NEDO-24209
79NED294
CLASS I
AUGUST 1979

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BROWNS FERRY NUCLEAR PLANT UNIT 1 RELOAD NO. 3

POOR ORIGINAL

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NEDO-24209
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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

BROWNS FERRY NUCLEAR PLANT UNIT 1

RELOAD NO. 3

Prepared by: A.M. Ervin

Approved by: R. E. Engel, Acting Manager
Reload Fuel Licensing

1139 014

NUCLEAR ENERGY PROJECTS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

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1. PLANT-UNIQUE ITEMS (1.0)*

Fuel Loading Error LHGR: Appendix A

Parameters Different from Reference 1: Appendix A

Item 3. Revised Reference Core Loading Pattern Format: Appendix B

New Bundle Loading Error Event Analysis Procedures: Appendix C

Items 7, 9, 10, 11, 13: Format change to include additional 8x8R/P8x8R column

Item 13. Stability Analysis Results: Reference 1 (Appendix D)

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

	Fuel Type	Number	Number Drilled
Irradiated	Initial Core Type 2	100	100
Irradiated	Initial Core Type 3	108	108
Irradiated	8DB274H	24	24
Irradiated	8DB274L	144	144
Irradiated	8DRB265L	88	88
Irradiated	8DRB265H	68	68
New**	P8DRB284	232	232
TOTAL		764	764

3. REFERENCE CORE LOADING PATTERN (3.3.1)

See Appendix B.

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	1.122
Fully Controlled	0.961
Strongest Control Rod Out	0.989
R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, Δk	0.000

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

**"General Electric Boiling Water Reactor Generic Reload Fuel Application," August 1979, (NEDE-24011-P-A-1).

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5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.027

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

Void Coefficient N/A,* (c/% Rg)	-7.31/-9.135
Void Fraction (%)	40.18
Doppler Coefficient N/A (c/°F)	-0.2316/-0.22
Average Fuel Temperature (°F)	1383
Scram Worth N/A ($)	-37.1/-29.6
Scram Reactivity	Figure 2

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

	7x7	8x8	8x8R/P8x8R
Peaking factors (local, radial and axial)	1.24 1.31 1.40	1.22 1.42 1.40	1.20 1.56 1.40
