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CLASS I
JUNE 1979

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BROWNS FERRY NUCLEAR POWER STATION UNIT 3 RELOAD NO. 2

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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
BROWNS FERRY NUCLEAR POWER STATION
UNIT 3 RELOAD NO. 2

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Operating Licenses II

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

Items different from or not included in Reference 1:

Fuel Loading Error LHGR: Appendix A

Safety/Relief Valve Capacity: Appendix A

Spring Safety Valve Capacity: Appendix A

New Bundle Loading Error Event Analysis Procedures: Reference 3

LHGR includes 0.02 penalty for R-factor uncertainty

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

	Fuel Type	Number	Number Drilled
Irradiated	8DB219-Initial Core Central	320	320
Irradiated	8DB219-Initial Core Peripheral	92	92
Irradiated	8DRB265L	208	208
New	P8DRB265L	144	144
Total		764	764

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	12,686 MWd/t
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	12,486 MWd/t
Assumed reload cycle core average exposure at end of cycle:	14,040 MWd/t
Core loading pattern:	Figure 1

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

BOC k_{eff}

Uncontrolled	1.108
Fully Controlled	0.949
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 "Generic Reload Fuel Application," NEDE-24011-P-A, Revision 0, August 1978.

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5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

600 ppm	Shutdown Margin (Δk) 0.040 (20°C, Xenon Free)

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

Void Coefficient N/A* (¢/% Rg)	-6.744/-8.43
Void Fraction (%)	40.29
Doppler Coefficient N/A (¢/°F)	-0.228/-0.217
Average Fuel Temperature (°F)	1337
Scram Worth N/A ($)	-37.713/-30.17
Scram Reactivity vs Time	Figure 2

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

Exposure	8x8	8x8R	P8x8R
Peaking factors (local, radial and axial)	1.22 1.4472 1.40	1.20 1.5858 1.40	1.20 1.5809 1.40
R-Factor	1.098	1.051	1.051
Bundle Power (MWt)	6.104	6.681	6.669
Bundle Flow (10^3 lb/hr)	108.2	108.6	109.0
Initial MCPR	1.22	1.22	1.23

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

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Recirculation Pump Trip

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.2)

Transient	Exposure	Power (%)	Core Flow (%)	ϕ (% NBR)	Q/A (% NBR)	P_{sl} (psig)	P_v (psig)	ΔCPR 8x8	ΔCPR 8x8R	ΔCPR P8x8R	Plant Response
Load Rejection without Bypass	BOC-EOC	104.5	100	229.5	109.3	1206	1232	0.15	0.15	0.16	Figure 3
Loss of 100°F Feedwater Heating	--	104.5	100	123	122.06	1012	1068	0.14	0.14	0.14	Figure 4
Feedwater Controller Failure	BOC-EOC	104.5	100	159.8	110.7	1154	1188	0.10	0.10	0.10	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	ΔCPR P8x8R	MLHGR 8x8	MLHGR 8x8R	MLHGR P8x8R	Limiting Rod Pattern
104	3.5	0.16	0.11	0.09	14.6	16.4	14.1	Figure 6
105	4.0	0.19	0.13	0.11	15.4	16.8	14.5	Figure 6
106*	4.5	0.21	0.14	0.13	15.5	17.0	14.6	Figure 6
107	5.0	0.23	0.16	0.14	15.3	17.0	14.7	Figure 6
108	5.5	0.25	0.17	0.15	15.1	16.8	14.7	Figure 6
109	6.0	0.26	0.18	0.17	14.9	16.6	14.7	Figure 6

*Indicates setpoint selected

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11. OPERATING MCPR LIMIT (5.2)

BOC3 to EOC3

1.28	8x8 Fuel
1.22	8x8R Fuel
1.23	P8X8R Fuel

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	1246	1280	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio, x_2/x_0 0.79

(105% Rod Line - Natural Circulation Power)

Channel Hydrodynamic Performance

	Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power)
8x8R/P8x8R	0.273
8x8	0.383

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

Reference 2.

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15. LOADING ERROR RESULTS (5.5.4)

Limiting Event: Rotated Bundle 8DRB265L

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

Plant specific analysis results

Parameter not bounded: Accident Reactivity Shape Function at 20°C

Resultant peak enthalpy: 278 cal/gm

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REFERENCES

1. General Electric Boiling Water Generic Reload Fuel Application, NEDE-24011-P-A, August 1978.

2. Loss-Of-Coolant Accident Analysis Report for Browns Ferry Nuclear Plant Unit 3, NEDO-24194, June 1979.

3. Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 3 Reload 1, NEDO-24128 (Appendix A), June 1978.

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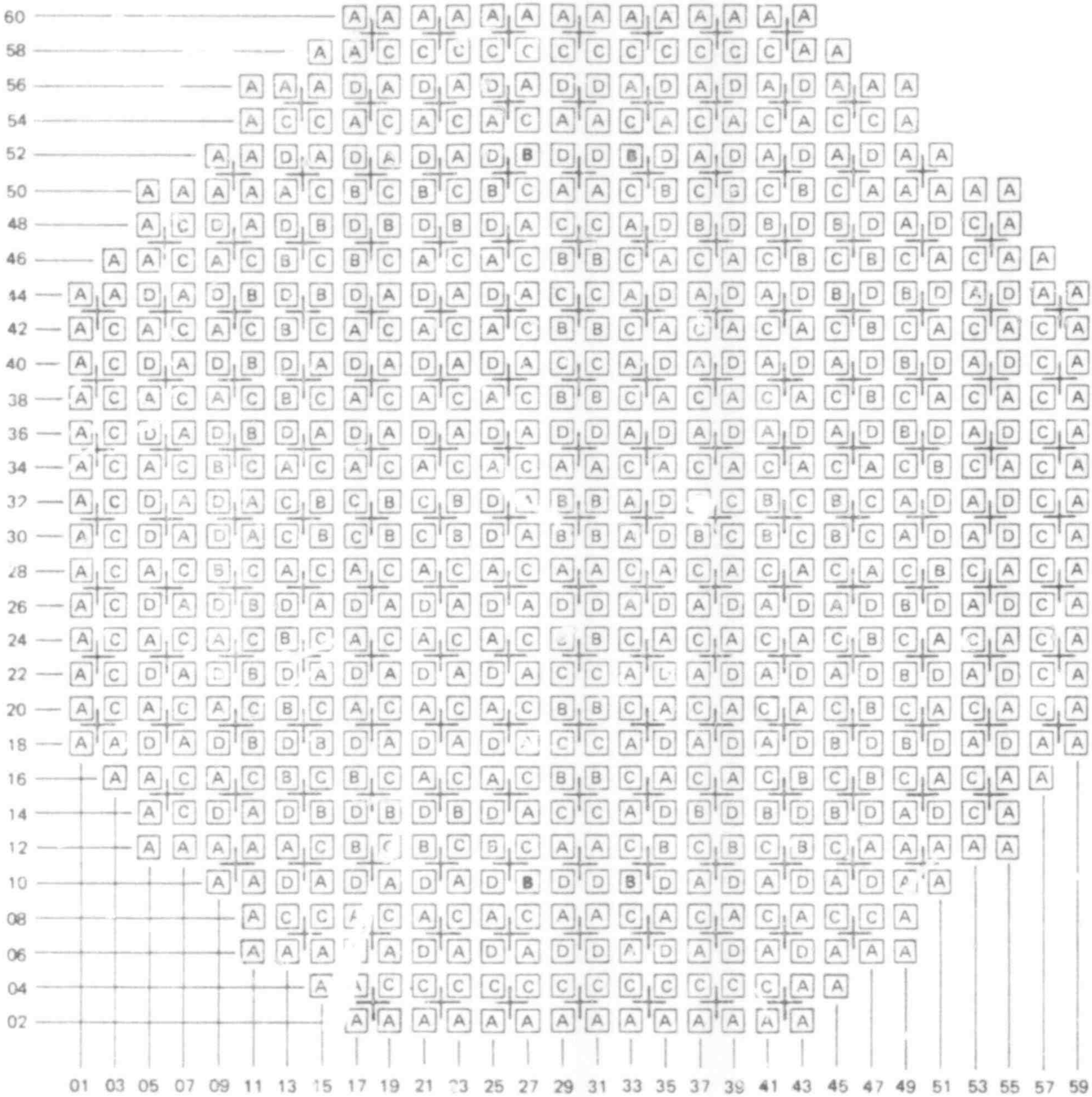

