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Class I
August 1980

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

BROWNS FERRY NUCLEAR POWER STATION

UNIT 3 RELOAD NO. 3

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GENERAL ELECTRIC

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1. PLANT-UNIQUE ITEMS (1.0)*

Items different from or not included in Reference 1:

Data for Section 4 provided by Tennessee Valley Authority (TVA):	Appendix A
Fuel Loading Error LHGR:	Appendix B
Safety/Relief Valve Capacity:	Appendix B
Spring Safety Valve Capacity:	Appendix B
Rated Steam Flow:	Appendix B
GETAB Analysis Initial Conditions:	Appendix B
New Bundle Loading Error Event Analysis Procedures:	Reference 3
Margin to Spring Safety Valves:	Appendix C

2. RELOAD FUEL BUNDLES (1.0, 3.3.1 and 4.0)

Fuel Type		Number	Number Drilled
Initial Core	8DB219	288	288
Reload 1	8DRB265L	208	208
Reload 2	P8DRB265L	144	144
New	P8DRB265L	124	124
	TOTAL	764	764

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	14,297 MWd/t
Assumed reload cycle core average exposure at end of cycle:	15,105 MWd/t
Core loading pattern:	Figure 1

*( ) Refers to areas of discussion in Reference 1.

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

See Appendix A for this data provided by The Tennessee Valley Authority.

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.04

6. RELOAD-UNIQUE TRANSIENT ANALYSIS INPUTS (3.3.2.1.5 AND 5.2)

Void Coefficient N/A* (¢/% Rg)	-6.97/-8.71
Void Fraction (%)	40.29
Doppler Coefficient N/A (¢/°F)	-0.228/-0.217
Average Fuel Temperature (°F)	1343
Scram Worth N/A ($)	-37.67/-30.13
Scram Reactivity vs Time	Figure 2

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

	8x8	8x8R	P8x8R
Exposure	EOC 4	EOC 4	EOC 4
Peaking factors (local, radial and axial)	1.22 1.42 1.40	1.20 1.55 1.40	1.20 1.55 1.40
R-Factor	1.098	1.051	1.051
Bundle Power (MWt)	5.987	6.550	6.526
Bundle Flow (10^3 lb/hr)	108.2	108.5	109.2
Initial MCPR	1.24	1.25	1.25

*N = Nuclear Input Data
A = Used in Transient Analysis

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Recirculation Pump Trip

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.2)

Transient	Exposure	Power (%)	Core Flow (%)	ϕ (% NBR)	Q/A (% NBR)	P_{sl} (psig)	P_v (psig)	ΔCPR 8x8	ΔCPR 8x8R	ΔCPR P8x8R	Plant Response
Load Rejection Without Bypass	BOC4-EOC4	104.5	100	239	111	1226+	1250	0.17	0.18	0.18	Figure 3
Loss of 100°F Feedwater Heating	--	104.5	100	12[illegible]	123	1013	1069	0.15	0.15	0.15	Figure 4
Feedwater Controller Failure	BOC4-EOC4	104.5	100	164	112	1155	1189	0.12	0.12	0.12	Figure 5

10. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (5.2.1)

Rod Block Reading	Rod Position (Feet Withdrawn)	ΔCPR** 8x8	ΔCPR** 8x8R/ P8x8R	MLHGR*** 8x8	MLHGR*** 8x8R/ P8x8R	Limiting Rod Pattern
104	3.5	0.10	0.10	14.8	16.2	Figure 6
105	4.0	0.12	0.11	15.2	16.5	Figure 6
106*	4.5	0.14	0.12	15.2	16.6	Figure 6
107	4.5	0.14	0.12	15.2	16.6	Figure 6
108	5.5	0.18	0.14	15.2	16.7	Figure 6
109	6.5	0.20	0.16	15.2	16.7	Figure 6
110	7.5	0.20	0.17	1[illegible].2	16.7	Figure 6

*Indicates setpoint selected.

**The initial MCPR (1.24) for the 8x8R and P8x8R fuel was 0.01 less than the operating limit MCPR (1.25). This is discussed on pp. B-114 and B-115 of Reference 1.

***A 2.2% peaking penalty for densification is included.

+Less than 25 psi margin to spring safety valves. One safety/relief valve is assumed out of service. See Appendices B and C.

11. OPERATING MCPR LIMIT (5.2)

BOC4 to EOC4

1.24	8x8 Fuel
1.25	8x8R Fuel
1.25	P8x8R Fuel

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	Power (%)	Core Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	104.5	100	1265	1299	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power): 0.85

Channel Hydrodynamic Performance

	Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power)
8x8R/P8x8R	0.29
8x8	0.36

14. LOSS-OF-COOLANT ACCIDENT RESULTS (5.5.2)

Reference 2.

15. LOADING ERROR RESULTS (5.5.4)

Limiting Event: Rotated Bundle P8DRB265L

MCPR: 1.08

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Doppler Reactivity Coefficient: Figure 9