R-Factor	1.10	1.098	1.052
Bundle Power (MWt)	5.517	5.980	6.569
Bundle Flow (10^3 lb/hr)	120.68	108.07	108.75
Initial MCPR	1.20	1.24	1.24

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Recirculation Pump Trip (REDY Code Only)

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

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9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Core Flow (%)	$\hat{\phi}$ (% NBR)	Q/A (% NBR)	P_{SL} (PSIG)	P_v (PSIG)	ΔCPR 7x7	ΔCPR 8x8	ΔCPR 8x8R/P8x8R	Plant Response
Generator Load Rejection without Bypass	EOC4	104.5	100	242	111.1	1199	1227	0.12	0.17	0.18	Figure 3
Loss of 100°F Feedwater Heating	-	104.5	100	123.7	123.3	1013	1069	0.13	0.15	0.15	Figure 4
Feedwater Controller Failure	EOC4	104.5	100	170.6	112.2	1157	1190	0.08	0.12	0.12	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/P8x8R	LHGR 7x7	LHGR 8x8	LHGR 8x8R/P8x8R	Limiting Rod Pattern
104	3.0	0.10	0.11	0.10	16.0	12.90	14.07	Figure 6
105	3.5	0.15	0.15	0.13	17.56	14.05	14.95	Figure 6
106	3.5	0.15	0.15	0.13	17.56	14.05	14.95	Figure 6
107	4.0	0.24	0.18	0.15	18.57	14.98	16.27	Figure 6
108	4.5	0.28	0.20	0.17	18.75	15.25	16.76	Figure 6
109	4.5	0.28	0.20	0.17	18.75	15.25	16.76	Figure 6
110	5.0	0.31	0.24	0.18	18.84	15.47	17.03	Figure 6

11. OPERATING MCPR LIMIT (5.2)

1.23 (7x7 fuel)

1.24 (8x8 fuel)

1.25 (8x8R/P8x8R)

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	1236	1272	Figure 7

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*Indicates setpoint selected.

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13. STABILITY RESULTS (5.4)

Decay Ratio:	Figure 8
Reactor Core Stability Decay Ratio, x_2/x_0:	0.757
Channel Hydrodynamic Performance Decay Ratio, x_2/x_0	
7x7 channel	0.262
8x8 channel	0.406
8x8R/P8x8R channel	0.327

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

P8DRB284

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
10000	12.0	1647	0.003
15000	12.0	1669	0.003
20000	11.8	1672	0.003
25000	11.?	1633	0.003
30000	10.8	1596	0.002

15. LOADING ERROR RESULTS (5.5.4)

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: Scram Reactivity Shape Function at 20°C

Resultant peak enthalpies (cal/g): 152.63

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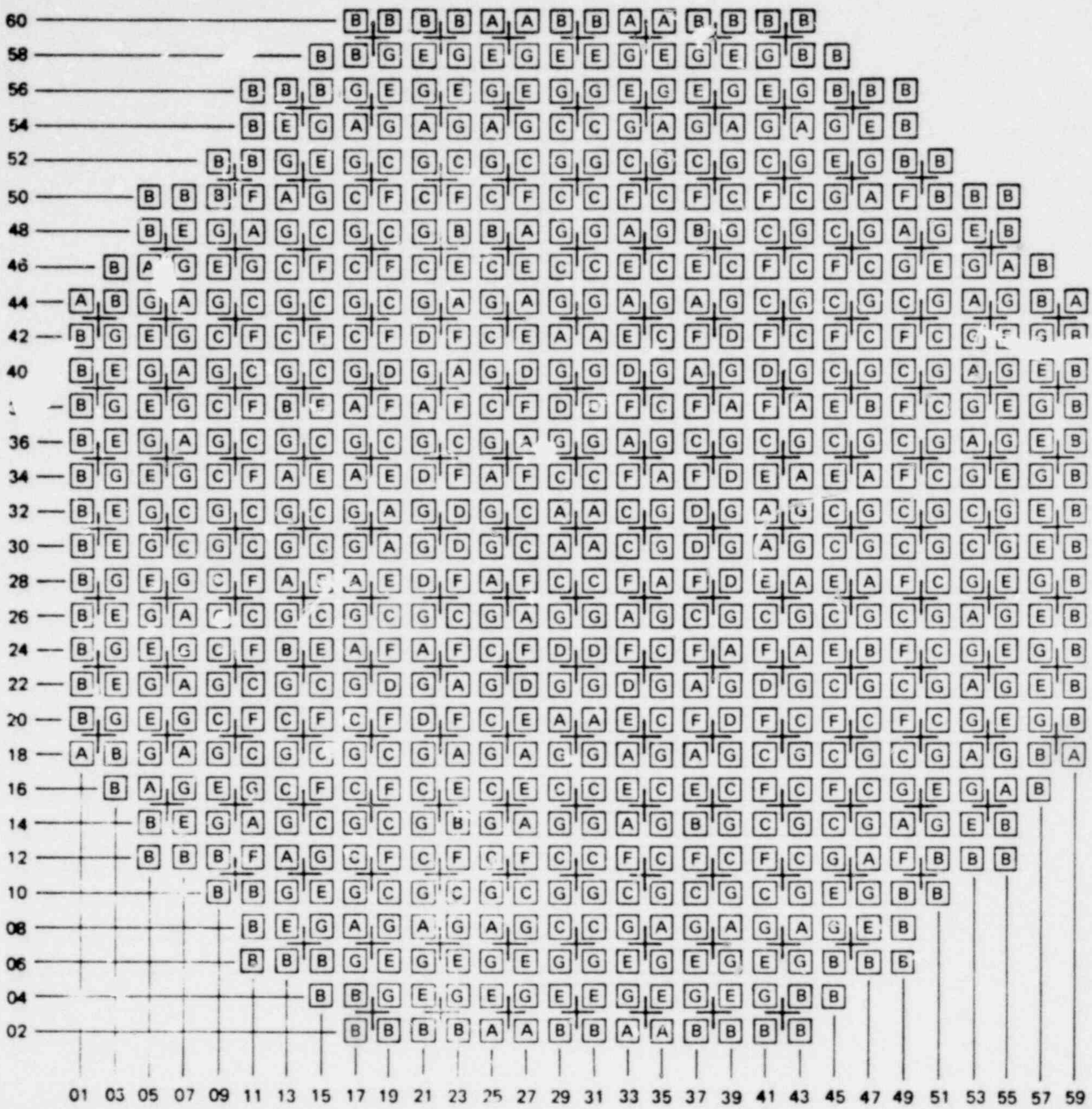

