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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR PEACH BOTTOM ATOMIC POWER STATION UNIT 3, RELOAD NO. 3

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GENERAL ELECTRIC

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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR PEACH BOTTOM ATOMIC POWER STATION UNIT 3 RELOAD 3

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1. PLANT-UNIQUE ITEMS (1.0)*

Appendix A - Loading Error Limiting LHGR

Appendix B - Pressurized Test Assembly

Appendix C - Fast Scram Control Rod Drive

Appendix D - New Methods - Fuel Loading Error

Bundle P8DRB284H description is documented in non-approved submittal, Reference 2

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

Fuel Type		Number	Number Drilled
Irradiated	7D250 Type II	52	0
	8DB274L	119	119
	8DB274H	68	68
	PTA	1	1
	8DRB283	252	252
New	P8DRB284H	272	272
Total		764	712

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 8363 MWd/t.

Assumed reload cycle exposure: 17160 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	1.124
Fully Controlled	0.9619
Strongest Control Rod Out	0.9869
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, Δk	0.0000

* ( ) refers to areas of discussion in Reference 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)
660	0.032

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

	EOC4	EOC4-2000 MWd/t*
Void Coefficient N/A** (¢/%Rg)	-8.26/-10.33	-9.44/-11.80
Void Fraction (%)	40.23	40.23
Doppler Coefficient N/A** (¢/°F)	-0.2366/-0.2248	-0.2279/-0.2165
Average Fuel Temperature (°F)	1356	1356
Scram Worth N/A** ($)	-35.48/-28.38	-33.93/-27.14
Scram Reactivity vs Time:	Figure 2a	Figure 2b

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

	7x7		8x8		8x8R		PTA/P8x8R	
	EOC4	EOC4-2000 MWd/t	EOC4	EOC4-2000 MWd/t	EOC4	EOC4-2000 MWd/t	EOC4	EOC4-2000 MWd/t
Peaking factors (local, radial and axial)	1.24	1.24	1.22	1.22	1.20	1.20	1.20	1.20
	1.28	1.30	1.37	1.43	1.49	1.56	1.47	1.56
	1.40	1.40	1.40	1.40	1.40	1.40	1.40	1.40
R-Factor	1.100	1.100	1.098	1.098	1.052	1.052	1.052	1.052
Bundle Power (MWt)	5.416	5.492	5.777	6.026	6.287	6.563	6.209	6.563
Bundle Flow (10^3 lb/hr)	123.4	122.8	111.1	109.4	111.1	109.3	111.8	109.5
Initial MCPR	1.23	1.22	1.29	1.24	1.30	1.24	1.32	1.24

*Mid Cycle Exposure Point

**N = Nuclear Input Data

A = Used in Transient Analysis

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8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Exposure Dependent Limits: From BOC4 to EOC4-2000 MWd/t and from EOC4-2000 MWd/t to EOC4.

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†	PTA & P8x8R	Plant Response
								7x7	8x8/8x8R		
Generator Load Rejection No Bypass	EOC4	101.0	100	287.8	113.5	1203	1230	0.16	0.23	0.25	Figure 3a
	EOC4-2000 MWd/t	104.5	100	159.1	109.4	1189	1217	0.05	0.10	0.10	Figure 3B
Loss of 100°F Feedwater Heating	--	104.5	100	126.4	125.1	1015	1071	0.15	0.17	0.17	Figure 4
Feedwater Controller Failure	BOC4	104.5	100	209.0	118.1	1149	1195	0.12	0.17	0.18	Figure 5a
	EOC4-2000 MWd/t	104.5	100	127.5	112.8	1143	1187	0.06	[illegible]	0.05	Figure 5b

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

Rod Block Set Point	Rod Position (Feet Withdrawn)	ΔCPR*			MLHGR (KW/ft)****			Limiting Rod Pattern
		7x7**	8x8	8x8R/ P8x8R/PTA	7x7**	8x8	8x8R/ P8x8R/PTA	
105	4.5	-	0.10	0.14	-	12.13	14.37	Figure 6
106	5.0	-	0.10	0.16	-	12.09	14.41	Figure 6
107***	6.0	-	0.11	0.20	-	12.01	14.53	Figure 6
108	8.0	-	0.14	0.25	-	11.70	16.89	Figure 6
109	9.0	-	0.15	0.27	-	12.06	17.99	Figure 6
110	12.0	-	0.16	0.33	-	12.56	18.09	Figure 6

*Based on an initial MCPR of 1.43 (8x8) and 1.34 (8x8R and P8x8R)

**7x7 fuel is located only on the core periphery and is not limiting; therefore its response is not given.

***Indicates setpoint selected.

****Includes the effects of densification power spiking.

†All ΔCPR values calculated from initial power of 104.5%.

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11. OPERATING MCPR LIMIT (5.2)

BOC4 to EOC4-2000 MWd/t	EOC4-2000 MWd/t to EOC4
1.23 (7x7 fuel)	1.23 (7x7 fuel)
1.24 (8x8 fuel)	1.30 (8x8 fuel)
1.27 (8x8R fuel)	1.30 (8x8R fuel)
1.27 (PTA/P8x8R fuel)	1.32 (PTA/P8x8R fuel)

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

	Power (%)	Core Flow (%)	P_{SL} (psig)	P_V (psig)	Plant Response
MSIV Closure (Flux Scram)	104.5	100	1271	1301	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio, x_2/x_o: (105% Rod Line - Natural Circulation Power)	0.90

Channel Hydrodynamic Performance	Decay Ratio, x_2/x_o (105% Rod Line - Natural Circulation Power)
8x8R/P8x8R/PTA Channel	0.29
8x8 Channel	0.40
7x7 Channel	<0.01

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

Exposure (MWd/t)	MAPLHGR (kW/ft) P8DRB284H	PCT (°F) P8DRB284H	Local Oxidation Fraction P8DRB284H
200	11.3	1812	0.007
1000	11.3	1813	0.007
5000	11.7	1858	0.008
10000	12.1	1894	0.009
15000	12.0	1897	0.009
20000	11.6	1858	0.008
25000	10.9	1777	0.006
30000	10.2	1700	0.004

15. LOADING ERROR RESULTS (5.5.4)

Limiting Events: Mislocated Bundle P8DRB284H MCPR $\geq$1.07
Rotated Bundle P8DRB284H MCPR $\geq$1.07

16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Doppler Reactivity Coefficient: Figure 9

Accident Reactivity Shape Function: Figures 10 and 11

Scram Reactivity Functions: Figures 12 and 13

Plant Specific Analysis Results

Parameters Not Bounded:

Scram Reactivity Functions: Cold and Hot Startup

Resultant Peak Enthalpies (cal/g):

Cold	Hot Startup
174	175

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17. REFERENCES

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

2. "General Electric Boiling Water Reactor Generic Reload Fuel Application," May 1979, (NEDE-24011-P-A, Amendment 3).

