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Class I
August 1979
Revision 1

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
BRUNSWICK STEAM ELECTRIC PLANT
UNIT 2 RELOAD 2

REVISION 1: REVISED TRANSIENT
AND GETAB ANALYSES

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1. PLANT-UNIQUE ITEMS (1.0)*

Rotated Bundle Analysis Procedure: Appendix A

Total Number and Capacity of Safety/Relief Valves: Reference 2

Fuel Loading Error LHGR: Appendix B

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

	Fuel Type	Number	Number Drilled
Irradiated	Initial Core Type 1	108	108
	Initial Core Type 3	176	176
	7DB230	4	4
	8DB274L	100	100
	8DB274H	40	40
New	8DRB265H	64	64
	8DRB283	68	68
Total		560	560

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 11,570 MWd/t

Assumed reload cycle exposure: 13,080 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	1.120
Fully Controlled	0.958
Strongest Control Rod Out	0.989

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	0.032

*( ) refers to areas of discussion in Reference 1.

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

	EOC	
Void Coefficient N/A* (¢/% Rg)	7.89/9.86	***
Void Fraction (%)	41.76	
Doppler Coefficient N/A (¢/% °F)	0.1937/0.1840	
Average Fuel Temperature (°F)	1538	
Scram Worth N/A ($)	38.75/31.00	
Scram Reactivity	Figure 2	

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

Exposure	7x7 EOC	8x8 EOC	8x8R EOC	
Peaking factors (local, radial and axial)	1.24/1.260/1.40	1.22/1.349/1.40	1.20/1.493/1.40	***
R-Factor	1.100	1.098	1.051	
Bundle Power (MWt)	5.379	5.752	6.362	***
Bundle Flow (10^3 lb/hr)	125.26	115.54	115.19	***
Initial MCPR	1.22	1.29	1.28	***

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

None

*N = Nuclear Input Data
A = Used in Transient Analysis
*** Denotes change from Rev. 0

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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
Generator Load Rejection w/o Bypass	BOC-EOC3	104	100	269.9	109.5	1169	1216	0.15	0.22	Figure 3 ***
Inadvertent HPCI Pump Start	---	104	100	122.4	113.1	1018	1067	0.11	0.14	Figure 4
Feedwater Controller Failure	BOC-EOC3	104	100	109.0	105.1	1028	1076	0.05	0.06	Figure 5 ***

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

Rod Block Reading	Rod Position (Feet Withdrawn)	ΔCPR 7x7	ΔCPR 8x8	ΔCPR 8x8R	MLHGR (kW/ft) 7x7	MLHGR (kW/ft) 8x8	MLHGR (kW/ft) 8x8R	Limiting Rod Pattern
104	4.0	0.13	0.10	0.19	18.0	15.3	12.5	Figure 6
105*	4.0	0.13	0.10	0.19	18.0	15.3	12.5	Figure 6
106	4.5	0.15	0.11	0.22	18.8	16.3	13.1	Figure 6
107	5.0	0.17	0.13	0.25	19.4	16.8	13.6	Figure 6
108	5.5	0.20	0.14	0.27	19.8	17.3	14.1	Figure 6
109	6.0	0.22	0.16	0.29	20.0	17.5	14.4	Figure 6
110	9.0	0.24	0.24	0.36	18.2	16.5	14.3	Figure 6

11. OPERATING MCPR LIMIT (5.2)

BOC3 - EOC3		
1.29	(8x8/8x8R fuel)	***
1.22+	(7x7 fuel)	***

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	Power (%)	Core Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	104	100	1214	1259	Figure 7 ***

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*Indicates setpoint selected.

***Denotes change from Rev. 0.

+If scoop tube block is set at 102.5% flow and 112% flow K_f curve is used.

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)	0.62

Channel Hydrodynamic Performance

	Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power)
8x8/8x8R channel	0.28
7x7 channel	0.13

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

8DRB265

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (^oF)	Local Oxidation Fraction
200	11.5	2154	0.030
1,000	11.6	2156	0.029
5,000	11.9	2192	0.032
10,000	12.0	2196	0.032
15,000	12.0	2200	0.033
20,000	11.8	2197	0.033
25,000	11.3	2138	0.027
30,000	10.7	2056	0.021

8DRB283

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (^oF)	Local Oxidation Fraction
200	11.2	2122	0.027
1,000	11.2	2117	0.026
5,000	11.8	2184	0.032
10,000	12.0	2197	0.033
15,000	11.9	2194	0.032
20,000	11.8	2197	0.033
25,000	11.3	2132	0.027
30,000	11.1	2106	0.025

15. LOADING ERROR RESULTS* (5.5.4, Appendix A)

Limiting Event: **Rotated** bundle 8DRB283 or 8DRB265H

MCPR: 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

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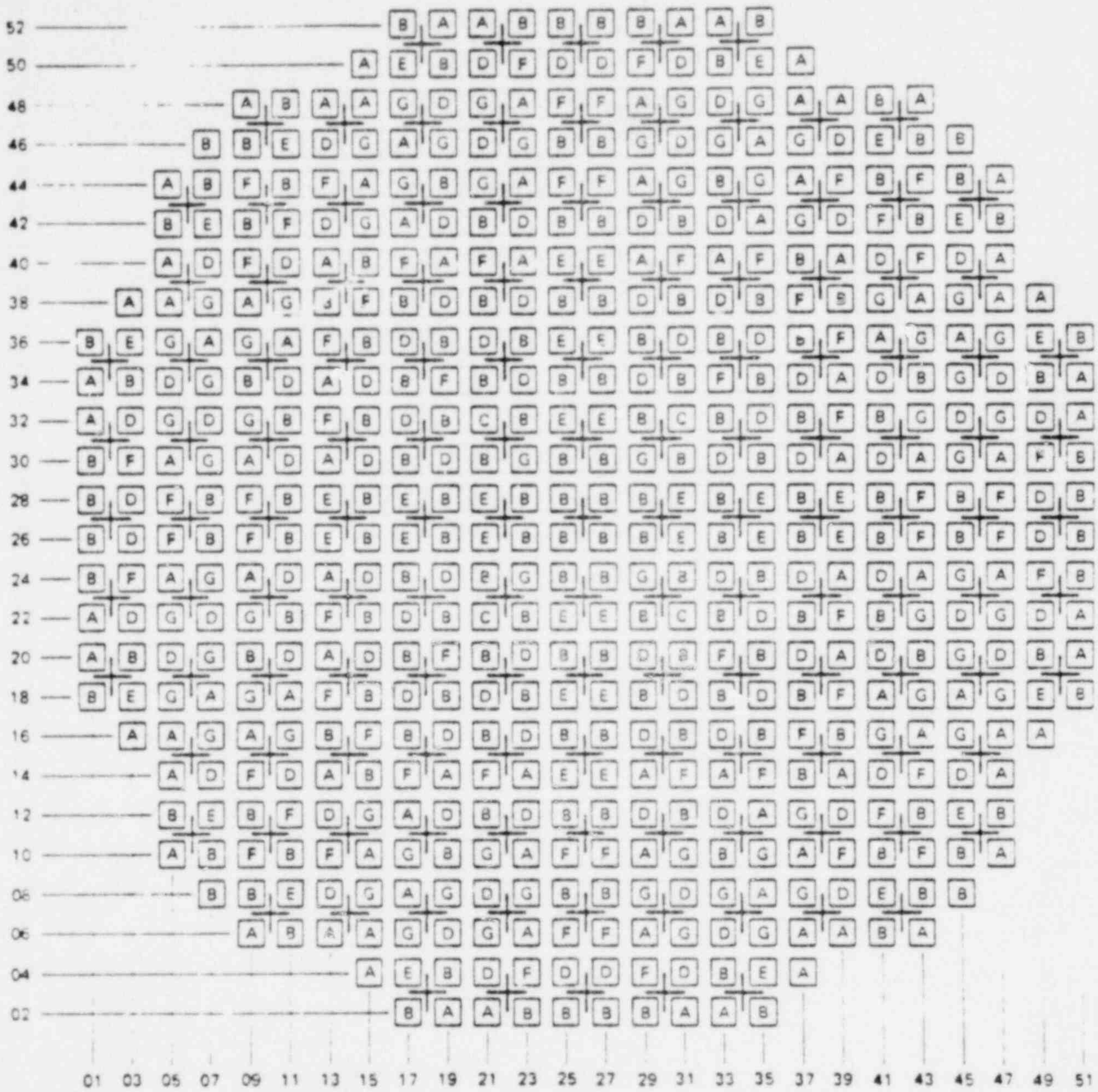
*Using New Rotated Bundle Analysis Procedures described in Appendix A.

