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GENERAL ELECTRIC BOILING WATER REACTOR RELOAD-3 (CYCLE 4) LICENSING AMENDMENT FOR DUANE ARNOLD ENERGY CENTER SUPPLEMENT 3: APPLICATION OF MEASURED SCRAM INSERTION TIMES

NEDO-24087-3 DAEC RELOAD-3 (CYCLE 4) LICENSING AMENDMENT (SUPPLEMENT 3)

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GENERAL ELECTRIC BOILING WATER REACTOR
RELOAD-3 (CYCLE 4) LICENSING AMENDMENT
FOR DUANE ARNOLD ENERGY CENTER
SUPPLEMENT 3: APPLICATION OF MEASURED SCRAM INSERTION TIMES

NUCLEAR ENERGY PROJECTS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

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CONTENTS OF THIS REPORT

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1. INTRODUCTION AND SUMMARY

Reference 1 contained the safety analysis for the Duane Arnold Energy Center (DAEC) cycle 4 based on scram insertion times as given by the Technical Specifications. Scram data from operating plants had indicated that these scram times are quite conservative; however, a sufficient data base with supporting statistical analysis to justify the use of more realistic scram time in plant safety analyses did not exist.

As part of a continuing program to provide operating margin improvements for DAEC to enable continued full power operation, operating data was collected and the necessary statistical analysis completed. From this analysis a revised scram insertion time specification was derived which would be unlikely to be exceeded during any scram.

This report describes the scram data base and statistical analysis, identifies the proposed scram insertion time limit and presents the results of the safety analysis which defines the MCPR operating limit based on the revised scram insertion time limit.

2. SCRAM INSERTION TIME ANALYSIS

Control rod scram time data from two similar operating BWR/4's were used to derive a more realistic scram insertion time specification to be used in the DAEC safety analysis to define the operating MCPR limit. The collection of the data is described in Section 2.1.

2.1 Data Base

The DAEC data base included four full core (89 control rod drives) individual drive scram tests over a four year operating period (1 full core scram test per year). Scram times were recorded at four insertion positions (≈5%, 20%, 50% and 90% insertion) for each individual control rod drive. DAEC scram times were also available from 8 full core reactor scrams in which the scram times were recorded for approximately 20 drives per reactor scram. This provides a data base of over 500 rod scram times specifically applicable to DAEC.

Scram time data from another BWR operating plant similar to DAEC (BWR/4 plant with the same number of control rods) with an identical control rod drive design were also used in the analysis to obtain a better estimate of the scram time variation between tests. This data base included scram times from 15 scram tests conducted over a two year period. Two of the 15 scram tests were full core (89 control rod drives) individual drive scram tests. The remaining 13 scram tests were from full core reactor scram tests in which the scram times were recorded for approximately 45 drives per reactor scram. Thus, over 1150 rod scram times were used to derive a more realistic scram time to be used in the plant safety analysis.

2.2 SUMMARY OF RESULTS

The core average scram insertion time specification assumed in safety analysis to determine the MCPR operating limit for each insertion position is greater than the measured DAEC average scram insertion time plus three standard deviations for the region of greatest importance (less than 50% inserted). The proposed average scram time specification for the three fastest control rods in a 2x2 array is greater than the measured DAEC average scram insertion time plus 2.6 standard deviations for this same region. The probability of exceeding the proposed specification limits is, therefore, acceptably low (probability <1%) and is unlikely to be exceeded during any scram.

2.3 CORE AVERAGE SCRAM INSERTION TIME SPECIFICATION

a. The proposed core average scram insertion time specification for each insertion position has been selected so that it is unlikely that the specification would be exceeded. The actual calculated difference between the proposed specification and the measured average (in terms of number of standard deviations) for each insertion position is given in Figure 2-1.

b. The DAEC average scram insertion time was calculated from the four full core individual drive tests. The data from these tests are the most representative of the population average since each of the four tests included scram times for all drives in the core. The distribution of these data is depicted in Figures 2-3, 2-4, 2-5 and 2-6.

c. The standard deviation for each insertion position was calculated from the average scram insertion times of the four full core individual drive scram tests and the eight full core scrams at DAEC.

d. The standard deviations calculated from the DAEC core average data are consistent with the standard deviations experienced at the other BWR.

2.4 AVERAGE SCRAM INSERTION TIME SPECIFICATION FOR THE THREE FASTEST CONTROL RODS IN A 2x2 ARRAY

a. The proposed specification for the average scram insertion time of the three fastest control rods in a 2x2 array is greater than the DAEC measured average scram insertion times by more than 2.5 standard deviations. The lower bound of the difference between the proposed specification and the measured average (in terms of number of standard deviations) for each insertion position is given in Figure 2-2.

b. The DAEC average scram insertion time of the fastest 3 rods in a 2x2 array was assumed to be equal to the average calculated for the core average scram insertion time specification. The real average of the fastest 3 rods in a 2x2 array would be less, and therefore, this is a conservative assumption. The data for the 3 fastest rods in all 2x2 arrays is shown in Figures 2-6, 2-7, 2-8, and 2-9.

c. The standard deviations used for this part of the analysis were calculated from the DAEC measured distribution of individual drive scram times. The standard deviation for the distribution of the averages of the three fastest control rods in a 2x2 array would be less than the calculated standard deviation of scram insertion times for individual drives. A precise calculation of the standard deviation of the average of the three fastest scram insertion times in a 2x2 array is not necessary since the average of the individual drive scram insertion times plus three standard deviations is approximately equal to the proposed specification. Therefore, there is a low probability (<1%) of exceeding the proposed technical specification for the average scram time of the three fastest control rods in a 2x2 array.

d. The standard deviations calculated from the DAEC individual drive measurements are consistent with the standard deviations experienced at the other BWR.

2.5 PROPOSED TECHNICAL SPECIFICATION SCRAM INSERTION TIME REQUIREMENT

The proposed new scram insertion time specification is given in Table 2-1.

Table 2-1

PROPOSED SCRAM TIME TECHNICAL SPECIFICATION

Control Rod Position	Core Mean Insertion Time (sec)	Average of Fastest 3 out of 4 Insertion Times in any 2x2 Array (sec)
46	≤0.361	≤0.383
36	≤0.917	≤0.972
26	≤1.463	≤1.556
06	≤2.686	≤2.847

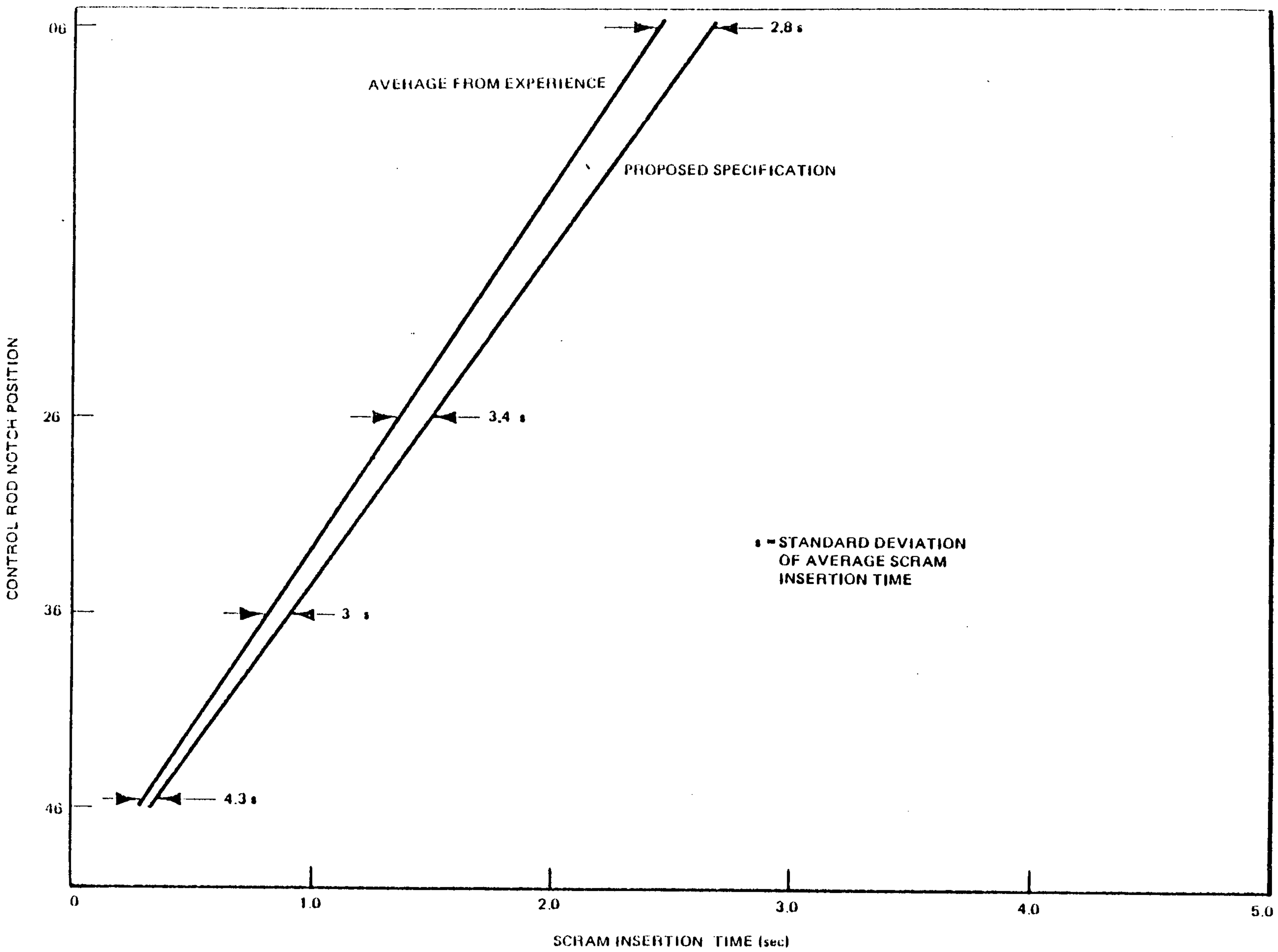

Figure 2-1. Scram Insertion Times, DAEC, Average Scram Insertion Time of All Operable Control Rods

