NEDO-24170
78NED402
CLASS I
DECEMBER 1978

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR COOPER NUCLEAR STATION UNIT 1 RELOAD 4

GENERAL ELECTRIC

NEDO-24170
78NED402
Class I
December 1978

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
COOPER NUCLEAR STATION
UNIT 1 RELOAD 4

Prepared: J. L. Rash

Approved: R. E. Engel, Manager
Operating Licenses I

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

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT
PLEASE READ CAREFULLY

This report was prepared by General Electric solely for Nebraska Public Power District (NPPD) for NPPD's use with the U.S. Nuclear Regulatory Commission (USNRC) for amending NPPD's operating license of the Cooper Nuclear Station. The information contained in this report is believed by General Electric to be an accurate and true representation of the facts known, obtained or provided to General Electric at the time this report was prepared.

The only undertakings of the General Electric Company respecting information in this document are contained in the contract between Nebraska Public Power District and General Electric Company for nuclear fuel and related services for the nuclear system for Cooper Nuclear Station and nothing contained in this document shall be construed as changing said contract. The use of this information except as defined by said contract, or for any purpose other than that for which it is intended, is not authorized; and with respect to any such unauthorized use, neither General Electric Company nor any of the contributors to this document makes any representation or warranty (express or implied) as to the completeness, accuracy or usefulness of the information contained in this document or that such use of such information may not infringe privately owned rights; nor do they assume any responsibility for liability or damage of any kind which may result from such use of such information.

1. PLANT-UNIQUE ITEMS (1.0)*

Bundle Loading Error Analysis - Appendix A
GETAB Analysis Initial Conditions - Appendix B
Densification Power Spiking - Appendix C

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

	Fuel Type	Number	Number Drilled
Irradiated	Initial Core 7DB250	152	152
Irradiated	8DB250 (R1)	72	72
Irradiated	8DB274L (R1&2)	112	112
Irradiated	8DB274L (R3)	24	24
Irradiated	8DRB283 (R3)	76	76
New	8DRB283 (R4)	112	112
	Total	548	548

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure:	15,753 MWd/t
Assumed reload cycle exposure:	16,230 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	1.110
Fully Controlled	0.952
Strongest Control Rod Out	0.986
R, Maximum Increase in Code 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.

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.036

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

	EOC5
Void Coefficient N/A* (¢/% Rg)	-8.15/-10.19
Void Fraction (%)	40.03
Doppler Coefficient N/A (¢/%°F)	-0.228/-0.217
Average Fuel Temperature (°F)	1352
Scram Worth N/A ($)	-38.78/-31.02
Scram Reactivity vs Time	Figure 2

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

Exposure	7x7 EOC5	8x8/8x8R EOC5
Peaking factors (local, radial and axial)	(1.24, 1.378, 1.4)	(1.19, 1.578, 1.4)
R-Factor	1.08	1.054
Bundle Power (MWt)	5.864	6.703
Bundle Flow (10^3 lb/hr)	119.85	109.86
Initial MCPR	1.20	1.22

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

None

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

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Core Flow (%)	$\hat{\phi}$ (% NBR)	$\hat{Q}/A$ (% NBR)	P_{sl} (psig)	P_v (psig)	ΔCPR 7x7	ΔCPR 8x8/8x8R	Plant Response
Turbine Trip Without Bypass	BOC-EOC5	104	100	226	107	1170	1194	0.09	0.14	Figure 3
Load Rejection Without Bypass	BOC-EOC5	104	100	248	108	1172	1196	0.10	0.15	Figure 4
Loss of 100°F Feedwater Heating	BOC-EOC5	104	100	124	117	1023	1069	0.13	0.14	Figure 5
Feedwater Controller Failure	BOC-EOC5	104	100	168	110	1123	1165	0.09	0.13	Figure 6

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

Rod Block Reading	Rod Position (Feet Withdrawn)	ΔCPR 7x7	ΔCPR 8x8/8x8R	MLHGR (Kw/ft) 7x7	MLHGR (Kw/ft) 8x8/8x8R	Limiting Rod Pattern
105*	3.5	0.16	0.10	17.3	16.1	Figure 7
106	4.0	0.23	0.12	17.5	16.4	Figure 7
107	4.5	0.31	0.13	17.5	16.4	Figure 7
108	5.0	0.33	0.14	17.3	16.2	Figure 7
109	5.5	0.33	0.15	17.2	16.2	Figure 7
110	7.0	0.32	0.19	16.7	15.5	Figure 7

*Indicates setpoint selected

11. OPERATING MCPR LIMIT (5.2)

<u>BOC to EOC5</u>

1.23 (8x8/8x8R fuel)

1.23 (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	1237	1276	Figure 8

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio:	Figure 9
Reactor Core Stability:	
Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power)	0.79
Channel Hydrodynamic Performance	Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power)
8x8/8x8R channel	0.37
7x7 channel	0.23

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

No new bundle types are added; no ECCS Tech Spec changes will be necessitated by this reload.

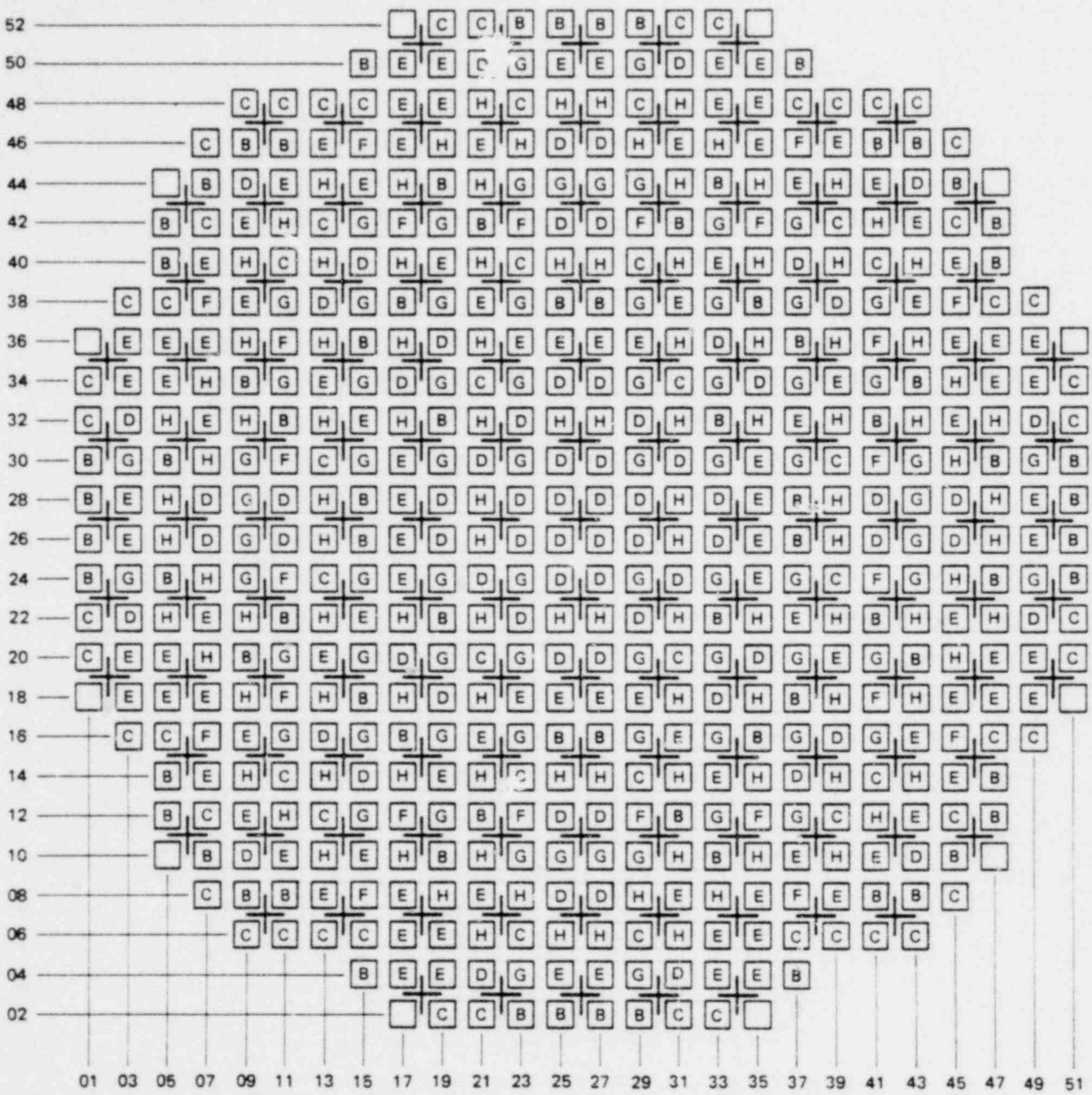
15. LOADING ERROR RESULTS (5.5.4)