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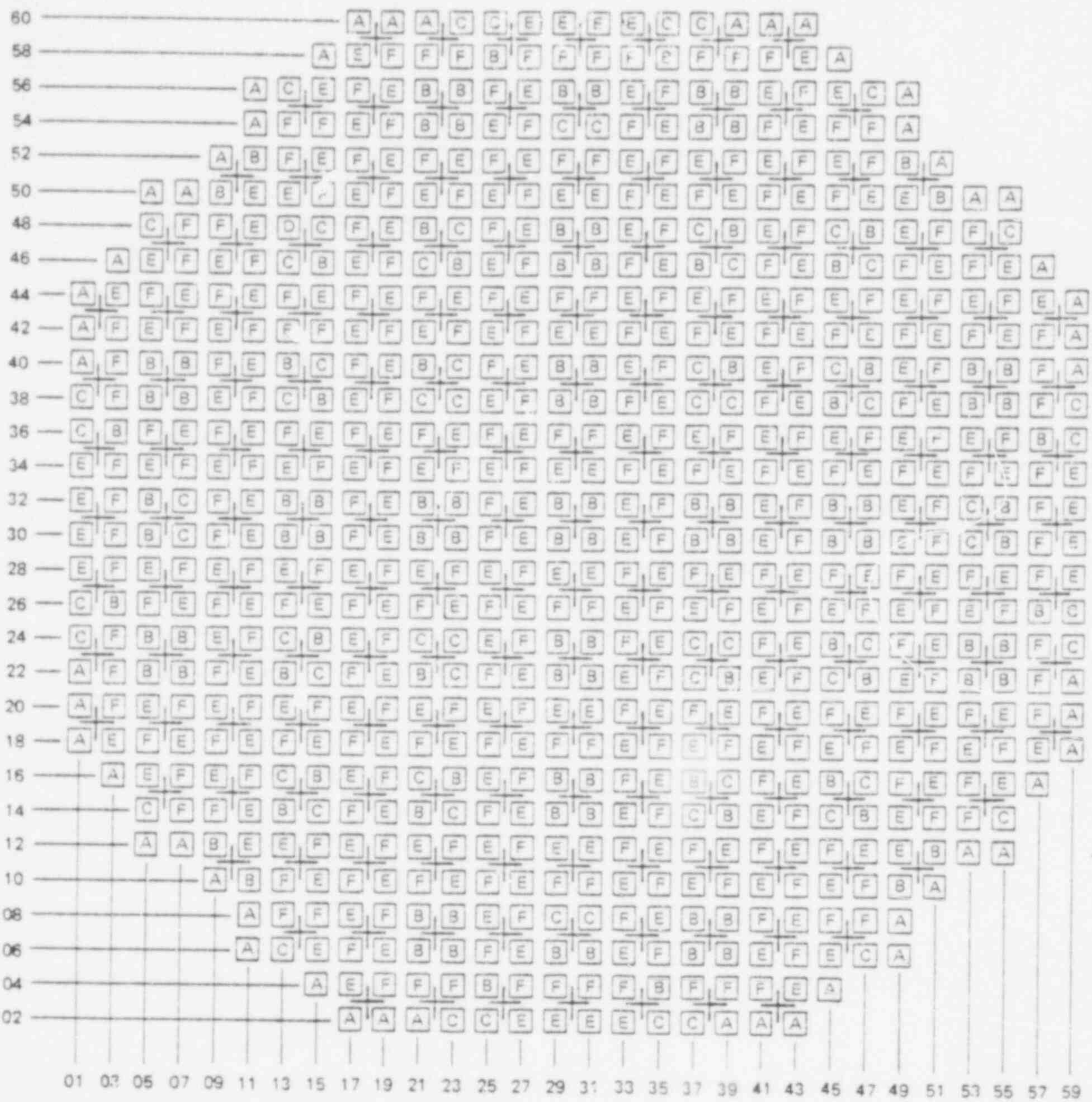

FUEL TYPE	
A = 7D250	E = 8DRB283
B = 8D8274L	F = P8DRB284H
C = 8D8274H	
D = PTA	

Figure 1. Reference Core Loading Pattern

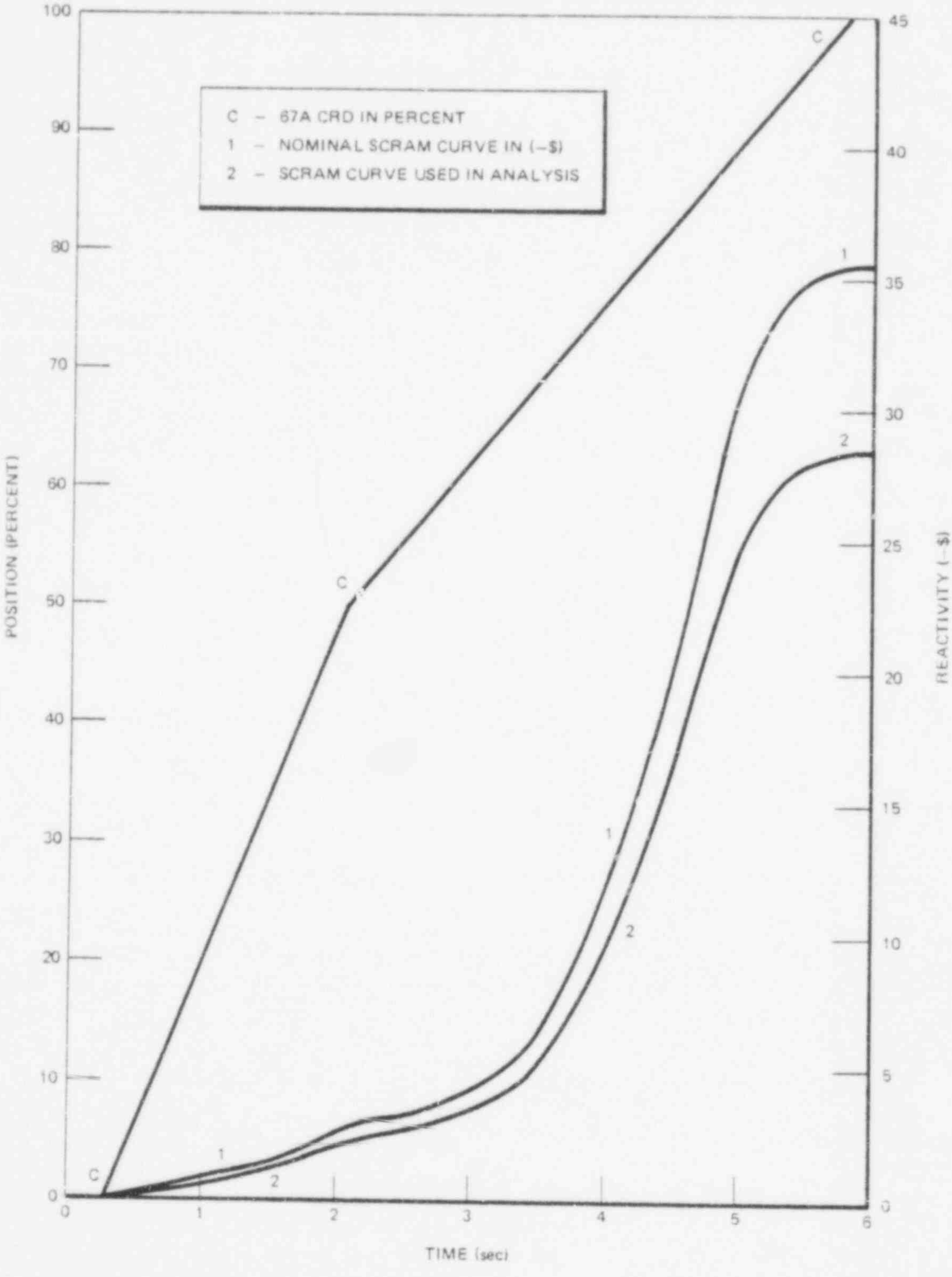

Figure 2a. Scram Reactivity and Control Rod Drive Specifications from EOC4-2000 MWd/t to EOC4

