11.79	11.91
25.00	27.56	11.05	11.17
35.00	38.58	9.69	9.74
45.00	49.60	7.86	8.09
49.56	54.63	5.62	5.91
49.59	54.66	—	5.90
49.68	54.76	—	5.85
49.73	54.81	—	5.83

NOTE:

Peak clad temperature (PCT) are ≤ 2127 °F at all exposure and local oxidation fractions are ≤ 0.065 at all exposures.

When in single loop operation, a MAPLHGR factor of 0.75 is substituted for the LOCA analysis factors of 1.0 and 0.86 contained in the flow dependent MAPLHGR curves (K_f) that are applied to the full power nodal exposure-dependent limits.

NRC approval for single loop operation is documented in Amendment No. 94, dated September 24, 1985, to Cooper Nuclear Station Facility Operating License.



Fuel Type			
A=GE9B–P8DWB302–10GZ–80M–150–T	(Cycle 14)	D=GE9B–P8DWB348–12GZ–80M–150–T	(Cycle 16)
B=GE9B–P8DWB320–10GZ1–80M–150–T	(Cycle 15)	E=GE9B–P8DWB348–11GZ–80M–150–T	(Cycle 17)
C=GE9B–P8DWB348–11GZ–80M–150–T	(Cycle 16)		

Figure 1 Reference Core Loading Pattern

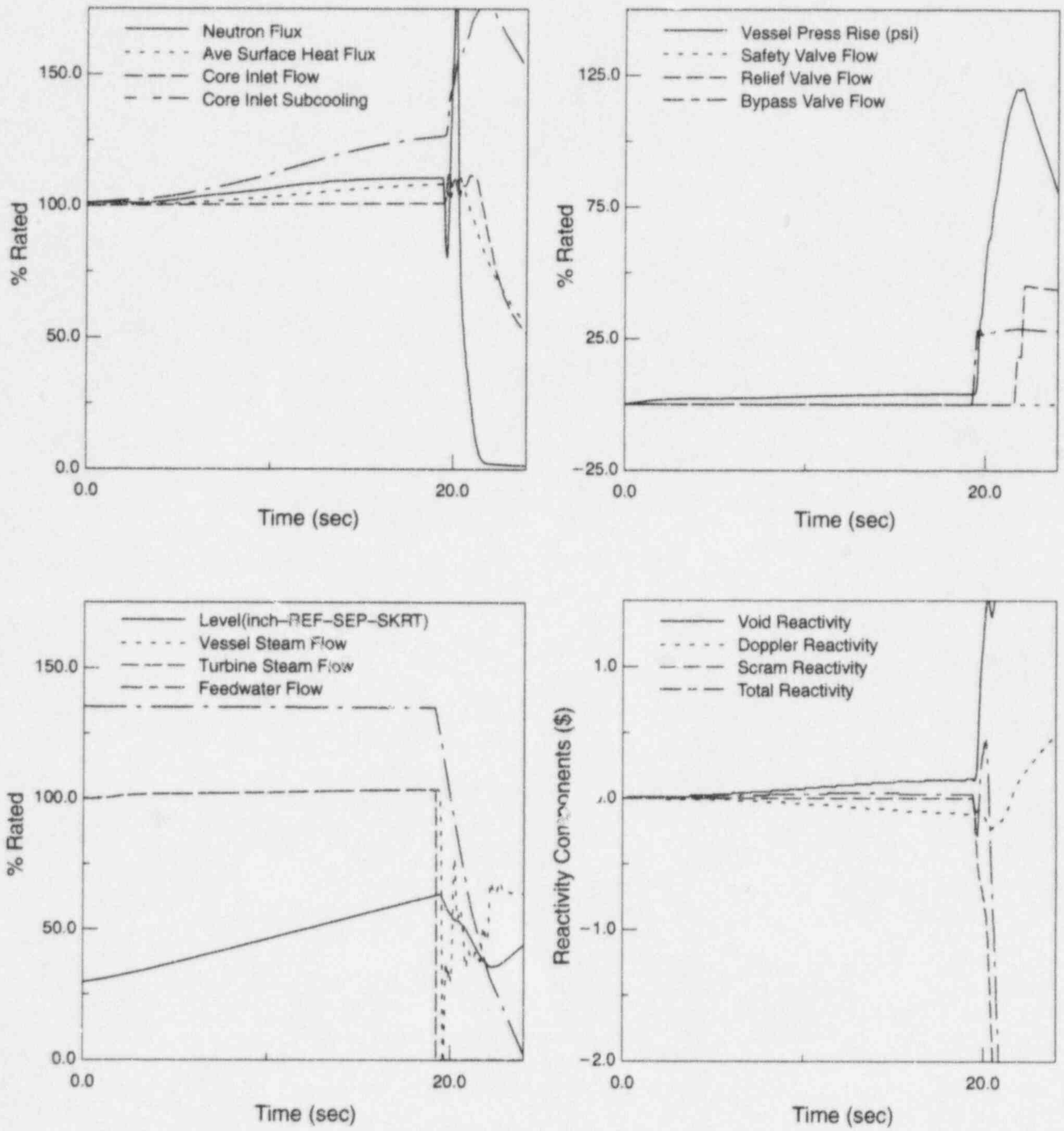


Figure 2 Plant Response to FW Controller Failure
(BOC17 to EHFP17–2205 MWd/MT)

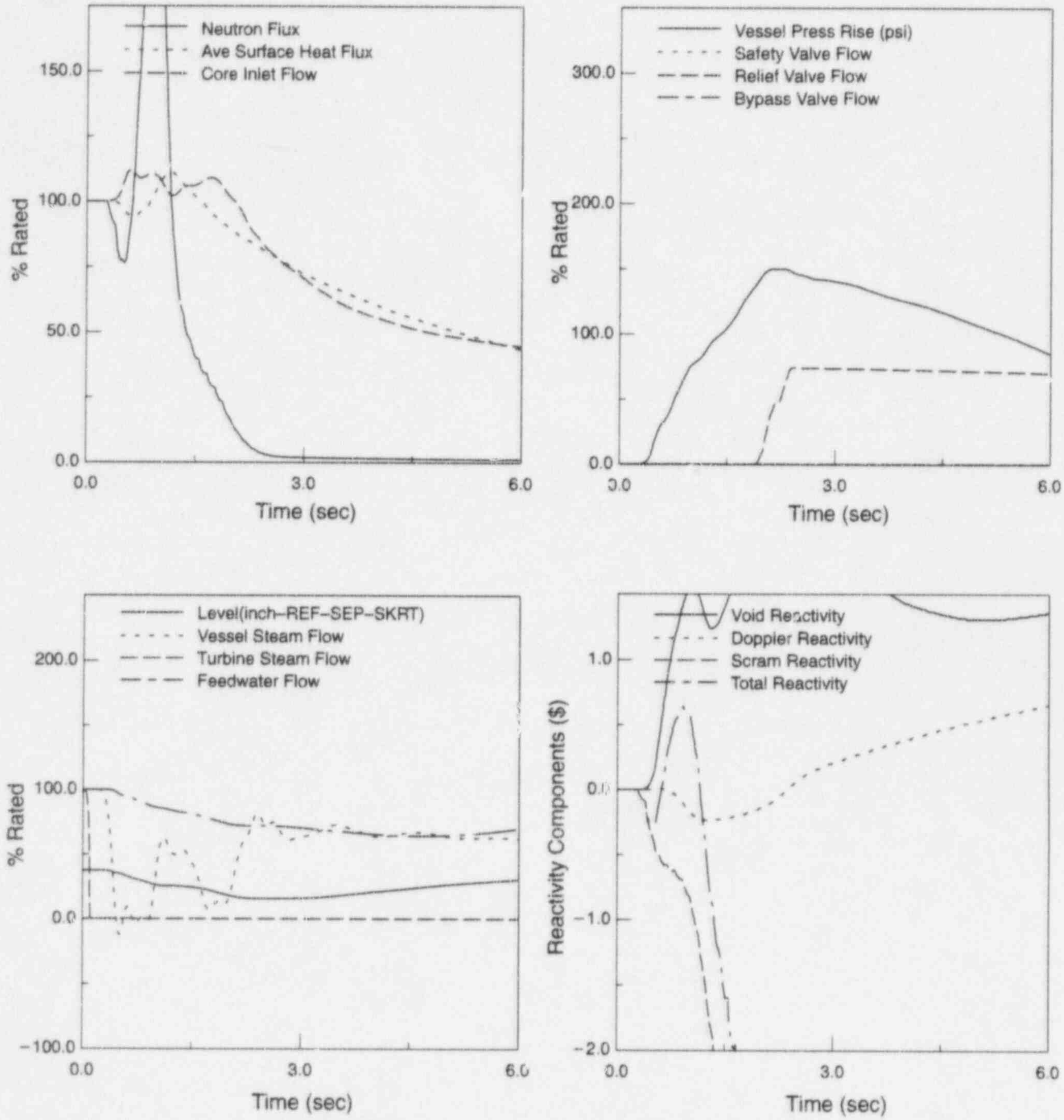


Figure 3 Plant Response to Turbine Trip w/o Bypass (BOC17 to EHFP17-2205 MWd/MT)