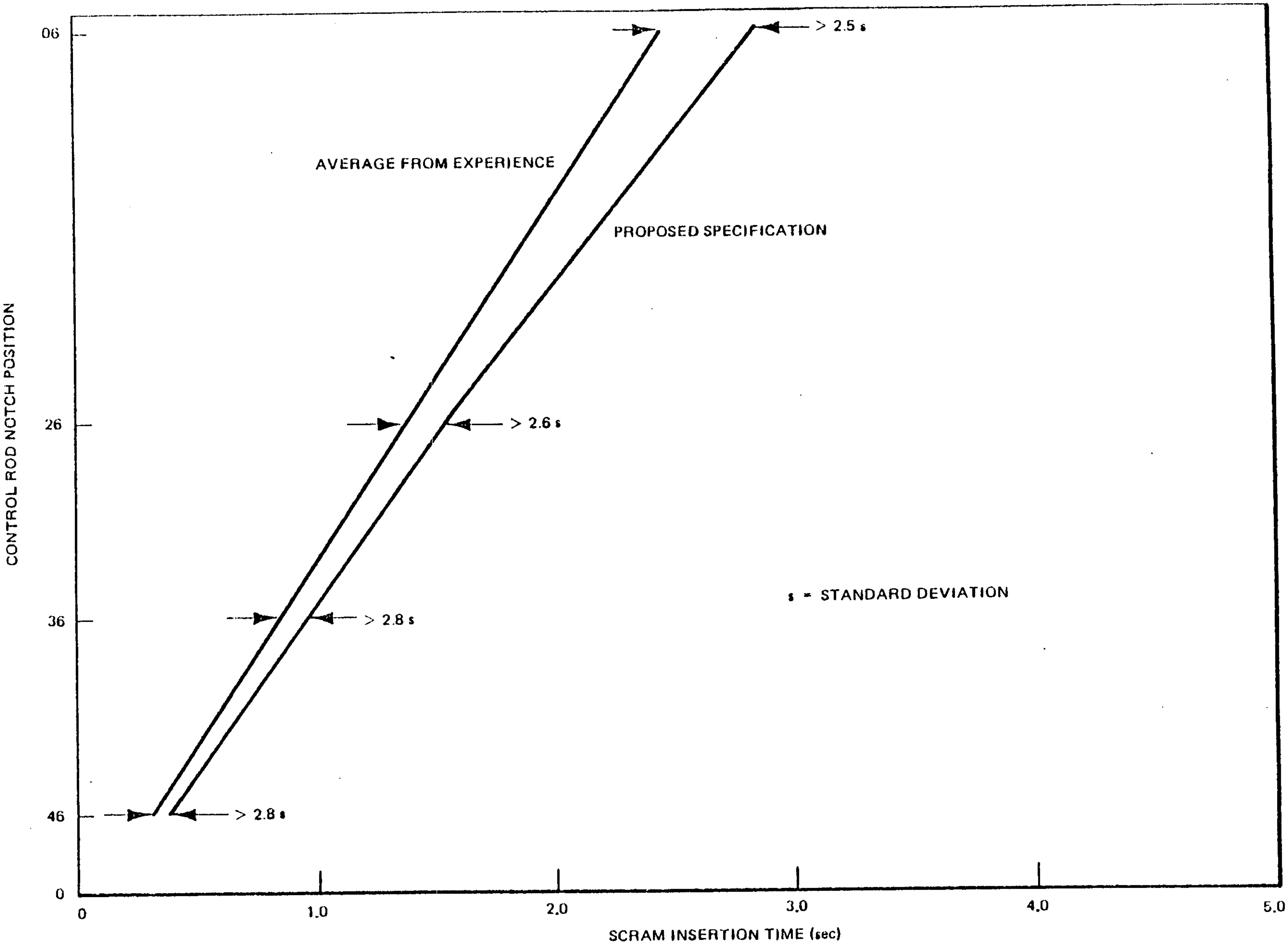

Figure 2-2. Scram Insertion Times, DAEC, Average of Fastest Three out of Four Control Rods in Any 2x2 Array

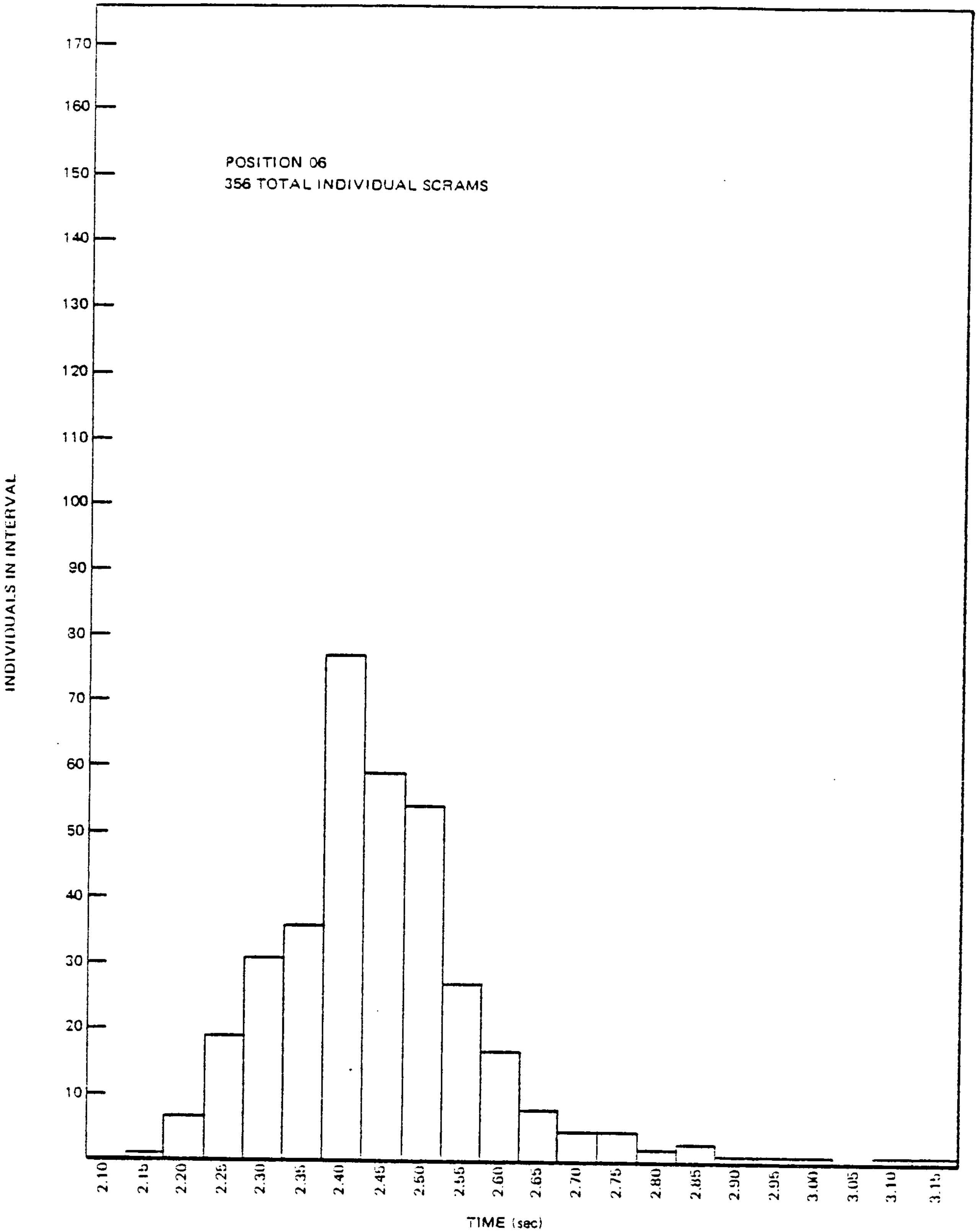

Figure 2-3. Histogram of DAEC Full Core Individual Drive Scram Insertion Time Data

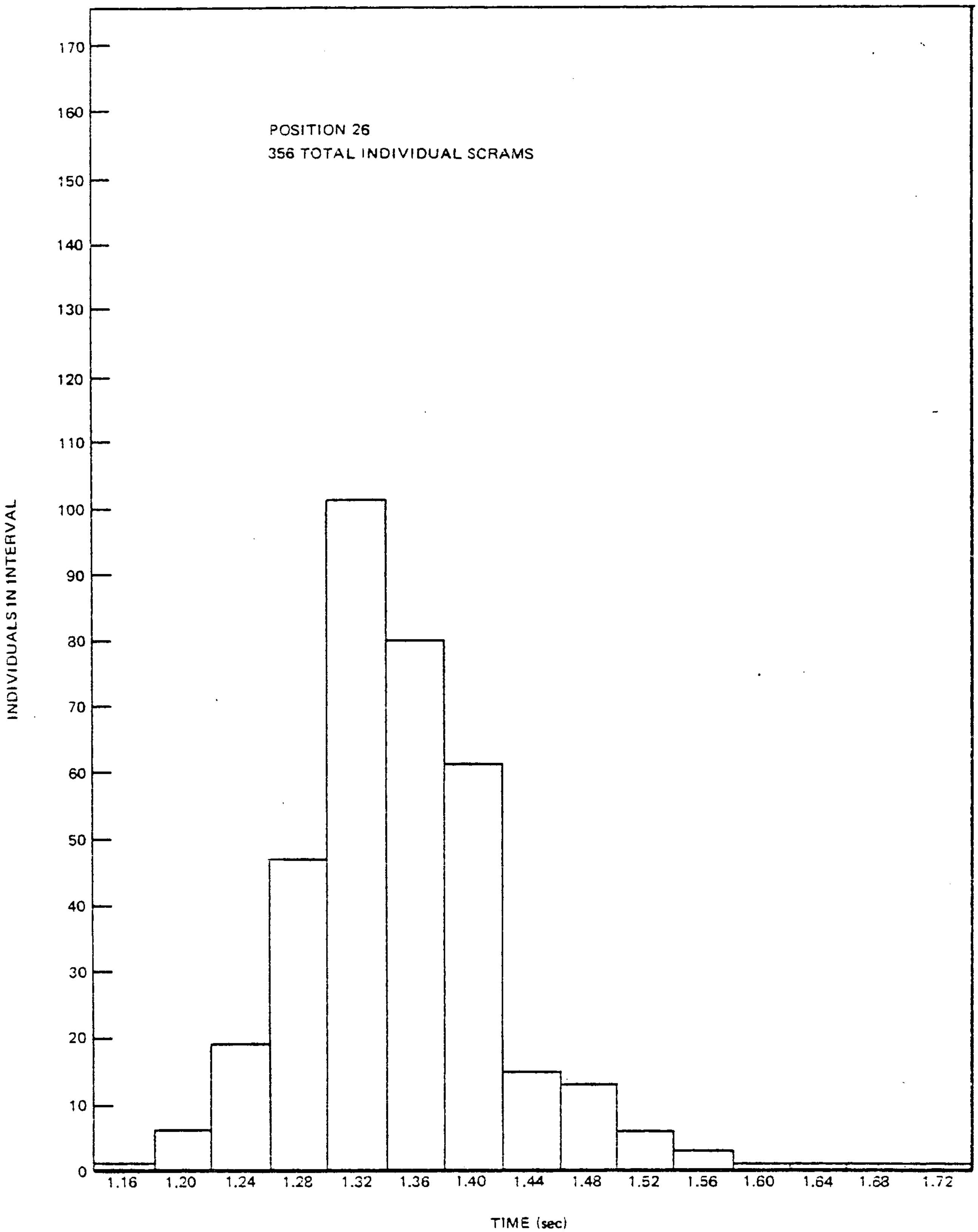

Figure 2-4. Histogram of DAEC Full Core Individual Drive Scram Insertion Time Data