Accident Reactivity Shape Functions: Figures 10 and 11

Scram Reactivity Functions: Figures 12 and 13

Plant specific analysis results

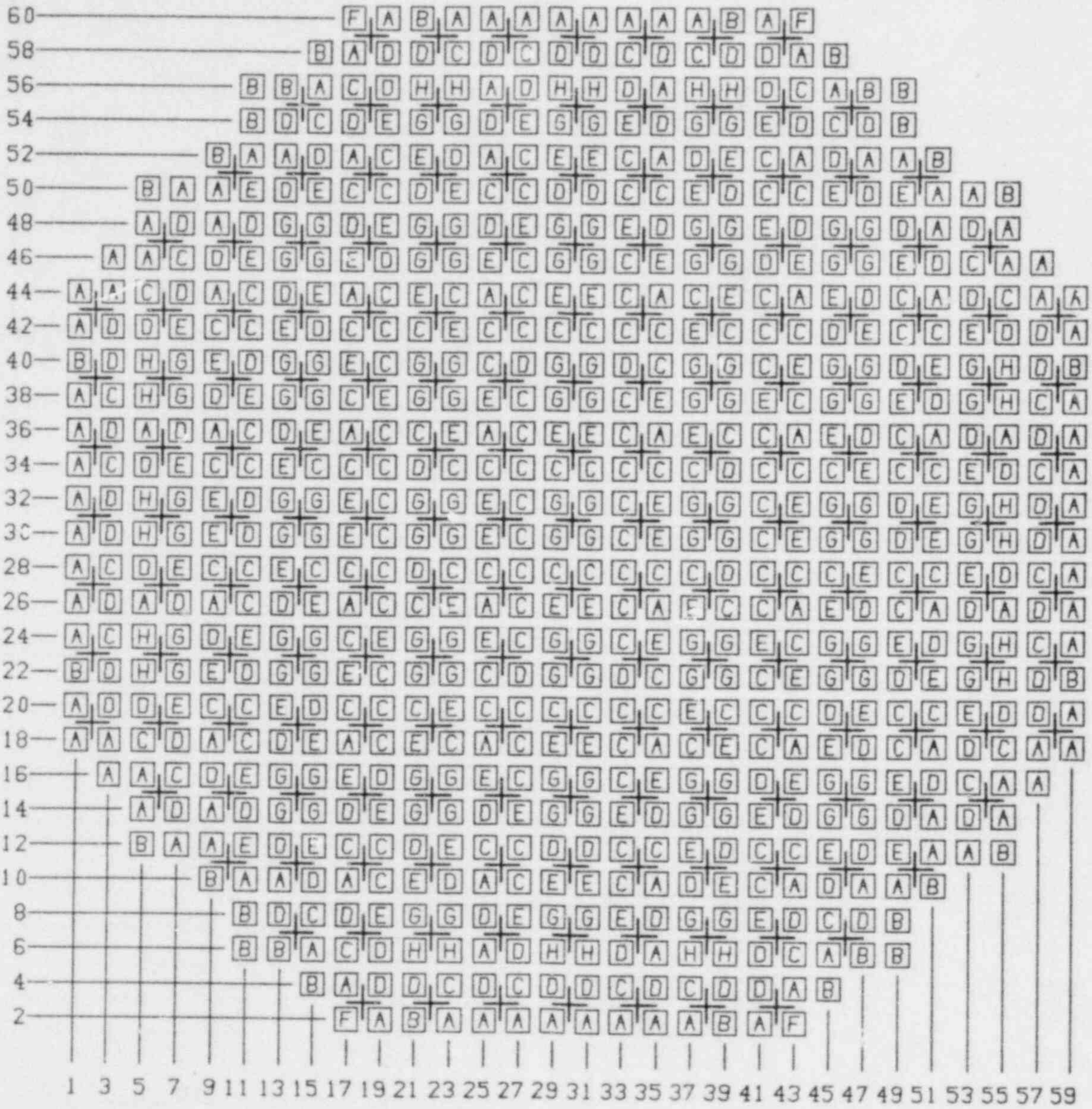
Parameter not bounded: None

REFERENCES

1. General Electric Boiling Water Generic Reload Fuel Application, NEDE-24011-P-A, August 1979.

2. Loss-Of-Coolant Accident Analysis Report for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A, July 1979.

3. Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 3 Reload 1, NEDO-24128 (Appendix A), June 1978.



FUEL TYPE	
A = 8DB219 ,IC*	E = P8DRB265L,R3
B = 8DB219 ,IC	F = 8DB219 ,IC
C = 8DRB265L ,R1	G = 8DB219 ,IC
D = P8DRB265L,R2	H = 8DRB265L ,R1

*IC = INITIAL CORE, RI = RELOAD 1, R2 = RELOAD 2, etc.

