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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

QUAD CITIES NUCLEAR POWER STATION

UNIT 2, RELOAD 9, CYCLE 10

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ACKNOWLEDGEMENTS

The engineering and reload licensing analysis, which form the technical basis of this Supplemental Reload Licensing Submittal, were performed in the Nuclear Fuel and Engineering Services Department by H. J. Pearson.

1. PLANT-UNIQUE ITEMS (1.0)*

Analysis Conditions: Appendix A

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

Fuel Type	Cycle Loaded	Number
Irradiated		
P8DGB263L **	6	8
P8DGB298 **	6	24
BP8DRB265H	7	200
BP8DRB282	8	72
BP8DRB283H	8	104
BP8DRB299	9	64
BP8DRB299L	9	88
New		
BD300C	10	92
BD316A	10	72
Total		724

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	21666 MWd/MT
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	21335 MWd/MT
Assumed reload cycle core average exposure at end of cycle:	22754 MWd/MT
Core Loading pattern:	Figure 1

*( ) Refers to area of in discussion 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.

** Barrier fuel.

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	1.102
Fully Controlled	0.958
Strongest Control Rod Out	0.981

R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, Delta k	0.007

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Delta k) (20 deg.C, Xenon Free)
600	0.043

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

(COLD WATER INJECTION EVENTS ONLY)

Void Fraction (%)	34.19
Average Fuel Temperature (degrees F)	987
Void Coefficient N/A* (cents/% Rg)	-4.61/-5.76
Doppler Coefficient N/A* (cents/deg. F)	-0.244/-0.232
Scram Worth N/A* ($)	**

Extended EOC with Increased Core Flow and Final Feedwater Temperature Reduction

Void Fraction (%)	28.50
Average Fuel Temperature (degrees F)	983
Void Coefficient N/A* (cents/% Rg)	-5.04/-6.29
Doppler Coefficient N/A* (cents/deg. F)	-0.273/-0.259
Scram Worth N/A* ($)	**

*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-8, May 1986.

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

Fuel Design	Peaking Factors Local	Radial	Axial	R-Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
Exposure: BOC10 to EOC10							
BP/P8x8R	1.20	1.83	1.40	1.051	[illegible].206	104.8	1.28
GE8x8EB	1.20	1.[illegible]	1.40	1.051	[illegible] 224	106.7	1.28
Exposure: Extended EOC with Increased Core Flow and Final Feedwater Temperature Reduction							
BP/P8x8R	1.20	1.82	1.40	1.051	6.159	114.7	1.31
GE8x8EB	1.20	1.83	1.40	1.051	6.177	116.7	1.31

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

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

9. OPERATING FLEXIBILITY OPTIONS (S.2.2.3)

Single-Loop Operation:	Yes
Load Line Limit:	Yes
Extended Load Line Limit:	Yes
Increased Core Flow	Yes
Flow Point Analyzed:	108%
Feedwater Temperature Reduction:	Yes
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	Delta CPR GE8x8EB	Figure
Exposure: BOC10to EOC10					
Turbine Trip w/o Bypass	535	122	[illegible]	0.21	2
Inadvertent HPCI	121	116	0.18	0.18	3
Feedwater Controller Failure	280	118	0.15	0.16	4
Exposure: Extended EOC with Increased Core Flow and Final Feedwater Temperature Reduction					
Load Rejection w/o Bypass	475	121	0.24	0.24	5
Inadvertent HPCI	118	114	0.14	0.14	6
Feedwater Controller Failure	276	122	0.18	0.18	7

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

Limiting Rod Pattern: Figure 8

Rod Block Reading	Rod Position (Feet Withdrawn)	Delta BP/P8x8R	Delta GE8x8EB	MLHGR BP/P8x8R	MLHGR GE8x8EB
104	3.0	0.07		15.59	16.59
105	3.5	0.10		16.61	17.61
106	4.0	0.12		17.23	18.23
107	4.0	0.12		17.23	18.23
108	7.0	0.20		17.26	18.26
109	12.0	0.26		17.26	18.26
110	12.0	0.26		17.26	18.26

Setpoint selected: 108

12. CYCLE MCPR VALUES (S.2.2)

Non-Pressurization Events

Exposure Range: BOC10 to EOC10

	BP/P8x8R	GE8x8EB
Inadvertent HPCI	1.22	1.22
Fuel Loading Error	--	1.18
Rod Withdrawal Error	1.24	1.24

Exposure Range: Extended EOC with Increased Core Flow and Final Feedwater Temperature Reduction

Inadvertent HPCI	1.18	1.18
Fuel Loading Error	--	1.15
Rod Withdrawal Error	--	1.16

Pressurization Events

Exposure Range: BOC10 to EOC10

	Option A		Option B	
	BP/P8x8R	GE8x8EB	BP/P8x8R	GE8x8EB
Turbine Trip w/o Bypass	1.31	1.31	1.26	1.27
Feedwater Controller Failure	1.25	1.26	1.20	1.20

Exposure Range: Extended EOC with Increased Core Flow and Final Feedwater Temperature Reduction

	Option A		Option B	
	BP/P8x8R	GE8x8EB	BP/P8x8R	GE8x8EB
Load Rejection w/o Bypass	1.34	1.35	1.29	1.30
Feedwater Controller Failure	1.28	1.29	1.23	1.24

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	Pressure Steam Line (psig)	Pressure Valve (psig)	Plant Response
MSIV Closure (Flux Scram)	1300	1324	Figure 9

14. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

Event	Delta CPR
Rotated Bundle Error	0.14

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

Control Rod Drop Accident Analysis is not required for banked position withdrawal sequence plants. NRC approval is documented in NEDE-24011-P-A-8-US, May 1986.

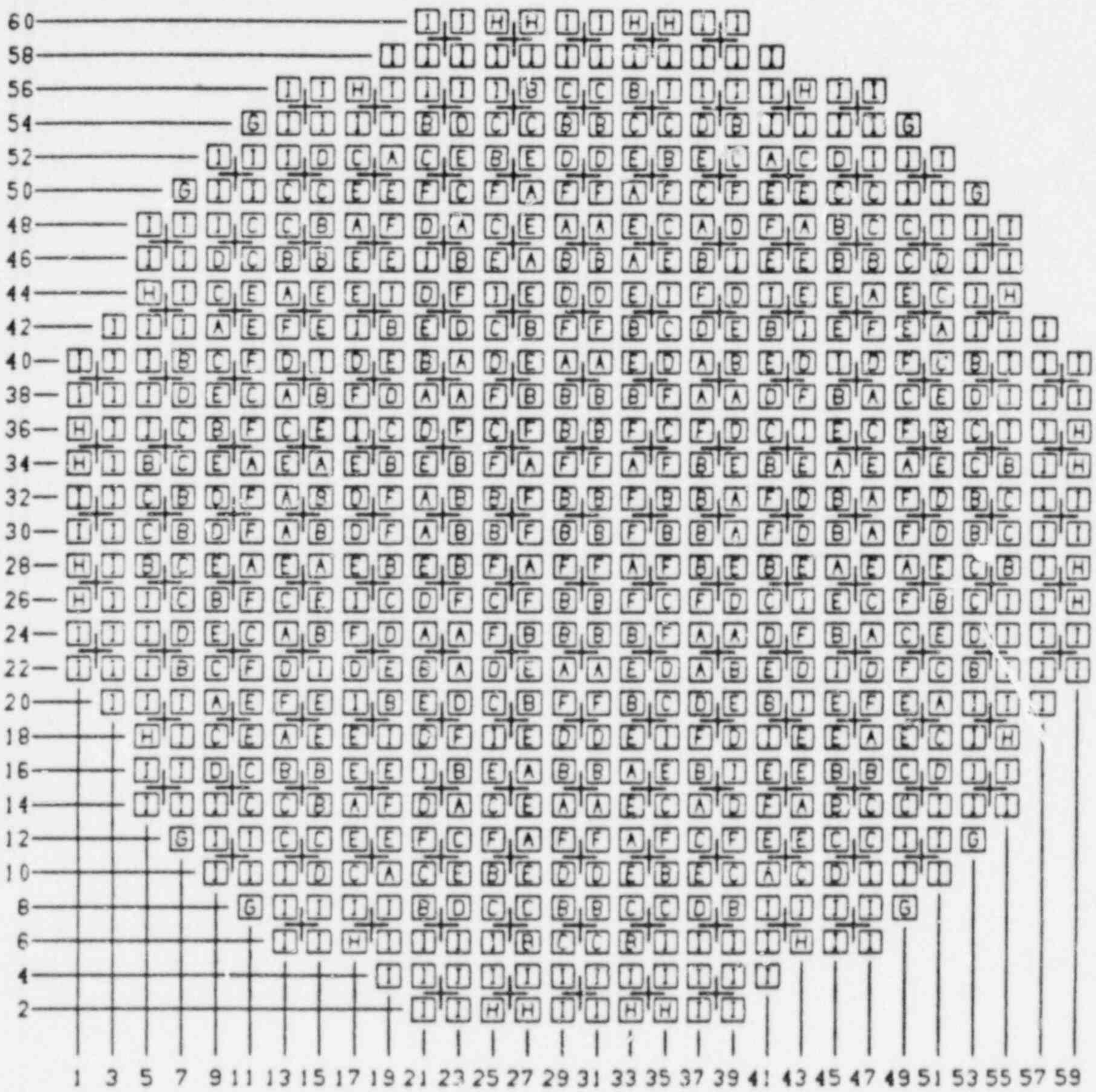
16. STABILITY ANALYSIS RESULTS (S.2.4)

BWR 2/3 plants are exempt from performing cycle-specific stability analyses.

17. LOSS-OF-COOLANT ACCIDENT RESULTS

LOCA Method Used: SAFER/GESTR-LOCA

See Quad Cities Nuclear Power Station Units 1 & 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis, NEDC-31345P, June 1987 (as amended).



FUEL TYPE	
A = BP8DRB282	F = BD316A
B = BP8DRB283H	G = P8DGB263L
C = BP8DRB299L	H = P8DGB298
D = BP8DRB299	I = BP8DRB265H
E = BD300C	

