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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BROWNS FERRY NUCLEAR PLANT UNIT 2, RELOAD NO. 4 (CYCLE 5)

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FOR
BROWNS FERRY NUCLEAR PLANT
(CYCLE 5)
UNIT 2, RELOAD NO. 4

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GENERAL ELECTRIC

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CONTENTS OF THIS REPORT
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1. PLANT UNIQUE ITEMS (1.0)*

Data for Sections 4 and 5 provided by Tennessee Valley Authority (TVA)	Appendix A
Safety/Relief Valve Capacity	Appendix B

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

Fuel Type	Cycle Loaded	Number	Number Drilled
Irradiated			
8DB274L	2	8	8
8DRB284L	3	232	232
8DB274L	3	36	36
P8DRB284L	4	240	240
New			
P8DRB284L	5	168	168
P8DRB265H	5	80	80
Total		764	764

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	19155 MWd/ST
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	18755 MWd/ST
Assumed reload cycle core average exposure at end of cycle:	18235 MWd/ST
Core loading pattern:	Figure 1

*( ) refers to area of discussion in "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-4, January 1982.

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

See Appendix A

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

See Appendix A

6. RELOAD UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 AND S.2) (REDY EVENTS ONLY)

	EOC 5
Void Fraction (%)	39.8
Average Fuel Temperature (°F)	1318
Void Coefficient N/A* (¢/% Rg)	-6.85/-8.56
Doppler Coefficient N/A (¢/°F)	-0.224/-0.213
Scram Worth N/A ($)	-46.31/-37.05

7. RELOAD UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (S.2)

Fuel Design	Peaking Factors (Local	Radial	Axial)	R-Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
BOC 5 to EOC 5							
P8x8R	1.20	1.51	1.40	1.051	6.352	108.0	1.27
8x8R	1.20	1.57	1.40	1.051	6.619	105.6	1.23
8x8	1.22	1.39	1.40	1.098	5.855	106.9	1.25

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

Transient Recategorization : No
Recirculation Pump Trip : Yes
Rod Withdrawal Limiter : No
Thermal Power Monitor : Yes**
Measured Scram Time : No
Number of Exposure Points : 1

*N = Nuclear input data
A = Used in transient analysis
**No credit for the thermal power monitors was used in the analysis.

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (S.2.2.1)

Transient	Flux (%NBR)	Q/A (%NBR)	ΔCPR P8x8R	ΔCPR 8x8R	ΔCPR 8x8	Figure
Exposure: BOC 5 to EOC 5 Load Rejection w/o Bypass	599	122	0.21	0.18	0.18	2
Exposure: BOC 5 to EOC 5 Loss of Feedwater Heater	122	122	0.13	0.13	0.12	3
Exposure: BOC 5 to EOC 5 Feedwater Controller Failure	385	120	0.16	0.15	0.14	4

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

Limiting Rod Pattern: Figure 5

Includes 2.2% Power Spiking Penalty: Yes

Rod Block Reading	Rod Position (feet withdrawn)	ΔCPR P8x8R/8x8R*	MLHGR (kW/ft) 8x8R/P8x8R
104	3.5	0.10	14.5
105	4.0	0.12	15.0
106	4.0	0.12	15.0
107	4.5	0.13	15.2
108	5.0	0.14	15.2
109	5.5	0.15	15.2
110	6.0	0.16	15.2

Set point selected is: 106

*The 8x8 fuel type is not limiting since it is highly-exposed, low-reactivity fuel located primarily on the periphery of the core and not adjacent to any control blades whose worth is near that of the error rod.

11. CYCLE MCPR VALUES (S.2)

Nonpressurization Events

Exposure Range: BOC 5 to EOC 5	P8x8R	8x8R	8x8
Loss of Feedwater Heating	1.20	1.20	1.19
Fuel Loading Error	1.22		
Rod Withdrawal Error	1.19	1.19	

Pressurization Events:

Exposure Range: BOC 5 to EOC 5	Option A			Option B		
	P8x8R	8x8R	8x8	P8x8R	8x8R	8x8
Load Rejection Without Bypass	1.34	1.30	1.30	1.24	1.22	1.22
Feedwater Controller Failure	1.28	1.27	1.26	1.25	1.24	1.23

12. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1218	1254	Figure 6

13. STABILITY ANALYSIS RESULTS (S.2.4)

Rod Line Analyzed: 105%

Decay Ratio:	Figure 7
Reactor Core Stability Decay Ratio, x_2/x_0	0.74
Channel Hydrodynamic Performance Decay Ratio, x_2/x_0	
Channel Type	
8x8R/P8x8R	0.29
8x8	0.38

14. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

Event	Initial MCPR	Resulting MCPR
Misoriented	1.20	1.07

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

Bounding Analysis Results:

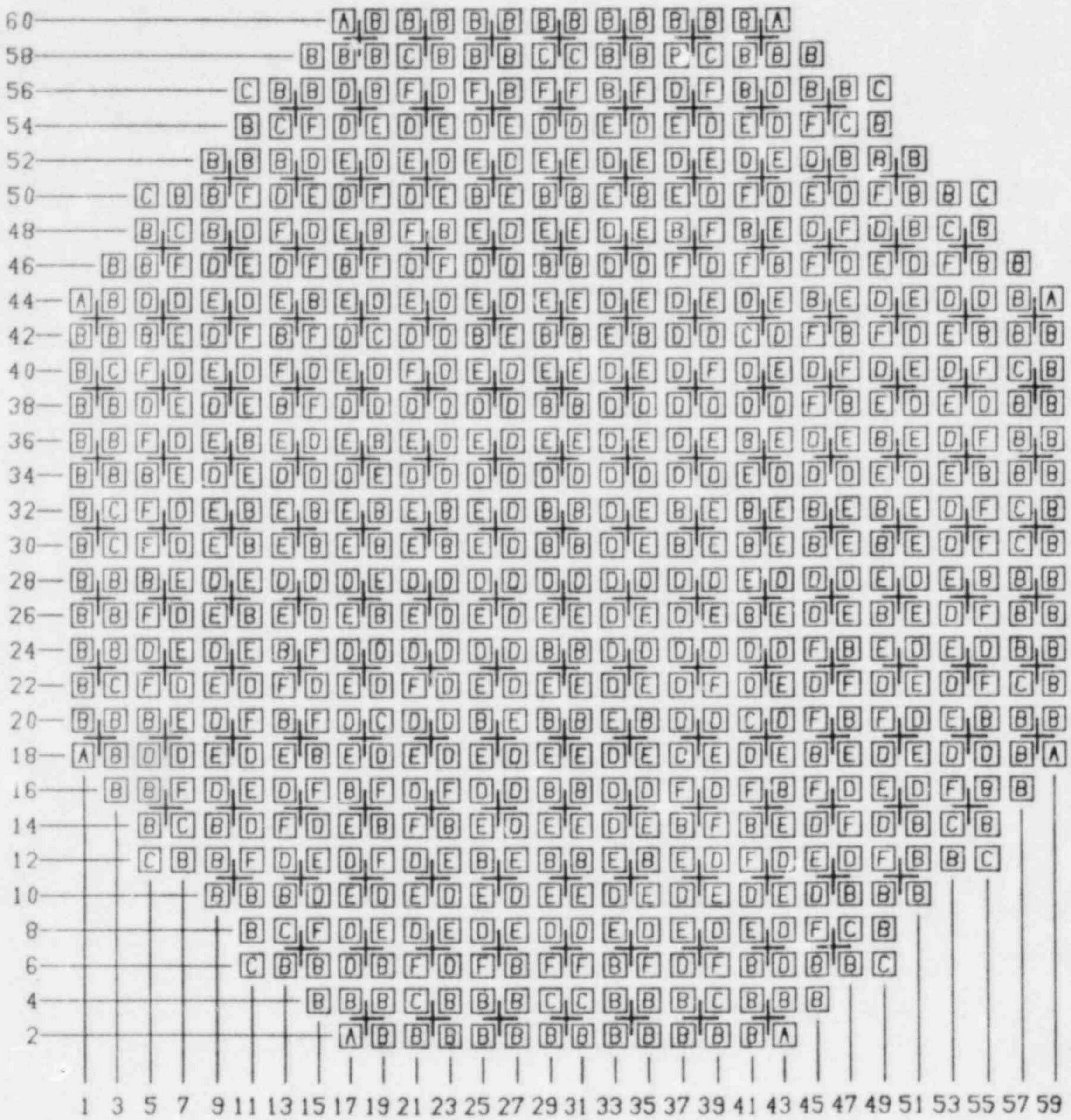
Doppler Reactivity Coefficient:	Figure 8
Accident Reactivity Shape Functions:	Figures 9 and 10
Scram Reactivity Functions:	Figures 11 and 12

Plant Specific Analysis Results:

Parameter(s) Not Bounded, Cold:	Accident Reactivity
Resultant Peak Enthalpy, Cold:	264.5 cal/gm
Parameter(s) Not Bounded, HSB:	None
Resultant Peak Enthalpy, HSB:	

16. LOSS-OF-COOLANT ACCIDENT RESULT (S.2.5.2)

Refer to "Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2", General Electric Company, February 1978 (NEDO-24088-1, as amended).