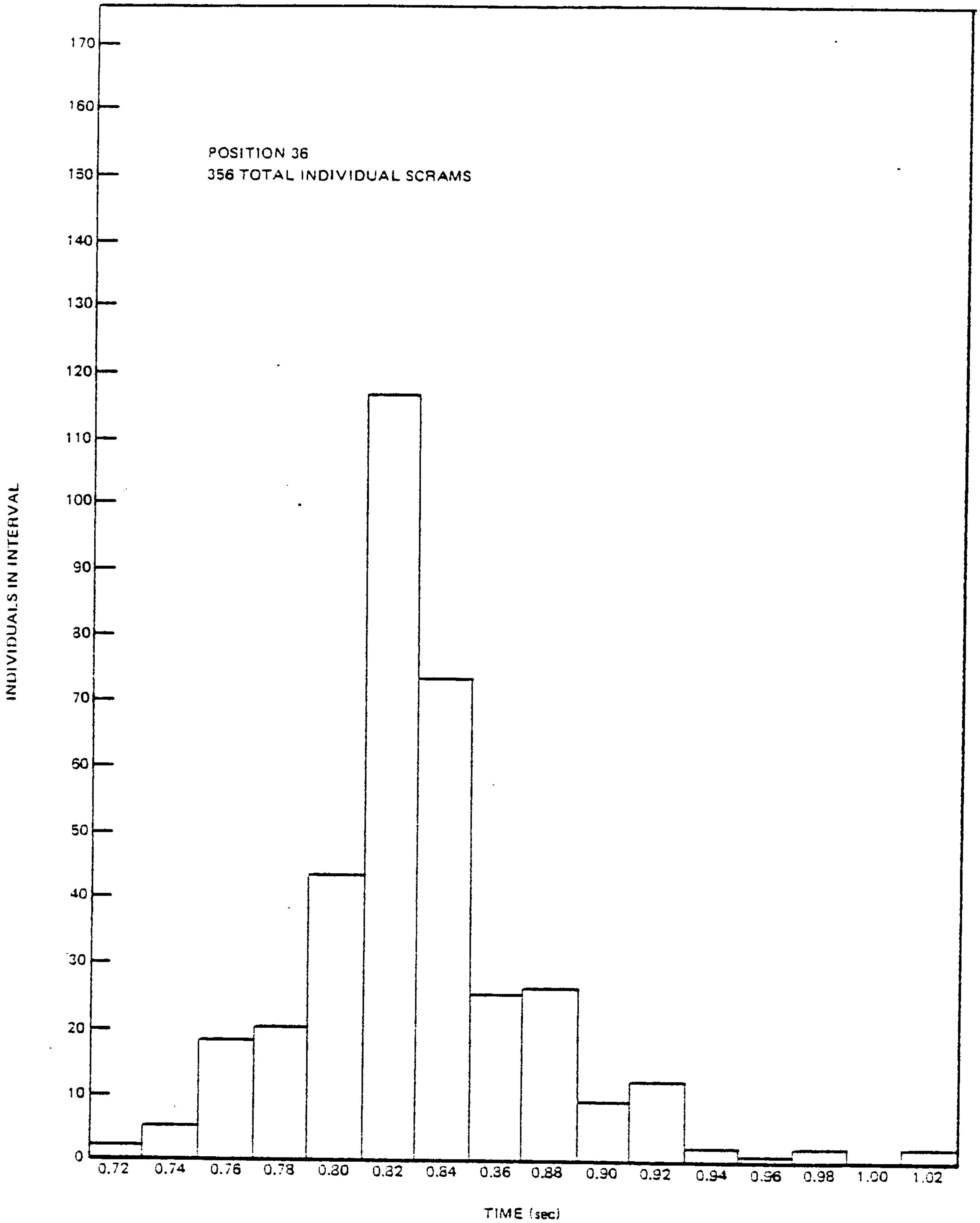

Figure 2-5. Histogram of DAEC Full Core Individual Drive Scram Insertion Time Data

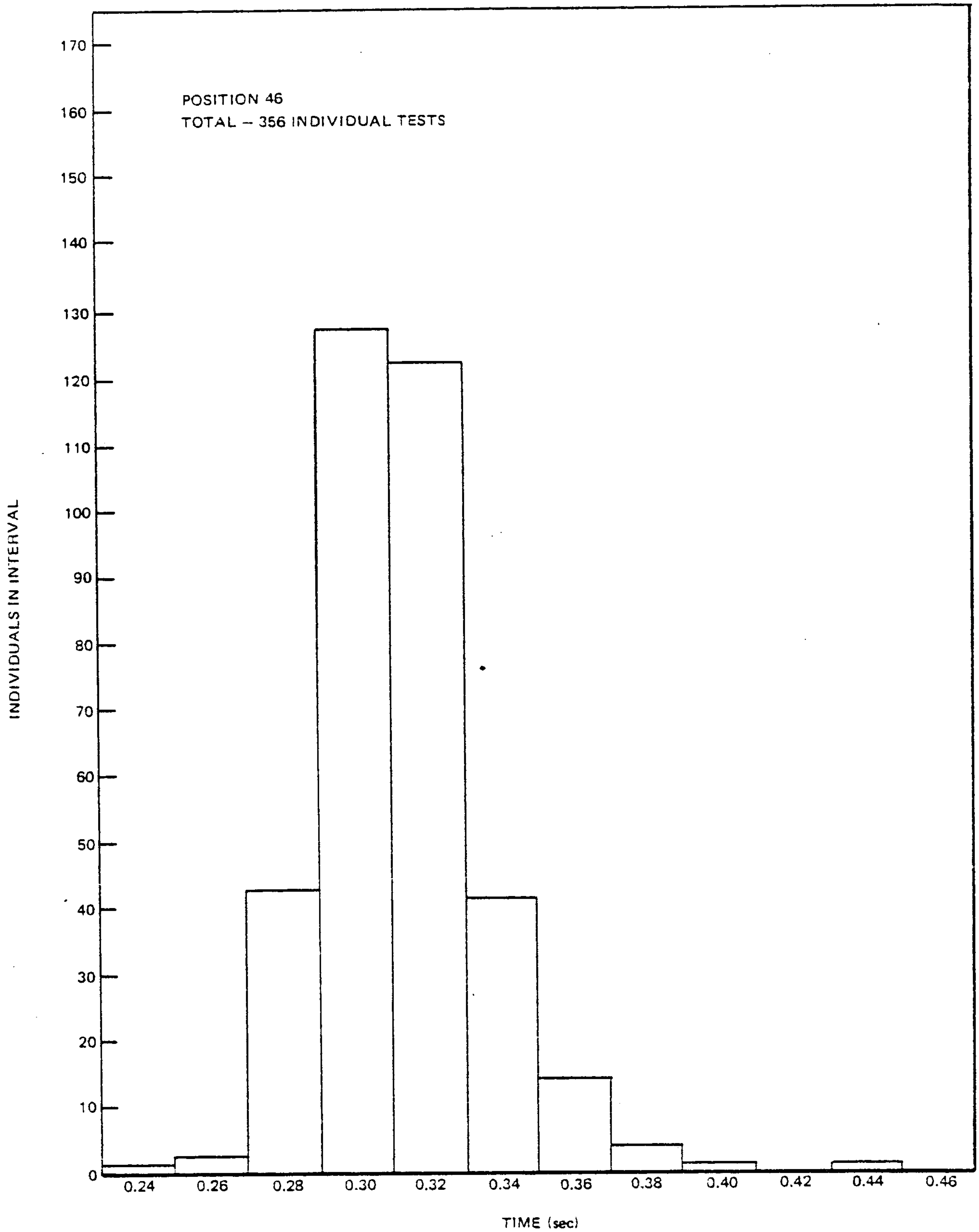

Figure 2-6. Histogram of DAEC Full Core Scram Insertion Time Data

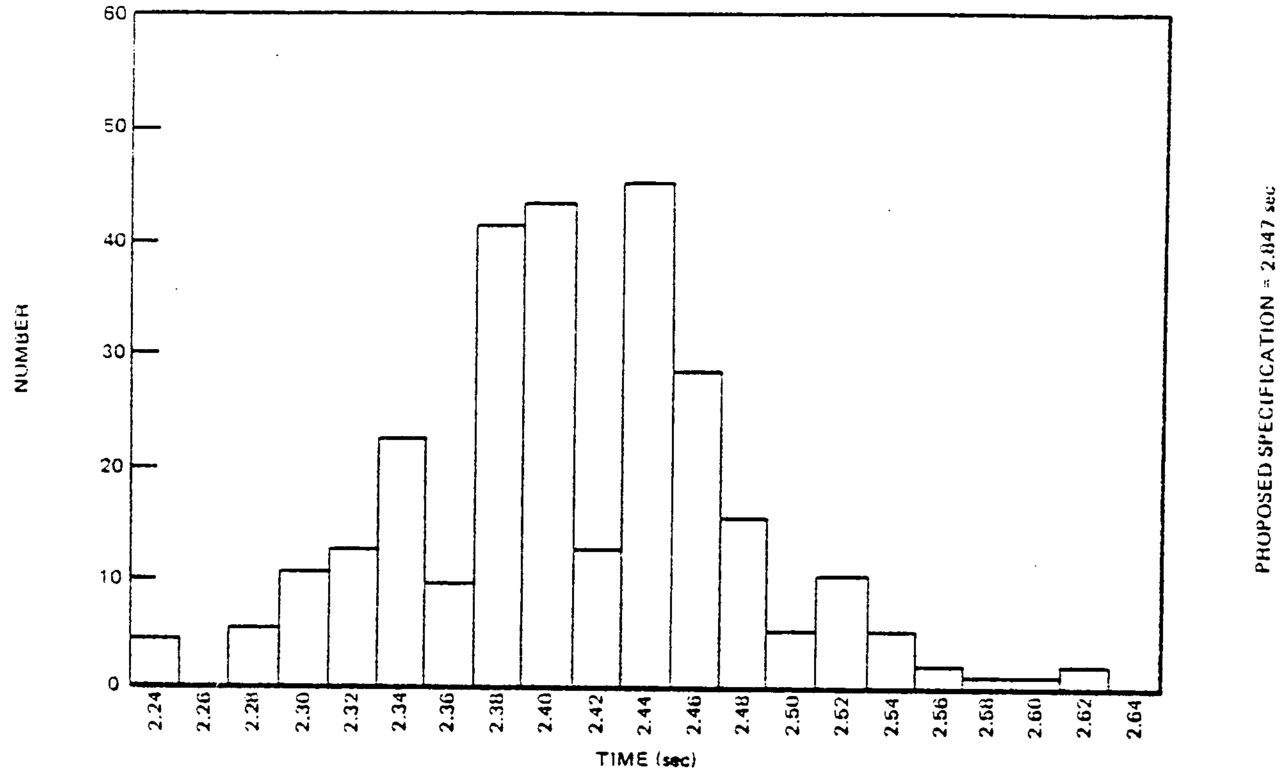

Figure 2-7. Histogram of DAEC Full Core Scram Insertion Data - Average of Fastest Three Control Rods in a 2x2 Array - Position 06