FUEL TYPE	
A = 8D219-1C – CENTRAL	E =
B = 8D219-1C – PERIPHERAL	F =
C = 8DRB265L	G =
D = P8DRB265L	H =

Figure 1. Reference Core Loading Pattern

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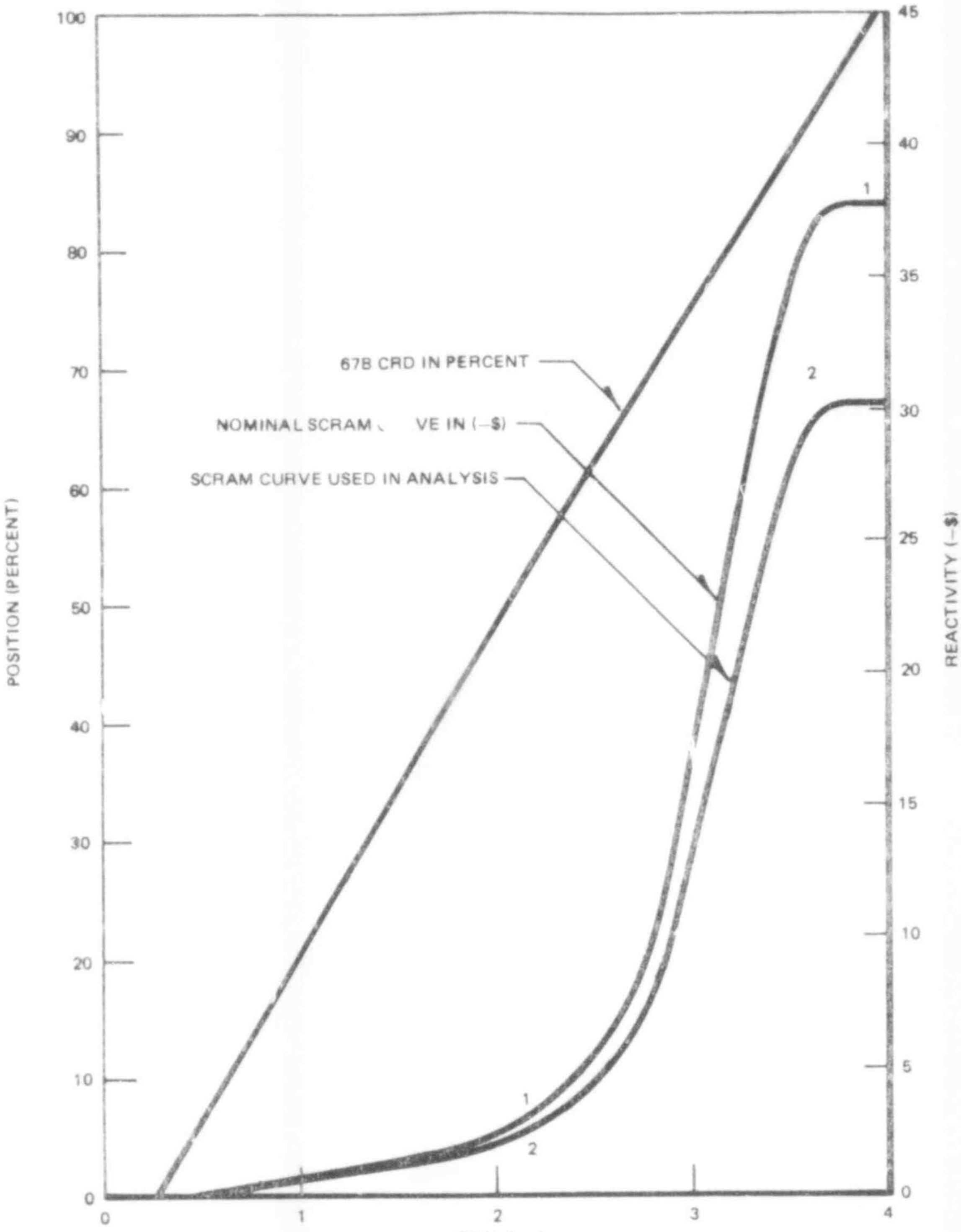

