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SUBJECT:
LICENSE NO DPR-49 APPL FOR AMEND: TECH SPECS PROPOSED CHANGE CONCERNING REVISIONS TO CYCLE 3 SAFETY ANALYSIS FOR RECATERGORIZED EVENTS.

PLANT NAME: DUANE ARNOLD

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March 1, 1978
IE-78-315
50-331

LEE LIU
VICE PRESIDENT - ENGINEERING

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

Dear Mr. Case:

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

This application consists of:

(1) Proposed Technical Specifications (RTS-107) reflecting the results of the safety and transient analyses discussed in (2) below.

(2) Duane Arnold Energy Center Unit 1 Cycle 3 Safety Analysis for Recategorized Events.

The enclosed analyses sets forth the bases to recategorize Turbine Trip without Bypass from an incident of moderate frequency to an infrequent incident. Although the anlayses have been conducted for DAEC Cycle 3 which will end shortly, it is applicable to subsequent cycles and we therefore request your review of this submittal. As stated in our December 13, 1977 letter we understand that the review of this item depends on NRC approval of new models based on the Peach Bottom tests.

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.

A001
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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

Lee Liu
Vice President-Engineering

LL/OCS/D
cc:Mr. D. Arnold
Mr. K. Meyer
Mr. R. Lowenstein
Mr. J. Keppler
Mr. R. Clark
Mr. L. Root
File A-117

Subscribed and Sworn to before me on this 1st day of March, 1978.

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

NEDO-24062
Class I
October 1977

DUANE ARNOLD ENERGY CENTER

UNIT 1

CYCLE 3 SAFETY ANALYSIS

FOR RECATEGORIZED EVENTS

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

GENERAL ELECTRIC

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PROPOSED CHANGE RTS-107 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 provides MCPR limits applicable for Cycle 3 operation.

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 the MCPR limits for 7 x 7 and 8 x 8 fuel as shown on the attached sheet.

III. Justification for Proposed Change

This change is proposed in order to incorporate the results of analyses obtained from the turbine trip combined with bypass system failure study. The results of this study are contained in the attached document entitled, "Duane Arnold Energy Center Unit 1 Cycle 3 Safety Analysis for Recategorized Events," NEDO-24062, Class 1, October 1977.

IV. Review Procedure

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

TABLE 3.12-2

MCPR LIMITS

Fuel Type	MCPR Limits
7 x 7	1.21
8 x 8	1.22

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1. INTRODUCTION

On 28 May 1976, the General Electric Company transmitted a letter[1] to the Nuclear Regulatory Commission (NRC) which provided the results of a study to determine the frequency of occurrence of presently analyzed abnormal operating transients based on actual operating data. The impetus for this study was the issuance of Reference 2 by the NRC.

Reference 2 establishes three categories of events based upon frequency of occurrence and qualifies them by acceptable event consequences. These categories are as follows:

Category A - Incidents of moderate frequency (once per year to once in 20 years),

Category B - Infrequent incidents (once in 20 years to once in 100 years),

Category C - Limiting faults (not expected to occur but postulated because occurrence of these events could result in the release of significant amounts of radioactive material).

These three categories were to replace the present "Transient" and "Design Basis Accident" categories.

Plant safety limits for Category A events caused by a single operator error or equipment malfunction are derived from the approved GETAB criteria[3] in which 99.9% of the rods in the core are expected to avoid boiling transition during the event. In addition, a commercial limit is established for plants with unpiped spring safety valves such that actuation of these safety valves is avoided. For Category B events, the GETAB Fuel Cladding Safety (Transient) Limit established for Category A events may be exceeded on a limited number of fuel rods and unpiped safety valves are allowed to open. The philosophy used to evaluate the Category C faults has not changed from that used to analyze the previously categorized "Design Basis Accidents."

The results of the study transmitted in the Reference 1 letter indicated that several abnormal operating events were considered to be infrequent incidents.

