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DRESDEN UNIT 3 CYCLE 9
PLANT TRANSIENT ANALYSIS

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EXXON NUCLEAR COMPANY, Inc.

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1.0 INTRODUCTION

This report presents the results of Exxon Nuclear Company's (ENC) evaluation of core-wide transient events for Dresden Station Unit 3 during Cycle 9 operation. Specifically, the evaluation determines the necessary thermal margin limit required 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 Cycle 9 as previously determined for Cycle 8[6].

2.0 SUMMARY

The determination of the Minimum Critical Power Ratio (MCPR) for Dresden Unit 3 Cycle 9 was based upon the consideration of various possible operational transients(1). A MCPR of 1.30 or greater for all 8x8 fuel types during Cycle 9 assures that at least 99.9% of the fuel rods in the core will avoid boiling transition during a full load rejection without condenser bypass at worst case (end-of-cycle) conditions, as well as other less limiting anticipated operational transients. This analysis was based upon the current Dresden 3 Operating License and associated Technical Specifications. The MCPR operating limits required for the more potentially limiting events are shown in Table 2.1. These values are equal to those reported for Cycle 8.

The maximum system pressure has been calculated for the containment isolation event, which is a rapid closure of all main steam isolation valves without scram on valve position or relief through the four electromatic relief valves. The safety valves of Dresden Unit 3 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

DRESDEN UNIT 3, CYCLE 9

	CPR/MCPR		
Transient	8x8 (ENC) XN-1 & XN-2	8x8R(GE)	8x8(GE)
Generator Load Rejection (w/o bypass)	.25/1.30	.25/1.30	.25/1.30
Increase in Feedwater Flow	.21/1.26	.21/1.26	.21/1.26
Loss of Feedwater Heating	.16/1.21	.16/1.21	.16/1.21

Maximum Pressure (psig)*

Transient	Vessel Dome	Vessel Lower Plenum	Steam Lines
MSIV Closure	1323.1	1347.6	1324.1

*Limit allowed is 1375 psig

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	300%	112.5%	1273
Increase in Feedwater Flow	260%	115.9%	1196
Loss of Feed-water Heating	120%	118.5%	1039
MSIV Closure w/flux scram	490%	132.1%	1348

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(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 capability is conservative for cases where control rods are partially inserted or reactor power is less than rated. Following requirements established in the Plant Operating License and associated Technical Specifications, observance of the MCPR operating limit of 1.30 or greater for all 8x8 fuel types protects against boiling transition during all anticipated transients at the Dresden Unit 3 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 3. 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 3.1, 3.2 and 3.3 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 calculated that the thermal margin (ΔCPR) required to prevent boiling transition for the nominal case for Cycle 9 was slightly less than previously calculated for the same case for Cycle 8. ENC had calculated this event for Cycle 8 to determine a ΔCPR which would not be exceeded in 95% of the possible outcomes of the event when four variables were considered:

- void reactivity
- scram worth
- control rod speed (average of all rods)
- scram time delay.

The standard deviations of the first two variables were 5% of their expected value. The standard deviations of the latter two variables were based upon plant test data:

(1) Average rod speed - one standard deviation equals 10.4 centimeters/sec.

(2) Scram time delays - one standard deviation equals 30 millisecs.

In the evaluation of Cycle 9, the cycle dependent neutronic and thermal hydraulic parameters were considered along with potential changes in control rod performance since the Cycle 8 analysis. While measured control rod speed and scam time delay slightly deteriorated between cycles, the void reactivity for Cycle 9 was less negative. The overall result was calculated ΔCPR's for Cycle 9 being slightly less than calculated for Cycle 8. Since neither the mean or standard deviation of ΔCPR is expected to be greater for Cycle 9, the calculated results for Cycle 8 are retained:

mean ΔCPR	.22
standard deviation	.016
95% ΔCPR	.250

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 will rise until the high vessel water trip setting is exceeded. To protect against spillover of subcooled water to the turbine, the turbine trips, with resultant 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 3.4, 3.5 and 3.6 display critical variables for this event.

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^{O}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% of rated before surface heat flux increased nearly as much. For conservatism, the thermal margin calculation assumed that the heat flux increased 120%, resulting in a predicted ΔCPR of 0.16 for each fuel type. Figures 3.7 and 3.9 depict the transient progression.

3.3 CALCULATIONAL MODEL

The plant transient model used to evaluate the load rejection and feed water 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 (CPR) of the limiting anticipated operational occurrence to the safety limit.

The safety limit for Dresden Unit 3 Cycle 9 was determined by the methodology presented in Reference 3, and used to determine the MCPR safety limit for Cycle 8 operation of Dresden Unit 3, to have the following value:

Dresden Unit 3 Cycle 9 MCPR Safety Limit = 1.05.

The input parameter values and uncertainties used to establish the safety limit are as presented in Reference 6.

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)	87.6
Core Bypass Flow (Mlb/hr)	10.4
Core Inlet Enthalpy (BTU/lbm)	522.3
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)	304.1
Recirculating Pump Flow (Mlb/hr)	17.1 [1]

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[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	298 msec (maximum)
Time to Sense Fast Turbine Control Valve Closure	80 msec (maximum)
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	±10 inches
High Level Trip	48 inches
Maximum Feedwater Runout Flow (3 pumps)	4966 lbm/sec
Maximum Feedwater Runout Flow (2 pumps)	3310.67 lbm/sec
Doppler Reactivity Coefficient (nominal)	-0.00228\$/°F/void fraction
Void Reactivity Coefficient (nominal)	-15.81\$/void fraction
Scram Reactivity Worth	-36.639\$*
Axial Power Distribution (Peak/average)	1.191 at x/L = .375
Delayed Neutron Fraction	.00519
Prompt Neutron Lifetime	4.66×10^{-5} sec
Recirculating Pump Trip Setpoint	1240 psig (vessel pressure)

