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CLASS I
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NEDO-24234

DUANE ARNOLD DAEC RELOAD 4 LICENSING SUBMITTAL

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR DUANE ARNOLD ATOMIC ENERGY CENTER RELOAD 4

29 pgs.

GENERAL ELECTRIC

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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
DUANE ARNOLD ATOMIC ENERGY CENTER
RELOAD 4

P. H. Henrikson
Sr. Licensing Engineer

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

NUCLEAR POWER SYSTEMS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT PLEASE READ CAREFULLY

1. PLANT-UNIQUE ITEMS (1.0)*

Margin to unpiped spring safety valves: Appendix A

New bundle loading error analyses procedures: Appendix B

Recirculation Pump Trip: "Basis for Installation of Recirculation Pump Trip System, Duane Arnold Energy Center", NEDO-24220, September 1979.

Measured Scram Time: "Safety Evaluation . . . Supporting Amendment No. 54 to License No. DPR-49", Docket 50-331, September 4, 1979.

ODYN Input and Results: Appendix C

Transient Operating Parameters: Appendix D

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

	Fuel Type		Number	Number Drilled
Irradiated	Initial Core	7DB212	4	4
Irradiated	Interim Reload	7DB230	4	4
Irradiated	Reload 1	8DB274L	32	32
		8DB274H	52	52
Irradiated	Reload 2	8DB274L	32	32
		8DB274H	68	68
Irradiated	Reload 3	8DB274H	88	88
New	Reload 4	P8DPB289	88	88
Total			368	368

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 14,169 MWd/t

Assumed reload cycle exposure: 15,330 MWd/t

Core loading pattern: Figure 1

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

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	1.116
Fully Controlled	0.958
Strongest Control Rod Out	0.986
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

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

	EOC 5
Void Coefficient N/A* (¢/% Rg)	-9.51/-11.89
Void Fraction (%)	42.3
Doppler Coefficient N/A (¢/°F)	-0.217/-0.206
Average Fuel Temperature (°F)	1359
Scram Worth N/A ($)	-37.23/-29.78
Scram Reactivity versus Time	Figure 2

*N = Nuclear Input Data

A = Used in Transient Analysis

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

Exposure	EOC 5 7x7	8x8	P8x8R
Peaking factor (local, radial, axial)	1.24, 1.29, 1.40	1.22, 1.43, 1.40	1.20, 1.58, 1.40
R-Factor	1.100	1.098	1.051
Bundle Power (MWt)	5.482	6.082	6.689
Bundle Flow (10^3 lb/hr)	124.4	111.9	113.5
Initial MCPR	1.20	1.21	1.22

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2.2)

Recirculation Pump Trip: Appendix C

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1

Transient	Exposure	Power (%)	Core Flow (%)	ϕ (% NBR)	Q/A (% NBR)	P_{SL} (psig)	P_v (psig)	ΔCPR 7x7/8x8/P8x8R	Plant Response
Turbine Trip No Bypass	EOC 5	104	100	142	104	1206	1214	0.01/0.02/0.03	Figure 3
Loss of 100°F Feedwater Heater	--	104	100	126	123	1022	1071	0.13/0.14/0.15	Figure 4
Feedwater Controller Failure	EOC 5	104	100	122	108	1148	1177	0.03/0.04/0.04	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 and P8x8R	LHGR 8x8 and P8x8R	Limiting Rod Pattern
105*	4.0	0.15/0.11	12.9/13.7	Figure 6
106	4.0	0.15/0.11	12.9/13.7	Figure 6
107	4.5	0.17/0.13	13.4/13.7	Figure 6
108	5.0	0.20/0.14	13.8/13.8	Figure 6
109	5.5	0.24/0.15	14.3/13.9	Figure 6
110	6.0	0.26/0.16	14.9/14.2	Figure 6

11. OPERATING MCPR LIMIT (5.2, Appendix C)

7x7	8x8	P8x8R
1.23	1.22	1.26

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	Power (%)	Core Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	104	100	1264	1291	Figure 7

*Indicates setpoint selected

†7x7 fuel is not significantly affected by the withdrawal of the error rod.

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio, x_2/x_o (Natural Circulation-105% Rod Line)	0.85

Channel Hydrodynamic Performance

	Decay Ratio (Natural Circulation-105% Rod Line)
8x8 Channel	0.28
8x8R Channel	0.21
7x7 Channel	0.14

14. LOSS-OF-COOLANT ACCIDENT RESULTS FOR NEW P8DPB289 (5.5.2)

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Local Oxidation Fraction
200	11.2	2128	0.027
1000	11.2	2127	0.027
5000	11.8	2173	0.030
10,000	12.0	2175	0.029
15,000	12.1	2186	0.030
20,000	11.8	2165	0.029
25,000	11.3	2091	0.022
30,000	11.1	2065	0.034

15. LOADING ERROR RESULTS (5.5.4)

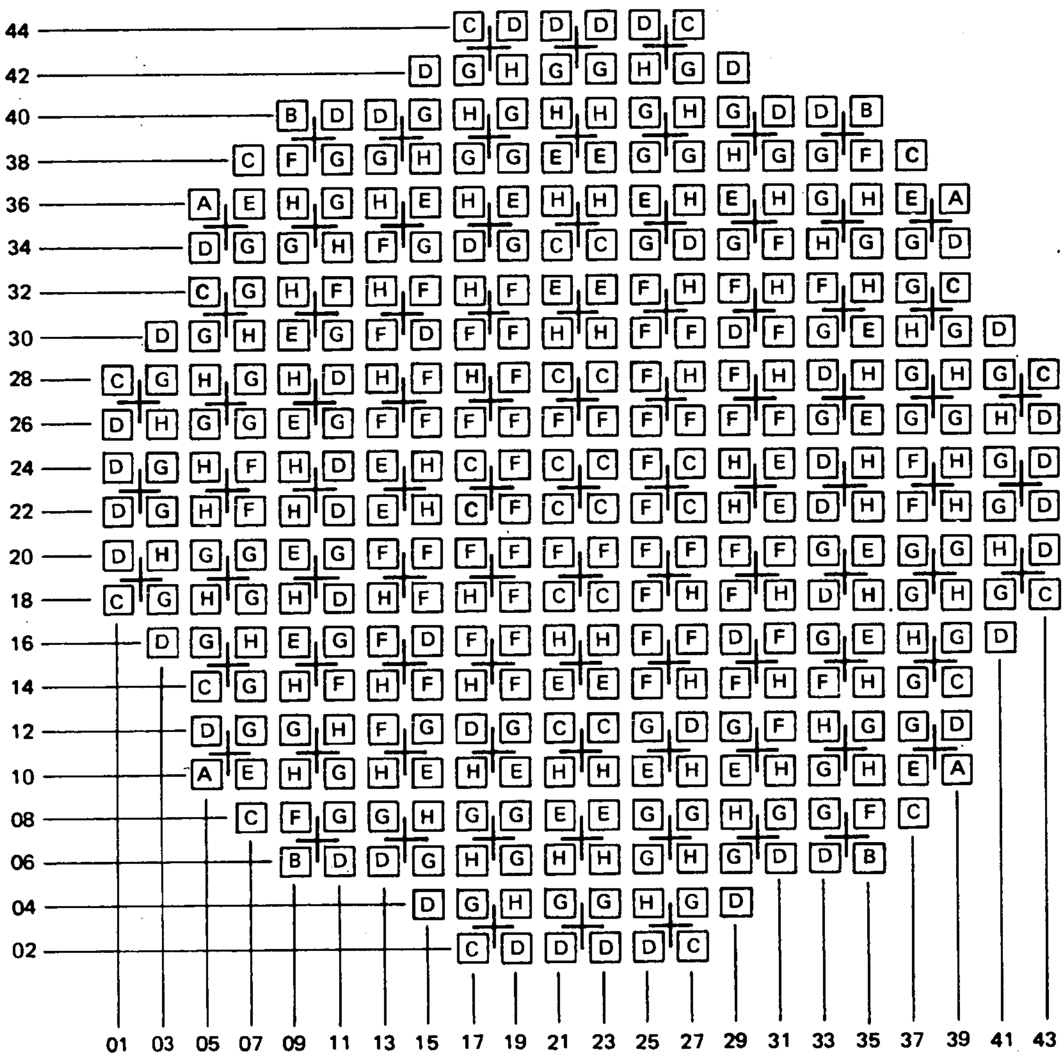
See Appendix B.

