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Revision 1

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BRUNSWICK STEAM ELECTRIC PLANT UNIT 2 RELOAD 2 (Recirculation Pump Trip Feature)

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GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

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1. PLANT-UNIQUE ITEMS (1.0)*

Rotated Bundle Analysis Procedure: Appendix A

Total Number and Capacity of Safety/Relief Valves: Reference 2

Fuel Loading Error LHGR: Appendix B

ODYN Transient Calculation Results: Appendix C

Recirculation Pump Trip Feature: Appendix D

Separate MCPR Limits Reported for 8 x 8 and 8 x 8R Fuels

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

	Fuel Type	Number	Number Drilled
Irradiated	Initial Core Type 1	108	108
	Initial Core Type 3	176	176
	7DB230	4	4
	8DB274L	100	100
	8DB274H	40	40
New	8DRB265H	64	64
	8DRB283	68	68
Total		560	560

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 11,570 MWd/t

Assumed reload cycle exposure: 13,080 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	1.120
Fully Controlled	0.958
Strongest Control Rod Out	0.989

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

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

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

	EOC
Void Coefficient N/A* (¢/% Rg)	7.46/9.49
Void Fraction (%)	41.76
Doppler Coefficient N/A (¢/% °F)	0.1937/0.1840
Average Fuel Temperature (°F)	1538
Scram Worth N/A ($)	38.75/31.00
Scram Reactivity	Figure 2

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

Exposure	7x7 EOC	8x8 EOC	8x8R EOC
Peaking factors (local, radial and axial)	1.24/1.284/1.40	1.22/1.448/1.40	1.20/1.585/1.40
R-Factor	1.100	1.098	1.051
Bundle Power (MWt)	5.481	6.175	6.753
Bundle Flow (10^3 lb/hr)	124.5	113.0	114.0
Initial MCPR	1.20	1.20	1.20

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Recirculation Pump Trip

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

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Flow (%)	φ (%)	Q/A (%)	P_{sl} (psig)	P_v (psig)	ΔCPR 7x7	ΔCPR 8x8/8x8R	Plant Response
Turbine Trip without Bypass	EOC3	104	100	133.0	100	1166	1197	0.01	0.03	Figure 3
Inadvertent HPCI Pump Start	----	104	100	121.2	112.9	1018	1067	0.11	0.13/0.14	Figure 4
Feedwater Controller Failure	EOC3	104	100	109.3	104.8	1028	1073	0.05	0.06	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	MLHGR (kW/ft) 7x7	MLHGR (kW/ft) 8x8	MLHGR (kW/ft) 8x8R	Limiting Rod Pattern
104	4.0	0.13	0.10	0.19	18.0	15.3	12.5	Figure 6
105*	4.0	0.13	0.10	0.19	18.0	15.3	12.5	Figure 6
106	4.5	0.15	0.12	0.22	18.8	16.3	13.1	Figure 6
107	5.0	0.17	0.13	0.25	19.4	16.8	13.6	Figure 6
108	5.5	0.20	0.14	0.27	19.8	17.3	14.1	Figure 6
109	6.0	0.22	0.16	0.29	20.0	17.5	14.4	Figure 6
110	9.0	0.24	0.24	0.36	18.2	16.5	14.3	Figure 6

11. OPERATING MCPR LIMIT (5.2)

BOC3 - EOC3	
1.20	(8x8 fuel)
1.26	(8x8R fuel)
1.20	(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	1213	1258	Figure 7

*Indicates setpoint selected

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power)	0.62

Channel Hydrodynamic Performance

	Decay Ratio, x_2/x_0 (105% Rod Line - Natural Circulation Power)
8x8/8x8R channel	0.28
7x7 channel	0.13

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

8DRB265

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Local Oxidation Fraction
200	11.5	2154	0.030
1,000	11.6	2156	0.029
5,000	11.9	2192	0.032
10,000	12.0	2196	0.032
15,000	12.0	2200	0.033
20,000	11.8	2197	0.033
25,000	11.3	2138	0.027
30,000	10.7	2056	0.021

8DRB283

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Local Oxidation Fraction
200	11.2	2122	0.027
1,000	11.2	2117	0.026
5,000	11.8	2184	0.032
10,000	12.0	2197	0.033
15,000	11.9	2194	0.032
20,000	11.8	2197	0.033
25,000	11.3	2132	0.027
30,000	11.1	2106	0.025

15. LOADING ERROR RESULTS* (5.5.4, Appendix A)

Limiting Event: Rotated bundle 8DRB283H or 8DRB265H

MCPR: 1.07**

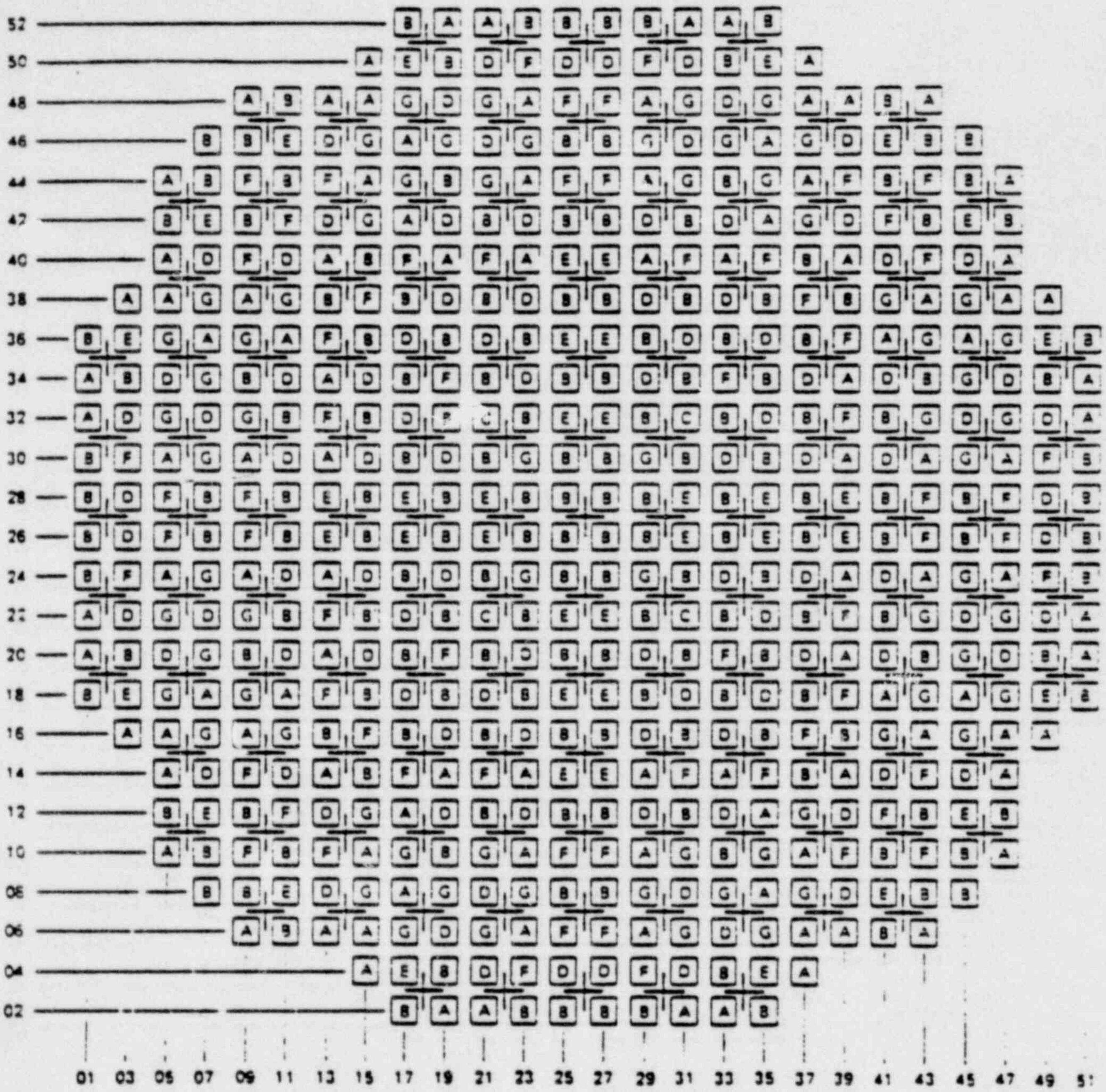
16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Doppler Reactivity Coefficient: Figure 9

Accident Reactivity Shape Functions: Figures 10 and 11

Scram Reactivity Functions: Figures 12 and 13

*Using New Rotated Bundle Analysis Procedures described in Appendix A.
**Includes added penalty of 0.02 imposed by NRC.



Fuel Type	
A=Init Core Type 1	E=8DB274H
B=Init Core Type 3	F=8DRB265H
C=Gen B (7DB230)	G=8DRB283
D=8DB274L	