FUEL TYPE	
A - INITIAL CORE TYPE 2	E = 8DRB265L
B - INITIAL CORE TYPE 3	F = 8DRB265H
C - 8DB274L	G = P8DRB284
D - 8DB274H	H -

Figure 1. Reference Core Loading Pattern

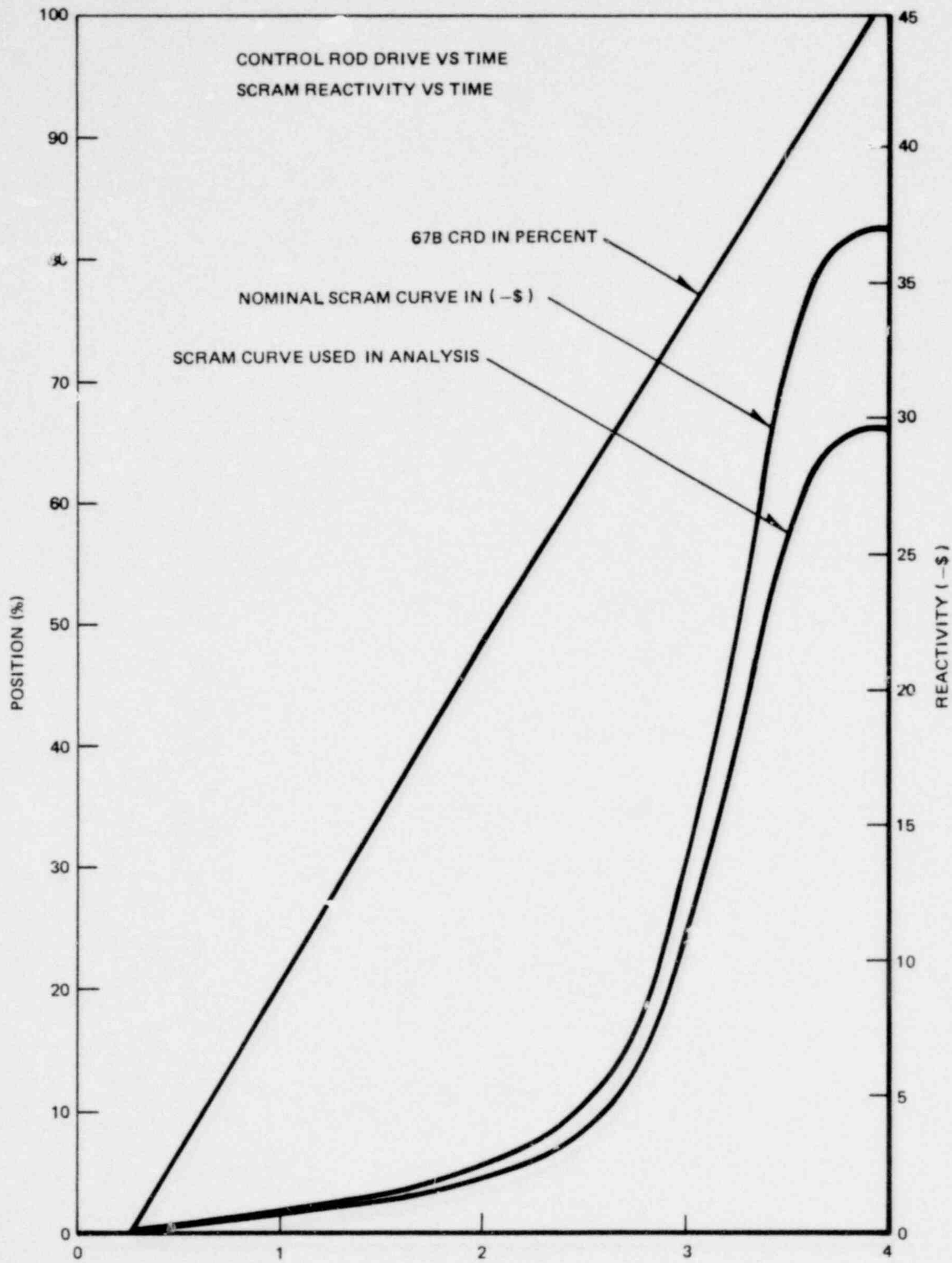

Figure 2. Scram Reactivity and Control Rod Drive Specification

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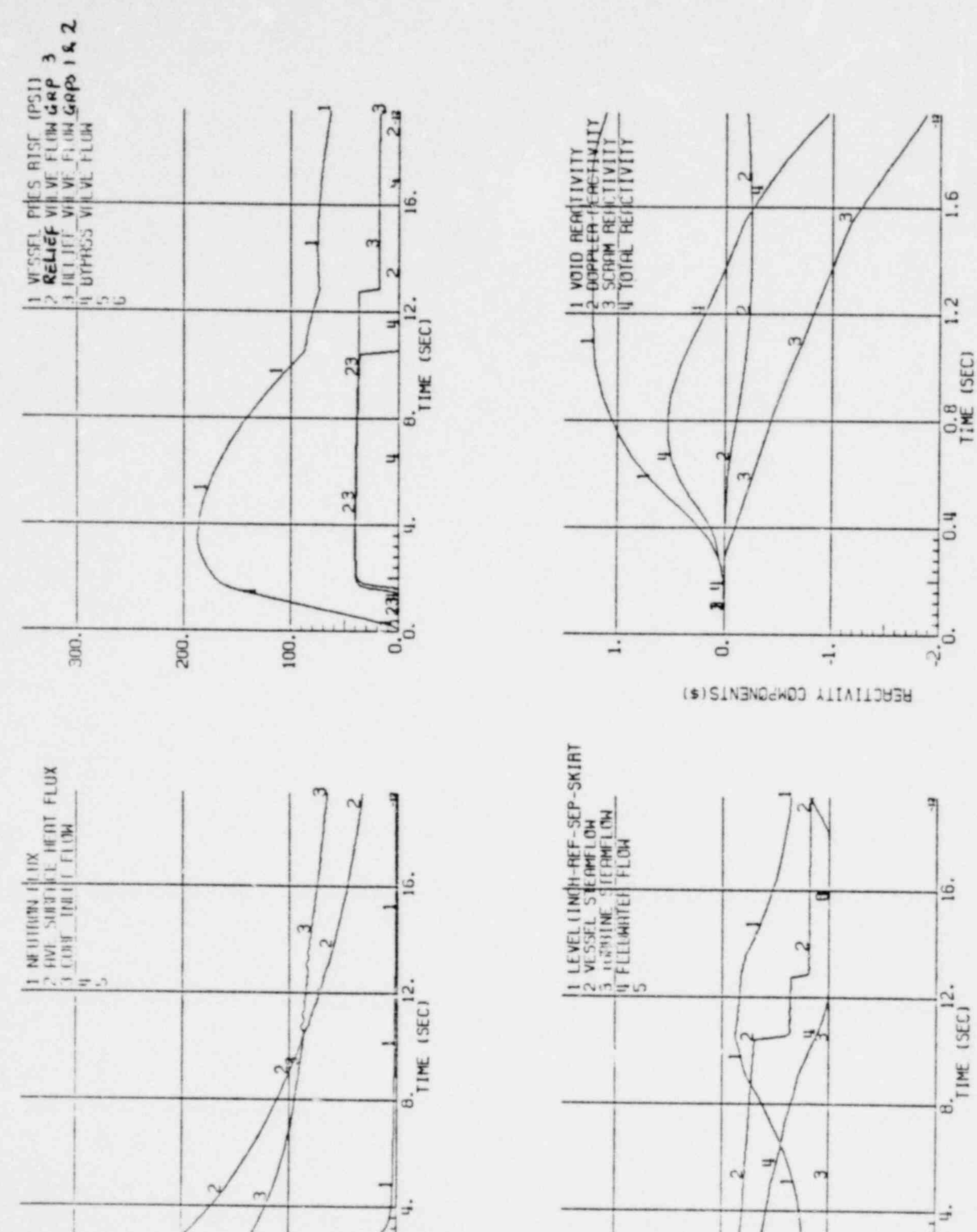