Figure 2. Scram Reactivity and Control Rod Drive Specifications

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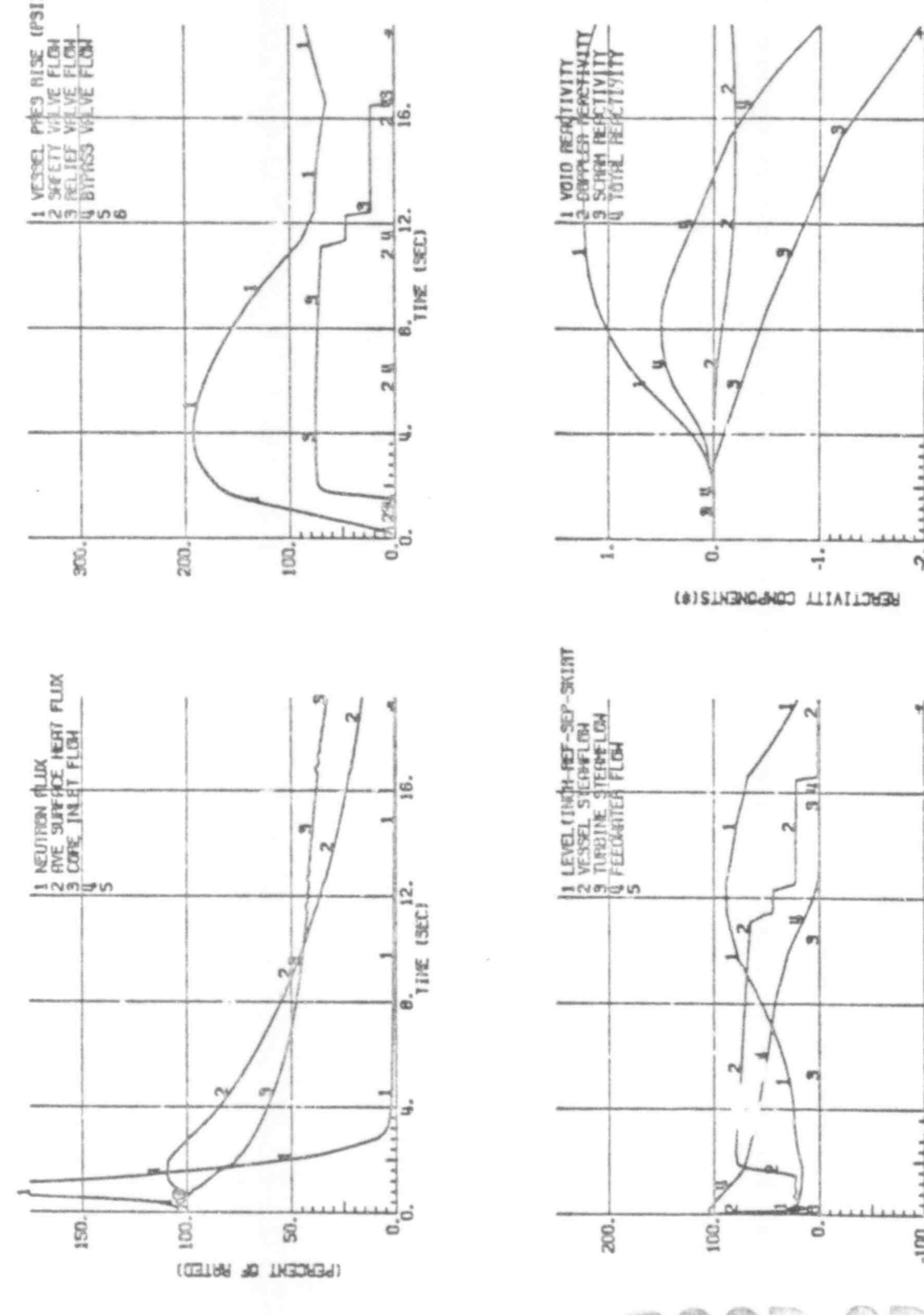

Figure 3. Plant Response to Generator Load Rejection without Bypass

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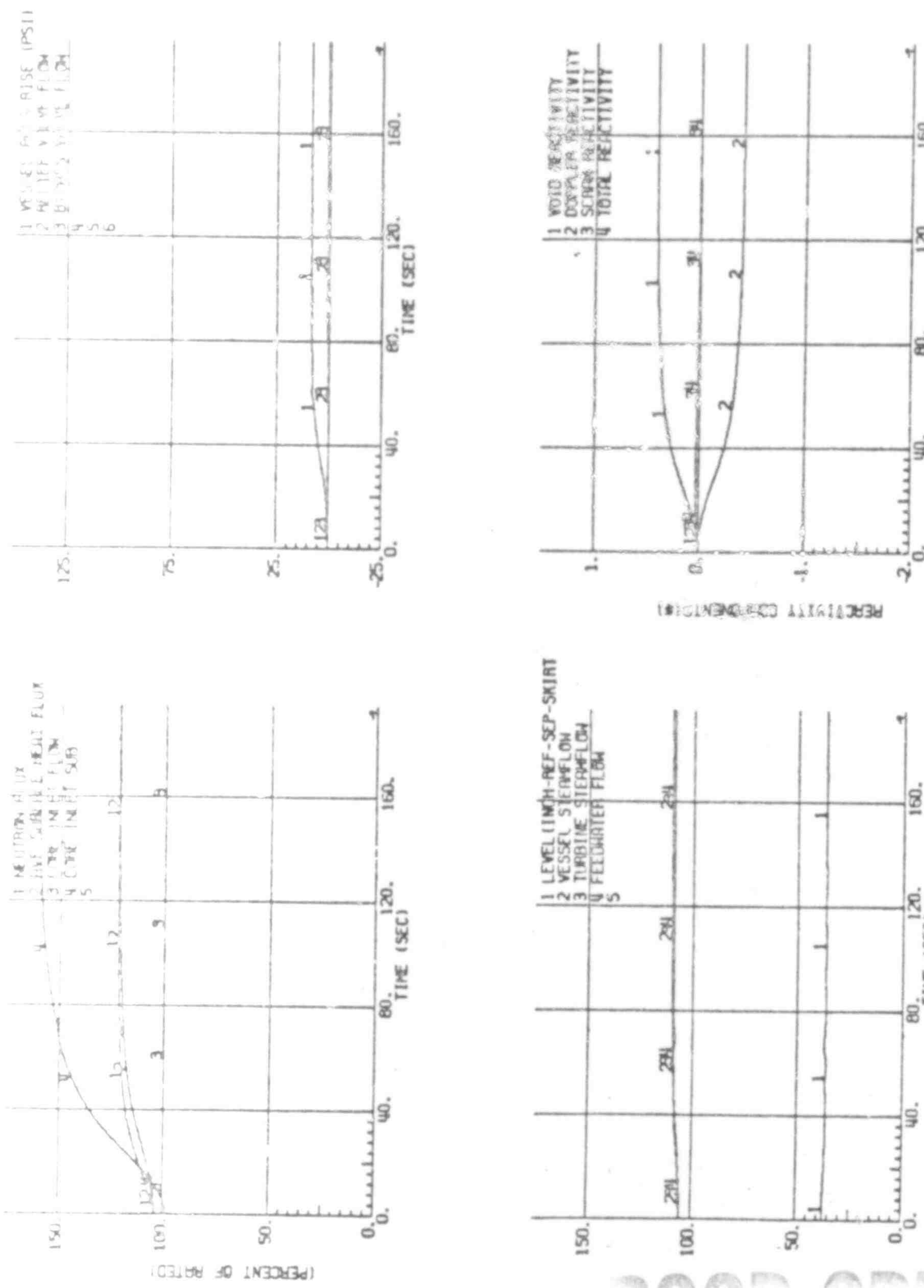

