23A1739
CLASS I
JUNE 1984

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR DUANE ARNOLD ATOMIC ENERGY CENTER UNIT 1, RELOAD 7

GENERAL ELECTRIC

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FOR
DUANE ARNOLD ATOMIC ENERGY CENTER
UNIT 1, RELOAD 7

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1. PLANT UNIQUE ITEM (1.0)*

Appendix A: Transient Analysis Input Param ters

Appendix B: Feedwater Controller Failure

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

Fuel Type	Cycle Loaded	Number	Number Drilled
Irradiated			
P8DPB289	5	36	36
P8DPB289	6	84	84
P8DRB299	7	40	40
P8DRB284H	7	88	88
New			
BP8DRB299	8	64	64
BP8DRB301L	8	56	56
Total		368	368

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	17954 MWD/ST
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	17754 MWD/ST
Assumed reload cycle core average exposure at end of cycle:	17499 MWD/ST
Core loading pattern:	Figure 1

*( ) Refers to area of discussion in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-6, April 1983. A letter "S" preceding the number refers to the appropriate section in the United States Supplement, NEDE-24011-P-A-6-US, April 1983.

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

Beginning of Cycle, k_{eff}	
Uncontrolled	1.112
Fully Controlled	0.960
Strongest Control Rod Out	0.987
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, Δk	0.003

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δ) (20°C, Xenon Free)
600	0.030

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

(COLD WATER INJECTION EVENTS ONLY)

Void Fraction (%)	41.8
Average Fuel Temperature (°F)	1280
Void Coefficient N/A* (¢/% Rg)	-9.11/-11.39
Doppler Coefficient N/A (¢/°F)	-0.225/-0.214
Scram Worth N/A ($)	**

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: BOC to EOC							
BP/P8x8R	1.20	1.47	1.40	1.051	6.493	112.9	1.23

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

**Generic exposure dependent values are used as given in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-6, April 1983.

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

Transient Recategorization:	No
Recirculation Pump Trip:	Yes
Rod Withdrawal Limiter:	No
Thermal Power Monitor:	No
Improved Scram Time:	Yes (ODYN Option B)
Exposure Dependent Limits:	No
Exposure Points Analyzed:	EOC

9. OPERATING FLEXIBILITY OPTIONS (S.2.2.3)

Single Loop Operation:	Yes
Load Line Limit:	Yes
Extended Load Line Limit:	Yes
Increased Core Flow:	No
Flow Point Analyzed:	N/A
Feedwater Temperature Reduction:	No

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

Exposure: BOC to EOC

Transient	Flux (% NBR)	Q/A (% NBR)	ΔCPR BP/P8x8R	Figure
Load Rejection Without Bypass	4[illegible]8	113	0.16	2
Loss of Feedwater Heater	121	117	0.14	3
Feedwater Controller Failure*	289	111	0.12	4

*See Appendix B

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: 105

12. CYCLE MCPR VALUES (S.2.2.)

Non-Pressurization Events

Exposure Range: BOC to EOC

	BP/P8x8R
Loss of Feedwater Heater	1.21
Fuel Loading Error	1.26*
Rod Withdrawal Error	1.23

Pressurization Events

Exposure Range: BOC to EOC

	Option A	Option B
	BP/P8x8R	BP/P8x8R
Load Rejection Without Bypass	1.28	1.20
Feedwater Controller Failure	1.24	1.21

*Includes a 0.02 penalty due to variable water gap R-factor uncertainty.

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1244	1275	Figure 5

14. STABILITY ANALYSIS RESULTS (S.2.4)

Rod Line Analyzed: Extrapolated Rod Block Line

Decay Ratio: Figure 6

Reactor Core Stability Decay Ratio, x_2/x_0: 0.84

Channel Hydrodynamic Performance Decay Ratio, x_2/x_0

Channel Type

BP/P8x8R 0.31

15. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

Event	Initial MCPR	Resulting MCPR
Misoriented	1.23	1.06

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

Bounding Analysis Results:

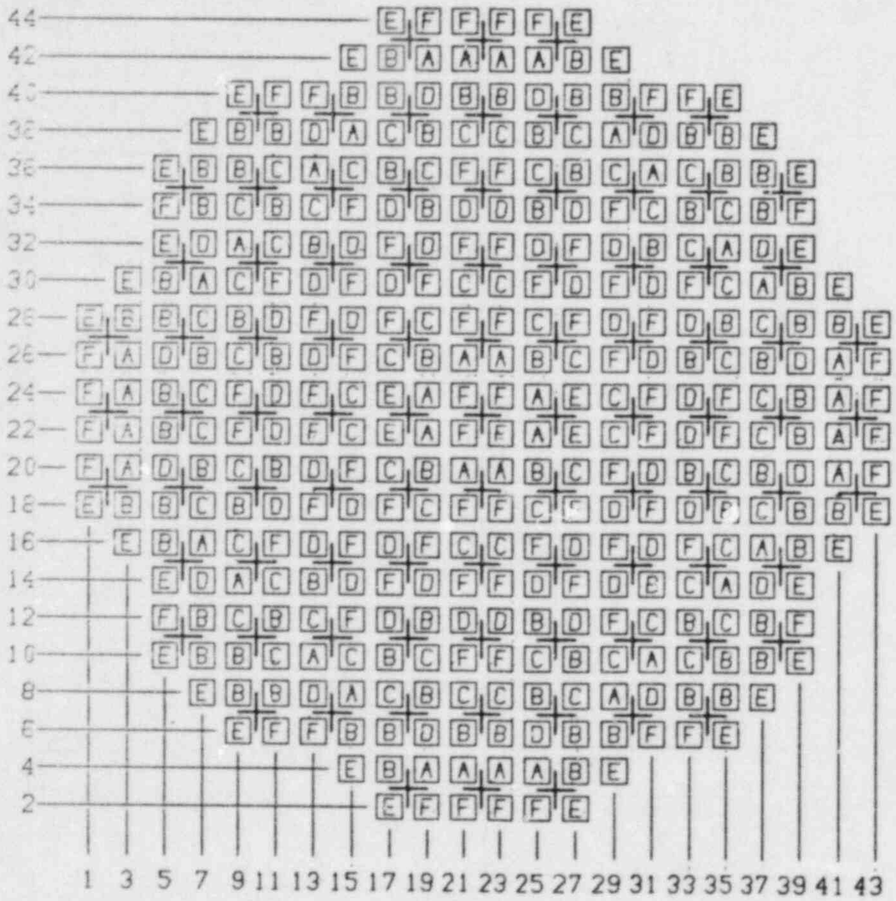
Doppler Reactivity Coefficient	Figure 7
Accident Reactivity Shape Functions:	Figures 8 and 9
Scram Reactivity Functions:	Figures 10 and 11

