GE Nuclear Energy

J11-03354-10
Revision 0
Class I
August 1998

Supplemental Reload Licensing Report
for
COOPER NUCLEAR STATION
Reload 18 Cycle 19

GE Nuclear Energy

J11-03354-10
Revision 0
Class I
August 1998

J11-03354-10, Rev. 0

Supplemental Reload Licensing Report

for

Cooper Nuclear Station

Reload 18 Cycle 19

Approved

G. A. Watford, Manager
Nuclear Fuel Engineering

Approved T R Brohaugh

T. R. Brohaugh
Fuel Project Manager

Important Notice Regarding

Contents of This Report

Please Read Carefully

This report was prepared by General Electric Company (GE) solely for Nebraska Public Power District (NPPD) for NPPD's use with the U. S. Nuclear Regulatory Commission (USNRC) to amend NPPD's operating license of the Cooper Nuclear Station. The information contained in this report is believed by GE to be an accurate and true representation of the facts known, obtained or provided to GE at the time this report was prepared.

The only undertakings of GE respecting information in this document are contained in the contract between NPPD and GE for nuclear fuel and related services for the nuclear system for Cooper Nuclear Station and nothing contained in this document shall be construed as changing said contract. The use of this information except as defined by said contract, or for any purpose other than that for which it is intended, is not authorized; and with respect to any such unauthorized use, neither GE nor any of the contributors to this document makes any representation or warranty (expressed or implied) as to the completeness, accuracy or usefulness of the information contained in this document or that such use of such information may not infringe privately owned rights; nor do they assume any responsibility for liability or damage of any kind which may result from such use of such information.

Acknowledgment

The engineering and reload licensing analyses, which form the technical basis of this Supplemental Reload Licensing Report, were performed by R.H. Szilard. The Supplemental Reload Licensing Report was prepared by R.H. Szilard. This document has been verified by Carmen Alonso.

The basis for this report is *General Electric Standard Application for Reactor Fuel*, NEDE-24011-P-A-13, August 1996; and the U.S. Supplement, NEDE-24011-P-A-13-US, August 1996.

1. Plant-unique Items

Appendix A: Analysis Conditions
Appendix B: Decrease in Core Coolant Temperature Events
Appendix C: SRV Tolerance Analysis
Appendix D: One Turbine Bypass Valve Out of Service

2. Reload Fuel Bundles

Fuel Type	Cycle Loaded	Number
Irradiated:		
GE9B-P8DWB348-11GZ-80M-150-T (GE8x8NB)	16	48
GE9B-P8DWB348-12GZ-80M-150-T (GE8x8NB)	16	24
GE9B-P8DWB348-11GZ-80M-150-T (GE8x8NB)	17	152
GE9B-P8DWB348-11GZ-80M-150-T (GE8x8NB)	18	4
GE9B-P8DWB350-10GZ-80U-150-T (GE8x8NB)	18	160
New:		
GE9B-P8DWB350-10GZ1-80U-150-T (GE8x8NB)	19	100
GE9B-P8DWB350-10GZ-80U-150-T (GE8x8NB)	19	60
Total		548

3. Reference Core Loading Pattern[1]

Nominal previous cycle core average exposure at end of cycle:	26737 MWd/MT (24255 MWd/ST)
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	26406 MWd/MT (23955 MWd/ST)
Assumed reload cycle core average exposure at beginning of cycle:	15794 MWd/MT (14328 MWd/ST)
Assumed reload cycle core average exposure at end of cycle:	27037 MWd/MT (24528 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	1.106
Fully controlled	0.967
Strongest control rod out	0.989
R, Maximum increase in cold core reactivity with exposure into cycle, Δk	0.001

5. Standby Liquid Control System Shutdown Capability

Boron (ppm) (at 20°C)	Shutdown Margin (Δk) (at 20°C, Xenon Free)
660	0.038

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

Exposure: BOC19 to EHFP19-2205 MWd/MT (2000 MWd/ST)							
	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	7.297	101.0	1.18

Exposure: EHFP19-2205 MWd/MT (2000 MWd/ST) to EHFP19							
	Peaking Factors						
Fuel Design	Local	Radial	Axial	R-Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
GE8x8NB	1.20	1.66	1.40	1.000	7.041	102.5	1.23

7. Selected Margin Improvement Options

Recirculation pump trip:	No
Rod withdrawal limiter:	No
Thermal power monitor:	No
Improved scram time:	Yes (ODYN Option B)
Measured scram time:	No
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
One turbine bypass valve out of service:	Yes
Safety/relief valves out of service:	No
Feedwater heaters out of service:	No
ADS out of service:	No
EOC RPT out of service:	No
Main steam isolation valves out of service:	No

9. Core-wide AOO Analysis Results

Methods used: GEMINI; GEXL-PLUS

Exposure range: BOC19 to EHFP19-2205 MWd/MT (2000 MWd/ST)				
			Uncorrected ΔCPR	
Event	**Flux (%NBR)**	**Q/A (%NBR)**	**GE8x8NB**	**Fig.**
FW Controller Failure	202	115	0.12	2
Load Reject w/o Bypass	290	115	0.11	3
Turbine Trip w/o Bypass	275	113	0.10	4

Exposure range: EHFP19-2205 MWd/MT (2000 MWd/ST) to EHFP19				
			Uncorrected ΔCPR	
Event	**Flux (%NBR)**	**Q/A (%NBR)**	**GE8x8NB**	**Fig.**
FW Controller Failure	293	120	0.17	5
Load Reject w/o Bypass	358	119	0.16	6
Turbine Trip w/o Bypass	335	117	0.15	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, Revision 1, May 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 new fuel design. The results obtained verified that the existing ARTS limits are still valid for this cycle.

11. Cycle MCPR Values [2]

In agreement with commitments to the NRC (letter from M.A. Smith to the Document Control Desk, *10CFR Part 21, Reportable Condition, Safety Limit MCPR Evaluation*, May 24, 1996) a cycle-specific Safety Limit MCPR calculation was performed, and has been reported in both the Safety Limit MCPR and Operating Limit MCPR shown below. This cycle specific SLMCPR was determined using the analysis basis documented in GESTAR with the following exceptions:

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

1. The reference core loading was analyzed.
2. The actual bundle parameters (e.g., local peaking) were used.
3. The full cycle exposure range was analyzed.

