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GENERAL ELECTRIC BOILING WATER REACTOR RELOAD-5 LICENSING SUBMITTAL FOR DRESDEN NUCLEAR POWER STATION UNIT 3

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GENERAL ELECTRIC BOILING WATER REACTOR

RELOAD-5 LICENSING SUBMITTAL

FOR

DRESDEN NUCLEAR POWER STATION

UNIT 3

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

GENERAL ELECTRIC

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1. INTRODUCTION

This document provides the supplemental information for Reload-5 at the Dresden Nuclear Power Station Unit 3. The technical bases, generic design information, and safety analyses are given in Reference 1.

The design reference core loading is based on the use of 20 8x8 bundles having a bundle average enrichment of 2.50 wt % U-235 and 156 8x8 bundles having a bundle average enrichment of 2.62 wt % U-235.

The objective of this outage is to load the reactor core to ensure sufficient reactivity to operate the 724-element core at a licensed power level of 2527 MWt for a nominal 5650 MWd/t cycle.

Analyses in this document and its references justify satisfaction of the outage objectives.

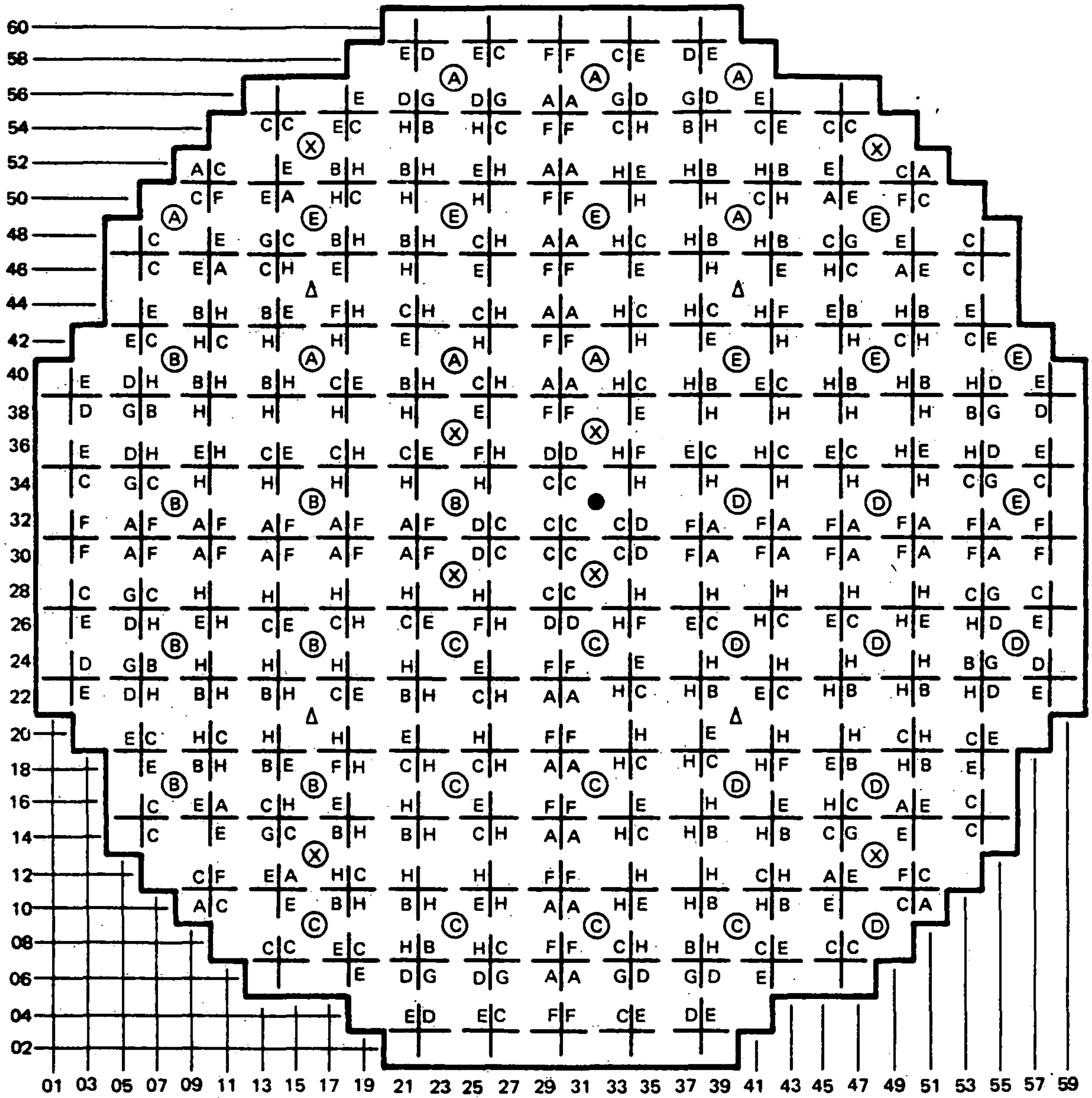
2. SUMMARY

The design reference core configuration for this license submittal 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		164
Reload-1	(7D230)	52
Reload-2	(8D250)	44
Reload-3	(8D250)	108
	(8D262)	32
Reload-4	(8D250)	60
	(8D262)	88
Reload-5	(8D250)	20
	(8D262)	156
	Total	724



Ⓐ LPRM LOCATION (LETTER INDICATES TIP MACHINE)
● LPRM LOCATION (COMMON LOCATION FOR ALL TIP MACHINES)
ⓧ IRM LOCATIONS
△ SRM LOCATIONS

BLANK – INITIAL FUEL
A – RELOAD 1 (7D230 GENERIC B)
B – RELOAD 2 (8D250)
C – RELOAD 3 (8D250)
D – RELOAD 3 (8D262)
E – RELOAD 4 (8D262)
F – RELOAD 4 (8D250)
G – RELOAD 5 (8D250)
H – RELOAD 5 (8D262)

Figure 2-1. Dresden 3 NPS Reload-5 Design Reference Core Loading

3. MECHANICAL DESIGN

The two types of Reload-5 fuel which will be employed have the same mechanical configuration and fuel bundle enrichments as the 8D262 and the 8D250 fuel assemblies described in Reference 1. Reload 5 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. The design criteria, models, and results from design evaluation presented in Section 3 of Reference 1 are applicable to the subject reload.

All Reload-5 fuel incorporates finger springs of the type described in Reference 1.

4. THERMAL-HYDRAULIC ANALYSES

4.1 STATISTICAL ANALYSIS

The statistical analyses of the reactor core were performed using the uncertainty inputs described in Section 4.5 of Reference 1. 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.

4.2 ANALYSIS OF ABNORMAL OPERATIONAL TRANSIENTS

The results of the limiting abnormal operational transient analyses and the Rod Withdrawal Error (RWE) are summarized in Table 4-1; the specific analyses are described in Section 6. The most severe transient from rated conditions for the 7x7 fuel is a RWE which has a maximum ΔCPR of 0.23. The most severe transient from rated conditions for the 8x8 fuel is a generator load rejection without bypass which has a maximum ΔCPR of 0.23. Addition of the ΔCPR to the Safety Limit MCPR gives the minimum initial MCPR to avoid violating the Safety Limit MCPR during the most severe transient from rated conditions. The GETAB analysis initial conditions for the abnormal operational transients are given in Table 4-2.

4.2.1 Operating Limit MCPR

Based on the Fuel Cladding Integrity Safety Limit and the results of the transient analyses, the Operating Limit MCPR is 1.29 for 8x8 fuel and 1.29 for 7x7 fuel.

Table 4-1

SUMMARY OF RESULTS
LIMITING TRANSIENTS

	Maximum ΔCPR 7x7	Maximum ΔCPR 8x8
Turbine Trip w/o Bypass (Rated Conditions)	0.16	0.22
Load Rejection w/o Bypass (Rated Conditions)	0.17	0.23
Loss of 145°F Feedwater Heating	0.16	0.18
Rod Withdrawal Error (107% RBM Set Point)	0.23	0.19

Table 4-2

GETAB TRANSIENT ANALYSIS
INITIAL CONDITION PARAMETERS*
(Abnormal Operating Transients)

	7x7	8x8
Peaking Factors (Local, Radial, Axial)	1.30, 1.52, 1.40	1.22, 1.67, 1.40
R-Factor	1.100	1.094
Bundle Power (MWt)	5.191	5.699
Nonfuel Power Fraction	0.035	0.035
Core Flow (Mlb/hr)	98	98
Bundle Flow (10^3 lb/hr)	113.56	109.43
Reactor Pressure (psia)	1030	1030
Inlet Enthalpy (Btu/lb)	522.5	522.5
Initial MCPR	1.25	1.31

*100% power/100% flow initial conditions
(Do not apply to RWE)

5. NUCLEAR CHARACTERISTICS

The bundle characteristics, analytical methods, and model descriptions presented in Subsections 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 176 Reload-5 fuel bundles by discharging a like number of fuel bundles from the Cycle 5 core.

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-5 fuel bundles are 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 0.25% 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 0.014 Δk calculated for the assumed refueling at a core average exposure of 15,134 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 Beginning of Cycle 6 (BOC-6) shutdown margin is 0.014 Δk. Thus, R, the differences between the BOC-6 and the minimum shutdown margin plus the effect of B_4C settling in the absorber tubes is 0.0004 Δk.

5.1.3 Standby Liquid Control System

A boron concentration of 600 ppm in the moderator water will bring the reactor subcritical by 0.033 Δk at 20°C, xenon free.