Plant Specific Analysis Results:

Parameter(s) not Bounded, Cold:	None
Resultant Peak Enthalpy, Cold:	N/A
Parameter(s) not Bounded, HSB:	Accident Reactivity
Resultant Peak Enthalpy, HSB:	274.8

17. <u>LOSS-OF-COOLANT ACCIDENT RESULT (S.2.5.2)</u>

See "Loss-of-Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant)," June 1984 (NEDO-21080-03-1A).



FUEL TYPE	
A = P8DRB299 B = P8DRB284H C = BP8DRB299	D = BP8DRB301L E = P8DPB289 F = P8DPB289

Figure 1. Reference Core Loading Pattern

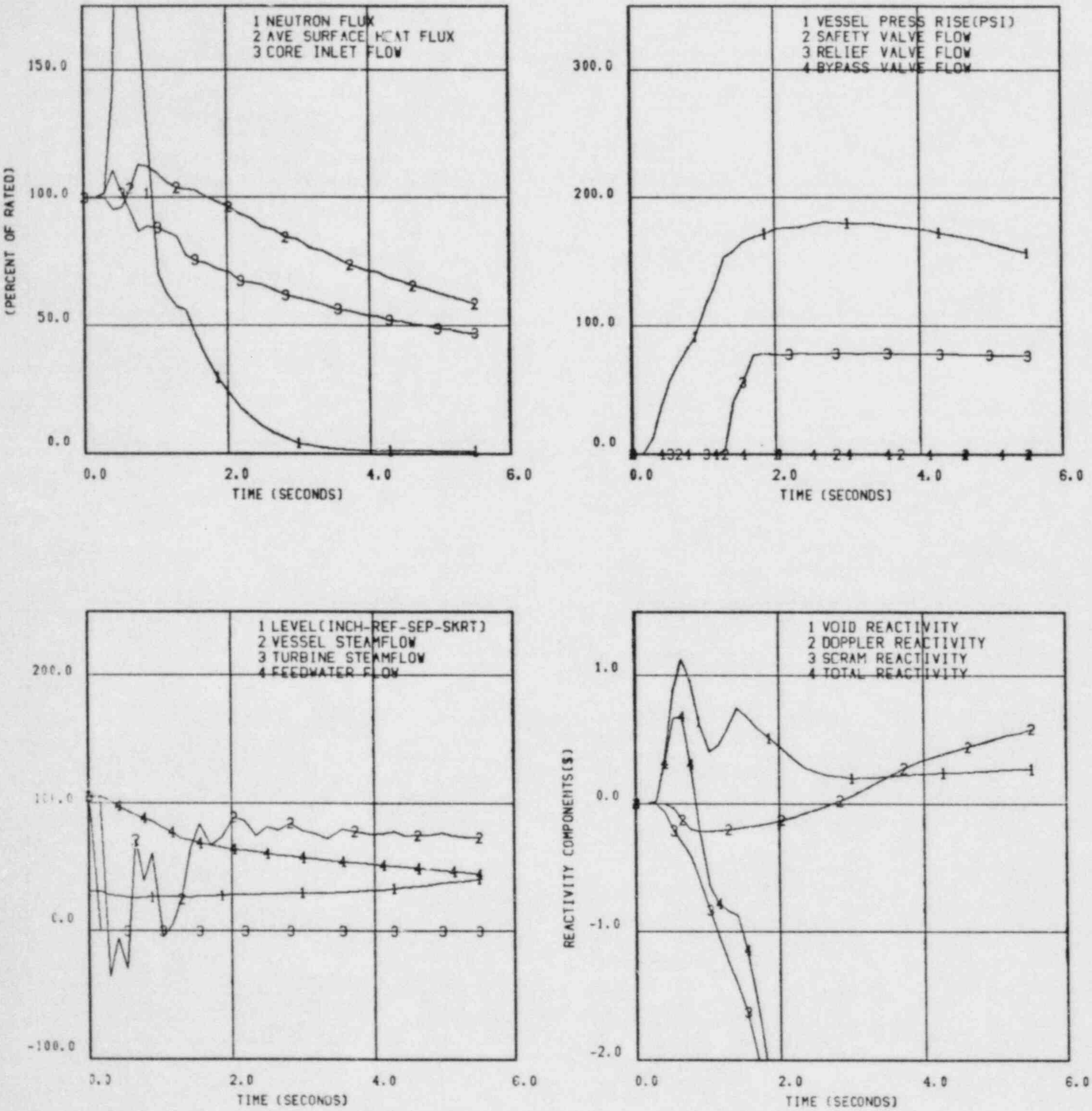


Figure 2. Plant Response to Generator Load Rejection Without Bypass