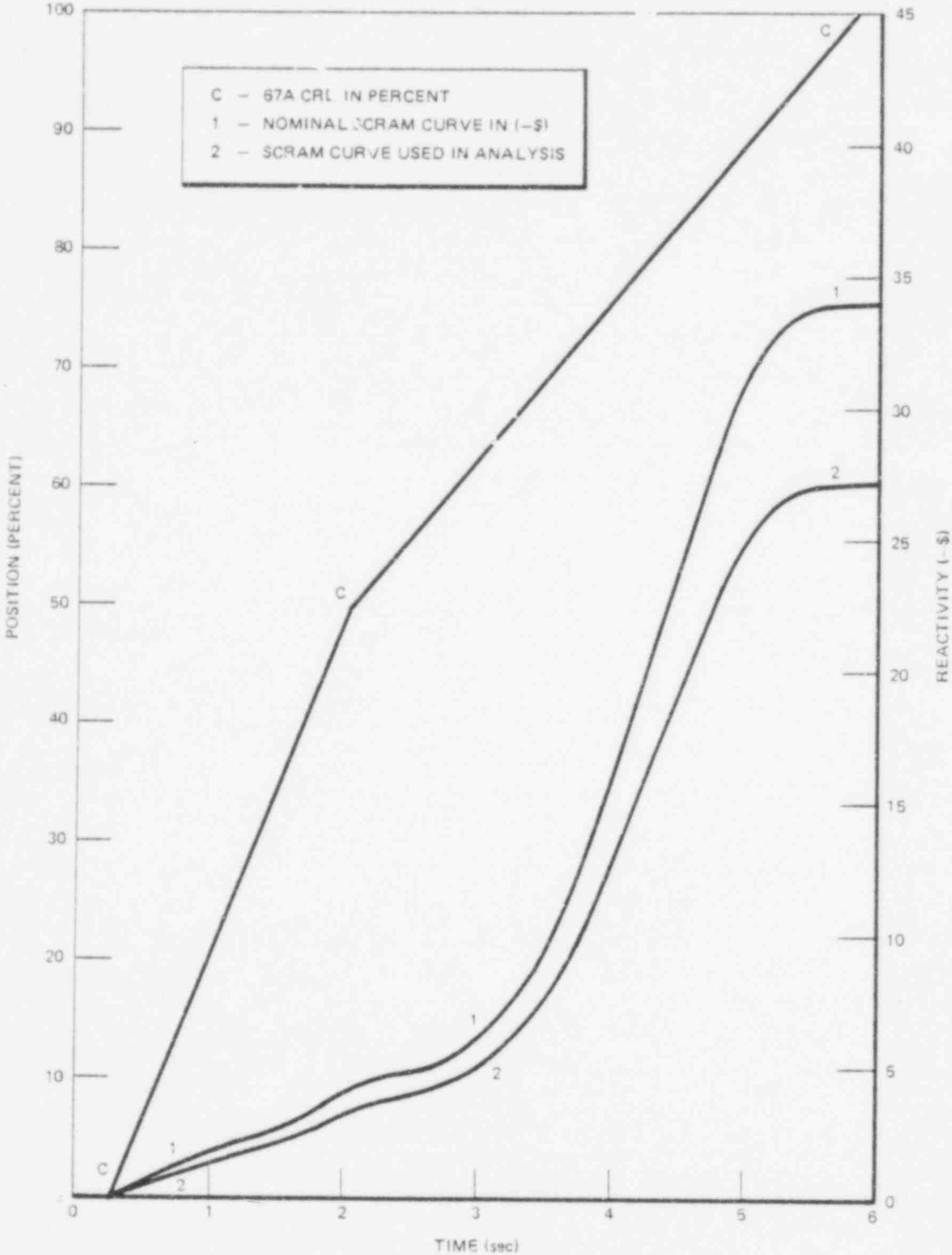

Figure 2b. Scram Reactivity and Control Rod Drive
Specifications from BOC4 to EOC4-2000 MWd/t