Figure 1. Reference Core Loading Pattern

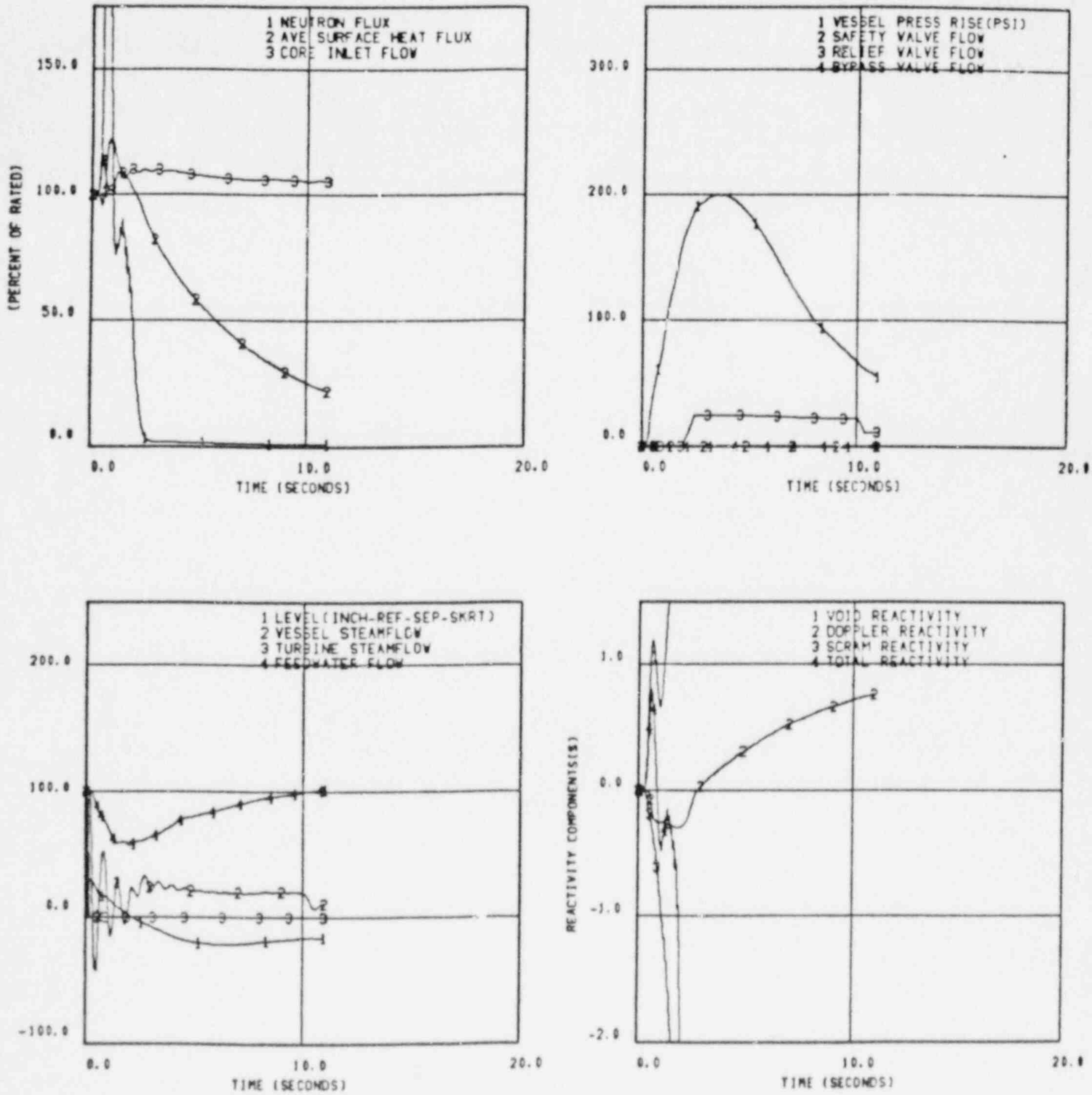


Figure 2. Plant Response to Turbine Trip Without Bypass

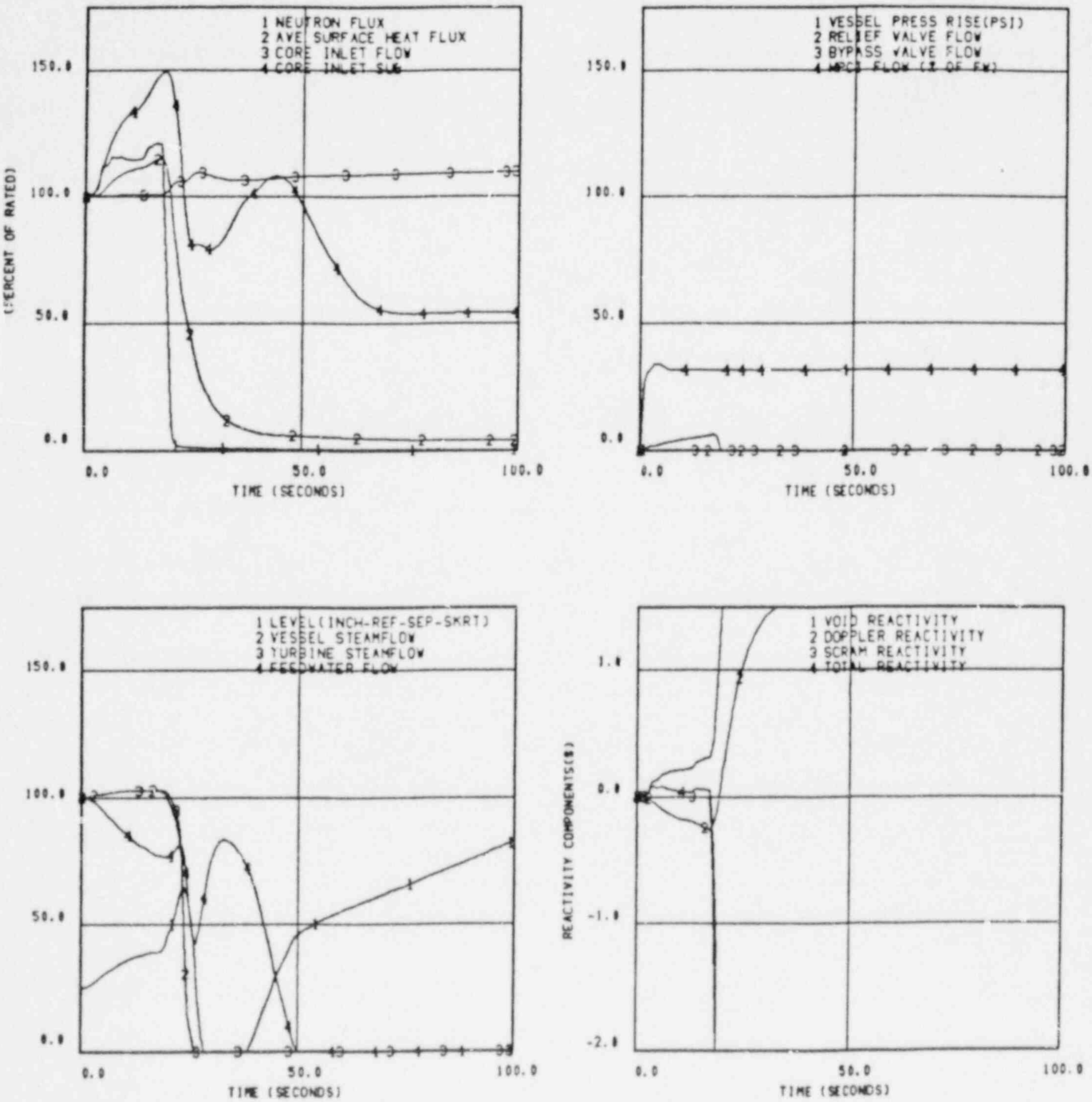


Figure 3. Plant Response to Inadvertent Activation of HPCI

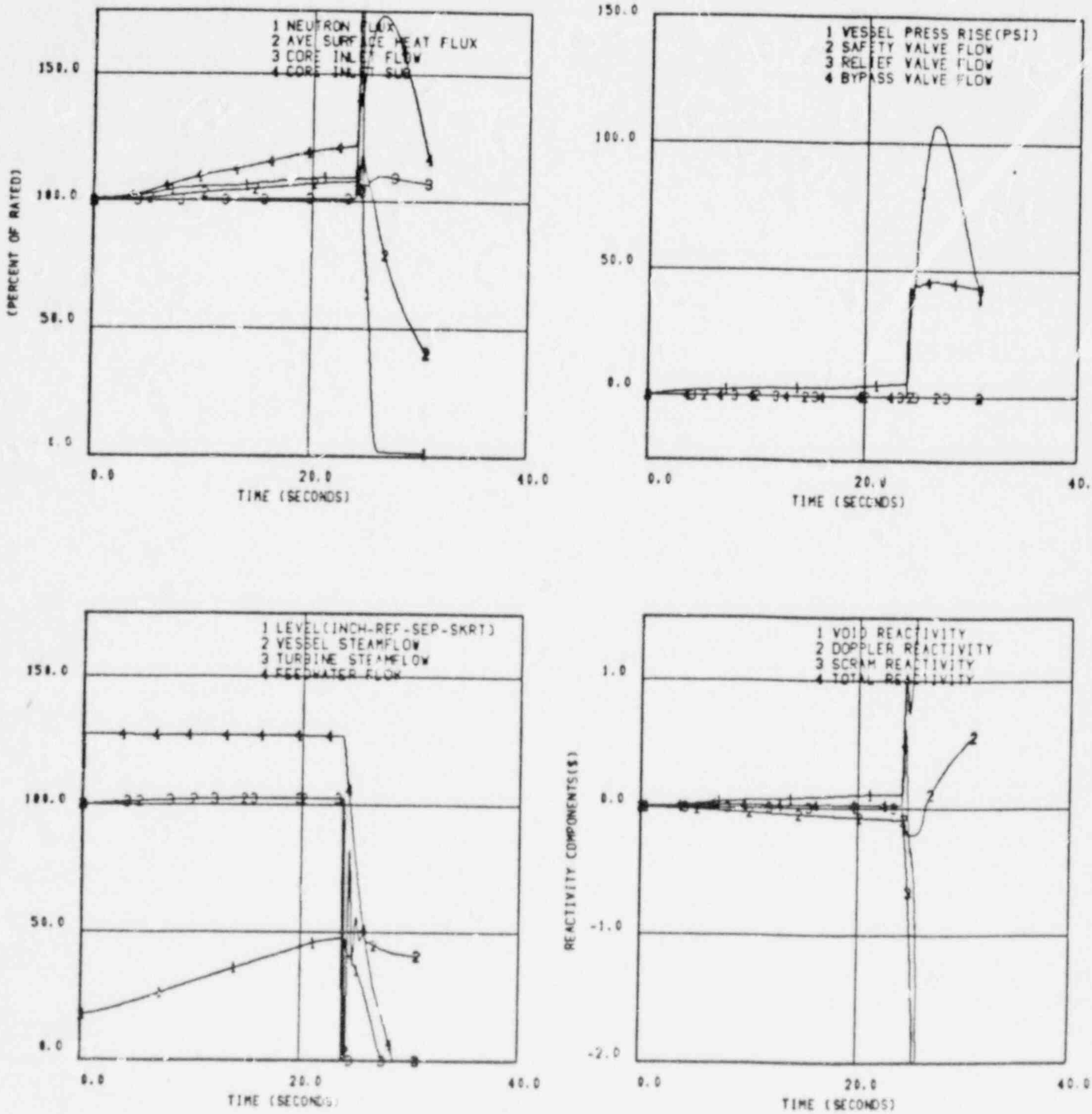


Figure 4. Plant Response to Feedwater Controller Failure

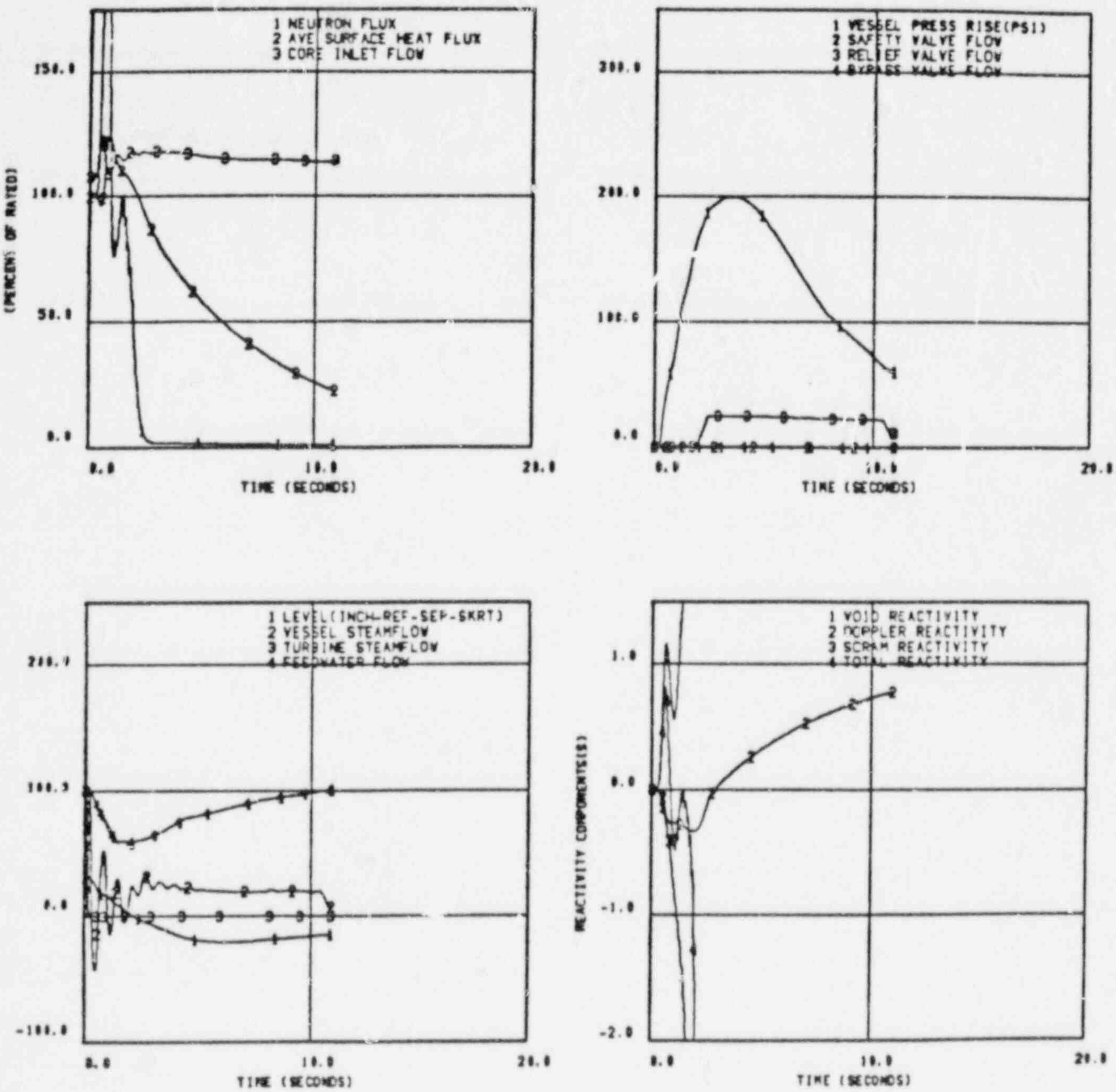


Figure 5. Plant Response to Generator Load Rejection Without Bypass (Extended EOC With Increased Core Flow and Feedwater Temperature Reduction)

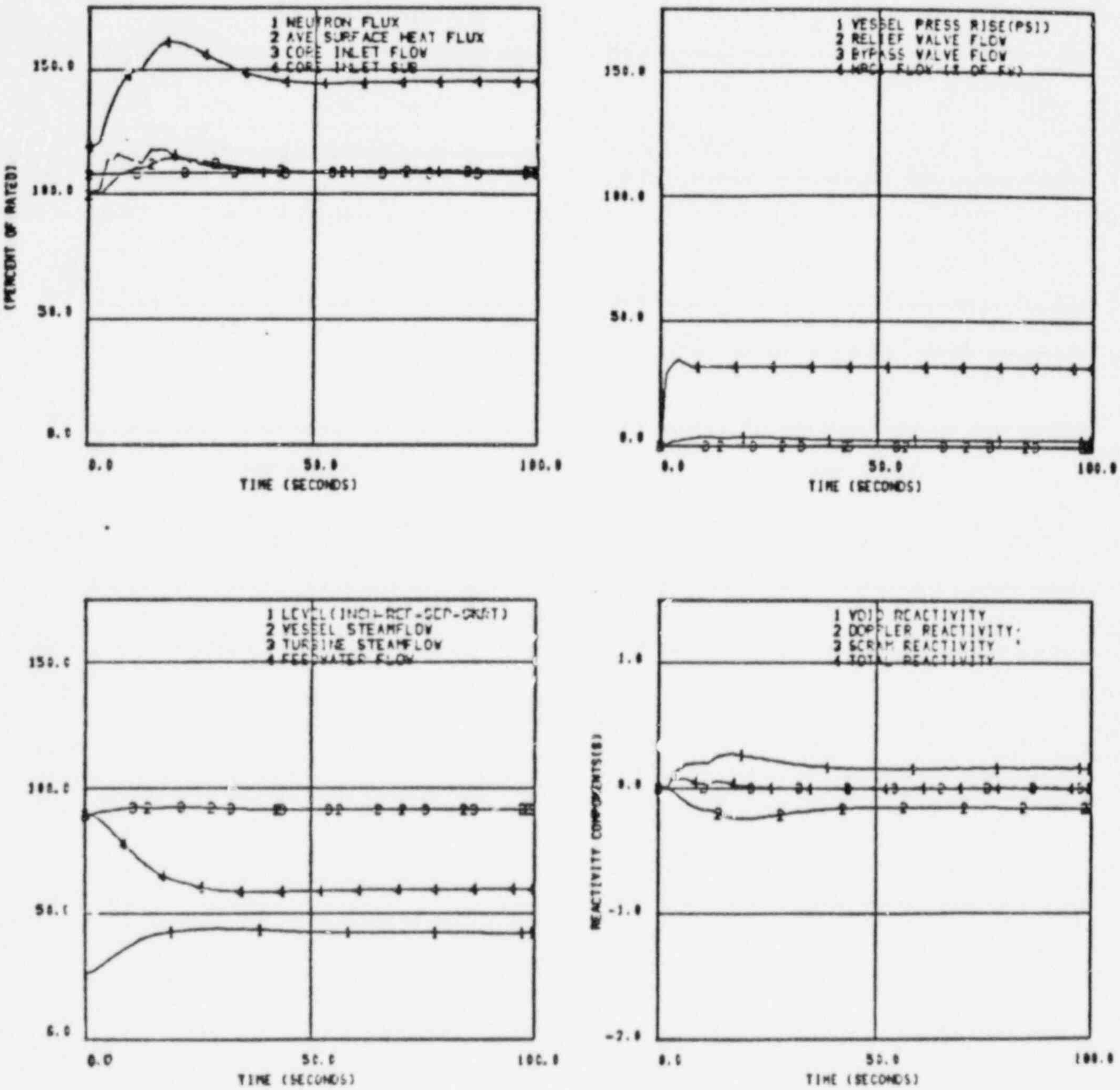


Figure 6. Plant Response to Inadvertent Activation of HPCI (Extended EOC with Increased Core Flow and Feedwater Temperature Reduction)

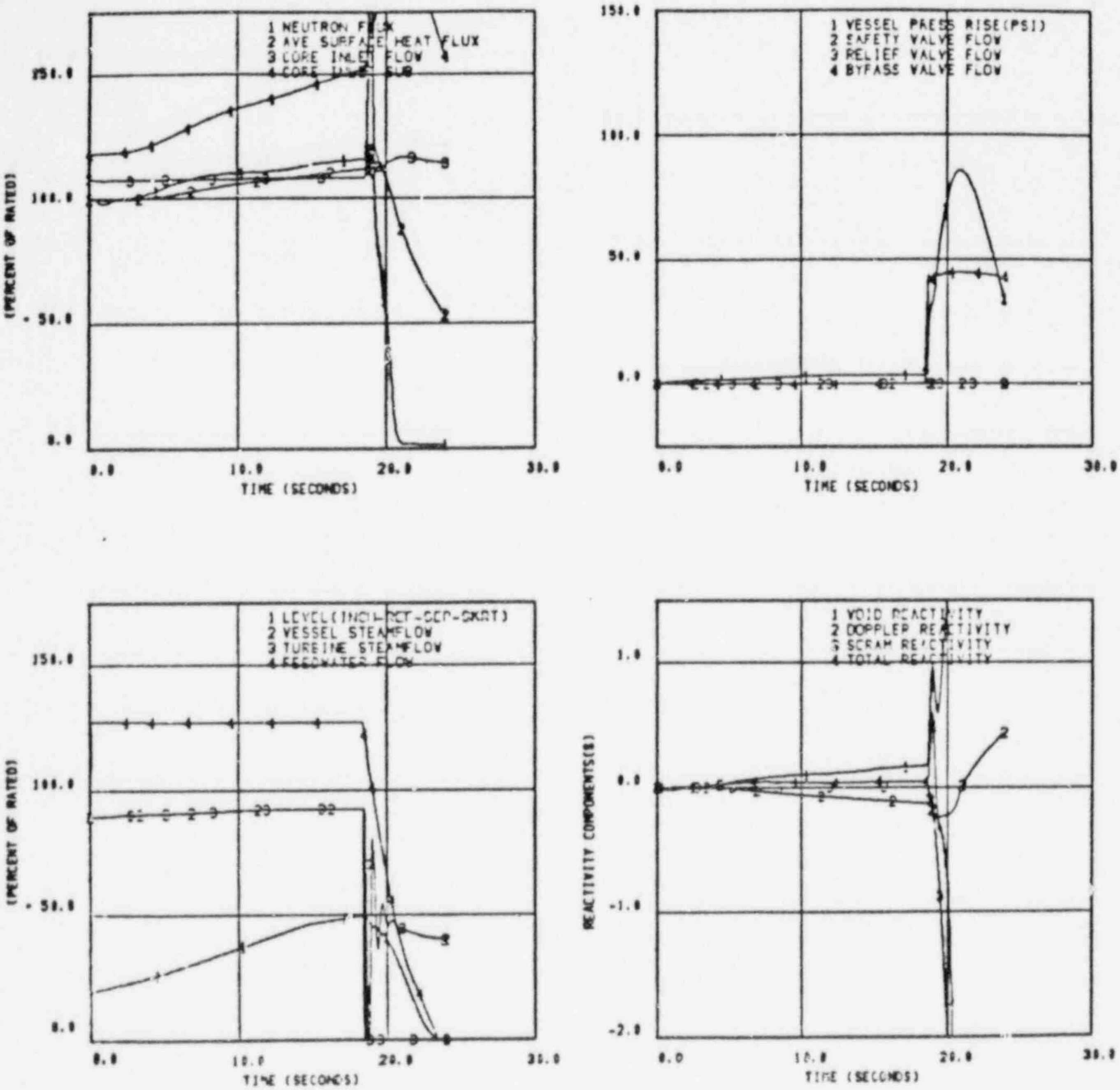


Figure 7. Plant Response to Feedwater Controller Failure (Extended EOC with Increased Core Flow and Feedwater Temperature Reduction)

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58
59						12		12		12					
55					0		0		0		0				
51						16		12		16					
47			0		10		0		0		10		0		
43		16		40		40		16		40		40		16	
39	0		10		10		10		10		10		10		0
35		40		44		36		40		36		44		40	
31	10		10		0		10		10		0		10		10
27		40		44		36		40		36		44		40	
23	0		10		10		10		10		10		10		0
19		16		40		40		16		40		40		16	
15			0		10		0		0		10		0		
11						16		12		16					
7					0		0		0		0				
3						12		12		12					

NOTES:

1. No. indicates number of notches withdrawn out of 48. Blank is a withdrawn rod.
2. Error rod is (18, 31).

Figure 8. Limiting Rod Pattern

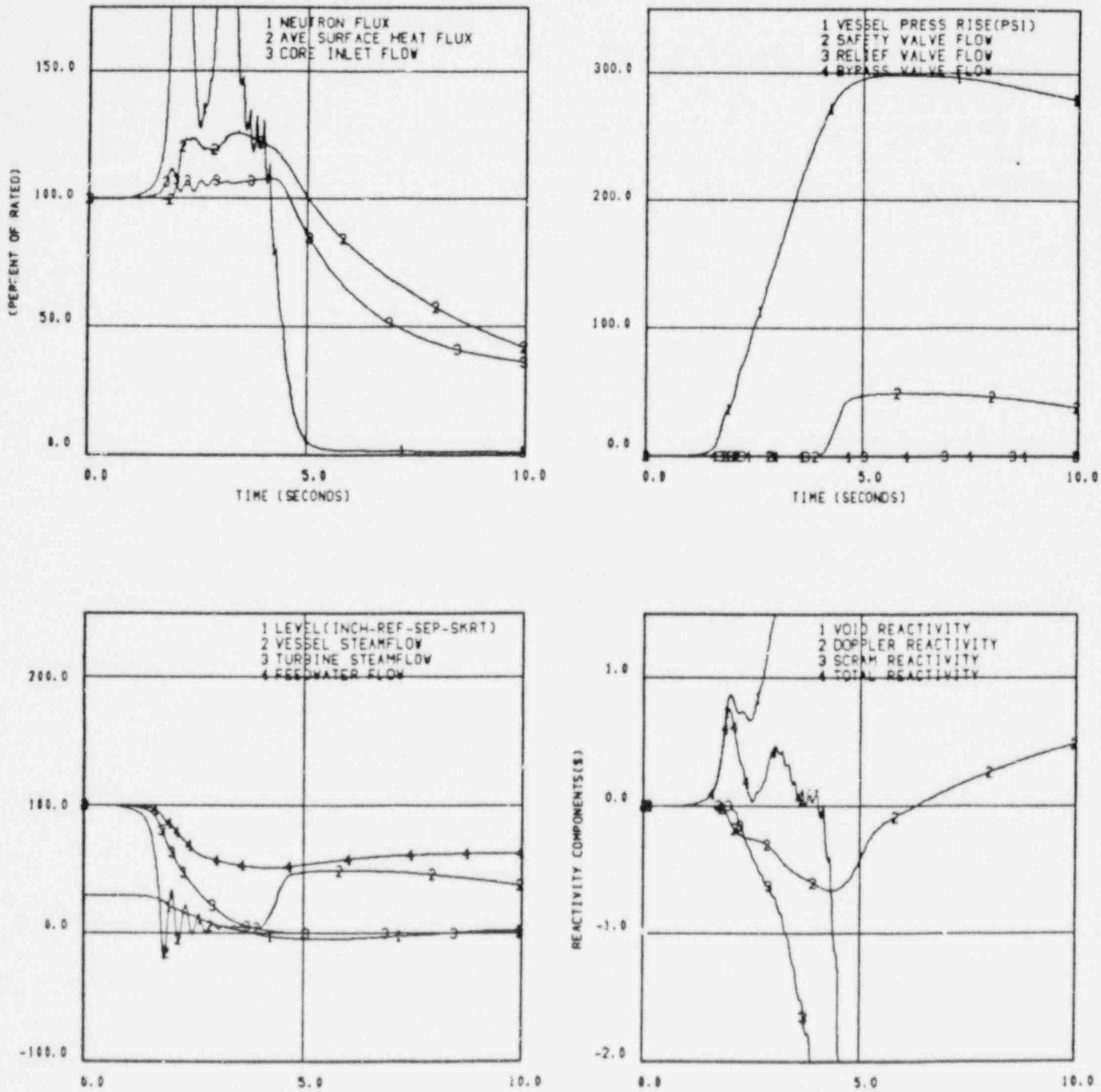


Figure 9. Plant Response to MSIV Closure, Flux Scram

APPENDIX A

ANALYSIS CONDITIONS

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

TABLE A-1

PLANT PARAMETERS

Parameter	Analysis Value	NEDE-24011 Value
Non-fuel Power Fraction	0.038	0.035
Relief Valve Capacity, lb/hr	558,000	645,000
Safety/Relief Valve, Lowest Setpoint, psig	1135	1135 + 1%

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EXTENDED OPERATING DOMAIN AND EQUIPMENT OUT-OF-SERVICE FOR QUAD CITIES NUCLEAR POWER STATION UNITS 1 AND 2

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ILLUSTRATIONS

SUMMARY

This two-part report documents a comprehensive set of analyses performed for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2 to:

(1) Support the expansion of the operating domain of the current power/flow map.

