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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

EDWIN I. HATCH NUCLEAR PLANT

UNIT 1 RELOAD 3

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IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

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

	EOC4
Void coefficient N/A* (¢/% R_{go})	-7.415/-9.269
Void Fraction (%)	40.12
Doppler Coefficient N/A (¢/&°F)	-0.231/-0.219
Average Fuel Temperature (°F)	1397
Scram Worth N/A ($)	-39.79/-31.83
Scram Reactivity vs Time	Figure 2

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

Exposure	7x7 EOC4	8x8/8x8R EOC4
Peaking factors (local, radial and axial)	1.27/1.36/1.40	1.17/1.59/1.40
R-Factor	1.081	1.051
Bundle Power (MWt)	5.798	6.768
Bundle Flow (10^3 lb/hr)	124.59	113.73
Initial MCPR	1.20	1.20

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

(1) Recirculation Pump Trip (RPT)

(2) Thermal Power Monitor (TPM)

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Flow (%)	ϕ (%)	Q/A (%)	P_{sl} (psig)	P_v (psig)	ΔCPR 7x7	ΔCPR 8x8/8x8R	Plant Response
TT w/o BP	BOC-EOC4	104.1	100	168.0	102.0	1149	1180	0.06	0.10	Figure 3
Loss of 100° FW Heating	BOC-EOC4	104.1	100	118.1	116.9	1014	1066	0.13	0.14	Figure 4
Feedwater Controller Failure	BOC-EOC4	104.1	100	135.4	106.8	1127	1161	0.06	0.07	Figure 5

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

1. PLANT-UNIQUE ITEMS (1.0)*

Revisions to Initial Conditions or Inputs: Appendix A

Fuel Loading Error LHGR: Appendix B

Densification Power Spiking: Appendix C

New Methods - Fuel Loading Error: Appendix D

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

	Fuel Type	Number	Number Drilled
Irradiated	Initial Core Type 1	68	68
	Initial Core Type 2	36	36
	Initial Core Type 3	32	32
	8DB250 (Reload 1)	92	92
	8DRB265H (Reload 2)	168	168
New	8DRB265H	164	164
Total		560	560

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 13,056 MWd/t

Assumed reload cycle exposure: 15,740 MWd/t

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)

BOC k_{eff}

Uncontrolled	1.110
Fully Controlled	0.950
Strongest Control Rod Out	0.983

R, Maximum increase in Cold Core Reactivity with Exposure Into Cycle, Δk 0.0036

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.036

*( ) refers to areas of discussion in "Generic Reload Fuel Application," NEDE-24011-P-A, Revision 0, August 1978.

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)	0.83

Channel Hydrodynamic Performance

	Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power)
8x8/8x8R channel	0.39
7x7 channel	0.23

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

Reference: 1*

15. LOADING ERROR RESULTS (5.5.4)

Loading Error Description	MCPR
Rotated Bundle (8x8R)	1.11
Mislocated Bundle (8x8R into 8x8)	≥ 1.07

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: Accident Reactivity (cold)

Resultant peak enthalpies: 197.35 cal/gm

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*Reference 1: "General Fuel Application", NEDE-24011-P-A, Subsection 5.5.2, August 1978.

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

Rod Block Reading*	Rod Position (Feet Withdrawn)	ΔCPR			MLHGR (kW/ft)			Limiting Rod Pattern
		7x7	8x8	8x8R	7x7	8x8	8x8R	
105%*	3.5	0.19	0.17	0.12	14.0	14.0	14.6	Figure 6
106	4.0	0.22	0.20	0.14	14.8	14.9	15.8	Figure 6
107	4.5	0.26	0.22	0.16	15.3	15.2	16.4	Figure 6
108	4.5	0.26	0.22	0.16	15.3	15.2	16.4	Figure 6
109	5.0	0.28	0.25	0.18	15.6	15.5	16.9	Figure 6
110	5.5	0.31	0.27	0.20	16.0	15.8	17.4	Figure 6

11. OPERATING MCPR LIMIT (5.2)

	BOC4 - EOC4
(7x7 fuel)	1.26
(8x8 fuel)	1.24
(8x8R fuel)	1.21

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	Power (%)	Core Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	104.1	100	1194	1232	Figure 7