Figure 1. Reference Core Loading Pattern

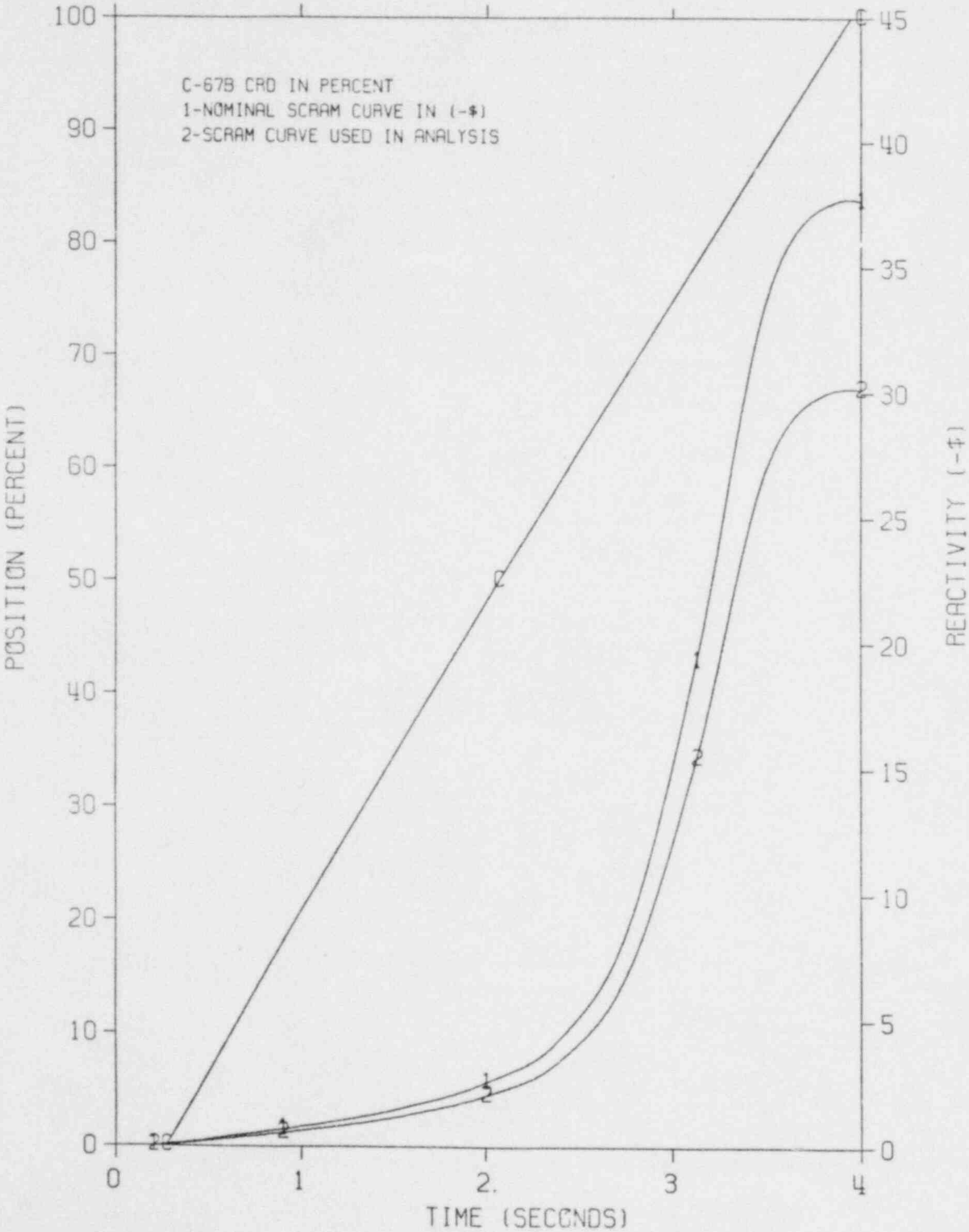


Figure 2. Scram Reactivity and Control Rod Drive Specifications

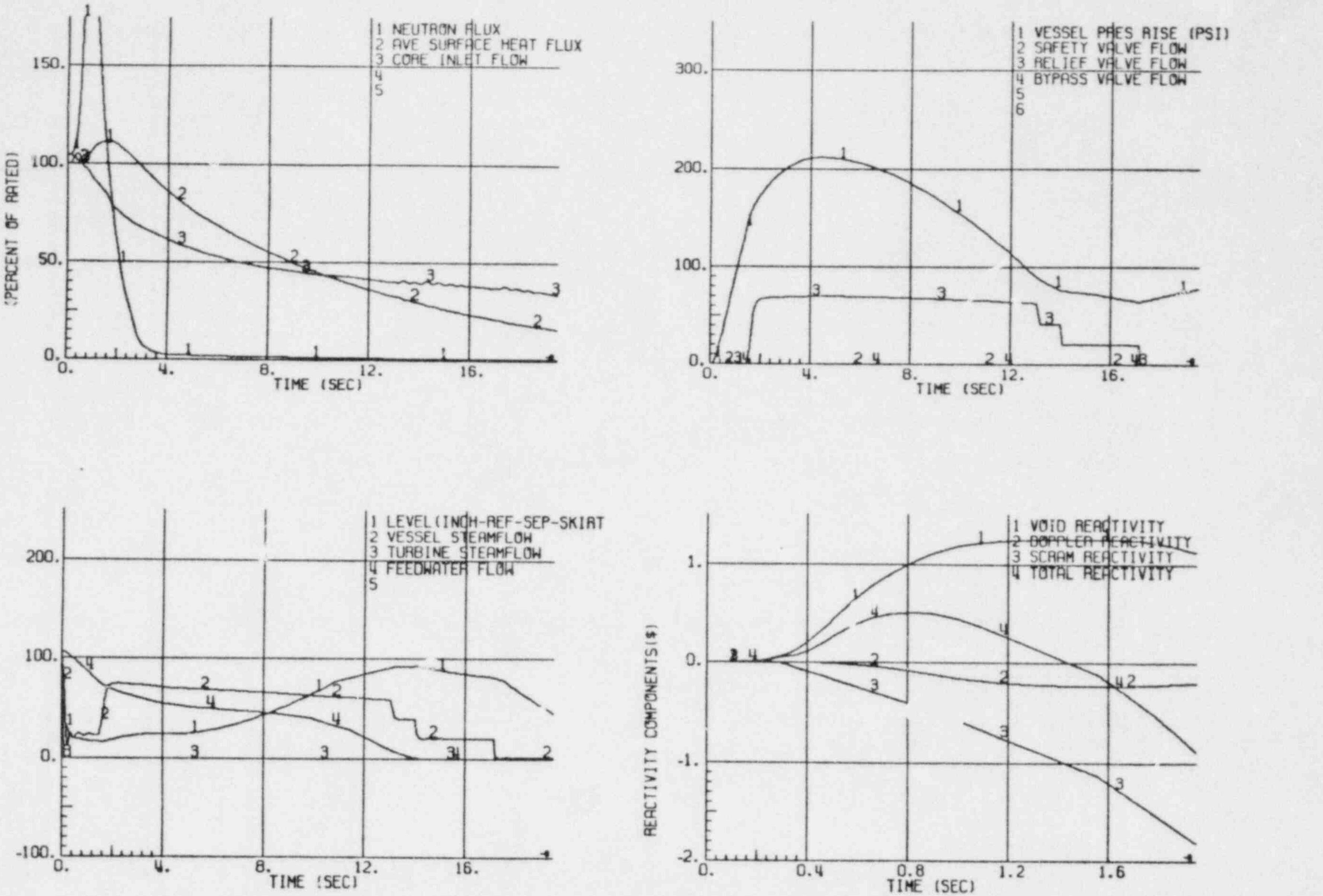


Figure 3. Plant Response to Generator Load Rejection Without Bypass

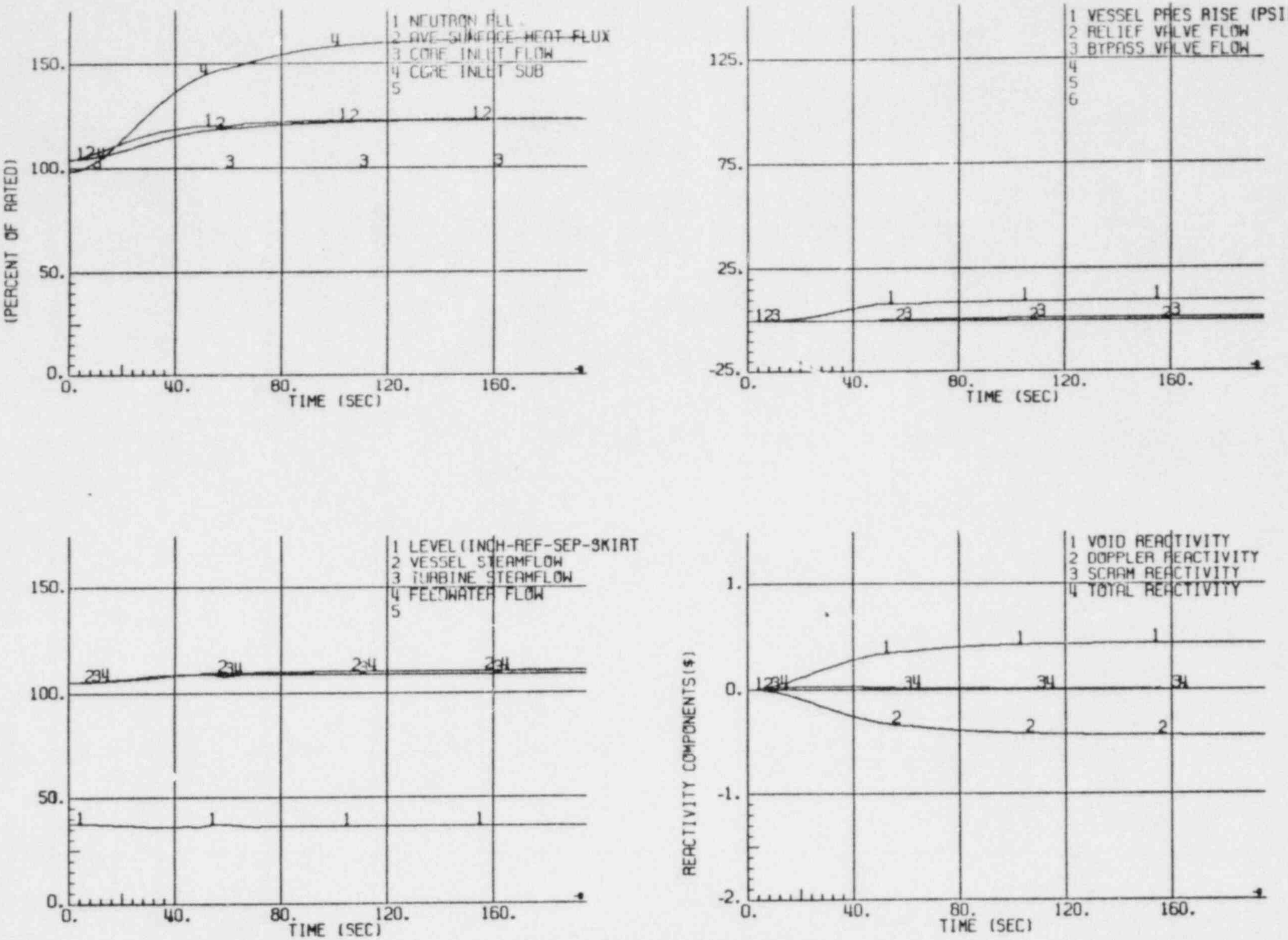


Figure 4. Plant Response to Loss of 100 Deg F Feedwater Heating

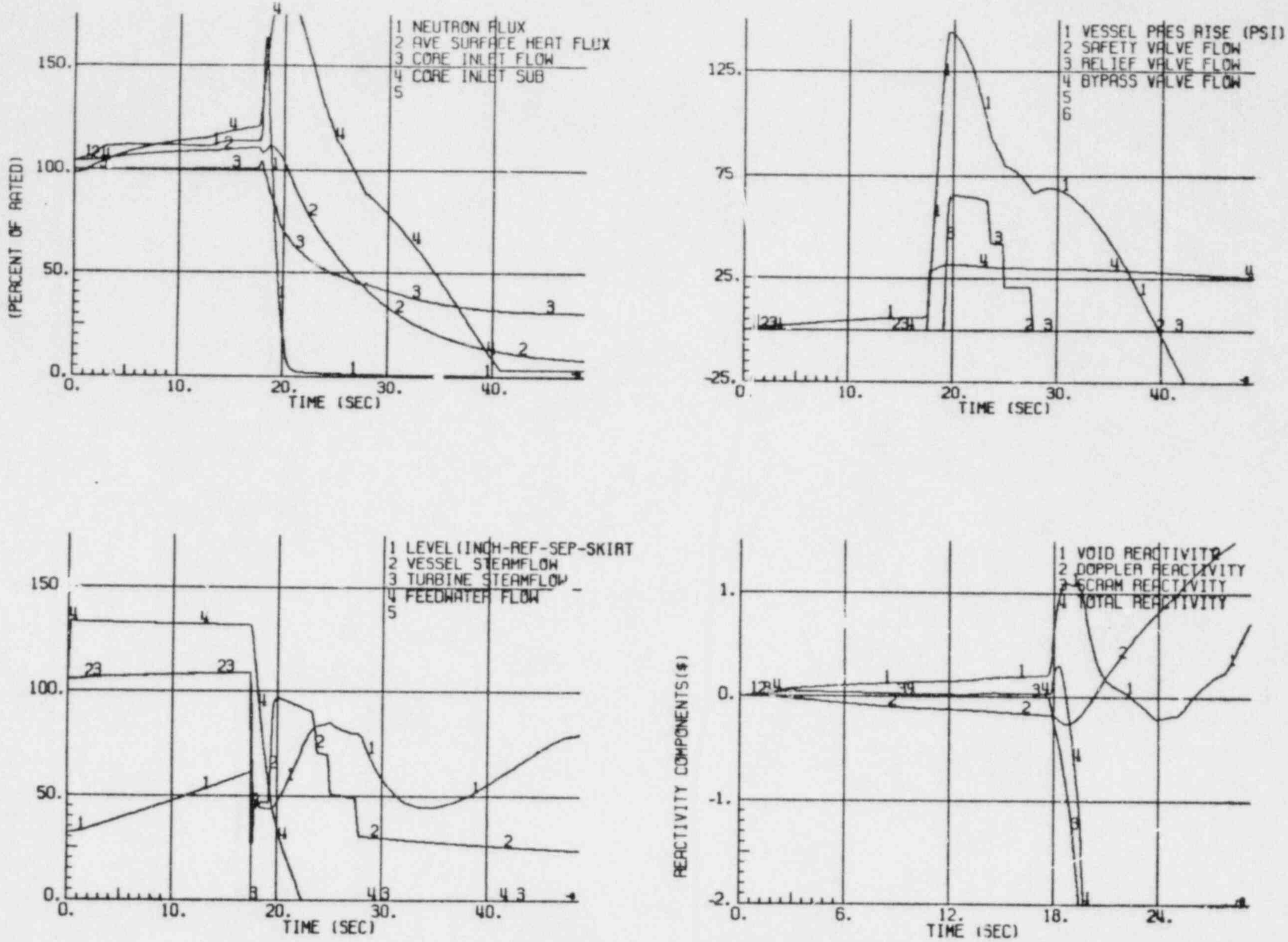


Figure 5. Plant Response to Feedwater Controller Failure, Maximum Demand

NOTES: 1. ROD PATTERN IS 1/4 CORE MIRROR SYMMETRIC (FULL CORE SHOWN)
2. NO. INDICATES NUMBER OF NOTCHES WITHDRAWN OUT OF 48.
BLANK IS A WITHDRAWN ROD
3. ERROR ROD IS (26, 35)

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58
59					6		4		4		6				
55				36		36		36		36		36			
51			6		6		2		2		6		6		
47		36		36		36		36		36		36		36	
43	6		6		10		14		14		10		6		6
39		36		36		36		40		36		36		36	
35	4		2		14		0		0		14		2		4
31		36		36		40		36		40		36		36	
27	4		2		14		0		0		14		2		4
23		36		36		36		40		36		36		36	
19	6		6		10		14		14		10		6		6
15		36		36		36		36		36		36		36	
11			6		6		2		2		6		6		
7				36		36		36		36		36			
3					6		4		4		6				