Safety limit: 1.06

Single loop operation safety limit: 1.07

Non-pressurization events:

Exposure range: BOC19 to EHFP19	
	GE8x8NB
Loss of 100°F feedwater heating	1.17
Fuel Loading Error (misoriented)	1.20
Fuel Loading Error (mislocated)	1.21
Rod withdrawal error (for RBM setpoint to 111%)	1.24

Pressurization events:

Exposure range: BOC19 to EHFP19-2205 MWd/MT (2000 MWd/ST) **Exposure point: EHFP19-2205 MWd/MT (2000 MWd/ST)**		
	Option A	**Option B**
	GE8x8NB	**GE8x8NB**
FW Controller Failure	1.29	1.22
Load Reject w/o Bypass	1.27	1.20
Turbine Trip w/o Bypass	1.25	1.18

Exposure range: EHFP19-2205 MWd/MT (2000 MWd/ST) to EHFP19 **Exposure point: EHFP19**		
	Option A	**Option B**
	GE8x8NB	**GE8x8NB**
FW Controller Failure	1.32	1.24
Load Reject w/o Bypass	1.31	1.23
Turbine Trip w/o Bypass	1.30	1.22

12. Overpressurization Analysis Summary

Event	**Psl (psig)**	**Pv (psig)**	**Plant Response**
MSIV Closure (Flux Scram)	1222	1246	Figure 8

13. Loading Error Results

Variable water gap misoriented bundle analysis: Yes [3]

Event	ΔCPR
Fuel loading error (Misoriented)	0.14
Fuel loading error (Mislocated)	0.15

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-13-US, August 1996. 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

Cooper Nuclear Station has implemented the Option 1-D stability solution, documented in the reference, *Application of the "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option 1-D) to the Cooper Nuclear Station, Licensing Topical Report,* GENE-A13-00395-01, Class I, November, 1996. The continued validity of the reference results for Cycle 19 has been confirmed.

16. Loss-of-Coolant Accident Results

LOCA method used: SAFER/GESTR-LOCA

See the Cooper Nuclear Station SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis, NEDC-32687P, Revision 1, March 1997. The LOCA analysis results presented in NEDC-32687P is conservatively analyzed for the GE8x8NB fuel types. This analysis yields a Licensing Basis peak clad temperature (PCT) of 1570°F, a peak local oxidation fraction of <0.4%, and a core-wide metal-water reaction of <0.1%. The MAPLHGR multiplier for single loop operation (SLO) is 0.77. The SLO multiplier of 0.77 ensures that the PCT for SLO will always be bounded by that of two-loop operation. NRC approval for single loop operation is documented in Amendment No. 94, dated September 24, 1985, to Cooper Nuclear Station Facility Operating License.

There is one new GE8x8NB fuel design loaded in Cycle 19. The most limiting and least limiting MAPLHGRs for the new fuel designs are as follows:

[3] Includes a 0.02 penalty due to variable water gap R-factor uncertainty.

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

Bundle Type: GE9B-P8DWB350-10GZ1-80U-150-T

Average Planar Exposure		MAPLHGR (kw/ft)	
(GWd/ST)	(GWd/MT)	Most Limiting	Least Limiting
0.00	0.00	11.59	11.61
0.20	0.22	11.63	11.65
1.00	1.10	11.71	11.74
2.00	2.20	11.84	11.88
3.00	3.31	11.99	12.03
4.00	4.41	12.14	12.17
5.00	5.51	12.26	12.30
6.00	6.61	12.39	12.43
7.00	7.72	12.53	12.57
8.00	8.82	12.66	12.71
9.00	9.92	12.81	12.87
10.00	11.02	12.85	12.91
12.50	13.78	12.79	12.87
15.00	16.53	12.51	12.54
20.00	22.05	11.78	11.78
25.00	27.56	11.05	11.05
35.00	38.58	9.74	9.75
45.00	49.60	7.92	7.96
49.61	54.68	5.66	5.70
49.68	54.76	--	5.67

52									D	D	D	D	D	D	D	D										
50								E	E	F	H	H	H	H	H	H	F	E	E							
48					E	D	D	F	A	H	A	H	A	A	H	A	H	A	F	D	D	E				
46				D	F	G	H	A	H	A	H	B	F	F	B	H	A	H	A	H	G	F	D			
44				E	F	H	A	H	H	F	B	H	F	F	H	B	F	H	H	A	H	F	E			
42			D	H	H	F	H	B	F	F	H	B	F	F	B	H	F	F	B	H	F	H	H	D		
40			D	H	A	H	B	F	B	H	B	F	B	B	F	B	H	B	F	B	H	A	H	D		
38		D	F	A	H	B	F	B	H	B	F	B	H	H	B	F	B	H	B	F	B	H	A	F	D	
36		F	A	H	H	F	B	H	F	F	B	H	F	F	H	B	F	F	H	B	F	H	H	A	F	
34	E	F	H	A	F	F	H	B	F	F	H	B	F	F	B	H	F	F	B	H	F	F	A	H	F	E
32	E	H	A	H	B	H	B	F	B	H	B	F	A	A	F	B	H	B	F	B	H	B	H	A	H	E
30	D	H	H	B	H	B	F	B	H	B	F	A	H	H	A	F	B	H	B	F	B	H	B	H	H	D
28	D	C	A	F	F	F	B	H	F	F	A	H	F	F	H	A	F	F	H	B	F	F	F	A	C	D
26	D	C	A	F	F	F	B	H	F	F	A	H	F	F	H	A	F	F	H	B	F	F	F	A	C	D
24	D	H	H	B	H	B	F	B	H	B	F	A	H	H	A	F	B	H	B	F	B	H	B	H	H	D
22	E	H	A	H	B	H	B	F	B	H	B	F	A	A	F	B	H	B	F	B	H	B	H	A	H	E
20	E	F	H	A	F	F	H	B	F	F	H	B	F	F	B	H	F	F	B	H	F	F	A	H	F	E
18		F	A	H	H	F	B	H	F	F	B	H	F	F	H	B	F	F	H	B	F	H	H	A	F	
16		D	F	A	H	B	F	B	H	B	F	B	H	H	B	F	B	H	B	F	B	H	A	F	D	
14			D	H	A	H	B	F	B	H	B	F	B	B	F	B	H	B	F	B	H	A	H	D		
12			D	H	H	F	H	B	F	F	H	B	F	F	B	H	F	F	B	H	F	H	H	D		
10				E	F	H	A	H	H	F	B	H	F	F	H	B	F	H	H	A	H	F	E			
8				D	F	G	H	A	H	A	H	B	F	F	B	H	A	H	A	H	G	F	D			
6					E	D	D	F	A	H	A	H	A	A	H	A	H	A	F	D	D	E				
4								E	E	F	H	H	H	H	H	H	F	E	E							
2										D	D	D	D	D	D	D	D									
	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51

