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TO:
Mr. Benard C. Rusche

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

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Duane Arnold

ENCLOSURE

Amdt. to ol/change to Appendix A tech specs for Duane Arnold Energy Center.

(18-P)

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11558

appl

IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office
CEDAR RAPIDS, IOWA

LEE LIU
VICE PRESIDENT – ENGINEERING

November 9, 1976
IE-76-1732

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 for DPR-49 (Appendix A to License) for the Duane Arnold Energy Center.

Operating limit MCPR's presently contained in the DAEC Technical Specifications for End-of-Cycle conditions limit power to about 80%. The enclosed amendment would allow an increased capability of about 5% power and increased maneuvering capability for the remainder of this cycle.

The enclosed analysis utilizes an increased bypass flow for a plugged core resulting in improved End-of-Cycle MCPR's. This change does not result in any change to the presently licensed safety limit MCPR.

This application has been reviewed and approved by the DAEC Operations Committee and the DAEC Safety Committee. This application does not involve a significant hazards consideration.

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

Iowa Electric Light and Power Company

By: ______________________
Lee Liu
Vice President, Engineering

LL/KAM/ms
Enc.
cc: K. Meyer
D. Arnold
R. Lowenstein
J. Shea (NRC)
J. Keppler (NRC)
L. Root
File A-117

Subscribed and Sworn to before me on this 9th day of November, 1976.

____
Notary Public in and for the State of Iowa.

Jean R. Smith
NOTARY PUBLIC
STATE OF IOWA
Commission Expires
September 30, 1978

11558

PROPOSED CHANGE RTS-76 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

Table 3.12-2 shows MCPR limits for 7 x 7 fuel to be 1.40 from "$\leq$ 1000 MWD/T to E.O.C." and 1.50 for 8 x 8 fuel.

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 "1.40" to "1.35" and "1.50" to "1.44".

Add reference "9" as shown on attached sheet 3.12-11.

III. Justification for Proposed Change

The above MCPR limits are the result of calculations with higher bypass flow rates than used in the calculations for the previous MCPR limits. Explanation of the new calculations is contained in General Electric Document Number NEDO-21082-01 dated October 1976.

It is Iowa Electric Light and Power Company's understanding that this calculation using higher bypass flow has previously been licensed on Brunswick-2, Brown's Ferry-1, 2 and 3, Fitzpatrick, Hatch and Vermont Yankee.

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.

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	1.26	1.26	1.35
8 x 8	1.30	1.34	1.44

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 One Licensing Submittal, January 1976.

9. Duane Arnold Energy Center Reload Number One Licensing Submittal, NEDO-21082-01, October 1976.

NEDO-21082-01
Class 1
Supplement 2
October 1976

GENERAL ELECTRIC BOILING WATER REACTOR

RELOAD NO. 1 LICENSING SUBMITTAL

FOR

DUANE ARNOLD ENERGY CENTER

License No. DPR-49

Docket No. 50-331

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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I. INTRODUCTION

Supplement 1 to NEDO 21082-01 of June 1976 provided analysis to support exposure dependent minimum critical power ratio operating limits. Supplement 1 was submitted and approved, and the plant is now operating in accordance with this submittal.

Supplement 2 provides transient analysis for cycle 2 incremental exposure beyond 2600 MWD/T, using an increased bypass flow rate for a plugged core. Full scale tests[4] have demonstrated that the total bypass flow for the plugged configuration is greater than was used in the previous analysis. The use of the new bypass flow data results in less bypass voiding, which in turn results in a change to the EOC scram curve and a less negative void coefficient.

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. Other analysis and limits identified in NEDO 21082-01 remain valid.

II. THERMAL-HYDRAULIC ANALYSIS

4.2 ANALYSIS OF ABNORMAL OPERATIONAL TRANSIENTS

The results of the most limiting pressure and power increase transients were evaluated to determine the largest decrease in MCPR. Other types of transients have an insignificant effect upon critical power and are, therefore, not reviewed in depth. The results of the transients analyzed are summarized in Table 4-3.

Addition of the MCPR 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 1.35 for 7x7 fuel and 1.44 for the 8x8 fuel for cycle 2 incremental exposure beyond 2600 MWD/T. 2

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.