(2) Justify continuous operation with one of the following equipment out-of-service: one feedwater heater string, one safety/relief valve, one recirculation loop.

The first part of the report addresses the technical bases to justify the proposed operating domain expansion, and the second part presents analyses results and associated technical specification limits to support plant operation with certain equipment out-of-service.

The standard operating domain for QCNPS Units 1 and 2 was previously modified to include the operating region above the rated rod line bounded by the 108% average power range monitor (APRM) rod block line, the rated power line and the rated core flow line (Reference 1). For this analysis, the standard operating envelope is modified to include the increased core flow (ICF) region (Figure A.1-1). This safety evaluation is also applicable to operation beyond nominal end-of-cycle with ICF and up to 100 degree F final feedwater temperature reduction (FFWTR) or beyond nominal end-of-cycle at less than rated core flow with reduced feedwater temperature. The cycle extension is then followed by a coastdown to 20% power.

As part of the expanded operating domain analysis, the limiting abnormal operational transients at rated core flow condition are reevaluated for 100% power and 108% core flow with and without FFWTR to support the ICF operation mode throughout the cycle and ICF/FFWTR operation

beyond nominal end-of-cycle. The operating limits obtained for ICF with FFWTR also bound operation at less than rated core flow with reduced feedwater temperature.

The loss-of-coolant accident (LOCA), the containment LOCA response, and the reactor stability compliance criteria were also evaluated to justify operation at ICF/FFWTR conditions. In addition, the effect of the increased pressure differences due to ICF on the reactor internal components, fuel channels, and fuel bundles was also analyzed to show that their design limits will not be exceeded. The effect of ICF on the flow-induced vibration response of the reactor internals was also evaluated to ensure that the response is within acceptable limits. The increase in the feedwater nozzle usage factors due to reduced feedwater temperature was also addressed.

Results of the analysis reported herein show that there is no impact on LOCA performance, containment design loads, reactor internal loadings capabilities, or feedwater nozzle fatigue resulting from operation in the expanded power/flow domain. The recirculation system performance data for QCNPS indicate that the currently achievable core flow rate may be less than 108% of rated. However, this is a system performance consideration rather than a safety concern. Therefore, the analyses were performed at the bounding condition of 108% rated core flow.

For the out-of-service equipment mentioned in Item 2 above, the analyses performed assumed a single failure only and established the licensing bases for continuous plant operation in the expanded power/flow map <u>excluding the ICF region</u>. The feedwater heater out-of-service (corresponding to a 100 degree F reduction in feedwater temperature) is included as part of the transient analysis input assumptions. Specific cycle independent operating MCPR limits are established to allow continuous plant operation with this equipment failure. In the case of a safety/relief valve (S/RV) out-of-service, transient analysis results showed that there is no impact on the calculated MCPR and that the overpressure margin to the ASME upset code limit is still satisfied. An analysis justifying recirculation system single loop-operation (SLO) was previously performed for QCNPS (Reference 2). This analysis has been reviewed and shown to be applicable with the new GE8x8EB fuel design. In addition, the impact on plant operating limits were also evaluated for SLO mode with or without one safety/relief valve out-of-service in the normal operating domain as well as in the region above the rated rod line. The analyses of the above-mentioned equipment out-of-service also showed that there is no impact on the LOCA containment response, reactor stability performance, or fuel peak clad temperature.

PART A

EXPANDED OPERATING DOMAIN ANALYSIS
FOR QUAD CITIES NUCLEAR POWER STATION UNITS 1 AND 2

A.1 INTRODUCTION

Most boiling water reactors (BWRs) have recirculation system pumping capabilities in excess of that required to provide 100% rated core flow. The use of increased core flow (ICF) above 100% rated core flow can provide greater operational flexibility in reaching and maintaining full power during the cycle and can extend the operating cycle at rated power. The magnitude of this extension is dependent on the characteristics of the core and on the maximum allowable core flow. In general, operation with ICF can extend full power operation by approximately one week.

Final feedwater temperature reduction (FFWTR) at the end-of-cycle (EOC) can further extend the operating cycle. In general, a 100 degree F reduction in feedwater temperature provides approximately two weeks of additional full thermal power operation. The ICF region (Figure A.1-1) is referred to as the extended operating domain.

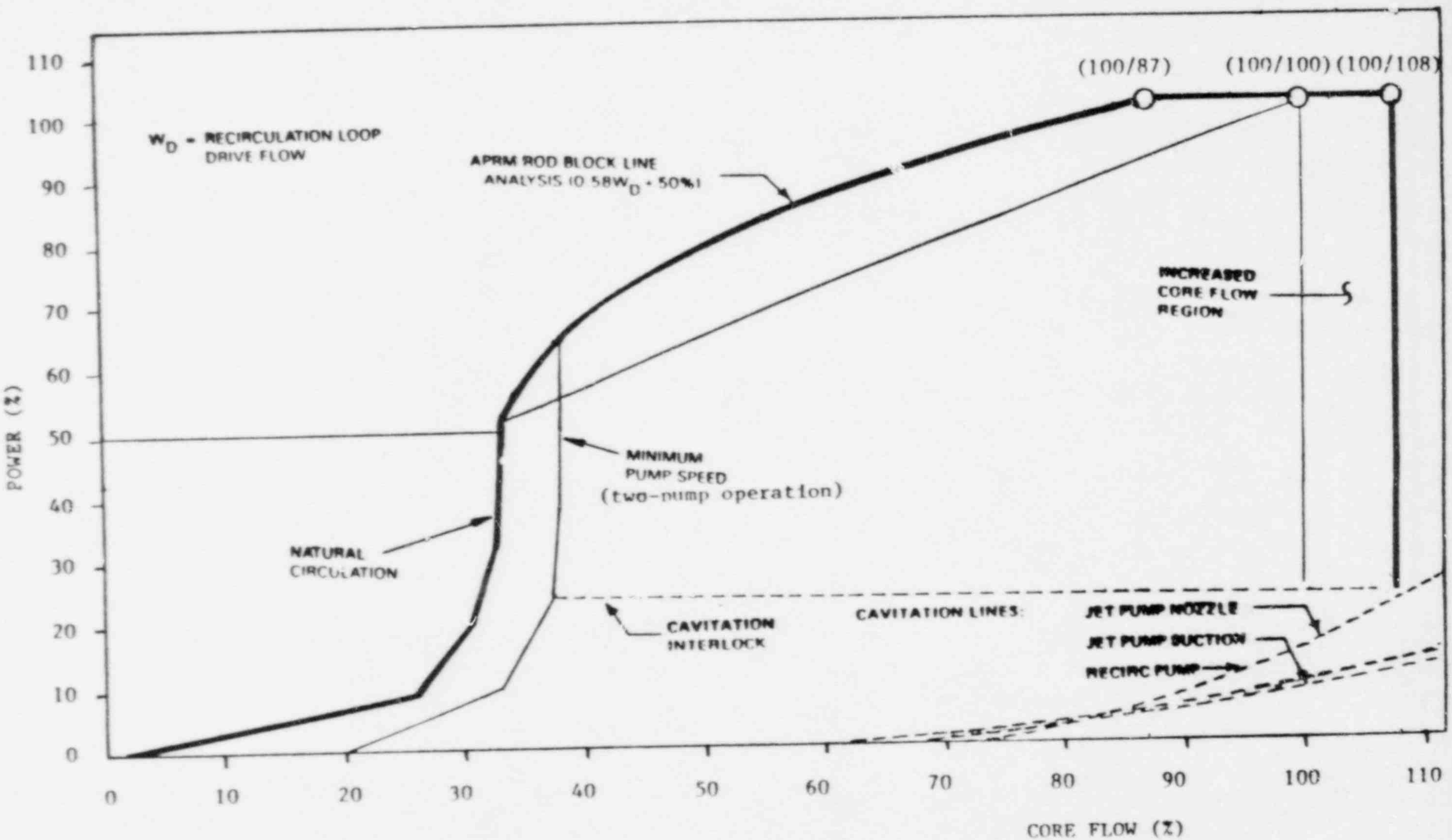

Figure A.1.1 Extended Operating Domain Power/Flow Map for QCNPS

A.2 PLANT OPERATIONAL STATUS

A.2.1 RECIRCULATION SYSTEM CAPABILITY

Based on recirculation system performance data for QCNPS, the maximum core flow is expected to be somewhat less than 108% of rated core flow. Given that the ICF capability results from a system performance consideration rather than a safety concern, the analyses are performed at the bounding condition of 108% of rated core flow.

A.2.2 MINIMUM END-OF-CYCLE COASTDOWN POWER LEVEL

Standard licensing transient analyses are performed at the full power, end-of-cycle (EOC), all-rods-out condition. Once an individual plant reaches this condition, it may shut down for refueling or it may be placed in a coastdown mode of operation. For QCNPS, this type of operation involves coasting down to a lower percent of rated power while maintaining constant core flow. For the purpose of this safety evaluation, coastdown to 20% power is assumed.

The power profile during this period is assumed to be a linear function with respect to exposure. It is expected that the actual profile will be a slow exponential curve. An analysis using the linear approximation, however, will be conservative, since it bounds the power level for any given exposure. In Reference 3, evaluations were made at 90%, 80% and 70% power level points on the linear curve. The results showed that the pressure margins from the limiting pressurization transient and the MCPR operating limits exhibit a larger margin for each of these points than the EOC full power, full flow case. The LOCA analysis results for the rated power/108% of rated core flow case is conservative for the coastdown period since the power will be decreasing and ICF will be maintained. Reference 4 presents the results of analyses to justify coastdown power operation.

A.3 SAFETY ANALYSIS

A.3.1 ABNORMAL OPERATIONAL TRANSIENTS

A.3.1.1 Limiting Transients

The limiting operational transients analyzed in the QCNPS Unit 1 Reload 9, Cycle 10 supplemental reload licensing submittal (Reference 5) were re-evaluated for ICF followed by final feedwater temperature reduction (FFWTR) as follows:

Nuclear transient data for 100% power, 108% core flow (100P/108F) with and without FFWTR (corresponding to a 100 degree F reduction in feedwater temperature) were developed for rated power at exposures beyond end-of-cycle 10 (EOC 10). These nuclear data were then used to analyze the load rejection without bypass (LRNBP), turbine trip without bypass (TTNBP) and the feedwater controller failure (FWCF) events at the 100P/108F condition. The transient events were analyzed based on core characteristics with both BP/P8x8R and GE8x8EB.

The results of the transient analyses are presented in Tables A.3-1, A.3-2 and A.3-3. As shown in Tables A.3-2 and A.3-3, the limiting event (LRNBP) from Reference 5 bounds all of the ΔCPR results from the ICF and ICF/FFWTR analyses with one exception: the BP/P8x8R limits for LRNBP and TTNBP at ICF with rated feedwater temperature. Therefore, the current technical specification MCPR operating limits should be modified to incorporate the changes as shown in Table A.3-3 for operation at ICF conditions. The transient performance responses for the LRNBP, TTNBP and FWCF events are shown in Figures A.3-1 through A.3-7.

A.3.1.2 Overpressurization Analysis

The limiting transient for the ASME code overpressurization analys [main steam isolation valve (MSIV) closure with flux scram (direct scram failure)], was evaluated for extended EOC10 conditions with ICF without

FFWTR (Table A.3-4 and Figure A.3-8). For this evaluation, ICF without FFWTR is more severe than with FFWTR, and the MSIV closure with flux scram event provides the most limiting overpressure transient response. The transient analysis (Table A.3-4) for the ICF condition produced a peak vessel pressure of 1327 psig, which is below the ASME Code upset limit of 1375 psig and, therefore, is acceptable.

A.3.1.3 Rod Withdrawal Error

The rod withdrawal error (RWE) transient was evaluated under ICF conditions. When ICF is employed, the rod block monitor (RBM) setpoint (which is flow biased) increases and gives a higher MCPR limi . Thus, the RBM should be clipped at flows greater than 100% of rated so that the ΔCPR values determined without ICF (Reference 5) apply. The clipping procedure includes an adjustment to the RBM circuit so that the high RBM trip setpoint at flows greater than 100% of rated is equal to the value at 100% rated fl w. These results are independent of whether FFWTR is implemented or not, and are therefore bounding.

A.3.2 FUEL LOADING ERROR

Operation with ICF and/or FFWTR does not impact the analysis of the rotated bundle fuel loading error event. Thus, the results reported in the Cycle 10 reload licensing submittal (Reference 5) are applicable for operation with ICF and/or FFWTR.

A.3.3 ROD DROP ACCIDENT

The rod drop accident (RDA) event is a startup accident evaluated at cold and hot standby core conditions which are unaffected by ICF and FFWTR operation. Therefore, there is no change to the RDA analysis bases as presented in Reference 5, and the RDA requirements of Reference 5 are applicable for operation with ICF and/or FFWTR.

A.3.4 LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSIS

For core flows lower than a critical value, boiling transition at the limiting fuel node (i.e., the high power node) can occur sooner than observed at rated operating conditions. This phenomenon is referred to as early boiling transition (EBT). If EBT occurs for the high power node at reduced flow, the resultant peak cladding temperature (PCT) can exceed the rated condition results. If there is no PCT margin to regulatory limits, it may be necessary to apply a maximum average planar linear heat generation rate (MAPLHGR) multiplier for operation in certain flow ranges. Low flow effects were generically addressed in Reference 6, which was approved by the Reference 7 NRC Safety Evaluation Report. It showed that no MAPLHGR multiplier is required for the QCNPS class of plant.

The effects of ICF and/or FFWTR on LOCA analyses are insignificant because the parameters which most strongly affect the calculated PCT (i.e., high power node boiling transition time and core reflooding time) have been shown to be relatively insensitive to core flow and feedwater temperature changes of this magnitude. Both of these modes of operation tend to slightly improve the results. With the lower initial core void fraction, there is more liquid mass to be lost out of the break before core uncovery results. The net effect of void fraction and other effects will result in a LOCA PCT change of less than 10 degrees F, which is insignificant in view of the large PCT margins from the new SAFER/GESTR LOCA analysis (Reference 8).

Therefore, it is concluded that the LOCA analysis results for QCNPS are applicable and insensitive to operation with ICF and/or FFWTR.

A.3.5 THERMAL-HYDRAULIC STABILITY

The core and channel hydrodynamic decay ratio were evaluated for ICF and/or FFWTR operation. If the reactor initially operates at ICF and at or below the rated rod line, both channel and core decay ratio will be less severe than for operation at rated core flow. With FFWTR alone, the channel decay ratio would improve because of the increased subcooling effect but the core decay ratio could be slightly more severe. Therefore, if only FFWTR is utilized, operation should be at or below the rated rod line. However, the combined effect of operating the reactor with ICF first and then with FFWTR would result in a lower overall core and channel decay ratio than for normal operation.

Therefore, it is concluded that the reactor core stability and the channel hydrodynamic stability performance with ICF and/or FFWTR are within the established criteria.