Table 5-1

NUCLEAR CHARACTERISTICS OF THE DESIGN REFERENCE CORE

Core Effective Multiplication and Control System Worth (No Voids, 20°C)	
BOC k_{eff}	
Uncontrolled	1.119
Fully Controlled	0.954
Strongest Control Rod Out	0.986
R, Maximum Increase in Core Reactivity With Exposure Into Cycle, Δk (including effects of inverted B_4C tubes in control rods)	0.0004
Reactivity Coefficients, Range During Operating Cycle	
Steam Void Coefficients at Average Voids; $(\Delta k/k)/\Delta V$, 1/% Void	-11.9×10^{-4} to -10.4×10^{-4}
Power Coefficient at Rated Conditions $(\Delta k/k)/(\Delta P/P)$	-0.057
Fuel Temperature Coefficient at 650°C $(\Delta k/k)/\Delta T$, 1/°F	-1.15×10^{-5} to -1.24×10^{-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 main steamline 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 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 Dresden 3 Reload 5, 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 the maximum values of the parameters after this reload will be well below the boundary value, the consequences of a rod drop excursion from any insequence 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 analysis will be submitted under a separate cover on a schedule consistent with NRC requirements.

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-5 bundle is inserted in an improper location; and

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

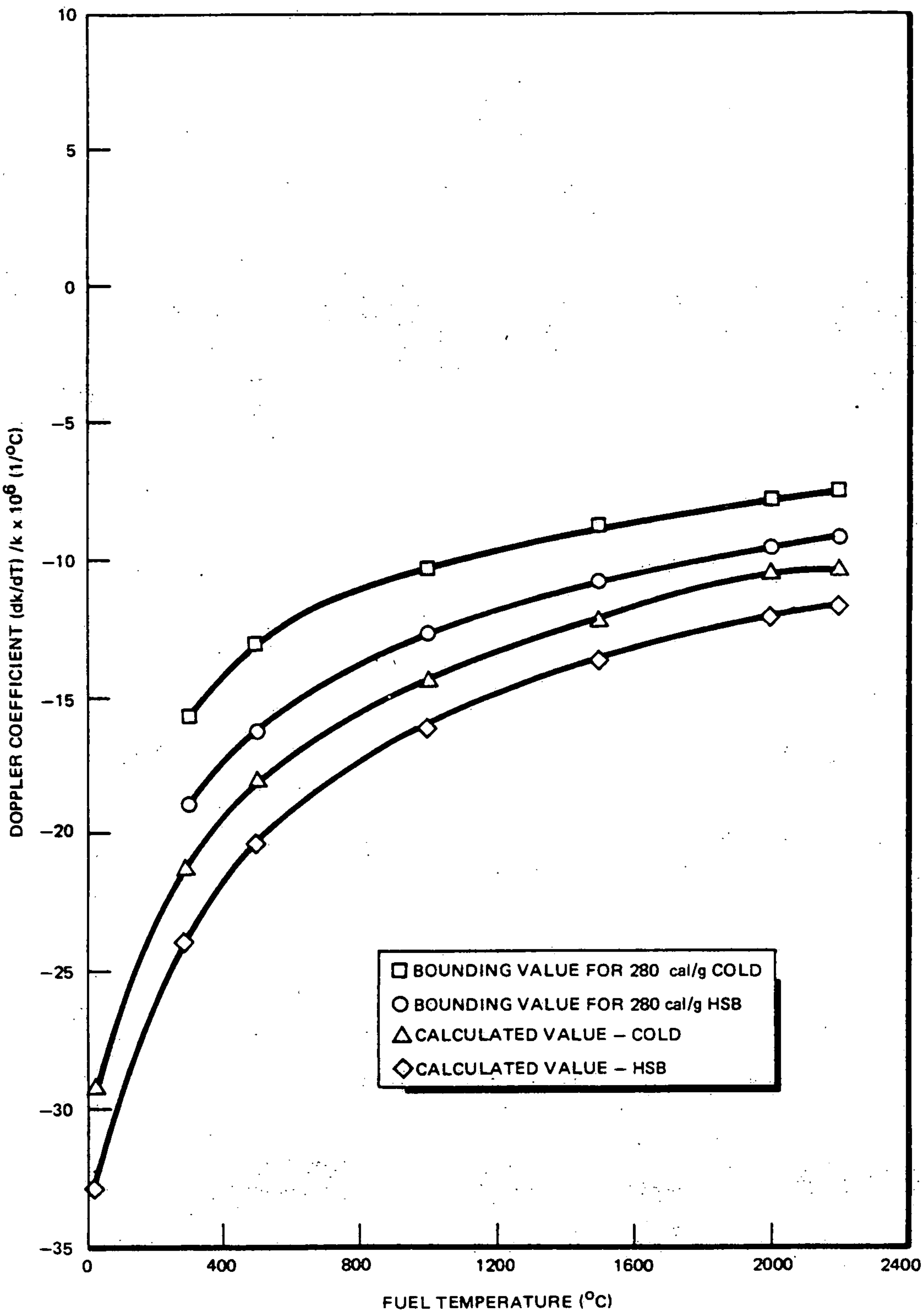


Figure 6-1. Doppler Coefficient Versus Average Fuel Temperature as a Function of Moderator Condition

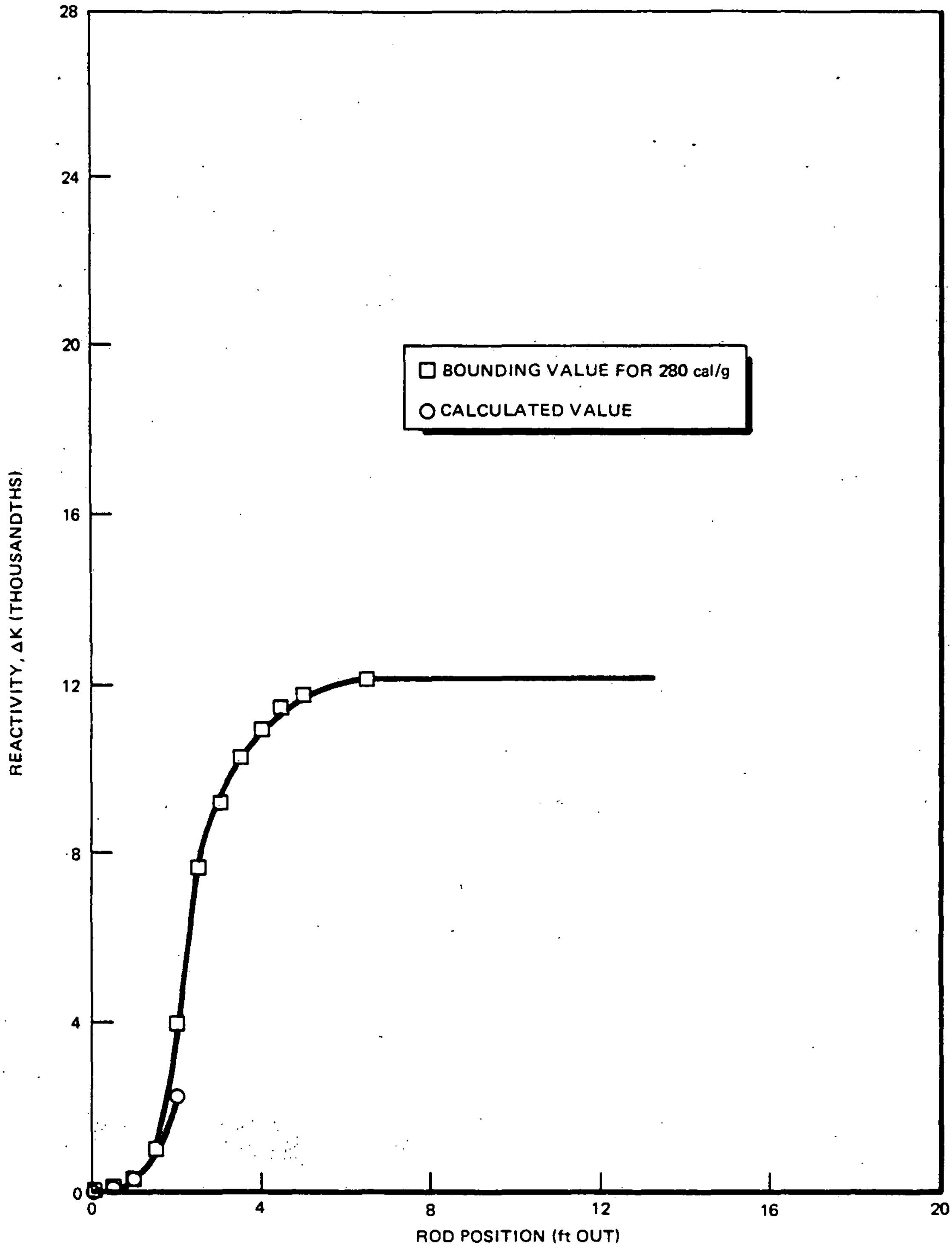


Figure 6–2. Accident Reactivity Shape Function Cold Startup, β = 0.005, P_L = 1.5