FUEL TYPE	
A = 8DB274L	D = P8DRB284L
B = 8DRB284L	E = P8DRB284L
C = 8DB274L	F = P8DRB265H

Figure 1. Reference Core Loading Pattern

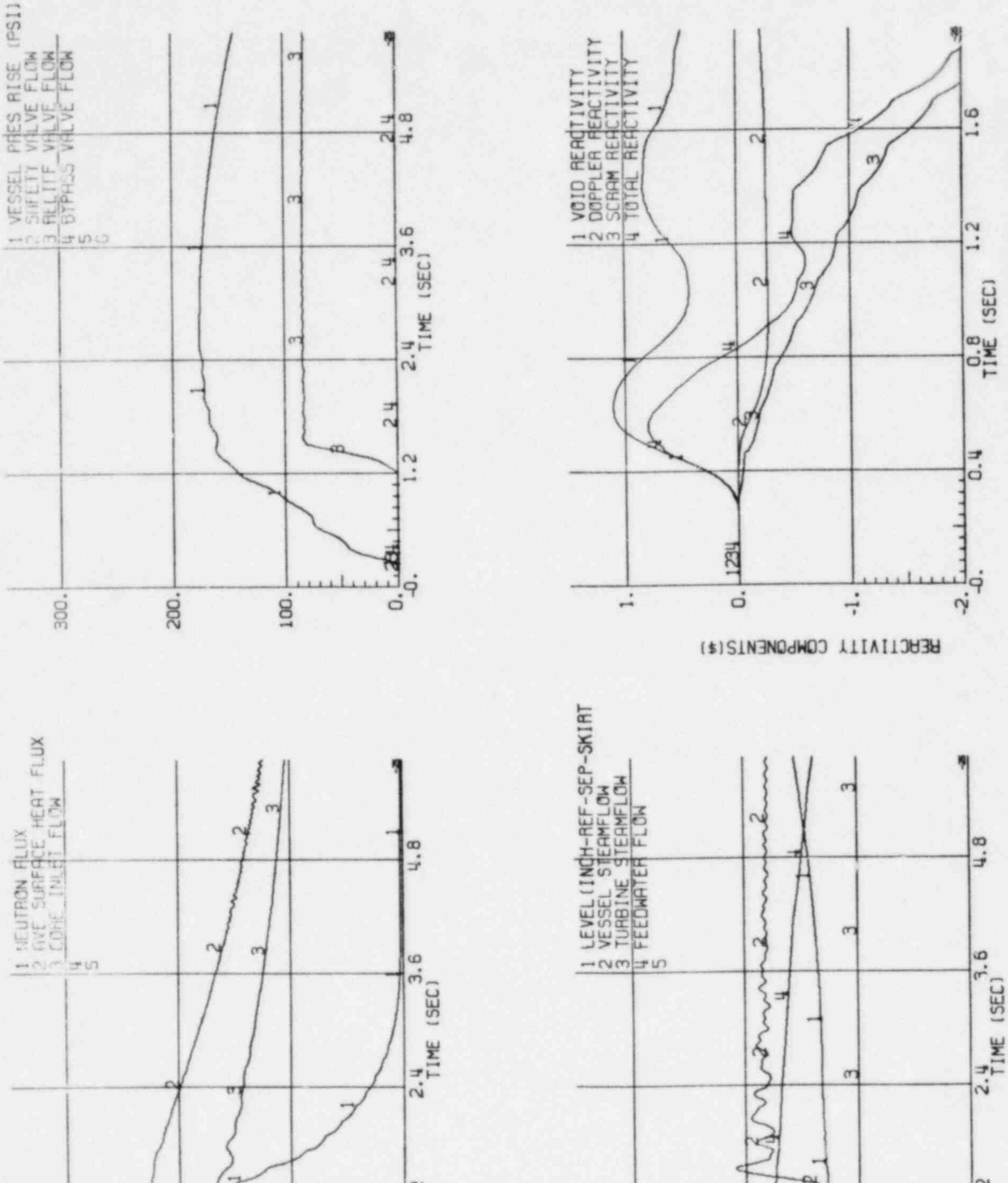


Figure 2. Plant Response to Generator Load Rejection Without Bypass

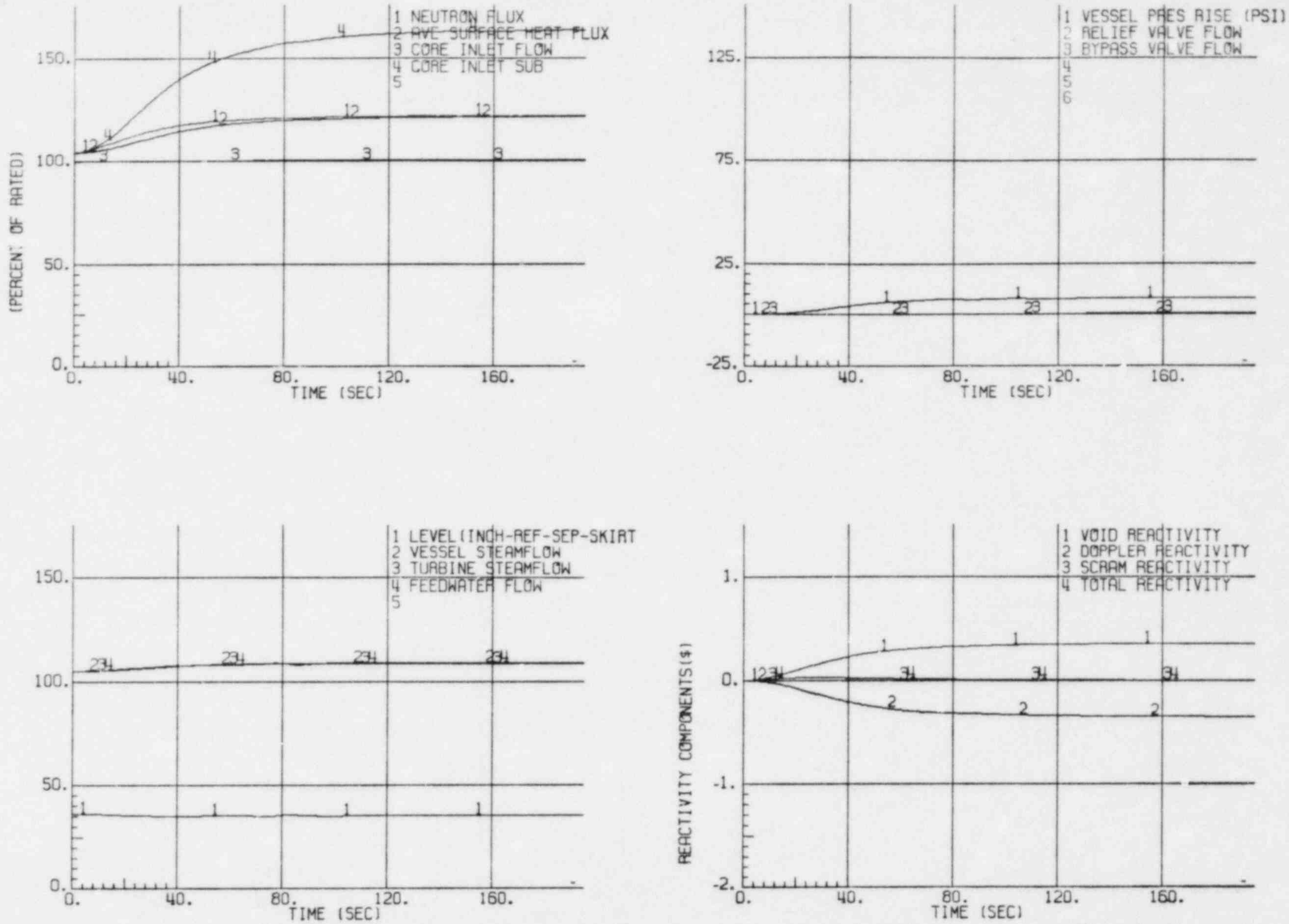


Figure 3. Plant Response to Loss of 100°F Feedwater Heating

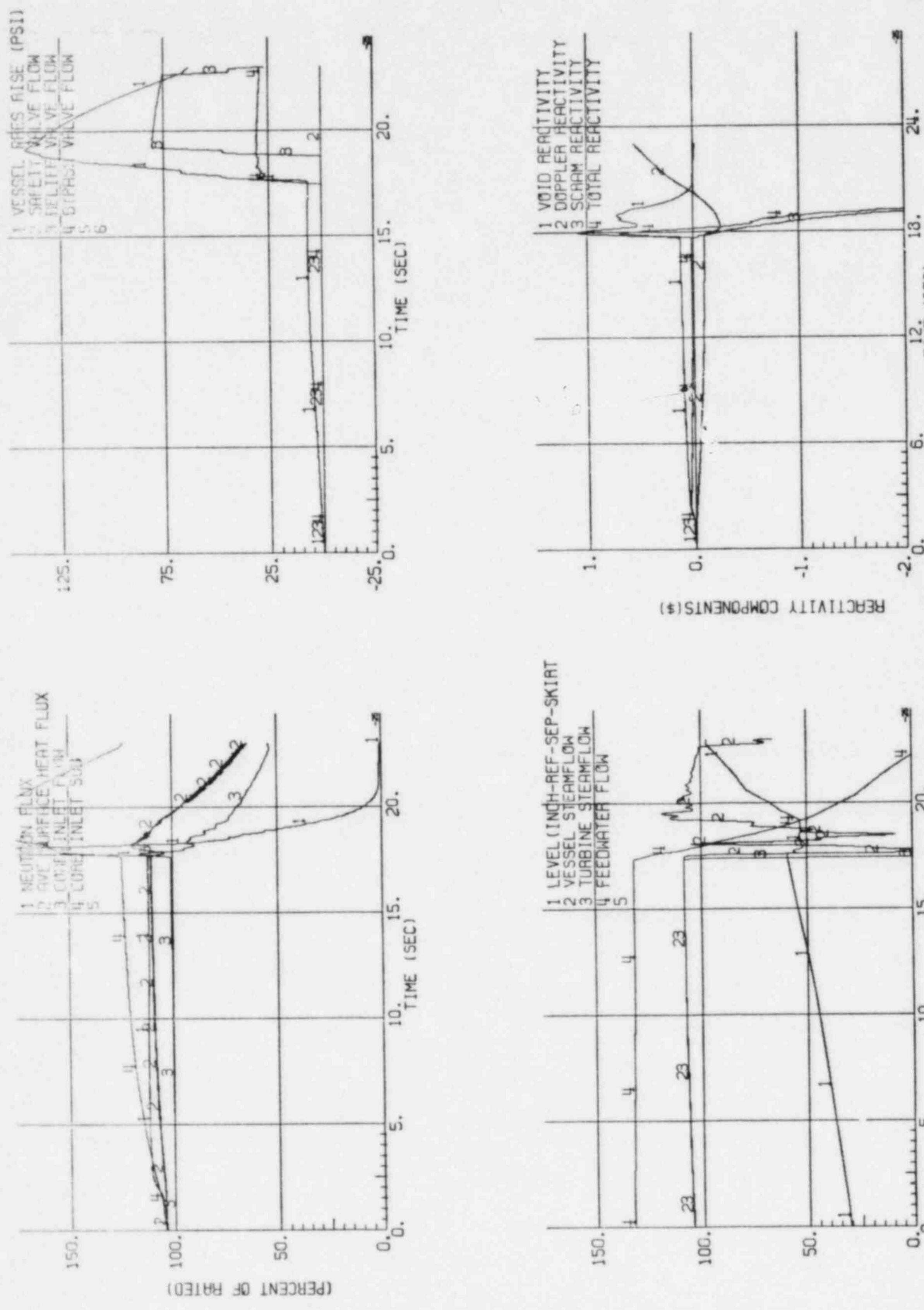


Figure 4. Plant Response to Feedwater Controller Failure

