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DESCRIPTION

ENCLOSURE

Trans of appl for amend to DPR-59, RTS-105 to incorporate proposed changes to tech specs (Appendix A to license.)

(27-P)

PLANT NAME: Duane Arnold Energy Center

RHF 1/3/77

(1-P)

FOR ACTION/INFORMATION

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December 14, 1977
IE-77-2256

LEE LIU
VICE PRESIDENT – ENGINEERING

RECEIVED DEC 22 1977 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20545

Dear Mr. Case:

Transmitted herewith in accordance with the requirements of 10CFR50.59 and 50.90 is an application for amendment of DPR-59, RTS-105 to incorporate proposed changes to the Technical Specifications (Appendix A to License) for the Duane Arnold Energy Center.

The present operating limit MCPR's result in an approximate 10% derate during the last 1000 MWD/t exposure. Utilization of an additional MCPR point from 1000 MWD/t to 500 MWD/t before end-of-cycle reduces the derate approximately 5% for thirty days. The enclosed Technical Specifications include the additional MCPR limit for the period from 1000 MWD/t to 500 MWD/t before end-of-cycle.

This change does not result in any change to the presently licensed safety limit MCPR of 1.06.

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

7800030206

LL/KAM/gan
Enclosure

Subscribed and Sworn before me on this 14th day of December.

Jean R. Smith
____
Notary Public in and for the State of Iowa

STATE OF IOWA
Commission Expires
September 30, 1978

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

PROPOSED CHANGE RTS-105 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

Table 3.12-2 contains MCPR limits for the 7 x 7 and 8 x 8 fuel for fuel exposure from Beginning of Cycle to greater than 1000 MWD/T and from less than or equal to 1000 MWD/T to End of Cycle.

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 fuel exposure headings, MCPR numbers and references as shown in the attached sheets.

III. Justification for Proposed Change

This proposed Technical Specification change is submitted in order to obtain MCPR operating relief for the fuel exposure range of less than or equal to 1000 MWD/T to greater than 500 MWD/T. The attached safety analysis supplements NEDO-21082-02-1 (June 1977, Licensing Submittal Supplement 1 for Partially Drilled Core).

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		
	B.O.C. to >1000 MWD/T	≤ 1000 MWD/T to >500 MWD/T	≤ 500 MWD/T to E.O.C.
7 x 7	1.27	1.27	1.29
8 x 8	1.27	1.32	1.37

3.12 REFERENCES

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

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

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

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

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

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

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

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

9. Duane Arnold Energy Center Reload Number Two Licensing Submittal, Supplement 1, Partially Drilled Core, NEDO-21082-02-01, Class I, June 1977.

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

11. Duane Arnold Energy Center Reload Number Two Licensing Submittal, Supplement 2, Additional MCPR Point, NEDO-21082-02-2, Class I, December 1977.

NEDO-21082-02-2

Class I

December 1977

Supplement 2

GENERAL ELECTRIC BOILING WATER REACTOR
RELOAD NUMBER 2 LICENSING SUBMITTAL
SUPPLEMENT 2
ADDITIONAL MCPR POINT
DUANE ARNOLD ENERGY CENTER

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

GENERAL ELECTRIC

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LIST OF TABLES

1. INTRODUCTION

This document provides 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 core loading has been described in Reference 4. The analysis in that document was done for all core support plate holes plugged and the 100 reload 2 bundles drilled. The analysis in Reference 7 was for 184 bundles drilled and is conservative for the actual core which has 200 drilled bundles. (Rod Withdrawal Error and Bundle Loading Error were done assuming all bundles drilled; this is the most conservative case for these analyses.)

Reference 7 also provided analysis to support exposure dependent minimum critical power ratio (MCPR) operating limits. The MCPR due to various transients was analyzed at different exposures resulting in greater flexibility throughout most of the cycle. Transient analyses were performed to obtain an operating limit MCPR from BOC to 1000 MWd/t before EOC, and for the last 1000 MWd/t before EOC. This submittal supplements the exposure dependent MCPR operating limits in Reference 7 by presenting results of analyses performed for an exposure of EOC minus 500 MWd/t.