* The value used in the analysis was 80% of the nominal

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

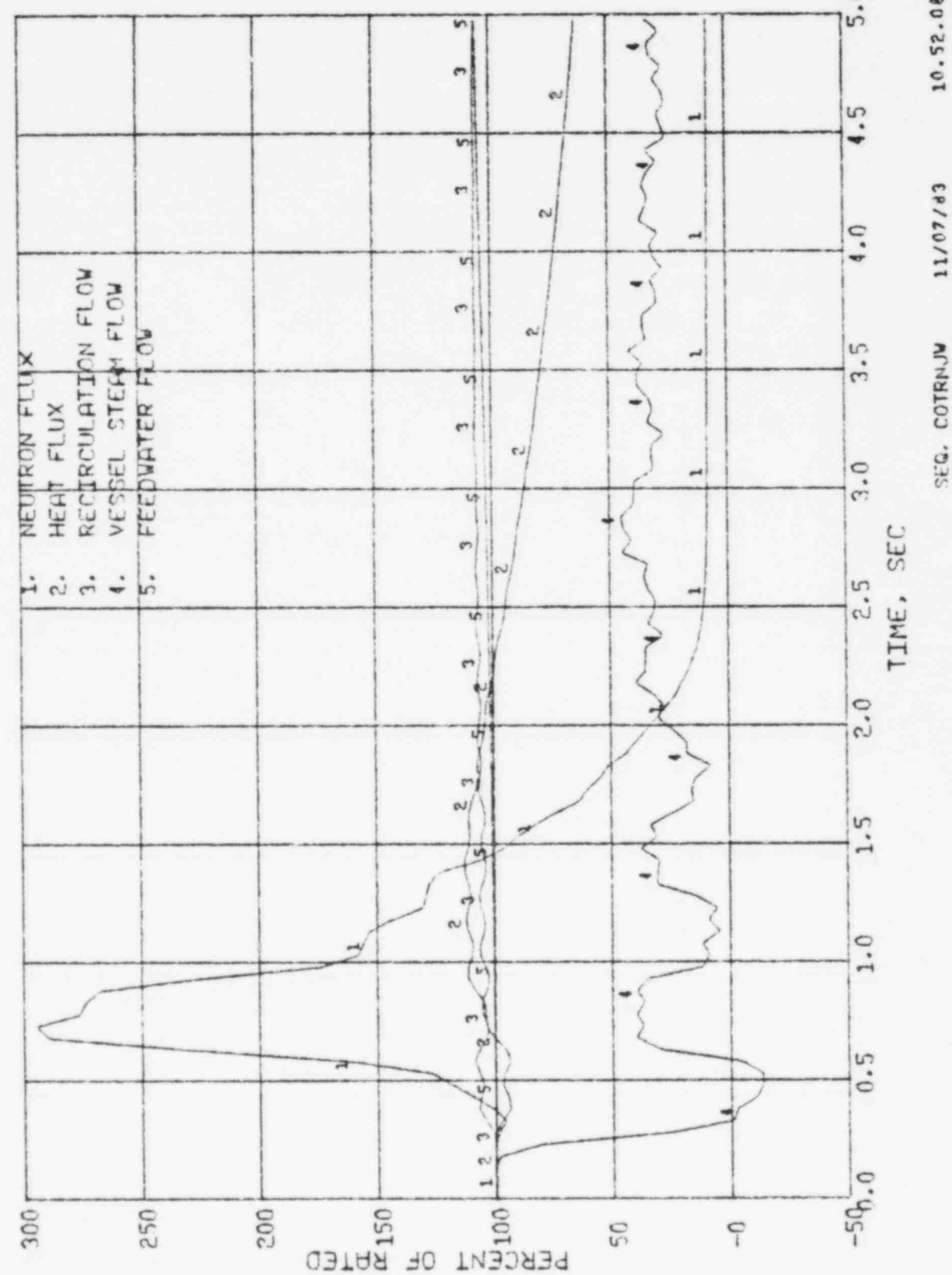

Figure 3.1 Generator Load Rejection w/o Bypass (Expected Power and Flows)