Figure 4. Plant Response to Loss of 100°F Feedwater Heating

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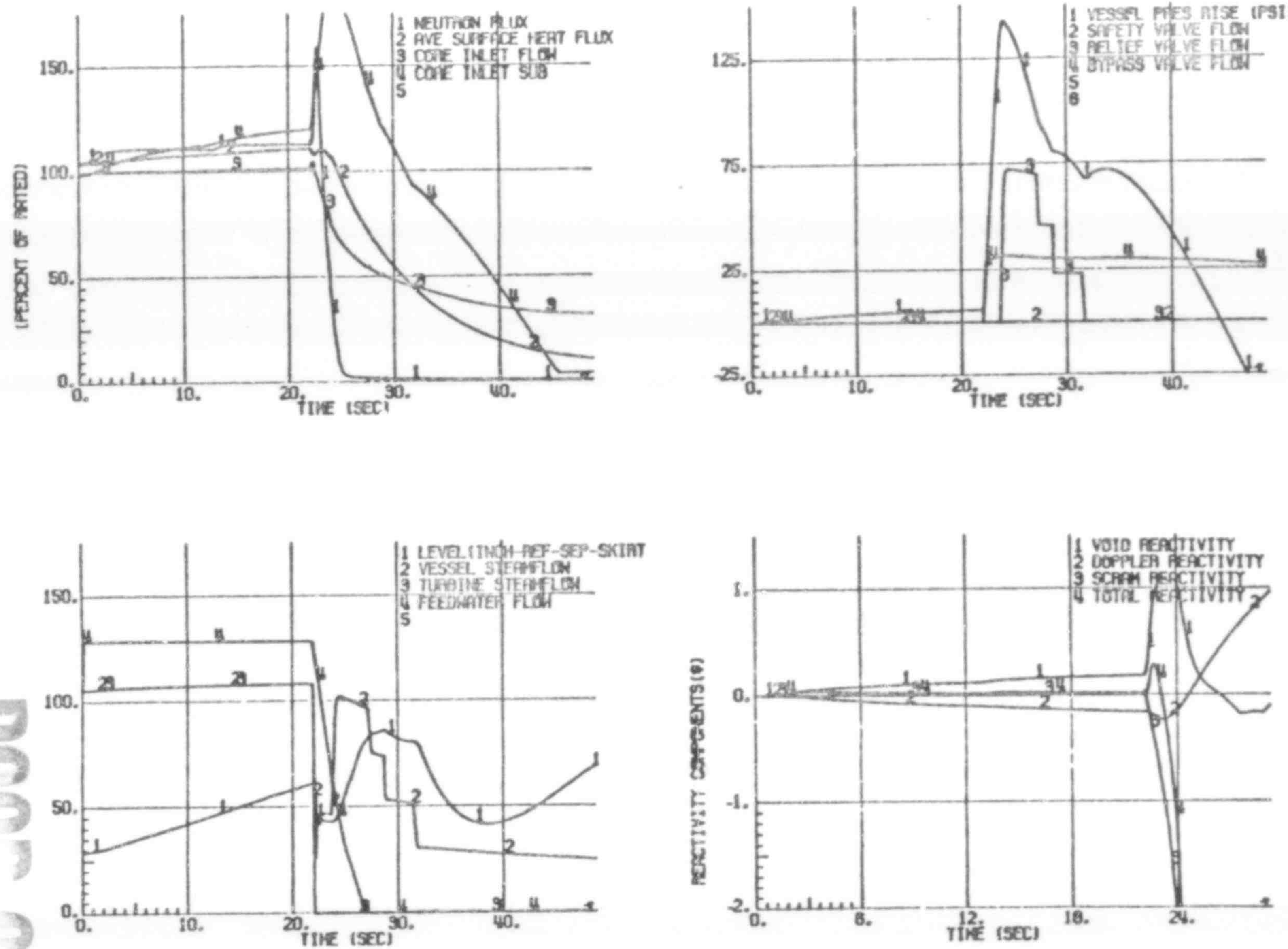

Figure 5. Plant Response to Feedwater Controller Failure, Maximum Demand

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	02	06	10	14	18	22	26	30
59								
55					42		36	
51				8		2		14
47			24		20			
43		14		4		16		0
39			30		26			
35		12		6		4		10
31			36		24		28	

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 THE 30-43 ROD.