2. SUMMARY

The core loading has been described in Reference 4 and the effects of the safety relief valve replacement have been presented in Reference 5. Reference 7 presented changes due to drilling of additional bundles, use of improved bundle loading error analysis, and use of the recently approved ECCS evaluation model (Reference 8).

3. MECHANICAL DESIGN

The basic mechanical design of the fuel is described in References 1, 4, and 7.

4. THERMAL-HYDRAULIC ANALYSES

Discussion 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 with bypass flow holes plugged, is given below and in References 4, 5, and 7.

4.1 STATISTICAL ANALYSIS

The statistical analysis is described in Reference 4.

4.1.1 Fuel Cladding Integrity Safety Limit

The fuel cladding integrity safety limit is a MCPR of 1.06.

4.1.2 Basis for Statistical Analyses

The basis for the statistical analysis is described in Reference 4.

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 4-1.

Addition of the CPR 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 1.21 for 7x7 and 1.22 for 8x8 fuels from BOC3 to 1 GWd/t before EOC3, 1.25 for 7x7 and 1.32 for 8.8 fuels 8.8 fuels from 1 GWd/t before EOC3 to 500 MWd/t before EOC3, and 1.29 for 7x7 and 1.37 for 8x8 fuels during the final 500 MWd/t of Cycle 3.

Table 4-1

SUMMARY OF RESULTS LIMITING ABNORMAL OPERATIONAL ΔCPR TRANSIENTS

Event	EOC3		EOC3 - 0.5 GWd/t		EOC3 - 1 GWd/t	
	7x7	8x8	7x7	8x8	7x7	8x8
Turbine Trip w/o Bypass	0.23	0.31	0.19	0.26	0.08	0.12
Loss of Feedwater Heater	0.14	0.15	0.14	0.15	0.14	0.15
Rod Withdrawal Error	0.15	0.16	0.15	0.16	0.15	0.16

4.3 TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

The magnitude of values used as initial input conditions for the transient analysis is shown in Table 4-2.

Table 4-2

GETAB TRANSIENT ANALYSIS

	7x7	8x8
INITIAL CONDITION PARAMETERS EOC		
Peaking Factors (Local, Radial and Axial)	(1.24, 1.20, 1.40)	(1.22, 1.27, 1.40)
R-Factor	1.100	1.098
Bundle Power, MWt	5.107	5.375
Nonfuel Power Fraction	0.04	0.04
Core Flow, Mlb/hr	49.0	49.0
Bundle Flow, 10^3 lb/hr	130.2	117.1
Reactor Pressure, psia	1035.0	1035.0
Inlet Enthalpy, Btu/lb	526.3	526.3
Initial MCPR	1.30	1.38
INITIAL CONDITION PARAMETERS EOC-1 GWd/t EVALUATIONS		
Peaking Factors (Local, Radial and Axial)	(1.24, 1.29, 1.40)	(1.22, 1.42, 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 4-2 GETAB TRANSIENT ANALYSIS (Continued)

INITIAL CONDITION PARAMETERS EOC-0.5 GWd/t

Peaking Factors (Local, Radial and Axial)	(1.24, 1.23, 1.40)	(1.22, 1.31, 1.40)
R-Factor	1.100	1.098
Bundle Power, MWt	5.241	5.560
Nonfuel Power Fraction	0.04	0.04
Core Flow, Mlb/hr	49.0	49.0
Bundle Flow, 10^3 lb/hr	125.8	115.7
Reactor Pressure, psia	1035.0	1035.0
Inlet Enthalpy, Btu/lb	526.3	526.3
Initial MCPR	1.26	1.33

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 and in Reference 4.

5.1 NUCLEAR CHARACTERISTICS OF THE CORE

The reactivity control characteristics of the core are presented in Reference 4. This section presents the results of the calculation of core average reactivity coefficients.