**Includes added penalty of 0.02 imposed by NRC.

POOR ORIGINAL



Fuel Type	
A=Init Core Type 1 B=Init Core Type 3 C=Gen B (7DB230) D=8DB274L	E=8DB274H F=8DRB265H G=8DRB283

Figure 1. Reference Core Loading Pattern

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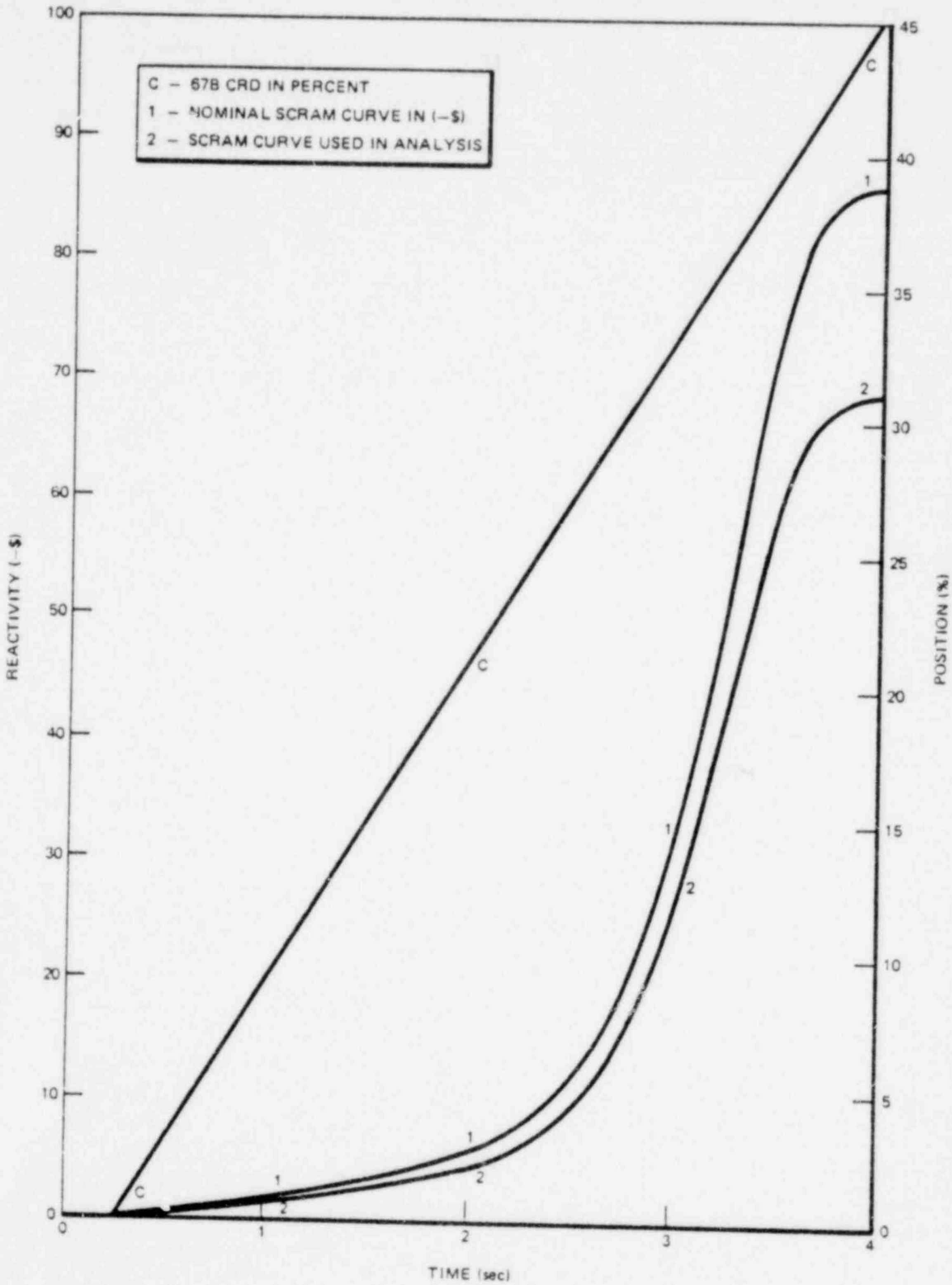

Figure 2. Scram Reactivity and Control Rod Drive Specifications

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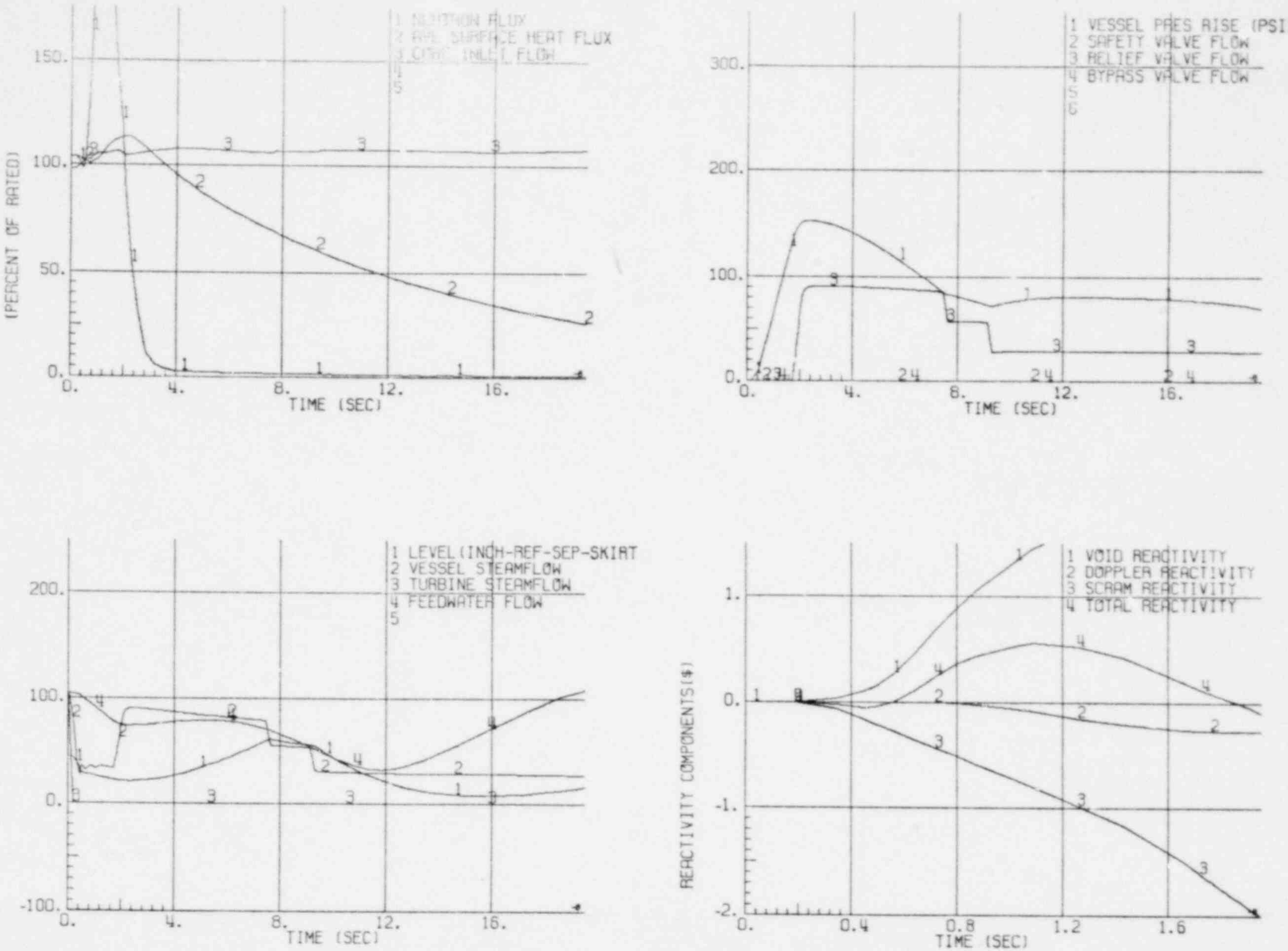