Fuel Type			
A=GE9B-P8DWB350-10GZ-80U-150-T	(Cycle 19)	E=GE9B-P8DWB348-12GZ-80M-150-T	(Cycle 16)
B=GE9B-P8DWB350-10GZ1-80U-150-T	(Cycle 19)	F=GE9B-P8DWB348-11GZ-80M-150-T	(Cycle 17)
C=GE9B-P8DWB348-11GZ-80M-150-T	(Cycle 17)	G=GE9B-P8DWB348-11GZ-80M-150-T	(Cycle 18)
D=GE9B-P8DWB348-11GZ-80M-150-T	(Cycle 16)	H=GE9B-P8DWB350-10GZ-80U-150-T	(Cycle 18)

Figure 1 Reference Core Loading Pattern

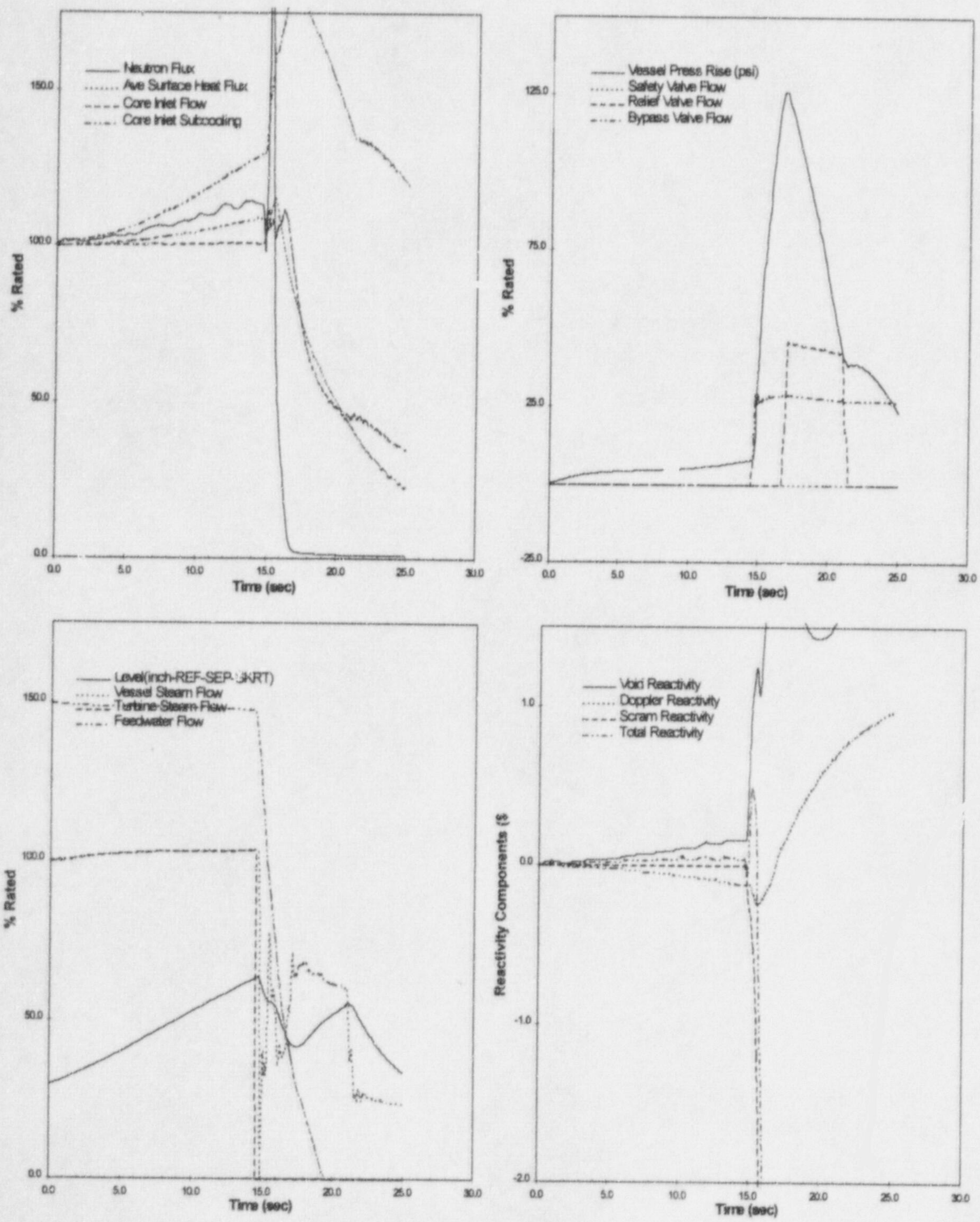

Figure 2 Plant Response to FW Controller Failure
(BOC19 to EHFP19-2205 MWd/MT (2000 MWd/ST))