Figure 1. Reference Core Loading Pattern

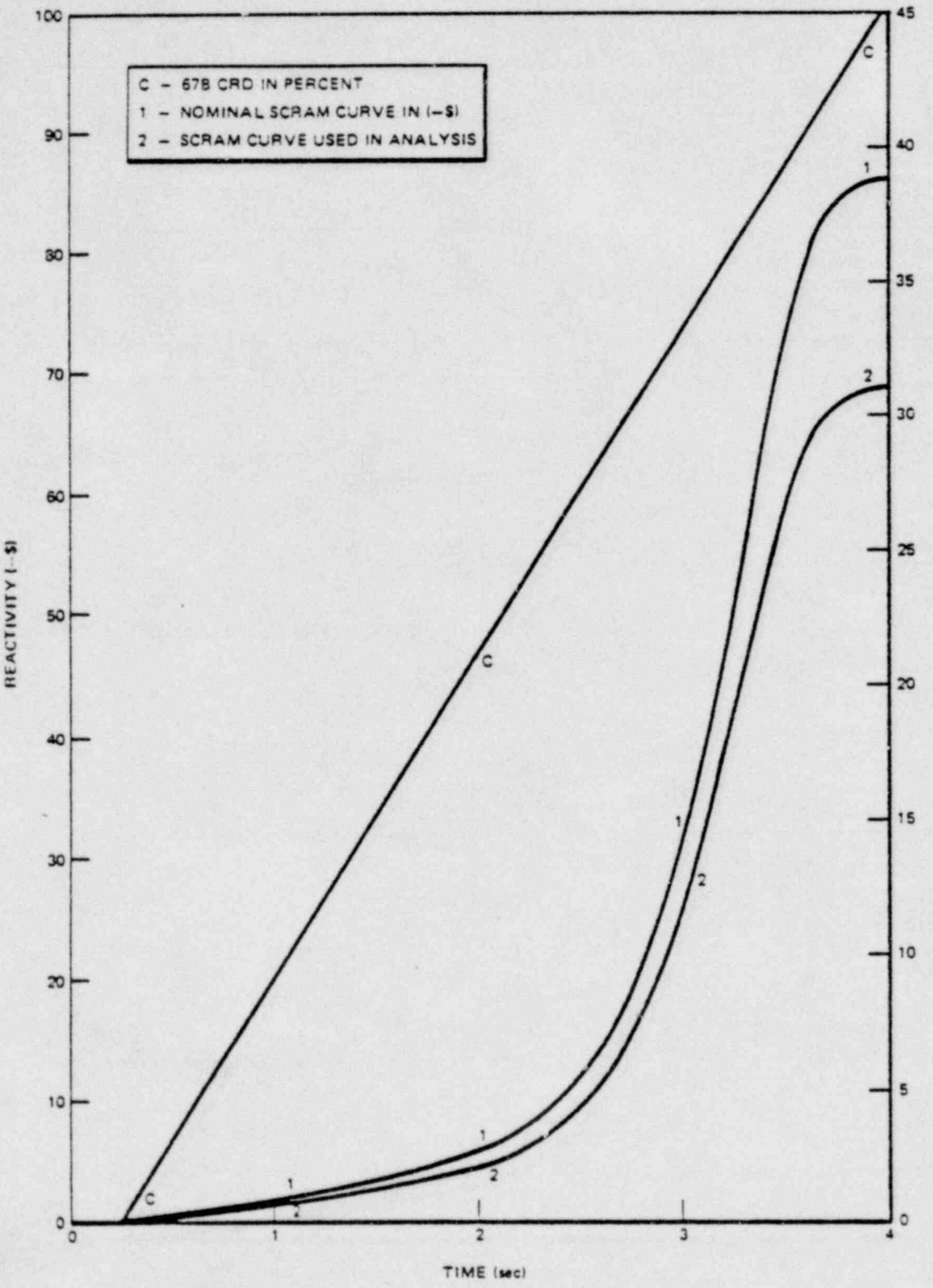


Figure 2. Scram Reactivity and Control Rod Drive Specifications

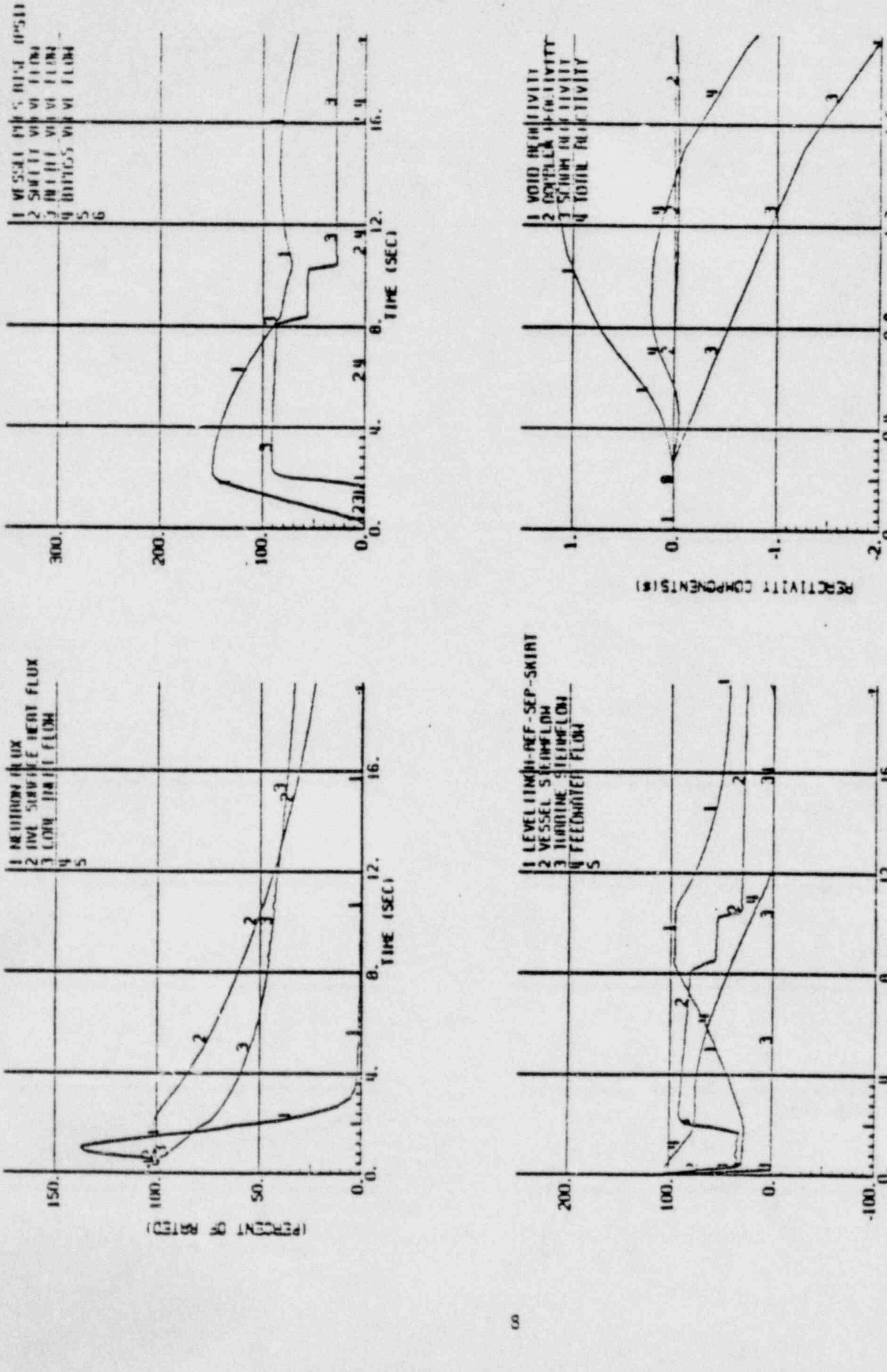


Figure 3. Plant Response to Turbine Trip Without Bypass

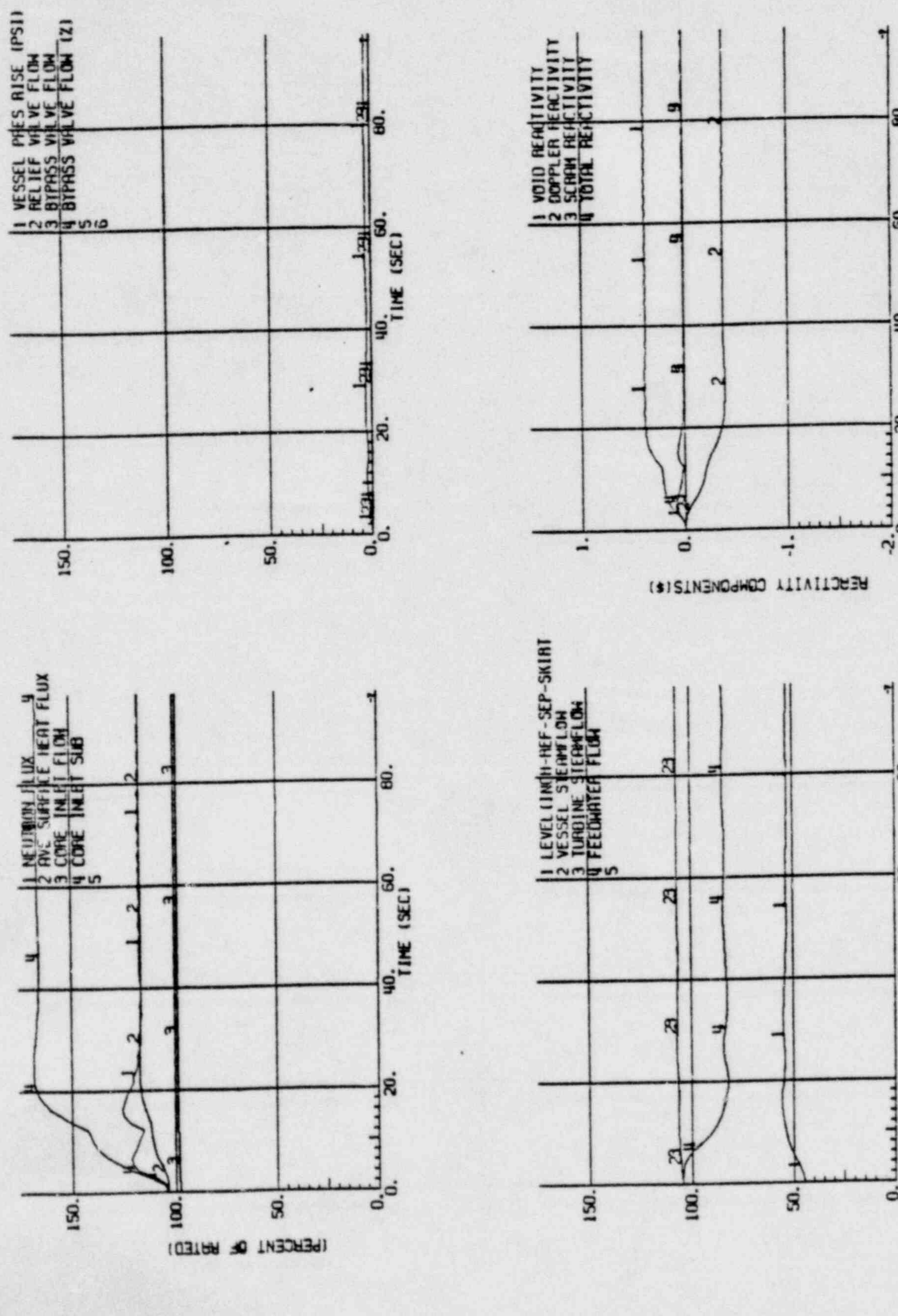


Figure 4. Inadvertent HPCI Pump Start

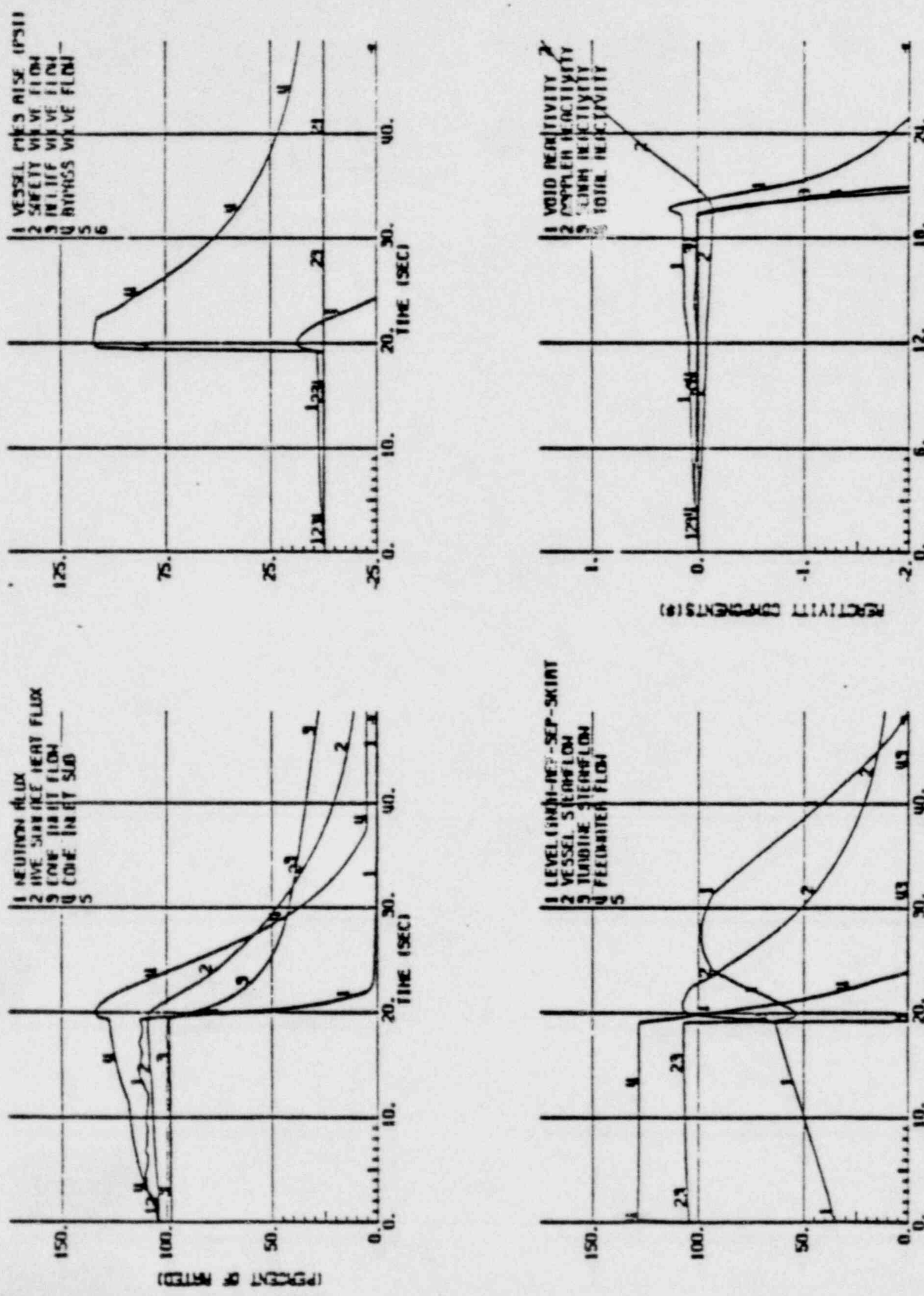


Figure 5. Plant Response to Feedwater Controller Failure

	02	06	10	14	18	22	26	30
51								
47								
43				34		6		
39							32	
35		34		4		2		
31					28			
27		6		2		0		
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 (22,27)