Figure 6. Limiting RWE Rod Pattern

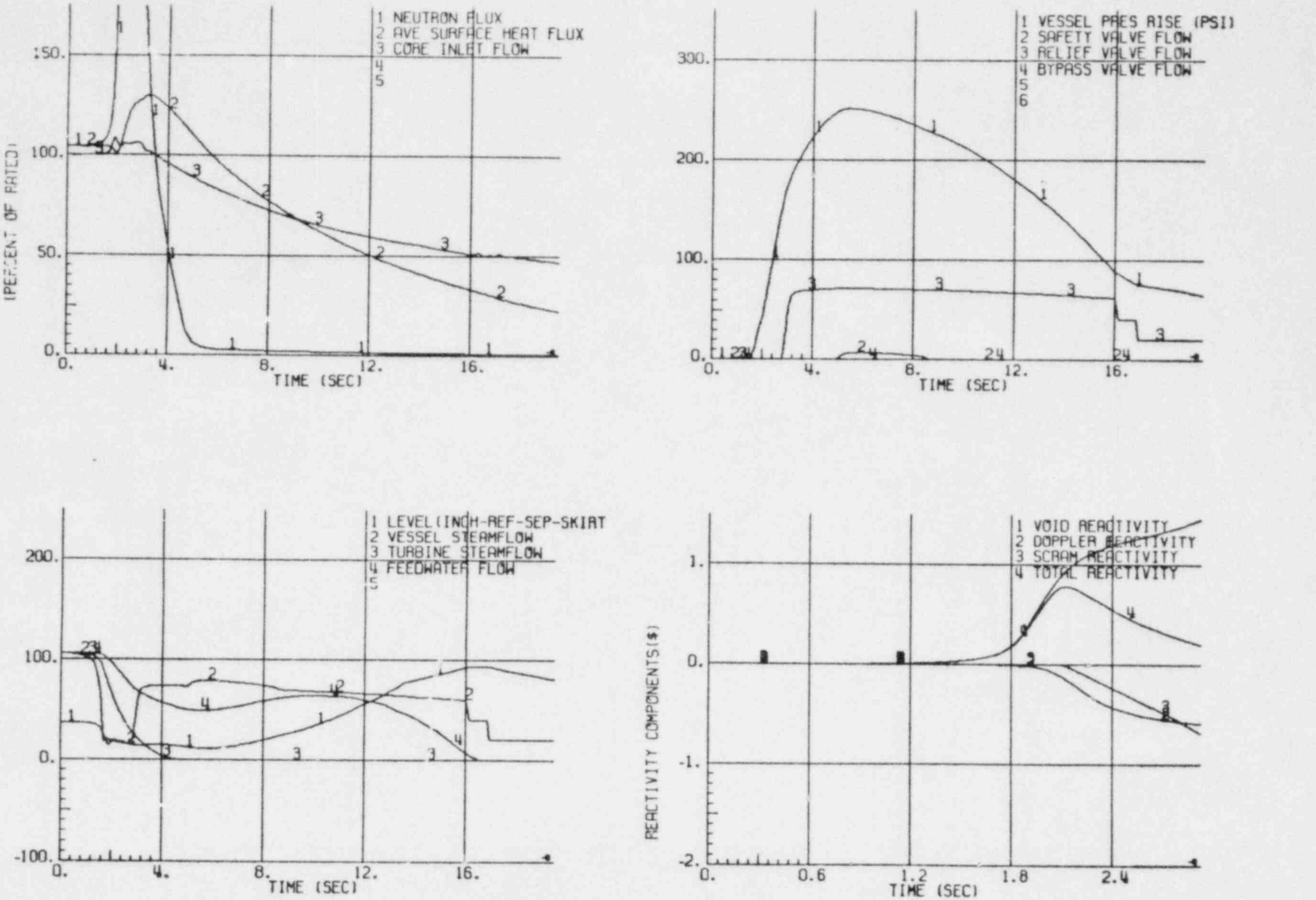


Figure 7. Plant Response to MSIV Closure

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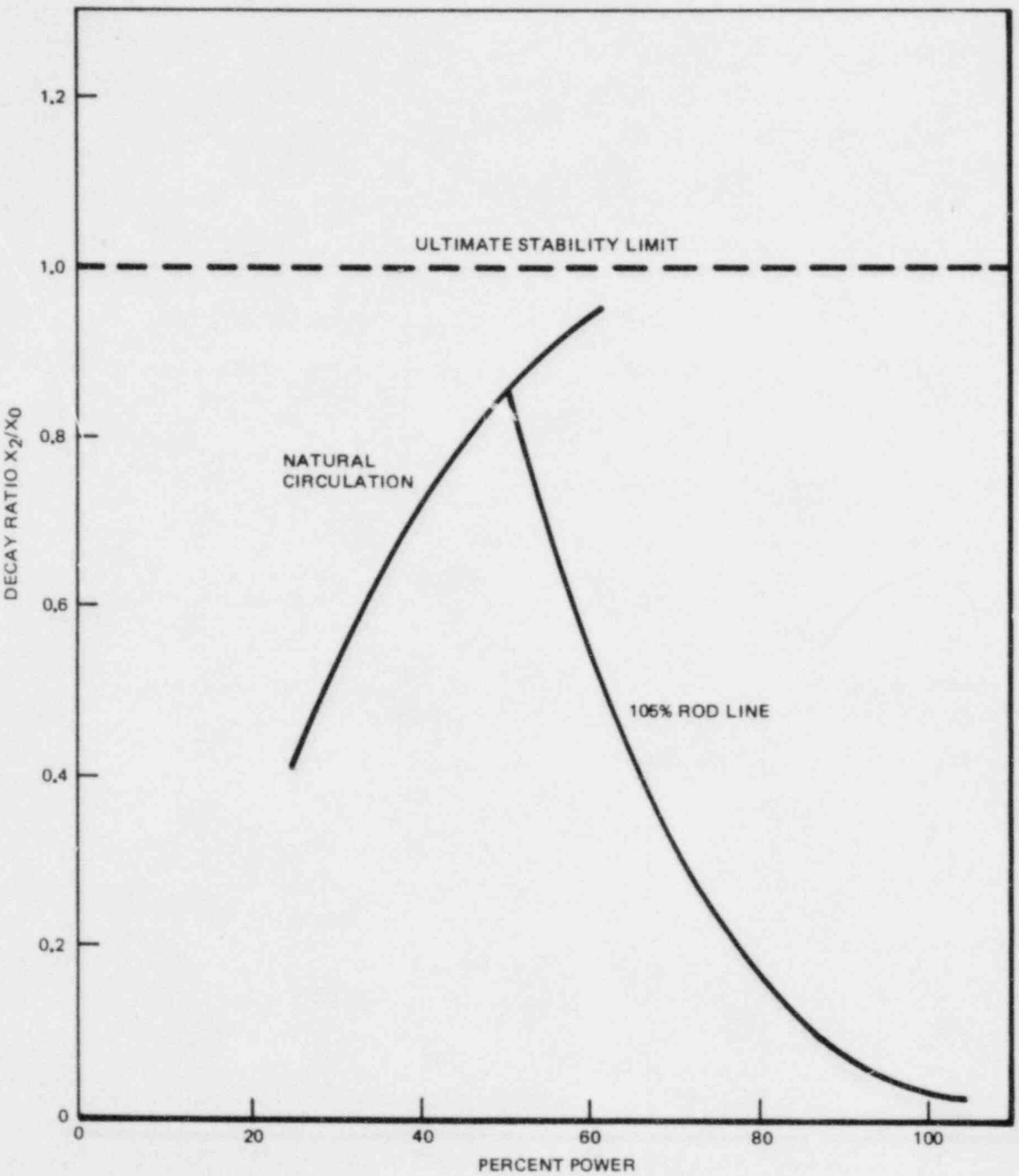