Table 4-3

SUMMARY OF TRANSIENTS ANALYZED

Event	Maximum ΔCPR 7x7	Maximum ΔCPR 8x8
Rod Withdrawal Error (RBM set to 107%)	0.19	0.15
Loss of Feedwater Heater at rated conditions	0.15	0.18
Turbine Trip w/o Bypass EOC2 Scram at rated conditions	0.28	0.37

2

Table 4-4

GETAB TRANSIENT ANALYSIS
INITIAL CONDITION PARAMETERS

C) Cycle 2 Incremental Exposure Beyond 2600 MWD/T, Increased Bypass flow.

	7x7	8x8	
Peaking factors (local, radial and axial)	(1.24, 1.212, 1.40)	(1.22, 1.206, 1.40)	2
R-Factor	1.084	1.102	
Bundle Power, MWt	5.157	5.127	
Non-fuel Power Fraction	0.04	0.04	
Core Flow, Mlb/hr	49.0	49.0	
Bundle Flow, 10^3 lb/hr	129.0	122.1	2
Reactor Pressure, psia	1035	1035	
Inlet Enthalpy, Btu/lb	526.3	526.3	
Initial MCPR	1.34	1.45	2

2

6.3.3 <u>Abnormal Operating Transients</u>

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 2. 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.

This report contains the transient analysis for the most limiting transients for cycle 2 incremental exposure beyond 2600 MWD/T, using increased bypass flow. The 2
following transients are the most limiting and an evaluation of these transients defines the operational bounds from safety considerations: (1) Turbine trip without bypass, and (2) loss of 100°F feedwater heating.

2

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)	7.18×10^6	105%
Rated Core Flow	(lb/hr)	49.0×10^6	100%
Dome Pressure	psig	1020	
Turbine Pressure	psig	960	
RV Set Point	psig	1101	
RV/Capacity (at Set Point)	No./%	6/74.7	
RV Time Delay	(msec)	400	
RV Stroke Time	(msec)	100	
SV Set Point	psig	1252	
SV Capacity	No./%	2/18.9	
Dynamic Void Coefficient	(¢/%Rg)	-11.49/-14.36 (nominal/analysis)	
Doppler Coefficient	(¢/°F)	-0.2192/-.2082 (nominal/analysis)	
Average Fuel Temperature	(°F)	1382	
Scram Reactivity Curve		Figure 6-6c	
Scram Worth	($)	-37.73/-30.18 (nominal-analysis)	

