XN-NF-82-84(NP)
Revision 1

Issue Date: 12/07/82

PLANT TRANSIENT ANALYSIS FOR

DRESDEN UNIT 2 CYCLE 9

Prepared by: Richard H Kelley (12/2/82)
R. H. Kelley
Plant Transient Analysis

Concur: G. C. Cooke 12/3/82
G. C. Cooke, Manager
Plant Transient Analysis

Approve: R. B. Stout 3 Dec 82
R. B. Stout, Manager
Licensing & Safety Engineering

Approve:
G. A. Sofer, Manager
Fuel Engineering & Technical Services

gf

EXXON NUCLEAR COMPANY, Inc.

8212290206 821221
PDR ADOCK 05000237
P PDR

NUCLEAR REGULATORY COMMISSION DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

This technical report was derived through research and development programs sponsored by Exxon Nuclear Company, Inc. It is being submitted by Exxon Nuclear to the USNRC as part of a technical contribution to facilitate safety analyses by licensees of the USNRC which utilize Exxon Nuclear-fabricated reload fuel or other technical services provided by Exxon Nuclear for light water power reactors and it is true and correct to the best of Exxon Nuclear's knowledge, information, and belief. The information contained herein may be used by the USNRC in its review of this report, and by licensees or applicants before the USNRC which are customers of Exxon Nuclear in their demonstration of compliance with the USNRC's regulations.

Without derogating from the foregoing, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this document.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

This report presents the results of Exxon Nuclear Company's (ENC) evaluation of core-wide transient events for Dresden Station Unit 2 during Cycle 9 operation. Specifically, the evaluation determines the necessary thermal margin limits to protect against the occurrence of boiling transition during the most limiting anticipated transient. Also, the evaluation demonstrates that vessel integrity will be protected during the most limiting pressurization event. The results are also incorporated in Reference 2.

This analysis was performed with the same methodology[1] used to establish thermal margin requirements for Dresden Unit 3 Cycle 8. The limiting expected transient, load rejection without condenser bypass, and maximum pressurization event, closure of all main steam isolation valves, were determined to be the same for Dresden Unit 2 as previously determined for Dresden Unit 3[6].

2.0 SUMMARY

The determination of the Minimum Critical Power Ratio (MCPR) for Dresden Unit 2 Cycle 9 was based upon the consideration of various possible operational transients(1). A MCPR of 1.31 or greater for all 8x8 fuel types during Cycle 9 adequately limits the occurrence of boiling transition during an end of cycle full load rejection without condenser bypass, as well as other less limiting anticipated operational transients. This assumes compliance with other related restrictions specified by the Dresden Unit 2 Operating License and associated Technical Specifications. The MCPR operating limits required for the more potentially limiting events are shown in Table 2.1.

The maximum system pressure has been calculated for the containment isolation event, which is a rapid closure of all main steam isolation valves, with an adverse scenario as specified by the ASME Pressure Vessel Code. The safety valves of Dresden Unit 2 have sufficient flow capacity and opening rates to prevent pressure from reaching the established transient safety limit of 1375 psig, which is 110% of design pressure. The maximum system pressures predicted during the event are shown in Table 2.1.

A summary of results of the transient analyses is shown in Table 2.2. This Table shows the relative maximum fuel power levels, core average heat fluxes, and maximum vessel pressures attained during the more limiting transient events.

Table 2.1 Thermal Margin Summary

Transient	ΔCPR/MCPR		
	8x8(XN-1)	8x8R(GE)	8x8(GE)
Generator Load Rejection (w/o bypass)	.26/1.31(1)	.26/1.31(1)	.26/1.31(1)
Increase in Feedwater Flow	.21	.21	.21
Loss of Feedwater Heating	.16	.16	.16

Maximum Pressure (psig)*

Transient	Vessel Dome	Vessel Lower Plenum	Steam Lines
MSIV Closure	1324.9	1349.3	1322.1

* Limit allowed is 1375 psig

---

(1) See Section 3.2.1 for basis of these values

Table 2.2 Results of Plant Transient Analyses

Event	Maximum Neutron Flux (% Rated)	Maximum Core Average Heat Flux (% Rated)	Maximum Vessel Pressure (psig)
Load Rejection(1) w/o Bypass	350.3	114.5	1281.1
Increase in Feedwater Flow	298.4	116.9	1207.2
Loss of Feed-water Heating	120.1	<120.0	1039.9
MSIV Closure w/flux scram	483.3	133.0	1349.3

----

(1) Nominal case, all other events are bounding case

3.0 TRANSIENT ANALYSIS FOR THERMAL MARGIN

3.1 DESIGN BASIS

The plant transient analysis determined that the most thermal margin limiting condition was operation at full reactor power. Reactor and plant conditions for this analysis are shown in Table 3.1. The most limiting point in cycle was end of full power capability when control rods are fully withdrawn from the core. The thermal margin limit established for end of full power conditions is conservative for cases where control rods are partially inserted or reactor power is less. Following requirements established in the Plant Operating License and associated Technical Specifications, observance of the MCPR operating limit of 1.31 or greater for all 8x8 fuel types protects against boiling transition during all anticipated transients at the Dresden Unit 2 for Cycle 9.

The calculational models used to determine thermal margin include ENC's plant transient(1), fuel performance(4), and core thermal-hydraulic(5) codes as described in previous documentation(1). Fuel pellet to clad gap conductances used in the analyses are based on previously submitted analyses(6). All calculational models have been benchmarked against appropriate measurement data, but the current evaluations are intentionally designed to provide a thermal margin which accounts for the random variability and uncertainty of critical parameters. For the limiting generator load rejection without bypass event, the variability of four critical parameters was statistically convoluted so that the calculated thermal margin bounds 95% of the possible outcomes. Table 3.2

summarizes the values used for important parameters. Table 3.3 provides the feedwater flow, recirculating coolant flow, and pressure regulation system settings used in the evaluation.

3.2 ANTICIPATED TRANSIENTS

ENC considered eight categories of potential transient occurrences for Jet Pump BWR's in XN-NF-79-71(1). Three of these transients have been evaluated here to determine the thermal margin for Cycle 9 at Dresden Unit 2. These transients are:

- generator load rejection w/o bypass
- increase in feedwater flow
- loss of feedwater heating

Other plant transient events are inherently non-limiting or clearly bounded by one of the above.

3.2.1 Generator Load Rejection without Condenser Bypass

This event is the most limiting of the class of transients characterized by rapid vessel pressurization. The turbine/generator control system causes a fast closure of the turbine control valves. Closure of these valves causes the reactor system to be pressurized while the reactor protection system scrams the reactor in response to the sensing of the fast closure of the control valves. Condenser bypass flow, which can mitigate the pressurization effect, is not allowed. The excursion of core power due to void collapse (by pressurization) is terminated by reactor scram since other mechanisms of power shutdown (Doppler feedback, pressure relief, etc.) are only partly successful. Figures 3.3, 3.4 and 3.5 depict the time variance of critical reactor and plant parameters during a load rejection event with expected void reactivity feedback and normal scram performance. ENC evaluated this event to determine a ΔCPR which would not be exceeded in 95% of the possible outcomes of the event when four variables were considered:

*[ ]

The standard deviations of the first two variables were [ ] The standard deviations of the latter two variables were based upon plant test data:

[ ]

The experimental design for the statistical analysis is given in Appendix B. The calculated results of the statistical evaluation were:

mean ΔCPR	.232
standard deviation	.014
95% ΔCPR	.260

3.2.2 Increase in Feedwater Flow

Failure of the feedwater control system is postulated to lead to a maximum increase of feedwater flow into the vessel. As the excessive feedwater flow subcools the recirculating water returning to the reactor core, the core power will rise and attain a new equilibrium if no other action is taken. Eventually, the inventory of water in the downcomer

*Brackets identify ENC proprietary information.

will rise until the high vessel water trip setting is exceeded. To protect against spillover of subcooled water to the turbine, the turbine trips, w . . esultant closure of the turbine stop valves. The power increase is terminated by scram, and pressure relief is obtained from the bypass valves opening. The present evaluation of this event assumed that all the conservative conditions of Table 3.2 were concurrent; no statistical evaluation was considered, and the ΔCPR calculated represents a bounding result. Though small differences exist between G.E. and ENC fuel, the highest ΔCPR of 0.21 reported is adequate to protect all fuel types against boiling transition. Figures 3.6, 3.7 and 3.8 display critical variables for this event for the critical 4 seconds following the turbine trip.

3.2.3 Loss of Feedwater Heating

The loss of feedwater heating leads to a gradual increase in the subcooling of the water in the reactor lower plenum. Reactor power slowly rises to the overpower trip point (120% of rated power). The gradual power change allows fuel thermal response to maintain pace with the increase in neutron flux. For this analysis, it was assumed that the initial feedwater temperature dropped 145°F linearly over a two minute period. The magnitude of the void reactivity feedback was assumed to be 25% lower than expected, so that the power response to subcooling was gradual, maximizing the thermal heat flux. Scram performance was assumed at its Technical Specification limit with scram worth 20% below expected. Reactor neutron flux reached 120.1% of rated and clad surface heat flux increased nearly as much. Calculation of thermal margin assumed that bundle power increased by 20% which predicted a ΔCPR of 0.16 for each fuel type. Figures 3.9 and 3.10 depict the transient progression.

