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IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office
CEDAR RAPIDS, IOWA

December 14, 1977
IE-77-2255

LEE LIU
VICE PRESIDENT – ENGINEERING

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-102 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. Evaluation of actual scram insertion times indicates that sufficient margins exist to redefine the limiting scram curve used in the DAEC safety analysis. Use of the enclosed proposed Technical Specifications would essentially delete the end-of-cycle derate. Accordingly, approval of these limits would result in a savings to the public of approximately $1,000,000 for the remainder of the cycle.

This change would 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

Lee Liu
Vice President, Engineering

773530054

Subscribed and Sworn before me on this 15th day of December

Marjorie E. McDonald

Notary Public in and for the State of Iowa

Marjorie E. McDonald
My Commission Expires
September 30, 1979

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

PROPOSED CHANGE RTS-102 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

Specification 3.3.C, Scram Insertion Times, provides average scram insertion times for various rod positions and supporting bases.

Table 3.12-2 provides MCPR limits for 7 x 7 and 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:

Delete pages 3.3-6, 3.3-17 through 3.3-22, and 3.12-9a and replace with the attached pages.

III. Justification for Proposed Change

This change is proposed in order to provide operating margin improvements for DAEC. The safety analysis for these proposed changes is included in NEDO-24075, 77NED354, Class I, November 1977, "Duane Arnold Energy Center Cycle 3 Safety Analysis for Application of Measured Scram Insertion Times".

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

1. Two specifications for scram insertion time are provided. If the most recent available scram time data set meets Specification 3.3.C.2, the operating MCPR limit shall be as given in Table 3.12-2a. If the most recent available scram time data set does not meet Specification 3.3.C.2 but does meet Specification 3.3.C.3, the operating MCPR limit shall be as given in Table 3.12-2b.

2. For application of the operating MCPR limits as specified in Table 3.12-2a, scram insertion time shall be as follows:

a. The average scram insertion time, based on the de-energization of the scram pilot valve at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

Rod Position	Average Scram Insertion Times (Sec)
46	0.361
36	0.917
26	1.468
06	2.686

b. The average scram insertion times for the three fastest control rods of all groups of four control rods in a 2 x 2 array shall be no greater than:

Rod Position	Average Scram Insertion Times (Sec)
46	0.383
36	0.972
26	1.556
06	2.847

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 950 psig (with saturation temperature) and the requirements of Specification 3.3.B.3.a met. This testing shall be completed prior to exceeding 40% power. Below 30% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. During all scram time testing below 30% power, the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

2. Whenever such scram time measurements are made (such as when a scram occurs and the computer is operable) an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3. For application of the operating MCPR limits as specified in Table 3.12-2b scram insertion times shall be as follows:

a. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

Rod Position	Average Scram Insertion Times (Sec)
46	0.37
36	1.10
26	1.87
06	3.41

b. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a 2 x 2 array shall be no greater than:

Rod Position	Average Scram Insertion Times (Sec)
46	0.39
36	1.17
26	1.98
06	3.62

c. The operating MCPR limits specified in Table 3.12-2a shall not be applied unless the scram insertion time specification in 3.3.C.2 is met.

4. Maximum scram insertion time for 90% insertion of any operable control rod should not exceed 7.00 seconds.

bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit [MCPR = 1.40 (7 x 7 array) or 1.50 (8 x 8 array) and LHGR = 18.5 KW/ft (7 x 7 array) or 13.4 KW/ft (8 x 8 array)]. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the DAEC Chief Engineer.

3. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the safety limit.

Two sets of scram insertion time specifications are provided:

a. That specified in Section 3.3.C.2 is based on analysis of data from DAEC and other plants with the same control drives and is the mean of this data plus a conservatism of approximately three standard deviations. When this specification is met, the operating MCPR limits given in Table 2.12-2a may be applied. Analysis of the most limiting transient (Rod Withdrawal Error) under these conditions shows that MCPR remains greater than the safety limit.

b. That specified in Section 3.3.C.3 is for use if Specification 3.3.C.2 cannot be met and the operating MCPR limits in Table 2.12-2a cannot be applied. If only the specification in Section 3.3.C.3 can be met, only the operating MCPR limits specified in Table 2.12-2b are to be used. Analysis of the most limiting transient (Turbine Trip Without Bypass) under these conditions shows that MCPR remains greater than the 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 requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

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3.3 AND 4.3 REFERENCES

1. Duane Arnold Energy Center Cycle 3 Safety Analysis for Application of Measured Scram Insertion Times, NEDO-24075, 77NED354, Class I, November 1977.

TABLE 3.12-2

MCPR LIMITS

TABLE 3.12-2a

(For application only if scram time Specification 3.3.C.2 is met)

Fuel Type	
7 x 7	1.27
8 x 8	1.27

TABLE 3.12-2b

(For application if scram time Specification 3.3.C.2 is not met)

Fuel Type	Exposure Remaining to End of Cycle	
	B.O.C to > 1000 MWD/T	≤ 1000 MWD/T to E.O.C.
7 x 7	1.27	1.29
8 x 8	1.27	1.37

NEDO-24075
77NED354
Class I
November 1977

DUANE ARNOLD ENERGY CENTER

UNIT 1

CYCLE 3 SAFETY ANALYSIS

APPLICATION OF MEASURED SCRAM INSERTION TIMES

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

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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LIST OF TABLES

1. INTRODUCTION AND SUMMARY

Reference 1 contained the safety analysis for the Duane Arnold Energy Center (DAEC) cycle 3 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 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	≤1.468	≤1.556
06	≤2.686	≤2.847

