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Amdt to OL/Change to Tech Specs: Consisting of revisions with regard to Cycle 3 operational & safety limits.........

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NRC FORM 195 (2-76)

IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office
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VICE PRESIDENT – ENGINEERING

January 31, 1977
IE-77-220

50-331

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20545

Dear Mr. Rusche:

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 Cycle 3 operational limits and safety limits.

In October 1976 Iowa Electric met with your staff to advise them that Iowa Electric did not then plan to make application for license amendment authorizing Cycle 3. We were then of the view that licensing action by NRC to authorize Cycle 3 would not be required; that the reload analyses would show that Technical Specification changes would not be needed and that no unreviewed safety questions would be associated with the reload. Our expectations, however, were not confirmed by the analyses. Upon receipt of the reload analyses, we concluded that Technical Specification changes would be required in order to assure you of maintenance of the same margins of safety as defined in the bases for the Technical Specifications. The analyses confirm, however, that no "unreviewed safety question" will be presented by this application; accordingly, we believe that this application does not involve significant hazards considerations.

This application incorporates General Electric Licensing Topical Report Generic Reload Application for 8x8 Fuel NEDO-20360 (Rev 1 Supplement 3 dated September 25, 1975) and its proprietary supplement NEDO-20360-IP (Rev 3 dated September 25, 1975) by reference pursuant to 10CFR50.32.

This application consists of:

1) Submittal in the format of Appendix A to NEDO 20360 which includes appropriate safety and transient analyses.

2) Proposed Technical Specifications reflecting the results of the above safety and transient analyses.

The DAEC is presently scheduling a shutdown March 12, 1977 with restart planned for April 16, 1977.

Iowa Electric is evaluating drilling of the used fuel for Cycle 3. The decision whether or not to drill will depend upon the availability of time to accomplish the drilling and any related licensing proceedings during the refueling shutdown. We will be in touch with you shortly concerning our plans for drilling.

Three signed and 40 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: [signature]
Lee Liu
Vice President, Engineering

LL/KAM/ms

cc: K. Meyer
D. Arnold
R. Lowenstein
J. Shea (NRC)
L. Root

Subscribed and Sworn to before me on this 1st day of February, 1977.

Marjorie E. McDonald
Notary Public in and for the State of Iowa

NOTARIAL SEAL IOWA
Marjorie E. McDonald
My Commission Expires
September 30, 1979

PROPOSED CHANGE RTS-80 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

Specifications 1.1, 1.2, 3.2, 3.3, 3.6, 3.12 and 5.2 contain Safety Limits, Limiting Conditions for Operation and Bases which are applicable for cycle 2.

II. Proposed Changes in Technical Specifications

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

Change as indicated in the attached sheets. These changes in general are as follows:

Change Minimum Critical Power Ratio (MCPR) safety limit from "$\geq$ 1.07" to "$\geq$ 1.06" on sheet 1.1-1. In the balance of the Technical Specifications change the value of MCPR from "1.07" to "safety limit".

Change the standard deviation range of the Bypass Void Effect on TIP from "3.85% to 5.05%" to "3.58% to 4.15%".

Change the equation for determining the trip level setting for the Rod Block Monitor (Flow Biased) from "$\leq (0.66W + 41)\frac{(*)}{(TPF)}$" to "$\leq (0.66W + 39)\frac{(*)}{(TPF)}$".

Change the Bases for the Safety/Relief Valves.

Change the description of the fuel in the reactor from "fuel assemblies of either 49 or 63 fuel rods each" to "fuel assemblies of an approved design".

III. Justification for Proposed Change

This change is proposed in order to incorporate into the Technical Specifications the new safety limits and limiting conditions for operation governing cycle 3 operation of the DAEC. The justification for the changes is contained in "General Electric Boiling Water Reactor Reload Number 2 Licensing Submittal," NEDO-21082-02, Class I, January 1977.

Other changes of an administrative nature were made where the value of MCPR was repeated on subsequent pages to reduce the possibility of missing these changes in future revisions to the Technical Specifications.

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.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the inter-related variables associated with fuel thermal behavior

Objective:

To establish limits which ensure the integrity of the fuel cladding.

Specifications:

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.06 shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit (Reactor Pressure ≤ 785 psig or Core Flow ≤ 10% of Rated)

When the reactor pressure is ≤ 785 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limits from being exceeded.

Specifications:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trips

1. APRM High Flux Scram When In Run Mode.

For operation with a peaking factor less than 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM scram trip setpoint shall be as shown on Fig. 2.1-1 and shall be:

$$S \leq (0.66W + 54)$$

with a maximum setpoint of 120% rated power at 100% rated recirculation flow or greater.

TABLE 1.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
Bypass Void Effect on TIP	3.58 to 4.15
R-Factor	1.6
Critical Power	3.6

during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than 2.61 (7 x 7 array) or 2.43 (8 x 8 array).

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than or equal to safety limit when the transient is initiated from MCPR $\geq$ values as indicated in Table 3.12.2.

2. <u>APRM High Flux Scram (Refuel or Startup & Hot Standby Mode)</u>.

For operation in these modes the APRM scram setting of 15 percent of rated power and the IRM High Flux Scram provide adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worths of individual rods are very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise.

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 880 psig.

3. <u>APRM Rod Block (Run Mode)</u>

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given power level at constant recirculation flow rate, and thus prevents a MCPR less than safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents excessive reactor power level increase resulting from control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the safety limit, thus preserving the APRM rod block safety margin.

4. IRM

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that the heat flux is in equilibrium with the neutron flux, and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents has been analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

B. Scram and Isolation on Reactor Low Water Level

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. Analyses show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than safety limit in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 21 inches below the normal operating range and is thus adequate to avoid spurious scrams.

C. Scram - Turbine Stop Valve Closure

The turbine stop-valve closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves.

With a scram setting at 10 percent of valve closure, the resultant increase in surface heat flux is such that MCPR remains above safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is by-passed when turbine steam flow is below 30 percent of rated, as measured by the turbine first stage pressure.

D. Turbine Control Valve Fast Closure (Loss of Control Oil Pressure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection. It prevents MCPR from becoming less than safety limit for this transient.

E. F. and J. Main Steam Line Isolation on Low Pressure, Low Condenser Vacuum, and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 880 psig has been provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity. Operation of the reactor at pressures lower than 880 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients

2.2 BASES

Reactor Coolant System Integrity

The discussion in section 3.6.D and 4.6.D Bases is applicable for discussion of pressure relief.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 135 psig.

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TABLE 3.2-C

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq (0.66W + 42) \left(\frac{*}{P.F}\right)$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$ 12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq (0.66W + 39) \left(\frac{*}{TPF}\right)$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$ 5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$ 108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

* 2.61 (7 x 7 array) or 2.43 (8 x 8 array)

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to safety limit. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, six IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criterion is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than safety limit.

The RMB rod block function provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 5 indicated on scale for APRM's and 5/125 full scale for IRM's.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
e. If Specifications 3.3.B.3a through d cannot be met, the reactor shall not be started, or if the reactor is in the run or startup modes at less than 30% rated power, it shall be brought to a shutdown condition immediately.	1) The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.
	2) The RWM computer on line diagnostic test shall be successfully performed.
f. The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range.	3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.
	4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.
	c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified.
4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:	5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).
a. Both RBM channels shall be operable: or	
b. Control rod withdrawal shall be blocked: or	
c. The operating power level shall be limited so that the MCPR will remain above safety limit assuming a single error that results in complete withdrawal of any single operable control rod.	

MCPR from becoming less than safety limit. The limiting power transient is that resulting from a turbine trip without bypass. Analysis of this transient shows that MCPR remains greater safety limit.

After initial fuel loading and subsequent refuelings when operating above 950 psig all control rods shall be scram tested within the constraints imposed by the Technical Specifications and before the 40% power level is reached.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on DAEC.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an

3.6.D & 4.6.D BASES:

Safety and Relief Valves

The pressure relief system has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4.1 of Subsection 4.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME code requirements is presented in Subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Report in FSAR Amendment No. 3 (response to AEC Question H.1.1) and is reverified in individual reload analyses.

Six relief valves and two safety valves are installed. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in greater than an 80 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 greater than 40 psi 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

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 3.12-1.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 3 and in References 4 and 5, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

1. Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.12.C are derived from the established fuel cladding integrity Safety Limit MCPR value, and an analysis of abnormal operational transients(1). For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The limiting transient which determines the required steady state MCPR limit is the turbine trip with failure of the turbine bypass. This transient yields the largest MCPR. When added to the safety limit MCPR value the required minimum operating limit MCPR of Specification 3.12.C are obtained.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4.4 of NEDO-20360(2) and on core parameters shown in Table 4.2.4 of Supplement 1 to Reference 1.

The evaluation of a given transient begins with the system initial parameters shown in Table 4.1 (page 4-7) of Reference 1 that are input to a GE core dynamic behavior transient computer program described in NEDO-10802(6). Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566(7). The principal result of this evaluation is the reduction in MCPR caused by the transient.

TABLE 3.12-2

MCPR LIMITS

Fuel Type	Exposure Remaining to End of Cycle		
	> 2000 MWD/T	≤ 2000 MWD/T, > 1000 MWD/T	≤ 1000 MWD/T to E.O.C.
7 x 7	1.27	1.34	1.35
8 x 8	1.35	1.42	1.43

3.12 REFERENCES

1. Duane Arnold Energy Center "Safety Analysis with Bypass Holes Plugged", June 9, 1975 and Supplement 1, June 16, 1975.

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. Duane Arnold Energy Center Reload Number Two Licensing Submittal, NEDO-21082-02, Class I, January 1977.

5.2 REACTOR

1. The core shall consist of not more than 368 fuel assemblies of an approved fuel design.

2. The reactor core shall contain 89 cruciform shaped control rods. The control material shall be boron carbide powder (B_4C).

NEDO-21082-02
Class I
January 1977
APPENDIX A to
NEDO-20360
REVISION 1
Supplement 3
September 1975

GENERAL ELECTRIC BOILING WATER REACTOR

RELOAD NUMBER 2 LICENSING SUBMITTAL

DUANE ARNOLD ENERGY CENTER

License Number DPR-49

Docket Number 50-331

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT

Please Read Carefully

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LIST OF ILLUSTRATIONS

LIST OF TABLES

1. INTRODUCTION

This document provides the supplemental information for Reload Number 2 at the Duane Arnold Energy Center (DAEC). 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 100 8x8 bundles having enrichment of 2.74 wt % U-235.

The objective of this outage is to load the reactor core to ensure sufficient reactivity to operate the 368-element core for an approximate 12-month cycle. The analysis for this submittal is done with the core support plate holes plugged and all 100 8x8 reload -2 bundles drilled. The remaining 268 bundles are not drilled in the analyses.

This licensing submittal provides analysis to support exposure dependent minimum critical power ratio (MCPR) operating limits. The Δ MCPR 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 BOC to 2000 MWD/t before EOC, 2000 MWD/t to 1000 MWD/t before EOC, and 1000 MWD/t to EOC. The results of these analyses continue to demonstrate the ability of the plant to operate safely within the constraints of the calculated MCPR operating limits.

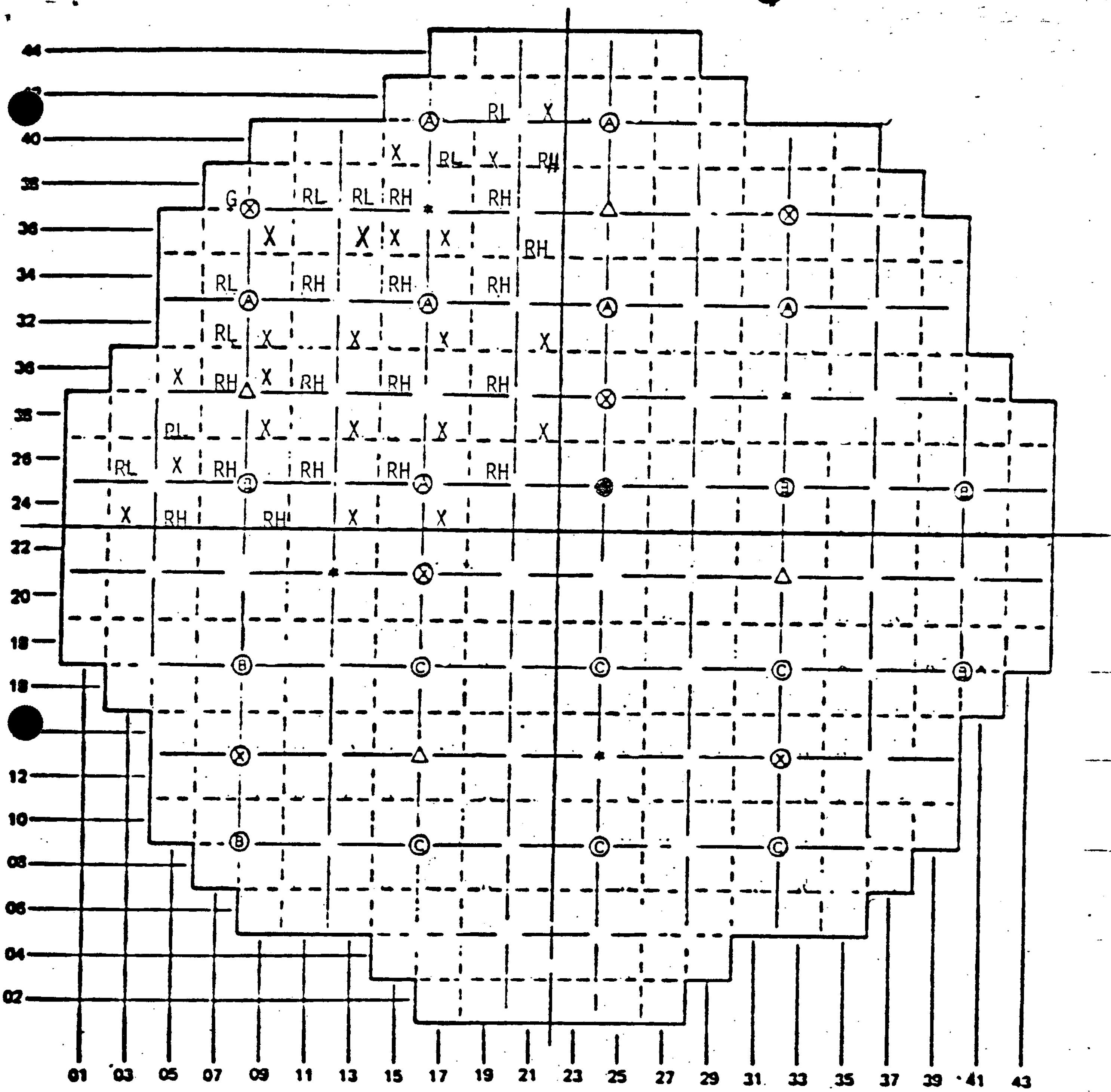
2. SUMMARY

The design reference core configuration for this license 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	152
Type 3 7D212	28
Interim Reload 7D230	4
Reload 1	
8D274H	52
8D274L	32
Reload 2	
8D274H	68
8D274L	32
TOTAL	368



BLANK – INITIAL CORE 7D212
G – INTERIM RELOAD 7D230
X – RELOAD 1 8D274
RH – RELOAD 2 8D274H
RL – Reload 2 8D274L

● 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-2 Design Reference Core Loading - Quarter Core Mirror Symmetry Upper Left Quadrant Only Shown

3. MECHANICAL DESIGN

The two types of Reload 2 fuel which will be employed have the same mechanical configuration and fuel bundle enrichments as the 8D274L and the 8D274H fuel assemblies described in Reference 1. Reload 2 incorporates the improved water rod design described in Sections 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 2 fuel assemblies will be provided with two bypass flow holes in the lower tie plate. Justification for operation with bypass flow holes in the reload fuel assemblies is given in Reference 2.

