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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
BROWNS FERRY NUCLEAR PLANT UNIT 1
RELOAD NO. 4
(CYCLE 5)

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

Plant Parameter Differences - Appendix A

- Safety Valves
- Safety/Relief Valves
- GETAB Initial Conditions
- Initial MCPR
- Fuel Loading Error LHGR

ODYN Code for Transient Analyses - Appendix B

New Bundle Loading Error Event Analyses Procedure - Appendix C

Densification Power Spiking - Appendix D

Lead Test Assemblies (LTAs) - Appendix E

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

	Fuel Type	Number	Number Drilled
Irradiated, Reload 1	8DB274H	8	8
Irradiated, Reload 1	8DB274L	112	112
Irradiated, Reload 2	8DRB265L	84	84
Irradiated, Reload 2	8DRB265H	68	68
Irradiated, Reload 3	P8DRB284L	232	232
New	P8DRB284L	220	220
New	P8DRB265L	36	36
New	GLTA-1	2	2
New	GLTA-2	2	2
Total		764	764

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle: 16,071 MWd/t

*( ) refers to areas of discussion in "General Electric Boiling Water Reactor Generic Reload Fuel Application", August 1979 (NEDE-24011-P-A-1).

Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	16,071 MWd/t
Assumed reload cycle core average exposure at end of cycle:	17,846 MWd/t
Core loading pattern:	Figure 1

4. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C (3.3.2.1, i.e., 3.3.2.1.2)

BOC k_{eff}	
Uncontrolled	1.115
Fully Controlled	0.955
Strongest Control Rod Out	0.990
R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, Δk	0.000

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.029

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

	EOC5
Void Coefficient N/A,* (c/% Rg)	-7.17/-8.96
Void Fraction	39.79
Doppler Coefficient N/A (c/°F)	-0.219/-0.208
Average Fuel Temperature (°F)	1383
Scram Worth N/A ($)	-46.31/-37.05
Scram Reactivity	Figure 2

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

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

	EOC5		
	8x8	8x8R	P8x8R
Peaking factors (local, radial and axial)	1.22 1.40 1.40	1.20 1.56 1.40	1.20 1.54 1.40
R-Factor	1.098	1.051	1.051
Bundle Power (MWt)	5.913	6.571	6.495
Bundle Flow (10^3 lb/hr)	107.2	108.0	108.4
Initial MCPR	1.25	1.25	1.26

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Recirculation Pump Trip

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Core Flow (%)	$\dot{\phi}$ (% NBR)	Q/A (% NBR)	P_{SL} (PSIG)	P_V (PSIG)	Nominal ΔCPR 8x8	Nominal ΔCPR 8x8R	Nominal ΔCPR P8x8R	Plant Response
Generator Load Rejection without Bypass	EOC5	104.5	100	599	121	1219	1230	0.18	0.18	0.20	Figure 3
Loss of 100°F Feedwater Heating	-	104.5	100	124	124	1013	1069	0.15	0.15	0.15	Figure 4
Feedwater Controller Failure	EOC5	104.5	100	367	120	1158	1189	0.15	0.15	0.16	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	LHGR** 8x8	LHGR** 8x8R/P8x8R	Limiting Rod Pattern
104	3.0	0.14	0.09	11.4	13.7	Figure 6
105	3.5	0.16	0.11	12.3	14.2	Figure 6
106*	4.0	0.19	0.14	13.0	15.2	Figure 6
107	4.5	0.22	0.16	13.4	15.9	Figure 6
108	4.5	0.27	0.16	13.4	15.9	Figure 6
109	5.0	0.29	0.19	13.5	16.3	Figure 6
110	5.5	0.32	0.21	13.6	16.7	Figure 6

11. CYCLE MCPR VALUES (5.2, APPENDIX C)

BOC5 to EOC5

Pressurization Events	Option A 8x8	Option A 8x8R	Option A P8x8R	Option B 8x8	Option B 8x8R	Option B P8x8R
Generator Load Rejection without Bypass	1.30	1.30	1.33	1.22	1.22	1.23
Feedwater Controller Failure	1.27	1.27	1.28	1.24	1.24	1.25

Non-Pressurization Events	8x8	8x8R	P8x8R
Loss of 100°F Feedwater Heating	1.22	1.22	1.22
Fuel Loading Error	1.22	1.22	1.22
Rod Withdrawal Error	1.26	1.21	1.21

*Indicates setpoint selected.
**Includes a 2.2% peaking penalty for fuel densification.

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	1238	1272	Figure 7

13. STABILITY RESULTS (5.4)

Decay Ratio:	Figure 8
Reactor Core Stability Decay Ratio, x_2/x_0:	0.83
Channel Hydrodynamic Performance Decay Ratio, x_2/x_o	
8x8 channel	0.39
8x8R/P8x8R channel	0.29

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

P8DRB284L, GLTA-1, and GLTA-2

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Local Oxidation Fraction
200	11.2	1685	0.004
1000	11.3	1667	0.003
5000	11.8	1671	0.003
10,000	12.0	1647	0.003
15,000	12.0	1669	0.003
20,000	11.8	1672	0.003
25,000	11.2	1633	0.003
30,000	10.8	1596	0.002
35,000	10.2	1469	0.001
40,000	9.5	1411	0.001

P8DRB265L

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Local Oxidation Fraction
200	11.6	1711	0.004
1000	11.6	1700	0.004
5000	12.1	1692	0.003
10,000	12.1	1663	0.003
15,000	12.1	1683	0.003
20,000	11.9	1683	0.003
25,000	11.3	1637	0.003
30,000	10.7	1579	0.002
35,000	10.2	1526	0.002
40,000	9.6	1463	0.001

15. LOADING ERROR RESULTS (5.5.4)

Limiting Event: Rotated Bundle P8DRB284L, MCPR $\geq$ 1.22

See Appendix C.

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Doppler Reactivity Coefficient: Figure 9