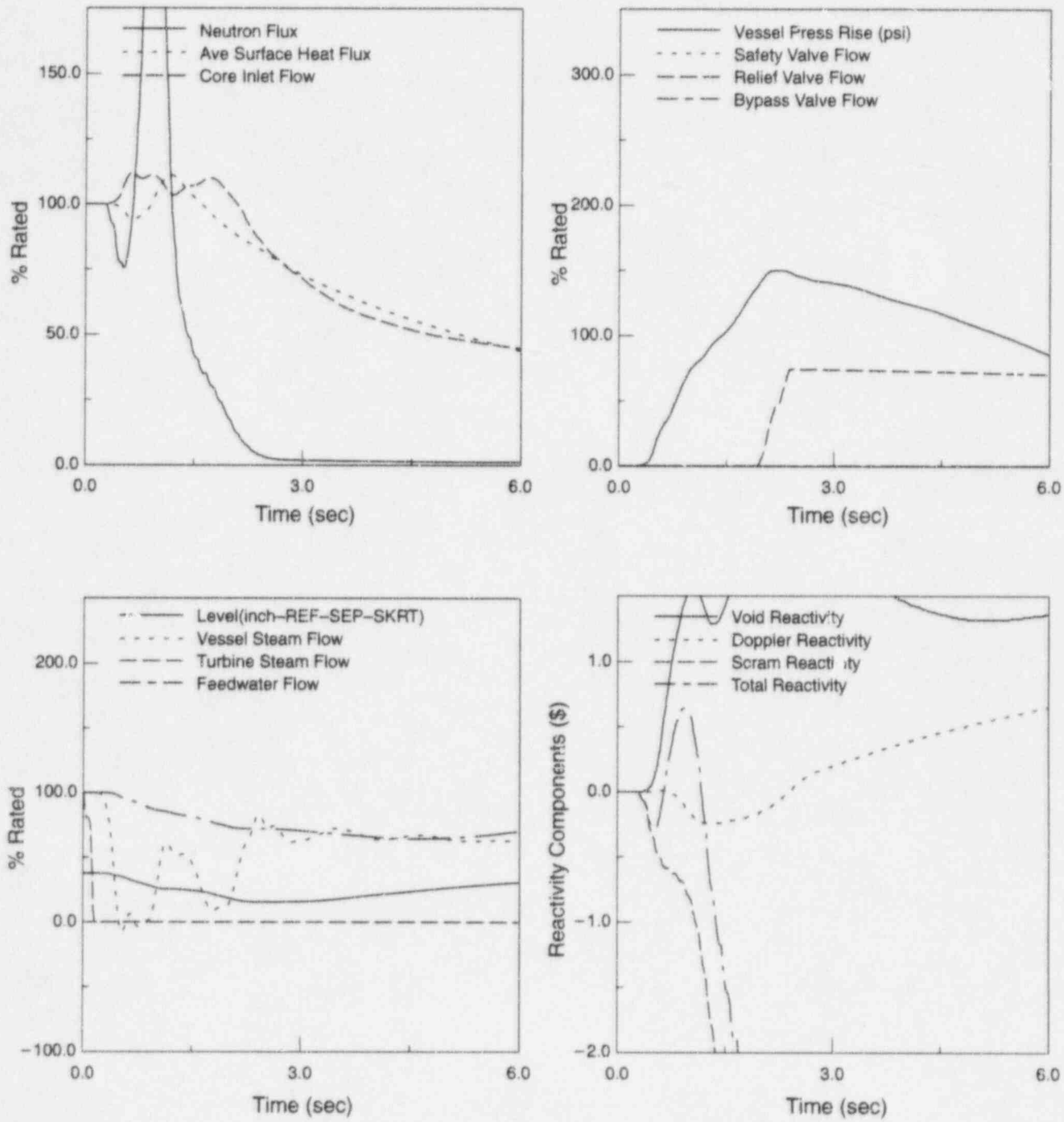


Figure 4 Plant Response to Load Reject w/o Bypass (BOC17 to EHFP17–2205 MWd/MT)