Figure 8. Decay Ratio

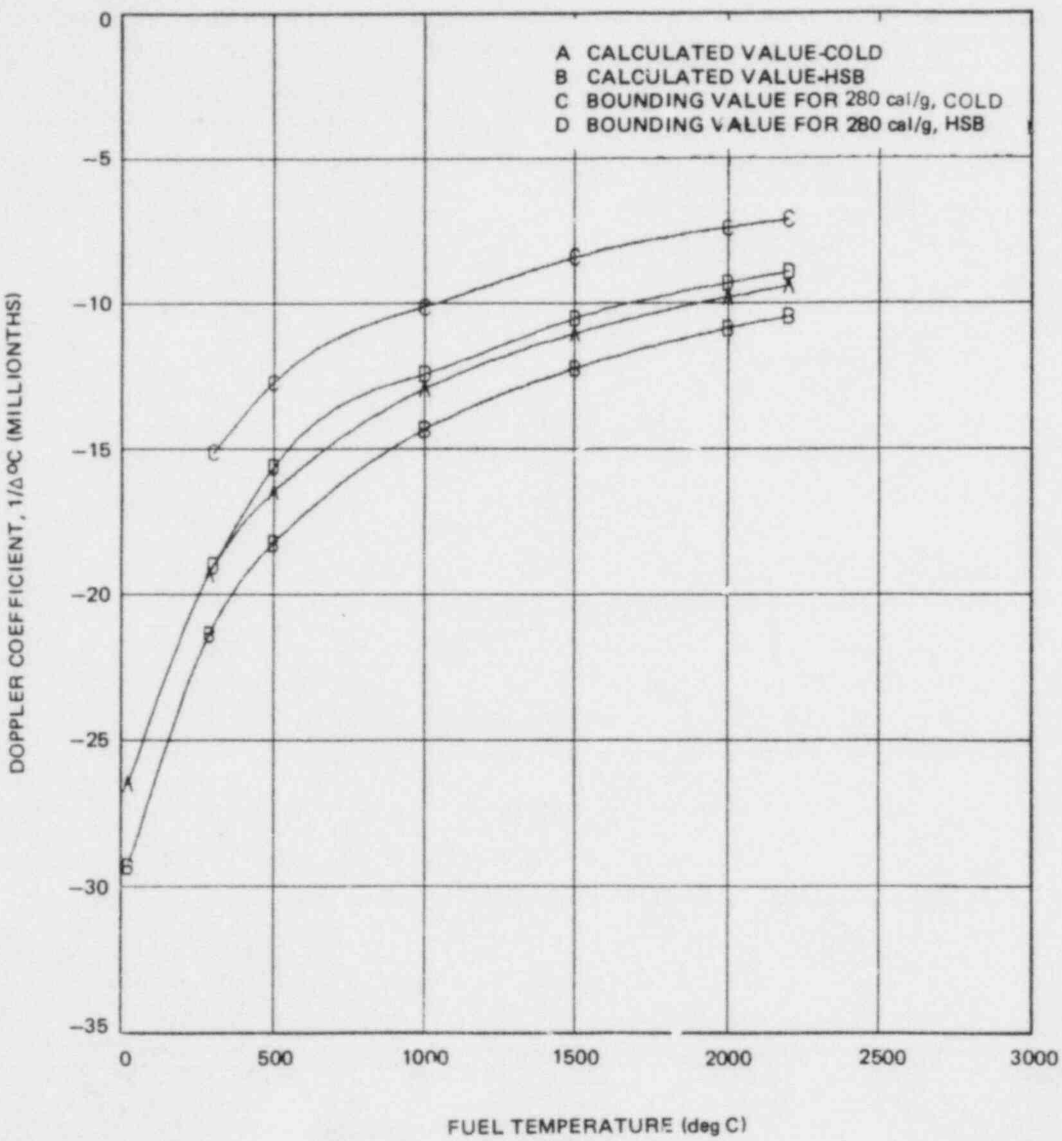

Figure 9. Doppler Reactivity Coefficient Comparison for RDA

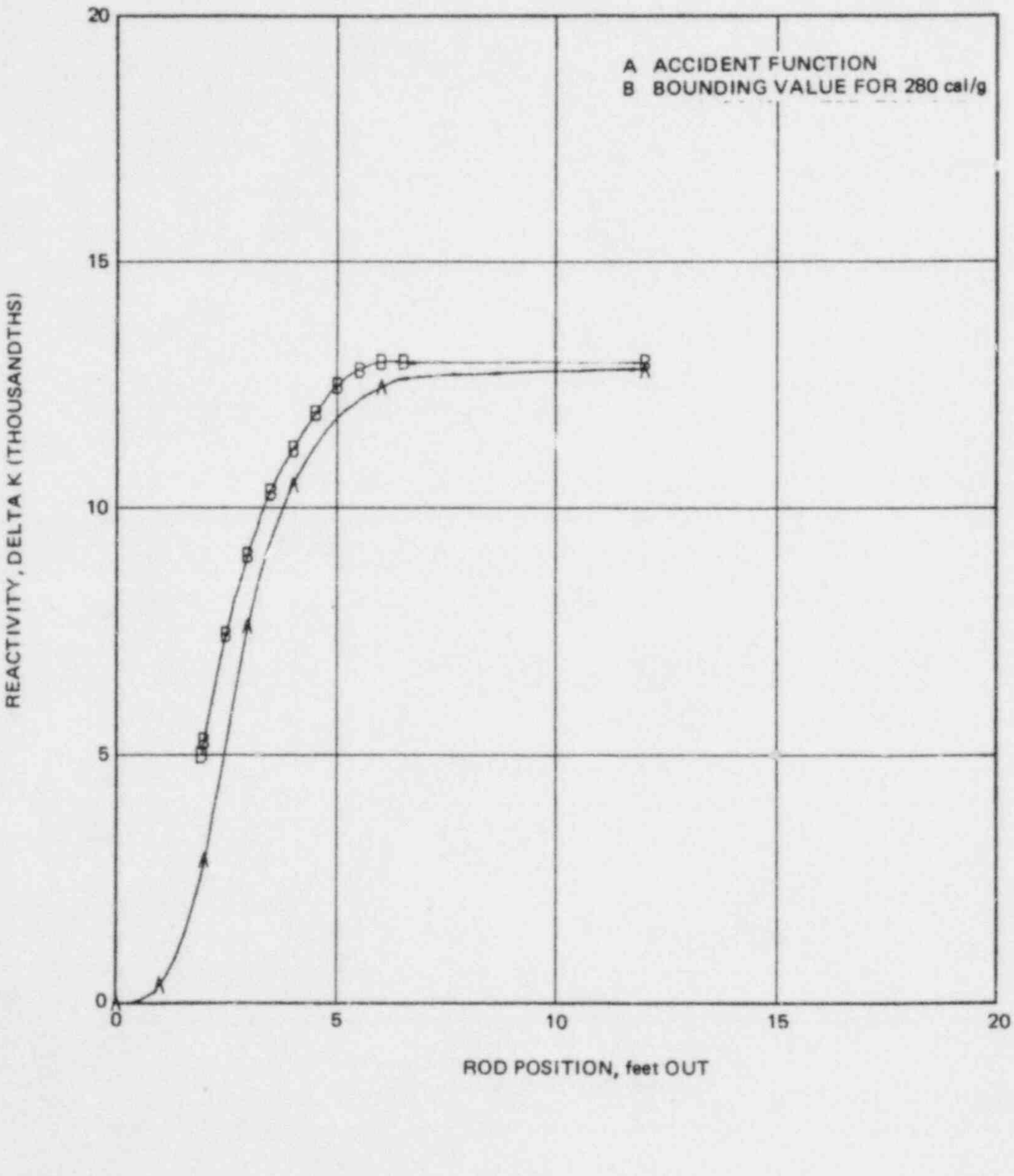

Figure 10. Accident Reactivity Shape Function at 20°C

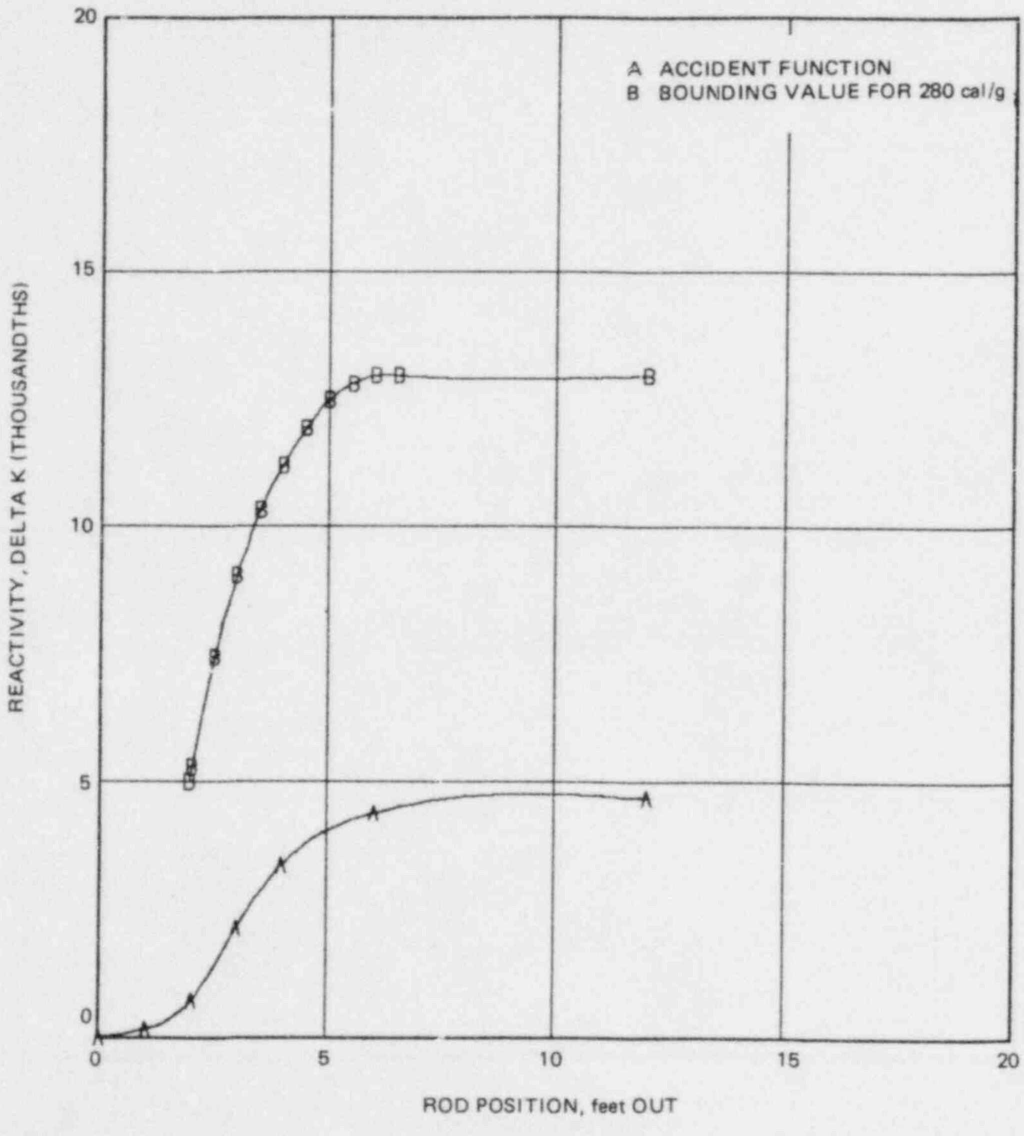

Figure 11. Accident Reactivity Shape Function at 286°C

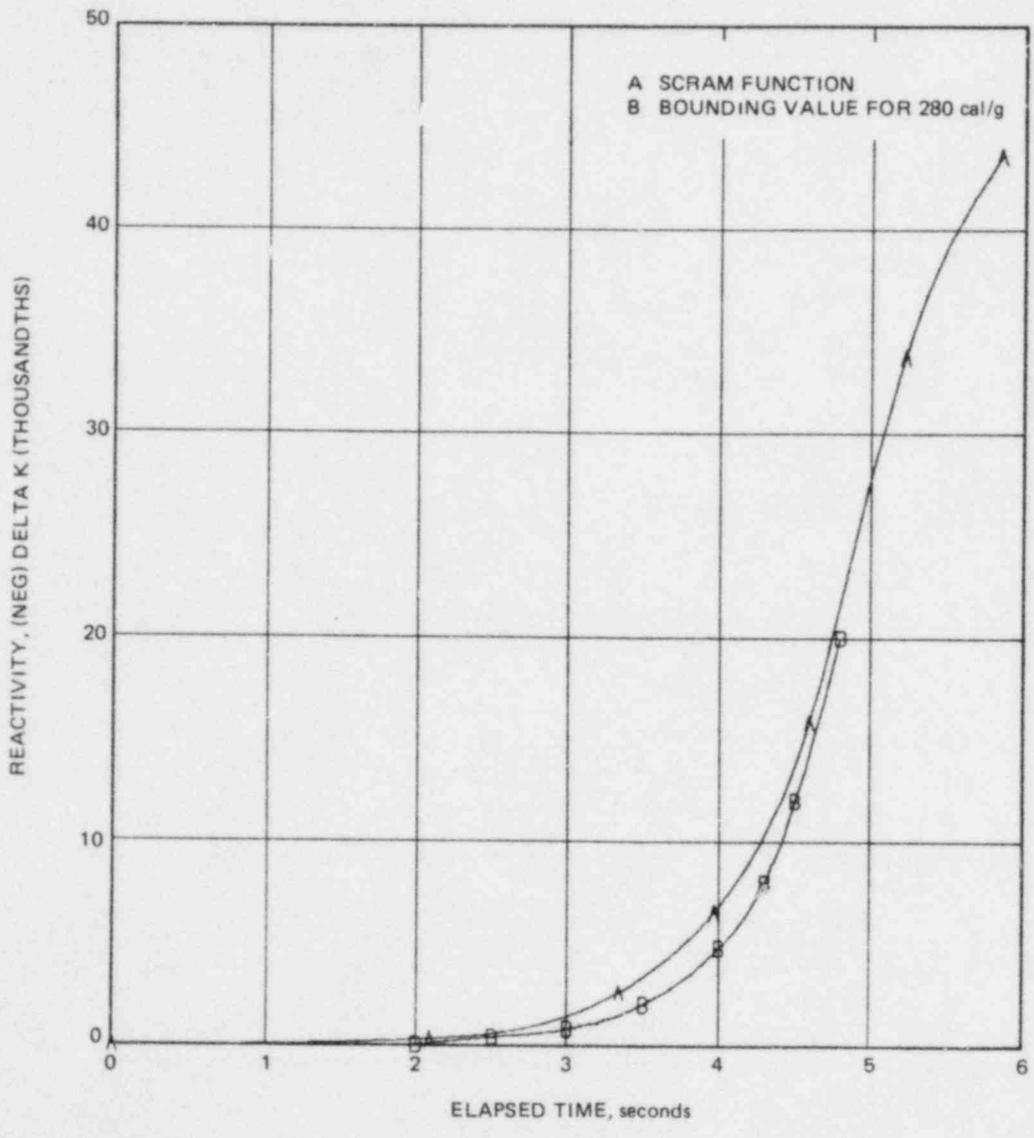

Figure 12. Scram Reactivity Function at 20°C

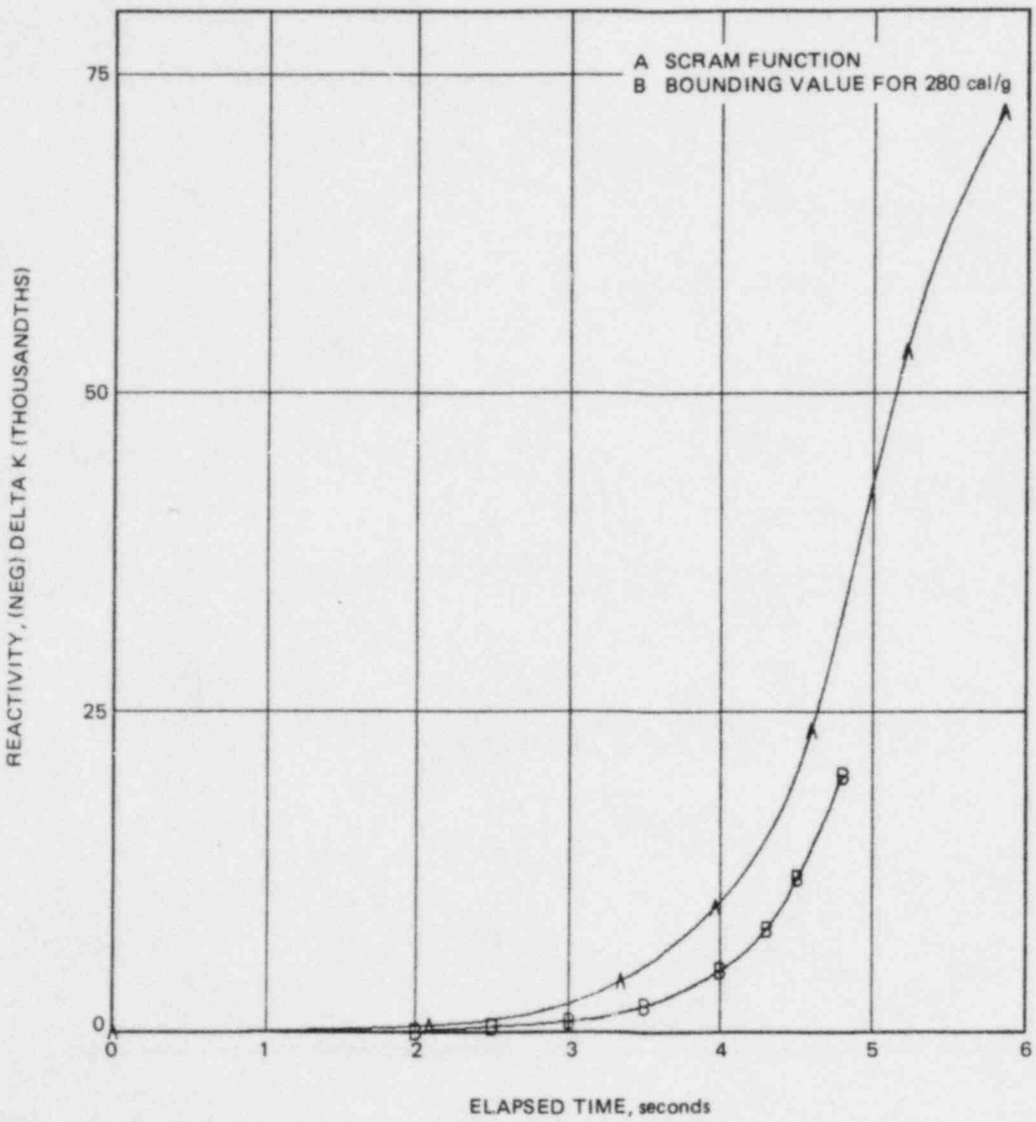

Figure 13. Scram Reactivity Function at 286°C

APPENDIX A
SHUTDOWN MARGIN DETERMINATION

A.1 BASES

The reference loading pattern, documented in item 3 of this supplemental reload submittal, is the basis for all reload licensing and operational planning and is comprised of the fuel bundles designated in item 2 of this supplemental submittal. It in turn is based on the best possible prediction of the core condition at the end of the present cycle and on the desired core energy capability for the reload cycle. It is designed with the intent that it will represent, as closely as possible, the actual core loading pattern.

A.2 CORE CHARACTERISTICS

The reference core is analyzed in detail to ensure that adequate cold shutdown margin exists. This section discusses the results of core calculations for shutdown margin.

A.2.1 Core Effective Multiplication and Control Rod Worth

Core effective multiplication and control rod worths were calculated using the TVA BWR simulator code (References A-1, A-3) in conjunction with the TVA lattice physics data generation code (References A-2, A-3) to determine the core reactivity with all rods withdrawn and with all rods inserted. A tabulation of the results is provided