4. THERMAL-HYDRAULIC ANALYSES

Discussions of thermal-hydraulic design requirements, hydraulic models, statistical analysis and uncertainties, and thermal hydraulics of mixed core loading are given in Section 4 of Reference 1. The analysis applicable to Duane Arnold, Cycle 3 with bypass flow holes plugged, is given below.

4.1 STATISTICAL ANALYSIS

Bounding statistical analysis was performed which provides a conservative safety limit Minimum Critical Power Ratio (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 DAEC.

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

The power distribution was generated by arranging the control rod pattern so that as many fuel assemblies as possible are at and near the MCPR limit as per the procedure described in Appendix IV, General Electric Thermal Analysis Basis (GETAB) Licensing Topical Report (Reference 3). For comparison purposes, actual

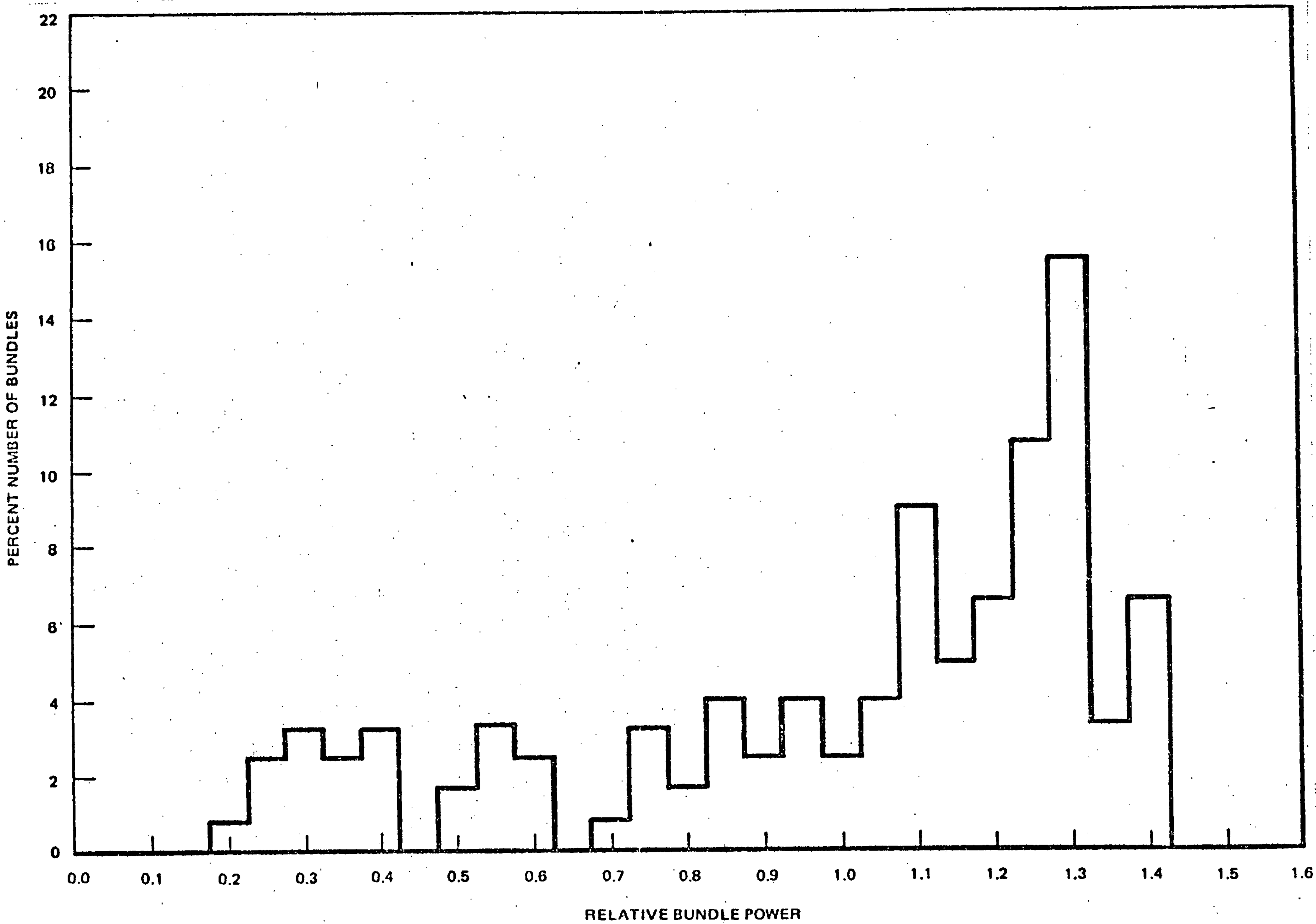


Figure 4-1. Relative Bundle Power Histogram for Power Distribution Used in Statistical Analysis

operating power distributions of typical BWR reload cores are shown in Figures 4-2 and 4-3.

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 and the nominal values of parameters used in the bounding statistical analysis are listed in Tables 4-1 and 4-2.

4.2 ANALYSIS OF ABNORMAL OPERATIONS 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 4-3.

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.

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 1.27 for 7x7 and 1.35 for 8x8 fuels from BOC3 to 2GWD/T before EOC3, 1.34 for 7x7 and 1.42 for 8x8 fuels to 1GWD/T before EOC3, and 1.35 for 7x7 and 1.43 for 8x8 fuels to EOC3.

4.3 TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

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

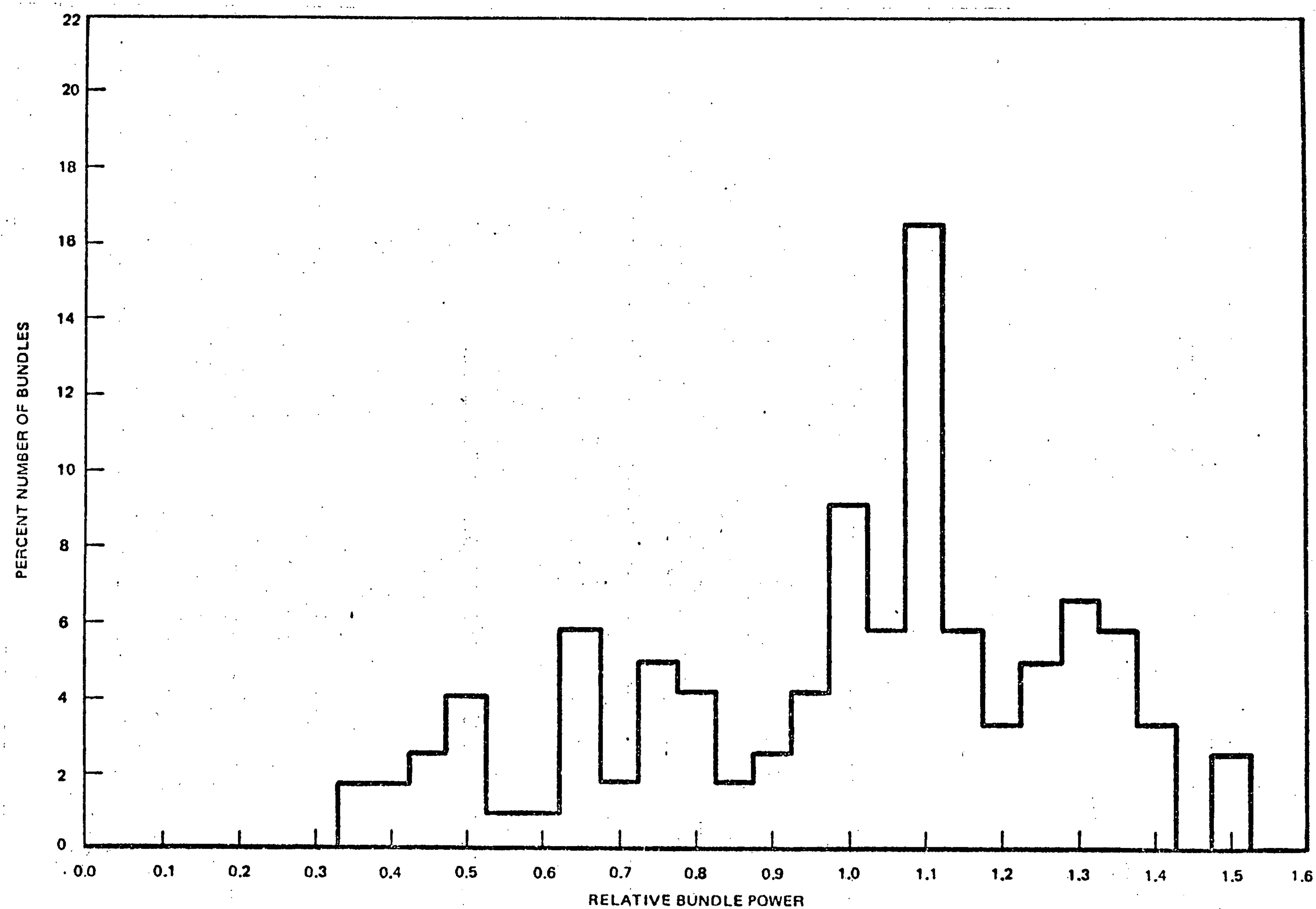


Figure 4-2. Relative Bundle Power Histogram for a Typical BWR Operating Power Distribution

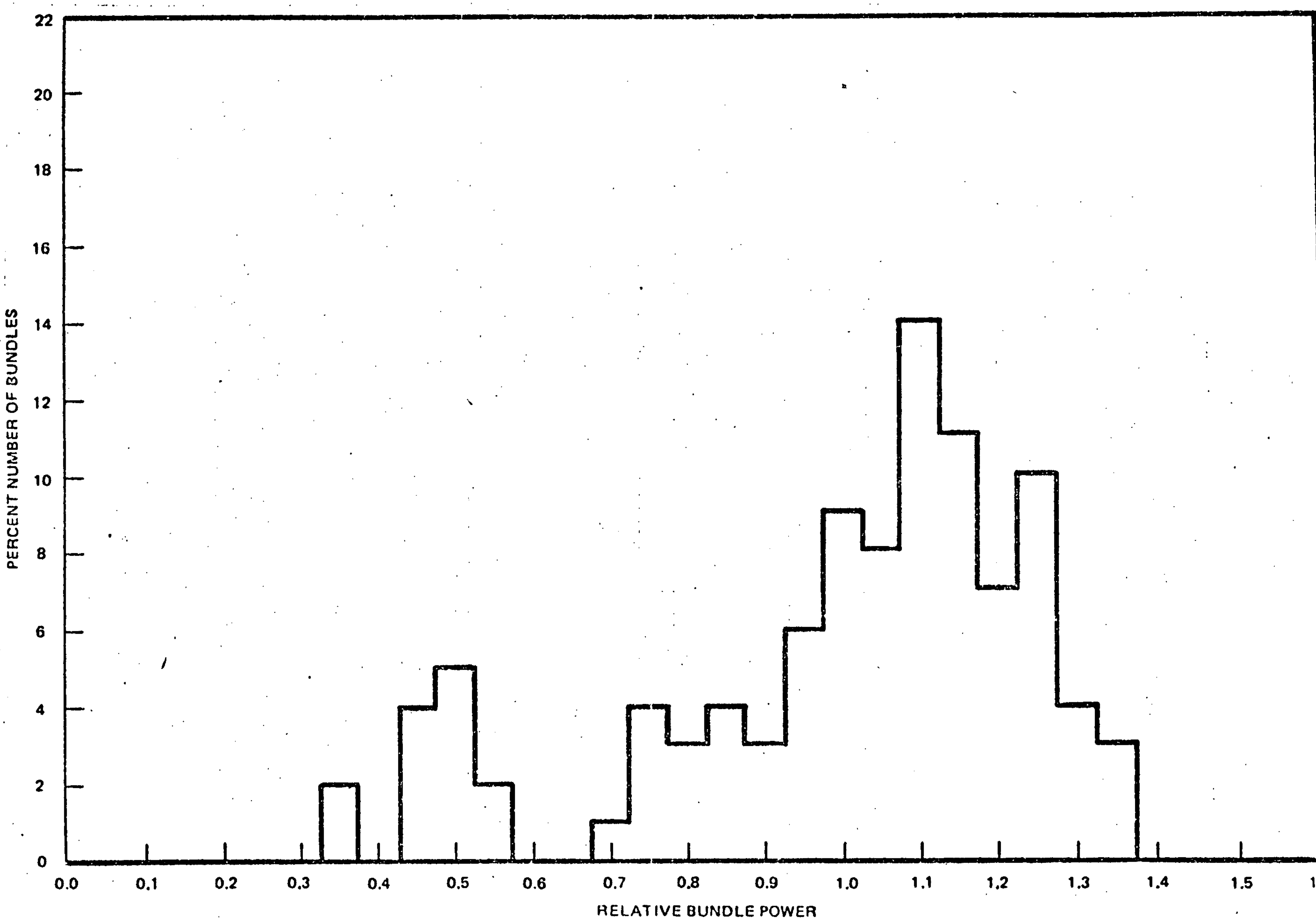


Figure 4-3. Relative Bundle Power Histogram for a Typical BWR Operating Power Distribution

Table 4-1

DESCRIPTION OF UNCERTAINTIES

Quantity	Standard Deviation (% of Point)	Comment
Feedwater Flow	1.76	This is the largest component of total power uncertainty.
Feedwater Temperatures	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.
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 Local Power Range Monitor (LPRM) system. LPRM readings are used to correct the power distribution and 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.

Table 4-1 (Continued)

Quantity	Standard Deviation (% of Point)	Comment
Bypass Void Effect on TIP	3.58-4.15	This accounts for additional uncertainty due to the bypass void content resulting from plugging of the core support plate leakage augmentation holes. The tip uncertainty introduced by the bypass-voids is zero in the bottom two thirds of the core (no-boiling in bypass region) and increases from 3.58% to 4.15% in the upper third of the core. The tip variations due to the bypass void in a given Monte Carlo trial are assumed (conservatively) to be perfectly correlated axially, so that each node receives an increment of the same sign, proportional to the corresponding nodal uncertainty.
R Ractor	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.
Critical Power	3.6	Uncertainty in the General Electric Critical Quality Boiling Length Correlation (GEXL) correlation in terms of critical power.

Table 4-2
NOMINAL VALUES OF PARAMETERS USED IN
THE STATISTICAL ANALYSIS

Core Thermal Power	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1078 ft^2
R-Factor	1.098 (7x7)
	1.100 (8x8)

Table 4-3

SUMMARY OF RESULTS

LIMITING ABNORMAL OPERATIONAL TRANSIENTS

	Event	ΔCPR 7x7	ΔCPR 8x8
1.	Turbine Trip without Bypass, Rated Conditions, EOC 3-2 GWD/T	0.21	0.29
2.	Turbine Trip without Bypass, Rated Conditions, EOC 3-1 GWD/T	0.28	0.36
3.	Turbine Trip without Bypass, Rated Conditions, EOC 3	0.29	0.37
4.	Loss of 100°F FW Heater, Rated Conditions	0.15	0.18
5.	Rod Withdrawal Error (RBM to 105%)	0.15	0.16

Table 4-4

GETAB TRANSIENT ANALYSIS

INITIAL CONDITION PARAMETERS EOC

	7x7	8x8
Peaking Factors (Local, Radial and Axial)	1.24, 1.16, 1.40	1.22, 1.22, 1.40
R-Factor	1.100	1.098
Bundle Power, MWt	4.931	5.182
Nonfuel Power Fraction	0.04	0.04
Core Flow, Mlb/hr	49.0	49.0
Bundle Flow, 10^3 lb/hr	132.0	118.9
Reactor Pressure, psia	1035	1035
Inlet Enthalpy, Btu/lb	526.3	526.3
Initial MCPR	1.35	1.44

Table 4-4 (continued)

GETAB TRANSIENT ANALYSIS
INITIAL CONDITION PARAMETERS
(EOC-1 GWD/T EVALUATIONS)

	7x7	8x8
Peaking Factors (local, radial and axial)	(1.24, 1.17, 1.40)	(1.22, 1.23, 1.40)
R-Factor	1.100	1.098
Bundle Power, MWt	4.974	5.225
Non-Fuel Power Fraction	0.04	0.04
Core Flow, Mlb/hr	49.0	49.0
Bundle Flow, 10^3 lb/hr	131.7	118.6
Reactor Pressure, psia	1035.0	1035.0
Inlet Enthalpy, Btu/lb	526.3	526.3
Initial MCPR	1.34	1.43

Table 4-4 (continued)

GETAB TRANSIENT ANALYSIS
INITIAL CONDITION PARAMETERS
(2 GWD/T BEFORE EOC-3 EVALUATIONS)

	7x7	8x8
Peaking Factors (local, radial and axial)	(1.24, 1.23, 1.40)	(1.22, 1.29, 1.40)
R-Factor	1.100	1.098
Bundle Power, MWt	5.207	5.492
Non-fuel Power Fraction	0.04	0.04
Core Flow, Mlb/hr	49.0	49.0
Bundle Flow, 10^3 lb/hr	130.2	116.8
Reactor Pressure, psia	1035.	1035.
Inlet Enthalpy, Btu/lb	526.3	526.3
Initial MCPR	1.27	1.35

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:

a. reactivity control characteristics; and

b. 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 100 Reload-2 fuel bundles by discharging a like number of fuel bundles from the Cycle 2 core.