3.3 CALCULATIONAL MODEL

The plant transient model used to evaluate the load rejection and feedwater increase event was ENC's advanced code, COTRANSA[1]. This one-dimensional neutronics model predicted reactor power shifts toward the core middle and top as pressurization occurred. This was accounted for explicitly in determining thermal margin changes in the transient. The loss of feedwater heating event was evaluated with the PTSBWR3[1] code since rapid pressurization and void collapse do not occur in this event.

3.4 SAFETY LIMIT

The safety limit is the minimum value of the critical power ratio (CPR) at which the fuel could be operated, where the expected number of rods in boiling transition would not exceed 0.1% of the heated rods in the core. Thus, the safety limit is the minimum critical power ratio (MCPR) which would be permitted to occur during the limiting anticipated operational occurrence as previously calculated. The MCPR operating limit is derived by adding the change in critical power ratio (ΔCPR) of the limiting anticipated operational occurrence to the safety limit.

The safety limit for Dresden Unit 2 Cycle 9 was determined by the methodology presented in Reference 3, and used to license Dresden-3, to have the following value:

Dresden Unit 2 Cycle 9 MCPR Safety Limit = 1.05.

The input parameter values and uncertainties used to establish the safety limit are presented in Appendix A.

Table 3.1 Design Reactor and Plant Conditions (Dresden 2)

Reactor Thermal Power (Mwt)	2527.0
Total Recirculating Flow (Mlb/hr)	98.0
Core Channel Flow (Mlb/hr)	87.4
Core Bypass Flow (Mlb/hr)	10.6
Core Inlet Enthalpy (BTU/lbm)	522.9
Vessel Pressures (psia)	
Dome	1020.0
Upper Plenum	1026.0
Core	1035.0
Lower Plenum	1049.0
Turbine Pressure (psia)	964.7
Feedwater/Steam Flow (Mlb/hr)	9.8
Feedwater Enthalpy (BTU/lbm)	320.6
Recirculating Pump Flow (Mlb/hr)	17.1 (1)

(1) Per pump

Table 3.2 Significant Parameter Values Used (1)

High Neutron Flux Trip	3032.4 MW
Control Rod Insertion Time	3.5 sec/90% inserted
Control Rod Worth	20% below nominal
Void Reactivity Feedback	10% above nominal (2)
Time to Deenergized Pilot Scram Solenoid Valves	283 msec
Time to Sense Fast Turbine Control Valve Closure	80 msec
Time from High Neutron Flux Trip to Control Rod Motion	290 msec
Turbine Stop Valve Stroke	100 msec
Turbine Stop Valve Position Trip	90% open
Turbine Control Valve Stroke (Total)	150 msec
Fuel/Clad Gap Conductance	
Core Average (Constant)	893 BTU/hr-ft^2-^{o}F
Limiting Assembly (variable*)	1430 BTU/hr-ft^2-^{o}F (at 8.475 kw/ft)
Safety/Relief Valve Performance Settings	Technical Specifications

(1) Generator load rejection w/o bypass event was evaluated statistically (see Section 3.2.1)

(2) 25% for calculations with point kinetics model

* Varies slightly with power and fuel type

Table 3.2 Significant Parameter Values Used (cont.)

Safety/Relief Valve Performance (cont.)	
Pilot Safety/Relief Valve Capacity	166.1 lbm/sec (at 1080 psig)
Power Relief Valves Capacity	620.0 lbm/sec (at 1120 psig)
Safety Valves Capacity	1432.0 lbm/sec (at 1240 psig)
Pilot Operated Valve Delay/Stroke	0.4/0.1 sec
Power Operated Valves Delay/Stroke	0.65/0.2 sec
MSIV Stroke Time	3.0 sec
MSIV Position Trip Setpoint	90% open
Condenser Bypass Valve Performance	
Total Capacity	1085.2 lbm/sec
Delay to Opening (from demand)	0.1 sec
Opening Time (Entire Bank with (Maximum Demand)	1.0 sec
% Energy Generated in Fuel	96.5%
Vessel Water Level (above Separator Skirt)	
Normal	30 inches
Range of Operation	$\pm$10 inches
High Level Trip	42 inches
Maximum Feedwater Runout Flow (3 pumps)	4966 lbm/sec
Maximum Feedwater Runout Flow (2 pumps)	3310.67 lbm/sec
Doppler Reactivity Coefficient (nominal)	-0.00232$/^{o}F/void fraction
Void Reactivity Coefficient (nominal)	-16.40$/void fraction
Scram Reactivity Worth	Figure 3.1
Axial Power Distribution	Figure 3.2
Delayed Neutron Fraction	.0051
Prompt Neutron Lifetime	4.93×10^{-5} sec
Recirculating Pump Trip Setpoint	1240 psig (vessel pressure)

Table 3.3 Control Characteristics

Sensor Time Constants	
Pressure	0.1 sec
Others	0.25 sec
Feedwater Control Mode	1-element
Feedwater Master Controller	
Proportional Band	100%
Reset	5 repeats/min
Feedwater 100% Mismatch	
Water Level Error	60 inches
Steam Flow (not used)	12 in equivalent
Flow Control Mode	Master Manual
Master Flow Control Settings	
Proportional Band	200%
Reset	8 repeats/min
Speed Controller Settings	
Proportional Band	350%
Reset	20 repeats/min
Pressure Setpoint Adjustor	
Overall Gain	5 psi/% demand
Time Constant	15 sec
Pressure Regulator Settings	
Lead	1.0 sec
Lag	6.0 sec
Gain	30 psid/100% demand