Figure 3. Plant Response to Generator Load Rejection Without Bypass

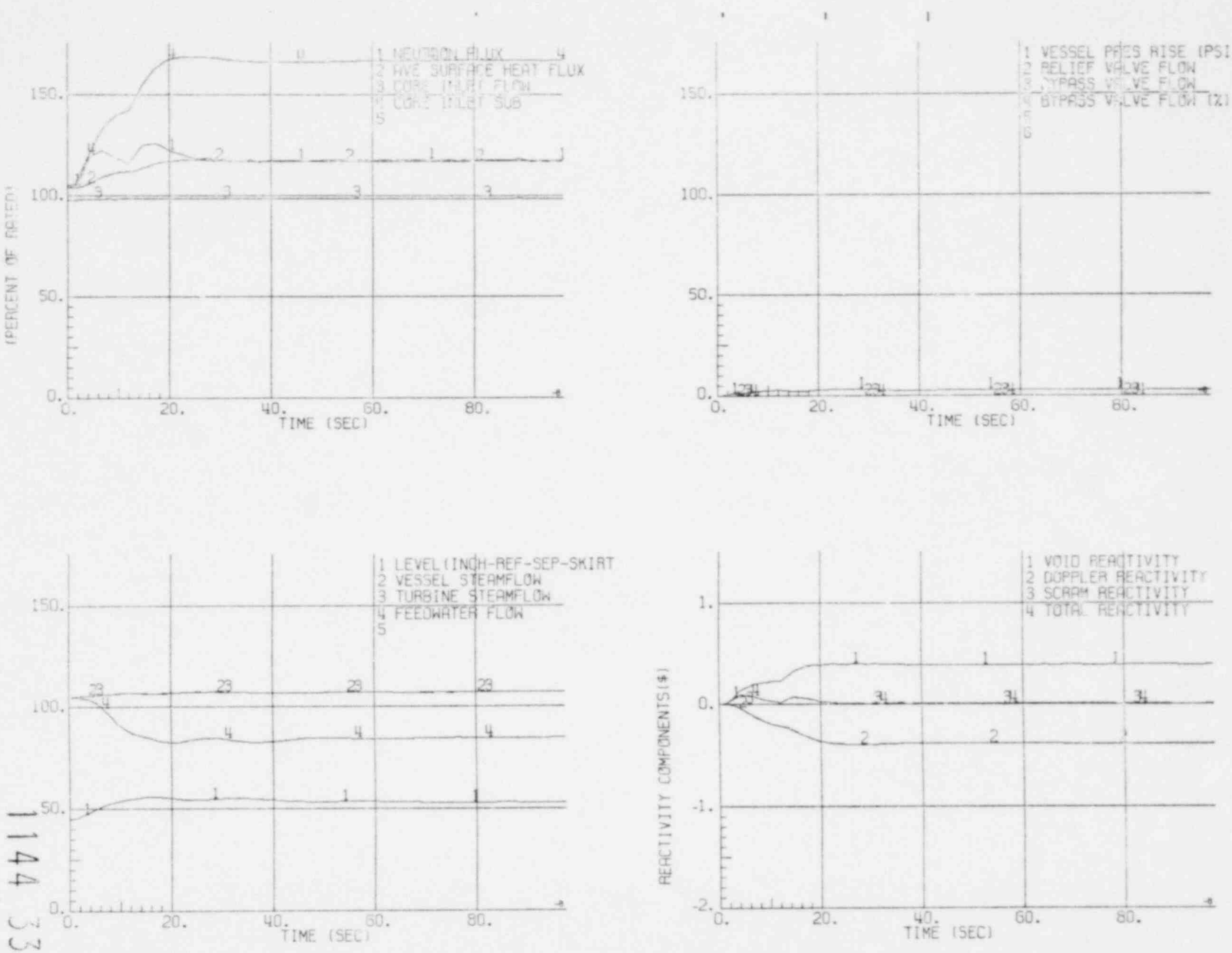

Figure 4. Plant Response to Inadvertent Startup of HPCI Pump

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POOR ORIGINAL

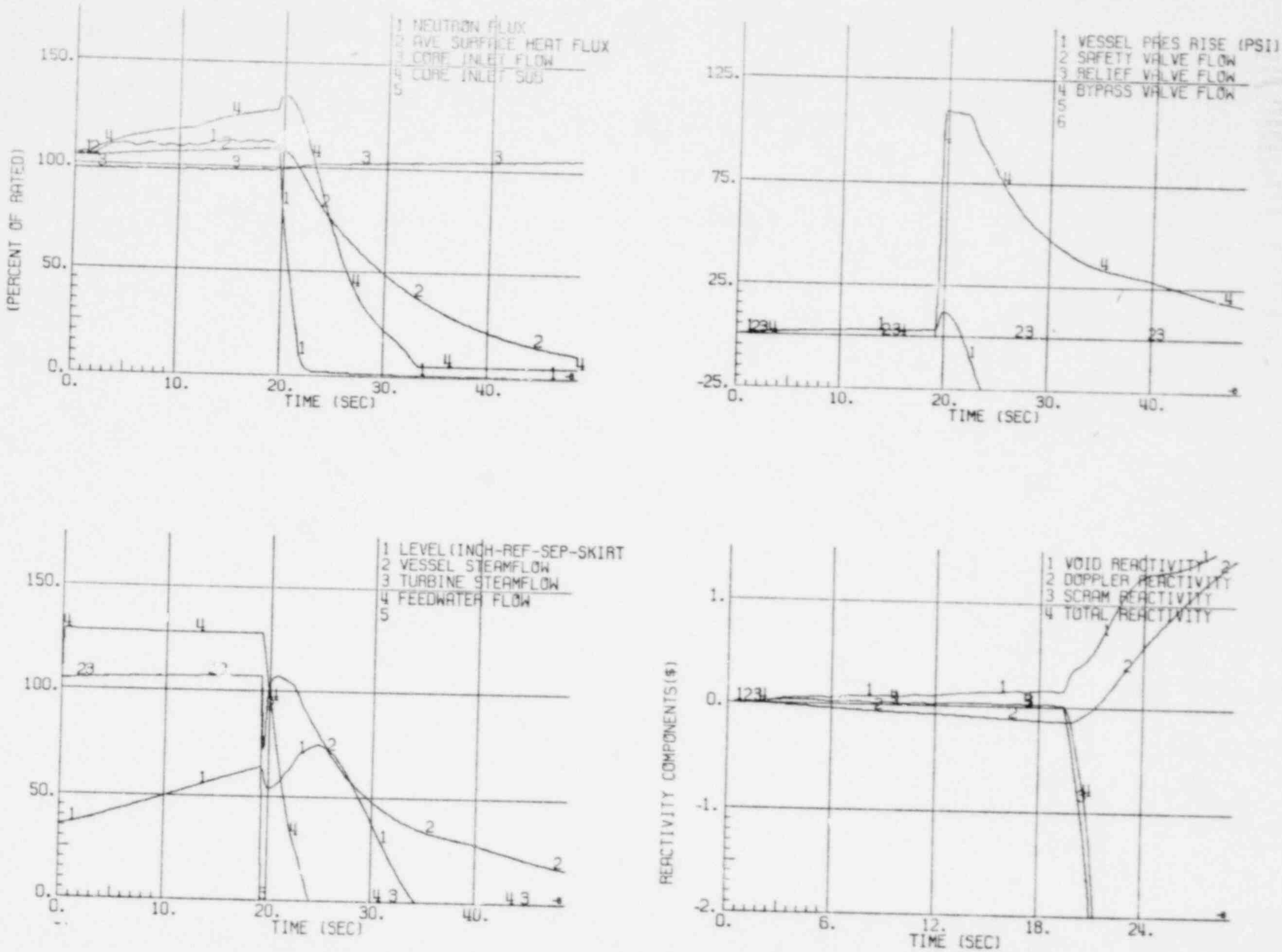

Figure 5. Plant Response to Feedwater Controller Failure

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	02	06	10	14	18	22	26	30
51								
47								
43				34		6		
39							32	
35		34		4		2		
31					28			
27		6		2		0		
23								

NOTES:
1. ROD PATTERN IS 1/4 CORE MIRROR SYMMETRIC. UPPER LEFT QUADRANT SHOWN ON MAP
2. NUMBERS INDICATE NUMBER OF NOTCHES WITHDRAWN OUT OF 48. BLANK IS A WITHDRAWN ROD
3. ERROR ROD IS (22,27)