in Table A.1. These three eigenvalues (effective multiplication of the core, uncontrolled, fully controlled, and with the strongest rod out) were calculated at the beginning-of-cycle 4 core average exposure corresponding to the minimum expected end-of-cycle 3 core average exposure. The core was assumed to be in a xenon-free condition.

Cold k_{eff} was calculated with the strongest control rod out at various exposures through the cycle. The value R is the difference between the strongest rod out k_{eff} at BOC and the maximum calculated strongest rod out k_{eff} at any exposure point. The strongest rod out k_{eff} at any exposure point is equal to or less than:

$$k_{eff}^{SRO} = (\text{Fully controlled } k_{eff})\ \text{BOC} + (\text{Strongest Rod Worth})\ \text{BOC} + R$$

A.2.2 Reactor Shutdown Margin

Technical Specifications require that the refueled core must be capable of being made subcritical with 0.38 percent Δk margin in the most reactive condition throughout the subsequent operating cycle with the most reactive control rod in its full out position and all other rods fully inserted.

The shutdown margin is determined by using the BWR simulator code to calculate the core multiplication at selected exposure points with the strongest rod fully withdrawn. The shutdown margin for the reloaded core is obtained by subtracting the k_{eff}^{SRO} given in Table A.1 from the critical k_{eff} of 1.0, resulting in a calculated cold shutdown margin of 1.1 percent Δk.

Table A.1

CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL ROD WORTHS - NO VOIDS, NO XENON, 20°C

Uncontrolled, K_{eff}^{UNC}	1.120
Fully Controlled, K_{eff}^{CON}	0.955
