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REVISION 1

ADVANCED NUCLEAR FUELS CORPORATION

WNP-2 CYCLE 5 PLANT TRANSIENT ANALYSIS

MARCH 1989
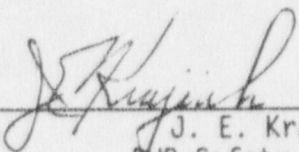
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WNP-2 CYCLE 5 PLANT TRANSIENT ANALYSIS

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SUMMARY OF REVISIONS

Revision 1 to ANF-89-01 was issued to address a reload batch size change from 144 to 136 assemblies and minor text changes which describe the reload batch size change. Feedwater controller failure calculated results at 47% power and 106% flow with normal scram speed and recirculation pump trip are also included for a 144 assembly reload batch size.

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ACKNOWLEDGMENT

1.0 INTRODUCTION

This report presents the results of the Advanced Nuclear Fuels Corporation (ANF) evaluation of system transient events for the Supply System Nuclear Project Number 2 (WNP-2) during Cycle 5 operation. Initially, the analysis the Cycle 5 core was assumed to contain 572 ANF 8x8 and 192 GE P8x8R fuel assemblies. This document has been revised, at the request of the Washington Public Power Supply System (WPPSS), to reflect a revised Cycle 5 core with eight fewer ANF assemblies or 564 ANF 8x8 and 200 GE P8x8R fuel assemblies. Since the load rejection without bypass (LRNB) is the limiting pressurization event, only the LRNB event with normal scram speed (NSS) and recirculation pump trip (RPT) operable was recalculated for the revised core loading.

This evaluation together with the analysis of final feedwater temperature reduction[1] (FFTR) and the analysis of core transient events[2] determines the necessary thermal margin (MCPR limits) to protect against boiling transition during the most limiting anticipated operational occurrence (AOO). The evaluation also demonstrates the vessel integrity for the most limiting pressurization event. This evaluation is applicable for core flows up to the maximum attainable with the recirculation flow control valve in its fully open position which is 106% of the rated core flow value at 100% power. The methodology used for these system transient analyses is detailed in References 3 and 4.

2.0 SUMMARY

The Minimum Critical Power Ratios (MCPR) calculated to protect against boiling transition during potentially limiting plant system transient events are shown in Table 2.1 for powers that bound allowable values. This table shows the LRNB results for the original and revised reload batch sizes. The system transient MCPR values of Table 2.1 for the LRNB and feedwater controller failure (FWCF) transients were obtained using a scram time based on WNP-2 measured values. The loss of feedwater heating (LOFH) transient results shown in Table 2.1 were obtained from a bounding analysis which is discussed in Section 3.2.3. The limiting AOO values for the cases of Table 2.1 are for the LRNB transient at End of Cycle (EOC) conditions; the limiting MCPR values are 1.34 for GE fuel and 1.31 for ANF fuel for the original reload batch size and 1.35 for GE fuel and 1.31 for ANF fuel with the revised reload batch size.

For previous WNP-2 cycles, ANF performed an analysis for the LRNB event at a cycle exposure of EOC -2000 MWd/MTU. Prior to the end of cycle, a large number of control blades are still inserted in the core. These analyses showed that this LRNB system transient was bounded by the control rod withdrawal event (CRWE) by a substantial margin. Thus, for the earlier cycles, plant operating limits were always based on the CRWE for cycle exposures up to EOC -2000 MWd/MTU. Based on this prior experience, the Cycle 5 MCPR limit up to EOC -2000 has been determined only by the CRWE.[2] Thus, the Cycle 5 CRWE defined MCPR limit is applicable up to EOC -2000 MWd/MTU, and for exposures beyond EOC -2000 MWd/MTU the limits in Table 2.1 are applicable.

Additional transient analyses were performed assuming the recirculation pump trip (RPT) was out of service, and using the technical specification scram speed (TSSS) and the results are reported herein. The critical power results for these events are presented in Section 3.0.

The maximum system pressure was calculated for the containment isolation event which is a rapid closure of all main steam isolation valves. This analysis shows that for WNP-2 Cycle 5 operation, the safety valve response

system pressures predicted during the event are below the ASME Code limit of 110% of design pressure (1375 psig) and are shown in Table 2.1. The analysis conservatively assumed six safety relief valves out of service.

The continued applicability of the previously established MCPR safety limit of 1.06 in Cycle 5 was confirmed for all fuel types using the methodology of Reference 6.

TABLE 2.1 THERMAL MARGIN SUMMARY FOR WNP-2 CYCLE 5

Transient	% Power/% Flow	Delta CPR/MCPR*	
		GE Fuel	ANF Fuel
Load Rejection** Without Bypass (572 ANF assemblies & 192 GE assemblies)	104/106	0.28/1.34	0.25/1.31
Load Rejection** Without Bypass (564 ANF assemblies & 200 GE assemblies	104/106	0.29/1.35	0.25/1.31
Feedwater Controller** Failure	47/106	0.23/1.29	0.20/1.26
Loss of Feedwater*** Heating	Not Applicable	0.09/1.15	0.09/1.15

MAXIMUM PRESSURE (PSIG)

Transient	Vessel Dome	Vessel Lower Plenum	Steam Line
MSIV Closure	1286	1315	1289

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*MCPR value using the 1.06 safety limit justified herein.

**These transients were evaluated with normal scram speed, RPT operable, and at the end of cycle.

***WNP-2 plant specific bounding value.

3.0 TRANSIENT ANALYSIS FOR THERMAL MARGIN

3.1 Design Basis

System analyses were performed at the increased core flow condition of 106% to determine the most limiting type of system transients for the establishment of thermal margins. As shown in Reference 5, system transients from the increased core flow condition bound transients from the nominal (100%) flow condition. Analysis of the LRNB was performed at the rated design 104% power/106% flow point. Since feedwater controller failure (FWCF) transients may be more severe at reduced power because of the larger change in feedwater flow, a FWCF transient was performed at the minimum power (47%) that allowed for increased core flow. The initial conditions used in the analysis for transients at the 104% power/106% flow point are as shown in Table 3.1. The most limiting exposure in cycle was determined to be at 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 in relation to cases where control rods are partially inserted.

The calculational models used to analyze these pressurization events include the ANF plant transient and core thermal-hydraulic codes as described in previous documentation.[3,4,6,7] Fuel pellet-to-clad gap conductances used in the analyses are based on calculations with RODEX2.[8] Recirculation pump trip (RPT) coastdown was input based on measured WNP-2 startup test data, and the COTRANSA system transient model for WNP-2 was benchmarked to appropriate WNP-2 startup test data. The hot channel performance is evaluated with XCOBRA-T[4] using COTRANSA supplied boundary conditions. Table 3.2 summarizes the values used for important parameters in the analysis.

3.2 Anticipated Transients

ANF transient analysis methodology for Jet Pump BWR's considers eight categories of potential system transient occurrences.[3] The three most limiting transients for WNP-2 are presented in this section; these transients are:

-   Load Rejection Without Bypass (LRNB)
-   Feedwater Controller Failure (FWCF)
-   Loss of Feedwater Heating (LOFH).

A summary of the transient analyses is shown in Table 3.3. Other plant transient events are inherently nonlimiting or clearly bounded by one of the above events.

3.2.1 Load Rejection Without Bypass

This event is the most limiting of the class of transients characterized by rapid vessel pressurization. The generator load rejection causes a turbine control valve trip, which initiates a reactor scram and a recirculation pump trip (RPT). The compression wave produced by the fast turbine control valve closure travels through the steam lines into the vessel and pressurizes the reactor vessel and core. Bypass flow to the condenser, which would mitigate the pressurization effect, is conservatively not allowed. The excursion of core power due to void collapse is primarily terminated by reactor scram and void growth due to RPT. Figures 3.1 through 3.10 depict the time variance of critical reactor and plant parameters from the analyses of several load rejection transients. Figures 3.1, 3.2 and 3.5 through 3.10 are load rejection results for the original reload batch size, and Figures 3.3 and 3.4 are load rejection results for the revised reload batch size. Transient analysis cases include the design basis power and increased core flow point with a matrix of cases which involve normal scram speed, technical specification scram speed, and recirculation pump trip (RPT) in service and out of service.

Analysis assumptions are:

-   Control rod insertion time based on WNP-2 measured data (normal scram speed) or minimum technical specification scram speed.

-   Integral power to the hot channel was increased by 10% for the pressurization transient, consistent with Reference 9.

Table 3.3 shows delta CPR values for a matrix of LRNB transients with the RPT out of service with both normal scram speed (NSS) and technical specification scram speed (TSSS).

ANF has previously analyzed the LRNB event for prior cycles at an exposure of EOC -2000 MWd/MTU. Since a significant number of control rods are inserted into the core up to end-of-cycle (EOC) minus 2000 MWd/MTU, this prior analytical experience has shown the CRWE to be clearly bounding from the beginning-of-cycle (BOC) up to this point. That is, the limiting delta CPR or MCPR limit throughout the earlier part of the cycle was set by the CRWE from BOC to EOC -2000 MWd/MTU. For Cycle 5 an LRNB calculation at EOC -2000 MWd/MTU has not been provided because the CRWE clearly sets the MCPR limit up to this exposure. For Cycle 5 exposures greater than EOC minus 2000 MWd/MTU, MCPR values defined in Table 3.3 are applicable.