*Indicates setpoint selected

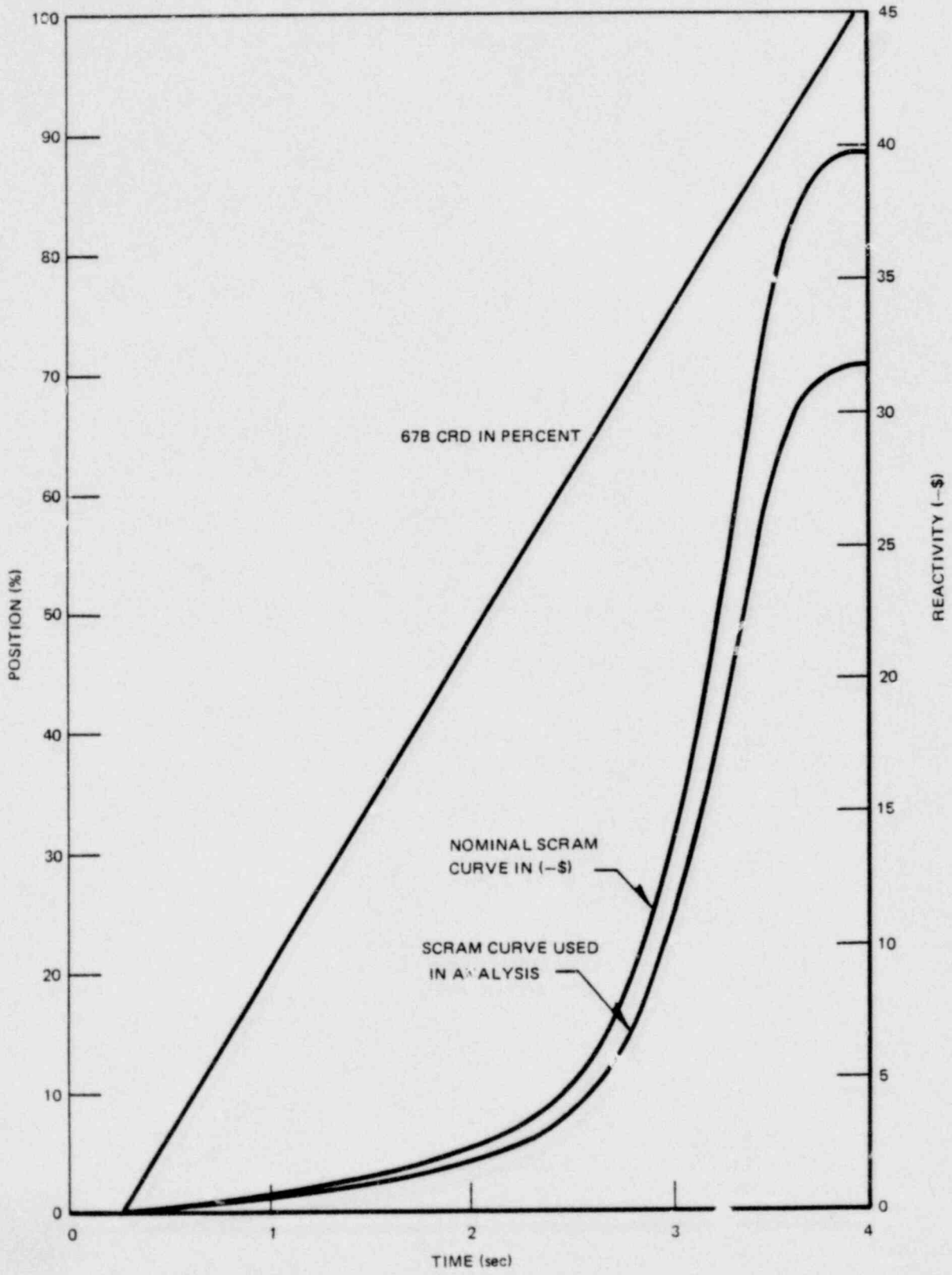
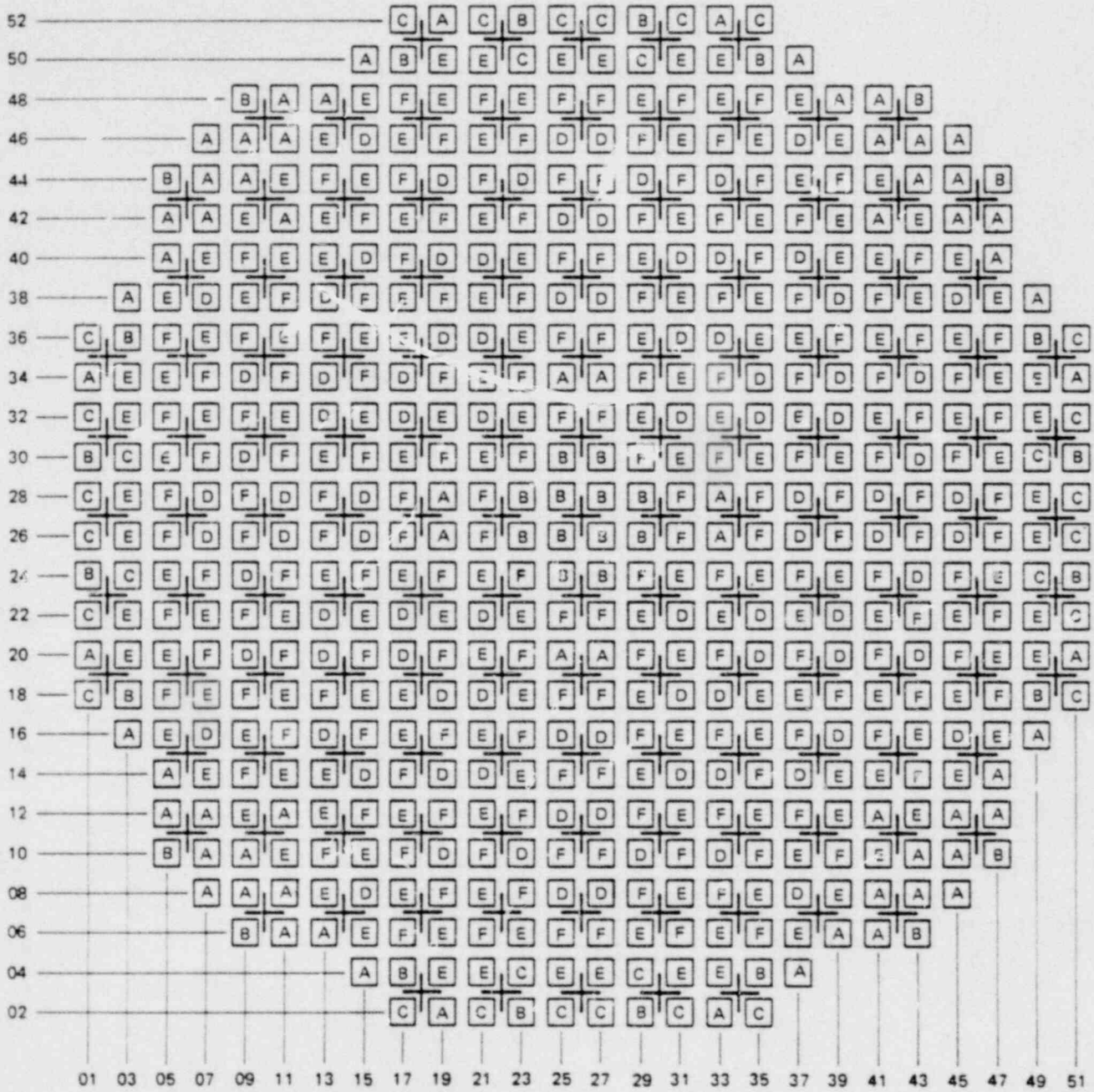


Figure 2. Scram Reactivity and Control Rod Drive Specifications



FUEL TYPE	
A = TYPE 1 (INITIAL CORE)	E = 8DR8265 (RELOAD 2)
B = TYPE 2 (INITIAL CORE)	F = 8DR8265 (RELOAD 3)
C = TYPE 3 (INITIAL CORE)	
D = 8D250 (RELOAD 1)	

