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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BROWNS FERRY UNIT 1 RELOAD 5

GENERAL ELECTRIC

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FOR
BROWNS FERRY
UNIT 1, RELOAD 5

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GENERAL ELECTRIC

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

A. Information for Sections 4 and 5 Provided by the Tennessee Valley Authority	Appendix A
B. Plant Parameter Differences	Appendix B
C. Increased Core Flow, 105% Rated	Appendix C

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

Fuel Type	Cycle Loaded	Number	Number Drilled
Irradiated			
8DB274L	2	17	17
8DRB265H	3	4	4
P8DRB284L	4	231	231
P8DRB284L	5	220	220
P8DRB265L	5	36	36
GLTA-1**	5	2	2
GLTA-2**	5	2	2
New			
P8DRB284L	6	44	44
P8DRB284Z	6	8	8
P8DRB265H	6	164	164
P8DRB284L	6	36	36
Total		764	764

*( ) Refers to area of discussion in "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (latest approved revision); a letter "S" preceding the number refers to the appropriate country-specific supplement.

** Previously described in "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Power Plant Unit 1 Reload No. 4 (Cycle 5)," Y1003J01A19, Rev. 1 (Appendix E), September 1982.

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	19074 MWd/St
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	18751 MWd/St
Assumed reload cycle core average exposure at end of cycle:	18045 MWd/St
Core loading pattern:	Figure 1

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.2.2)

(REDY Events Only)

	EOC 6
Void Fraction (%)	39.2
Average Fuel Temperature (°F)	1273
Void Coefficient N/A* (¢/% Rg)	-7.10/-8.88
Doppler Coefficient N/A* (¢/°F)	-0.228/-0.217
Scram Worth	**

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

**Generic, exposure independent values are used as given in "General Electric Boiling Water Reactor Generic Reload Fuel Application", NEDE-24011-P-A-1, Amendment 10, April 1981.

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
BOC 6 to EOC 6							
P8x8R	1.20	1.52	1.40	1.051	6.411	114.6	1.28
8x8R	1.20	1.55	1.40	1.051	6.512	112.5	1.25
8x8	1.22	1.41	1.40	1.098	5.947	112.3	1.24

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.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

9. OPERATING FLEXIBILITY OPTIONS (S.2.2.3)

Single Loop Operation:	Yes
Load Line Limit:	No
Extended Load Line Limit:	No
Increased Core Flow:	Yes
Flow Point Analyzed:	105%
Feedwater Temperature Reduction:	No

*No credit for the thermal power monitor was used in the analysis.

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

Transient	Flux (% NBR)	Q/A (% NBR)	ΔCPR P8x8R	ΔCPR 8x8R	ΔCPR 8x8	Figure
Exposure: BOC 6 to EOC 6 Load Rejection Without Bypass	611	123	0.21	0.18	0.17	2
Exposure: BOC to EOC Loss of Feedwater Heater	123	123	0.14	0.14	0.14	3
Exposure: BOC 6 to EOC 6 Feedwater Controller Failure	398	121	0.16	0.15	0.14	4

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

(Generic Bounding Analysis Results)

Rod Block Reading	ΔCPR (all fuel types)
104	0.13
105	0.16
106	0.19
107	0.22
108	0.28
109	0.32
110	0.36

Setpoint Selected: 106

12. CYCLE MCPR VALUES (S.2.2)

Non-Pressurization Events

Exposure Range: BOC to EOC

	P8x8R	8x8R	8x8
Loss of Feedwater Heater	1.21	1.21	1.21
Fuel Loading Error	1.25		
Rod Withdrawal Error	1.26	1.26	1.26

Pressurization Events

Exposure Range: BOC 6 to EOC 6

	Option A			Option B		
	P8x8R	8x8R	8x8	P8x8R	8x8R	8x8
Load Rejection Without Bypass	1.34	1.30	1.29	1.24	1.22	1.21
Feedwater Controller Failure	1.28	1.27	1.26	1.25	1.24	1.23

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1220	1257	Figure 5

14. STABILITY ANALYSIS RESULTS (S.2.4)

Rod Line Analyzed: 105%

Decay Ratio:	Figure 6
Reactor Core Stability Decay Ratio, x_2/x_0:	0.77
Channel Hydrodynamic Performance Decay Ratio, x_2/x_0:	
Channel Type	
P8x8R/8x8R	0.29
8x8	0.38

15. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

Event	Initial MCPR	Resulting MCPR
Misoriented	1.23	1.07

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

Bounding Analysis Results:

Doppler Reactivity Coefficient:	Figure 7
Accident Reactivity Shape Functions:	Figures 8 and 9
Scram Reactivity Functions:	Figures 10 and 11