Figure 3. Plant Response to Generator Load Rejection Without Bypass

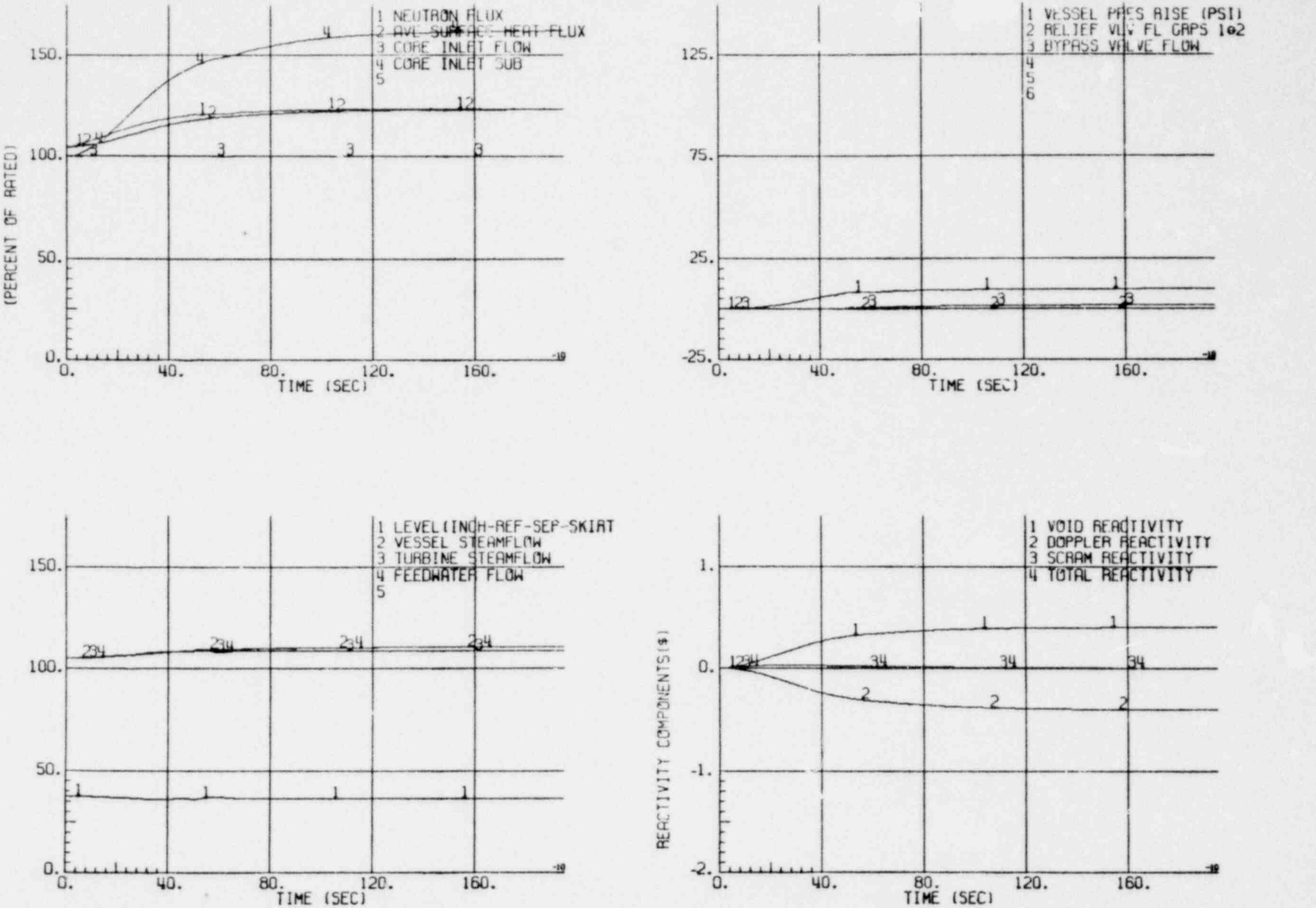

Figure 4. Plant Response to Loss of 100°F Feedwater Heating

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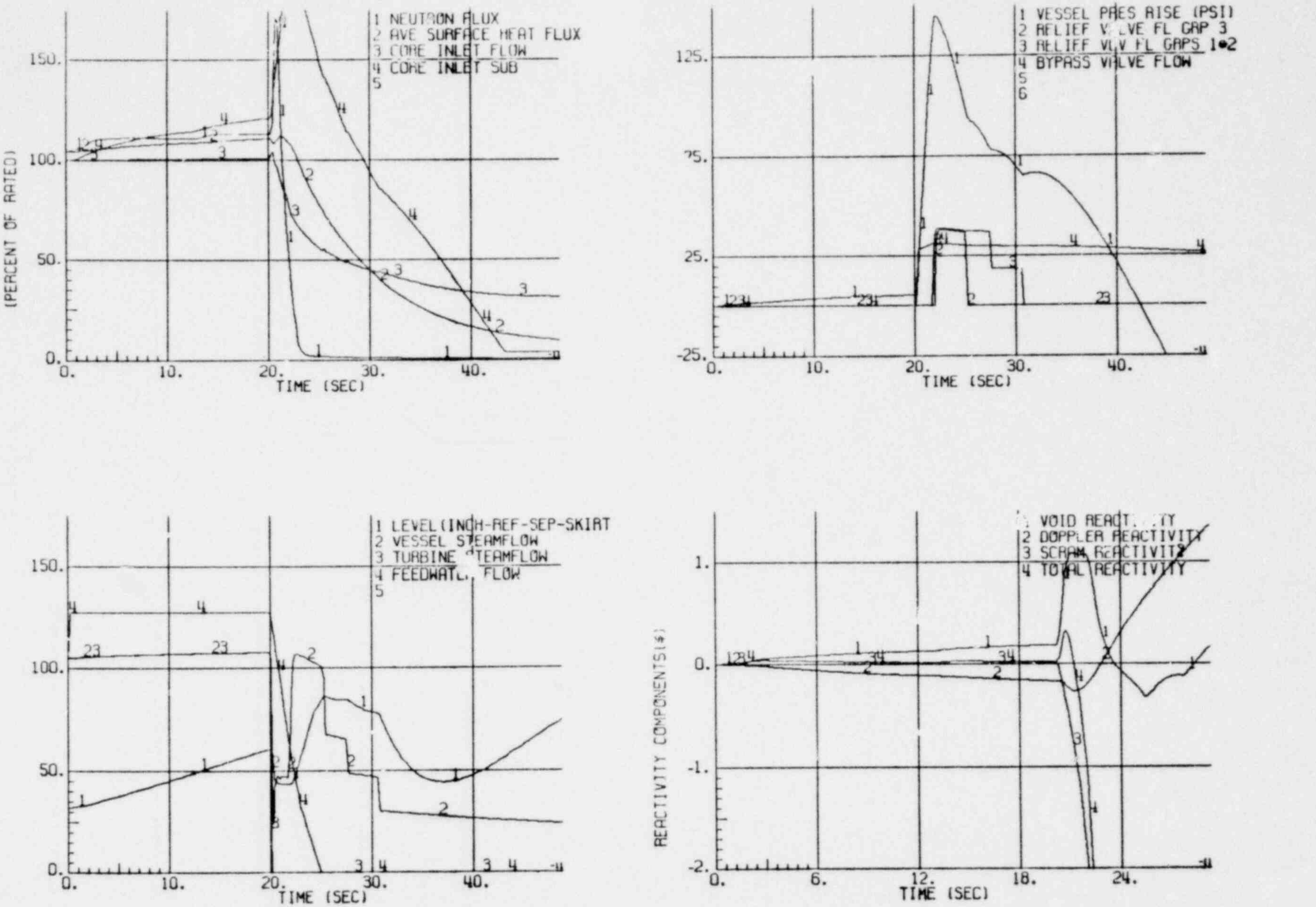

Figure 5. Plant Response to Feedwater Controller Failure

	02	06	10	14	18	22	26	30
59						00		00
55							32	
51				00		10		00
47					34			
43		00		18		00		16
39			36		28		26	
35		00		02		00		02
31			30		38		38	

NOTES:

1. Rod Pattern Is 1/4 Core Mirror Symmetric.
2. Numbers Indicate Number of Notches Withdrawn.
3. Error Rod Is (22-43).

Figure 6. Limiting RWE Rod Pattern

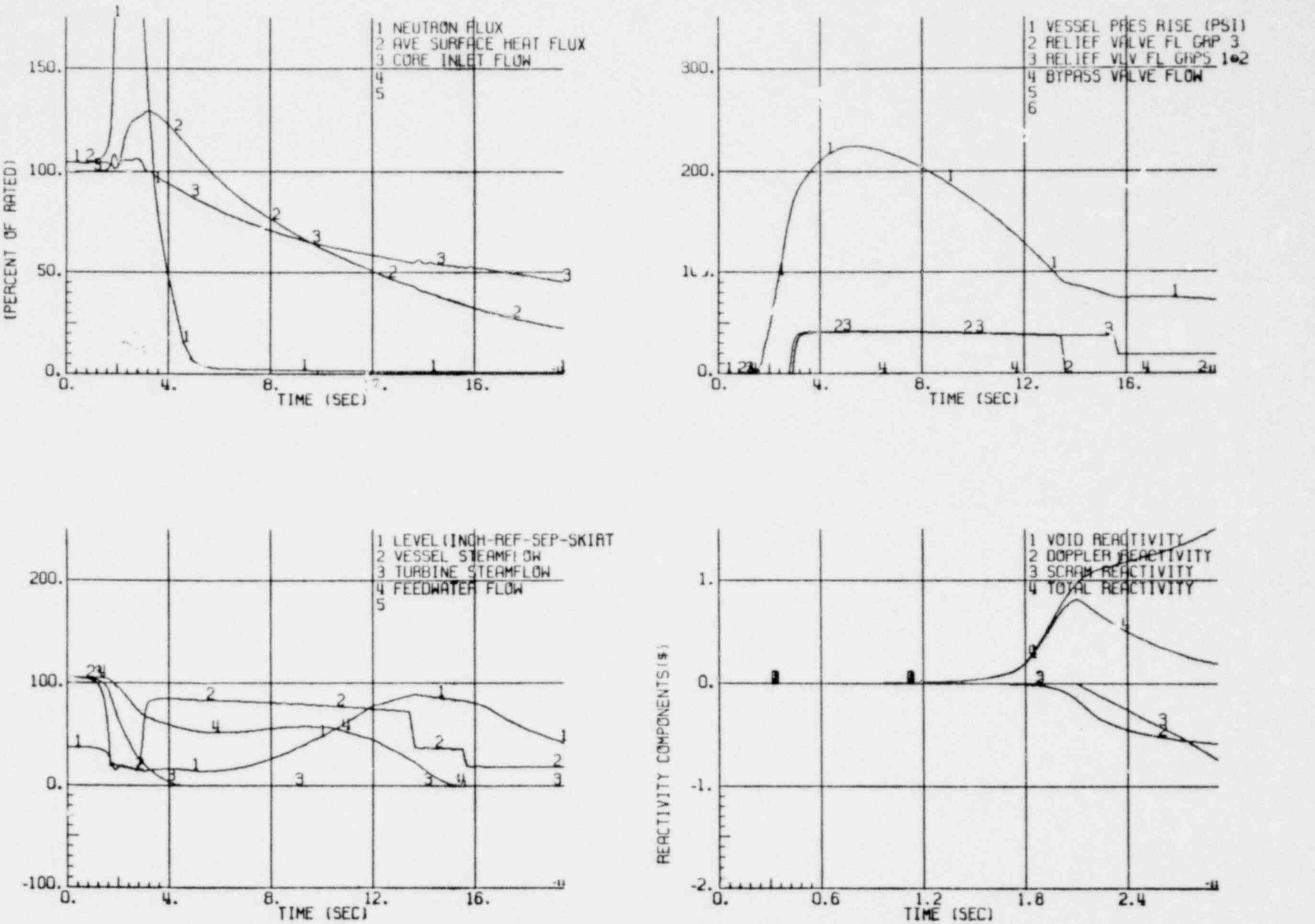

Figure 7. Plant Response to MSIV Closure

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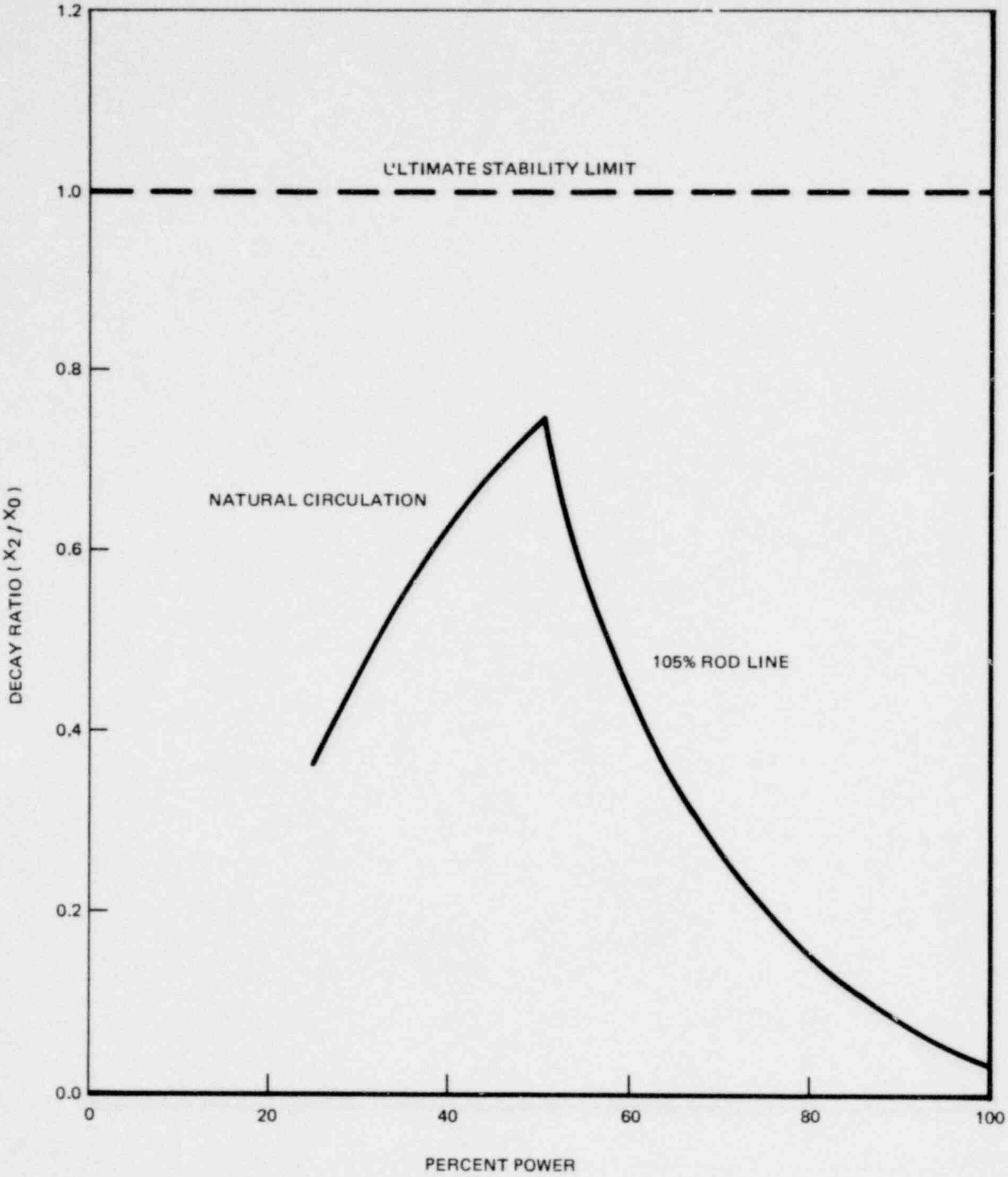