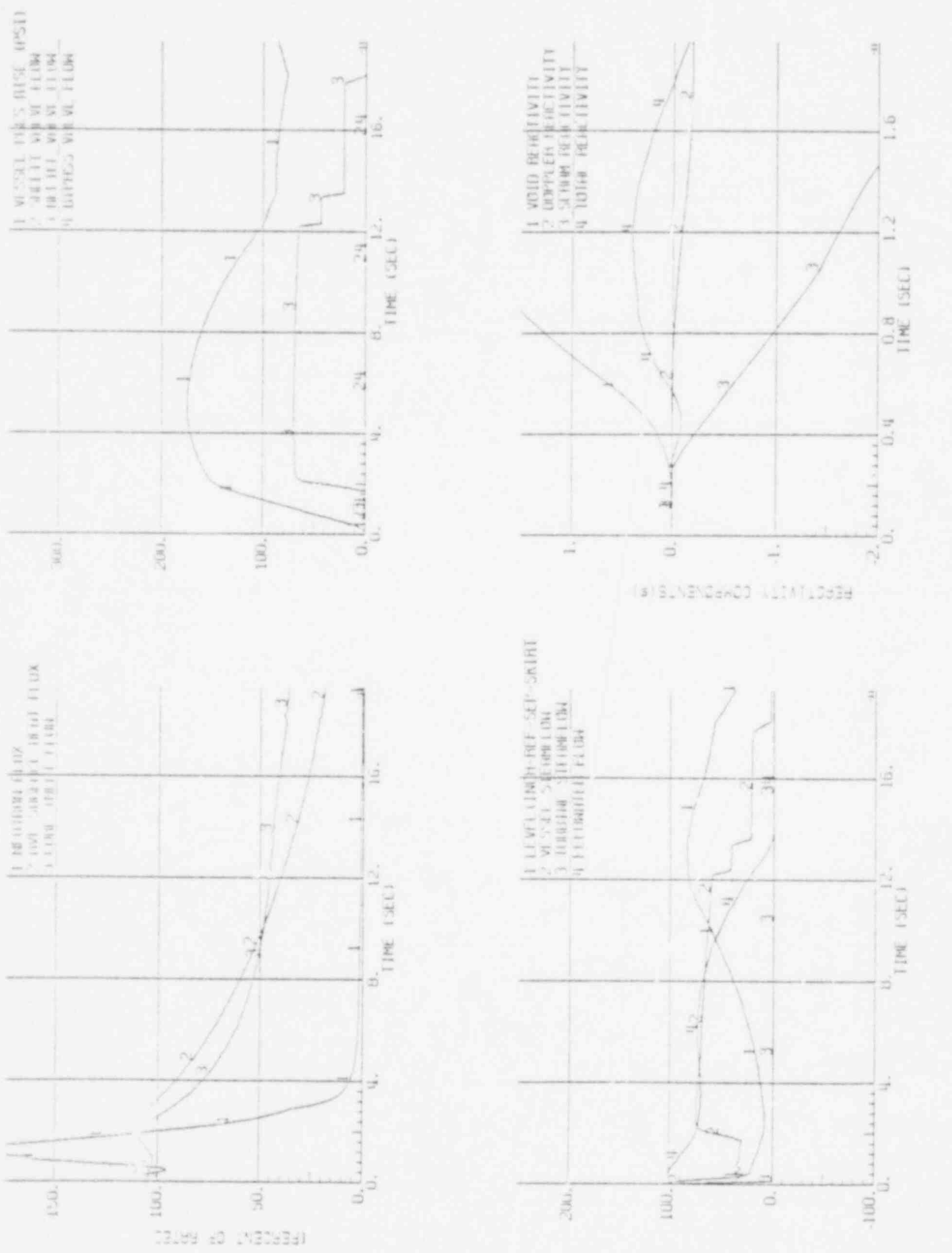

Figure 3a. Generator Load Rejection No Bypass, EOC4

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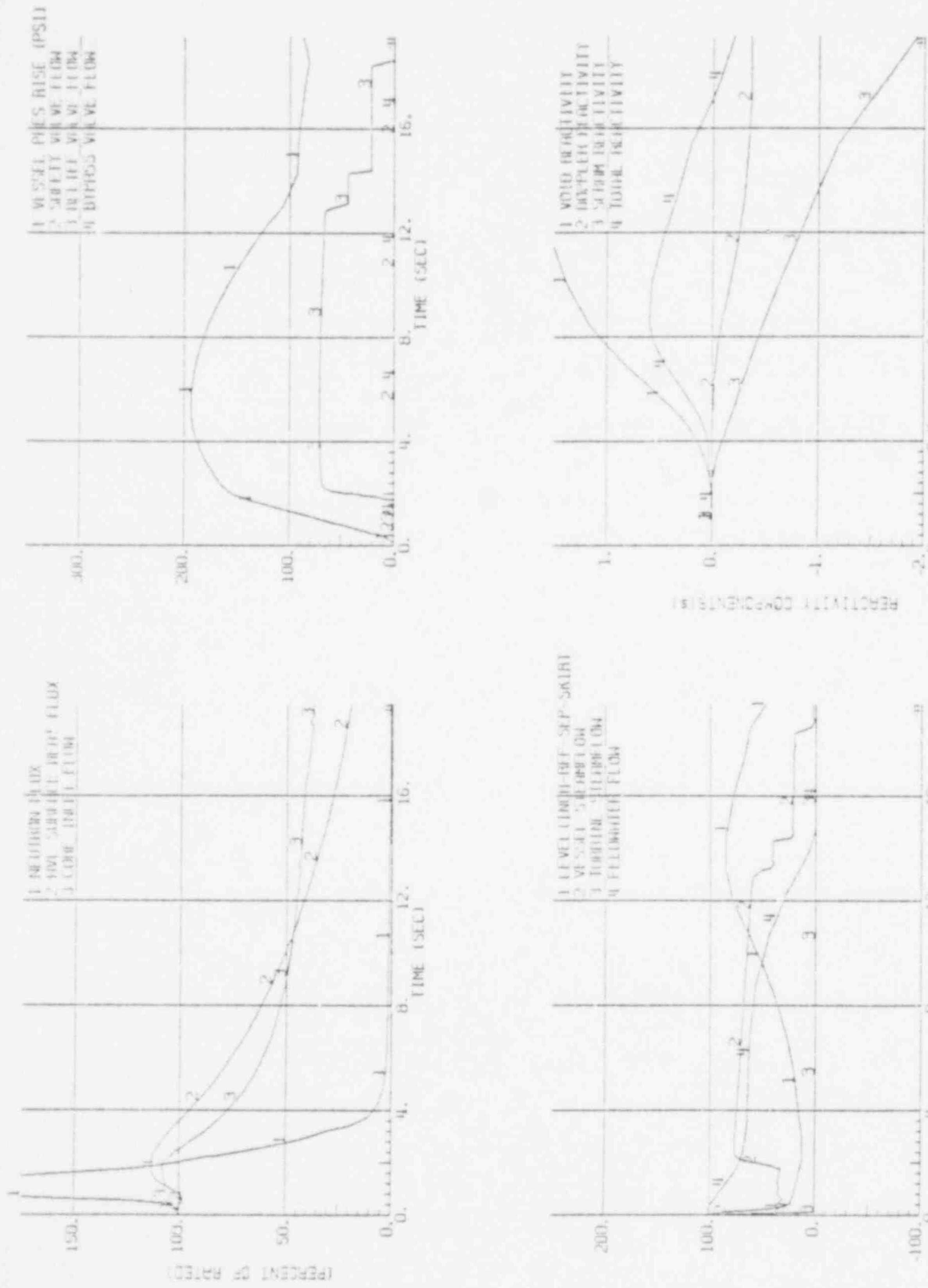

Figure 3b. Generator Load Rejection No Bypass, EOC4-2000 MWd/t

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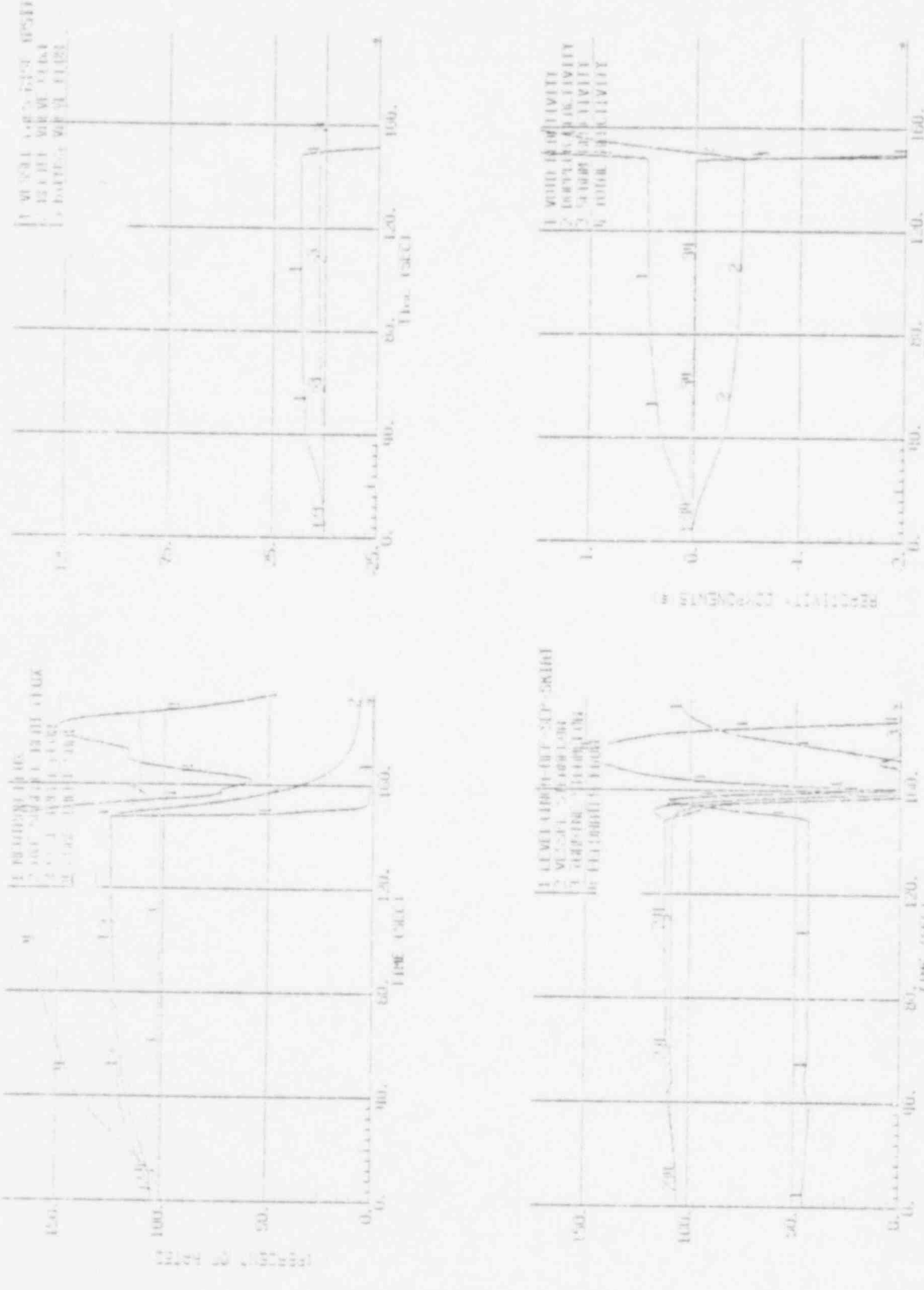