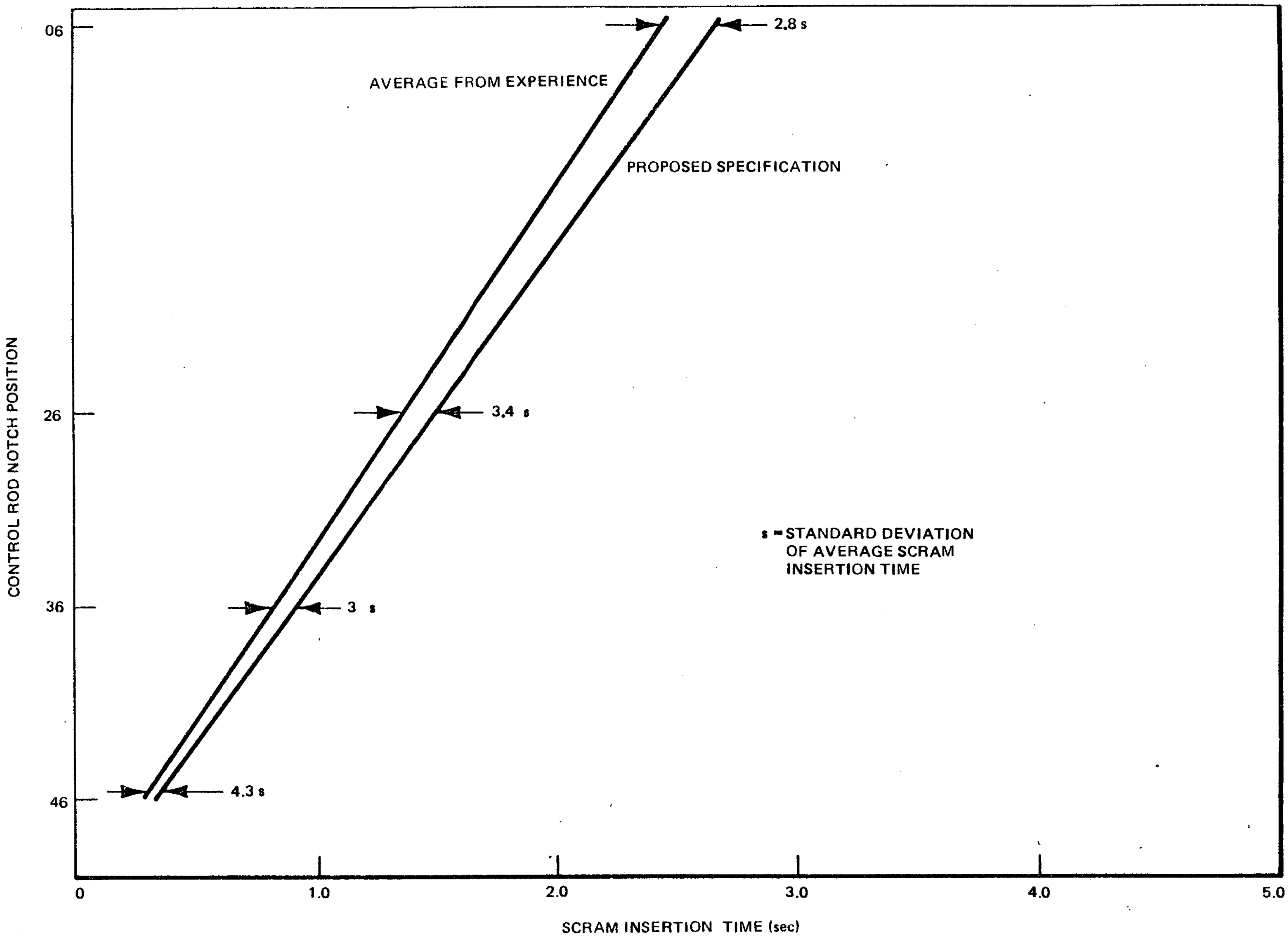

Figure 2-1. Scram Insertion Times DAEC
(Average Scram Insertion Time of All Operable Control Rods)

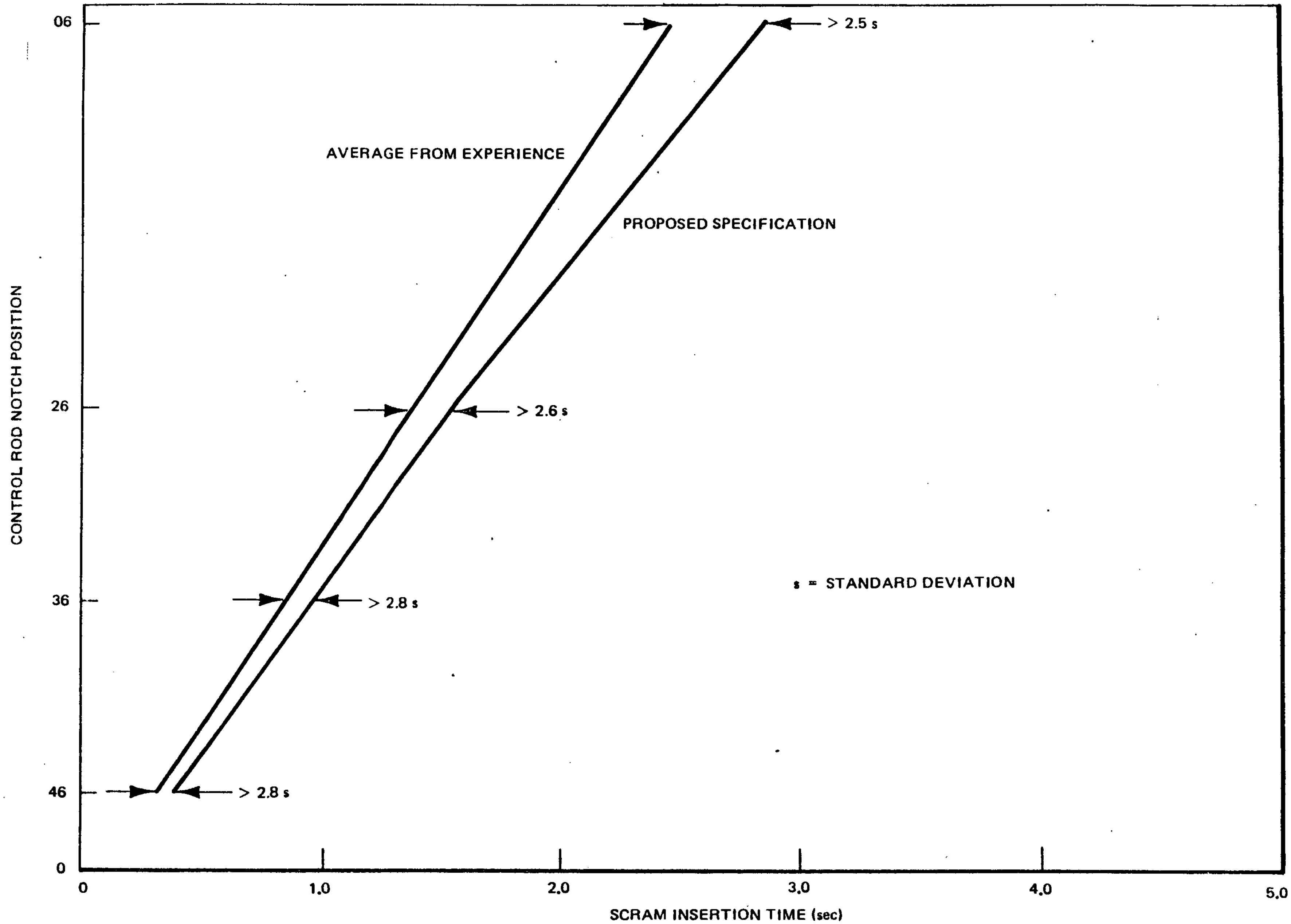

Figure 2-2. Scram Insertion Times, DAEC, Average of Fastest Three out of Four Control Rods in Any 2x2 Array