5.1.1 Core Effective Multiplication, Control System Worth and Reactivity Coefficients

A calculation of the typical nuclear characteristics of the reconstituted core is given in Table 5-1. The nuclear characteristics of the Reload-2 fuel bundles are identical to those previously loaded. Therefore, the total control system worth, 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.

Table 5-1

NUCLEAR CHARACTERISTICS OF THE DESIGN REFERENCE CORE

Core Effective Multiplication and Control System Worth (No Voids, 20°C)	
Beginning of Cycle (BOC) k_{eff}	
Uncontrolled	1.1120
Fully Controlled	0.9490
Strongest Control Rod Out	0.9805
R, Maximum Increase in Core Reactivity with Exposure Into Cycle, Δk	0.0057
Reactivity Coefficients, Range During Operating Cycle	
Steam Void Coefficient at Average Voids $(\Delta k/k)/\Delta V$, 1/% Void	-1.564×10^{-3} to -1.426×10^{-3} to
Power Coefficient at Rated Conditions $(\Delta k/k)/(\Delta P/P)$	≤ -0.066
Fuel Temperature Coefficient at 650°C $(\Delta k/k)/\Delta T$, 1/°F	-1.127×10^{-5} to -1.245×10^{-5}

A minimum shutdown margin of 0.0138 Δk is calculated at the most reactive condition throughout the subsequent operating cycle with the strongest control rod fully withdrawn and all other rods fully inserted. The BOC-2 shutdown margin is 0.0195 Δk. Thus R, the difference between the BOC-2 and the minimum shutdown margin is 0.0057 Δk.

5.1.3 Standby Liquid Control System

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

6. SAFETY ANALYSIS

6.1 INTRODUCTION

The safety analysis for reloads consists of three categories: (1) generic safety analysis, which is applicable to all reloads; (2) bounding analysis; and (3) 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 (Reference 3) 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 Steam Line Break Accident

The consequences of the main 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 a reload, the referenced analysis applies.

6.3.2.2 Refueling Accident

The description and analyses of the refueling accident provided in the Final Safety Analysis Report (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-2, 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 analyses given in Reference 1 are applicable to this reload. These analyses were performed for the Reload-2 fuel in accordance with Appendix K of 10CFR Part 50.

Table 6-1 shows the variation of Maximum Average Planar Linear Head Generation Rate (MAPLHGR), Peak Cladding Temperature (PCT), and maximum oxidation fraction versus exposure for the Duane Arnold Energy Center Reload-2 fuel with the bypass flow holes plugged.

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:

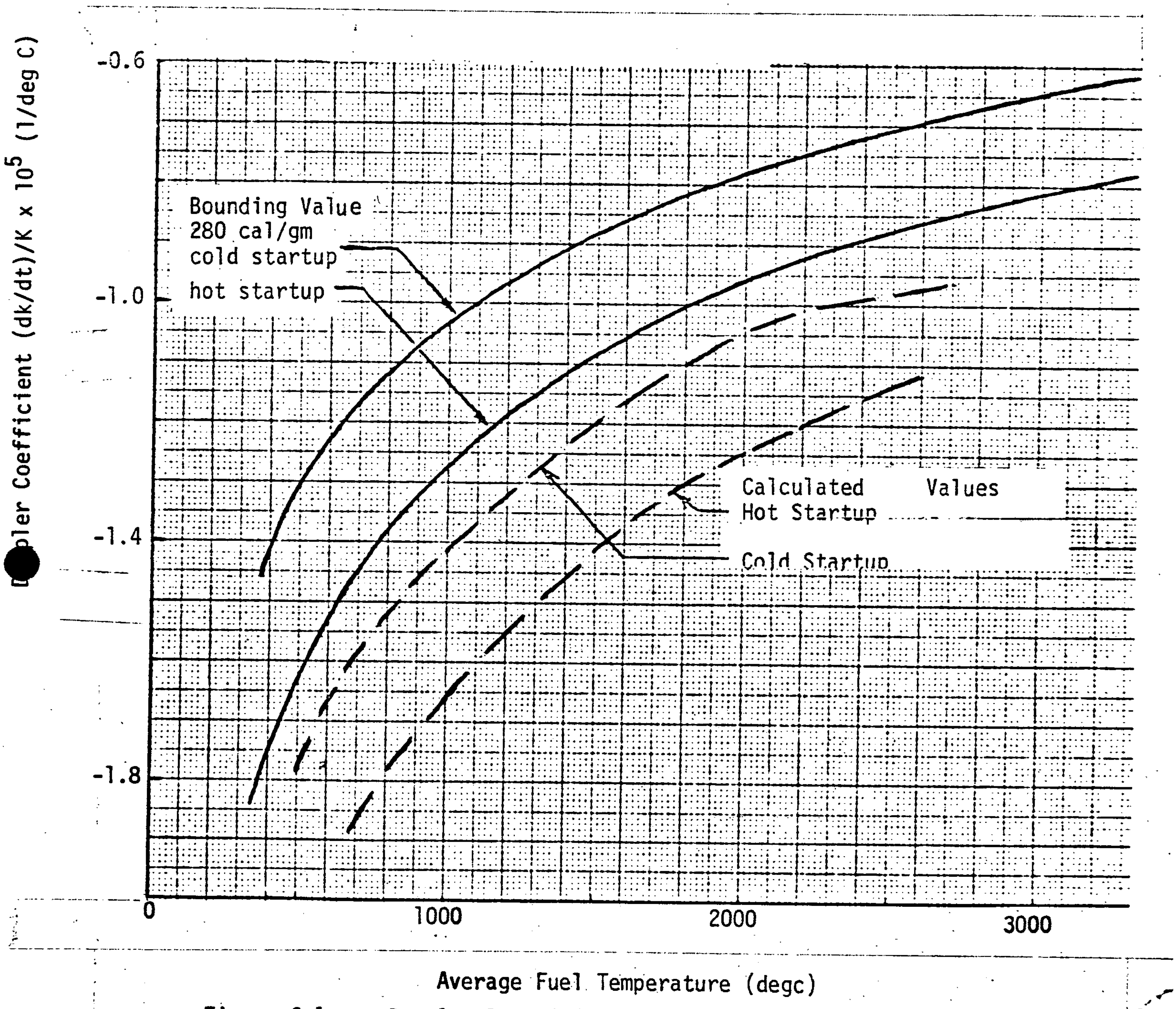


Figure 6-1 Doppler Reactivity Coefficient versus Average Fuel Temperature as a Function of Moderator Condition

