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TO: Mr. Edson G. Case

FROM: Iowa Elect Light & Power Co.
Cedar Rapids, Iowa
Lee Liu

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DESCRIPTION

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ENCLOSURE

License No. DPR-49 Appl for Amend tech specs proposed change concerning amended tech specs operational limits for Cycle 4. consisting of 1) Boiling Water Reactor Reload - 3 Licensing Amendment for Duane Arnold Energy Center (NEDO-24087, 77NED359, Class 1, Dec., 1977) which includes appropriate safety and transient analyses. adn 2) Proposed Tech Specs (RTS-106) reflecting the results of the above safety and transient analyses...

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IOWA ELECTRIC LIGHT AND POWER COMPANY

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LEE LIU
VICE PRESIDENT - ENGINEERING

January 4, 1978
IE-78-7

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U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Case:

Transmitted herewith in accordance with the requirements of 10CFR50.59 and 50.90 is an application for amendment of DPR-49 and the Technical Specifications (Appendix A to License) for the Duane Arnold Energy Center (DAEC) for amended Technical Specifications operational limits for Cycle 4.

This application consists of:

1) Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center (NEDO-24087, 77NED359, Class 1, December, 1977) which includes appropriate safety and transient analyses.

2) Proposed Technical Specifications (RTS-106) reflecting the results of the above safety and transient analyses.

The safety and transient analyses have been conducted in the same manner as previously submitted on our docket. (Application dated January 31, 1977 IE-77-220 as amended and Application dated June 17, 1977 IE-77-1192 as amended.) The ECCS analysis conducted for Cycle 3 (Application dated June 24, 1977 IE-77-1230) is applicable to Cycle 4 and is conservative. The proposed Technical Specifications maintain the same safety margins, (with the exception of Bundle Loading Error which is an unresolved item from Cycle 3) as presently licensed for Cycle 3 during Cycle 4 and therefore does not constitute an unreviewed safety question nor a significant hazards consideration. This change does not result in any change to the presently licensed safety limit MCPR of 1.06.

We expect that the present operational limits contained in the DAEC technical specifications will permit operation at about 98% of power during approximately the first seven months of operation in Cycle 4. Our Operations Committee and Safety Committee has reviewed operation with the new fuel loading and has concluded that operation with the new fuel loading in accordance with present technical specifications will not present any unreviewed safety question under 50.59, 10 CFR.

780100031

Accordingly, operation of the plant with the new fuel after refueling is authorized subject to existing technical specifications.

It is desirable, however, that the technical specifications be amended as requested in the enclosure as promptly as feasible to enable operation at full power with the new fuel reload.

The DAEC is presently scheduling a shutdown March 10, 1978 with restart planned for April 8, 1978.

This application has been reviewed and approved by the DAEC Operations Committee and the DAEC Safety Committee.

Three signed and 37 additional copies of this application are transmitted herewith. This application consisting of the foregoing letter and enclosures hereto, is true and accurate to the best of my knowledge and belief.

Iowa Electric Light and Power Company

BY: ______________________

Lee Liu
Vice President, Engineering

LL/KAM/gan

cc: K. Meyer
D. Arnold
R. Lowenstein
R. Clark (NRC)
L. Root
J-60a

Subscribed and Sworn to before me on this 5th day of January, 1978.

Sue E. Scott

Notary Public in and for the State of Iowa

NOTARIAL SEAL IOWA
SUE E. SCOTT
My Commission Expires
September 30, 1978

PROPOSED CHANGE RTS-106 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix A of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Table 3.12-2 contains MCPR limits for 7 x 7 and 8 x 8 fuel related to various fuel burnups and Specifications 3.6.D and 4.6.D Bases concerning safety/relief valve margin.

II. Proposed Changes in Technical Specifications

The licensees of DPR-49 propose the following changes in the Technical Specifications set forth in I above:

Delete sheets 3.6-24, 3.12-9a and 3.12-11, and replace with the attached sheets.

III. Justification for Proposed Change

This change is proposed in order to incorporate into the Technical Specifications the new MCPR limits and safety/relief valve pressure margins governing cycle 4 operation of the DAEC. The justification for the changes is contained in "Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, NEDO-24087, 77 NED 359, Class 1, December 1977."

IV. Review Procedure

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

the direct scram (valve position scram) results in an 88 psi margin to the code allowable overpressure limit of 1375 psig if a flux scram is assumed. In addition, the generic analyses have been conducted which show an approximate 20 psi sensitivity increase for each relief valve failure.

The analysis of the plant isolation transient (Turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in FSAR paragraphs 14.5.1.2 and 14.5.1.3 and is evaluated in each reload analyses. These analyses show that the six relief valves assure a 25 psi or greater margin below the setting of the safety valves. Therefore, the safety valves will not open. These analyses verify that peak system pressure is limited to greater than a 125 psi margin to the allowed vessel overpressure of 1375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to

TABLE 3.12-2

MCPR LIMITS

Fuel Type	Exposure Remaining to End of Cycle			
	B.O.C. to >2000 MWD/T	≤ 2000 MWD/T to > 1000 MWD/T	≤ 1000 MWD/T to > 500 MWD/T	≤ 500 MWD/T to E.O.C.
7 x 7	1.22	1.22	1.26	1.30
8 x 8	1.22	1.29	1.34	1.38

3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-02-1A, Class I, July 1977, Appendix A.

2. General Electric BWR Generic Reload Application for 8 x 8 Fuel, NEDO-20360, Revision 1, November 1975.

3. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-19735, August 1973.

4. Supplement 1 to Technical Reports on Densifications of General Electric Reactor Fuels, December 14, 1973 (AEC Regulatory Staff).

5. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

6. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

7. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Part 50, Appendix K, NEDE-20566 (Draft), August 1974.

8. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, NEDO-24087, 77 NED 359, Class 1, December 1977.

NEDO-24087
77NED359
Class I
Dec 1977

GENERAL ELECTRIC BOILING WATER REACTOR

RELOAD-3 LICENSING AMENDMENT

FOR

DUANE ARNOLD ENERGY CENTER

BOILING WATER REACTOR SYSTEMS DEPARTMENT • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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LIST OF TABLES

1. INTRODUCTION

This document provides information for Reload-3 (Cycle 4) at the Duane Arnold [redacted] Energy Center. The technical bases, generic design information, and safety analyses are given in Reference 1.

The design reference core loading is based on Reload-3 fuel consisting of 88 8x8 fuel bundles having an enrichment of 2.74 wt % U-235 with 80 mil channels. The objective of this document is to describe the reloading of the reactor core to insure sufficient reactivity to operate the 368 bundle core at a licensed power level of 1593 MWt for an approximate one year cycle.

Analyses in this document and its references justify satisfaction of these objectives. The design basis bypass flow configuration assumed in the analyses in this document is installation of the alternate flow path in 276 of the 368 assemblies in the core (i.e., all 1-in. bypass flow holes in the core support plate are plugged and two 9/32-in. holes are drilled in the lower tieplate of 276 assemblies).

This licensing submittal provides analysis to support exposure dependent minimum critical power ratio (MCPR) operating limits. The ΔCPR due to various transients has been analyzed at different exposures resulting in greater flexibility throughout most of the cycle. Transient analyses have been performed to obtain an operating limit MCPR from:

(1) BOC to 2000 MWd/t before EOC

(2) from 2000 MWd/t before EOC to 1000 MWd/t before EOC

(3) from 1000 MWd/t before EOC to 500 MWd/t before EOC

(4) and for the last 500 [redacted] MWd/t before EOC.

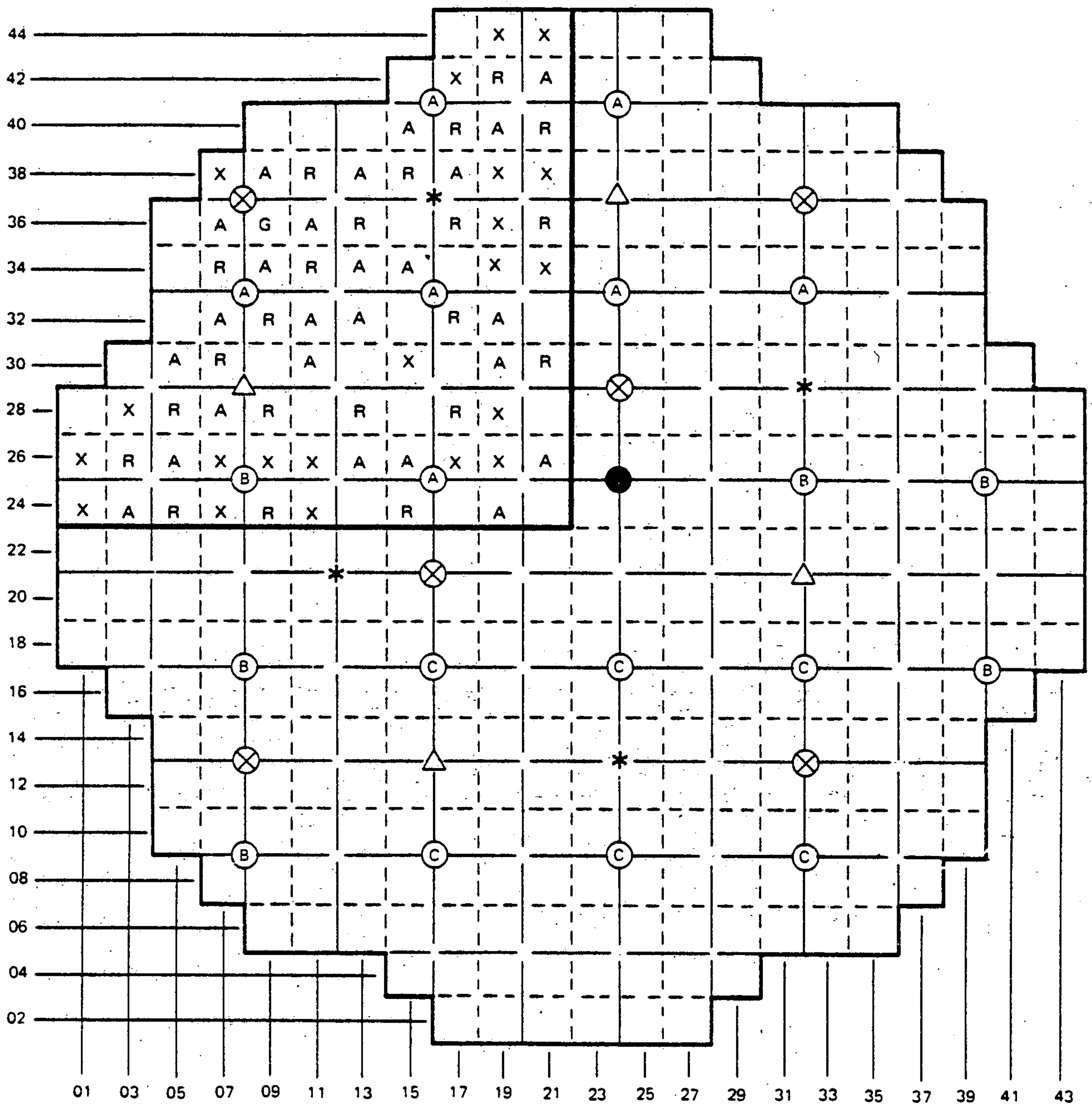
2. SUMMARY

The design reference core configuration for this reload description consists of bundles defined in Table 2-1. The relative location of each fuel bundle type is shown in Figure 2-1.

Table 2-1

FUEL TYPE AND NUMBER

Fuel Type	Number
Initial Core	
Type 2 7D212	74
Type 3 7D212	18
Interim Reload 7D230	4
Reload 1	
8D274H	52
8D274L	32
Reload 2	
8D274H	68
8D274L	32
Reload 3	
8D274H	88
Total	368



BLANK – INITIAL CORE 7D212
G – INTERIM RELOAD 7D230
X – RELOAD 1 8D274
A – RELOAD 2 8D274
R – RELOAD 3 8D274H

● LPRM LOCATION (COMMON LOCATION FOR ALL TIP MACHINES)
○ LPRM LOCATION (LETTER INDICATES TIP MACHINE)
⊗ IRM LOCATION
△ SRM LOCATION
✱ SOURCE LOCATION

Figure 2-1. DAEC R-3 Design Reference Core Loading – Quarter Core Mirror Symmetry Upper Left Quadrant Only Shown

The following bundles, listed in Table 2-2, will be provided with 2 bypass flow holes in the lower tie plate:

Table 2-2

BUNDLES WITH BYPASS FLOW HOLES

Reload 3	
8D274H	88
Reload 2	
8D274H	68
8D274L	32
Total R2	100
Reload 1	
8D274H	52
8D274L	32
Total R1	84
Interim Reload	
7D230	4
Total	276

3. MECHANICAL DESIGN

The Reload 3 fuel which will be employed has the same mechanical configuration and fuel bundle enrichment as the 8D274H fuel assembly described in Reference 1. Reload 3 incorporates the improved water rod design described in Section 3.1 of Reference 1. The design criteria, models, and results from design evaluation presented in Section 3 of Reference 1 are applicable to the subject reload.

The Reload 3 fuel assemblies will be provided with two bypass flow holes in the lower tie plate.

4. THERMAL-HYDRAULIC ANALYSES

Discussions of thermal-hydraulic design requirements, hydraulic models, statistical analysis and uncertainties, and thermal hydraulics of mixed core loading is given in Section 4 of Reference 1. The analysis applicable to DAEC, Cycle 4, is given below.

4.1 STATISTICAL ANALYSIS

Bounding statistical analyses were performed which provide a conservative safety limit MCPR applicable to all the reload cycles for BWR/4 class plants. The results of the analyses show that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition if the MCPR is 1.06 or greater.

4.1.1 Fuel Cladding Integrity Safety Limit

Based on the results of the statistical analysis, the fuel cladding integrity safety limit is a MCPR of 1.06.

4.1.2 Basis for Statistical Analyses

The reactor core selected for the statistical analysis is a typical 251-764 reload core. The large core analysis results conservatively apply for these classes of reactors.

The histogram of relative bundle powers used in the statistical analysis is shown in Figure 4-2 of Reference 1.

The power distribution was generated by arranging the control rod pattern so that as many fuel assemblies as possible are at or near the MCPR limit in accordance with the procedure described in Reference 2. For comparison purposes, actual operating power distributions of typical BWR reload cores are shown in Figures 4-1 and 4-2.

It can be seen that the power distribution used in the statistical analysis is clearly skewed more to the high power side than the actual operating

power distributions, thus yielding a conservative value of the 99.9% statistical limit MCPR.

The uncertainty inputs used in the bounding statistical analysis are listed in Table 4-1. Nominal values of parameters used in the bounding statistical analysis are listed in Table 4-3 of Reference 1.

4.2 ANALYSIS OF ABNORMAL OPERATIONAL TRANSIENTS

The results of the most limiting pressure and power increase transients were evaluated to determine the largest decrease in MCPR. Other types of transients have an insignificant effect upon critical power and are, therefore, not reviewed in depth. The results of the transients analyzed are summarized in Table 4-2.

Addition of the ΔCPR to the safety limit MCPR gives the minimum operating MCPR required to avoid violating the safety limit should this limiting transient occur.

Table 4-1

DESCRIPTION OF UNCERTAINTIES

Quantity	Standard Deviation (% of Point)	Comment
Feedwater Flow	1.76	This is the largest component of total reactor power uncertainty.
Feedwater Temperature	0.76	These are the other significant parameters in core power determination
Reactor Pressure	0.5	
Core Inlet Temperature	0.2	Affect quality and boiling length.
Core Total Flow	2.5	Flow is not measured directly, but is calculated from jet pump ΔP. The listed uncertainty in total core flow corresponds to 11.2% standard deviation in each individual jet pump flow.
Channel Flow Area	3.0	This accounts for manufacturing and service induced variations in the free flow area within the channel.

Table 4-1 (Continued)

Quantity	Standard Deviation (% of Point)	Comment
Friction Factor Multiplier	10.0	Accounts for uncertainty in the correlation representing two-phase pressure losses.
Channel Friction Factor Multiplier	5.0	Represents variation in the pressure loss characteristics of individual channels. Flow area and pressure loss variations affect the core flow distribution, influencing the quality and boiling length in individual channels.
TIP Readings	8.7	These sets of data are the base from which gross power distribution is determined. The assigned uncertainties include all electrical and geometrical components plus a contribution from the analytical extrapolation from the chamber location to the adjacent fuel assembly segment. Also included are uncertainties contributed by the LPRM system. LPRM readings are used to correct the power distribution calculations for changes which have occurred since the last TIP survey. The assigned uncertainty affects power distribution in the same manner as the base TIP reading uncertainty.
R Factor	1.6	This is the last of the three-power distribution-related uncertainties. It is a function of the uncertainty in local fuel rod power and is discussed in detail in Reference 2.
Critical Power	3.6	Uncertainty in the GEXL correlation in terms of critical power.

Table 4-2

SUMMARY OF RESULTS OF LIMITING ABNORMAL OPERATIONAL TRANSIENTS

	ΔCPR							
	BOC to EOC-2		EOC-2 to EOC-1		EOC-1 to EOC-.5		EOC-.5 to EOC	
	7x7	8x8	7x7	8x8	7x7	8x8	7x7	8x8
Load Rejection w/o bypass	0.06	0.10	0.16	0.23	0.20	0.28	0.24	0.32
Turbine Trip w/o bypass	0.05	0.09	0.15	0.22	0.19	0.26	0.23	0.30
Rod Withdrawal Error* (RBM AT 105%)	0.16	0.11	0.16	0.11	0.16	0.11	0.16	0.11
Loss of Feedwater* Heating (100°F)	0.14	0.16	0.14	0.16	0.14	0.16	0.14	0.16
Feedwater Controller Failure	0.04	0.05	0.09	0.13	0.12	0.18	0.16	0.22

*EOC analysis applied throughout cycle.