Table A.3-1

CORE-WIDE TRANSIENT ANALYSIS RESULTS AT ICF AND ICF/FFWTR

Transient Description[a]	Figure Number	Fuel Exposure[b]	Power (% NBR)	Flow (% NBR)	Rated Feedwater Temperature Reduction[c] (^{o}F)	Maximum Neutron Flux (% NBR)	Maximum Core Avg. Surface Heat Flux (% Initial)	Maximum Vessel Press. (Psig)	Maximum Steam Line Press. (Psig)
LRNBP	Ref. 5	EOC	100	100	0	505.5	120.1	1230	1200
LRNBP	A.3-1	EOC+128	100	108	0	517.7	120.3	1232	1198
LRNBP	A.3-2	EOC+582	100	108	100	439.0	118.9	1214	1181
TTNBP	A.3-3	EOC	100	100	0	494.9	119.8	1229	1199
TTNBP	A.3-4	EOC+128	100	108	0	514.3	120.2	1231	1197
TTNBP	A.3-5	EOC+582	100	108	100	440.7	117.9	1211	1178
FWCF	Ref. 5	EOC	100	100	0	229.9	116.0	1138	1104
FWCF	A.3-6	EOC+128	100	108	0	229.8	115.8	1140	1102
FWCF	A.3-7	EOC+582	100	108	100	215.8	120.6	1120	1083

a. LRNBP = Load rejection with no bypass, TTNBP = Turbine trip with no bypass
FWCF = Feedwater controller failure at maximum demand

b. EOC = End-of-Cycle, EOC+128 = End-of-Cycle + 128 MWD/MT
EOC+582 = End-of-Cycle + 582 MWD/MT

c. Reduction in feedwater temperature from nominal rated feedwater temperature (340^{o}F)

Table A.3-2

CORE-WIDE ΔCPR RESULTS

Transients	Exposure	P/F	FFWTR	Uncorrected BP/8X8R	Uncorrected GE8X8EB	Option A BP/P8X8R	Option A GE8X8EB	Option B BP/P8X8R	Option B GE8X8EB
LRNBP	EOC	100/100	No	0.19	0.19	0.26	0.26	0.21	0.21
	EOC + 128	100/108	No	0.20	0.19	0.27	0.26	0.22	0.21
	EOC + 582	100/108	Yes	0.18	0.18	0.25	0.25	0.19	0.20
TTNBP	EOC	100/100	No	0.19	0.19	0.25	0.26	0.20	0.21
	EOC + 128	100/108	No	0.20	0.19	0.26	0.26	0.21	0.21
	EOC + 582	100/108	Yes	0.17	0.17	0.23	0.23	0.18	0.18
FWCF	EOC	100/100	No	0.14	0.14	0.20	0.20	0.15	0.15
	EOC + 128	100/108	No	0.14	0.15	0.20	0.21	0.15	0.16
	EOC + 582	100/108	Yes	0.17	0.17	0.23	0.23	0.18	0.18

Table A.3-3

MCPR OPERATING LIMITS
FOR QCNPS UNIT 1, EOC 10

Transient	Option A BP/P8X8R	Option A GE8X8EB	Option B BP/P8X8R	Option B GE8X8EB
LRNBP (100P/100F)	1.33	1.33	1.28	1.28
LRNBP (100P/108F, Rated Feedwater Temperature)	1.34	1.33	1.29	1.28
LRNBP (100P/108F, FFWTR)	1.32	1.32	1.26	1.27

Table A.3-4

OVERPRESSURIZATION ANALYSIS RESULTS

TRANSIENT	Initial Power (%)	Initial Flow (%)	Maximum Steamline Pressure (PSIG)	Maximum Vessel Pressure (Psig)	Figure Number
MSIV Closure - Flux Scram (Reference 5, EOC)	100	100	1295	1319	Reference 5
MSIV Closure - Flux Scram (ICF w/o FFWTR, EOC+128 MWD/MT)	100	108	1290	1327	Figure A.3-8

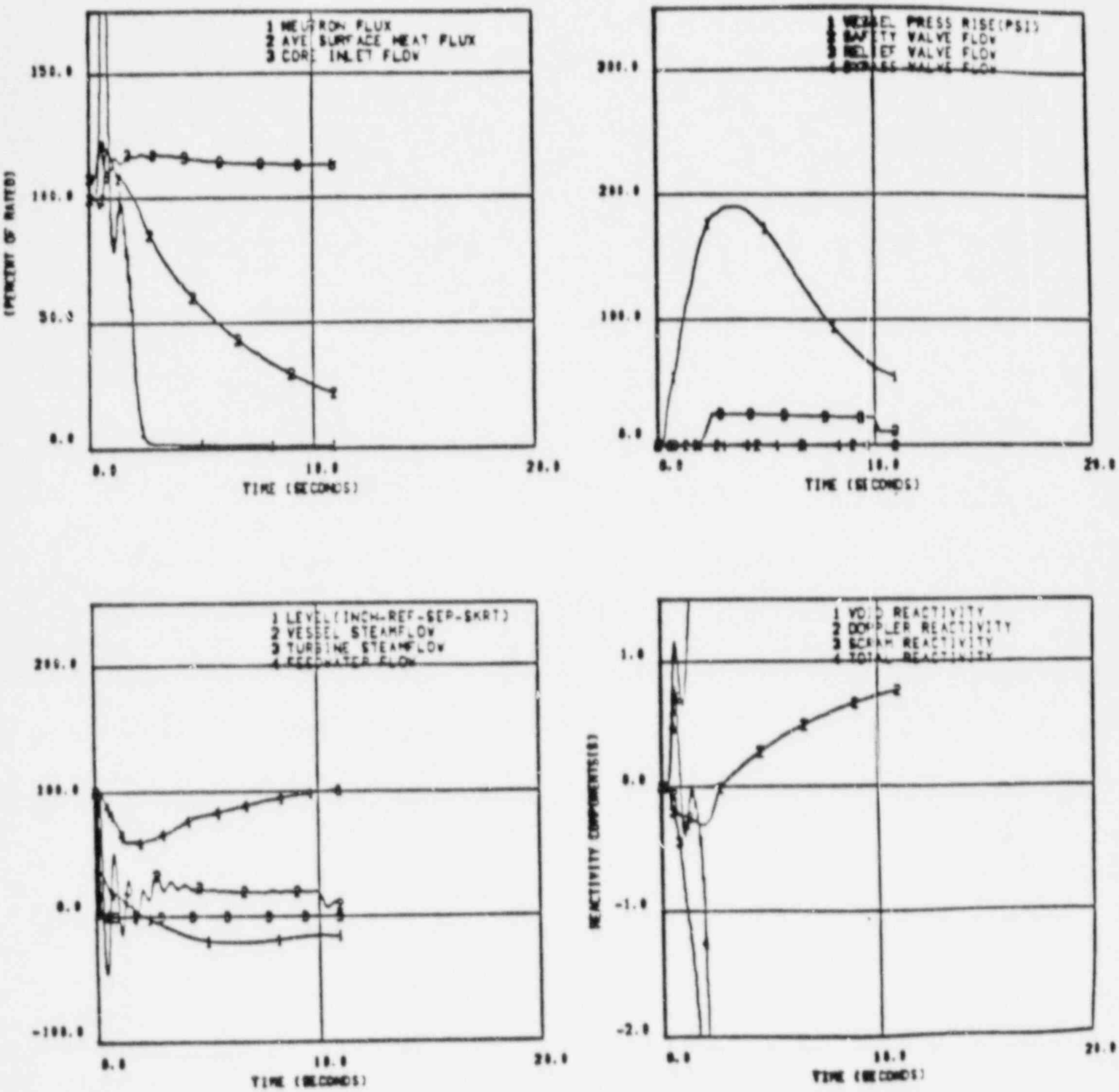

Figure A.3.1 - Plant Response to Load Rejection without Bypass, 100% Power 108% Flow, Rated Feedwater Temperature, EOC+128 MWD/MT

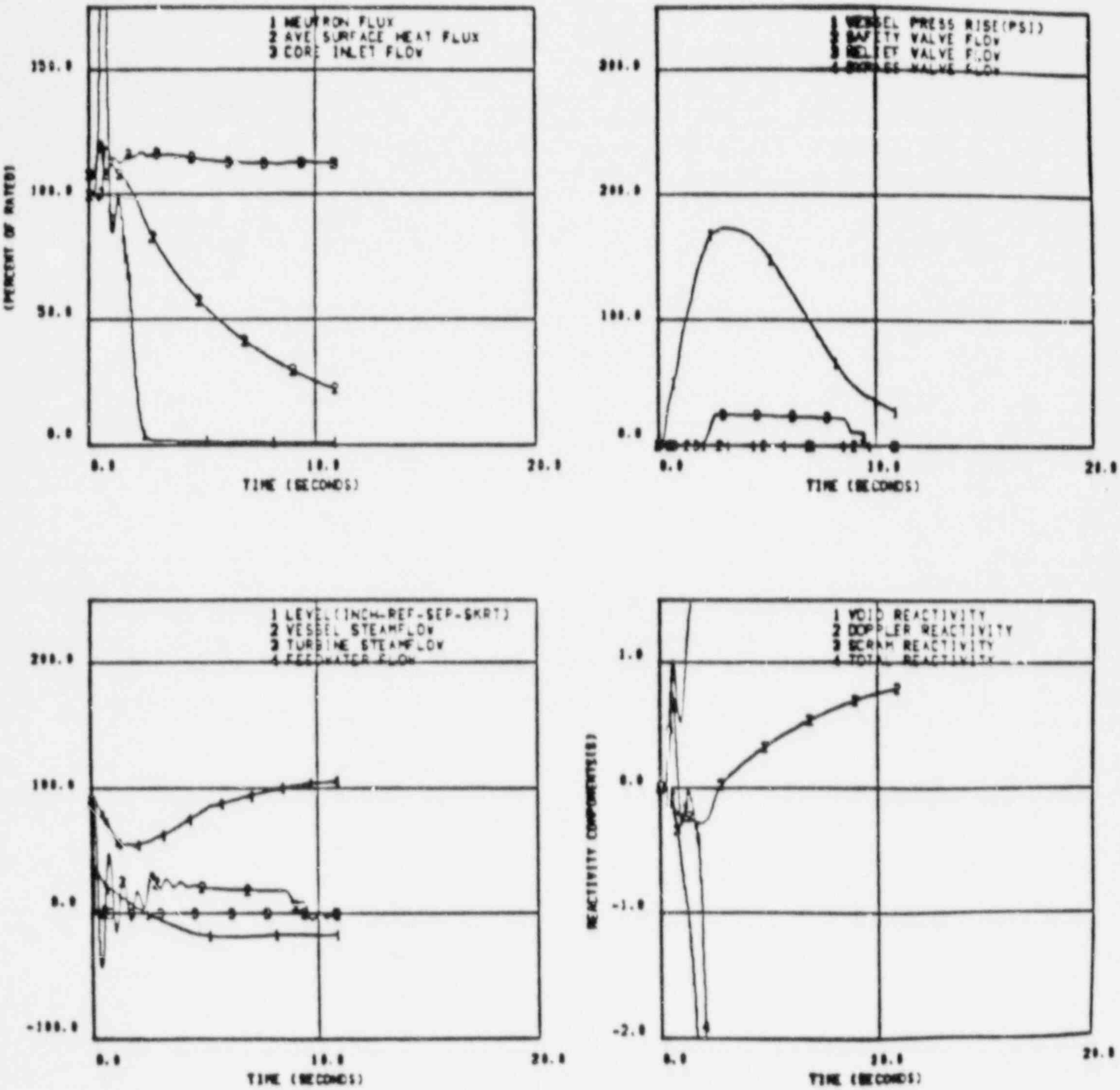

Figure A.3.2 - Plant Response to Load Rejection Without Bypass, 100% Power 108% Flow, FFWTR at EOC+582 MWD/MT.

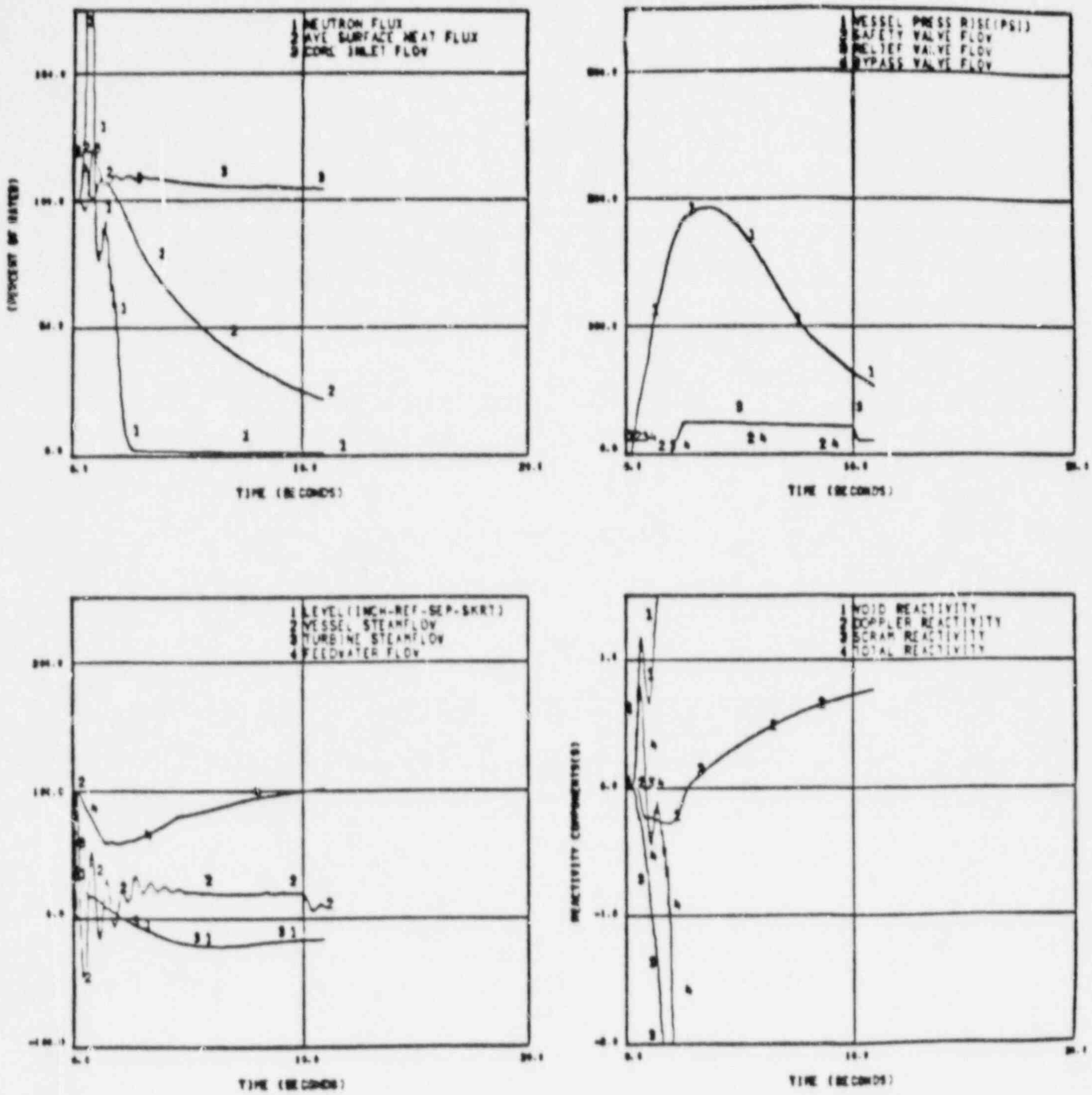

Figure A.3.3 - Plant Response to Turbine Trip without Bypass, 100% Power 100% Flow, Rated Feedwater Temperature, EOC.

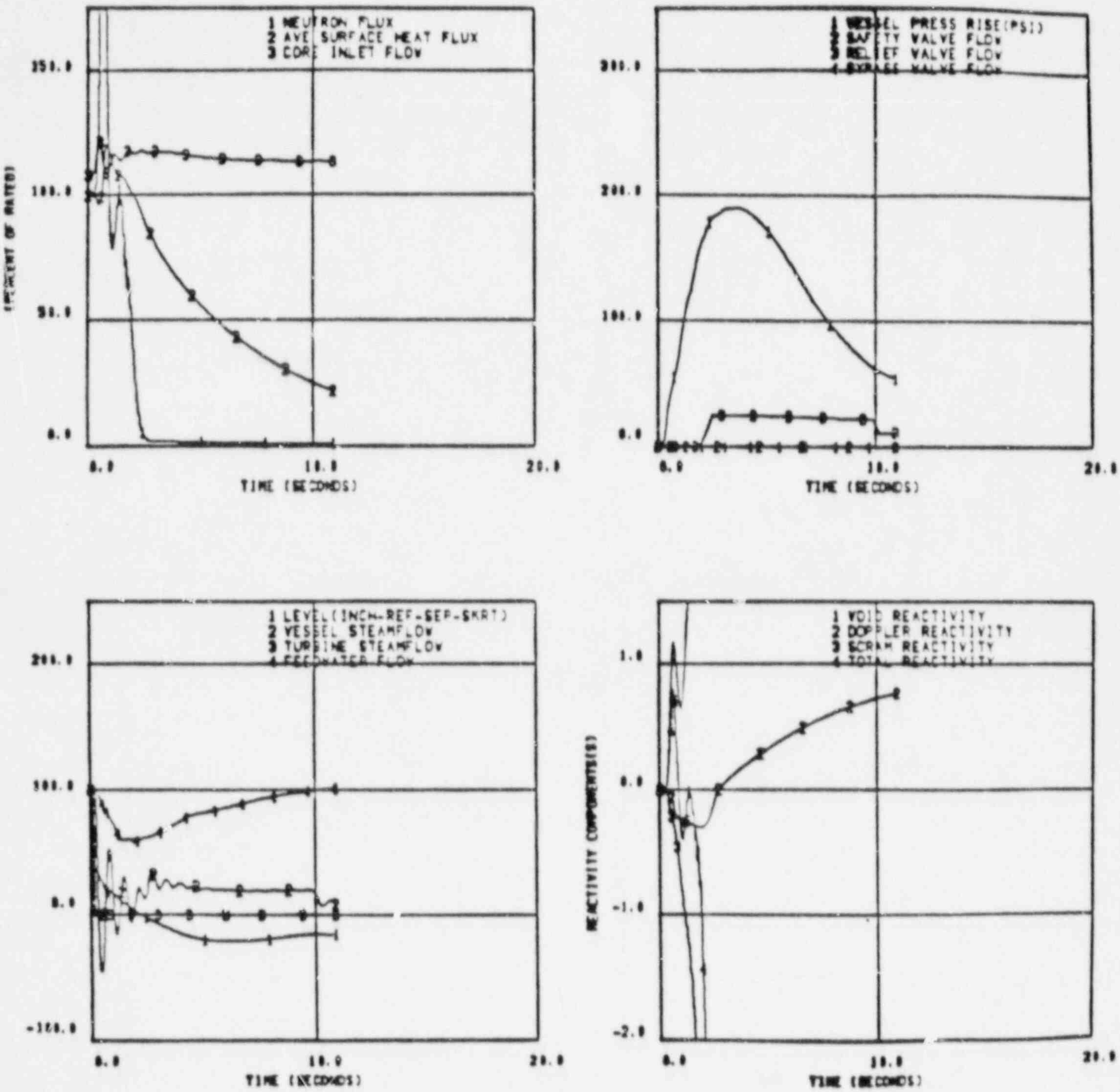

Figure A.3.4 – Plant Response to Turbine Trip without Bypass, 100% Power 108% Flow, Rated Feedwater Temperature, EOC+128 MWD/MT.