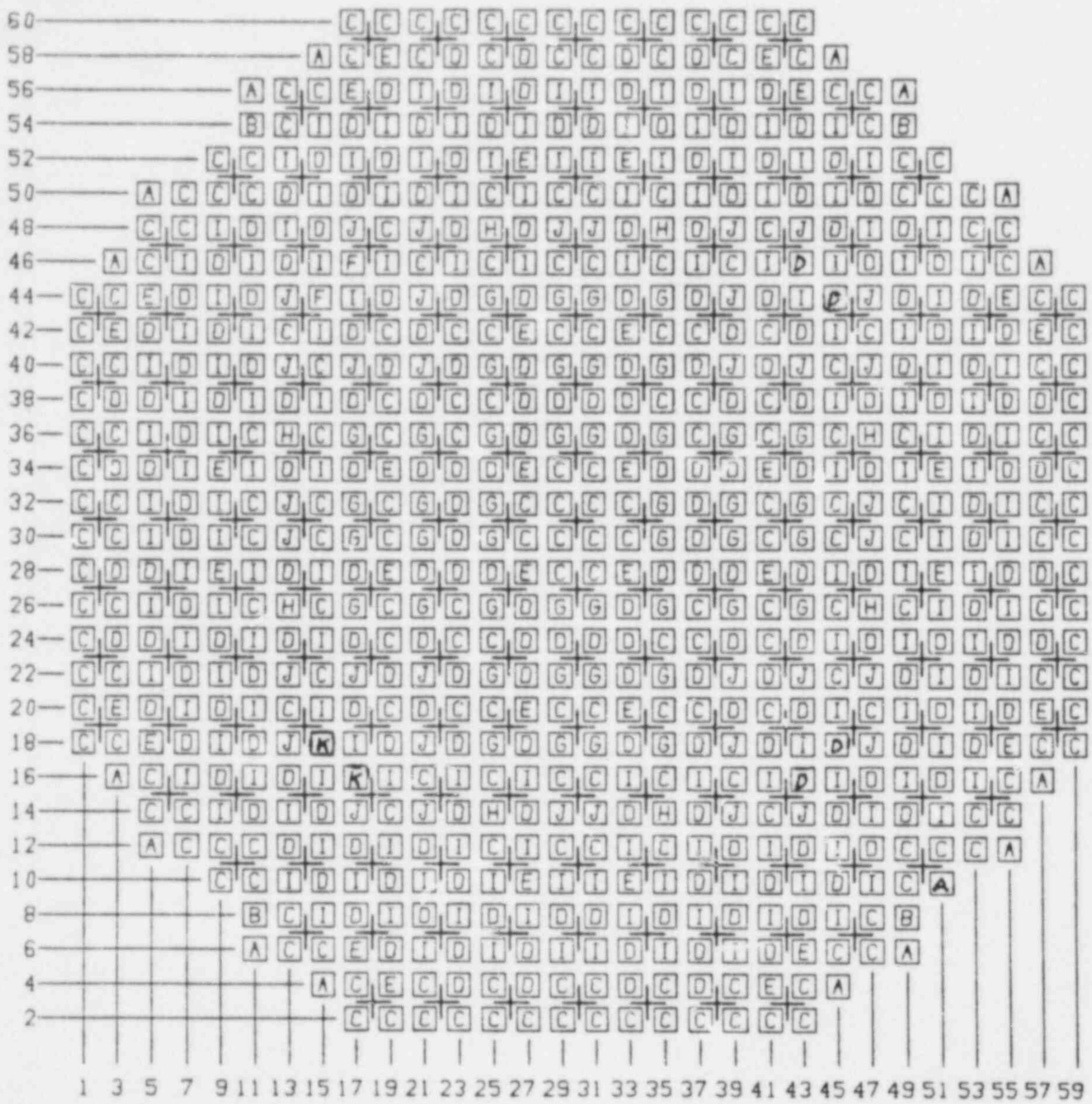
Plant Specific Analysis Results:

Parameter(s) not Bounded, Cold:	Accident Reactivity Scram Reactivity
Resultant Peak Enthalpy, Cold:	167.2
Parameter(s) not Bounded, HSB:	Accident Reactivity
Resultant Peak Enthalpy, HSB:	243.7

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

"Loss-of-Coolant Accident Analysis Report for Browns Ferry Unit 1", General Electric Company, NEDO-24056, Rev. 1, May 1983.

LIST OF FIGURES



FUEL TYPE	
A = 8DB274L B = 8DRB265H C = P8DRB284L D = P8DRB284L E = P8DRB265L	F = GLTA-1 G = P8DRB284L H = P8DRB284Z I = P8DRB265H J = P8DRB284L K = GLTA-2