A list of events considered to be moderate frequency, Category A incidents is given in Table 1-1; events considered to be infrequent incidents are listed in Table 1-2. Of the events in Category B, Turbine Trip without Bypass has been analyzed in detail to determine the consequence of occurrence upon fuel integrity if this event was not used to establish the Category A operating MCPR limit. The results of this analysis, documented in Reference 4, showed that no fuel failures were calculated to occur because of these events. Until similar documentation is provided for the other Category B events, these events will be limited by and will be used to determine the Category A operating limit.

This report documents the results of the safety analysis performed for the Duane Arnold Energy Center Nuclear Plant (DAEC) using the above recategorized basis. The analysis was performed for the half drilled core configuration documented in Reference 5 at the end-of-cycle 3 (EOC3) conditions. Recategorization does not affect the statistical Fuel Cladding Safety Limit or the results of the accident (Category C) analyses given in Reference 6. Therefore, only the results of the Category A and Category B analyses are given in this document.

Table 1-1

CATEGORY A

INCIDENTS OF MODERATE FREQUENCY

Transient	Frequency (Events/Plant Yr)
Generator Load Rejection	0.75
Turbine Trip	1.33
MSIV Closure	1.0
Pressure Regulator Failure-Closed	N/A
Inadvertent Start of HPCI Pump	0.20
Loss of Feedwater Flow	0.75
Pressure Regulator Failure-Open	0.67
One Recirculation Pump Trip	1.33
Two Recirculation Pump Trip	0.25
Recirculation Flow Controller Failure-Decreasing	0.25
Recirculation Flow Controller Failure-Increasing	0.25
Abnormal Start of Idle Recirculation Pump	0.10
Loss of RHR Shutdown Cooling	N/A
Feedwater Controller Failure (Maximum Demand)	0.5
Inadvertent Opening of Safety/Relief Valve	0.1
Loss of Condenser Vacuum	0.67
Loss of Auxiliary Power	0.25

Table 1-2

CATEGORY B

INFREQUENT INCIDENTS

Incident	Frequency (Events/Plant Yr)
Generator Load Rejection with Bypass Failure	0.0036
Turbine Trip with Bypass Failure	0.0064
Loss of Feedwater Heating	*
Rod Withdrawal Error at Power	*
Rod Withdrawal Error during Startup	*

*The data base required to establish the frequency of these events is not yet available. For the present these events are conservatively considered to be members of Category A.

2. CATEGORY A - MODERATE FREQUENCY EVENTS

The results of the most limiting moderate frequency pressure and power increase transients were evaluated to determine the largest decrease in MCPR. In addition, the results of the infrequent loss of 100° feedwater heating and local rod withdrawal error events were conservatively evaluated as moderate frequency events as described above.

The initial conditions used in the DAEC GETAB analysis are given in Table 2-1. Input parameters to the transient analyses are the same as those documented in Table 6-1 of Reference 5. The digital computer model described in Reference 6 was used in these analyses. Results of the analyses are summarized in Table 2-2. A review of these results shows that the limiting Category A events for DAEC at the end of Cycle 3 are the feedwater controller failure event with a ΔMCPR of 0.21 for the reload 8x8 fuel, and feedwater controller failure and rod withdrawal error, each with a ΔMCPR of 0.15, for the remaining 7x7 initial core fuel. The limiting event for exposures up to 1000 MWd/t before the end of Cycle 3 is the rod withdrawal error with a ΔMCPR of 0.16 for the 8x8 fuel and 0.15 for the 7x7 fuel, as shown in Reference 5. The turbine trip with bypass and feedwater controller failure event are less severe than the turbine trip without bypass event; therefore, the mid-cycle ΔMCPR reported in Reference 5 is conservatively applied to the less severe events in this report. Adding the results of these limiting events to the Fuel Cladding Safety Limit of 1.06 results in new DAEC Cycle 3 MCPR operating limits of 1.22 for 8x8 fuel and 1.21 for 7x7 fuel for exposures up to 1000 MWd/t before the end of Cycle 3, and 1.27 for 8x8 fuel and 1.21 for 7x7 fuel for exposures from 1000 MWd/t before the end of Cycle 3 to the end of Cycle 3. Details of the event descriptions and results are given below.