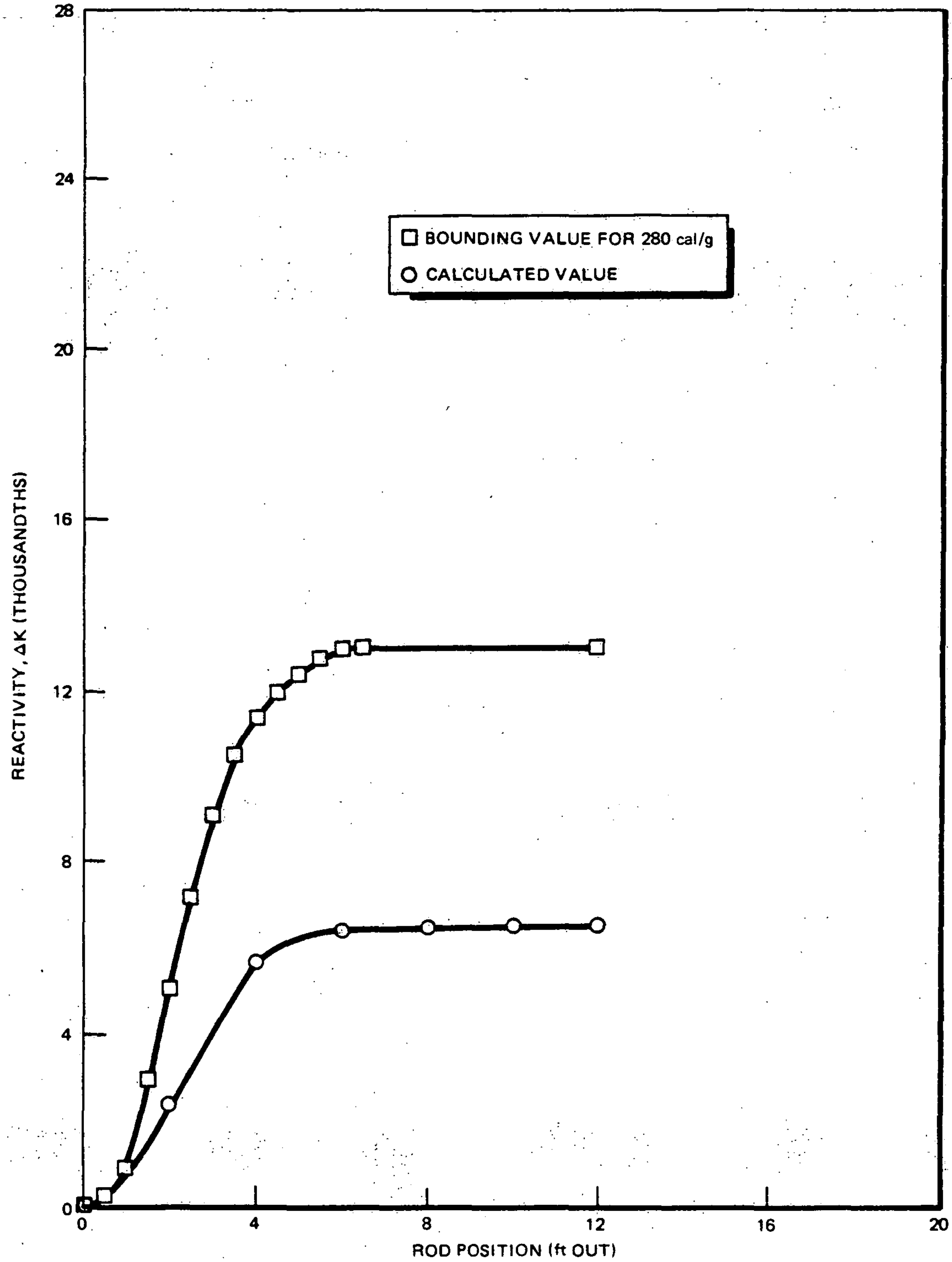


Figure 6-3. Accident Reactivity Shape Function Hot Startup, $\beta = 0.005$, $P_L = 1.3$

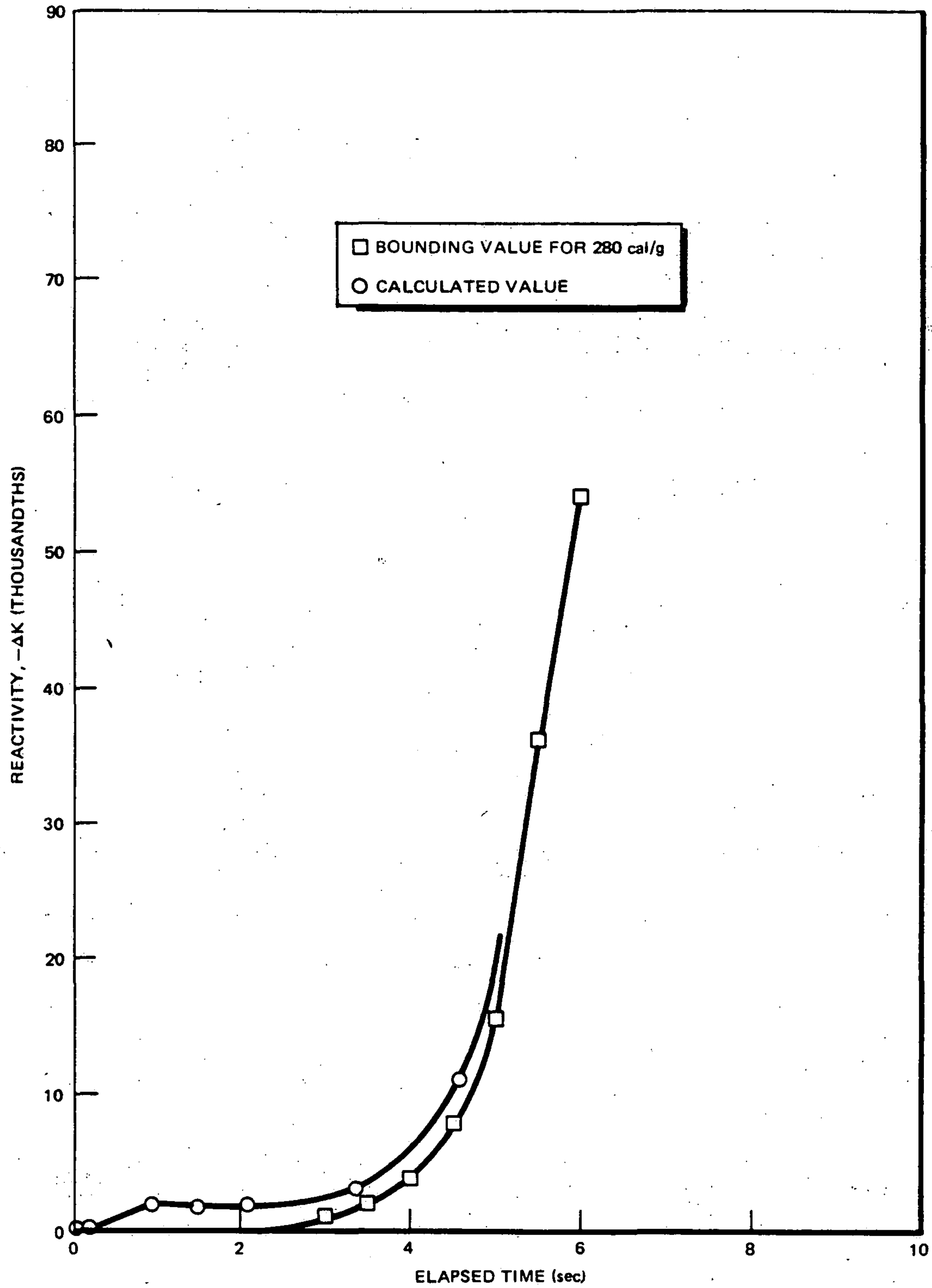


Figure 6-4. Scram Reactivity Function for 20°C (RDA)