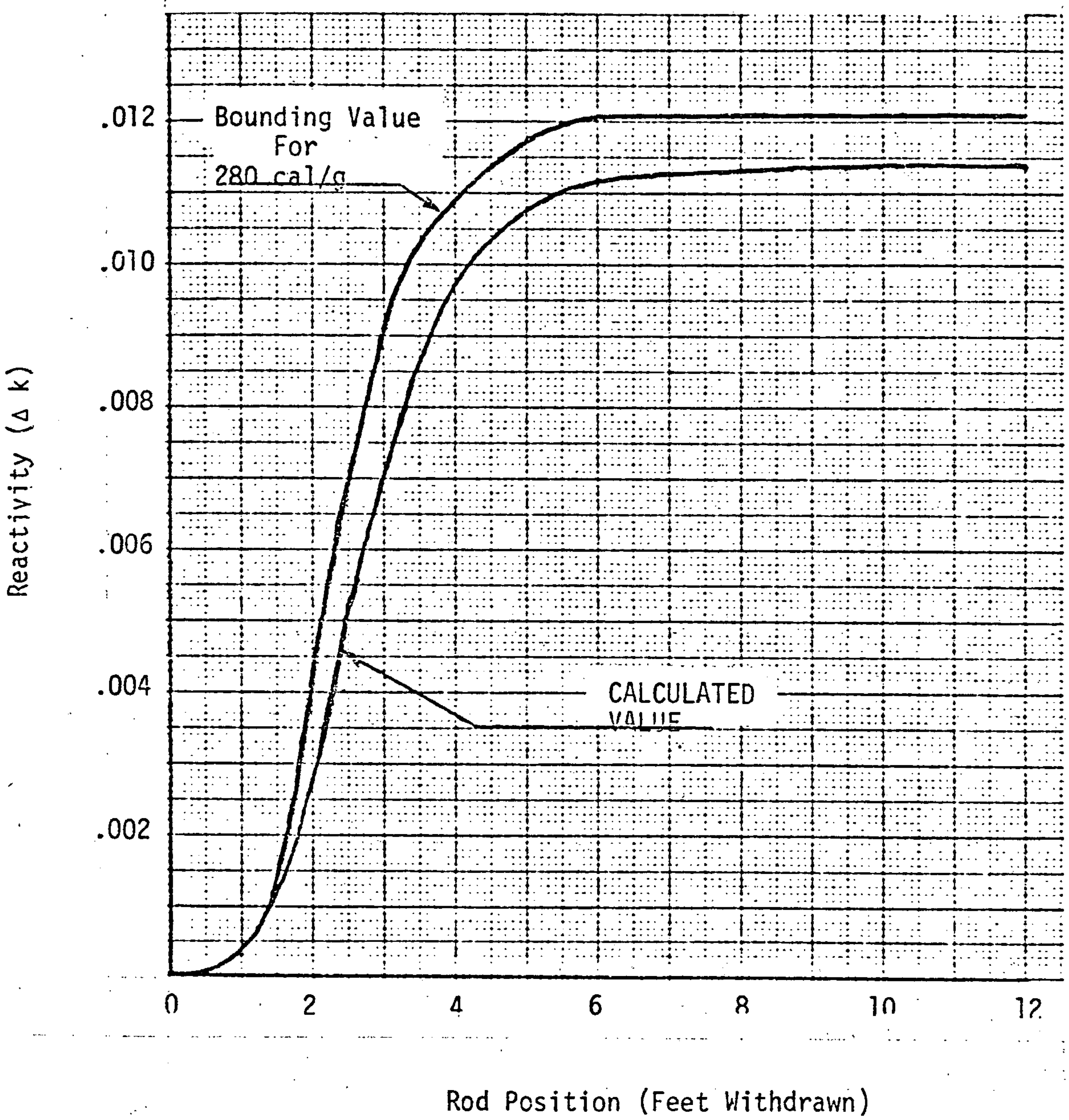


Figure 6-2 Accident Reactivity Function for 20°C

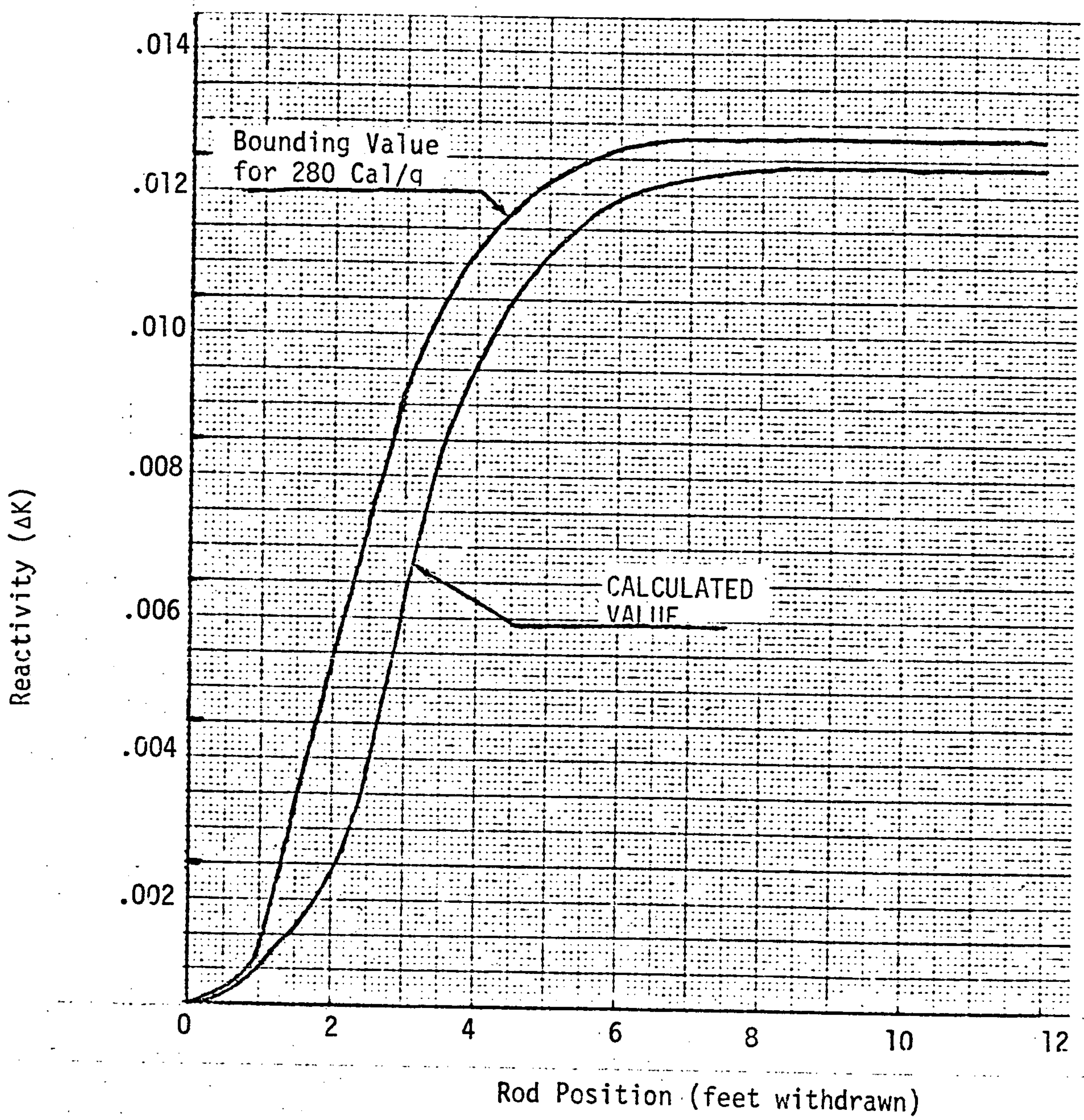


Figure 6-3 Accident Reactivity Function for 286°C

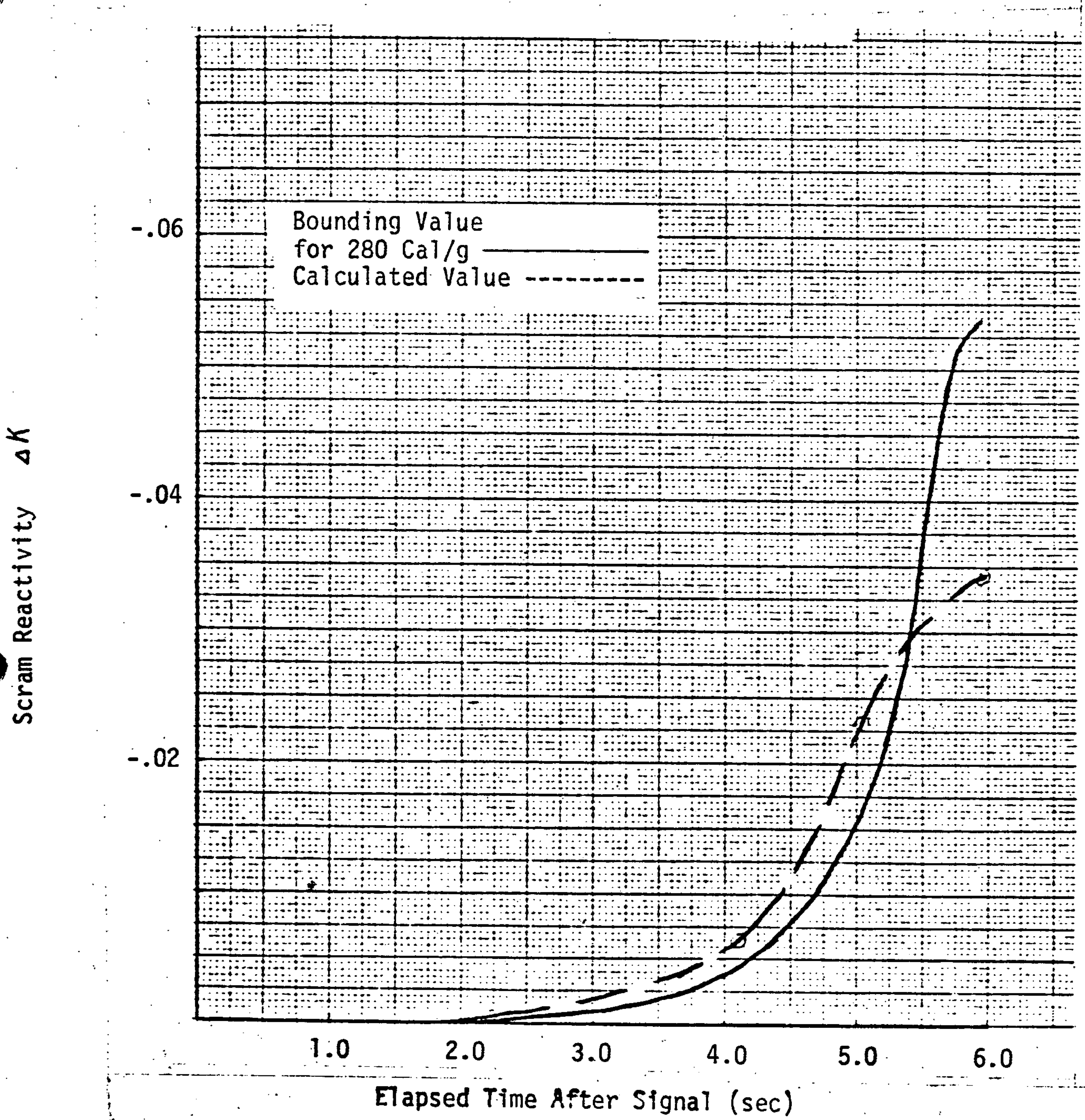


Figure 6-4 Scram Reactivity Function for 20°C

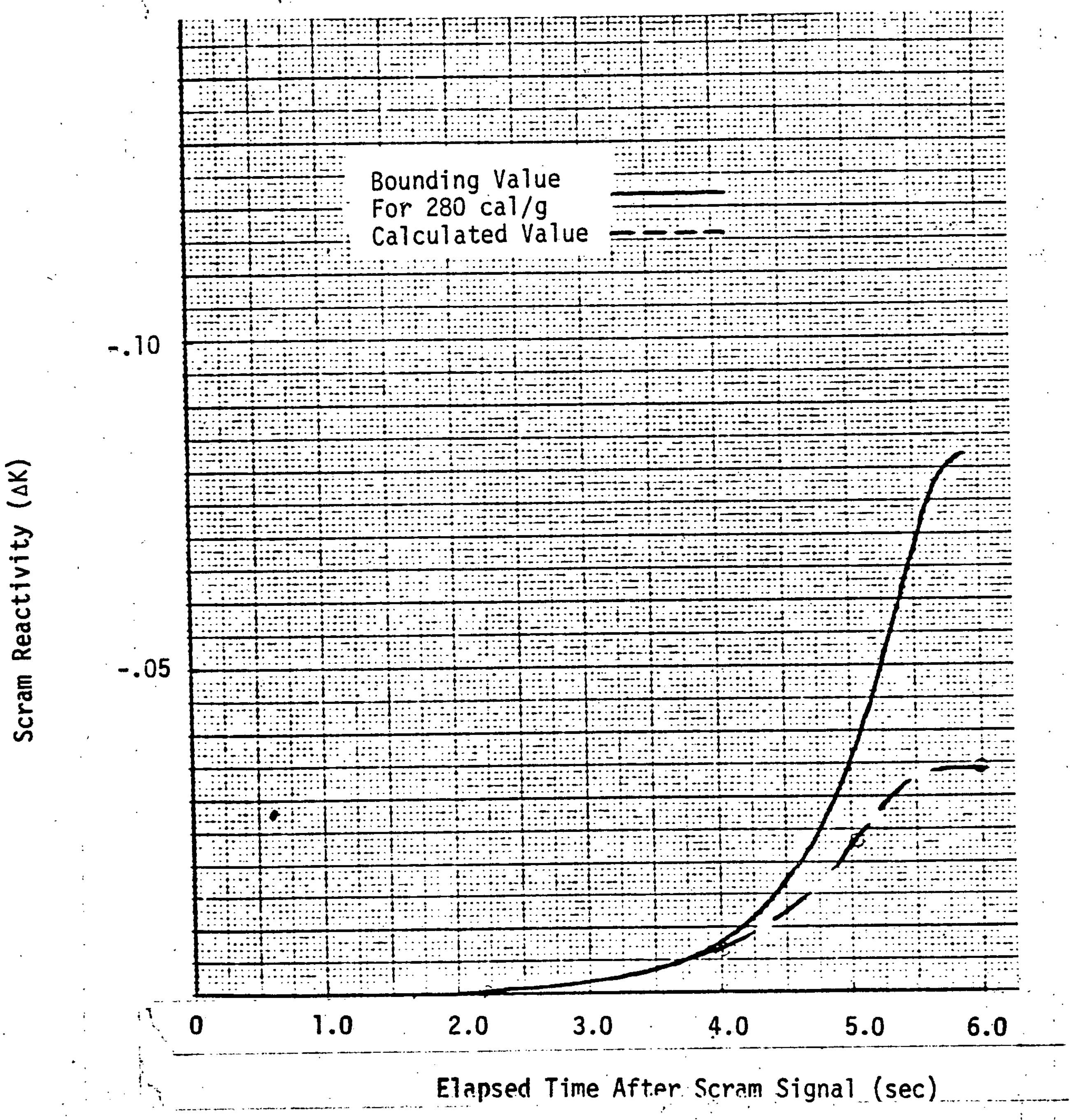


Figure 6-5 Scram Reactivity Function for 286°C

Table 6-1

MAPLHGR, PCT, OXIDATION FRACTION

VERSUS EXPOSURE - PLUGGED

8D274 Fuel

Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200.0	10.5	2198	0.052
1000.0	10.5	2197	0.051
5000.0	10.6	2198	0.049
10000.0	10.7	2197	0.046
15000.0	10.7	2198	0.046
20000.0	10.5	2196	0.047
25000.0	10.4	2198	0.048
30000.0	10.4	2197	0.047

a. 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

b. 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 of 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 1.08 in the misplaced reload bundle. This Linear Heat Generation Rate is below the value at which 1% plastic strain of 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. Therefore, fuel failure is not expected for this event.

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 Duane Arnold Energy Center Cycle 3. 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.

Transient analysis have been performed to obtain an operating limit MCPR from BOC3 to 2 GWD/T before EOC3, 2 GWD/T to 1 GWD/T before EOC3, and 1 GWD/T before EOC3 to EOC3.
The following transients are most limiting and an evaluation of these transients defines the operational bounds from safety considerations: (1) Turbine Trip without bypass, and (2) loss of 100F feedwater heating.

6.3.3.1.2 Input Data and Operating Conditions

The input data and operating conditions are shown in Table 6-2 and represent the nominal basis for these analyses. Each transient is considered at these conditions unless otherwise specified.

6.3.3.1.3 Transient Summary

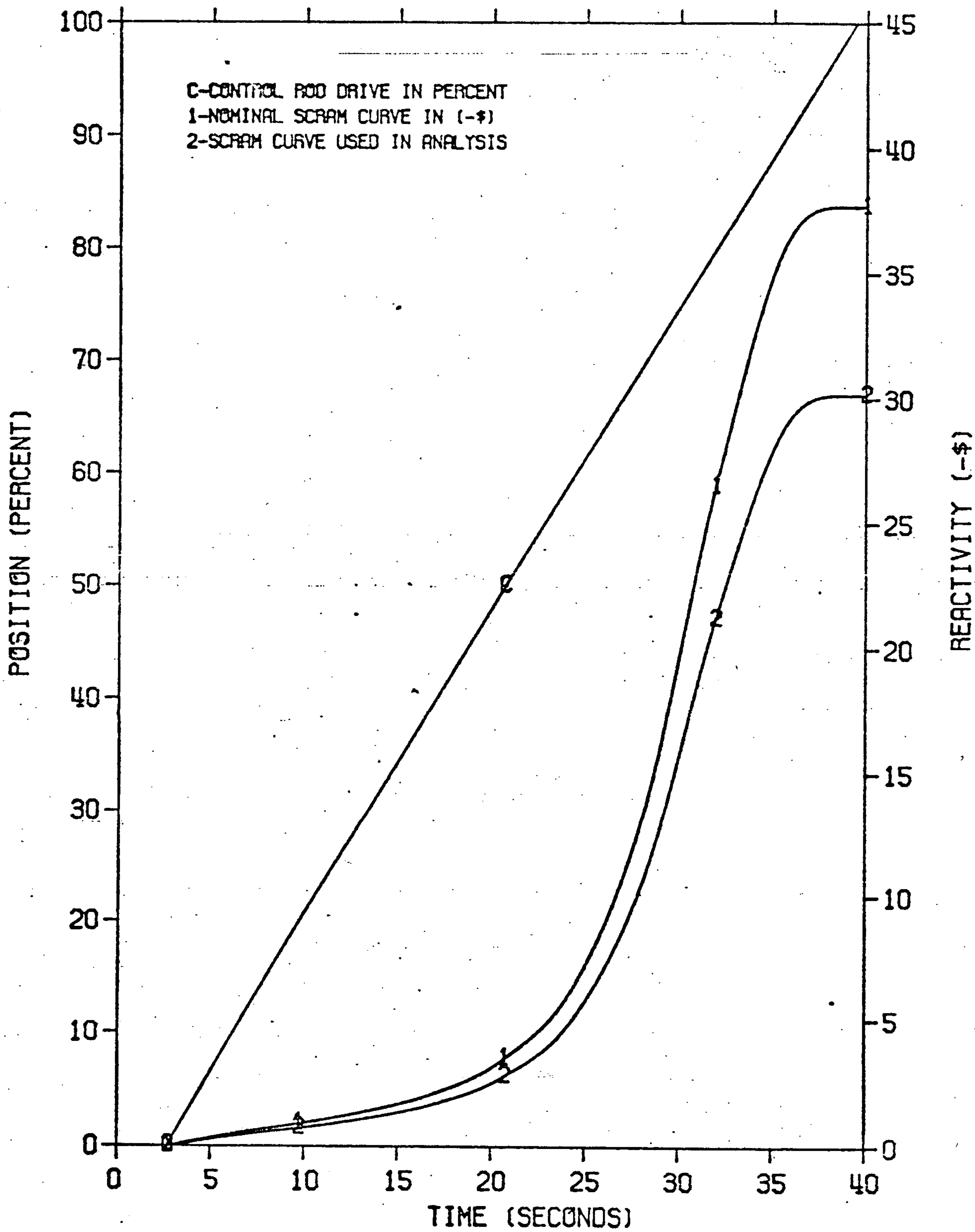
A summary of the transients analyzed and their consequences is provided in Table 6-3.

Table 6-2

TRANSIENT INPUT PARAMETERS

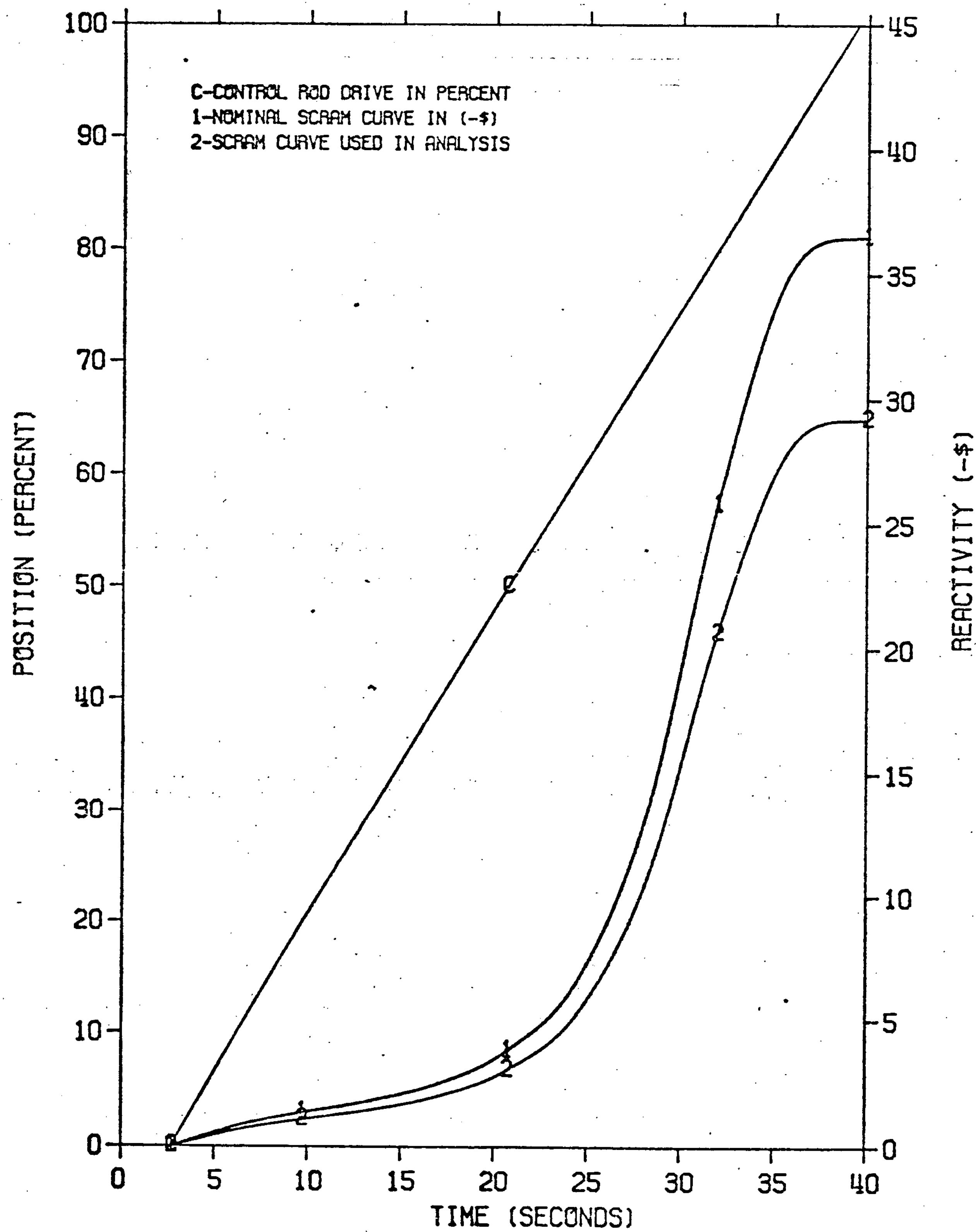
Thermal Power	(MWt)	1657	104%
Rated Steam Flow	(lb/hr)	7.16×10^6	105%
Rated Core Flow	(lb/hr)	49.0×10^6	100%
Dome Pressure	psig	1020	
Turbine Pressure	psig	960	
RV Set Point	psig	1090	
RV/Capacity (at Set Point)	No./%	6/74.7	
RV Time Delay	(msec)	400	
RV Stroke Time	(msec)	100	
SV Set Point	psig	1240	
SV Capacity	No./%	2/18.9	