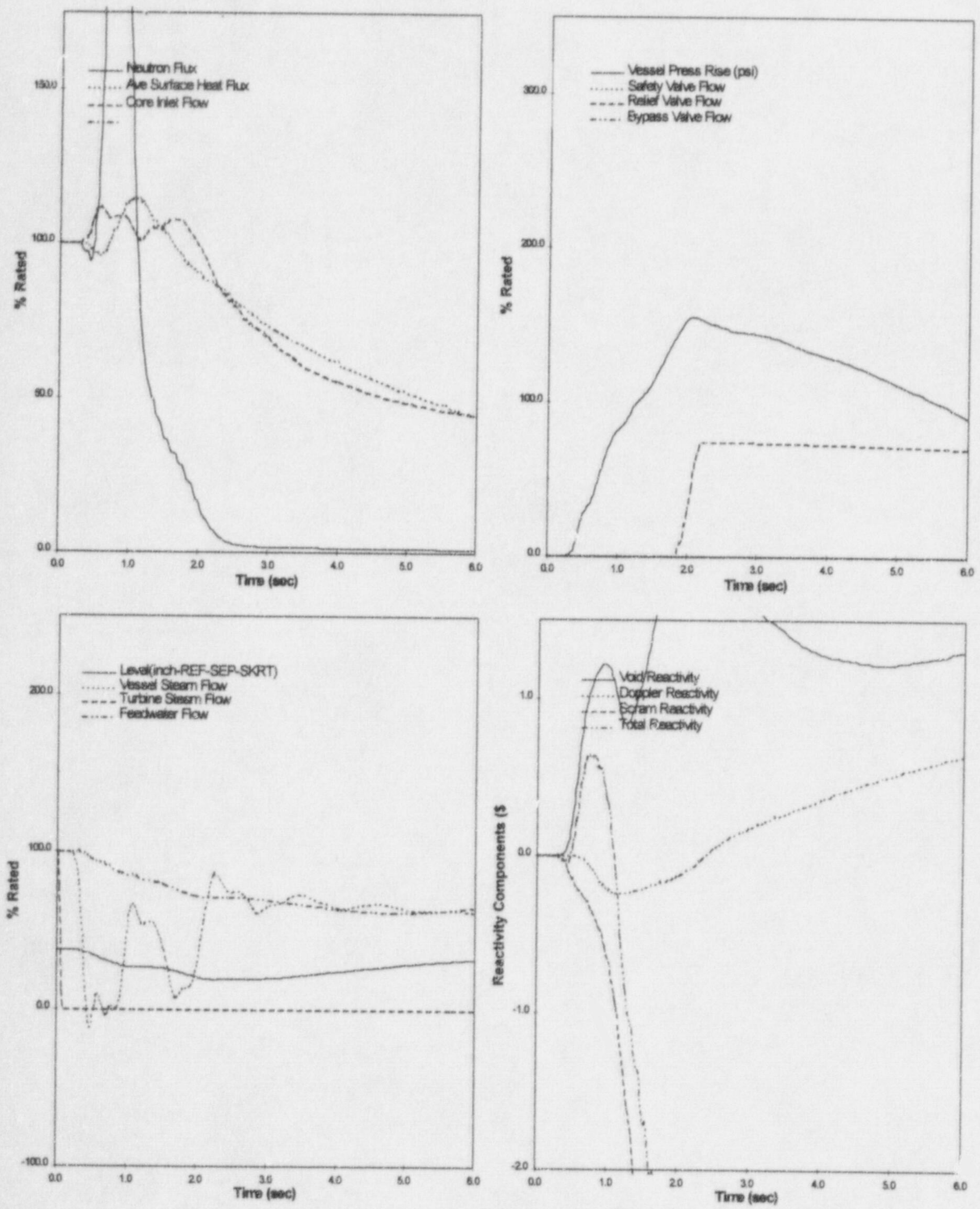

Figure 3 Plant Response to Load Reject w/o Bypass
(BOC19 to EHFP19-2205 MWd/MT (2000 MWd/ST))

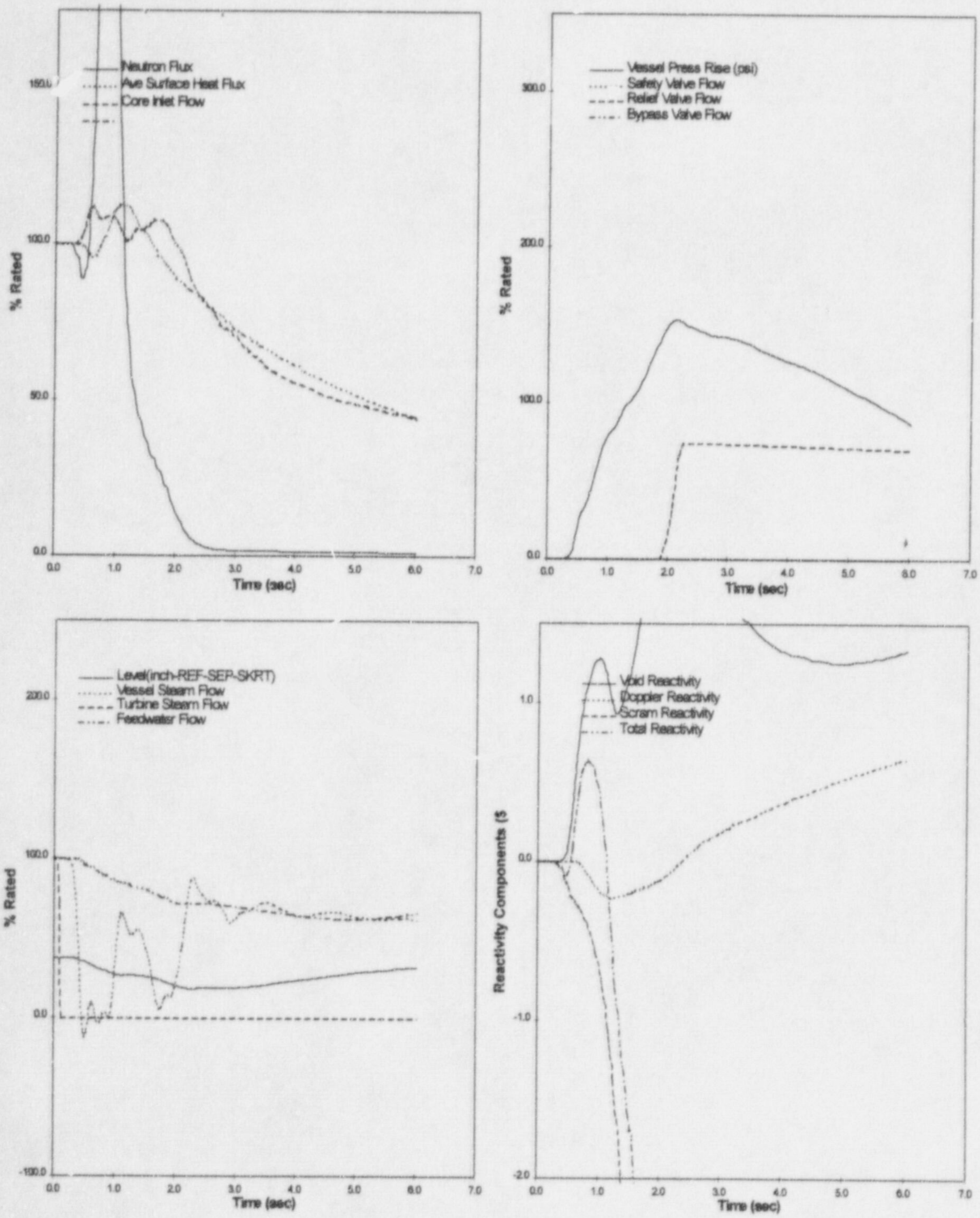

Figure 4 Plant Response to Turbine Trip w/o Bypass (BOC19 to EHFP19-2205 MWd/MT (2000 MWd/ST))