2.1 TURBINE TRIP WITH OPERABLE BYPASS

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are: moisture separator and heater drain tank high levels, large vibration, loss of control fluid pressure, loss of condenser vacuum and reactor high water level.

The following sequence of events occurs for a turbine trip:

a. The turbine stop valves close over a period of approximately 0.1 second.

b. A reactor scram is initiated from position switches on the turbine stop valves at 10% closure.

c. The turbine bypass valves are opened by the turbine control system. Delay after start of stop valve closure is 0.1 second.

d. The pressure continues to rise until the pressure relief set points are reached, some or all of the safety/relief valves briefly discharge steam to the suppression pool.

This event would produce a transient as shown in Figure 2-1. The initial reactor power is at a level corresponding to 105% of NBR steam flow, the neutron flux peaks at 198.7% rated, the average surface heat flux peaks at 105.9% rated.

The peak streamline pressure is limited to 1142 psig as a result of the high-pressure actuation of the six safety/relief valves, which provides a 98-psi margin to the 1240-psig set point of the first spring safety valve.

2.2 FEEDWATER CONTROLLER FAILURE

An event that can directly cause excess coolant inventory is one in which feedwater flow is increased. The most severe applicable event is a feedwater controller failure in the maximum demand direction. The transient was initiated from a level corresponding to 105% of NBR steam flow. The feedwater controller was assumed to fail such as to demand maximum feedwater valve opening resulting in a maximum runout flow of 135% of NBR rated feedwater flow at a system pressure of 1060 psig. With excess feedwater flow, the water level rises to the high level trip setpoint, at which time the main turbine and feedwater pumps are tripped and a reactor scram is initiated. Figure 2-2 shows the results of this transient.

2.3 LOSS OF 100°F FEEDWATER HEATING

The transient analysis documented in Reference 5 is not affected by recategorization and is therefore still valid (Figure 2-3). The resultant MCPR is given in Table 2-2.

2.4 LOCAL ROD WITHDRAWAL ERROR

The analysis documented in Reference 5 is not affected by recategorization.

Table 2-1
GETAB TRANSIENT ANALYSIS
INITIAL CONDITION PARAMETERS

EOC3 Evaluation	7x7	8x8
Peaking Factors (local, radial and axial)	(1.24, 1.282, 1.40)	(1.22, 1.364, 1.40)
R-Factor	1.100	1.098
Bundle Power, MWt	5.447	5.791
Nonfuel Power Fraction	0.04	0.04
Core Flow, Mlb/hr	49.0	49.0
Bundle Flow, 10^3 lb/hr	128.1	114.3
Reactor Pressure, psia	1035.0	1035.0
Inlet Enthalpy, Btu/lb	526.3	526.3
Initial MCPR	1.22	1.28

GETAB TRANSIENT ANALYSIS
INITIAL CONDITION PARAMETERS

1 GWd/t Before EOC3 Evaluation	7x7	8x8
Peaking Factors (local, radial and axial)	(1.24, 1.286, 1.40)	(1.22, 1.423, 1.40)
R-Factor	1.100	1.098
Bundle Power, MWt	5.464	6.042
Nonfuel Power Fraction	0.04	0.04
Core Flow, Mlb/hr	49.0	49.0
Bundle Flow, 10^3 lb/hr	128.0	112.7
Reactor Pressure, psia	1035.0	1035.0
Inlet Enthalpy, Btu/lb	526.3	526.3
Initial MCPR	1.21	1.22