Figure 4. Loss of 100°F Feedwater Heating

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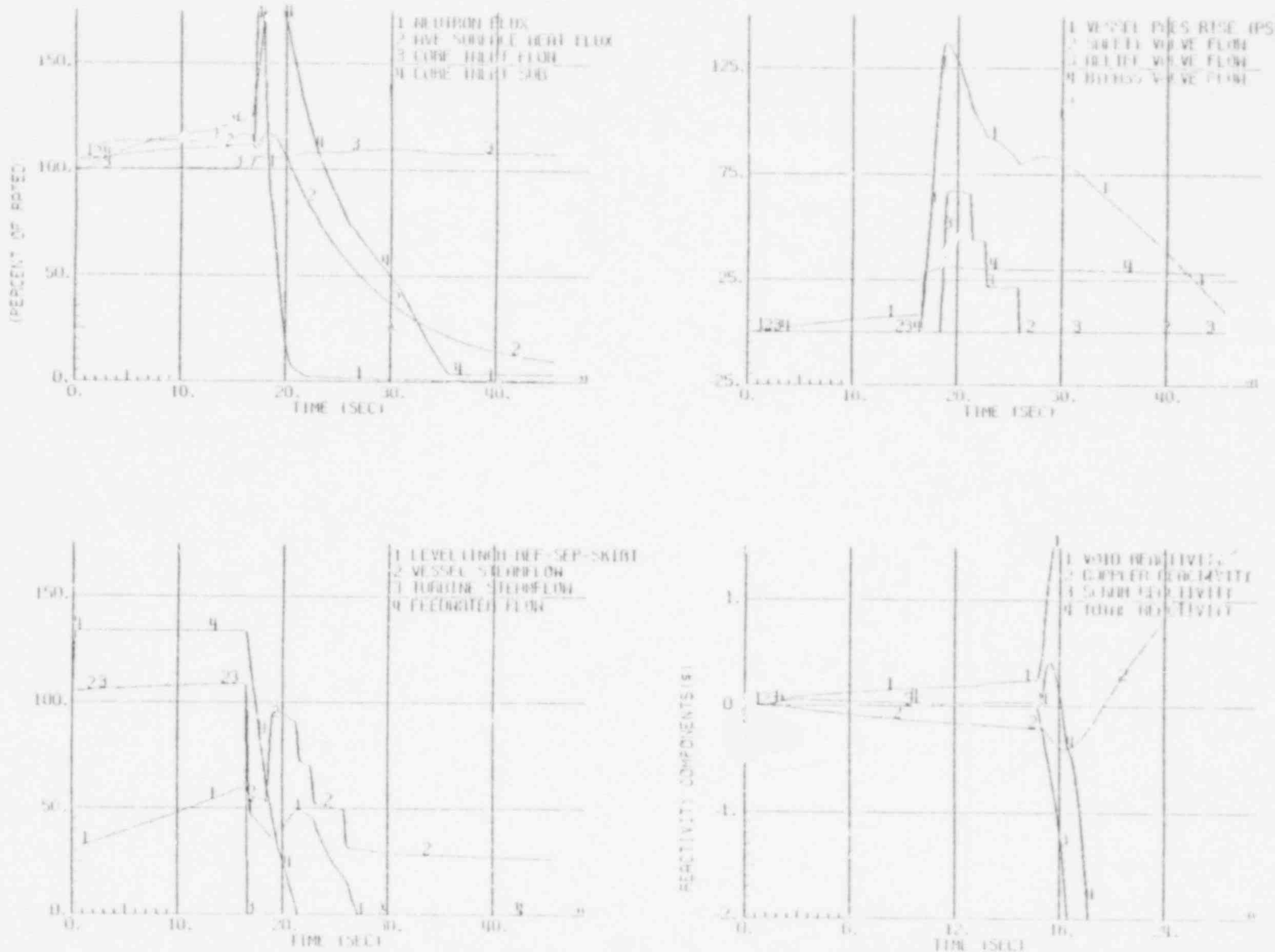

Figure 5a. Feedwater Controller Failure, Maximum Demand, EC04

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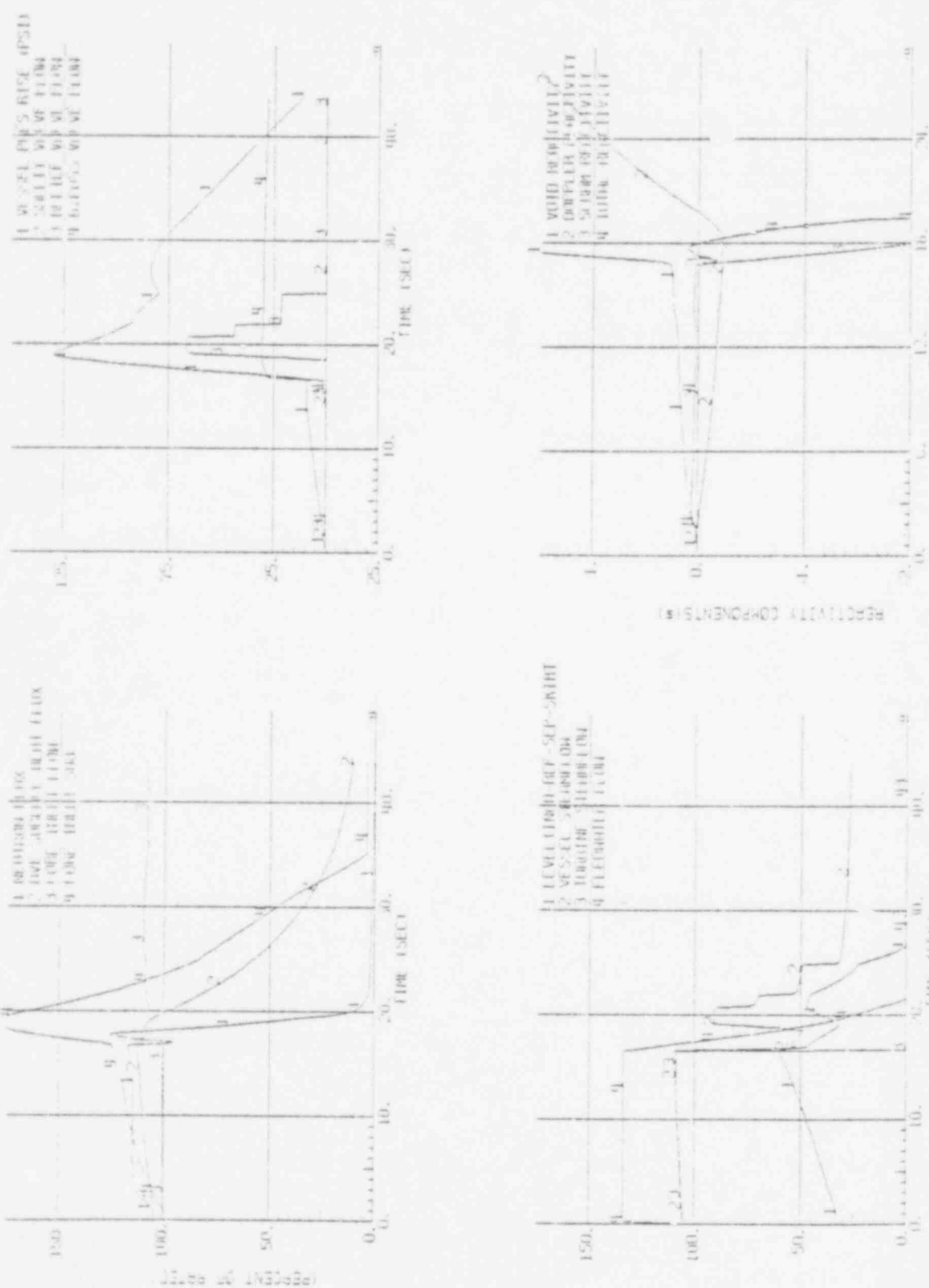

Figure 5b. Feedwater Controller Failure, Maximum Demand, EC04-2000 MWd/t

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	02	06	10	14	18	22	26	30
59					26		10	
55				10		10		8
51			26		20		8	
47		10		10		30		30
43	26		20		8		8	
39		10		30		42		42
35	10		8		8		0	
31		8		30		42		42

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 ROD (26,35).

Figure 6. Limiting RWE Rod Pattern

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Figure 7. MSIV Closure, Flux Scram