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Doppler Reactivity Coefficients: Figure 9

Accident Reactivity Shape Functions: Figures 10 and 11

Scram Reactivity Functions: Figures 12 and 13



FUEL TYPE	
A = INITIAL CORE, 7DB212	E = RELOAD 2, 8DB274L
B = INTERIM CORE, 7DB230	F = RELOAD 2, 8DB274H
C = RELOAD 1, 8DB274L	G = RELOAD 3, 8DB274H
D = RELOAD 1, 8DB274H	H = RELOAD 4, P8DPB289

Figure 1. Reference Core Loading Pattern

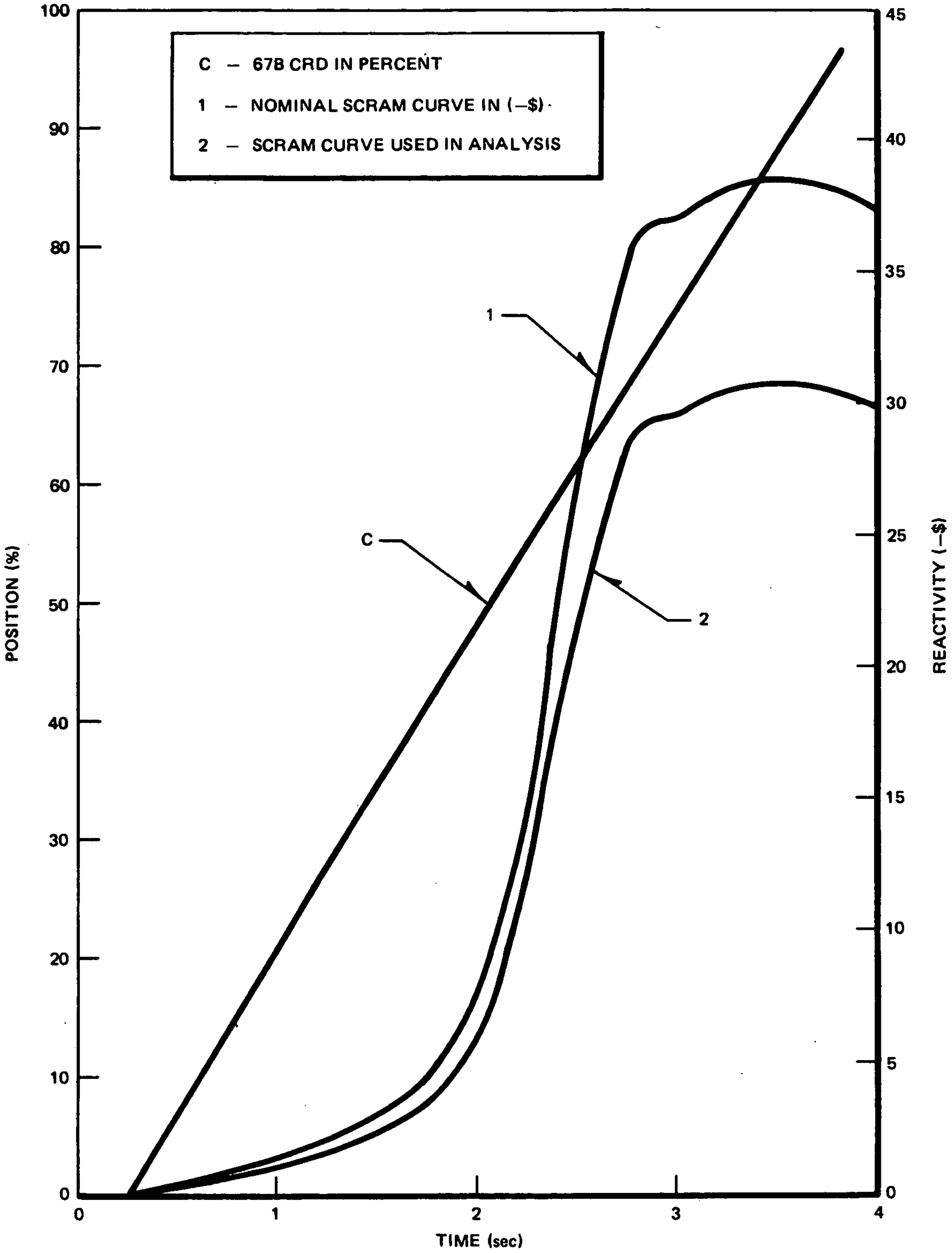


Figure 2. Scram Reactivity and Control Rod Drive Specification

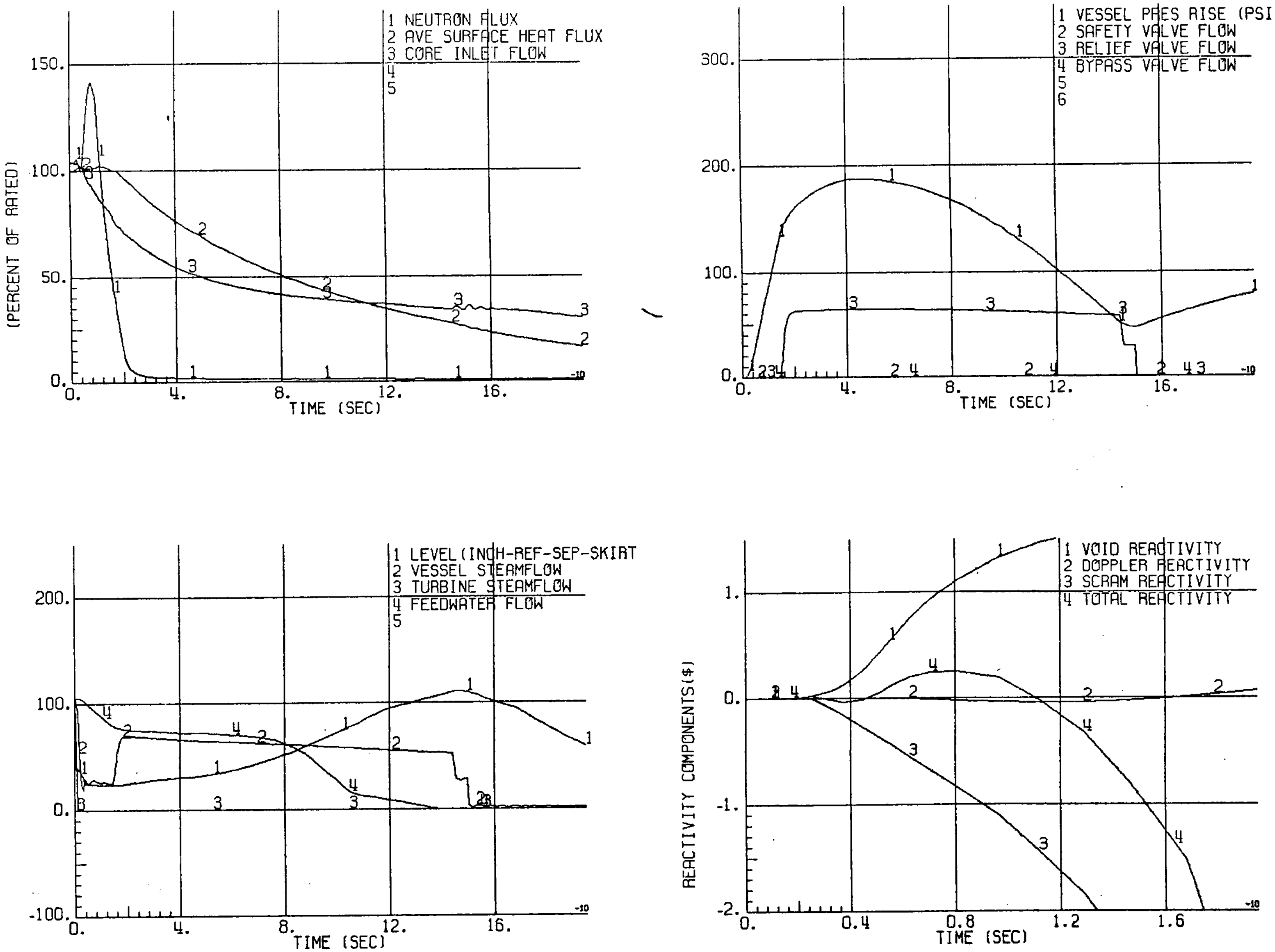


Figure 3. Plant Response to Turbine Trip Without Bypass (REDY)