3.2.2 Feedwater Controller Failure

Failure of the feedwater control system is postulated to lead to a maximum increase in 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 level trip setting is exceeded. To protect against wet steam entering the turbine, the turbine trips upon reaching the high level setting, closing the turbine stop valves. The compression wave that is created, though mitigated by bypass flow, pressurizes the core and causes a power excursion. The power increase is terminated by reactor scram, RPT, and pressure relief from the bypass valves opening. The evaluation of this event was performed using the scram and integral power assumptions discussed in Section 3.2.1. Sensitivity results have shown that EOC conditions are bounding because rods are inserted for lower cycle exposures, and high flows are bounding because of higher axials in the core.

Reference 11 showed that the LRNB is more limiting at full power than the FWCF. Because the total change in feedwater flow is the greatest from reduced

power condition, the FWCF was analyzed from reduced power conditions. The FWCF was analyzed with the feedwater flow rate increasing at a rate between 10 and 25 percent of nuclear boiler rated (NBR) flow per second. The FWCF transient event was analyzed from the lowest allowed power (47%) at increased core flow. Figures 3.11 through 3.14 present key variables. The delta CPR values for the co-resident fuel types for 47% power/106% flow transient are shown in Table 3.3. Table 3.3 shows that the delta CPR/MCPR values for the FWCF are less than the delta CPR/MCPR value for the 104/106 LRNB event with RPT operable and inoperable with normal scram speed.

3.2.3 Loss of Feedwater Heating

Loss of Feedwater Heating (LOFH) events were evaluated for Cycle 5 with the ANF core simulator model XTGBWR[10] by representing the reactor in equilibrium before and after the event. Actual and projected operating statepoints were used as initial conditions. Final conditions were determined by reducing the feedwater temperature by 100°F and increasing core power such that the calculated eigenvalue remain unchanged.

Based on a bounding value analysis, a MCPR limit of 1.15 for WNP-2 with a MCPR safety limit of 1.06 is supported (i.e., a delta CPR of 0.09). As shown in Appendix A of this report, the WNP-2 MCPR safety limit for Cycle 5 continues to be 1.06; hence, the LOFH transient requires a MCPR limit of 1.15 for WNP-2.

3.3 Calculational Model

The plant transient codes used to evaluate the pressurization transients (generator load rejection and feedwater flow increase) were the ANF advanced codes COTRANSA[3] and XCOBRA-T.[4] This axial one-dimensional 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. All pressurization transients were analyzed on a bounding basis using COTRANSA in conjunction with the XCOBRA-T hot channel model. The XCOBRA-T code was used consistent with the benchmarking methodology.

3.4 Safety Limit

The MCPR 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 fuel rods in the core. The operating limit MCPR is established such that in the event the most limiting anticipated operational transient occurs, the safety limit will not be violated.

The safety limit for all fuel types in WNP-2 Cycle 5 was confirmed by the methodology presented in Reference 6 to have the Cycle 2 value of 1.06. The input parameters and uncertainties used to establish the safety limit are presented in Appendix A of this report.

3.5 Final Feedwater Temperature Reduction

Reference 1 presents final feedwater temperature reduction (FFTR) analysis with thermal coastdown for WNP-2 for Cycles 3 and 4. The FFTR analysis was performed for a 65°F temperature reduction. These FFTR analyses are applicable after the all rods out condition is reached with normal feedwater temperature. The FFTR analysis results show that delta CPR changes for the LRNB and FWCF transients are conservatively bounded by adding 0.02 to the delta CPR values for these transients at normal feedwater temperatures.

TABLE 3.1 DESIGN REACTOR AND PLANT CONDITIONS FOR WNP-2

Reactor Thermal Power (104%)	3464 MWt
Total Recirculating Flow (106%)	115.0 Mlb/hr
Core Channel Flow	102.4 Mlb/hr
Core Bypass Flow	12.3 Mlb/hr
Core Inlet Enthalpy	527.8 BTU/lbm
Vessel Pressures	
Steam Dome	1036. psia
Upper Plenum	1049. psia
Core	1056. psia
Lower Plenum	1073. psia
Turbine Pressure	978. psia
Feedwater/Steam Flow	14.8 Mlb/hr
Feedwater Enthalpy	391.1 BTU/lbm
Recirculating Pump Flow (per pump)	16.3 Mlb/hr

TABLE 3.2 SIGNIFICANT PARAMETER VALUES USED IN ANALYSIS FOR WNP-2

High Neutron Flux Trip	126.2%		
Void Reactivity Feedback	10% above nominal*		
Time to Deenergized Pilot Scram			
Solenoid Valves	200 msec		
Time to Sense Fast Turbine			
Control Valve Closure	80 msec		
Time from High Neutron Flux			
Time to Control Rod Motion	290 msec		

	Normal	Tech Spec	
Scram Insertion Times**	0.404 sec	0.430 sec	to Notch 45
	0.660 sec	0.868 sec	to Notch 39
	1.504 sec	1.936 sec	to Notch 25
	2.624 sec	3.497 sec	to Notch 5

Turbine Stop Valve Stroke Time	100 msec
Turbine Stop Valve Position Trip	90% open
Turbine Control Valve Stroke	
Time (Total)	150 msec
Fuel/Cladding Gap Conductance	
Core Average (Constant)	587. BTU/hr-ft^2-F
Safety/Relief Valve Performance	
Settings	Technical Specifications
Relief Valve Capacity	228.2 lbm/sec (1091 psig)
Pilot Operated Valve Delay/Stroke	400/100 msec

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*For rapid pressurization transients a 10% multiplier on integral power is used; see Reference 9 for methodology description.

**Slowest measured average control rod insertion time to specified notches for each group of 4 control rods arranged in a 2x2 array.

TABLE 3.2   SIGNIFICANT PARAMETER VALUES USED IN ANALYSIS FOR WNP-2
(Continued)

MSIV Stroke Time	3.0 se
MSIV Position Trip Setpoint	85% open
Condenser Bypass Valve Performance	
Total Capacity	990. lbm/sec
Delay to Opening (80% open)	300 msec
Fraction of Energy Generated in Fuel	0.965
Vessel Water Level (above Separator Skirt)	
High Level Trip (L8)	73 in
Normal	49.5 in
Low Level Trip (L3)	21 in
Maximum Feedwater Runout Flow	
Two Pumps	5799. lbm/sec
Recirculating Pump Trip Setpoint	1170 psig
	Vessel Pressure

TABLE 3.2   SIGNIFICANT PARAMETER VALUES USED IN ANALYSIS FOR WNP-2
(Continued)

Control Characteristics

Sensor Time Constants	
Steam Flow	1.0 sec
Pressure	500 msec
Others	250 msec
Feedwater Control Mode	Three-Element
Feedwater 100% Mismatch	
Water Level Error	48 in
Steam Flow Equiv.	100%
Flow Control Mode	Manual
Pressure Regulator Settings	
Lead	3.0 sec
Lag	7.0 sec
Gain	3.3%/psid

TABLE 3.3   RESULTS OF SYSTEM PLANT TRANSIENT ANALYSES

Event	Maximum Neutron Flux (% Rated)	Maximum Core Average Heat Flux (% Rated)	Maximum System Pressure (psig)	Delta CPR GE Fuel	Delta CPR ANF Fuel
LRNB RPT Operable, NSS* (original reload batch)	403	121	1169	0.28	0.25
LRNB RPT Operable, NSS (revised reload batch)	406	121	1169	0.29	0.25
LRNB RPT Inoperable, NSS	501	127	1181	0.35	0.31
LRNB RPT Operable, TSSS**	454	127	1174	0.35	0.31
LRNB RPT Inoperable, TSSS	594	132	1189	0.41	0.35
FWCF (47% Power/106% Flow), NSS RPT Operable	163	54	1026	0.23	0.20
FWCF (47% Power/106% Flow), NSS RPT Inoperable	217	55	1023	0.29	0.25
MSIV Closure With Flux Scram	708	133	1315	N/A	

NOTES: 1. All results are for the design power and increased flow point (104% power/106% flow) unless otherwise noted.

2. Since there is a small delta CPR increase associated with the LRNB results for the reduced reload batch size, it is conservative to add 0.01 to all of the original reload batch size LRNB results to conservatively set limits for the reduced reload batch size.

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*Normal Scram Speed (NSS).