Table 2-2
CATEGORY A
EVENT RESULTS

EVENT	INITIAL POWER (%) NBR	$\hat{\emptyset}$ (%) NBR (EOC3)	$\hat{Q}/A$ (%) NBR (EOC3)	$\hat{P}_{LINE}$ (PSIG) (EOC3)	CPR for Exposures Up to 1000 MWd/t Before EOC3: 7x7	CPR for Exposures Up to 1000 MWd/t Before EOC3: 8x8	CPR for Exposures 1000 MWd/t Before EOC3 to EOC3: 7x7	CPR for Exposures 1000 MWd/t Before EOC3 to EOC3: 8x8
Turbine Trip with Bypass	104	198.7	105.9	1142			0.10	0.14
FW Controller Failure	104	225.0	113.0	1146	0.05	0.06	0.15	0.21
Loss of 100°F Feedwater Heating	104	121.0	119.0	1023	0.14	0.15	0.14	0.15
Rod Withdrawal Error (RBM Set to 105%)	--	--	--	--	0.15	0.16	0.15	0.16

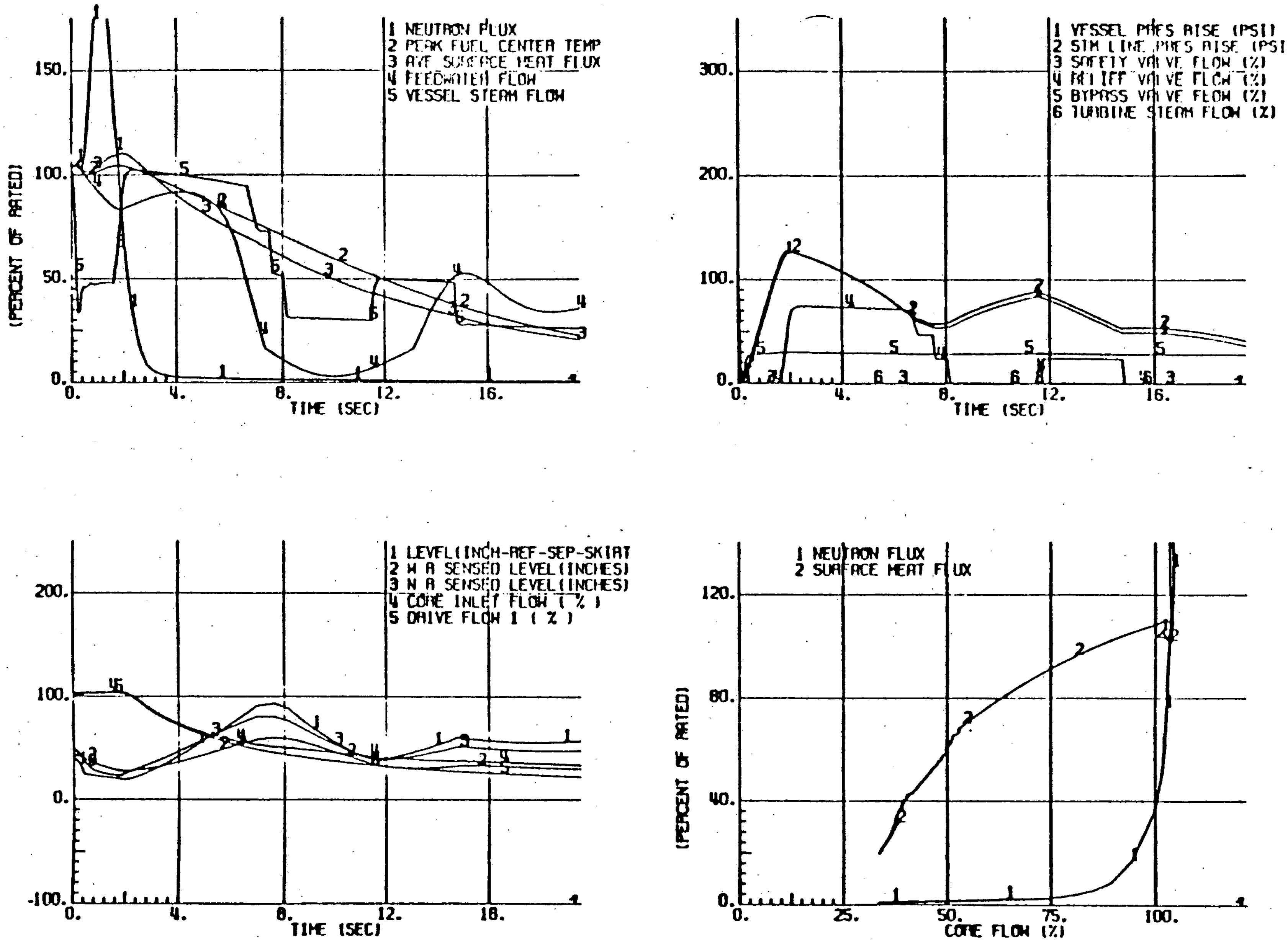

Figure 2-1. Turbine Trip with Bypass (DAEC, EOC3, 104% Power, 100% Flow, Half Drill, Trip Scram)