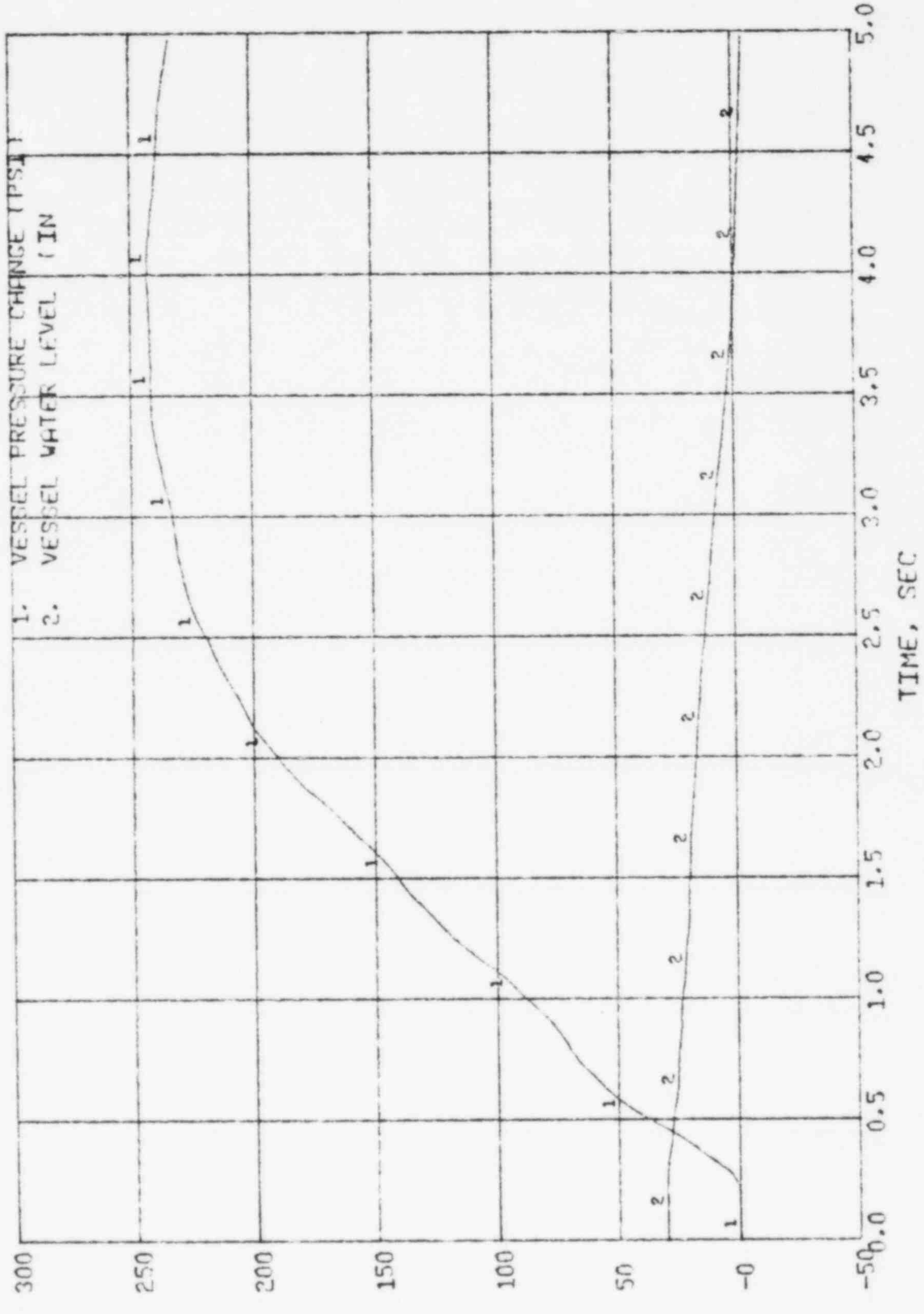

Figure 3.2 Generator Load Rejection w/o Bypass
(Expected Vessel Pressure and Level)

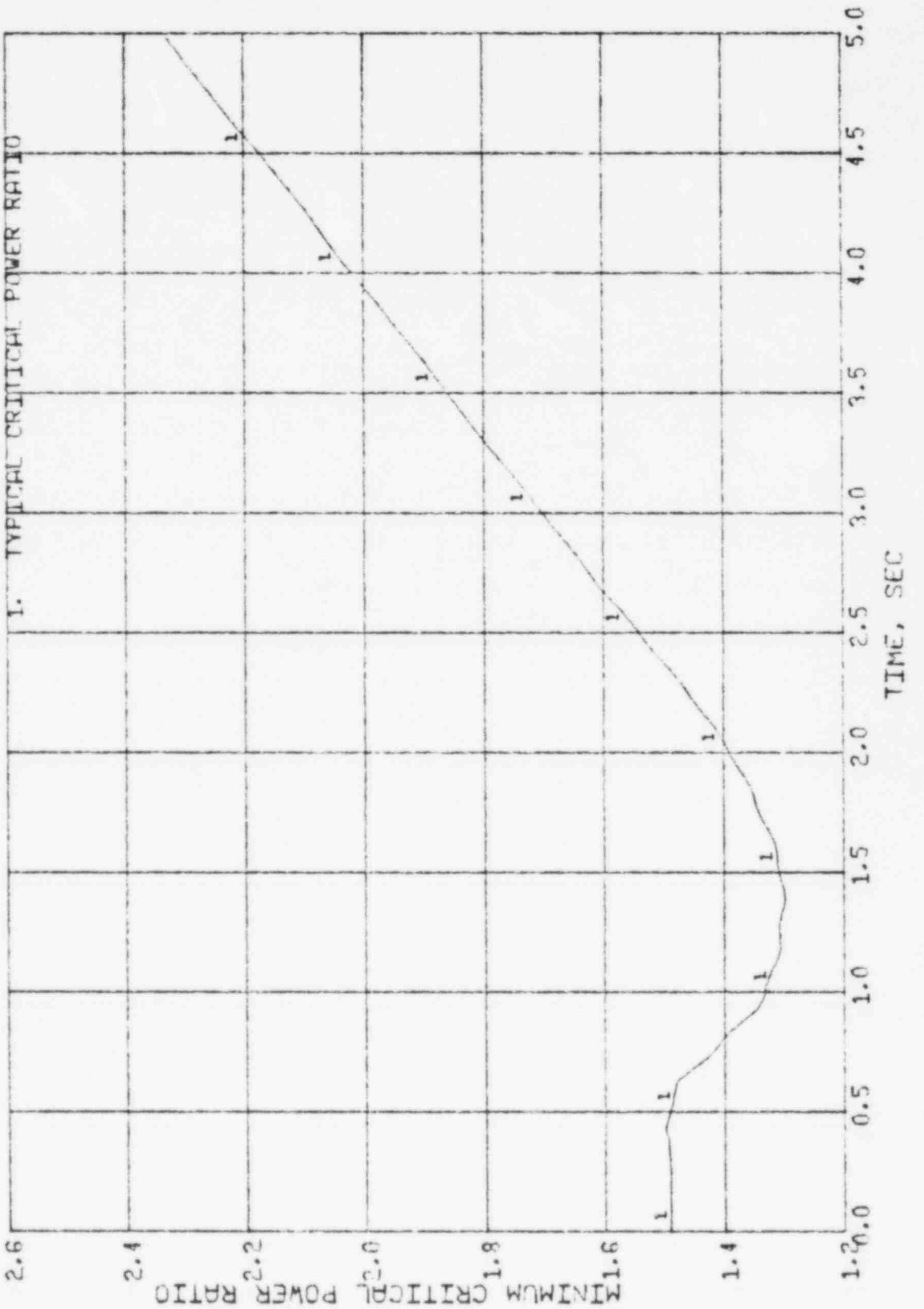

Figure 3.3 Generator Load Rejection w/o Bypass
(Expected CPR for a Typical Fuel Assembly)