Figure 8. Decay Ratio

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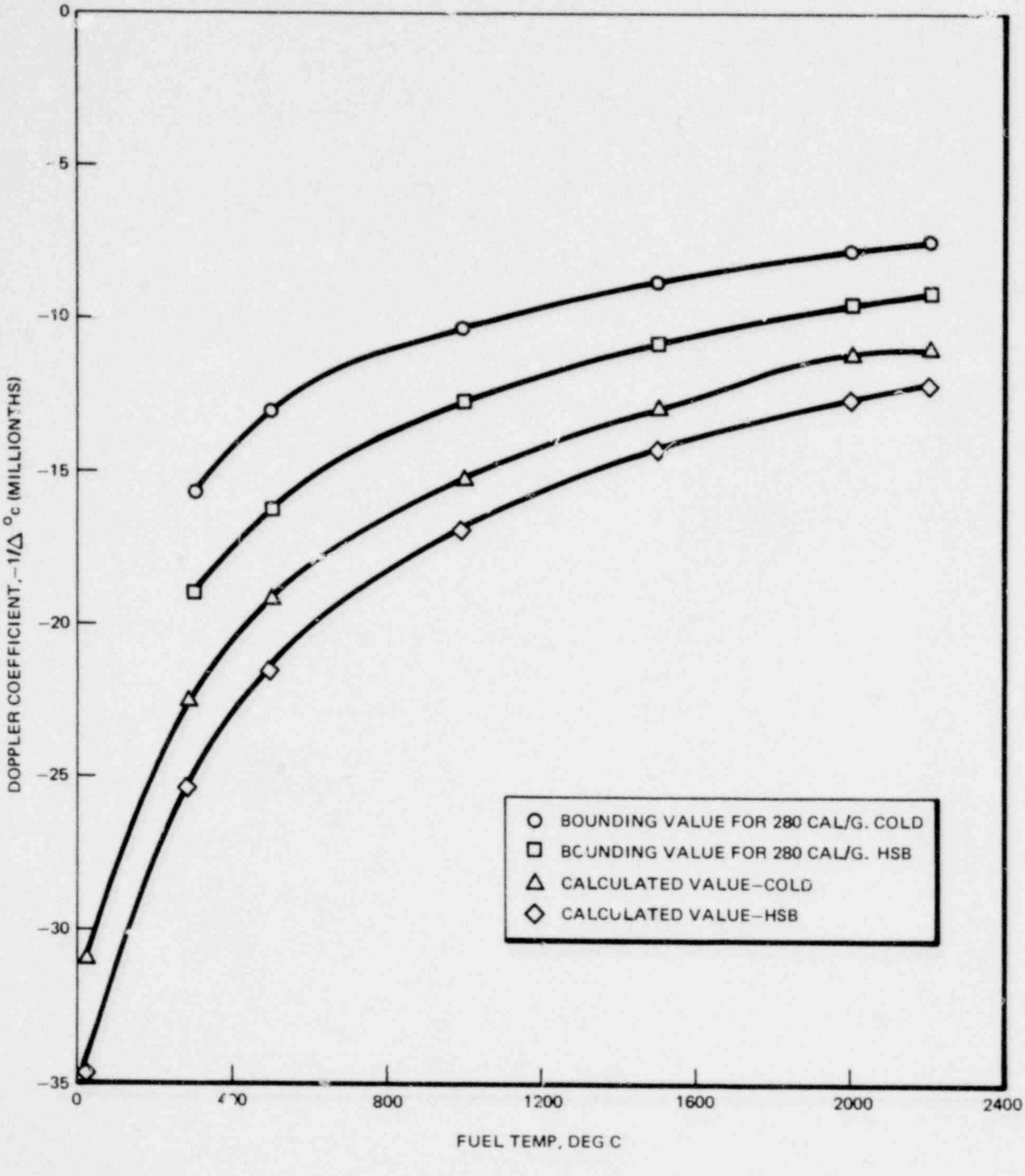

Figure 9. Doppler Reactivity Coefficient Comparison for RDA

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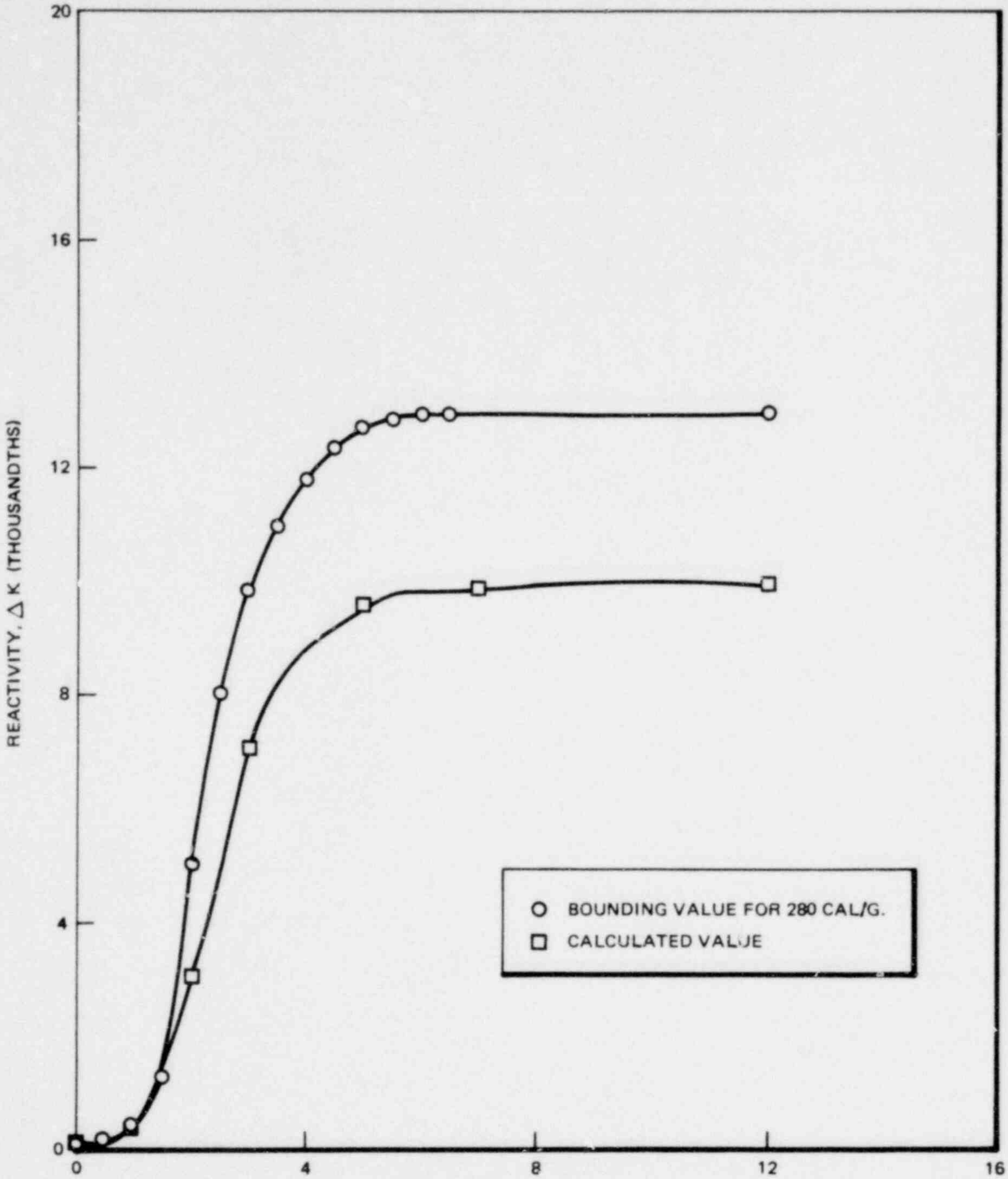