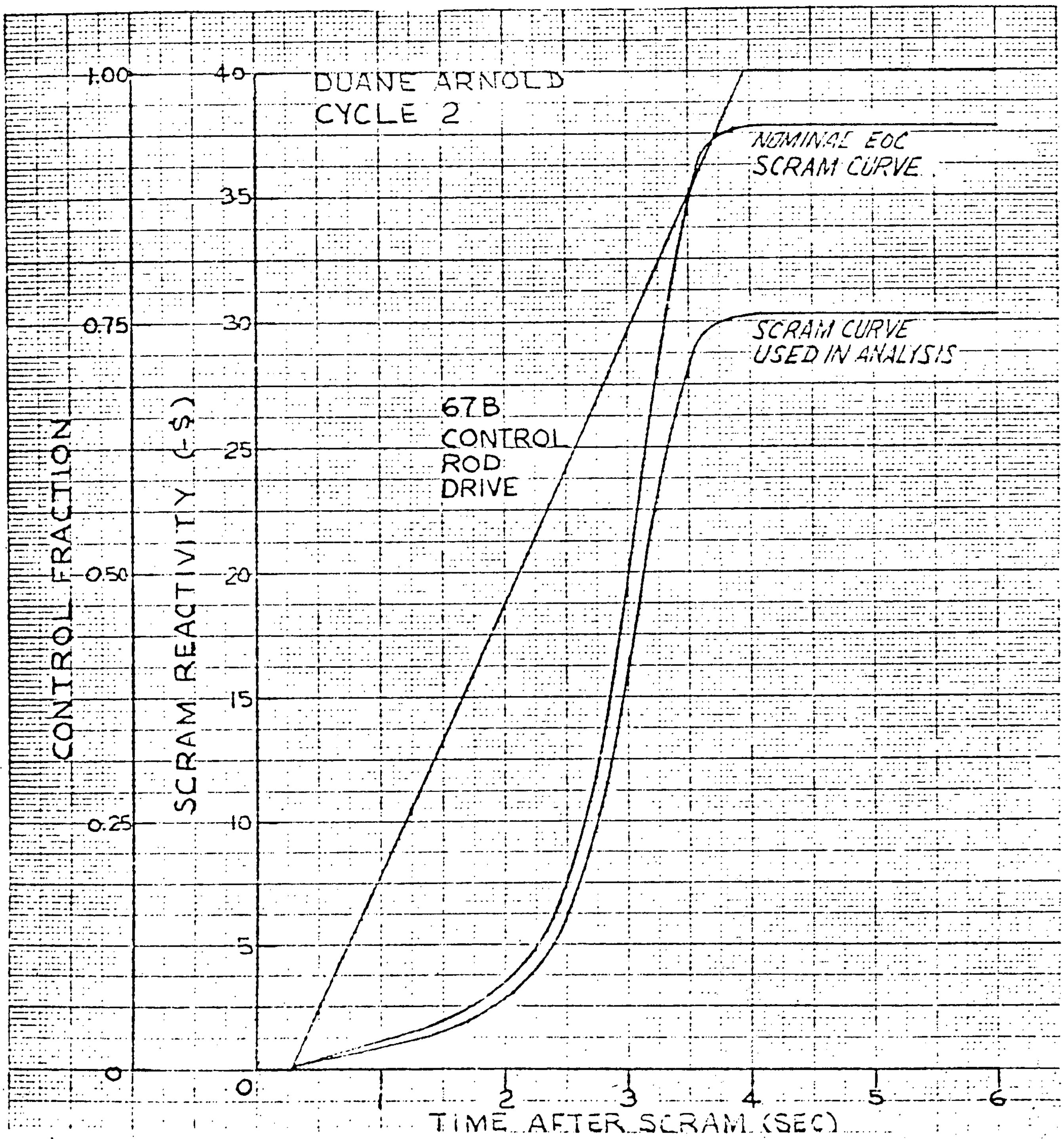

Figure 6-6. Control Rod Drive and Scram Curves at End of Cycle.

Table 6-3

TRANSIENT DATA SUMMARY

Transient	Power (%)	Core Flow (%)	$\hat{\phi}$ ref %	$\hat{Q}/A$ ref %	$\hat{P}_{sl}$ (psig)	$\hat{P}_v$ (psig)	ΔCPR 8x8	ΔCPR 7x7
Turbine Trip w/o **Bypass**-T Scram EOC	104	100	751	123	1211	1249	0.43	0.33
Increased Bypass Flow	104	100	483	121	1209	1250	0.37	0.28
Loss of Feedwater **Heater** EOC	104	100	124	119	--	--	0.19	0.16
Increased Bypass Flow	104	100	121	119	--	--	0.18	0.15

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 for EOC Conditions.

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 for EOC conditions.

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.