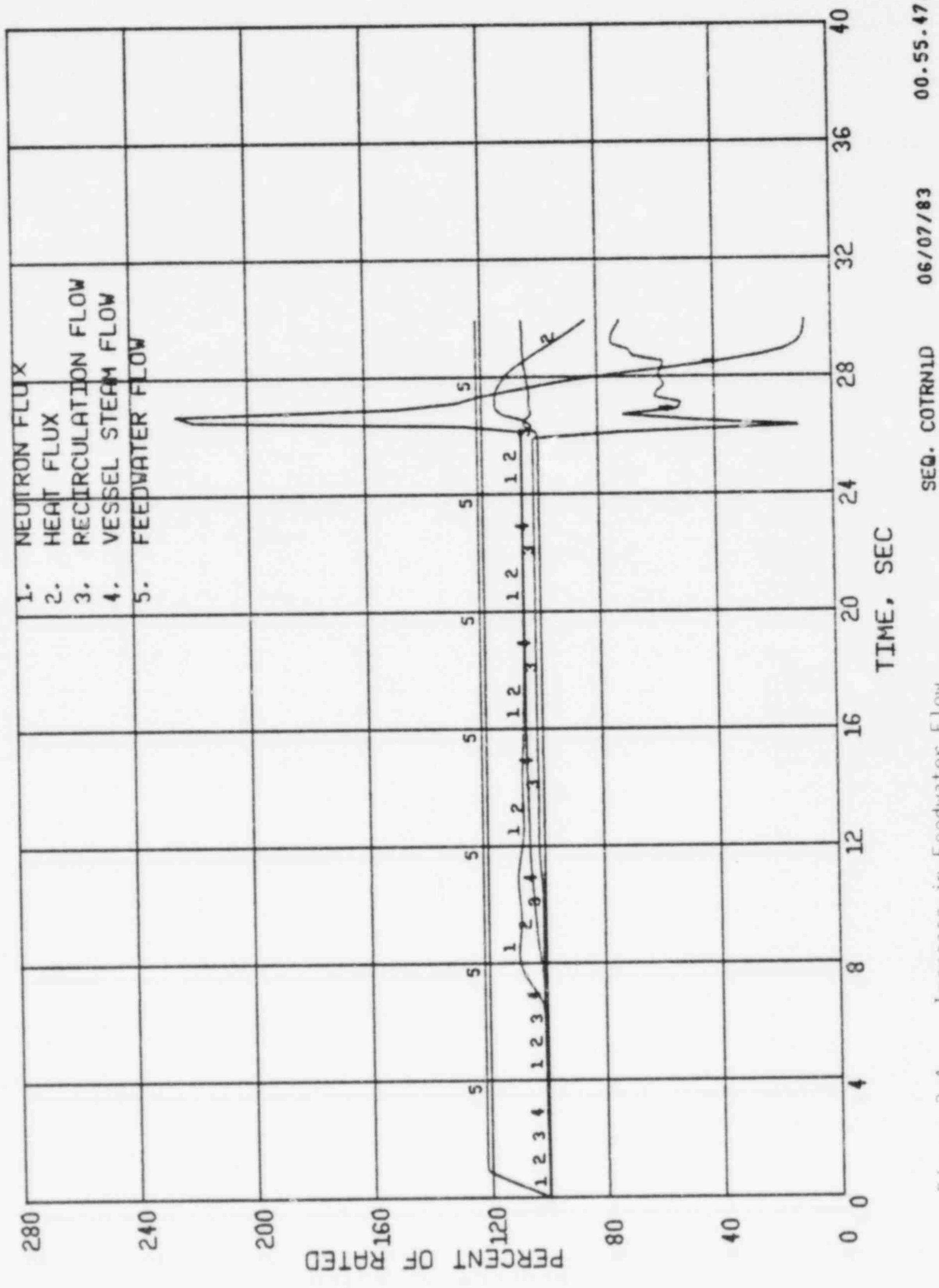

Figure 3.4 Increase in Feedwater Flow (Power and Flows)