4.2.1 Operating Limit MCPR

Based on the fuel cladding integrity safety limit and the results of the abnormal operational transient analyses, the operating limit MCPR is given in Table 4-3.

4.3 TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

The magnitude of values used as initial input conditions for the transient analysis is given in Table 4-4.

Table 4-3

MCPR OPERATING LIMIT

Applicable Exposure Range	7x7	8x8
BOC4 to 2GWd/t before EOC4	1.22	1.22
From 2 GWd/t to 1 GWd/t before EOC4	1.22	1.29
From 1 GWd/t to 0.5 GWd/t before EOC4	1.26	1.34
From 0.5 GWd/t to EOC4	1.30	1.38

Table 4-4

GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

	BOC4 to EOC4-2 GWd/t		EOC4-2 GWd/t to EOC4-1 GWd/t		EOC4-1 GWd/t to EOC4-0.5 GWd/t		EOC4-0.5 GWd/t to EOC4	
	7x7	8x8	7x7	8x8	7x7	8x8	7x7	8x8
Peaking Factors (local, radial)	(1.24,1.29)	(1.22,1.445)	(1.24,1.26)	(1.22,1.345)	(1.24,1.235)	(1.22,1.305)	(1.25,1.20)	(1.22,1.268)
R-Factor	1.100	1.098	1.100	1.098	1.100	1.098	1.100	1.098
Bundle Power, MWt	5.478	6.134	5.351	5.708	5.245	5.538	5.096	5.381
Bundle Flow, 10^3 lb/hr	124.2	111.7	125.0	114.4	125.7	115.5	126.7	116.5
Initial MCPR	1.20	1.20	1.23	1.30	1.26	1.34	1.30	1.38

Core Power, MWt	1593.0
Core Flow, Mlb/hr	49.0
Reactor Pressure, psia	1035.0
Inlet Enthalpy, Btu/lb	526.3
Non-Fuel Power Fraction	0.04
Axial Peaking Factor	1.40

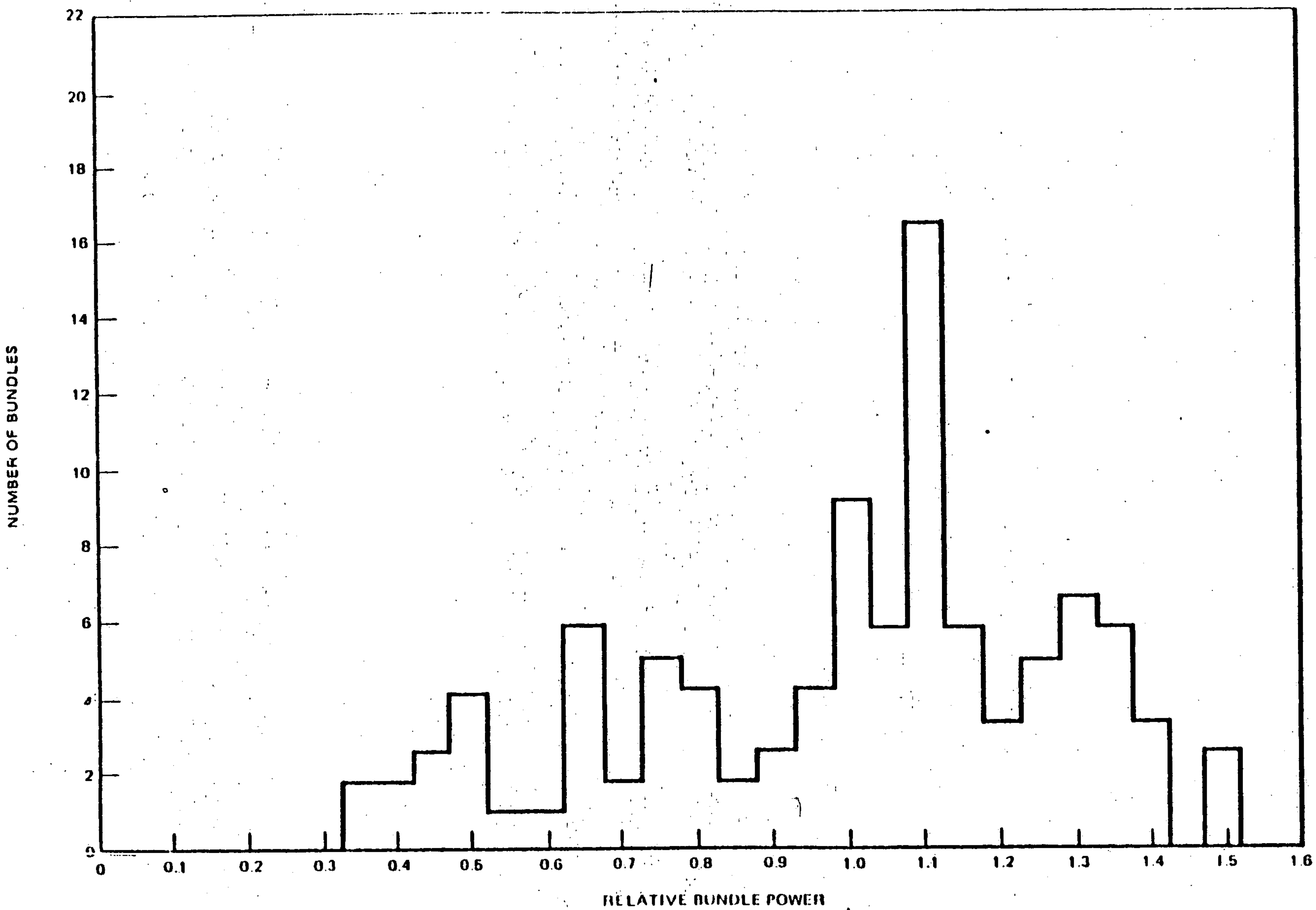


Figure 4-1. Relative Bundle Power Histogram for a Typical BWR Operating Power Distribution

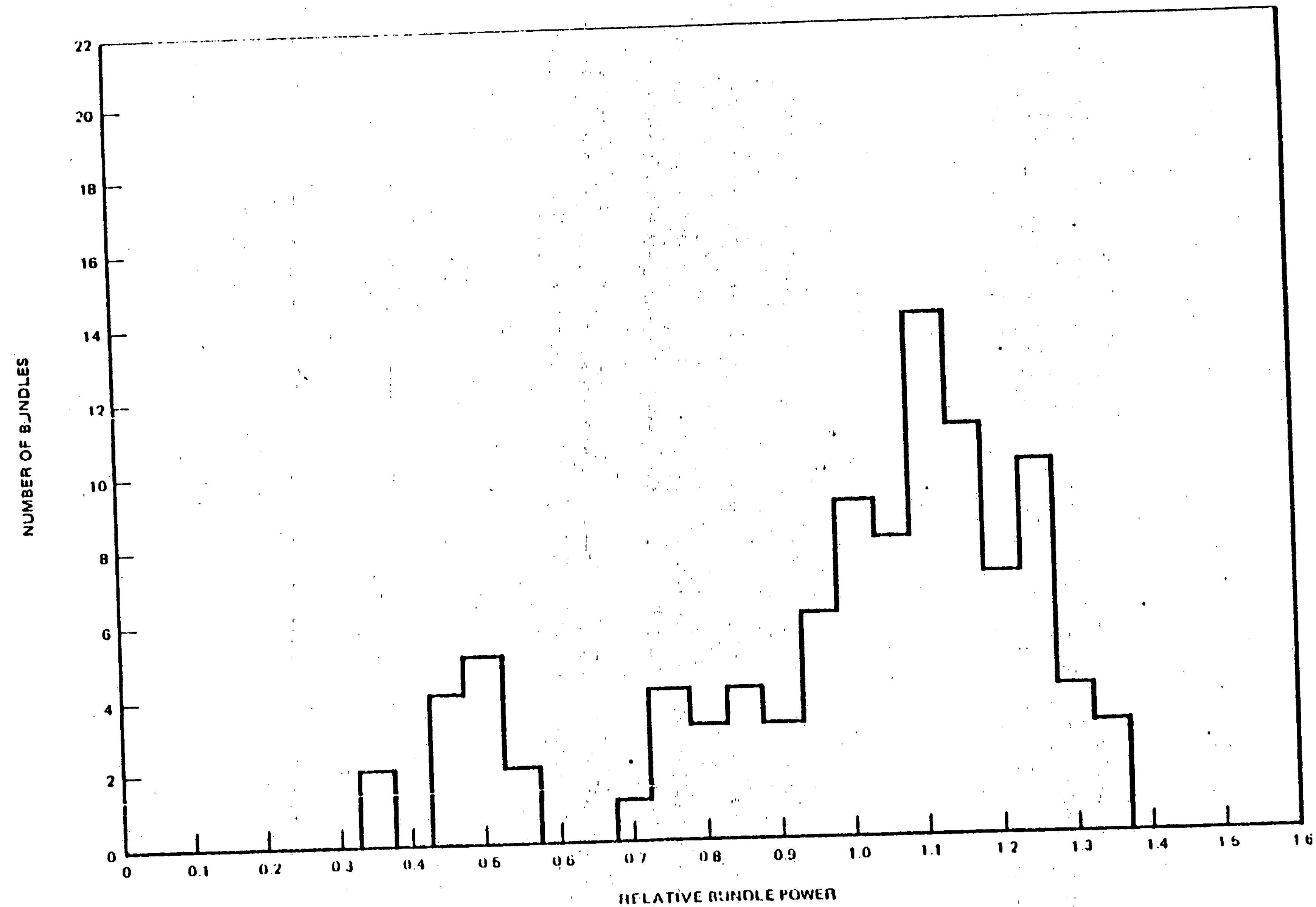


Figure 4-2. Relative Bundle Power Histogram for a Typical BWR Operating Power Distribution

5. NUCLEAR CHARACTERISTICS

The bundle characteristics, analytical methods, and model descriptions presented in Sections 5.1 through 5.4 of Reference 1 are applicable to this reload. Results of specific reload core calculations are given below.

5.1 NUCLEAR CHARACTERISTICS OF THE CORE

This section presents the results of the calculation on:

1. reactivity control characteristics; and

2. core average reactivity coefficients.

The core characteristics were calculated using the design reference loading pattern shown in Figure 2-1. The loading pattern was designed to accommodate 88 Reload-3 fuel bundles by discharging a like number of fuel bundles from the Cycle 3 core. Twenty eight initial core type bundles discharged at EOC-1B or EOC-2 will be re-inserted to replace a like number of bundles of the same type.

5.1.1 Core Effective Multiplication, Control System Worth and Reactivity Coefficients

A tabulation of the typical nuclear characteristics of the reconstituted core is given in Table 5-1. The nuclear characteristics of the Reload-3 fuel bundles are similar or identical to those previously loaded. Therefore, the total control system worth and the temperature and void dependent behavior of the reconstituted core will not differ significantly from those values previously reported.

5.1.2 Reactor Shutdown Margin

The reconstituted core fully meets the established technical specification criteria in that it may be maintained subcritical by at least 0.38% Δk in the most reactive condition throughout the subsequent operating cycle with the strongest control rod fully withdrawn and all other rods fully inserted.

A minimum shutdown margin of 0.012 Δk calculated for the assumed refueling at a previous cycle exposure increment of 4940 MWd/t is the most reactive condition throughout the subsequent operating cycle with the strongest control rod fully withdrawn and all other rods fully inserted. The BOC 4 shutdown margin is 0.012 Δk. Thus, R, the difference between the BOC and the minimum shutdown margin, is 0.0 Δk. Due to the relatively large shutdown margin calculated, the Cycle 3 exposure increment can be as low as 4700 MWd/t at the refueling outage.

5.1.3 Standby Liquid Control System

A boron concentration of 600 ppm in the moderator water will bring the reactor subcritical by $\geq$ 0.03 Δk at 20°C, xenon free.

Table 5-1

NUCLEAR CHARACTERISTICS OF THE DESIGN REFERENCE CODE

Core Effective Multiplication and Control System Worth (No Voids, 20°C)	
BOC k_{eff}	
Uncontrolled	1.119
Fully Controlled	0.957
Strongest Control Rod Out	0.988
R, Maximum Increase in Core Reactivity with Exposure into Cycle, Δk	0.0
Reactivity Coefficients, Range During Operating Cycle	
Steam Void Coefficient at average Voids; $(\Delta k/k)/\Delta V$, 1/%Void	-1.270×10^{-3} to -1.225×10^{-3}
Least Negative Power Coefficient at Rated Conditions $(\Delta k/k)/(\Delta P/P)$	-0.053
Fuel Temperature Coefficient at 650°C $(\Delta k/k)/\Delta T$, 1/°F)	-1.228×10^{-5} to -1.268×10^{-5}

6. SAFETY ANALYSIS

6.1 INTRODUCTION

The safety analysis for reloads consists of three categories: (a) generic safety analysis, which is applicable to all reloads; (b) bounding analysis; and (c) specific analysis applicable only to the current reload. Wherever a bounding analysis is applied for an accident or transient, the key parameters need only to be compared with the worst case and, if they are within "bounds," all limits and margins applicable to the accidents or transients will be met.

6.2 MODEL APPLICABILITY TO 8x8 FUEL

Information on the applicability to the 8x8 design of existing models used for safety analyses is given in Reference 1.

6.3 RESULTS OF SAFETY ANALYSES

6.3.1 Core Safety Analyses

The General Electric Thermal Analysis Basis (GETAB) (Reference 2) is used to establish thermal margins in reload cores. The operating limits, margins, and fuel damage limits previously used are applicable to this reload. Where necessary, further discussions of these and other controlling factors are presented below.

6.3.2 Accident Analyses

6.3.2.1 Main Steamline Break Accident

The consequences of the mean steam line break analysis depend on the basic thermal-hydraulic parameters of the overall reactor, as discussed in Reference 1. Because these parameters do not normally change as a result of reload, the referenced analysis applies.

6.3.2.2 Refueling Accident

The description and analyses of the refueling accident provided in the DAEC FSAR and discussed in Reference 1 apply to this reload. The factors involved are such that the conclusions of these evaluations remain valid.

6.3.2.3 Control Rod Drop Accident

The technical bases (bounding analyses) which are presented in Reference 1 were used to verify that the results of a rod drop excursion in the reloaded core would not exceed the design criteria. For application to Duane Arnold Energy Center Reload-3, the actual Doppler coefficient, accident reactivity shape functions and scram reactivity functions are compared with the technical bases in Figures 6-1 through 6-5. Since all values were not within bounding limits, a plant specific analysis has been performed and the results indicate the consequences of a rod-drop excursion from any in-sequence control rod would be below the 280 cal/gm design limit. Further, the radiological consequences will be no greater than those evaluated in Reference 1.

6.3.2.4 Loss-of-Coolant Accident

The Loss of Coolant Accident (LOCA) analysis for Duane Arnold Reload-2 (Reference 3) is to be applied to Reload-3. Since the Reload-3 core will contain more drilled bundles than the Reload-2 core, the previous LOCA analysis is conservative when applied to the Reload-3 core. No new fuel types will be inserted into the Reload-3 core, therefore, new MAPLHGR's are not required.