	2	6	10	14	18	22	26	30
59						10		10
55					40		36	
51				10		2		6
47			40		36		36	
43		10		2		6		8
39			36		40		44	
35		10		6		8		0
31	40		36		36		44	

NOTES:
1. Rod pattern is 1/4-core mirror symmetric.
2. Numbers indicate number of notches withdrawn out of 48. Blank is a withdrawn rod.
3. Error rod is (30, 35).

Figure 5. Limiting RWE Rod Pattern

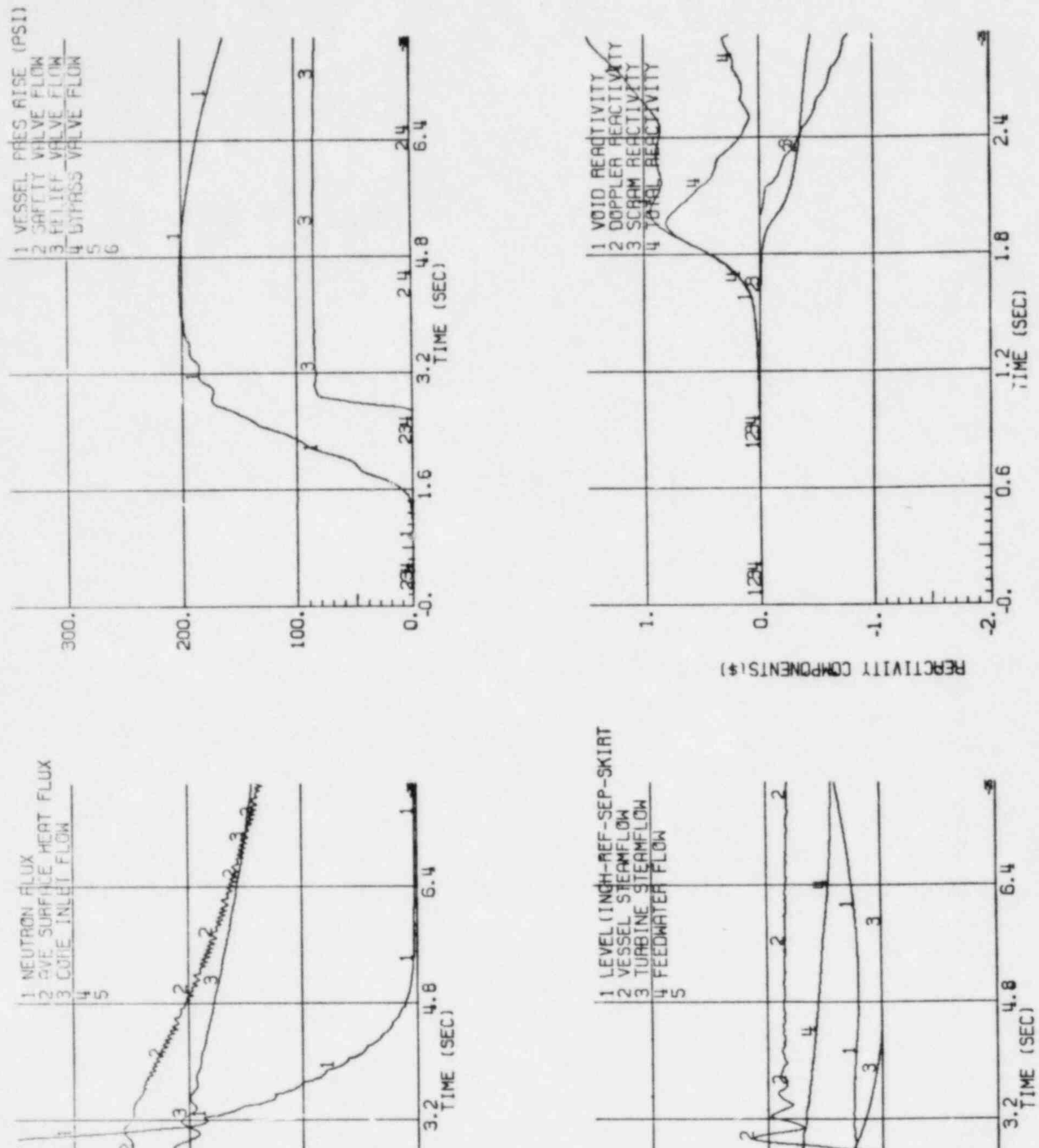


Figure 6. Plant Response to MSIV Closure (Flux Scram)