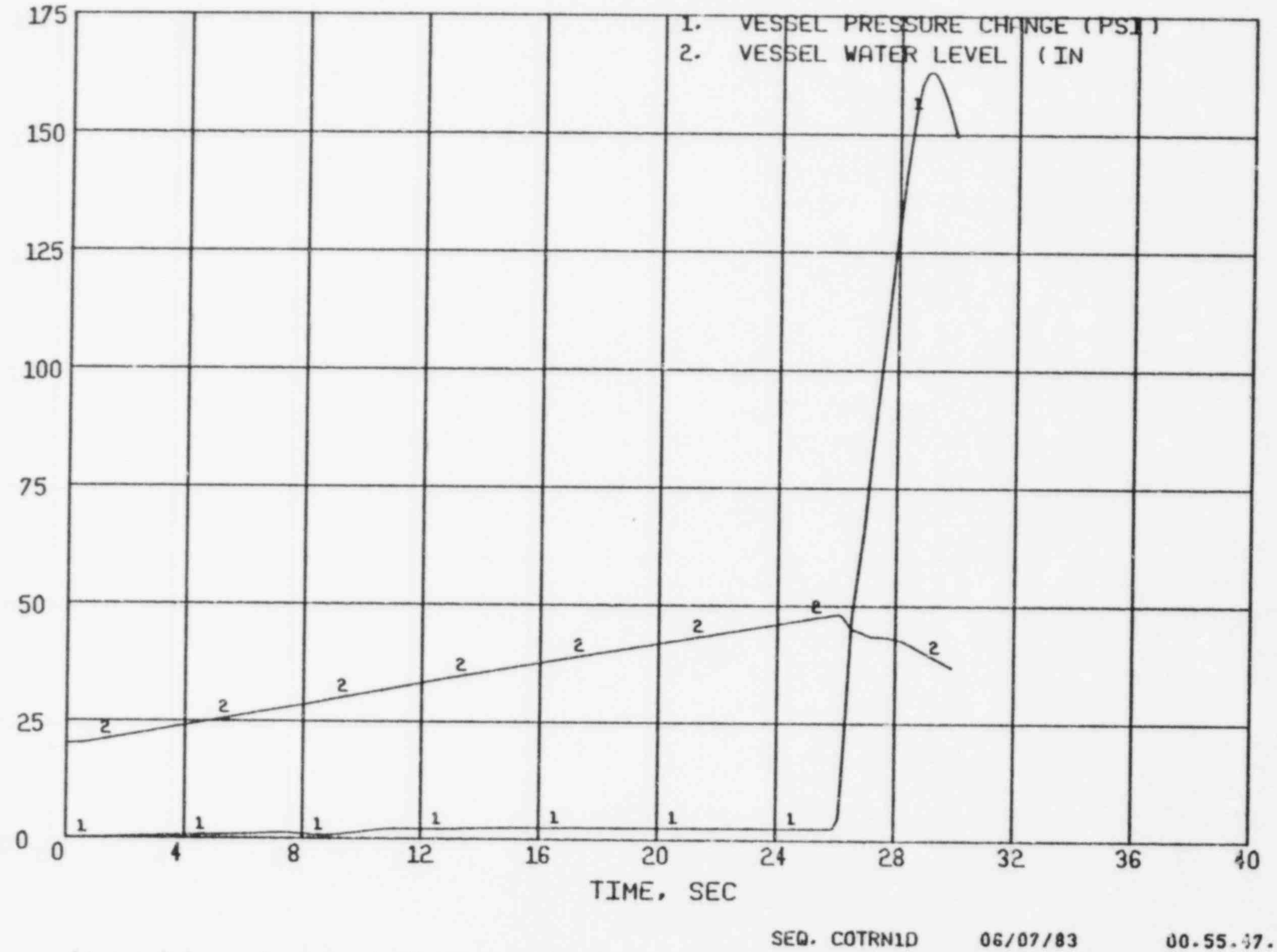

Figure 3.5 Increase in Feedwater Flow (Vessel Pressure and Level)