6.3.2.5 Loading Error Accident

6.3.2.5.1 Event Description

A loading error for the reference core configuration is defined as:

(1) a reload bundle is rotated 180 degrees in a location near the center of the core or a bundle is inserted in an improper location; and

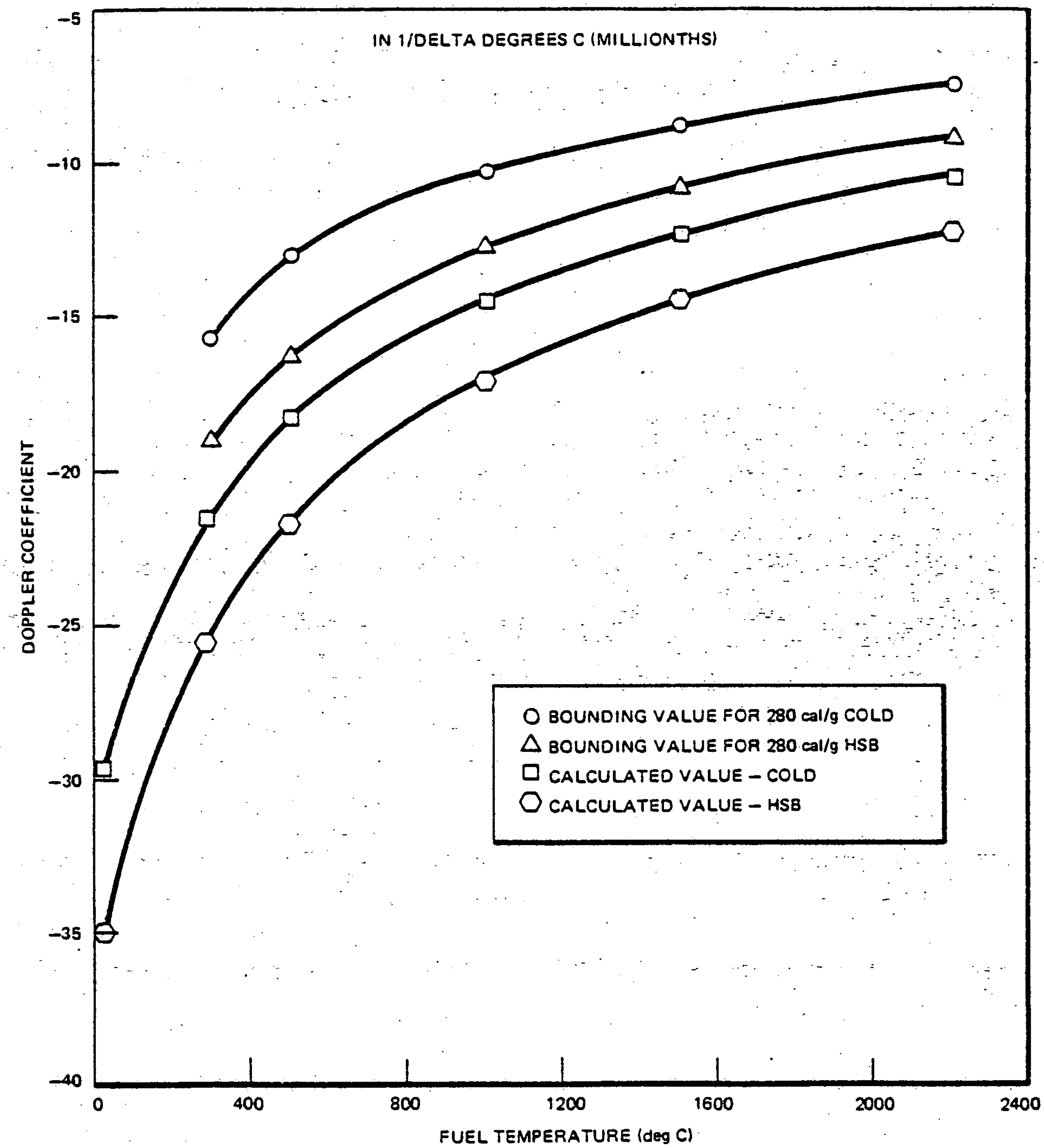


Figure 6-1. Fuel Doppler Coefficient

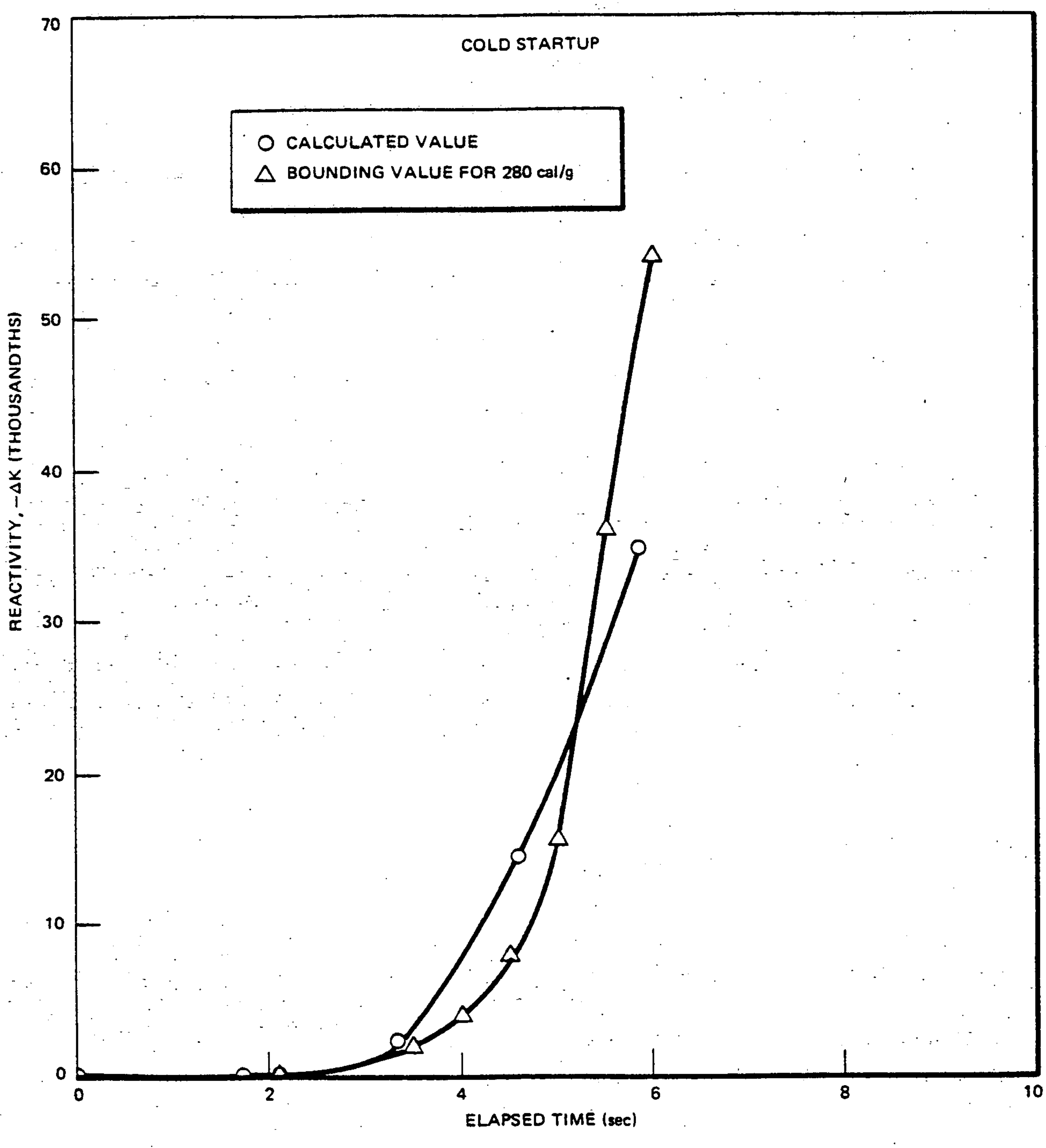


Figure 6-2. Scram Reactivity Function

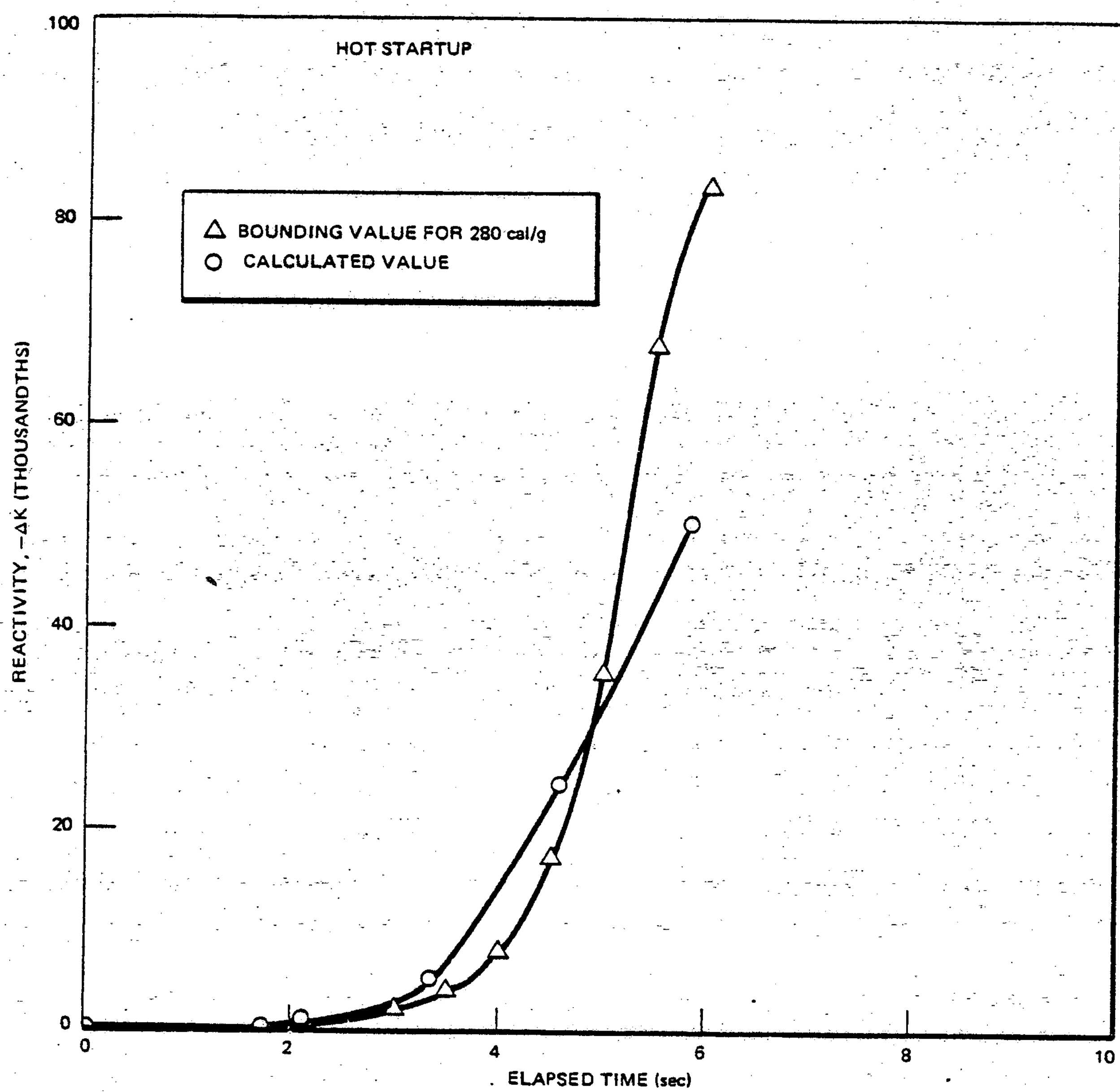


Figure 6-3. Scram Reactivity Function

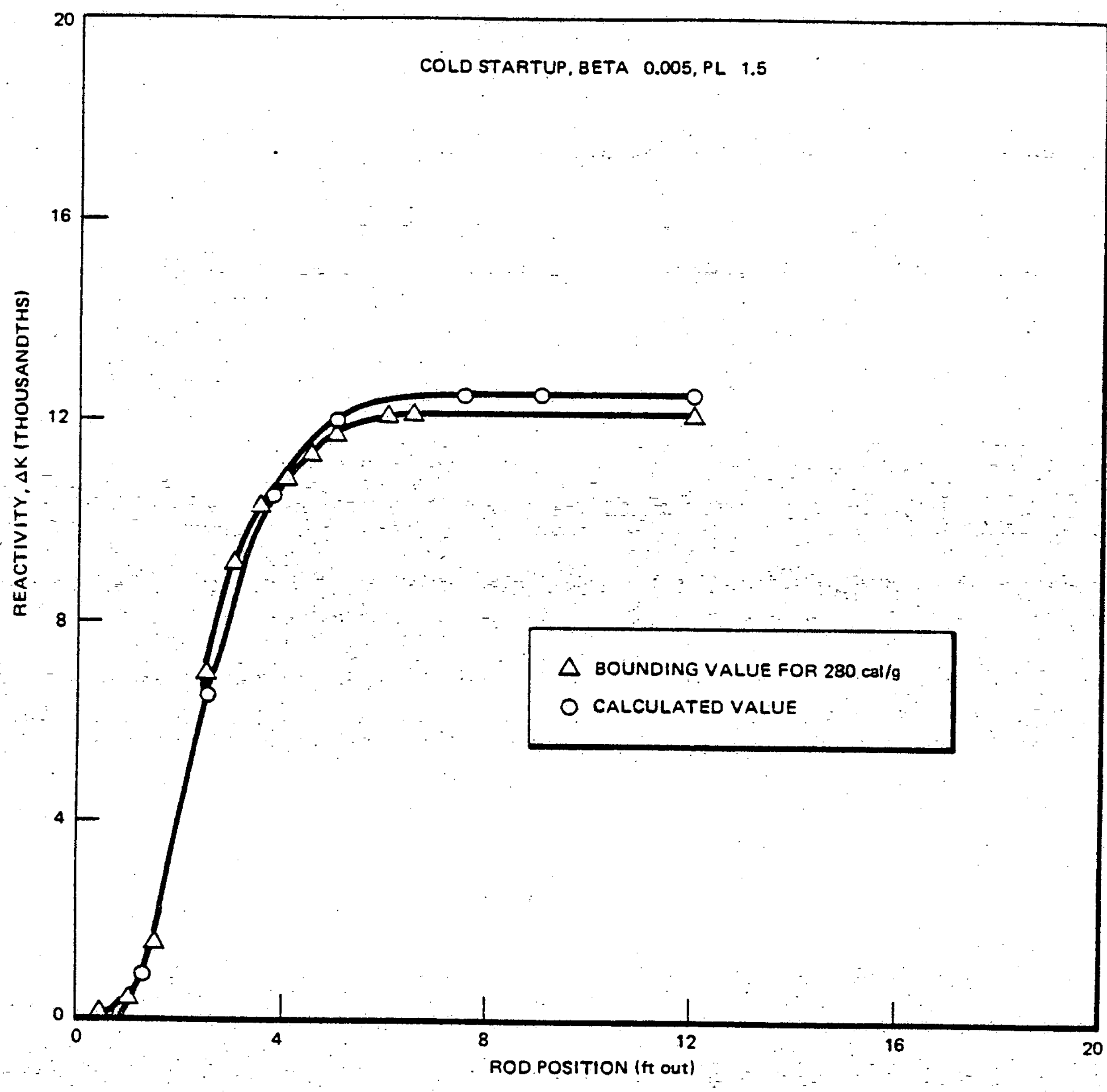


Figure 6-4. Accident Reactivity Shape Function

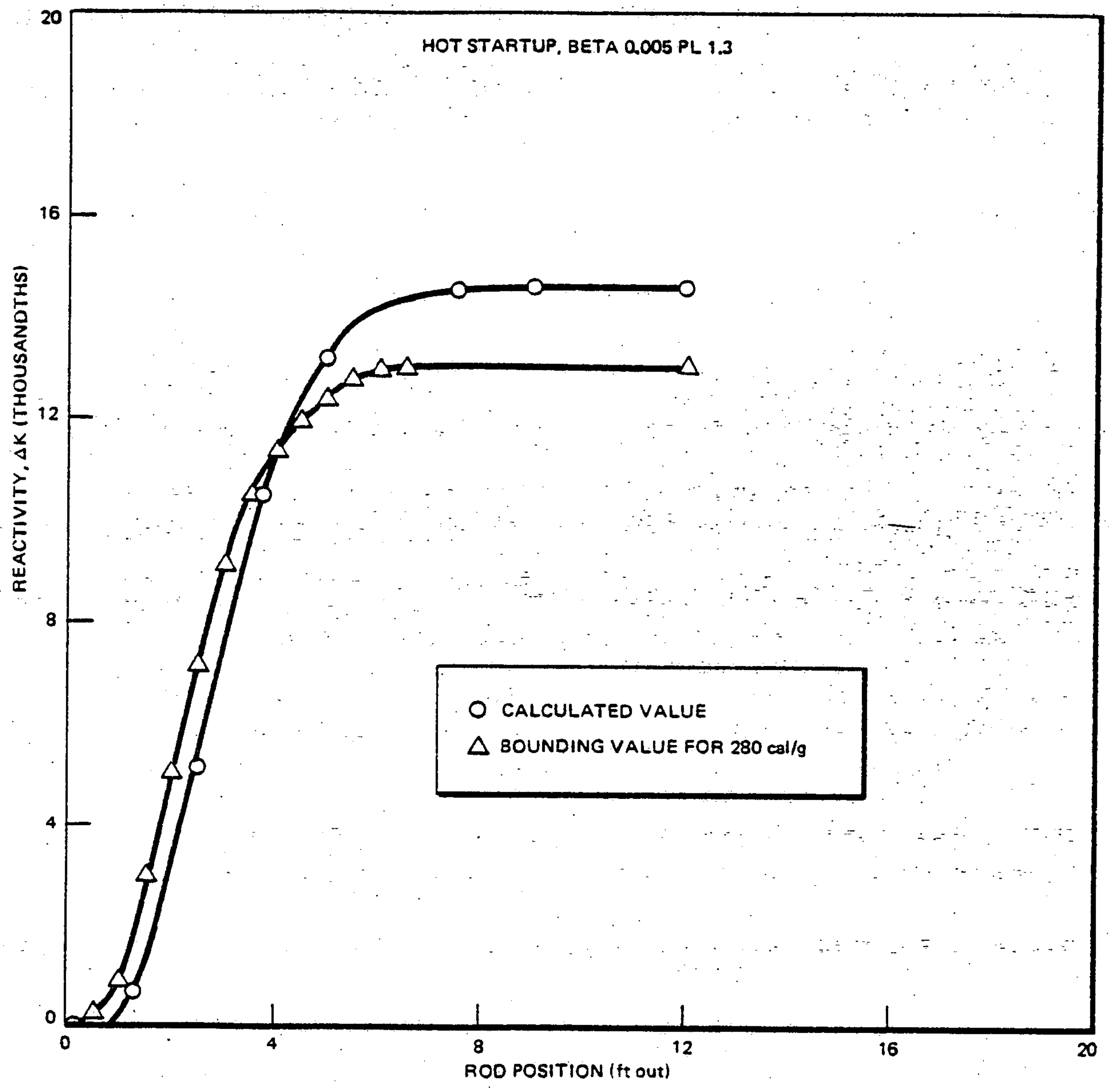


Figure 6-5. Accident Reactivity Shape Function

(2) the error is not discovered in the subsequent core verification and the reactor is operated.

Since two independent errors are assumed to occur, the single-error criterion is violated, so the event is not classified as an abnormal operational transient. The following are the results and consequences for a worst-case error.