		EOC3	EOC3-1 GWD/T	EOC3-2 GWD/T
Dynamic Void Coefficient	(-c/%Rg)	13.27	14.47	14.40
Doppler Coefficient	(-c/°F)	0.2152	0.2092	0.2016
Average Fuel Temperature	(°F)	1435	1435	1435
Scram Reactivity Curve		Fig 6.6	Fig 6.6.a	Fig 6.6.b
Scram Worth	(-$)	30.16	29.20	28.48



FIGURE

6-6 SCRAM REACTIVITY CURVE: EOC3



FIGURE

6-6a SCRAM REACTIVITY CURVE: 1 GWD/T BEFORE EOC3

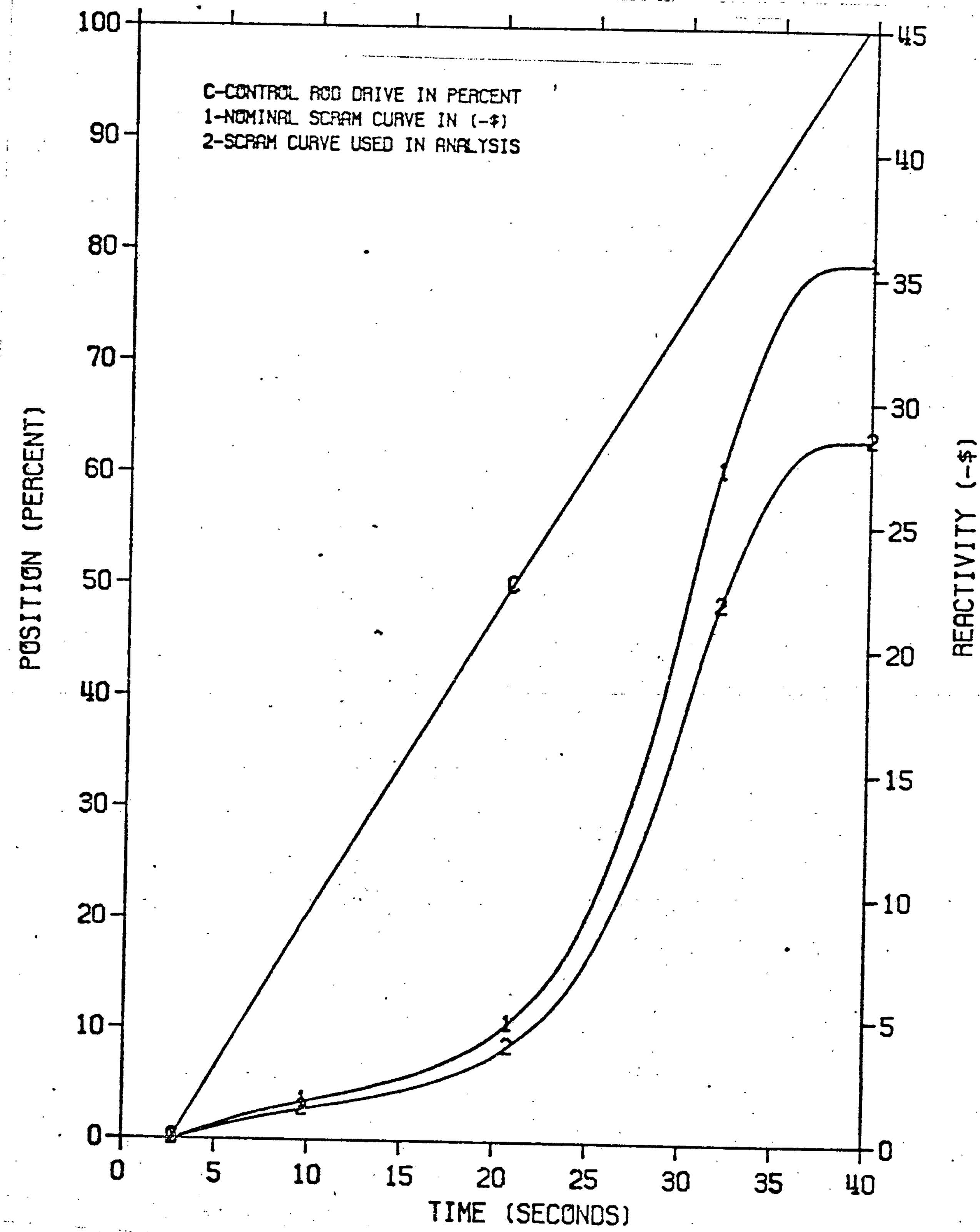


FIGURE 6.6.b SCRAM REACTIVITY CURVE: 2 GWD/T BEFORE EOC3

Table 6-3

TRANSIENT DATA SUMMARY

Transient	Power (%)	Core Flow (%)	ϕ (% ref)	Q/A (% ref)	Psl (psig)	Pv (psig)	ΔCPR 8x8	ΔCPR 7x7
Turbine Trip w/o Bypass								
EOC	104	100	439	119	1200	1243	.37	.29
1 GWD/T Before EOC	104	100	465	118	1199	1241	.36	.28
2 GWD/T Before EOC	104	100	371	114	1188	1230	.29	.21
Loss of Feedwater Heater	104	100	121	118	1024	1073	.18	.15

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 Turbine Trip 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 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. Figure 6-7 illustrates this transient.

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.

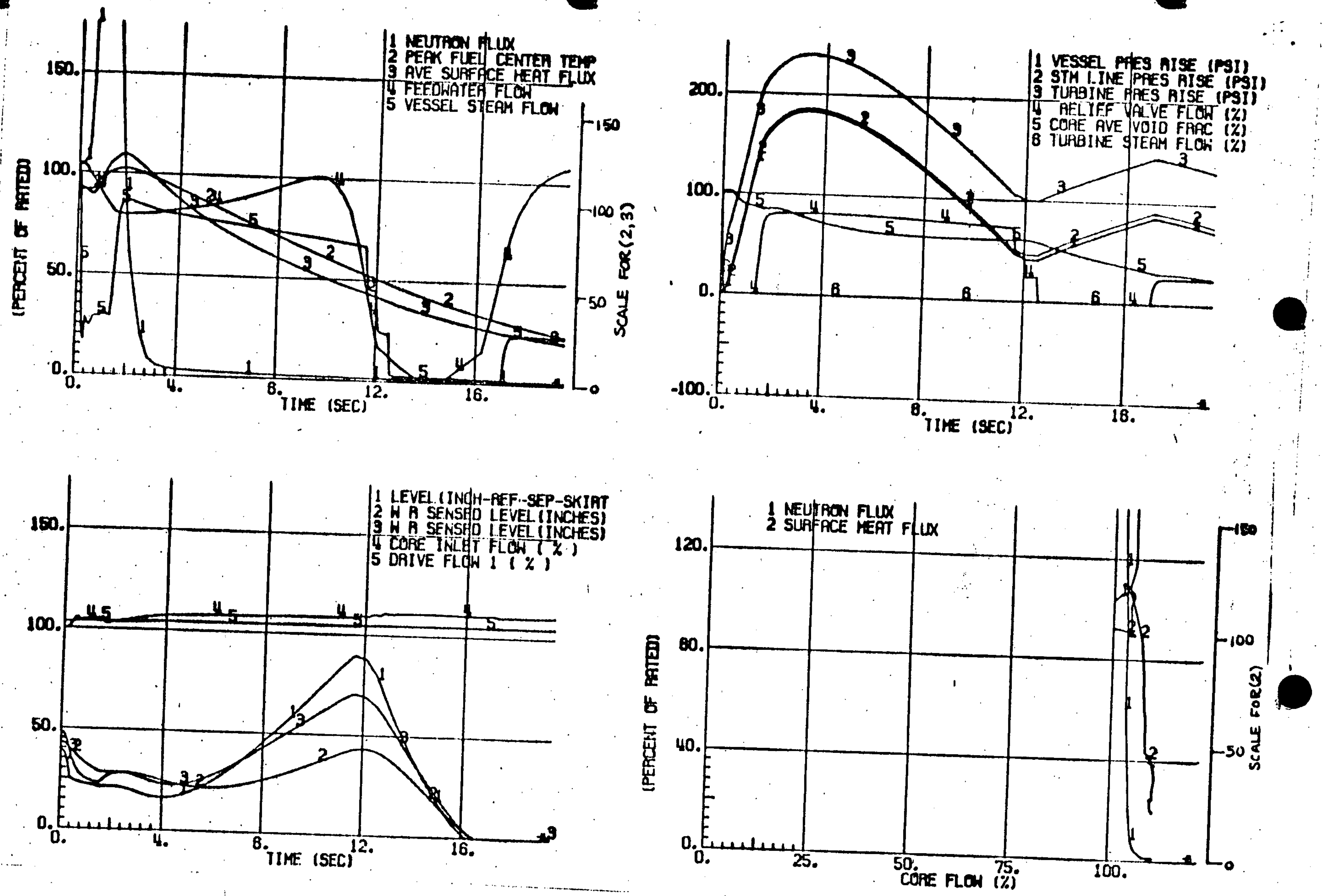


FIGURE 6-7
TURBINE TRIP WITHOUT BYPASS-TRIP SCRAM

END OF CYCLE 3
104% POWER, 100% FLOW

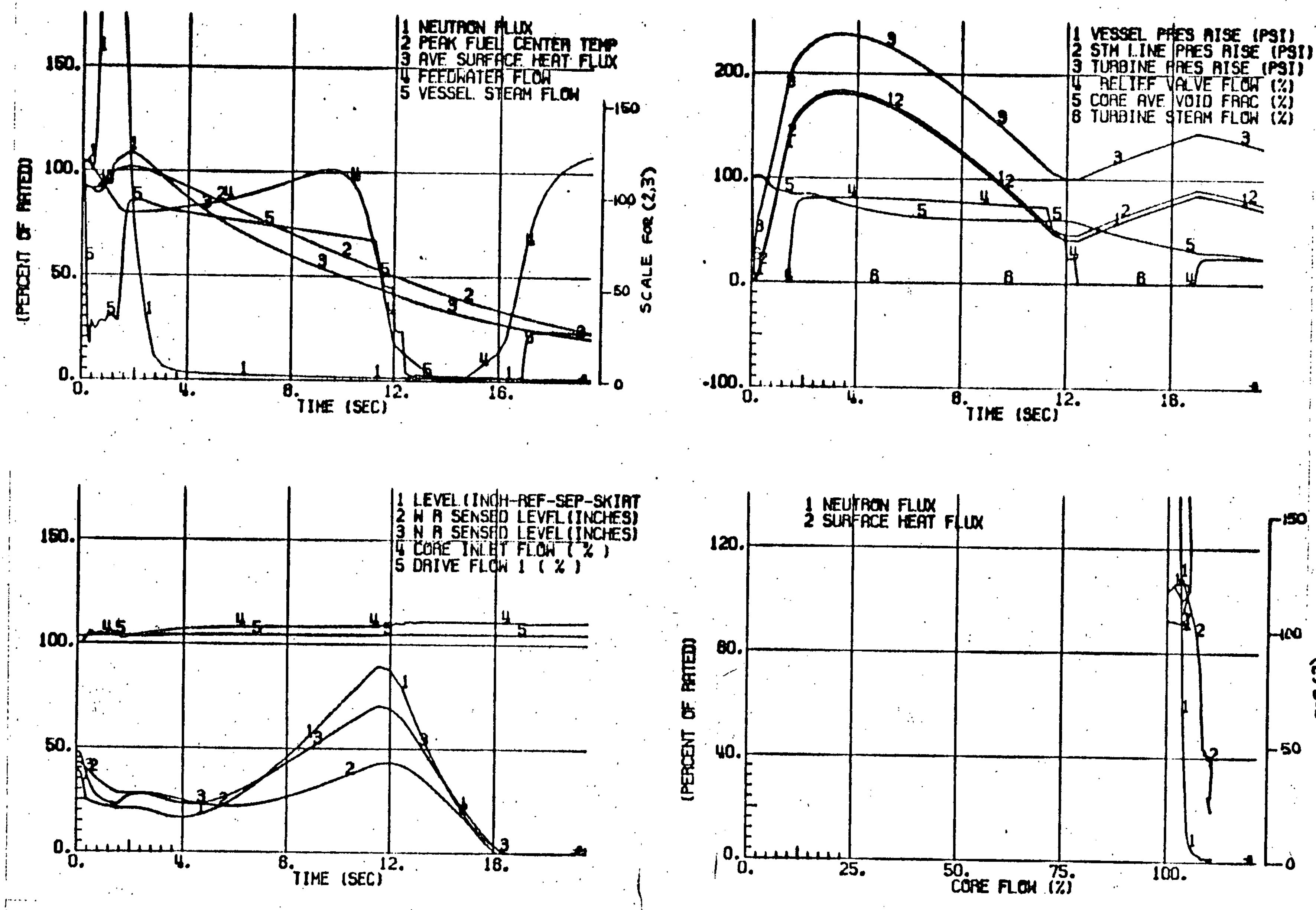


FIGURE 6-7a
TURBINE TRIP WITHOUT BYPASS-TRIP SCRAM

EOC3-1000 MWD/T
104% POWER, 100% FLOW

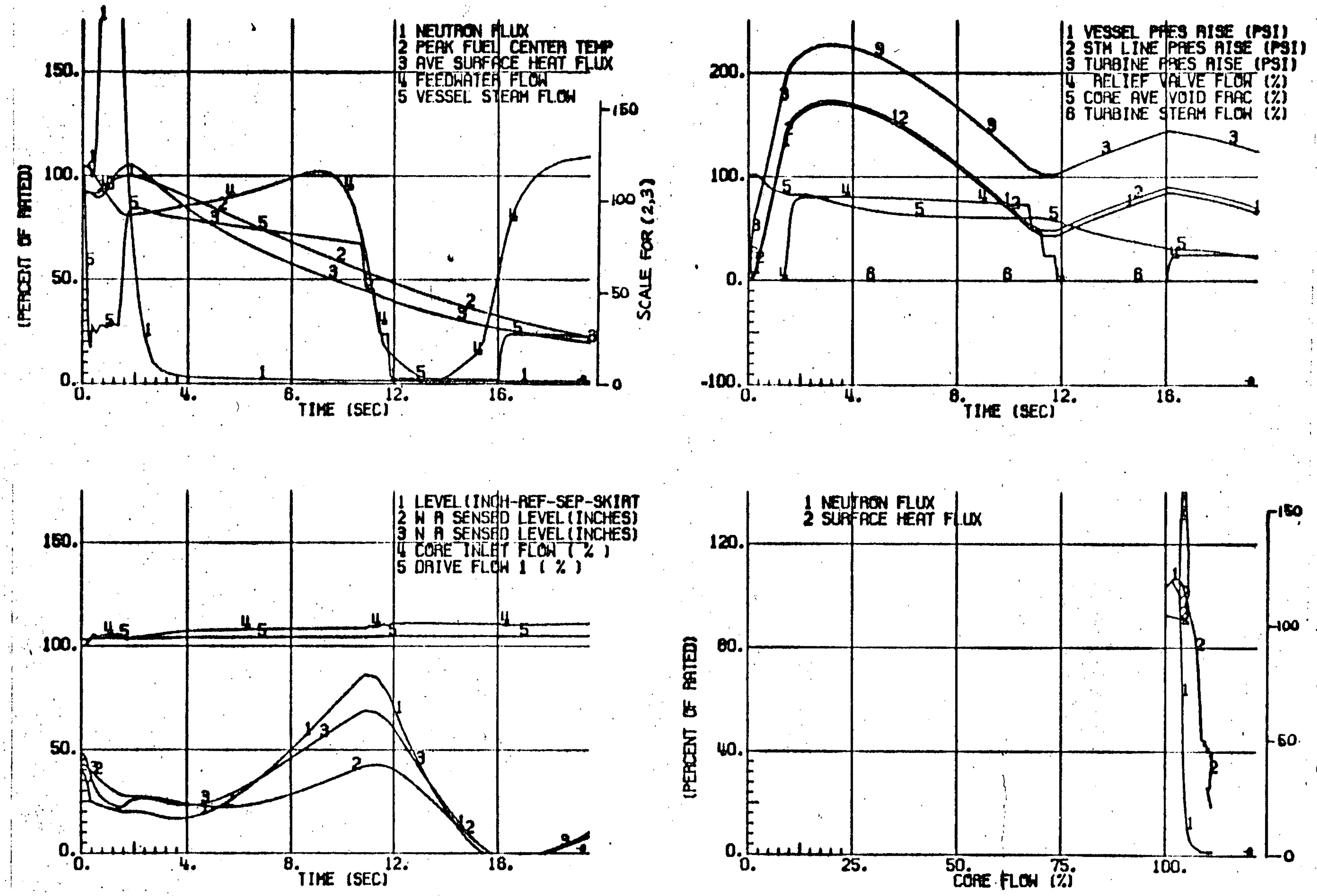


FIGURE 6-7b
TURBINE TRIP WITHOUT BYPASS-TRIP SCRAM

EOC3-2000 MWD/T
104% POWER, 100% FLOW

Neutron flux, the precursor of heat flux, and the resulting ΔMCPR which determines the design basis operating critical power ratio is given in Table 6-3.