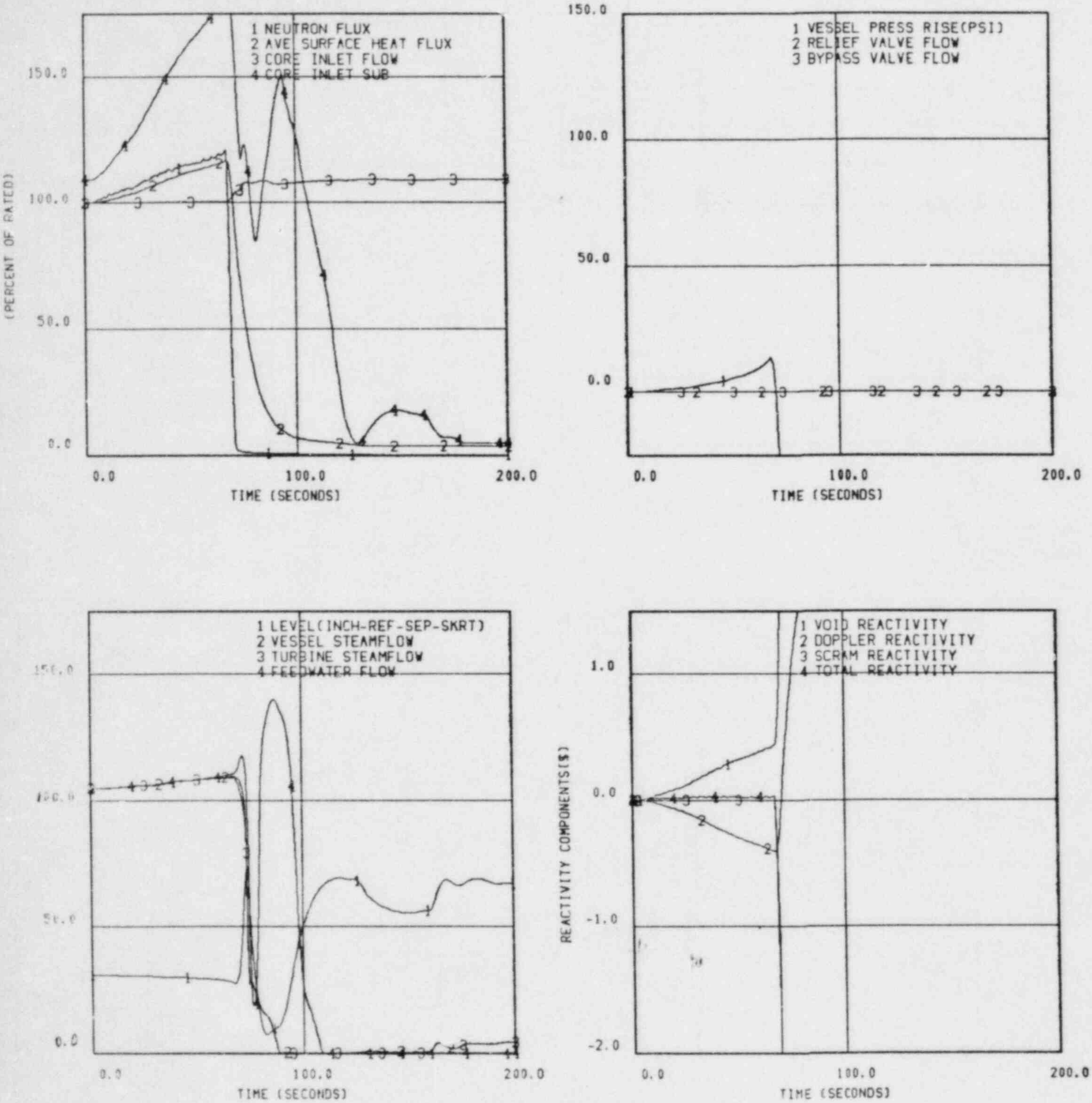


Figure 3. Plant Response to Loss of 100°F Feedwater Heating

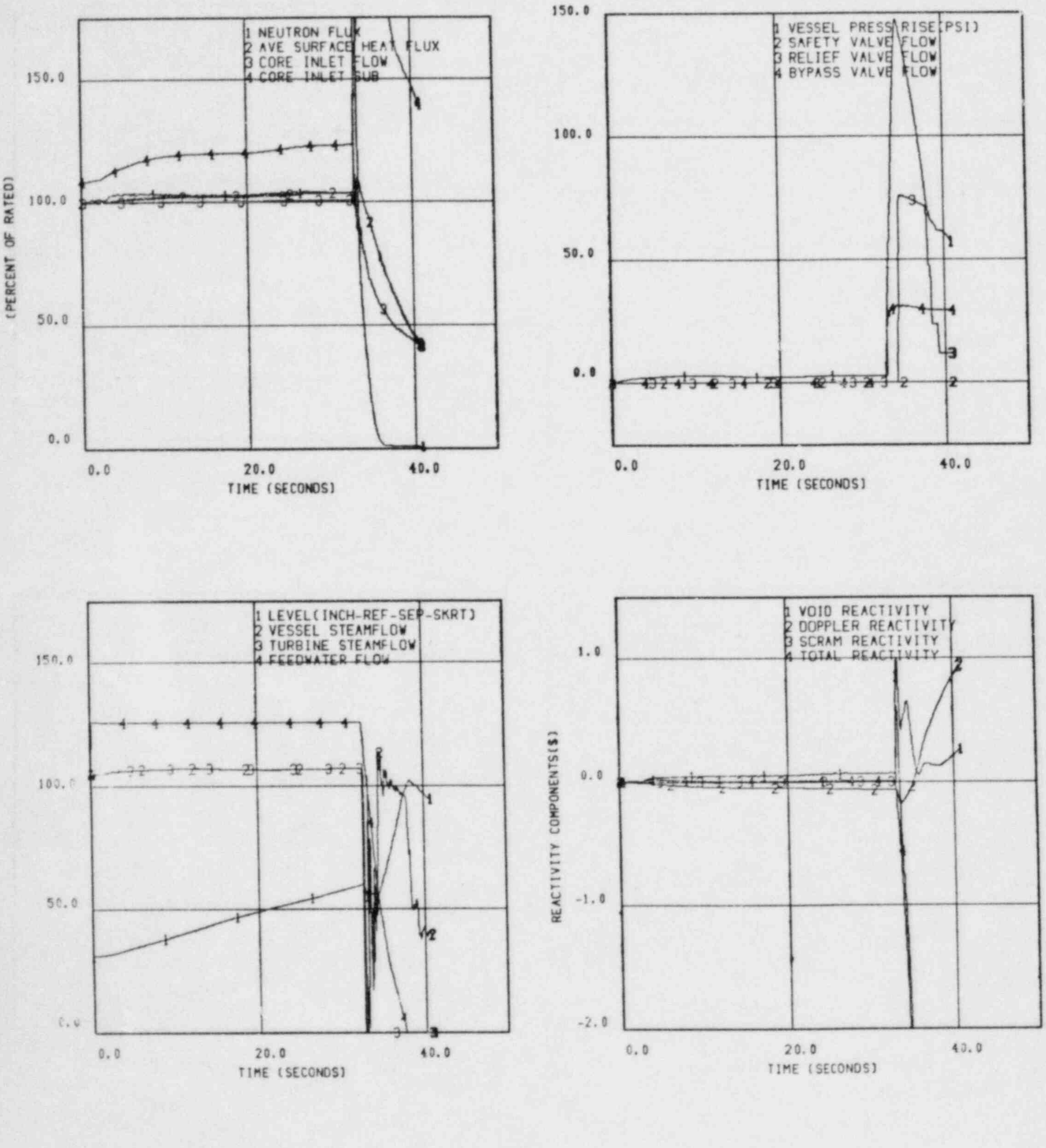


Figure 4. Plant Response to Feedwater Controller Failure

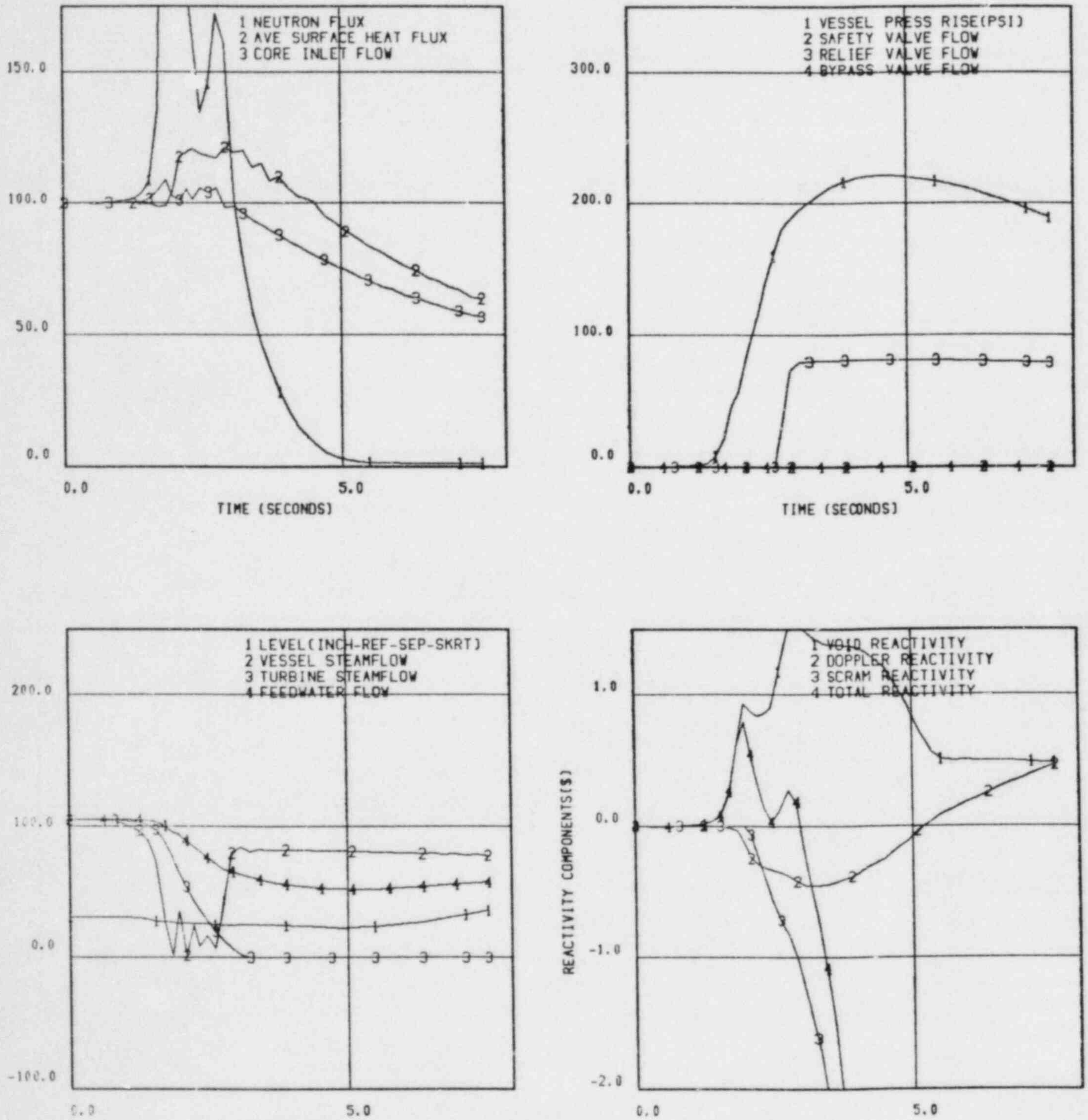


Figure 5. Plant Response to MSIV Closure (Flux Scram)