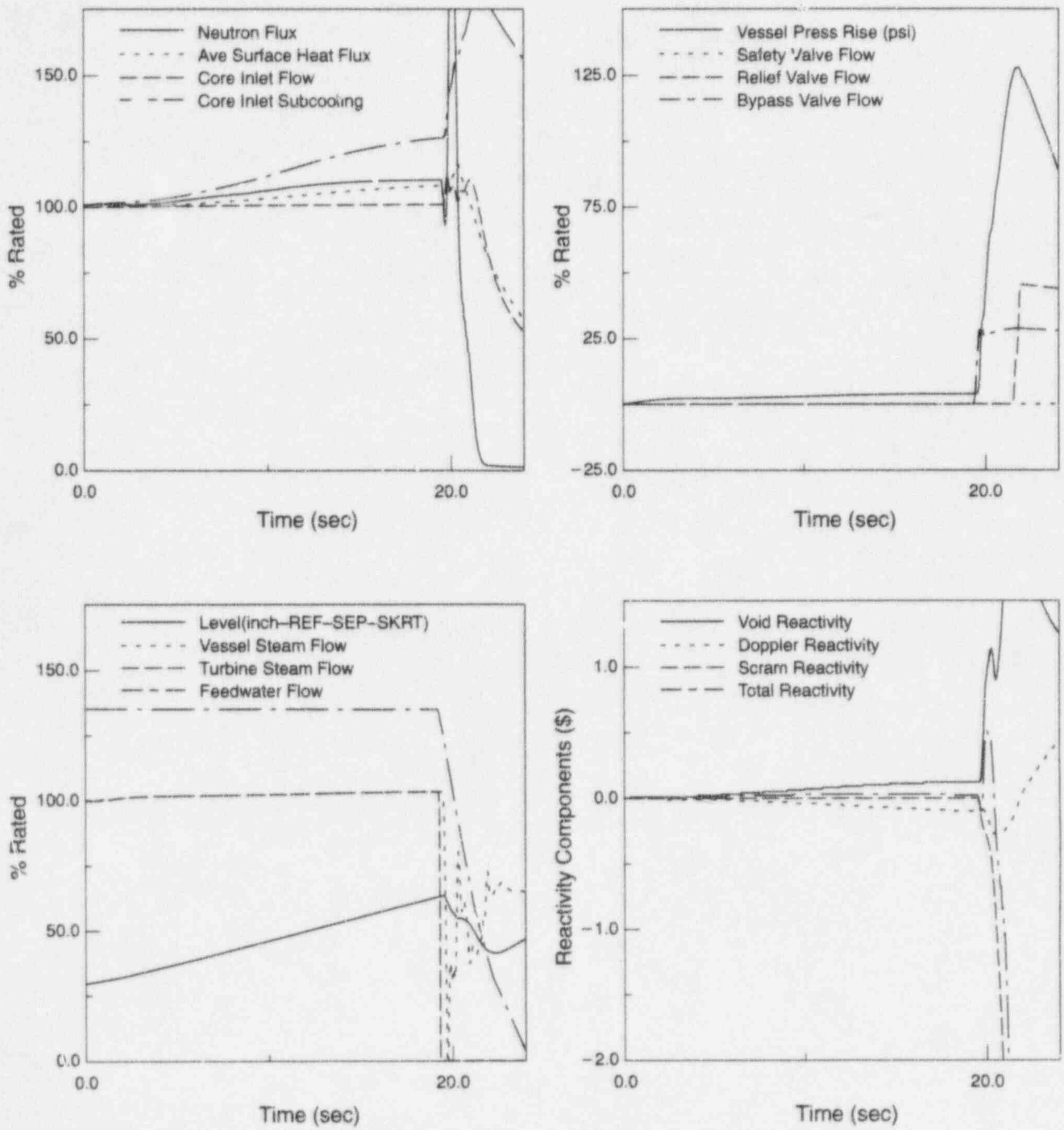


Figure 5 Plant Response to FW Controller Failure (EHFP17-2205 MWd/MT to EHFP17)