Figure 1. Reference Core Loading Pattern

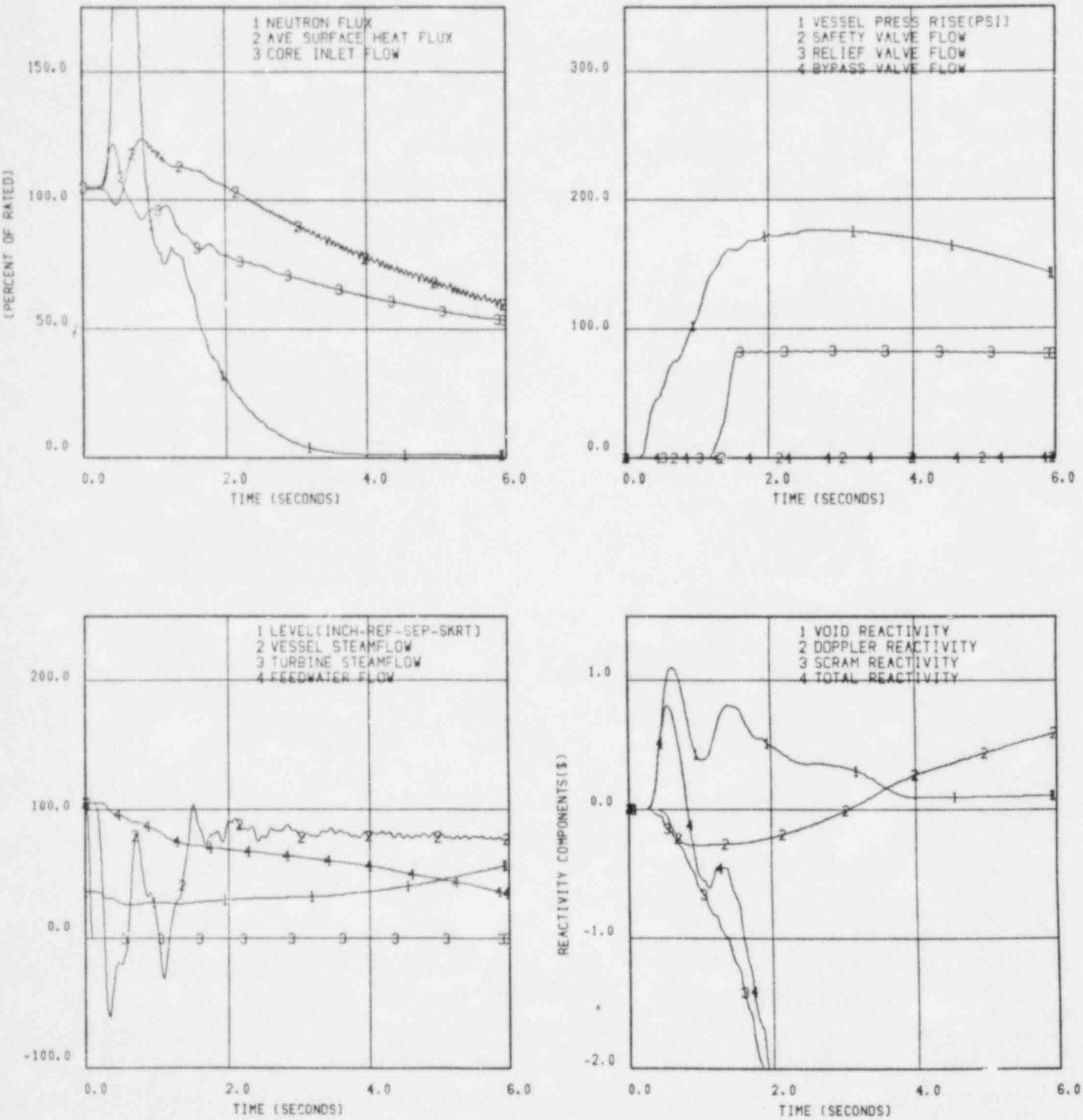


Figure 2. Plant Response to Generator Load Rejection, Without Bypass

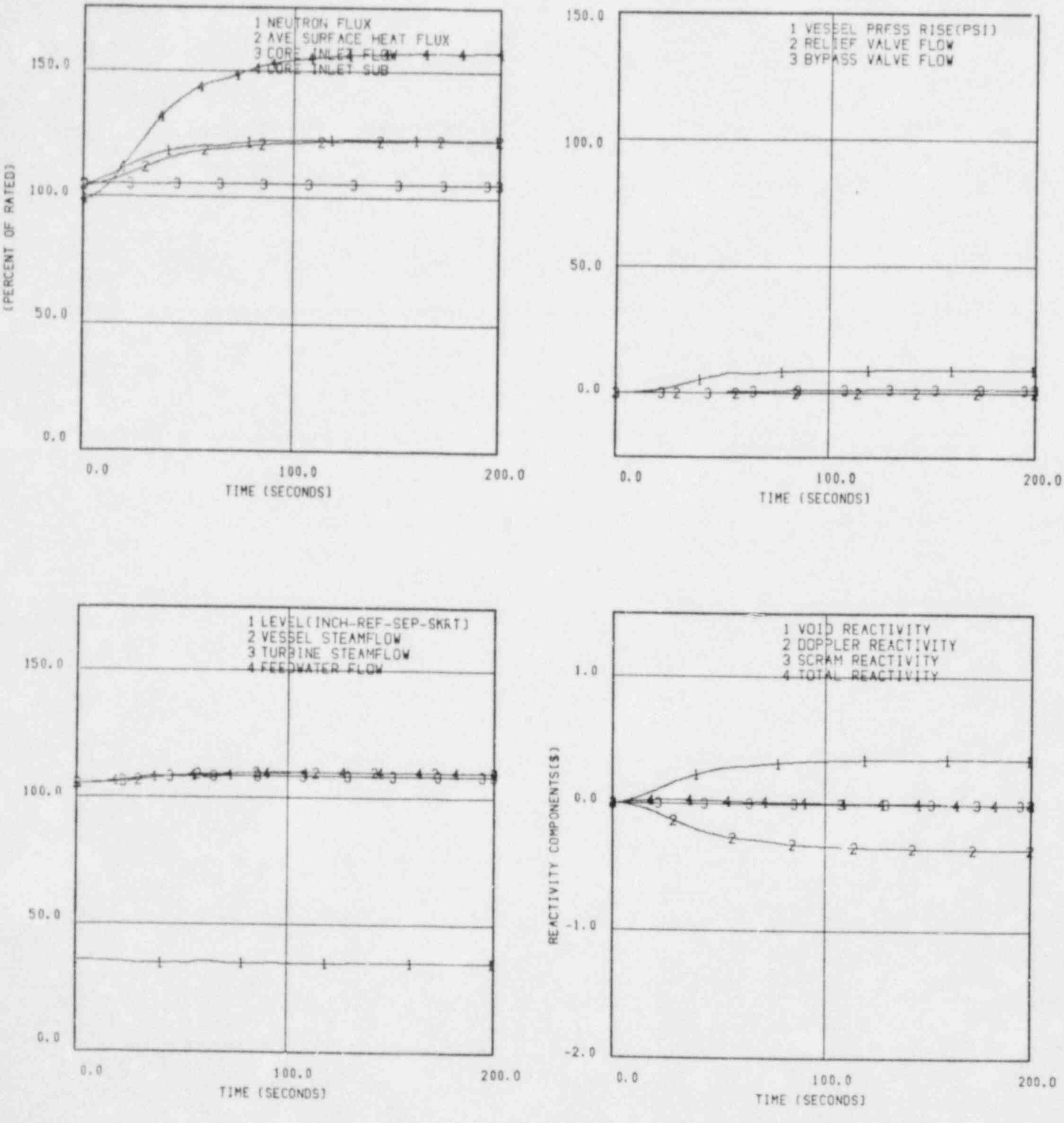


Figure 3. Plant Response to Loss of 100°F Feedwater Heating

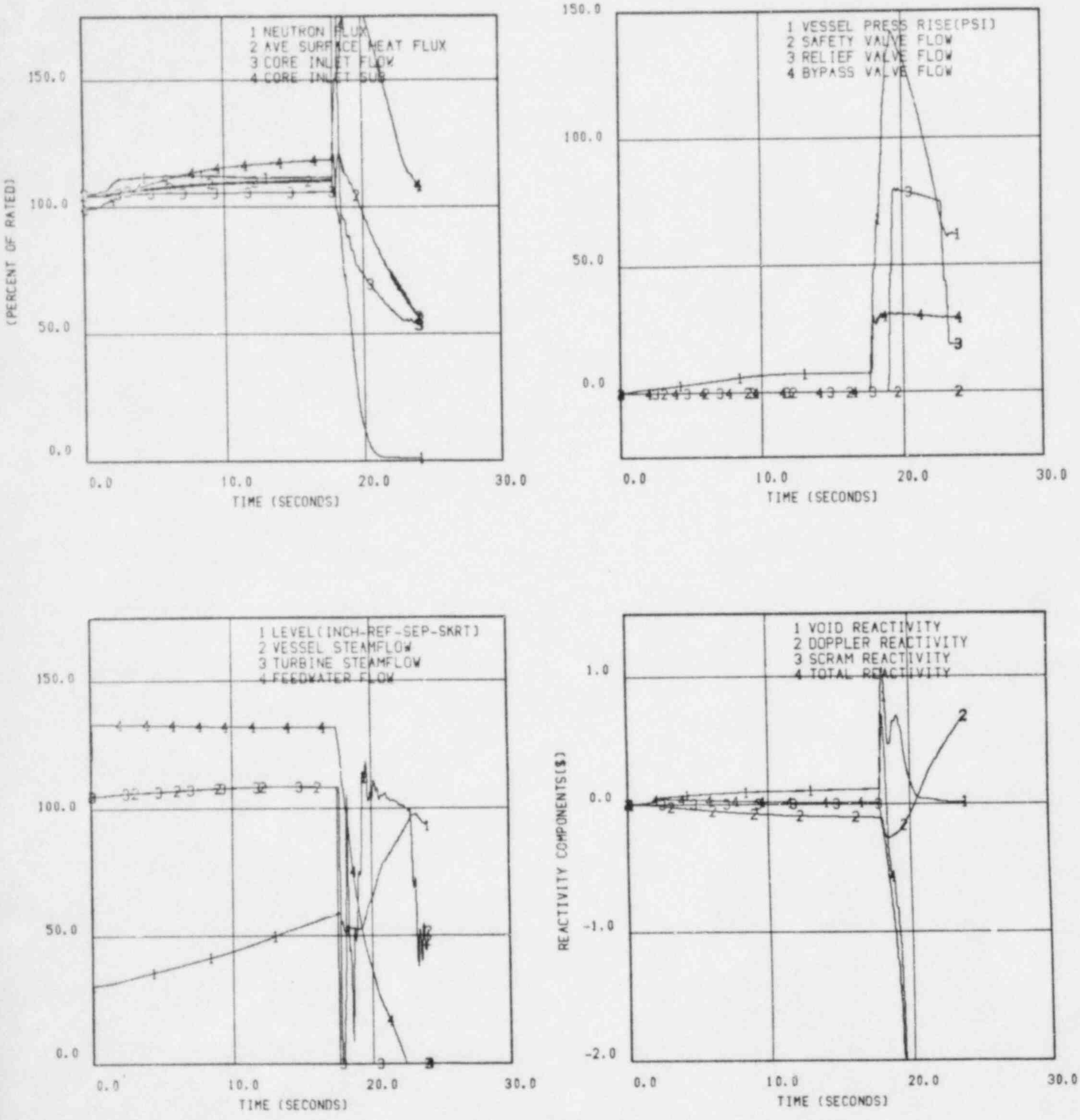


Figure 4. Plant Response to Feedwater Controller Failure

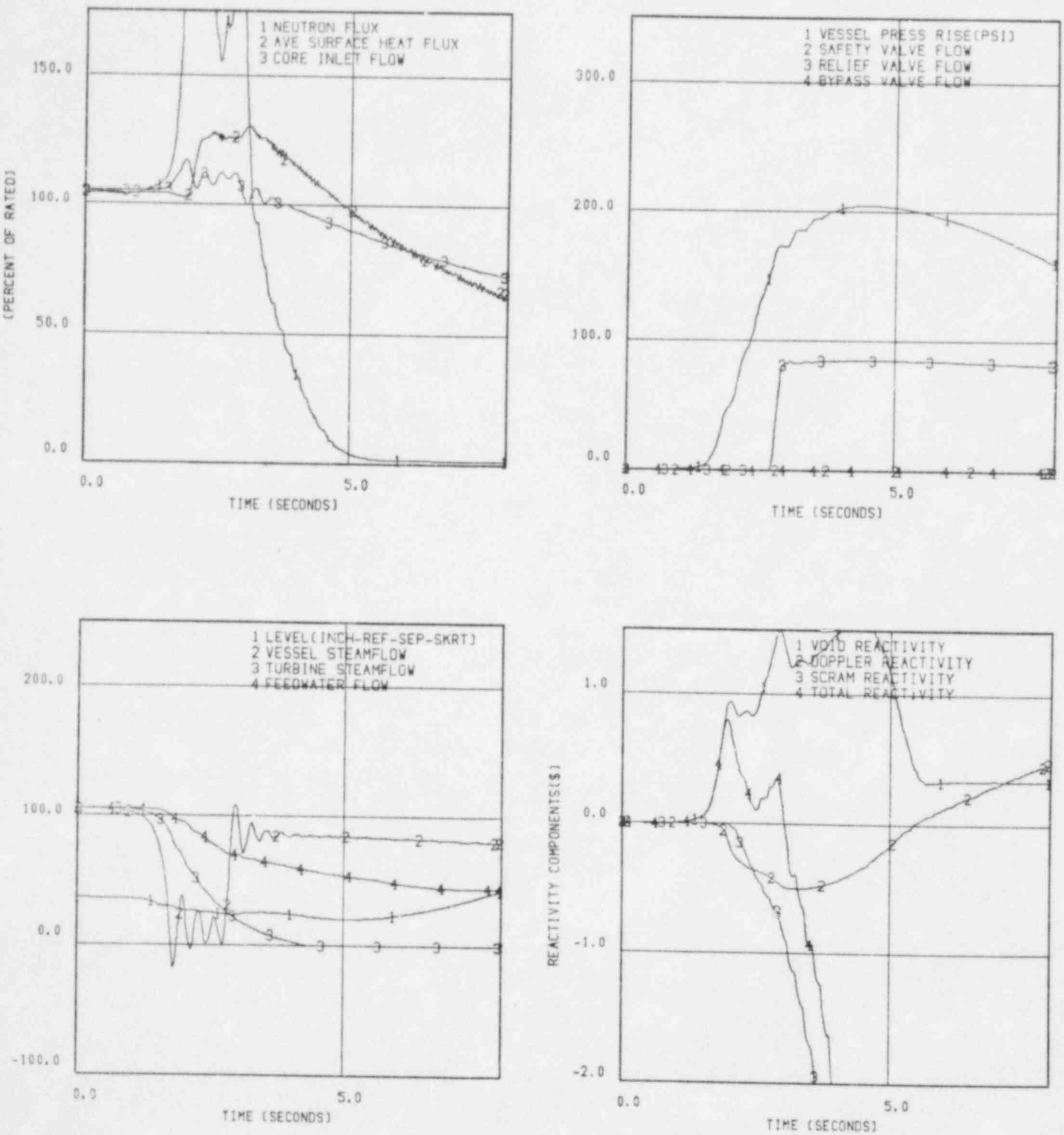


Figure 5. Plant Response to MSIV Closure