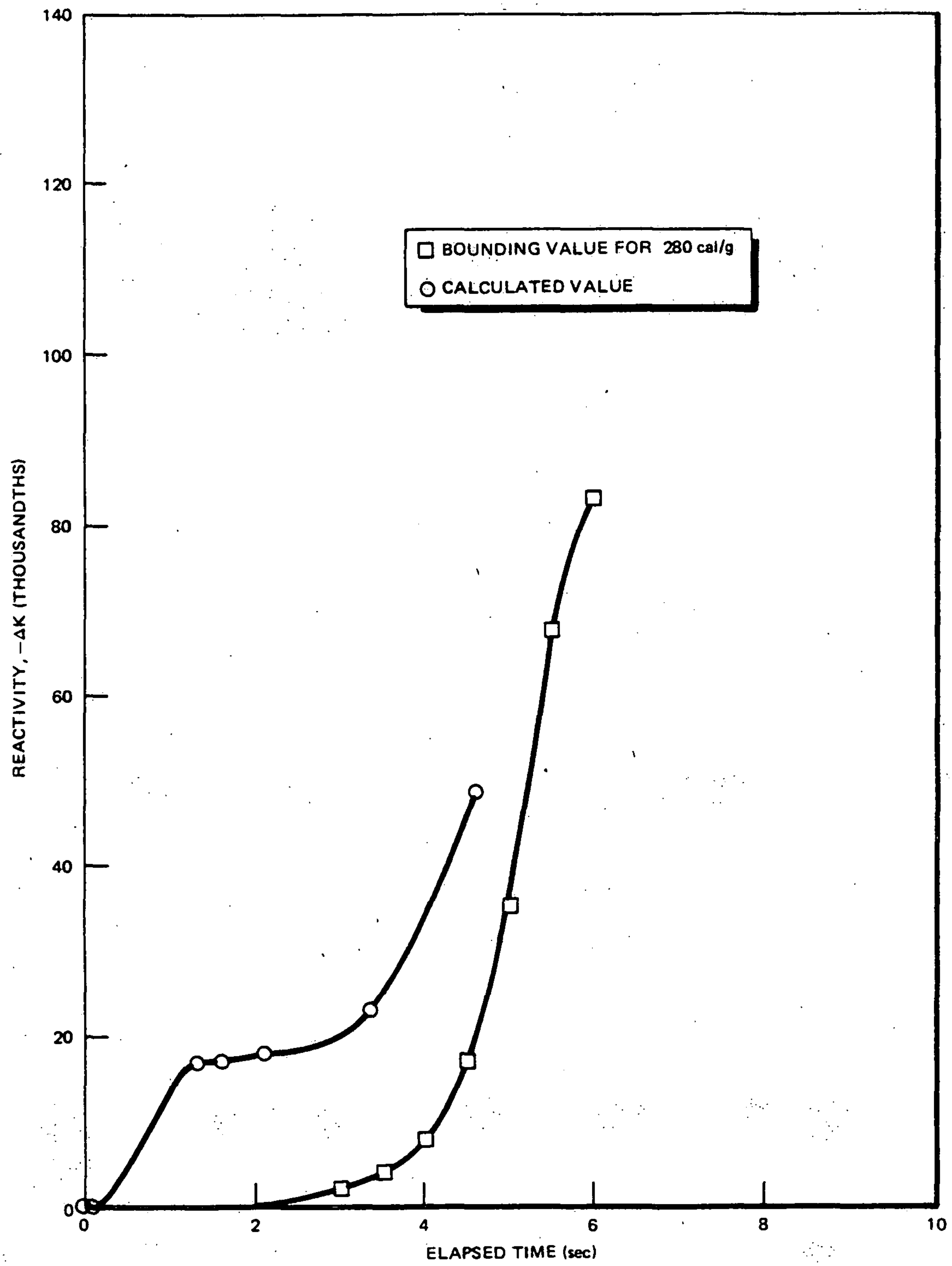


Figure 6-5. Scram Reactivity Function for 286°C (RDA)

Since two independent errors are assumed to occur, the single-error criterion is violated; thus, 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 18.87 kW/ft and a minimum critical power ratio (MCPR) of 0.95* in the misplaced reload (8D262) bundle. The peak LHGR is not large enough to cause fuel damage. Since only a single assembly has an MCPR lower than the safety limit MCPR, the number of rods in the core expected to experience boiling transition is small (<0.01%). Thus, the results of this accident are far less severe than the major accidents.

Fuel bundles adjacent to the misplaced bundle are insignificantly affected by the presence of the misplaced 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 Dresden 3 Cycle 6. 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.

6.3.3.1.2 Input Data and Operating Conditions

The input data and operating conditions are shown in Table 6-1 and represent the nominal basis for these analyses. Each transient is considered at these conditions unless otherwise specified.

*From an initial MCPR of 1.25.

Table 6-1

TRANSIENT INPUT PARAMETERS

Thermal Power	(MWt)	2527
Rated Steam Flow	(lb/hr)	9.76×10^6
Rated Core Flow	(lb/hr)	98.0×10^6
Dome Pressure	(psig)	1005
Turbine Pressure	(psig)	935
RV Setpoint (1)	(psig)	1@1136, (3) 2@1141, 2@1146
RV/Capacity (at Setpoint)	(No./%)	5/29.2
RV Time Delay	(msec)	650/400 (3)
RV Stroke Time	(msec)	200/100 (3)
SV Setpoint (1)	(psig)	2@1252, 2@1262, 4@1272
SV Capacity	(No./%)	8/52.5
Void Coefficient (2)	(¢/%Rg)	-9.552, -8.637
Void Fraction (2)	(%)	36.04, 33.92
Doppler Coefficient (2)	(¢/°F)	-0.208, -0.213
Average Fuel Temperature	(°F)	1203
Scram Reactivity Curve		Figure 6-6
Scram Worth (2) ($)	(%)	-29.2, -30.1

(1) Includes 1%

(2) 2000 MWd/t before EOC-6 and Cycle 6 limiting case respectively

(3) Target Rock combination safety/relief valve

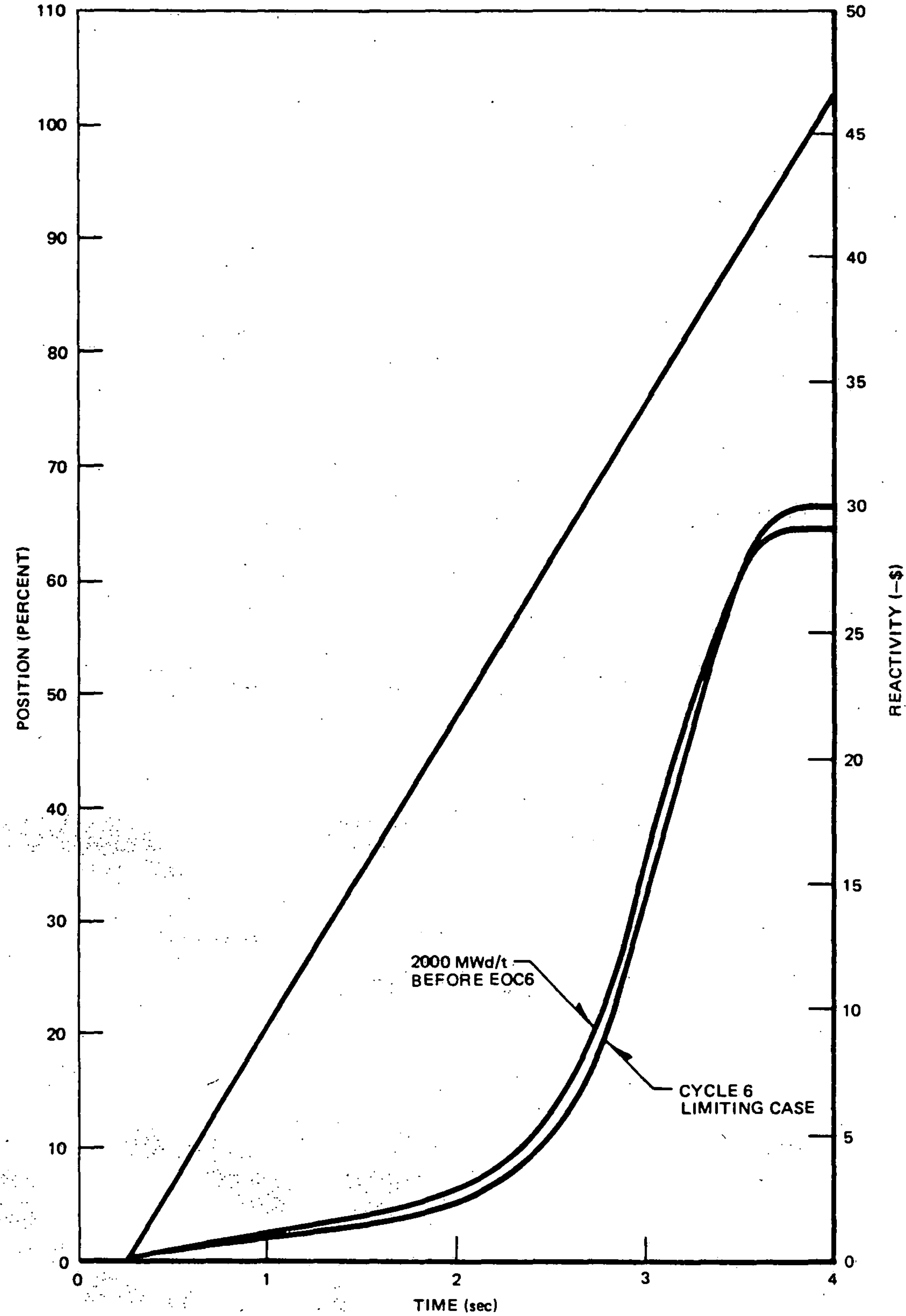


Figure 6-6. Dresden 3 Cycle 6 Scram Reactivity and Control Rod Drive Specification (Transients)

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

The primary characteristic of this transient is a pressure increase due to the obstruction 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 pressure switches sensing control valve fast closure and by a void increase after the relief valves have automatically opened on high pressure. Figure 6-7a and b illustrate this transient for the EOC6-2000 MWd/t and the limiting Cycle 6 cases.

The parameters of concern are the peak steamline pressure margin to the first spring safety valve setpoint and the peak average surface heat flux correlated to MCPR.

For the limiting Cycle 6 case, the neutron flux (the precursor of heat flux) rises to a peak of 259.2% with a corresponding peak heat flux of 111.0%. The change in critical power ratio (ΔCPR) is presented in Table 4-1.

The peak steamline pressure for the worst Cycle 6 case is limited to 1215 psig as a result of the high-pressure actuation of the four electromatic valves and one combination of safety/relief valve, which provides a 25 psi margin to the 1240 psig setpoint of the first spring safety valve.