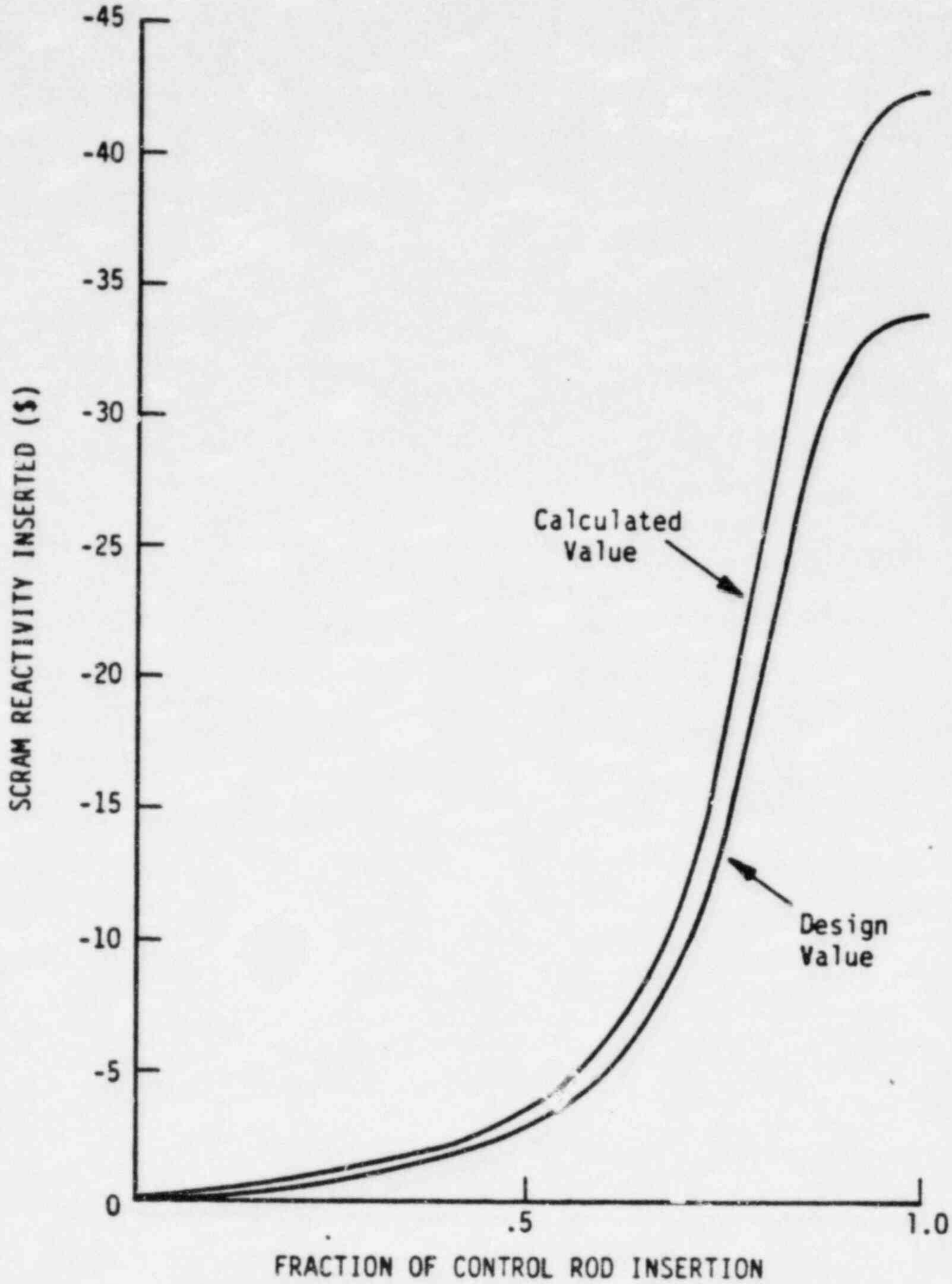

Figure 3.1 Scram Reactivity

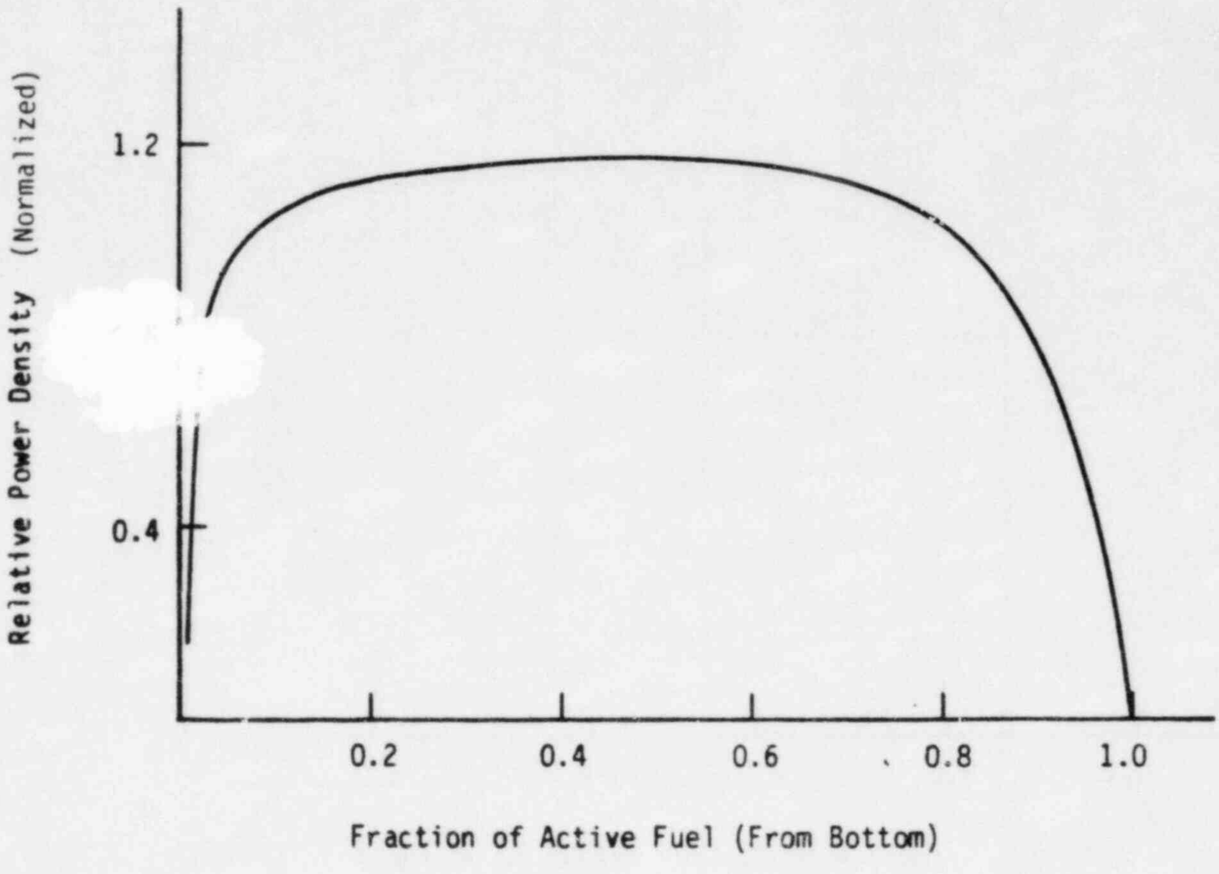

Figure 3.2 Axial Power Distribution

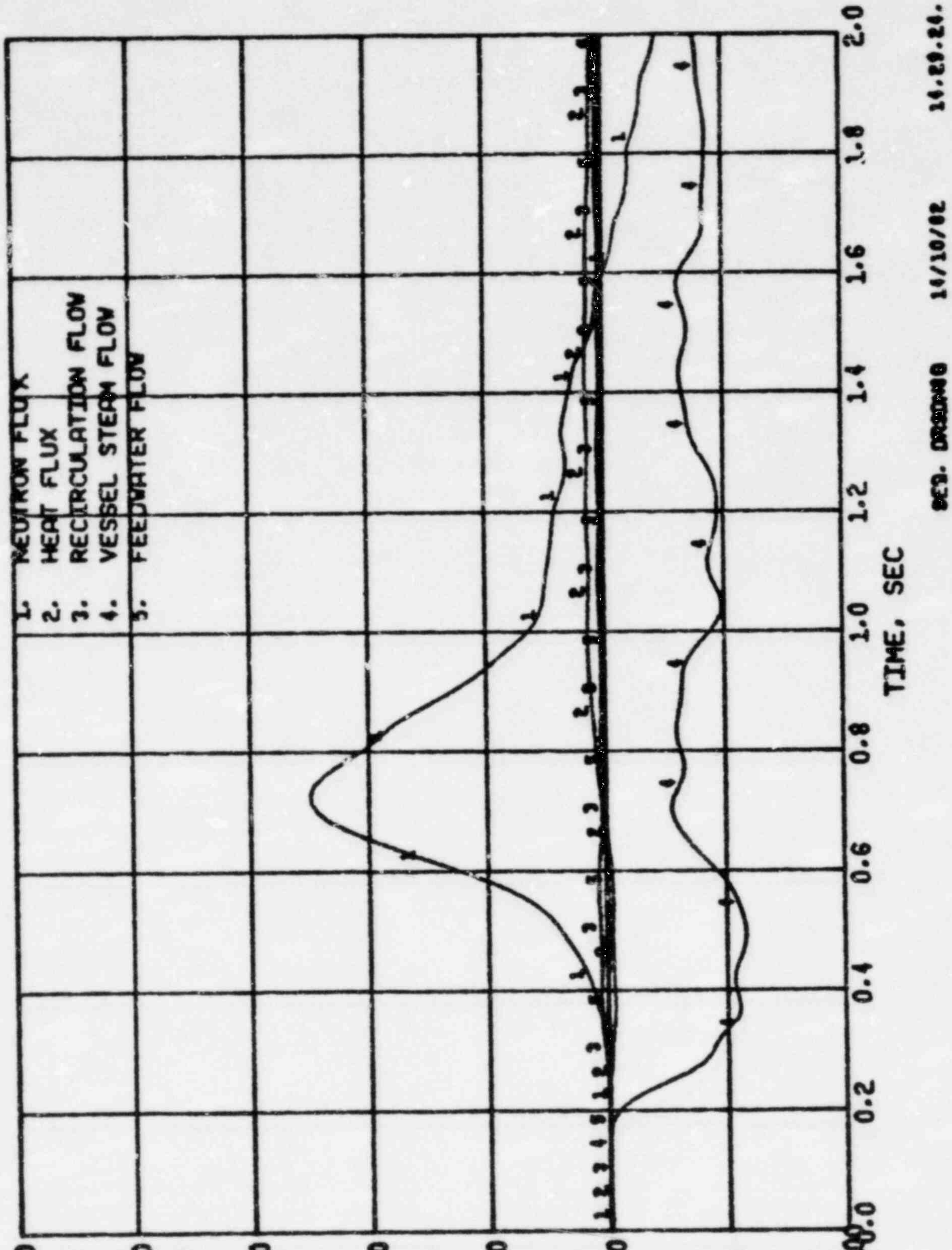

Figure 3.3 Generator Load Rejection w/o Bypass (Expected Powers and Flows)