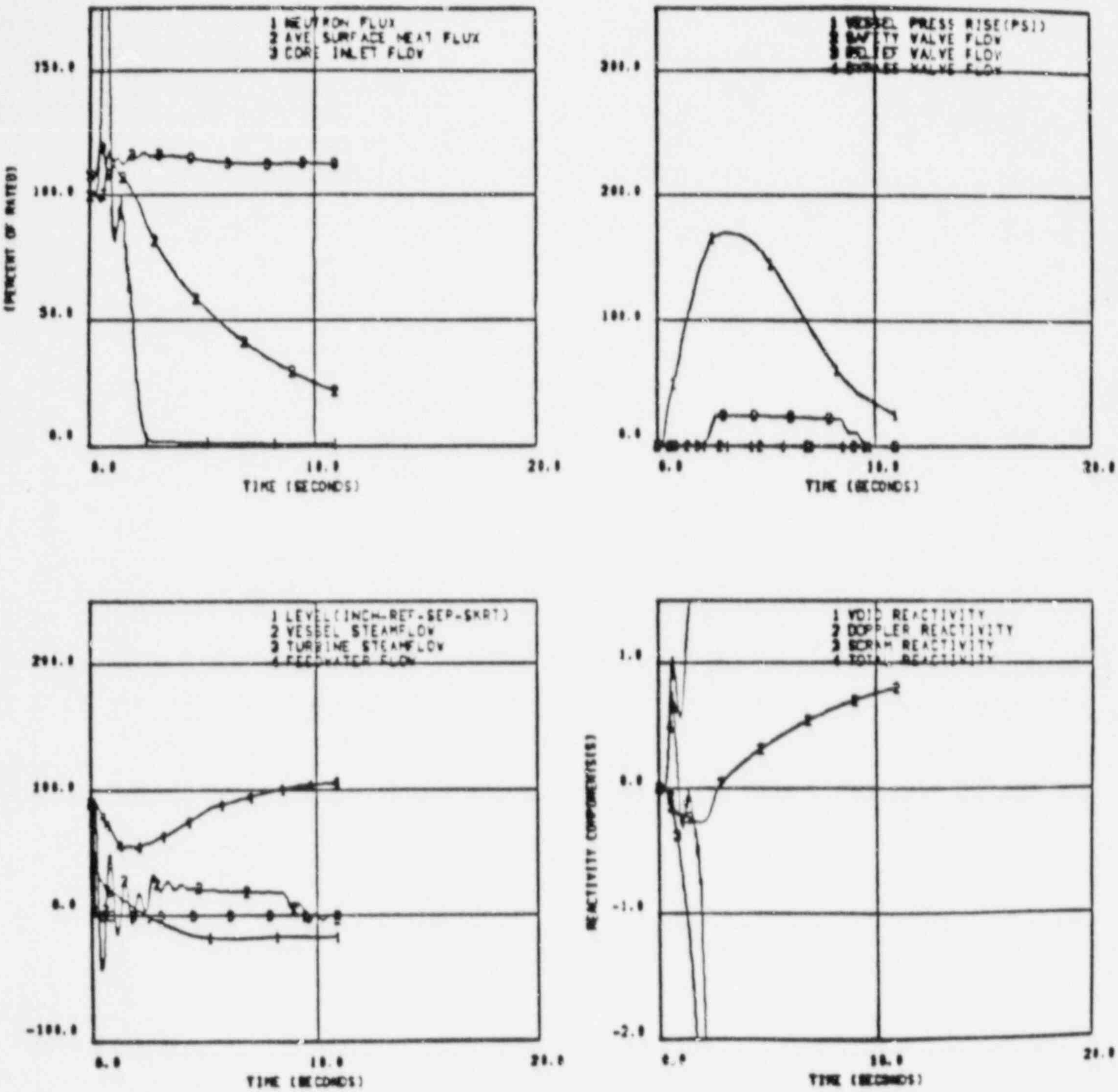

Figure A.3.5 - Plant Response to Turbine Trip without Bypass, 100% Power 108% Flow, FFWTR at EOC+582 MWD/MT.

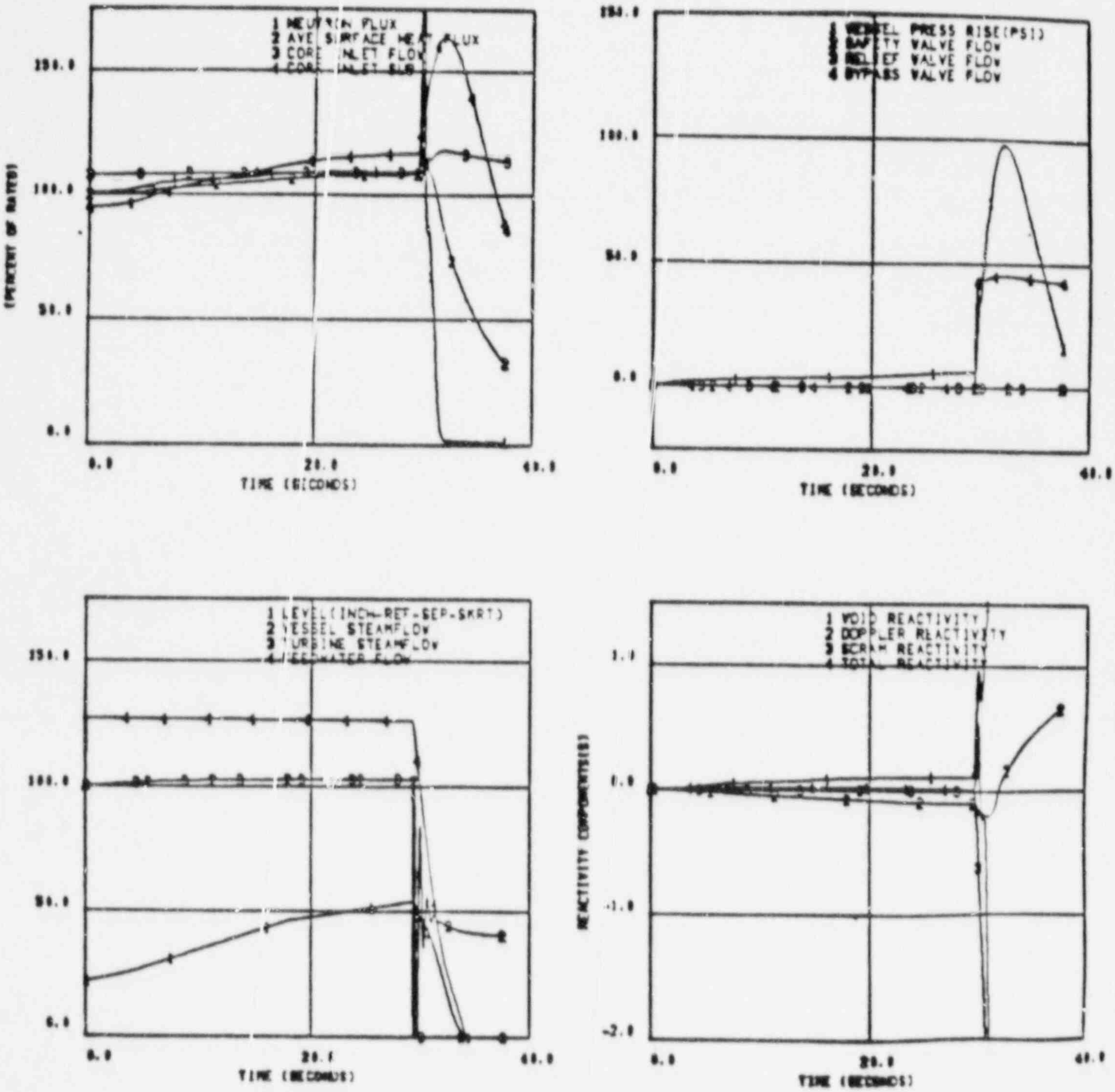

Figure A.3.6 - Plant Response to Feedwater Controller Failure, 100% Power 108% Flow, Rated Feedwater Temperature, EOC+128 MWD/MT.

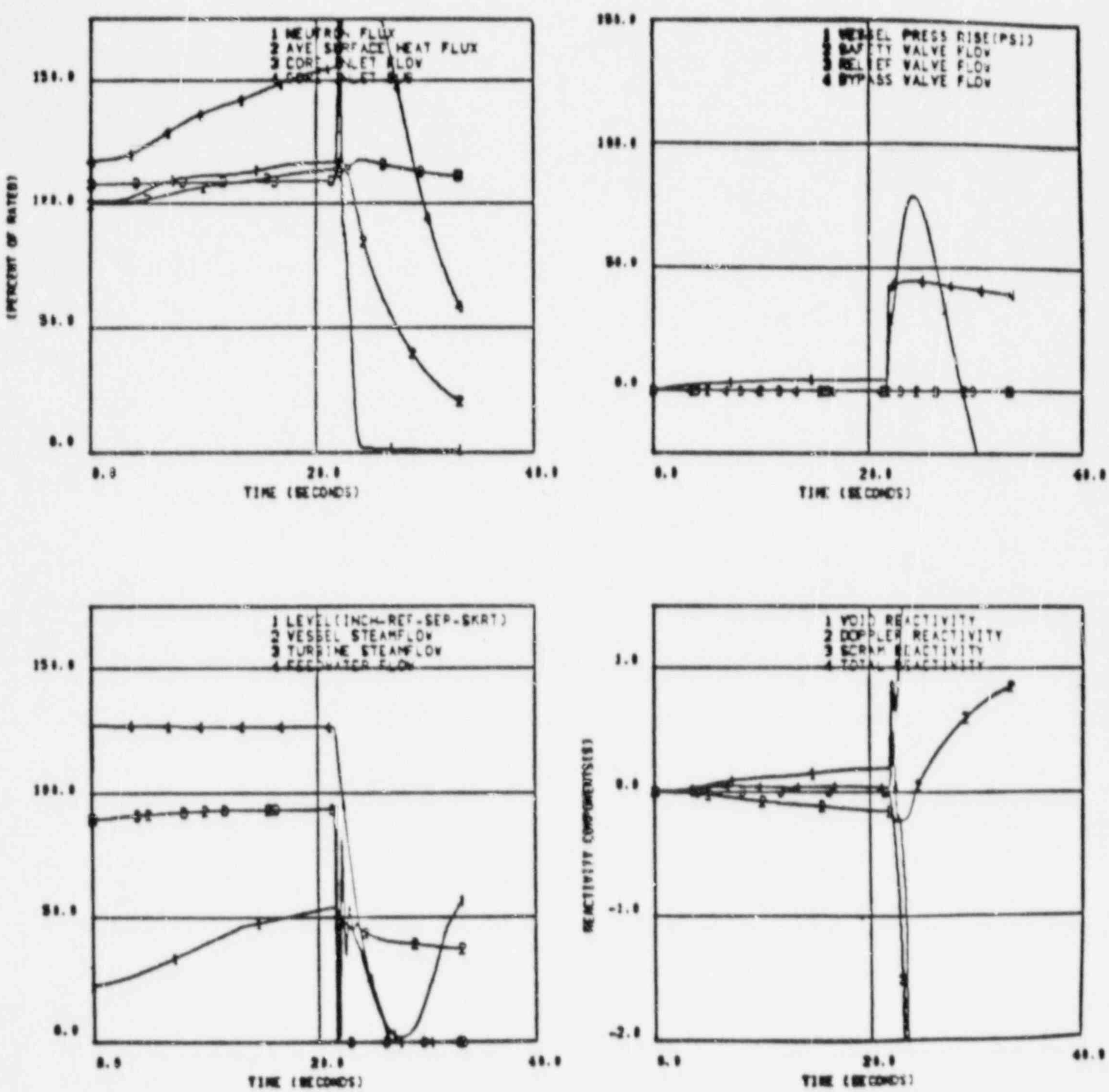

Figure A.3.7 - Plant Response to Feedwater Controller Failure, 100% Power 108% Flow, FFWTR at EOC+582 MWD/MT.

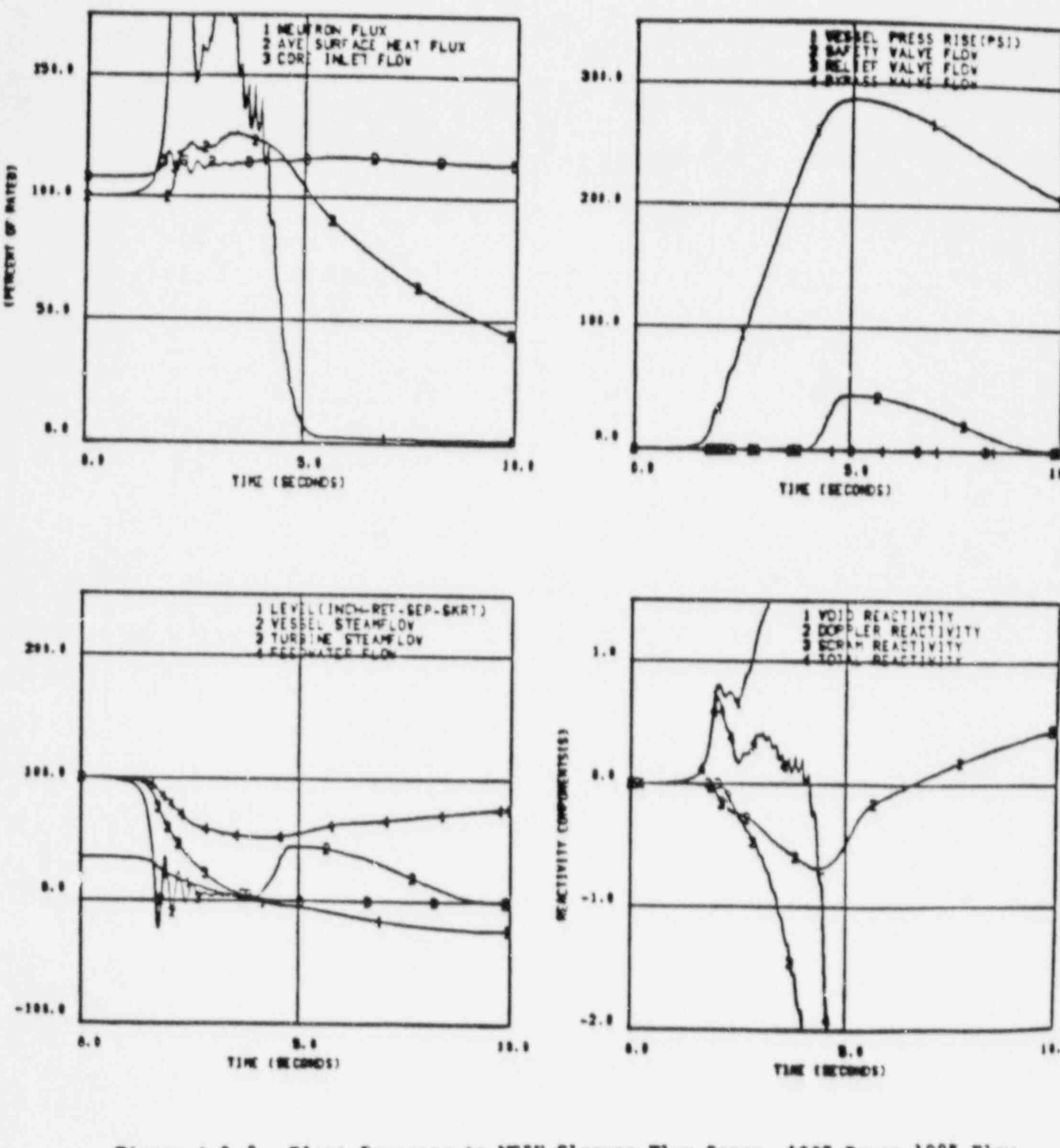

Figure A.3.8 - Plant Response to MSIV Closure Flux Scram, 100% Power 108% Flow, Rated Feedwater Temperature EOC+128 MWD/MT.

A.4 MECHANICAL EVALUATION OF REACTOR INTERNALS AND FUEL ASSEMBLY

A.4.1 LOADS EVALUATIONS

Evaluations were performed to determine the bounding acoustic and flow-induced loads, reactor internal pressure difference loads, and fuel-support loads for ICF and/or FFWTR operation.

A postulated sudden break in the recirculation line is accompanied by the propagation of a decompression wave which originates at the break and propagates back toward the vessel. Once in the vessel, the wave would broaden and lose intensity. However, it can create lateral loads on the vessel internals located opposite the recirculation suction line connections to the vessel. The pressure wave amplitude will be larger if the subcooling to the downcomer is increased and, therefore, the lateral loads could increase. The high velocity flow patterns in the downcomer resulting from a recirculation suction line break also create lateral loads on the reactor vessel internals. These loads are proportional to the square of the critical mass flow rate out of the break. The additional subcooling in the downcomer resulting from FFWTR operation can lead to an increase in the flow-induced loads. The reactor internals most impacted by acoustic and flow-induced loads are the core shroud, shroud support and jet pumps.

A reactor internals pressure difference (RIPD) analysis was performed to evaluate the effect of ICF operation on the reactor internal components loadings. The increased internal pressure differences across the reactor internals were computed for the 108% rated core flow at normal, upset and faulted conditions for the reactor internals impact evaluation.

Based on the results from plant-specific fuel lift analyses performed at 108% core flow, the resulting impact of ICF operation on the fuel-support loads and fuel bundle lift for QCNPS were evaluated. Fuel-support loads and fuel bundle lift were evaluated for upset, faulted and fatigue load combinations. It was shown that the fuel bundle lift is expected to be minimal, and the design basis vertical loads on the fuel assembly and its supports remain valid.

A.4.2 LOADS IMPACT

A.4.2.1 Reactor Internals

The reactor internals most affected by ICF and/or FFWTR operation are the core plate, shroud support, shroud, top guide, shroud head, steam dryer, control rod guide tube, control rod drive housing and jet pump. These and other components were evaluated using the bounding pressure differential loads, as calculated in Section A.4.1, under normal, upset and faulted conditions. It is concluded that the stresses produced in these and other components are within the allowable design limits given in the Final Safety Analysis Report or the ASME Code, Section III.

A.4.2.2 Fuel Assemblies

The fuel assemblies, including fuel bundles and channels, were evaluated for ICF operation considering the effects of loads discussed in Section A.4.1 under normal, upset and faulted load combinations. Results of the evaluation demonstrate that the fuel assemblies are adequate to withstand ICF effects up to 108% core flow.

The fuel channels were also evaluated under normal, upset and faulted conditions for ICF operation. The channel wall pressure gradients were found to be within the allowable design limits.

A.5 FLOW-INDUCED VIBRATION

To ensure that the flow-induced vibration response of the reactor internals is acceptable, a single reactor of each product line and size undergoes an extensive vibration test during initial plant startup. After analyzing the results of such test and assuring that all responses fall within acceptable limits of the established criteria, the reactor is classified as a valid prototype in accordance with Regulatory Guide 1.20. All other reactors of the same product line and size undergo a less rigorous confirmatory test to assure similarity to the base test.

Both QCNPS Units 1 and 2 are BWR/3 251 inch size plants. Reactor internal vibration measurements were conducted at QCNPS Unit 1. Since Unit 2 internal components are similar to Unit 1, the same approximate vibration levels are expected at Unit 2. Test data from Unit 1 were analyzed based on current GE standard design bases procedures and acceptance criteria based on a maximum allowable alternating stress intensity of 10,000 psi. The results showed that, at the rated core flow condition (98 Mlb/hr), the maximum vibration amplitude observed was 40% of the acceptance criteria. In addition, the amplitude of vibration is assumed to be proportional to the square of the flow velocity. Therefore, ICF operation at 108% of rated core flow would increase the vibration level to approximately 65% of the acceptance criteria. Therefore, it is concluded that the vibration level remains acceptable for operation at 108% core flow.

A.6 FEEDWATER NOZZLE FATIGUE ANALYSIS

An evaluation of the effect of FFWTR and end-of-cycle coastdown on feedwater nozzle fatigue was performed for the QCNPS with the following assumptions:

(1) An 18-month fuel cycle.

(2) FFWTR to 230 degrees F (equivalent to 100 degree F reduction) for 14 days was followed by a coastdown to 70% power over a period of 12 weeks. The feedwater temperature at the end of the coastdown was 210 degrees F.