Table 6-2

DRESDEN 3 CYCLE 6 TRANSIENT DATA SUMMARY

Transient	Power (%)	Core Flow (%)	ϕ (%)	Q/A (%)	P_{SL} (psig)	P_V (psig)
Generator Load Rejection, w/o Bypass, Trip Scram EOC6 -2000 MWd/t	100	100	266.6	110.7	1211	1245
Cycle 6 Limiting Case	98	100	259.2	111.0	1215	1250
Turbine Trip, w/o Bypass, Trip Scram EOC6 -2000 MWd/t	100	100	248.3	109.2	1211	1245
Cycle 6 Limiting Case	98	100	243.3	109.3	1214	1247
Loss of 145°F Feedwater Heating	100	100	120.8	119.5	999	

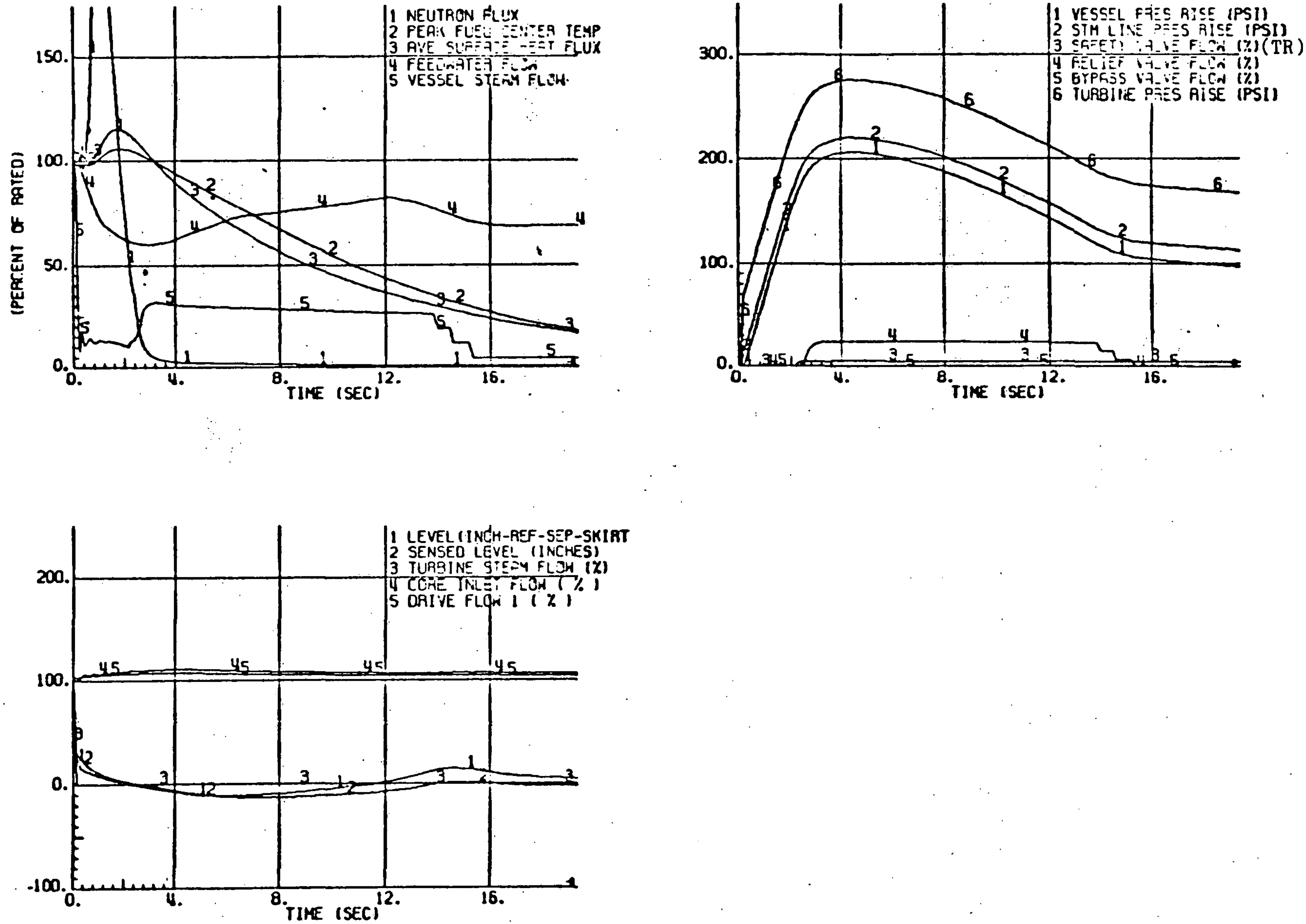


Figure 6-7a. Generator Load Rejection Without Bypass -2000 MWd/t Before EOC-6 (100% Power)

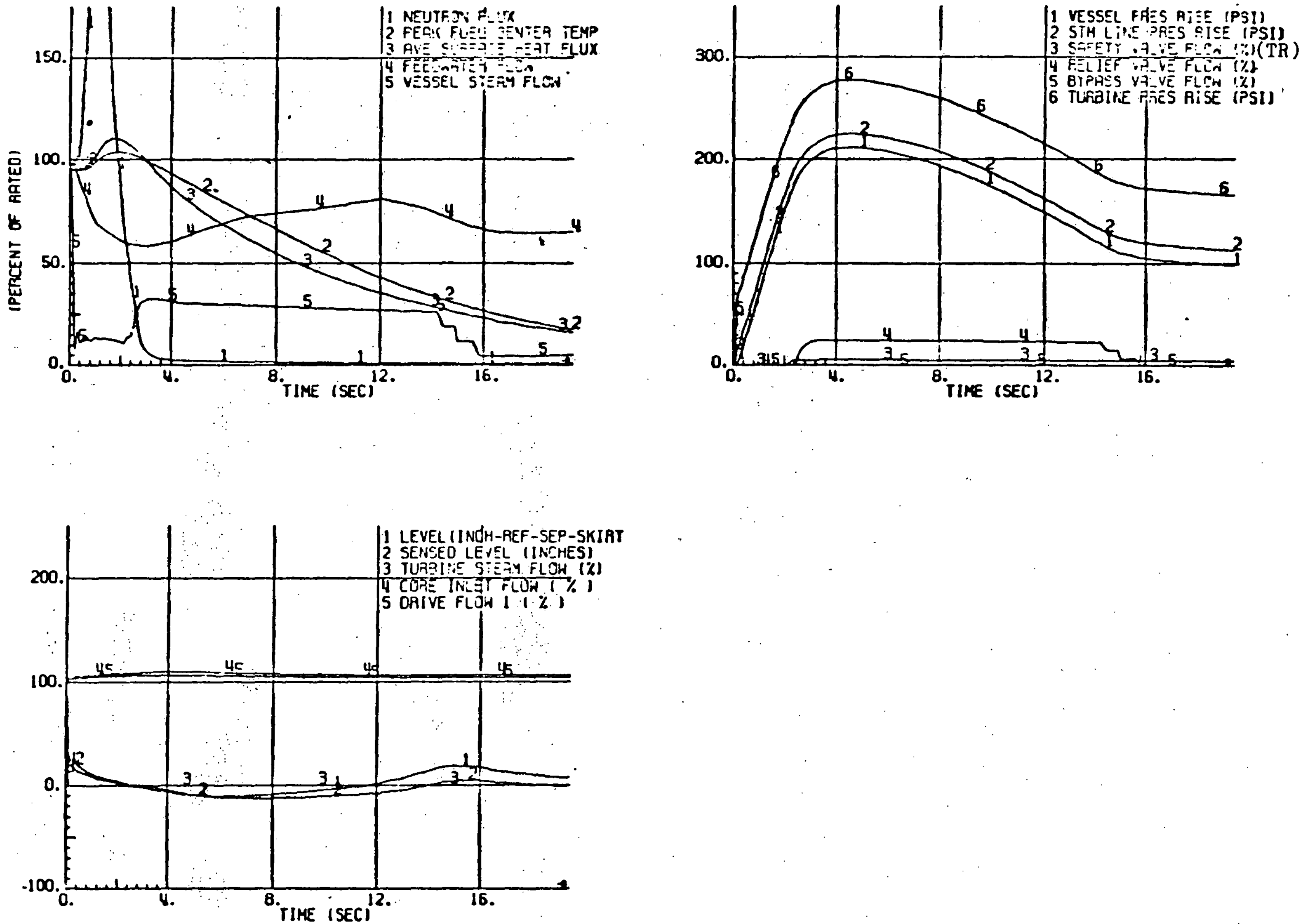


Figure 6-7b. Generator Load Rejection Without Bypass - Cycle 6 Limiting Case (98% Power)

6.3.3.2.2 Turbine Trip With Failure of the Bypass Valves

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 relief valves have automatically opened on high pressure. Figure 6-8a and 8b illustrate this transient for the EOC 6 -2000 MWd/t and the limiting Cycle 6 cases.

The parameters of concern are the peak steamline pressure margin to the first spring safety valve setpoint and the peak average surface heat flux correlated to MCPR.

For the limiting Cycle 6 case, the neutron flux (the precursor of heat flux) rises to a peak of 243.3% with a corresponding peak heat flux of 109.3%. The change in critical power ratio (ΔCPR) is presented in Table 4-1.

The peak steamline pressure for the worst Cycle 6 case is limited to 1214 psig as a result of the high-pressure actuation of the four electromatic valves and one combination of safety/relief valve, which provides a 26 psi margin to the 1240 psig setpoint of the first spring safety valve.

6.3.3.2.3 Loss of Feedwater Heating

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

Feedwater heating 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. The delay in the flow from the tripped feedwater heater to the feedwater sparger is ignored, thereby adding conservatism to the analysis.