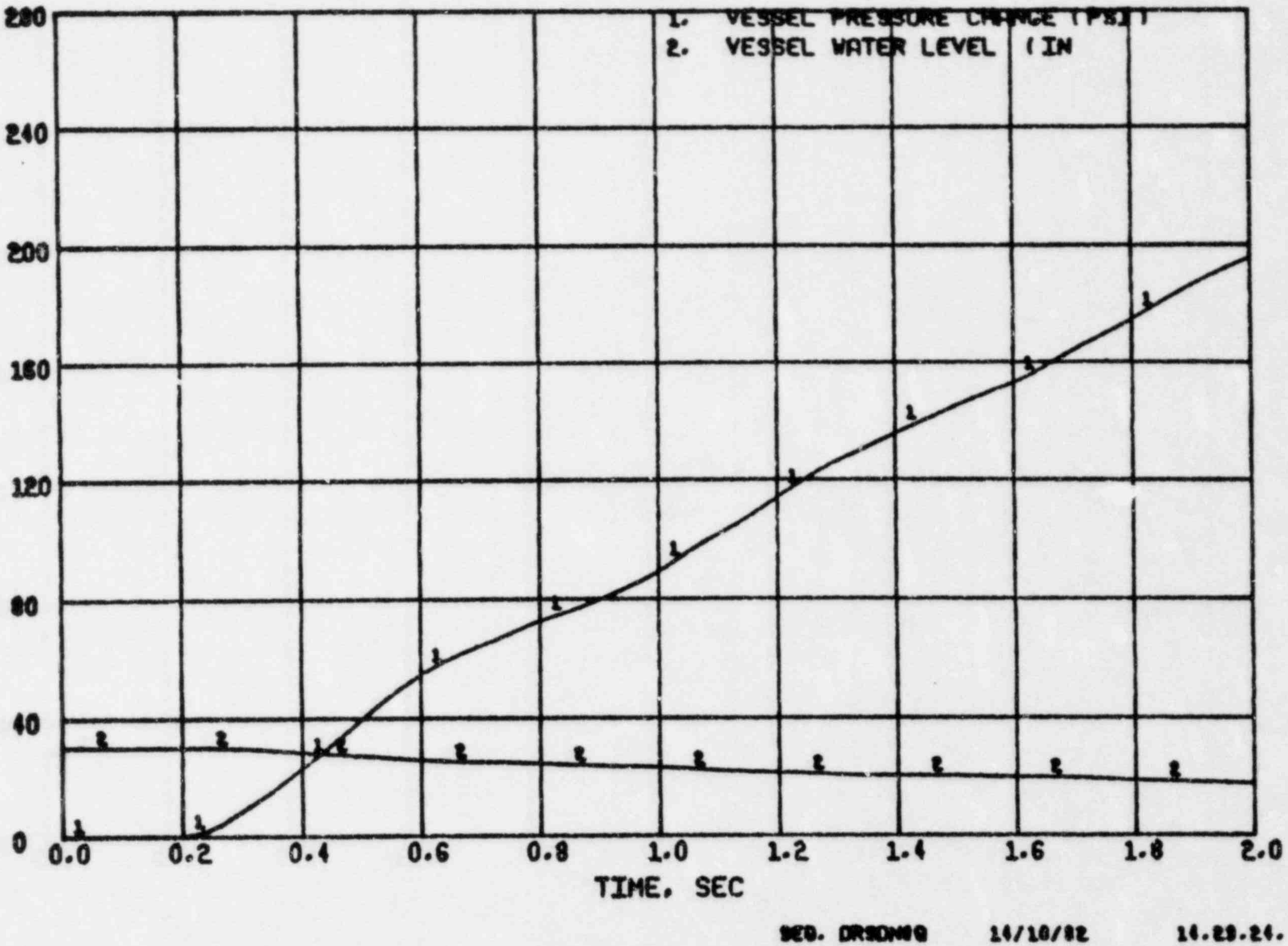

Figure 3.4 Generator Load Rejection w/o Bypass (Expected Vessel Pressure and Level)

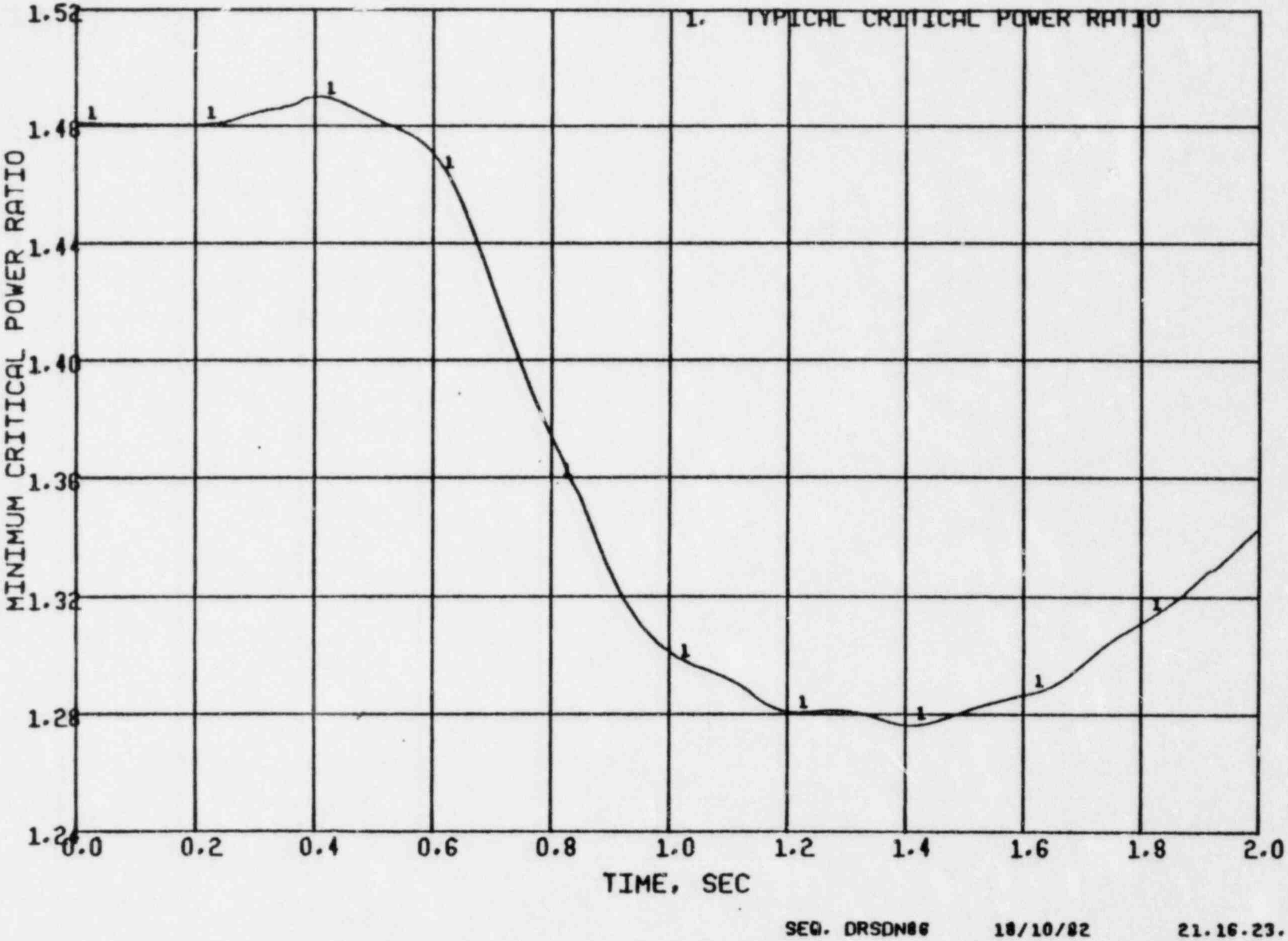

Figure 3.5 Generator Load Rejection w/o Bypass (Typical CPR)

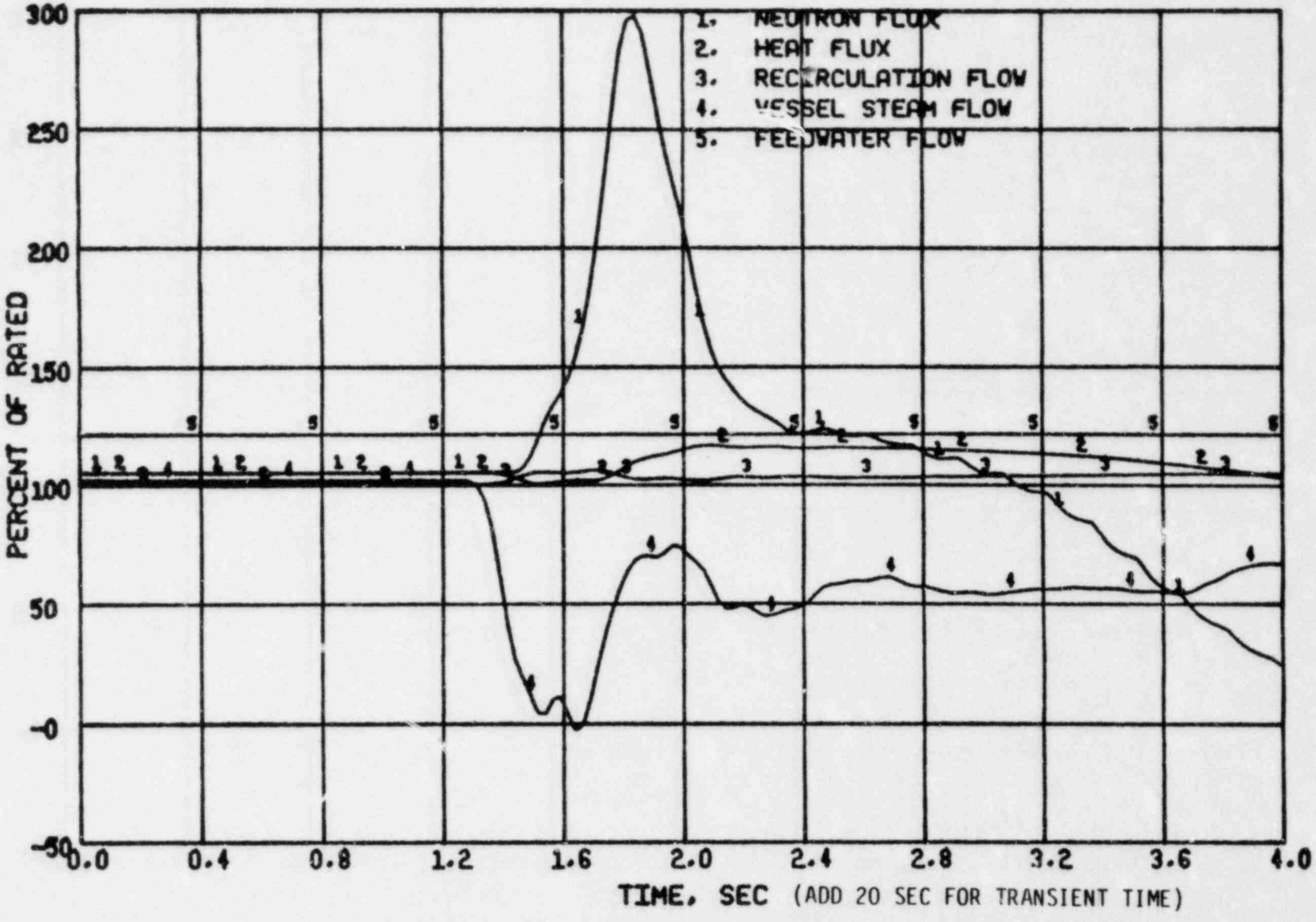

Figure 3.6 Increase in Feedwater Flow (Power and Flows from 20-24 Seconds)