Limiting Event: Rotated Bundle (8DRB283)

MCPR: 1.07

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Doppler Reactivity Coefficient:	Figure 10
Accident Reactivity Shape Functions:	Figures 11 and 12
Scram Reactivity Functions:	Figures 13 and 14
Plant specific analysis results	
Parameter not bounded:	Accident Reactivity Shape Function, Cold Startup
Resultant peak enthalpy:	214 cal/gm



FUEL TYPE	
A = NOT USED	E = RELOAD 1 AND 2 8DB274
B = INITIAL CORE 7DB250	F = RELOAD 3 8DB274
C = INITIAL CORE 7DB250	G = RELOAD 3 8DRB283
D = RELOAD 1 8DB250	H = RELOAD 4 8DRB283

Figure 1. Reference Core Loading Pattern

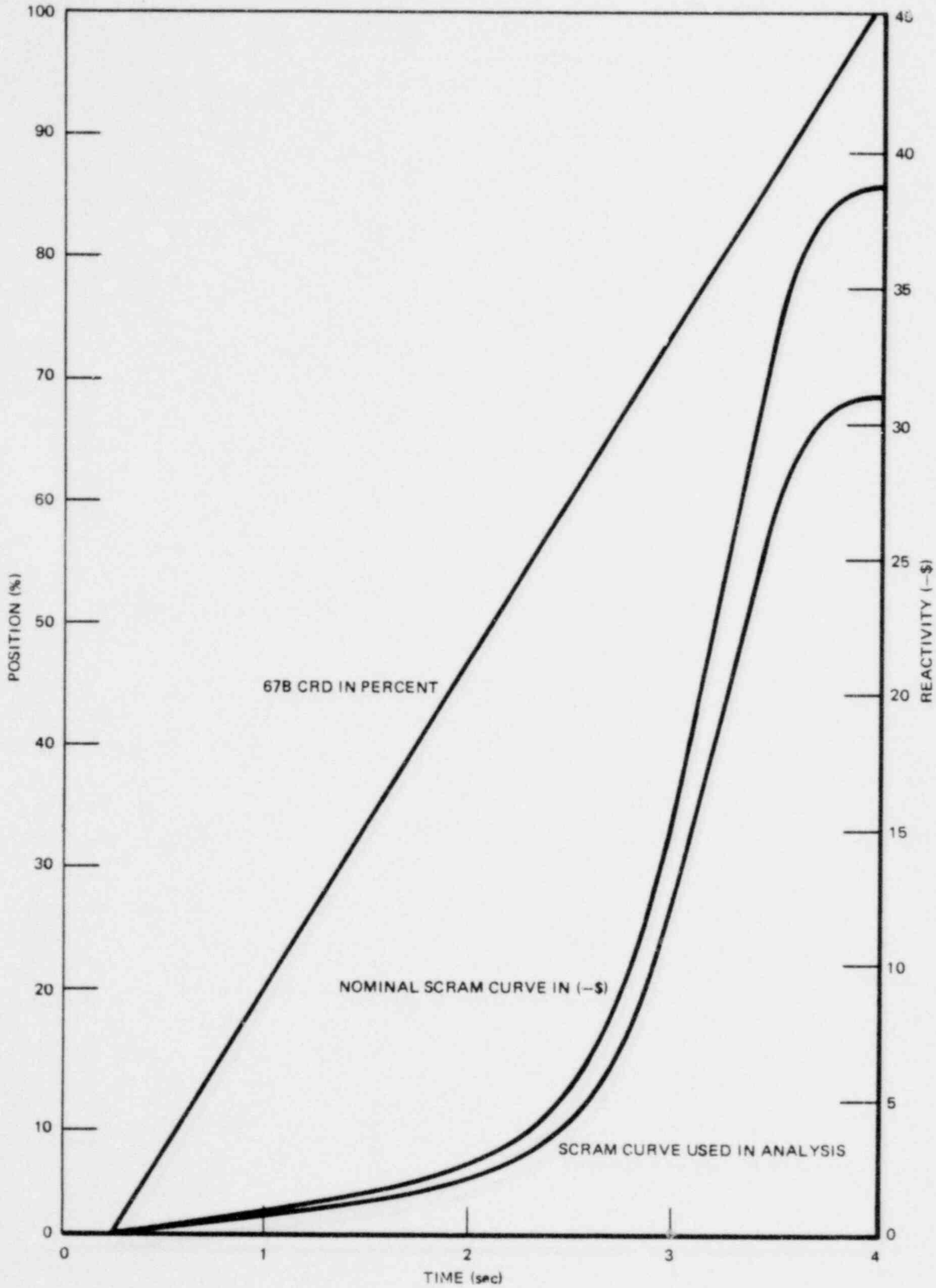


Figure 2. Scram Reactivity and Control Rod Drive Specifications

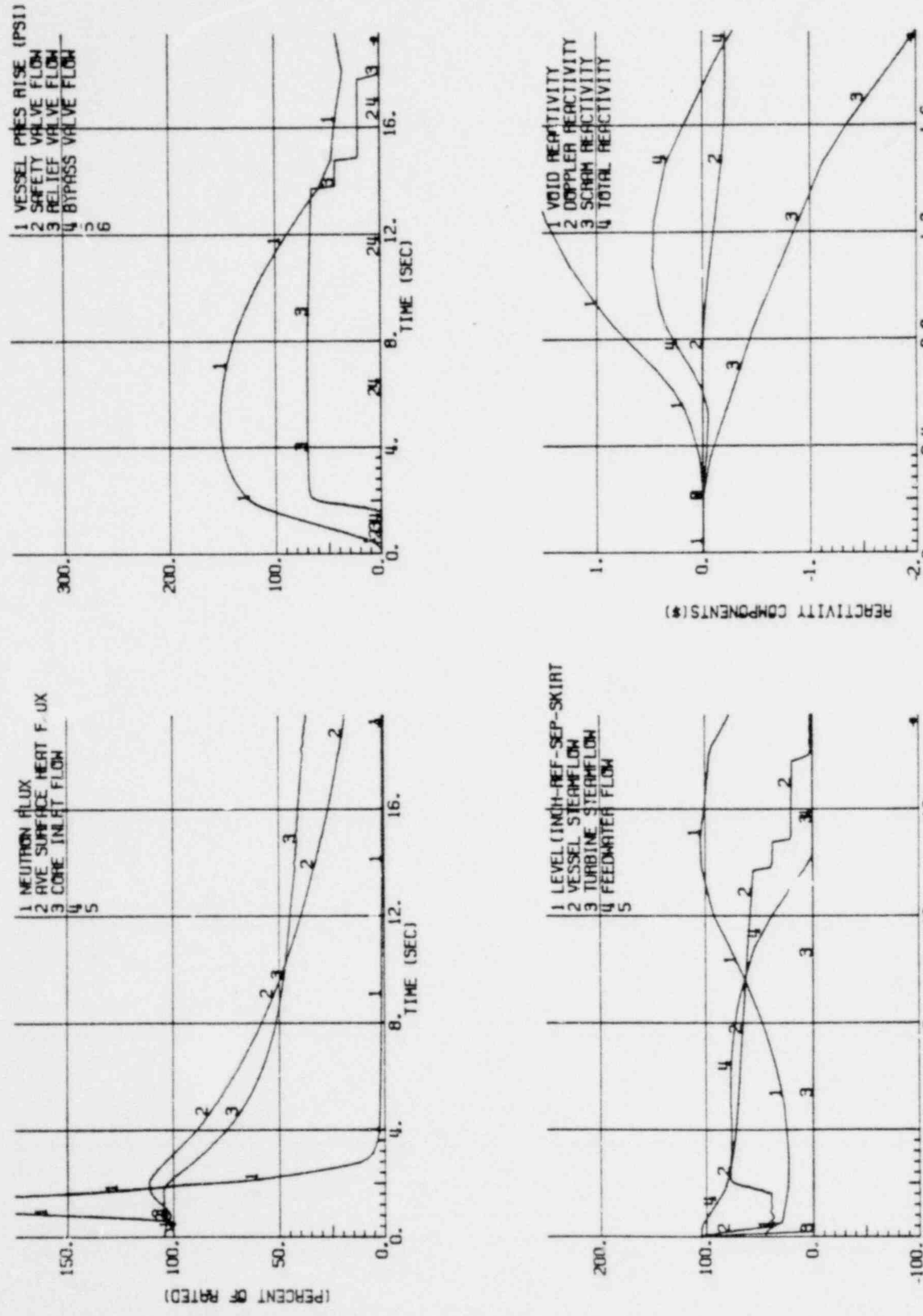


Figure 3. Plant Response to Turbine Trip Without Bypass

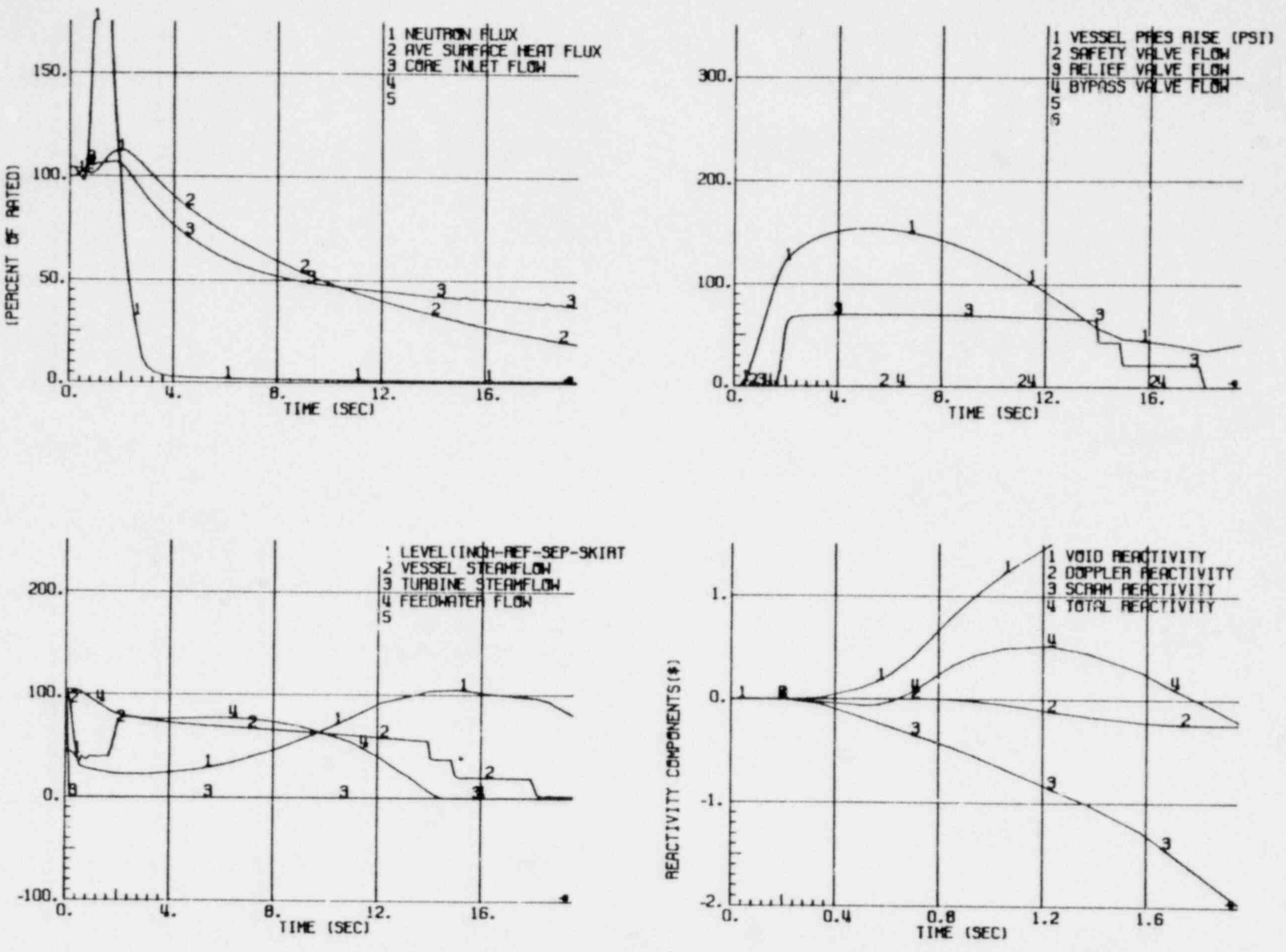


Figure 4. Plant Response to Generator Load Rejection, without Bypass