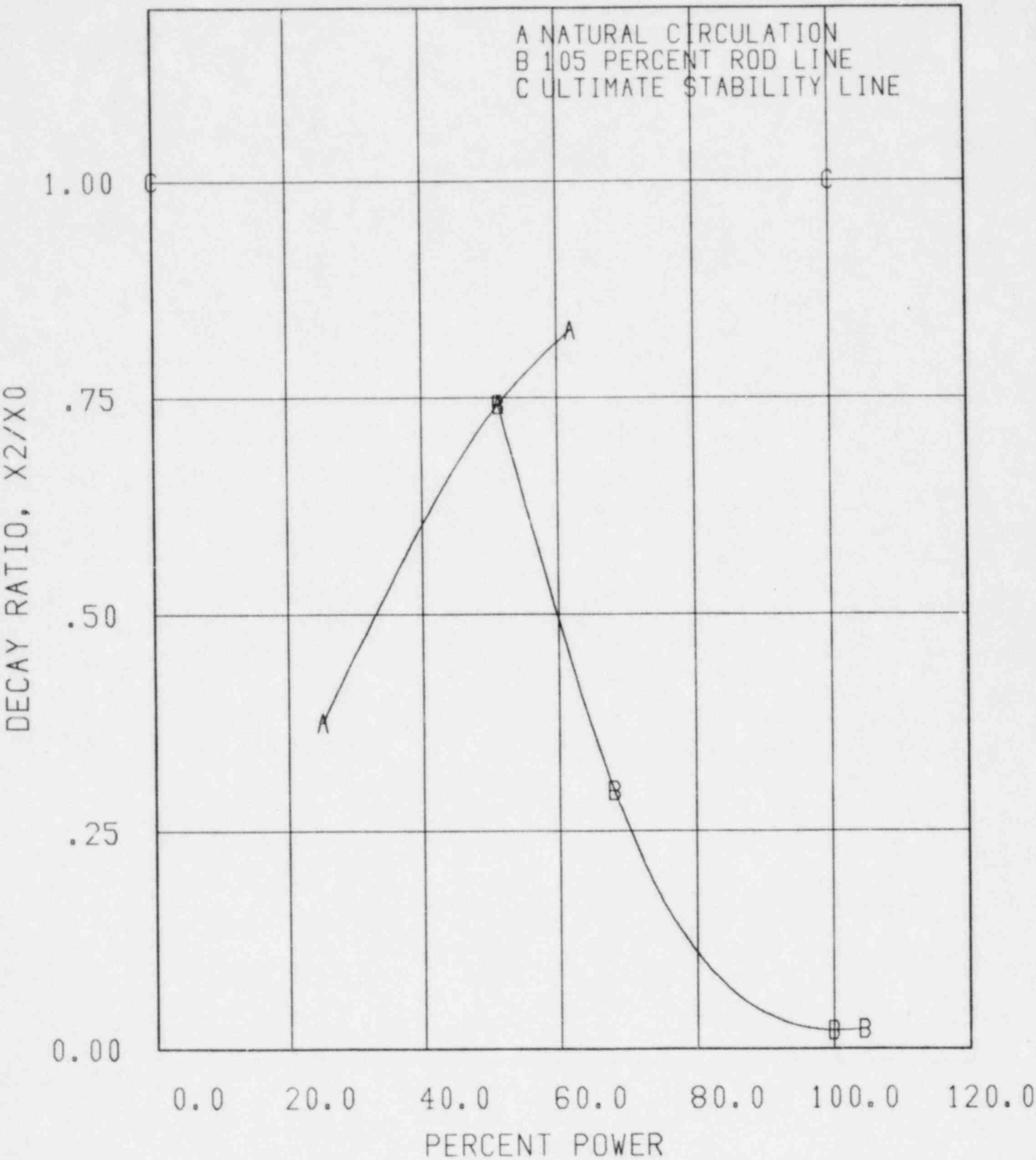


Figure 7. Reactor Core Decay Ratio

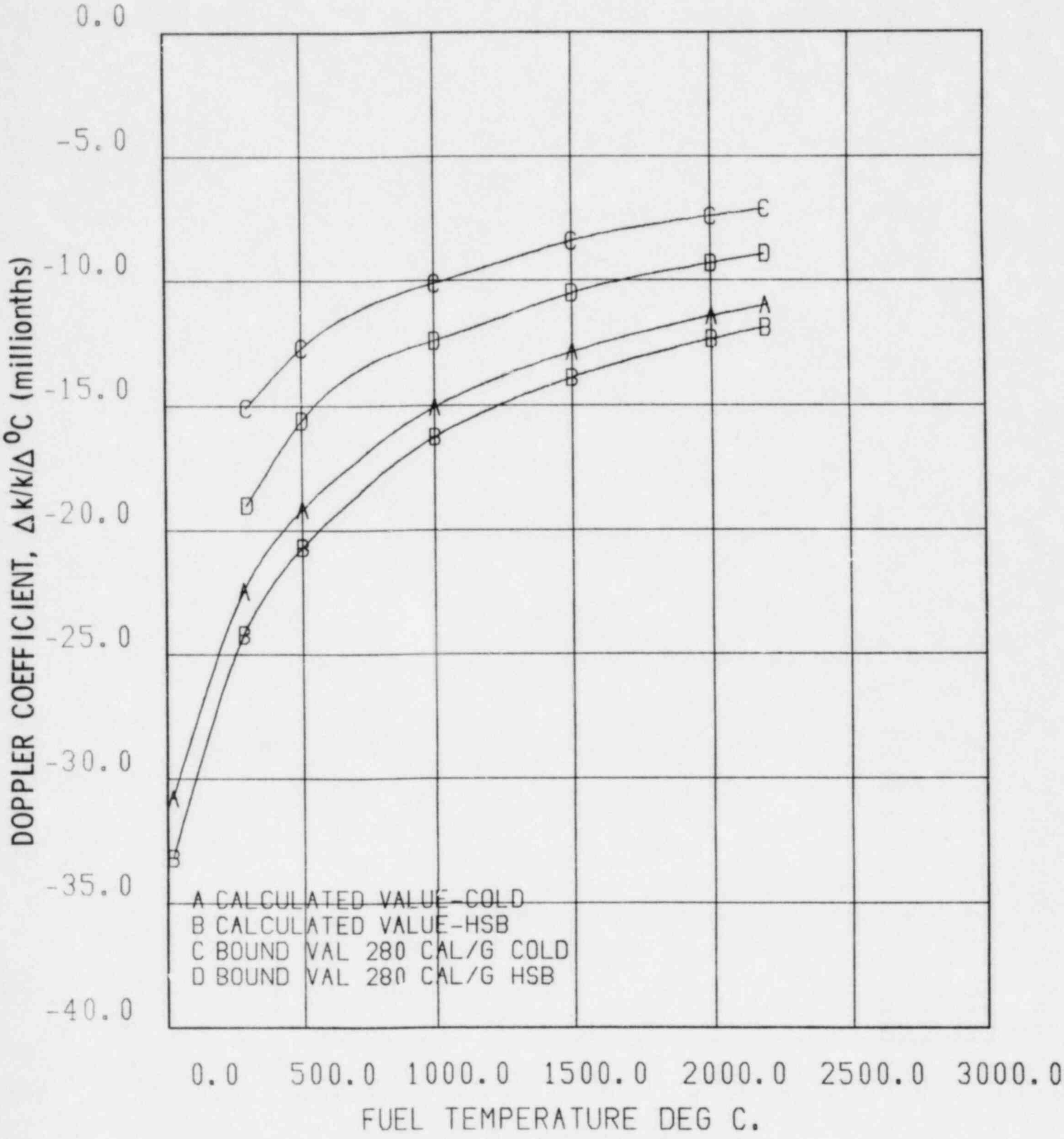


Figure 8. Doppler Reactivity Coefficient Comparison for RDA

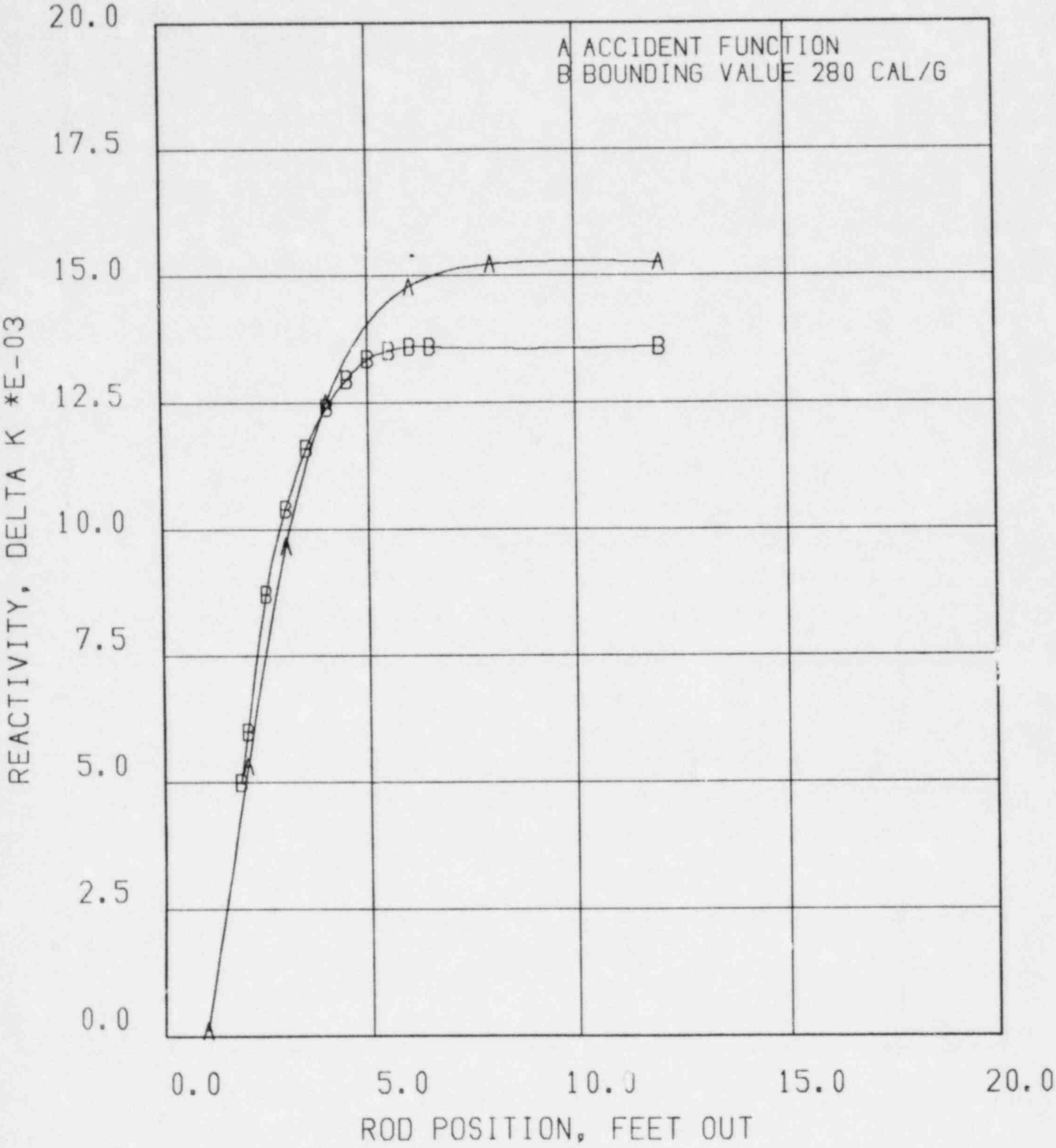


Figure 9. Accident Reactivity Shape Function Cold Startup

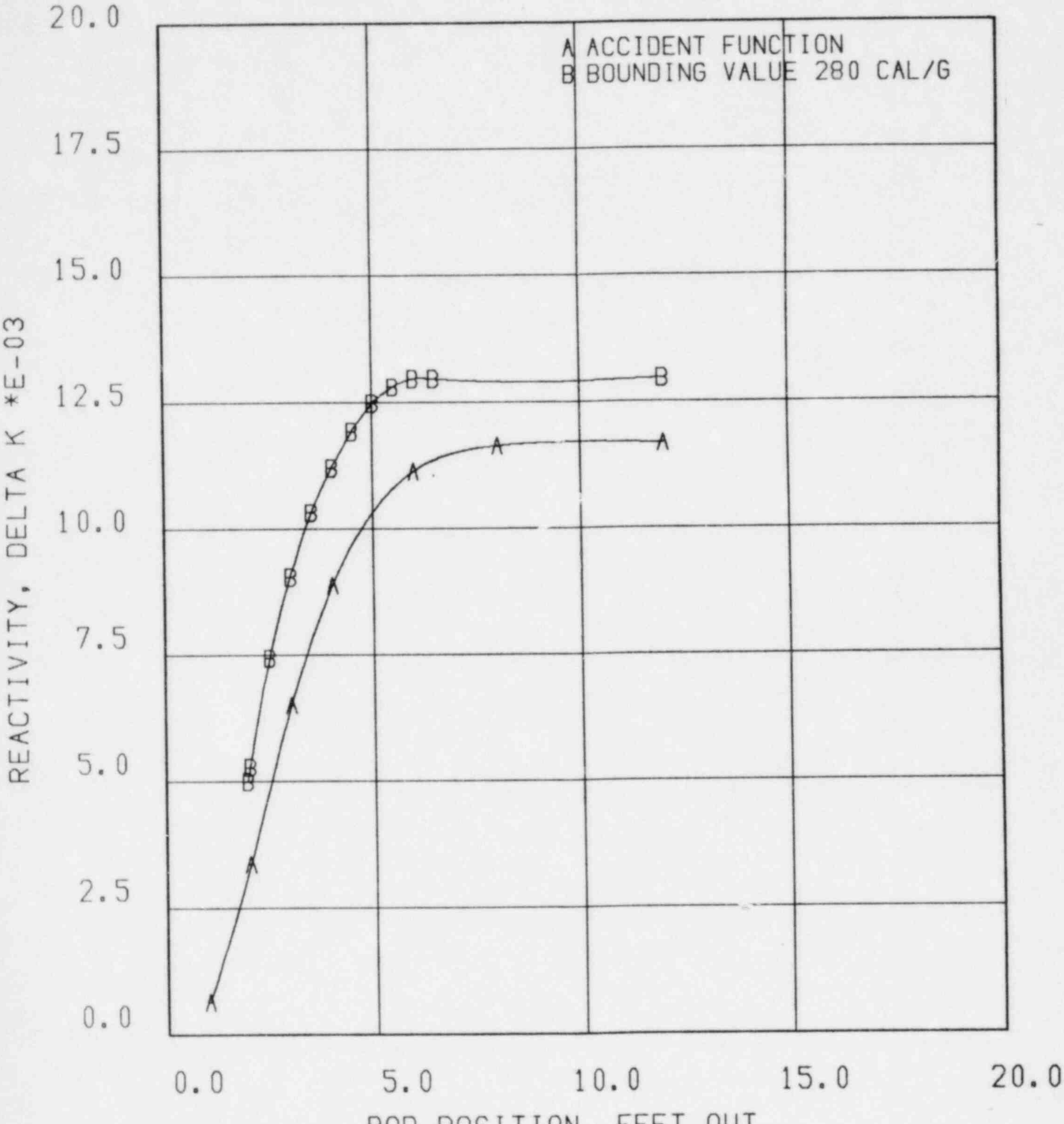


Figure 10. Accident Reactivity Shape Function Hot Startup

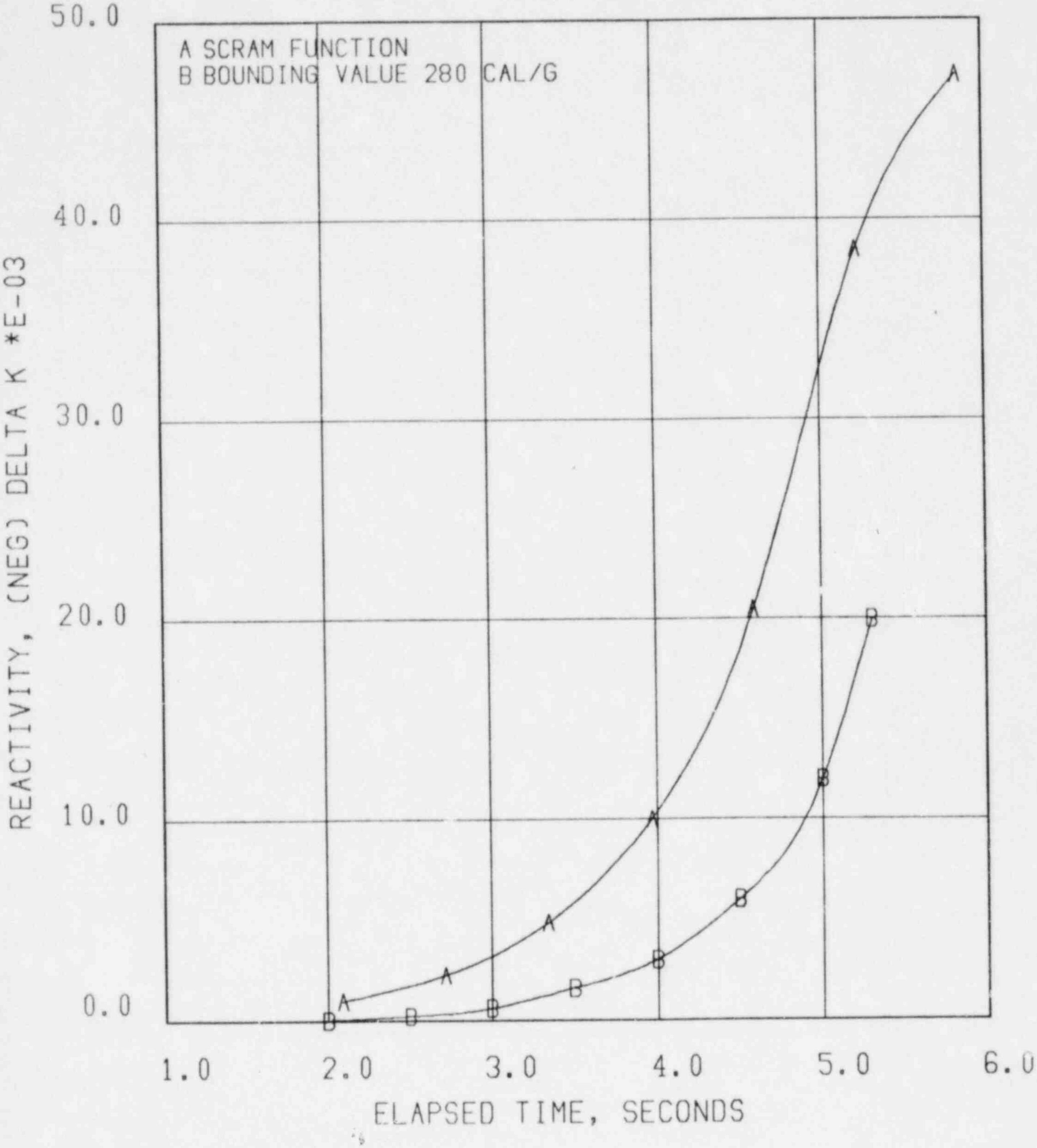


Figure 11. Scram Reactivity Function Cold Startup

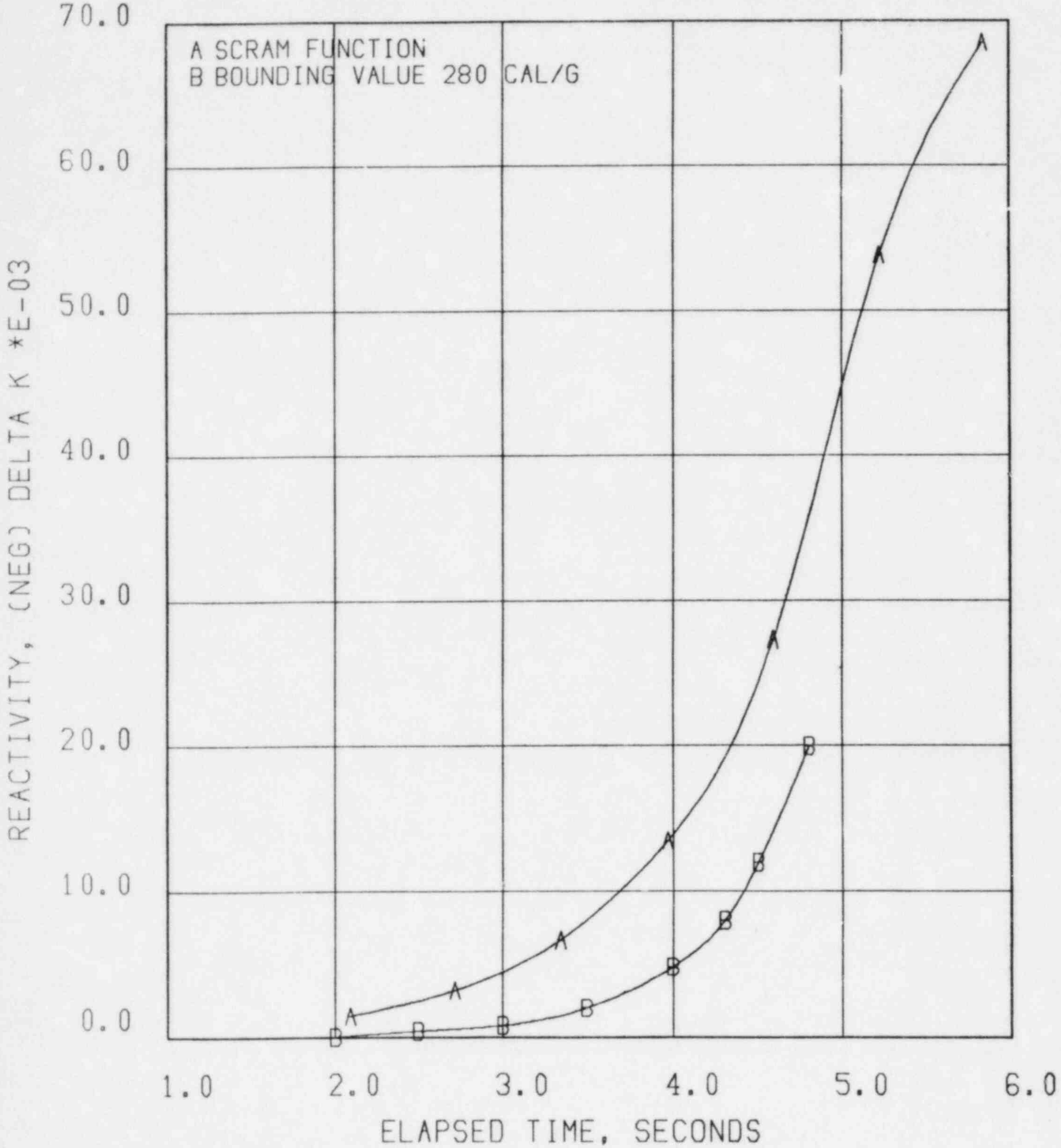


Figure 12. Scram Reactivity Function Hot Startup