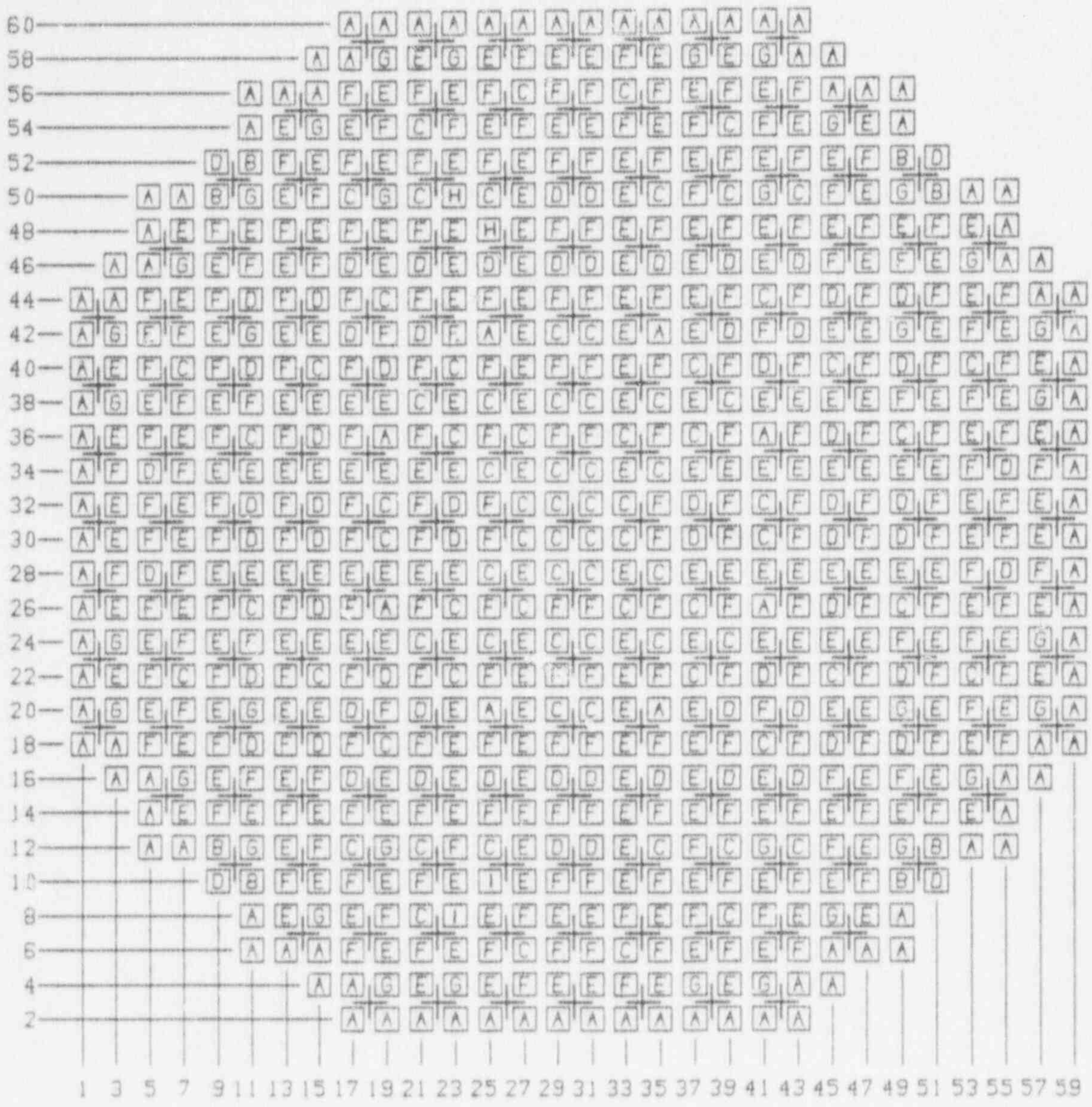
Accident Reactivity Shape Functions: Figures 10 and 11

Scram Reactivity Functions: Figures 12 and 13

Plant Specific Analysis Results

Parameter(s) not bounded: Accident Reactivity Shape Function - Cold

Resultant peak enthalpy (cal/g): 274.3



FUEL TYPE	
A = 8DB274L, RELOAD 1	E = P8DRB284L, RELOAD 3
B = 8DB274H, RELOAD 1	F = P8DRB234L, NEW
C = 8DRB265L, RELOAD 2	G = P8DRB265L, NEW
D = 8DRB265H, RELOAD 2	H = GLTA-1, NEW
	I = GLTA-2, NEW