6.3.2.5.2 Results and Consequences

Analysis of the loading error accident results in a peak linear heat generation rate (LHGR) of 16.5 kW/ft and a minimum critical power ratio (MCPR) of 1.02* for the rotated bundle. The misplacing of an 8x8 bundle in a 7x7 location results in a peak LHGR of 18.6 kW/ft and an MCPR of 1.02*. The LHGR's are below the value at which 1% plastic strain of the cladding occurs. Fuel damage is not expected to occur with a LHGR lower than that needed to cause a 1% plastic strain in the cladding (see Section 3.2.1 of Reference 1). Therefore, fuel failure is not expected for this event.

Fuel bundles adjacent to the misloaded bundle are insignificantly affected by the presence of the misloaded bundle.

6.3.3 Abnormal Operating Transients

6.3.3.1 Transients and Core Dynamics

6.3.3.1.1 Analysis Basis

This subsection contains the analyses of the most limiting abnormal operational transients for DAEC, Cycle 4. All transients which are the basis of the existing license were reviewed, and those transients which have been limiting in the past with respect to safety margins and are significantly sensitive to the core transient parameter deviations were reanalyzed.

* from an initial MCPR of 1.22

6.3.3.1.2 Input Data and Operating Conditions

The input data and operating conditions are shown in Tables 6-1a and 6-1b. Each transient is considered at these conditions unless otherwise specified.

6.3.3.1.3 Transient Summary

A summary of the transients analyzed and their consequences is provided in Table 6-2.

6.3.3.2 Transient Descriptions

The abnormal operating transients which are limiting according to safety criteria and which also are sensitive to nuclear core parameter changes have been analyzed and are evaluated in the following narrative.

6.3.3.2.1 Generator Load Rejection with Failure of the Bypass Valves

This transient produces the most severe reactor isolation. The primary characteristic of this transient is a pressure increase due to the obstruction

Table 6-1a

TRANSIENT INPUT PARAMETERS

Thermal Power	(MWt)	1657	104%
Steam Flow	(lb/hr)	7.18×10^6	105%
Rated Core Flow	(lb/hr)	49.0×10^6	100%
Dome Pressure	psig	1020	
Turbine Pressure	psig	960	
RV Set Point	psig	1090	
RV/Type/Capacity (at Set Point)	No./Type/%NBR	6/Target Rock/72.0	
RV Time Delay	(msec)	400	
RV Shroke Time	(msec)	100	
SV Set Point	psig	1240	
SV Capacity	No./%NBR	2/18.9	

Table 6-1b

TRANSIENT INPUT PARAMETERS

Parameter	Units	Exposure	Exposure	Exposure	Exposure
Cycle Exposure	GWD/T	EOC4	EOC4-0.5	EOC4-1.	EOC4-2.
Void Coeff	$-¢/\%R_g$				
NDP		9.35	9.68	9.89	9.50
TAP		11.69	12.10	12.37	11.88
Doppler Coeff	-¢/°F				
NDP		0.2302	0.2273	0.2240	0.2170
TAP		0.2187	0.2159	0.2128	0.2062
Average Fuel Temperature	°F	1359	1359	1359	1359
Scram Worth	-$				
NDP		39.01	38.92	38.58	37.49
TAP		31.21	31.14	30.86	29.99
CRD Spec		67B	67B	67B	67B
Scram Reactivity		Fig 6-6a	Fig 6-6b	Fig 6-6c	Fig 6-6d

NDP - Nuclear Dynamic Parameter
TAP - Transient Analysis Parameter

of steam flow by the turbine control valves. The pressure increase causes a significant void reduction which yields a pronounced positive void reactivity effect. The net reactivity is sharply positive and causes a rapid increase in neutron flux until the net reactivity is forced negative by scram initiated from closure of the turbine control valves and by a void increase after the safety-relief valves have automatically opened on high pressure. Figures 6-7a through 6-7d illustrate this transient.

The parameters of concern are the peak vessel pressure margin to the ASME Pressure Vessel Code Limit and the peak average surface heat flux correlated to MCPR. These are given in Table 6-2.