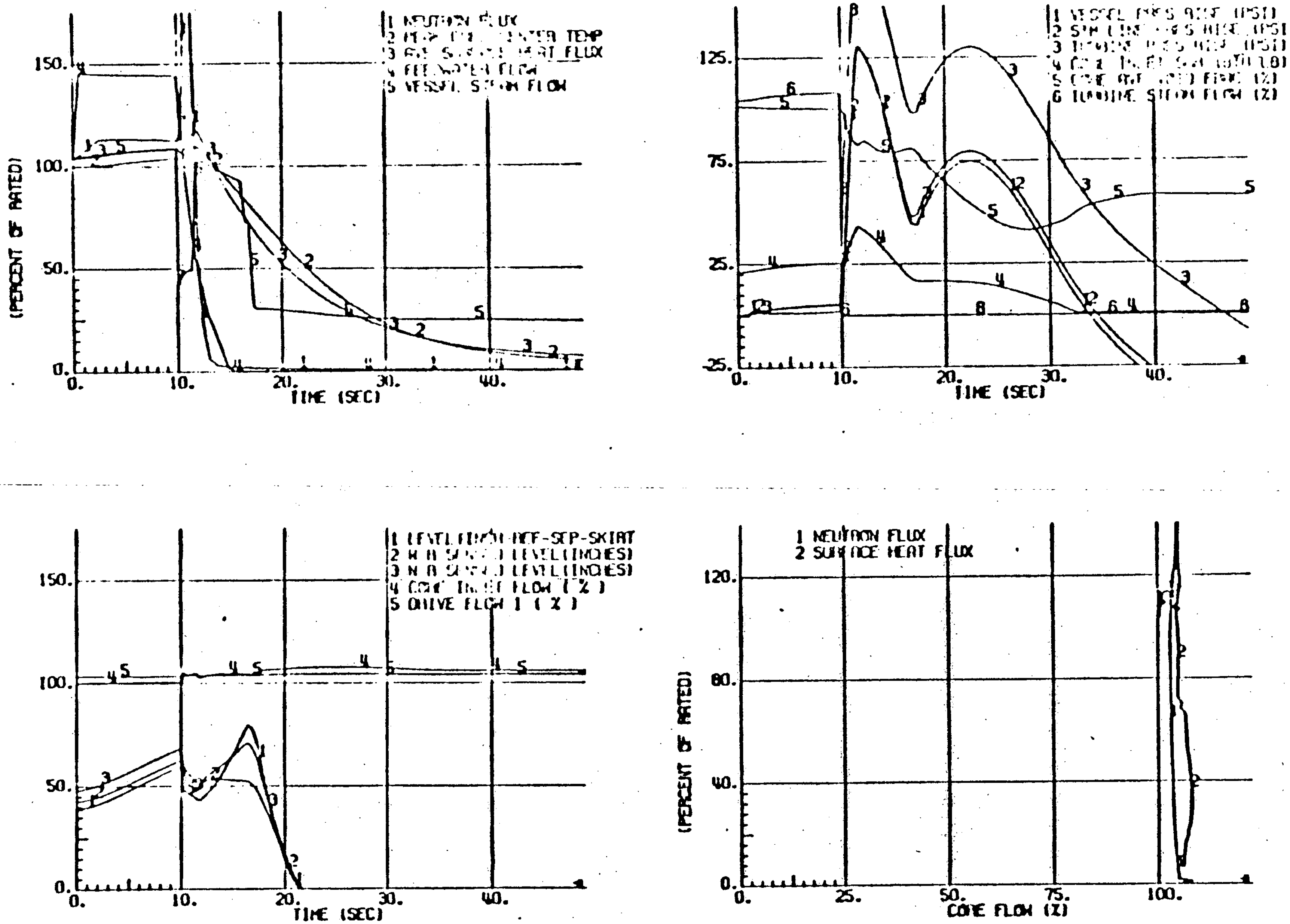

Figure 2-2 Feedwater Controller Failure (DAEC, EOC3, 104% Power, 100% Flow, Half Drill)