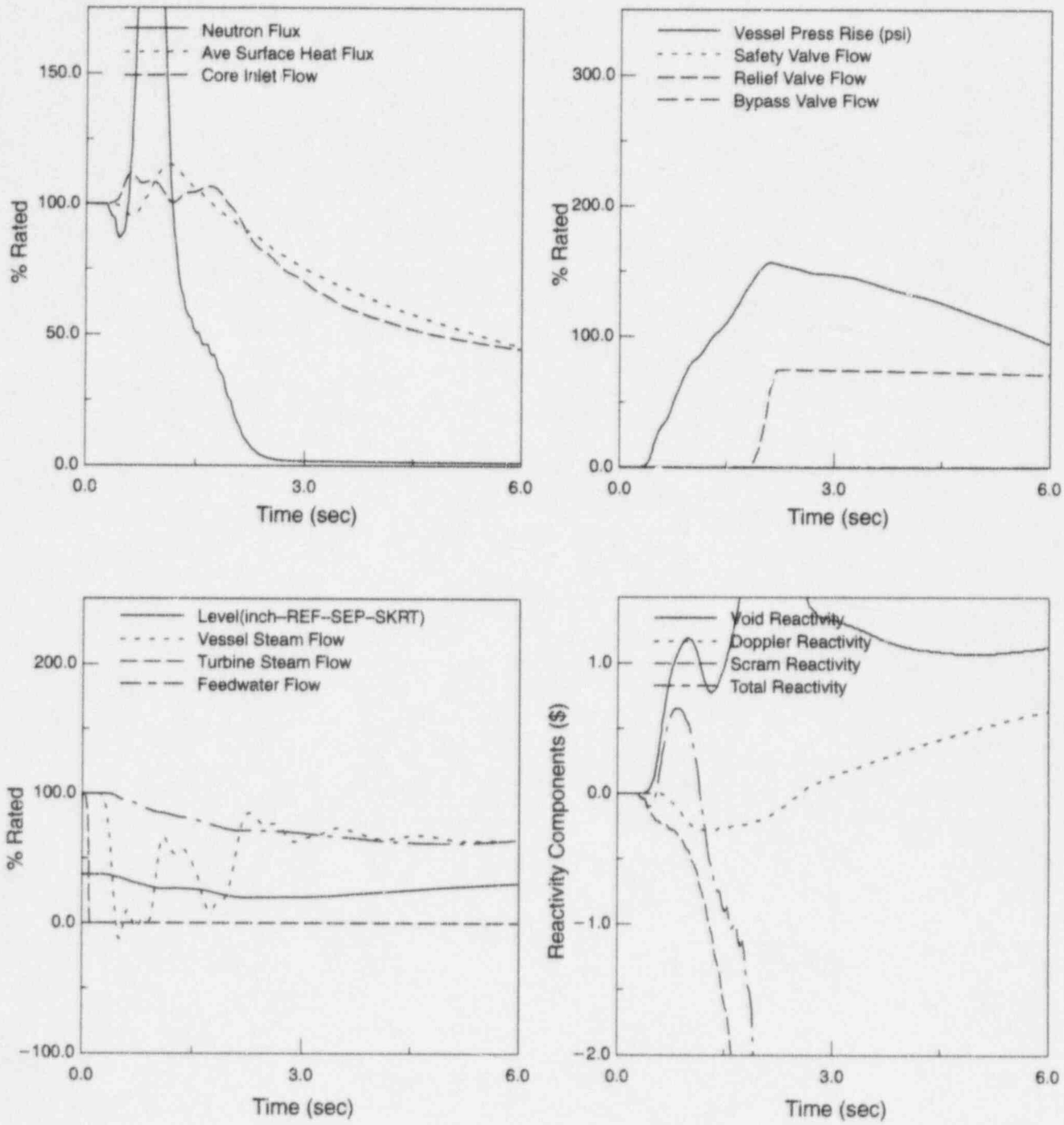


Figure 6 Plant Response to Turbine Trip w/o Bypass (EHFP17-2205 MWd/MT to EHFP17)