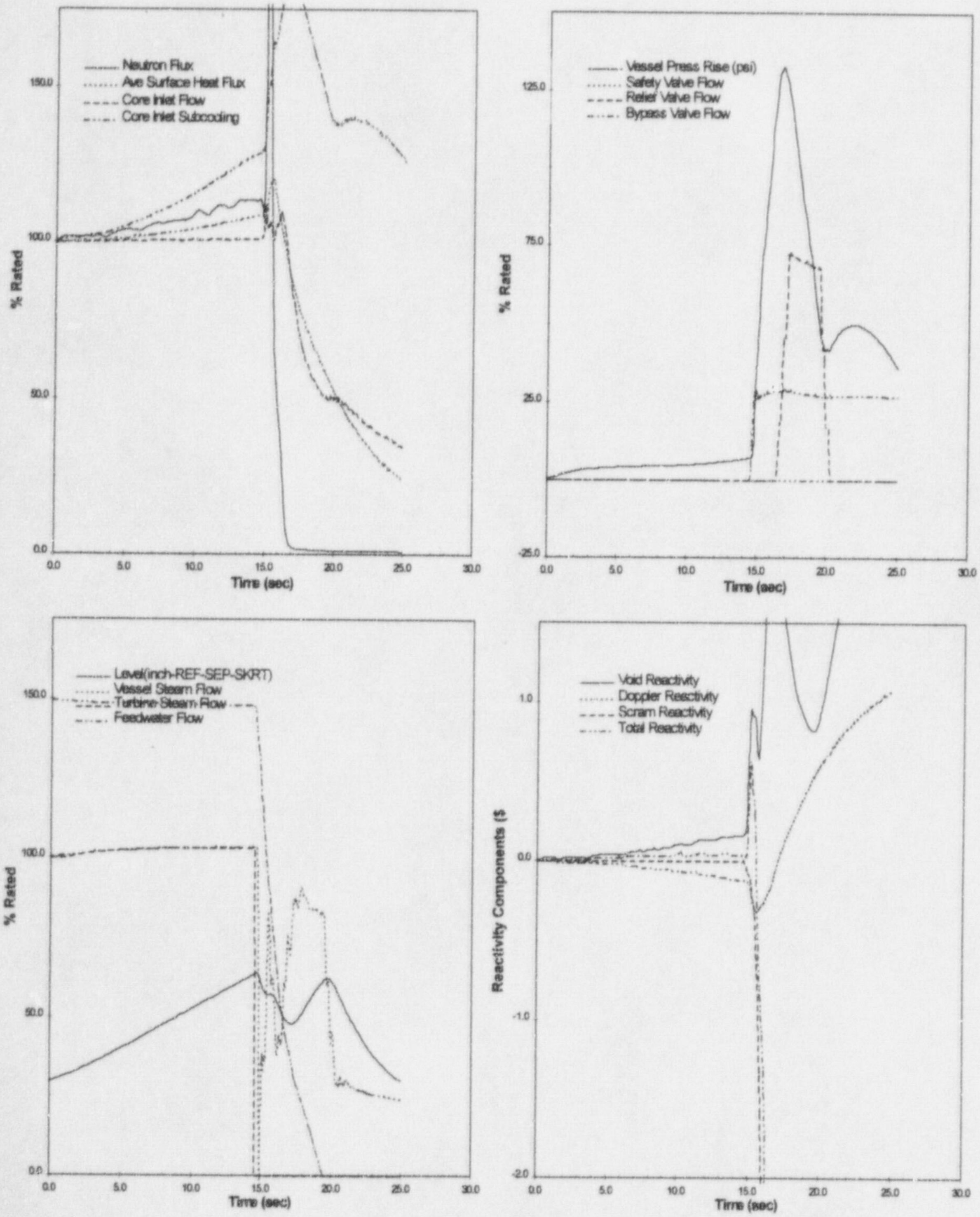

Figure 5 Plant Response to FW Controller Failure (EHFP19-2205 MWd/MT (2000 MWd/ST) to EHFP19)