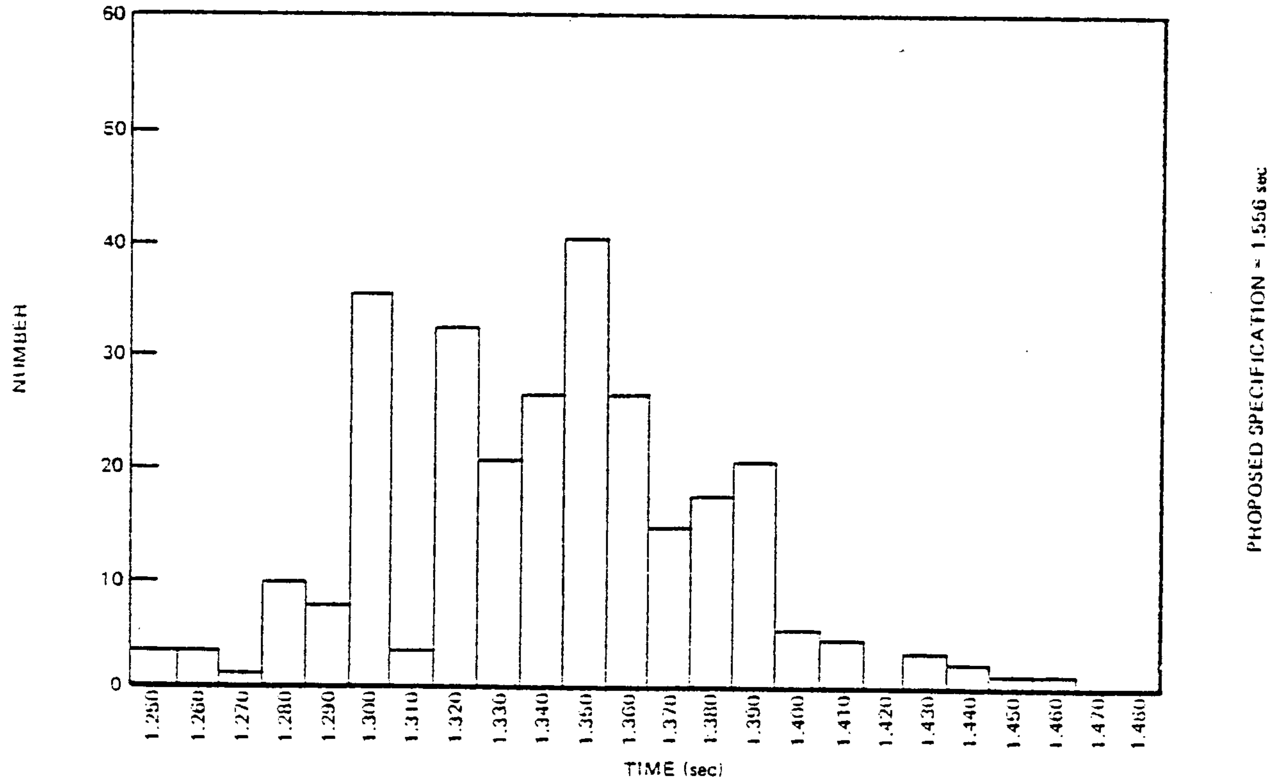

Figure 2-8. Histogram of DAEC Full Core Scram Insertion Data - Fastest Three Control Rods in a 2x2 Array - Position 26

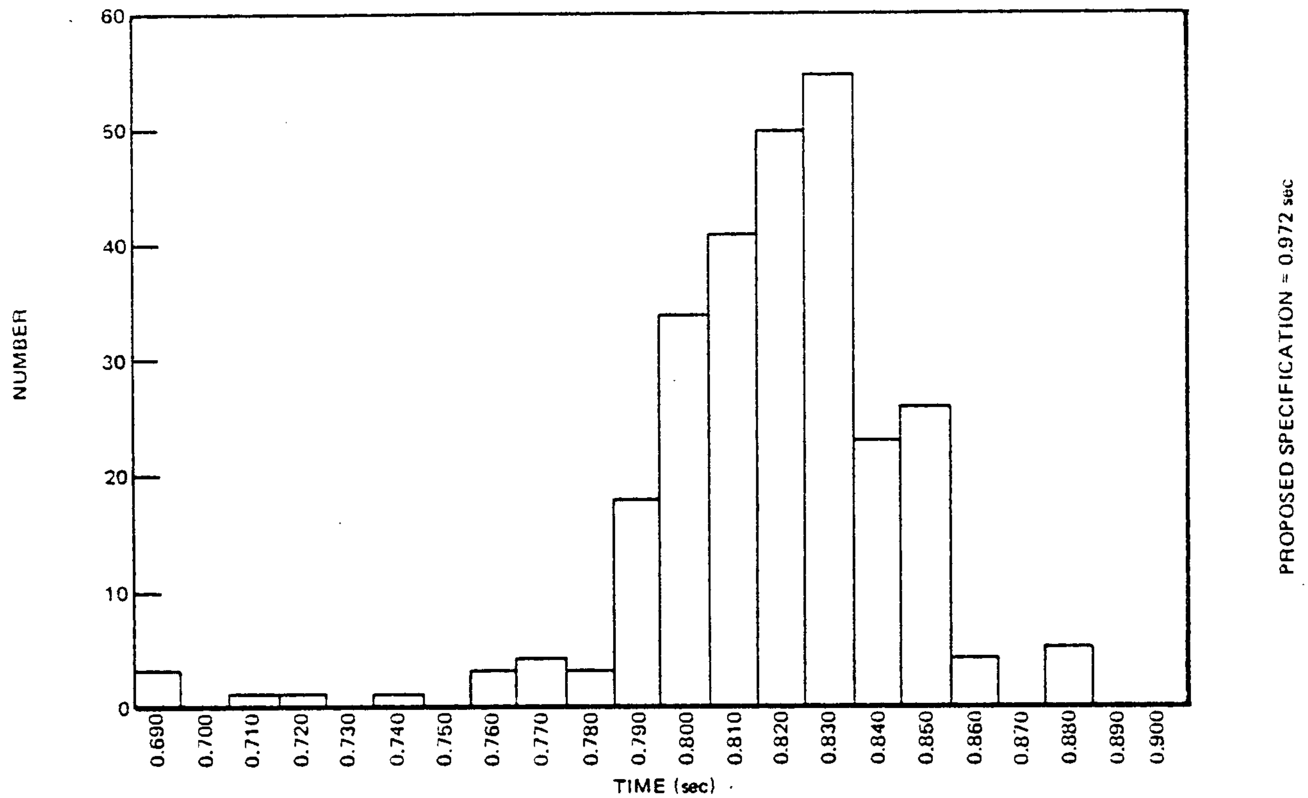

Figure 2-9. Histogram of DAEC Full Core Scram Insertion Data - Average of Fastest Three Control Rods in a 2x2 Array - Position 36

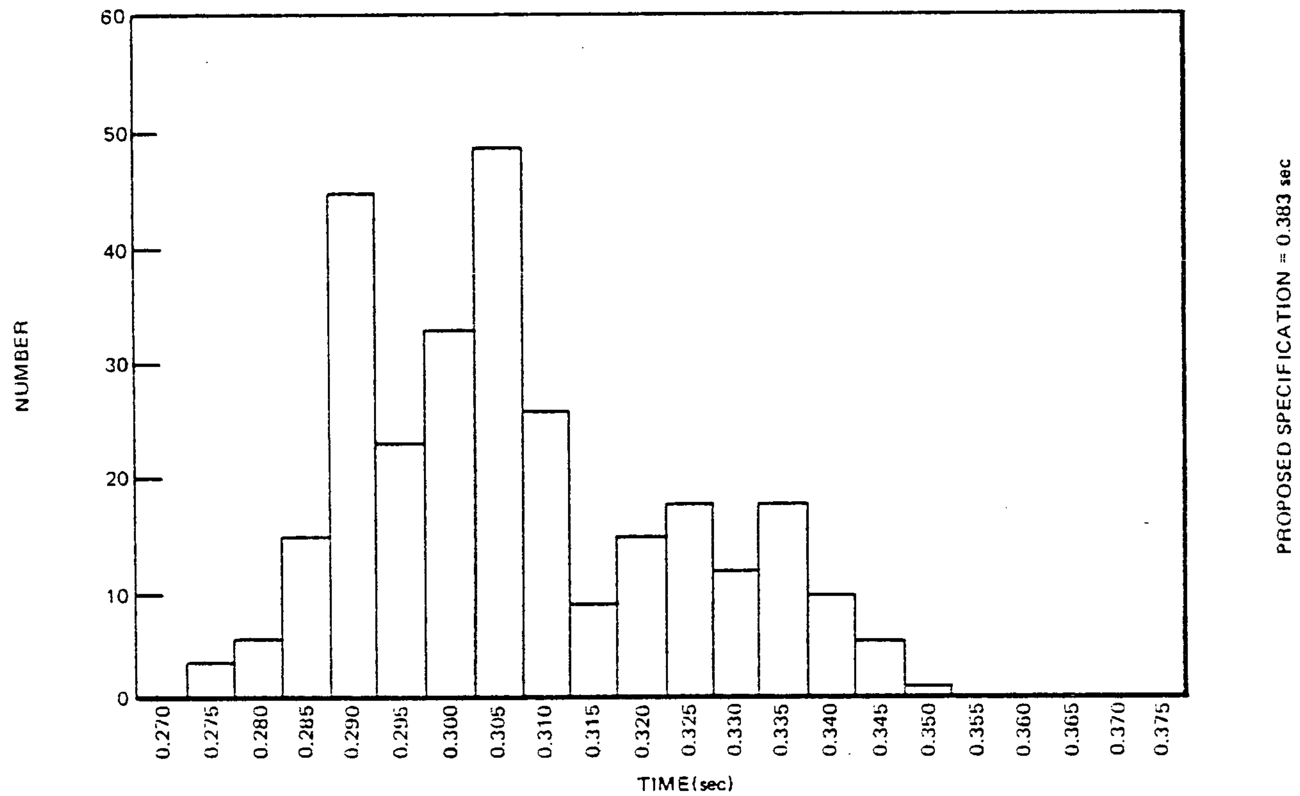

Figure 2-10. Histogram of DAEC Full Core Scram Insertion Data - Average of Fastest Three Control Rods in a 2x2 Array - Position 46

3. 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 2. The analysis applicable to Duane Arnold Cycle 4 is given below and in Reference 1.

3.1 STATISTICAL ANALYSIS

The statistical analysis is described in Reference 1.

3.1.1 Fuel Cladding Integrity Safety Limit

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

3.1.2 Basis for Statistical Analyses

The basis for the statistical analysis is described in Reference 1.

3.2 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 3-1.

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.

3.2.1 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 8x8 fuel types and 1.22 for 7x7 for EOC-2000 MWd/t to EOC. The limit for BOC to EOC-2000 MWd/t and its basis is given in References 1, 3 and 4.

3.3 TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

The magnitude of values used as initial input conditions for the transient analysis is shown in Table 3-2. Those transients which are not sensitive to scram reactivity insertion rate (RWE and LOFWH) were not re-analyzed. Initial condtions for these transients are given in Reference 1.