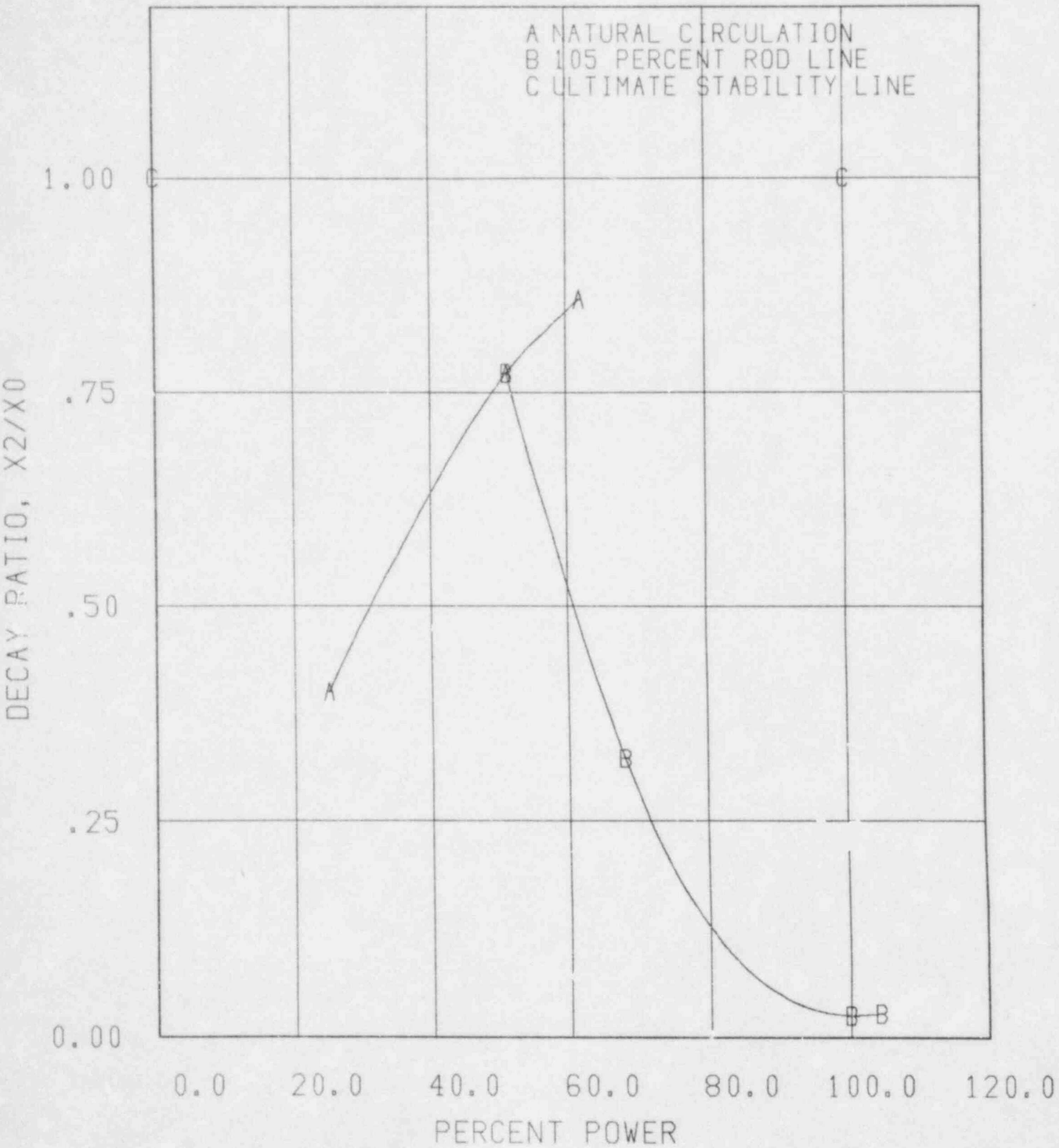


Figure 6. Reactor Core Decay Ratio

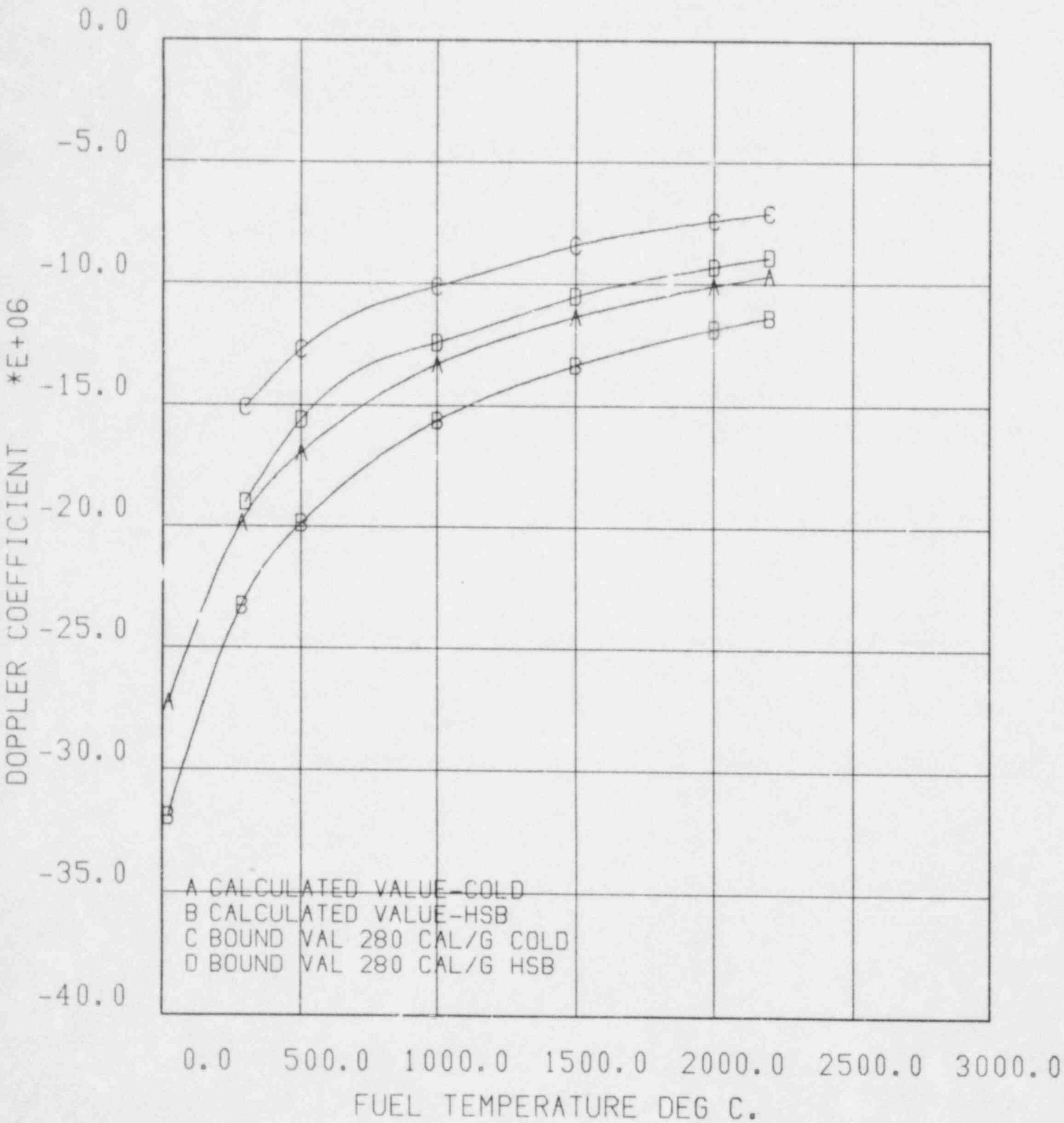


Figure 7. Fuel Doppler Coefficient in 1/Δ°C

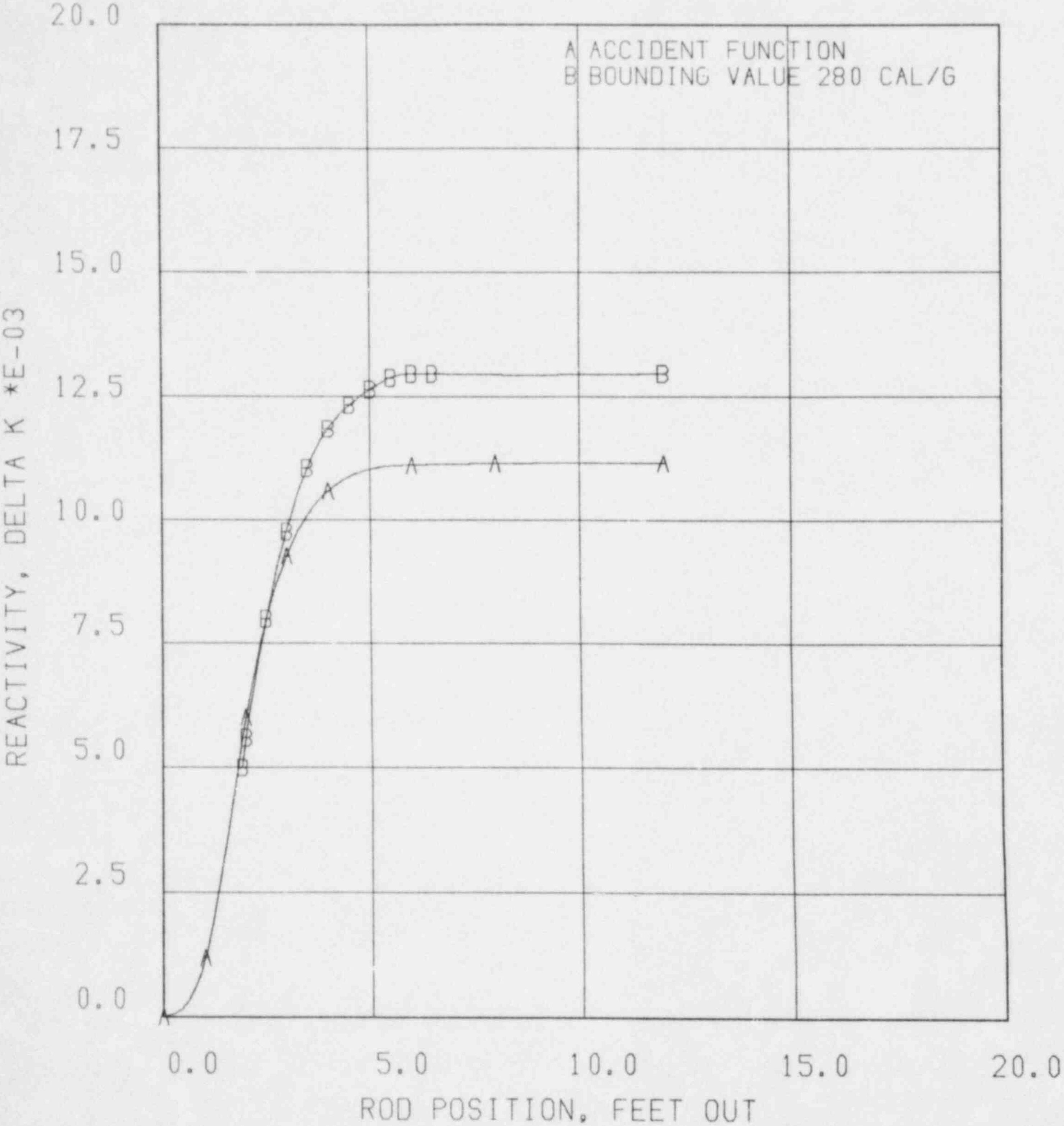


Figure 8. Accident Reactivity Shape Function, Cold Startup

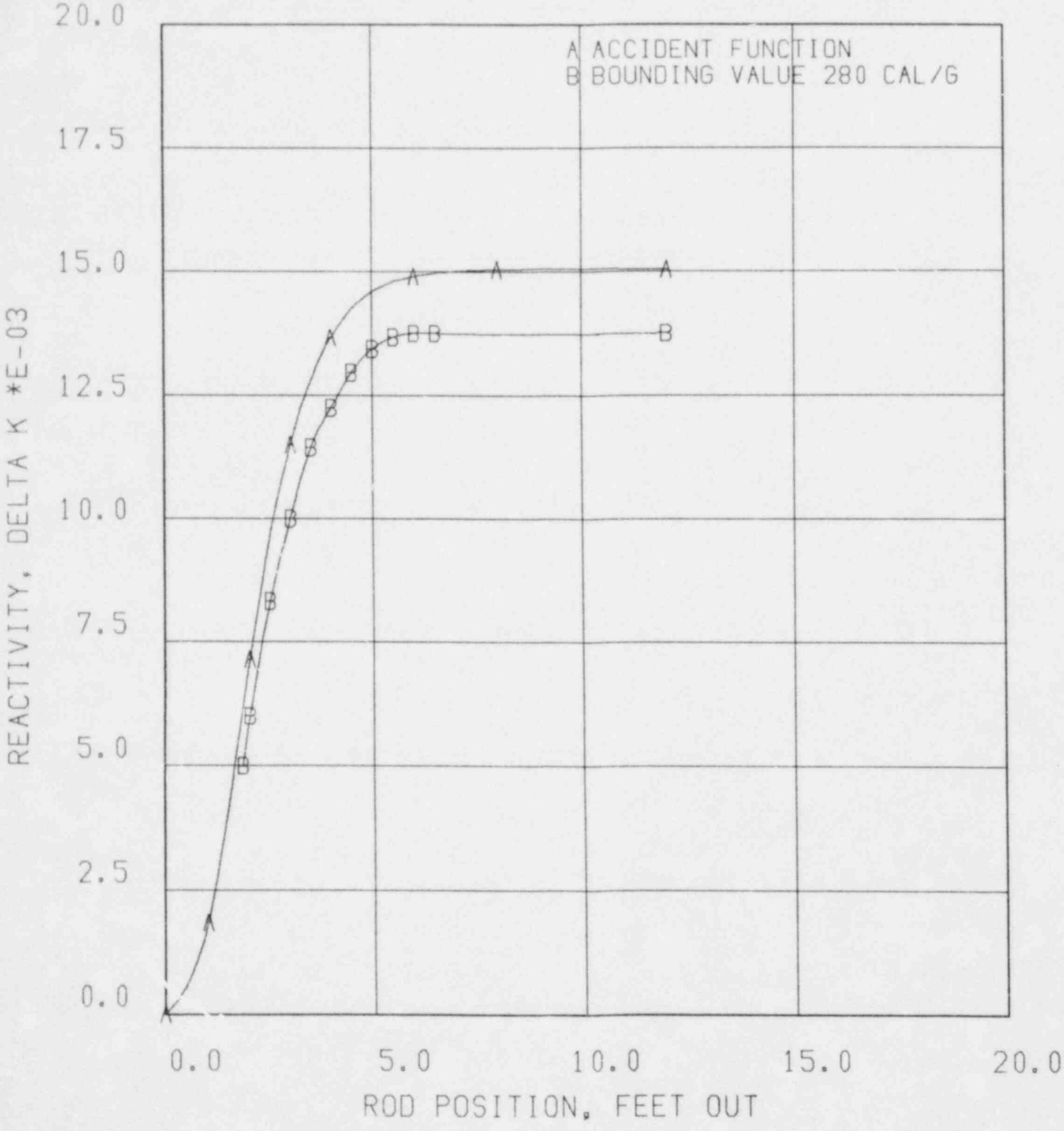


Figure 9. Accident Reactivity Shape Function, Hot Startup

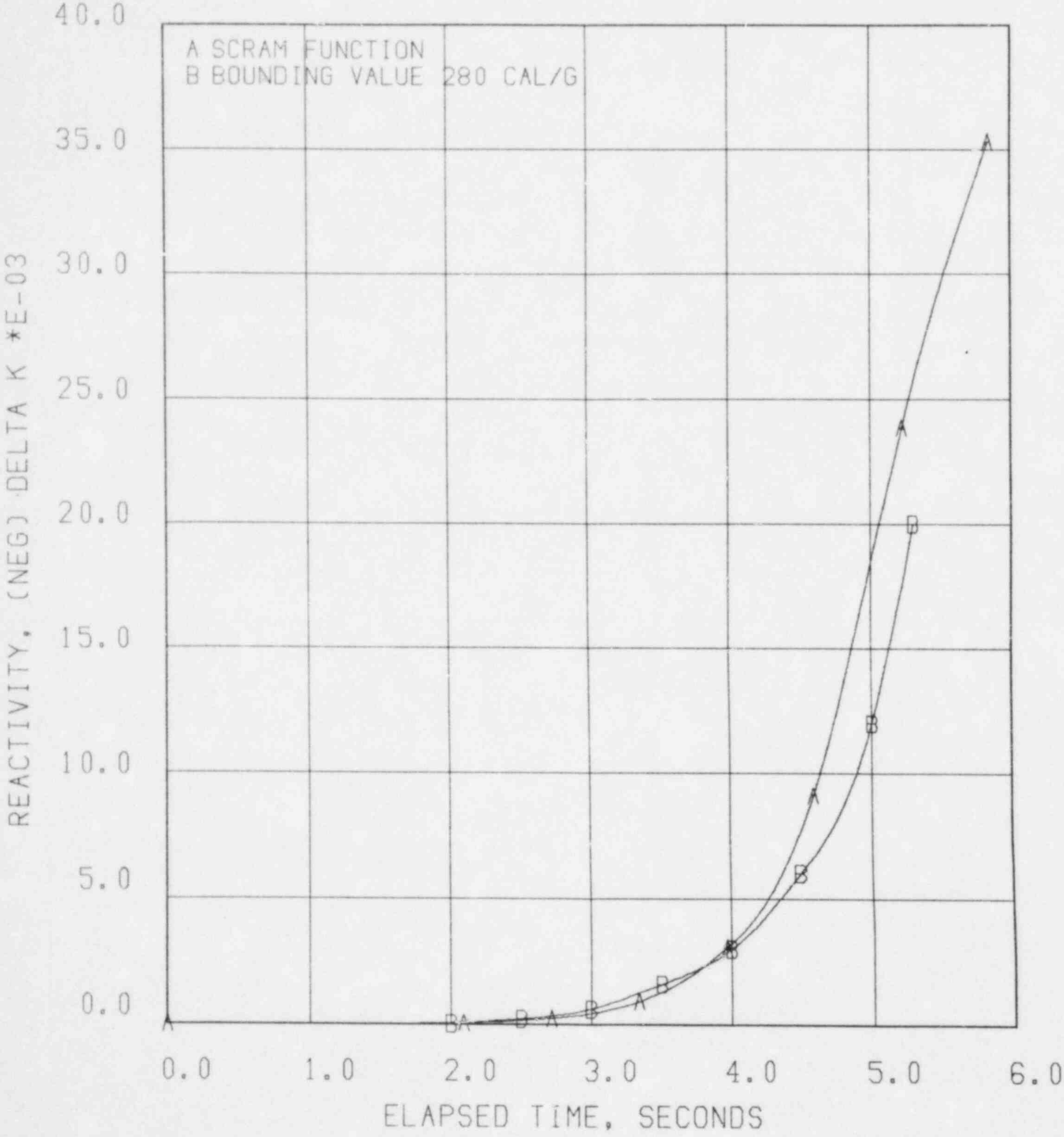


Figure 10. Scram Reactivity Function, Cold Startup