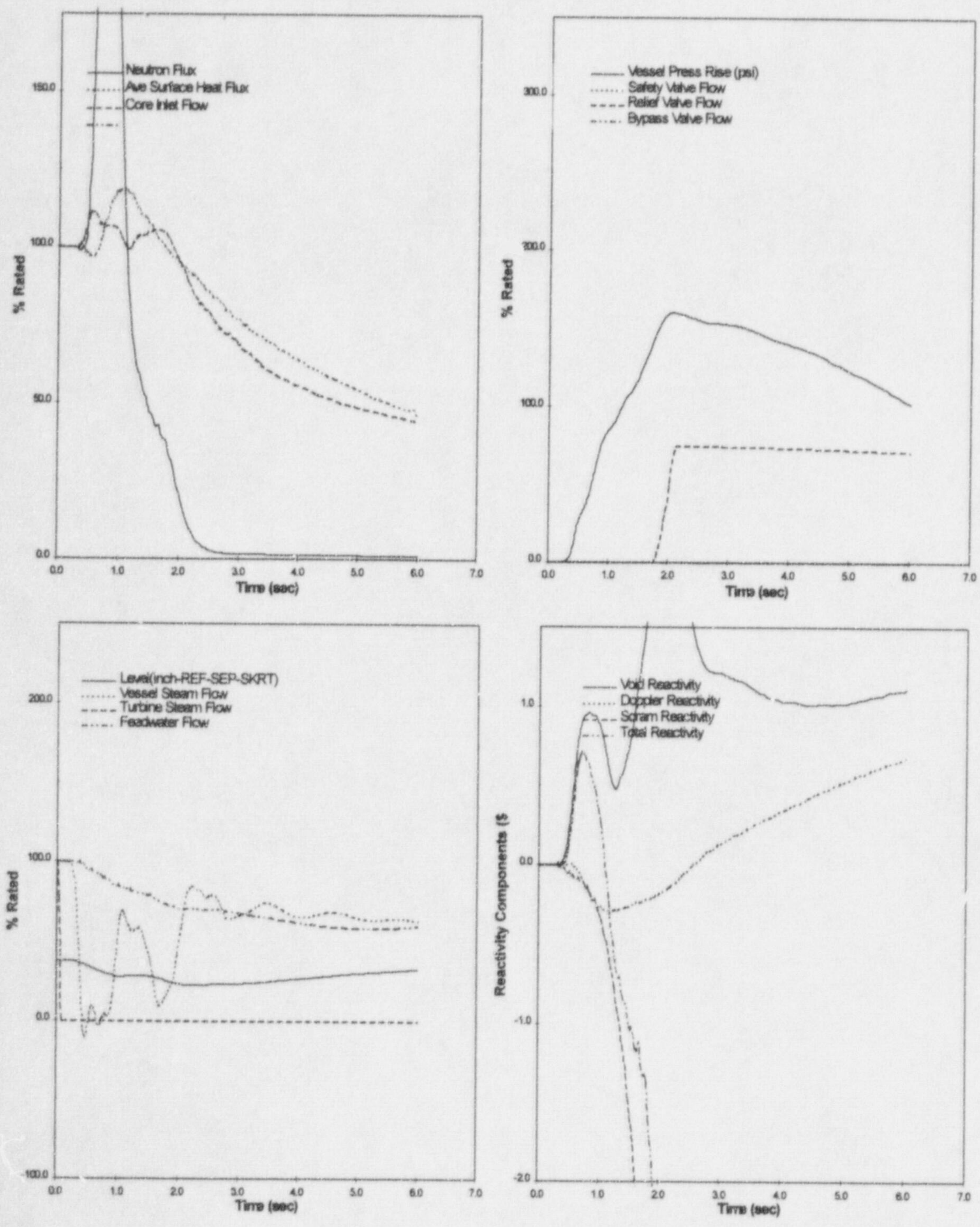

Figure 6 Plant Response to Load Reject w/o Bypass (EHFP19-2205 MWd/MT (2000 MWd/ST) to EHFP19)

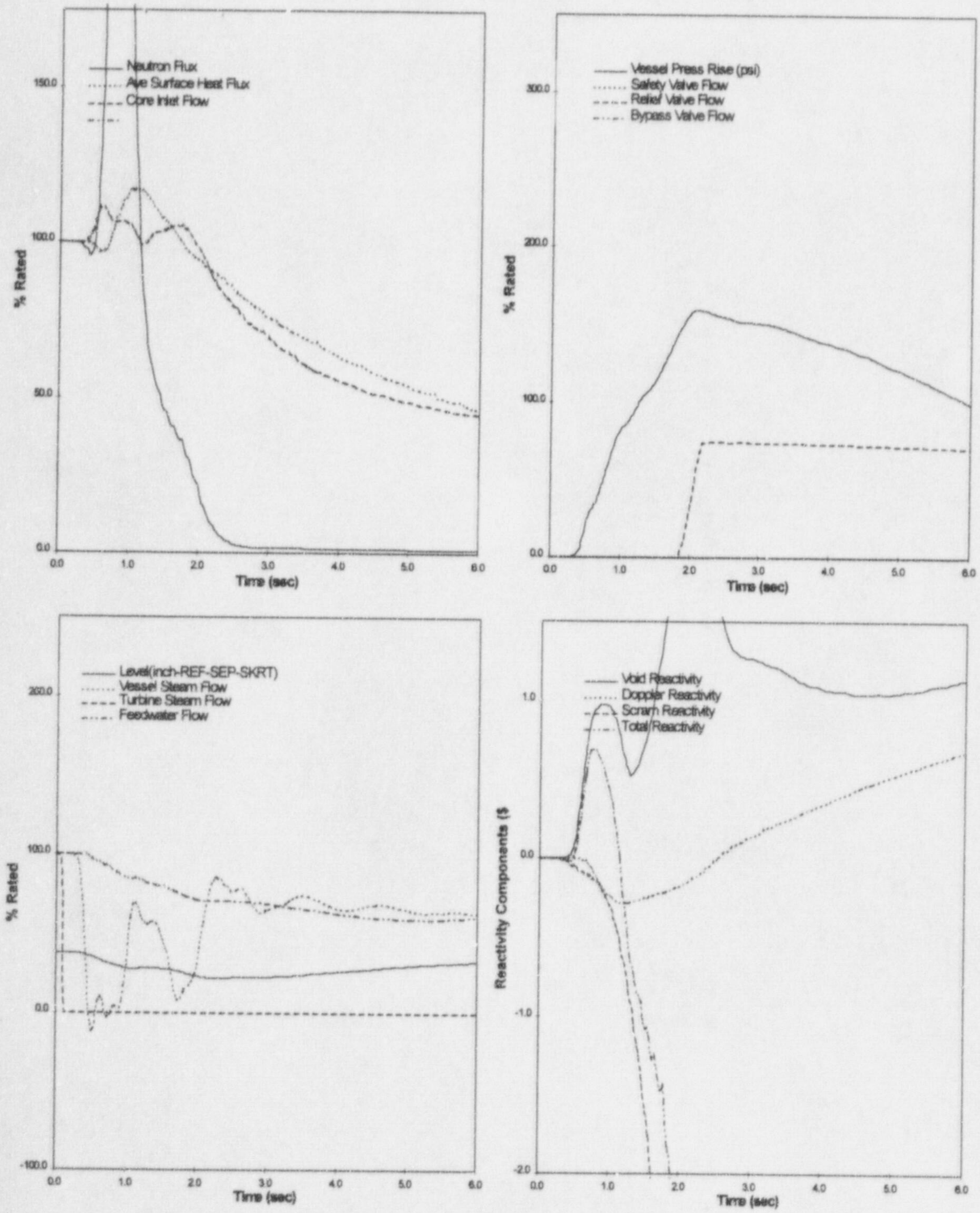

Figure 7 Plant Response to Turbine Trip w/o Bypass (EHFP19-2205 MWd/MT (2000 MWd/ST) to EHFP19)