**Technical Specification Scram Speed (TSSS).
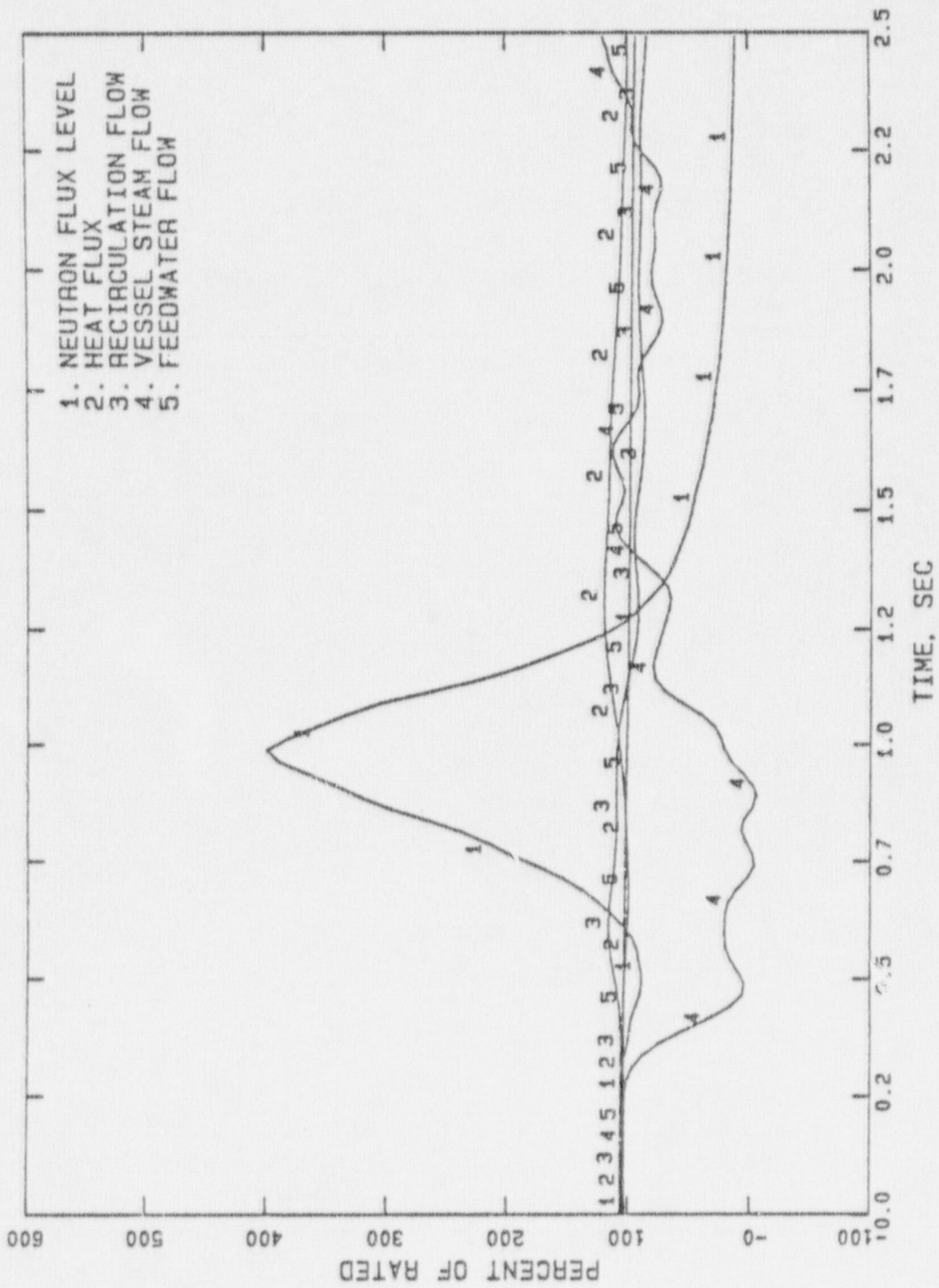
FIGURE 3.1 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT OPERABLE, NORMAL SCRAM SPEED
(ORIGINAL RELOAD BATCH SIZE)
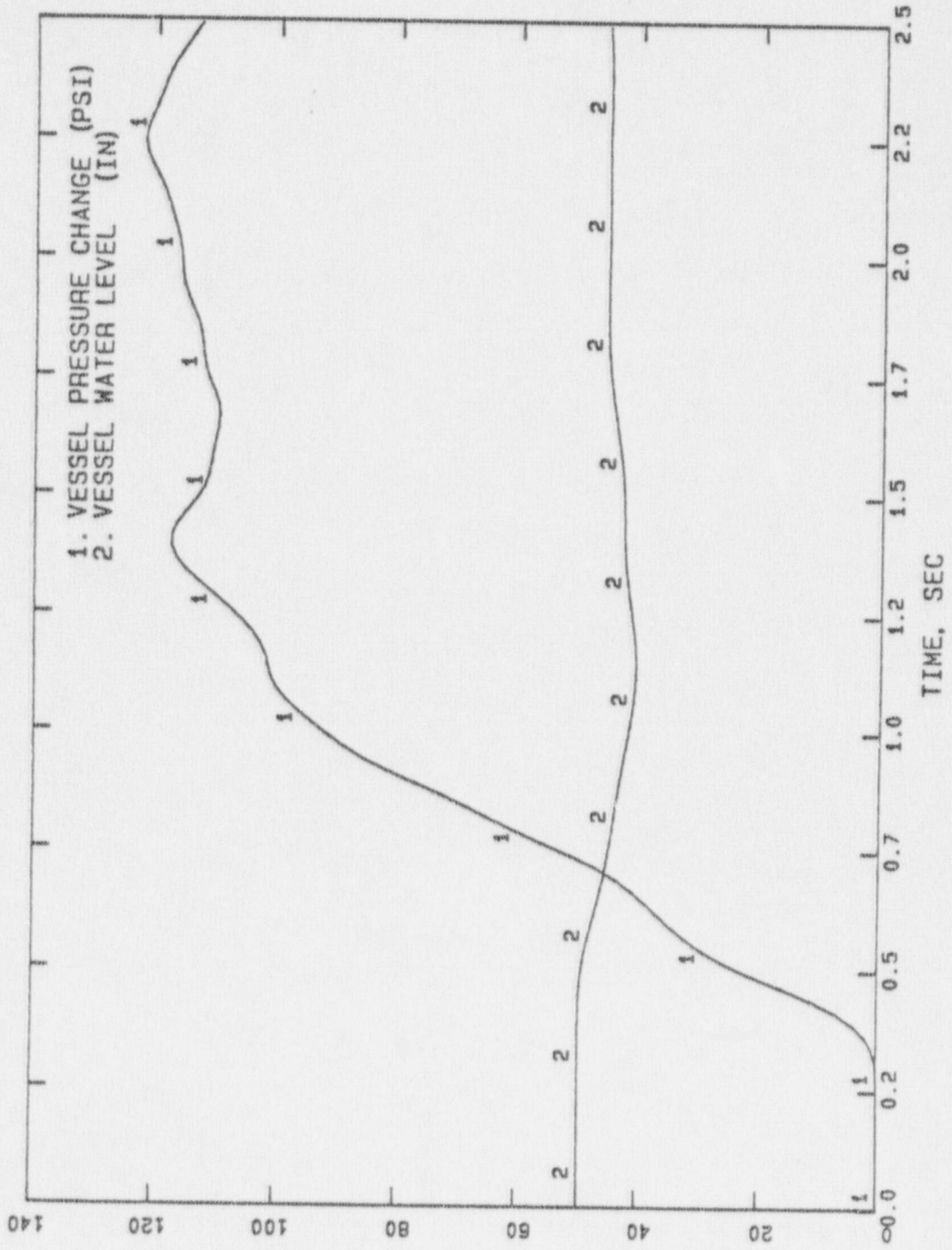
FIGURE 3.2 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT OPERABLE, NORMAL SCRAM SPEED
(ORIGINAL RELOAD BATCH SIZE)
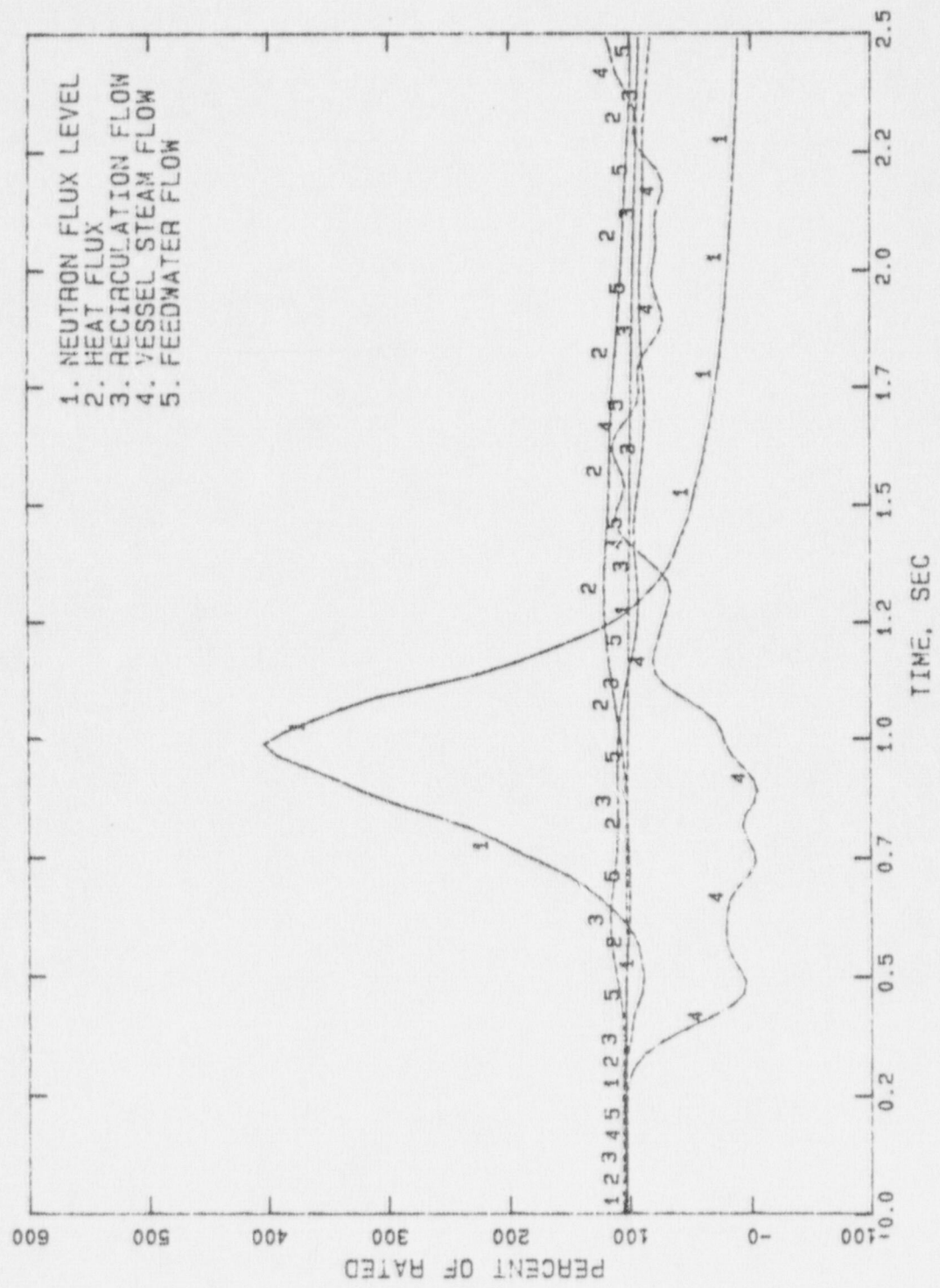