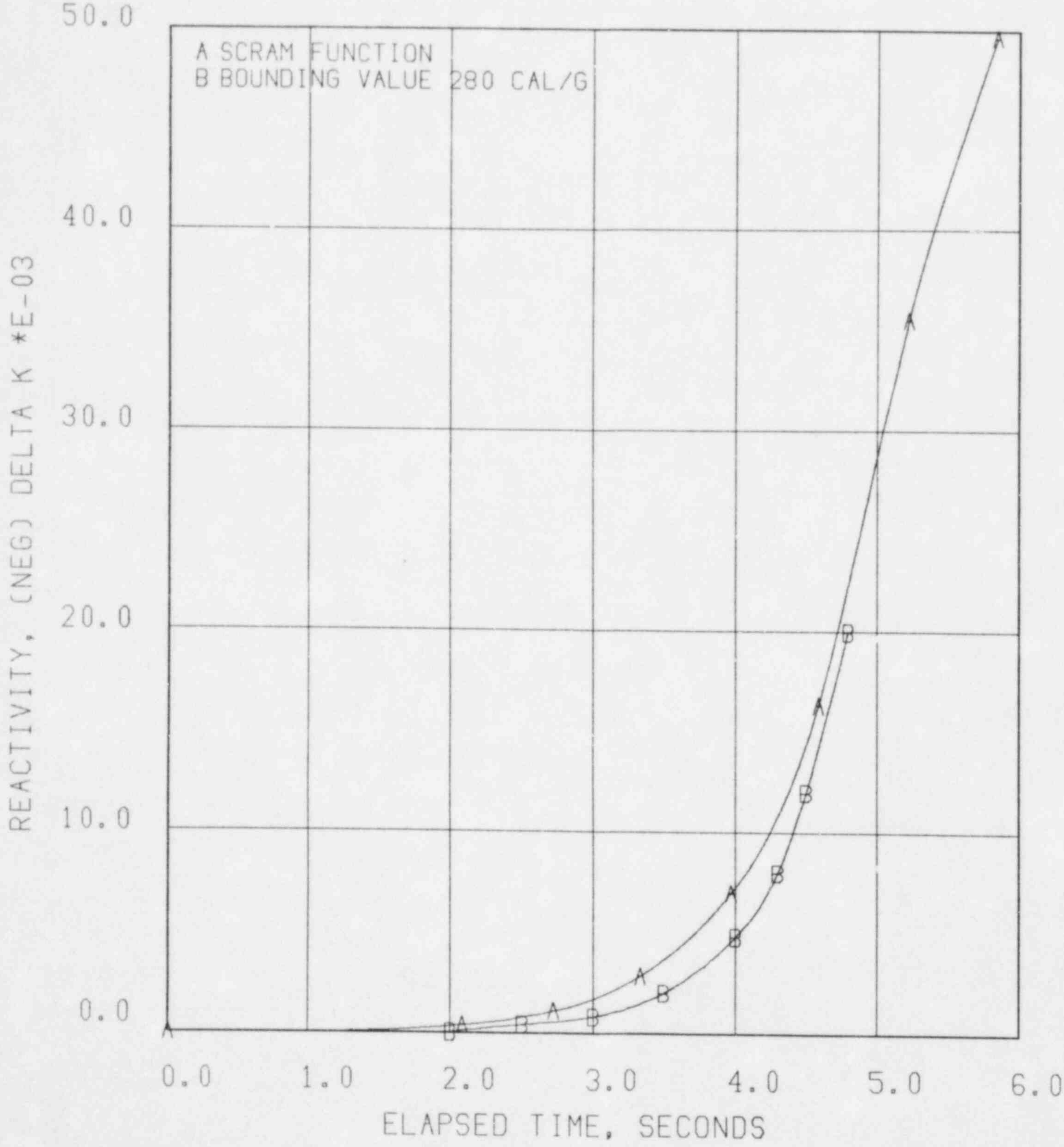


Figure 11. Scram Reactivity Function, Hot Startup

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 (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 6 core average exposure corresponding to the minimum expected End-of-Cycle 5 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:

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