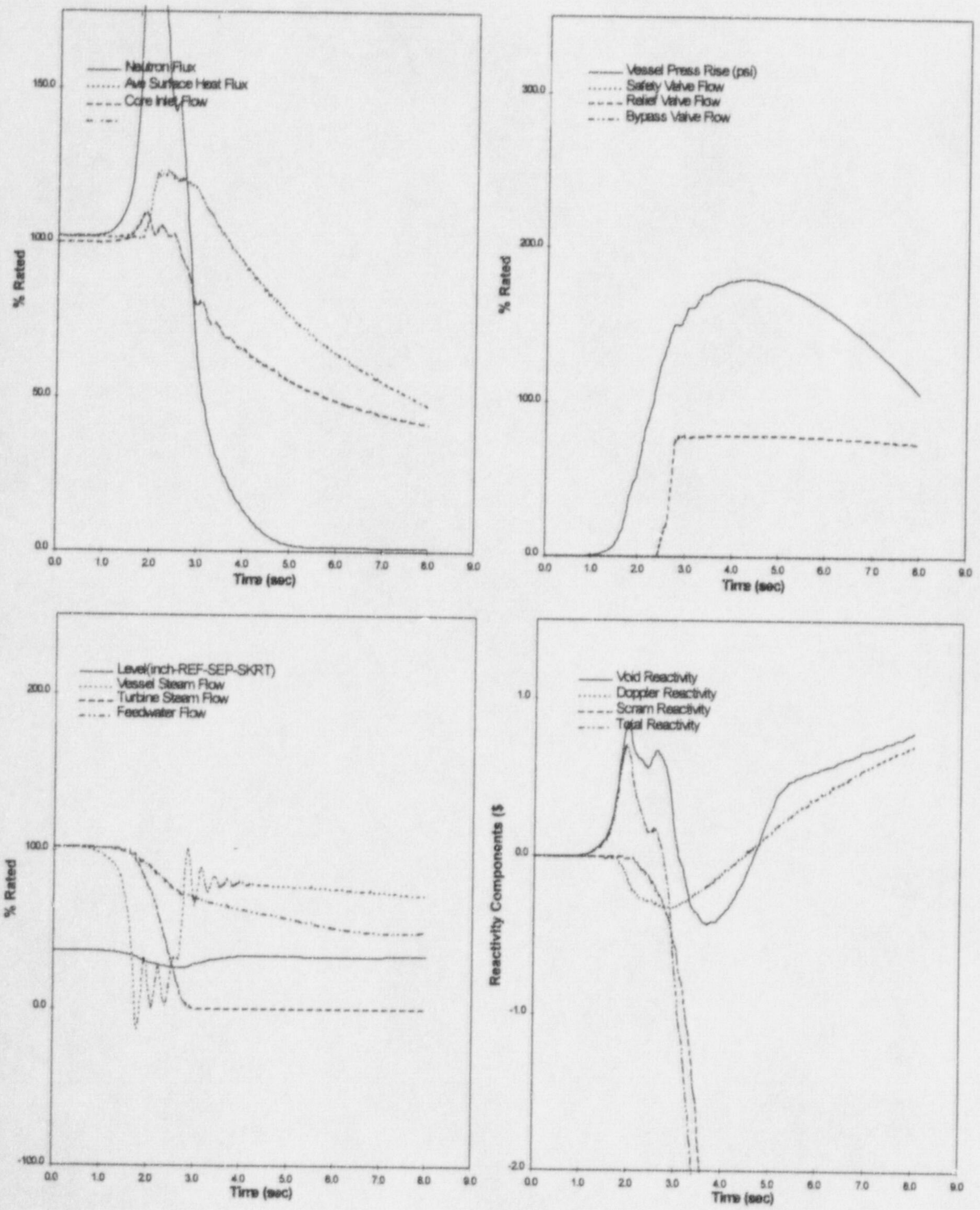

Figure 8 Plant Response to MSIV Closure (Flux Scram)

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, 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 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-to-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 Cycle 19, the Inadvertent HPCI analysis was shown to be bounded by the LFWH event. This was done by showing the core inlet subcooling due to feedwater temperature reduction from HPCI plus the core inlet subcooling due to excess feedwater from HPCI is less than the core inlet subcooling for the LFWH event.

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-13-US, August 1996.

Appendix C
SRV Tolerance Analysis

The limiting overpressure event for Cooper is the main steam isolation valve closure with flux scram (MSIVF). The Cycle 19 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 19 reload is 1246 psig.

An SRV tolerance analysis was previously completed and reported in Reference C-1. To determine the applicability of Reference C-1 results to Cycle 19, 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 1305 psig at the vessel bottom, which is significantly below the vessel overpressure limit of 1375 psig. Thus, the Cycle 19 upper limit case meets the ASME code requirement for the overpressure protection.

This evaluation demonstrates compliance to vessel overpressure limits for Cycle 19 with the upper limit SRV pressure. Thus, the applicability of Reference C-1 can be extended to Cycle 19.