Figure 6. Limiting RWE Rod Pattern

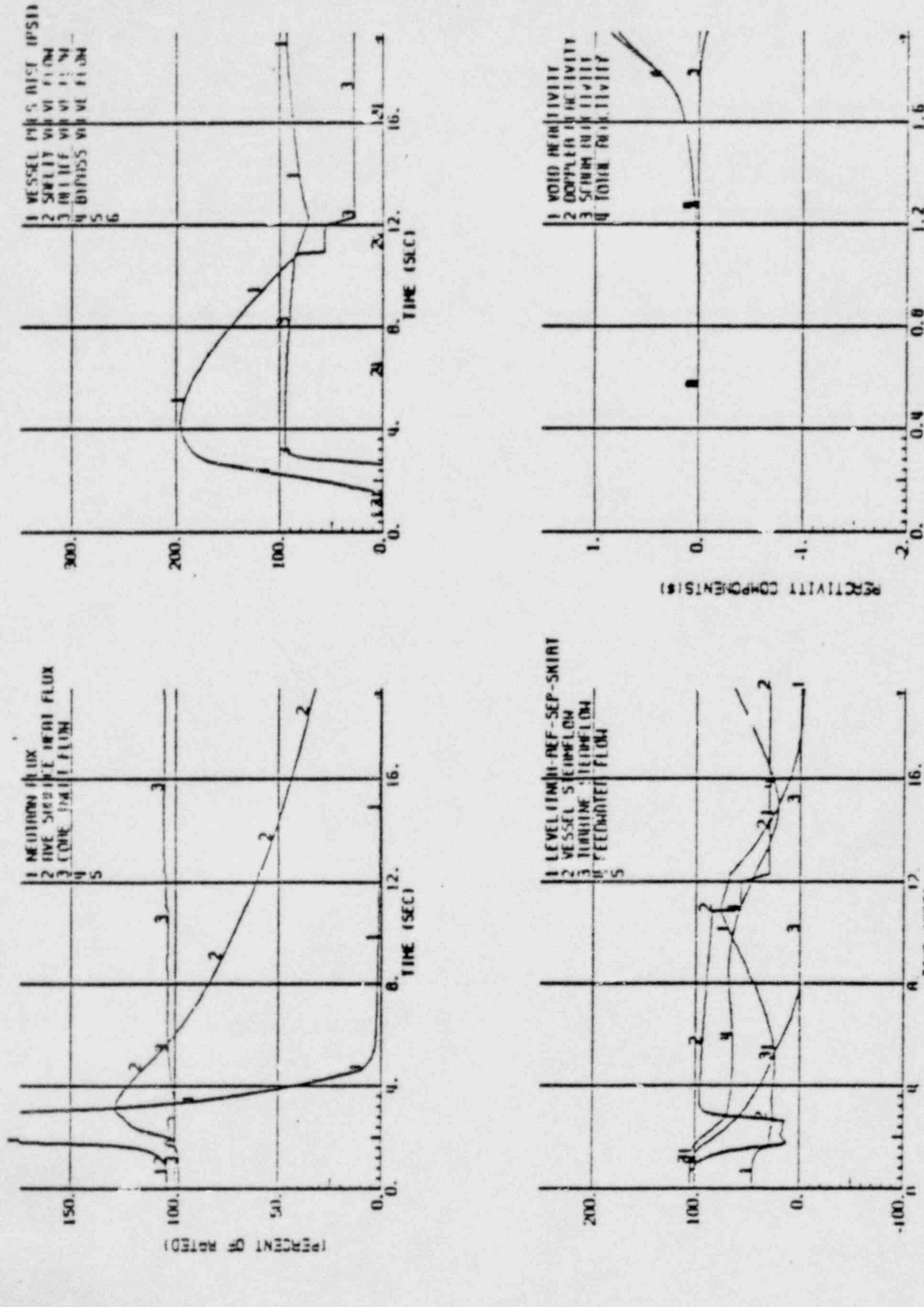


Figure 7. Plant Response to MSIV Closure

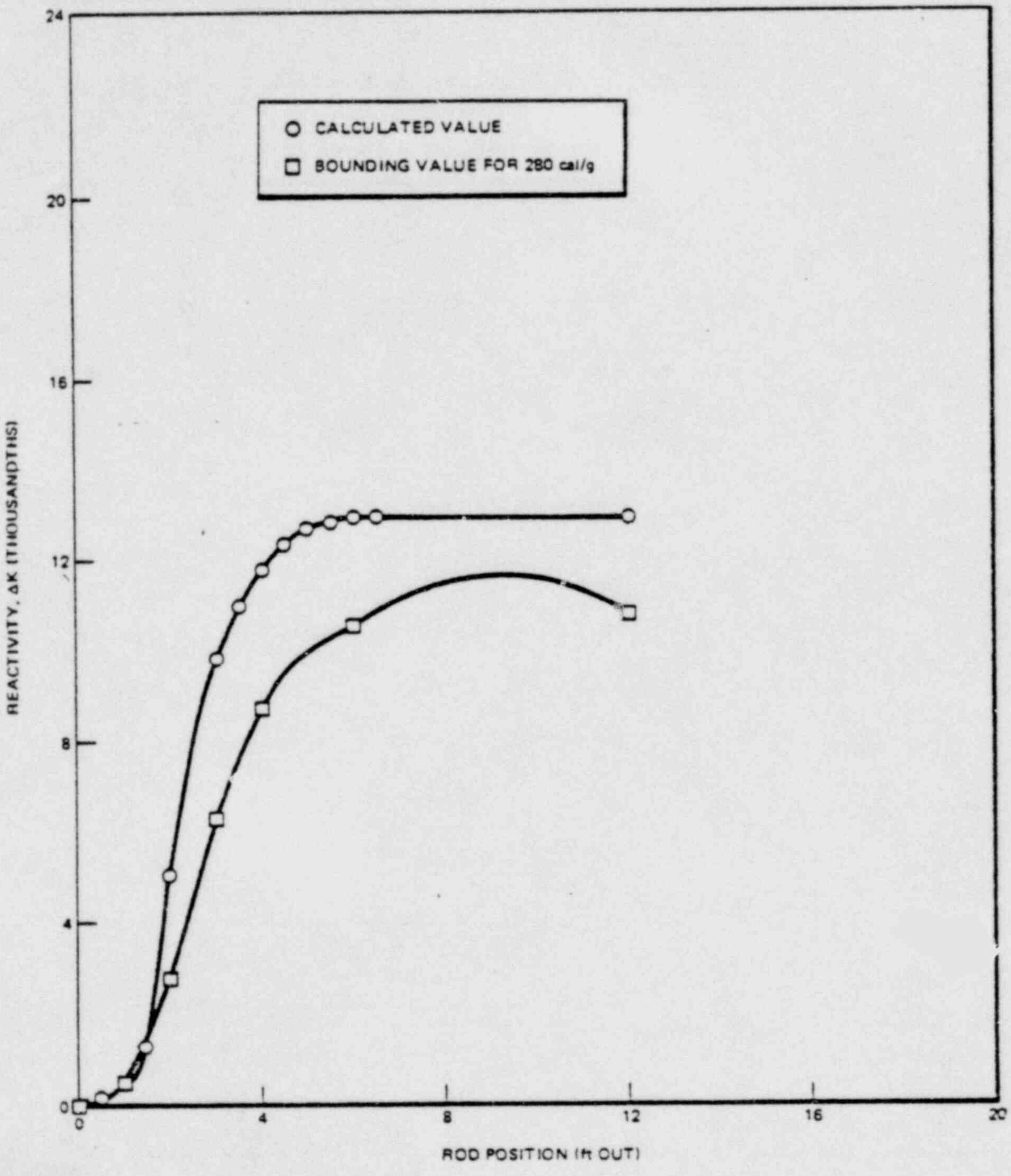


Figure 10. RDA Reactivity Shape Function

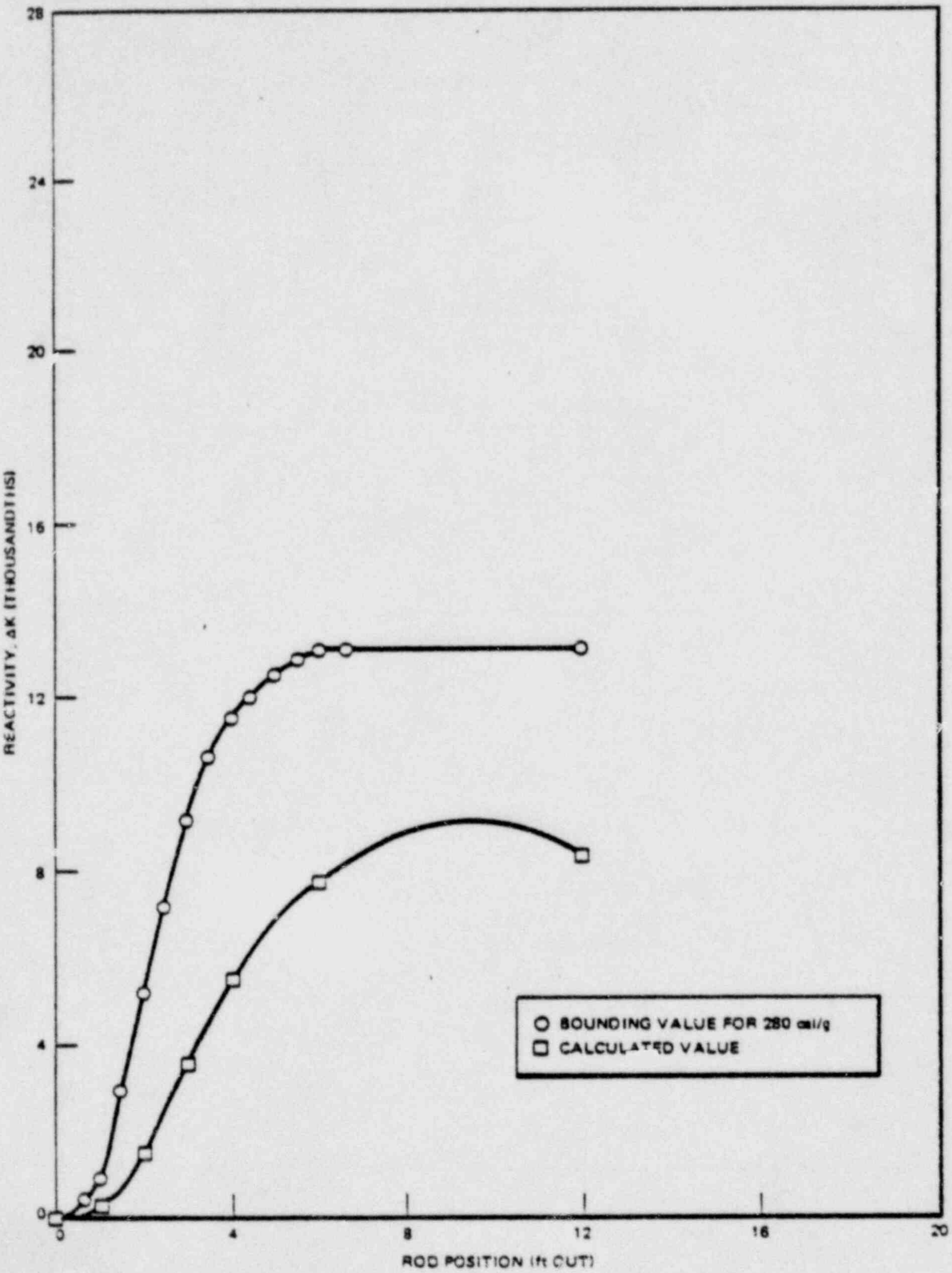


Figure 11. RDA Reactivity Shape Function at 286°C

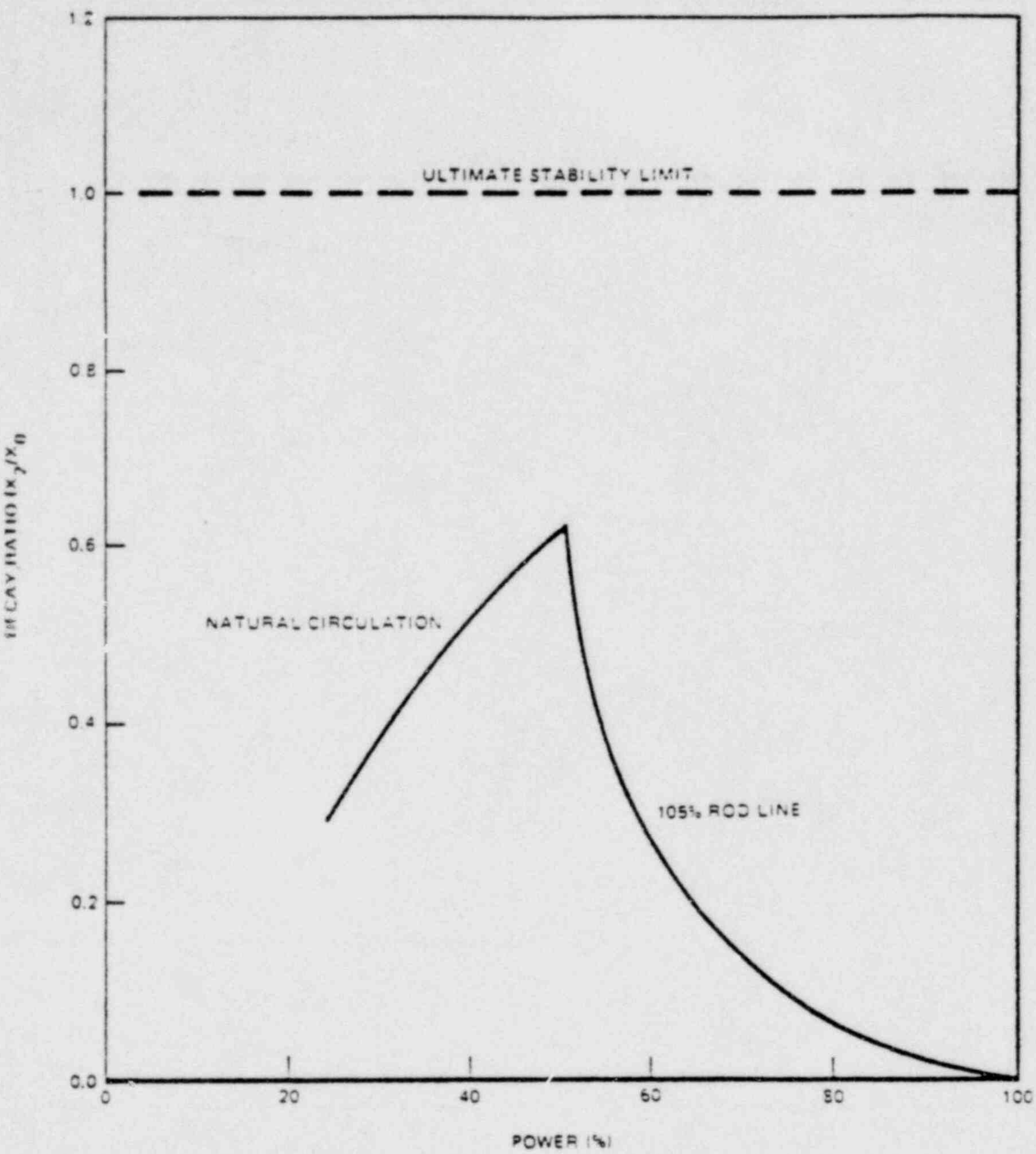


Figure 8. Decay Ratio

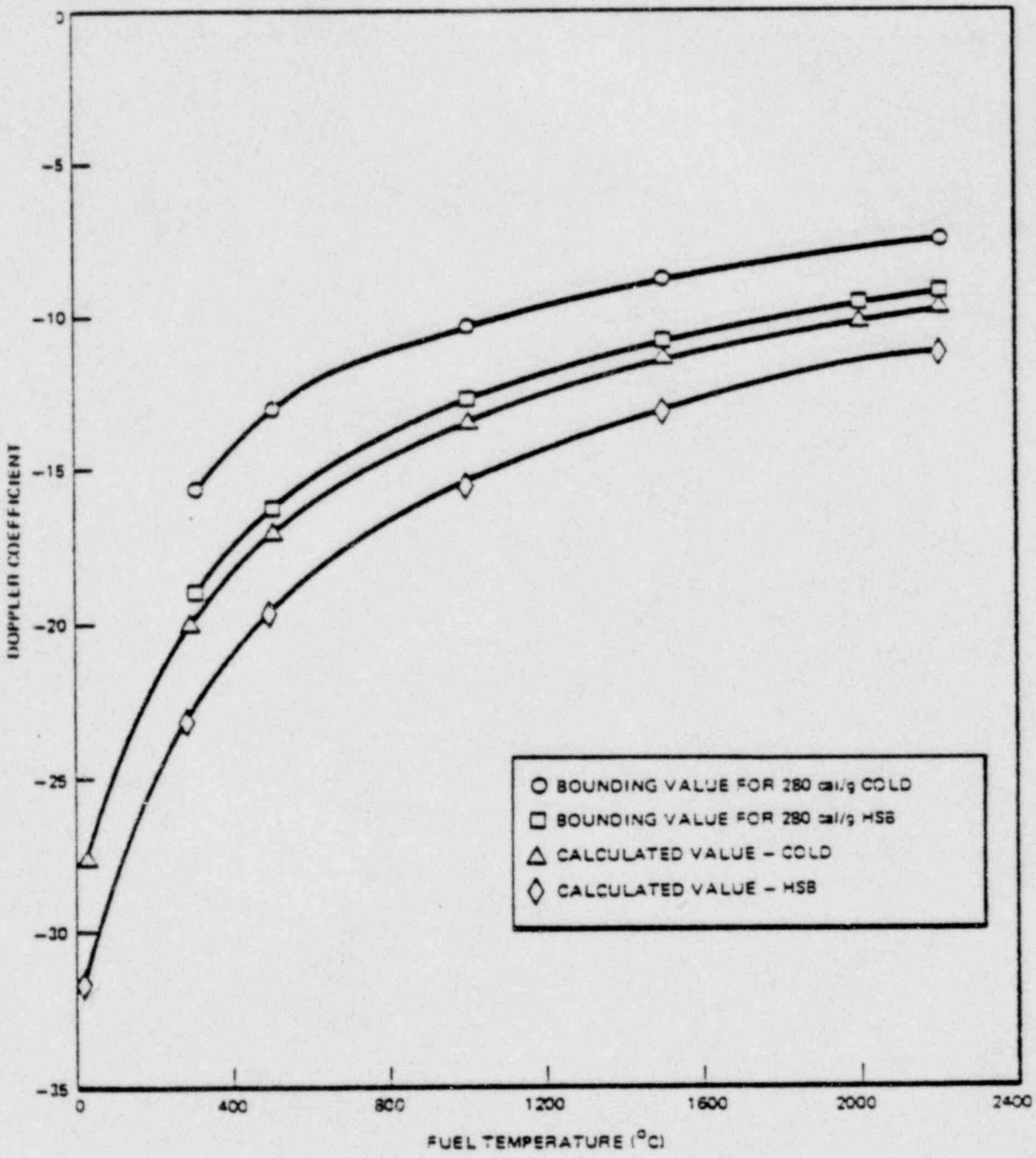


Figure 9. Doppler Reactivity Coefficient Comparison for RDA

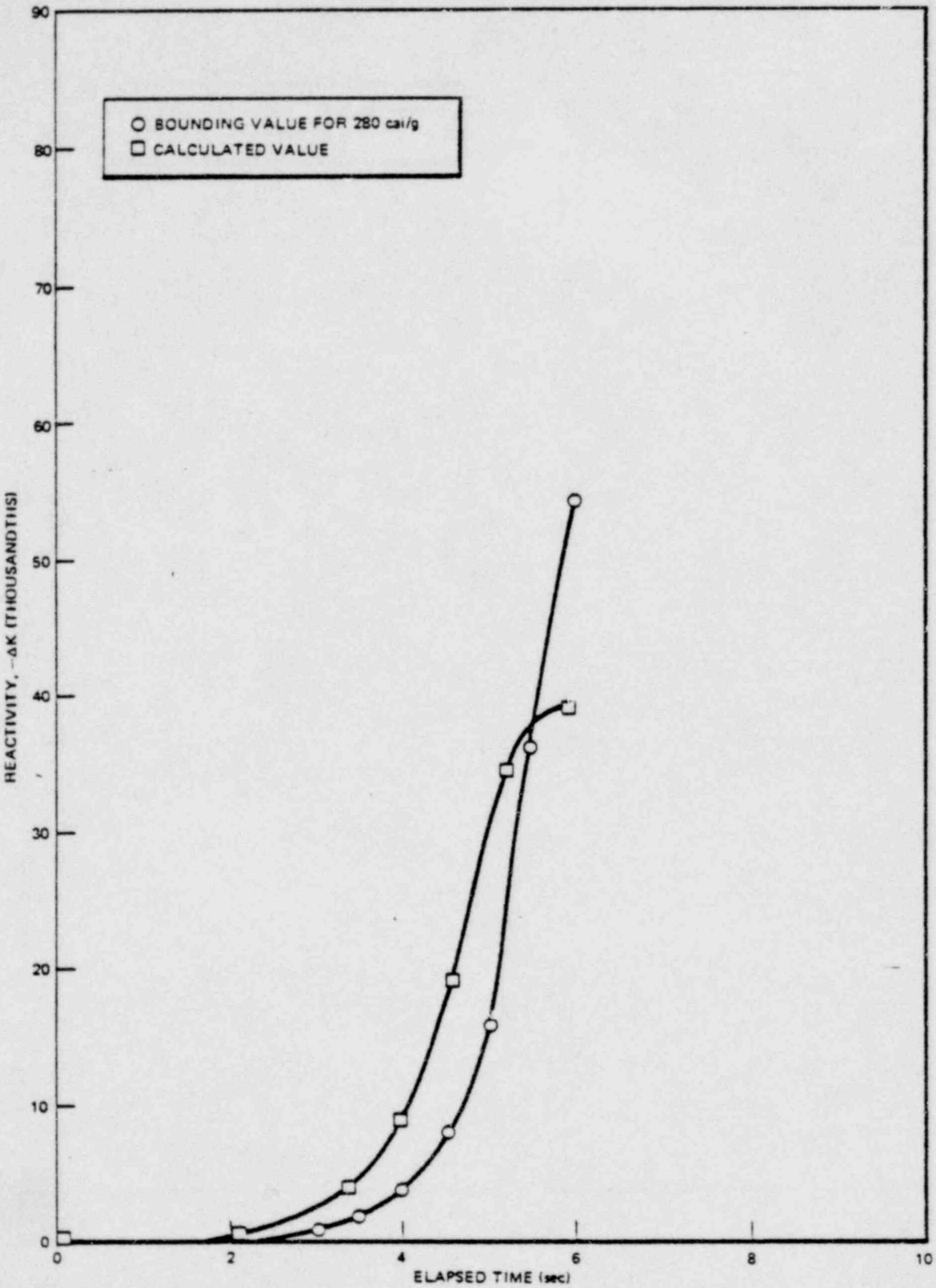


Figure 12. RDA Scram Reactivity Function at 20°C

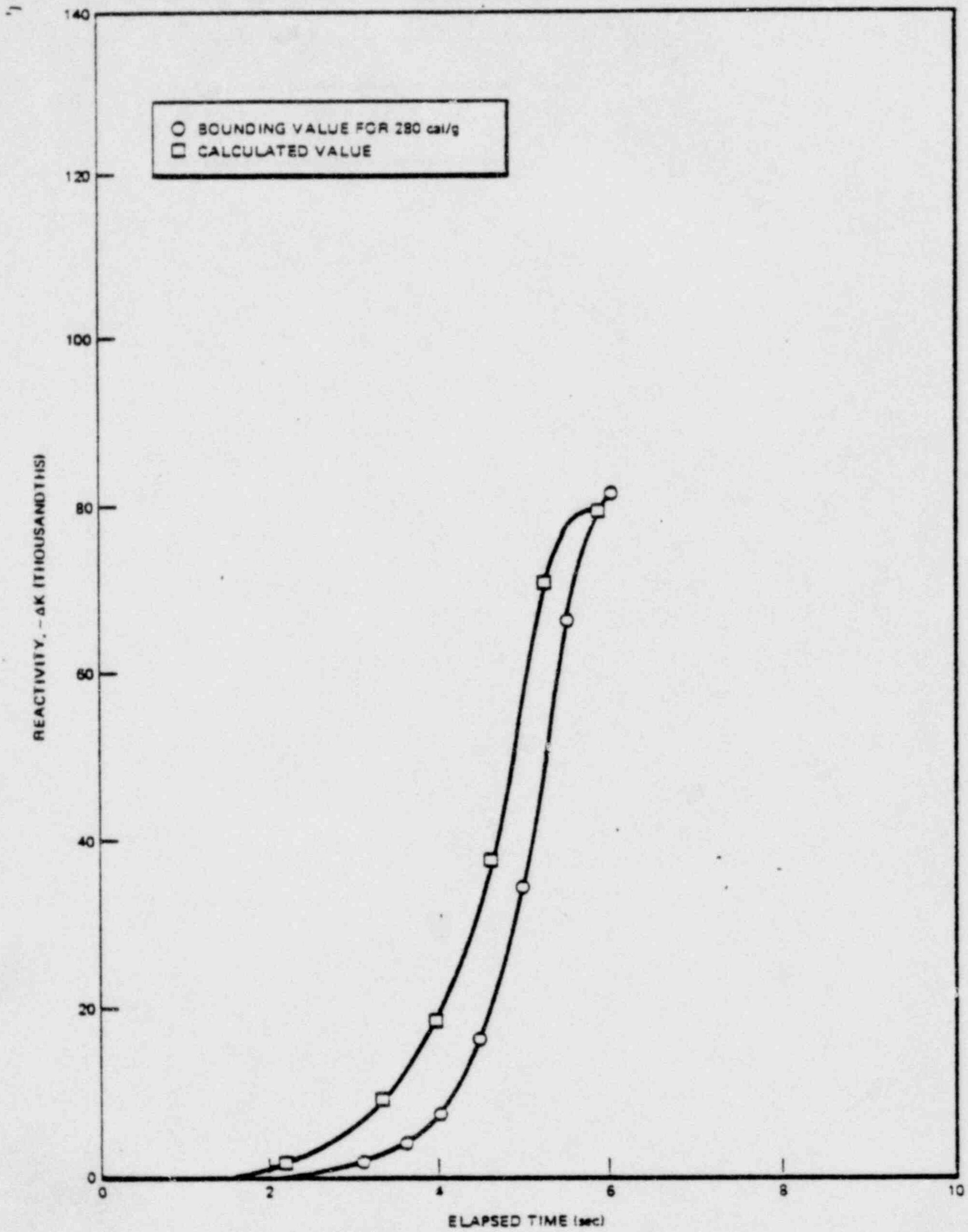


Figure 13. RDA Scram Reactivity Function at 286°C

REFERENCES

1. "General Electric Boiling Water Generic Fuel Application," NEDE-24011-P, Revision 3, March 1978.

2. Letter No. NG-77-1060 from E. E. Utley (CP&L) to A. Schwencer (NRC), September 20, 1977.

APPENDIX A

NEW BUNDLE LOADING ERROR EVENT ANALYSES PROCEDURES

The bundle loading error analyses results presented in Section 15 in this supplement are based on new analyses procedures for the rotated bundle loading error events. The use of this new analysis procedure is discussed below.

A.1 NEW ANALYSES PROCEDURE FOR THE ROTATED BUNDLE LOADING ERROR EVENT