5.1.1 Core Effective Multiplication, Control System Worth and Reactivity Coefficients

The core effective multiplication and control system worth are discussed in Reference 4. The reactivity coefficients for the half-drilled core are given in Reference 7.

5.1.2 Reactor Shutdown Margin

The reactor shutdown margin is discussed in Reference 4.

5.1.3 Standby Liquid Control System

The standby liquid control system is discussed in Reference 4.

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 control rod drop accident is discussed in Reference 4.

6.3.2.4 Loss-of-Coolant Accident (LOCA)

The LOCA analysis is presented in Reference 8.

6.3.2.5 Loading Error Accident

The loading error analysis is presented in Reference 6 and 7.

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 previously been performed to obtain an operating limit MCPR from BOC3 to 1 Gwd/t before EOC3, and 1 GWd/t before EOC3 to EOC3. New analyses have been performed to obtain an operating limit MCPR from 1000 MWd/t before EOC3 to 500 MWd/t before EOC3.

6.3.3.1.2 Input Data and Operating Conditions

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

6.3.3.1.3 Transient Summary

A summary of the transients analyzed and their consequences is provided in Table 6-2. Figure 6-1 illustrates the transients analyzed at EOC3-500 MWd/t.

6.3.3.2 Transient Description

The transient descriptions are presented in Reference 7.

6.3.4 ASME Vessel Pressure Code Compliance

This analysis is presented in Reference 7.

6.3.5 Thermal-Hydraulic Stability Analysis

This analysis is presented in Reference 7.

Table 6-1

TRANSIENT INPUT PARAMETERS

Thermal Power	(MWt)	1657	104% Rated
Steam Flow	(lb/hr)	7.16×10^6	105% NBR
NBR Core Flow	(lb/hr)	49.0×10^6	100% NBR
Dome Pressure	psig	1020	
Turbine Pressure	psig	960	
RV Set Point (nominal/analysis)	psig	1090/1101	
RV/Capacity (at Set Point)	No./%NBR	6/72.0 (Old Value 6/74.7)	
RV Time Delay	(msec)	400	
RV Stroke Time	(msec)	100	
SV Set Point (nominal/analysis)	psig	1240/1253	
SV/Capacity (at Set Point)	No./%NBR	2/18.9	

		EOC3	EOC3- 1 GWd/t	EOC-3 500 MWd/t
Dynamic Void Coefficient	(-c/%Rg)	11.80	12.01	12.13
Doppler Coefficient	(-c/°F)	0.2186	0.2110	0.2015
Average Coefficient	(°F)	1435	1435	1432
Scram Reactivity	(°F)	Reference 7	Reference 7	Figure 6-2
Scram Reactivity	(-$)	30.16	29.20	30.95

Table 6-2

TRANSIENT DATA SUMMARY

Transient	Power (%)	Core Flow (%)	ϕ (% reference)	Q/A (% reference)	Psl (Psig)	ΔCPR 8x8	ΔCPR 7x7
Turbine Trip w/o Bypass							
EOC	104	100	351	115	1196	0.31	0.23
1 GWd/t Before EOC	104	100	198	105	1171	0.12	0.08
500 MWd/t Before EOC	104	100	322	112	1188	0.26	0.19
Loss of Feedwater Heater	104	100	121	119	1023	0.15	0.14