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 1.5% 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. The results of the SLCS evaluation are given in Table A-2.

Table A-1

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

Uncontrolled, K_{eff}^{UNC}	1.117
Fully Controlled, K_{eff}^{CON}	0.956
Strongest Control Rod Out, K_{eff}^{SRO}	0.982
R, Maximum Increase in Cold Core Reactivity With Exposure Into Cycle, Δk	0.003

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

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

APPENDIX B

PLANT PARAMETER DIFFERENCES

Only 12 of the 13 safety/relief valves were considered operable. The capacity was 78.1% at a reference pressure of 1123 psig.

APPENDIX C
INCREASED CORE FLOW

The licensing analyses for Cycle 6 were done with a core flow of 105% of rated flow which will bound operation at rated conditions.

The conclusions regarding LOCA analysis, reactor internals pressure drop, and flow-induced vibration as discussed in Reference C-1 are applicable to Cycle 6.

The flow-biased instrumentation for the rod block monitor should be signal clipped for a setpoint of 106% since flow rates higher than rated would otherwise result in a ΔCPR higher than reported for the rod withdrawal error.

REFERENCE:

C-1. "Safety Review of Browns Ferry Nuclear Plant Unit No. 1 at Core Flow Conditions Above Rated Flow During Cycle 5", NEDO-22135, May 1982.