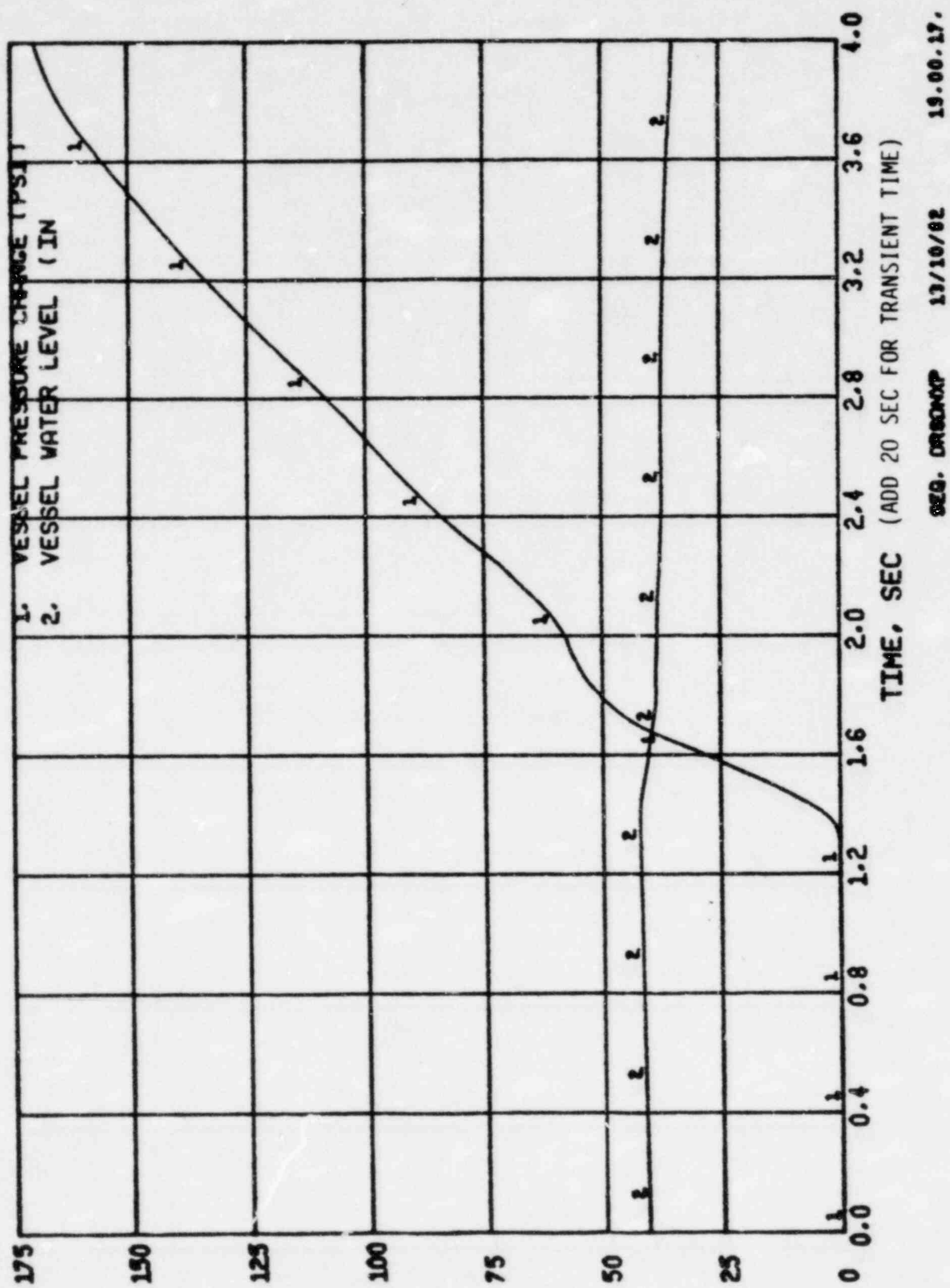

Figure 3.7 Increase in Feedwater Flow (Vessel Pressure and Level from 20-24 Seconds)

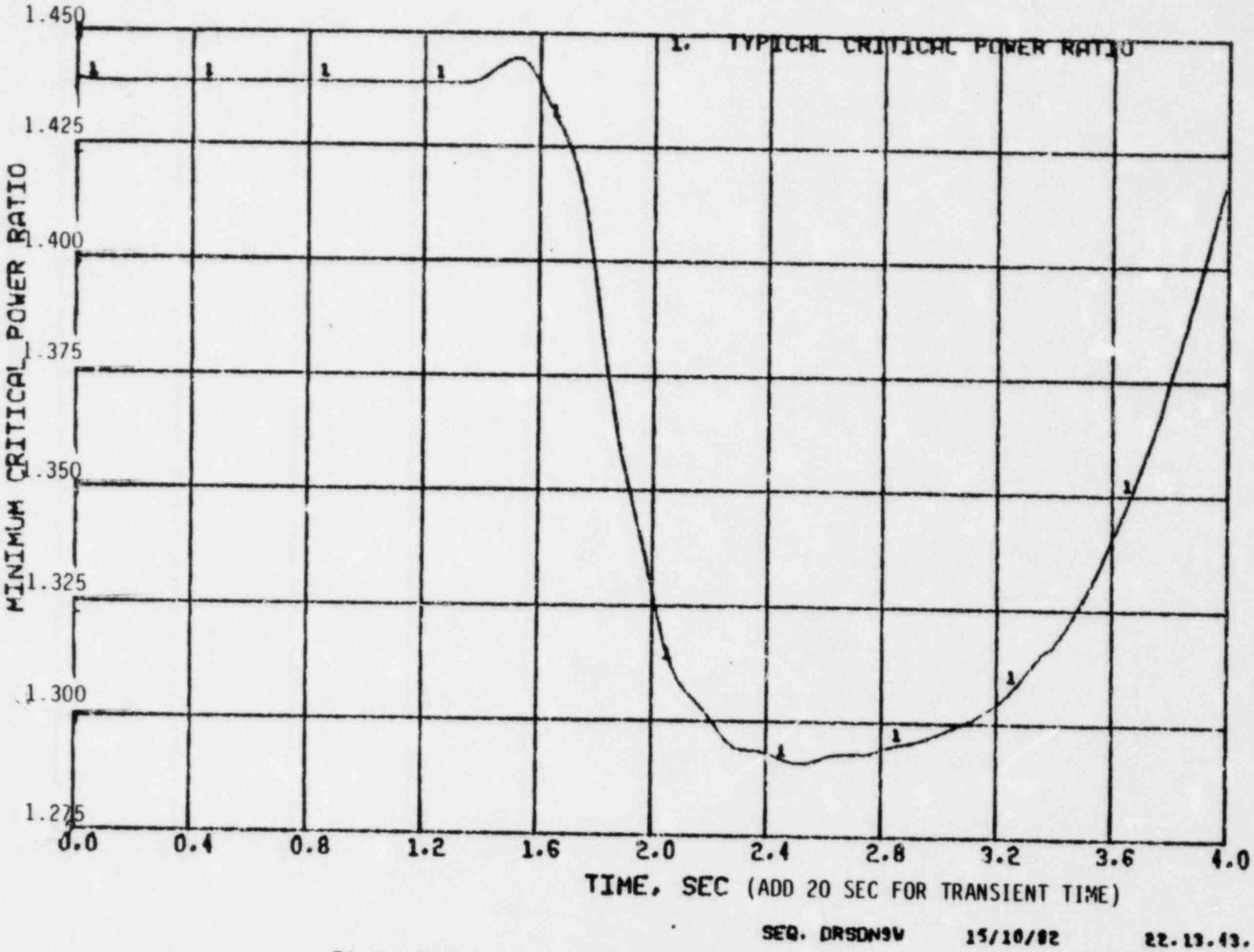

Figure 3.8 Increase in Feedwater Flow (Typical CPR)