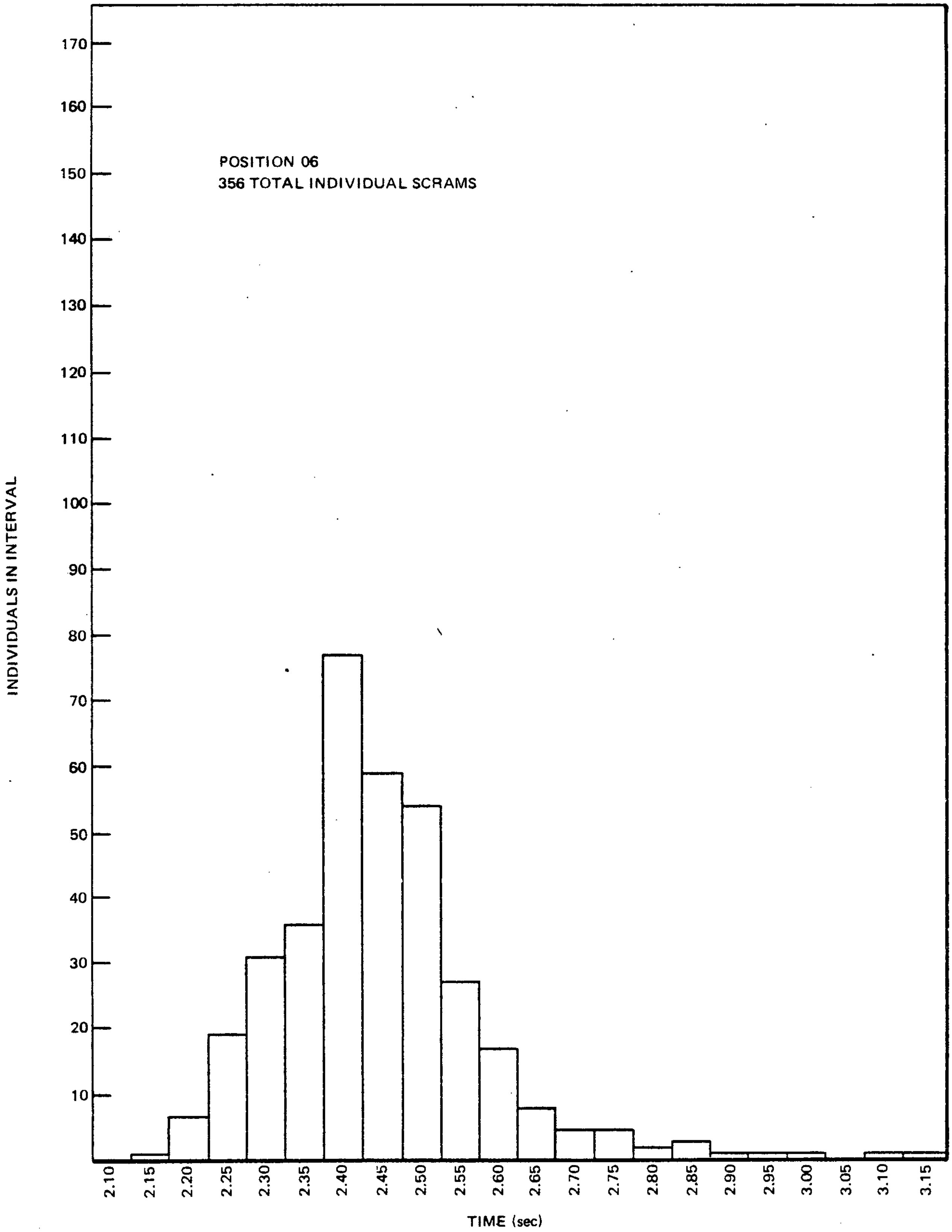

Figure 2-3. Histogram of DAEC Full Core Individual Drive Scram Insertion Time Data

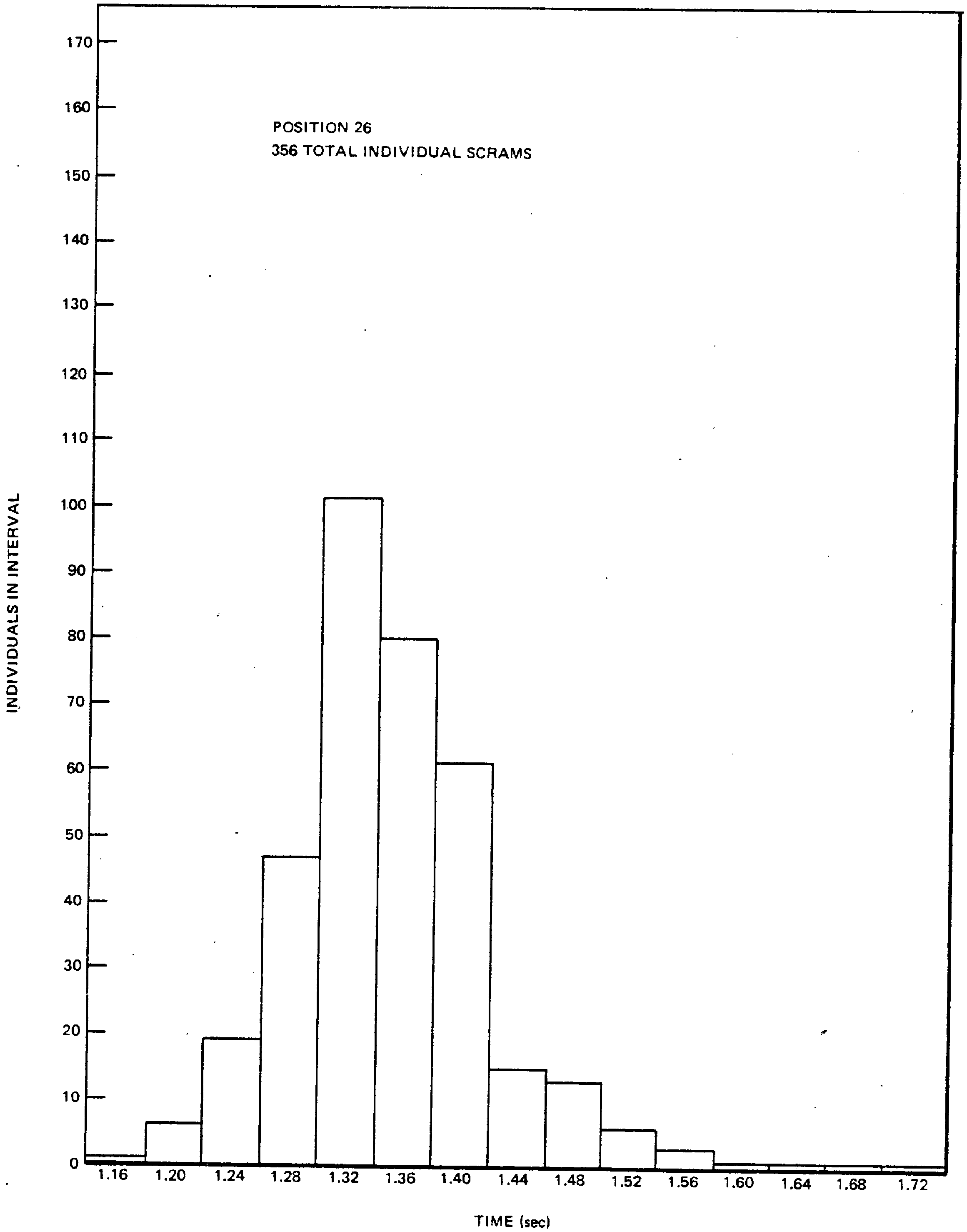

Figure 2-4. Histogram of DAEC Full Core Individual Drive Scram Insertion Time Data

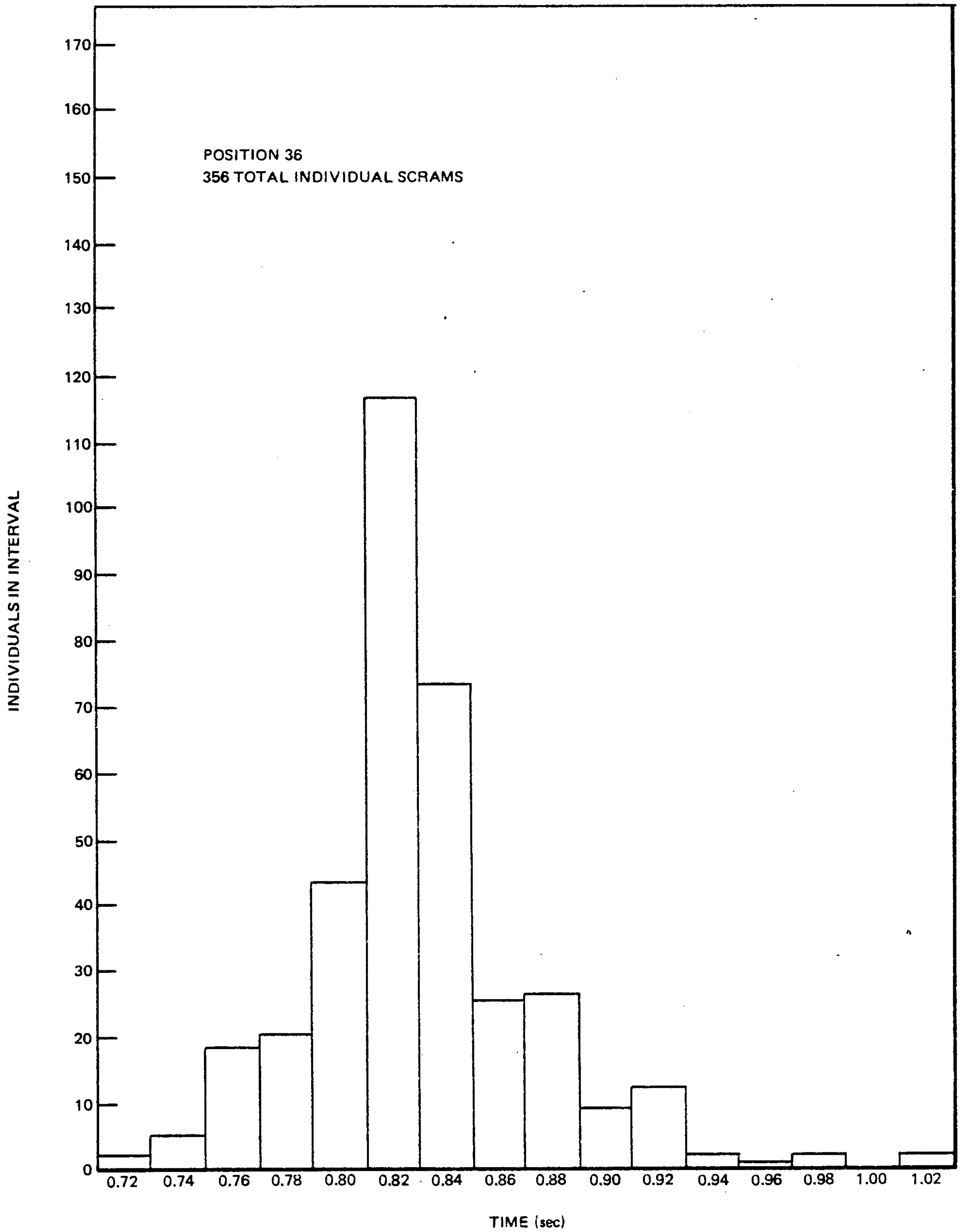

Figure 2-5. Histogram of DAEC Full Core Individual Drive Scram Insertion Time Data