Figure 1. Reference Core Loading Pattern

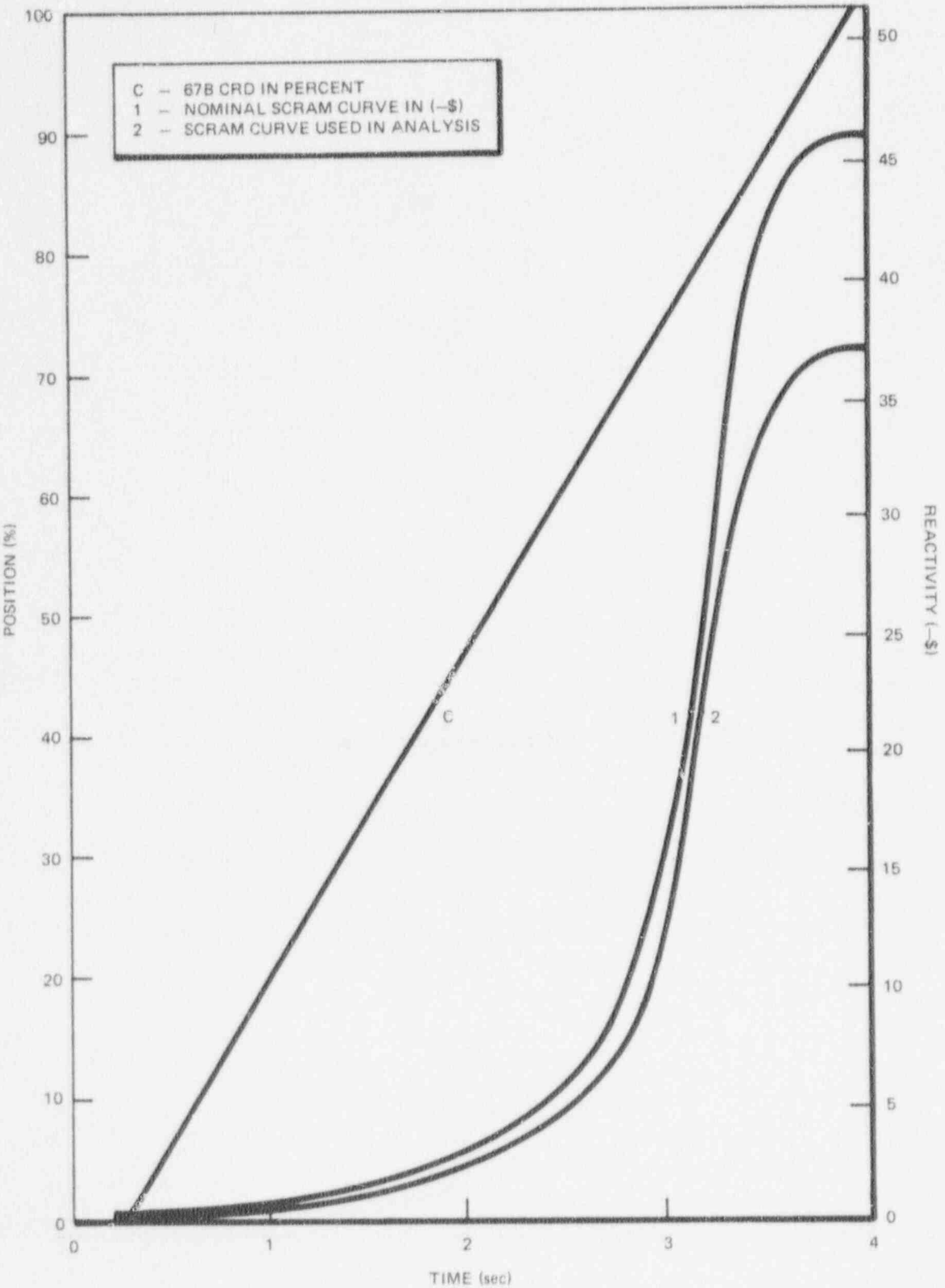


Figure 2. Scram Reactivity and CRD Specifications

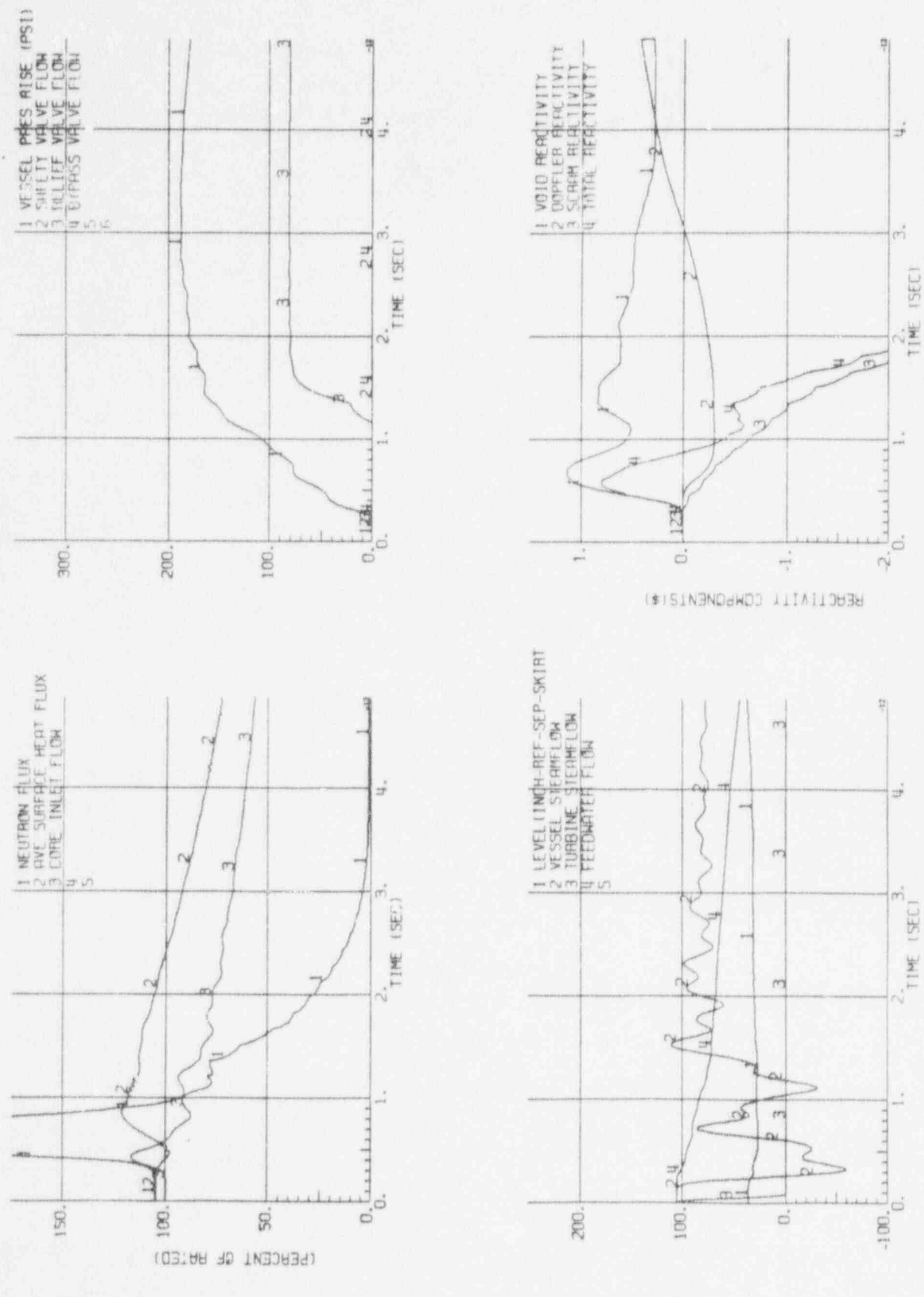


Figure 3. Plant Response to Generator Load Rejection, Without Bypass

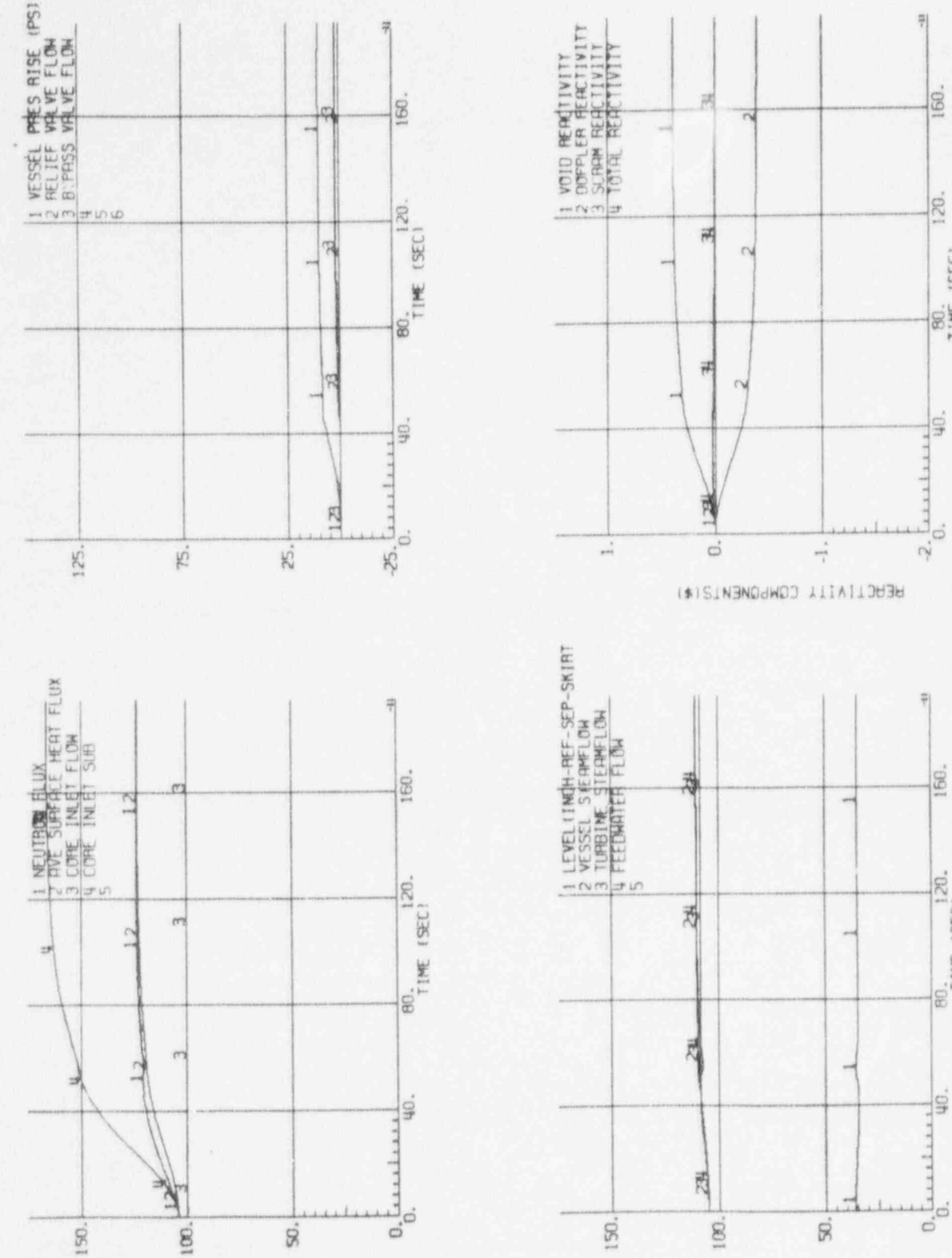


Figure 4. Plant Response to Loss of 100°F Feedwater Heating

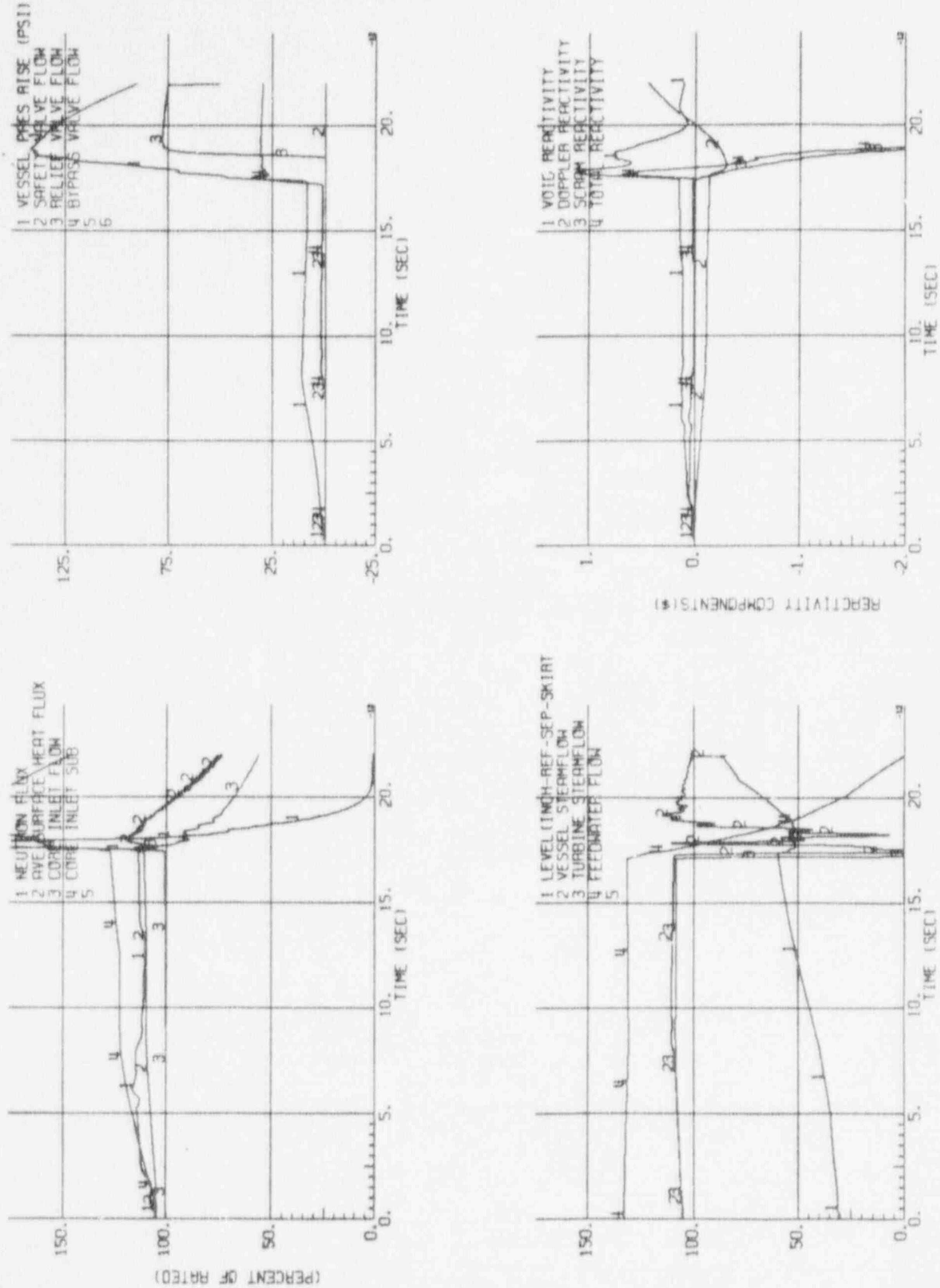


Figure 5. Plant Response to Feedwater Controller Failure

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58
59					0		4		4		0				
55				24		32		24		32		24			
51			0		0		8		8		0		0		
47				24		36		44		36		24			
43	0		4		8		0		0		8		4		0
39		24		32		44		36		44		32		24	
35	0		0		0		8		8		0		0		0
31		28		24		24		32		24		24		28	
27	0		0		0		8		8		0		0		0
23		24		32		44		36		44		32		24	
19	0		4		8		0		0		8		4		0
15				24		36		44		36		24			
11			0		0		8		8		0		0		
7				24		32		24		32		24			
3					0		4		4		0				

NOTES:

1. No. indicates number of notches withdrawn out of 48. Blank is a withdrawn rod.

2. Error rod is (26,43).

Figure 6. Limiting RWE Rod Pattern

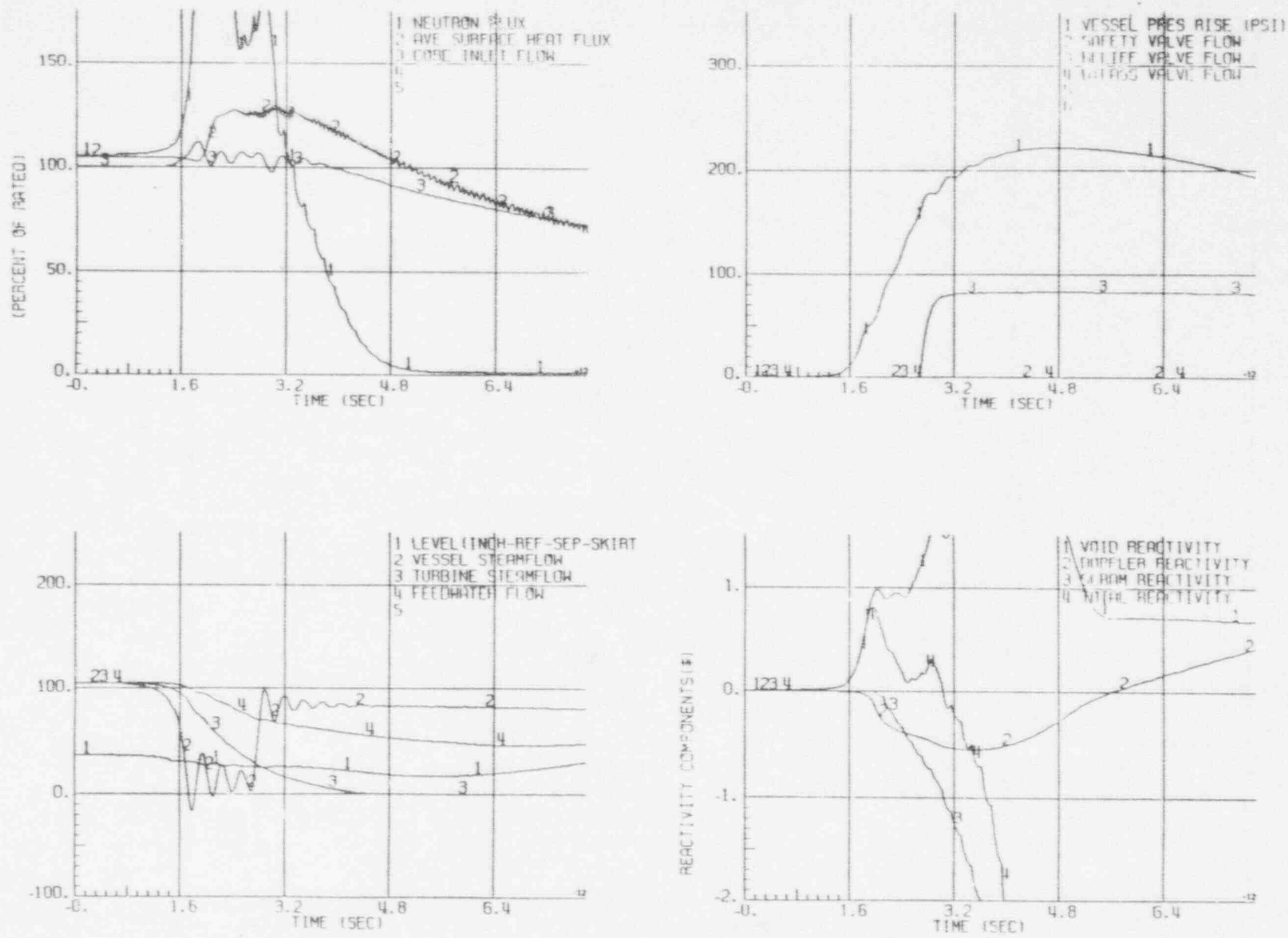


Figure 7. Plant Response to MSIV Closure

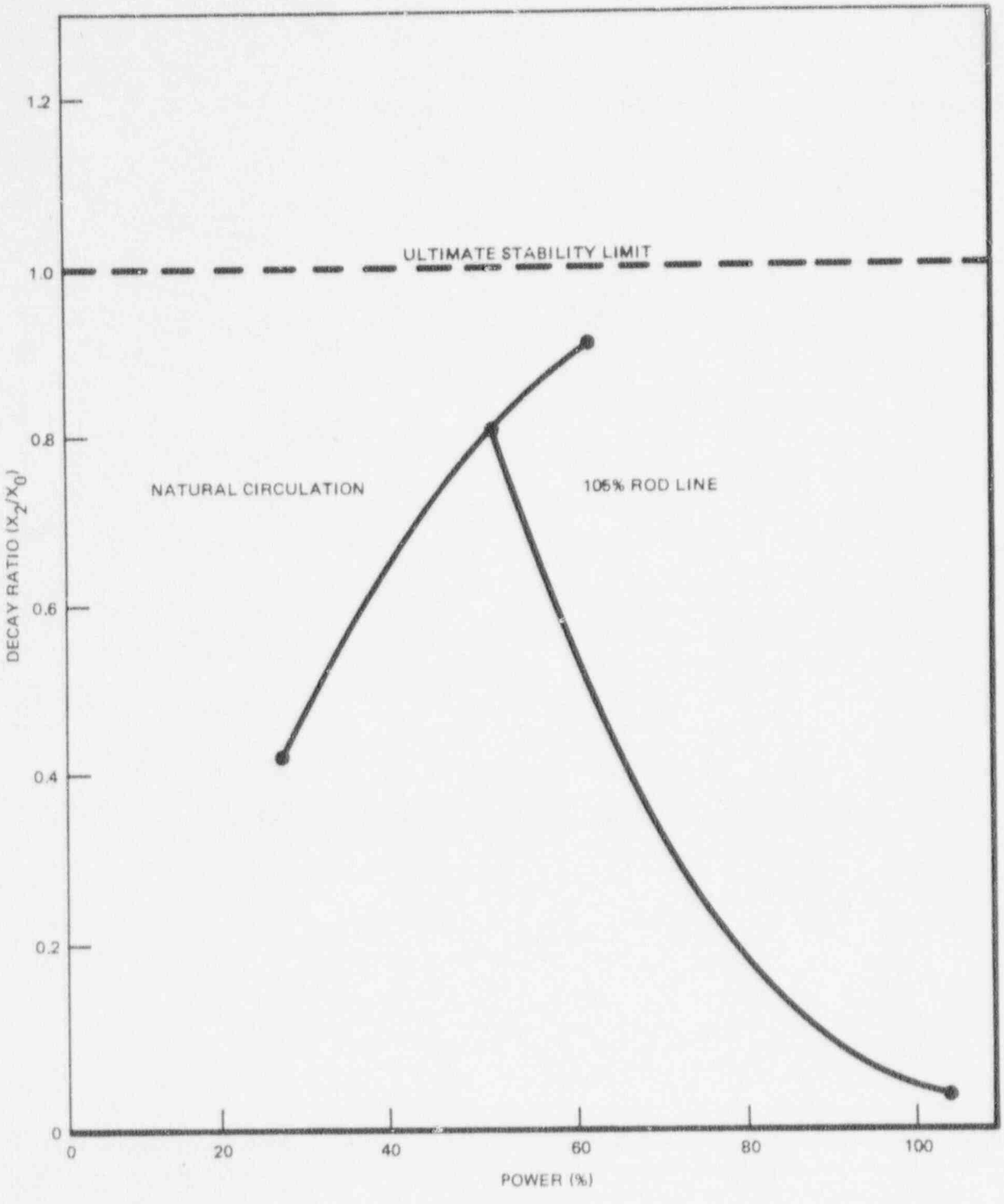


Figure 8. Decay Ratio

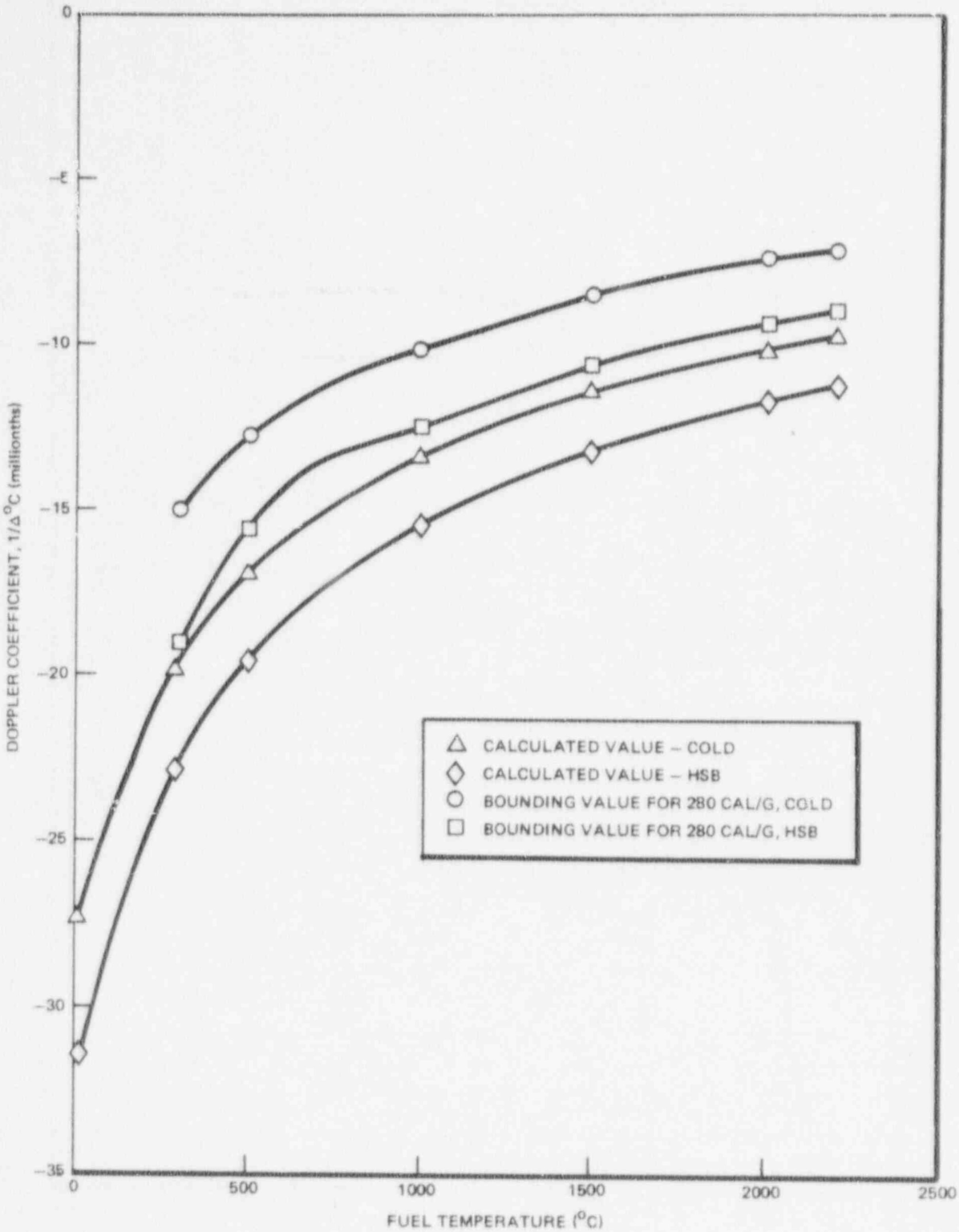


Figure 9. Doppler Reactivity Coefficient Comparison for RDA

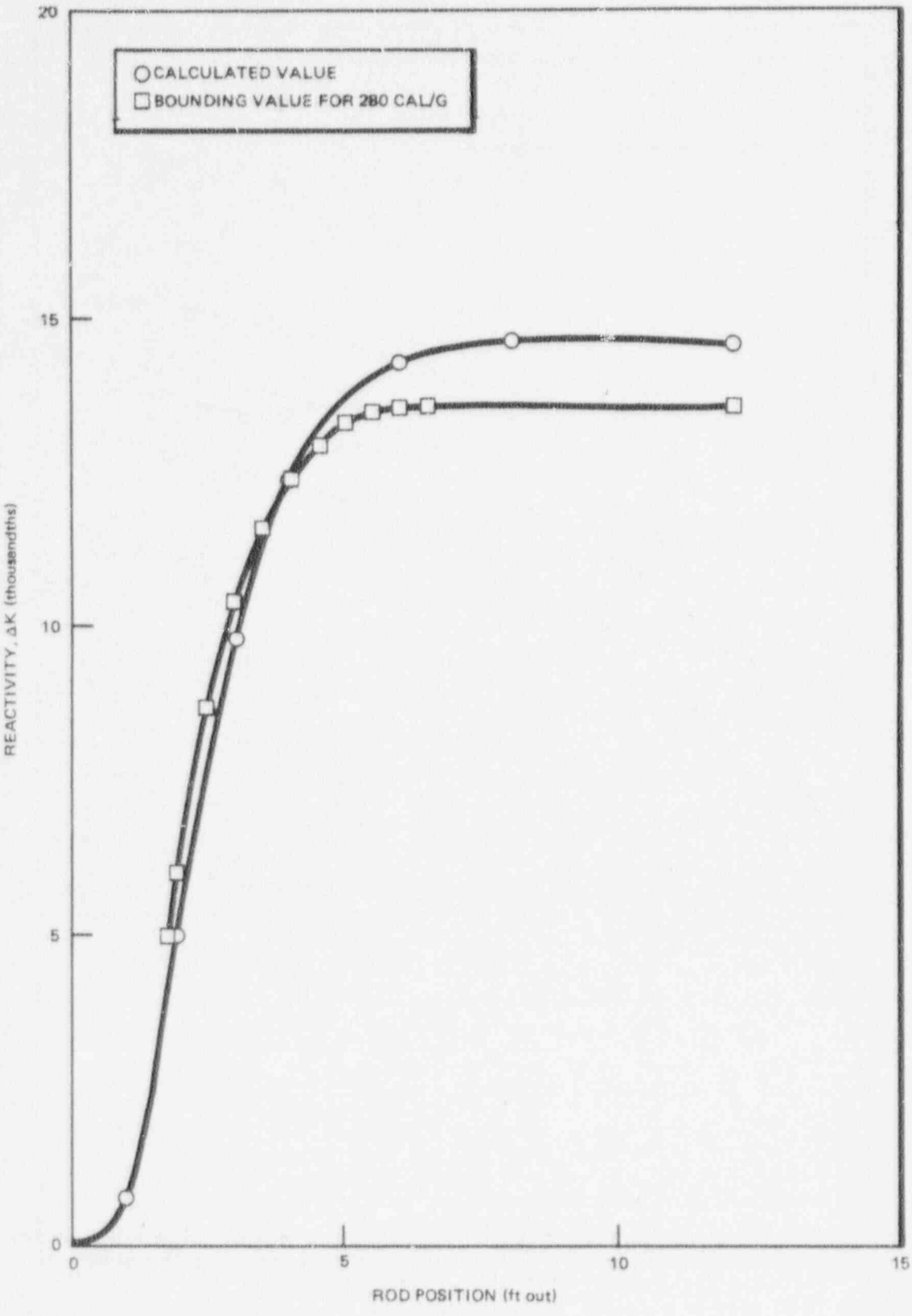


Figure 10. RDA Reactivity Shape Function - Cold

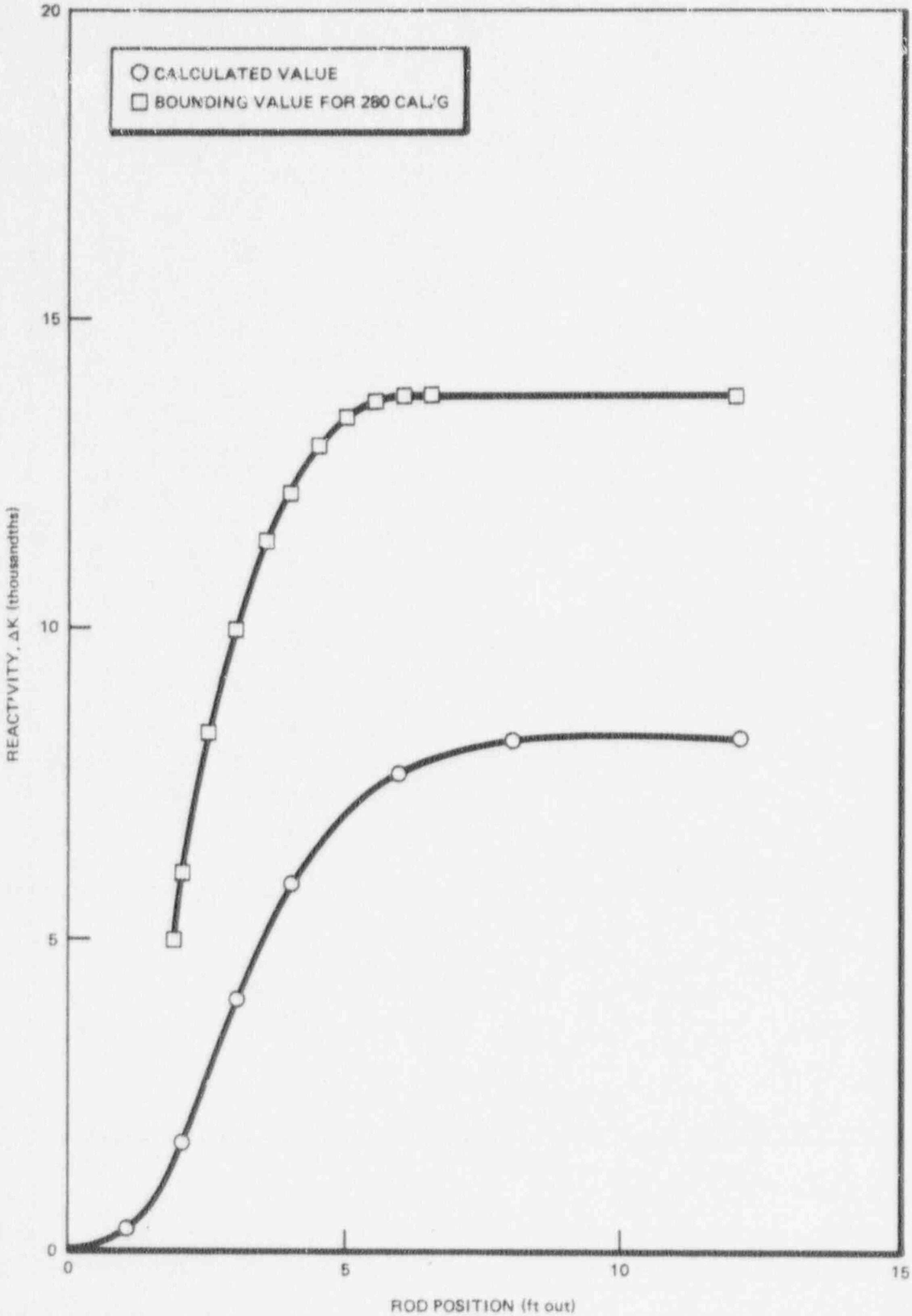


Figure 11. RDA Reactivity Shape Function at 286°C

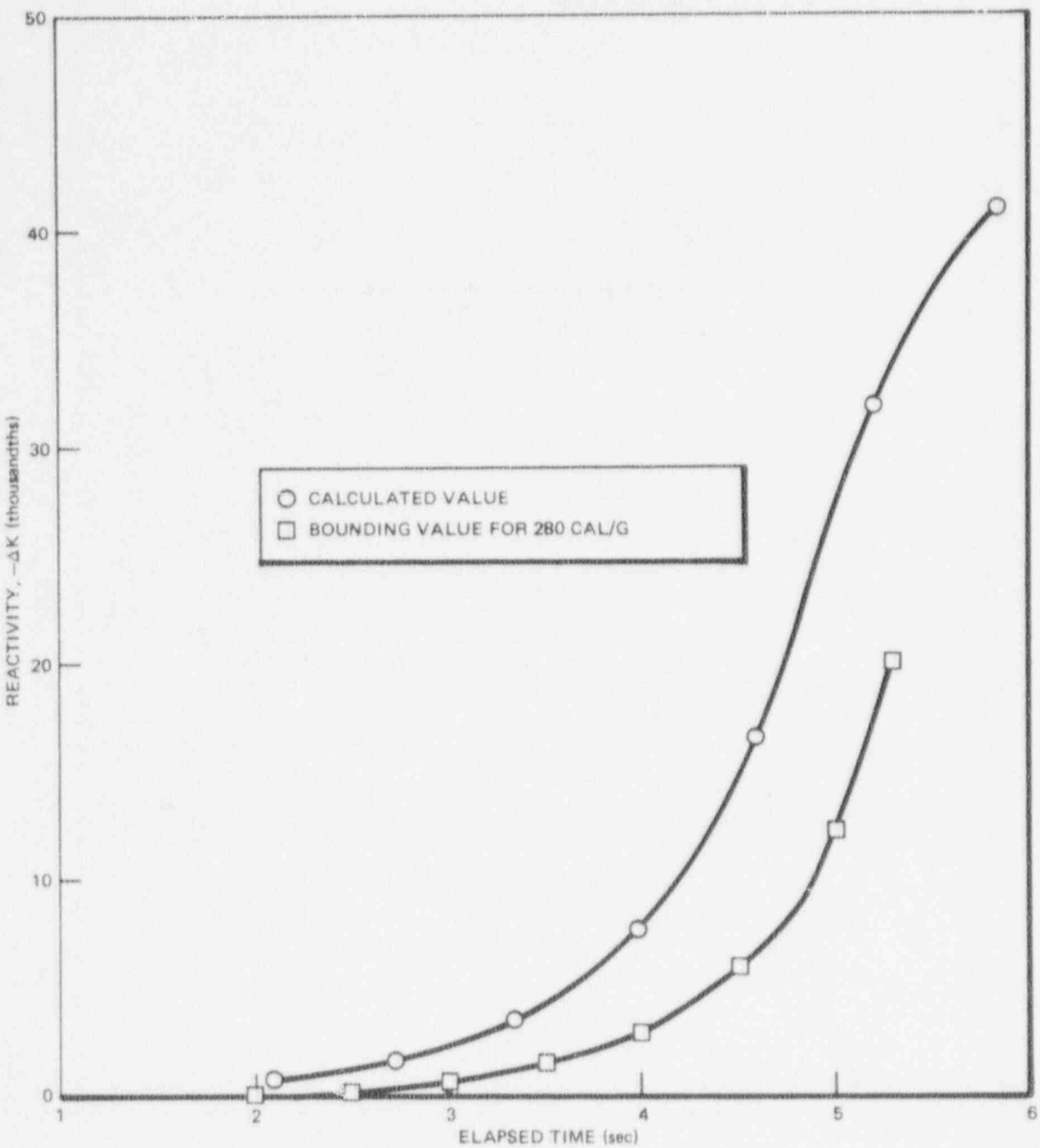


Figure 12. RDA Scram Reactivity Function - Cold

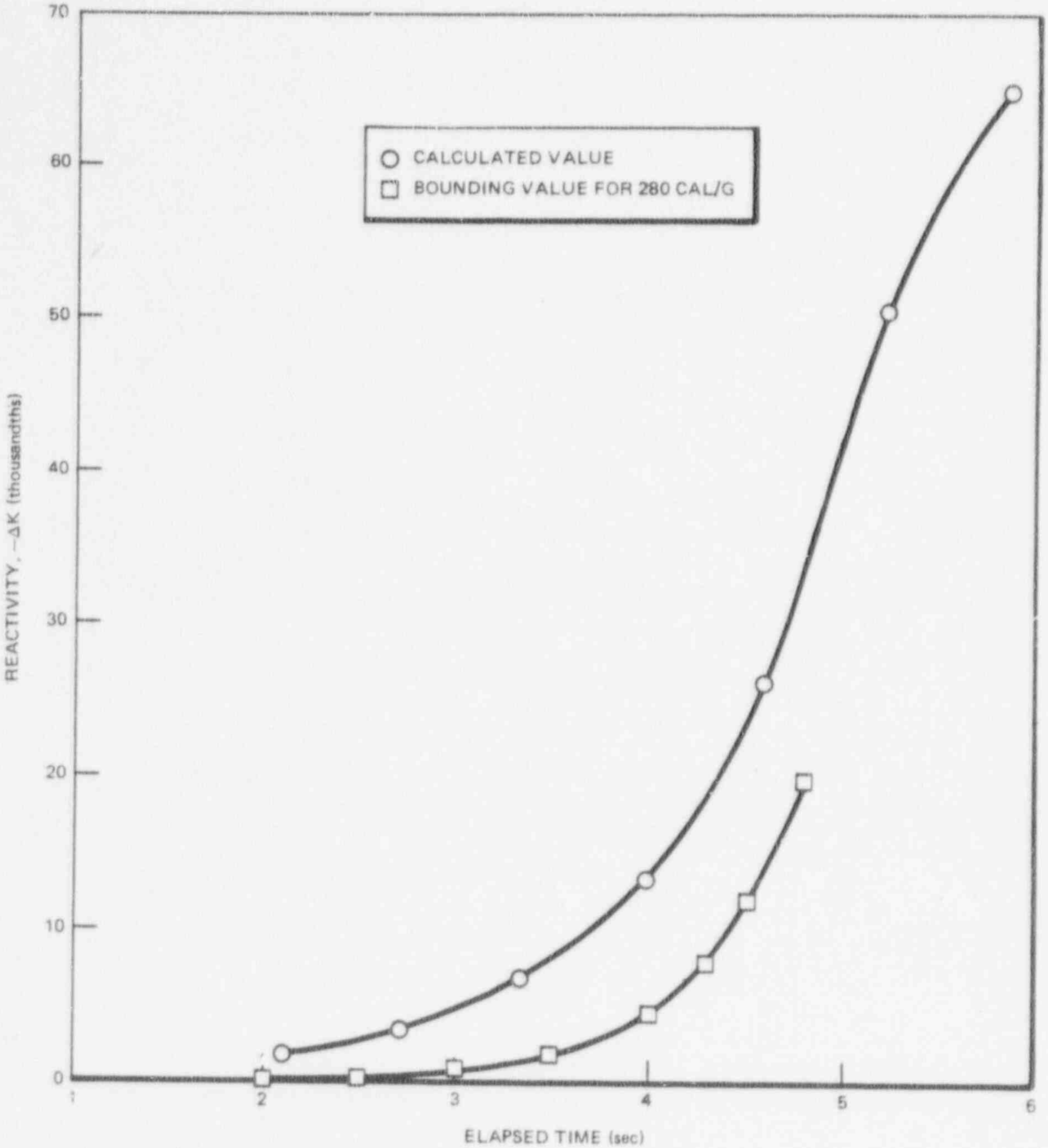


Figure 13. RDA Scram Reactivity Function at 286°C

APPENDIX A

PLANT PARAMETER DIFFERENCES

Safety Valves: None

Safety/Relief Valves: 13 installed
12 used in analysis
Capacity at setpoint used in analysis, 77.46%*

GETAB Initial Conditions

See the revision to Table 5-8 of NEDE-24011-P-A (page 5-66) enclosed with the letter, J. F. Quirk (GE) to Olan D. Parr (NRC), "General Electric Co. Licensing Topical Report NEDE-24011-P-A, Generic Reload Fuel Application, Appendix D, Second Submittal," February 28, 1979.

Initial MCPR

The initial MCPR for the P8x8R fuel was less than the operating limit MCPR based on nominal ΔCPRs. This is discussed on pp. B-114 and B-115 of the "Generic Reload Fuel Application," NEDE-24011-P-A-1.

Fuel Loading Error Results

LHGR: 17.5 kW/ft including a 2.2% power peaking penalty due to fuel densification.

*The value formerly used was 76.25%. More recent calculations yielded the value of 77.46%.

APPENDIX B
ODYN TRANSIENT CODE