The rotated bundle loading error event analyses results presented in this supplement are based on the new analyses procedure described in References A-1 and A-2. This new method of performing the analyses is based on a more detailed analysis model, which reflects more accurate analyses than that used in previous analyses of this event.

The principle difference between the previous analyses procedure and the new analyses procedure is the modeling of the water gap along the axial length of the bundle. The previous analyses used a uniform water gap, whereas the new analyses utilize a variable water gap which is representative of the actual condition. 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 ΔCPR 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.

In the new analyses, the axial alignment of a 180° rotated bundle conservatively ignores the presence of the channel fastener. The more limiting condition of assuming that the spacer buttons are in contact with the top guide is assumed. There is no known loading that could bend or break the channel spacer button during the insertion of a 180° rotated bundle, since both the top guide and spacer button are chamfered to provide lead-in. For a properly assembled bundle, no mechanism exists which could invalidate the assumption that a 180° rotated bundle leans to one side.

It should be noted that proper orientation of bundles in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. Five separate visual indications of proper bundle orientation exist:

(1) The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.

(2) The identification boss on the fuel assembly handle points toward the adjacent control rod.

(3) The channel spacing buttons are adjacent to the control rod passage area.

(4) The assembly identification numbers which are located on the fuel assembly handles are all readable from the direction of the center of the cell.

(5) There is cell-to-cell replication.

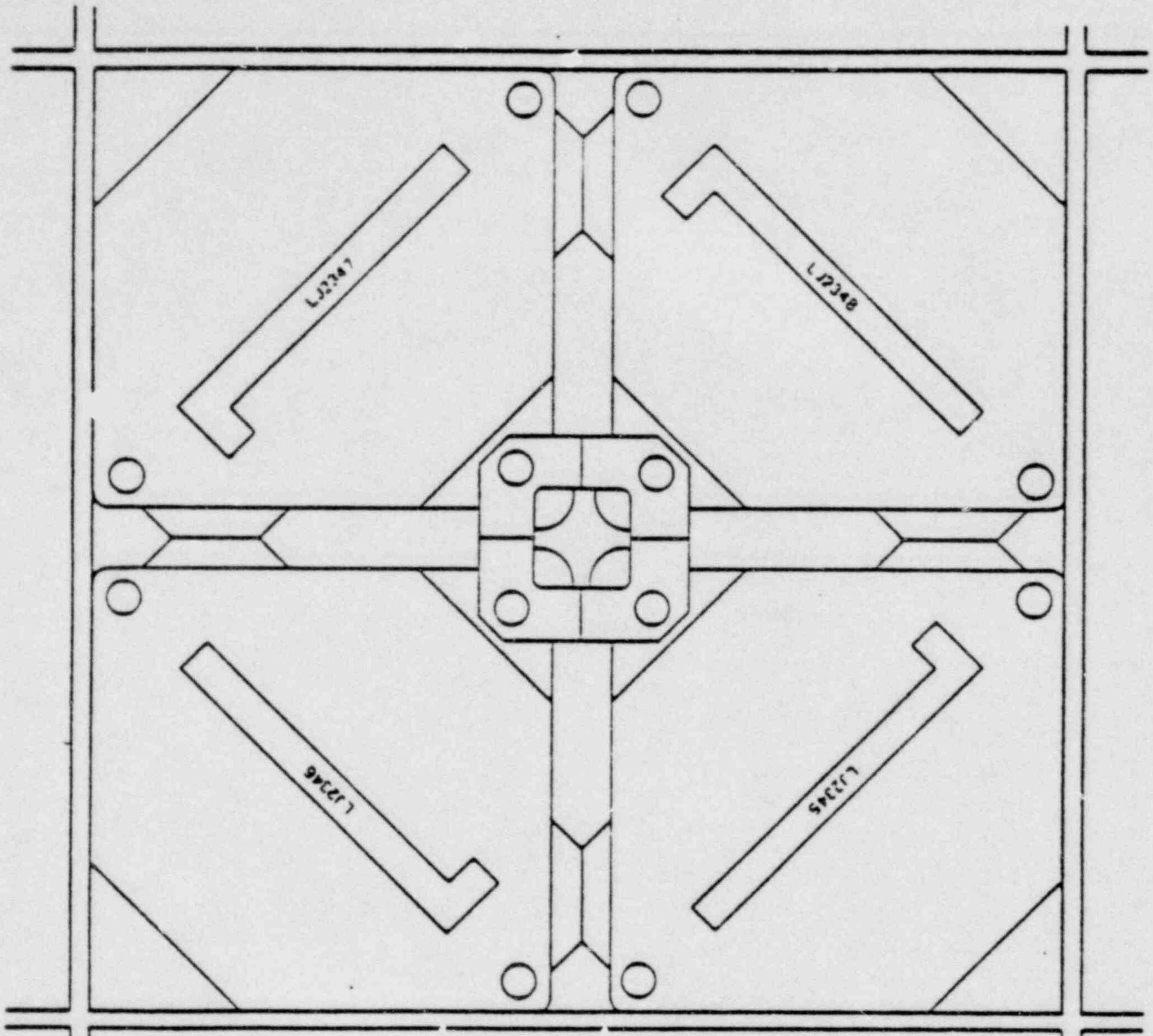
Experience has demonstrated that these design features are clearly visible so that any misloaded bundle would be readily identifiable during core loading verification. Figures A-1, A-2 and A-3 denote a normally loaded bundle, a 180° rotated bundle, and a 90° rotated bundle, respectively. Actual experience (References A-1 and A-2) has demonstrated that the probability of a rotated bundle is low.