Figure 6. Limiting RWE Rod Pattern

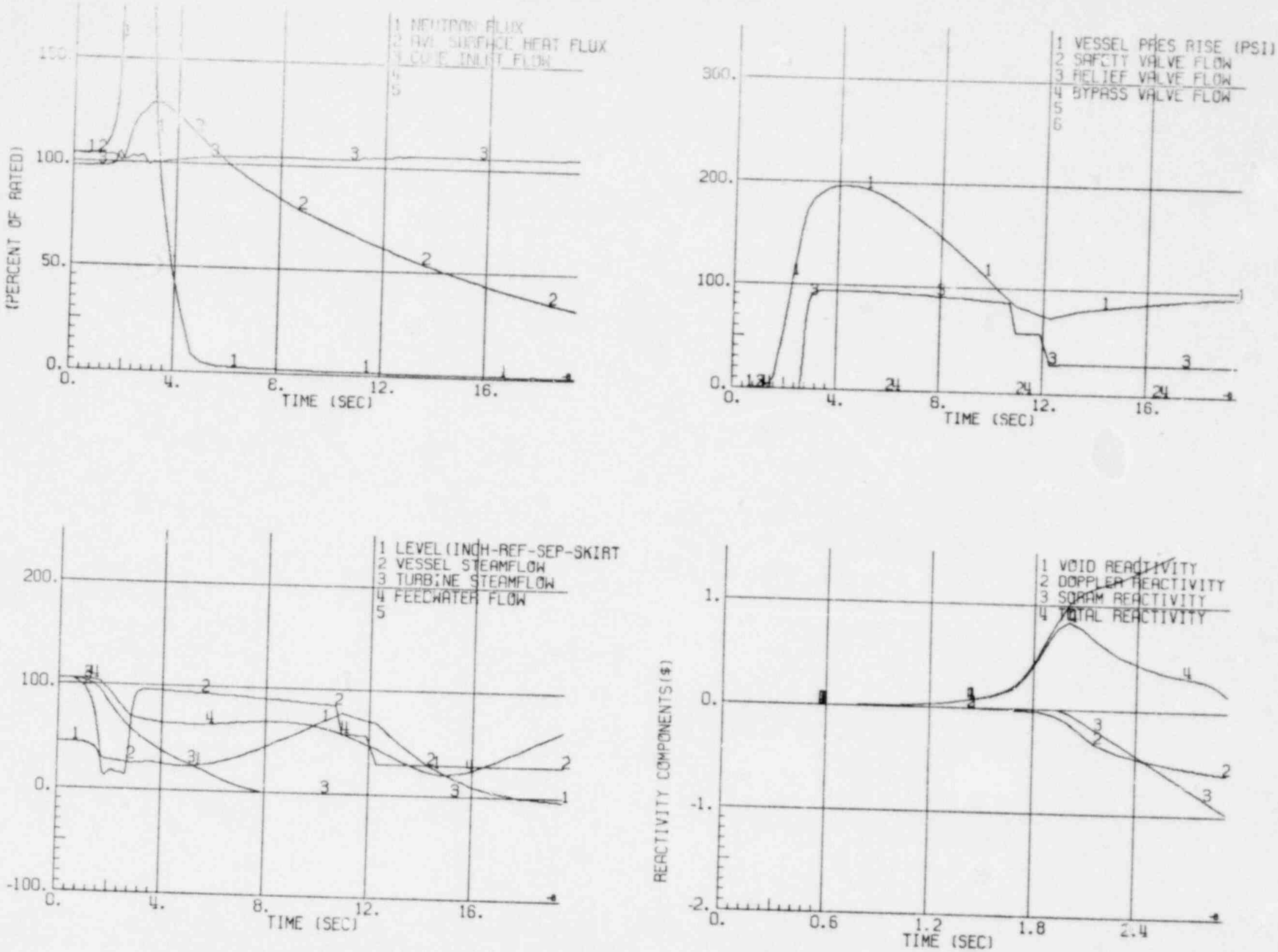

Figure 7. Plant Response to MSIV Closure

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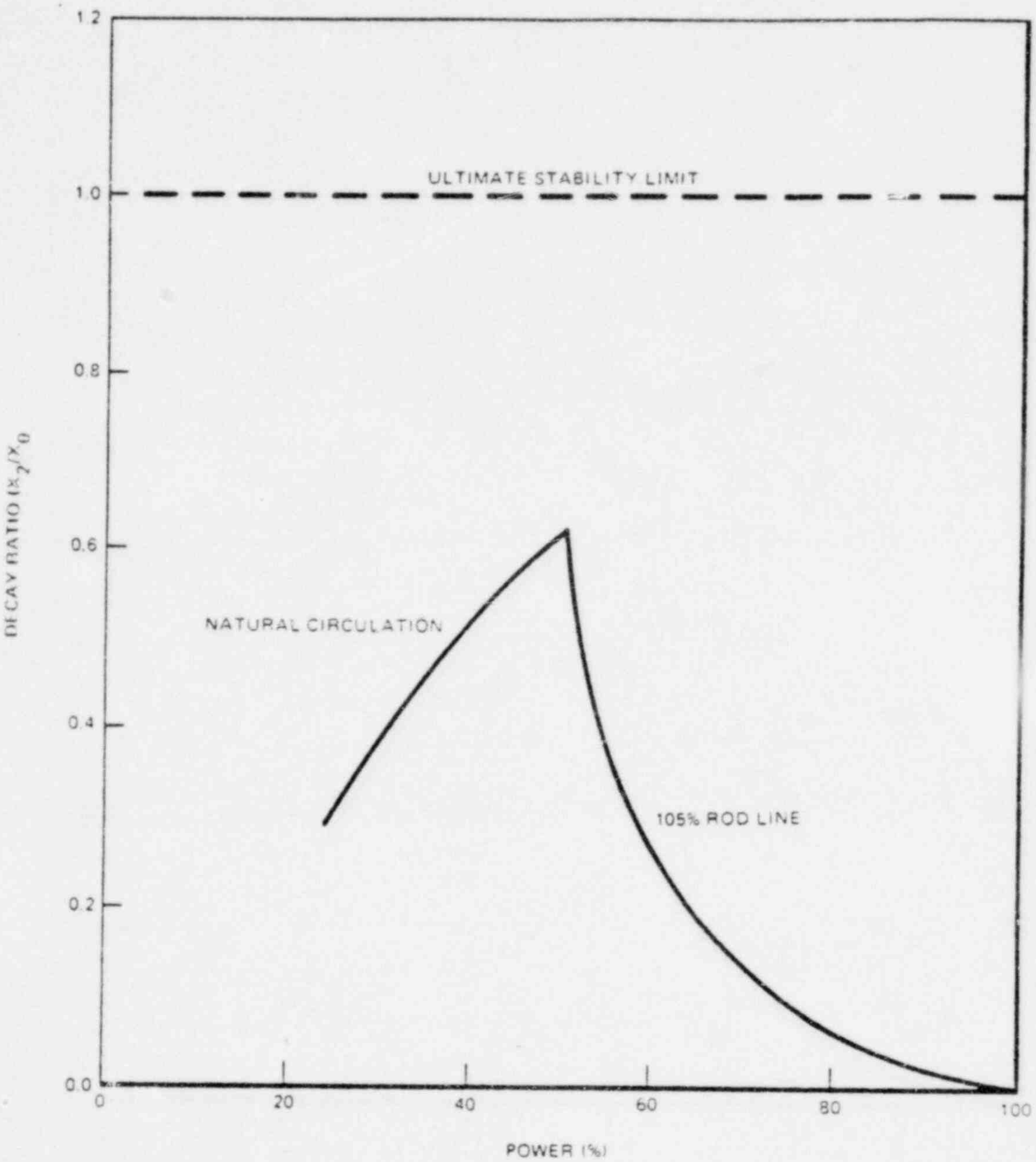