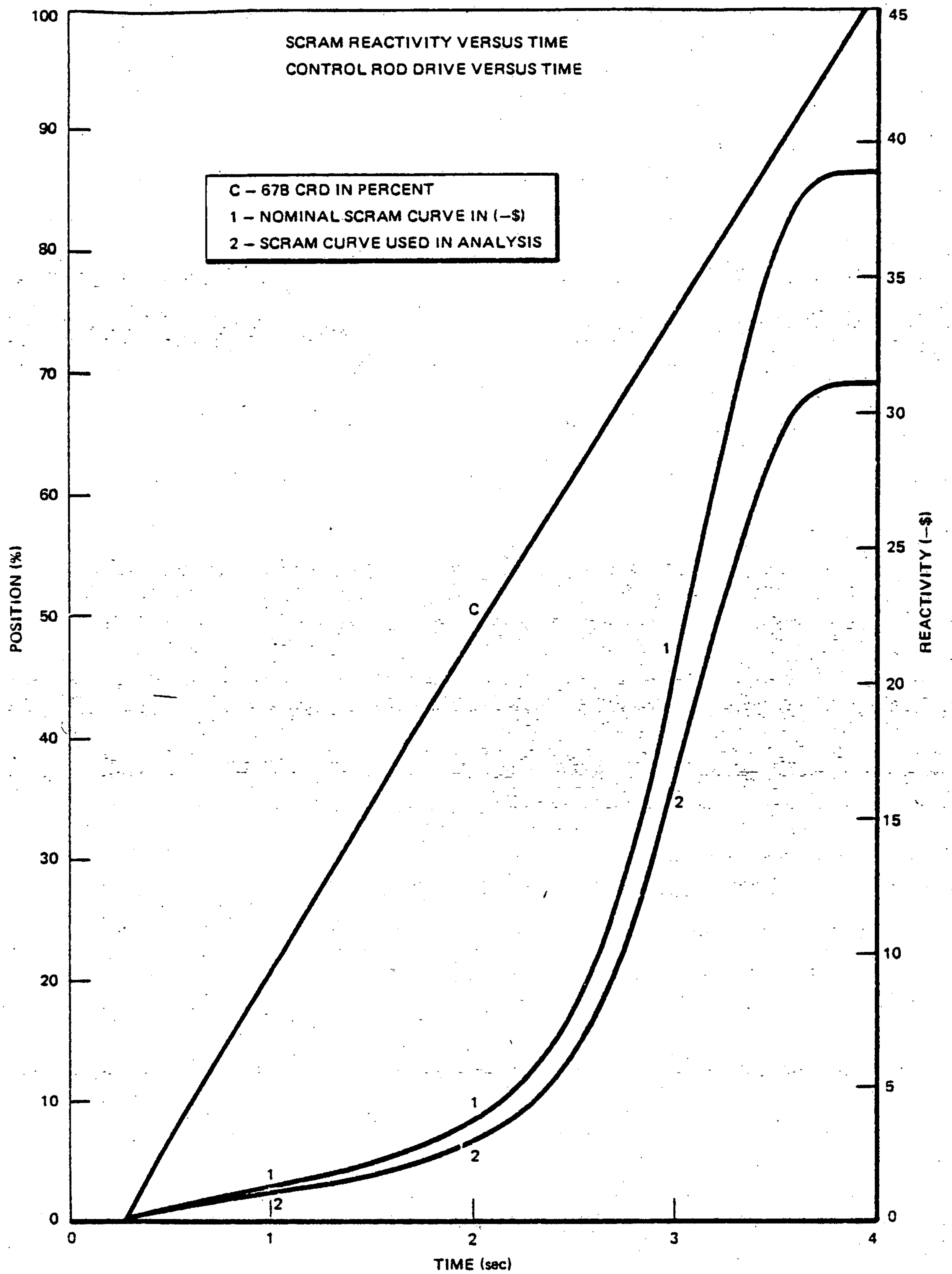


Figure 6-6a. Duane Arnold EOC4 GWd/t

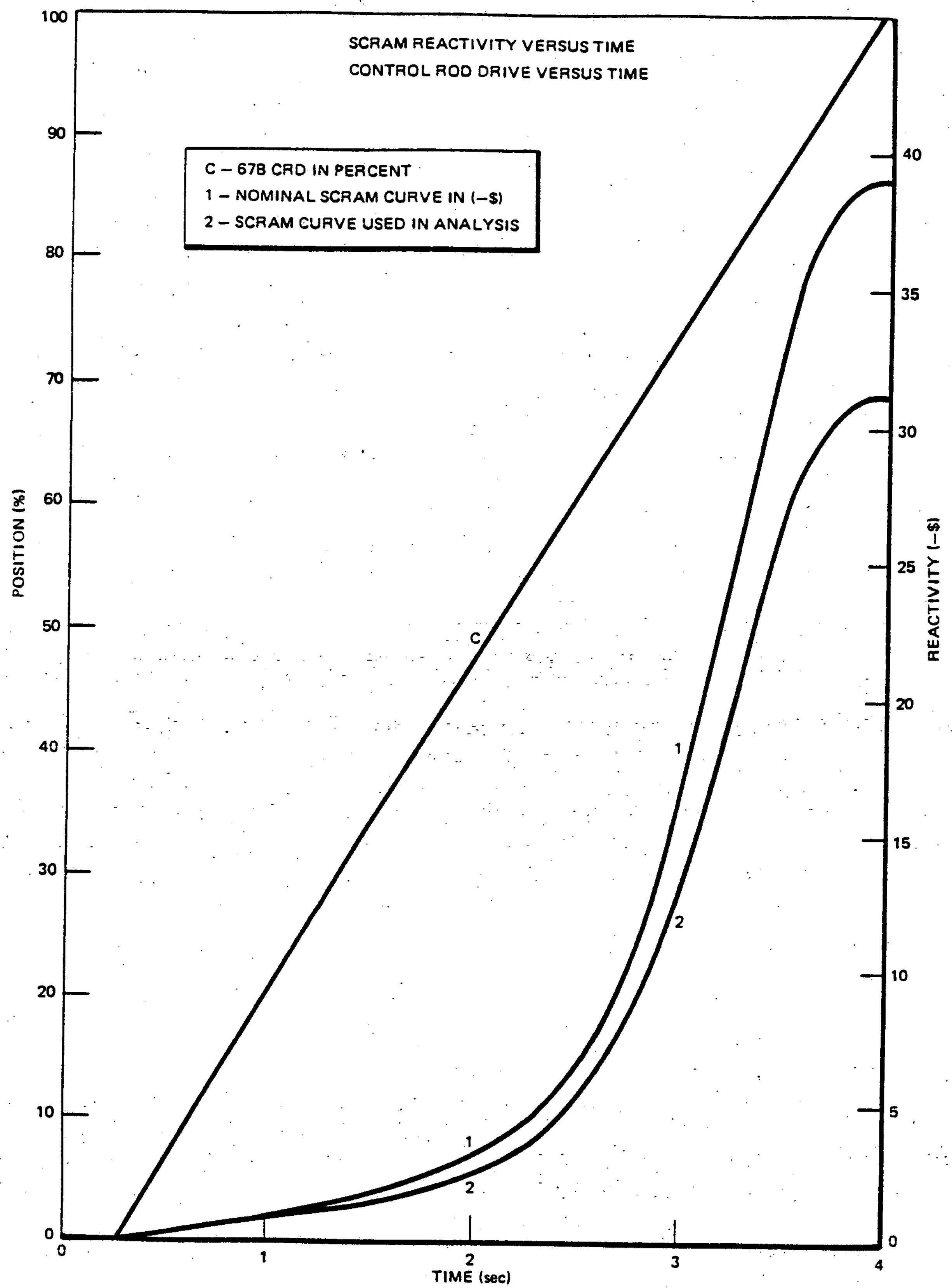


Figure 6-6b. Duane Arnold EOC4-0.5 GWd/t

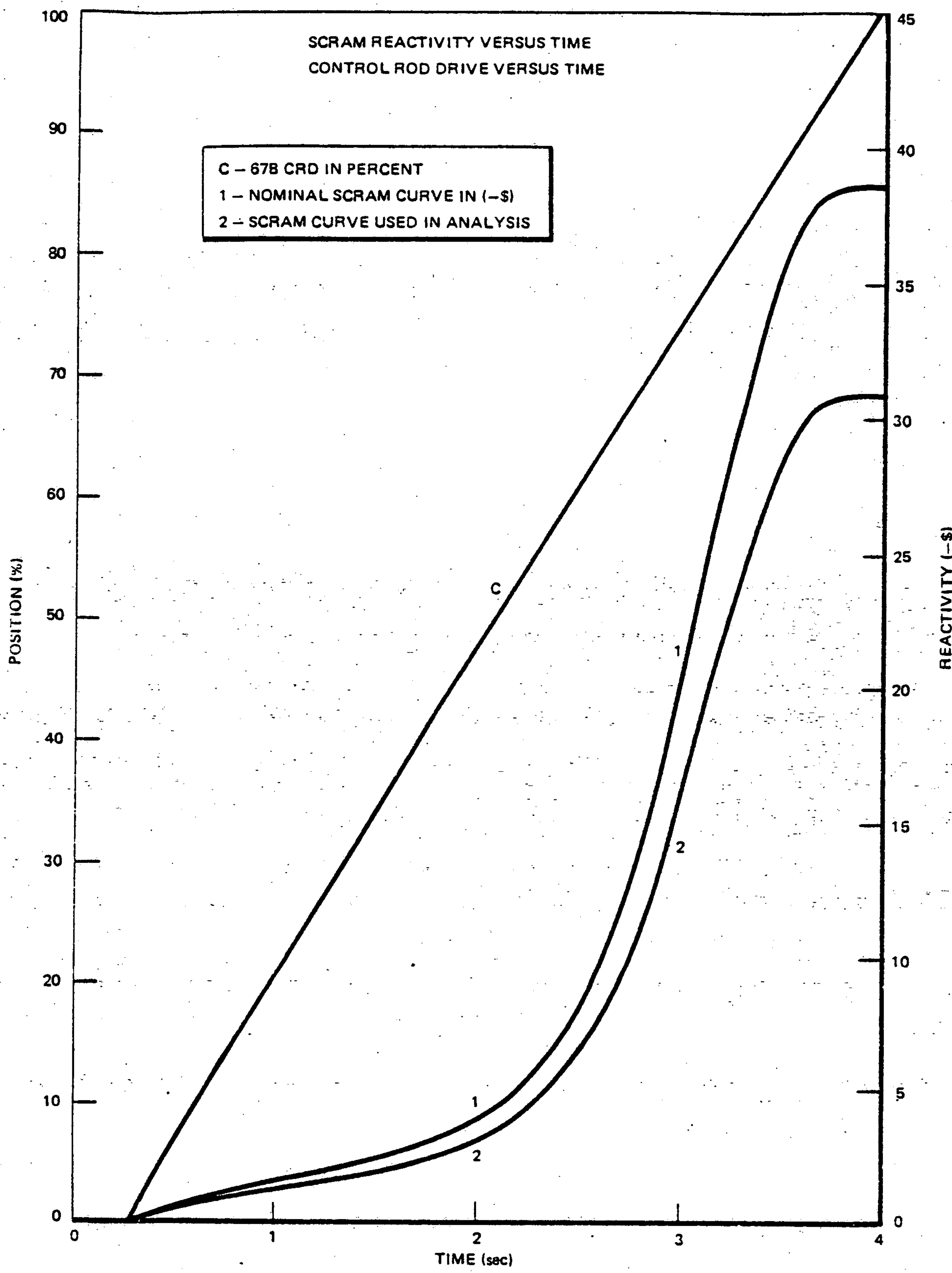


Figure 6-6c. Duane Arnold EOC4-1 GWd/t

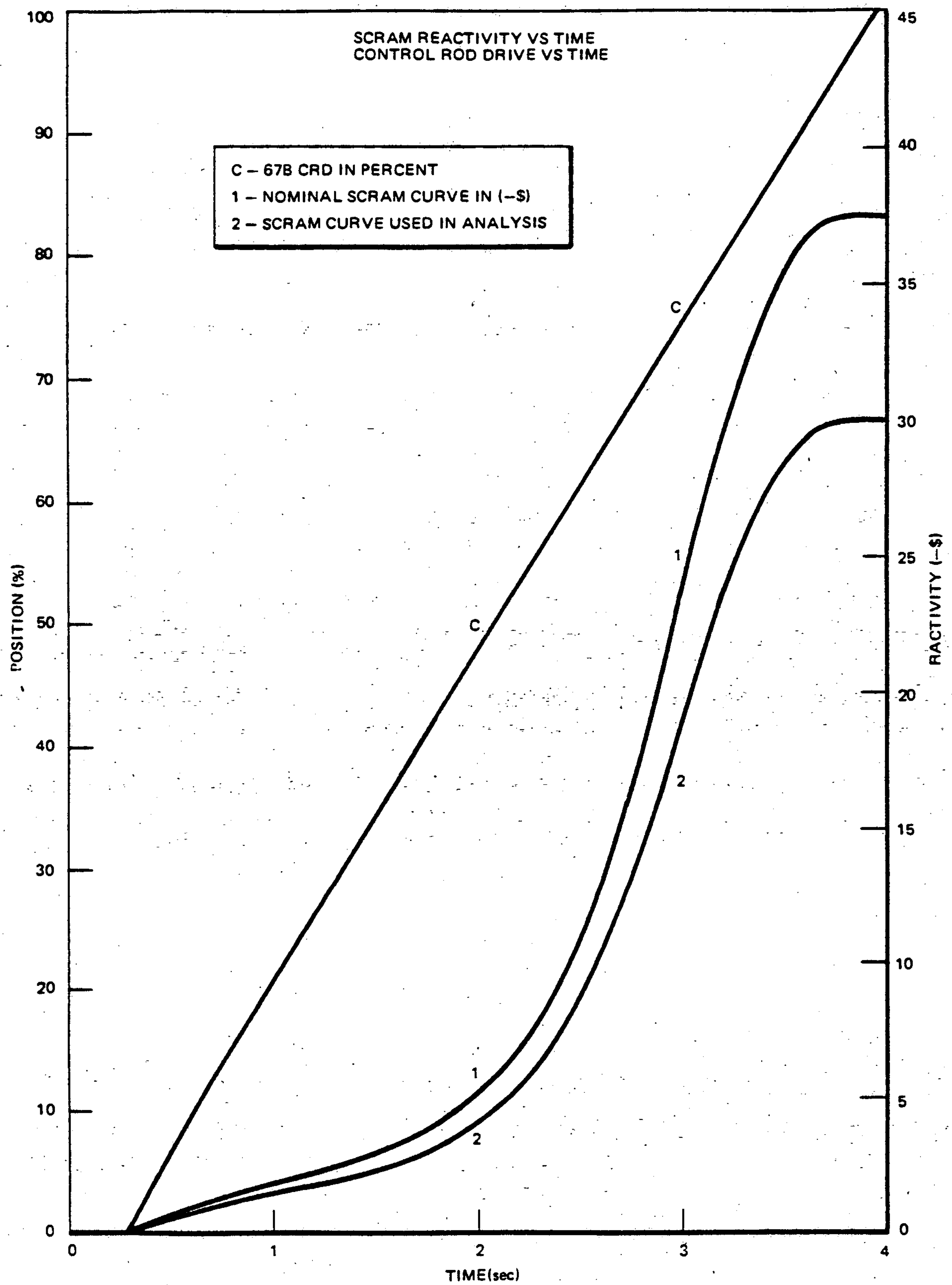


Figure 6-6d. Duane Arnold EOC4-2 GWd/t

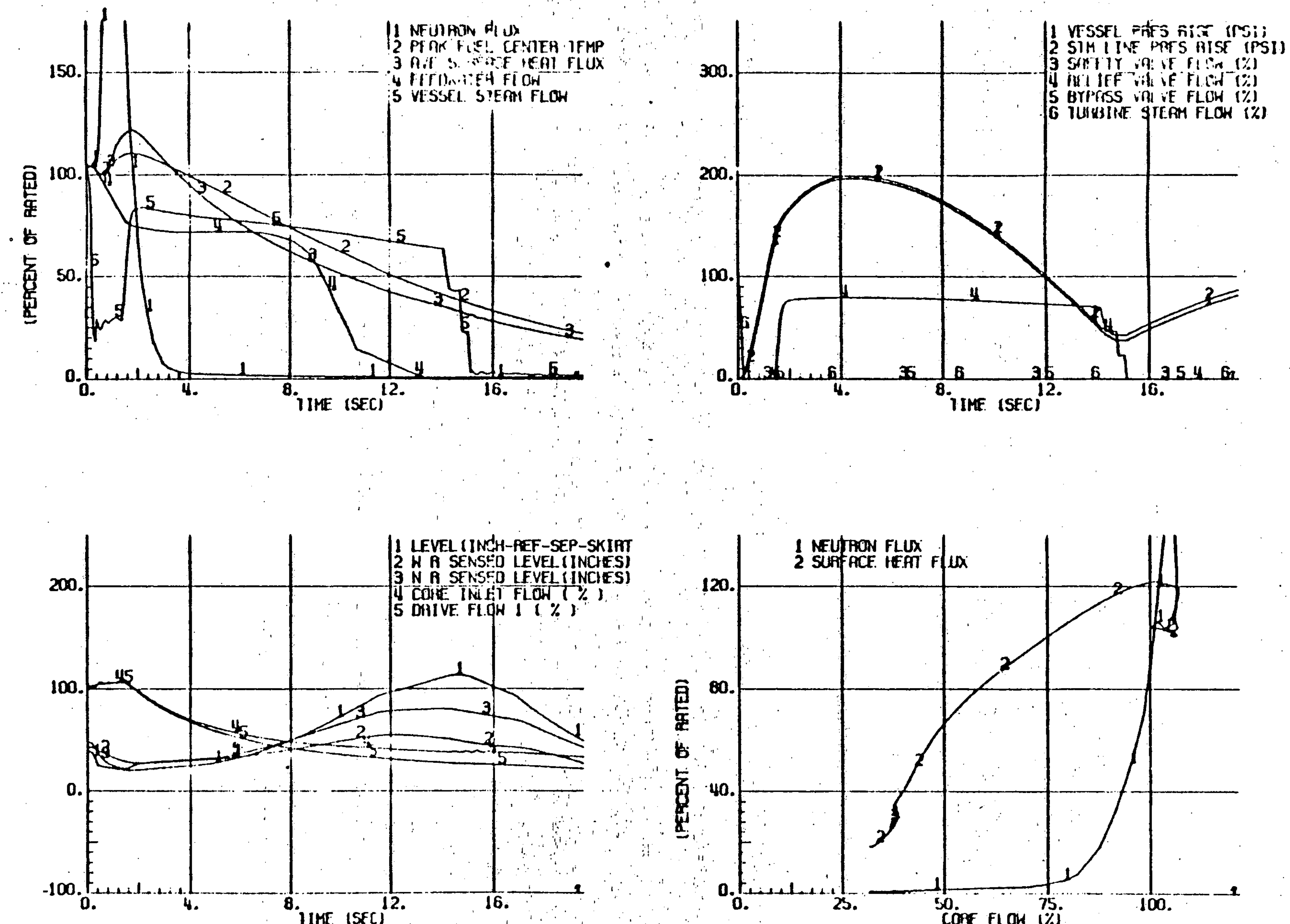


Figure 6-7a. Generator Load Rejection, Without Bypass EOC4

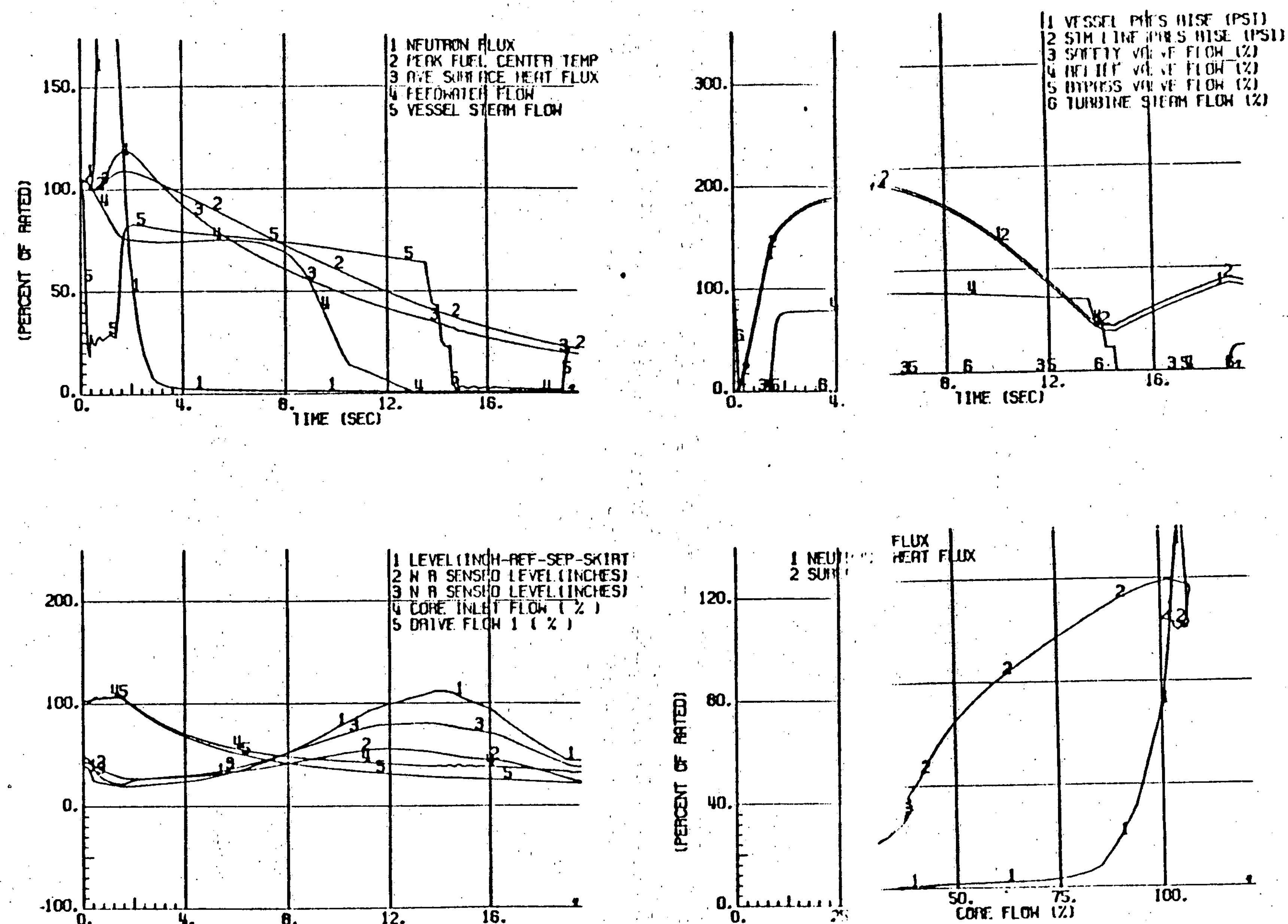


Figure 6-7b. Generator Load Rejection, Without Bypass EOC4-0.5 GWd/t

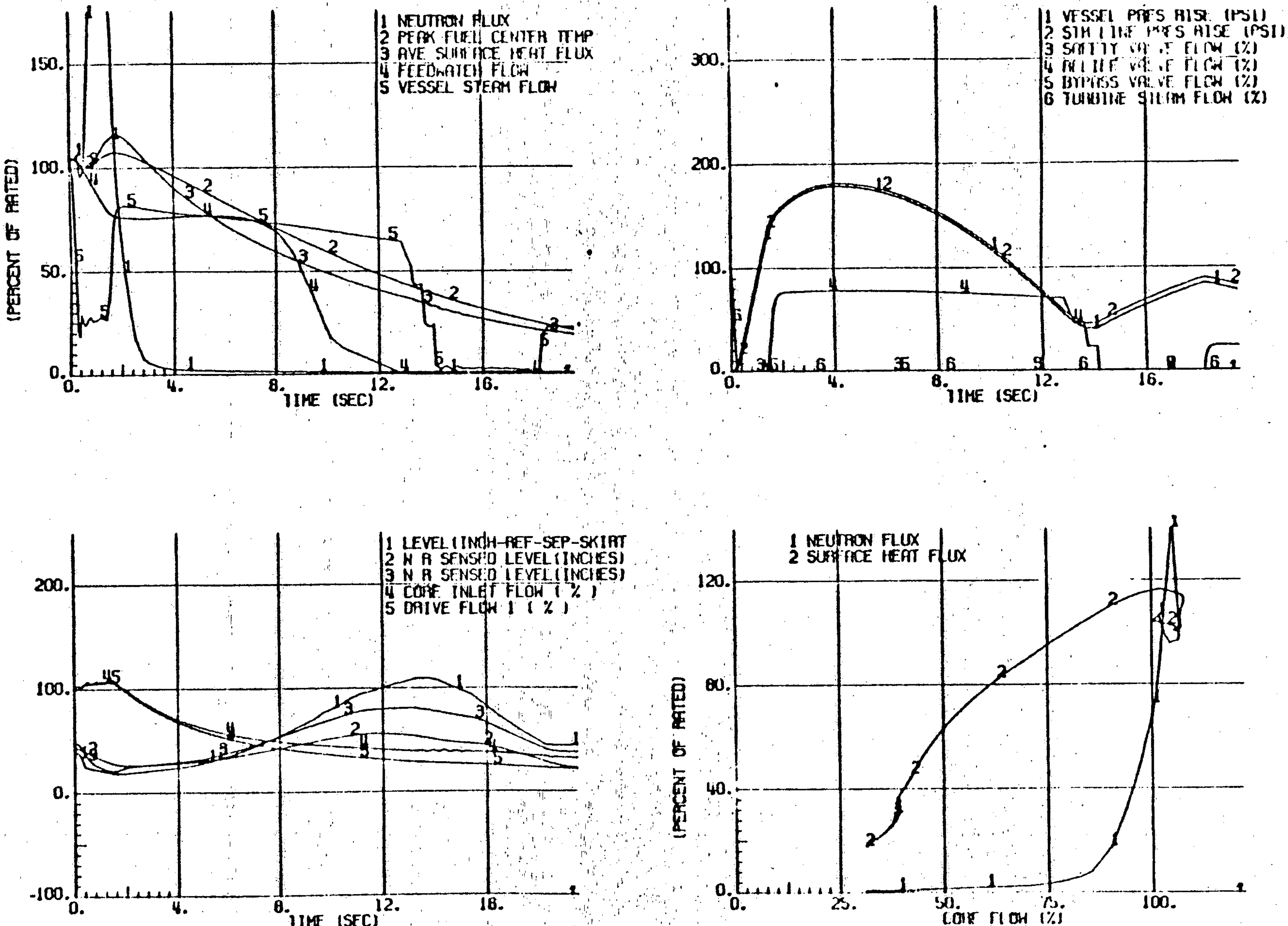


Figure 6.7c. Generator Load Rejection, Without Bypass EOC4-1.0 GWd/t

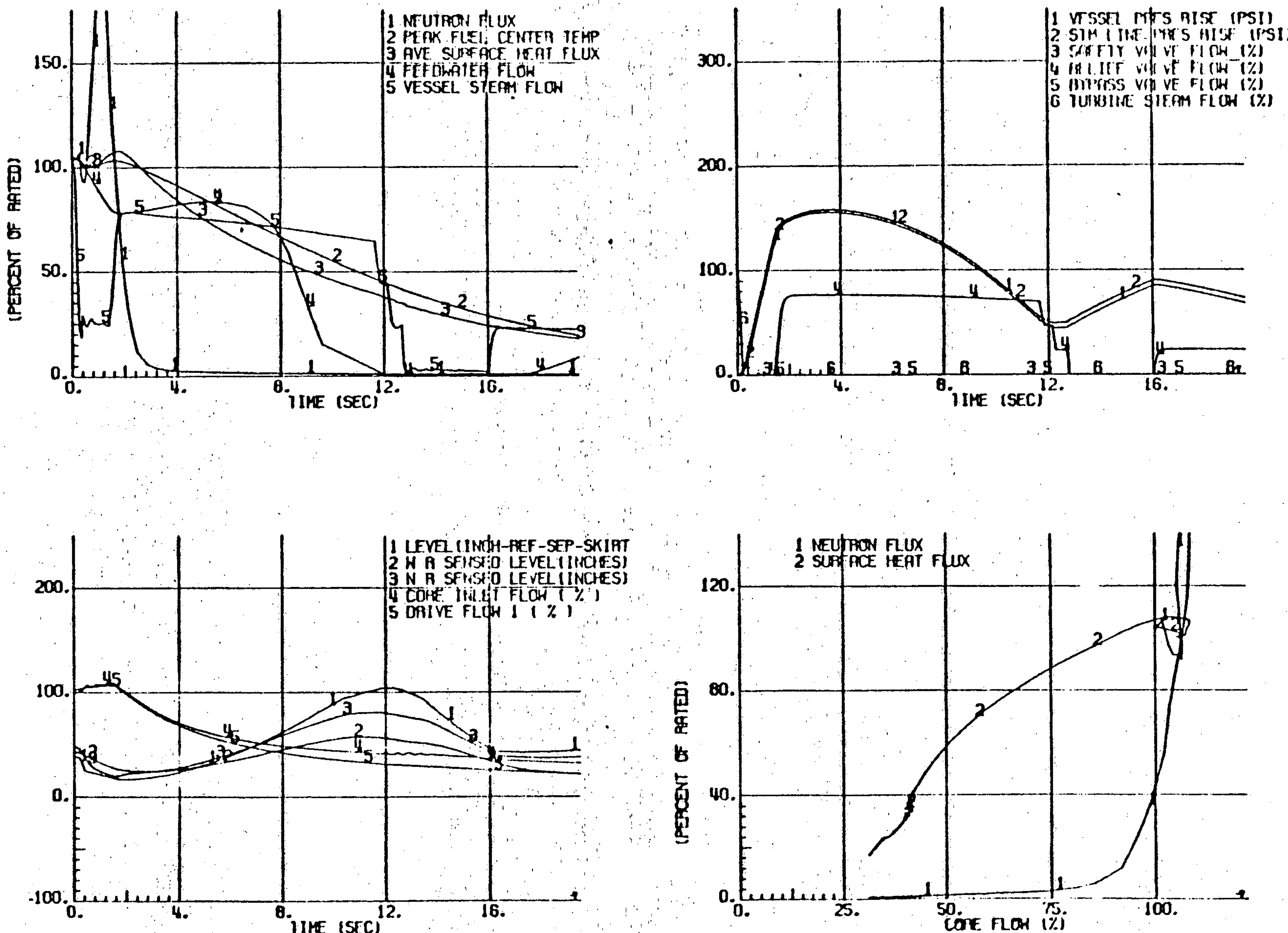


Figure 6-7d. Generator Load Rejection, Without Bypass EOC4-2.0 GWd/t

Table 6-2

DAEC CYCLE 4 TRANSIENT SUMMARY

	Pwr %	Core Flow %	Exposure Before EOC4	$\hat{\phi}$% Init	$\hat{Q}$/A% Init	$\hat{P}_{SL}$ Psig	$\hat{P}_V$ Psig
Load Rejection w/o Bypass	104	100	EOC4	386.5	117.3	1215	1240
Load Rejection w/o Bypass	104	100	EOC4-0.5	355.9	114.7	1204	1231
Load Rejection w/o Bypass	104	100	EOC4-1.0	305.0	111.9	1196	1223
Load Rejection w/o Bypass	104	100	EOC4-2.0	194.7	103.8	1173	1205
Feedwater Controller Failure	104	100	EOC4	221.5	112.5	1147	1193
Feedwater Controller Failure	104	100	EOC4-0.5	195.8	110.1	1145	1190
Feedwater Controller Failure	104	100	EOC4-1.0	165.0	107.2	1142	1187
Feedwater Controller Failure	104	100	EOC4-2.0	118.4	104.5	1138	1182
Loss of Feedwater Heater	104	100	EOC4	120.7	118.6	1023	1071
Turbine Trip w/o Bypass	104	100	EOC4	351.3	115.7	1212	1237
Turbine Trip w/o Bypass	104	100	EOC4-0.5	320.3	113.1	1202	1228
Turbine Trip w/o Bypass	104	100	EOC4-1.0	273.5	110.4	1194	1221
Turbine Trip w/o Bypass	104	100	EOC4-2.0	179.0	102.7	1172	1204

6.3.3.2.2 Loss of a Feedwater Heater

The loss of a feedwater heater is analyzed in FSAR's and other submittals because it constitutes the most limiting cool water transient.