FIGURE 3.3 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT OPERABLE, NORMAL SCRAM SPEED
(REVISED RELOAD BATCH SIZE)
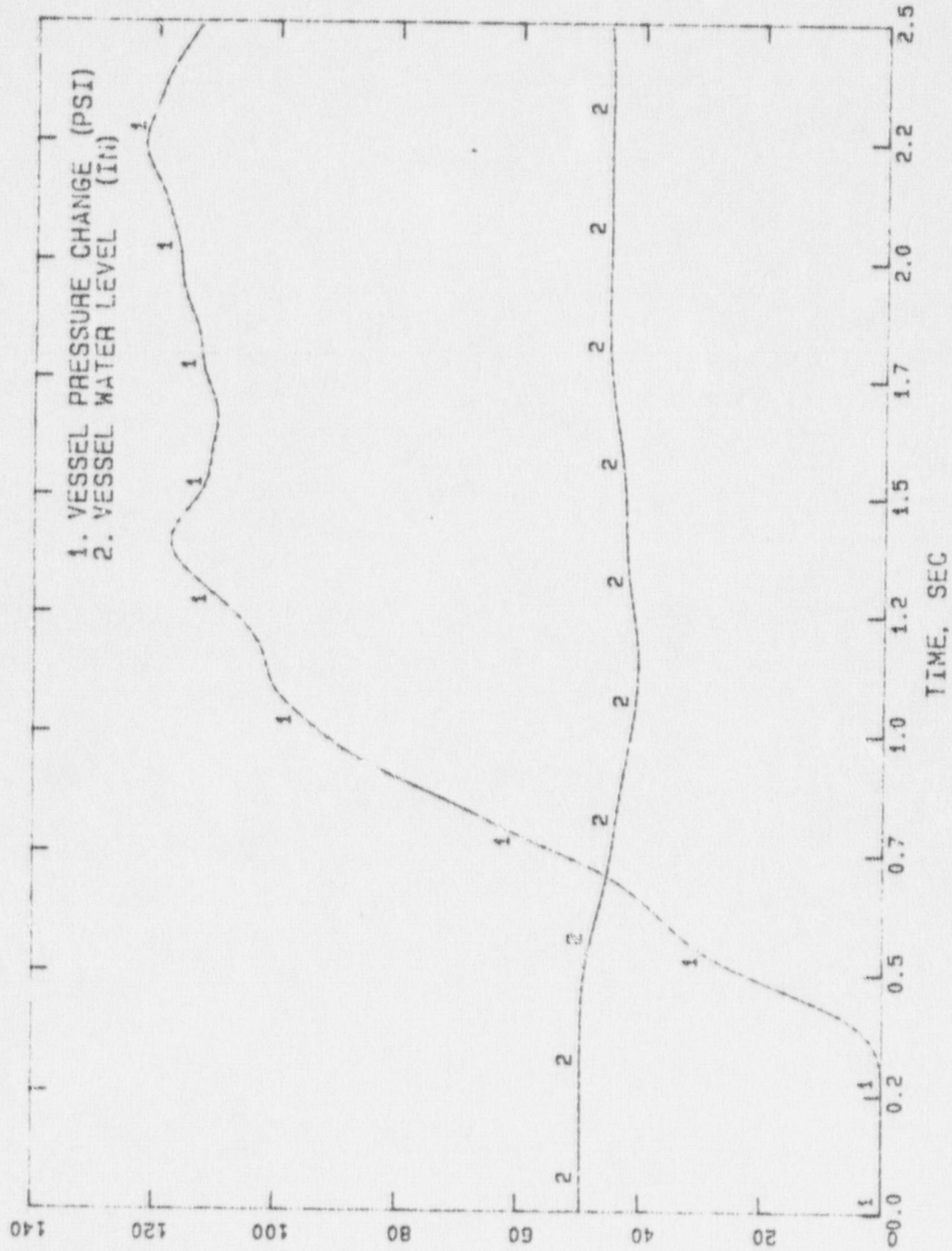
FIGURE 3.4 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT OPERABLE, NORMAL SCRAM SPEED
(REVISED RELOAD BATCH SIZE)
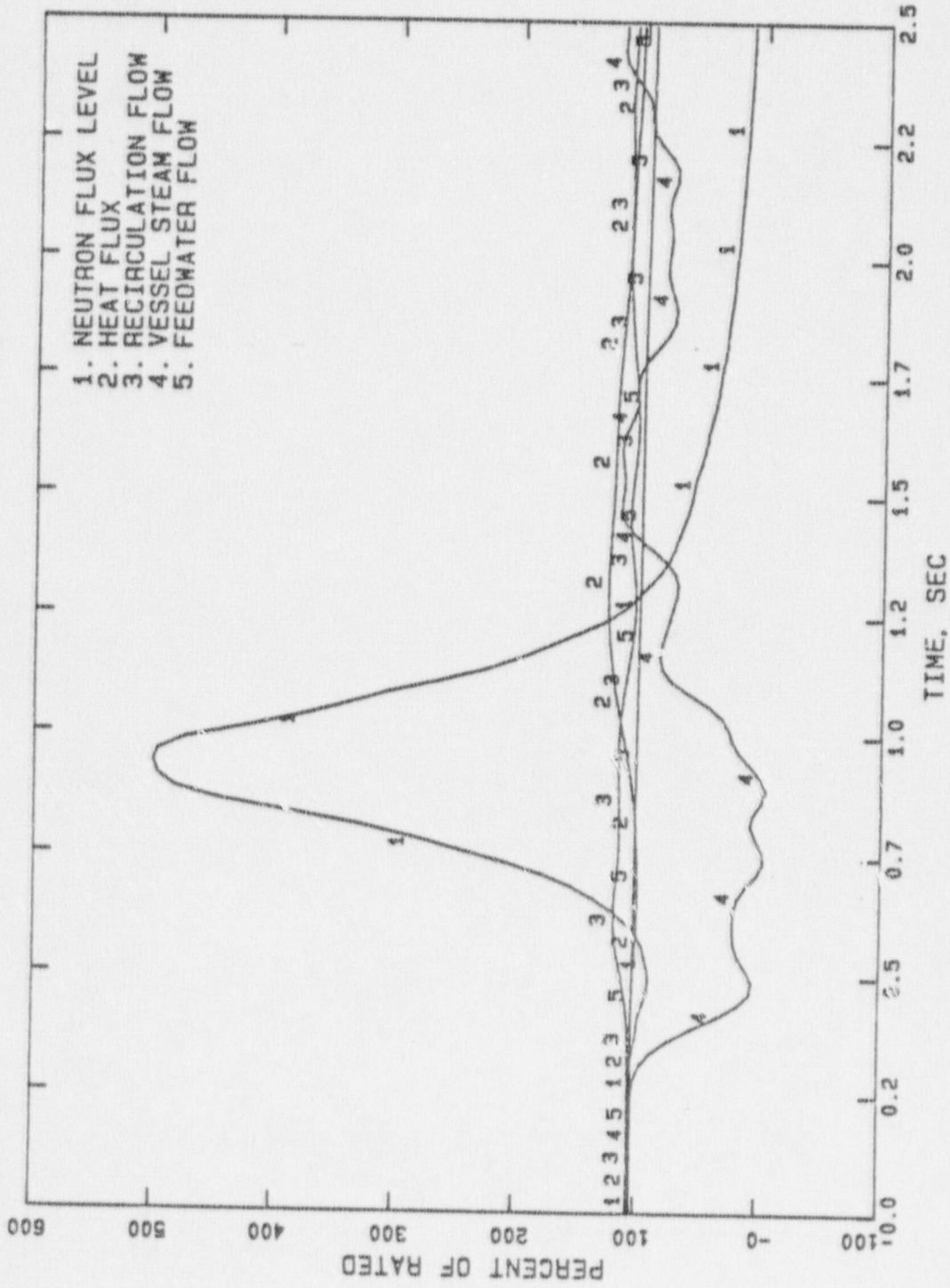
FIGURE 3.5 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT INOPERABLE, NORMAL SCRAM SPEED
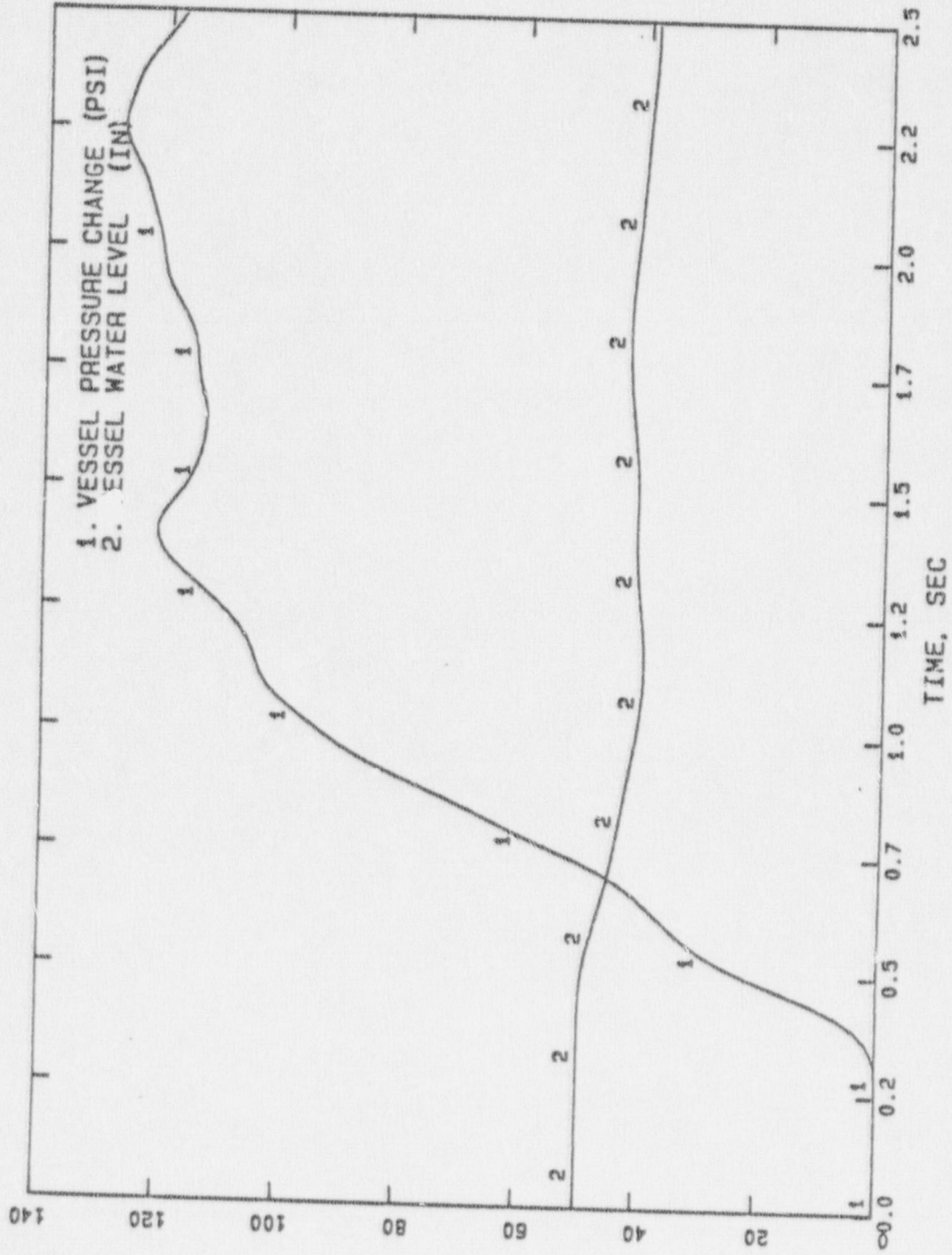