Figure 8. Decay Ratio

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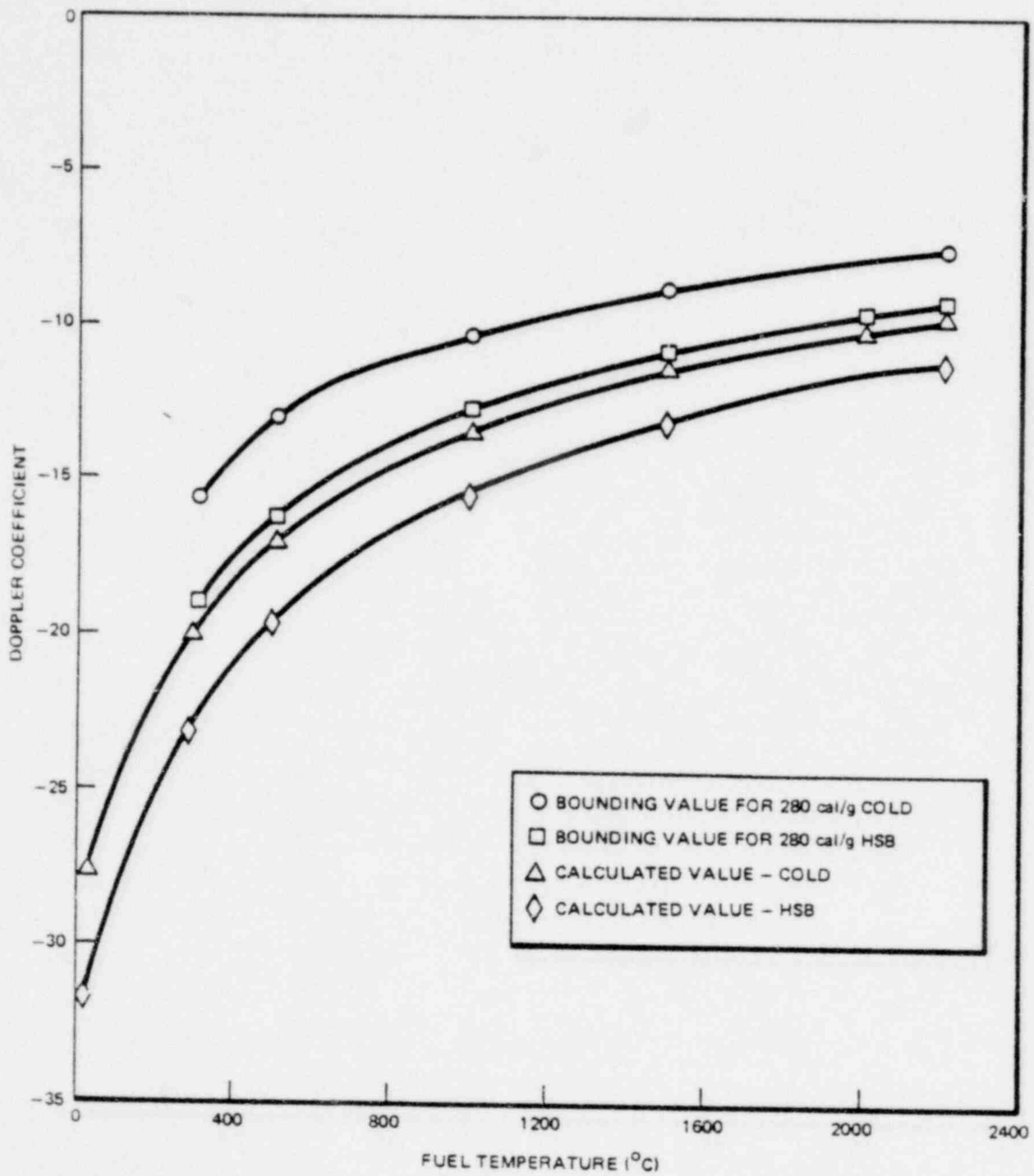