A feedwater heater can be lost if the steam extraction line to the heater is shut and the heat supply to the heater is removed, producing a gradual cooling

of the tubes. The reactor will receive cooler feedwater flow, which will produce an increase in core inlet subcooling and, due to the negative void reactivity coefficient, an increase in core power.

Figure 6-8 shows the response of the plant in manual flow control mode to the loss of 100°F of the feedwater heating capability of the plant. This represents an upper limit on the effect expected from a single heater (or group of heaters) which can be tripped or bypassed by a single event. The reactor was assumed to be operating at maximum power when the heater was lost. The traces neglect the flow delay time of approximately 25 sec between the heaters and the feedwater sparger during which the plant would continue at steady-state conditions. The power increases due to the cooler feedwater entering the core. Fuel Surface peak heat flux and other parameters of interest are given in Table 6-2.

6.3.3.3 Rod Withdrawal Error

Assumptions and descriptions of rod withdrawal error (RWE) are given in Reference 1. Table 6-3 gives the results of the worst-case condition for DAEC Reload No. 3.

The rod block monitor (RBM) setpoint of 105% is selected to allow for failed instruments for the worst allowable situation. This case demonstrates that, even if the operator ignores all alarms during the course of this transient, the RBM will stop rod withdrawal so that critical power ratio (CPR) will not go below the 1.06 MCPR safety limit.

6.3.3.2.4 Feedwater Controller Failure

Of the class of events resulting in an increase in coolant inventory, the failure of the feedwater controller during maximum flow demand is the most severe. This event begins with the feedwater controller being forced to its upper limit while the plant is operating in a manual flow control mode. The feedwater pumps begin increasing the feedwater flow toward the maximum pump capability (feedwater pump runout capability is conservatively assumed to be 135% of NBR flow at a system design pressure of 1060 psig for DAEC, Cycle 4.

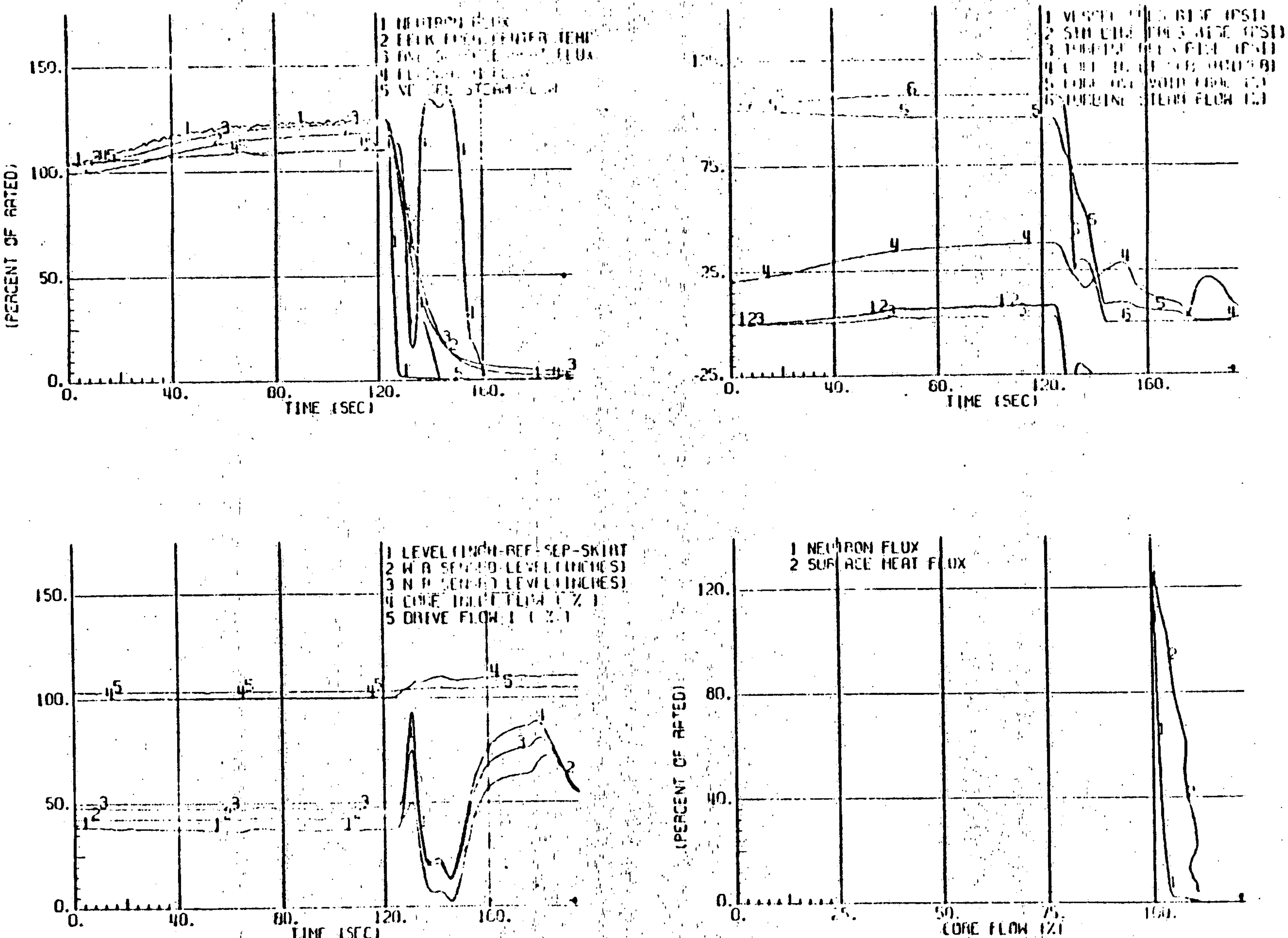


Figure 6-8. Loss of 100°F Feedwater Heating, MFC, EOC4

Table 6-3

LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE)

TRANSIENT SUMMARY

RBM Setpoint	ROD Position (Feet Withdrawn)	ΔCPR 7x7	ΔCPR 8x8	MLHGR 7x7	MLHGR 8x8
105	3.5	0.16	0.11	15.18	14.63
106	4.0	0.22	0.15	16.10	15.28
107	4.0	0.22	0.15	16.10	15.28
108	4.5	0.27	0.18	16.90	15.94
109	5.0	0.30	0.21	17.60	16.62
110	5.0	0.30	0.21	17.60	16.62

The influx of excess feedwater flow results in an increase in core subcooling, which reduces the void fraction and thus induces an increase in reactor power. The excess feedwater flow also results in a rise in the reactor water level, which eventually leads to high water level main turbine and feedwater turbine trip, and turbine bypass valves are actuated. Reactor scram trip is actuated from main turbine stop valve position switches. Relief valves open as steamline pressures reach relief valve setpoints. Figures 6-9a through 6-9d illustrate this transient. The peak neutron flux, and average surface heat flux, are shown in Table 6-2.

6.3.3.2.5 Turbine Trip With Failure of the Bypass Valves

This transient produces a reactor isolation which is slightly less severe than that produced by the load rejection without bypass. The primary characteristic of this transient is a pressure increase due to the obstruction of steam flow by the turbine stop valves. The pressure increase causes a significant void reduction which yields a pronounced positive void reactivity effect. The net reactivity is sharply positive and causes a rapid increase in neutron flux until the net reactivity is forced negative by scram initiated from 90% open switches on the turbine stop valves and by a void increase after the safety/relief valves have automatically opened on high pressure. Figures 6-10a through 6-10d illustrate this transient.

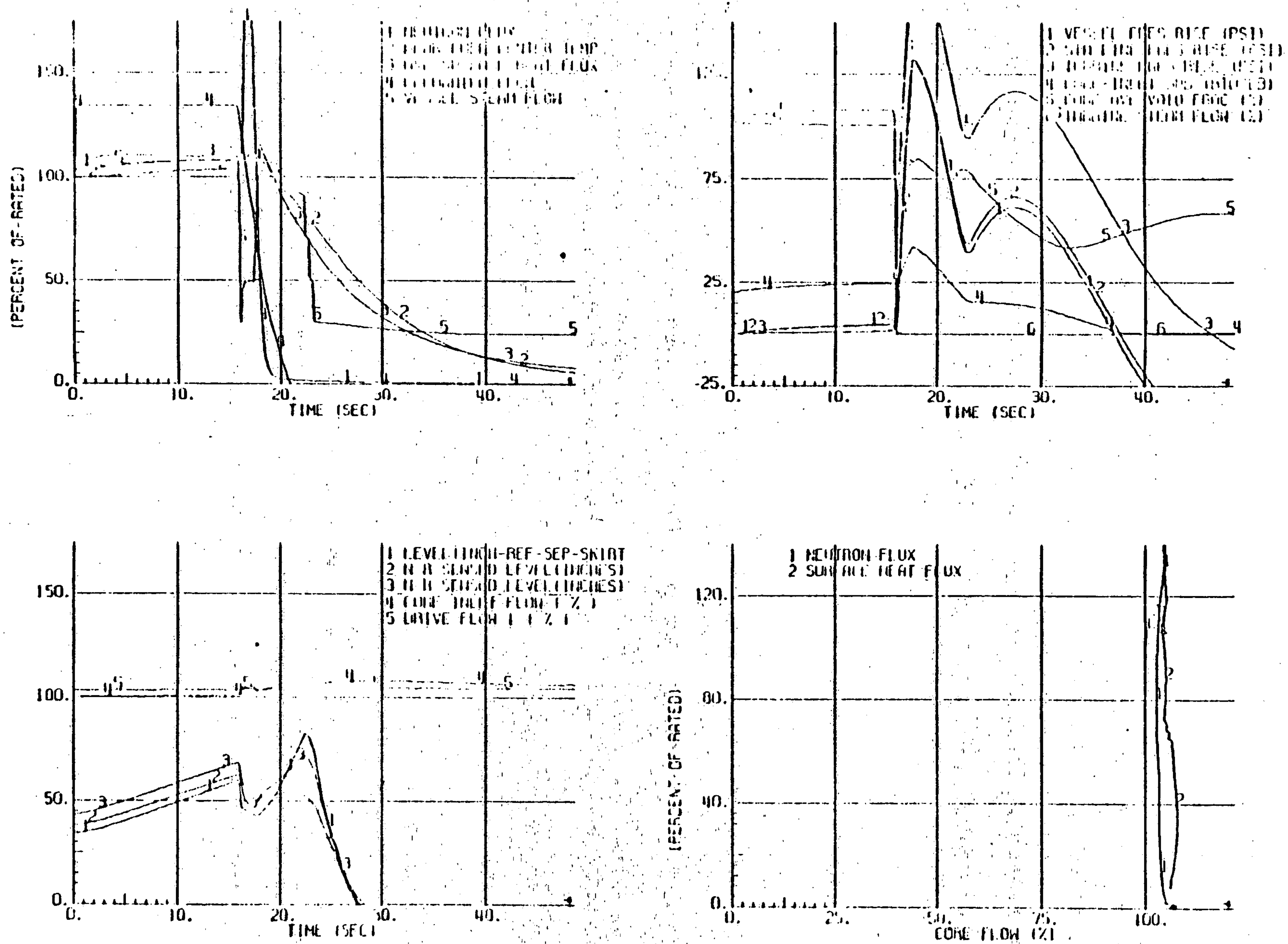


Figure 6-9a. Feedwater Controller Failure, Maximum Demand, with High Level Turbine Trip, EOC4

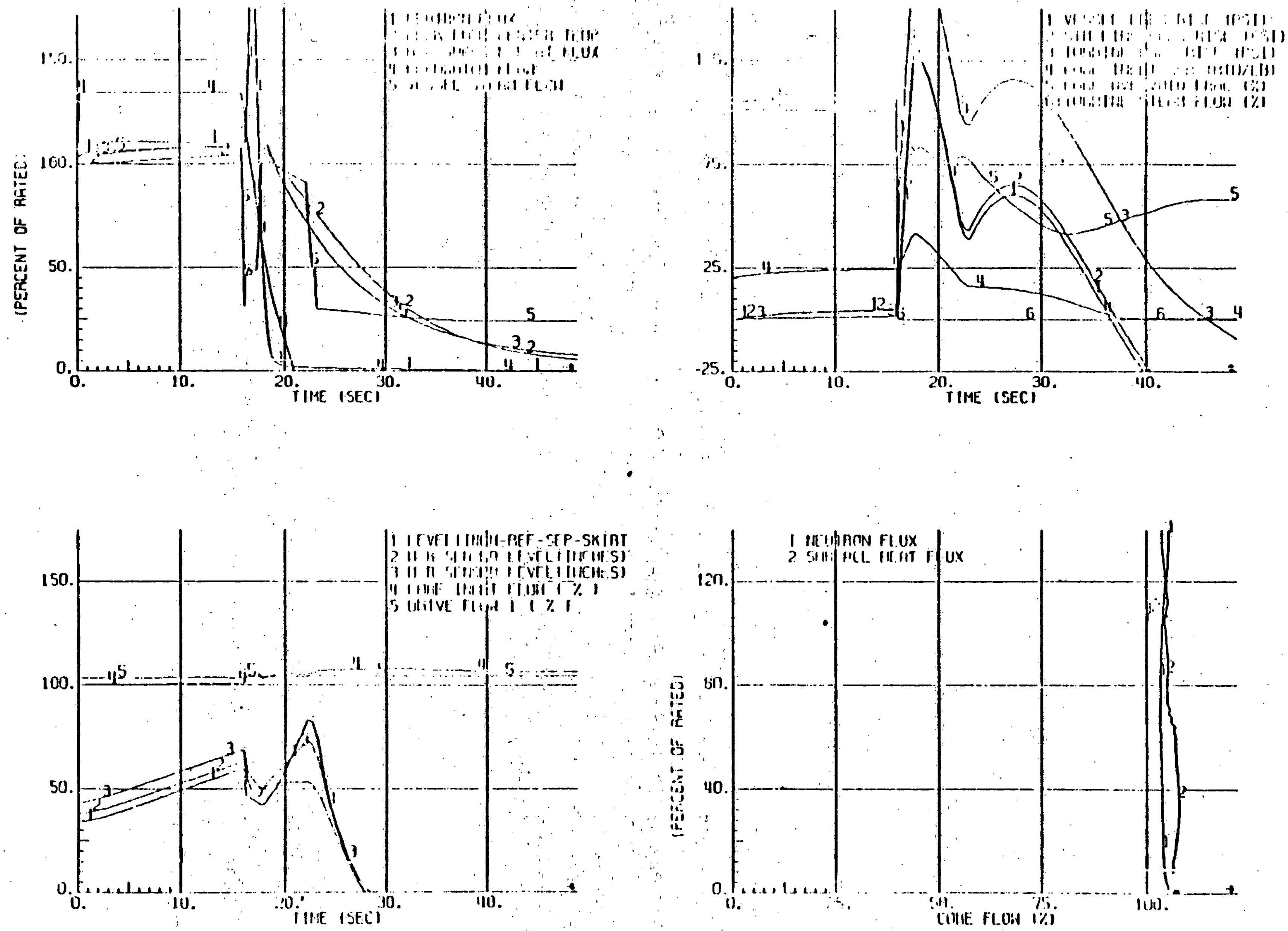


Figure 6-9b. Feedwater Controller Failure, Maximum Demand, with High Level Turbine Trip, EOC4-0.5 GWd/t