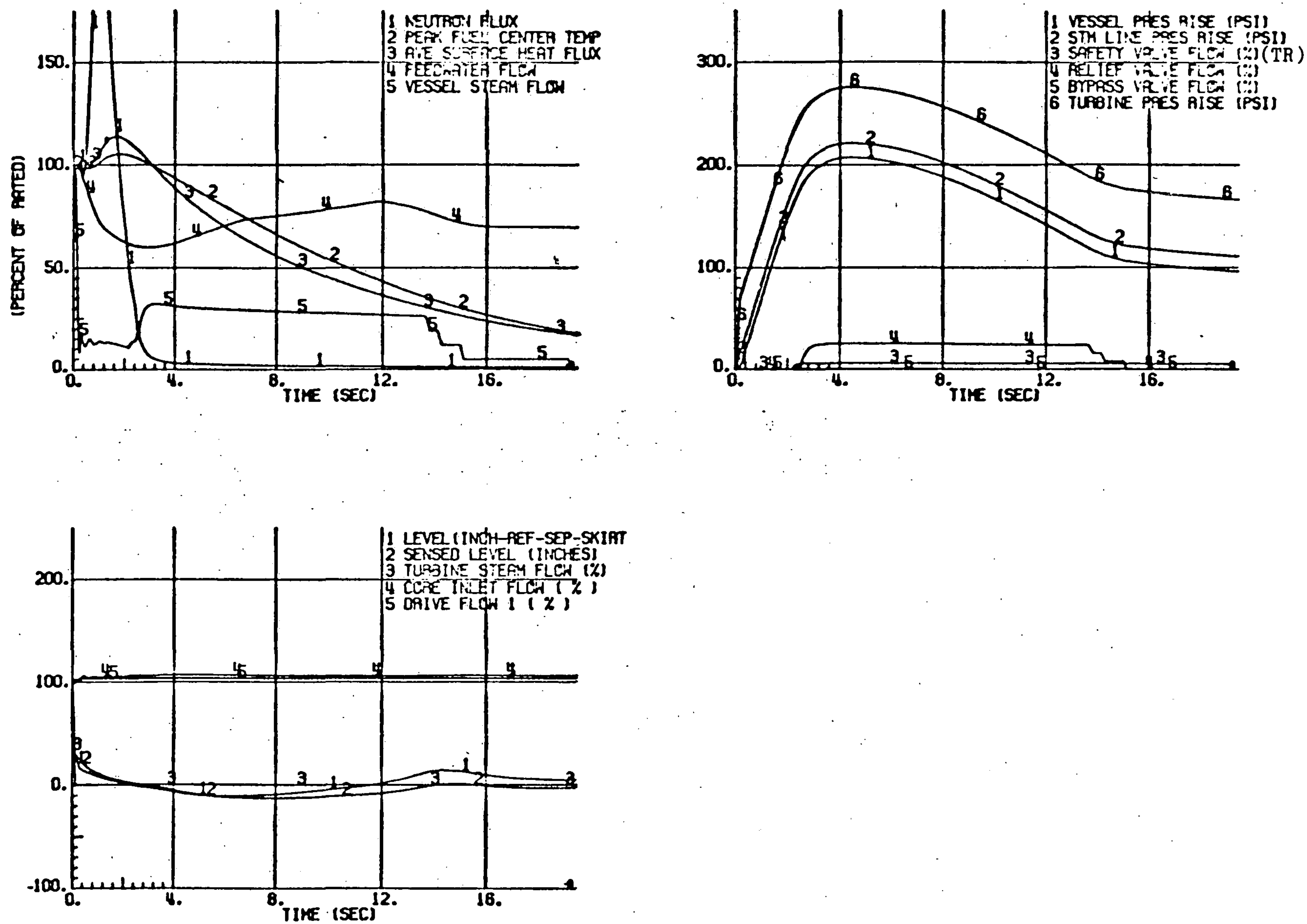


Figure 6-8a. Turbine Trip Without Bypass -2000 MWd/t Before EOC-6 (100% Power)

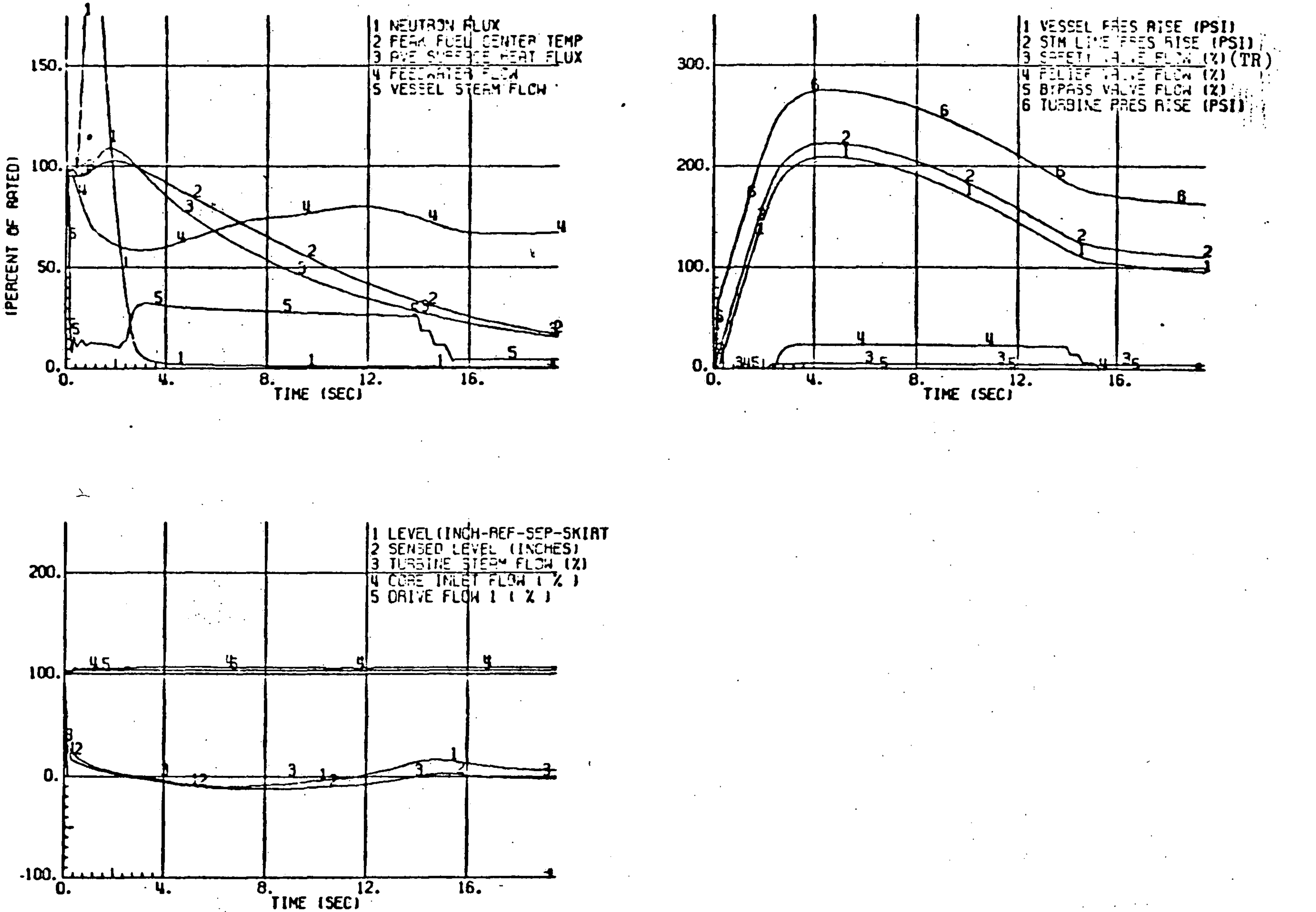


Figure 6-8b. Turbine Trip Without Bypass - Cycle 6 Limiting Case (98% Power)

Figure 6-9 shows the response of the plant to the loss of 145°F of the feedwater heating capability of the plant. This represents the maximum expected single heater (or group of heaters) to be tripped or bypassed by a single event. The reactor is assumed to be at maximum power conditions on manual flow control when the heating capability was lost. Note that in manual flow control mode the core flow remains essentially constant throughout the transient. Neutron flux increases above the initial value, however, in order to produce the same steam flow with the higher inlet subcooling. The reactor settles out with a neutron flux 120.8% of initial power and fuel average surface heat flux peaks at 119.5% of its initial value. The change in critical power ratio (ΔCPR) is presented in Table 4-1.

6.3.3.2.4 Plant Operation

The operating plan for Dresden 3, Cycle 6 is to start up and operate out to 2000 MWd/t before EOC-6 at 100% power, then reduce power to 98% by coasting down and operating at this power level out to EOC-6. Previous analyses of all rods out coastdown at EOC have consistently shown that operating limits as determined by pressure transients are not exceeded. These analyses are applicable to Dresden 3 for Cycle 6 operation.