Table 3-1

SUMMARY OF RESULTS LIMITING ABNORMAL OPERATIONAL ΔCPR TRANSIENTS

	EOC4	
Event	7x7	8x8
Rod Withdrawal Error* (RBM @ 105%)	0.16	0.11
Loss of Feedwater Heater** (100°F)	0.14	0.15
Turbine Trip w/o Bypass	0.09	0.13
Load Rejection w/o Bypass	0.09	0.14
Feedwater Controller Failure	0.06	0.09
Turbine Trip with Bypass	0.01	0.03
Load Rejection with Bypass	0.01	0.03

Table 3-2

GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

	7x7	8x8
Peaking factors (local and radial)	1.24, 1.29	1.22, 1.45
R-Factor	1.100	1.098
Bundle Power, MWt	5.474	6.132
Bundle Flow, 10^3 lb/hr	124.3	111.7
Initial MCPR	1.20	1.20
Core Power, MWt	1593.0	
Core Flow, Mlb/hr	49.0	
Reactor Pressure, psia	1035.0	
Inlet Enthalpy, Btu/lb	526.3	
Nonfuel Power Fraction	0.04	
Axial Peaking Factor	1.40	

*Results from Reference 1.
**Results from Reference 4.

4. ABNORMAL OPERATING TRANSIENTS

4.1 TRANSIENTS AND CORE DYNAMICS

4.1.1 Analysis Basis

This subsection contains the analyses of the most limiting abnormal operational transients for Duane Arnold Energy Center Cycle 4 using the proposed new scram insertion time specification. The control rod drive specifications are given in Figure 4-6.

4.1.2 Input Data and Operating Conditions

The input data and operating conditions are shown in Table 4-1 and represent the nominal basis for these analyses. Each transient is considered at these conditions unless otherwise specified.

4.1.3 Transient Summary

A summary of the transients analyzed and their consequences is provided in Table 4-2.

4.2 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.

4.2.1 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 4-1.

The parameters of concern are the peak vessel pressure margin to the ASME Pressure Vessel Code Limit and the peak average surface heat flux correlated to MCPR. These are given in Table 4-2.

4.2.2 Loss of a Feedwater Heater

The loss of a feedwater heater was analyzed in Reference 1. This analysis is conservative for the new scram insertion time.

4.2.3 Rod Withdrawal Error

The rod withdrawal error was analyzed in Reference 1. The rod withdrawal error analysis is unchanged by the control rod scram insertion time.

4.2.4 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 sec.

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 sec.

d. The pressure continues to rise until the pressure relief setpoints 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 4-2.

The parameters of concern are the peak vessel pressure margin to the ASME Pressure Vessel Code Limit and the peak average surface heat flux correlated to MCPR. These are given in Table 4-2.

4.2.5 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 in a feedwater controller failure is 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 4-3 shows the results of this transient.

The parameters of concern are the peak vessel pressure margin to the ASME Pressure Vessel Code Limit and the peak average surface heat flux correlated to MCPR. These are given in Table 4-2.

4.2.6 Generator Load Rejection 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 control valves. The pressure increase causes a significant void reduction which yields a pronounced positive void reactivity effect. The net reactivity is sharply positive and causes a rapid increase in neutron flux until the net reactivity is forced negative by scram initiated from closure of the turbine control valves and by a void increase after the safety-relief valves have automatically opened on high pressure. Figure 4-4 illustrates this transient.

The parameters of concern are the peak vessel pressure margin to the ASME Pressure Vessel Code Limit and the peak average surface heat flux correlated to MCPR. These are given in Table 4-2.

4.2.7 Generator Load Rejection with Operable Bypass

Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine-generator rotor. The closing causes a sudden reduction in steam flow, which results in a nuclear system pressure increase and scram. Figure 4-5 shows the calculated results of this transient.

The parameters of concern are the peak vessel pressure margin to the ASME Pressure Vessel Code Limit and the peak average surface heat flux correlated to MCPR. These are given in Table 4-2.

Table 4-1

TRANSIENT INPUT PARAMETERS

Thermal Power	(MWt)	1657	104% Rated
Steam Flow	(lb/hr)	7.18×10^6	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	
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	

		Analysis	Nominal
Dynamic Void Coefficient	(-c/%Rg)	11.69	9.35
Doppler Coefficient	(-c/°F)	0.2187	0.2302
Average Fuel Temperature	(°F)	1359	1359
Scram Reactivity Curve		Fig. 4-6	Fig. 4-6
Scram Worth	(-$)	31.58	39.48

Table 4-2

TRANSIENT DATA SUMMARY

Transient	Power (%)	Flow (%)	ϕ (% reference)	Q/A (% reference)	Psl (psig)	Pv+ (psig)
Turbine Trip without Bypass	104	100	253	105	1178	1215
Load Rejection without Bypass	104	100	277	106	1179	1217
Loss of Feedwater Heater*	104	100	121	119	1023	1071
Feedwater Controller Failure	104	100	163	105	1140	1184
Turbine Trip with Bypass	104	100	144	100	1137	1181
Load Rejection wtih Bypass	104	100	152	100	1137	1181

*Results of analysis from Reference 1.

+Without ATWS recirc pump trip, which is most conservative for vessel pressure. All other data are with ATWS pump trip.

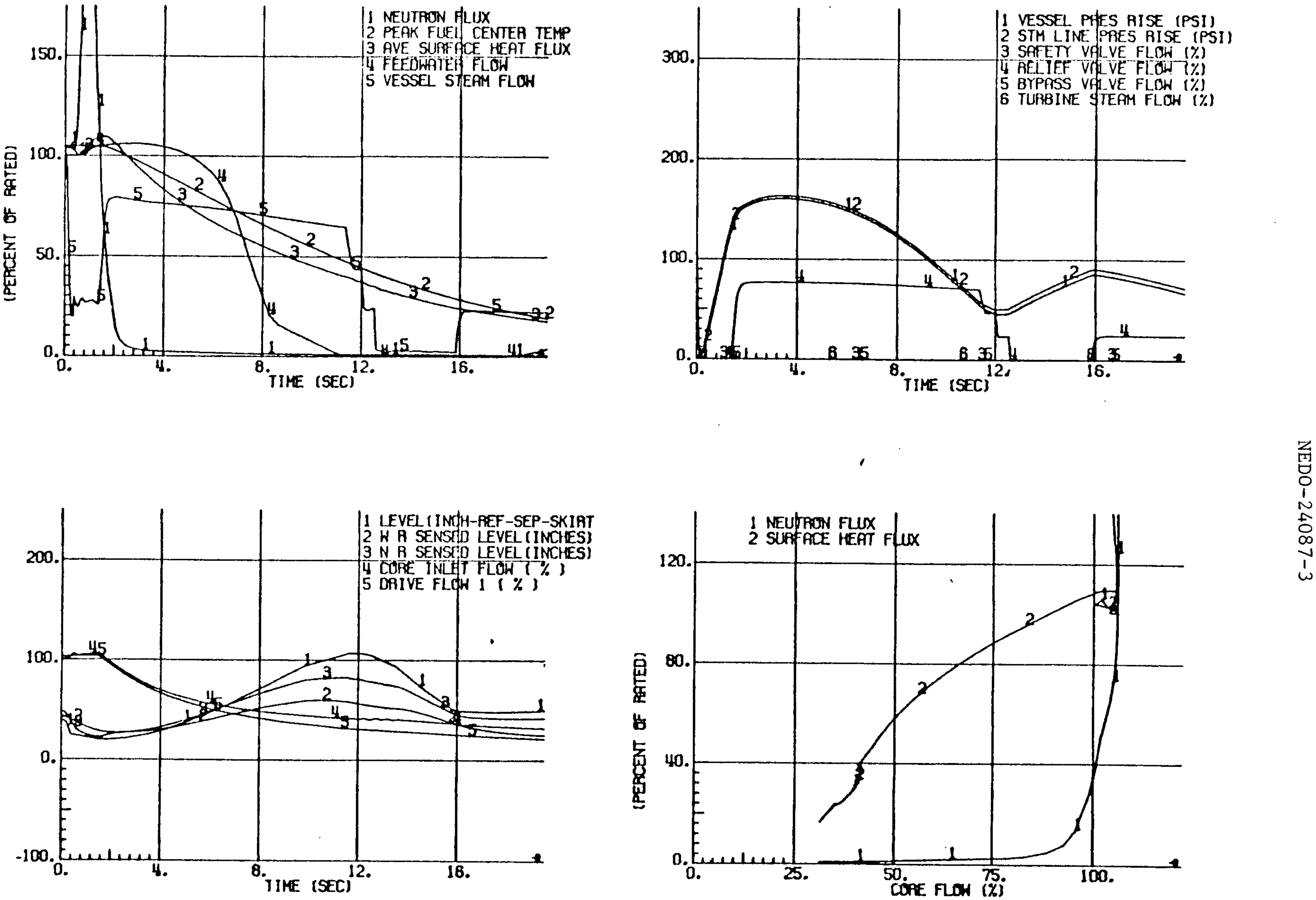

Figure 4-1. Duane Arnold EOC4 Turbine Trip without Bypass, Measured Scram Insertion Time

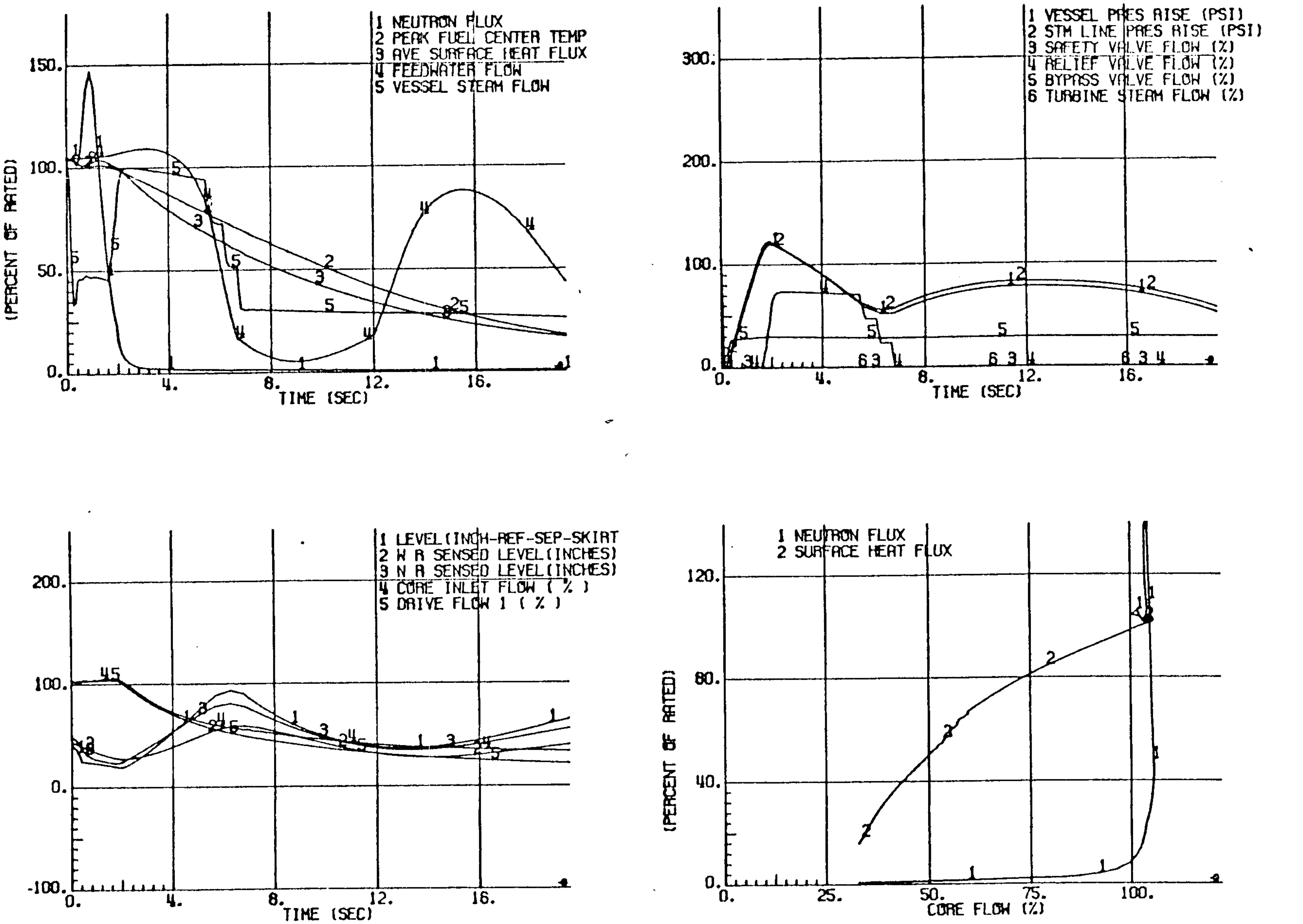

Figure 4-2. Duane Arnold EOC4 Turbine Trip with Bypass, Measured Scram Insertion Time