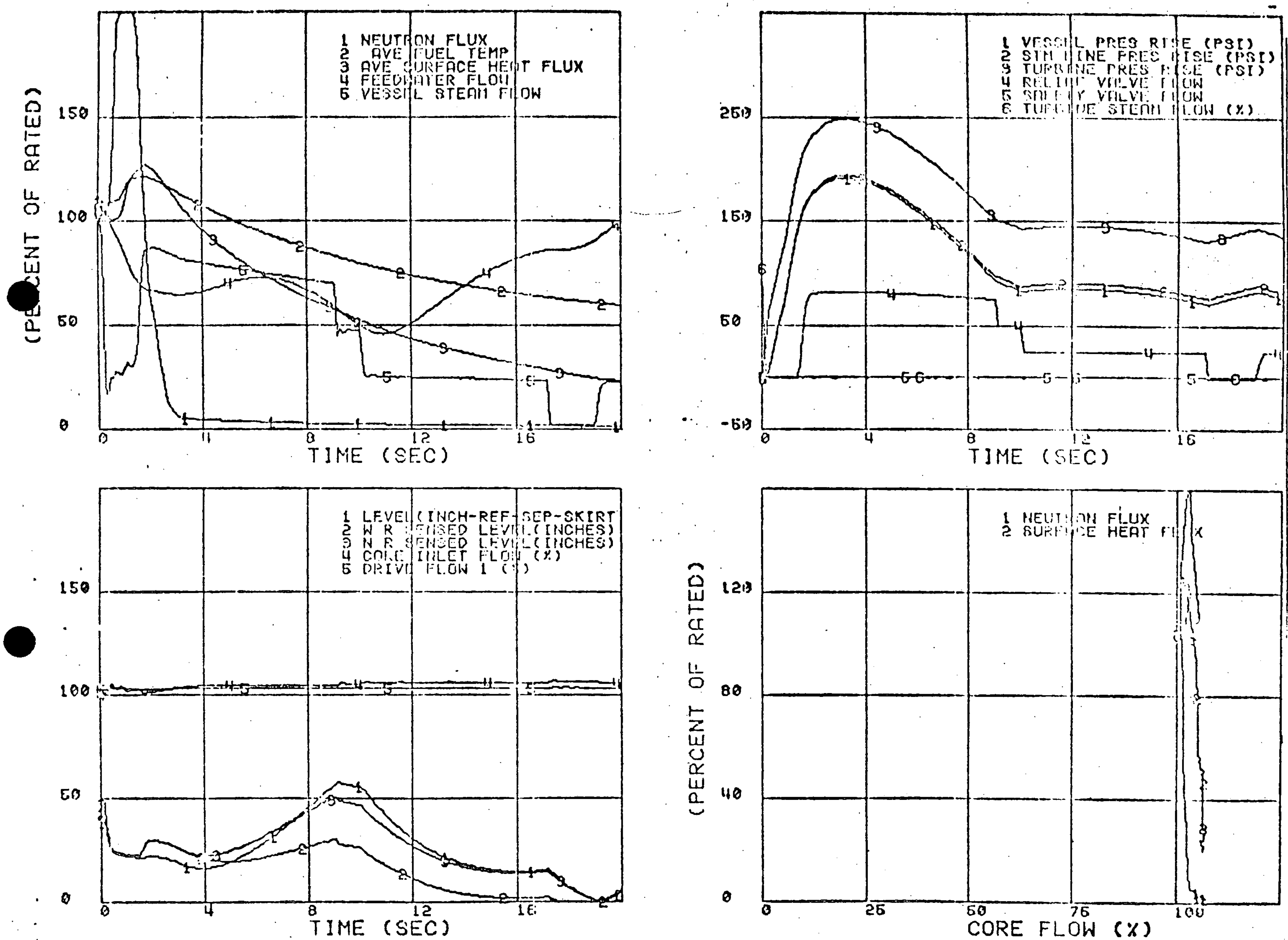

Figure 6.7c Turbine Trip Without Bypass- Increased Core Bypass Flow

Neutron flux, the precursor of heat flux, rises to a peak of 751% of initial power with a corresponding peak heat flux of 123% of its initial value. The resulting ΔCPR from this transient is 0.43 for 8x8 fuel and 0.33 for 7x7 fuel which determines a design basis operating critical power ratio (CPR) of 1.50 and 1.40, respectively.

The peak steamline pressure is limited to 1211 psig as a result of the high-pressure actuation of the six safety/relief valves which provides a 29-psi margin to the 1240-psig set point of the first spring safety valve.

6.3.3.2.1.1 Turbine Trip With Failure of Bypass Valves for cycle 2 incremental Exposure Beyond 2600 MWD/T, Using Increased Core Bypass Flow.

The time responses of the key variables are shown in Figure 6.7c.

The magnitudes of the output parameters are a funxtion of the characteristics of the relief valves and core transient parameters. The specific values of the key output parameters are summarized in Table 6-3. 2

The turbine trip without bypass transient proves to be the limiting transient with respect to the pressure margin of the peak steamline pressure to the lowest setpoint of the spring safety valves (1240 psig). The 31 psi margin for cycle 2 exposures beyond 2600 MWD/T provides surplus margin beyond the 25 psi required margin.

6.3.3.2.2 Loss of a Feedwater Heater for EOC Conditions

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 reactor flux peaks at 124% of initial value and fuel average surface heat flux peaks at 119% of its initial value; a high flux scram occurs 93 seconds after the transient begins. Fuel thermal margins are not exceeded; transient CPR is 0.19 for 8x8 fuel and 0.16 for 7x7 fuel. Transient consequences are milder for lower initial power levels.

6.3.3.2.2.1 Loss of a Feedwater Heater for cycle 2 Incremental Exposure Beyond 2600 MWD/T, Using Increased Core Bypass Flow. 2

The time response of this transient is illustrated in Figure 6.8c.

The specific values of the output parameters are listed in Table 6-3.