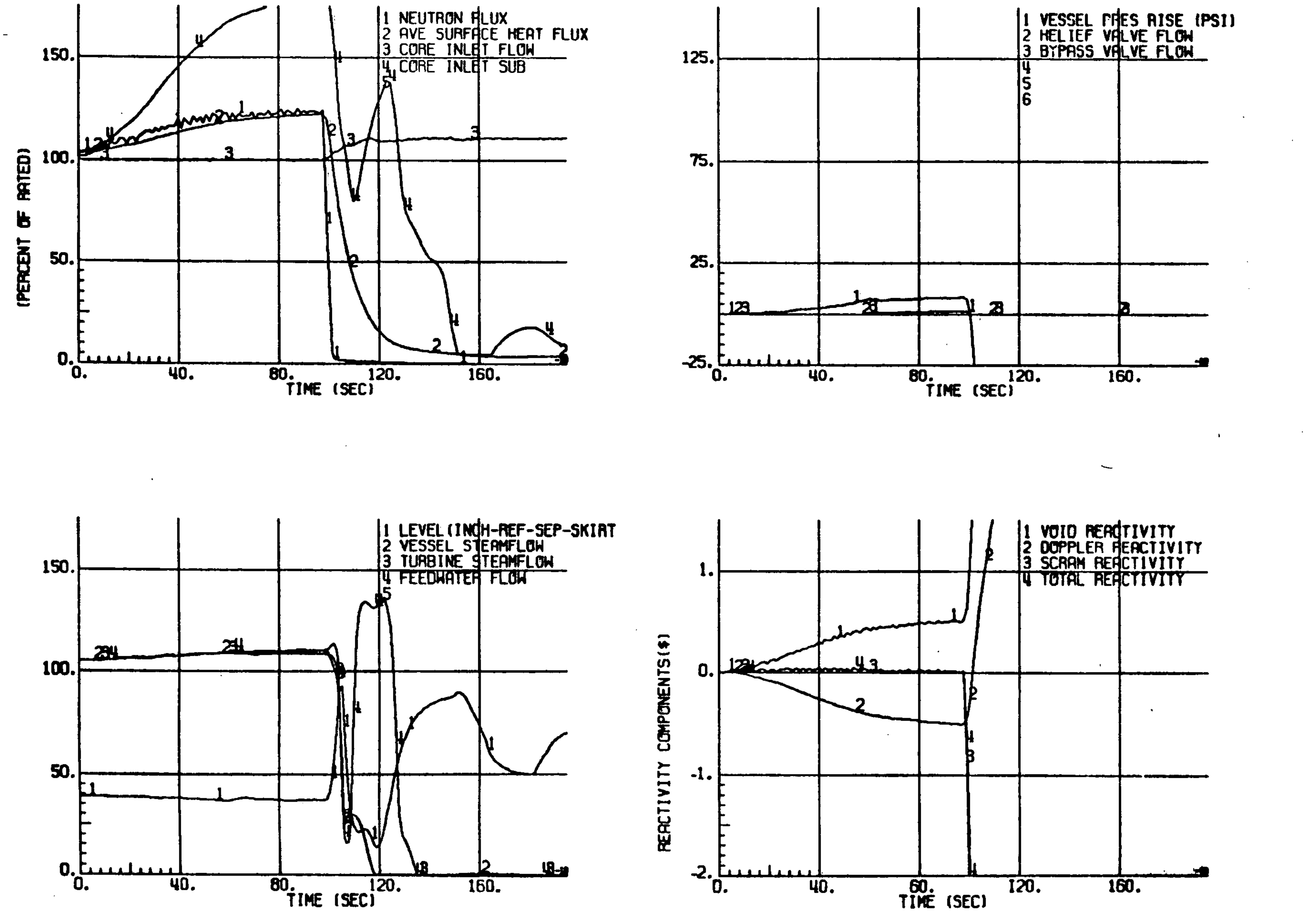


Figure 4. Plant Response to Loss of 100°F Feedwater Heating

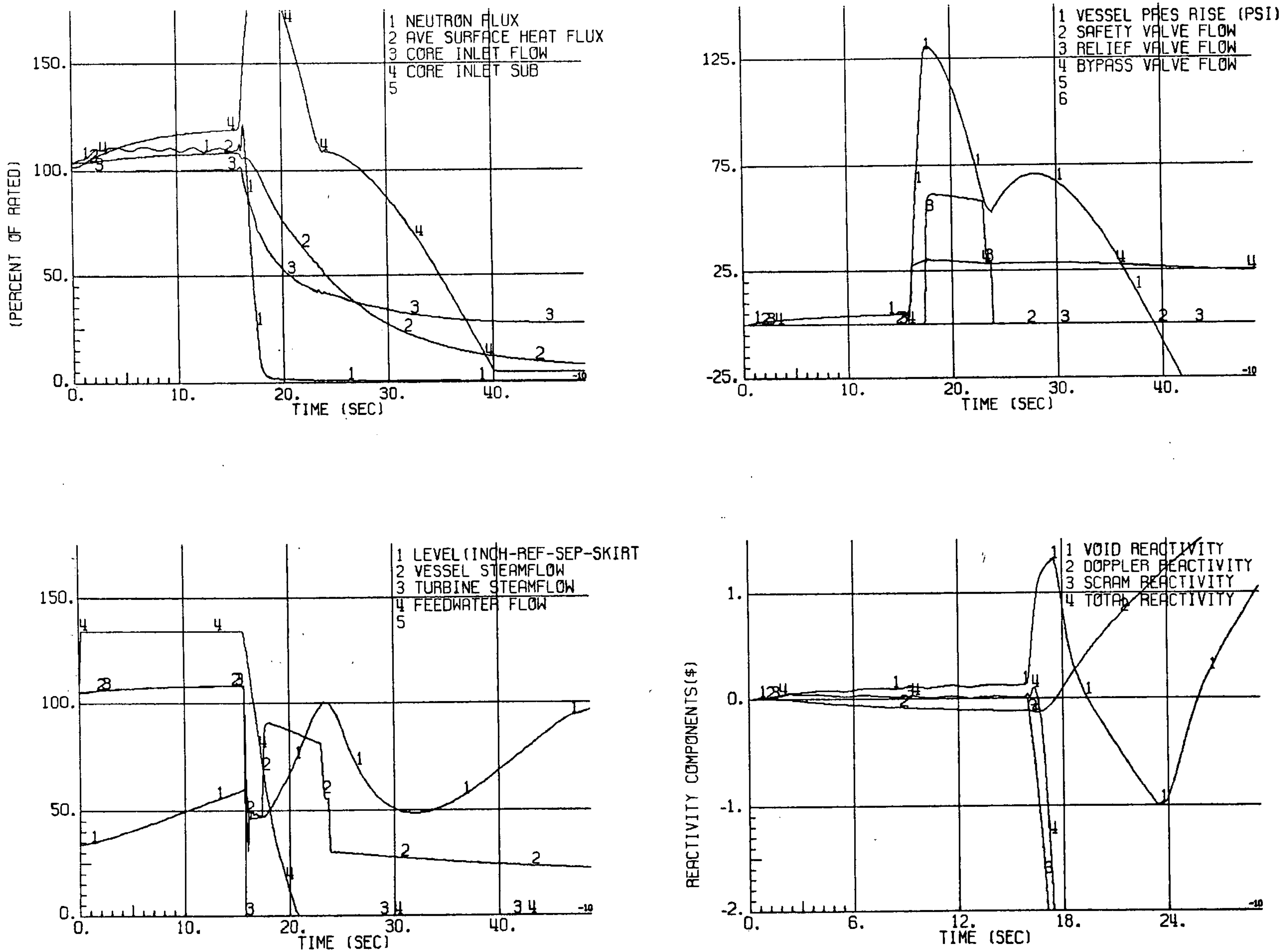


Figure 5. Plant Response to Feedwater Controller Failure (REDY)

	02	06	10	14	18	22
43						
39						
35					8	
31				44		30
27			8		0	
23				30		30

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 18-27.