FIGURE 3.6 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT INOPERABLE, NORMAL SCRAM SPEED
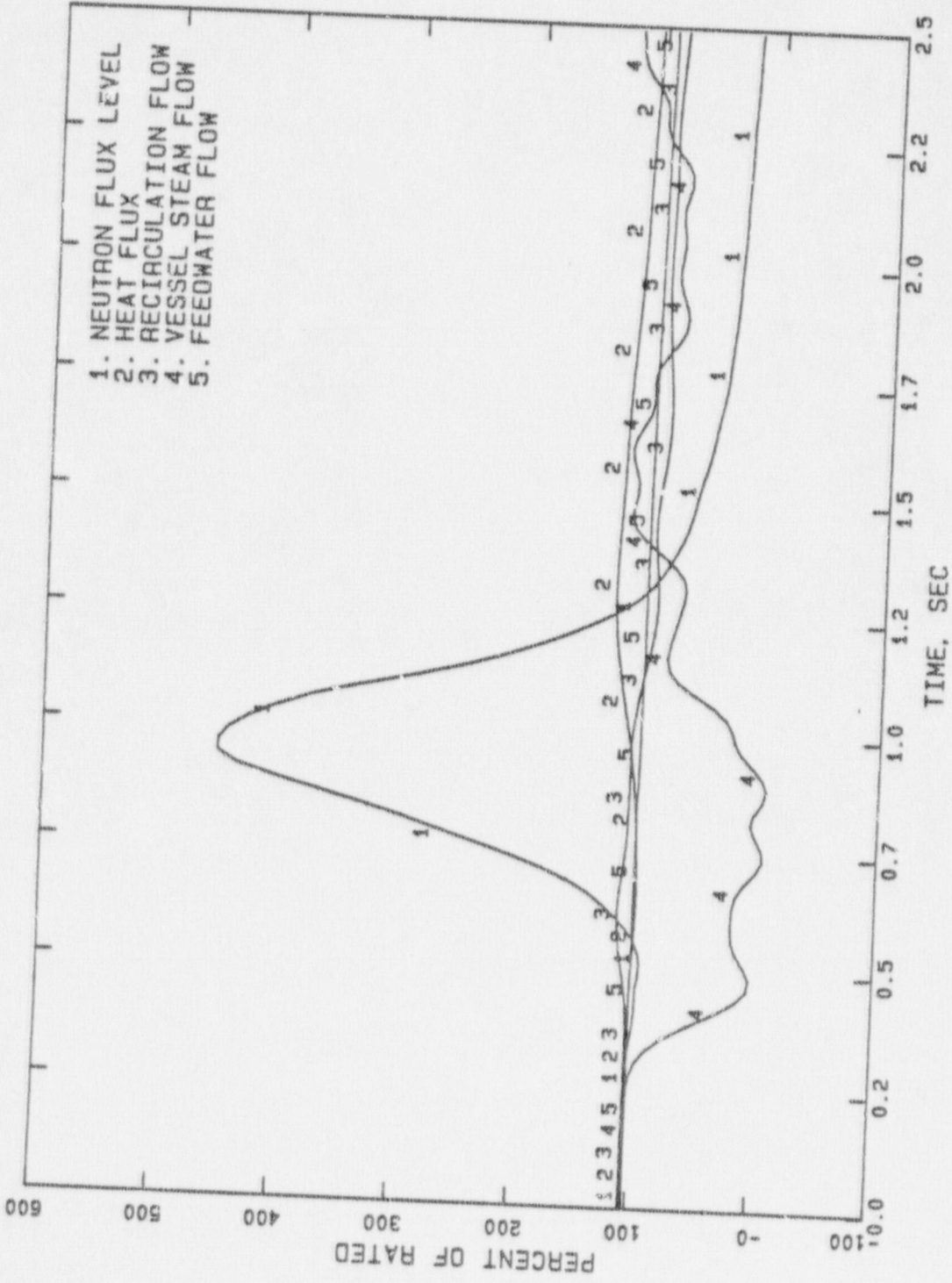
FIGURE 3.7 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT OPERABLE, TECH. SPEC. SCRAM SPEED
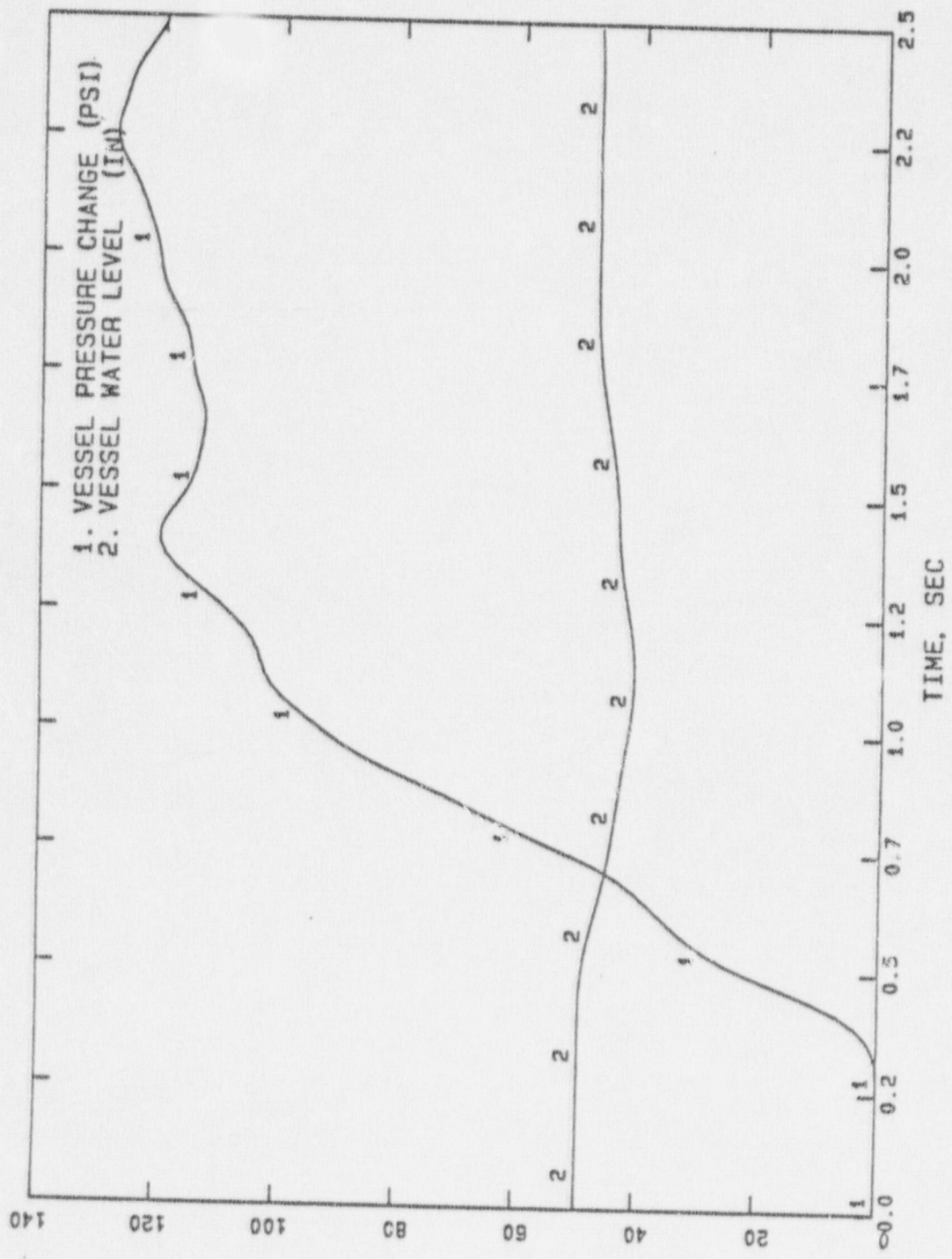
FIGURE 3.8 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT OPERABLE, TECH. SPEC. SCRAM SPEED
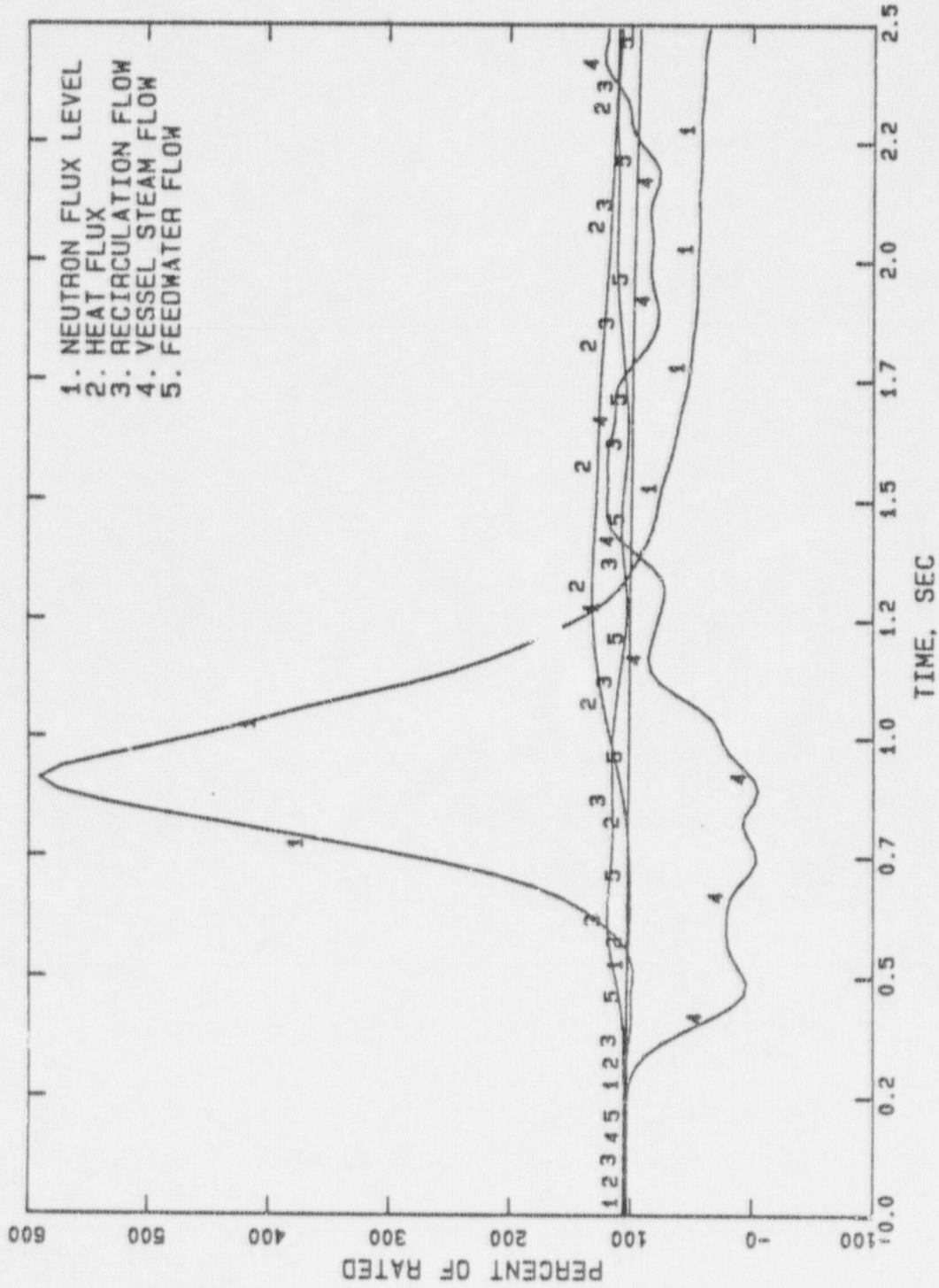