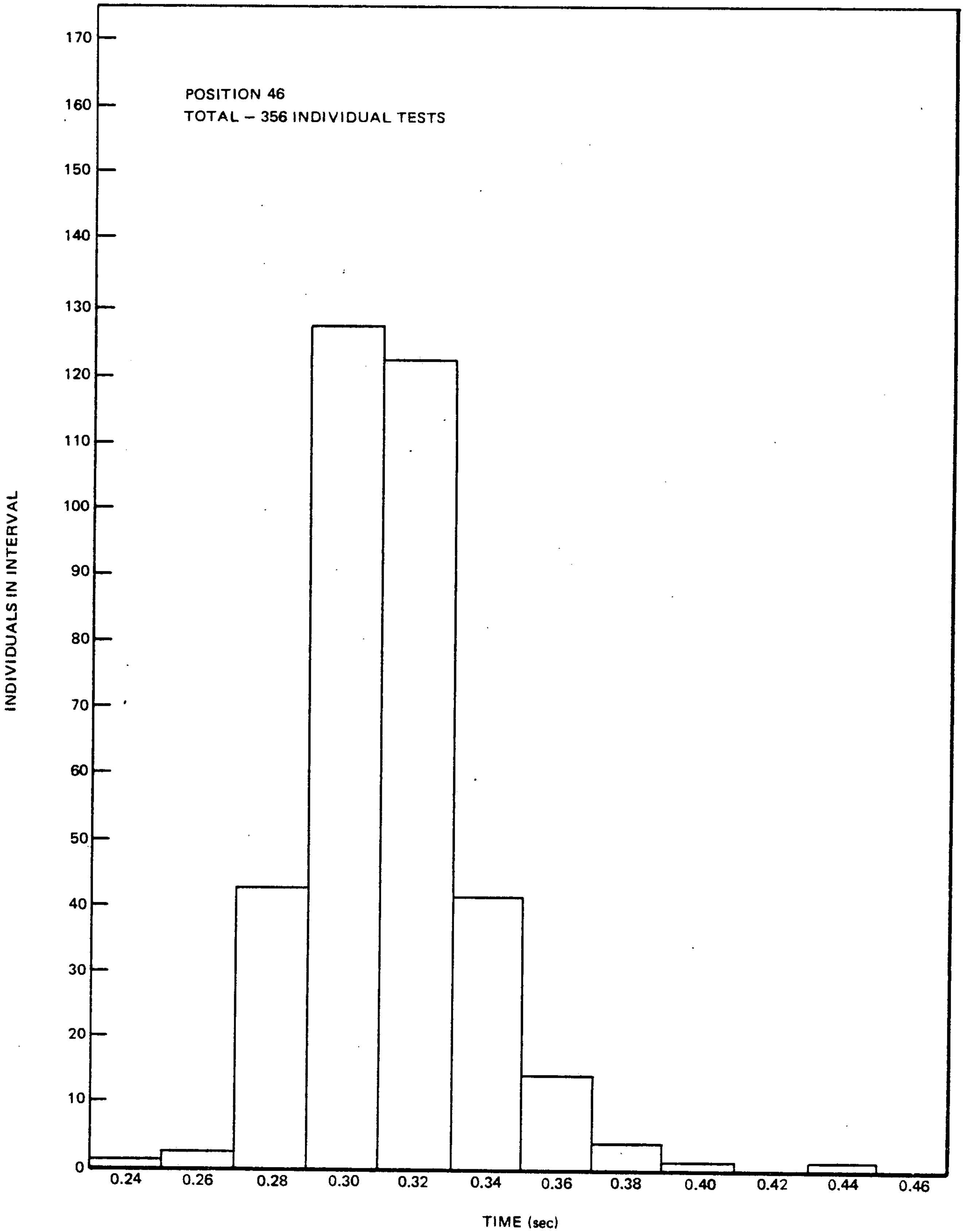

Figure 2-6. Histogram of DAEC Full Core Scram Insertion Time Data

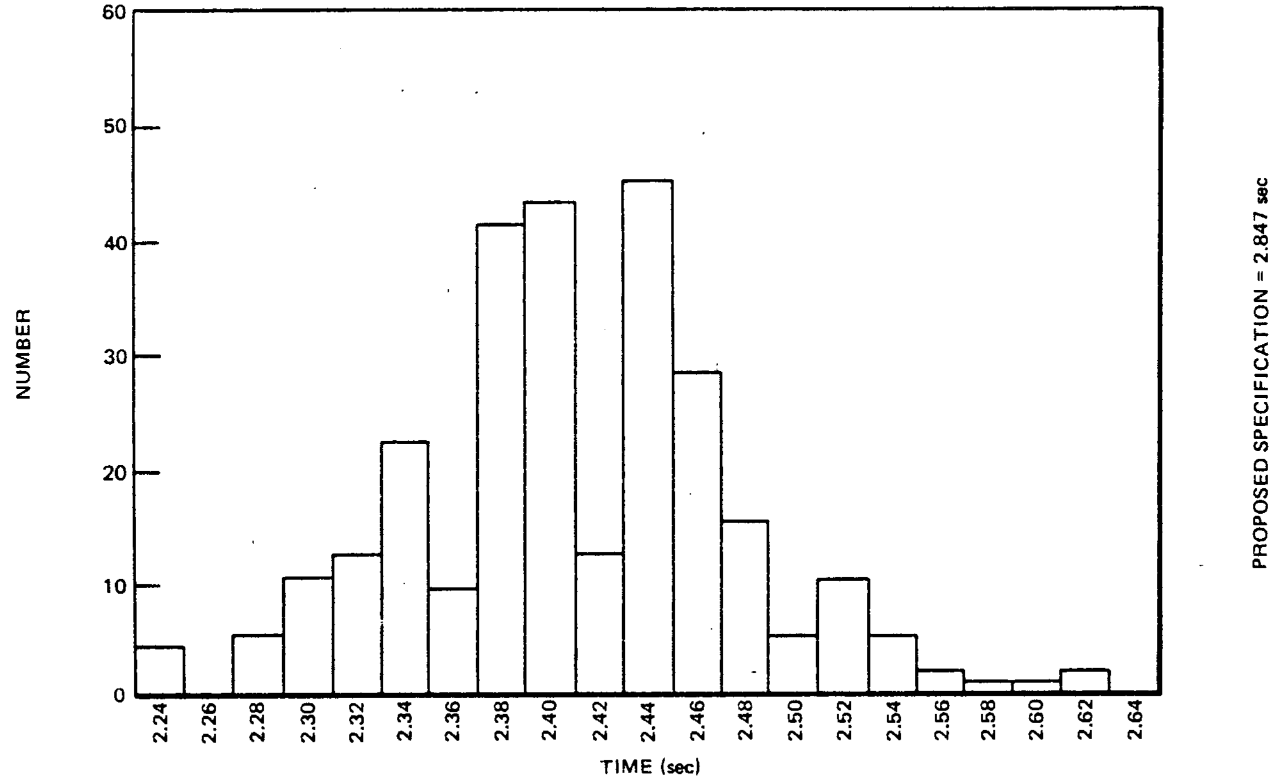

Figure 2-7. Histogram of DAEC Full Core Scram Insertion Data - Average of Fastest Three Control Rods in a 2x2 Array - Position 06