All rapid pressurization and overpressure protection events have been analyzed using the ODYN transient code as specified in Reference B-1. Code overpressure protection analysis results are deterministic as discussed in Reference B-2. The ΔCPR values given for the pressurization events in Section 9 are the plant-specific deterministic values calculated by ODYN based on the initial MCPR given in Item 7 of this submittal. These ΔCPRs may be adjusted to reflect either Option A or Option B ΔCPRs by employing the conversion method described in Reference B-2. These adjustments are based on conservatism factors applied to the ratio ΔCPR/ICPR. The MCPR for the event is determined by adding the ΔCPR to the safety limit. Section 11 presents both the MCPRs for the nonpressurization events, as well as the adjusted MCPRs (Option A and Option B) for the pressurization events.

The operating limit MCPR is the maximum MCPR of the following events:

(1) turbine trip or load rejection without bypass based on ODYN;

(2) feedwater controller failure event based on ODYN;

(3) loss of feedwater heating event;

(4) rod withdrawal error event;

(5) bundle loading error accident;

(6) minimum required by LOCA; and

(7) minimum required by Reference B-3, Appendix C, Page C-65

where Items 3 through 7 are calculated as described in Reference B-3 but the MCPRs for the pressurization events analyzed with ODYN have been adjusted as follows:

(1) MCPRs are adjusted for Option B for all plants choosing to operate under Option B which meet all scram specifications given in Reference B-4.