FIGURE 3.9 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT INOPERABLE, TECH. SPEC. SCRAM SPEED
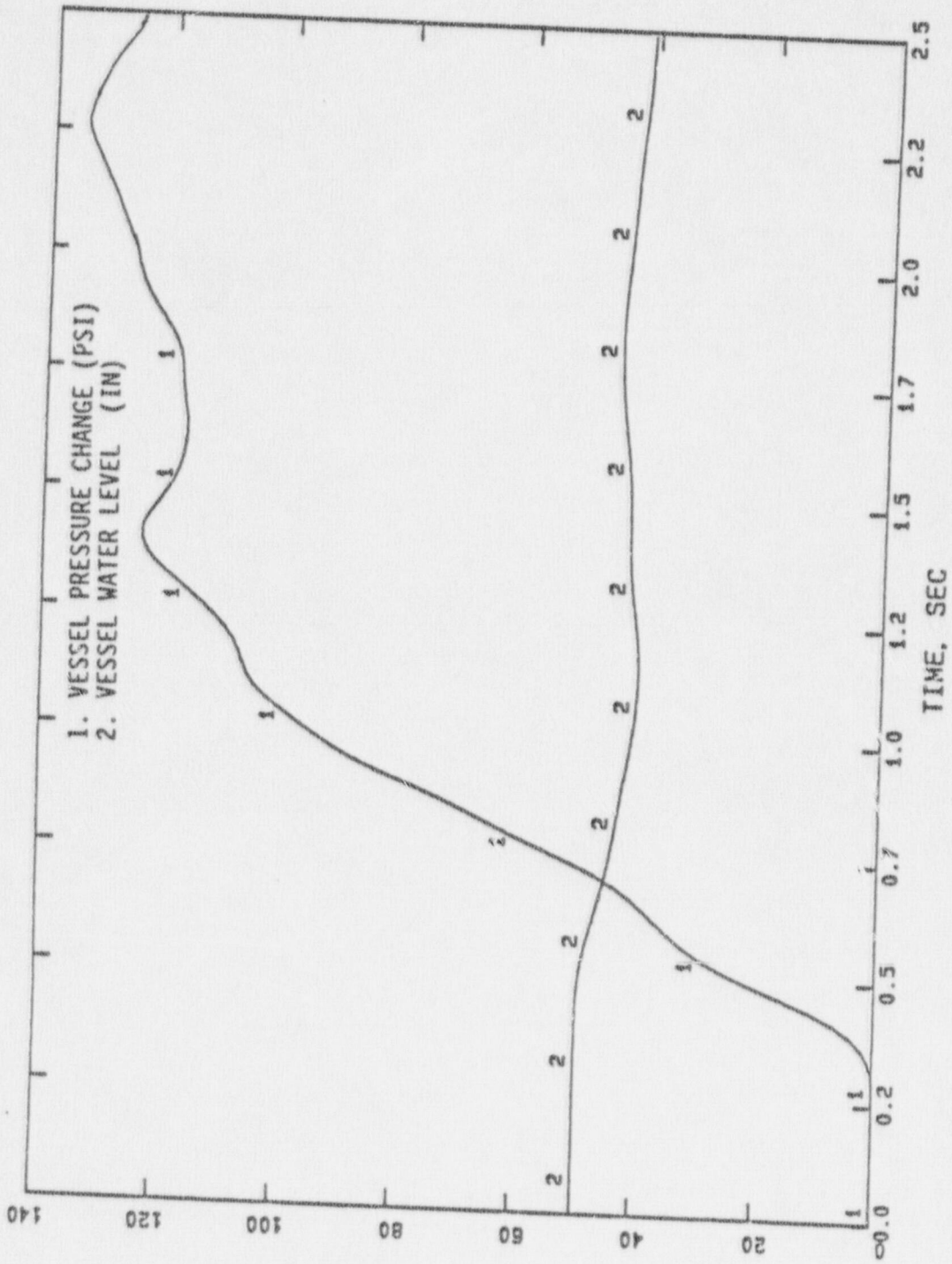
FIGURE 3.10 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT INOPERABLE, TECH. SPEC. SCRAM SPEED
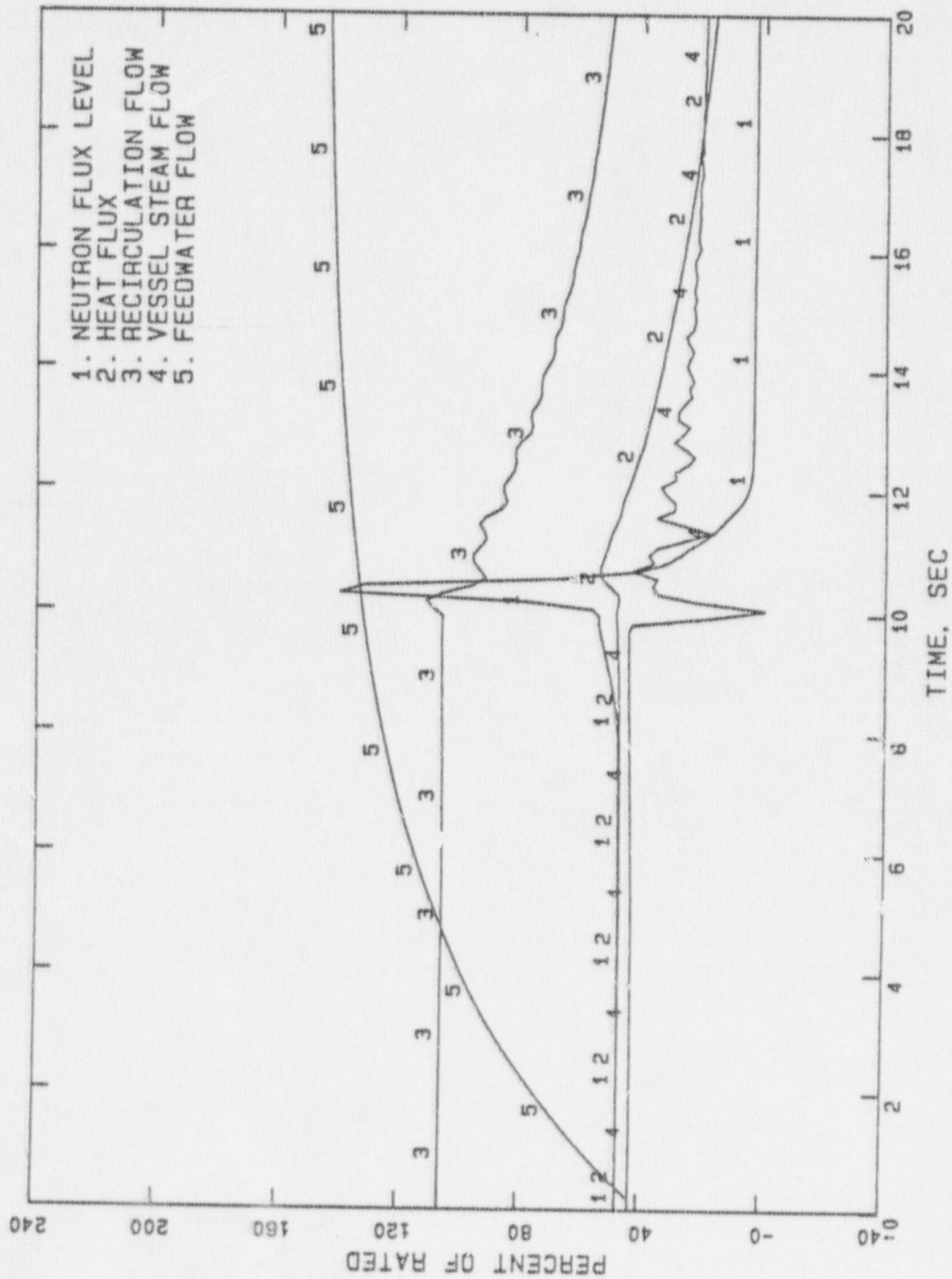
FIGURE 3.11 FEEDWATER CONTROLLER FAILURE RESULTS FOR 47% POWER AND 106% FLOW, RPT OPERABLE, NORMAL SCRAM SPEED
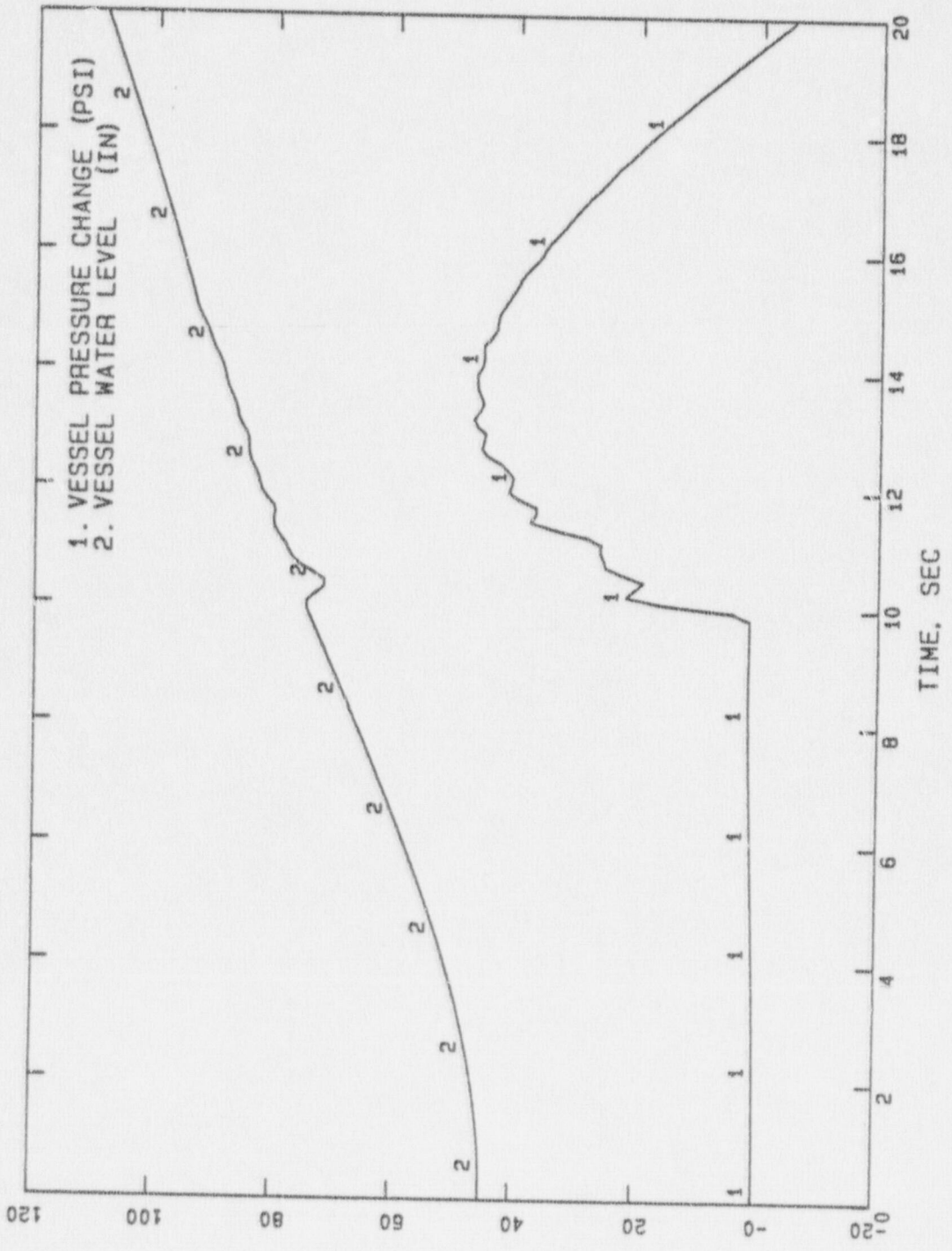