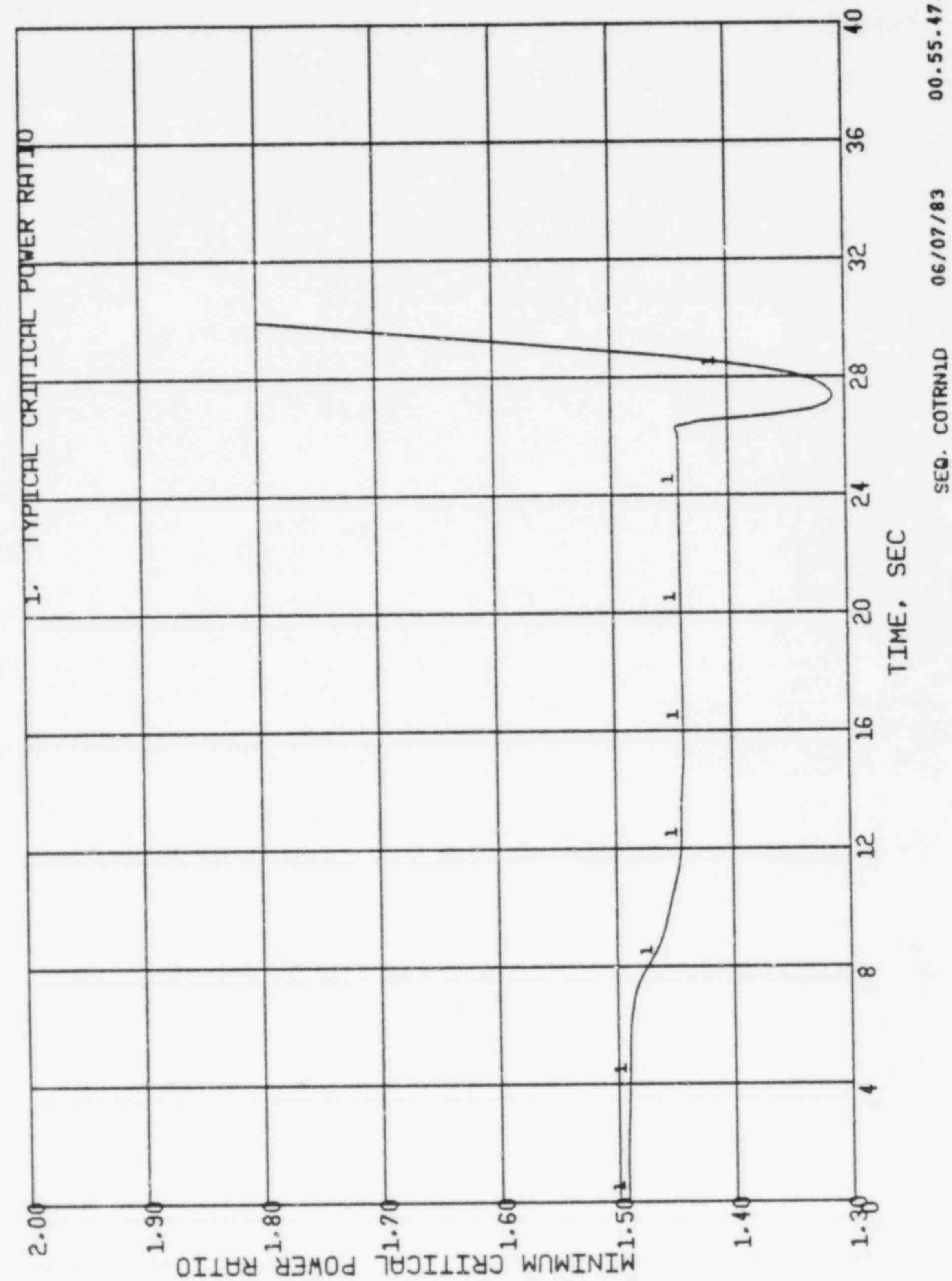

Figure 3.6 Increase in Feedwater flow (Typical CPR)

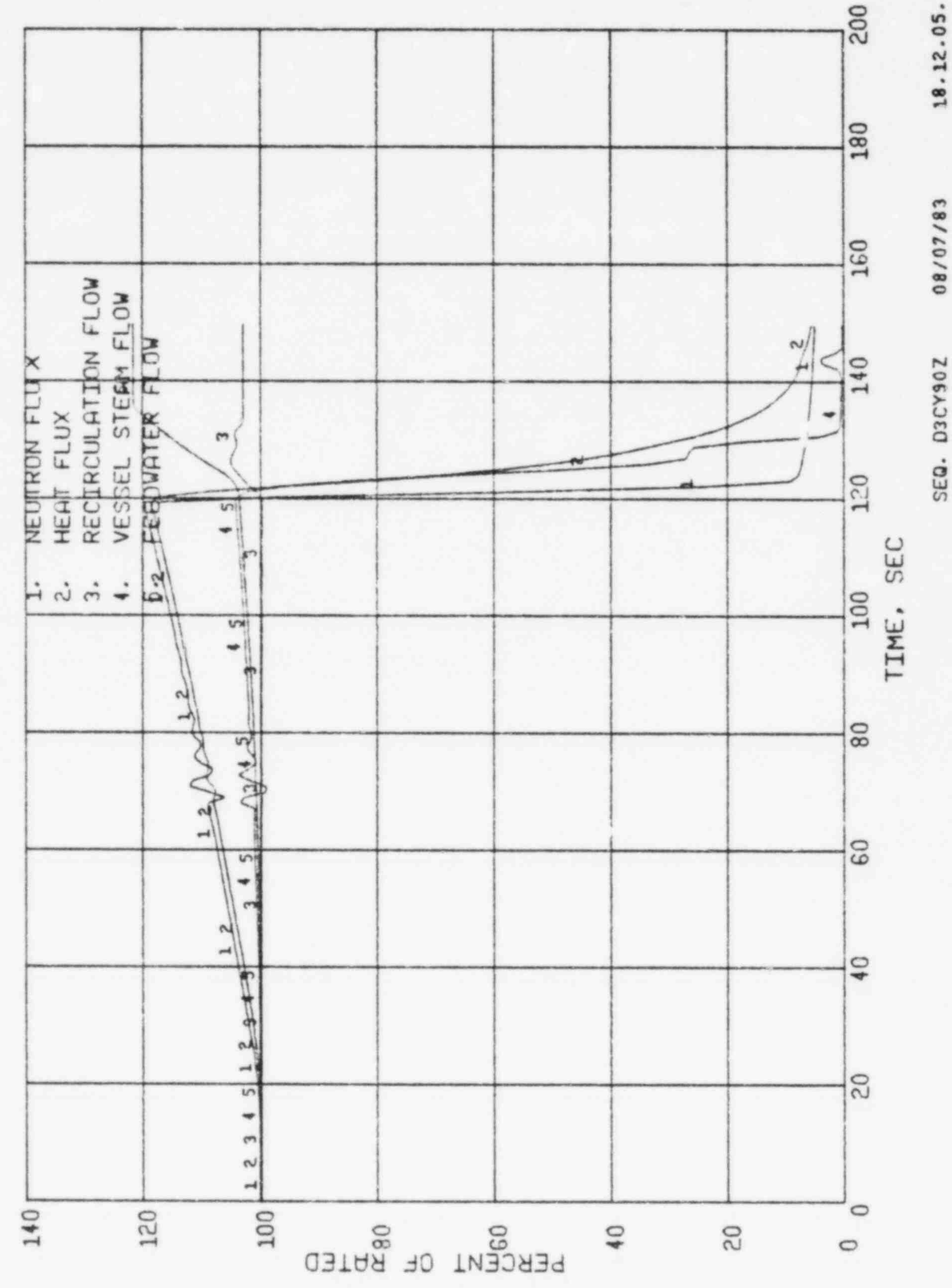

Figure 3.7 Loss of Feedwater Heating (Power and Flows)