Figure 9. Doppler Reactivity Coefficient Comparison for RDA

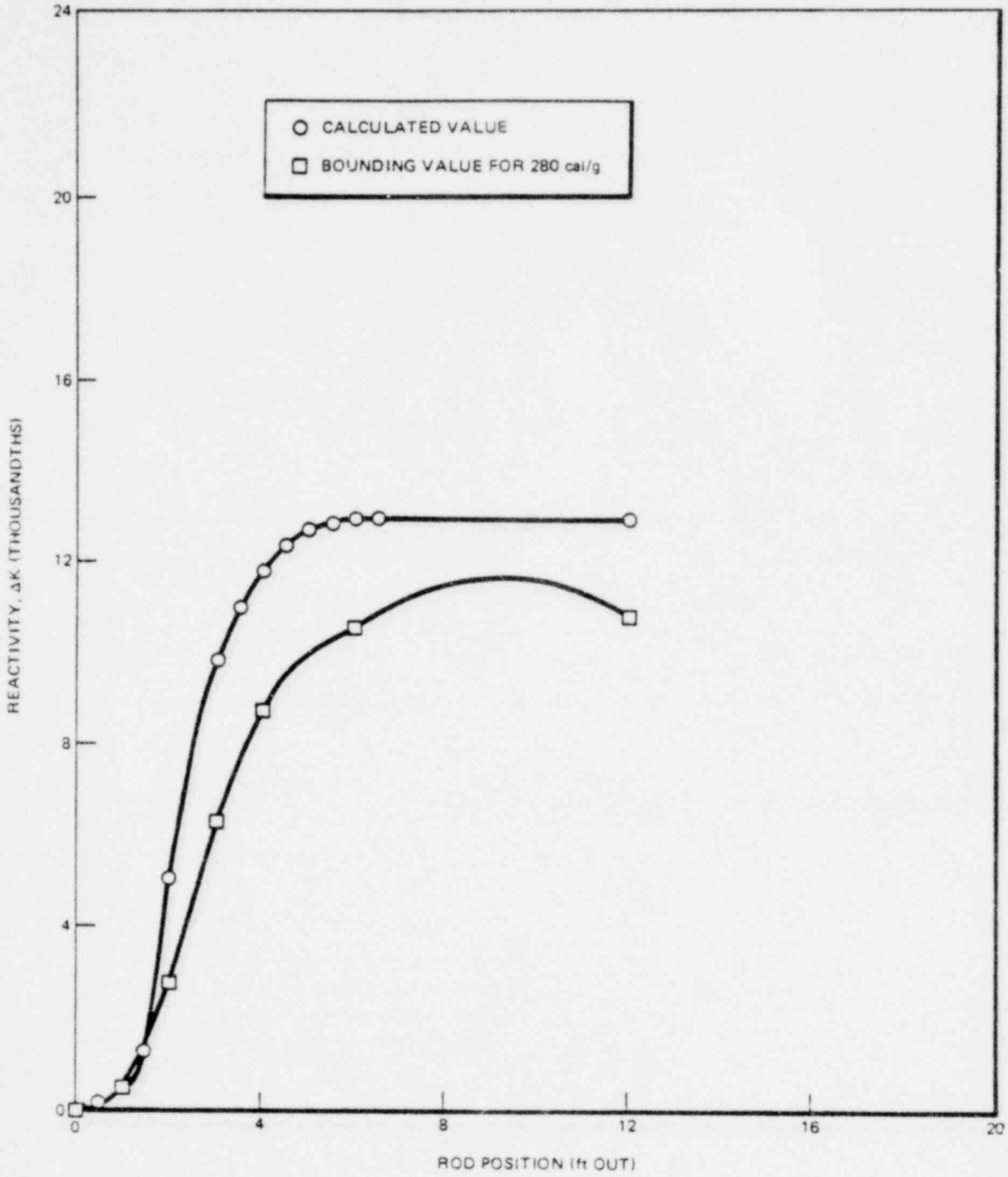

Figure 10. RDA Reactivity Shape Function

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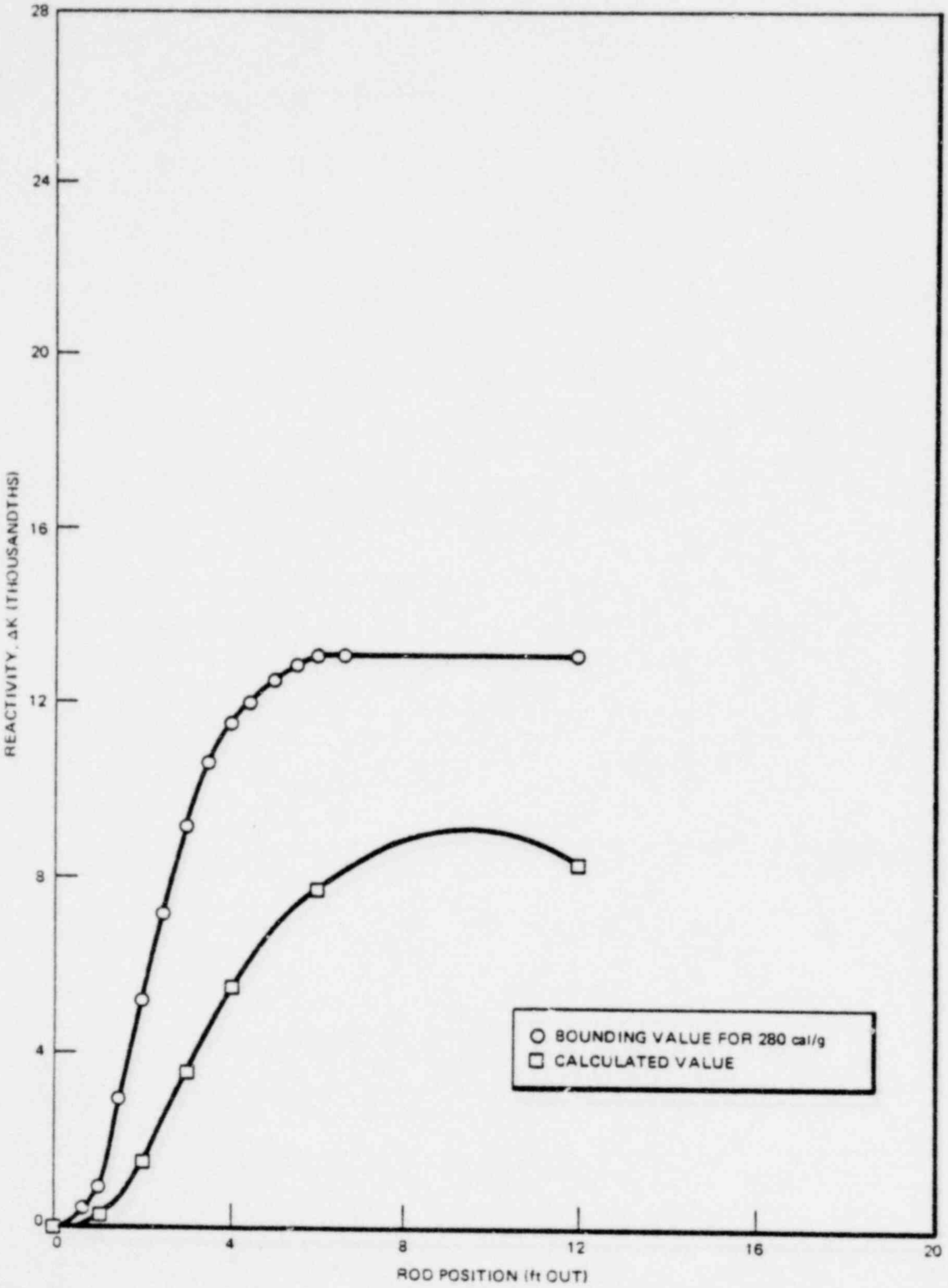

Figure 11. RDA Reactivity Shape Function at 286°C

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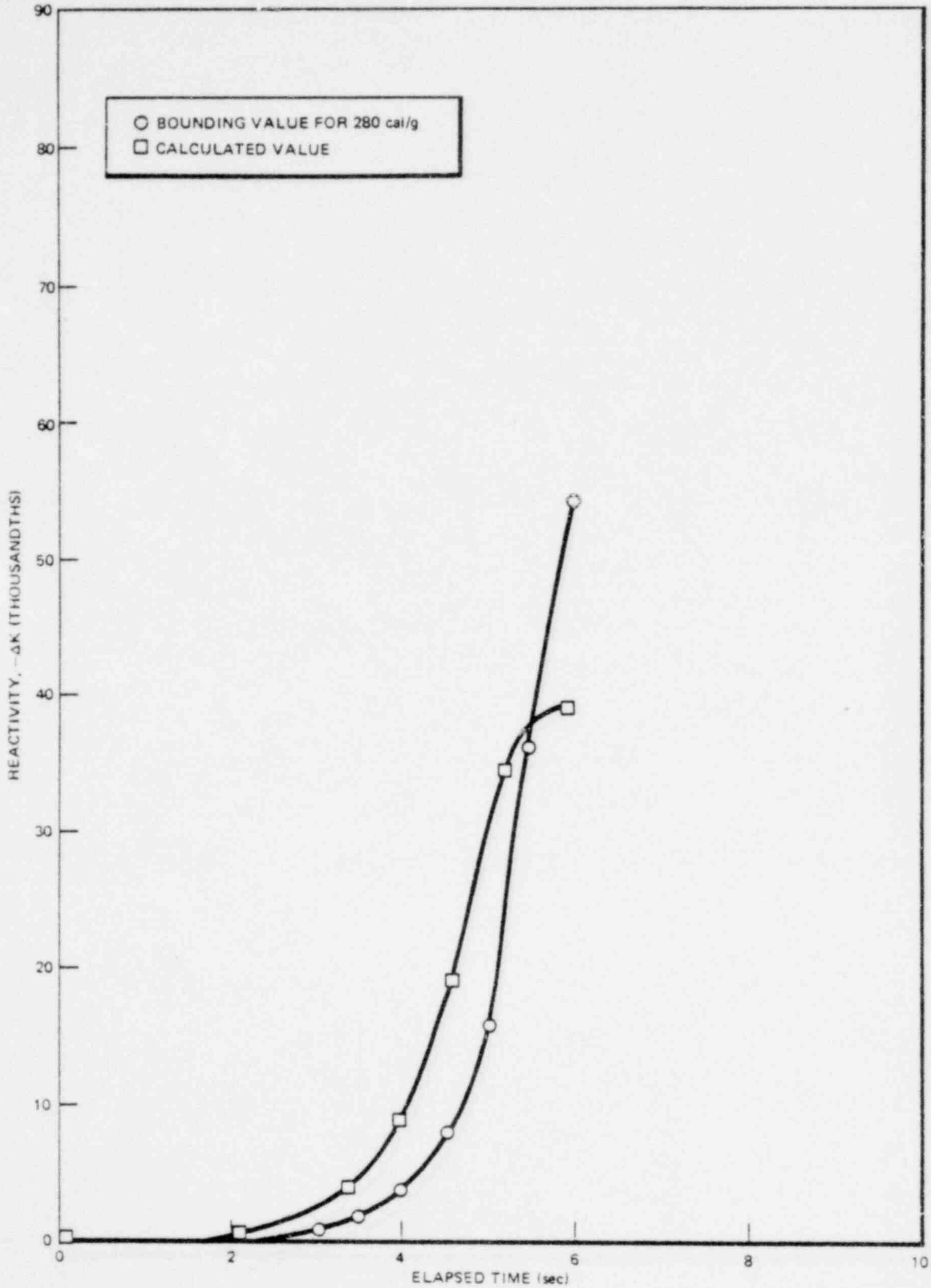