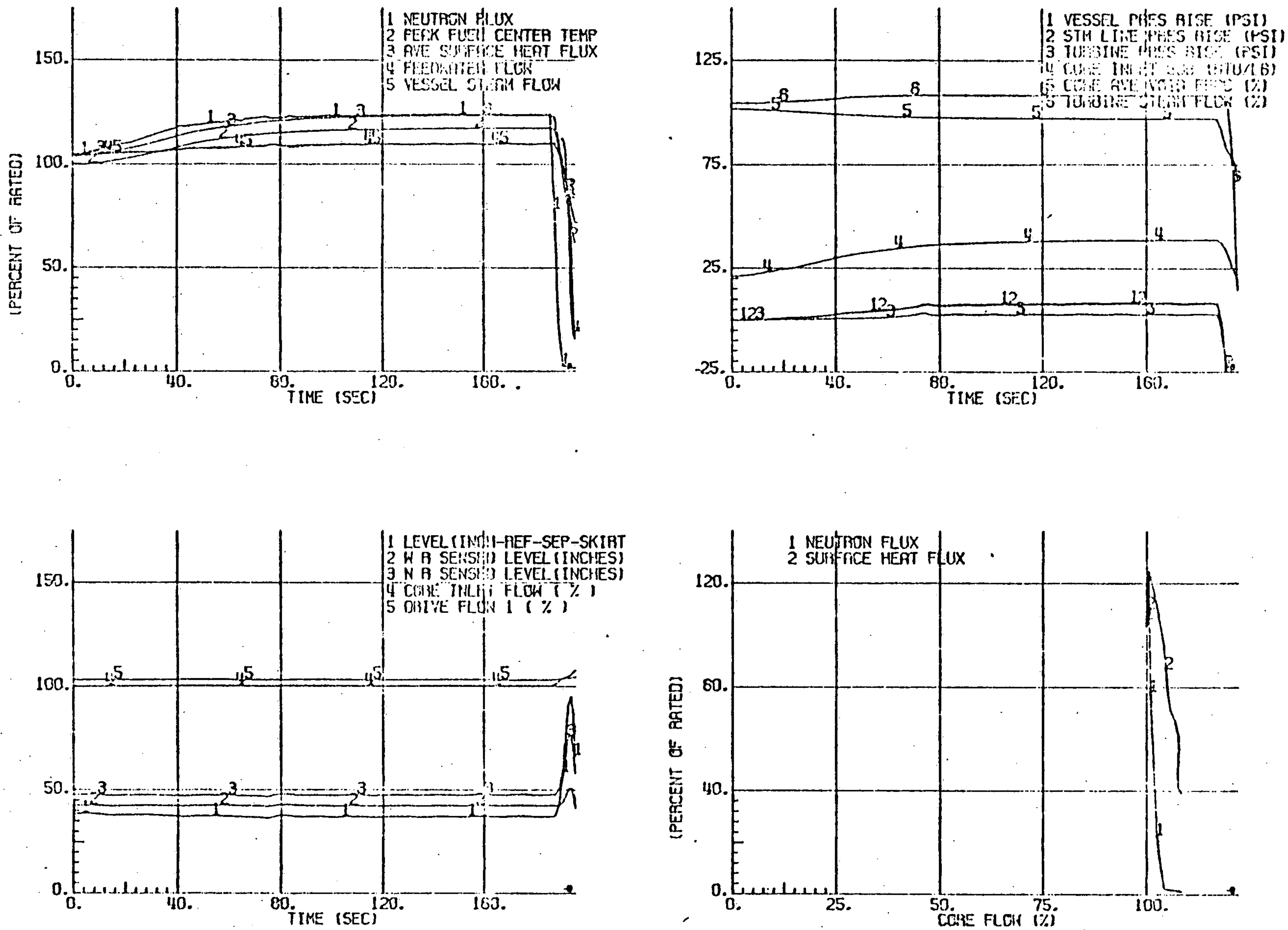

Figure 2-3. Loss of 100% Feedwater Heater, Manual Flow Control, FSCRAM (DAEC, EOC3, 104% Power, 100% Flow, Half Drill)

3. CATEGORY B - INFREQUENT INCIDENTS

The only event analyzed herein as part of this category is the turbine trip combined with bypass system failure. This section is included to cover the transient response of infrequent incidents when MCPR limits have been applied to Category A (moderate frequency) incidents. The initial conditions and input parameters used in the analyses are given in Section 2. The most severe Category B event is Turbine Trip without Bypass. The results of the analyses are given in Table 3-1. Subtracting the ΔCPR for this event from the Category A MCPR operating limit yields a MCPR of 0.98 (1.27 - 0.29) for 8x8 fuel and 1.00 (1.21 - 0.21) for 7x7 fuel. However, as shown in Reference 4, continued fuel integrity is assured for this event. Details of the analysis are given below.

3.1 TURBINE TRIP WITH FAILURE OF THE BYPASS VALVES

The primary characteristic of the turbine trip without bypass 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. Core net reactivity is sharply positive and causes a rapid increase in neutron flux until the net reactivity is forced negative by the scram initiated from the position switches on the turbine stop valves.

This unlikely event would produce a transient as shown in Figure 3-1. The initial reactor power is at a level corresponding to 105% of rated steam flow, the neutron flux peaks at 351% rated, the average surface heat flux peaks at 115% of initial. This transient is slightly more severe than that which results from full load rejection with bypass failure.

Table 3-1
CATEGORY B
EVENT RESULTS

EVENT	INITIAL POWER (%)	Ø (%) NBR	QA (%) Initial	ΔCPR 7x7	ΔCPR 8x8	Rods Subject to Boiling Transition (%)
Turbine Trip w/o Bypass	104.1	351	115	0.21	0.29	0.88