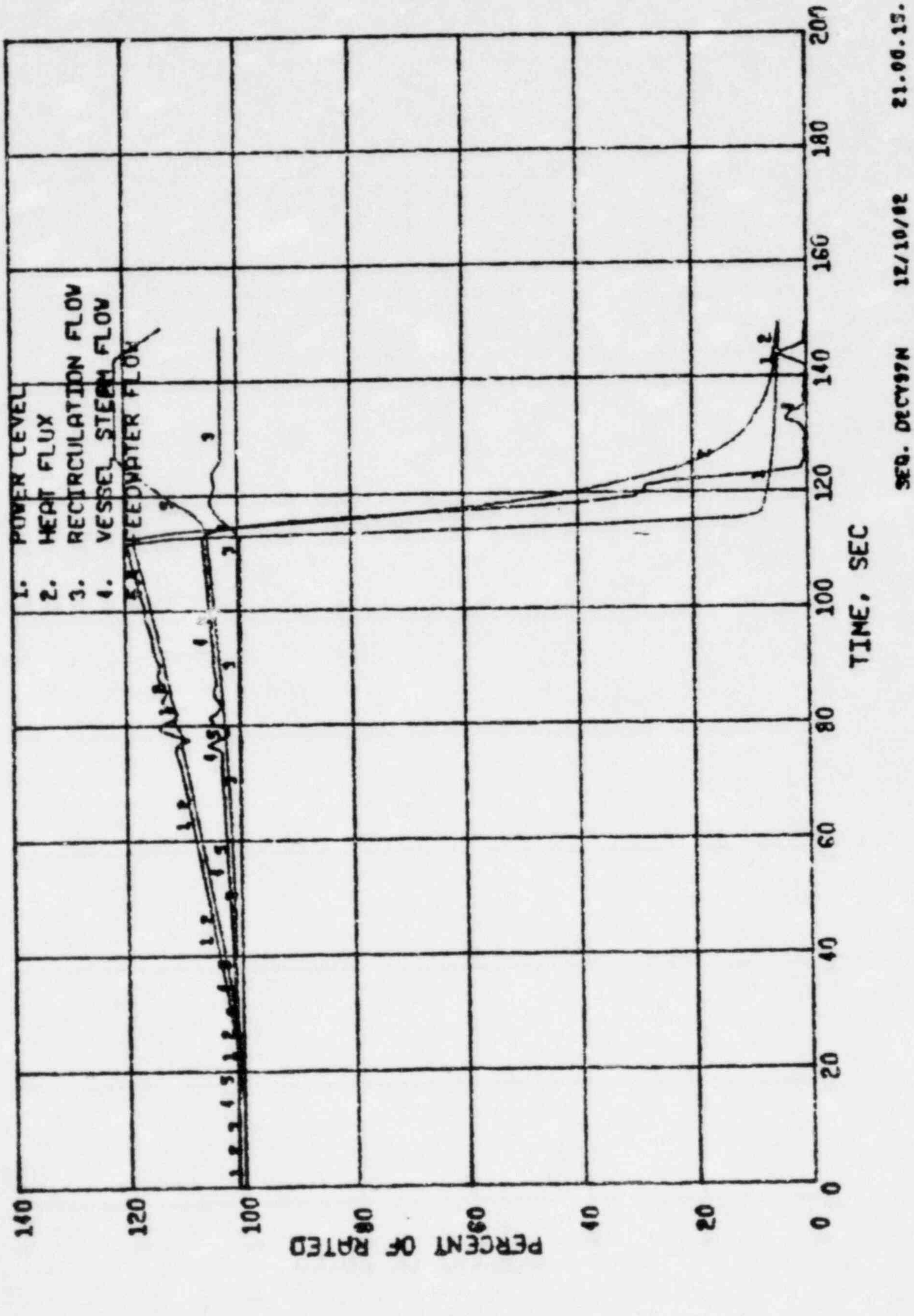

Figure 3.9 Loss of Feedwater Heating (Power and Flows)

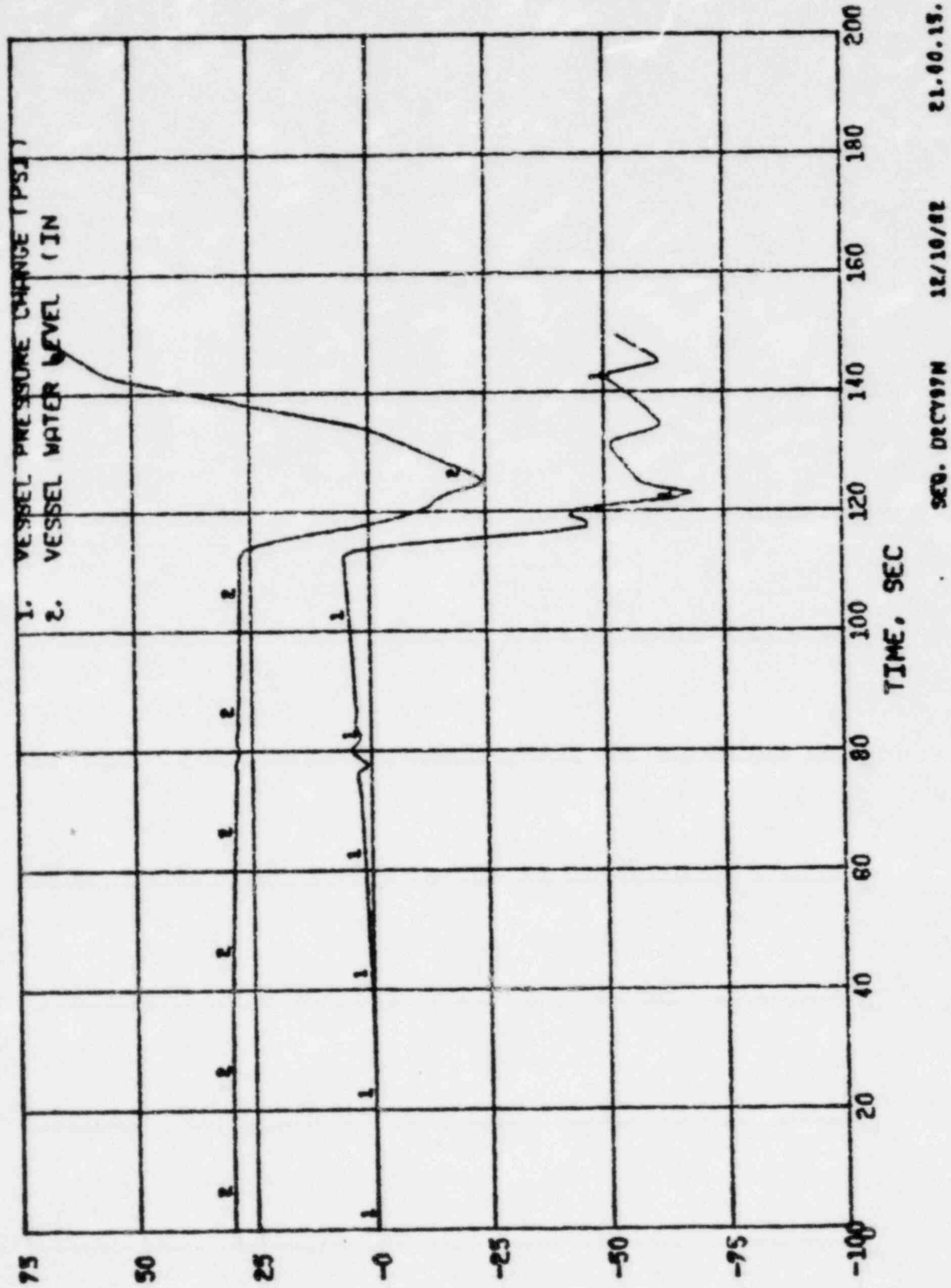

Figure 3.10 Loss of Feedwater Heating (Vessel Pressure and Level)

4.0 MAXIMUM OVERPRESSURIZATION

4.1 DESIGN BASIS

The reactor conditions used in the evaluation of the maximum pressurization event are those shown in Table 3.1. In addition to the conservative assumptions shown in Table 3.2, ENC assumed that the four power actuated relief valves were not available to vent steam as the ASME Pressure Vessel Code does not allow credit for power operated relief valves. Also, the most critical active component (scram on MSIV closure) was failed during the transient.

4.2 PRESSURIZATION TRANSIENTS

ENC has evaluated several pressurization events, and has determined that closure of all main steam isolation valves without direct scram is most limiting for maximum vessel pressure. Though the closure rate of the MSIVs is substantially slower than turbine stop or control valves, the compressibility of the fluid in the steam lines causes the severity of the compression wave of the slower closure to be nearly as great as the faster turbine stop or control valves closures. Essentially, the rate and magnitude of steam velocity reduction is concentrated toward the end of valve stroke, generating a substantial compression wave. Once the containment is isolated, the subsequent core power production must be absorbed in a smaller volume than if turbine isolation occurred. Calculations have determined that the overall result is to cause containment isolation to be more limiting than turbine isolation.

4.3 CLOSURE OF ALL MAIN STEAM ISOLATION VALVES

This calculation assumed all four steam lines were isolated at the containment boundary within 3 seconds. Due to the valve characteristics and steam compressibility, the vessel pressure response is not noted until about 3 seconds after beginning of valve stroke. Since scram performance was degraded to its Technical Specification limit for this analysis, effective power shutdown is delayed until after 5 seconds. Due to limitations in steam venting capacity, (i.e. power operated relief valves failures), significant pressure relief is not realized until after 5 seconds, preventing that mechanism from assisting in power shutdown. Thus, substantial thermal power production enhances the pressurization. Pressures reach the recirculating pump trip setpoint (1240 psig) before the pressurization has been reversed by the lifting of the safety valves. Loss of coolant flow leads to enhanced steam production as less subcooled water is available to absorb core thermal power. The maximum pressure calculated in the steam lines was 1337 psia occurring near the vessel at about 6.75 seconds. The maximum vessel pressure was 1364 psia occurring in the lower plenum at about 6.5 seconds. Figures 4.1 and 4.2 illustrate the progression of the transient.

The calculation was performed with ENC's advanced plant simulator code, COTRANSA, which includes a one-dimensional neutronics model.