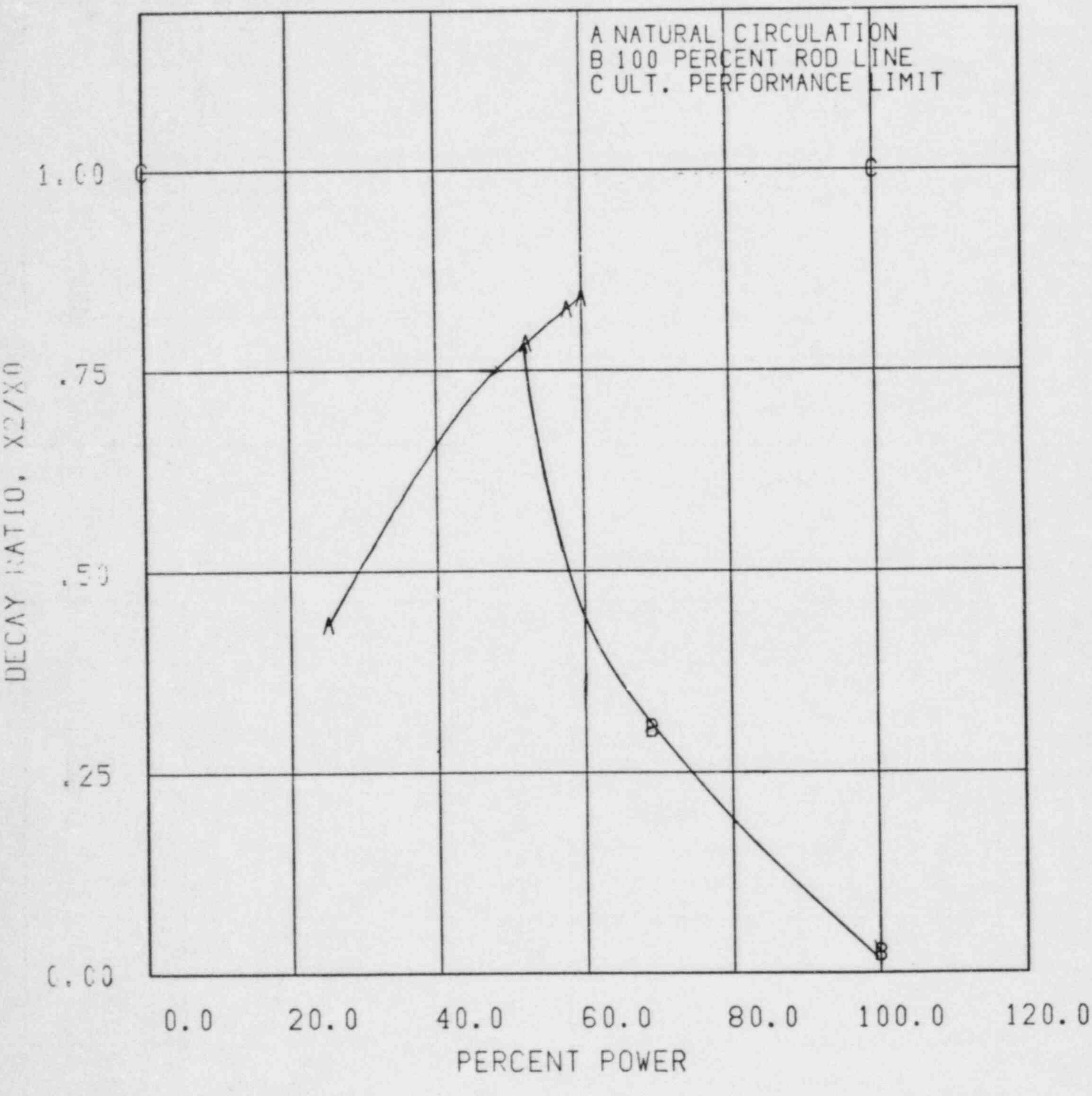


Figure 6. Reactor Core Decay Ratio

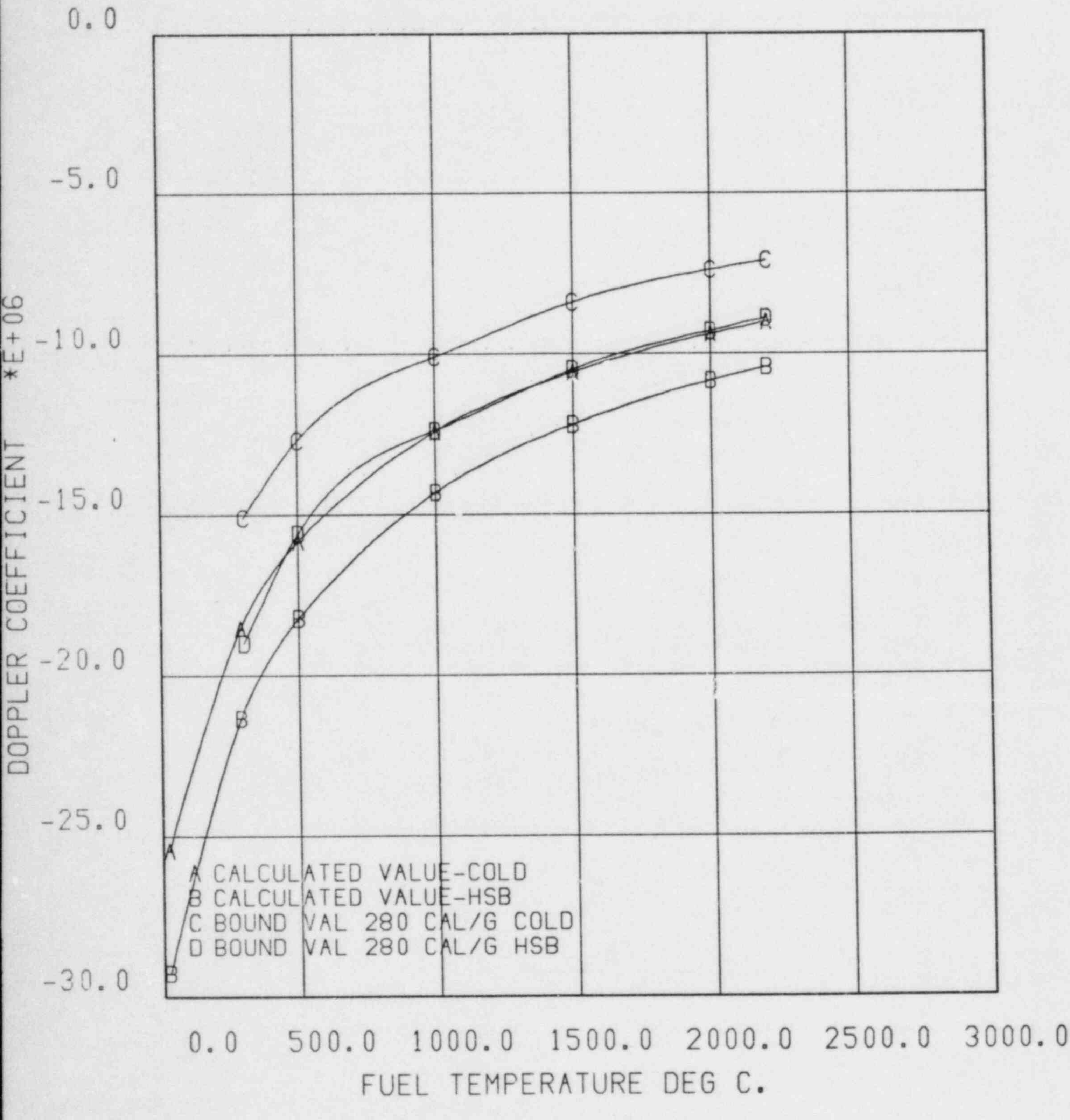