A.6.1 METHODS AND ASSUMPTIONS

The fatigue experienced by the feedwater nozzle results from two phenomena: (1) system cycling and (2) rapid cycling. System cycling is caused by major temperature changes associated with system transients. Rapid cycling is caused by small, high frequency temperature fluctuations caused by mixing of relatively colder nozzle annulus water with the hot reactor water. The colder water impinging the nozzle originates from leakage past the thermal sleeve secondary seal and from the boundary layer of cold water formed by heat transfer through the thermal sleeve.

FFWTR affects only the rapid cycling fatigue usage for two reasons: (1) the transient temperature variation associated with these modes of operation is small and thus does not affect the system cycling usage factor, and (2) the time spent at a reduced feedwater temperature is a significant contributor to rapid cycling fatigue usage. An updated rapid cycling analysis performed in Reference 10 was revised to include the condition for FFWTR and coastdown to 70% power.

The feedwater duty map, (Table 3-2 in Reference 10), was modified to include the additional indices shown in Table A.6-1 for FFWTR operation with coastdown. These additional indices model the coastdown as a four-step process. The temperatures and flow rates are set for each step to give conservative results. The percentage of time spent in the FFWTR and coastdown is subtracted from the percentage spent in normal operation.

Seal ring refurbishment time is determined so that by the end of the feedwater nozzle life, the sum of the system cycling and the rapid cycling usage factor for each the feedwater nozzle locations will not exceed the allowable value of 1.0. It is assumed that the system cycling usage factor is linearly dependent on the number of years since the beginning of feedwater nozzle life. After each year, the total rapid cycling usage factor from the beginning of life is compared to the maximum allowable rapid cycling usage factor for each year, which is determined as follows:

$$U_{max} = \frac{UFMAX - SCUF}{LIFE} \cdot yrs$$

where:

U_{max} = Maximum allowable rapid cycling usage factor from beginning of life for specific year

UFMAX = Maximum allowable rapid plus system cycling usage factor by the end of the feedwater nozzle life

SCUF = Total system cycling usage factor for the feedwater nozzle life

LIFE = Feedwater nozzle design life (years)

yrs = Number of years since beginning of feedwater nozzle life

If the total rapid cycling usage factor since beginning of life exceeds U_{max} for any feedwater nozzle locations, seal ring refurbishment is assured at the end of the previous year. This method is illustrated in Figure A.6-1.

A.6.2 RESULTS

The analysis documented in Reference 10 indicated that refurbishment of the thermal sleeve seals after 11 years would be necessary to keep the 40-year total fatigue usage (system cycling plus rapid cycling) below a value of 1.0. Keeping the refurbishment schedule constant for the analysis, the 40-year total fatigue usage was calculated as shown in Table A.6-2. The fatigue damage per cycle for FFWTR operation is conservatively estimated by taking the difference between the FFWTR fatigue and the normal operation fatigue and dividing that quantity by the number of cycles in 40 years.

If FFWTR and coastdown were used for every cycle during the plant's life, the 40-year total fatigue usage factor would be greater than 1.0, assuming that the seals were replaced after 11 years. Satisfactory fatigue usage can be achieved by reducing the refurbishment interval to seven (7) years, as noted in Table A.6-2 assuming FFWTR at the end of every cycle, the refurbishment interval is impacted by four (4) years.

The results of this analysis are based on the expected coastdown operational strategry (Section A.6) and on leakage correlations developed during testing of the triple-sleeve design. A shorter end-of-cycle coastdown period or a smaller temperature reduction would increase the refurbishment interval. Also, the leakage is based on several geometric factors and assumed corrosion rates for the sleeve and safe-end materials. The resulting fatigue results are conservative for the expected plant operation mode. Since leakage is the primary contributor to rapid cycling fatigue, a more accurate evaluation of rapid cycling could be made by monitoring seal leakage and considering actual plant performance.

Table A.6-1

FEEDWATER DUTY MAP INDICES ADDED FOR FFWTR AND COASTDOWN

Cycle Index	Feedwater Flow (% rated)	Feedwater Temperature (°F)	Region A Temperature (°F)	Time Year (%)
20	100	225	546	2.56
21	100	220	546	3.84
22	92.5	215	546	3.84
23	85	210	546	3.84
24	77.5	205	546	3.84

Notes: (1) The feedwater temperature is based on a lower value of a +3% variation on the nominal temperature.

(2) The time spent at this mode of operation (2.56 + 3.84 + 3.84 + 3.84 + 3.84 = 17.92%) was subtracted from normal operation (i.e., index 1 = 65.20 - 17.92 = 47.28%).

Table A.6-2

NOZZLE FATIGUE USAGE FOR A 11-YEAR SEAL REFURBISHMENT PERIOD FOR FFWTR AND COASTDOWN (LOCATION I)

	Normal Operation	18-Month Cycle FFWTR/Coastdown to 70% Power Each Cycle
40 Year Total Fatigue Usage	0.5735	2.1843
Additional Usage Due To FFWTR and Coastdown	--	1.6108
Additional Usage Per Cycle	--	0.0600

Note: The total 40-year usage factor for FFWTR operation after every cycle can be kept to below 1.0 by refurbishing the seals after 7 years (at location D).

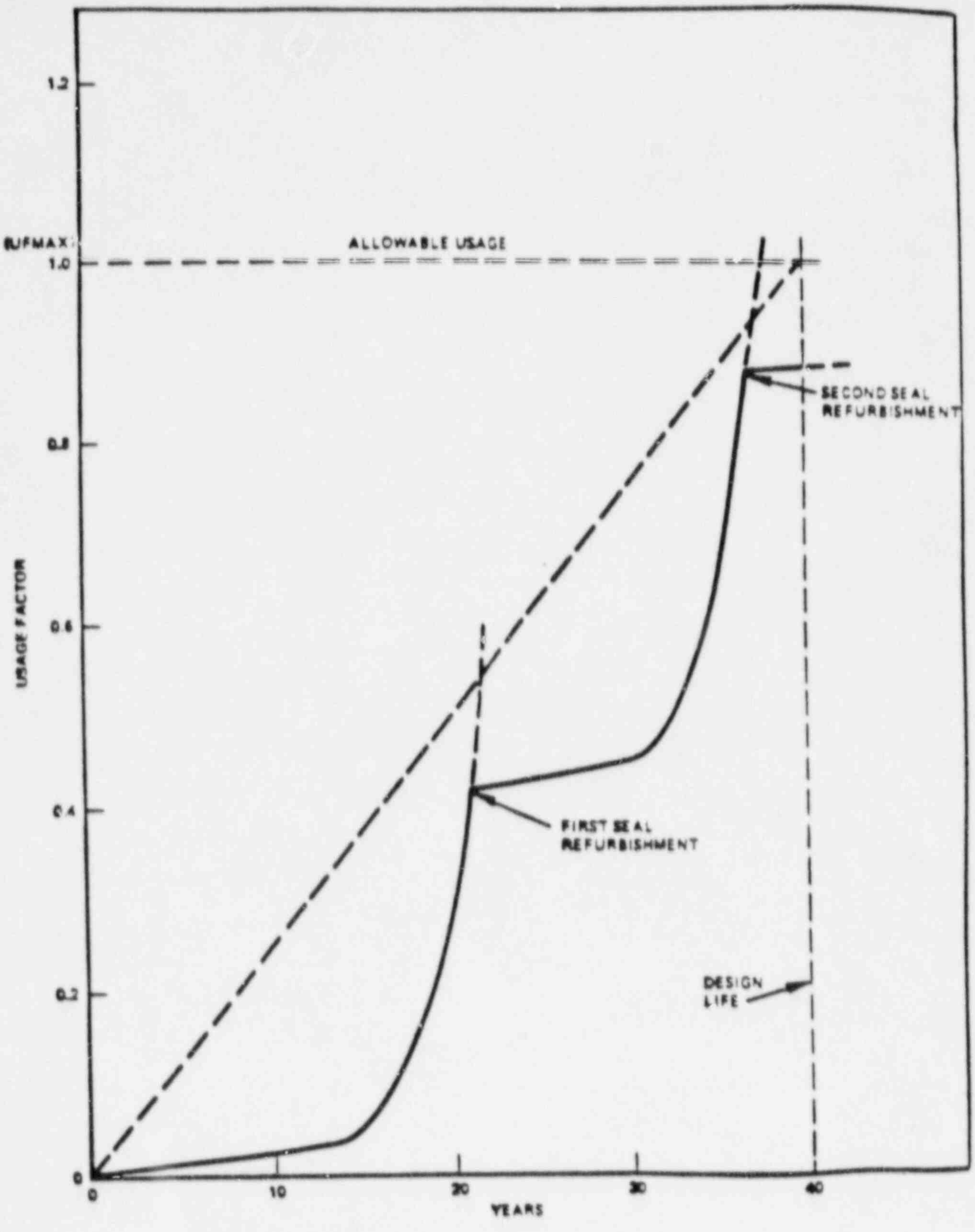

Figure A.6.1 Example of Linear Method of Determining Seal Refurbishment Intervals

A.7 CONTAINMENT ANALYSIS

The impact on the containment LOCA response was evaluated for QCNPS with regard to operation in the expanded power/flow map (i.e., including operation above the rated rod line and in the ICF region with and/or without FFWTR).

The important containment parameters considered in the analysis include:

(1) Drywell pressure and temperature
(2) Suppression chamber airspace pressure and temperature
(3) Drywell-suppression chamber differential pressure
(4) Suppression pool temperature
(5) Annulus pressurization loads
(6) Hydrodynamic loads.

Results of the analysis showed that the peak values of drywell pressure and temperature, suppression chamber airspace pressure and temperature, suppression pool temperature, and annulus pressurization loads are bounded by the values reported in the Plant-Unique Load Definition (PULD) (Reference 11). Major containment hydrodynamic loads postulated to occur in a hypothetical LOCA were evaluated and included pool swell load, condensation oscillation (CO), and chugging loads. All these dynamic loads are bounded by their corresponding design values except for the vent line thrust load.

The peak calculated value for the vent line thrust load is 21% higher than the PULD-reported value. This load represents only a part of the total maximum vent system discharge load (i.e., the vent thrust load is just one component of the maximum vent discharge load combination). From QCNPS Plant-Unique Analysis Report (Reference 12), the margin for the maximum vent system discharge load (allowable versus calculated) is estimated at 25%. The vent line thrust load contributes 19% to the total load combination. Therefore, while this vent thrust load component exceeds the PULD-value, the total vent discharge load remains well within the existing design margin.

S/RV loads on the containment are not affect(.ecause there is no change in the S/RV setpoints or reactor operating pressure associated with operation in the extended operating domain.

PART B

PLANT EQUIPMENT OUT-OF-SERVICE ANALYSIS FOR QUAD CITIES NUCLEAR POWER STATION UNITS 1 AND 2

B.1 INTRODUCTION

The purpose of this section is to present the results of a study prepared for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2 to establish the licensing bases for continued plant operation with a single failure of the following equipment:

(1) Last-stage feedwater heater string
(2) One safety/relief valve (S/RV)
(3) One recirculation pump loop

The ability to operate at full power or at a reduced power level throughout an entire or partial reactor fuel cycle with one of the above equipment out-of-service would be of significant economic value. Operational flexibility and capacity factor are increased because the plant can continue to operate until the out-of-service equipment can be repaired or until the next convenient outage occurs. Assuming only a single equipment failure, the resultant operating MCPR limits are applicable to the expanded power/flow map (Figure A.1-1) with the exception of the increased core flow region.

To establish the technical specification operating limits for each of the equipment assumed out-of-service, one or more of the following concerns need to be addressed:

(1) Core-wide transient performance
(2) Containment dynamic loads
(3) Feedwater nozzle fatigue
(4) Loss-of-Coolant accident (LOCA)

B.2 FEEDWATER HEATER OUT-OF-SERVICE

This analysis justifies operation with 100 degree F reduction in feedwater temperature in the expanded operating domain with the exception of the ICF region (Figure A.1-1). The feedwater heater out-of-service analysis supports a contingency operating mode allowing continued operation with reduced feedwater temperature over a full fuel cycle.

Operation with feedwater heater out-of-service is similar to operation with FFWTR, except that (1) the duration of operation can be longer and, (2) operation can occur at any time during the cycle. Therefore, transient analyses are performed to develop a cycle independent operating MCPR limit applicable to plant operation at the reduced feedwater temperature. In addition, the impact on other safety analyses and design bases such as containment, LOCA and feedwater nozzle fatigue is evaluated.

B.2.1 ABNORMAL TRANSIENTS EVALUATION

Operating with a feedwater heater out-of-service could potentially impact plant transient analysis as follows:

(1) The direct effect of reduced feedwater temperature is to increase the core inlet subcooling which in turn affects the core pressurization rate and reactivity during postulated transients.

(2) The potential change in core inlet conditions can affect the reactor nuclear parameters such as the power shape and core void fraction. Changes in these parameters can affect the plant responses for the transient events analyzed.

To establish cycle-independent operating limits for reactor operation with a feedwater heater out-of-service, a bounding end-of-cycle (EOC) exposure condition is used to develop nuclear input to the transient analysis model. The severity of the transient results is strongly dependent on the effectiveness of the control rod scram action. For this reason, the EOC bounding exposure condition assumes a more top-peaked axial power distribution than the nominal power shape, thus yielding a poorer scram response. Analyses with this bounding power shape result in a ΔCPR 0.04 worse than similar analyses with the nominal power shape and, therefore, should provide reasonable conservatisms for operating MCPR limits in future cycles.

The limiting transient event from Reference 5, [load rejection without bypass (LRNBP)] was analyzed with the feedwater heater out-of-service. With reduced feedwater temperature, the LRNBP will be less severe because of the reduced core steaming rate and lower initial void fraction. The feedwater controller failure (FWCF) event, although not limiting in terms of ΔCPR, has the potential to become more severe with a feedwater heater out-of-service and could become the limiting transient. Therefore, both LRNBP and FWCF were reanalyzed with the bounding power shape at 100% core power/100% core flow with a 100 degree F feedwater temperature reduction.

The results for the above transient analyses are presented in Table B.2-1 and time histories of the key parameters are shown in Figures B.2-1 through B.2-4. Table B.2-2 presents the ΔCPR results for the events analyzed. To account for plant operation in the region above the rated rod line, the above transients were also evaluated at 100% power/87% flow and found to be less limiting than the rated condition case. As expected, the LRNBP event with feedwater heater out-of-service is less limiting than the base case with feedwater heater operable. The opposite trend is observed for the FWCF event. However, with feedwater heater out-of-service, both the LRNBP and FWCF event yield similar operating limits for this operating condition.

Therefore, the operating limits associated with feedwater heater out-of-service are:

1.37 (Option A) and 1.32 (Option B)

The above limits are valid for all future cycles at QCNPS Units 1 and 2 loading current GE fuel designs provided that:

(1) The standard reload licensing LRNBP and TTNBP events result in operating limit MCPR (OLMCPR) values less than or equal to 1.37 and 1.32 for Option A and B, respectively.

(2) The standard reload licensing analysis FWCF event results in OLMCPR less than 1.34 and 1.29 for Option A and Option B, respectively. This condition is imposed to assure that the ΔCPR spread between the FWCF and the LRNBP/TTNBP observed for the bounding power shape is maintained.

These two criteria are not expected to be restrictive, since they represent conservative limits obtained with the bounding power shape. The current Cycle 10 reload licensing analysis (Reference 5) results are also included in Table B.2.2 and confirm the 0.04 CPR margin establisned in the bounding basis.

B.2.2 LOCA ANALYSIS

Operation with a feedwater heater out-of-service increases the subcooling in the downcomer and at the core inlet. This could cause an increase in blowdown flow out of a postulated break in the recirculation line during the early stages of a LOCA. This increase in subcooling and in blowdown flow out of the break can cause several small effects on the ECCS thermal-hydraulic analysis:

(1) The decay in core inlet flow could occur more rapidly because of the higher inventory loss and cause a slightly earlier fuel cladding dryout (boiling transition).

(2) The core uncovery time could change slightly because of two competing effects: more mass loss out of the break, but more mass in the core (due to lower initial core void fraction).

(3) The sensible heat in the vessel and internals could be slightly lower which could affect the overall system depressurization response.

The LOCA analysis of these effects demonstrated the insensitivity to changes in feedwater temperature of this magnitude. The net impact of void fraction and other effects results in a LOCA PCT change of less than 10 degrees F, which is insignificant when compared to the conservatism in the standard LOCA analysis.

B.2.3 FEEDWATER NOZZLE FATIGUE ANALYSIS

An evaluation of the effect of a feedwater heater out-of-service on feedwater nozzle fatigue was performed for QCNPS Units 1 and 2 with the following assumptions:

(1) An 18-month fuel cycle.

(2) Assuming a feedwater heater out-of-service for various lengths of time at the end of a fuel cycle, which causes a 100 degree F drop in the feedwater temperature.

A relationship was determined for incremental fatigue damage as a function of time spent at the lower feedwater temperature. As part of the expanded operating domain analysis (Section A.6), a feedwater nozzle fatigue study was performed for QCNPS operation with ICF/FFWTR. Both ICF/FFWTR and feedwater heater out-of-service operation involve the same physical phenomena and fatigue mechanism to the feedwater nozzle. Therefore, the methods and assumptions previously described in Section A.6.1 remain applicable to the feedwater heater out-of-service condition. Also, the basis for the results of this analysis are identical to the ICF/FFWTR analysis described in Section A.6.2.

Table B.2-3 shows the modification of the indices for feedwater heater out-of-service operation. The percentage of time spent in the feedwater heater out-of-service mode was subtracted from the percentage of time spent in normal operation. A relationship of incremental fatigue damage as a function of time spent in the feedwater heater out-of-service mode was developed from these results.

The analysis documented in Reference 10 indicated that refurbishment of the thermal sleeve seals after 11 years is necessary to keep the 40-year total fatigue usage (system cycling plus rapid cycling) below a value of 1.0. Keeping the refurbishment schedule constant for the analysis, the 40-year total fatigue usage was calculated as shown in Table B.2-4. The fatigue damage per cycle for FFWTR operation is conservatively estimated by taking the difference between the FFWTR fatigue and the normal operation fatigue and dividing that quantity by the number of cycles in 40 years.

If operation with a feedwater heater out-of-service were implemented for every cycle during the plant's life, the 40-year total fatigue usage factor would be greater than 1.0, assuming that the seals were replaced after every 11 years. Satisfactory fatigue usage can be achieved by reducing the refurbishment interval to 8 years, as noted in Table B.2-4. The impact on seal refurbishment was less severe for this mode of operation than for ICF/FFWTR.

B.2.4 CONTAINMENT LOADS EVALUATION

The containment analysis results for ICF with FFWTR (Section A.7) are applicable to the feedwater heater out-of-service analysis because the resulting core subcooling increase following ICF and FFWTR bounds the case for a feedwater heater out-of-service. Given that FFWTR and feedwater heater out-of-service both result in a temperature reduction of 100 degrees F, the addition of ICF will increase the core inlet subcooling and yield conservative results if applied to the feedwater heater out-of-service analysis.