Figure 12. RDA Scram Reactivity Function at 20°C

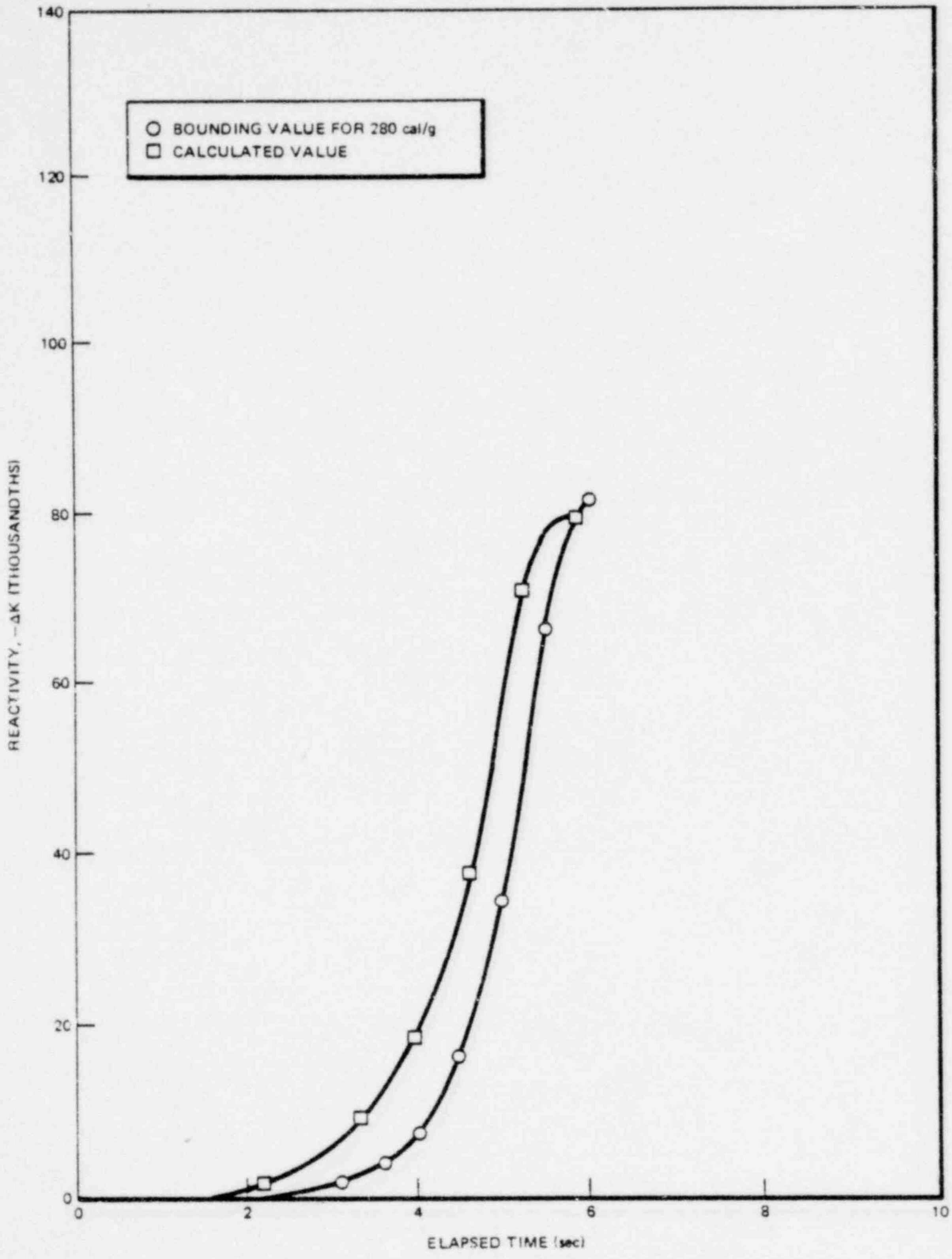

Figure 13. RDA Scram Reactivity Function at 286°C

REFERENCES

1. "General Electric Boiling Water Generic Fuel Application," NEDE-24011-P, Revision 3, March 1978.

2. Letter No. NG-77-1060 from E. E. Utley (CP&L) to A. Schwencer (NRC), September 20, 1977.

APPENDIX A

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 the rotated bundle loading error events. The use of this new analysis procedure is discussed below.

A.1 NEW ANALYSES PROCEDURE FOR THE ROTATED BUNDLE LOADING ERROR EVENT

The rotated bundle loading error event analyses results presented in this supplement are based on the new analyses procedure described in References A-1 and A-2. This new method of performing the analyses is based on a more detailed analysis model, which reflects more accurate analyses than that used in previous analyses of this event.

The principle difference between the previous analyses procedure and the new analyses procedure is the modeling of the water gap along the axial length of the bundle. The previous analyses used a uniform water gap, whereas the new analyses utilize a variable water gap which is representative of the actual condition. 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 ΔCPR 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.

In the new analyses, the axial alignment of a 180° rotated bundle conservatively ignores the presence of the channel fastener. The more limiting condition of assuming that the spacer buttons are in contact with the top guide is assumed. There is no known loading that could bend or break the channel spacer button during the insertion of a 180° rotated bundle, since both the top guide and spacer button are chamfered to provide lead-in. For a properly assembled bundle, no mechanism exists which could invalidate the assumption that a 180° rotated bundle leans to one side.

It should be noted that proper orientation of bundles in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. Five separate visual indications of proper bundle orientation exist:

(1) The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.

(2) The identification boss on the fuel assembly handle points toward the adjacent control rod.

(3) The channel spacing buttons are adjacent to the control rod passage area.

(4) The assembly identification numbers which are located on the fuel assembly handles are all readable from the direction of the center of the cell.

(5) There is cell-to-cell replication.

Experience has demonstrated that these design features are clearly visible so that any misloaded bundle would be readily identifiable during core loading verification. Figures A-1, A-2 and A-3 denote a normally loaded bundle, a 180° rotated bundle, and a 90° rotated bundle, respectively. Actual experience (References A-1 and A-2) has demonstrated that the probability of a rotated bundle is low.

The new analyses procedure results show that the minimum CPR for the most limiting rotated bundle in the core is greater than the safety limit.

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REFERENCES

A-1 Letter, R. E. Engel (GE) to D. Eisenhut (NRC), "Fuel Assembly Loading Error," MFN-219-77, June 1, 1977.

A-2 Letter, R. E. Engel (GE) to D. Eisenhut (NRC), "Fuel Assembly Loading Error," MFN-457-77, November 30, 1977.