The new analyses procedure results show that the minimum CPR for the most limiting rotated bundle in the core is greater than the safety limit.

REFERENCES

A-1 Letter, R. E. Engel (GE) to D. Eisenhut (NRC), "Fuel Assembly Loading Error," MFN-219-77, June 1, 1977.

A-2 Letter, R. E. Engel (GE) to D. Eisenhut (NRC), "Fuel Assembly Loading Error," MFN-457-77, November 30, 1977.



NOTE BUNDLE NUMBERS ARE FOR ILLUSTRATIVE PURPOSES ONLY

Figure A-1. Normal Loading

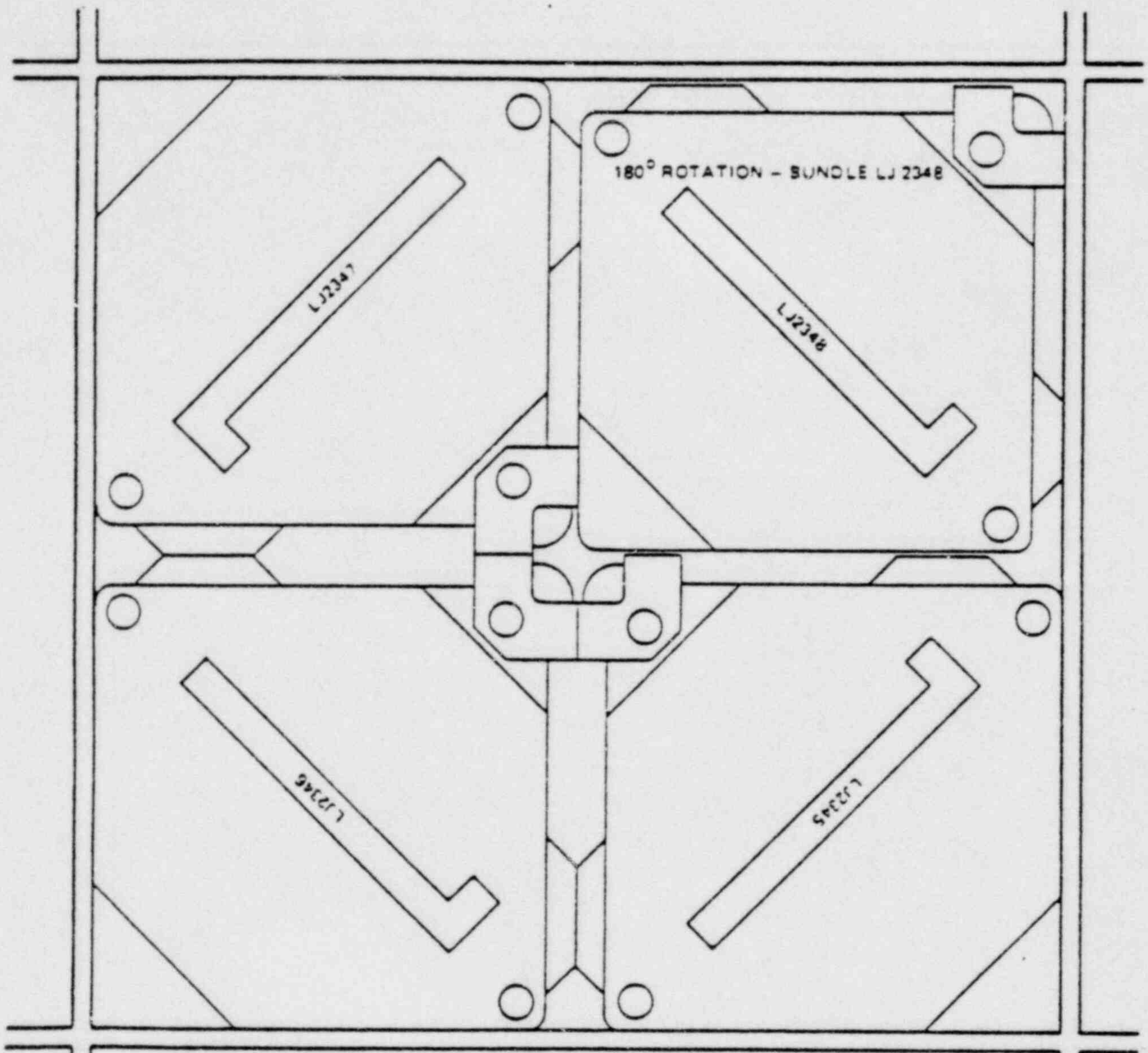


Figure A-2. Rotated Bundle, 180 Degree Rotation

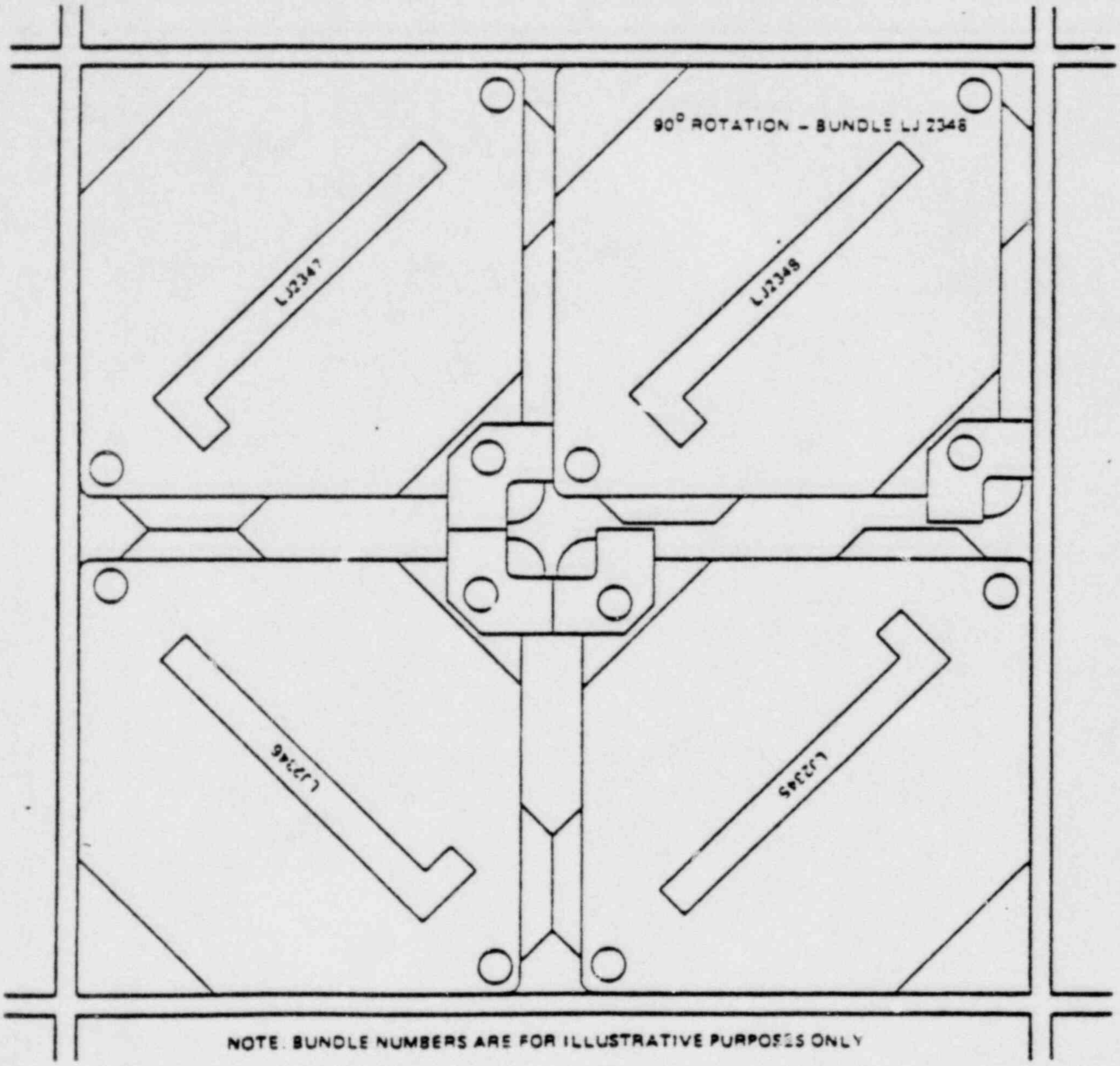


Figure A-3. Rotated Bundle, 90 Degree Rotation

APPENDIX B

Fuel Loading Error LHGR: 15.5 kW/ft

APPENDIX C

For the past several months, General Electric, with the approval of the Nuclear Regulatory Commission in cooperation with BWR Owners and EPRI, has been engaged in a program of confirmation transient testing which has resulted in the development and qualification of an improved transient model. A description of the improved transient computer model (ODYN), its qualification and its general licensing application have been transmitted to the U.S. Nuclear Regulatory Commission in References C-1 through C-4.

At the staff's request, ODYN analyses of the limiting fast pressurization transients at end of cycle 4 with Recirculation Pump Trip are being supplied in this appendix. Transients analyzed with ODYN in support of recirculation pump trip are the Load Rejection without Bypass (LR w/o BP), the Turbine Trip without Bypass (TT w/o BP), and the Feedwater Controller Failure (FWCF). For different transients under different conditions, the ΔCPR calculated using ODYN may be larger or smaller than that calculated using REDY. Table C-1 presents the results of the ODYN analysis. The analyses presented in this appendix differ from the standard licensing calculational procedure in that the assumed initial MCPR for each transient is equal to the safety limit CPR plus the ΔCPR for that transient. These transient-dependent initial CPR's are given in Table C-1, and Figures C-1 and C-2 depict the transients.