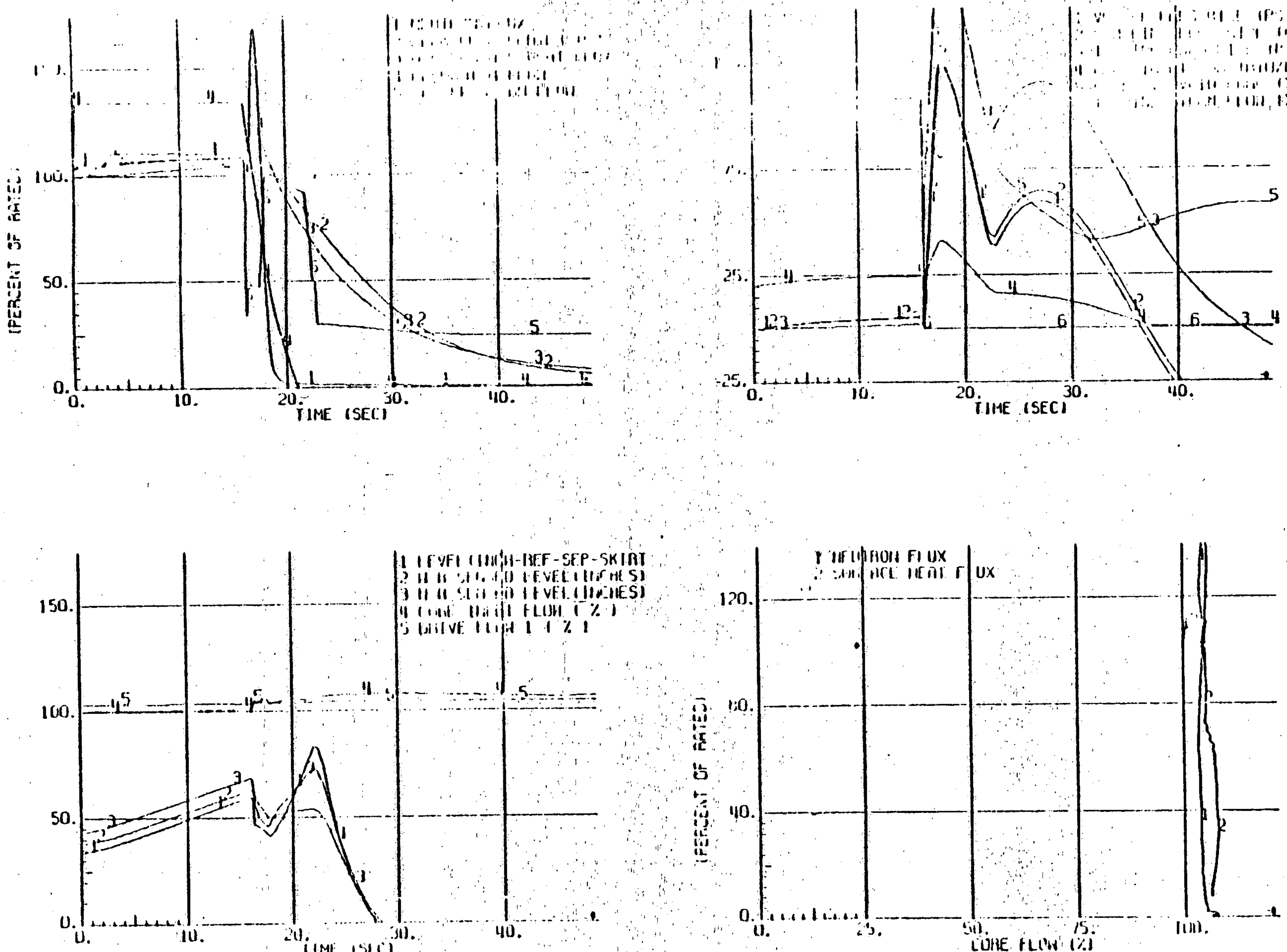


Figure 6-9c. Feedwater Controller Failure, Maximum Demand, with Higb Level Turbine Trip, EOC4-1.0 GWd/t

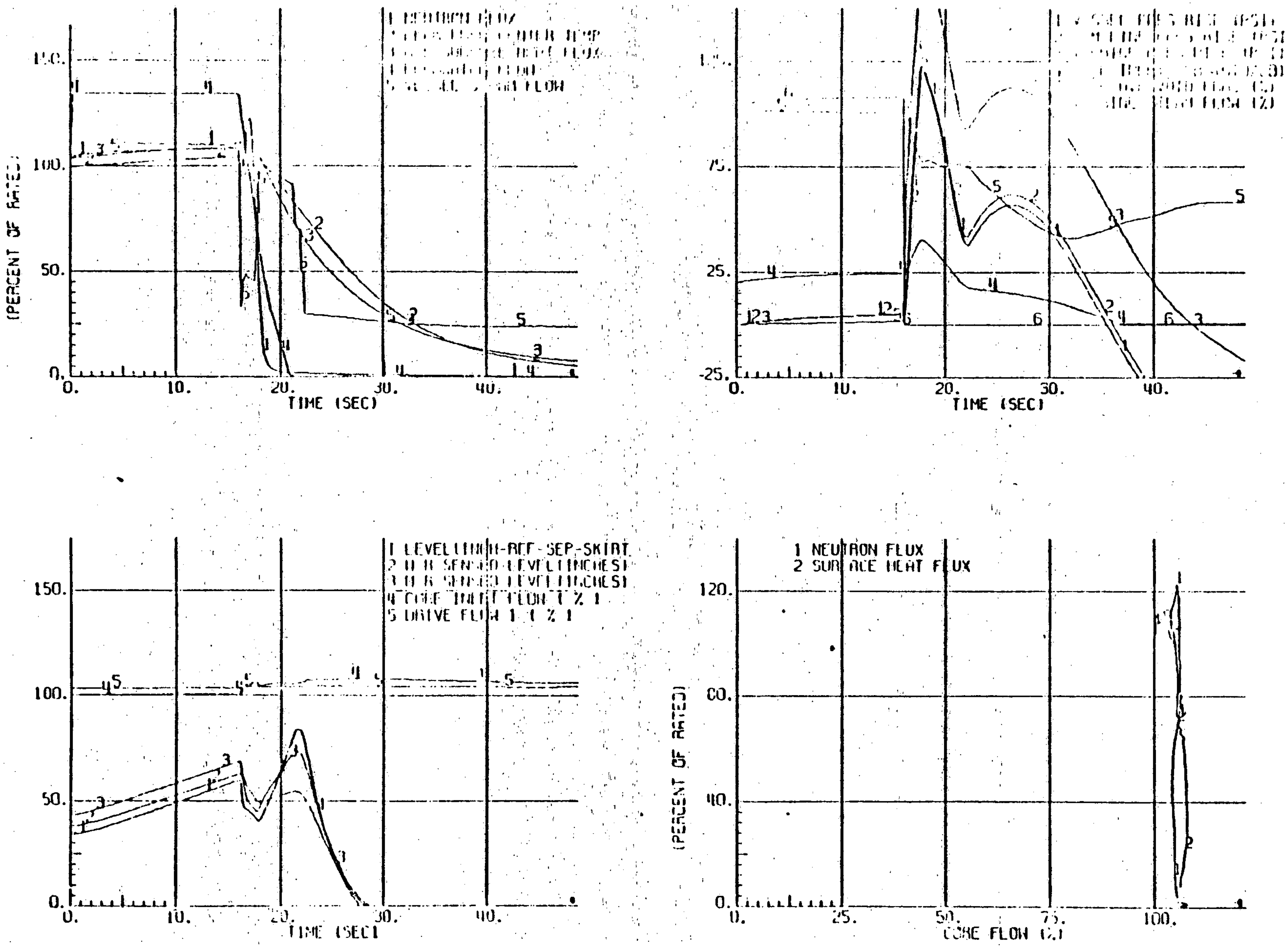


Figure 6-9d. Feedwater Controller Failure, Maximum Demand, with High Level Turbine Trip, EOC4-2.0 GWd/t

The parameters of concern are the peak vessel pressure margin to the first spring safety valve set point and the peak average surface heat flux correlated to MCPR. Neutron flux, heat flux, peak steamline pressure and peak vessel pressure are given in Table 6-2.

6.3.4 ASME Vessel Pressure Code Compliance

6.3.4.1 All Main Steam Line Isolation Valve Closure-Flux Scram (Safety Valve Adequacy)

The pressure relief system must prevent excessive overpressurization of the primary system process barrier and the pressure vessel to preclude an uncontrolled release of fission products.

The DAEC pressure relief system includes 6 dual function safety/relief valves and 2 spring safety valves located on the main steam lines within the drywell between the reactor vessel and the first isolation valve. These valves provide the capacity to limit nuclear system overpressurization.

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from the consequences of pressure in excess of the vessel design pressure:

(a) A peak allowable pressure of 110% of the vessel design pressure is allowed (1375 psig for a vessel with a design pressure of 1250 psig).

(b) The lowest qualified safety/relief valve setpoint must be at or below vessel design pressure.

(c) The highest safety/relief valve setpoint must not be greater than 105% of vessel design pressure (1313 psig for a 1250 psig vessel).