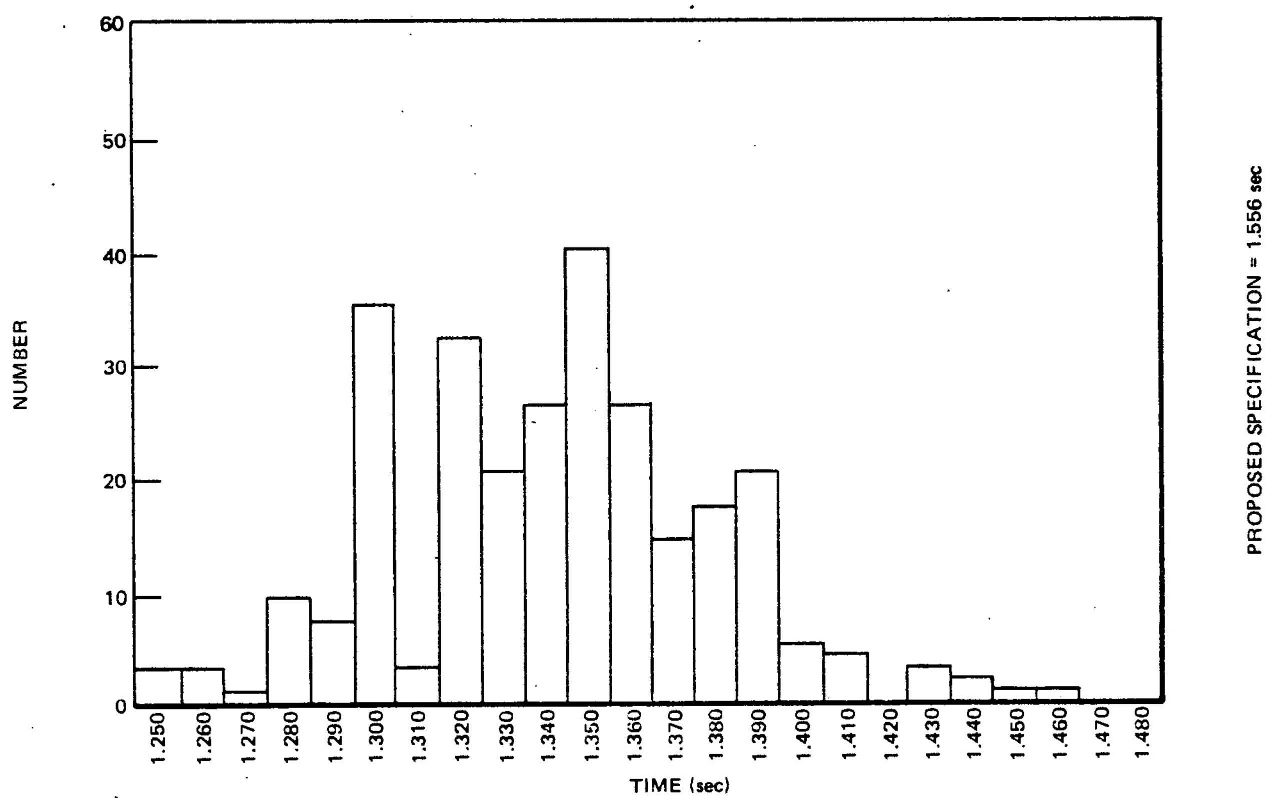

Figure 2-8. Histogram of DAEC Full Core Scram Insertion Data - Fastest Three Control Rods in a 2x2 Array - Position 26

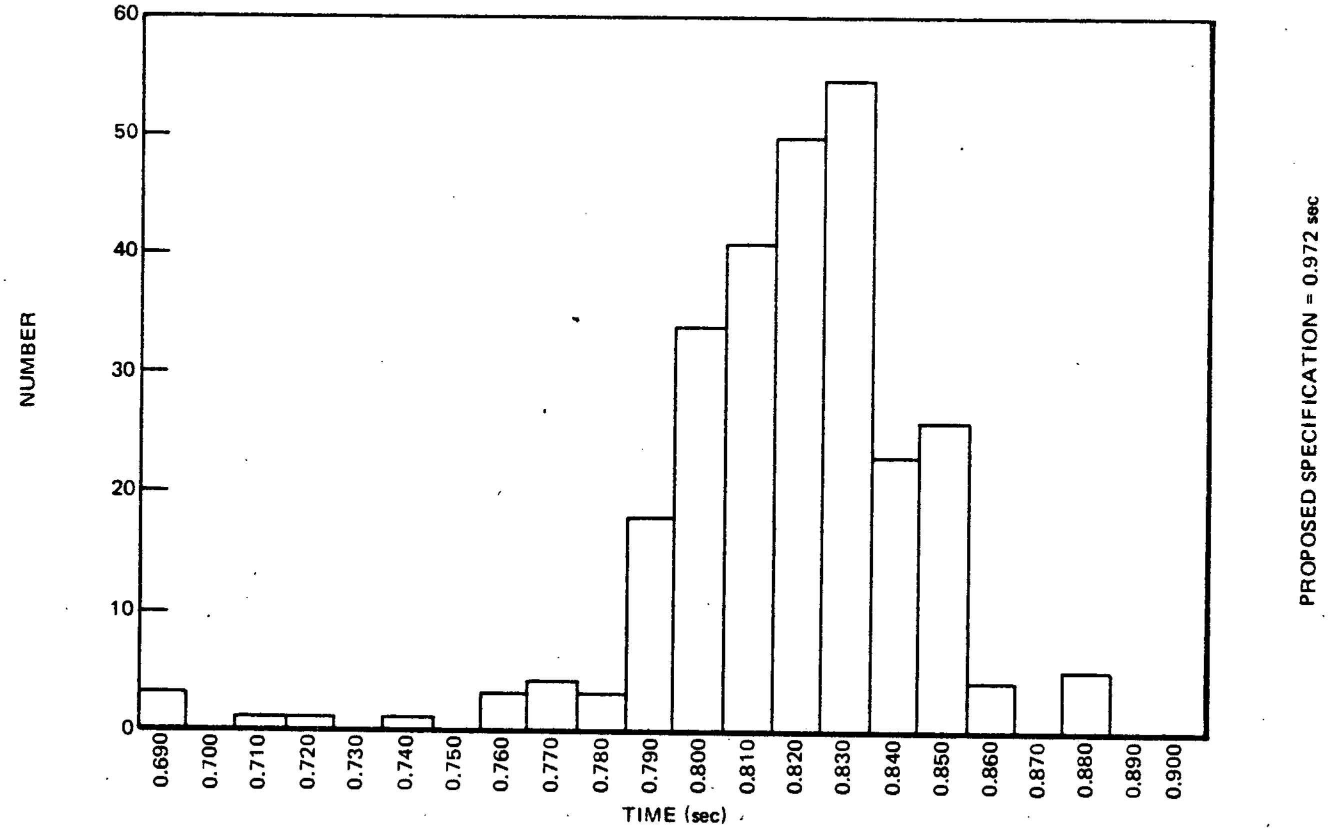

Figure 2-9. Histogram of DAEC Full Core Scram Insertion Data - Average of Fastest Three Control Rods in a 2x2 Array - Position 36