Table C-1

CORE-WIDE TRANSIENT ANALYSIS RESULTS

(ODYN ANALYSES WITH RECIRCULATION PUMP TRIP)

Transient	Exposure	Power (%)	Flow (%)	$\hat{\phi}$ (% initial)	$\hat{Q/A}$ (% initial)	$\hat{P}_{SL}$ (psia)	$\hat{P}_V$ (psig)	8x8/8x8R ΔCPR	7x7 ΔCPR
Turbine Trip without Bypass	EOC3	104	100	280.6	107.2	1187	1212	0.12	0.08
Feedwater Controller Failure	EOC3	104	100	108.3	106.0	1070	1093	0.07	0.05

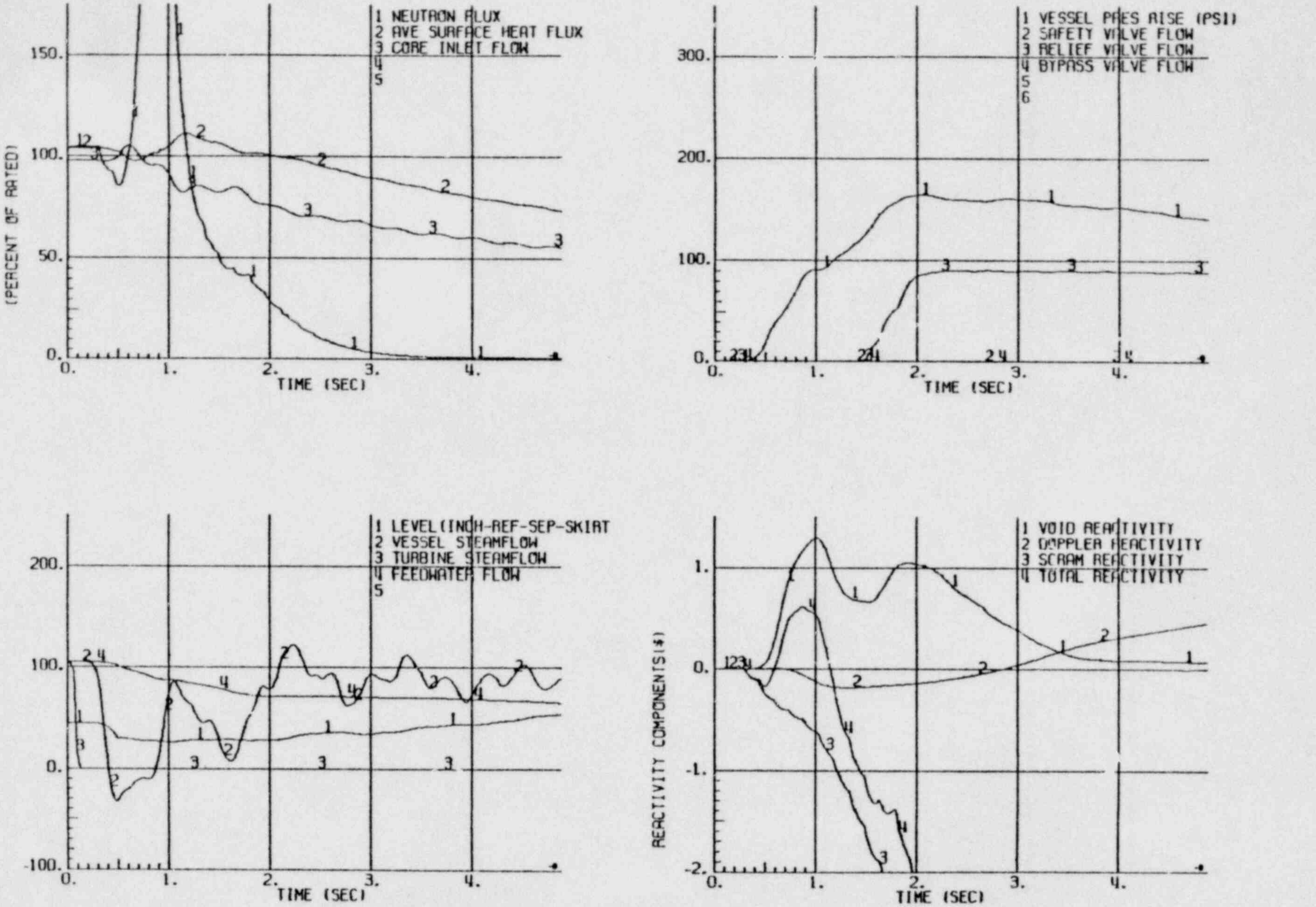


Figure C-1. Plant Response to Turbine Trip Without Bypass, Trip Scram (ODYN Analysis with RPT)

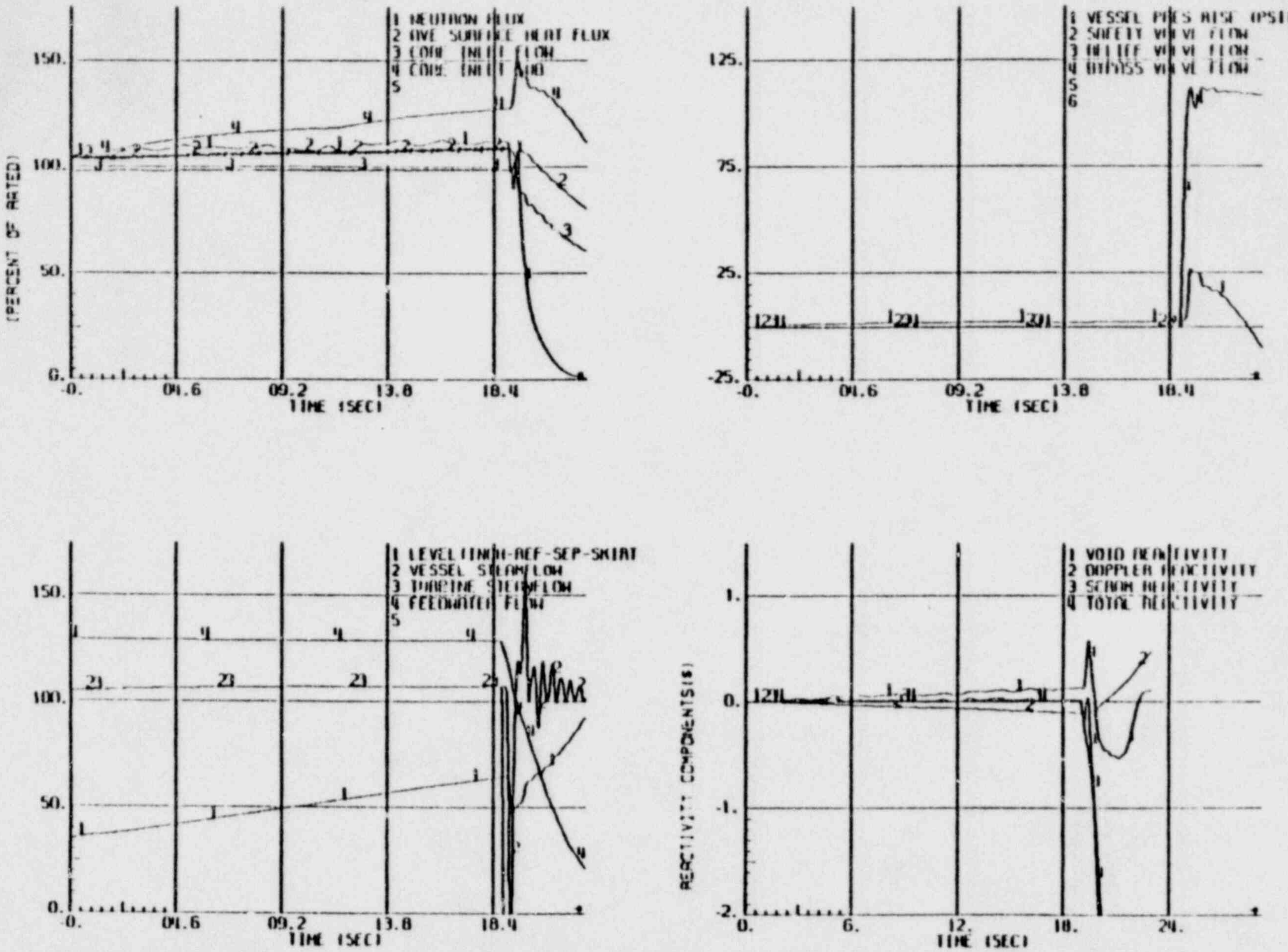


Figure C-2. Plant Response to Feedwater Controller Failure

REFERENCES

C-1. Letter MFN 462-77, E. D. Fuller to D. F. Ross, "Transmittal of ODYN Computer Model Description," dated December 2, 1977

C-2. Letter MFN 058-78, E. D. Fuller to D. F. Ross, "General Electric Proposal for Licensing Basis Criteria," dated February 7, 1978

C-3. Letter MFN 014-78, E. D. Fuller to D. F. Ross, "Transmittal of Draft ODYN Qualification Report," dated January 13, 1978

C-4. Letter MFN 136-78, E. D. Fuller to D. F. Ross, "Application Submittal for ODYN Transient Model," dated March 31, 1978

APPENDIX D

RECIRCULATION PUMP TRIP FEATURE

D.1 INTRODUCTION

Significant improvement in thermal margin can be realized if the severity of core-wide pressurization transients is reduced. The Recirculation Pump Trip (RPT) feature accomplishes this by rapidly cutting off power to the recirculation pump motors any time turbine control valve or turbine stop valve fast closure occurs. This results in a rapid reduction in recirculation flow and increases the core void content during the core-wide pressurization transients, thereby reducing the peak transient power and heat flux. A more detailed discussion of the effect of RPT is included in Section D-2.

Basically, the RPT system consists of pressure switches* installed in the turbine control valves and the position switches* in turbine stop valves. When these valves close, redundant breakers between the motor generator sets and the recirculation pump motors are tripped; this releases the recirculation pumps to coast down under their inertia. Adding RPT will result in a reduction in CPR for transients involving stop valve or control valve closures.

D.2 EFFECT OF RPT ON PLANT PERFORMANCE

D.2.1 Dynamic Characteristics

An inherent design characteristic of the boiling water reactor (BWR) is the relationship of the core average moderator density to neutron moderation, which is represented by a negative void reactivity coefficient. This negative void reactivity coefficient permits load following through control of the recirculation flow without control rod movement. To increase power, core flow is increased, which decreases the void fraction and icnreases the neutron moderation and reactor power.

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*These are the same switches which initiate scram on control valve fast closure or stop valve closure. By using the same signal to initiate RPT, the necessary hardware modifications are minimized and the scram trip and RPT are initiated simultaneously.

The negative void reactivity characteristic of the BWR dictates the necessity for reactivity control during certain operational pressurization events. The two most limiting events analyzed in a typical plant safety analysis are the rapid turbine stop valve closure (turbine trip) or control valve closure (generator load rejection) with assumed bypass failure. In these events, the dome pressure increases rapidly, causing a reduction in the core average void fraction, which increases moderation and results in a positive power increase. This is reflected in decreased margins to pressure and thermal limits.

The physical phenomenon which causes the reduced margins is that the void reactivity feedback, which is due to the pressurization, momentarily can add positive reactivity to the system faster than the control rods add negative scram reactivity.

The BWR design provides a system for which reactivity changes have an inverse relationship to the steam void content in the moderator. This void feedback effect is one of the inherent safety features of the BWR system. Any system input which increases reactor power (either in a local or gross sense) produces additional steam voids, which reduces the reactivity and thereby reduces the power. The void feedback mechanism contributes to the stable regulation of core reactivity and permits load following without use of control rods by varying the recirculation flow. The practical constraints on the void coefficient are that it must be large enough to prevent power oscillation due to spatial xenon changes yet small enough that pressurization transients do not unduly limit plant operation.

The basic phenomenon associated with void feedback is the decrease in neutron moderation resulting from an increase in void fraction. A spectral shift in the neutron flux occurs wherein the thermal flux, and hence the fission rate, decreases and the epithermal flux, and hence the resonance capture rate, increases. Conversely, a decrease in void fraction causes an increase in reactivity. The void coefficient is predominantly the function of three variables for any fixed bundle geometry: (1) the average voids; (2) enrichment; and (3) exposure. As each of these three parameters increases, the absolute magnitude of the void coefficient increases and becomes more negative.

For pressurization transients, the rate of flux rise is dependent on the magnitude of the void coefficient. The more negative the void coefficient, the greater the flux rise rate. The rate at which the negative reactivity can be added to the core by the scram determines the severity of the transient. The scram reactivity depends on the ability of the control rods to be in the high flux regions of the core. The minimum scram reactivity occurs at end of cycle when control rods are fully withdrawn from the core. In this situation, it takes a longer time for the control rod to travel to a high importance region in the core. For this reason, the pressurization transients are most severe near the end of the cycle.