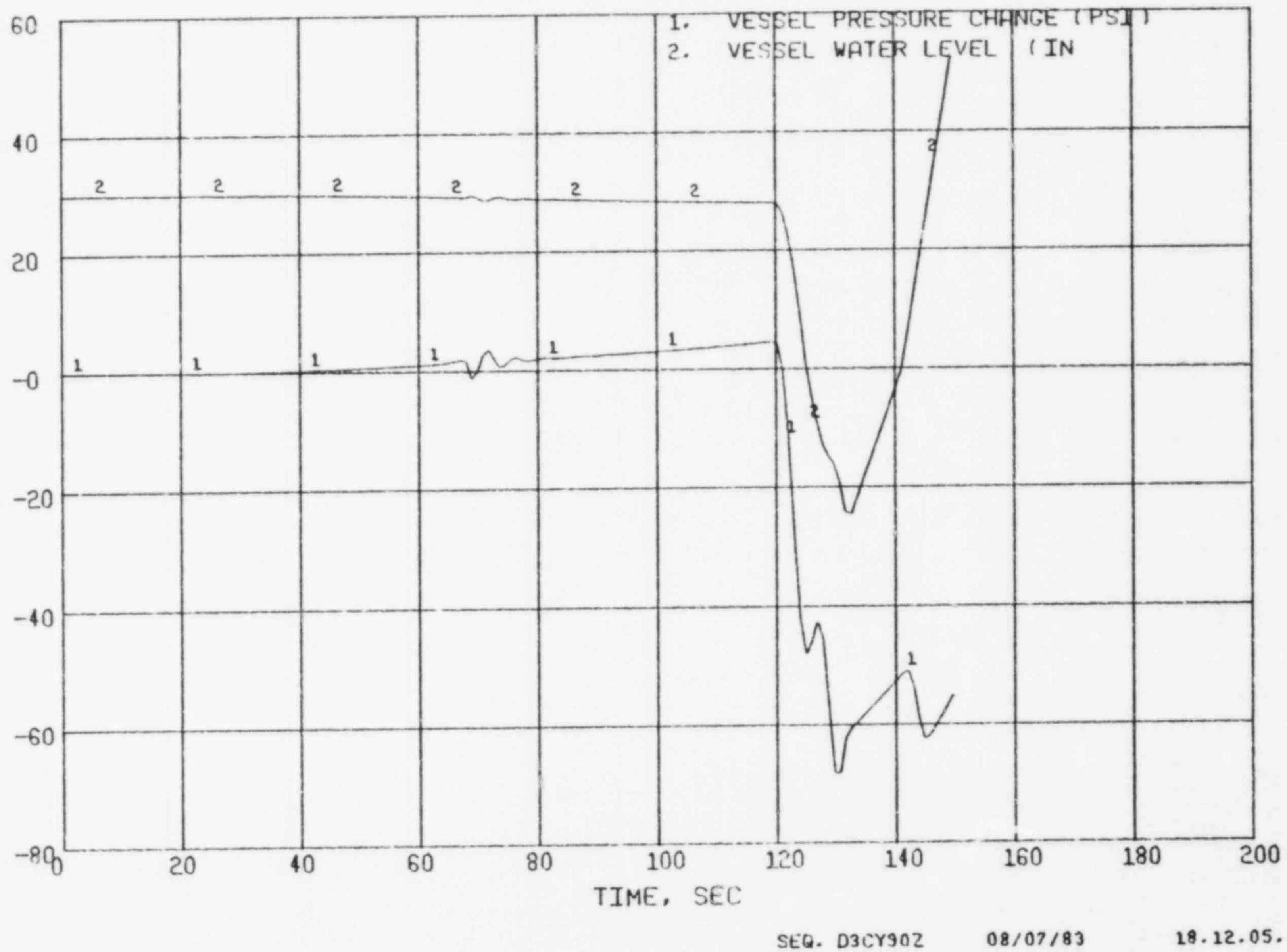

Figure 3.8 Loss of Feedwater Heating (Vessel Pressure and Level)

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 the 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 1324.1 psig occurring near the vessel at about 6.75 seconds. The maximum vessel pressure was 1347.6 psig 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.