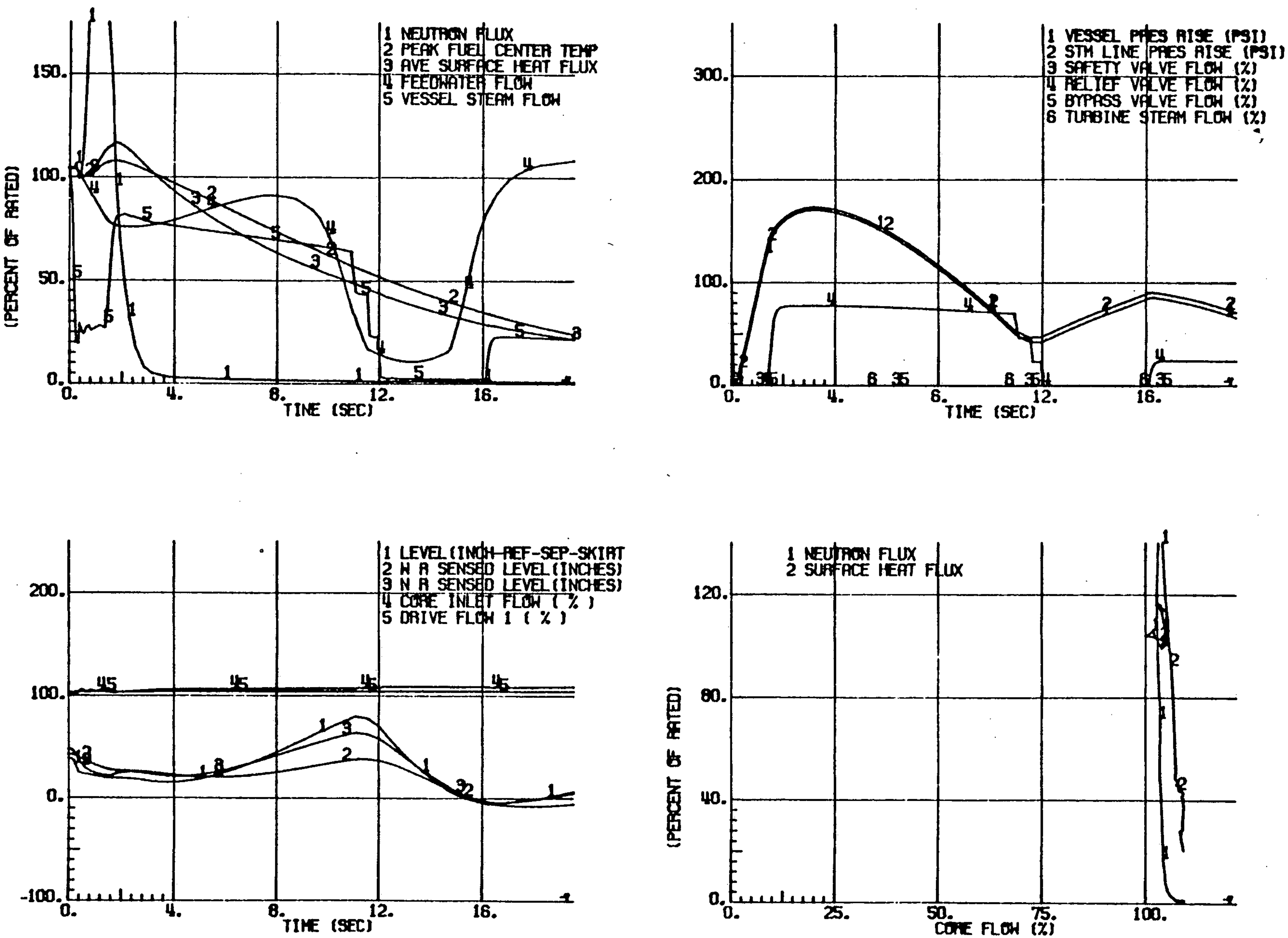

Figure 6-1. Turbine Trip Without Bypass, Trip Scram (DAEC; EOC3-500 MWd/t, 104% Power, 100% Flow, Half Drill)