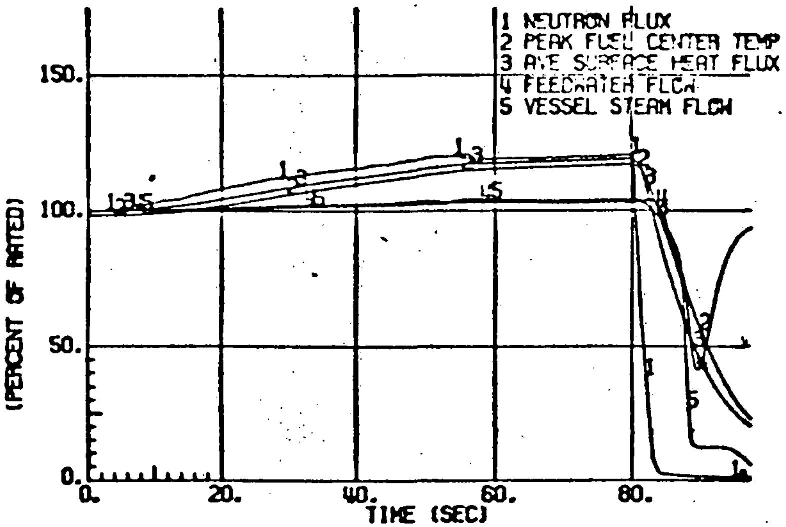
6.3.3.2.5 Rod Withdrawal Error

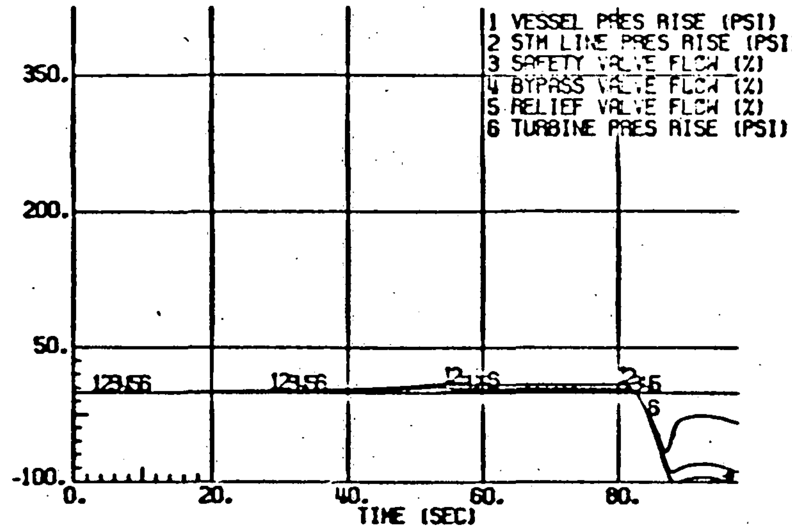
Assumptions and descriptions of rod withdrawal error are given in Reference 1. Table 6-3 shows the results of the worst case condition for Dresden 3 Reload 5. The rod block monitor (RBM) setpoint of 107% is selected to allow for failed instruments for the worst allowable situation. This case demonstrates that even if the operator ignores all alarms during the course of this transient, the critical power ratio (CPR) does not go below the 1.06 MCPR safety limit.

6.3.4 <u>ASME Vessel Pressure Code Compliance</u>

<u>All Main Steam Line Isolation Valve Closure-Flux Scram (Safety Valve Adequacy)</u>

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.





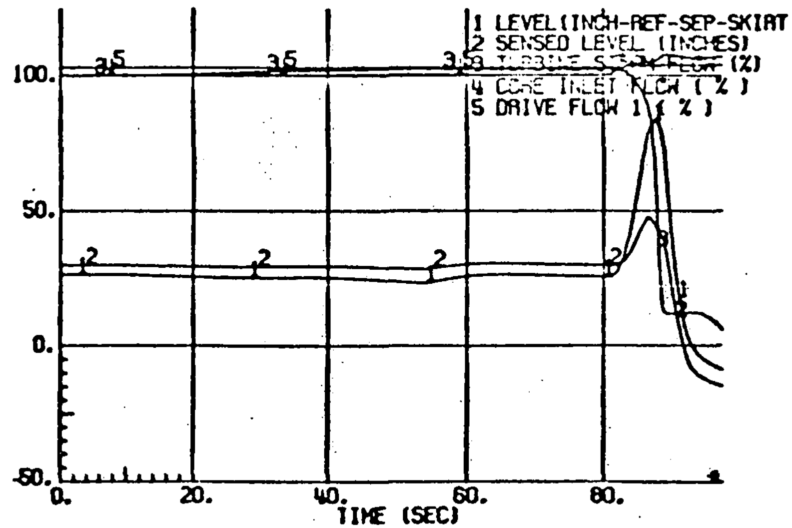


Figure 6-9. Loss of 145°F Feedwater Heating

Table 6-3

RWE AND RBM ANALYSIS

(WITH INSTRUMENT FAILURE)

Rod Block Setpoint	MCPR/ΔCPR* 7x7	8x8	Rod Position Ft Withdrawn
1.04	1.2829/0.1553	1.1690/0.1110	4.00
1.05	1.2610/0.1772	1.1550/0.1250	4.50
1.06	1.2410/0.1972	1.1420/0.1380	5.00
1.07	1.2098/0.2284	1.1042/0.1758	6.00
1.08	1.1480/0.2902	1.0130/0.2670	9.00
1.09	1.1329/0.3053	0.9764/0.3036	10.00
1.10	1.1240/0.3142	0.9520/0.3280	11.00

*Based on an estimated MCPR of: 1.4382 (7x7)
1.2800 (8x8)

Table 6-4

ROD WITHDRAWAL ERROR SUMMARY

Rod Position Ft Withdrawn	MLHGR. Kw/ft 8x8	7x7	MCPR 8x8	7x7
0	13.40	14.54	1.280	1.438
2	13.43	14.58	1.236	1.376
4	14.95	15.00	1.169	1.283
6	14.50	14.85	1.104	1.210
8	13.68	14.61	1.060	1.167
10	15.78	16.65	0.976	1.133
12	17.31	18.46	0.930	1.120

The Dresden 3 pressure relief system includes 4 electromatic relief valves, one Target Rock dual-purpose safety/relief valve and 8 spring safety valves located on the main steamlines 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 valve setpoint must not be greater then 105% of vessel design pressure (1313 psig for a 1250 psig vessel).

Dresden 3 safety/relief and spring 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 relief valves are assumed to be inactive.

The event which satisfies this specification is the closure of all main steamline isolation valves with indirect (flux) scram. The initial conditions assumed are those specified in Table 6-1. Figures 6-10a and b graphically illustrate the event for the two cases. An abrupt pressure and power rise occurs as soon as the reactor is isolated. For the worst case, the safety valves open to limit the pressure rise in the steamline at the valves to 1277 psig and at the bottom of the vessel to 1311 psig. This response provides a 64 psi margin to the vessel code limit to 1375 psig. Thus, requirement (a) is satisfied and adequate overpressure protection is provided by the pressure relief system.