Figure 6. Limiting RWE Rod Pattern

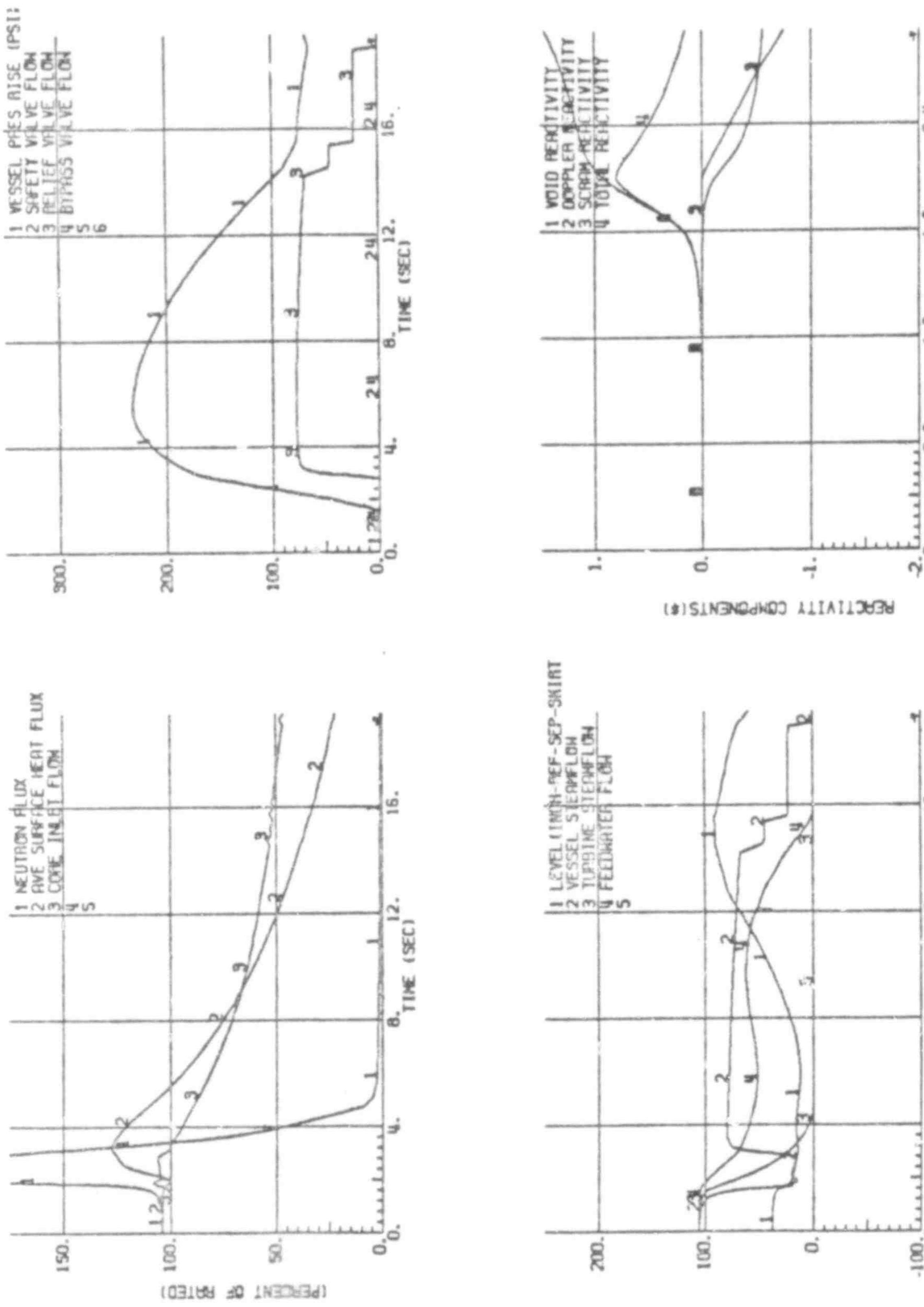

Figure 7. Plant Response to MSIV Closure

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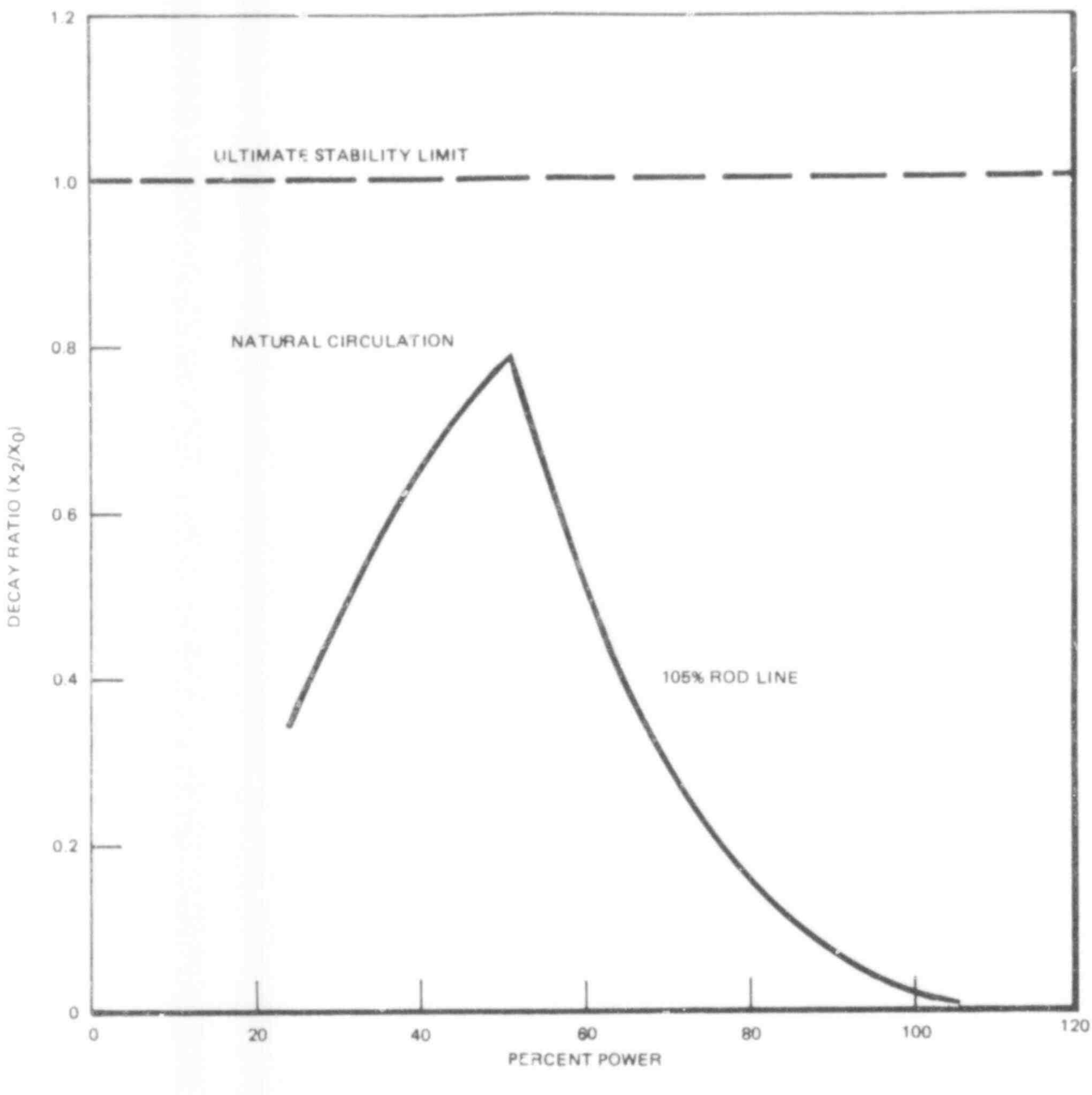