Reference

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

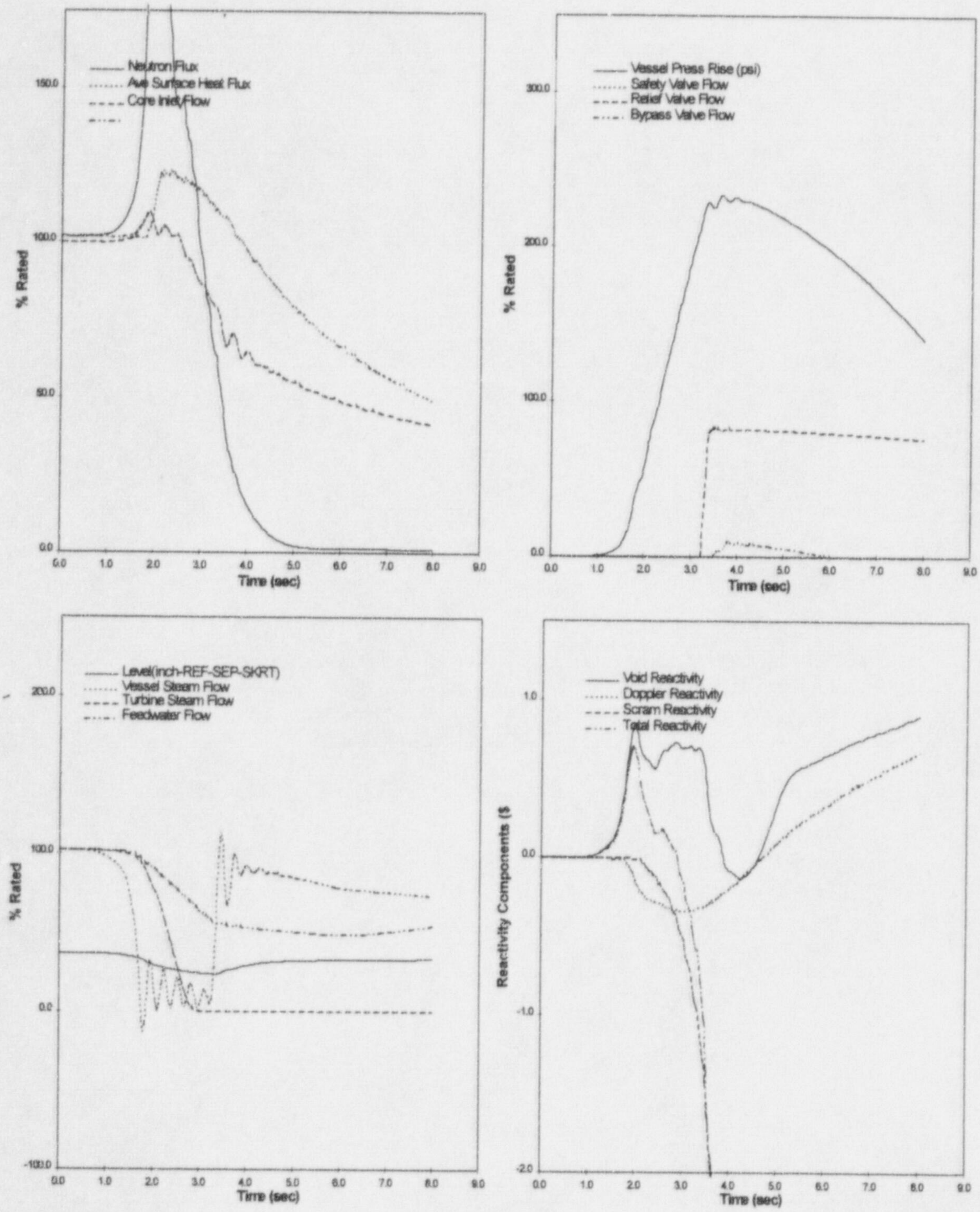

**Figure C-1 Plant Response to MSIV Closure (Flux Scram)
(SRV Tolerance Analysis)**

Appendix D

One Turbine Bypass Out of Service

In order to support continued operation of Cooper Nuclear Station with the possibility that one bypass valve is unavailable, the turbine bypass valve (BPV) out of service operation was evaluated. The objective of this evaluation was to calculate the MCPR for the limiting event with one BPV unavailable and determine whether the calculated MCPR specified for the most limiting event for Cycle 19 is affected.

The effect of one BPV unavailable is to reduce the pressure relief capability in the early part of a pressurization event (i.e., before the relief and safety valves can open) and thus result in an increase in the ΔCPR. The limiting pressurization events that are analyzed on a cycle-specific basis for Cooper are the turbine trip without bypass, the load reject without bypass, and the feedwater controller failure events. The turbine trip without bypass and the load reject without bypass events are not affected by one BPV being unavailable because the analyses do not take credit for any BPV's being available. Therefore, only the feedwater controller failure event (FWCF) was analyzed.

The same conditions that were used for the Cycle 19 reload analysis for the FWCF were used, except that one BPV was assumed to be unavailable. End of Cycle 19 conditions were used as these are most stringent. A conservative representation for the BPV opening characteristic was assumed. Both Option A and Option B scram conditions were analyzed and the results are provided below. Figure D-1 shows the reactor response for the FWCF event with one BPV unavailable.

With one BPV unavailable, the MCPRs are as follows:

Exposure range: BOC19 to EHFP19		
	Option A	**Option B**
GE8x8NB	1.34	1.26

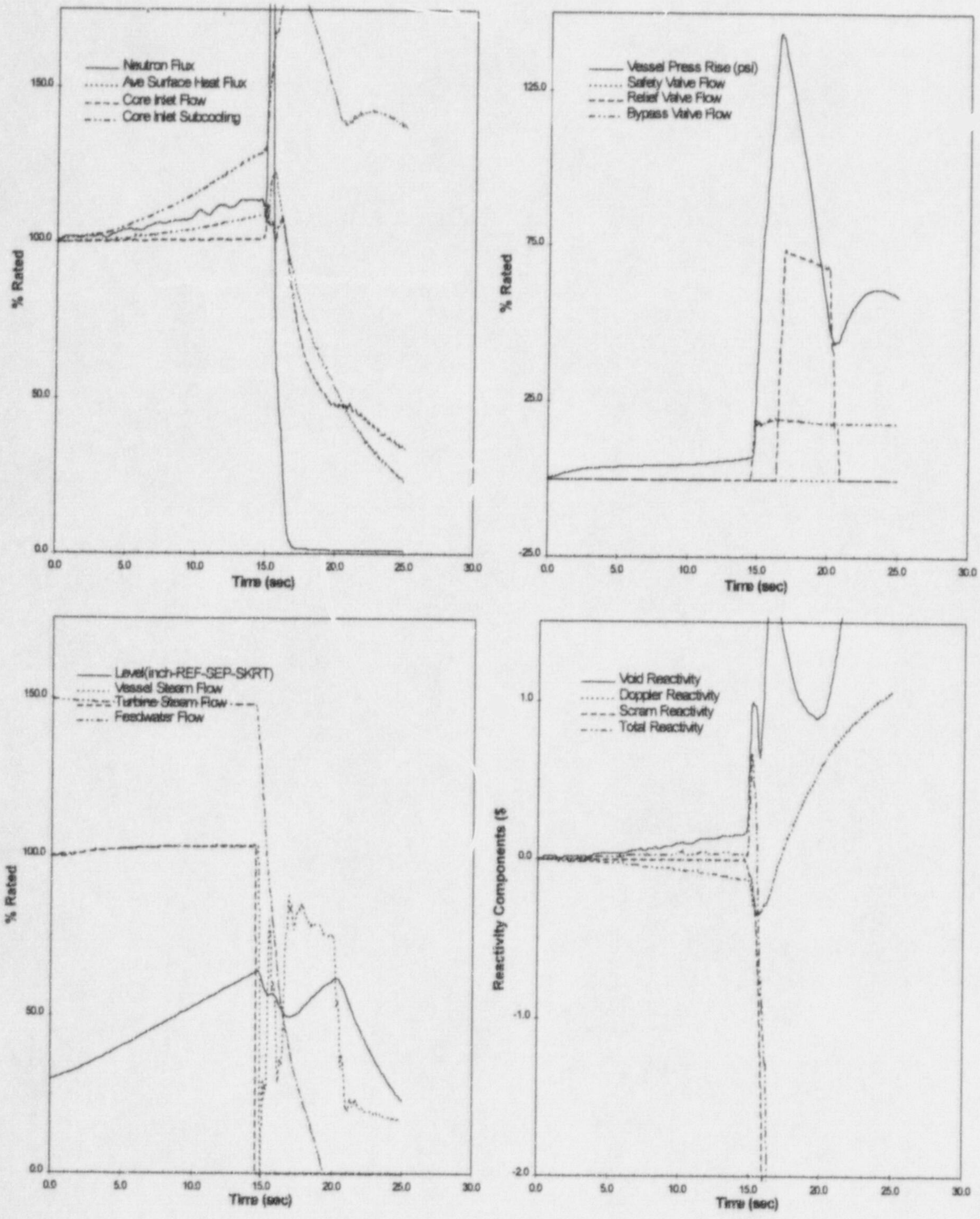

Figure D-1 Plant Response to FW Controller Failure (One Turbine Bypass Valve Out of Service)