Figure 6. Limiting RWE Rod Pattern

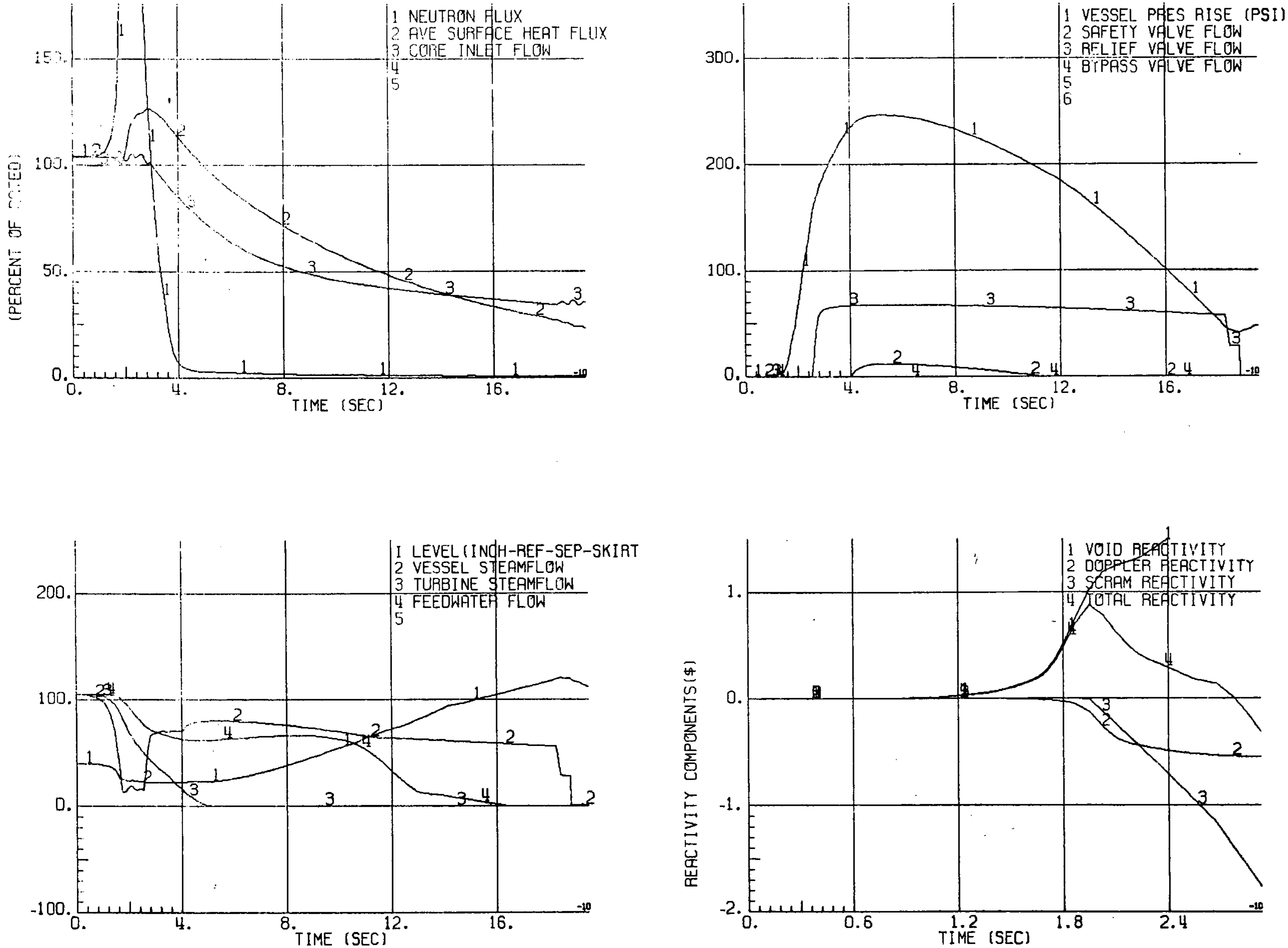


Figure 7. Plant Response to MSIV Closure

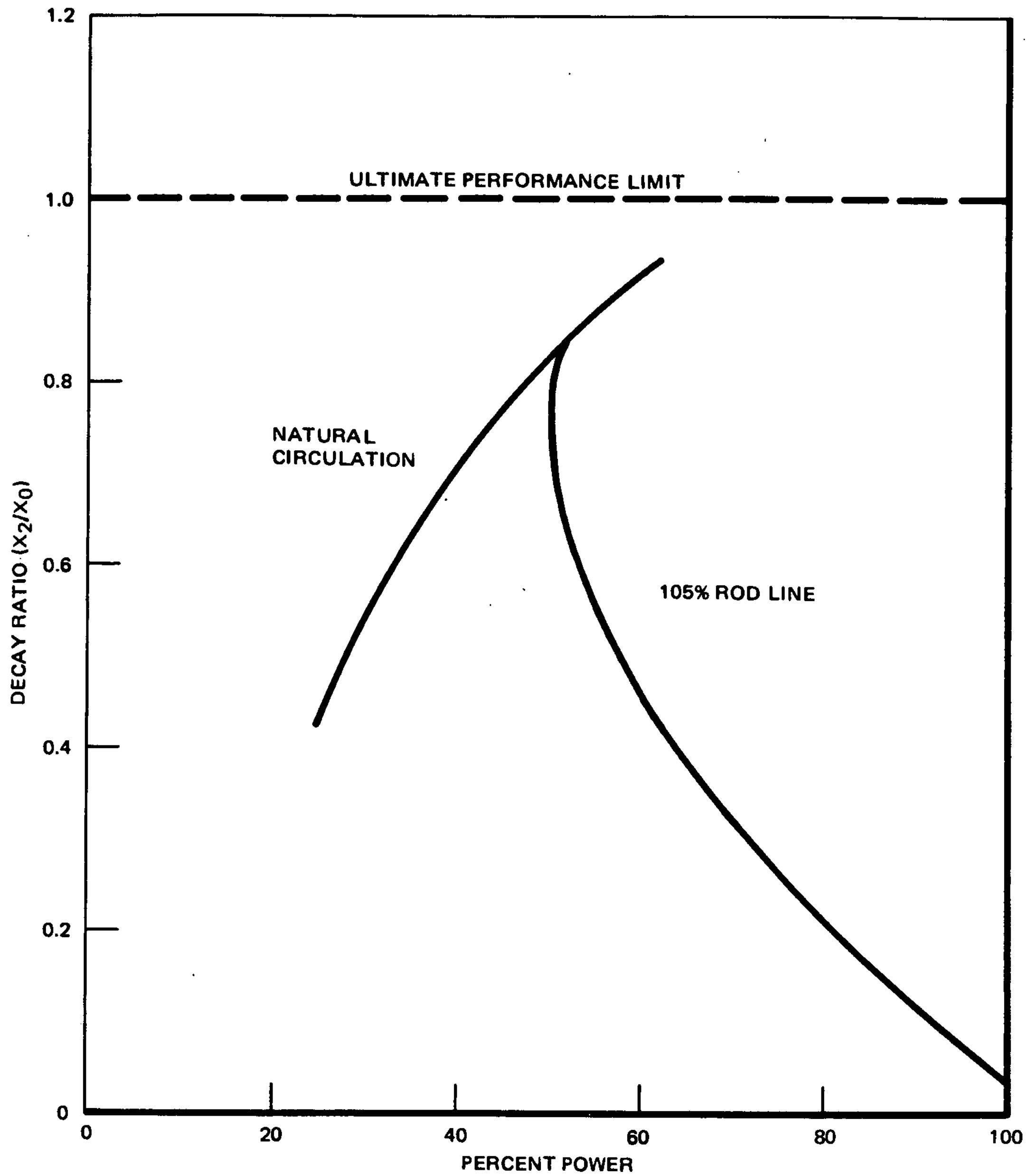


Figure 8. Decay Ratio

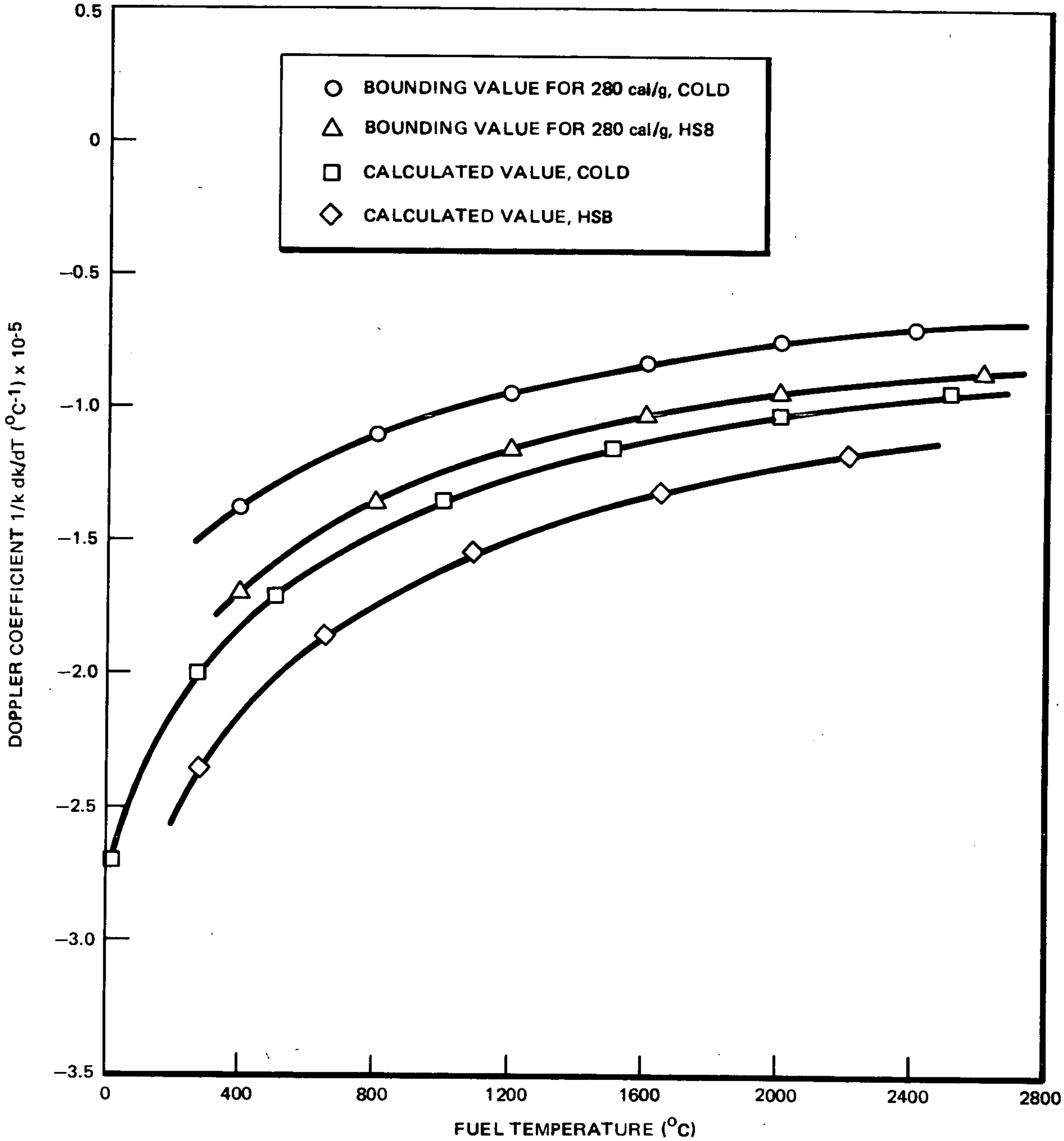


Figure 9. Doppler Reactivity Coefficient Comparison for RDA

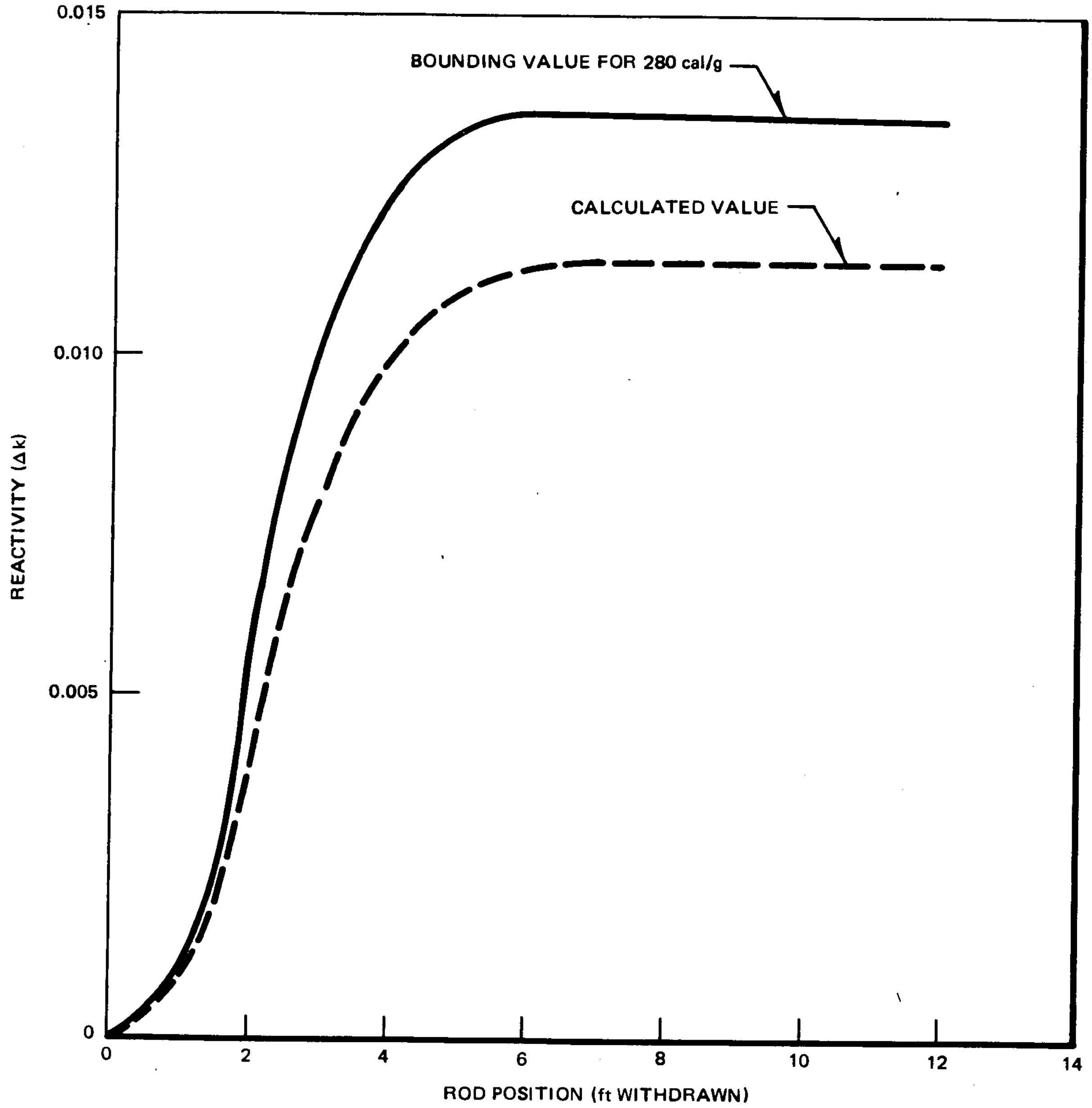


Figure 10. RDA Reactivity Shape Function at 20°C

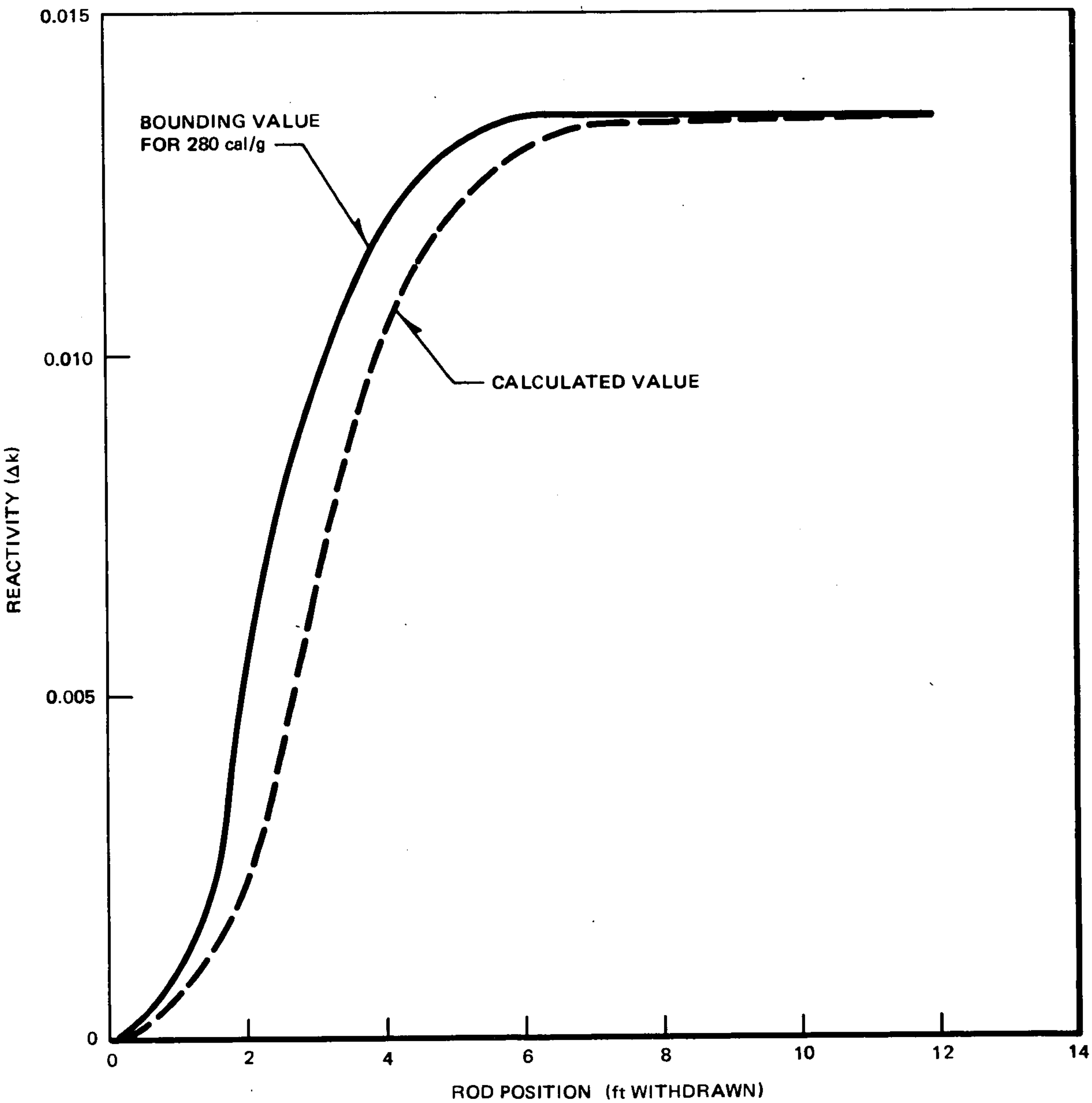


Figure 11. RDA Reactivity Shape Function at 286°C

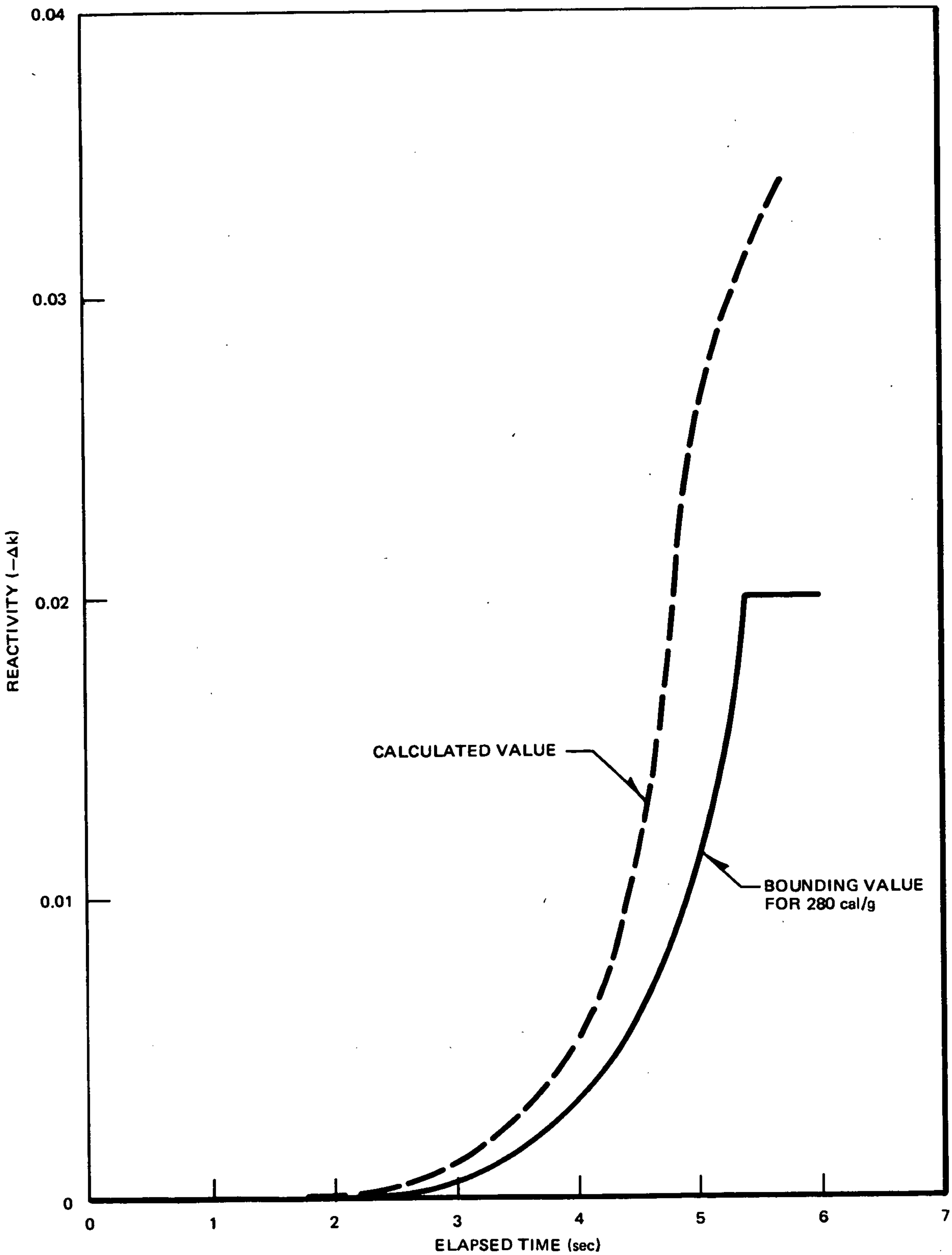


Figure 12. RDA Scram Reactivity Function at 20°C