Therefore, the containment evaluation performed to support ICF with FFWTR is applicable to this feedwater heater out-of-service analysis.

Table B.2-1

TRANSIENT ANALYSIS RESULTS FOR QCNPS AT 100P/100F
FEEDWATER HEATER OUT-OF-SERVICE

Transient Description	Maximum Neutron Flux (% NBR)	Maximum Core Ave. Surface Heat Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pres. (psig)
LRNBP w/ FWH[a]	529.3	121.7	1193	1223
LRNBP w/o FWH	404.5	117.9	1174	1204
FWCF w/ FWH	267.8	118.0	1109	1141
FWCF w/o FWH	224.0	120.7	1086	1117

[a] FWH: feedwater heater

Table B.2-2

OPERATING MCPR RESULTS FOR QCNPS AT 100P/100F
FEEDWATER HEATER OUT-OF-SERVICE

Transient Description	Exposure 1 = Bounding 2 = EOC10	ΔCPR[a]	$OLMCPR_A$	$OLMCPR_B$
LRNBP W/FWH[b]	1	0.23	1.37	1.32
FWCF W/FWH	1	0.18	1.31	1.26
LRNBP W/O FWH	1	0.21	1.34	1.29
FWCF W/O FWH	1	0.21	1.34	1.29
LRNBP W/FWH	2	0.19	1.33	1.28
FWCF W/ FWH	2	0.14	1.27	1.22

a Uncorrected for Option A and Option B.

b FWH: feedwater heater.

Table B.2-4

NOZZLE FATIGUE USAGE FOR A 11-YEAR SEAL REFURBISHMENT PERIOD FOR FEEDWATER HEATER OUT-OF-SERVICE (LOCATION I)

	Normal Operation	18 Month Cycle FWHOOS Operation for 1314 Hours Each Cycle (10% per year)
40-Year Total Fatigue Usage	0.5735	1.5305
Additional Usage Due to FWHOOS	--	0.9570
Additional Usage Per Cycle	--	0.0360

Note: The total 40-year usage factor for 1314 hours of feedwater heater out-of-service operation during every cycle can be kept below 1.0 by refurbishing the seals after 8 years (at location D).

Table B.2-3

FEEDWATER DUTY MAP INDICES ADDED FOR FEEDWATER HEATER OUT-OF-SERVICE

Cycle Index	Feedwater Flow (% rated)	Feedwater Temperature (°F)	Region A Temperature (°F)	Time Year (%)
20	100	225	546	10

Notes: (1) The feedwater temperature is based on a lower value of a +3% variation on the nominal temperature.
(2) One case was arbitrarily analyzed: 10% time per year = 1314 hours/cycle.
(3) The time spent at this mode of operation (10%) was subtracted from normal operation (i.e., index 1 = 65.20 - 10 = 55.20).

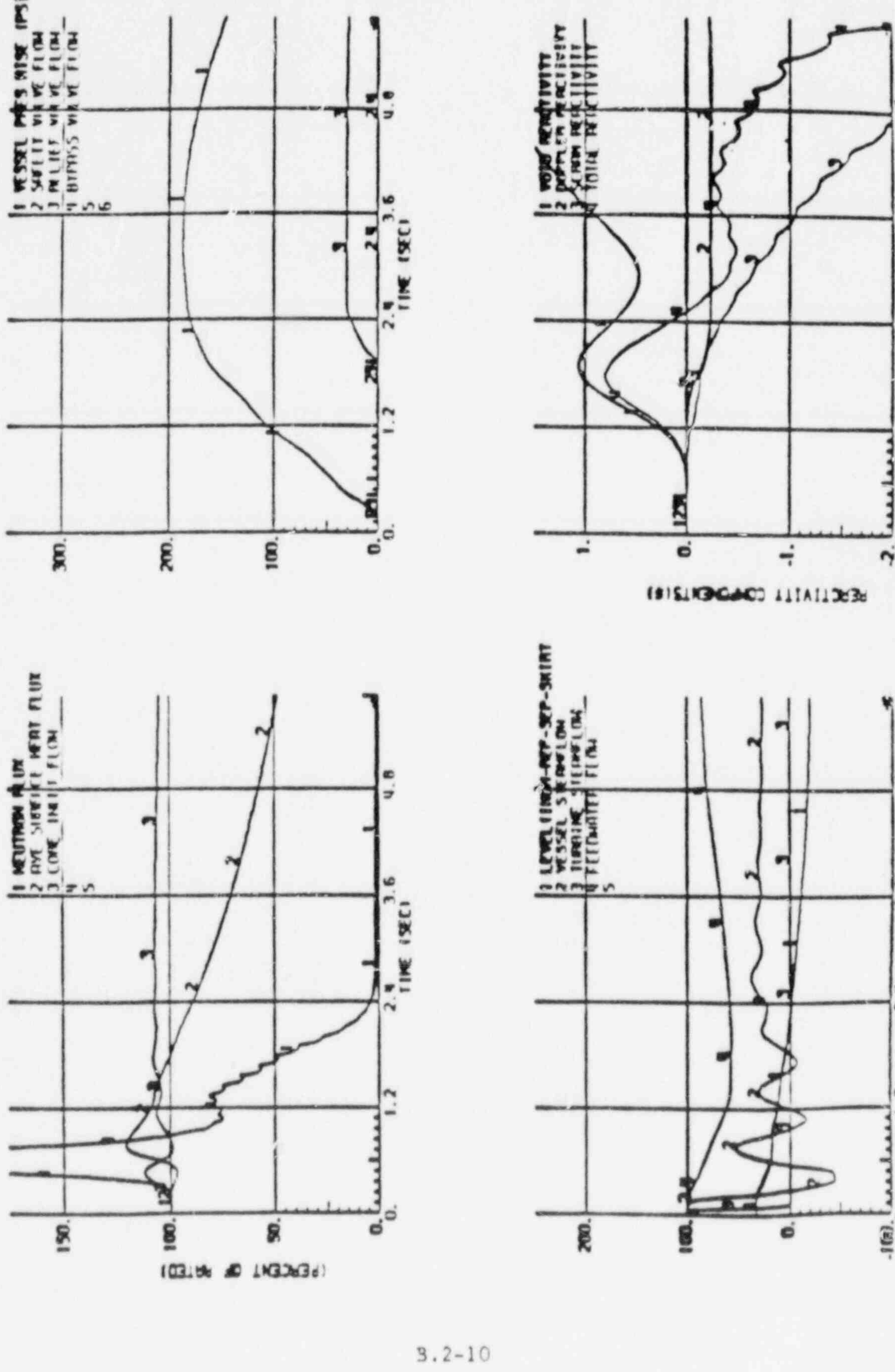

Figure B.2.1 - Plant Response to Load Rejection without Bypass, Feedwater Heater In-Service (Bounding Power Shape)

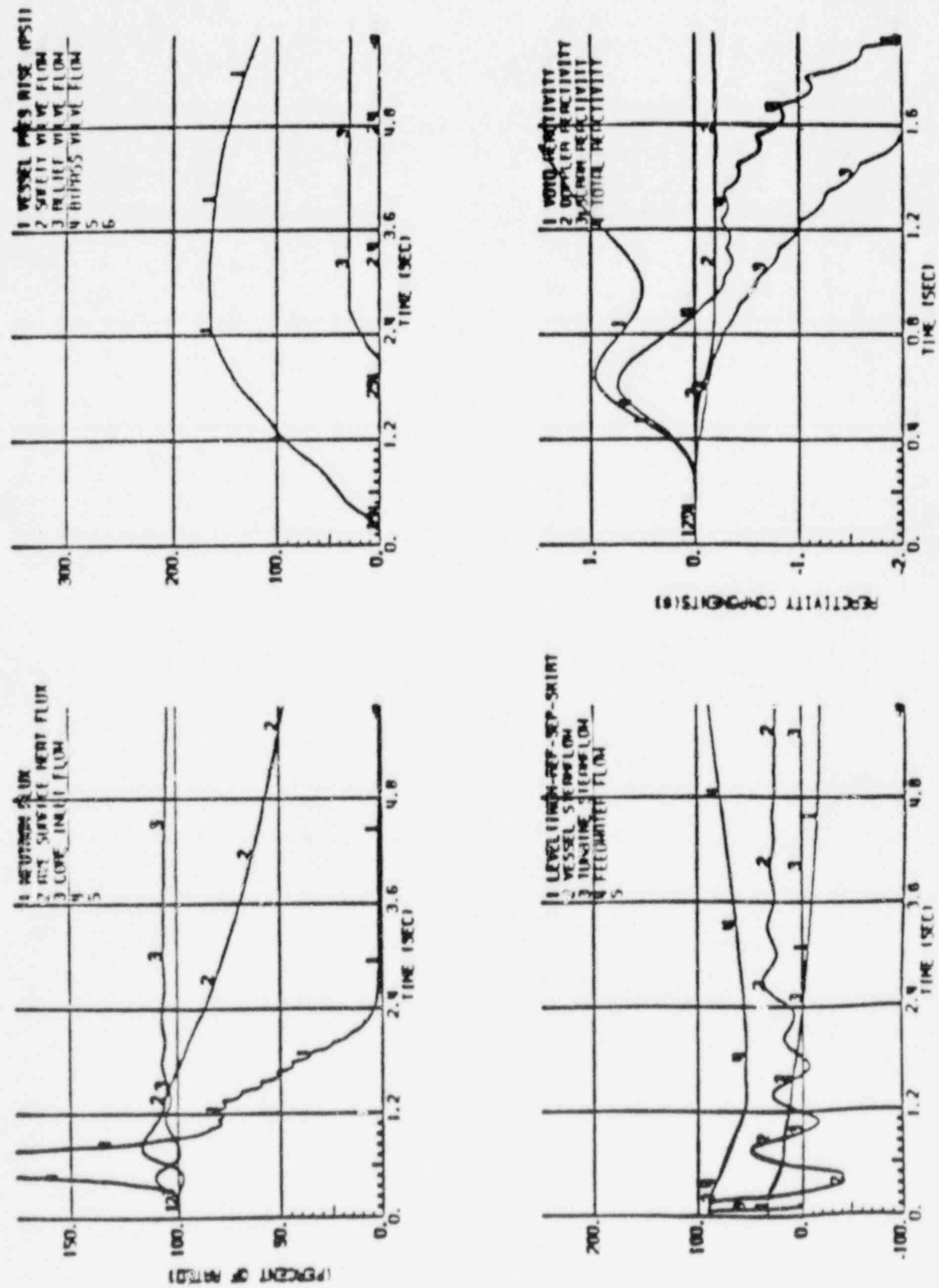

Figure B.2.2 - Plant Response to Load Rejection without Bypass, Feedwater Heater Out-of-Service (Bounding Power Shape)

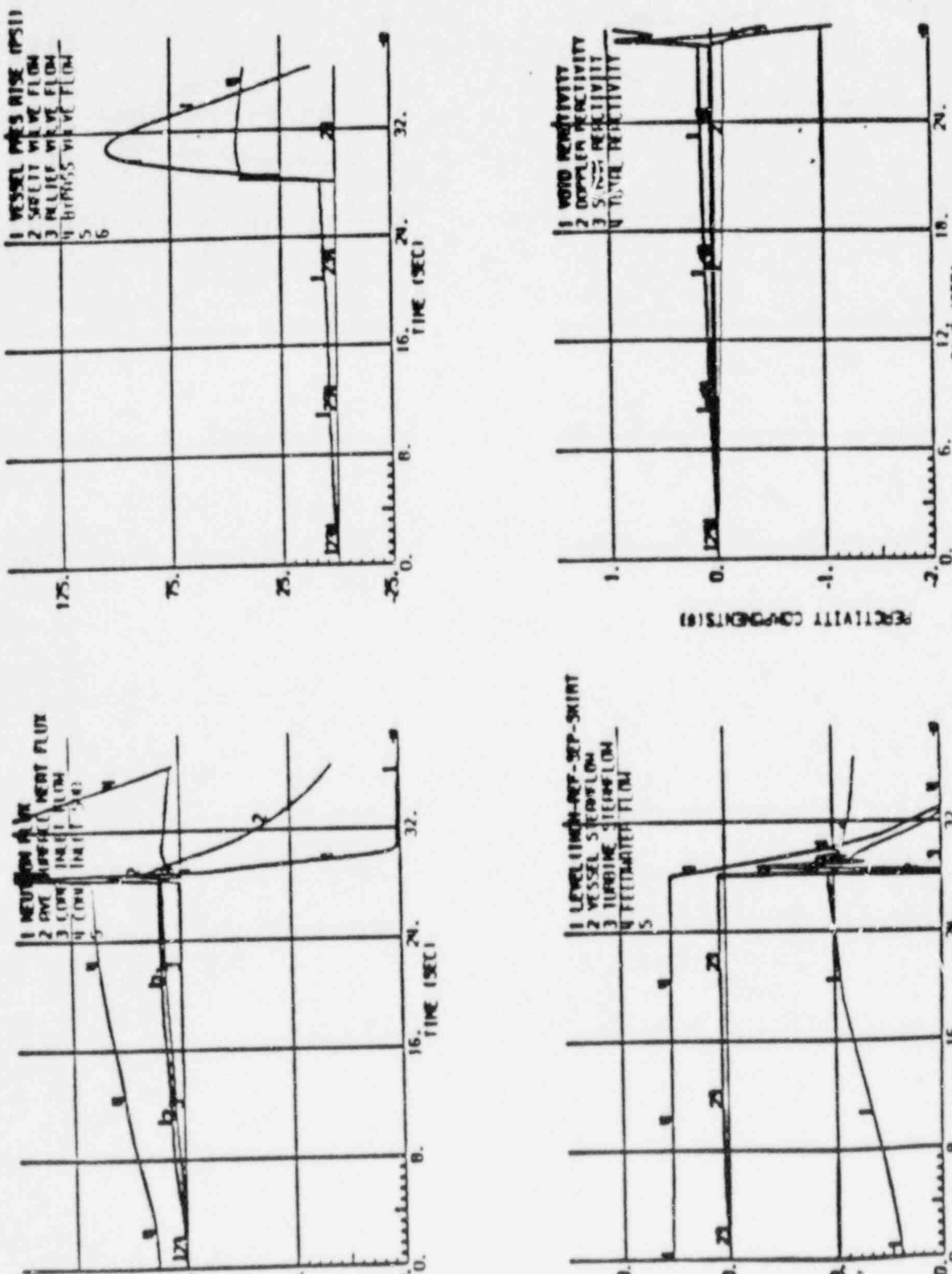

Figure B.2.3 - Plant Response to Feedwater Controller Failure, Feedwater Heater In-Service (Bounding Power Shape)

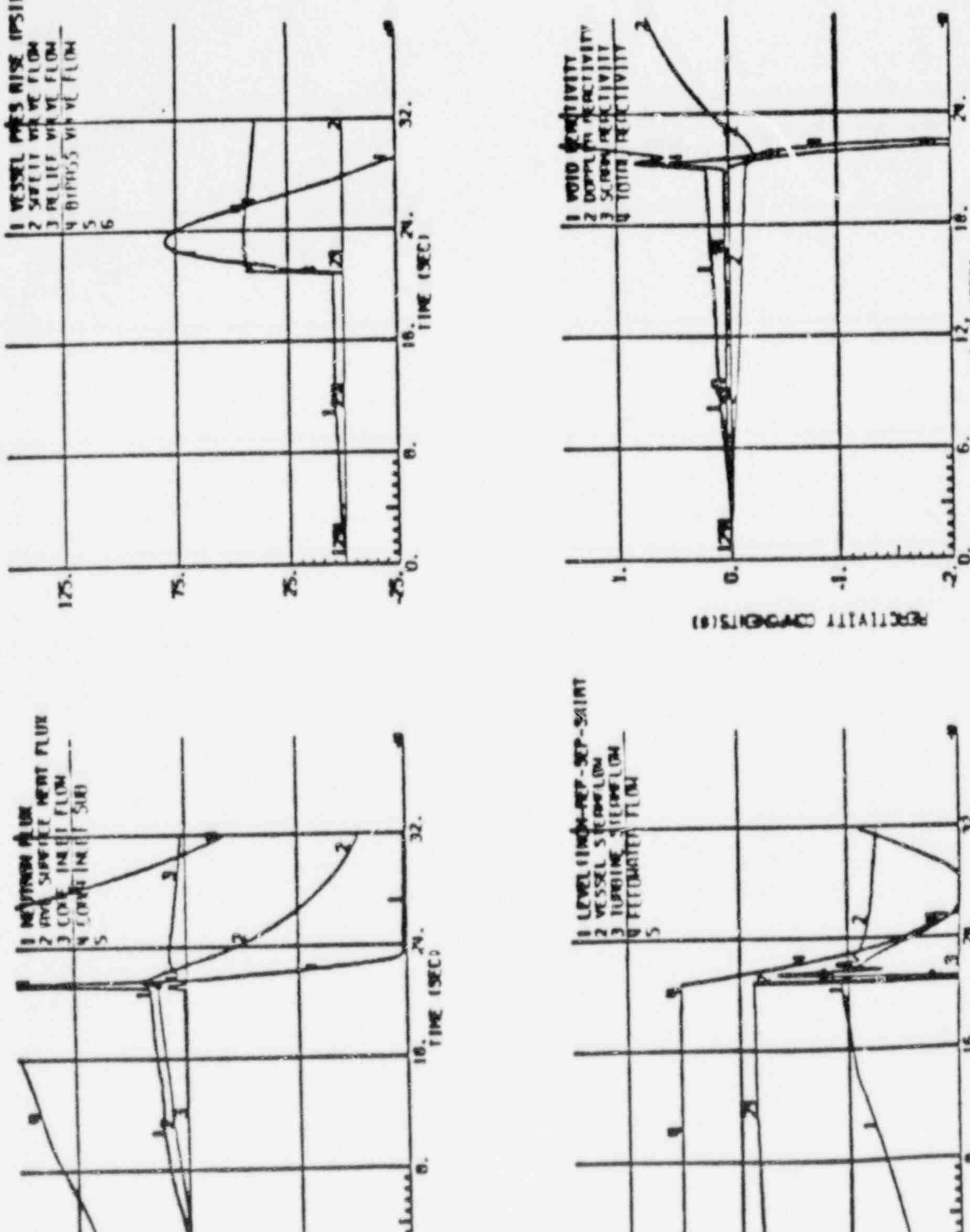

Figure B.2.4 - Plant Response to Feedwater Controller Failure, Feedwater Heater Out-of-Service (Bounding Power Shape)

B.3 ONE SAFETY/RELIEF VALVE OUT-OF-SERVICE

This analysis provides the technical bases for operation of QCNPS Units 1 and 2 with one safety/relief valve (S/RV) out-of-service. In particular, the accident and transient considerations for operation with one S/RV out-of-service are presented.