The degree to which the pressure and thermal margins are reduced during pressurization events depends on the tradeoff between the negative scram and positive void reactivities. Typically, at beginning of cycle (BOC), control rods are partially inserted; this permits a prompt shutdown of the system without a significant decrease in margins. As the fuel cycle proceeds toward end of cycle (EOC), the control rods are withdrawn until, ideally, they are all withdrawing. Hence, the effectiveness of scram reactivity for shutdown of certain pressurization transients is decreased as the core approaches EOC conditions.

As discussed above, margins are decreased when the positive void reactivity feedback is inserted at a rate faster than the negative scram reactivity.

Analyses have shown that the transient severity can be significantly reduced by a rapid reduction in core flow. This increases the core void fraction during pressurization transients and consequently minimizes the power rise experienced. The rapid reduction in core flow necessary to accomplish this effect can be achieved by the prompt tripping of both recirculation pumps. The RPT system described in Section D.3 has been developed to accomplish this goal.

D.2.2 Thermal Limits Consideration

One of the operating fuel thermal limits, the minimum critical power ratio (MCPR), is established such that the most severe abnormal operational transient is not expected to subject more than 0.1% of the fuel rods to boiling transition. This is known as the General Electric Thermal Analysis Basis (GETAB). GETAB statistically correlates a calculated MCPR as the condition at which less than 0.1% of the fuel rods are expected to experience boiling transition. This value

is incorporated into the plant technical specifications as the fuel cladding integrity safety limit. An operating limit MCPR is established such that the most severe abnormal operational transient will not result in violating the safety limit. The difference between the actual plant operating critical power ratio (CPR) and the operating limit MCPR s a measure of the thermal margin.

If the normal operating CPR at the licensed power level cannot be maintained above the operating limit MCPR, a plant derate will be imposed to assure that the resultant change in CPR from a worst-case abnormal operational transient will not decrease the MCPR below the safety limit. A reduction in severity of the worst transient allows a reduction in the operating limit. Usually either a turbine or generator trip without bypass is the limiting thermal event near EOC. The RPT system is intended to provide improved thermal margin for these limiting events.

D.2.3 Overpressure Protection Considerations

The RPT system has no effect on overpressure protection considerations since it is not initiated during the event (MSIV Closure with Indirect Scram) which demonstrated compliance with the ASME vessel overpressure protection limit.

D.3 RECIRCULATION PUMP TRIP DESCRIPTION

D.3.1 System Function

The RPT system, which is designed to improve fuel thermal margin, trips both recirculation pumps upon sensing stop valve closure or fast control valve closure. The reduced core flow reduces the void collapse in the core during two of the most limiting pressurization events (i.e., turbine and generator trips). Tripping of the recirculation pumps results in a smaller net positive void reactivity addition to the system during these pressurization events. This results in a lower power increase and consequently a lower operating MCPR limit. Although the reduction in core flow in itself may cause a slight decrease in thermal margins, the effect of reduced flow on the power increase is a considerably more dominant effect and the net result is to reduce the thermal severity of the event.

In order for the RPT system to effectively counteract the void collapse effects from pressurization transients, the pump trip must occur very soon after the turbine/generator trip, and the pumps must coast down at a relatively fast rate. If the pump trip and coastdown do not occur quickly, the positive void reactivity feedback caused by the pressurization effects will dominate the transient and no margin improvement will be seen from tripping of the pumps.

Analyses have been performed which demonstrate that the RPT system is made most effective by installing and tripping a line breaker between the recirculation pump drive motor/generator and the pump motor. Although a motor/generator field breaker trip has cost advantages over a line breaker, the response characteristics from such a trip do not achieve significant improvements in thermal margins. Upon tripping the field breakers, the drive motor generator continues to momentarily supply some reduced power to the pump motor due to the time required for the generator field and line current to drop to zero. This results in reduced effectiveness of the system.

In order to achieve the desired improvements in thermal margins for the turbine/generator trips, the supply current to the pump motor must be terminated in less than approximately 200 milliseconds after receipt of the signal from the switches in the turbine stop valves or in the turbine control valves. The line breaker pump trip does achieve the desired system goal.

D.3.2 System Description

The RPT system includes all equipment that trips recirculation pump motors from their power supplies in response to a turbine/generator trip or load rejection. The RPT system is designed to be of quality consistent with the reactor protection system functions which provide protection for the same events. The system consists of turbine control and stop valve closure sensors, separate division logic and two circuit breakers for each pump motor. The RPT system is designed to be operable whenever the turbine generator trip scram is operable (i.e., above approximately 30% reactor thermal pressure). Existing turbine first-stage pressure sensors will prevent RPT initiation for turbine-generator trips occurring below the existing 30% power bypass of turbine and generator trip scram signals.

The RPT system design includes two separate trip divisions with each having two separate trip channels, sensors and associated equipment for each measured variable. The system is designed to meet the single-failure criterion such that any single trip channel (sensor and associated equipment) or system component failure shall not prevent the system from performing its intended safety function. Electromechanical relays used as the logic elements within the system and the system logic are of the failsafe type (i.e., trip on loss of electrical power).

The RPT system is designed to accomplish the desired function and to minimize the effect of this additional system on plant availability. The system logic is designed such that it will not cause the inadvertent trip of more than one pump given a single component failure in the system. Each trip division shall be clearly identified to reduce the possibility of inadvertent trip of the recirculation pump during routine maintenance and test operations. Redundant sensor circuits in each division (sensors, wiring, transmitter, amplifiers, etc.) are electrically, mechanically, and physically independent so that they are unlikely to be disabled by a common cause except for an electrical power failure.

Capability is provided for testing and calibrating the system logic quarterly and circuit breakers once per refueling outage. Provisions are made to allow closure of stop valve and fast closure of turbine control valve separately at least one valve at a time (for normal routine valve test purposes) without causing a pump motor trip. The system input sensors and the division logic are capable of being checked one channel or division at a time. The sensors and system logic test or calibration during power operation will not initiate pump trip action at the system level.

D.4 RPT LOGIC DIAGRAMS, CIRCUITS AND TESTABILITY

The purpose of RPT is to reduce the severity of the reactivity transient caused by either of two postulated events: (1) turbine trip with failure of the bypass valve, and (2) generator load rejection with failure of the bypass valves.

RPT is not required for any other postulated events; therefore, the logic begins with the sensing of stop valve closure (turbine trip) or fast closure of the turbine control valves (generator load rejection). RPT is initiated following either event for turbine power levels greater than 30%, but is independent of the operation of the bypass valves.

Figure D-1 is the logic diagram for RPT System A that trips pump A. System B is the same except for the nomenclature changes indicated in parentheses.

The logic is "two-out-of-two or two-out-of-two" for both stop valve closure and fast closure of the turbine control valves. This means, for example, that the closure of stop valves 1 and 2 will trip both recirc pumps through System A or stop valves 3 and 4 will trip both pumps through System B. This logic provides the testability feature with which any one stop valve or control valve can be closed and system status tested by observing relay contact status without causing RPT operation. The entire logic of one division of the RPT system may be tested without tripping the pumps by placing that system control switch briefly in the "inop" position for the duration of the test; the test is initiated by closing the two stop valves which initiate that system to the 10% closed (90% open) position. Successful completion of the test is indicated by annunciation of "RPT initiate" as the annunciation relays are energized. During this brief interval, the redundant RPT system is, of course, continuously available to perform its safety function.

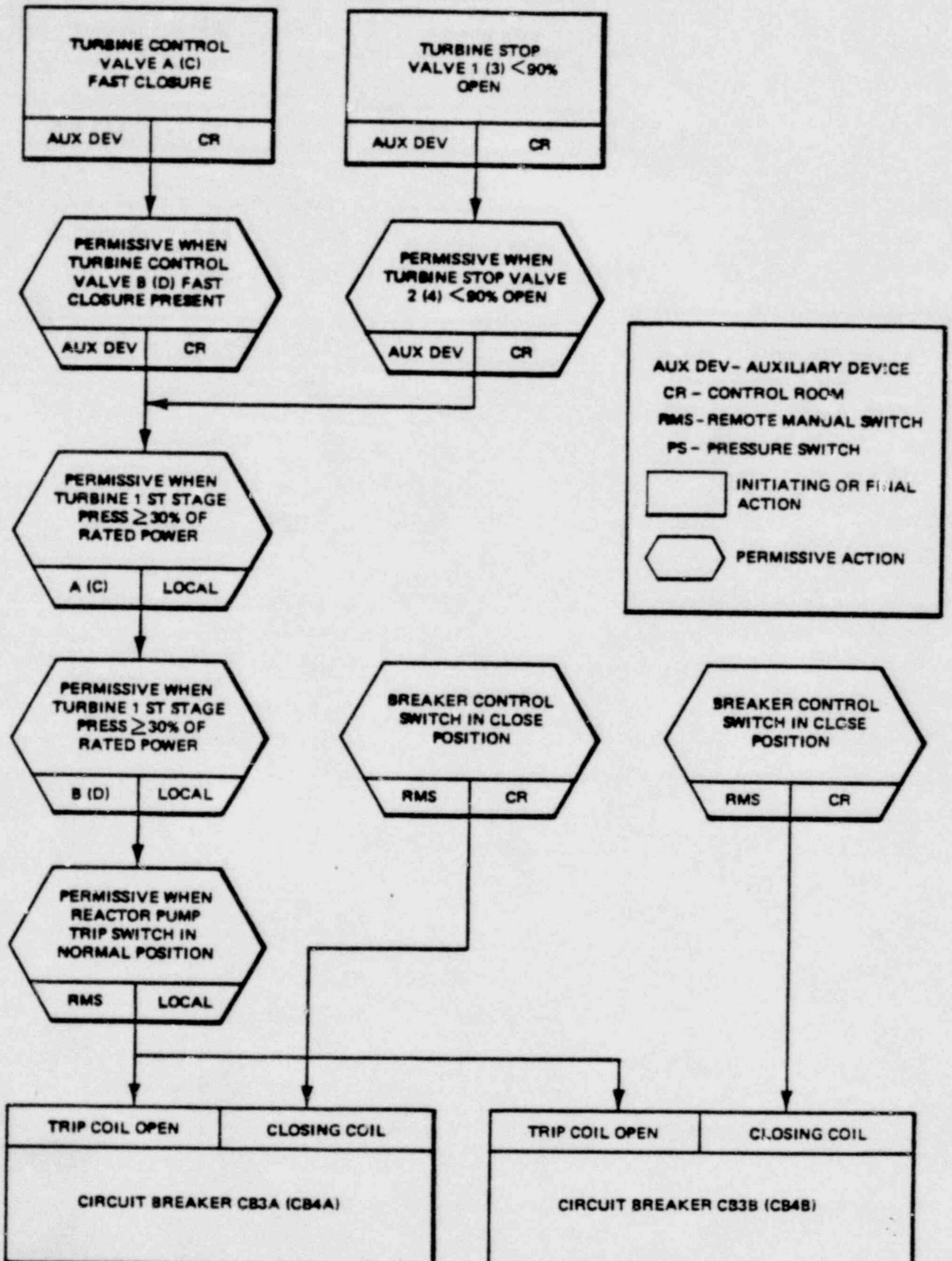

Figure D-1. Recirc Pump Trip System A
Typical for System B Except as Shown ( )