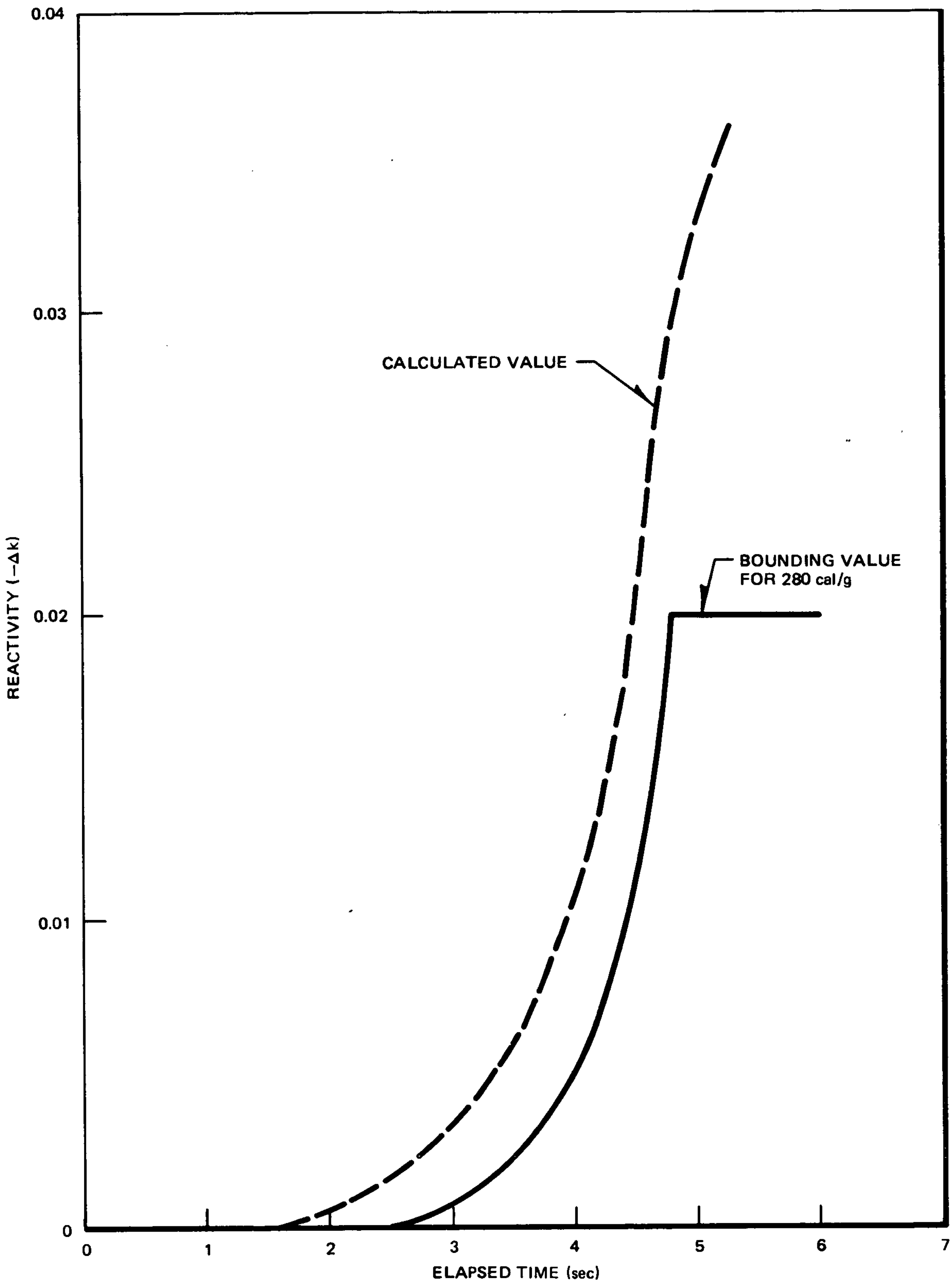


Figure 13. RDA Scram Reactivity Function at 286°C

APPENDIX A
MARGIN-TO-SPRING SAFETY VALVES

The rationale for changing the basis for providing pressure margin to the spring safety valves is presented in:

J. F. Quirk (GE) letter to Olan D. Parr (NRC), "General Electric Licensing Topical Report NEDE-24011-P-A, 'Generic Reload Fuel Application', Appendix D, Second Submittal", dated February 28, 1979.

On this basis the plant can operate at full power throughout the cycle.