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 shutdown margin exists. This section discusses the results of core calculations for shutdown margin (including the liquid poison system).

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 (Reference 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 5 core average exposure corresponding to the minimum expected end-of-cycle 4 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 BCC 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% 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 1.4% Δk.

A.2.3 Standby Liquid Control System

The standby liquid control system (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from a full power and minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state.

The SLCS shutdown margin is determined by using the BWR simulator code to calculate the core multiplication for the cold, xenon-free, all rods out conditions at the exposure point of maximum cold reactivity, with the soluble boron concentration given in the technical specifications. The resulting k-effective is subtracted from the critical k-effective of 1.0 to obtain the SLCS shutdown margin. Table A-2 gives the results of the SLCS evaluation.

Table A-1

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

Uncontrolled, k_{eff}^{UNC}	1.115
Fully Controlled, k_{eff}^{CON}	0.955
Strongest Control Rod Out, k_{eff}^{SRO}	0.986
R, Maximum Increase in Cold Core Reactivity With Exposure Into Cycle, Δk	0.000

Table A-2

STANDBY LIQUID CONTROL SYSTEM CAPABILITY

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.023

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 BWRs", TVA-TR78-02A, 1978.

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

APPENDIX B

SAFETY/RELIEF VALVE CAPACITY AT SET POINT (NO./%): 12/77.6*

*Assumed one safety/relief valve out of service.
Reference pressure was 1105 + 1% psig.