The peak streamline pressure is limited to 1200 psig as a result of the high-pressure actuation of the six safety/relief valves which provides a 40-psi margin to the 1240-psig set point of the first spring safety valve.

6.3.3.2.2 Loss of a Feedwater Heater

The loss of a feedwater heater is analyzed in FSARs and other submittals because it constitutes the most limiting cool water injection 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. The delay in the flow from the tripped feedwater heater to the feedwater sparger is ignored, thereby adding conservatism to the analysis.

Figure 6-8 shows the response of the plant to the loss of 100°F of the feedwater heating capability of the plant. This represents the maximum expected single heater (or group of heaters) which can be tripped or bypassed by a single event. The reactor is assumed to be at maximum power conditions on manual flow control when the heater was lost. Note that in manual flow control mode the core flow remains constant throughout the transient. Neutron flux, however, increases above the initial value in order to produce the same steam flow with the higher inlet subcooling. The peak neutron flux and average surface heat flux are shown in Table 6-3. Fuel thermal margins are not exceeded; transient ΔCPR is shown in Table 6-3. Transient consequences are milder for lower initial power levels.

6.3.3.2.3 Rod Withdrawal Error

Assumptions and descriptions of rod withdrawal error are given in Reference 1. Figures 6-9 through 6-11 show the results of the worst case condition for Duane Arnold Energy Center Reload-2. The rod block monitor (RBM) set point of 105% is selected to allow

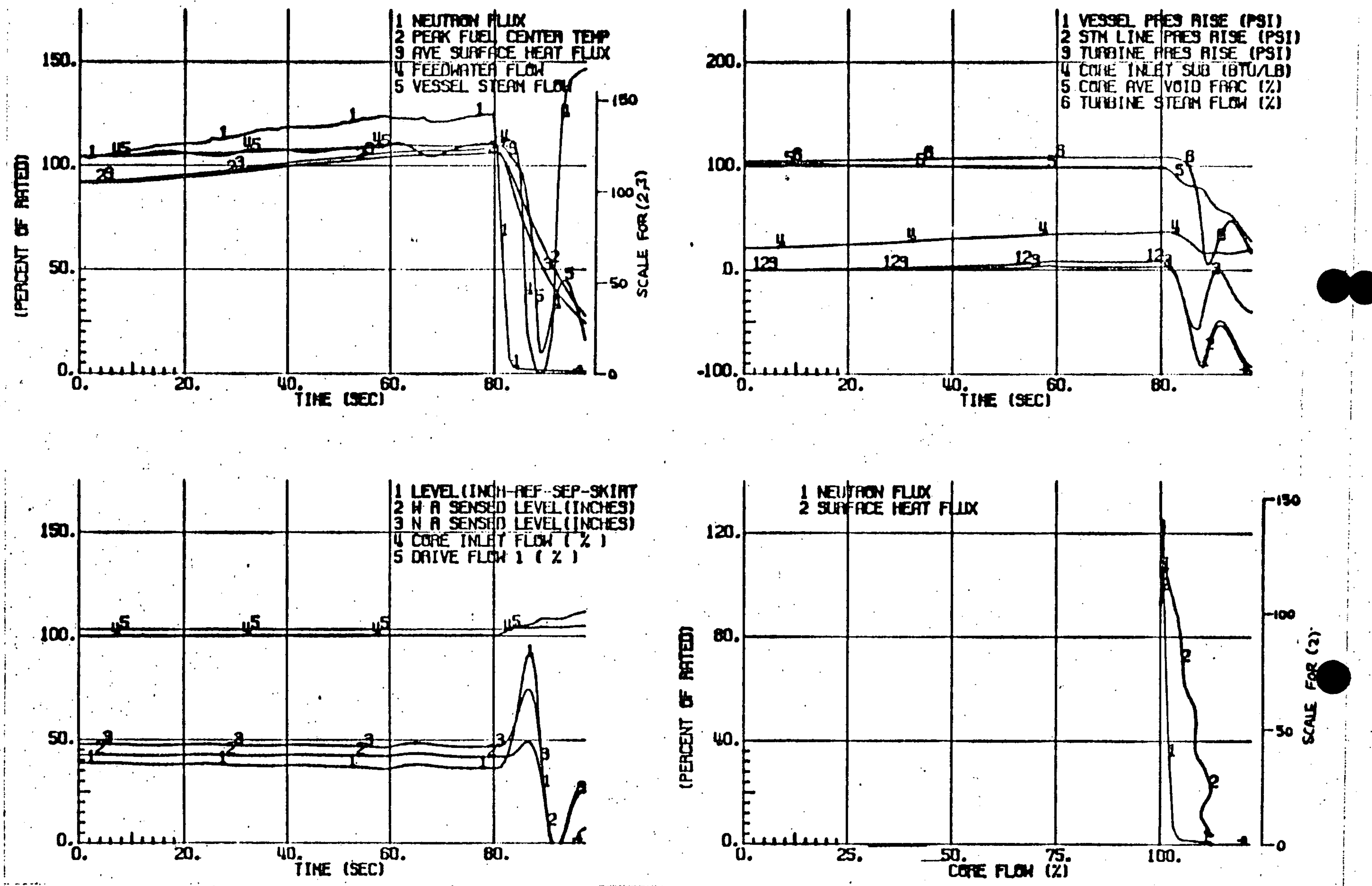


FIGURE 6-8
LOSS OF 100F FEEDWATER HEATER, MFC, FSCRAM

END OF CYCLE 3
104% POWER, 100% FLOW

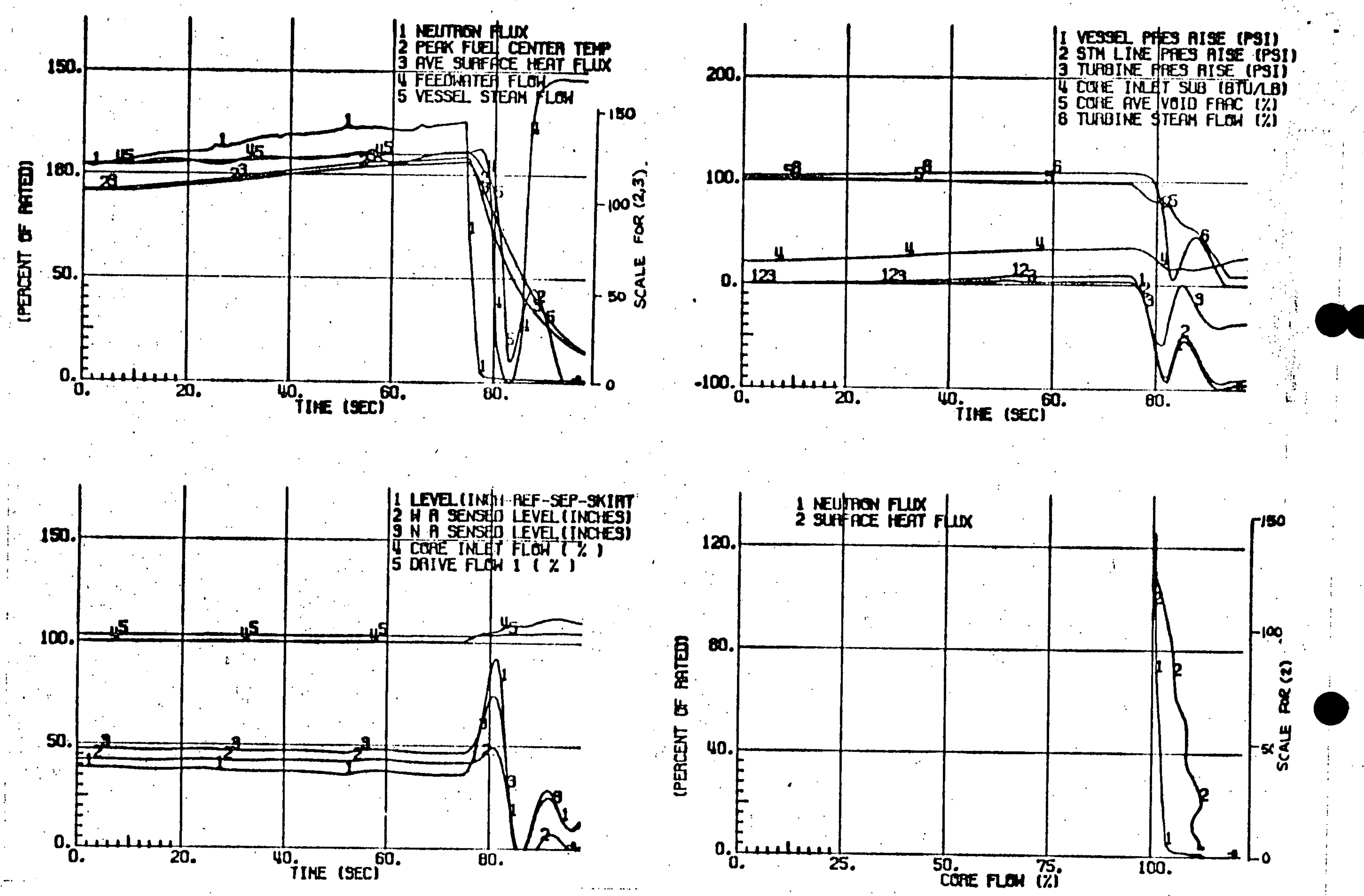


FIGURE 6-8a
LOSS OF 100F FEEDWATER HEATER, MFC, FSCRAM

EOC3-1000 MWD/T
104% POWER 100% FLOW

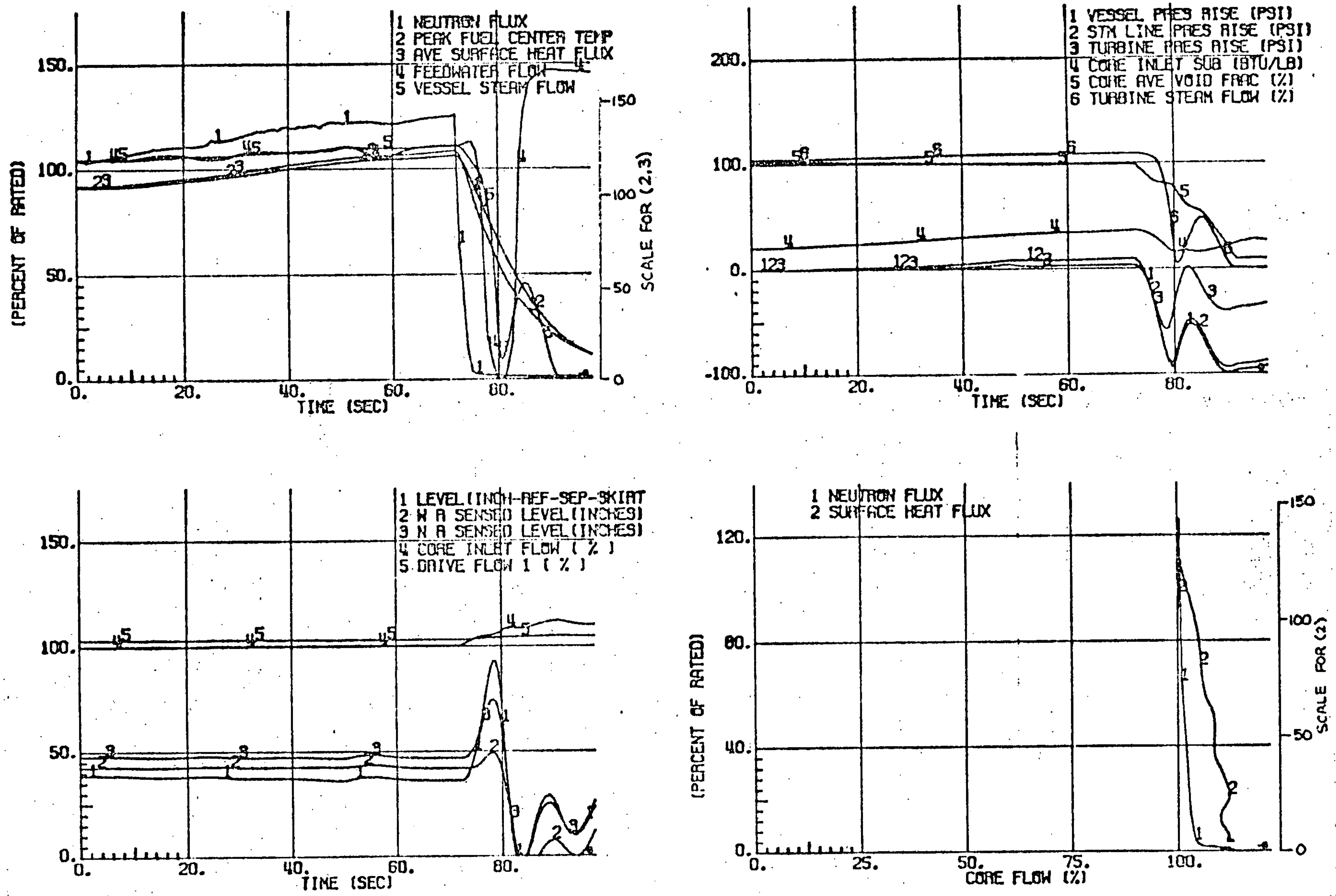


FIGURE 6-8b
LOSS OF 100F FEEDWATER HEATER, MFC, FSCRAM

EOC3-2000 MWD/T
104% POWER 100% FLOW

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 when the critical power ratio (CPR) is 1.22, still greater than the 1.06 MCPR safety limit.

6.3.4 ASME Vessel Pressure Code Compliance

All Main Steamline 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 Duane Arnold Energy Center pressure relief system includes six dual function safety/relief valves and two 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 valve set point must be at or below vessel design pressure.

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

Duane Arnold Energy Center's safety/relief and spring safety valves are set to self-actuate at the pressures shown in Table 6-2, thereby satisfying b. and c., above. Requirement a. is evaluated by considering the most severe isolation event with indirect scram.