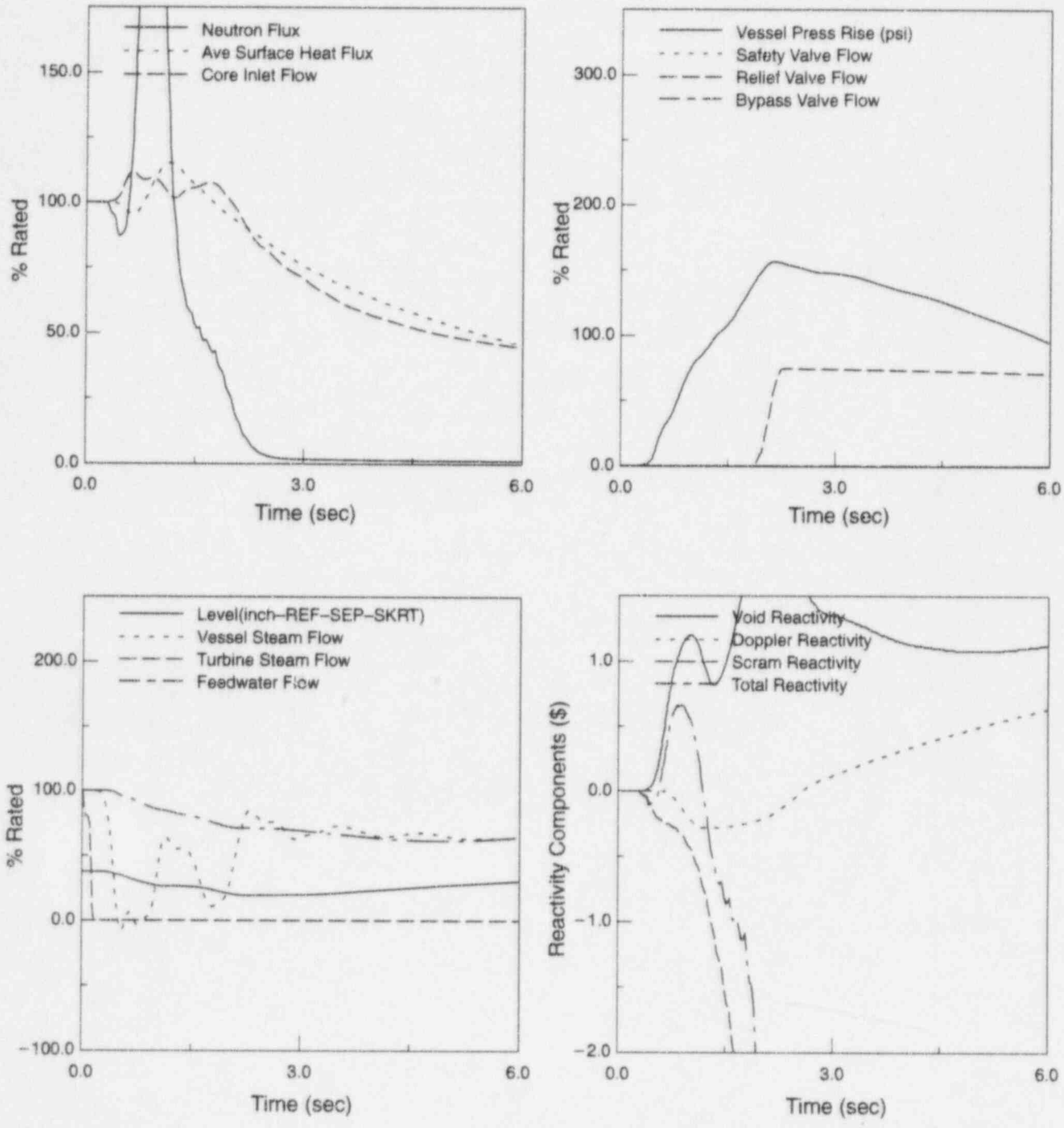


Figure 7 Plant Response to Load Reject w/o Bypass (EHFP17–2205 MWd/MT to EHFP17)