The core response to the limiting anticipated event is given in Table A-1 and Figure A-1.

Table A-1
CORE-WIDE TRANSIENT ANALYSIS RESULTS

	Exposure	Power (%)	Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure Trip Scram	BOC-EOC	104	100	1153	1193	Figure A-1

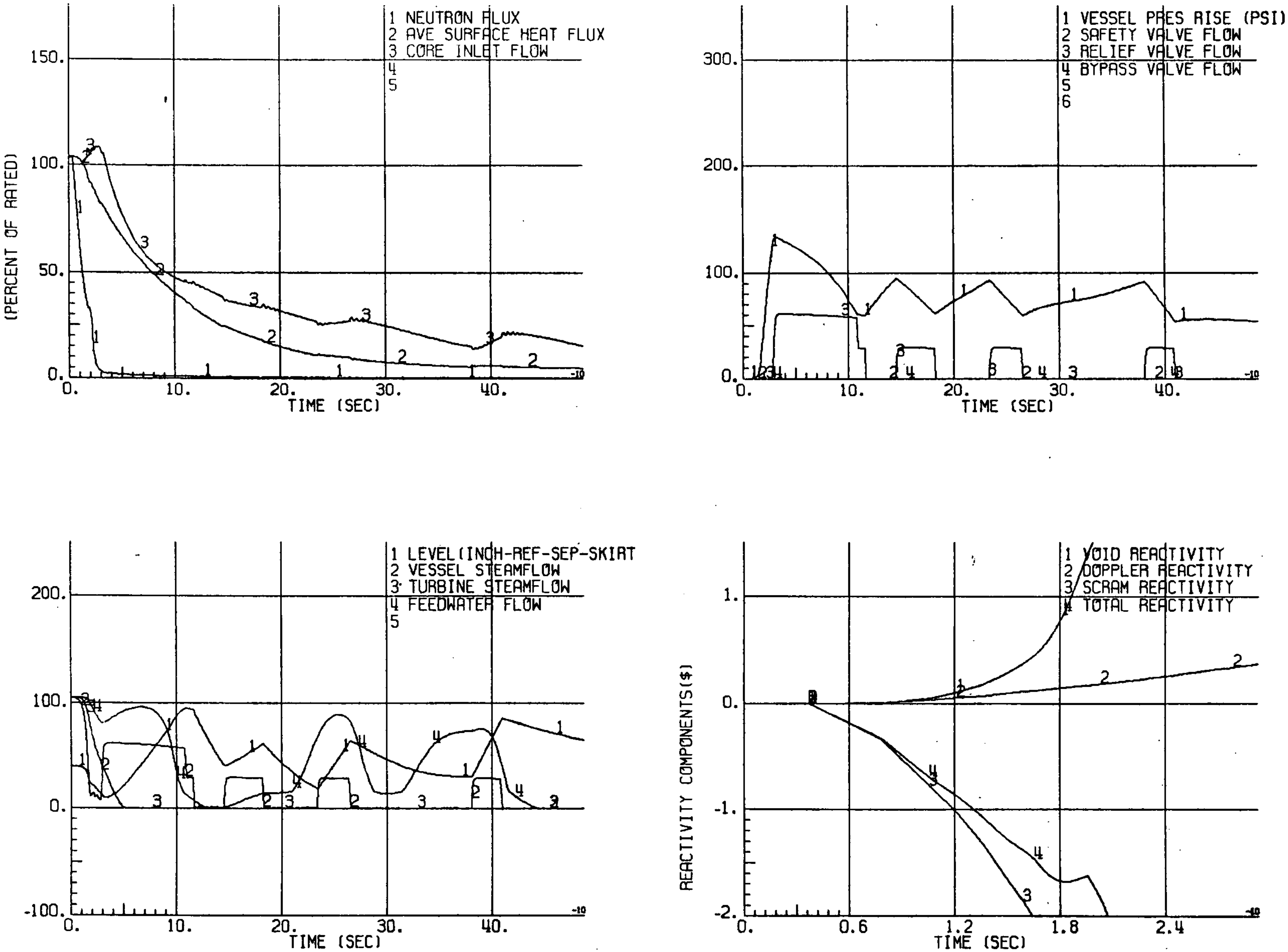


Figure A-1. Plant Response to MSIV Closure, Position Scram

APPENDIX B

NEW BUNDLE LOADING ERROR EVENT ANALYSES PROCEDURES

The bundle loading error analyses results 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.

B.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 B-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 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 for the P8DPB289 bundle a 17.7 kW/ft LHGR (includes densification spiking penalty of 2.2%) and 0.19 ΔCPR (includes a 0.02 penalty due to variable water gap R-factor uncertainty) with a CPR of 1.07.

B.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 indicates that the minimum CPR for mislocated bundles (e.g., P8DPB289 into 8DB274) is greater than the safety limit (1.07) for all exposures throughout Cycle 5.