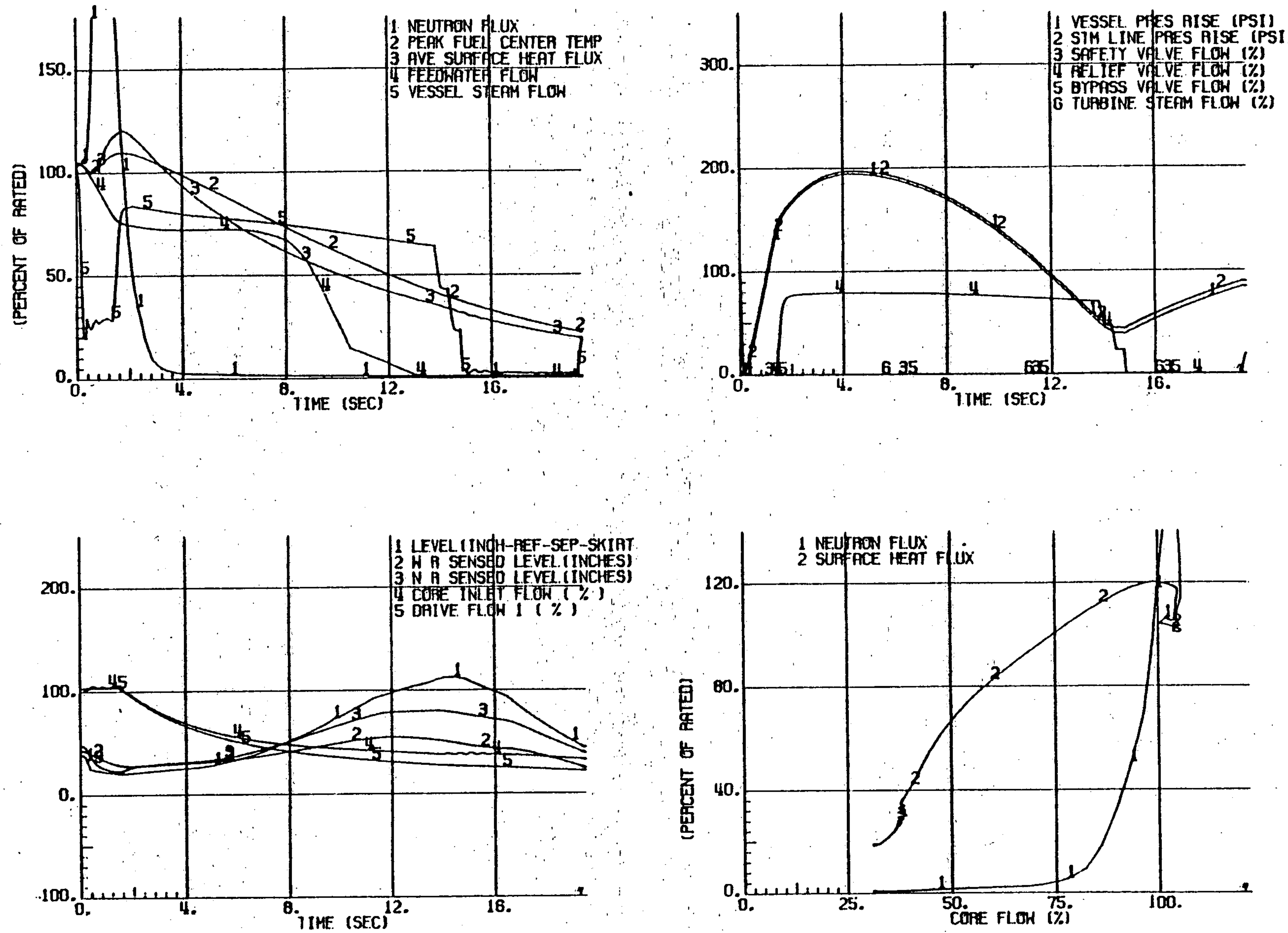


Figure 6-10a. Turbine Trip without Bypass, Trip Scram, EOC4

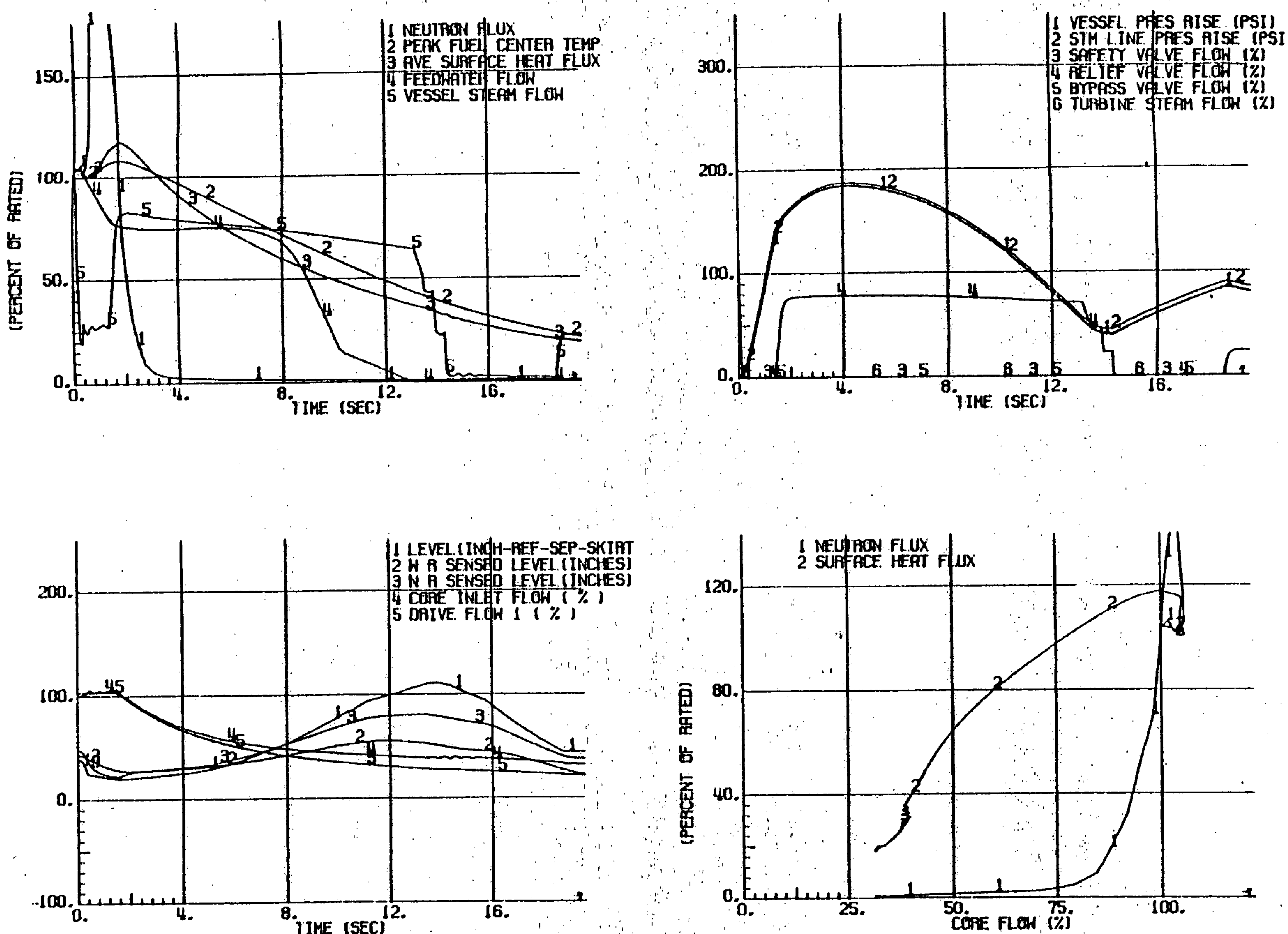


Figure 6-10b. Turbine Trip without Bypass, Trip Scram, EOC4-0.5 GWd/t

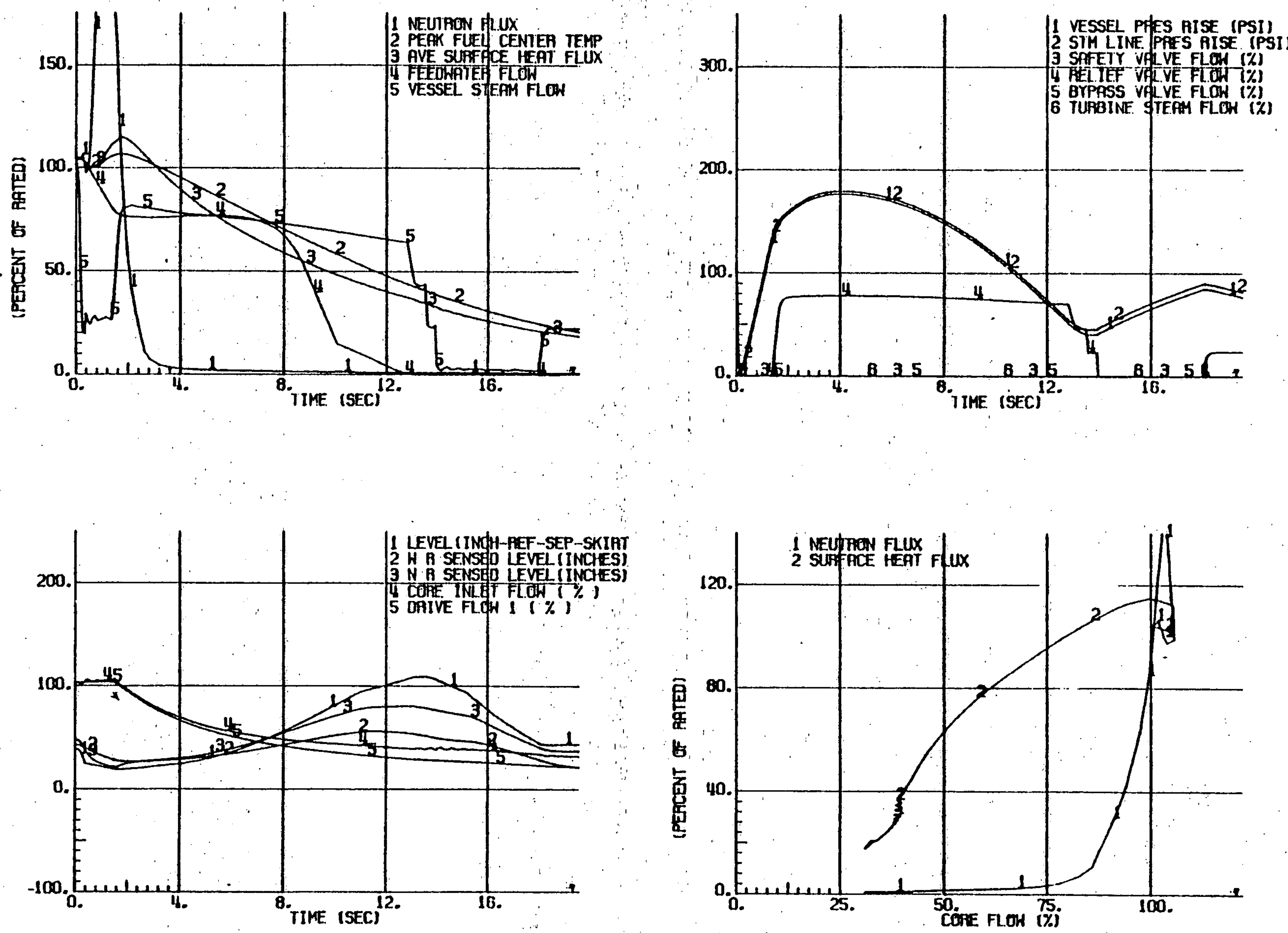


Figure 6-10c. Turbine Trip without Bypass, Trip Scram, EOC4-1.0 GWd/t

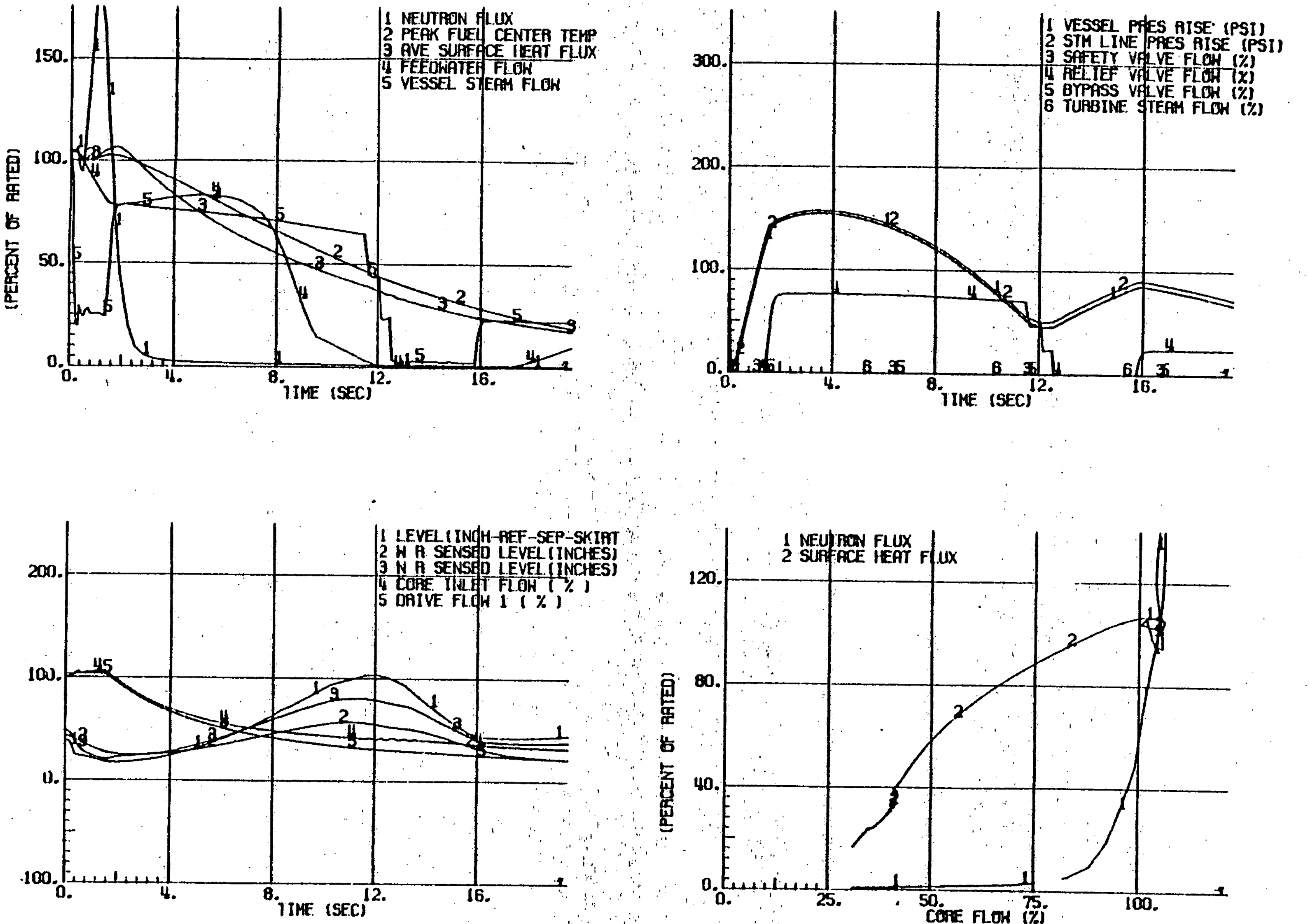


Figure 6-10d. Turbine Trip without Bypass, Trip Scram, EOC4-2.0 GWd/t

The DAEC safety/relief and safety valves are set to self-actuate at the pressures shown in Table 6-1, thereby satisfying (b) and (c), above. Requirement (a) is evaluated by considering the most severe isolation event with indirect scram.

The event which satisfies this specification is the closure of all main steamline isolation valves with indirect (flux) scram and assumes a drive motor trip at a reactor pressure of 1120 psi plus a 300 millisecond delay. The initial conditions assumed are those specified in Table 6-1. Figure 6-11 graphically illustrates the event. An abrupt pressure and power rise occur as soon as the reactor is isolated. Reactor shutdown is initiated when neutron flux reaches scram level. The safety/relief valves open to limit the pressure rise in the steamline at the valves to 1259 psig and at the bottom of the vessel to 1287 psig. This response provides an 88 psi margin to the vessel code limit of 1375 psig. Thus, requirement (a) is satisfied and adequate overpressure protection is provided by the pressure relief system. The effect on peak vessel pressure of having one valve out of service is given in Reference 4. This effect is expected to be less than 20 psi.

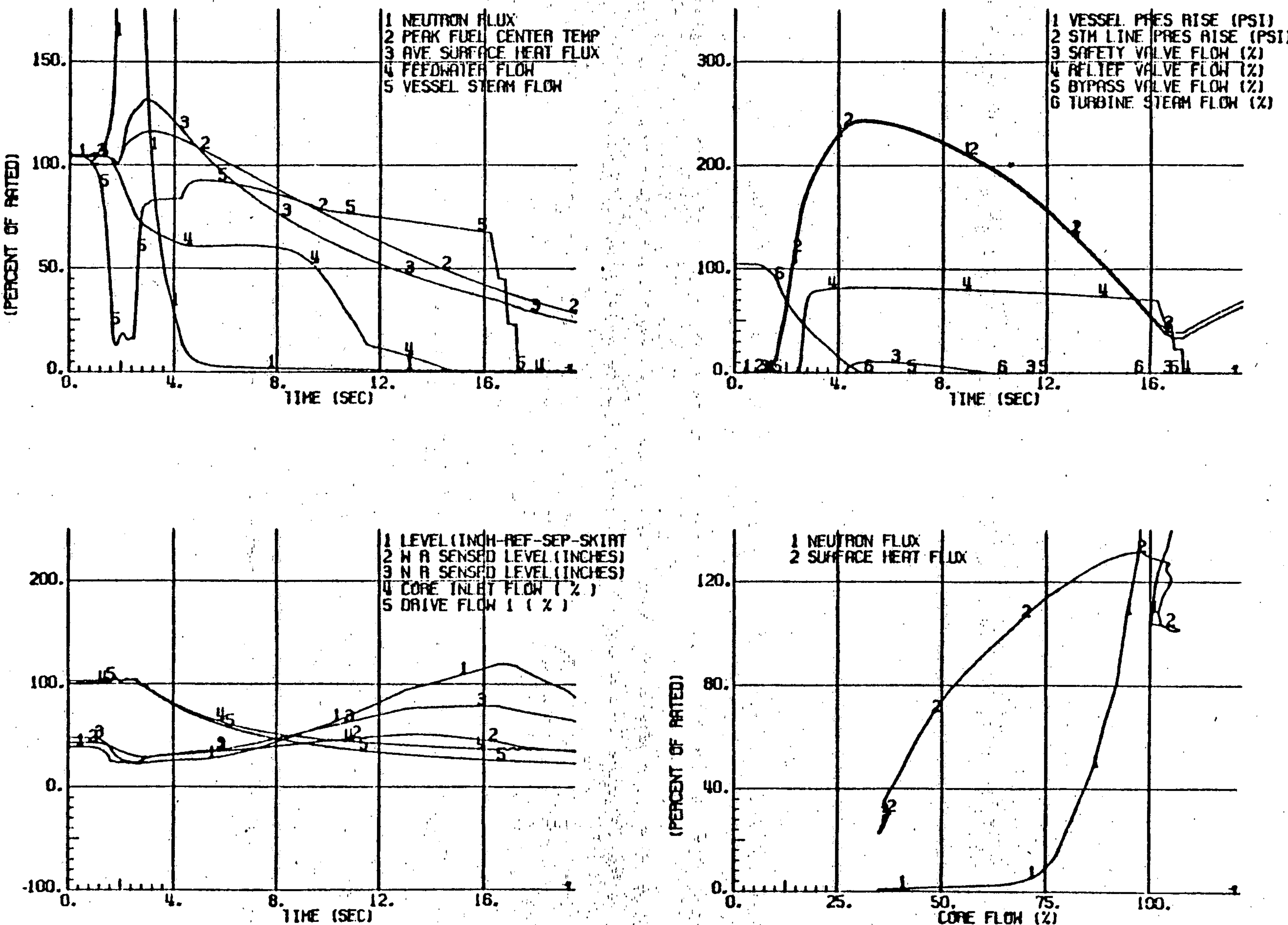


Figure 6-11. Duane Arnold EO Cycle 4 MSIV Closure, Flux Scram

6.3.5 Thermal-Hydraulic Stability Analysis

Descriptions of the types of thermal-hydraulic stability considered and the analytical method used for evaluation are given in Reference 1. The results for DAEC Reload-3 are given below.

6.3.5.1 Channel Hydrodynamic Conformance to the Ultimate Performance Criteria

The channel performance calculation yields decay ratios as presented below:

Channel Hydrodynamic Performance	Intersection of 105% Rod Line - Natural Circulation Power
Decay Ratio, X_2/X_o	
8x8 Channel	0.38
7x7 Channel	0.23

At this most responsive condition, the most responsive channels are clearly within the bounds of the ultimate performance criteria of ≤ 1.0 decay ratio at all attainable operating conditions.

6.3.5.2 Reactor Conformance to Ultimate Performance Criteria

The decay ratios determined from the limiting reactor core stability conditions are presented in Figure 6-12. The most responsive case is again the 105% rod line - natural circulation condition.

Reactor Core Stability	Intersection of 105% Rod Line - Natural Circulation Power
Decay Ratio, X_2/X_o	0.81

These calculations show the reactor to be in compliance with the ultimate performance criteria, including the most responsive condition at 105% Rod Line - Natural Circulation Power.

6.3.5.3 Channel Hydrodynamic Conformance to the Operational Design Guide

Channel Hydrodynamic Performance	Rated Conditions	Low End of Flow Control Range
Decay Ratio, X_3/X_0		
8x8	<0.01	<0.02
7x7	<0.01	<0.01

The most responsive channel is in conformance with the operational design guide of <0.5 decay ratio.

6.3.5.4 Reactor Core Conformance to Operational Design Guide

The calculated value of the decay ratio of the reactor power dynamic response for rated operating conditions and for the low end of the flow control range at the corresponding nominal power (74% power, 57% flow) are presented below.

Reactor Core Performance	Rated Conditions	Low End of Flow Control Range
Decay Ratio	0.024	0.025

As noted earlier, Figure 6-12 describes the variation of decay ratio over the entire power flow range.

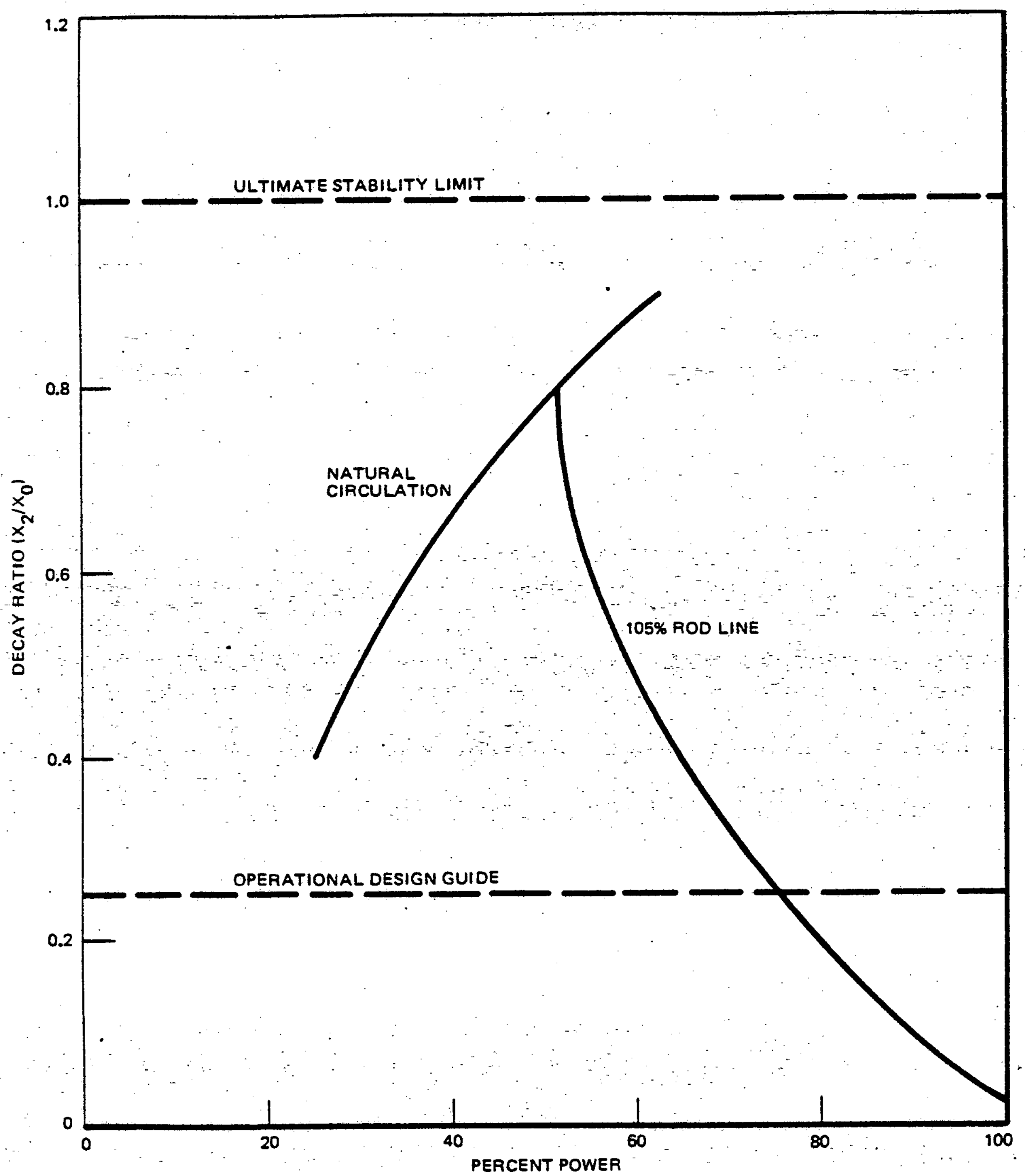


Figure 6-12. DAEC Cycle 4 Decay Ratio

7. REFERENCES

1. GE/BWR Generic Reload Licensing Application for 8x8 Fuel, Rev 1 Supplement 4, April 1976 (NEDO-20360).

2. General Electric Thermal Analysis Basis (GETAB): Data Correlation and Design Application, General Electric Company BWR Systems Department, November 1973 (NEDE-10958, Class III).

3. Loss of Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant), NEDO-21082-02-1A, Appendix A, Revision 1, July 1977.

4. Letter, Ivan F. Stuart to Victor Stello, Jr., "Code Overpressure Protection Analysis - Sensitivity of Peak Vessel Pressures to Valve Operability," December 23, 1975.