Figure 1. Reference Core Loading Pattern

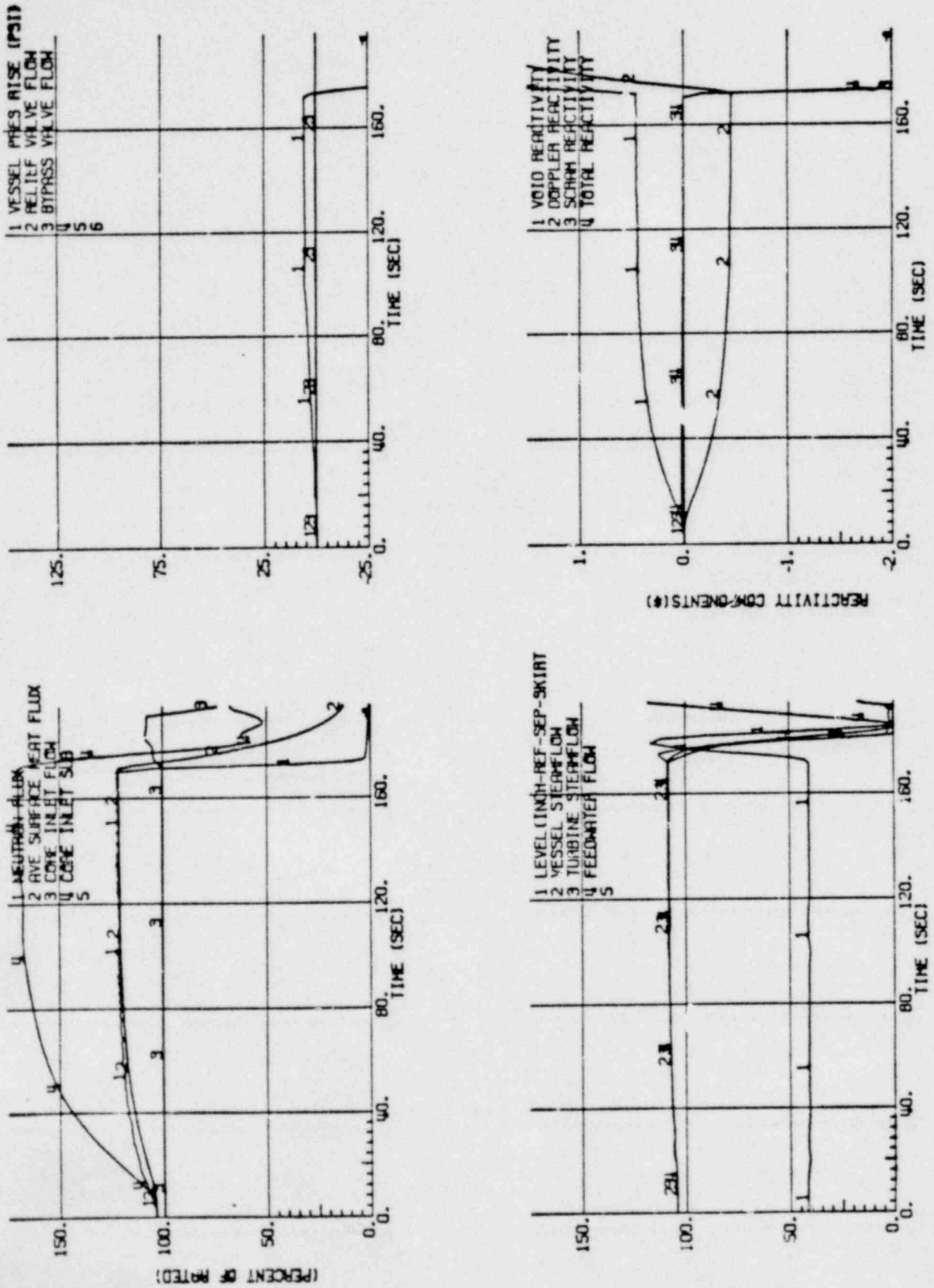


Figure 4. Plant Response to Loss of 100° Feedwater Heater

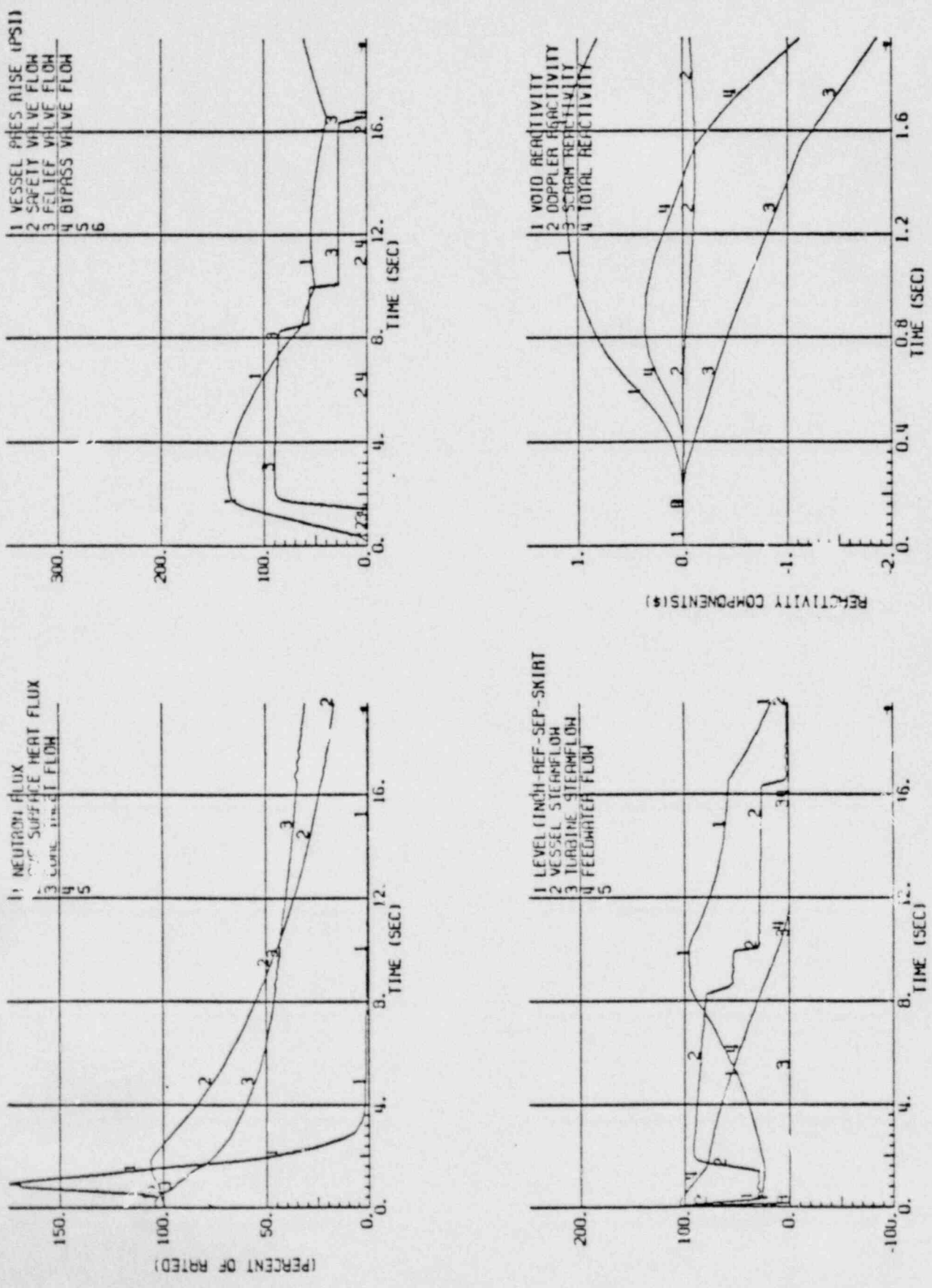


Figure 3. Plant Response to Turbine Trip Without Bypass

	01	03	05	07	09	11	13	15	17	19	21	23	25	27	29
01															
03					8		8								
05				20		20									
07			4		4										
09		20				28									
11			4		0		4								
13		20		28		46									
15															
17															
19															
21															
23															
25															
27															
29															

Notes:
1. Rod Pattern is 1/4 core mirror symmetric upper left quadrant shown on map.
2. No. indicate number of notches withdrawn out of 48. Blank is a withdrawn rod.
3. Error rod is (18,35).