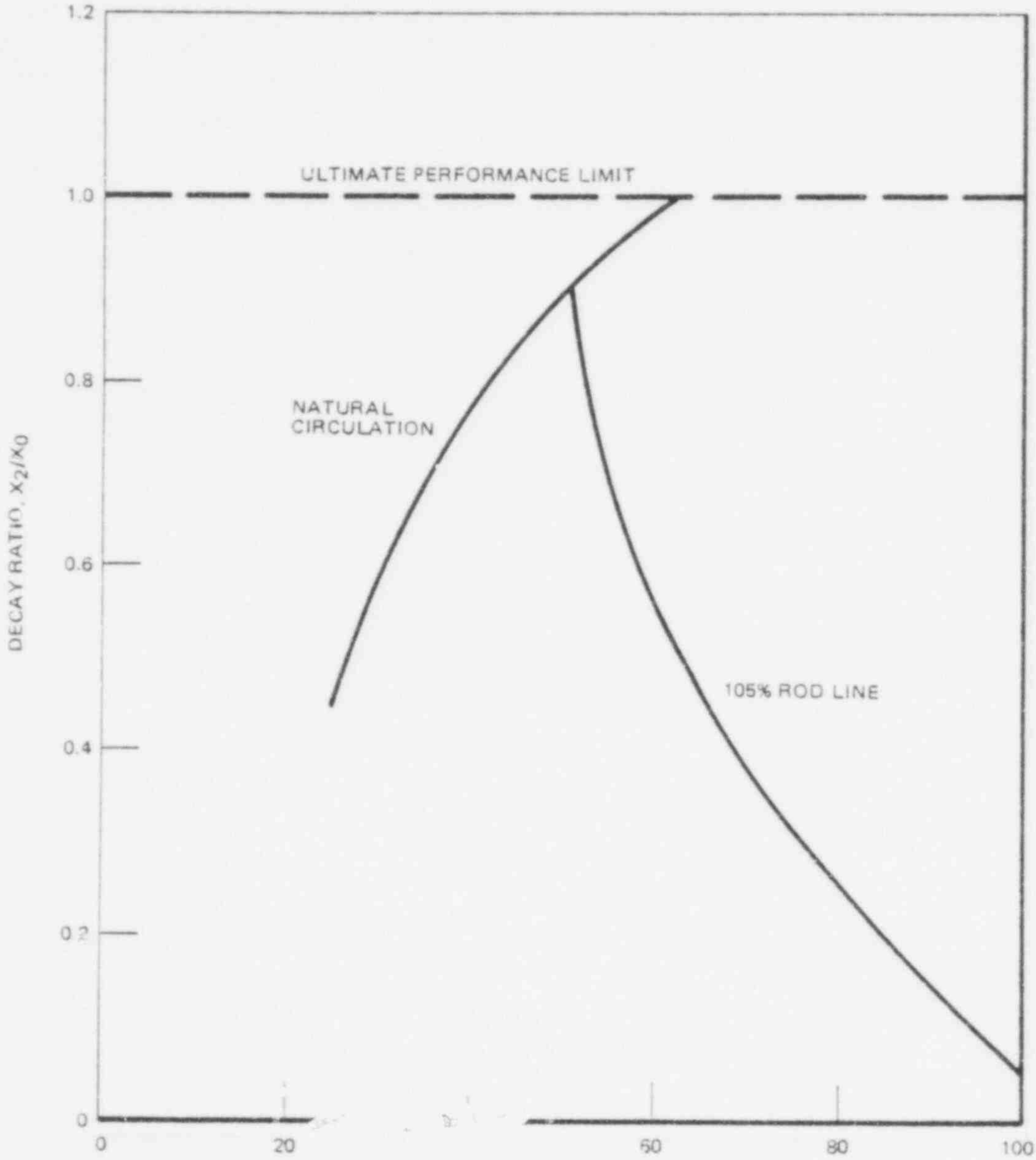

Figure 8. Decay Ratio

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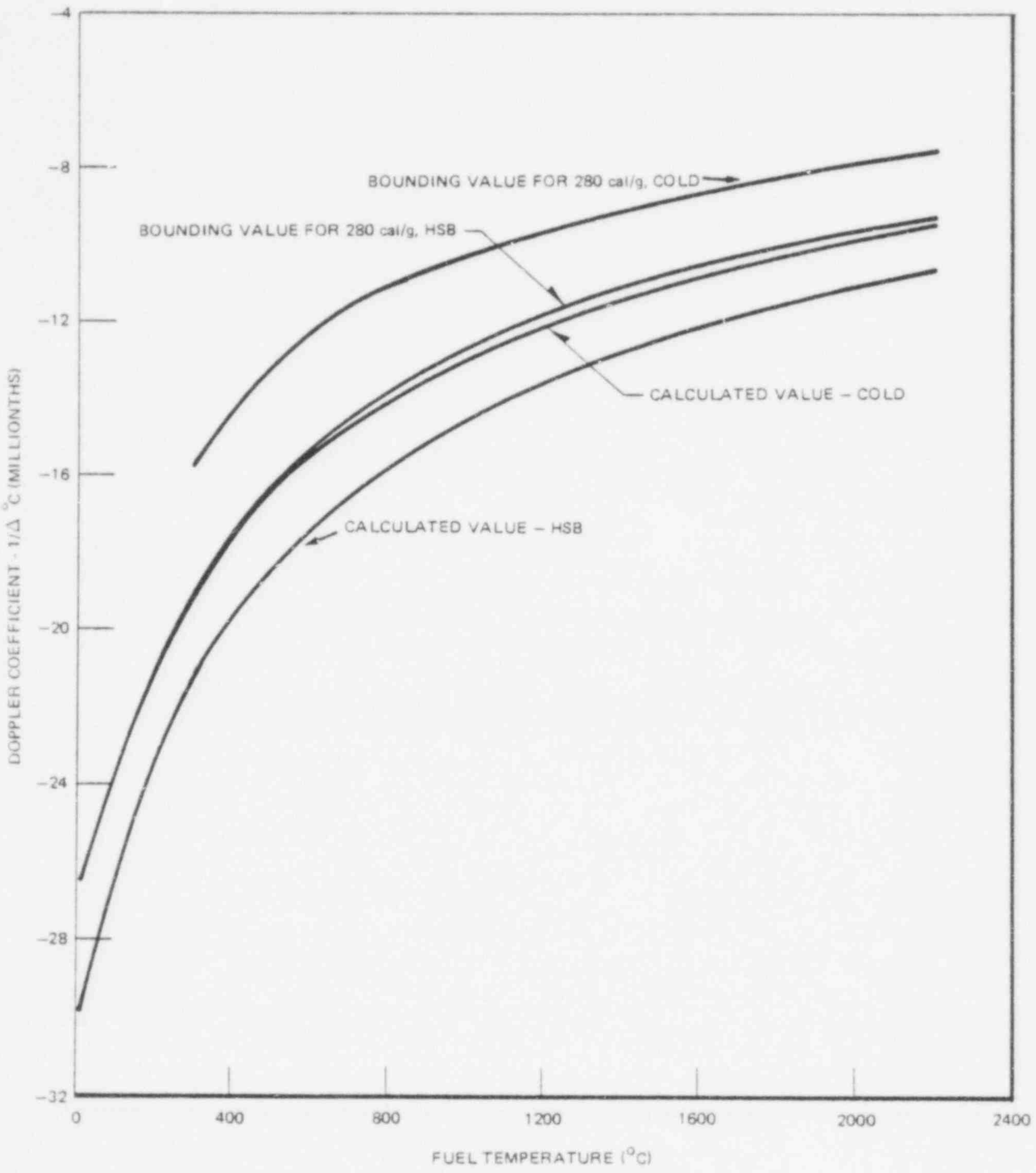

Figure 9. Doppler Reactivity Coefficient Comparison for RDA

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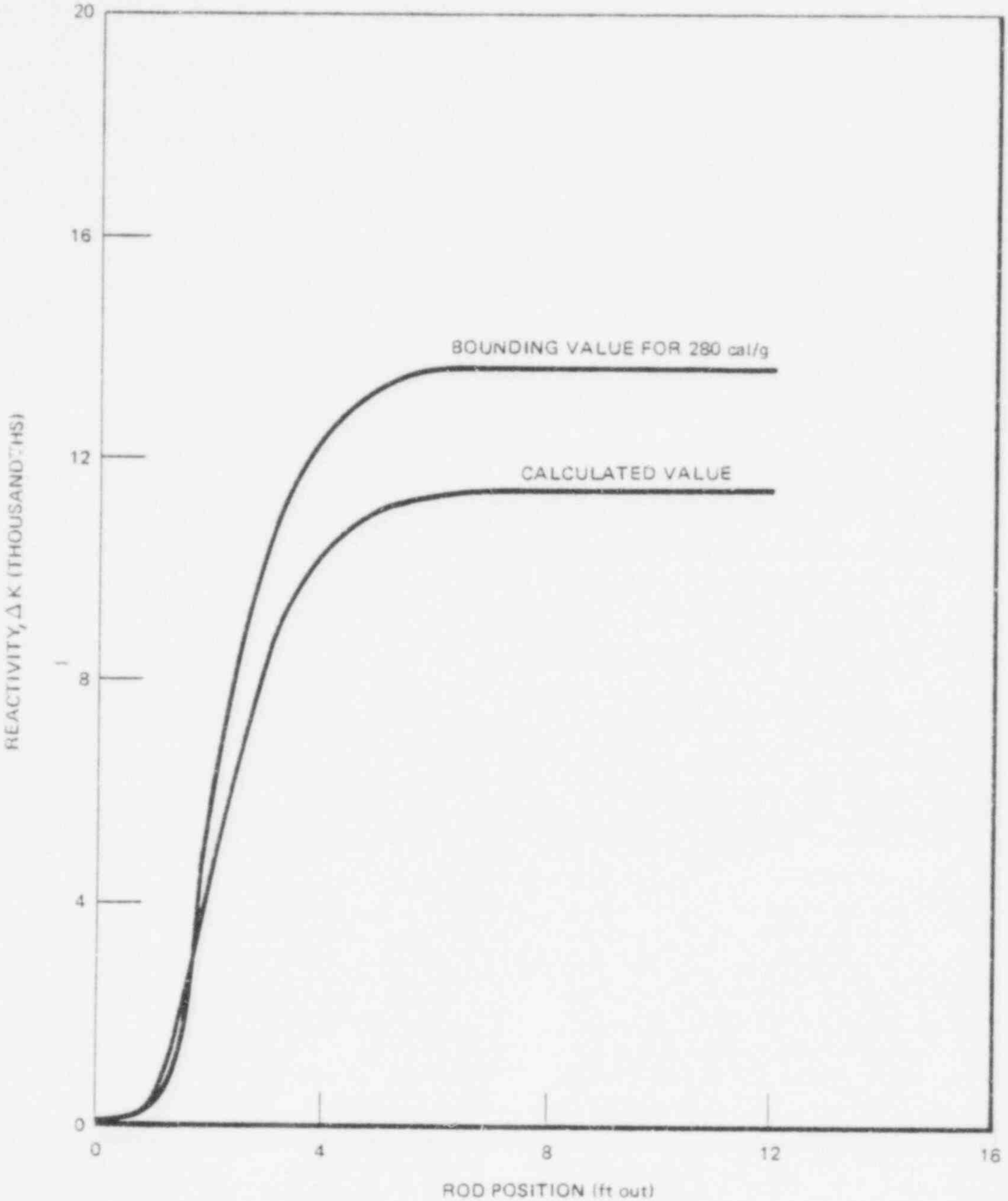