Figure 10. RDA Reactivity Shape Function at 20°C

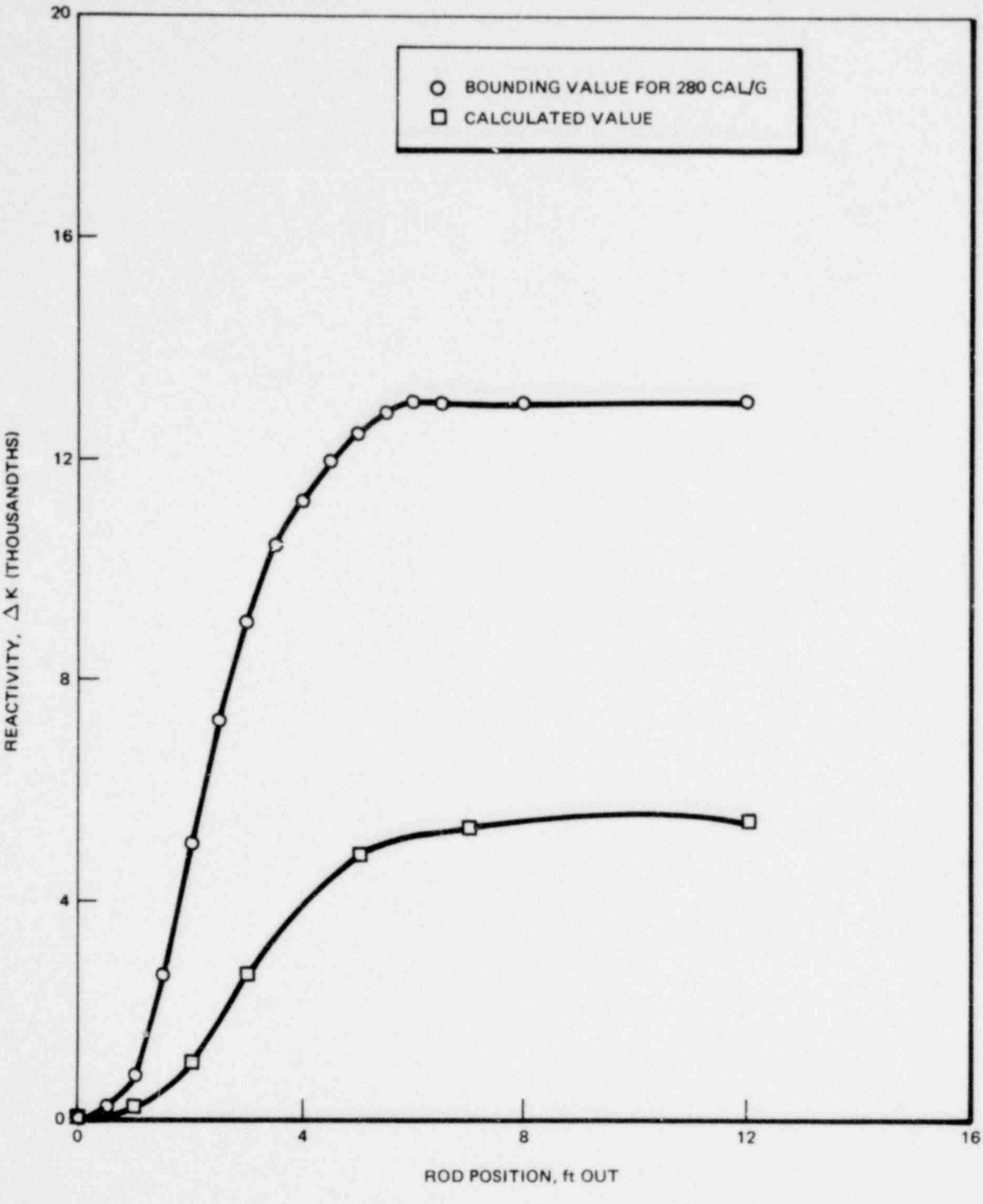

Figure 11. RDA Reactivity Shape Function at 286°C

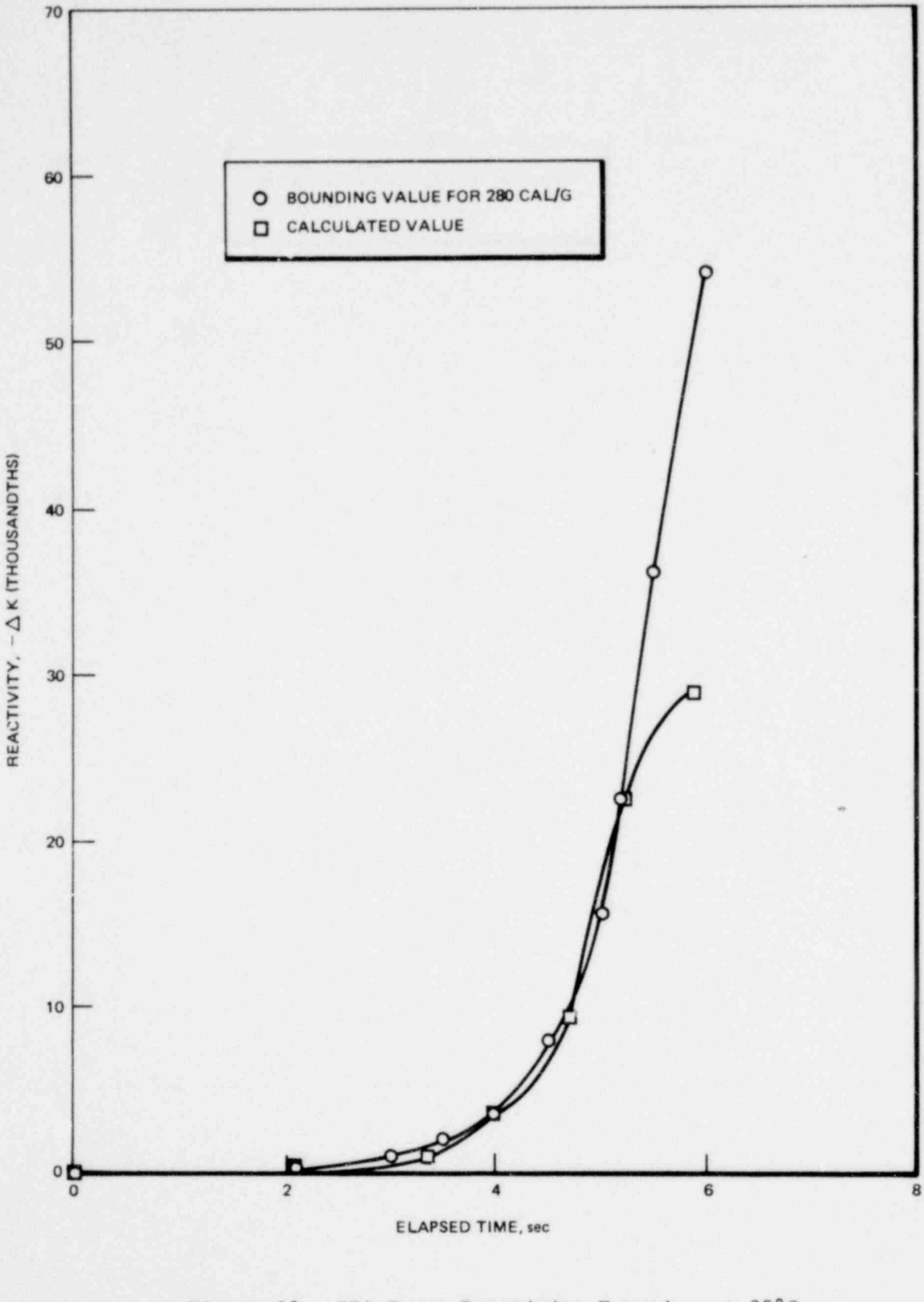

Figure 12. RDA Scram Reactivity Function at 20°C

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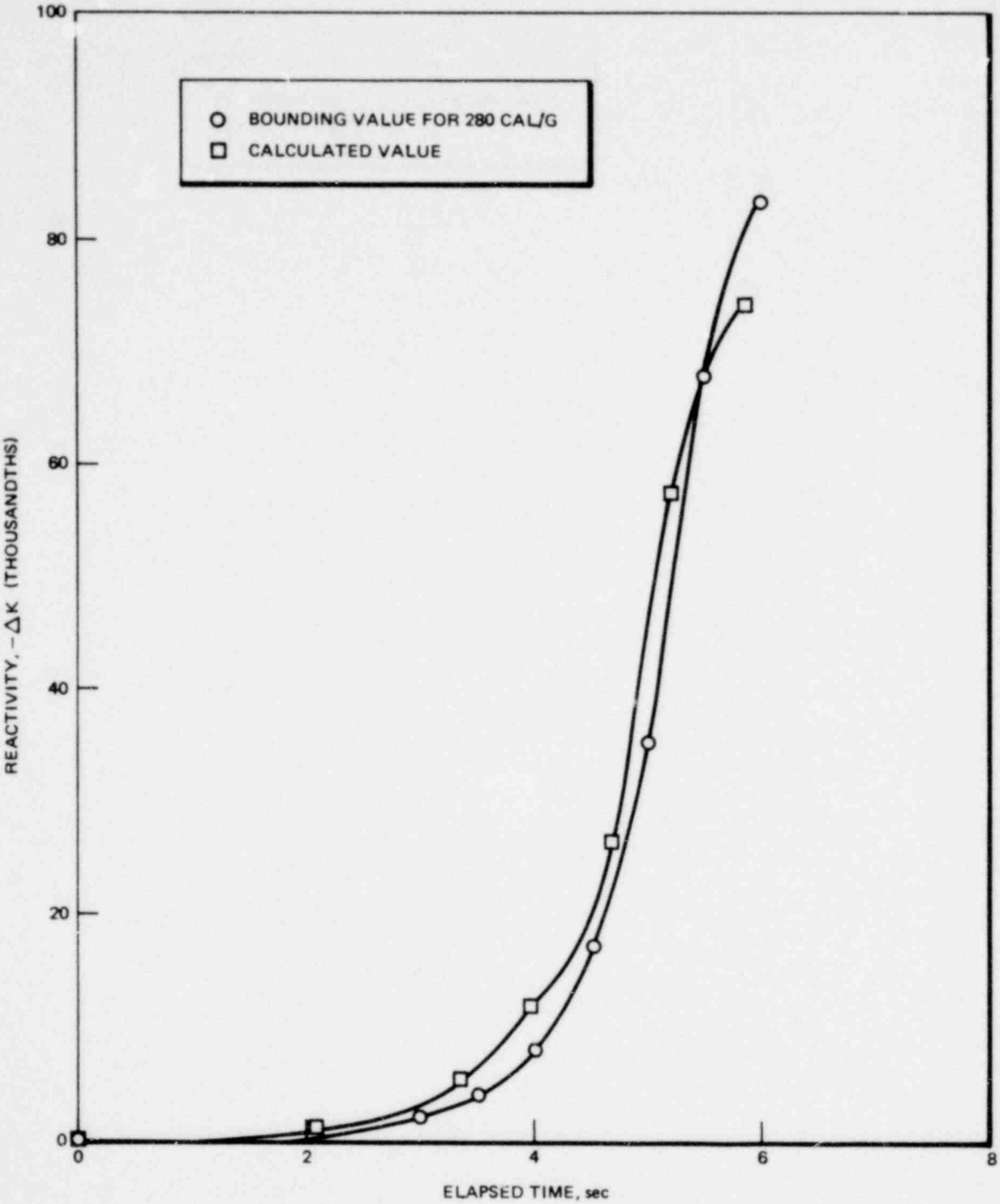

Figure 13. RDA Scram Reactivity at 286°C

APPENDIX A

Fuel Loading Error LHGR* (kW/ft): 18.0

Parameters Different From Reference 1

Number of Safety Valves: 0

Safety/Relief Valve Capacity at Setpoint** (No./%): 12/76.246

GETAB Analysis Initial Conditions

Reactor Pressure (psia): 1035

Inlet Enthalpy (Btu/lb): 521.5

*Includes 0.02 penalty for R-Factor uncertainty
**13S/R valves installed, however 1 valve assumed to be out of service

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APPENDIX B

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	15,377 MWd/t
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	15,377 MWd/t
Assumed reload cycle core average exposure at end of cycle:	16,640 MWd/t
Core loading pattern:	Figure 1

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APPENDIX C

NEW BUNDLE LOADING ERROR EVENT ANALYSES PROCEDURES

The bundle loading error analyses results presented below 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.

C.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 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.

Analysis of the most limiting rotated bundle starting from an initial CPR of 1.19 (which includes the 2% allowance for uncertainties as required by the NRC) results in a minimum CPR greater than 1.07.

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C.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 B-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 gives the following results:

Limiting Events:	MCPR