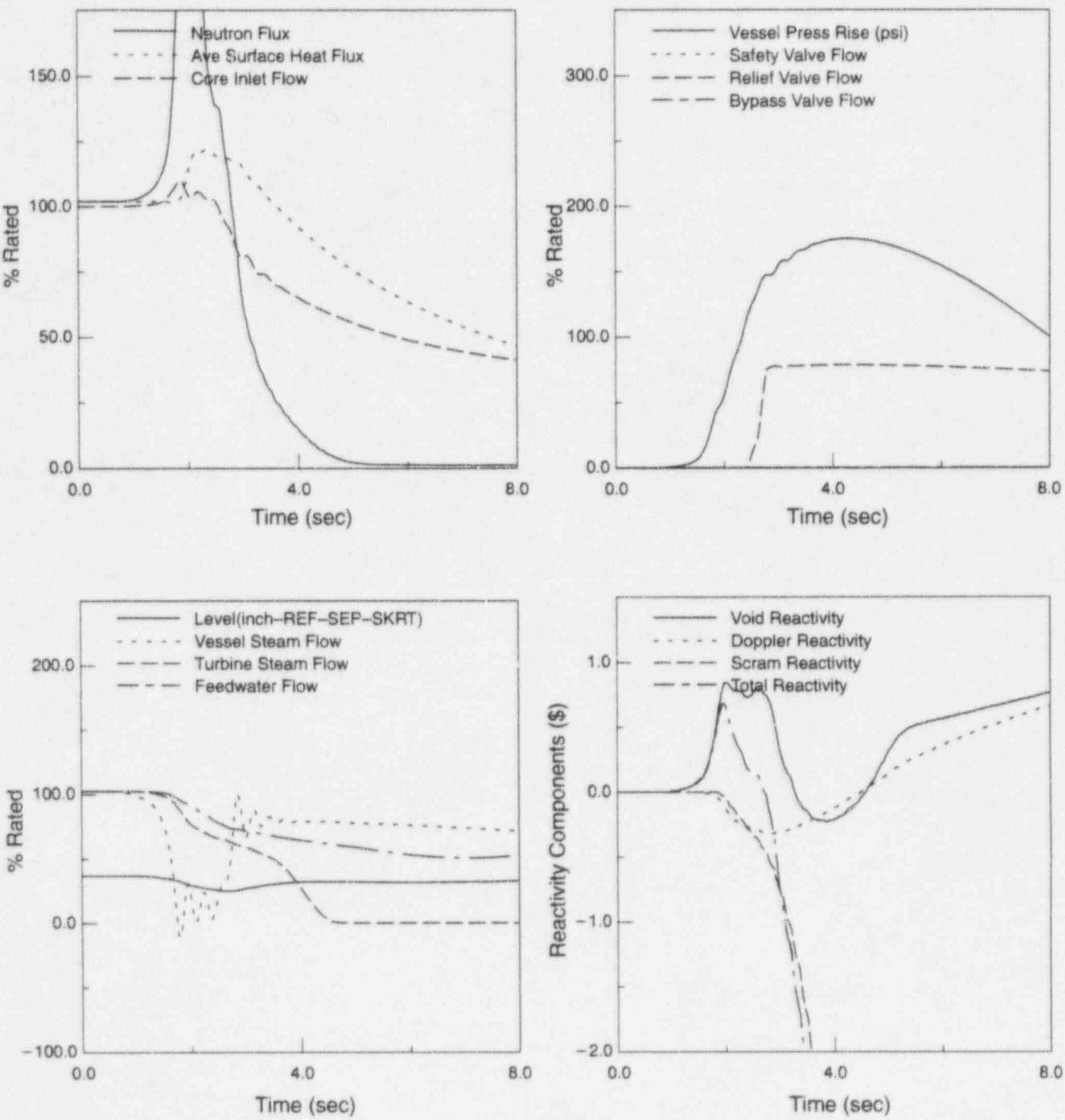


Figure 8 Plant Response to MSIV Closure (Flux Scram)

Appendix A

Analysis Conditions

To reflect actual plant parameters accurately, the values shown in Table A–1 were used this cycle.

Table A–1

STANDARD	
Parameter	**Analysis Value**
Thermal power, MWt	2381.0
Core flow, Mlb/hr	73.5
Reactor pressure, psia	1035.0
Inlet enthalpy, BTU/lb	520.4
Non–fuel power fraction	0.038
Steam flow analysis, Mlb/hr	9.56
Dome pressure, psig	1005.0
Turbine pressure, psig	955.1
No. of Safety/Relief Valves	8
No. of Single Spring Safety Valves	3
Relief mode lowest setpoint, psig	1113.0
Safety mode lowest setpoint, psig	1277.0

Appendix B

Decrease in Core Coolant Temperature Events

The loss–of–feedwater heating (LFWH) and the HPCI inadvertent startup anticipated operational occurrences (AOO) are the only cold water injection events checked on a cycle–by–cycle basis.

The LFWH 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, flux, and Q/A normally reported in Section 9 are not outputs of the BWR Simulator Code; therefore, these items are not included in this document for the LFWH event.

For the HPCI event, the CPR is presented in Section 11. The transient analysis inputs used for the HPCI AOO are given in Table B–1.

Table B–1

Void fraction (%)	43.66
Average fuel temperature (°F)	1099
Void coefficient N/A* (¢/%RG)	–8.14/–10.18
Doppler coefficient N/A* (¢/°F)	–0.191/–0.181
Scram worth N/A* ($)	**

References

B–1. *Steady state Nuclear Methods*, NEDE–30130–P–A and NEDO–03130–A, April 1985.

B–2 *General Electric Standard Application for Reactor Fuel*, NEDE–24011–P–A, February 1991.

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* N=Nuclear input data: A=Used in transient analysis.

** Generic exposure–independent values are used in General Electric Standard Application for Reactor Fuel, NEDE–24011–P–A–10, February 1991.

Appendix C

SRV Tolerance Analysis

The limiting overpressure event for Cooper is the main steam isolation valve closure with flux scram (MSIVF). The Cycle 17 reload evaluation was performed with the SRV and SV opening pressures at 3% above their nominal values. The peak vessel pressure reported for the Cycle 17 reload is 1242 psig.

An SRV tolerance analysis was previously completed and reported in Reference C–1. To demonstrate the applicability of Reference C–1 results to Cycle 17, an additional MSIVF event was analyzed with SRV opening pressure of 1210 psig (SRV upper limit). Except for the SRV opening pressure, this evaluation used the same analysis conditions as in the standard MSIVF analysis. Figure C–1 shows the reactor response for the MSIVF event with the upper limit SRV opening pressure set to 1210 psig. The peak vessel pressure for this case is 1302 psig at the vessel bottom, which is significantly below the vessel overpressure limit of 1375 psig. Thus, the Cycle 17 Upper limit case meets the ASME code requirement for the overpressure protection.