Figure 10. Accident Reactivity Shape Function at 20°C

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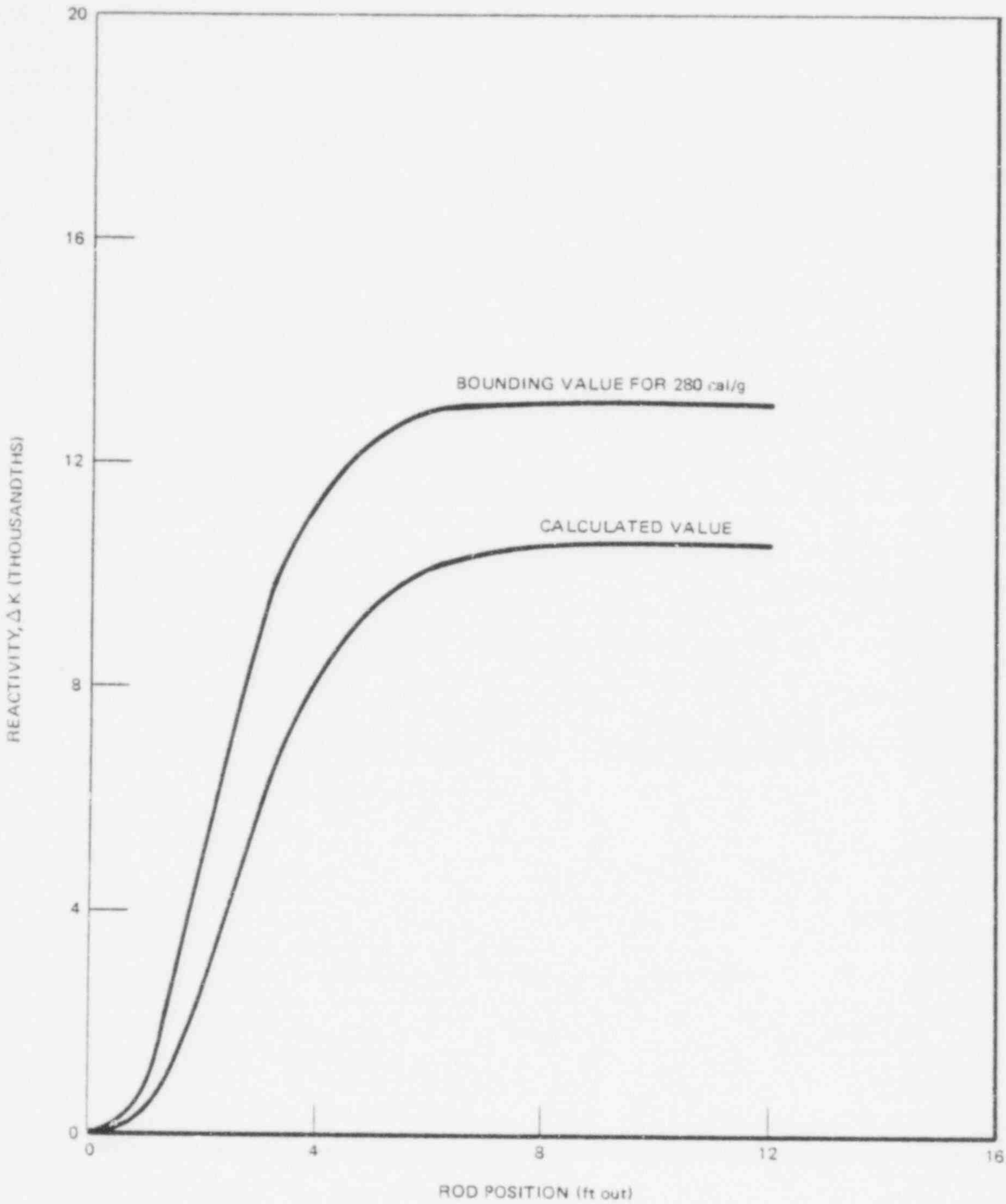

Figure 11. Accident Reactivity Shape Function at 286°C

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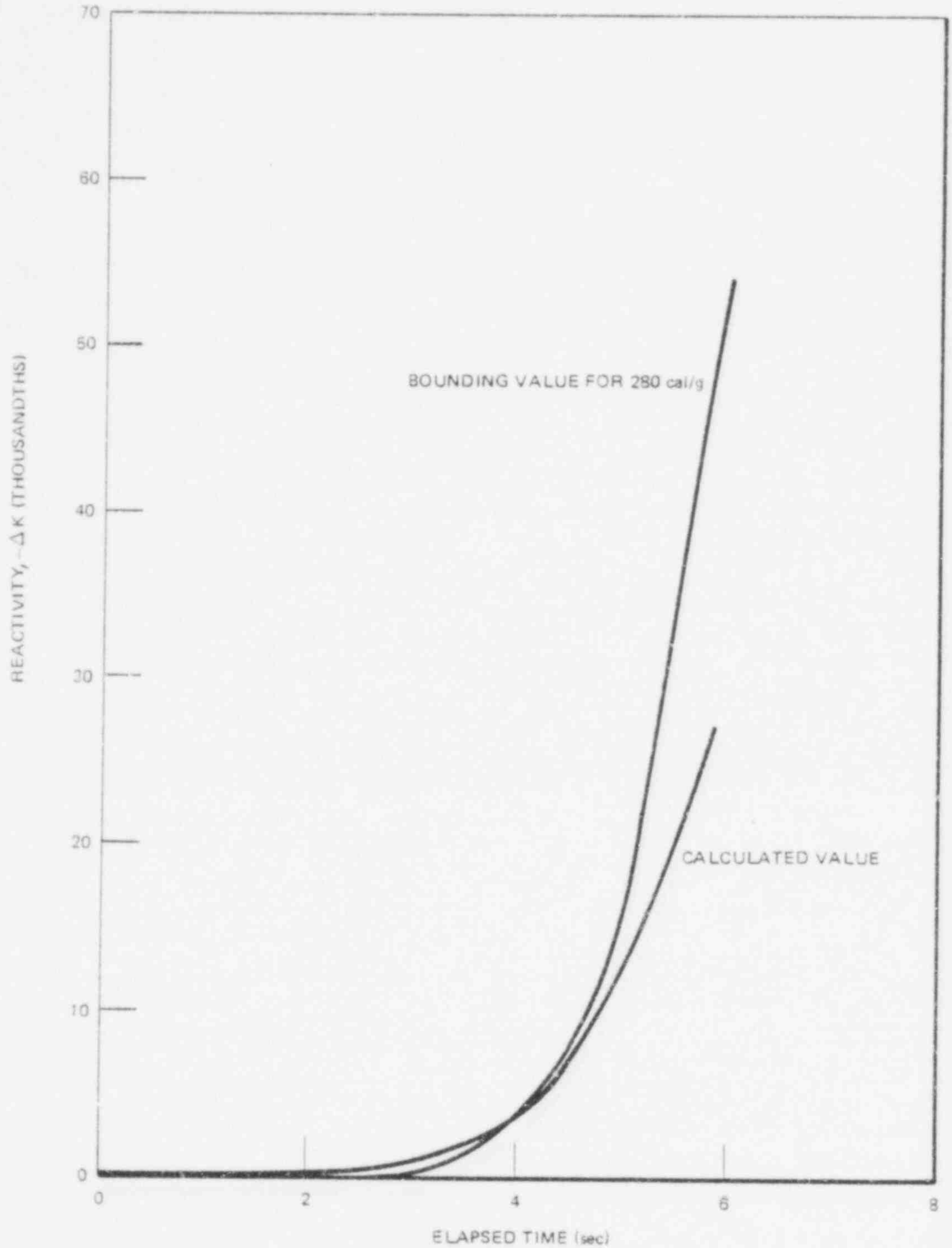