Figure 7. Fuel Doppler Coefficient in 1/Δ°C

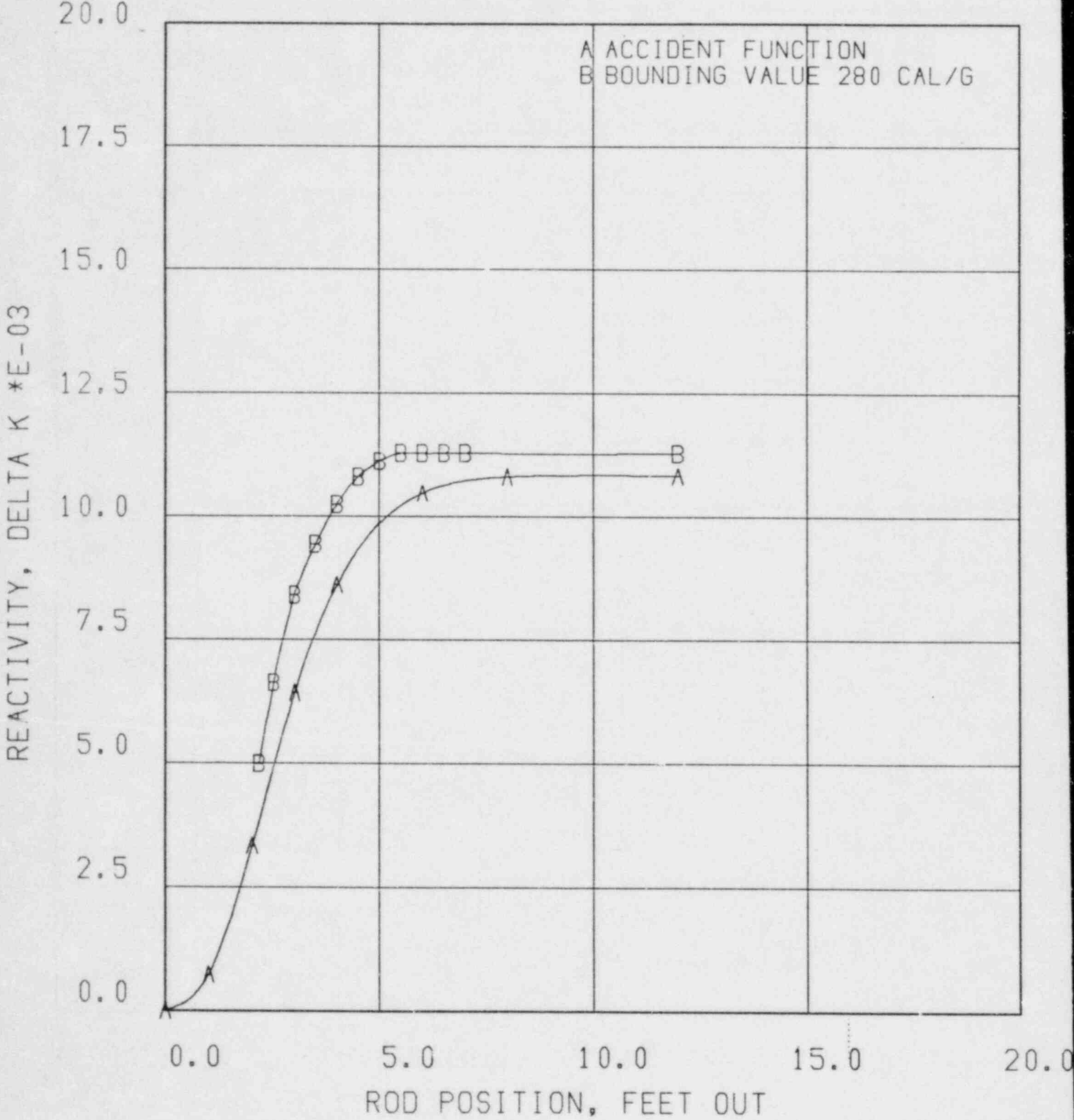


Figure 8. Accident Reactivity Shape Function (Cold Startup)