REFERENCES

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

APPENDIX C

ODYN INPUTS AND RESULTS

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS

Transient	Exposure	Power (%)	Flow (%)	ϕ (% NBR)	Q/A (% NBR)	P_{SL} (psig)	P_v (psig)	ΔCPR 7x7/8x8/P8x8R	Plant Response
Turbine Trip no Bypass	EOC 5	104	100	214	106	1221	1236	0.01/0.03/0.04	C3
Feedwater Controller Failure	EOC 5	104	100	191	110	1154	1178	0.03/0.06/0.07	C5

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

	EOC 5
Void Coefficient N/A (¢/% R_g)	-9.51/-11.89
Void Fraction (%)	44.2
Doppler Coefficient N/A (¢/°F)	-0.217/-0.206
Average Fuel Temperature (°F)	1359
Scram Worth	-37.23/-29.78

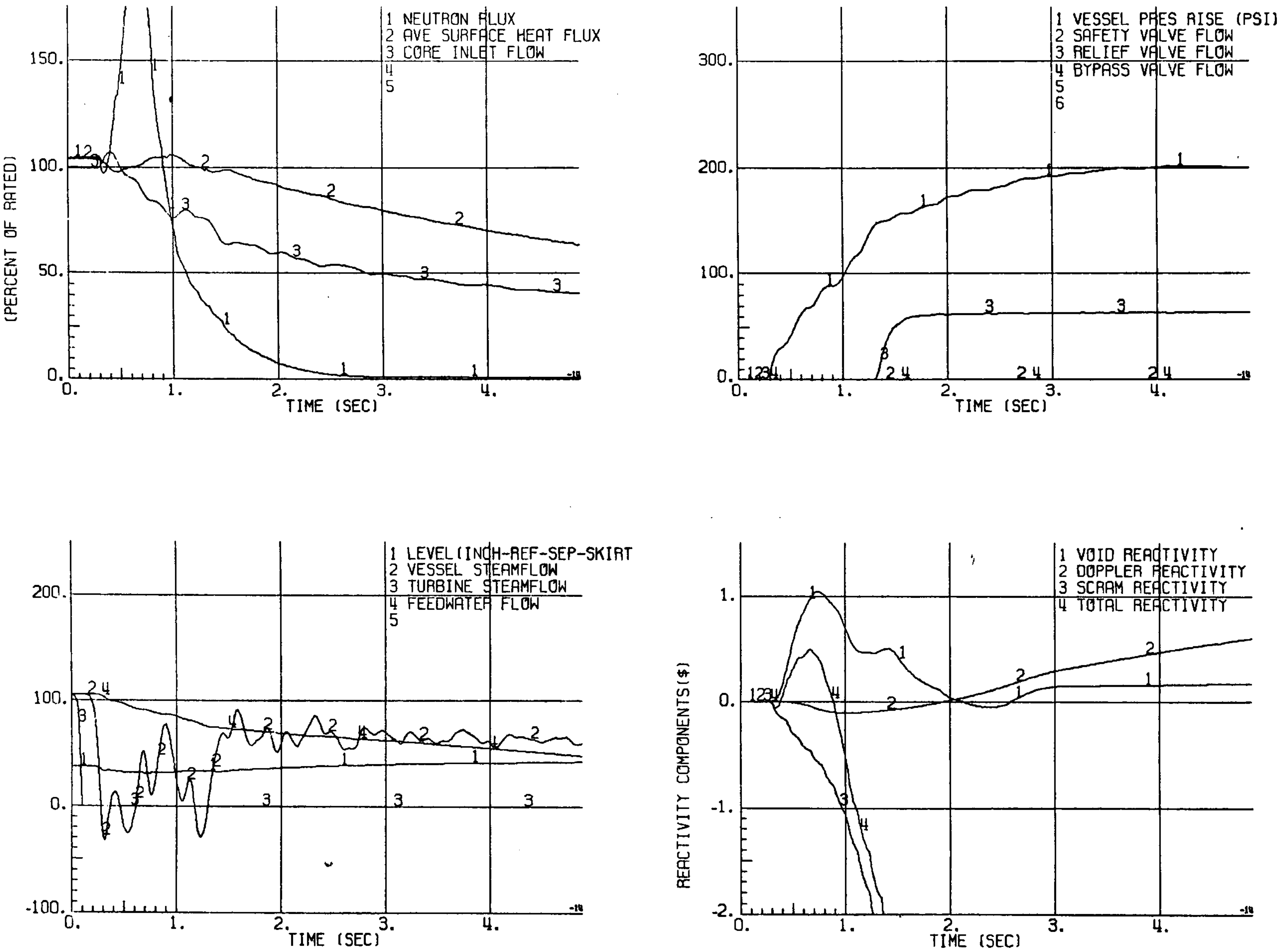


Figure C-1. Plant Response to Turbine Trip Without Bypass (ODYN)

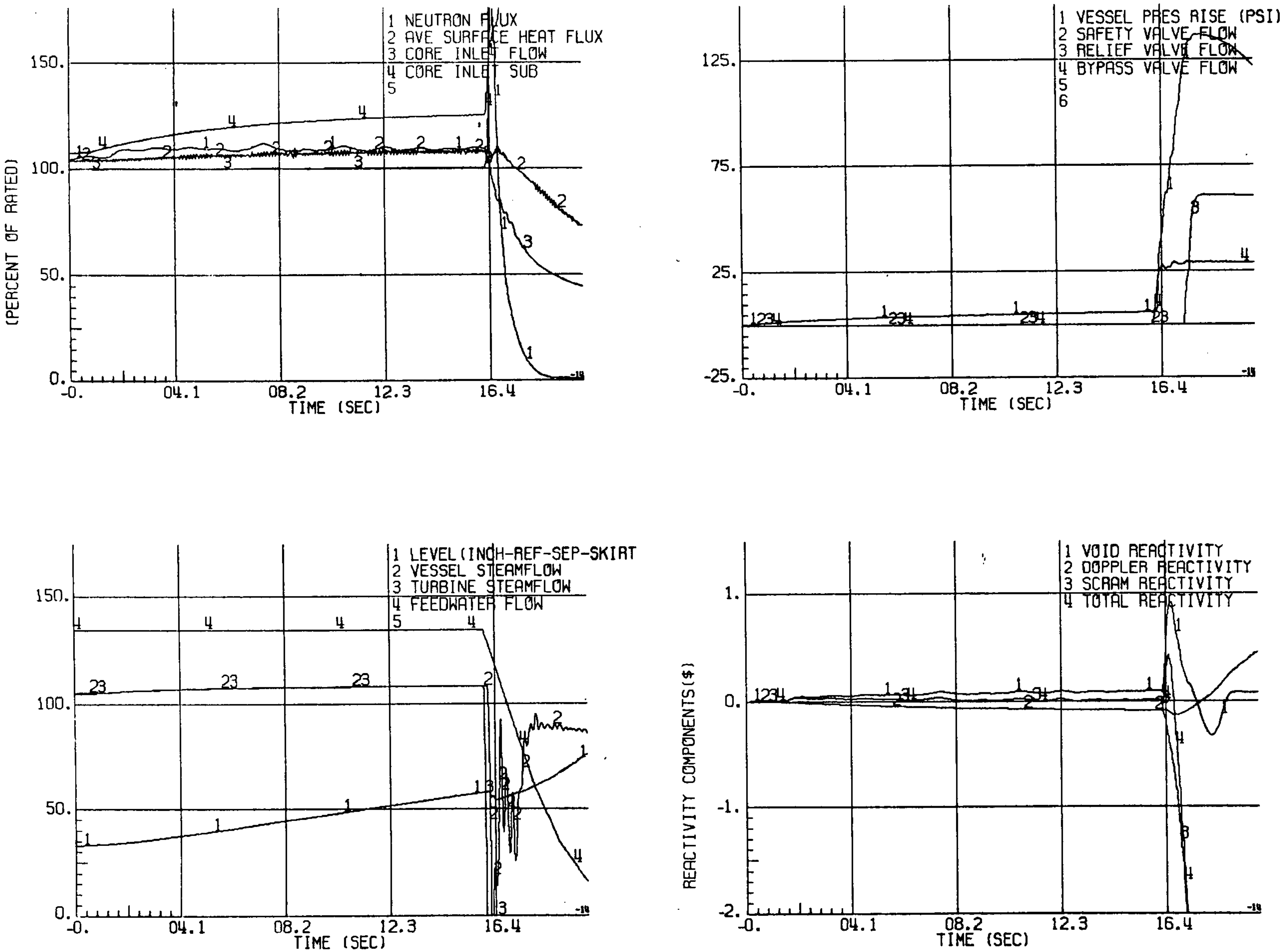


Figure C-2. Plant Response to Feedwater Controller Failure (ODYN)

APPENDIX D
TRANSIENT OPERATING PARAMETERS

Rated Steam Flow: 7.19×10^6 lb/hr

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