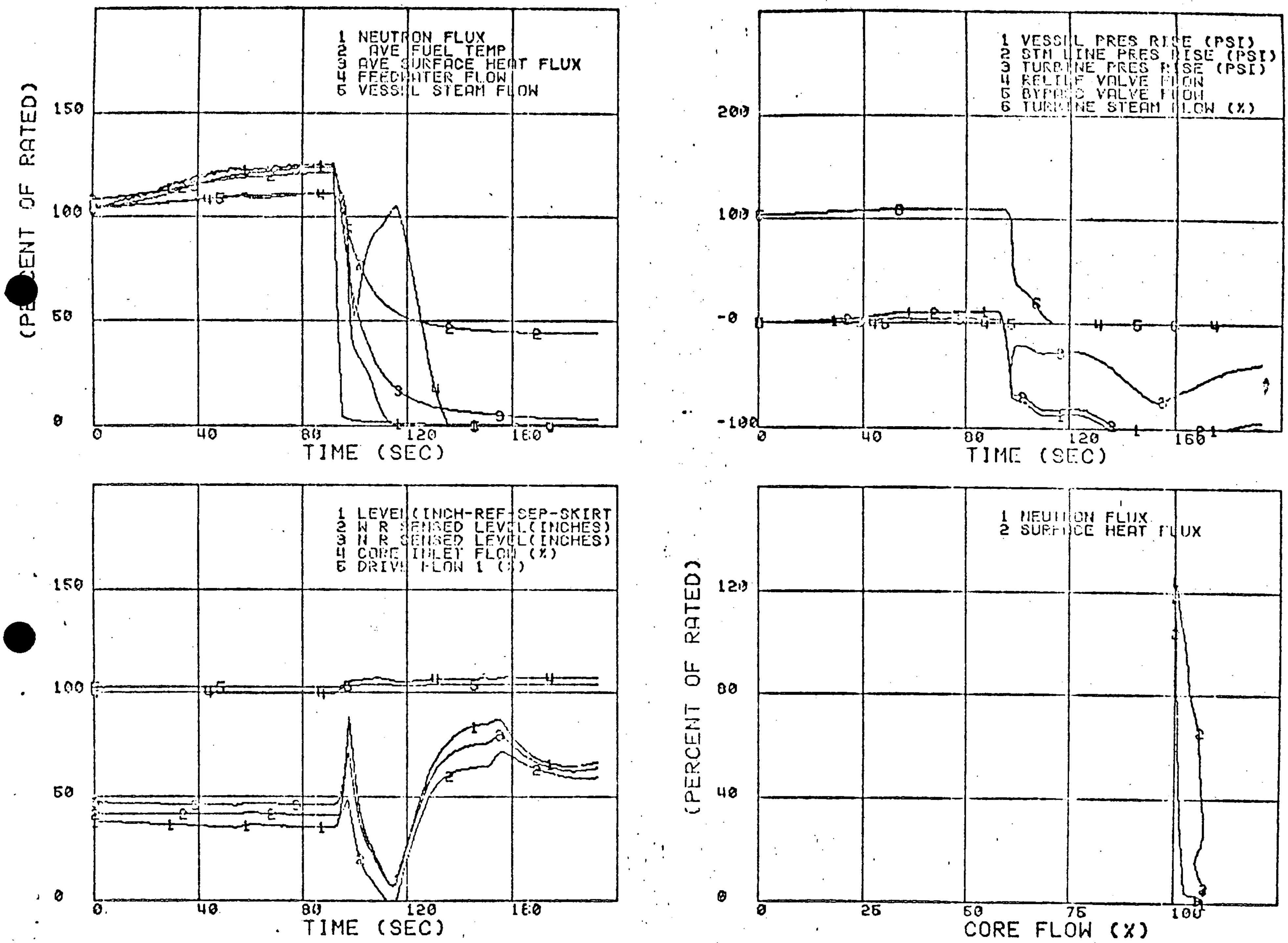

Figure 6.8c. Loss of 100°F Feedwater Heater, Increased Bypass Flow

REFERENCES

1. *GE/BWR Generic Reload Licensing Application for 8x8 Fuel, Revision 1, Supplement 3,* September 1975 (NEDO-20360).

2. *Duane Arnold Energy Center, Safety Analysis with Bypass Flow Holes Plugged,* NEDC-20932, June 1975.

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

4. "Supplemental Infomation for Plant Modification to Eliminate Significant In-Core Vibration," NEDE-21156, January 1976 (Proprietary). 2

5. "Duane Arnold Energy Center Reload No. 1 Licensing Submittal," NEDO-21082-01, Supplement 1, June 1976.