This evaluation demonstrate compliance to vessel overpressure limits for cycle 17 with the upper limit SRV pressure. Thus, the applicability of Reference C–1 can be extended to Cycle 17.

Reference

C–1. *SRV Setpoint Tolerance Analysis for Cooper Nuclear Station*, General Electric Company, NEDC–31628P, October 1988.

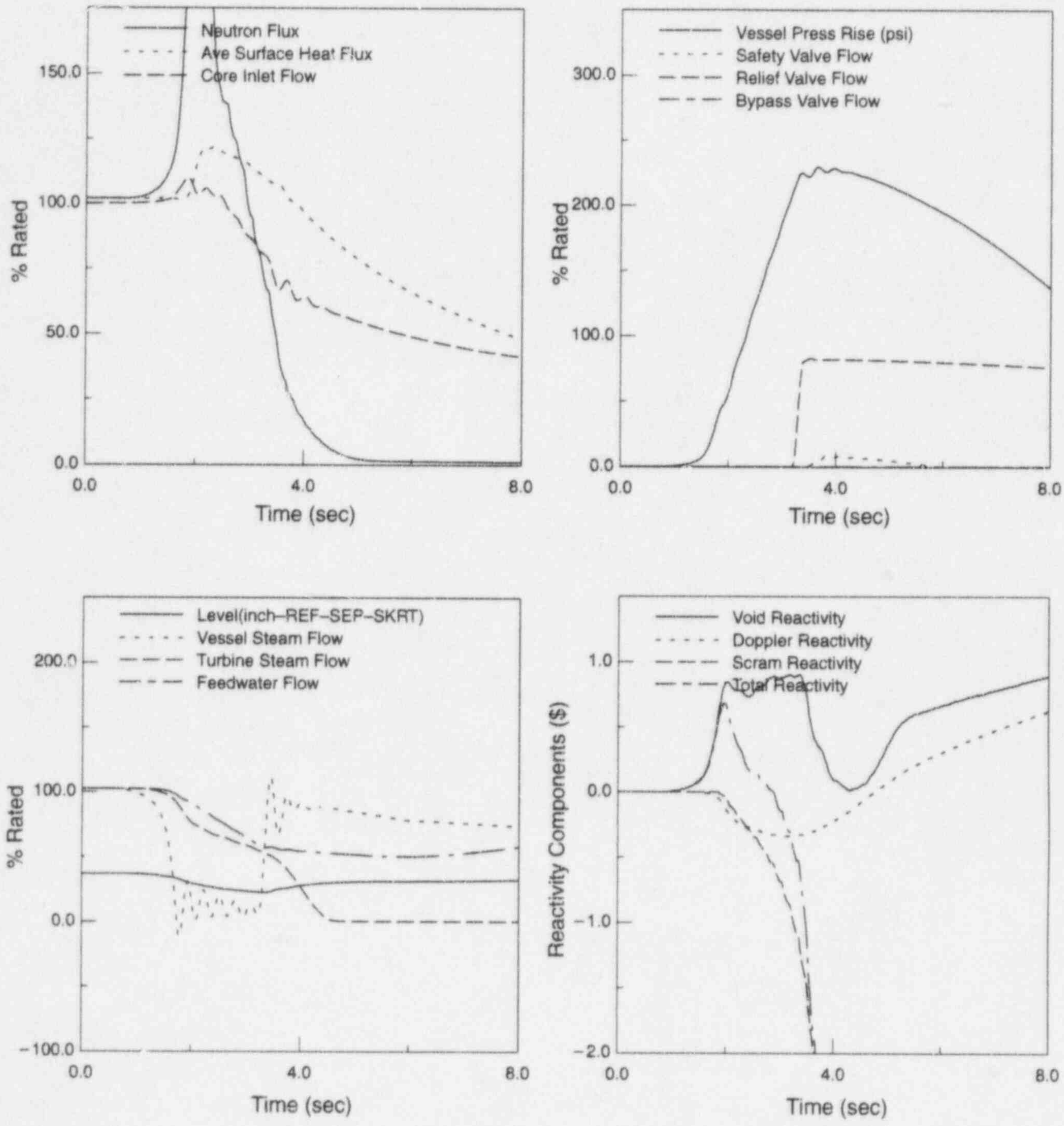

Figure C-1 Plant Response to MSIV Closure (Flux Scram)
(SRV Tolerance Analysis)