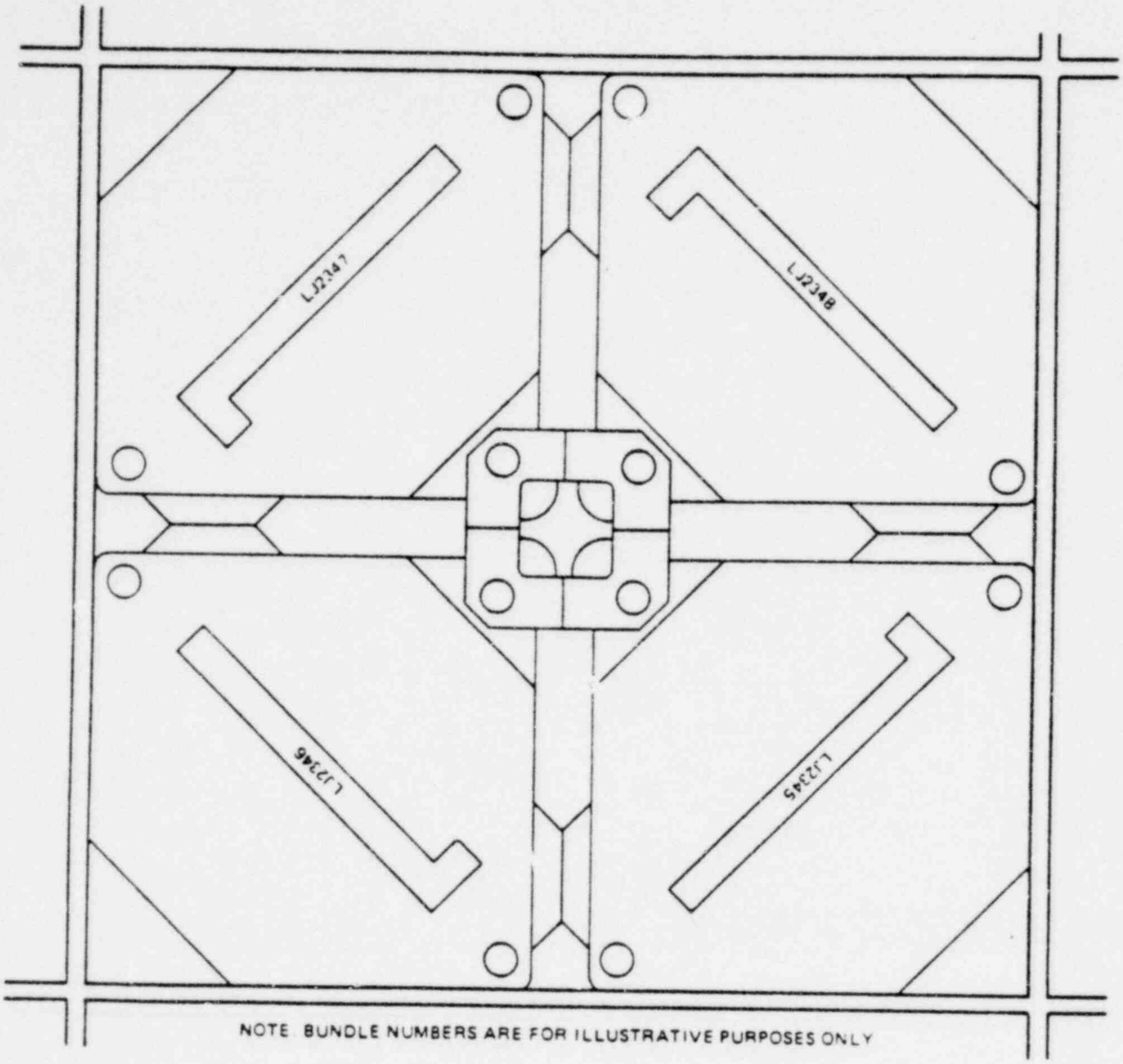

Figure A-1. Normal Loading

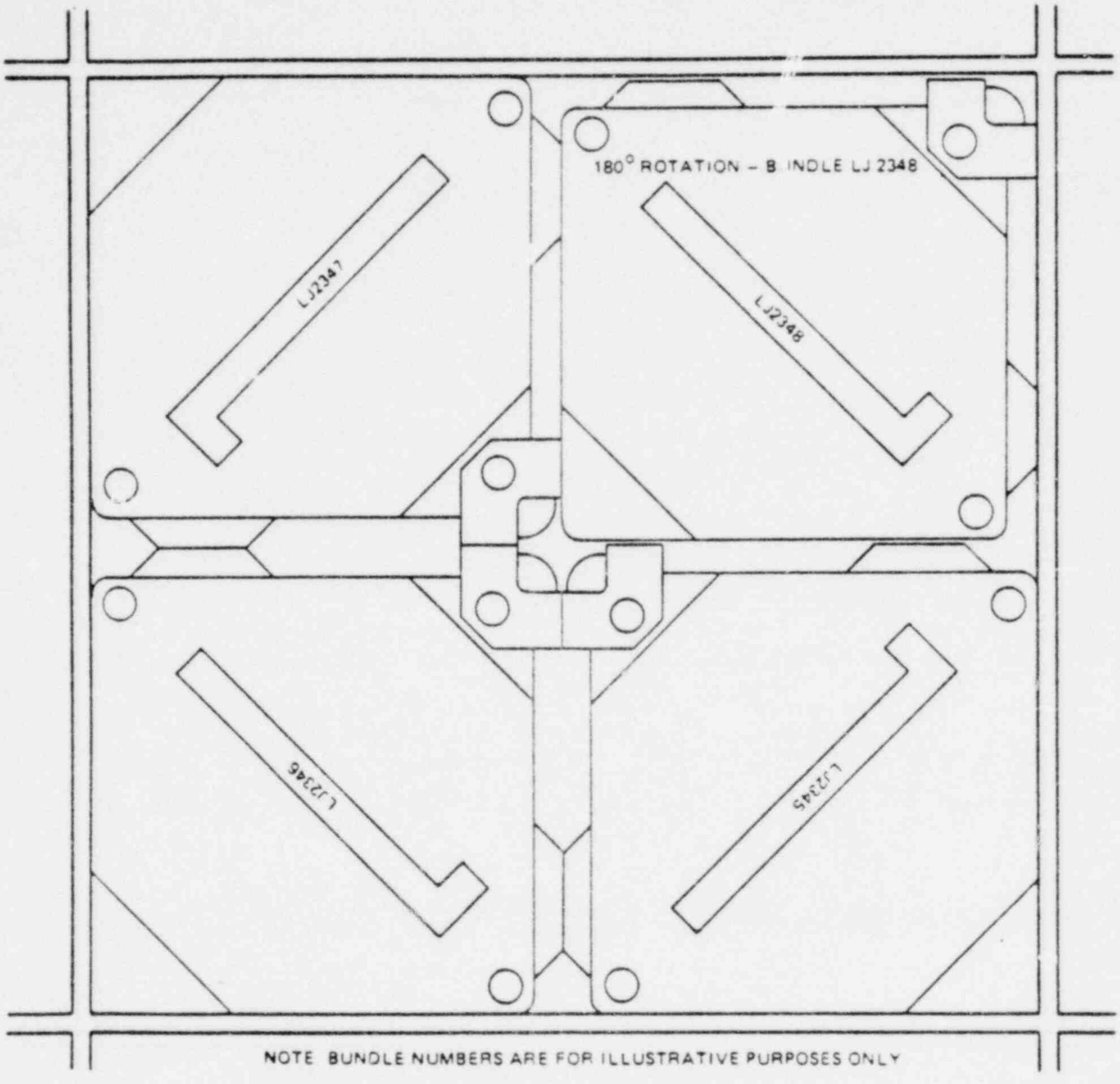

Figure A-2. Rotated Bundle, 180 Degree Rotation

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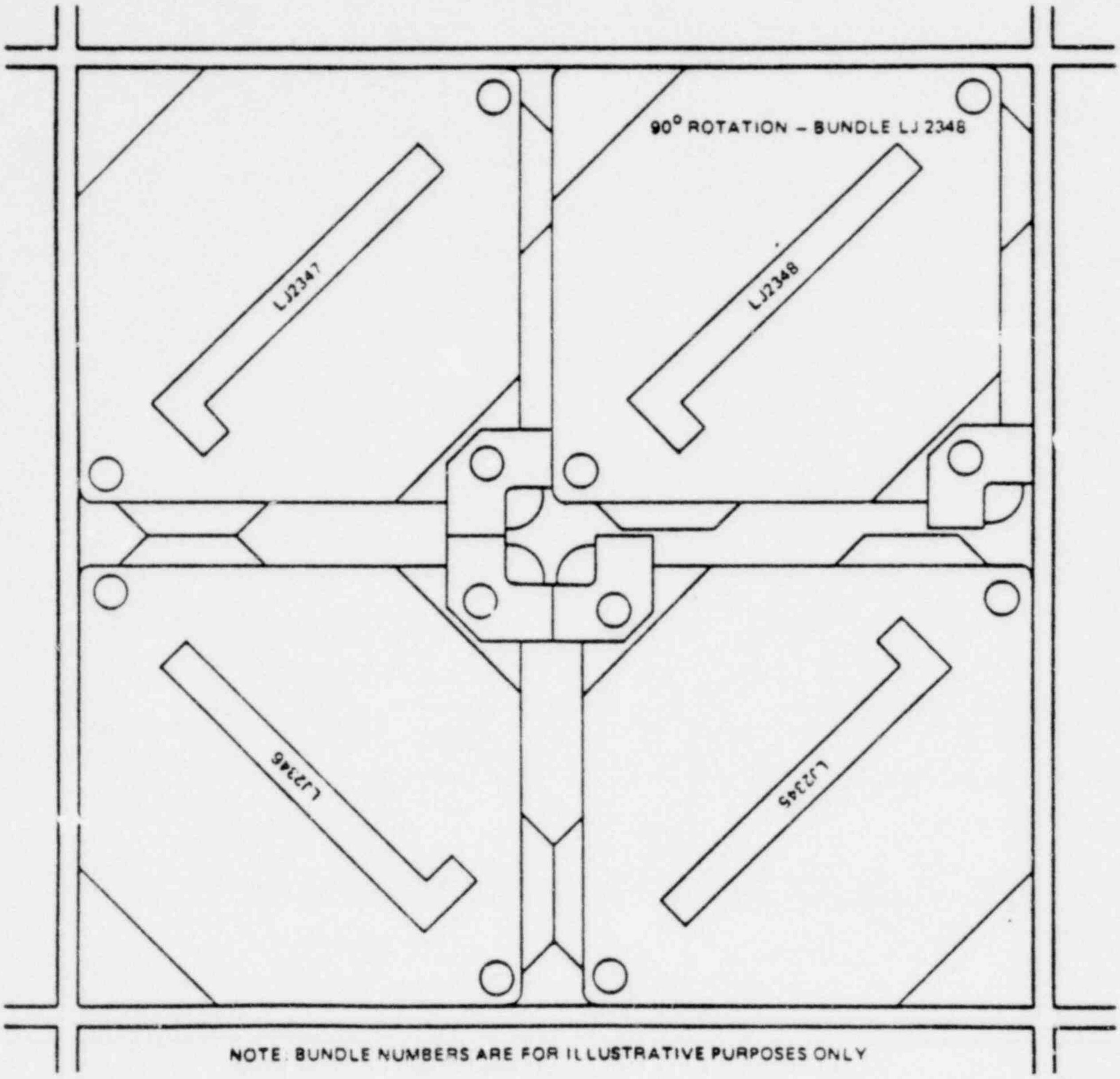

Figure A-3. Rotated Bundle, 90 Degree Rotation

APPENDIX B

Fuel Loading Error LHGR: 15.5 kW/ft

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