Figure 6. Limiting RWE Rod Pattern

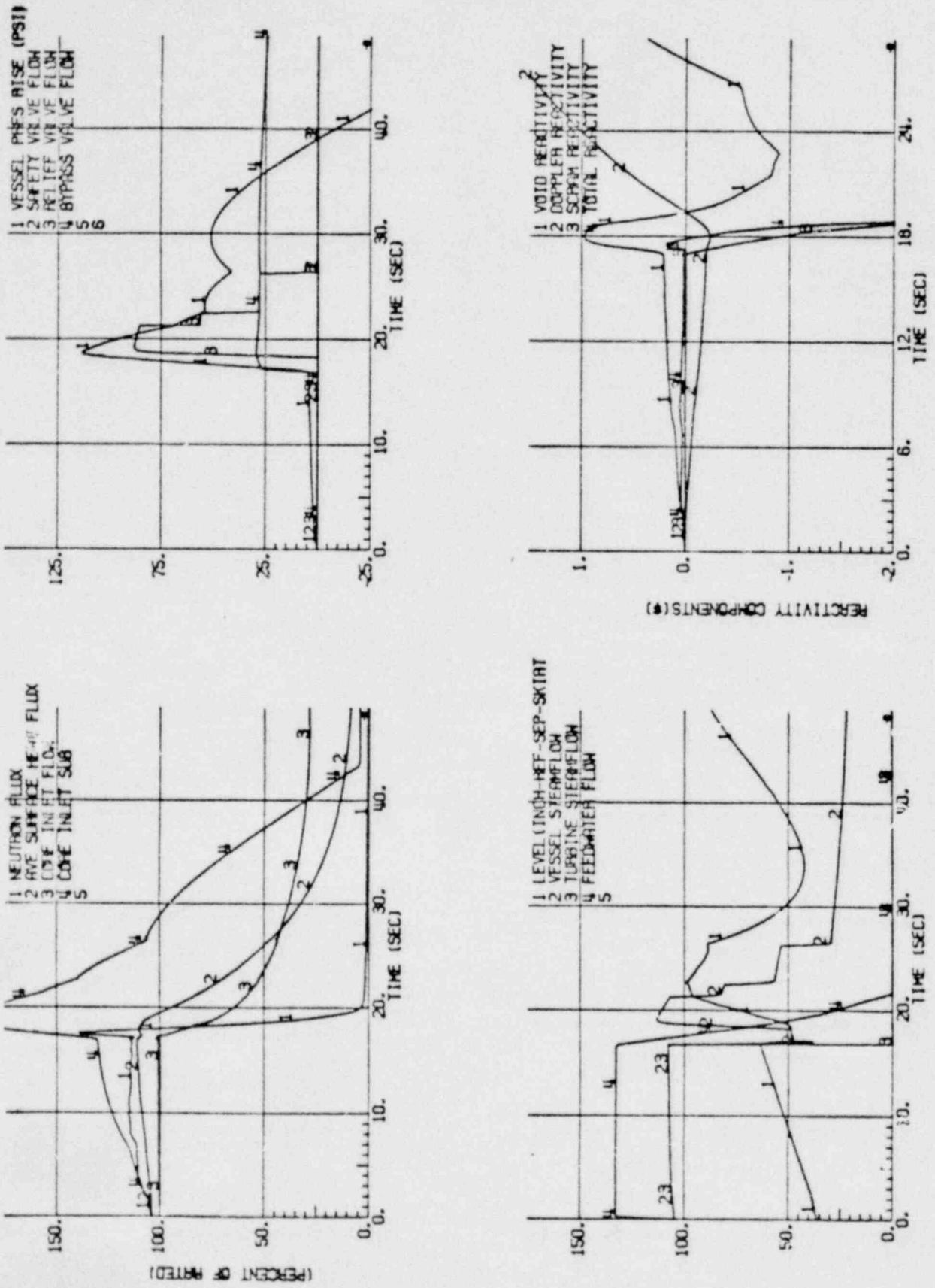


Figure 5. Plant Response to Feedwater Controller Failure

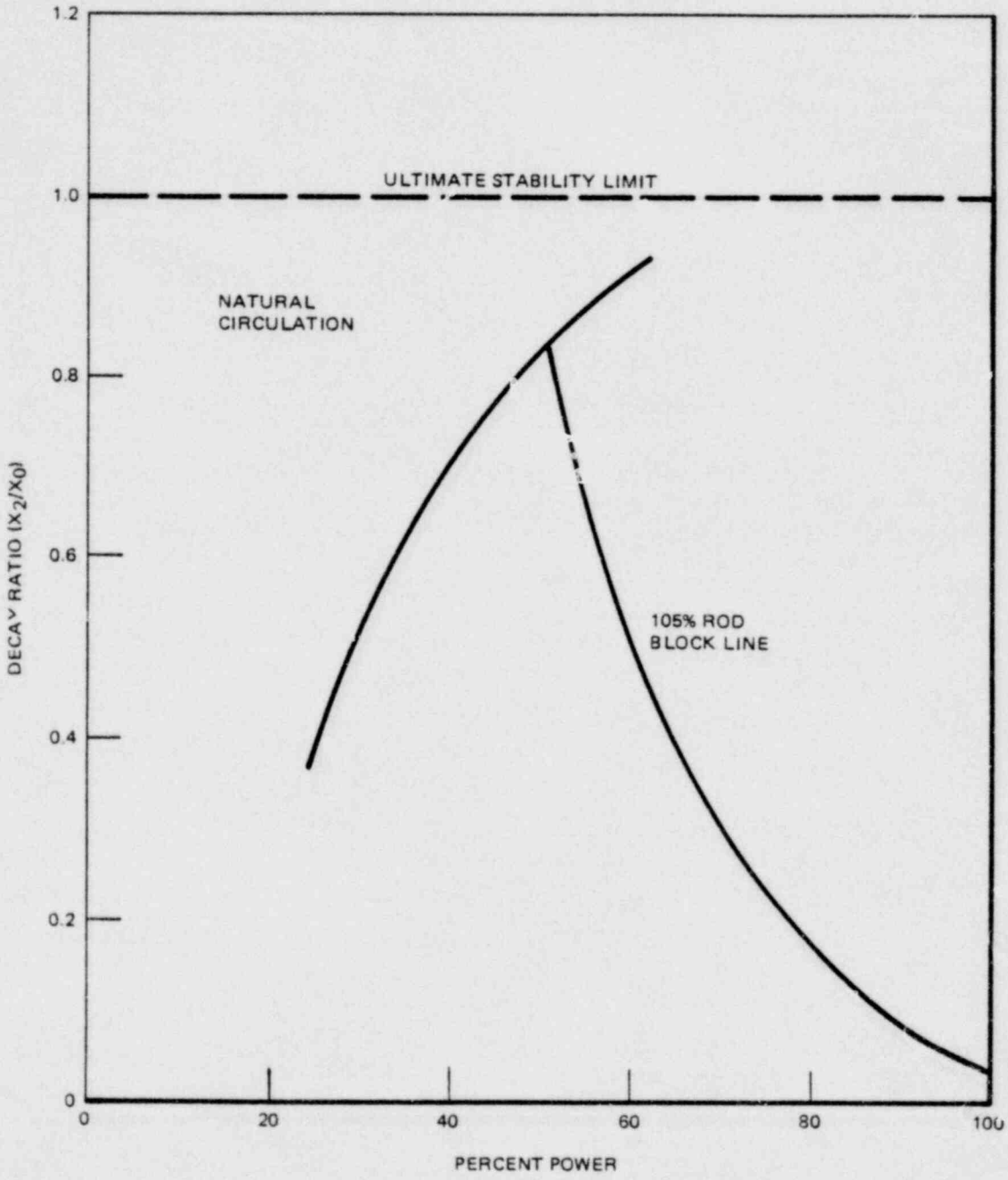


Figure 8. Decay Ratio

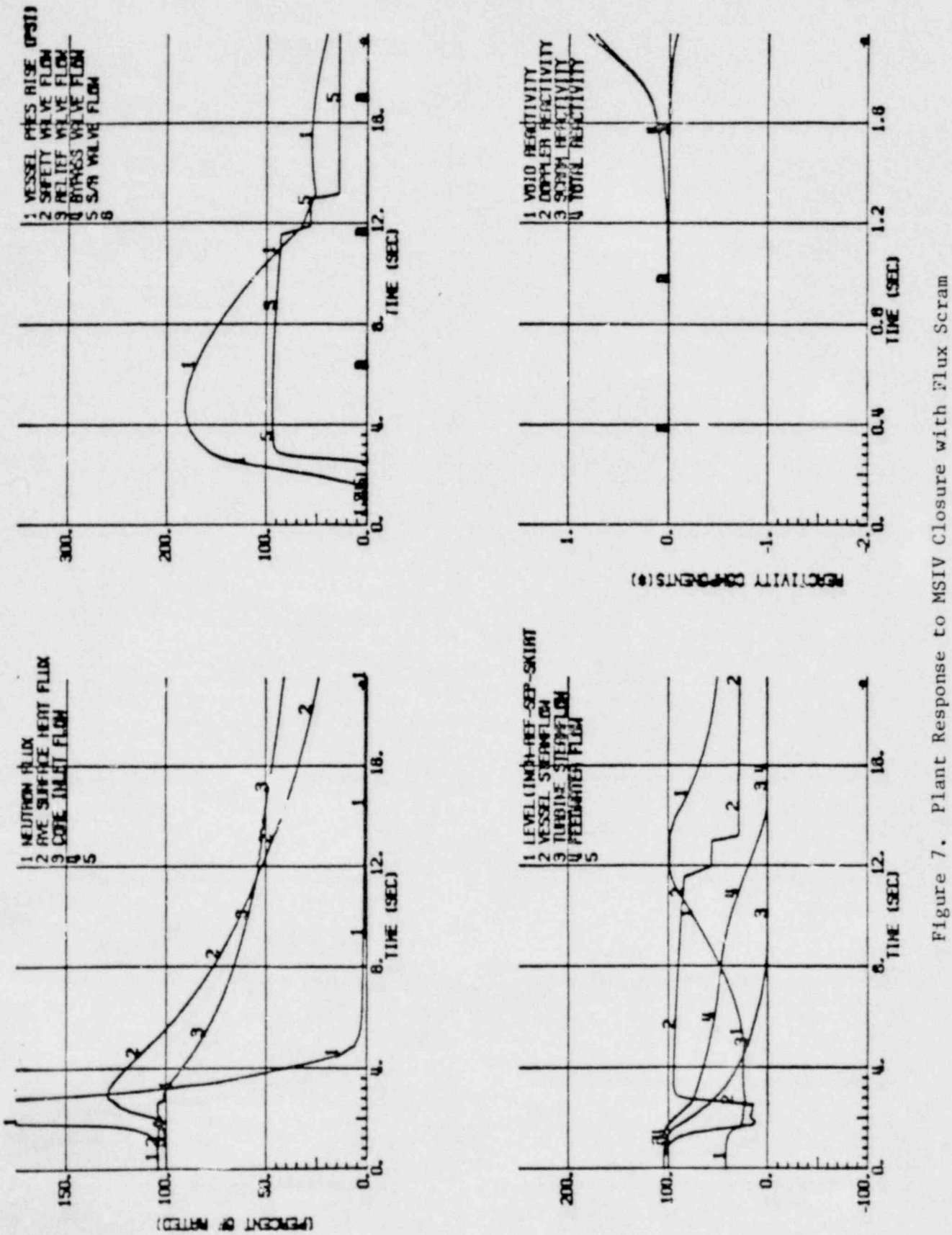


Figure 7. Plant Response to MSIV Closure with Flux Scram

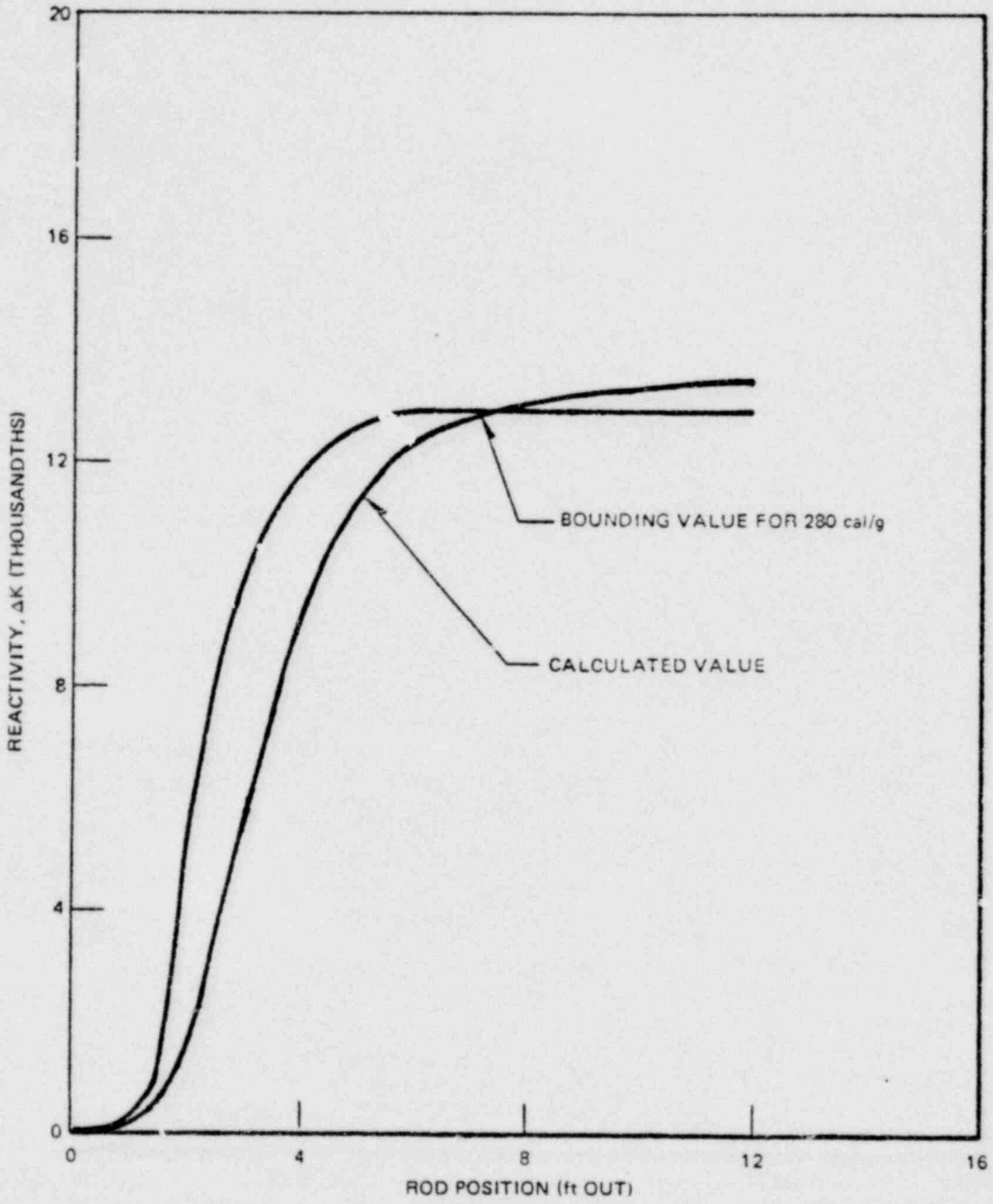


Figure 10. RDA Reactivity Shape Function at 20°C

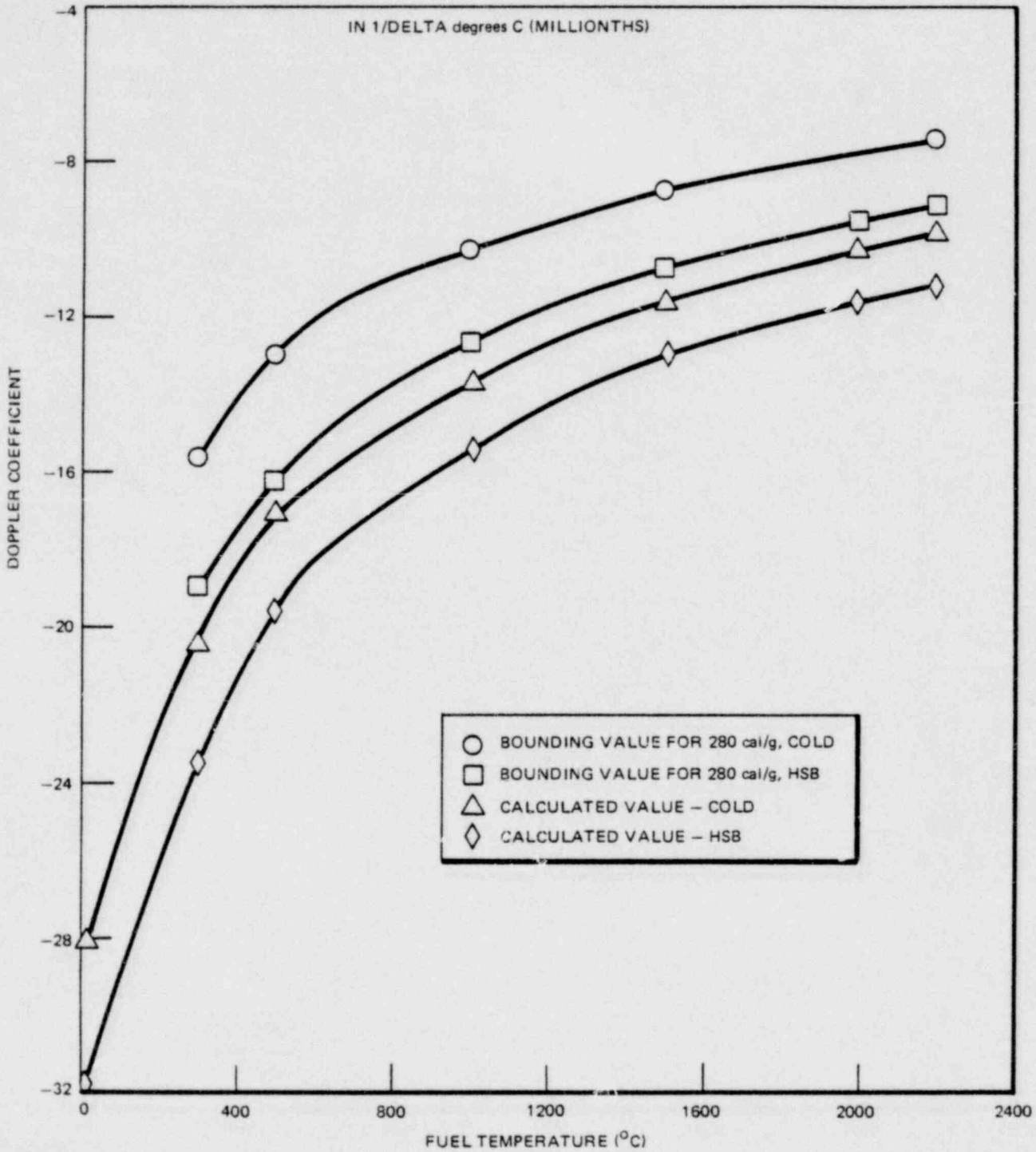


Figure 9. Doppler Reactivity Coefficient Comparison for RDA

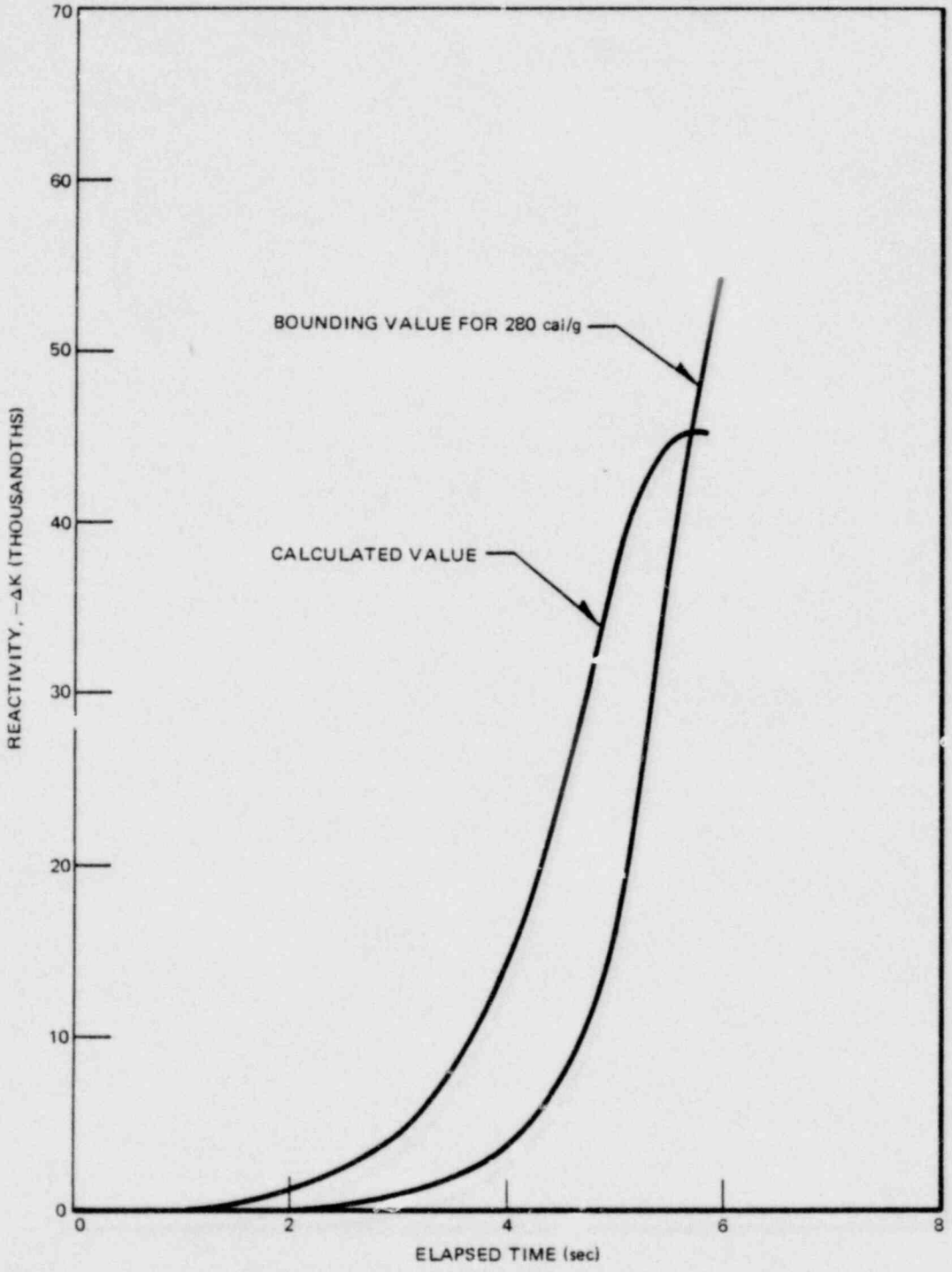


Figure 12. RETA Scram Reactivity Function at 20°C

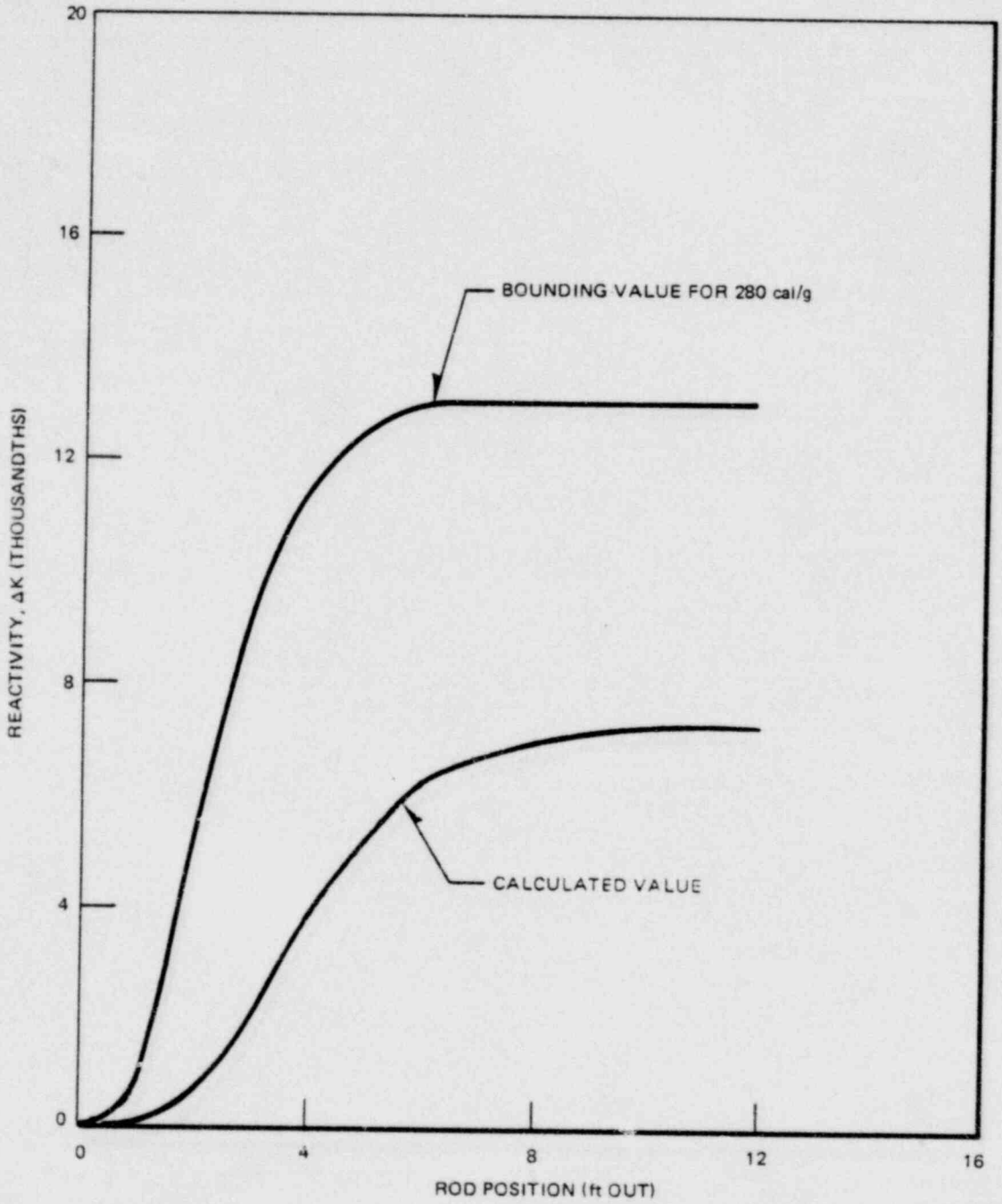


Figure 11. RDA Reactivity Shape Function at 286°C

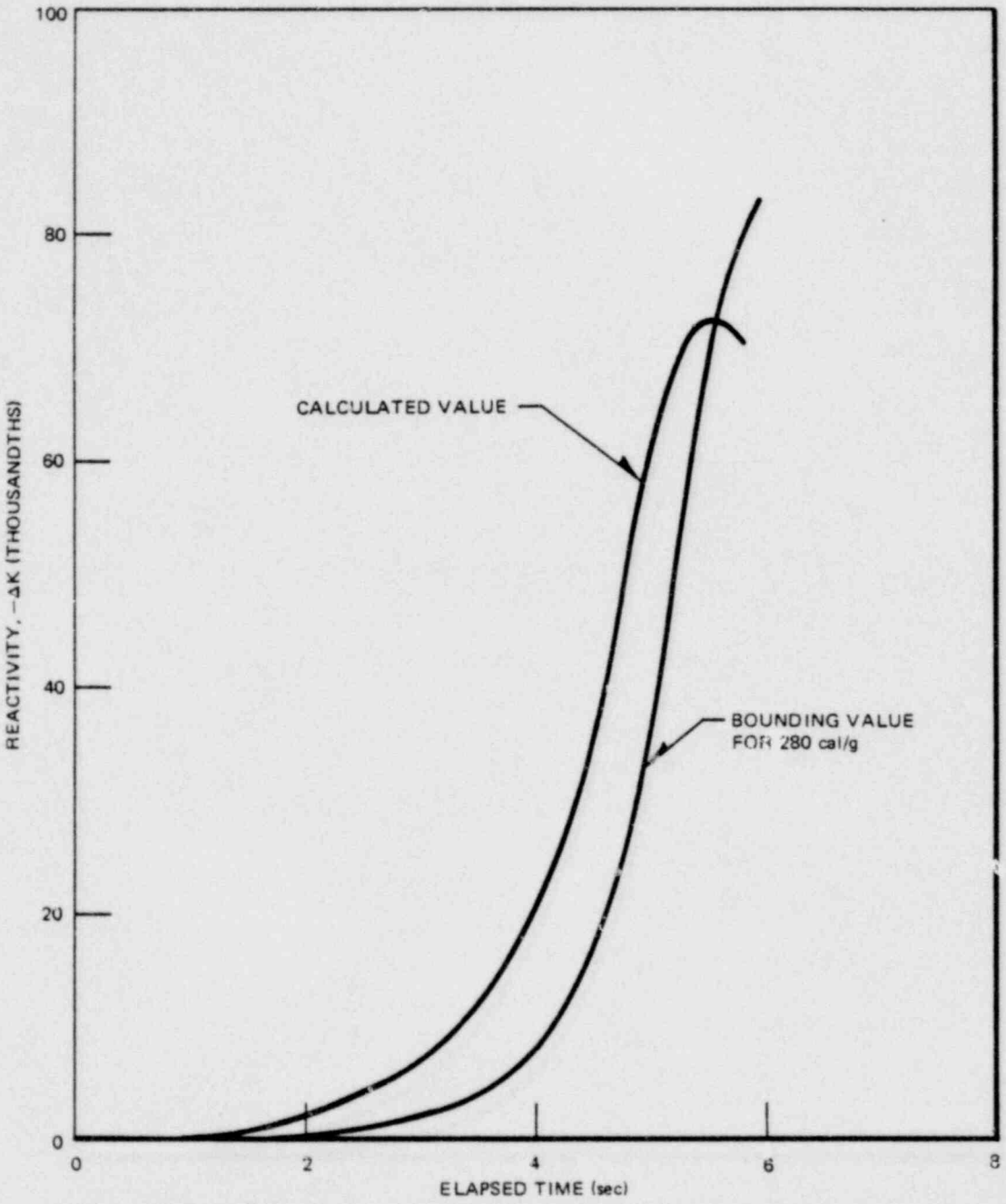


Figure 13. RDA Scram Reactivity Function at 286°C

APPENDIX A

CHANGES IN PARAMETERS FROM NEDE-24011

GETAB Transient Analysis Initial Condition Parameters:

Reactor Pressure, psia	1035.0