FIGURE 3.12 FEEDWATER CONTROLLER FAILURE RESULTS FOR 47% POWER AND 106% FLOW, RPT OPERABLE, NORMAL SCRAM SPEED
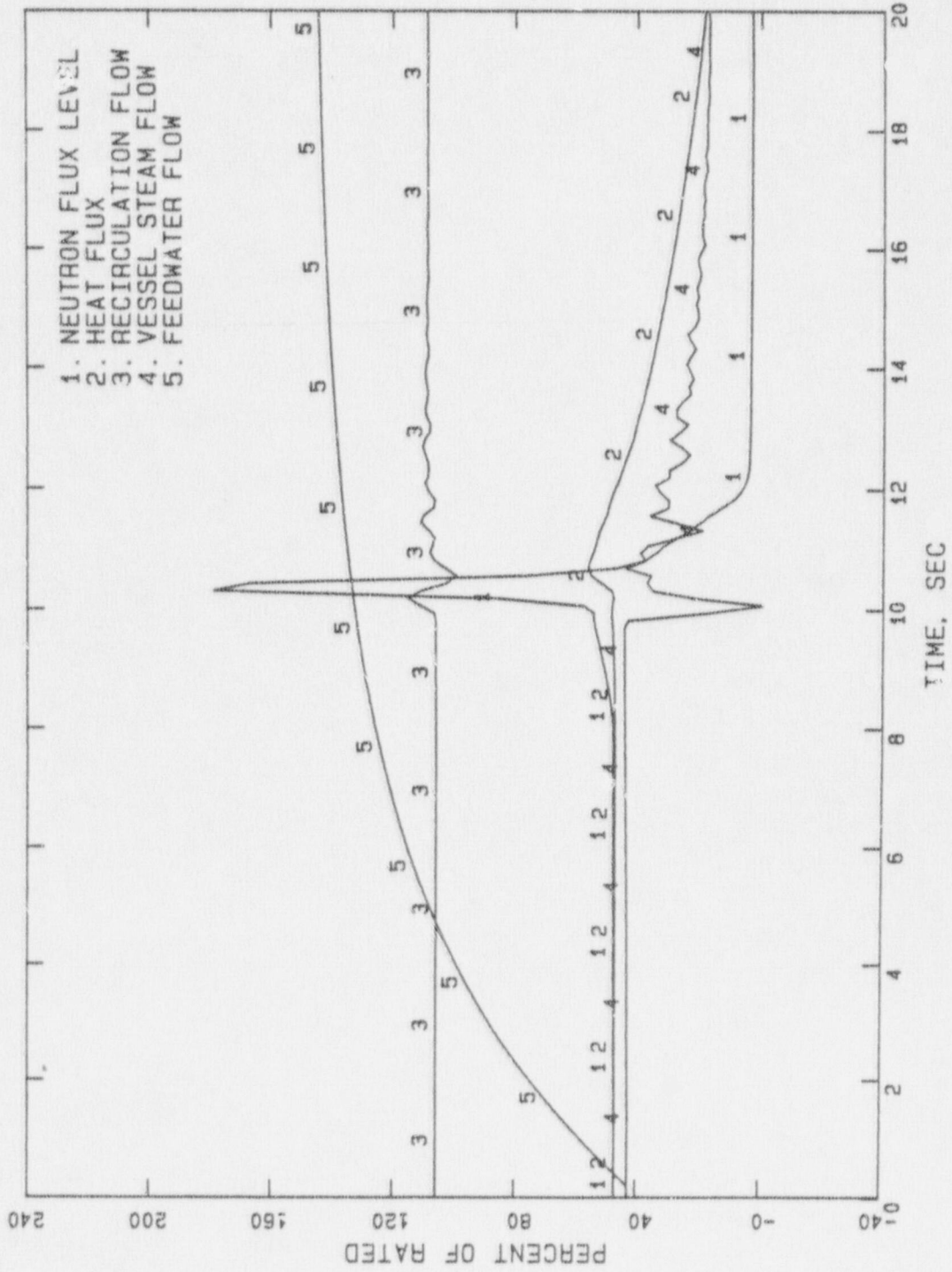
FIGURE 3.13 FEEDWATER CONTROLLER FAILURE RESULTS FOR 47% POWER AND 106% FLOW, RPT INOPERABLE, NORMAL SCRAM SPEED
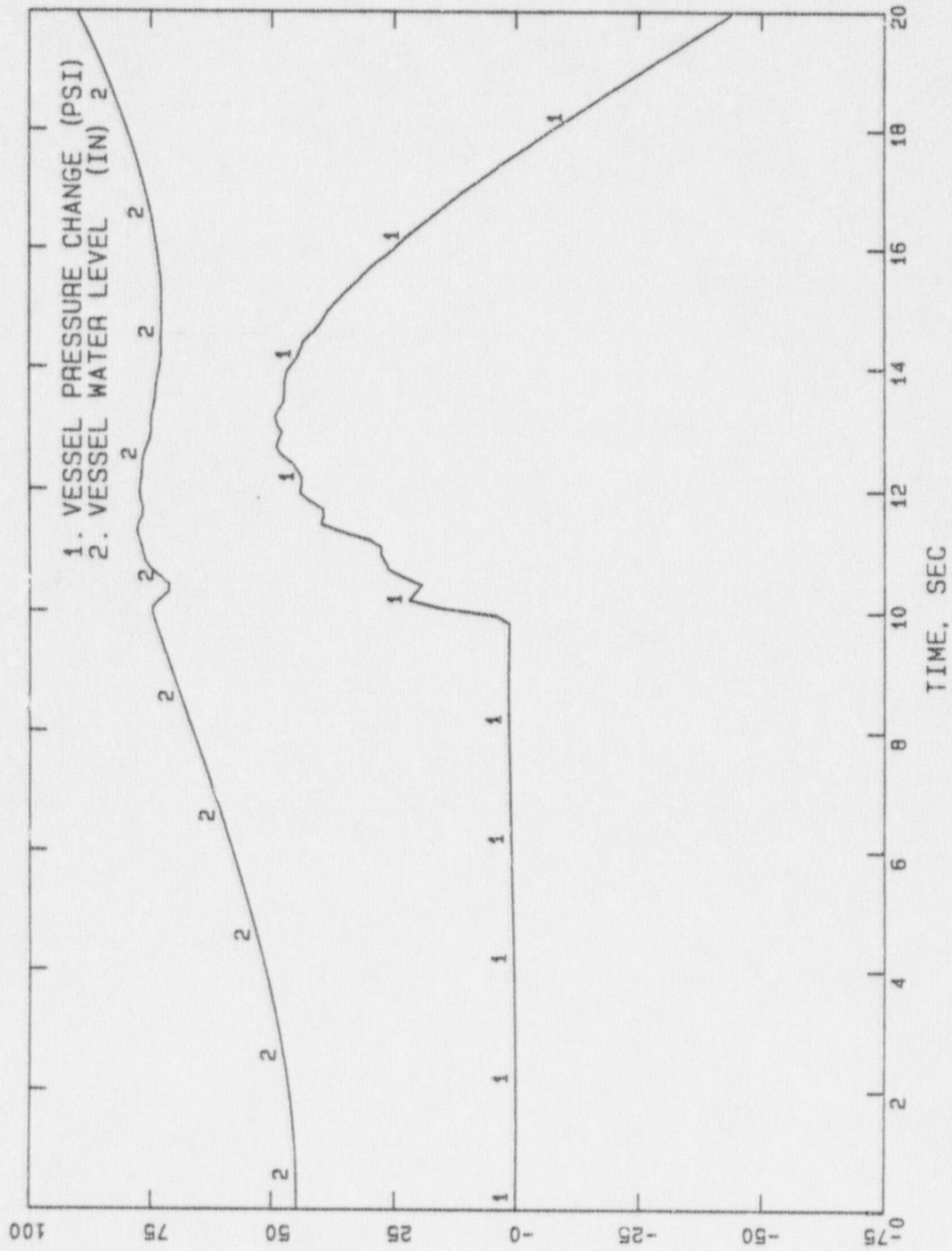
FIGURE 3.14 FEEDWATER CONTROLLER FAILURE RESULTS FOR 47% POWER AND 106% FLOW, RPT INOPERABLE, NORMAL SCRAM SPEED