Figure 12. Scram Reactivity Function at 20°C

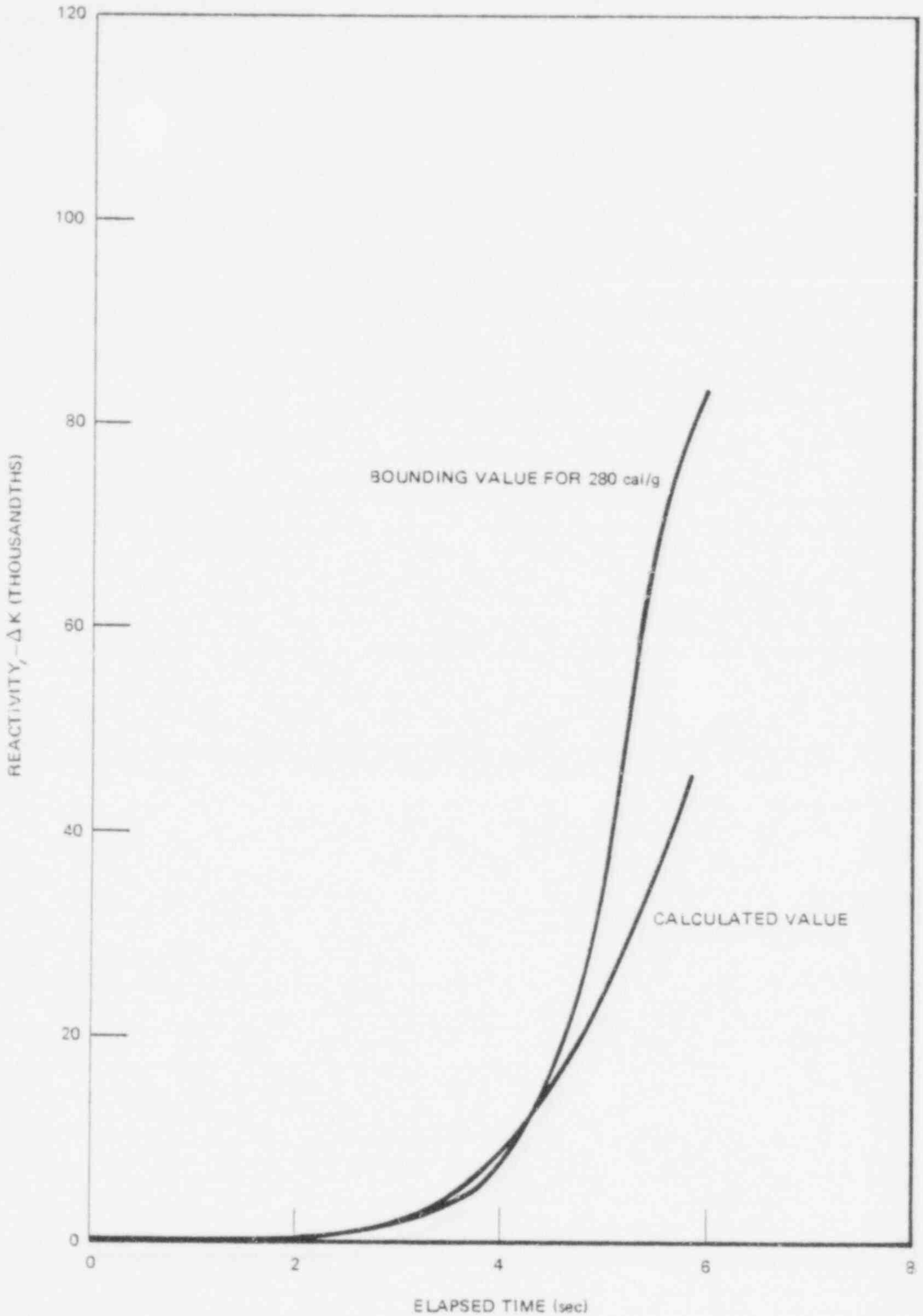

Figure 13. Scram Reactivity Function at 286°C

APPENDIX A
LOADING ERROR LIMITING LHGR

This appendix provides the limiting linear heat generation rate (LHGR) resulting from the bundle loading error (BLE) analysis.

Limiting Event	LHGR (kW/ft)*
Misplaced Bundle P8DRB284H	18.4

*Limiting bundle results including the effects of densification power spiking (factor = 1.022).

APPENDIX B
PRESSURIZED TEST ASSEMBLY

The pressurized test assembly (PTA) was described in NEDO-21363-1 Supplement 1, dated November 1976 as amended by NEDO-21363-4 Supplement 4, dated January 1977. In addition to describing the PTA, these licensing documents provided a safety analysis for installation of the PTA in the Peach Bottom 3 reactor for cycle 2 and subsequent cycles. It is planned that the PTA will remain in the core during cycle 4. The safety analysis performed for cycle 4 includes consideration of the PTA as part of the reloaded core. Although the GETAB analysis was performed for the PTA as a separate fuel type, its effect on the remaining safety analysis is insignificant since the PTA configuration is basically the same as the reload 3 8x8R bundle.

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APPENDIX C
FAST SCRAM CONTROL ROD DRIVE

The fast scram control rod drive (FSCRD) was described in NEDO-21363-2 Supplement 2. In addition to describing the FSCRD, the licensing document provided the results of a safety review and evaluation which considered any effects the presence of the FSCRD would have on the plant safety analysis. It was determined that the inclusion of the FSCRD did not introduce an unreviewed safety question and had no effect on parameters used in the plant safety analysis. For cycle 4 the FSCRD may be left installed for another cycle of operation.

NEDO-21363-2A, "General Electric Boiling Water Reactor Reload 1 Licensing Amendment for Peach Bottom Atomic Power Station Unit 3 Fast Scram Control Rod Drive, Second Supplement," July 1979, provides results of evaluation of the FSCRD which was operated in Peach Bottom Unit 3 during cycle 2 and subsequently disassembled and inspected. The report provides performance results and a report of the effects of the reactor environment on the drive mechanism. A safety evaluation is also provided which demonstrates that continued operation of the currently installed FSCRD, during cycle 4, does not introduce an unreviewed safety question and has no adverse effect on parameters used in the plant safety analysis.

APPENDIX D
NEW BUNDLE LOADING ERROR EVENT ANALYSES PROCEDURES

The bundle loading error analyses results presented in Section 15 in the supplement 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.

D.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 D-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.

The number presented in Section 15 represents the minimum CPR of the most limiting rotated bundle starting from an initial CPR of 1.22 which includes the 2% allowance for uncertainties as required by the NRC.

D.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 D-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 the mislocated bundle in the core is greater than the safety limit.

REFERENCES

D-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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