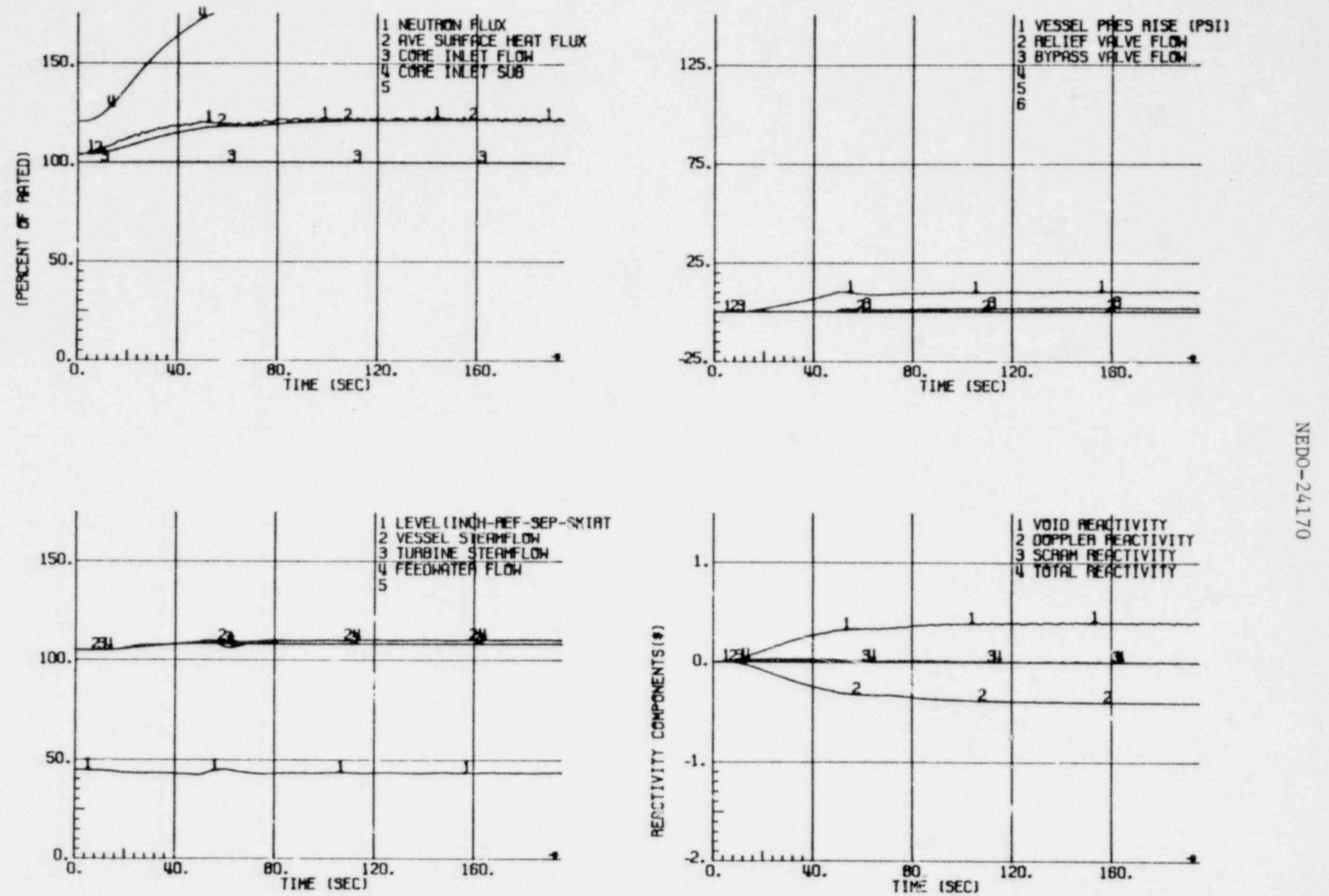


Figure 5. Plant Response to Loss of 100°F Feedwater Heating, MFC

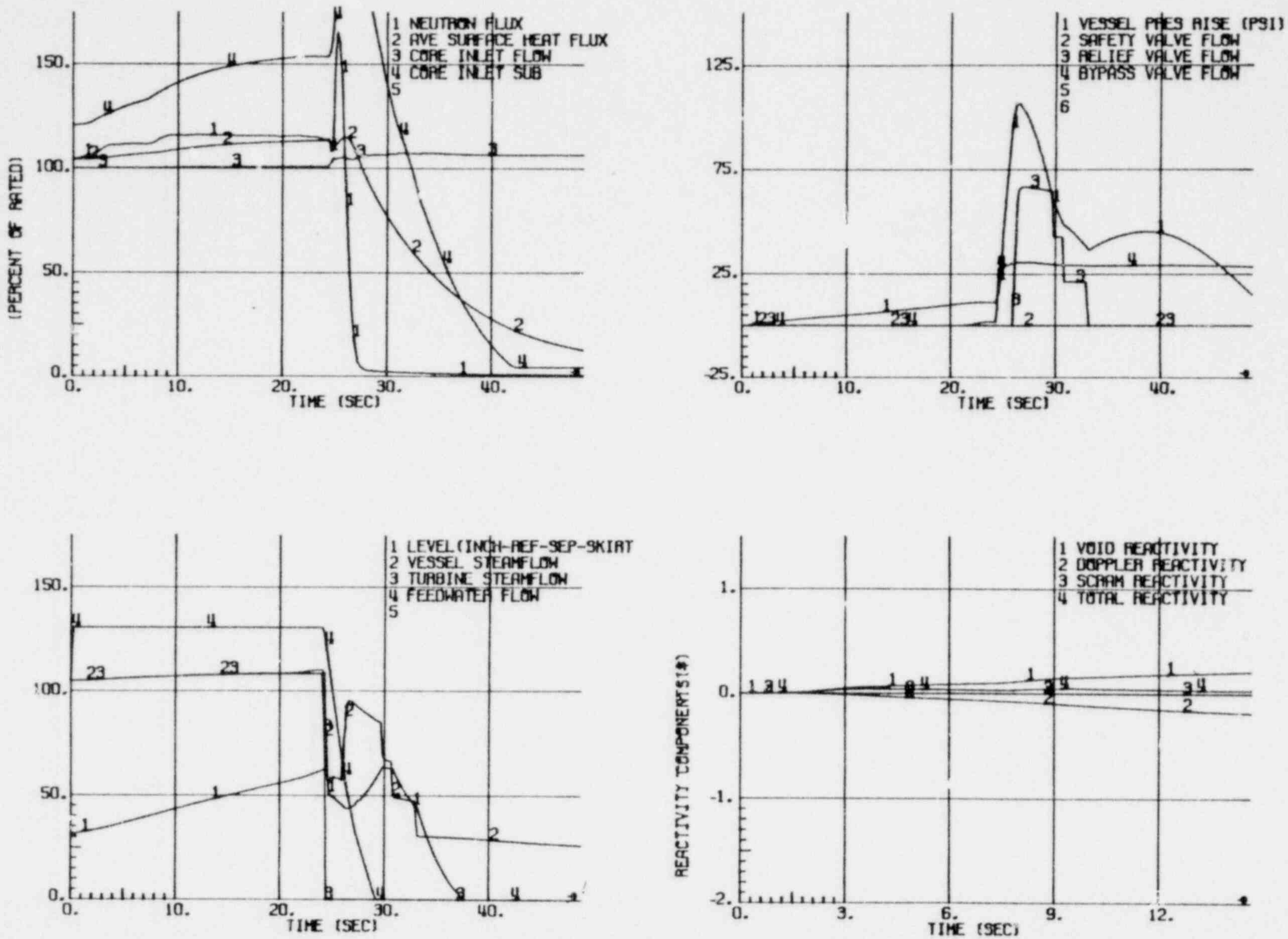


Figure 6. Plant Response to Feedwater Failure, Maximum Demand, with High Level Turbine Trip

	02	06	10	14	18	22	26	30
51								
47					10		10	
43				30		18		
39			6		10		6	
35		30		40		42		
31			14		0		18	
27		40		42		42		
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 (18,31).