Strongest Control Rod Out, K_{eff}^{SRO}	0.989
R, Maximum Increase in Cold Core Reactivity With Exposure Into Cycle, Δk	0.000

REFERENCES

A-1. S. L. Forkner, G. H. Meriwether, and T. D. Beu, "Three-Dimensional LWR Core Simulation Methods," TVA-TR78-03A, 1978

A-2. B. L. Darnell, T. D. Beu, and G. W. Perry, "Methods for the Lattice Physics Analysis of LWR's," TVA-TR78-02A, 1978

A-3. "Verification of TVA Steady-State BWR Physics Methods," TVA-TR79-01A, 1979

APPENDIX B

Fuel Loading Error LHGR*: Rotated Bundle, 16.9 kW/ft; Misplaced Bundle, 18.1 kW/ft

Safety/Relief Valve Capacity at Setpoint (No./%): **10/63.6****

Spring Safety Valve Capacity at Setpoint (No./%): 2/14.2

Rated Steam Flow: 14.09 x 10^6 lb/hr

GETAB Analysis Initial Conditions

Reactor Pressure: 1035 psia

Inlet Enthalpy: 521.5 Btu/lb

*2.2% peaking penalty for densification is included.
**Assumes one safety/relief vlave out of service.

APPENDIX C
MARGIN TO SPRING SAFETY VALVES

The rationale for changing the basis for providing pressure margin to the spring safety valves is presented in Reference C-1. This change has been accepted by the NRC (Reference C-2).

On this basis the plant can operate at full power throughout the cycle.

REFERENCES

C-1. J. F. Quirk (GE) letter to Olan D. Parr (NRC), "General Electric Licensing Topical Report NEDE-24011-P-A, 'Generic Reload Fuel Application,' Appendix D, Second Submittal," dated February 28, 1979.

C-2. Letter, T. A. Ippolito (NRC) to D. L. Peoples (Commonwealth Edison Co.) enclosing a Safety Evaluation supporting Amendment No. 42 to Facility Operating License No. DPR-25 Dresden Nuclear Power Station Unit 3, dated April 16, 1980.