Inlet Enthalpy, Btu/lb	523.7

Plant Operating/Input Parameters:

S/RV Capacity, %	85.9

APPENDIX B
FUEL LOADING ERROR LHGR'S

Mislocated Bundle	LHGR
8x8R into I.C. 7x7	18.84
Rotated Bundle	
8x8R	15.18

APPENDIX C

DENSIFICATION POWER SPIKING

Reference C-1 documents the NRC Staff position that " . . . it (is) acceptable to remove the 8x8 and 8x8R spiking penalty factor from the plant Technical Specification for these operating BWR's for which it can be shown that the predicted worst case maximum transient LHGR's, when augmented by the power spike penalty, do not violate the exposure-dependent safety limit LHGR's".

The Hatch-1 Reload-3 license submittal contains the required information to remove the power spiking penalty from the Hatch-1 Technical Specifications. All calculated maximum transient LHGR's, including those for Rod Withdrawal Error and Fuel Loading Error, are less than the 1% plastic diametral strain LHGR limits, as given in Reference C-2, by more than the power spike penalty of 2.2%.

REFERENCES

C-1 "Safety Evaluation of the GE Methods for the Consideration of Power Spiking Due to Densification Effects in BWR 8x8 Fuel Design and Performance", Reactor Safety Branch, DOR, May 1978.

C-2 Licensing Topical Report, "GE Boiling Water Reactor, Generic Reload Fuel Application", NEDE-24011-P-A, May 1977.

APPENDIX D

NEW BUNDLE LOADING ERROR EVENT ANALYSES PROCEDURES

The bundle loading error analyses results presented in Section 15 in this supplement are based on new analyses procedures for both the rotated bundle and the mislocated bundle loading error events. The use of these new analyses procedures is discussed below.

D.1 NEW ANALYSIS PROCEDURE FOR THE ROTATED BUNDLE LOADING ERROR EVENT