B.3.1 ABNORMAL TRANSIENT EVALUATIONS

Operation of QCNPS Units 1 and 2 with one S/RV out-of-service could affect the change in critical power ratio (ΔCPR) in the event of an abnormal operating transient. The decrease in relief capacity could lead to higher pressures during a pressurization event, which could lead to a larger ΔCPR. The failure of one S/RV could also result in a higher peak vessel pressure, thereby reducing the margin to the ASME upset code limit for a pressure vessel.

The transients which yields the most limiting ΔCPR for QCNPS Unit 1 Cycle 10 is the load rejection without bypass (LRNBP). This event was reanalyzed with the most limiting relief valve disabled (i.e., the relief valve with the lowest setpoint). The valve setpoints used in this analysis are given in Table B.3-1. For the overpressure criteria, the main steamline isolation valve (MSIV) closure transient with high flux scram was analyzed with the lowest setpoint spring safety valve out-of-service.

B.3.1.1 Impact on Delta CPR Analysis

The LRNBP transient was analyzed using the ODYN computer program with full relief capacity (as a base case) and with the lowest setpoint relief valve out-of-service. The results showed no change in the ΔCPR due to the reduced relief capacity. Plots of typical transient responses are shown in Figures B.3-1 and B.3-2.

From the transient responses, it can be seen that the peak neutron flux occurs about 0.7 second before the relief valves open for both analyzed cases. Because QCNPS has two relief valves with the same low setpoint, disabling one relief valve affects only the pressure relief capacity and not the time of valve initiation. Because the neutron flux is decreasing rapidly at the time when the relief valves open, a change in the overall relief capacity will not affect the CPR result.

In summary, with one relief valve out-of-service there is negligible impact on the MCPR limit. The ΔCPR for this operating condition will be bounded by reload licensing calculations. This conclusion is valid for current General Electric fuel types and analysis methods as applied to QCNPS Units 1 and 2.

B.3.1.2 Impact on Overpressure Criteria

Reference 9 documents the results of GE sensitivity studies which show the effect of a S/RV out-of-service is a peak pressure increase of less than 20 psi.

The adequacy of the S/RV capacity based on ASME code requirements is demonstrated by the MSIV closure transient with high flux scram and without credit for relief valve operation. With the lowest setpoint spring safety valve out-of-service, this transient event still shows an adequate margin of 54 psi to the ASME upset code limit of 1375 psig. The time response of key variables for this transient is shown in Figure B.3-3.

B.3.2 LOCA ANALYSIS

If the out-of-service valve has an automatic depressurization function (ADS), there can be a potential impact on the calculated peak cladding temperature (PCT) for small break sizes of less than approximately 0.2 ft^2. With a worst case postulated single failure of the High Pressure Core Injection (HPCI) System, a small effect may be seen because the small break transient is dominated by the time required to depressurize the reactor to the operating pressure of the low pressure Emergency Core Cooling System (ECCS).

The effect of one ADS valve out-of-service was accounted for in the Reference 8 LOCA analysis because only four of the five ADS valves were used for the break spectrum analysis. The results showed much lower PCT values for small breaks than for the large breaks, which are the most limiting LOCA cases for this plant. For the large breaks cases, the reactor vessel is rapidly depressurized prior to the actuation of the ADS. Therefore, one ADS valve out-of-service has no impact on the calculated PCT.

Table B.3-1

VALVE SETPOINTS USED FOR TRANSIENT ANALYSIS

Setpoint (psig)	Type	No.
1105 + 1%	RV	1*
1125 + 1%	RV	2
1125 + 1%	S/RV	1

* One S/RV assumed out-of-service

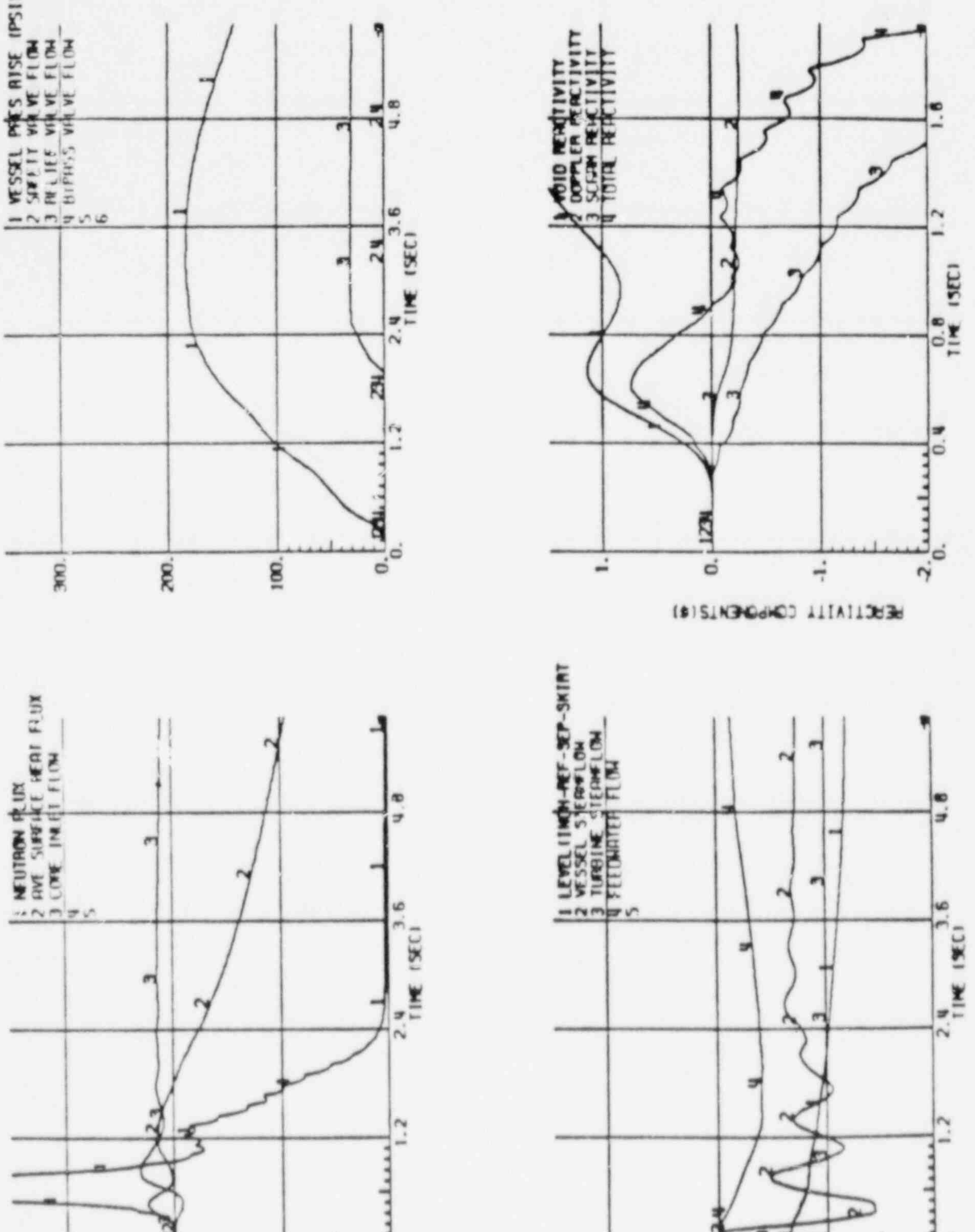

Figure B.3.1 - Plant Response to Load Rejection without Bypass, Full Relief Capacity

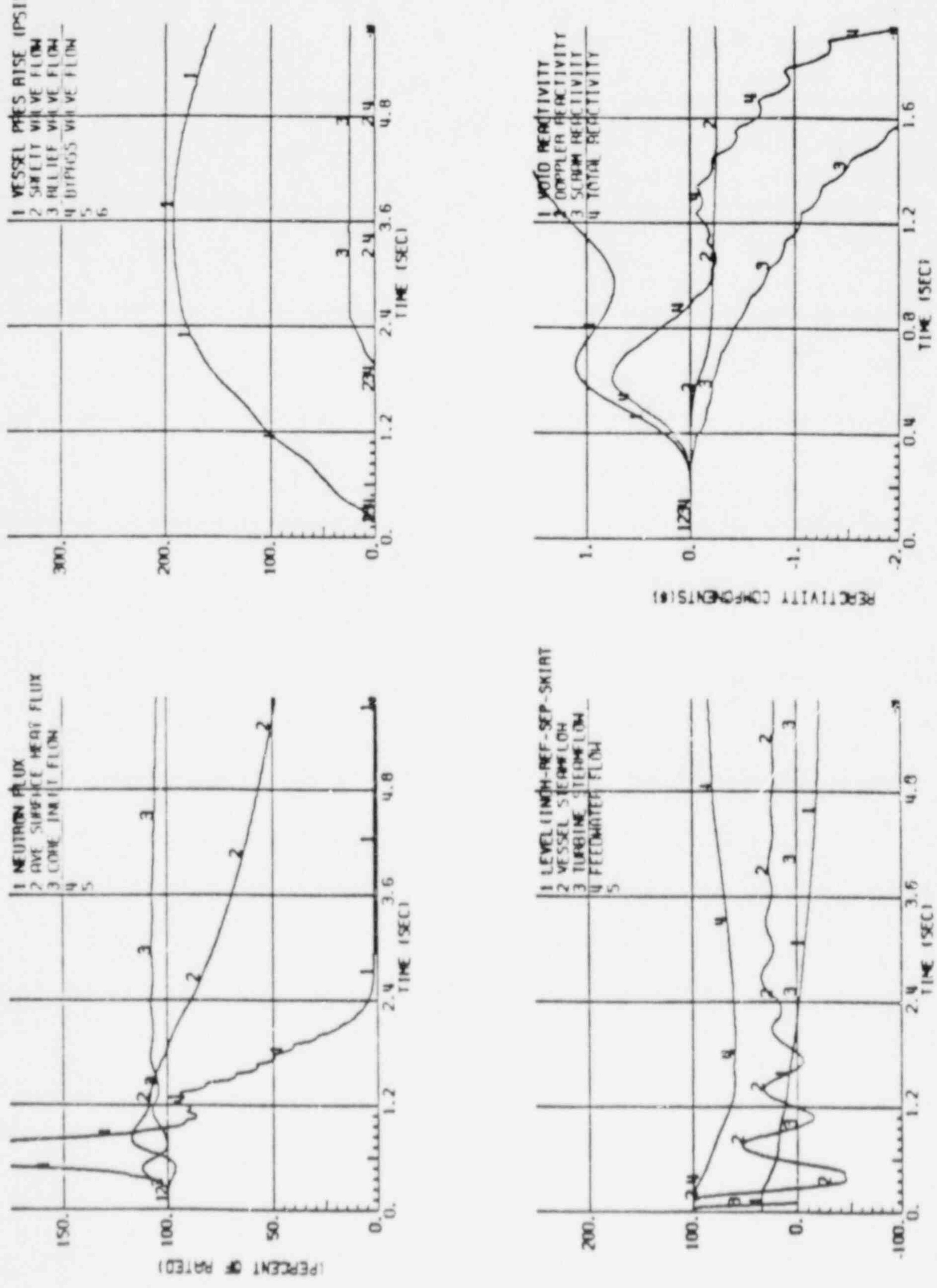

Figure B.3-2 Plant Response to Load Rejection Without Bypass, One S/RV Out-of-Service

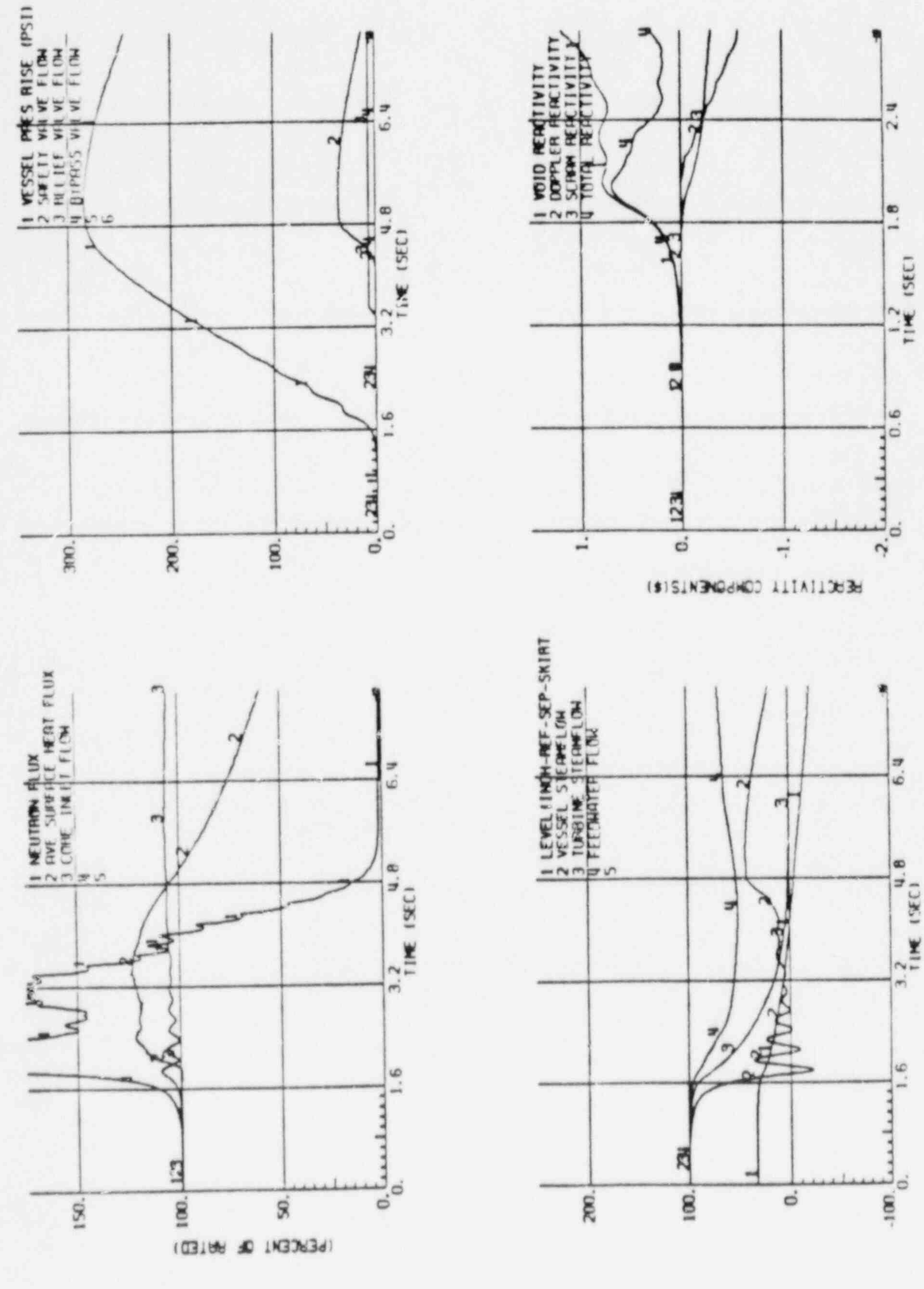

Figure B.3-3 Plant Response to MSIV Flux Scram, One S/RV Out-of-Service

B.4 ONE RECIRCULATION PUMP OUT-OF-SERVICE

From a plant availability/outage planning standpoint, the capability of operating at reduced power with a single recirculation loop is highly desirable in the event that maintenance of a recirculation pump or other components render one loop inoperable. To justify the single-loop operation (SLO), accidents and abnormal operational transients associated with power operation were reviewed for the single loop case with one pump in operation. This SLO analysis was previously performed for QCNPS Units 1 and 2 and documented in Reference 2.

To support the introduction of the GE8x8EB fuel design and the additional operating domain above the rated rod line (Figure A.1-1), the issues addressed by the referenced SLO analysis are reviewed to ensure their applicabilities with these operational changes. In addition, the impact on safety limits for SLO in the region above the rated rod line with one safety/relief valve (S/RV) out-of-service is also addressed here.

B.4.1 MINIMUM CPR FUEL CLADDING INTEGRITY SAFETY LIMIT

Except for the total core flow and the traversing in-core probe (TIP) measurements, the uncertainties used in the statistical analyses to determine the fuel cladding integrity MCPR safety limit do not depend on whether coolant flow is provided by one or two recirculation pumps.

Since the core flow uncertainty and the TIP noise uncertainty are not affected by the proposed operational changes (i.e., the GE8x8EB fuel design and operation above the rated rod line with or without one S/RV out-of-service), the conclusions shown in the referenced SLO analysis are still applicable.

B.4.2 MINIMUM CPR OPERATING LIMIT

The referenced SLO analysis (Reference 2, Paragraph 6.B.3) demonstrated that, within the normal operating domain, the consequences of abnormal operation transients from one-loop operation will be considerably less severe than those analyzed for a two-loop operation mode. Operation with one recirculation loop results in a maximum power output significantly below (by 20 to 30%) that which is attainable with a two-pump operation. Thus, for pressurization, flow decrease and cold water increase transients, the results for two-pump operation cases bound both the thermal and overpressure consequences of one-loop operation. The introduction of GE8x8EB fuel in the core is not expected to alter the above conclusion. The observed transient performance trend (one-pump case bounded by two-pump case) remains applicable to the QCNPS cores with the new GE8x8EB fuel design.

The failure of the S/RV with the lowest setpoint for two-pump operation was previously shown to have no impact on the MCPR operating limits and the vessel overpressure criteria (Section B.3). For SLO, the same conclusion remains applicable because the peak neutron flux would still occur before any S/RV actuation. In addition, the lower initial power level for SLO mode would reduce the severity of the vessel peak pressure compared with the two-pump case.

The above conclusions are also applicable for plant operation in the region above the rated rod line.

B.4.3 STABILITY ANALYSIS

The introduction of GE8x8EB fuel design to QCNPS Units 1 and 2 cores will result in an insignificant impact on the core and channel decay ratio for reactor operation with one recirculation loop. Therefore, the conclusions stated in the Reference 2 regarding this subject are still applicable to QCNPS Units 1 and 2.

B.4.4 LOCA ANALYSES

The LOCA analysis documented in Reference 2, Paragraph 6.B.6, imposed a MAPLHGR reduction factor of 0.84 to GE 8x8 retrofit fuel type.

Based on analysis experiences in using the SAFER/GESTR LOCA evaluation models, the GE8x8EB fuel design has been shown to have larger margins to the PCT limit than the 8x8R and BP/P8x8R fuel types. This is primarily due to the decrease in initial stored energy of the GE8x8EB fuel attributable to the increased initial pressurization level. The Reference 8 analysis concluded, with SAFER/GESTR, no MAPLHGR multiplier is required for SLO at QCNPS Units 1 and 2.

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