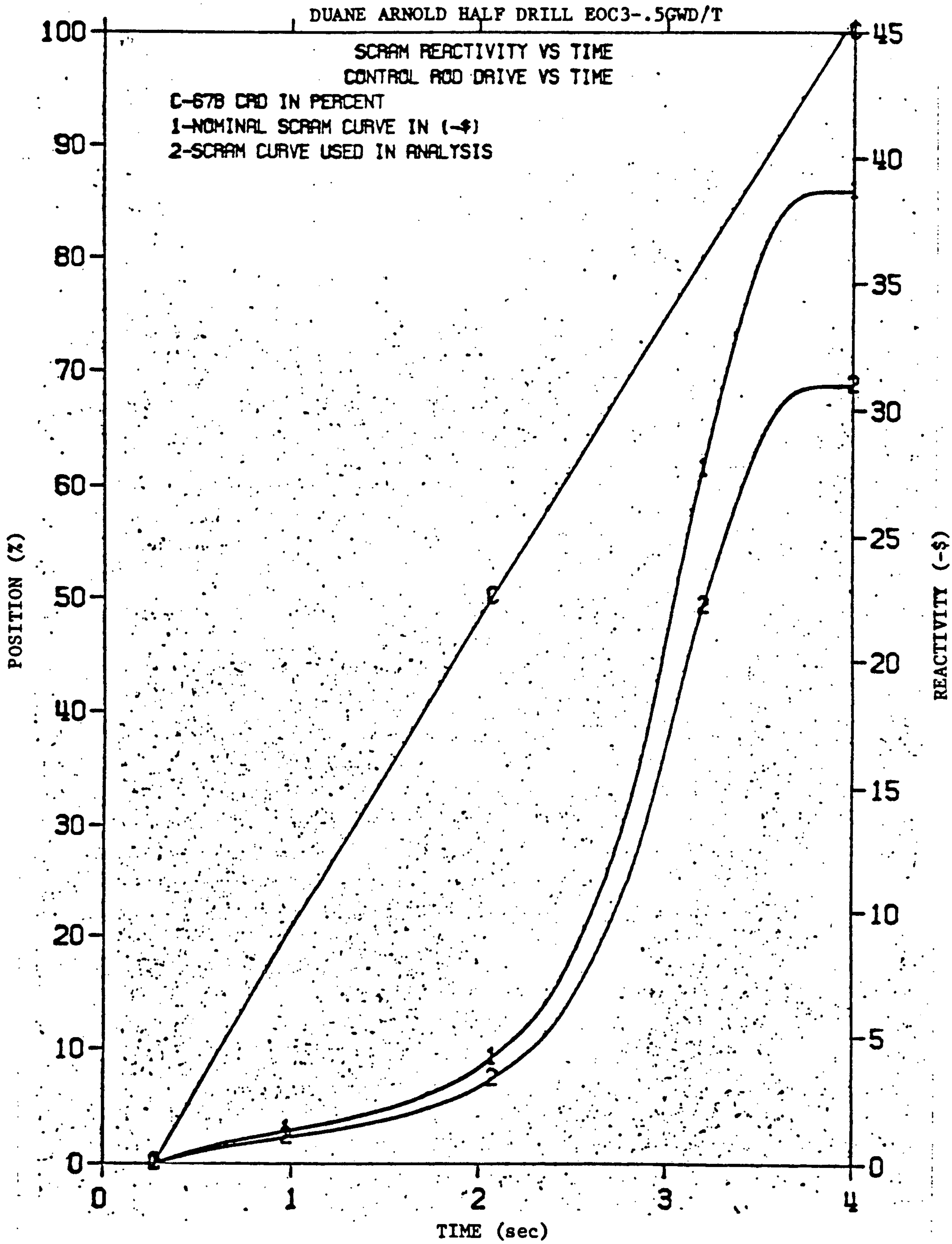

Figure 6-2. Scram Reactivity vs Time

REFERENCES

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

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).

4. General Electric Boiling Water Reactor Reload 2 Licensing Submittal Duane Arnold Energy Center, January 1977 (NEDO-21082-02).

5. Safety Evaluation, Safety/Relief Valve Replacement, Duane Arnold Energy Center, May 1977.

6. Letter, Ronald Engel to Darrell Eisenhut, "Fuel Assembly Loading Error," June 1, 1977.

7. General Electric Boiling Water Reactor Reload No. 2 Licensing Submittal Supplement 1 Partially Drilled Core, Duane Arnold Energy Center, June 1977 (NEDO-21082-02-1).

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