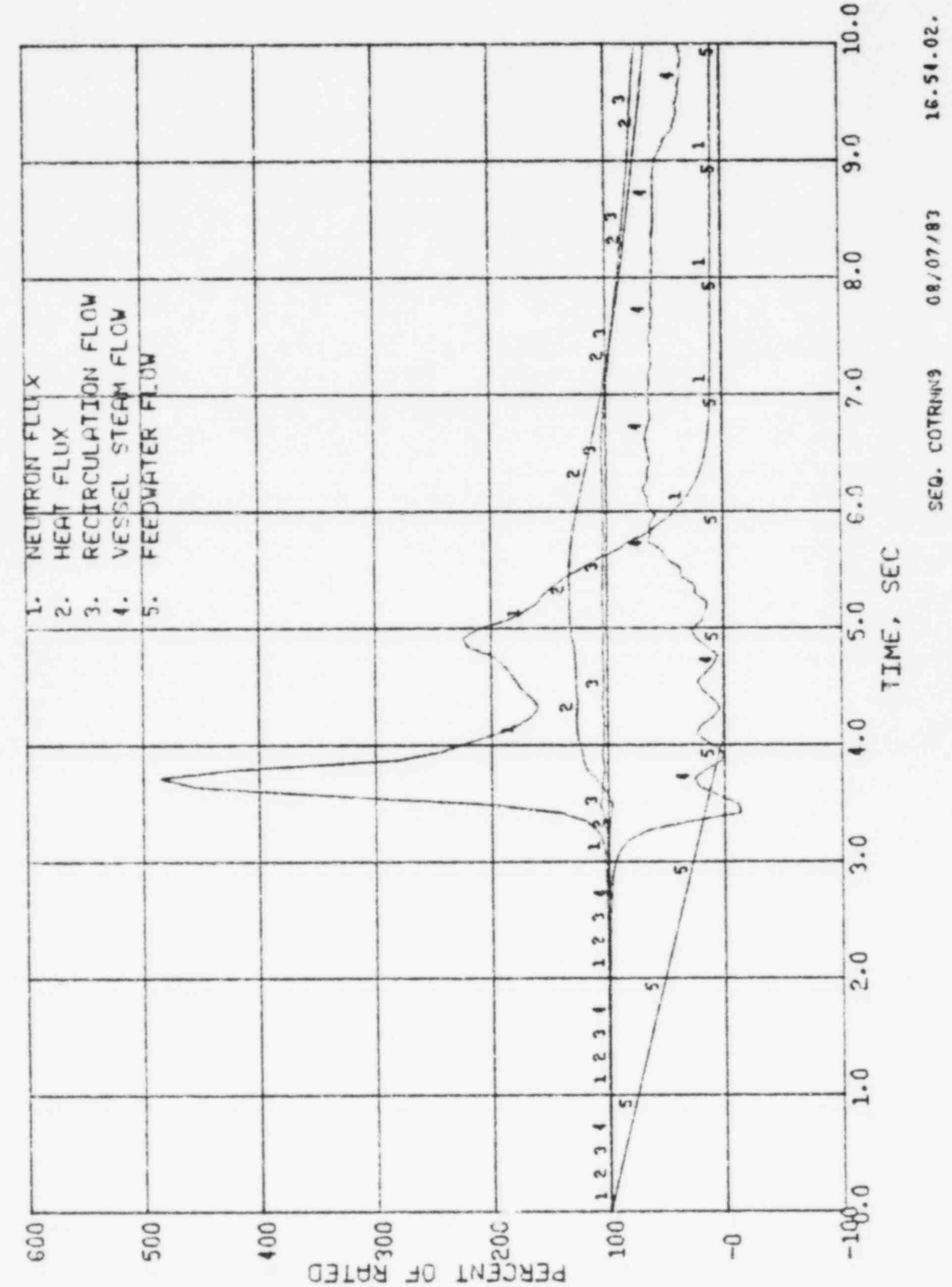

Figure 4.1 MSIV Closure Without Direct Scram (Power and Flows)

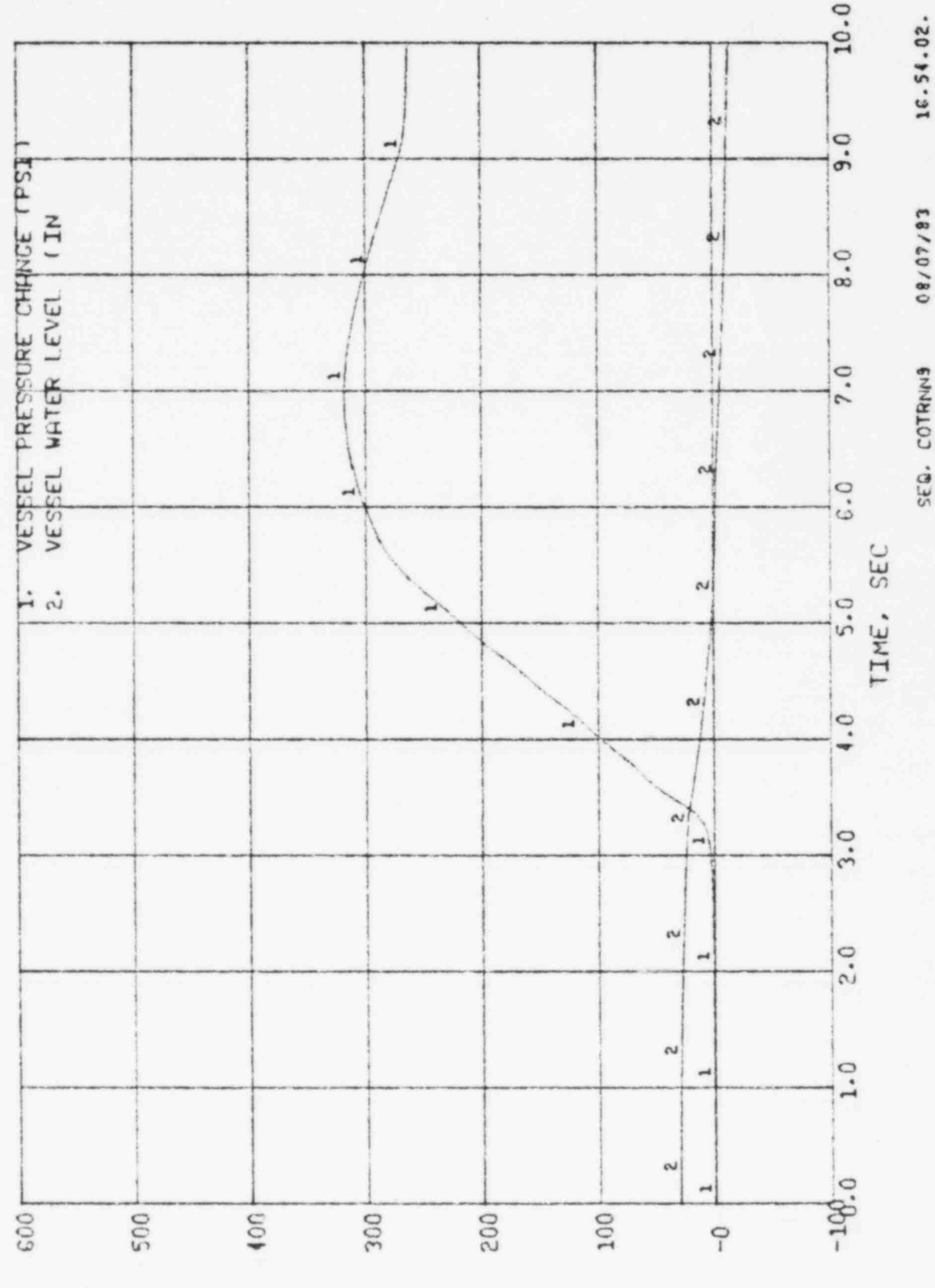

Figure 4.2 MSIV Closure Without Direct Scram
(Vessel Pressure and Level)

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 3 Cycle 9 Reload Analysis", XN-NF-83-47, Exxon Nuclear Company Inc., Richland, Washington 99352, August, 1983.

(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.

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