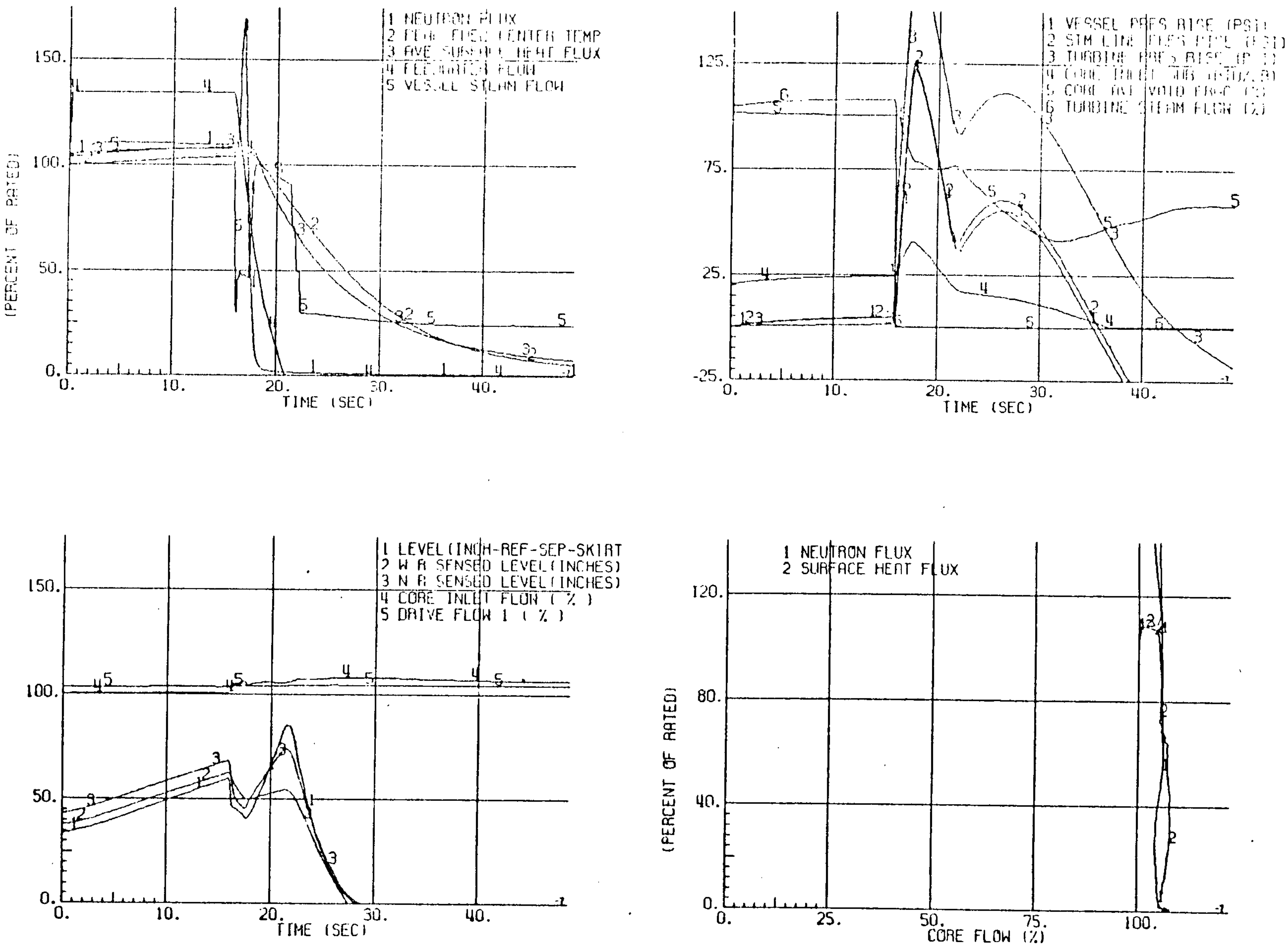

Figure 4-3. Duane Arnold EOC4 Feedwater Controller Failure, Maximum Demand with High Level Turbine Trip, Measured Scram Insertion Time

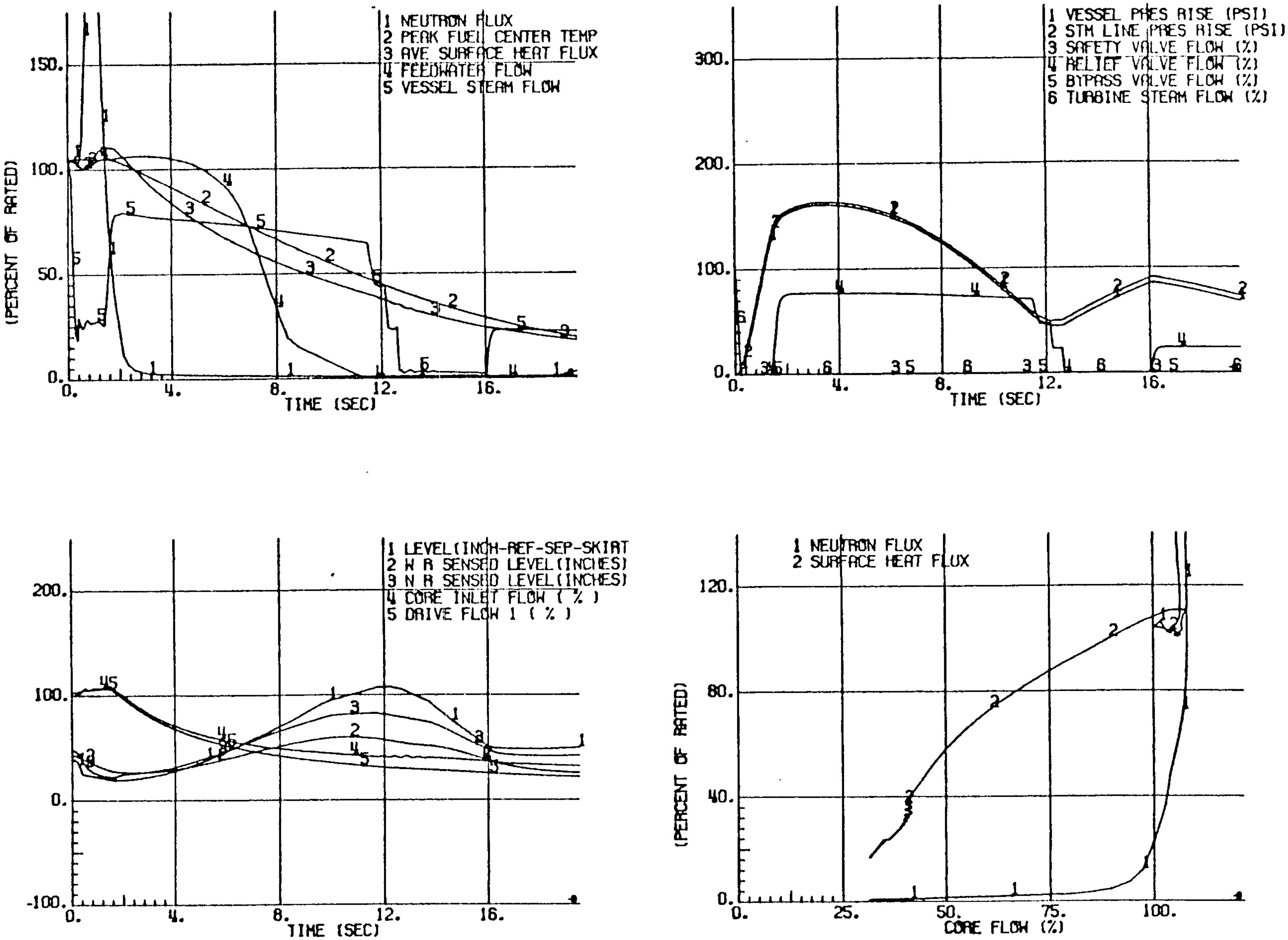

Figure 4-4. Duane Arnold EOC4 Load Rejection without Bypass, Measured Scram Insertion Time