(2) MCPRs are determined by a linear interpolation between the Option A MCPR and the Option B MCPR for all plants choosing to operate under Option B which do not meet the scram time specification. This interpolation is based on the tested measured scram time and is described in Reference B-4.

REFERENCES

B-1. Letter, R. P. Denise (NRC) to G. G. Sherwood (GE), January 23, 1980.

B-2. Letter, R. H. Buchholz (GE) to P. S. Check (NRC), "ODYN Adjustment Method for Determination of Operating Limits", January 19, 1981.

B-3. "Generic Reload Fuel Application", NEDE-24011-P-A-1, August 1979.

B-4. Letter (with attachment), R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model", September 5, 1980.

APPENDIX C

BUNDLE LOADING ERROR EVENT ANALYSES

The bundle loading error analyses, procedures, and results for the rotated bundle are presented below. The mislocated bundle loading error event analysis is no longer being reported as discussed in Reference C-2.

C.1 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 C-1. This new method of performing the analysis is based on a more accurate detailed analytical model.

The principal 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 rotated 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 results of the analysis indicate that the limiting event is a rotated F8DRB284L bundle resulting in a 17.5 kW/ft LHGR and a 0.15 ΔCPR (includes a 0.02 penalty due to variable water gap R-factor uncertainty) with a minimum CPR of >1.07. The LHGR value includes a 2.2% power peaking penalty due to fuel densification.

REFERENCES

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

C-2. Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "Change in General Electric Methods for Analysis of Mislocated Bundle Accident", November 14, 1980.