Figure 8. Decay Ratio

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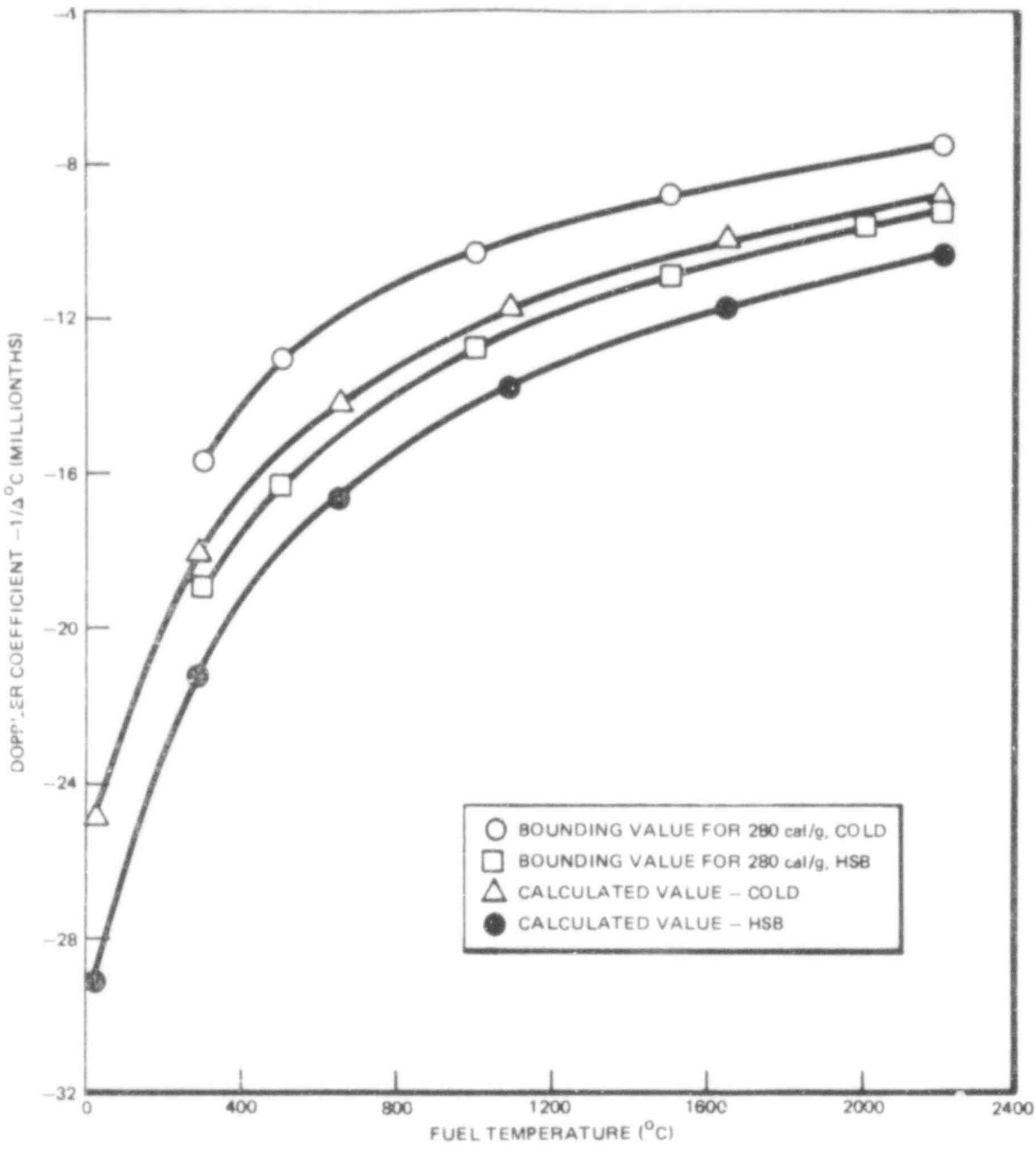

Figure 9. Doppler Reactivity Coefficient Comparison for RDA

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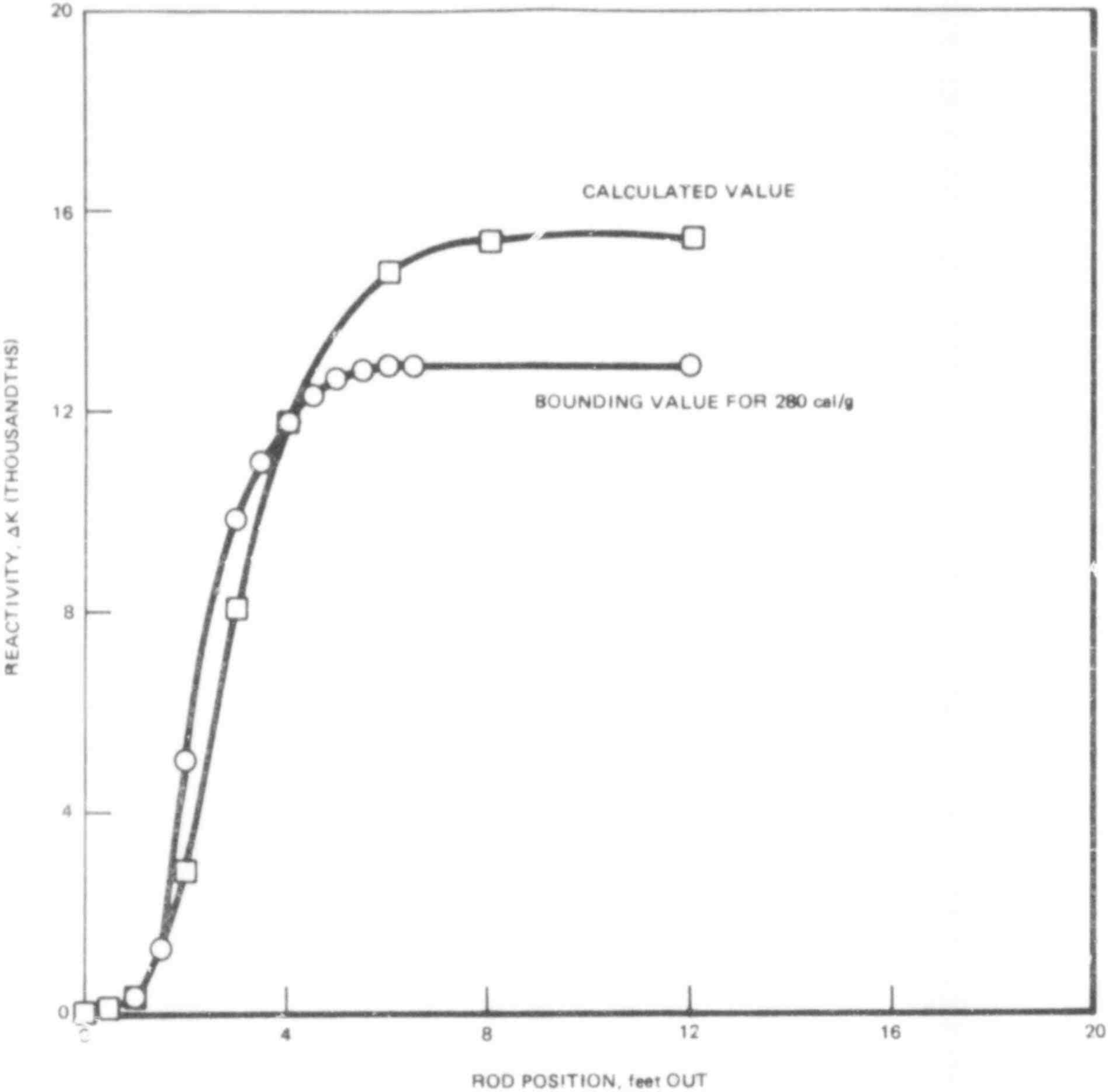