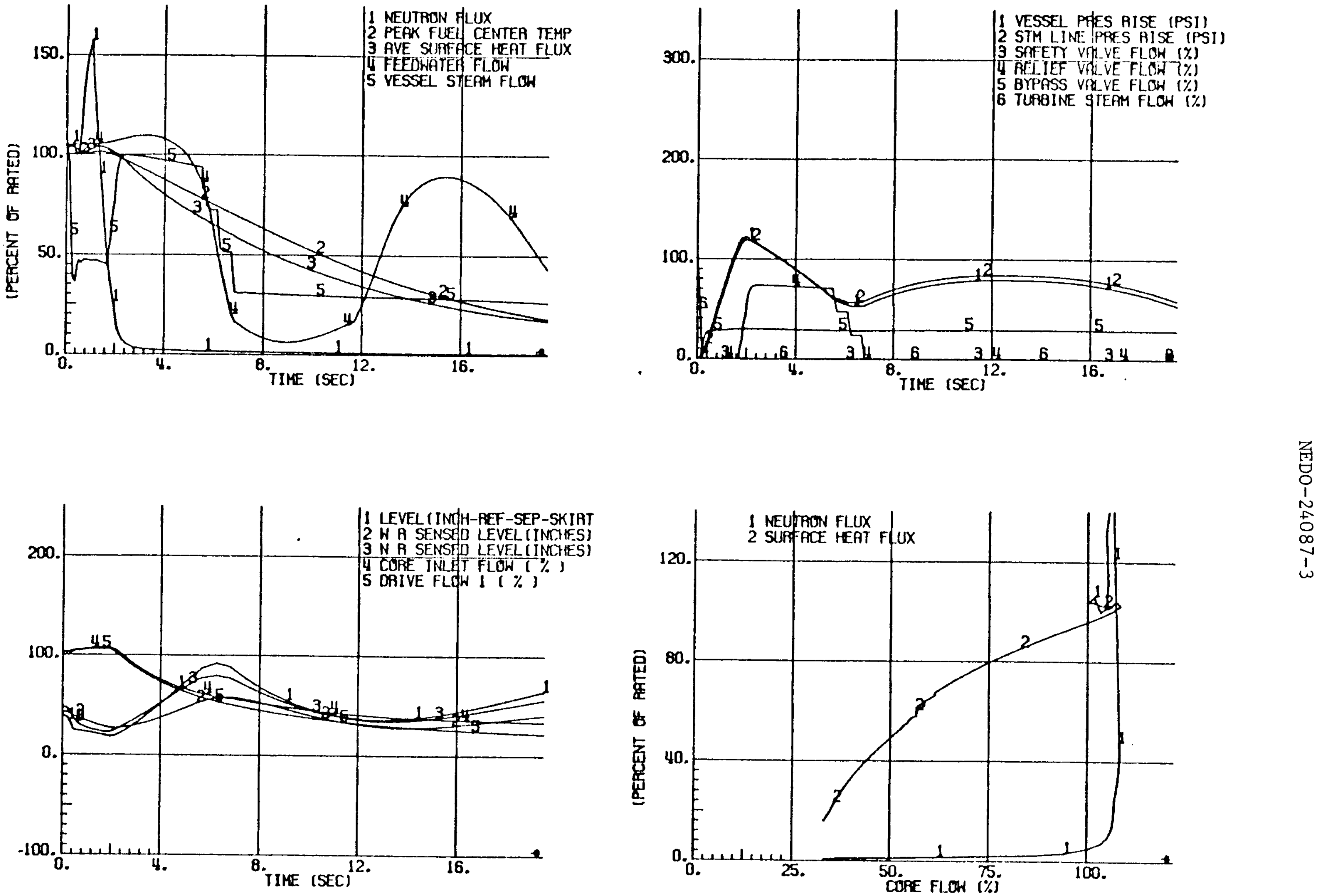

Figure 4-5. Duane Arnold EOC4 Load Rejection with Bypass, Measured Scram Insertion Time

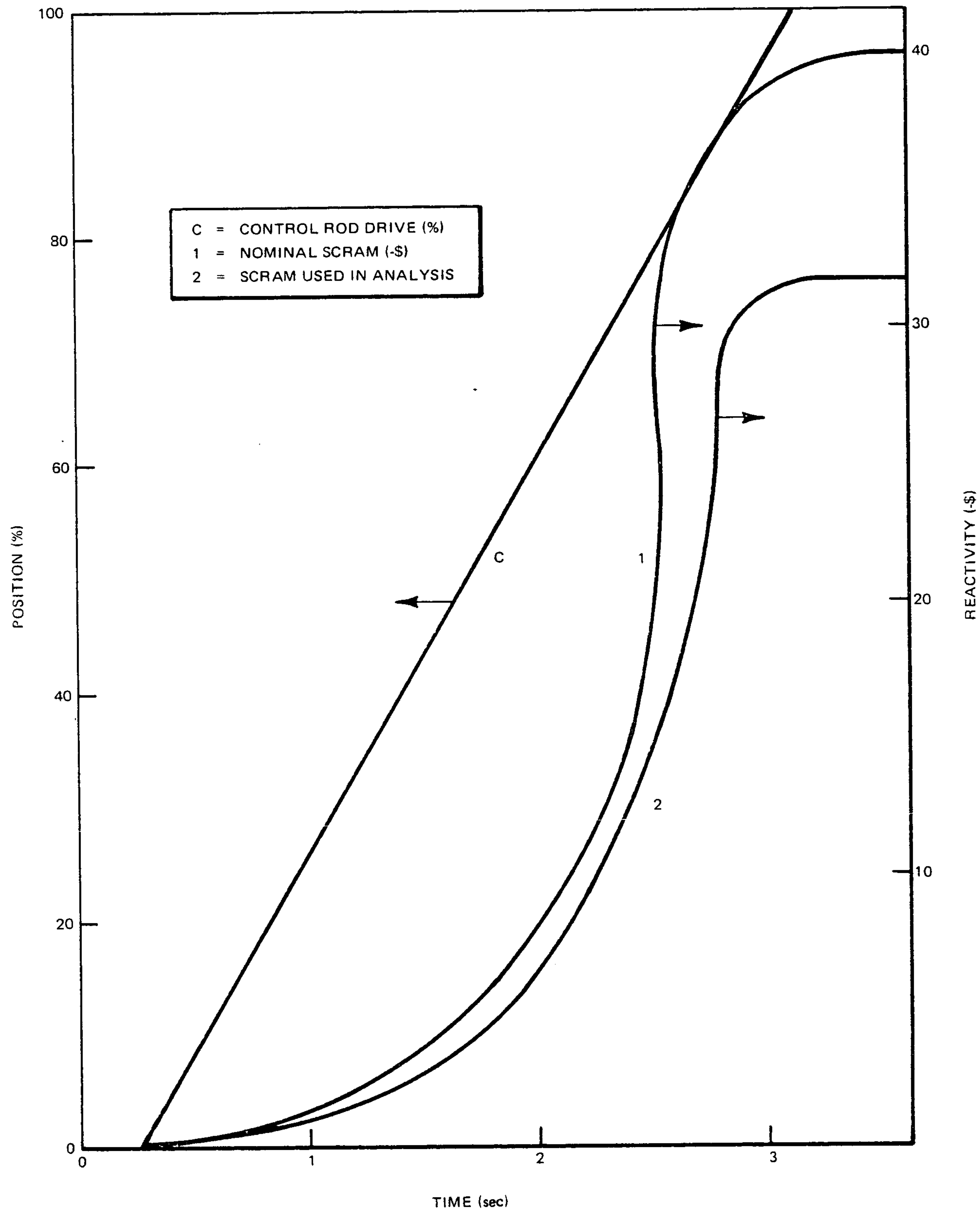

Figure 4-6. Control Rod Drive Specification and Scram Reactivity, DAEC EOC4, Measured Scram Insertion Time

5. TECHNICAL SPECIFICATION CHANGES

The following technical specification changes will be required.

a. MCPR Operating Limit

b. Scram Insertion Times

6. REFERENCES

1. "General Electric Boiling Water Reactor, Reload-3 Licensing Amendment for Duane Arnold Energy Center," December 1977 (NEDO-24087).

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

3. "General Electric Boiling Water Reactor Reload-3 (Cycle 4) Licensing Amendment for Duane Arnold Energy Center Supplement 2: Revised Fuel Loading Accident Analysis," June 1977 (NEDO-24087-2).

4. "General Electric Boiling Water Reactor Reload-3 (Cycle 4) Licensing Amendment for Duane Arnold Energy Center Supplement 5: Revised GETAB Operating Limits for Loss of Feedwater Heating," June 1978 (NEDO-24087-5).

APPENDIX A
ODYN RESULTS

For the past several months, General Electric, with the approval of the Nuclear Regulatory Commission in cooperation with BWR Owners and EPRI, has been engaged in a program of confirmation transient testing which has resulted in the development and qualification of an improved transient model. A description of the improved transient computer model (ODYN), its qualification and its general licensing application have been transmitted to the U.S. Nuclear Regulatory Commission in References A-1 through A-4.

At the staff's request, ODYN analyses of the limiting fast pressurization transients are being supplied in this appendix. Transients analyzed with ODYN in support of this submittal are the Load Rejection without Bypass (LR w/o BP) and the Feedwater Controller Failure (FWCF). For different transients under different conditions, the ΔCPR calculated using ODYN may be larger or smaller than that calculated using REDY. Table A-1 presents the results of the ODYN analysis of the Load Rejection without Bypass and the Feedwater Controller Failure at the end of cycle 4. The analyses presented in this appendix differ from the standard licensing calculational procedure in that the assumed initial MCPR for each transient is equal to the safety limit CPR plus the ΔCPR of that transient. These transient-dependent initial CPR's are given in Table A-1. Figures A-1a, b and c depict the Load Rejection transient. Plots depicting the Feedwater Controller Failure Transient are not available. Additional ODYN analyses for DAEC cycle 4 may be found in References A-5 and A-6.

Table A-1

DAEC CYCLE 4 TRANSIENT SUMMARY
ODYN ANALYSES WITH MEASURED SCRAM TIMES

	Power Power (%)	Flow (%)	Exposure	$\hat{\phi}$ (% initial)	$\hat{Q}/A$ (% initial)	P_{sl} (psia)	$\hat{P}_v$ (psig)	8x8 Initial CPR	8x8 ΔCPR	7x7 Initial CPR	7x7 ΔCPR
Load Rejection without Bypass	104	100	EOC4	503	115	1200	1229	1.26	0.20	1.21	0.15
Feedwater Controller Failure	104	100	EOC4	422	114	1152	1196	1.23	0.17	*	*

*Analysis not performed.