Figure 7. Limiting RWE Rod Pattern

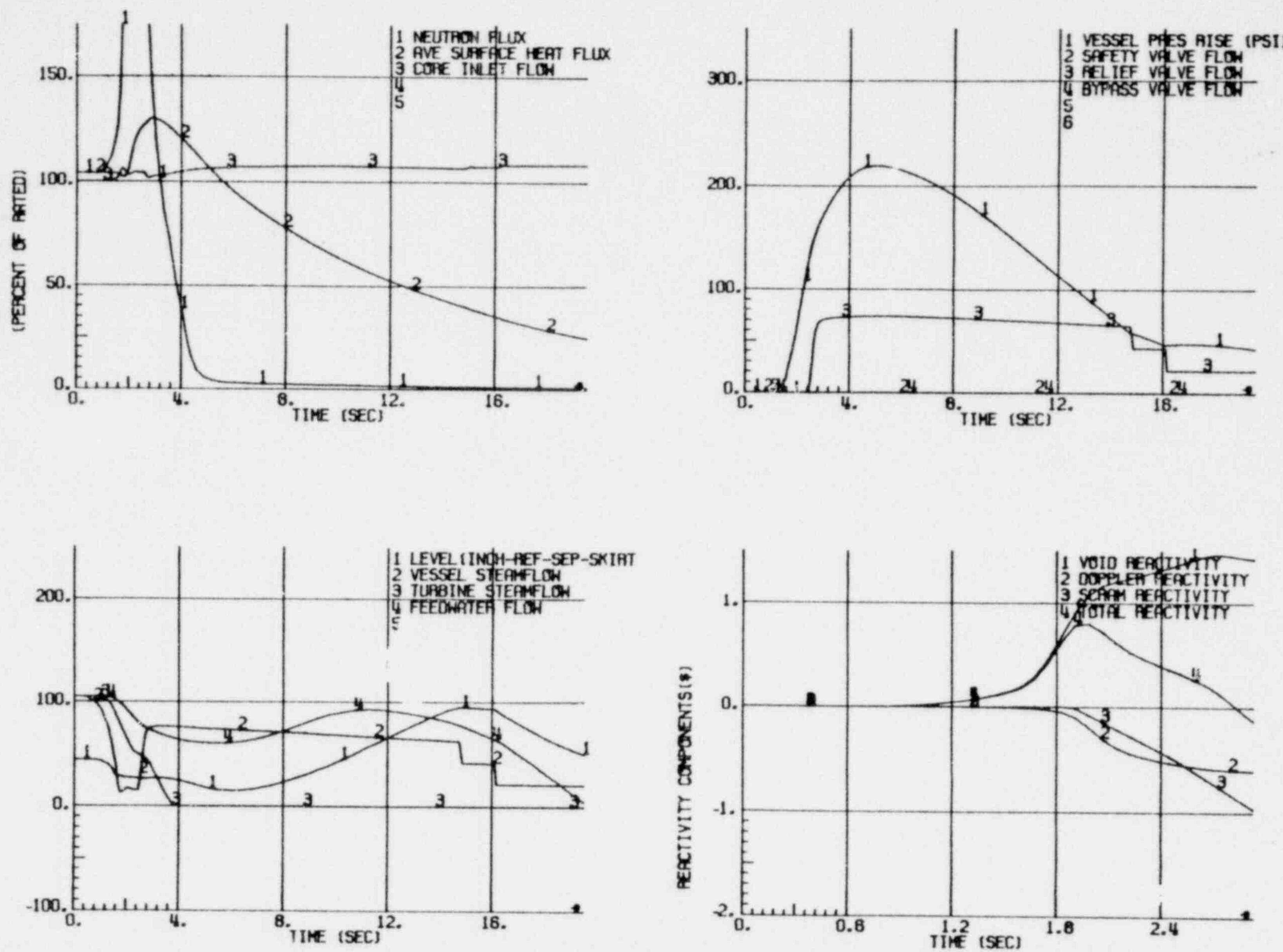


Figure 8. Plant Response to MSIV Closure, Flux Scram

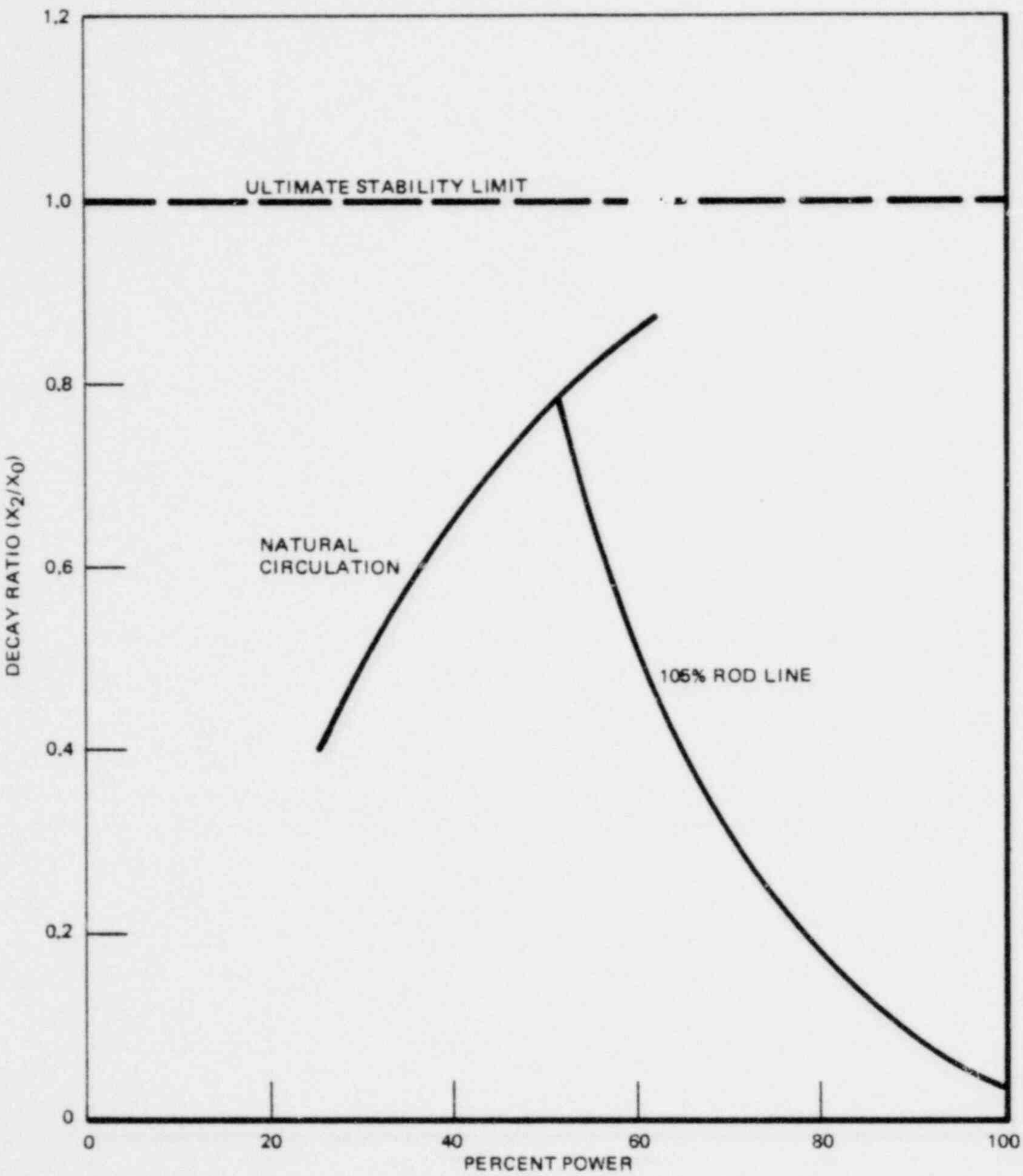


Figure 9. Decay Ratio

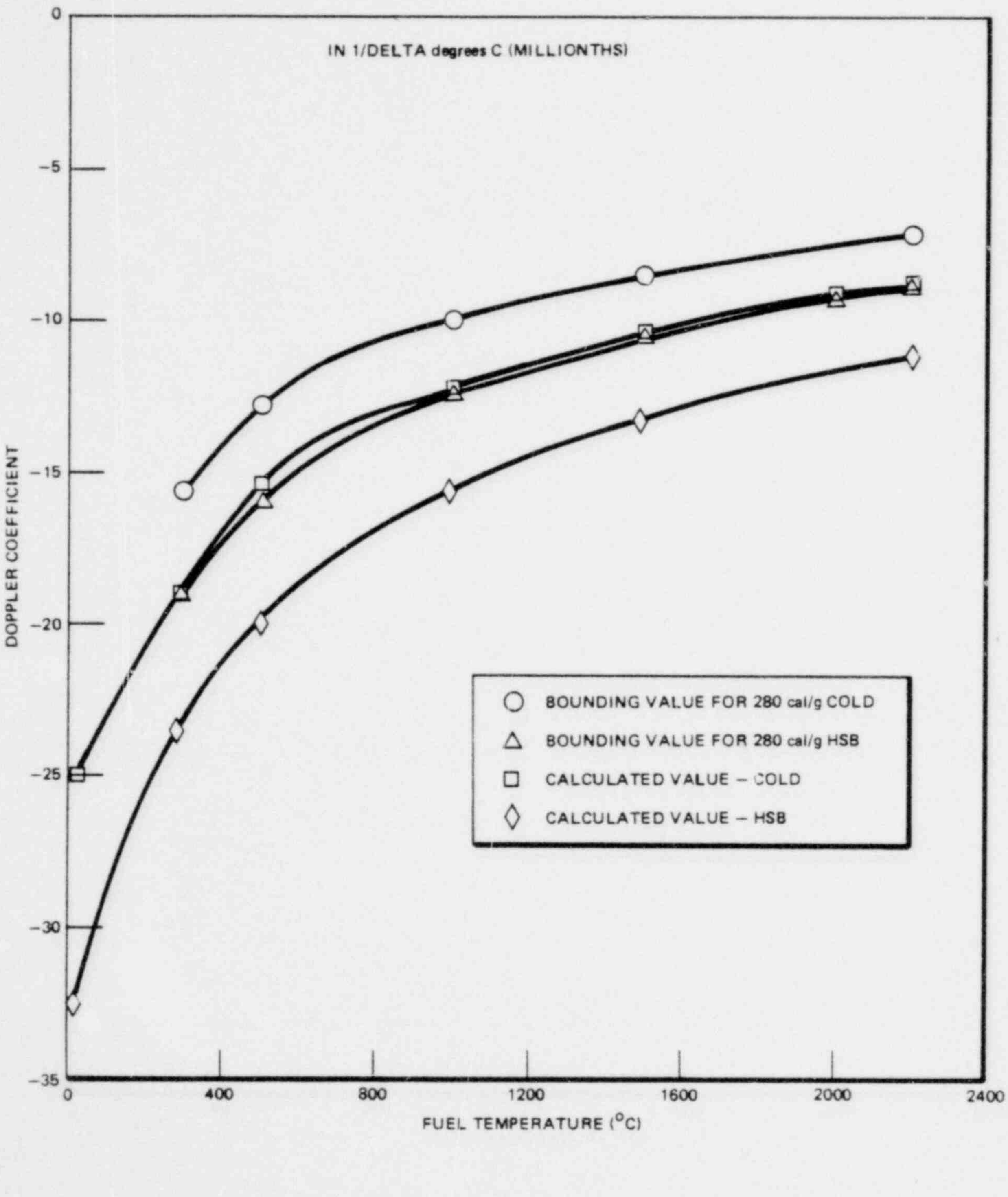


Figure 10. Fuel Doppler Reactivity Coefficient Comparison for RDA

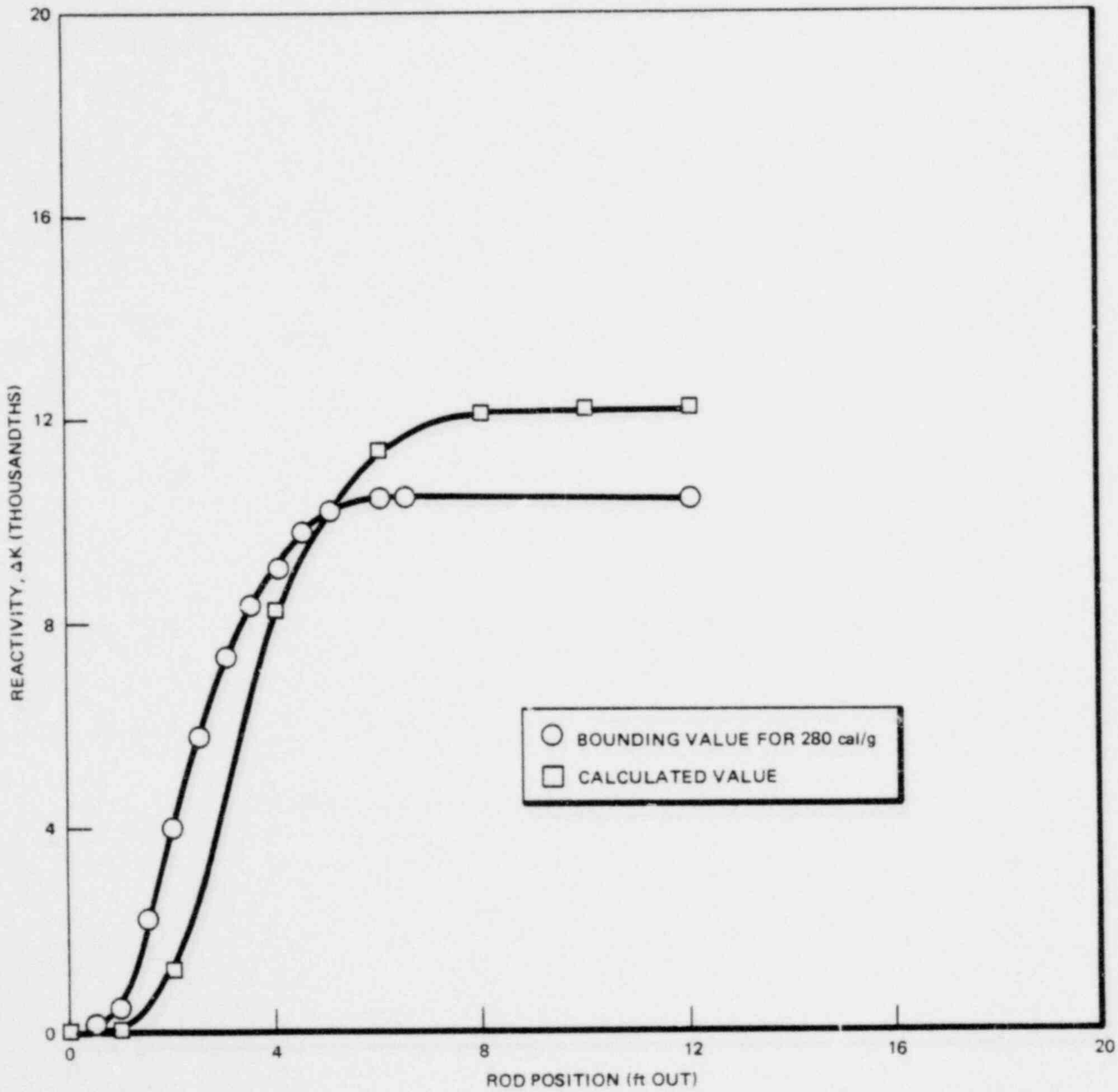


Figure 11. RDA Reactivity Shape Function at 20°C

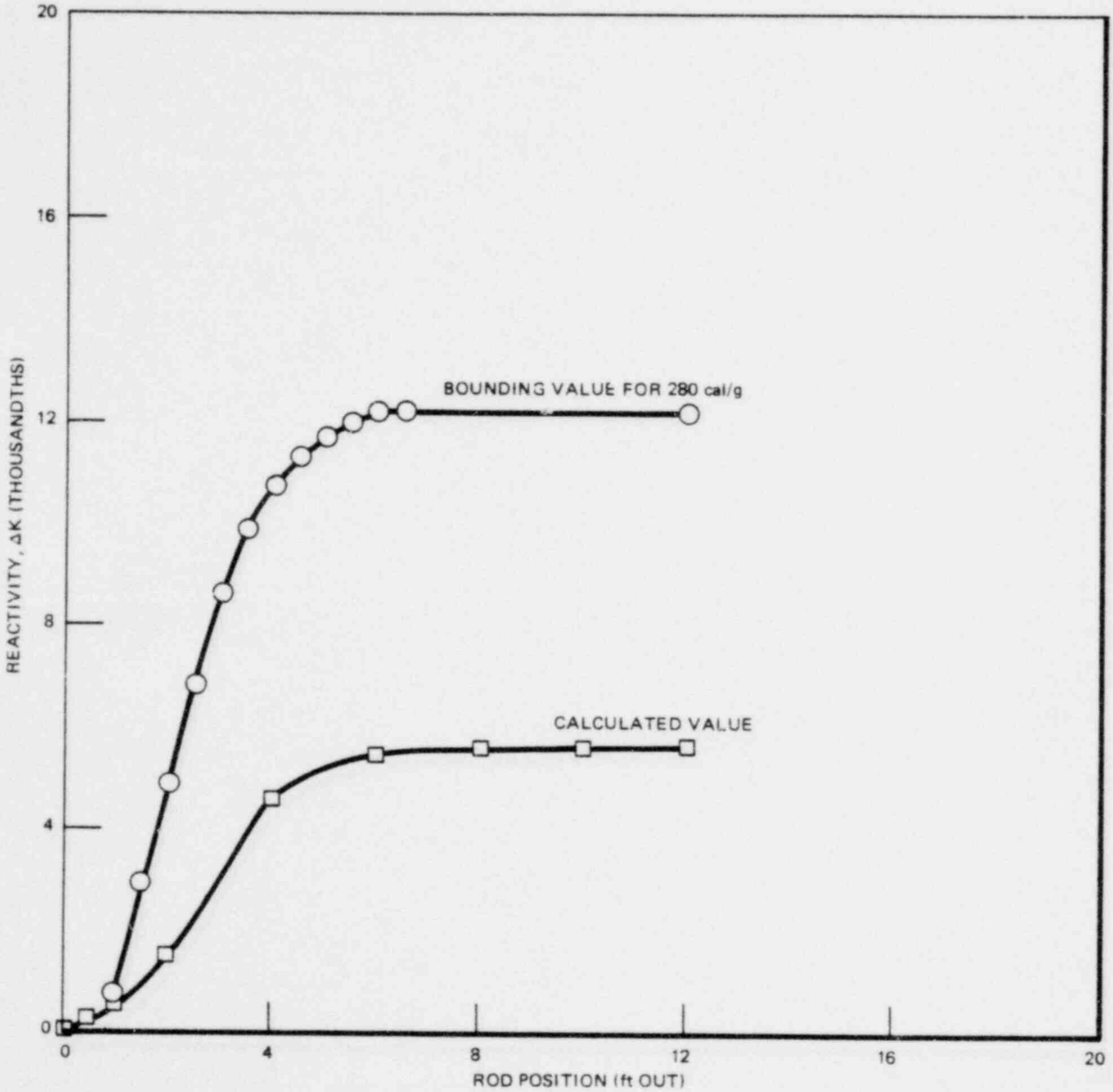


Figure 12. RDA Reactivity Shape Function at 286°C

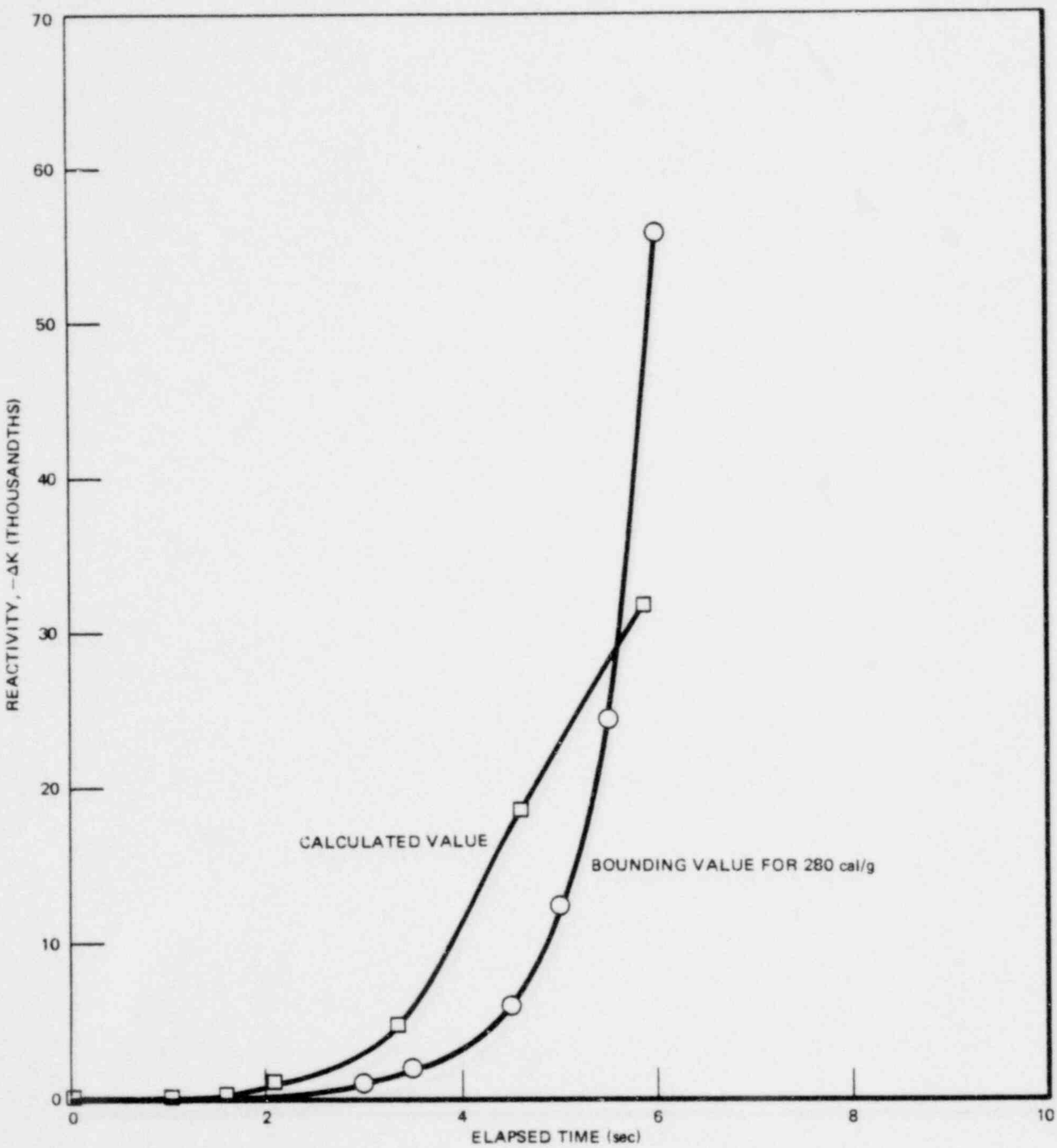


Figure 13. RDA Scram Reactivity Function at 20°C

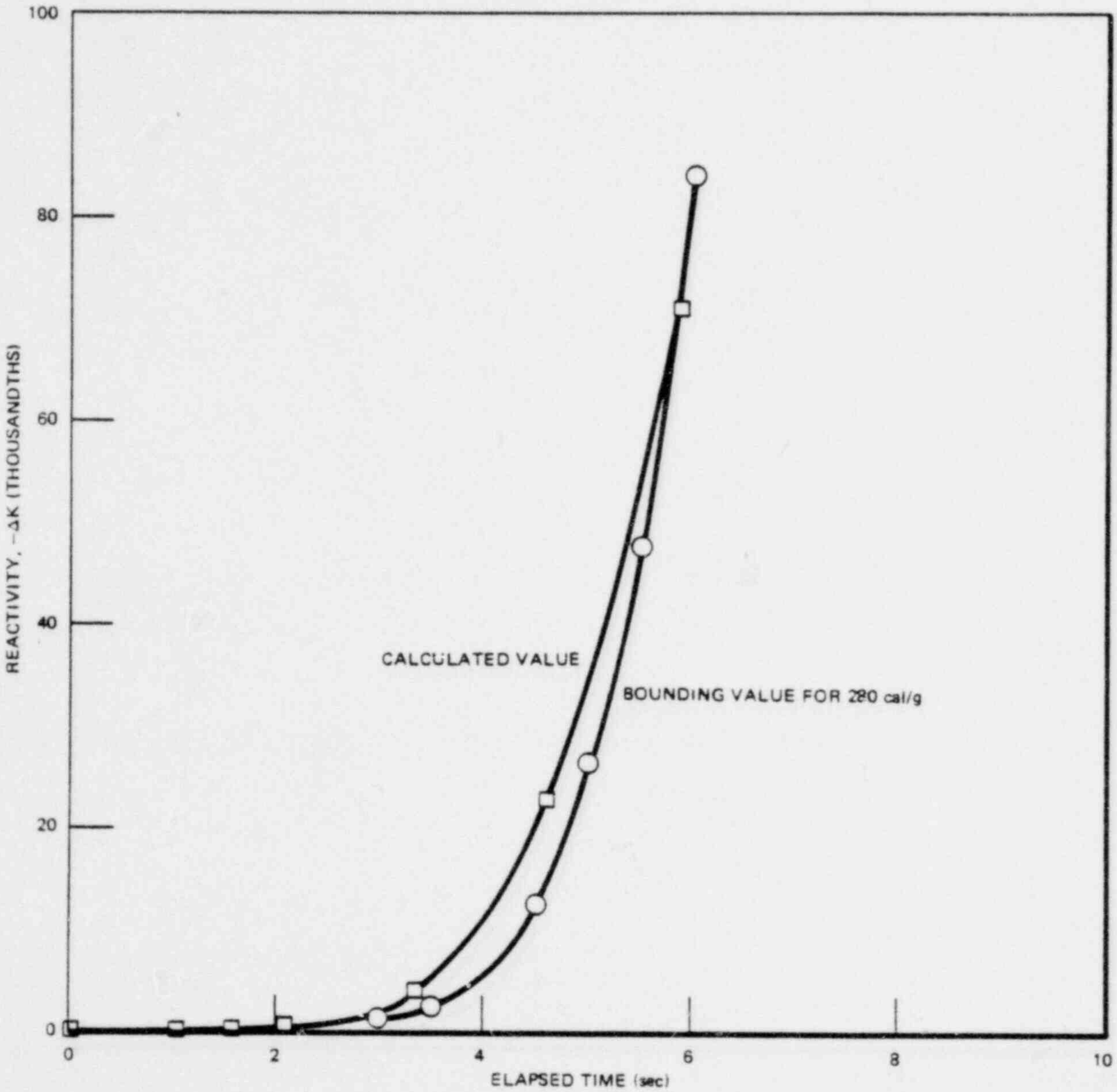


Figure 14. RDA Scram Reactivity Function at 286°C

APPENDIX A
BUNDLE LOADING ACCIDENT