Figure 10. Accident Reactivity Shape Function at 20°C

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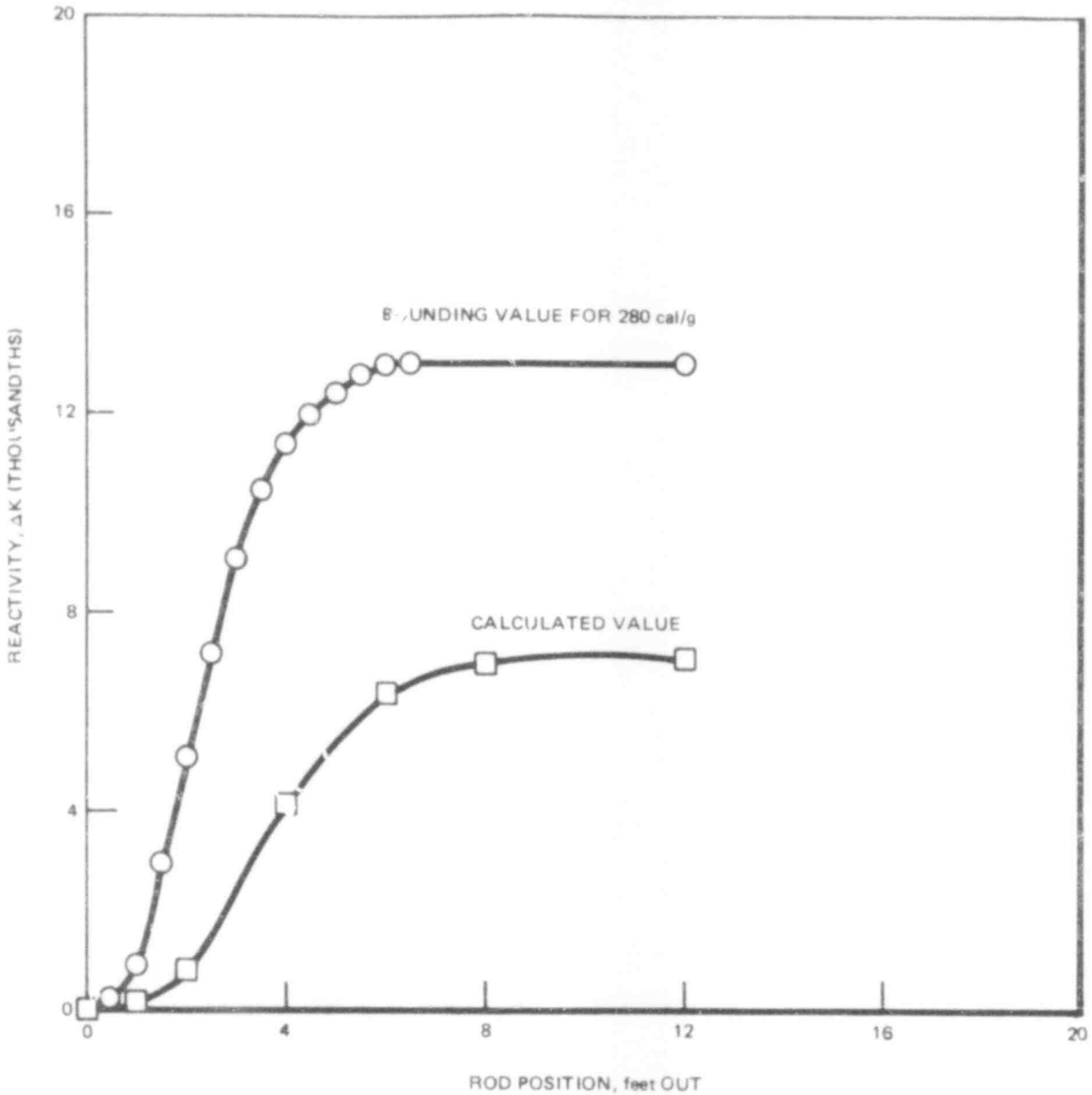