Rotated 8DB274	$\geq$1.07
Mislocated 8DRB265	$\geq$1.07

REFERENCES

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

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APPLICABLE TO:
PUBLICATION NO. NEDO-24209
T. I. E. NO. 79NED294
TITLE SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BFNP UNIT 1 RELOAD NO. 3
ISSUE DATE September 19

ERRATA And ADDENDA SHEET

NO. 1
DATE September 1979

NOTE: Correct all copies of the applicable publication as specified below.

ITEM	REFERENCES (SECTION, PAGE PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
01	Page 3	Replace with new page 3.

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9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Expo[illegible]e	Power (%)	Core Flow (%)	[illegible]	Q/A (% NBR)	P_{SL} (PSIG)	P_V (PSIG)	ΔCPR 7x7	ΔCPR 8x8	ΔCPR 8x8R/P8x8R	Plant Response
Generator Load Rejection without B[illegible]ass	EOC4	10[illegible]5	100	[illegible]2	111.1	1199	1227	0.12	0.17	0.18	Figure 3
Loss of [illegible]0°F Feedwater Heating	–	10[illegible]	100	123.7	12[illegible].3	1013	1069	0.13	0.15	0.15	Figure 4
Feedwater Controller Failure	EOC4	104.5	100	170.6	112.2	1157	1190	0.08	0.12	0.12	Figure 5

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

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106*	3.5	0.15	0.15	0.13	17.56	14.05	14.95	Figure 6
107	4.0	0.24	0.18	0.15	18.57	14.98	16.27	Figure 6
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109	4.5	0.28	0.20	0.17	18.75	15.25	16.76	Figure 6
110	5.0	0.31	0.24	0.18	18.84	15.47	17.03	Figure 6

]

11. OPERATING MCPR LIMIT (5.2)

1.23 (7x7 fuel)

1.24 (8x8 fuel)

1.25 (8x8R/P8x8R)

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	1236	1272	Figure 7

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*Indicates setpoint selected.

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