APPENDIX D
DENSIFICATION POWER SPIKING

Reference D-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 those operating BWRs for which it can be shown that the predicted worst-case minimum transient LHGRs, when augmented by the power spike penalty, do not violate the exposure-dependent safety limit LHGRs".

The Browns Ferry-1 Reload-4 submittal contains the required information to remove the power spiking penalty from the Technical Specifications. Section 10 (Rod Withdrawal Error), Appendix A (Fuel Loading Error) and Appendix C (New Loading Error Event Analyses Procedures) include the densification effect in the calculated LHGR.

REFERENCES

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

APPENDIX E

LEAD TEST ASSEMBLIES

In Spring 1981, Browns Ferry-1 will load four lead test assemblies (LTAs) which are exactly the same as the standard P8DRB284L reload bundle except for a small axial section of increased Gadolinia content in some rods. Test measurements will be performed on these bundles during Cycle 5 to benchmark the effect of this increased Gadolinia content. All approved thermal-mechanical and reload methods described in NEDE-24011-P-A, "General Electric Standard Application for Reload", will hold for these LTAs. Results of the reload analyses are given in Sections 2 through 16 of this report.

Since the LTAs are essentially the same as the standard P8DRB284L reload bundles except for the small axial section of increased Gadolinia, the LHGR, CPR, and MAPLHGR limits for the standard bundles will apply. For the nodal regions of increased Gadolinia, appropriate local peaking factors will be provided for the process computer.