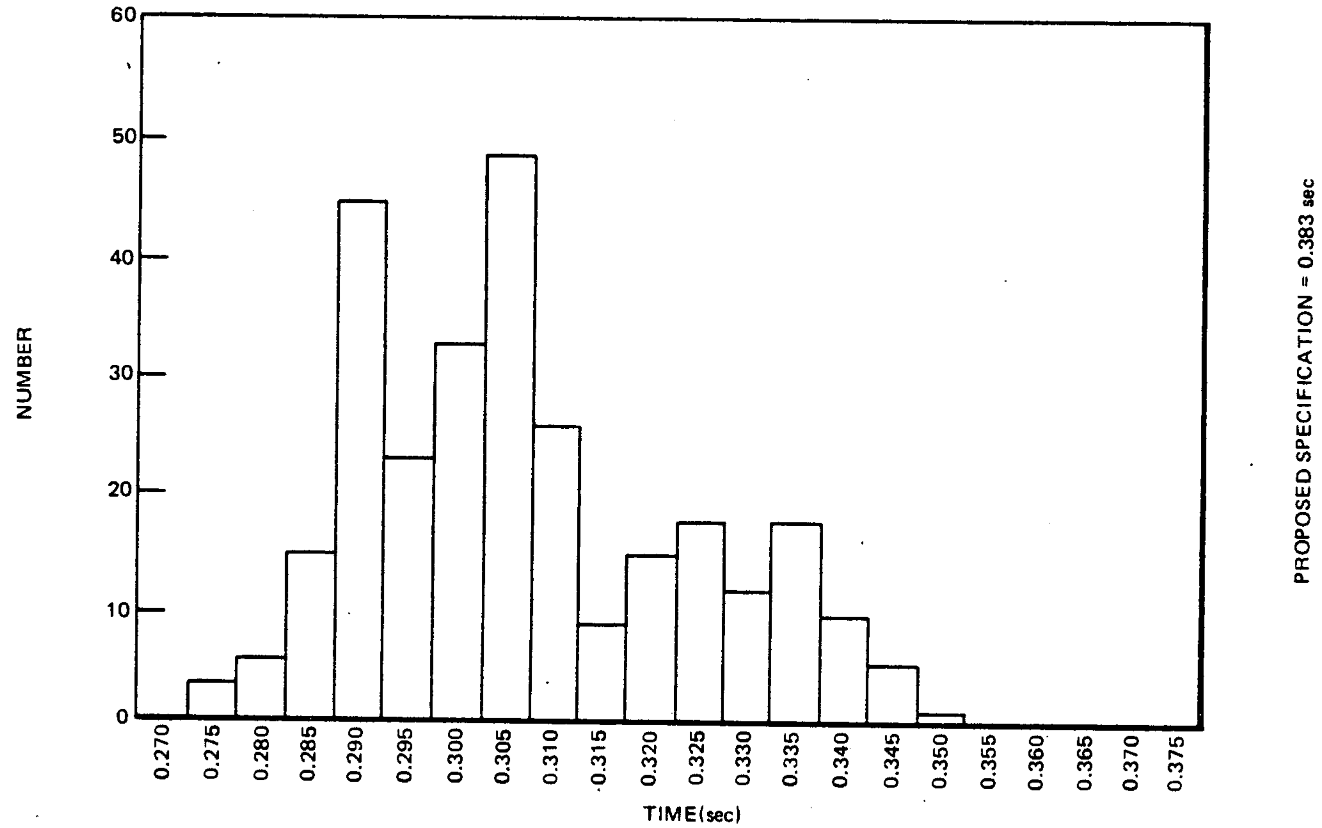

Figure 2-10. Histogram of DAEC Full Core Scram Insertion Data - Average of Fastest Three Control Rods in a 2x2 Array - Position 46

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 3, is given below and in References 1, 3 and 4.

3.1 STATISTICAL ANALYSIS

The statistical analysis is described in Reference 3.

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 3.

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 7x7 and 1.22 for 8x8 fuels.

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.

Table 3-1

SUMMARY OF RESULTS LIMITING ABNORMAL OPERATIONAL ΔCPR TRANSIENTS

	EOC3	
Event	7x7	8x8
Rod Withdrawal Error	0.15	0.16
Loss of Feedwater Heater*	0.14	0.15
Turbine Trip w/o Bypass	0.08	0.12
Feedwater Controller Failure	0.04	0.07
Turbine Trip with Bypass	<0.01	0.02

Table 3-2

GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

	7x7	8x8
Peaking factors (local, radial and axial)	(1.24, 1.285, 1.40)	(1.22, 1.44, 1.40)
R-Factor	1.100	1.098
Bundle Power, MWt	5.524	6.115
Non-fuel Power Fraction	0.04	0.04
Core Flow, Mlb/hr	49.0	49.0
Bundle Flow, 10^3 lb/hr	122.7	110.9
Reactor Pressure, psia	1035.0	1035.0
Inlet Enthalpy, Btu/lb	526.3	526.3
Initial MCPR	1.20	1.20

*Results of bounding analysis from Reference 1.

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 3 using the proposed new scram insertion time specification. The control rod drive specifications are given in Figure 4-4.

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 initial reactor power is at a level corresponding to 105% of rated steam flow, the neutron flux peaks at 249% initial, the average surface heat flux peaks at 104% of initial.

The peak streamline pressure is limited to 1171 psig as a result of the high-pressure actuation of the six safety/relief valves which provides a 69-psi margin to the 1240-psig set point of the first spring safety valve.

4.2.2 Loss of a Feedwater Heater

The loss of a feedwater heater was analyzed in Reference 1. This analysis is consevative for the new scram insertion time. Since the loss of feedwater heater does not affect the MCPR limit this transient was not reanalyzed.

4.2.3 Rod Withdrawal Error

The rod withdrawal error was analyzed for the fully drilled core (most conservative case). The results were not measurably different from those presented in Reference 3; therefore the analysis presented in Reference 3 is applicable to the half-drilled core. 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 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 4-2. The initial reactor power is at a level corresponding to 105% of NBR steam flow, the neutron flux peaks at 140% of initial, the average surface heat peaks at 100% initial.

The peak streamline pressure is limited to 1137 psig as a result of the high-pressure actuation of the six safety/relief valves, which provides a 103-psi margin to the 1240-psig set point of the first spring safety valve.

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 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 neutron flux peaks at 153% of initial and the average surface heat flux peaks at 104% initial.

The peak streamline pressure is limited to 1138 psig as a result of the high-pressure actuation of the six safety/relief valves, which provides a 102-psi margin to the 1240-psig set point of the first spring safety valve.

Table 4-1

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	
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)	11.67	9.34
Doppler Coefficient	(-c/°F)	0.2186	0.2301
Average Fuel Temperature	(°F)	1435	1435
Scram Reactivity Curve		Fig. 4-4	Fig. 4-4
Scram Worth	(-$)	31.51	39.39

Table 4-2

TRANSIENT DATA SUMMARY

Transient	Power (%)	Core Flow (%)	ϕ (% reference)	O/A (% reference)	Psl (psig)
Turbine Trip without Bypass	104	100	249	104	1171
Loss of Feedwater Heater*	104	100	121	119	1023
Feedwater Controller Failure	104	100	153	104	1138
Turbine Trip with Bypass	104	100	140	100	1137

*Results of bounding analysis from Reference 1.