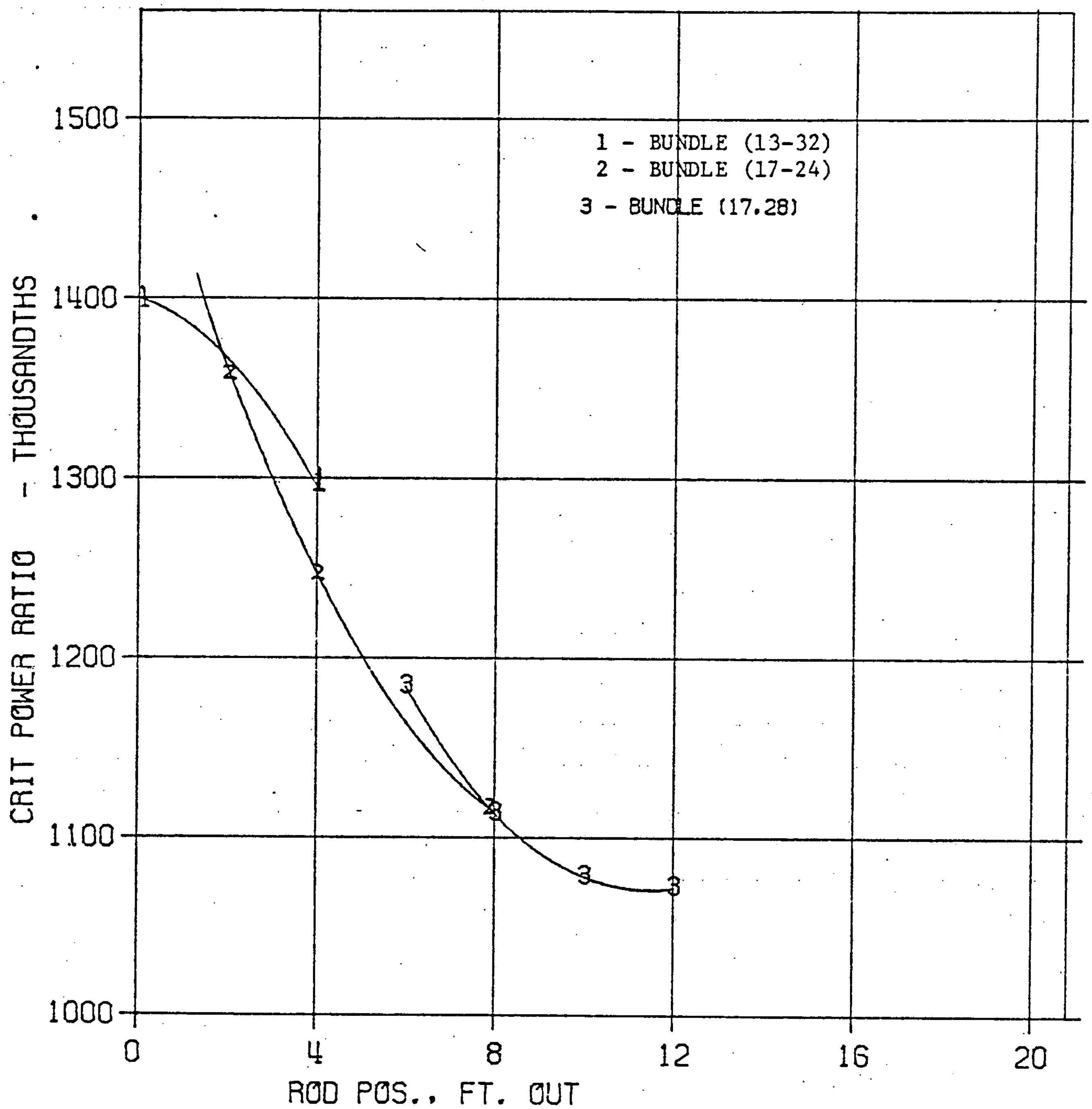


FIGURE 6-9a MCPR VS. ROD POSITION - 8x8

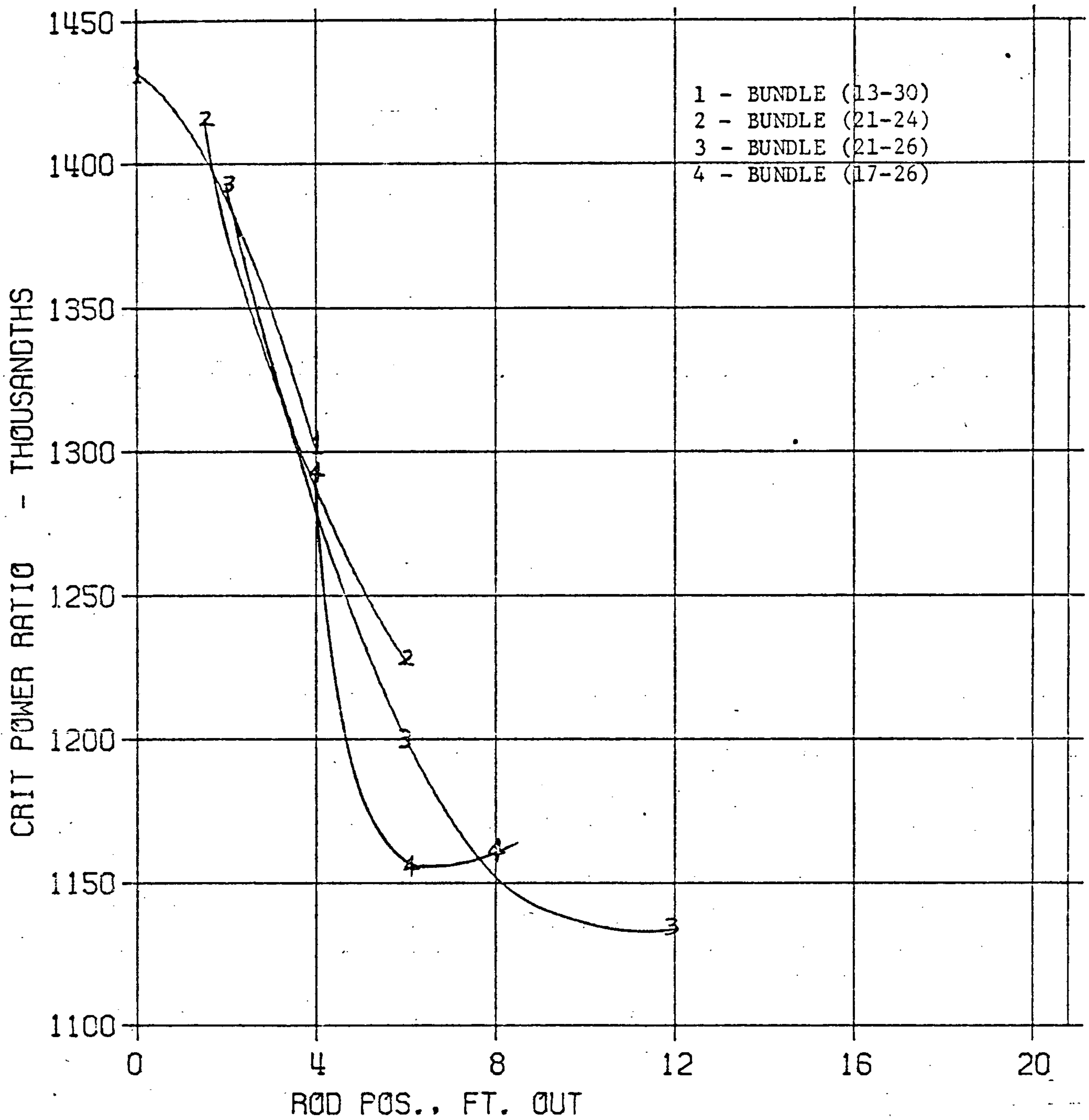


FIGURE 6-9b MCPR VS. ROD POSITION - 7x7

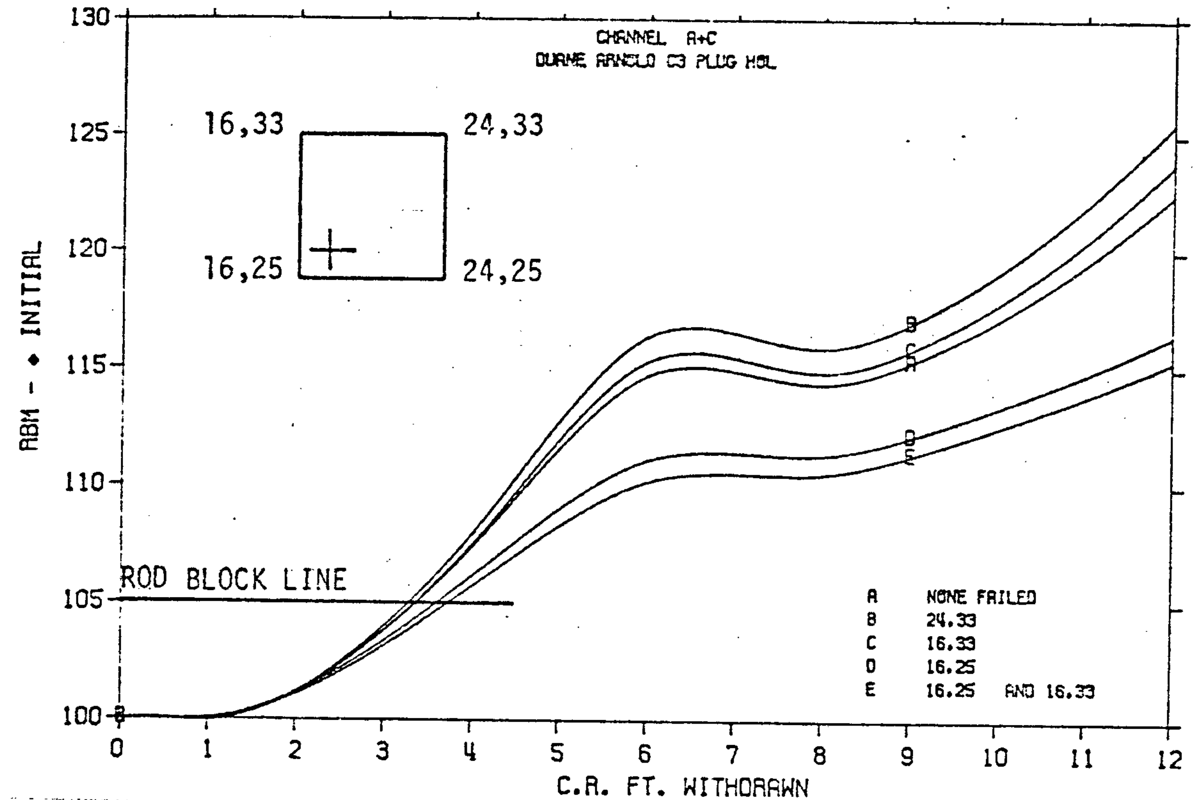


Figure 6-10. RBM Response to Control Rod Motion for Rod Withdrawal Error

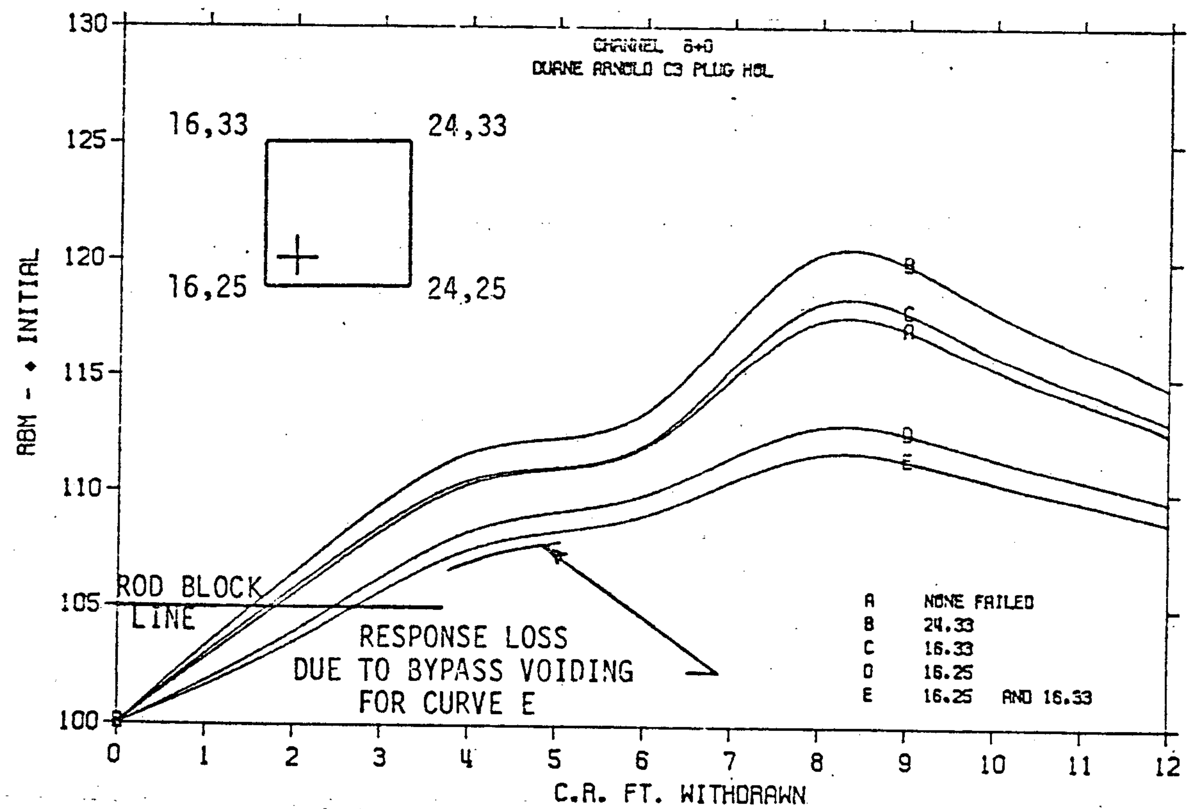


Figure 6-11. RBM Response to Control Rod Motion for Rod Withdrawal Error--Limiting Case, Channel B+D

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 6-2. Figures 6-12, 6-12a, and 6-12b graphically illustrate the event for exposures at EOC3, 1 GWD/T before EOC3, and 2 GWD/T before EOC3. The response indicates a $\geq$ 84 psi margin to the vessel code limit of 1375 for EOC3, $\geq$ 90 psi for 1 GWD/T before EOC3, and $\geq$ 100 psi for 2 GWD/T before EOC. Thus, requirement a. is satisfied and adequate overpressure protection is provided by the pressure relief system.

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 Duane Arnold Energy Center Reload 2 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	105% Rod Block - Natural Circulation Power
Decay Ratio, X_2/X_0	
8x8 Channel	0.30
7x7 Channel	0.16

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 at all attainable operating conditions.