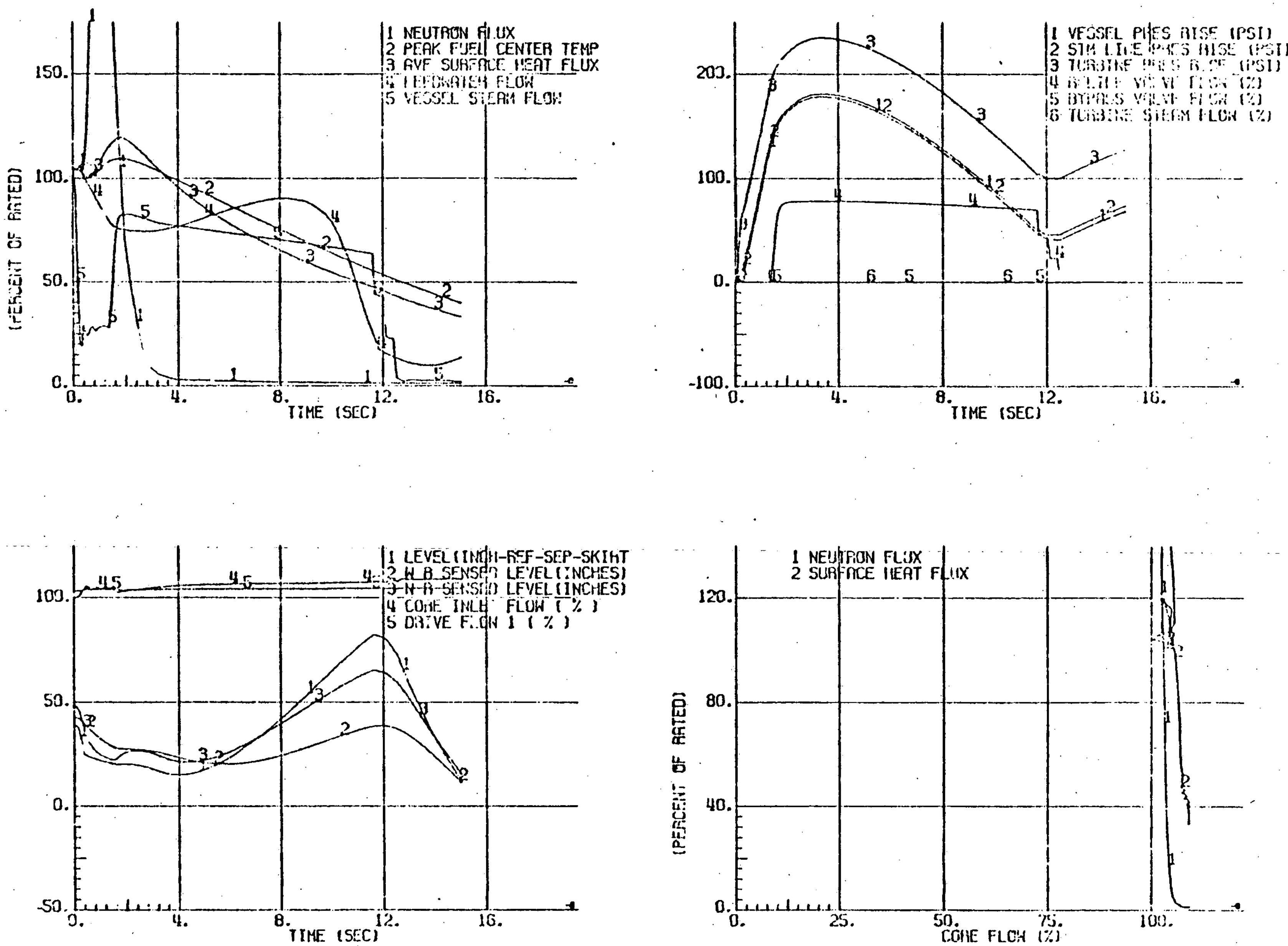

Figure 3-1. Turbine Trip without Bypass - Trip Scram (DAEC, EOC3, 104% Power, 100% Flow, Half Drill)

4. REFERENCES

1. Letter, E. A. Hughes to R. C. DeYoung, "Transient Recategorization per Regulatory Guide 1.70, Revision 2," 28 May 1976.

2. Nuclear Regulatory Commission, Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," September 1975.

3. "General Electric Thermal Analysis (GETAB): Data, Correlation, and Design Application," January 1977, (NEDO-10958-A).

4. Letter, E. A. Hughes to R. C. DeYoung, "Turbine Trip without Bypass Analyzed as an Infrequent Incident," 5 October 1976.

5. "General Electric Boiling Water Reactor, Reload Number 2 Licensing Submittal, Supplement 1, Partially Drilled Core, Duane Arnold Energy Center," June 1977, (NEDO-21082-02-1).

6. R. B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," February 1973, (NEDO-10802).