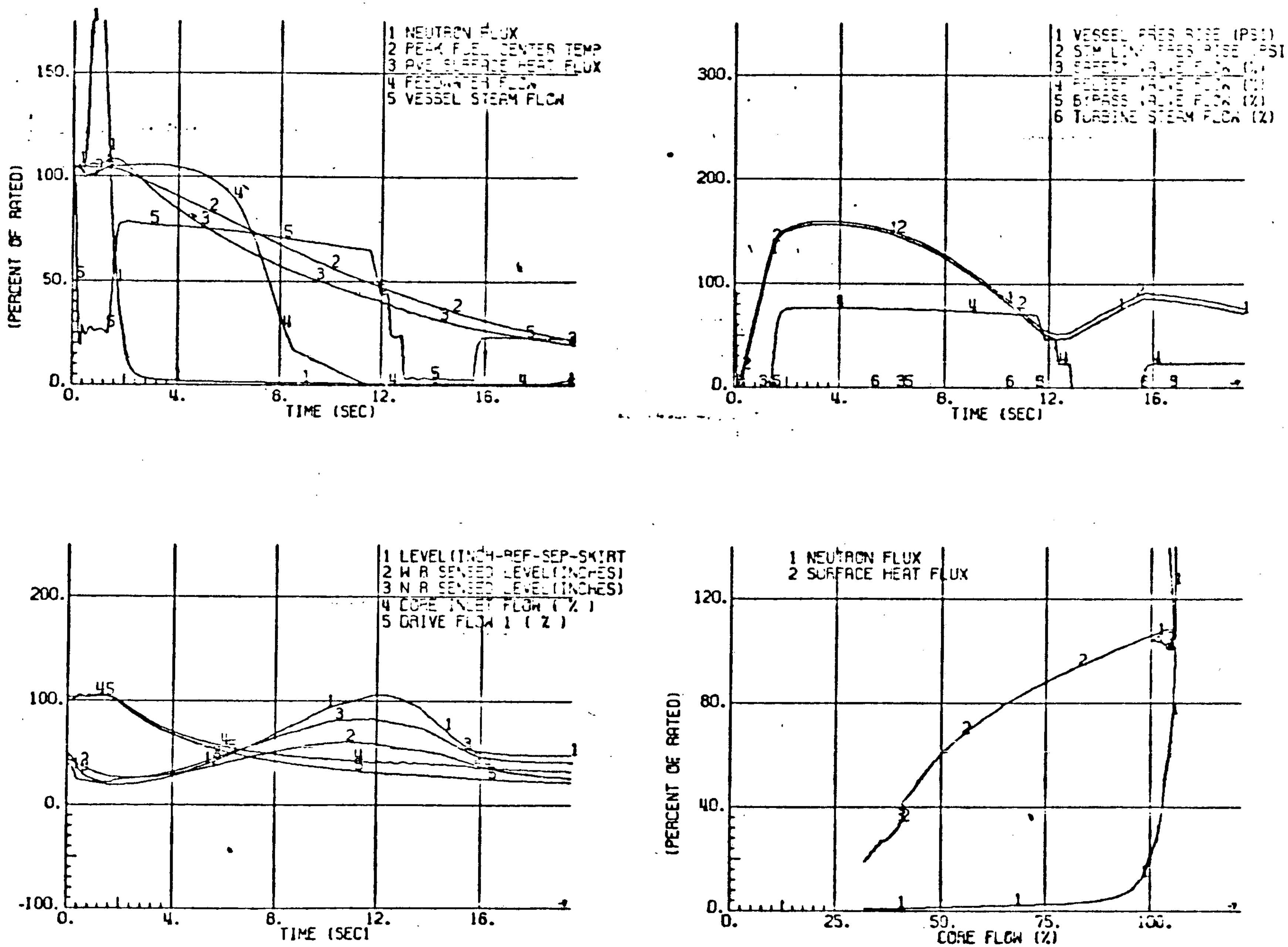

Figure 4-1. Turbine Trip Without Bypass-Trip Scram

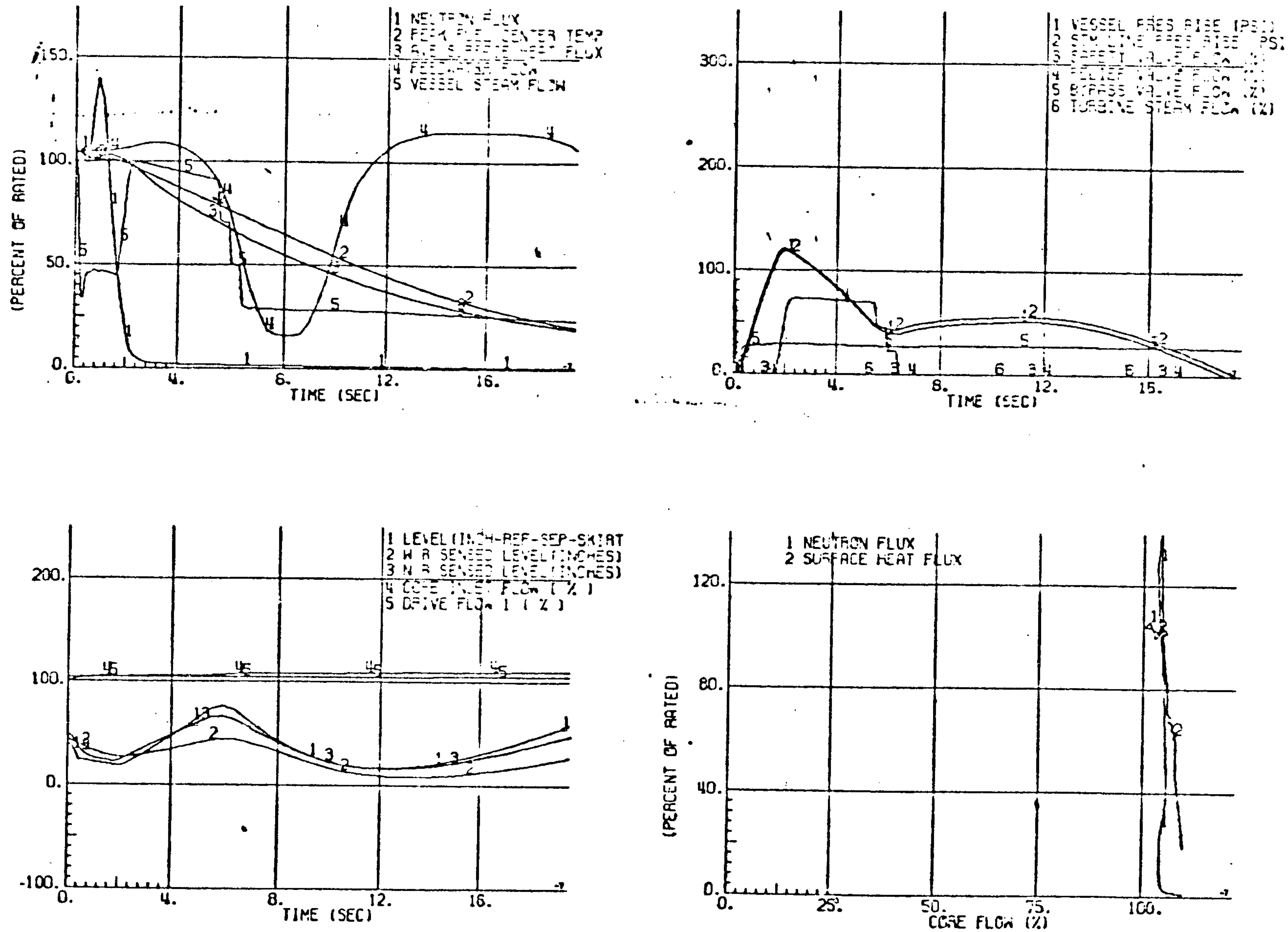

Figure 4-2. Turbine Trip With Bypass-Trip Scram

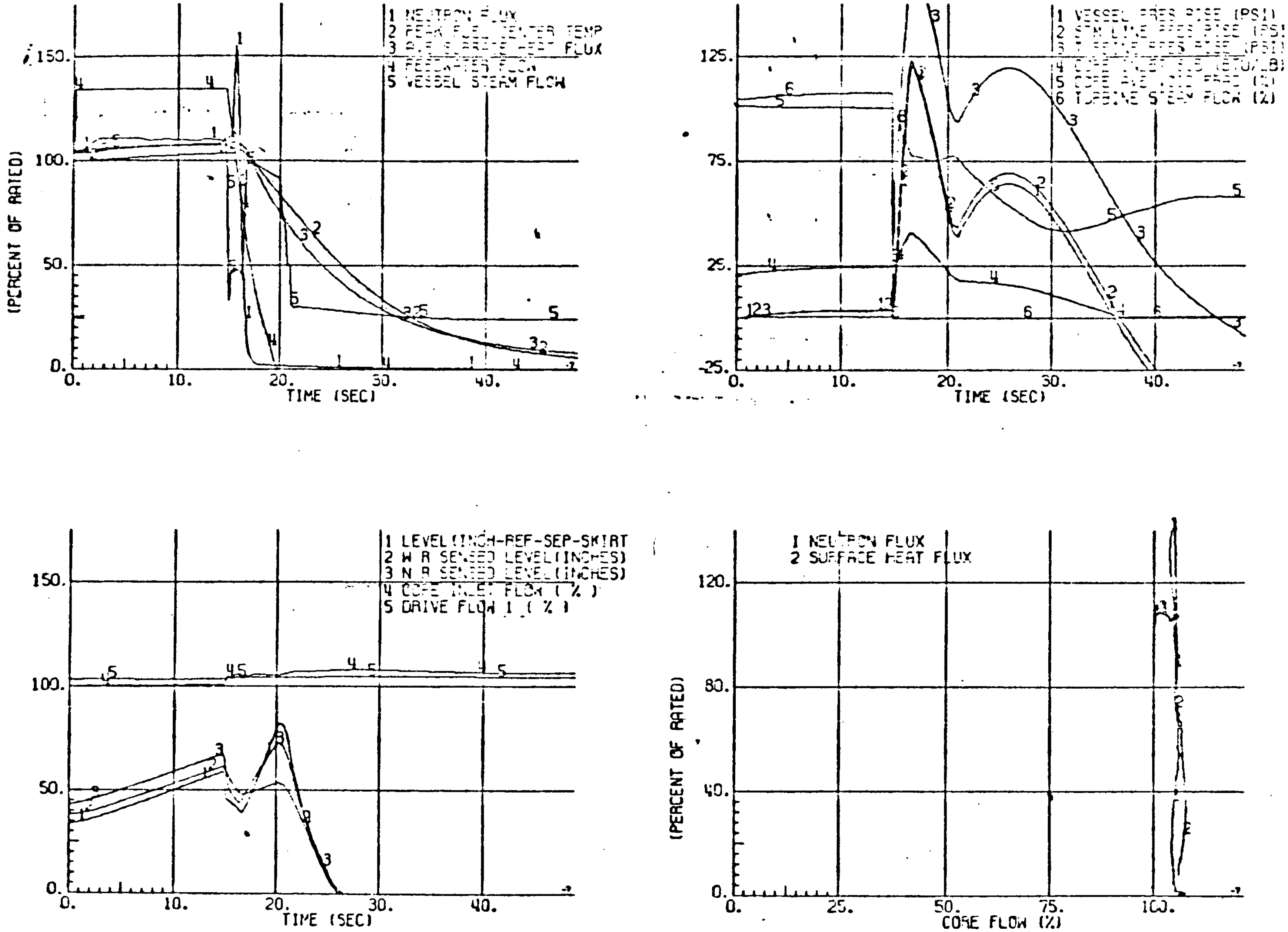

Figure 4-3. Feedwater Controller Failure Maximum Demand With High Level Turbine Trip

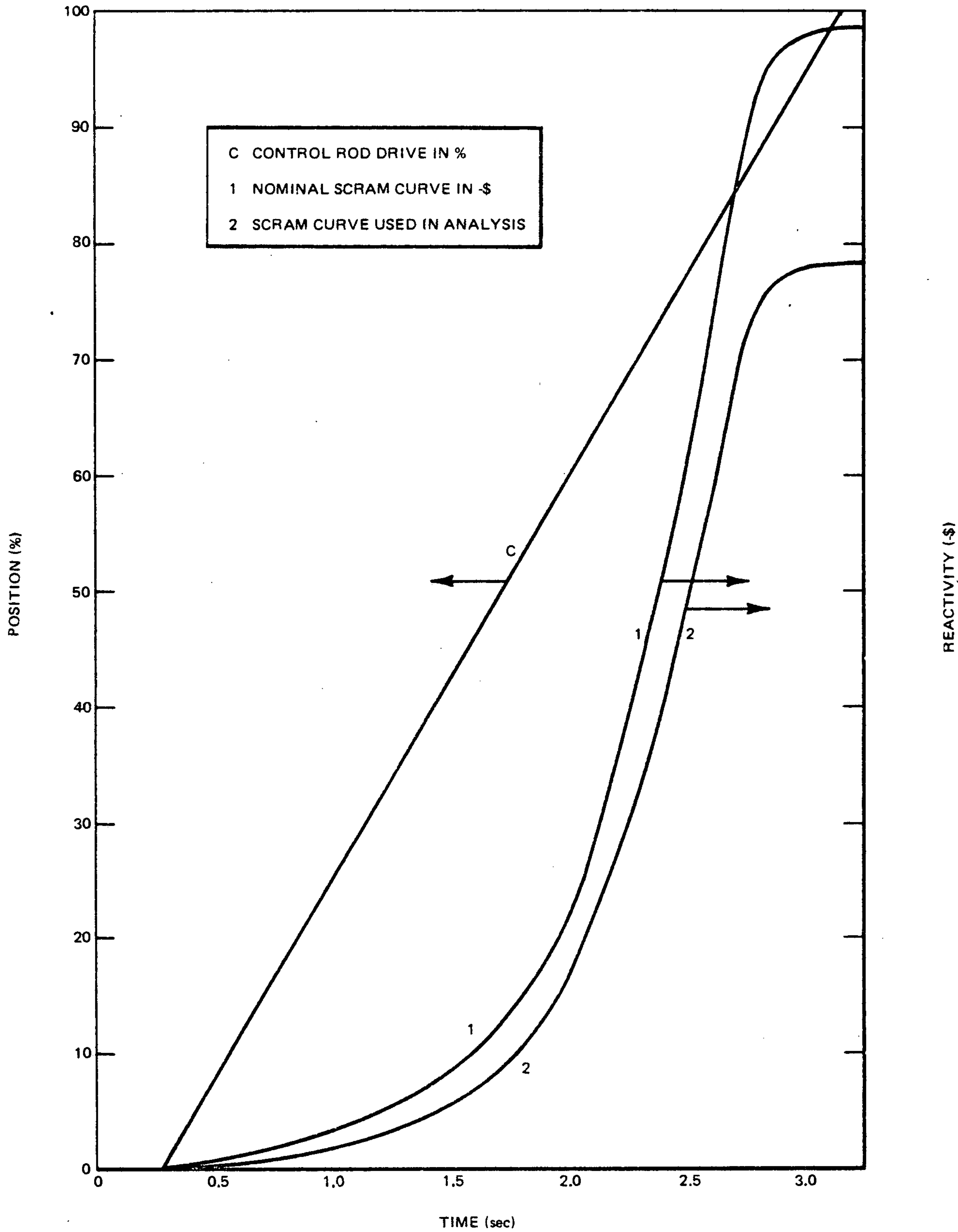

Figure 4-4. Control Rod Drive Specification and Scram Reactivity, DAEC, Cycle 3

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 Number 2 Licensing Submittal Supplement 1, Partially Drilled Core, Duane Arnold Energy Center," Supplement 1, June 1977, (NEDO-21082-02-1).

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 2 Licensing Submittal Duane Arnold Energy Center, January 1977 (NEDO-21082-02).

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

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TECHNICAL INFORMATION EXCHANGE

TITLE PAGE

AUTHOR	SUBJECT	TIE NUMBER
Jim Rash	DUANE ARNOLD	77NED354
		DATE November 1977
TITLE DUANE ARNOLD ENERGY CENTER UNIT 1 CYCLE 3 SAFETY ANALYSIS APPLICATION OF MEASURED SCRAM INSERTION TIMES		GE CLASS I
		GOVERNMENT CLASS —
REPRODUCIBLE COPY FILED AT TECHNICAL SUPPORT SERVICES, R&UO, SAN JOSE, CALIFORNIA 95125 (Mail Code 211)		NUMBER OF PAGES 24

SUMMARY

This report describes the creation of the data base and the statistical analysis, specifies the proposed scram insertion time limit and presents analyses to support MCPR limits based on the new scram insertion time limits.

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DOCUMENT NUMBER NEDO-24075

INFORMATION PREPARED FOR Nuclear Energy Division

SECTION BWR PD

BUILDING AND ROOM NUMBER K-2606 MAIL CODE 682