The loading error accidents ηislocated bundle and rotated bundle) for Cooper Cycle 5 have been analyzed using the "revised methods" (Reference A-1). These analysis methods have been previously applied to Cooper (References A-2 and A-3) and these applications approved by the NRC (References A-4 and A-5).

The loading errors will not influence the operating CPR limit for 7x7 fuels; however the analysis of a rotated 8DRB28: bundle indicates more severe consequences than any abnormal operational transient. The operating CPR limit for 8x8 and 8x8R bundles will therefore be established by the rotated bundle accident. This limit includes a bias of 0.02 for R factor uncertainties as required by the NRC (Reference A-1). The most severe MLHGR predicted for a misloaded bundle is 16.4 kW/ft. This value includes an allowance of 2.2% for power spiking due to fuel densification (see Appendix C).

REFERENCES

A-1 Letter, D.G. Eisenhut (NRC) to R.E. Engel (GE), transmitting Safety Evaluation Report on "new calculational procedures. . .for the fuel bundle loading error analyses", May 8, 1978.

A-2 Letter, J. Pilant (NPPD) to George Lear (NRC), April 14, 1978.

A-3 Letter, J. Pilant (NPPD) to T.A. Ippolito (NRC), August 16, 1978.

A-4 Letter, George Lear (NRC) to J.M. Pilant (NPPD), May 2, 1978.

A-5 Letter, T.A. Ippolito (NRC) to J. Pilant (NPPD), August 25, 1978.

APPENDIX B
GETAB INITIAL CONDITIONS

Table 5-8 of Reference B-1 states the "Nonvarying Plant GETAB Analysis Initial Conditions". The Cooper core pressure is given as 1045 psia. A value of 1035 psia, which more nearly reflects actual plant data, was assumed for this submittal.

Reference B-1 will be revised to eliminate this discrepancy.

REFERENCES

B-1 Licensing Topical Report, "General Electric Boiling Water Reactor, Generic Reload Fuel Application", NEDE-24011-P-A, May 1977.

APPENDIX C
DENSIFICATION POWER SPIKING

Reference C-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 BWR's for which it can be shown that the predicted worst case maximum transient LHGR's, when augmented by the power spike penalty, do not violate the exposure-dependent safety limit LHGR's".

The Cooper Reload-4 submittal contains the required information to remove the power spiking penalty from the Cooper Technical Specifications. Section 10, Rod Withdrawal Error, and Appendix A (Bundle Loading Accident) include the densification effect in the calculated LHGR of the 8x8 fuels.

REFERENCES

C-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.

GENERAL ELECTRIC