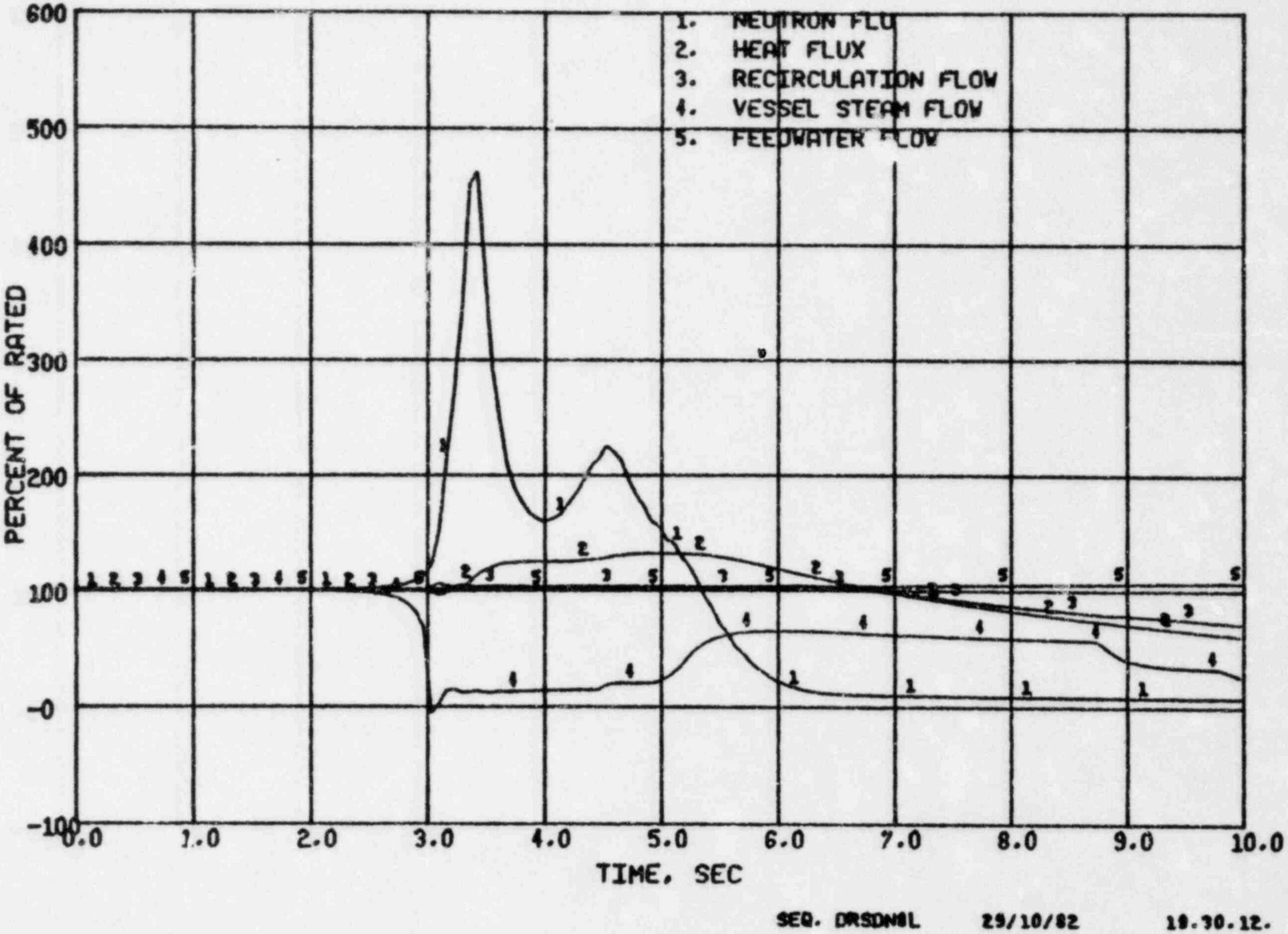

Figure 4.1 MSIV Closure without Direct Scram (Power and Flows)

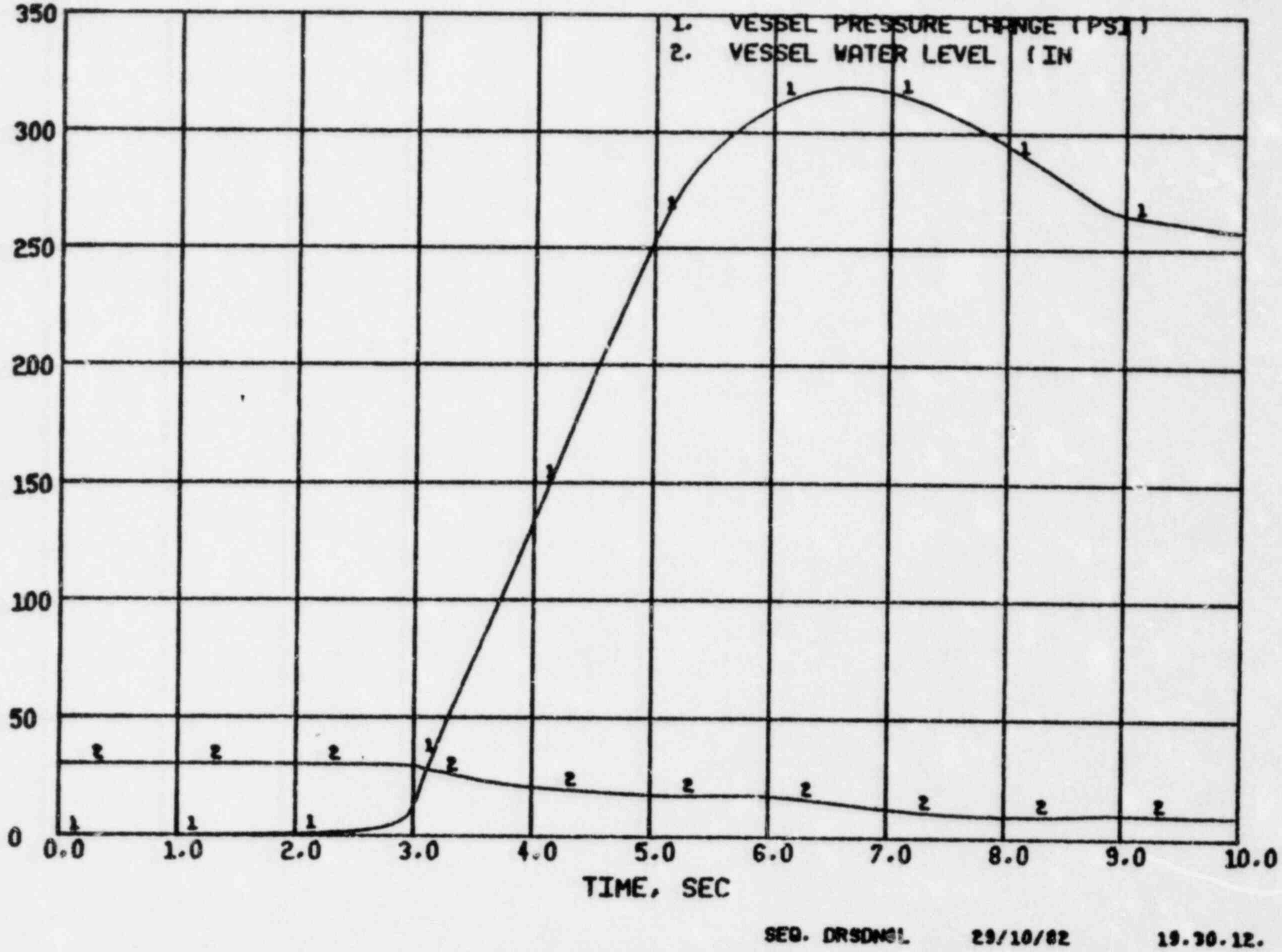

Figure 4.2 MSIV Closure without Direct Scram (Vessel Pressure and Level)

5.0 REFERENCES

(1) "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors", XN-NF-79-71(P), Revision 2, Exxon Nuclear Company Inc., Richland, Washington 99352, November 1981.

(2) "Dresden Unit 2 Cycle 9 Reload Analysis", XN-NF-82-77(P), Revision 1, Exxon Nuclear Company Inc., Richland, Washington 99352, December 1982.

(3) "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors", XN-NF-524(P), Exxon Nuclear Company Inc., Richland, Washington 99352, November 1979.

(4) "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option", XN-CC-33(A), Revision 1, Exxon Nuclear Company Inc., Richland, Washington 99352, November 1975.

(5) "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies", XN-NF-79-59(P), Exxon Nuclear Company Inc., Richland, Washington 99352, 1979.

(6) "Dresden-3 Cycle 8 Plant Transient Analysis Report", XN-NF-81-78, Revision 1, Exxon Nuclear Company Inc., Richland, Washington 99352, December 1981.

APPENDIX A

DRESDEN UNIT 2, CYCLE 9

SAFETY LIMIT CALCULATION PARAMETERS, INPUT VALUES, AND UNCERTAINTIES

A.1 REACTOR SYSTEM UNCERTAINTIES

The reactor system uncertainties used in the Dresden Unit 2 Cycle 9 safety limit calculation are the generic values listed in Table 5.1 of XN-NF-524(P)(3).

A.2 FUEL RELATED UNCERTAINTIES

Fuel related uncertainties used in the Dresden Unit 2 Cycle 9 safety limit calculation are listed in Table A-1. The values listed in Table A-1 are for Dresden Unit 2 Cycle 9 with the exception of the XN-3 correlation uncertainty, which is generic.