Figure 11. Accident Reactivity Shape Function at 286°C

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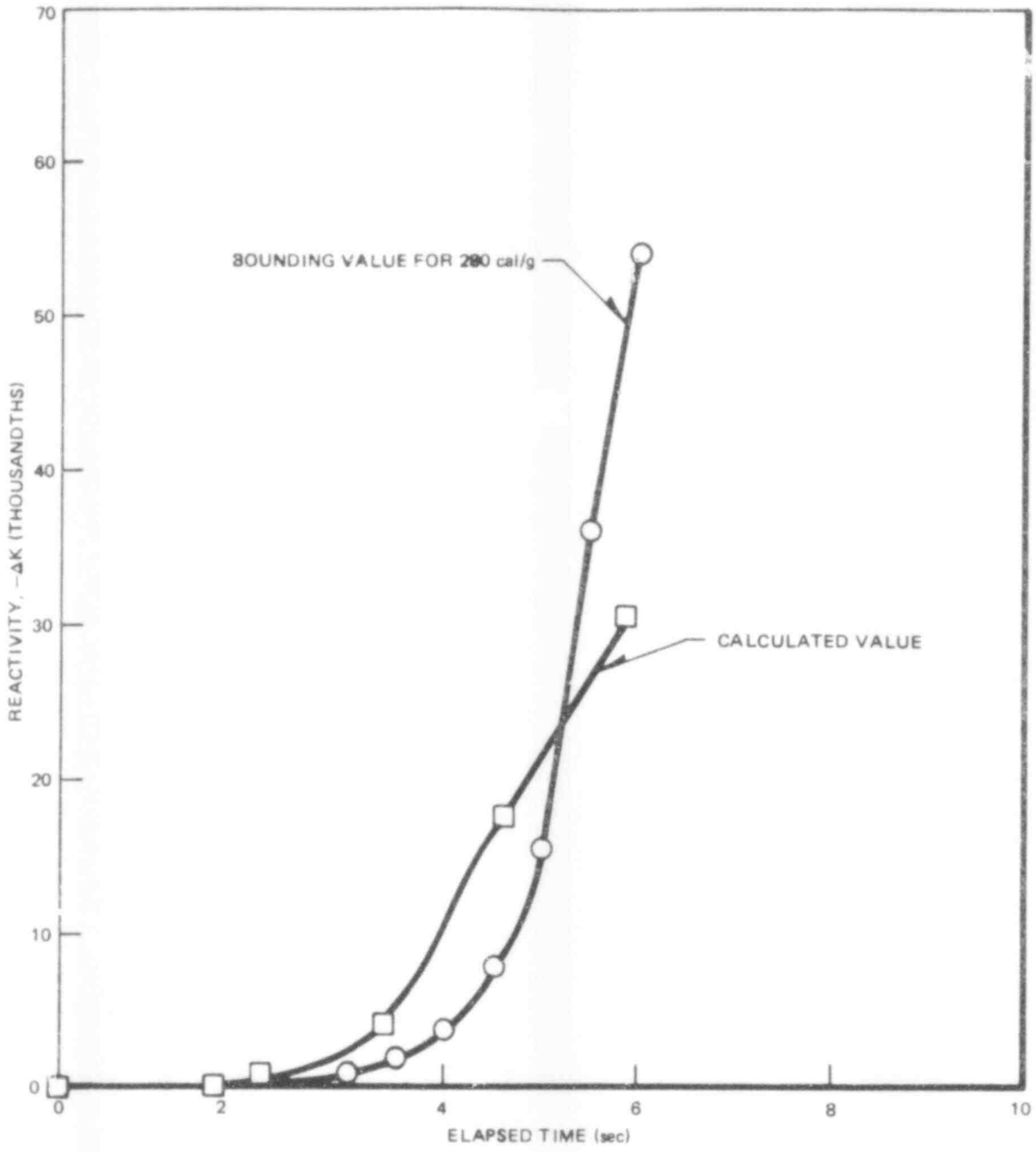

Figure 12. Scram Reactivity Function at 20°C

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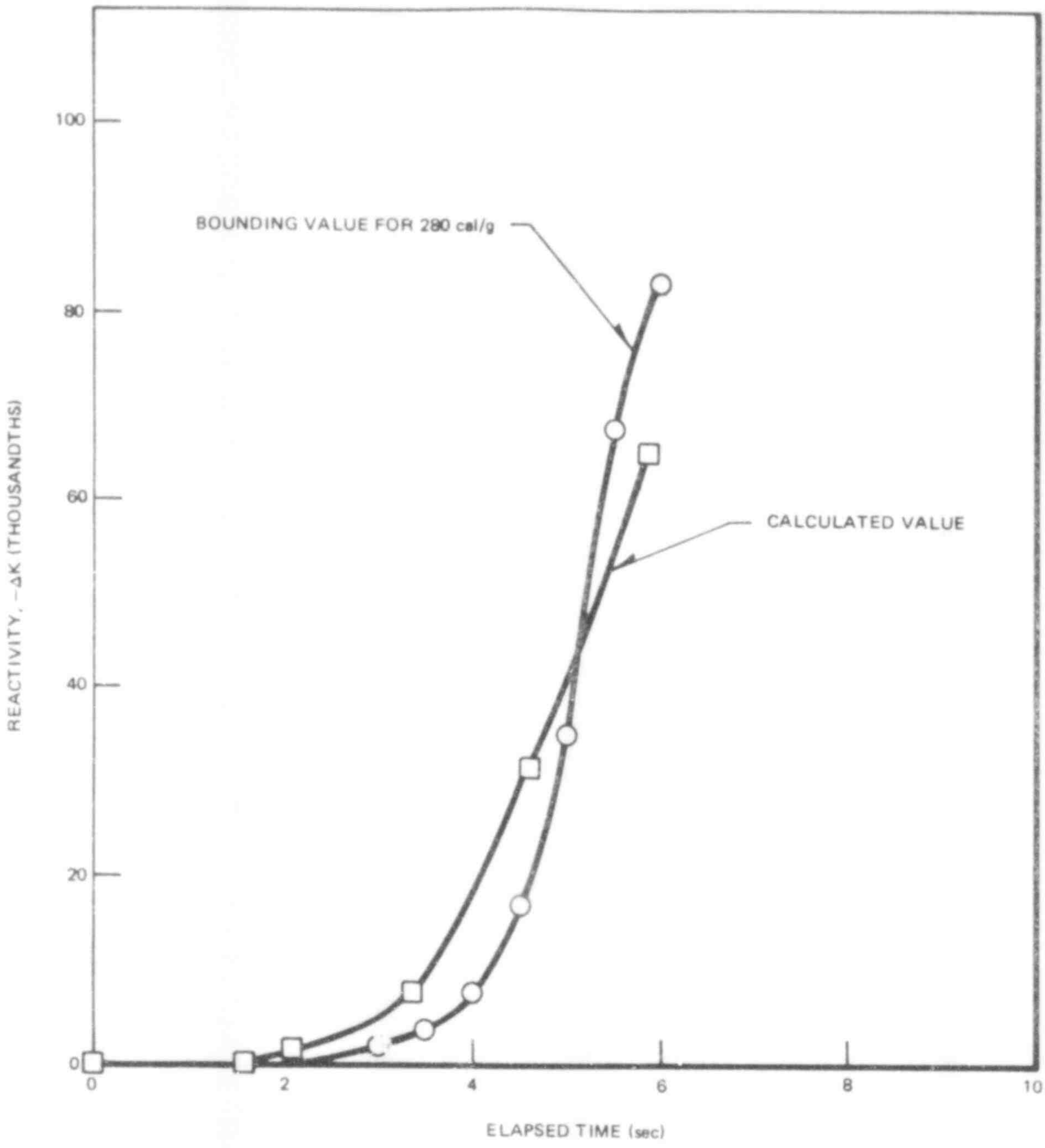

Figure 13. Scram Reactivity Function at 286°C

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APPENDIX A

Fuel Loading Error LHGR: 16.02 kW/ft

Safety/Relief Valve Capacity at Setpoint (No./%): 11/70

Spring Safety Valve Capacity at Setpoint (No./%): 2/14.2

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