The rotated bundle loading error event analysis results presented in this supplement are based on the new analysis procedure described and approved in Reference D-1. This new method of performing the analysis is based on a more accurate detailed analytical model.

The principle difference between the previous analysis procedure and the new analysis procedure is the modeling of the water gap along the axial length of the bundle. The previous analysis used a uniform water gap, whereas the new analysis utilizes a variable water gap which is more representative of the actual condition, since the interfacing between the top guide and the fuel spacer buttons, caused by misorientation, causes the bundle to lean. The effect of the variable water gap is to reduce the power peaking and the R-factor in the upper regions of the limiting fuel rod. This results in the calculation of a reduced CPR for the rotaced bundle. The calculation was performed using the same analytical models as were previously used. The only change is in the simulation of the water gap, which more accurately represents the actual geometry.

The number presented in Section 15 represents the minimum CPR during Cycle 4 of the most limiting rotated bundle.

D.2 NEW ANALYSIS PROCEDURE FOR THE MISLOCATED BUNDLE LOADING ERROR EVENT

The mislocated bundle loading error event analyses results presented in this supplement are based on the new analysis procedure described in Reference D-1. This

new method of performing the analysis employs a statistically corrected Haling procedure and analyzes every bundle in the core.

The use of the statistically corrected Haling analyses procedure indicates that the minimum CPR for mislocated 8DR265 bundles in the core is greater than the safety limit (1.07) for all exposures throughout Cycle 4.

REFERENCES

D-1 Safety Evaluation Report (letter), D. G. Eisenhut (NRC) to R. E. Engel (GE), MFN-200-78, dated May 8, 1978.