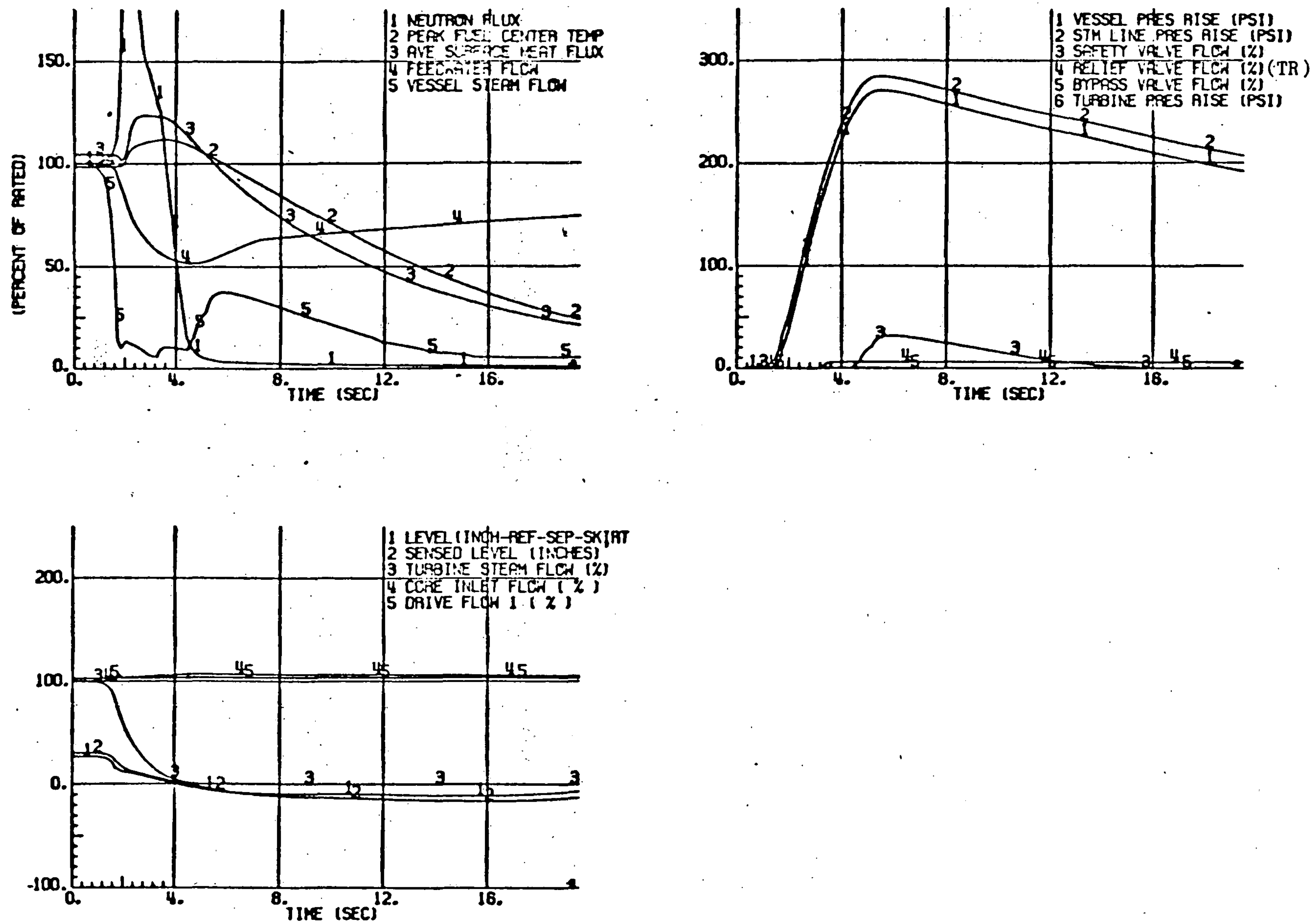


Figure 6-10a. MSIV Closure Flux Scram -2000 MWd/t Before EOC-6

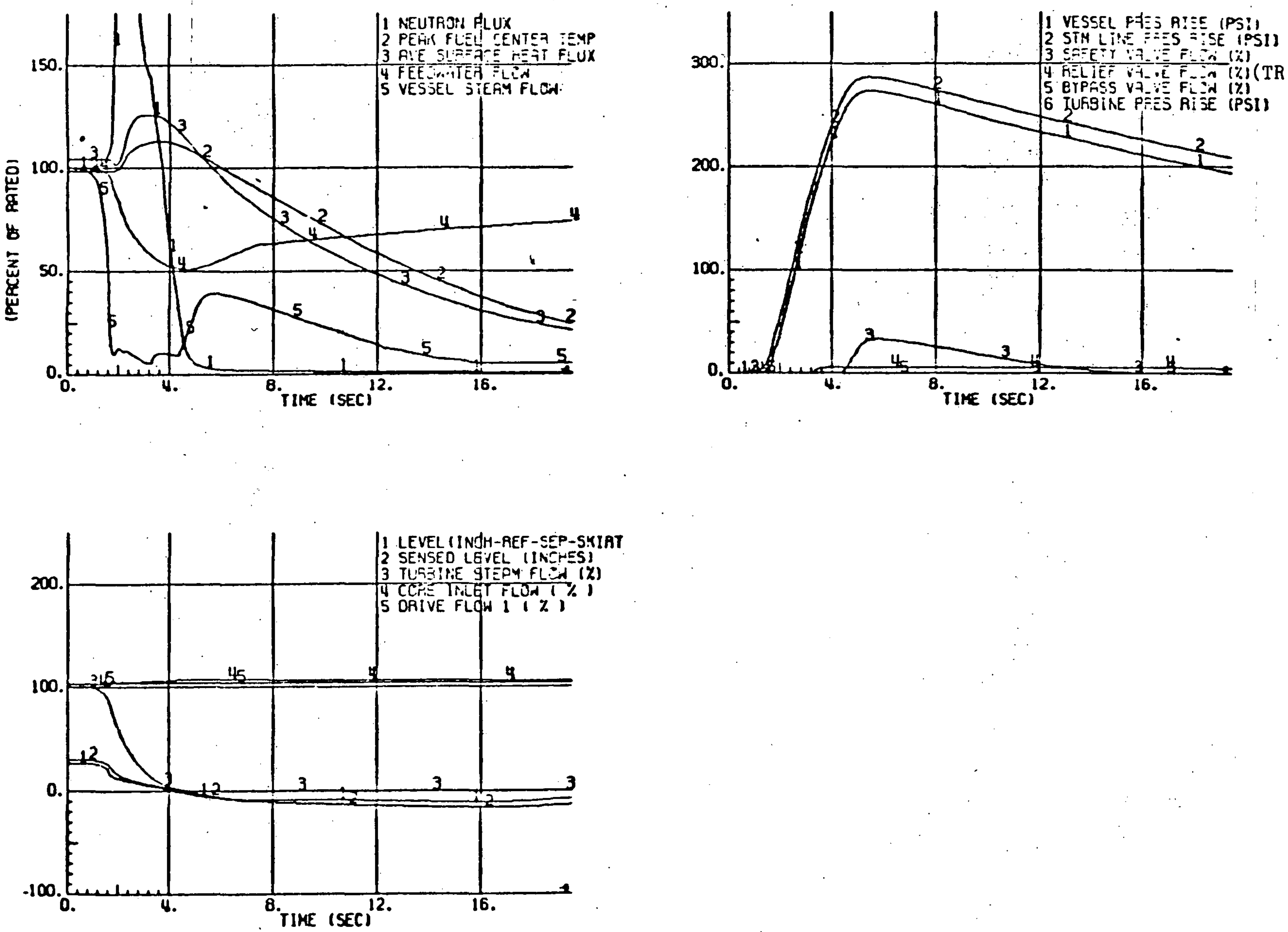


Figure 6-10b. MSIV Closure Flux Scram - Cycle 6 Limiting Case

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 Dresden 3 Reload-5 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	100% Rod Line - Natural Circulation Power
Decay Ratio, X_2/X_0	
8x8 Channel	0.09
7x7 Channel	0.01

At this most responsive condition, the most responsive channels are clearly within the bounds of the ultimate performance criteria of $\leq$ 1.0 decay ratio.

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 6-11. The most responsive case is again 100% rod line - natural circulation condition.

Reactor Core Stability	100% Rod Line - Natural Circulation Power
Decay Ratio, X_2/X_0	0.48

These calculations show the reactor to be in compliance with the ultimate performance criteria, including the most responsive condition at 100% rod line - natural circulation power.

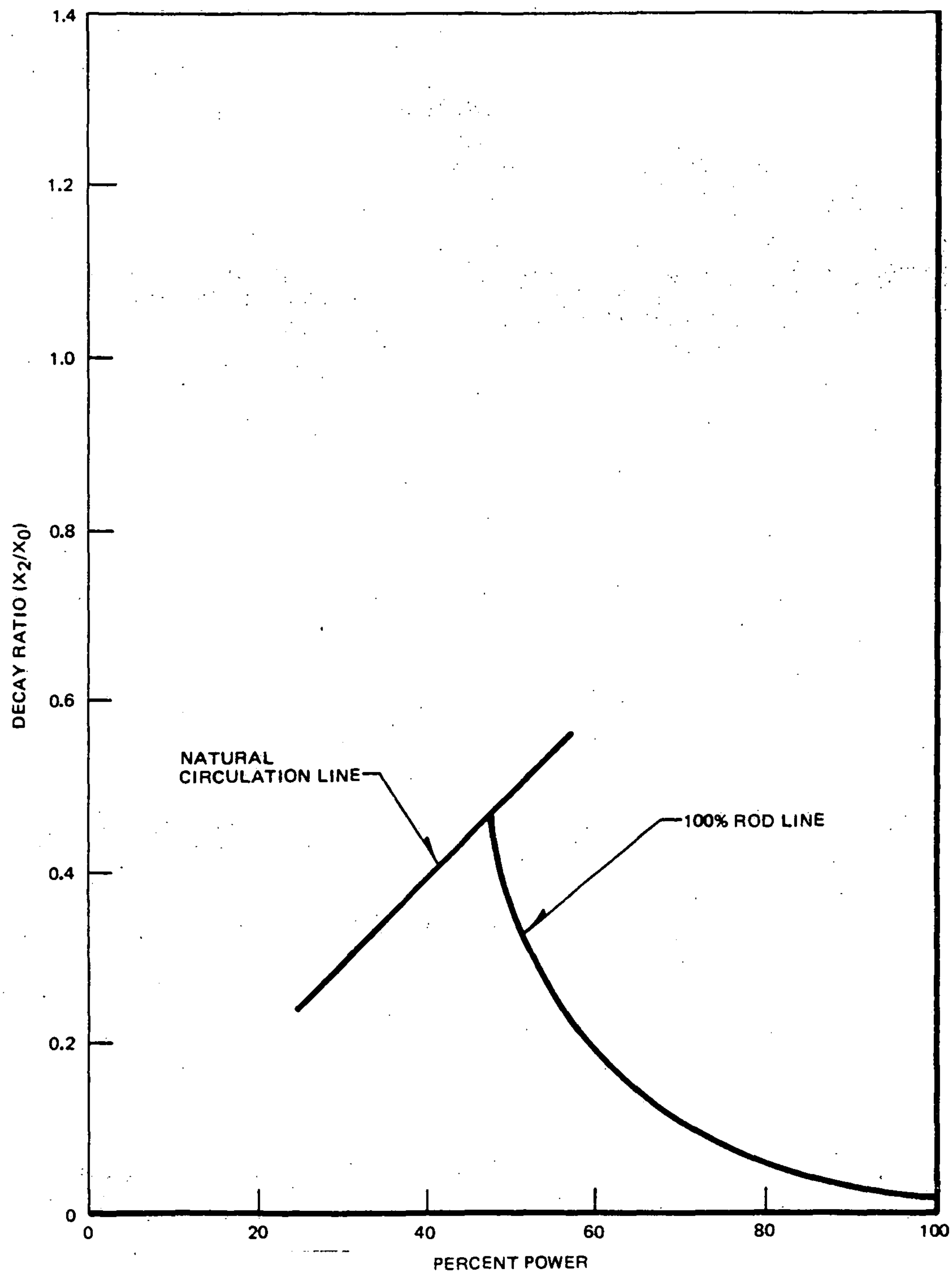


Figure 6-11. Decay Ratio

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_2/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 $\leq$ 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 (55% power, 39% flow) are presented below.

Reactor Core Performance	Rated Conditions	Low End of Flow Control Range
Decay Ratio	<0.02	0.25

As noted earlier, Figure 6-11 describes the variation of decay ratio over the entire power flow range.

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).

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