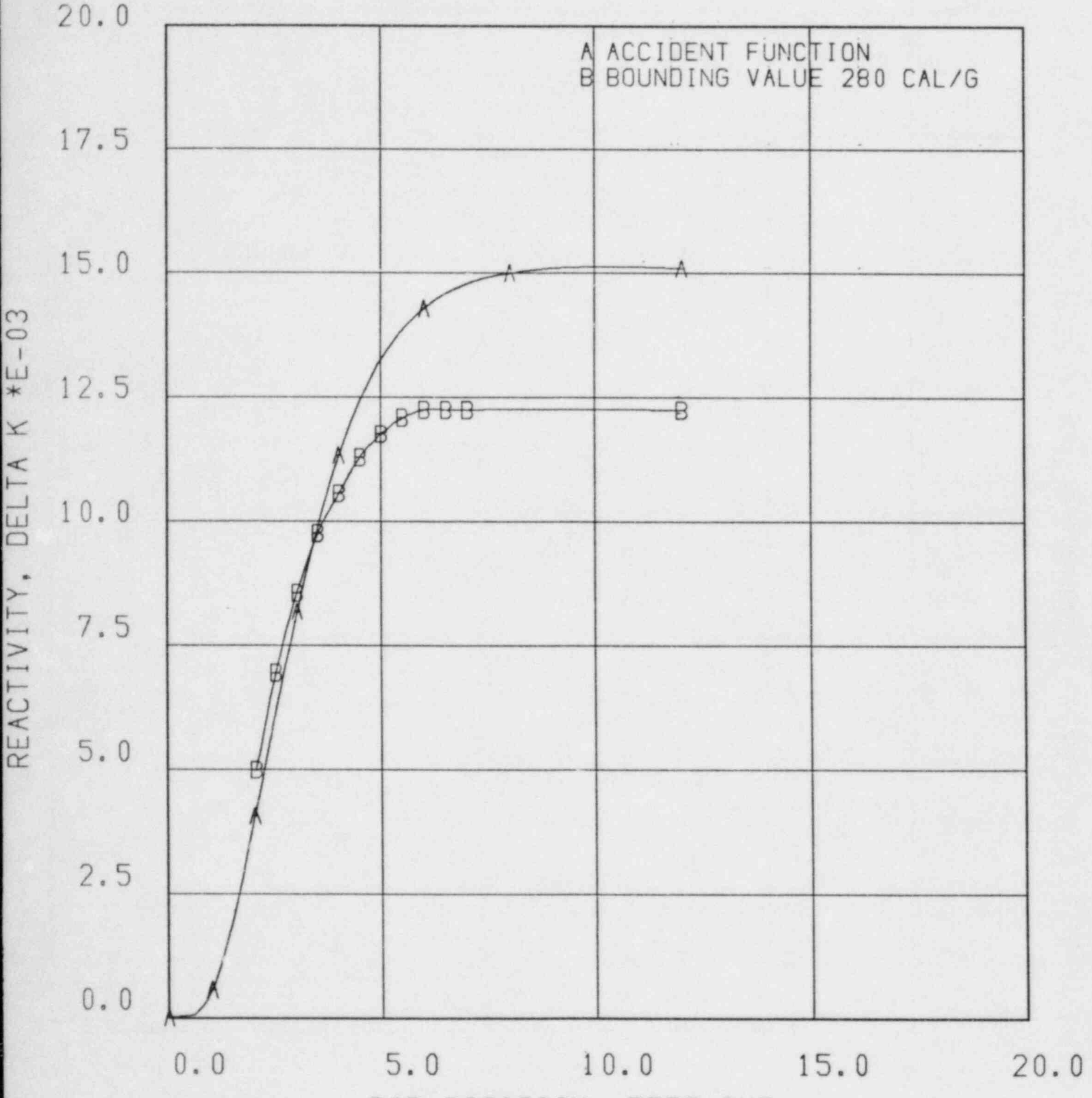


Figure 9. Accident Reactivity Shape Function (Hot Startup)