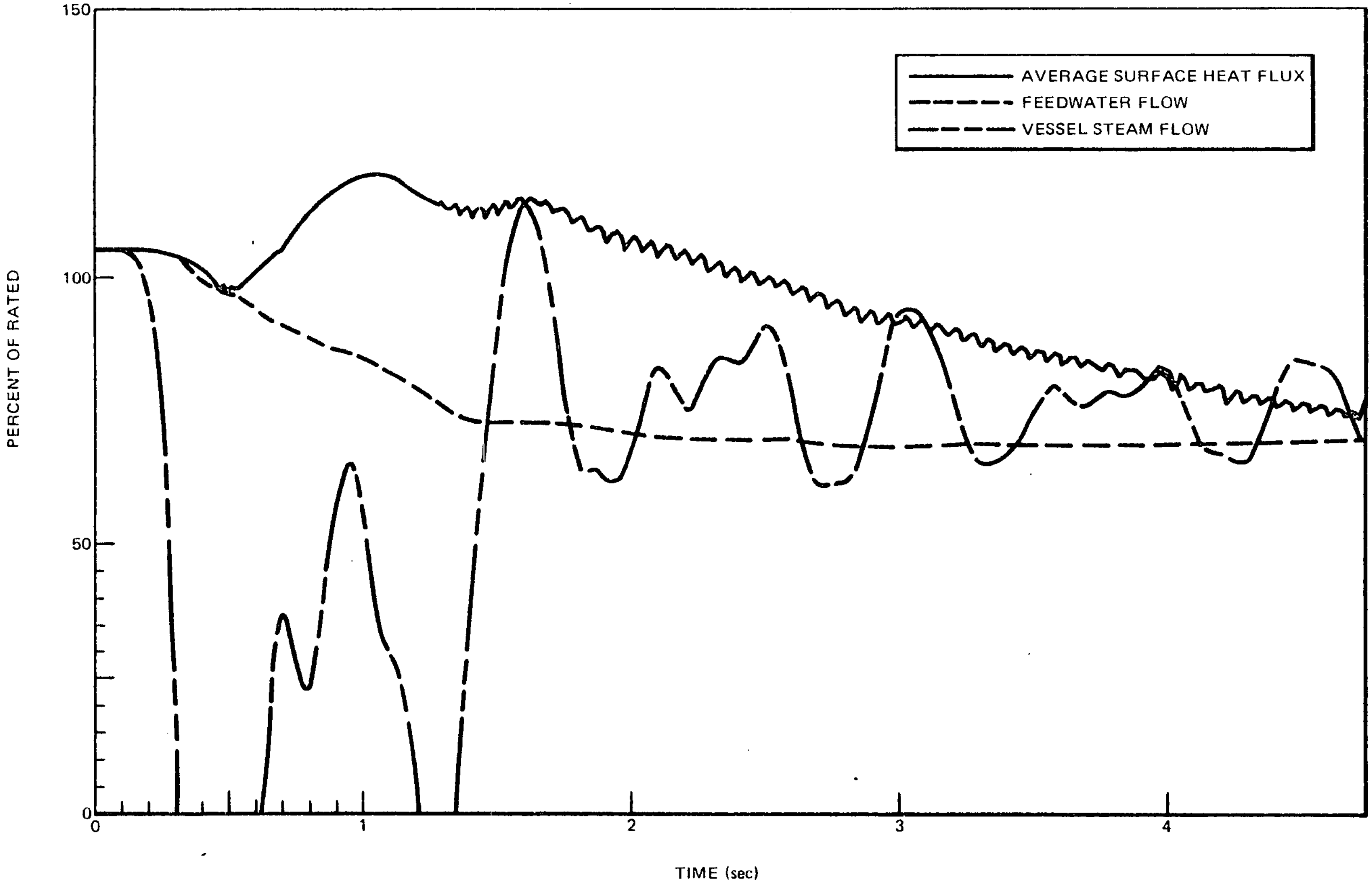

Figure A-1a. Load Rejection w/o Bypass (ODYN), Duane Arnold EOC4 Measured Scram

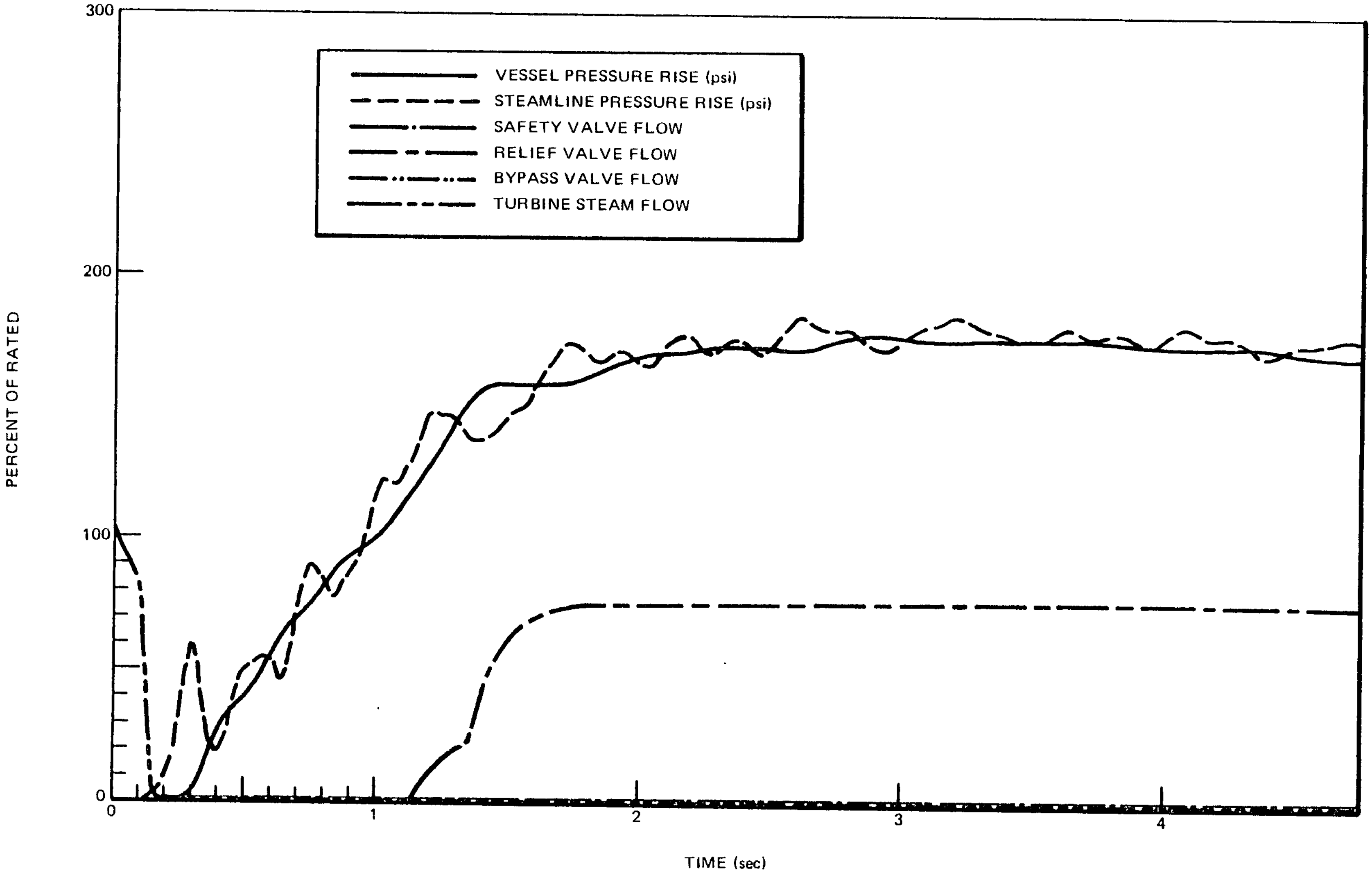

Figure A-1b. Load Rejection w/o Bypass (ODYN), Duane Arnold EOC4 Measured Scram

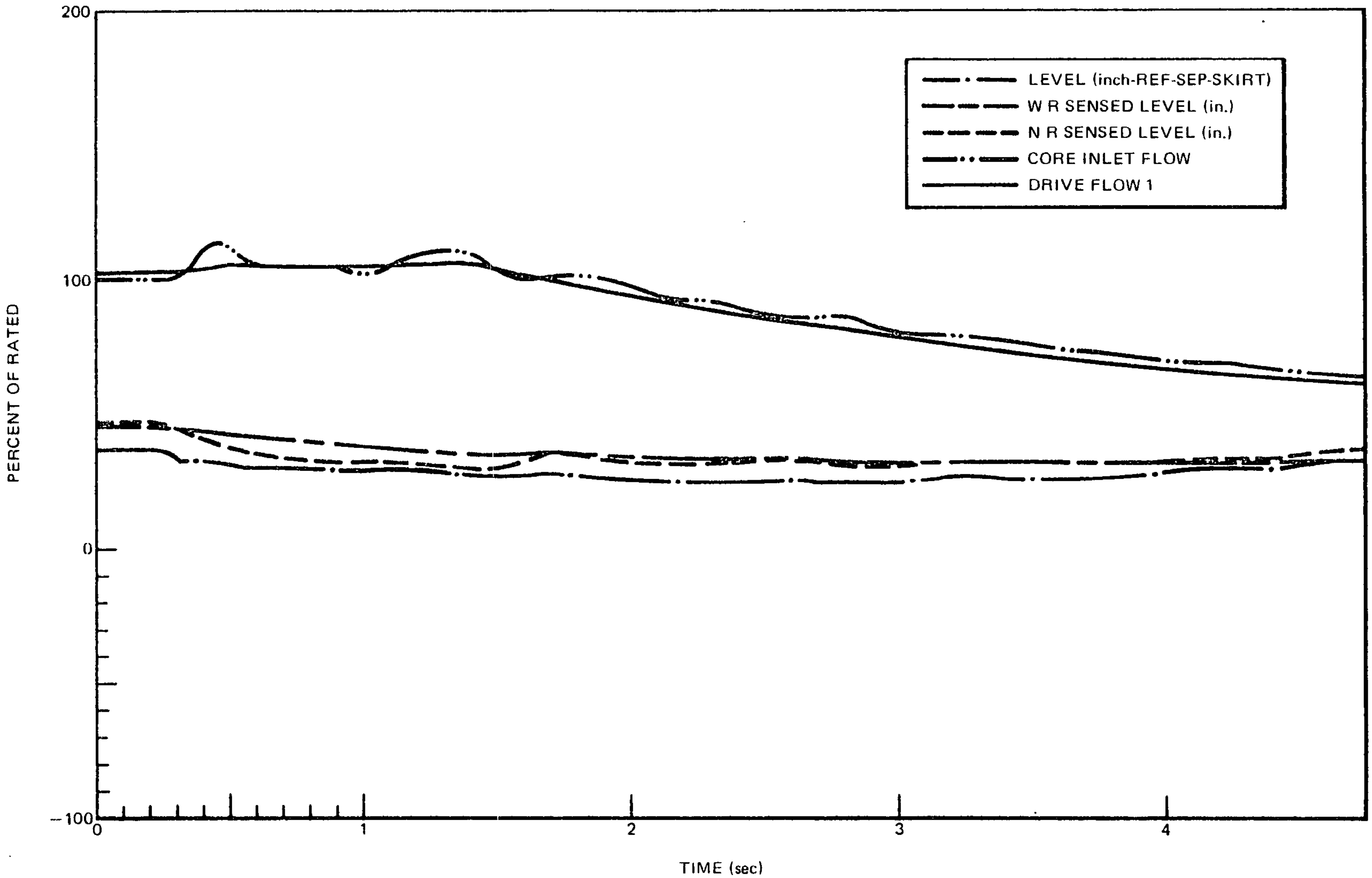

Figure A-1c. Load Rejection w/o Bypass (ODYN), Duane Arnold EOC4 Measured Scram

REFERENCES

A-1. Letter MFN 462-77, E. D. Fuller to D. F. Ross, "Transmittal of ODYN Computer Model Description," dated December 2, 1977.

A-2. Letter MFN 058-78, E. D. Fuller to D. F. Ross, "General Electric Proposal for Licensing Basis Criteria," dated February 7, 1978.

A-3. Letter MFN 014-78, E. D. Fuller to D. F. Ross, "Transmittal of Draft ODYN Qualification Report," dated January 13, 1978.

A-4. Letter MFN 136-78, E. D. Fuller to D. F. Ross, "Application Submittal for ODYN Transient Model," dated March 31, 1978.

A-5. "General Electric Boiling Water Reactor Reload-3 (Cycle 4) Licensing Amendment for Duane Arnold Energy Center, Supplement 1: Recirculation Pump Trip, Appendix A: ODYN Analyses," to be issued June 1978 (NEDO-24087-1A).

A-6. "General Electric Boiling Water Reactor Reload-3 (Cycle 4) Licensing Amendment for Duane Arnold Energy Center, Supplement 4: Safety Analysis for Reclassified Events," to be issued June 1978 (NEDO-24087-4).

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SUMMARY

This report describes the data base and statistical analysis, identifies the proposed scram insertion time limit and presents the results of the safety analysis which defines the MCPR operating limit based on the revised scram insertion time limit.

By cutting out this rectangle and folding in half, the above information can be fitted into a standard card file.

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