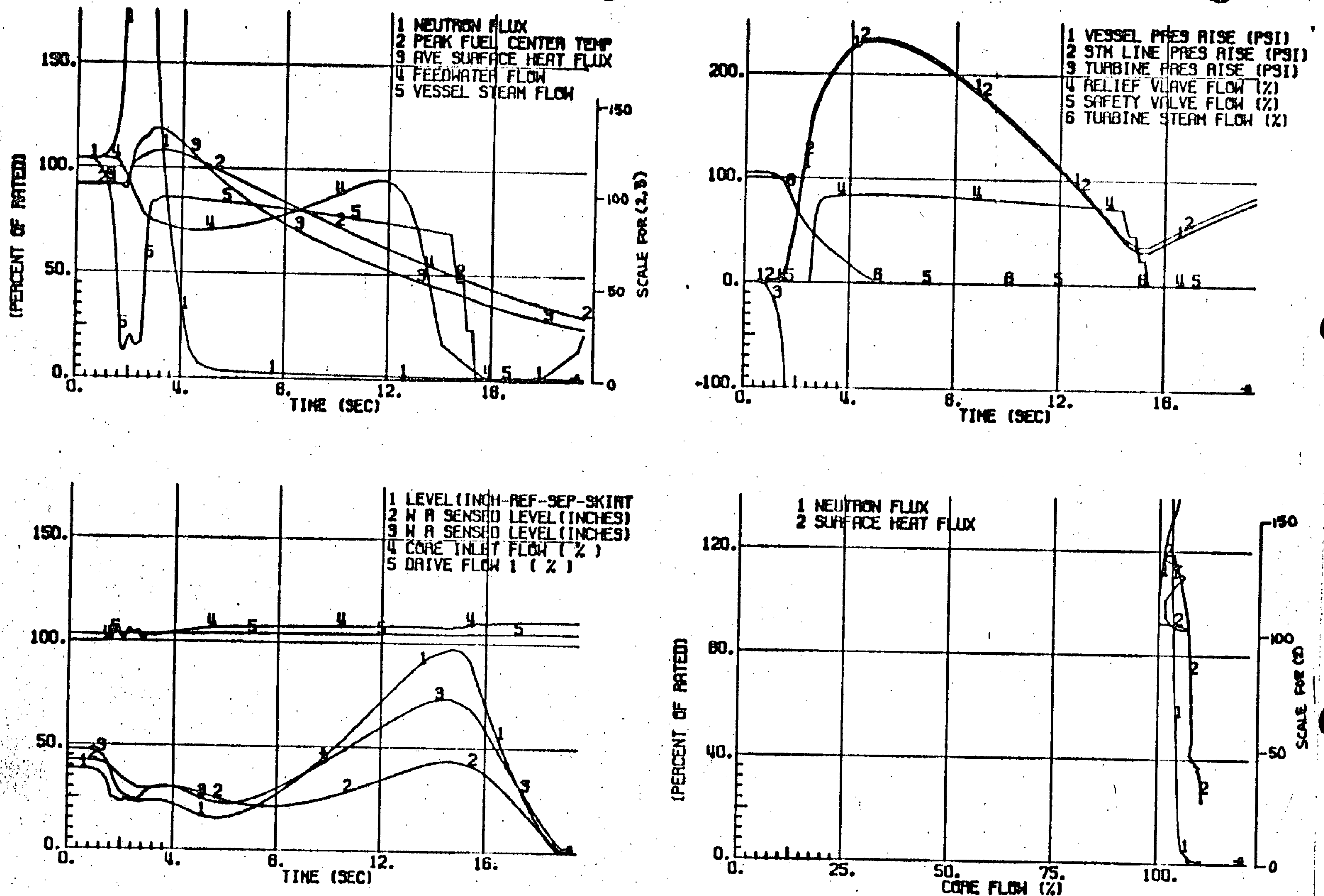


FIGURE 6-12 EOC3
MSIV CLOSURE FLUX SCRAM 100% POWER 100% FLOW

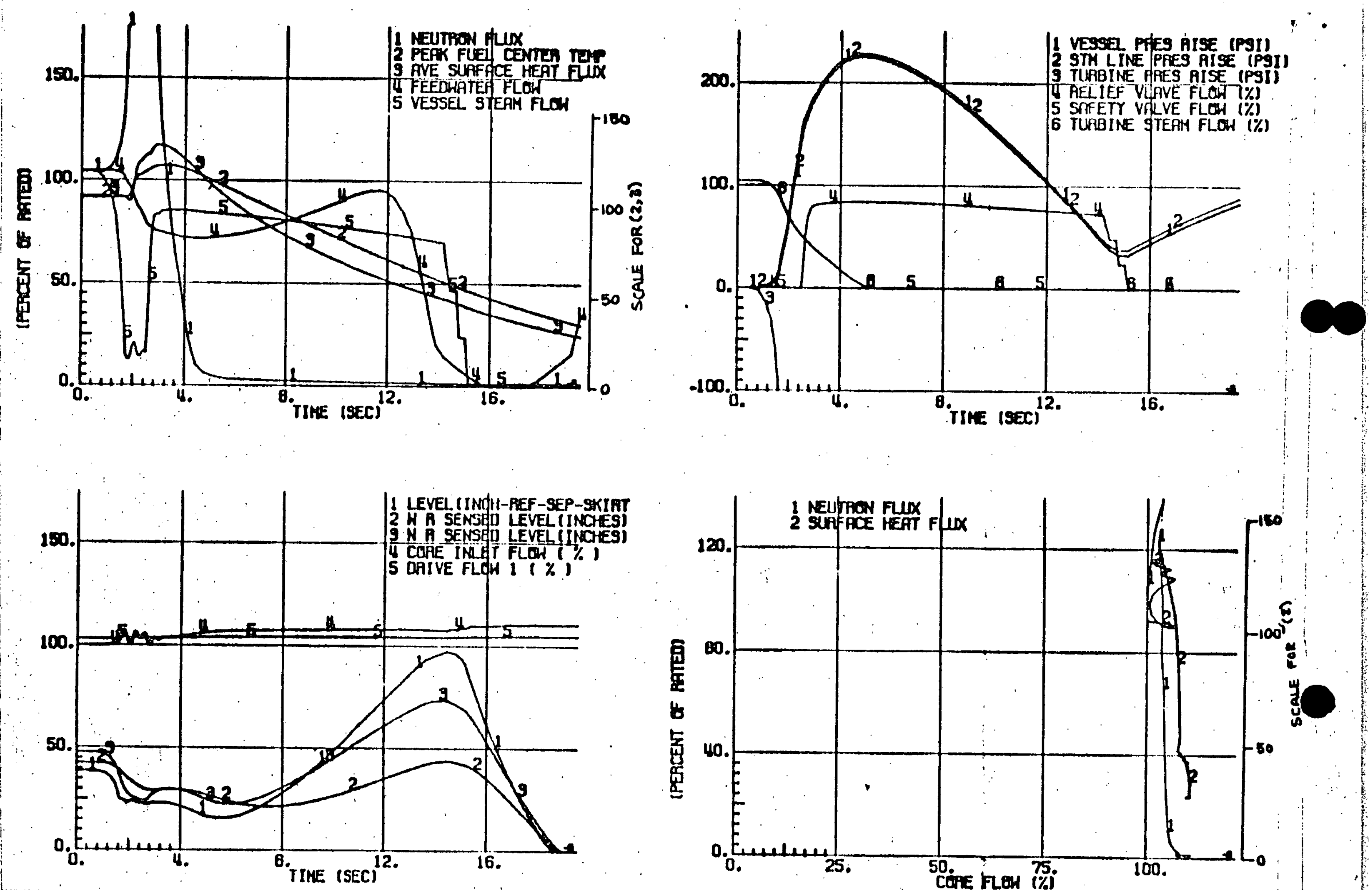


FIGURE 6-12a
MSIV CLOSURE FLUX SCRAM

EOC3-1000MWD/T
100% POWER 100% FLOW

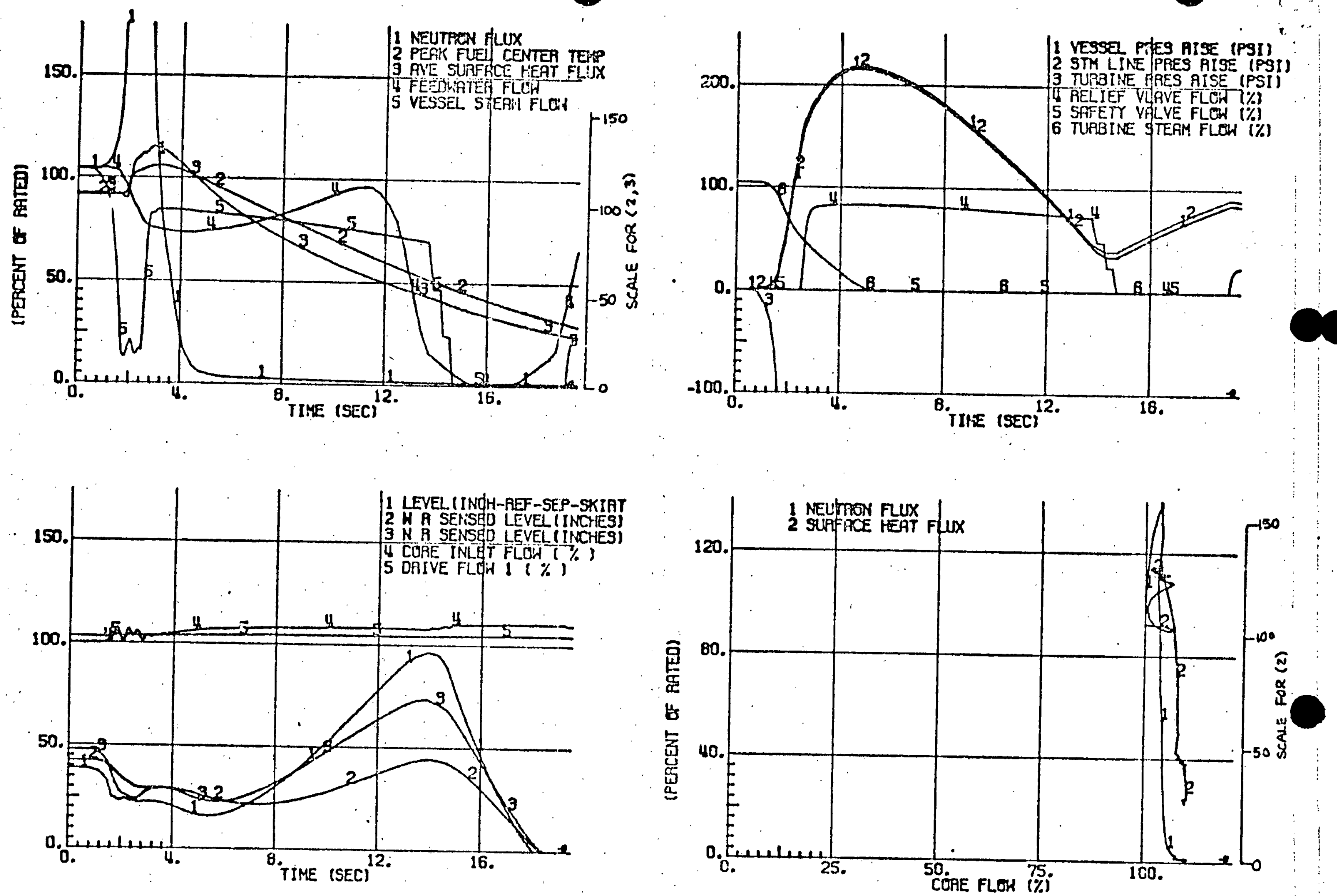


FIGURE 6-12b
MSIV CLOSURE FLUX SCRAM

EOC3-2000 MWD/T
100% POWER 100% FLOW

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 6-13. The most responsive case is again the 105% rod block-natural circulation condition.

Reactor Core Stability	105% Rod Block - Natural Circulation Power
Decay Ratio, X_2/X_0	0.78

These calculations show the reactor to be in compliance with the ultimate performance criteria, including the most responsive condition at 105% rod block - 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_2/X_0		
8x8 Channel	<0.01	0.06
7x7 Channel	<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 values of the decay ratio of the reactor power dynamic response for rated operating conditions and for the low end of the normal flow control range at the corresponding nominal power (66% power, 51% flow) are presented below.

Reactor Core Performance	Rated Conditions	Low End of Flow Control Range
Decay Ratio	<0.01	0.25

As noted earlier, Figure 6-13 describes the variation of decay ratio over the entire power flow range.

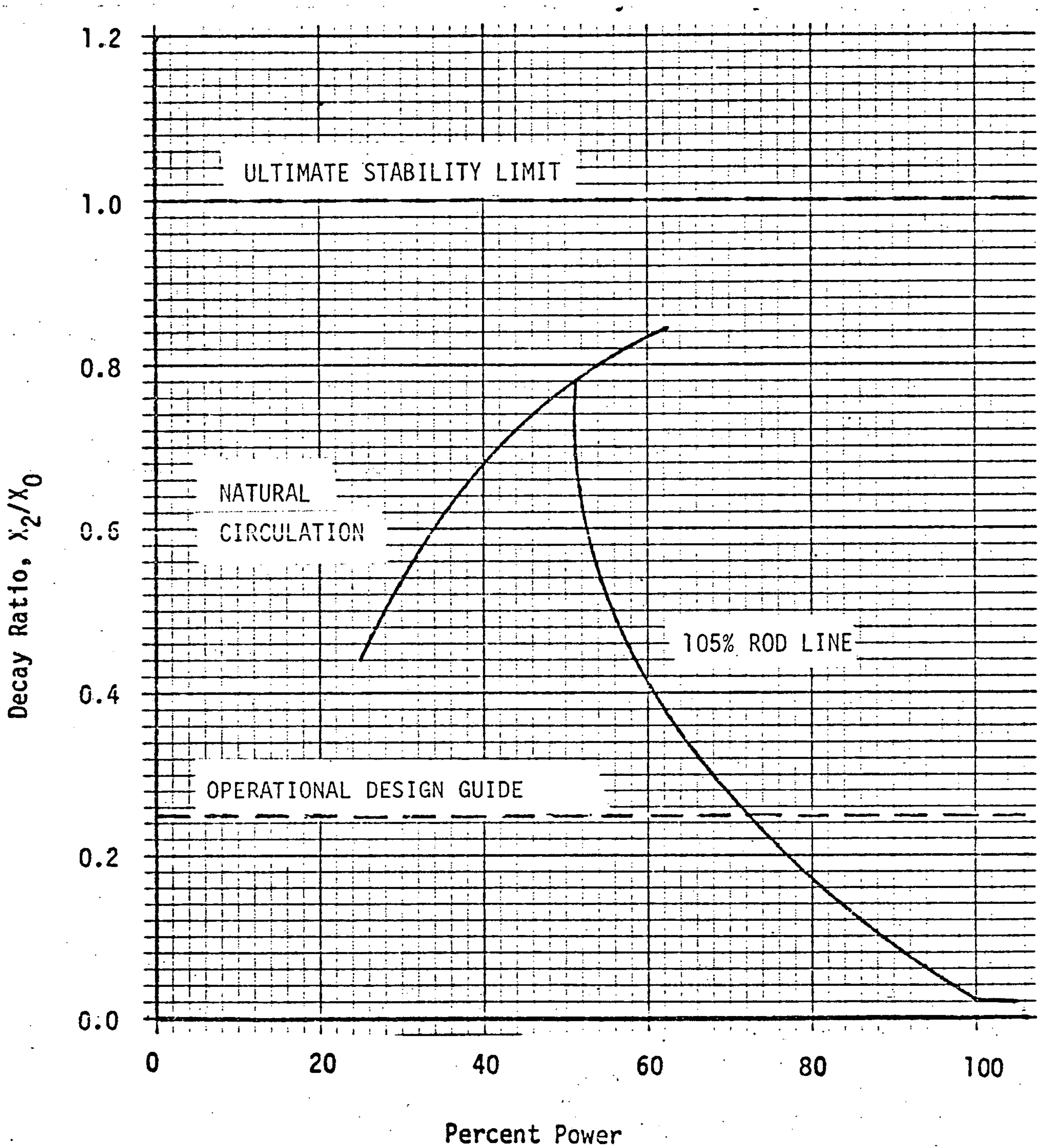


Figure 6-13. DUANE ARNOLD NUCLEAR POWER STATION RELOAD 2 DECAY RATIO

7. TECHNICAL SPECIFICATIONS

Technical Specifications and Bases should be submitted separately prior to reactor startup.

These changes should be based on this submittal as well as any other items requiring change. Several possible changes are listed below:

a. MCPR values

b. Transient results

c. Uncertainties in Fuel Cladding Safety Limit

d. Rod Block Monitor Setpoint

e. Shutdown margin for Standby Liquid Control System

REFERENCES

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

2. Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration, January 1976 (NEDE 21156-Class III).

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