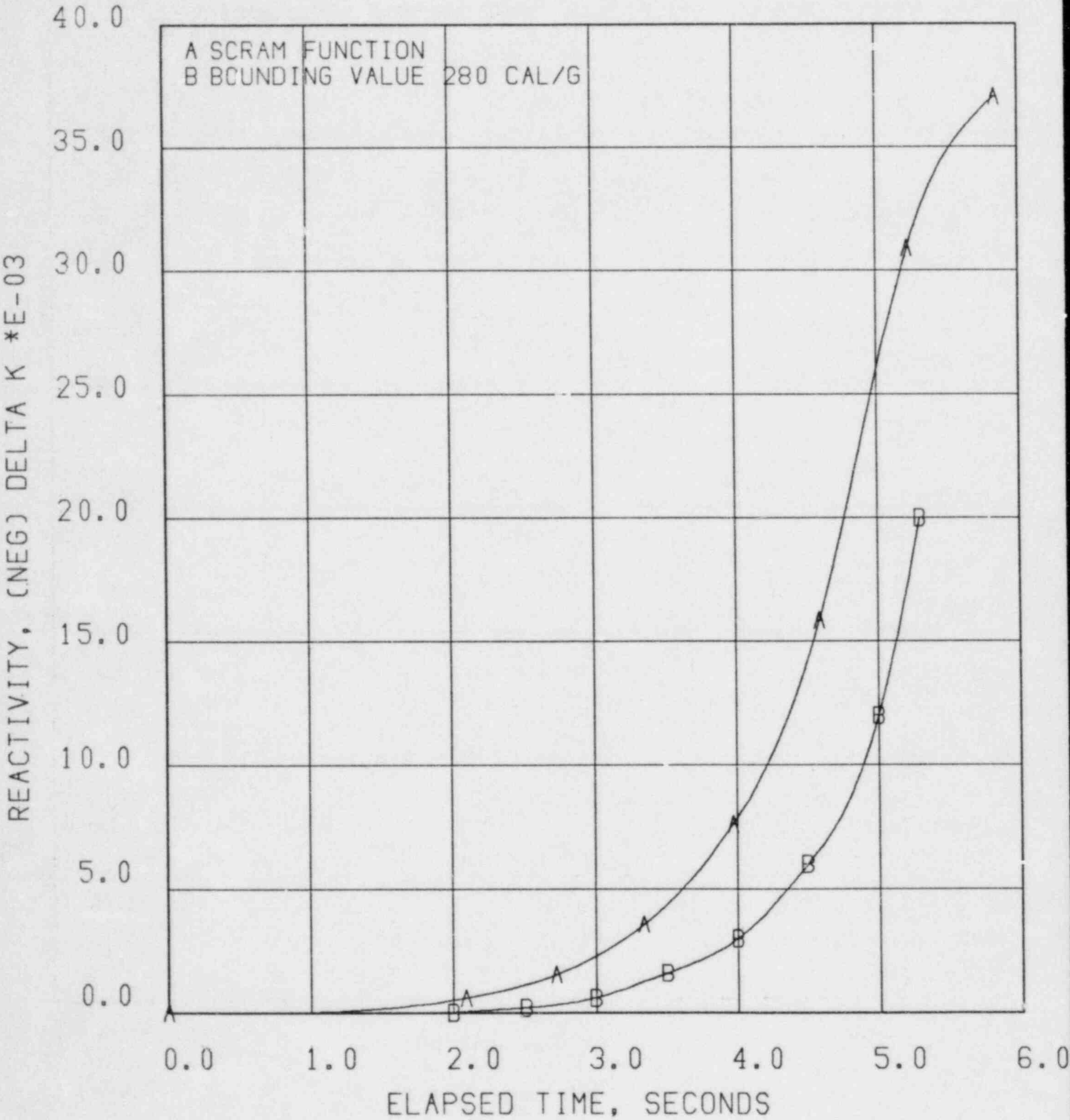


Figure 10. Scram Reactivity Function (Cold Startup)

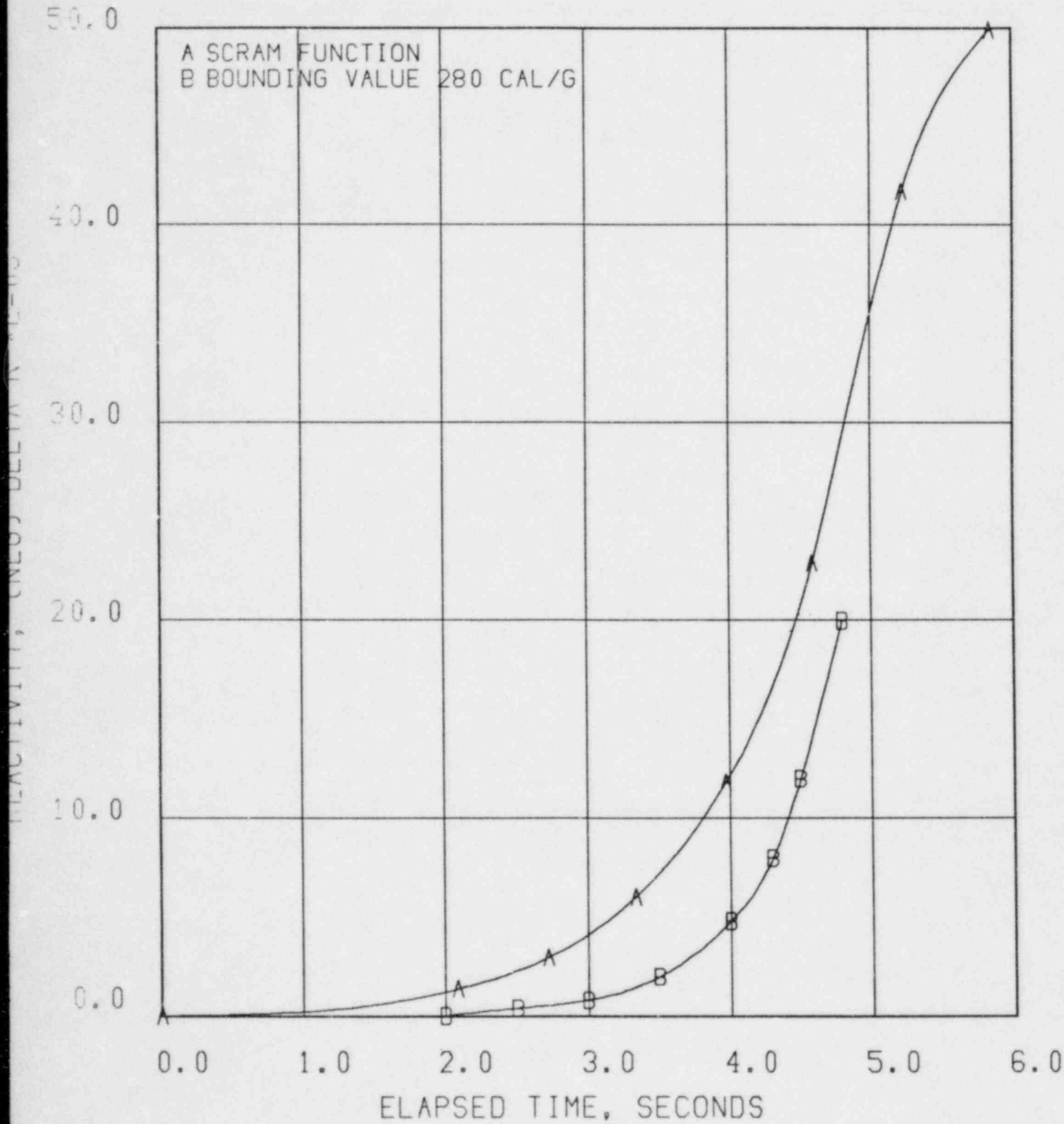


Figure 11. Scram Reactivity Function (Hot Startup)

APPENDIX A
TRANSIENT ANALYSIS INPUT PARAMETERS

The values listed below were used as inputs to the licensing analyses rather than the values provided in Reference A-1, in order to reflect actual plant operating parameters.

	Analysis Value
Dome Pressure (psig)	1026
Turbine Pressure (psig)	975
Reactor Pressure (psia)	1055
Inlet Enthalpy (BTU/lb)	528.6
S/RV Lowest Setpoint (psig)	1110+1%

Reference

A-1. NEDE-24011-P-A-6-US, "General Electric Standard Application for Reactor Fuel (U.S. Supplement)", April 1983.

APPENDIX B

FEEDWATER CONTROLLER FAILURE

The Feedwater Controller Failure (FWCF) event was analyzed at the 100% power/87% flow point for the Extended Load Line Limit Analysis, since this point was found to be more conservative than the 100% power/100% flow point.

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