A.3 NOMINAL INPUT PARAMETER VALUES

Nominal values of input parameters used in the Dresden Unit 2 Cycle 9 safety limit calculation are listed in Table A-2.

A.3.1 RADIAL POWER HISTOGRAM

The radial power histogram used in the Dresden Unit 2 Cycle 9 safety limit calculation is given in Figure A-1. The radial power histogram was chosen from a representative group of histograms. The histogram was then biased in a manner which would produce a worse (larger) value of the predicted safety limit. The peak value for the histogram was chosen such that the limiting bundle MCPR would conservatively remain greater than the expected MCPR operating limit under steady-state, full-power, full-flow conditions.

A.3.2 LOCAL PEAKING DISTRIBUTION

The local peaking distribution used in the Dresden Unit 2 Cycle 9 safety limit calculation is shown in Figure A-2. The local peaking distribution was chosen from the predicted distributions covering the range of Dresden Unit 2 Cycle 9 exposures. The chosen distribution was used because it was found to produce the worst (largest) value of the safety limit of the group of distributions.

A.3.3 AXIAL POWER DISTRIBUTION

The axial power distribution used in the Dresden Unit 2 Cycle 9 safety limit calculation was:

$$F_A\ (X/L) = 0.30 + 1.10 \sin (\pi X/L)$$

where X/L = relative axial position. This axial power distribution was chosen because it is conservative with respect to the predicted axial power distributions of MCPR limiting bundles.

A.4 SAFETY LIMIT RESULTS

The final Dresden Unit 2 Cycle 9 safety limit calculation used 500 Monte Carlo trials. The MCPR of the safety limit calculation using the nominal input parameters was 1.05 or less for all fuel types. With those conditions, the number of rods in the core which are expected to avoid boiling transition is greater than 99.9%. Thus, a safety limit of 1.05 for Dresden Unit 2 Cycle 9 satisfies the requirement that at least 99.9% of the rods in the core must be expected to avoid boiling transition when the reactor is at the safety limit.

Table A-1 Dresden Unit 2 Cycle 9 Safety Limit Fuel Related Uncertainties

Parameter	Standard Deviation (% of Nominal)	Assumed Probability Distribution Type
XN-3 correlation	4.1	Normal
Assembly Radial Peaking Factor	5.18	Normal
Rod Local Peaking Factor	2.46	Normal
Assembly Flow Rate	2.8	Normal

Table A-2 Dresden Unit 2 Cycle 9 Safety Limit Nominal Input Parameters

Parameter	Value
Core Pressure	1035 psia
Core Power	3277 MW
Core Inlet Enthalpy	521.8 BTU/lbm
Total Core Flow	98.0 Mlbm/hr
Feedwater Temperature	$320^{o}F$
Feedwater Flow Rate	12.4 Mlbm/hr

Hydraulic Demand Curve*

$$G = 1.540 + (-8.851 \times 10^{-2}) \times LHGR + (1.908 \times 10^{-3}) \times LHGR^2 \quad \text{(8x8 fuel)}$$

where G = Assembly Mass Flux [$Mlbm/ft^2$-hr]

LHGR = Assembly power [kw/ft]

---

* Reference 2, Section 3.3

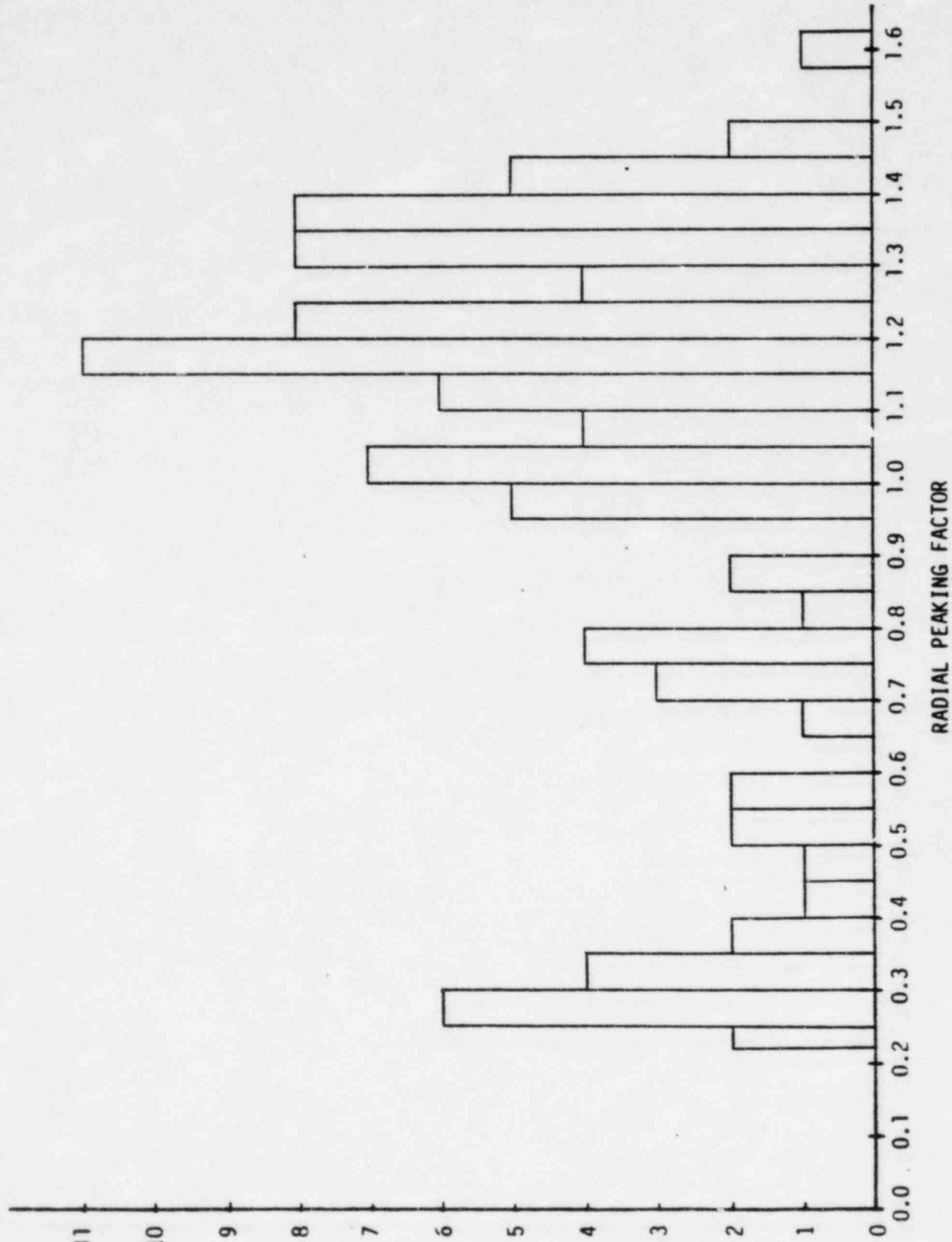

Figure A-1 Dresden-2 Cycle 9 Safety Limit Radial Power Histogram

L 1.06	ML 1.02	ML 0.92	M 1.08	M 1.06	M 1.08	ML 0.92	ML 1.01
ML 1.08	ML* 0.90	M 1.03	H 1.00	H 0.98	ML* 0.73	M 1.03	ML 0.92
ML 1.03	M 1.09	H 1.00	H 0.94	H 0.95	H 0.96	ML* 0.73	M 1.08
ML 1.01	M 1.06	H 0.97	H 0.94	W 0.00	H 0.95	H 0.98	M 1.06
ML 1.03	M 1.08	H 0.99	H 0.94	H 0.94	H 0.94	H 1.00	M 1.08
ML 1.08	ML 0.93	M 1.03	H 0.99	H 0.97	H 1.00	M 1.03	ML 0.92
L 1.07	ML* 1.01	ML 0.93	M 1.08	M 1.06	M 1.09	ML* 0.90	ML 1.02
LL 1.01	L 1.07	ML 1.08	ML 1.03	ML 1.01	ML 1.03	ML 1.08	L 1.06

W
I
D
E

W I D E

Figure A-2 Dresden-2 Cycle 9 Safety Limit Local Peaking

APPENDIX B

DRESDEN UNIT 2, CYCLE 9 EXPERIMENTAL DESIGN

The experimental design used in the construction of the response surface is provided in the attached Table B-1. This design is a Box-Behnken type for N=4 as described in XN-NF-81-22(P). The variables are defined as follows:

X1 - [ ]

X2 - [ ]

X3 - [ ]

X4 - [ ]

The coded values (+2, 0, -2) for the variables are as described in XN-NF-81-22(P), except that the standard deviations used are specific to Dresden Unit 2 (see Section 3.2.1 of main text). The sign of the coded variable values for the first three variables was chosen such that a positive value would increase the observed CPR. The opposite applies to the fourth (X4) variable.

Table B-1 ΔCPR Experimental Design

Coded Values of Predictor Variables

XN-NF-82-84(NP)
Revision 1

Issue Date: 12/07/82

PLANT TRANSIENT ANALYSIS FOR

DRESDEN UNIT 2 CYCLE 9

Distribution

J. C. Chandler

G. C. Cooke

N. F. Fausz

R. H. Kelley

L. C. O'Malley/Edison (60)

Document Control (5)