4.0 MAXIMUM OVERPRESSURIZATION

Maximum system pressure has been calculated for the containment isolation event (rapid closure of all main steam isolation valves) with an adverse scenario as specified by the ASME Pressure Vessel Code. This analysis showed that the safety valves of WNP-2 have sufficient capacity and performance to prevent pressure from reaching the established transient pressure safety limit of 110% of the design pressure. The maximum system pressures predicted during the event are shown in Table 2.1. This analysis also assumed six safety relief valves out of service.

4.1 Design Bases

The reactor conditions used in the evaluation of the maximum pressurization event are those shown in Table 3.1. The most critical active component (scram on MSIV closure) was assumed to fail during the transient. The calculation was performed with the ANF advanced plant simulator code COTRANSA,[3] which includes an axial one-dimensional neutronics model.

4.2 Pressurization Transients

ANF has evaluated several pressurization events and has determined that closure of all main steam isolation valves (MSIVs) without direct scram is the most limiting. Since the MSIVs are closer to the reactor vessel than the turbine stop or turbine control valves, significantly less volume is available to absorb the pressurization phenomena when the MSIVs are closed than when turbine valves are closed. The closure rate of the MSIVs is substantially slower than the turbine stop valves or turbine control valves. The impact of this smaller volume is more important to this event than the slower closure speed of the MSIV valves relative to turbine valves. Calculations have determined that the overall result is to cause MSIV closures to be more limiting than turbine isolations.

4.3 Closure of All Main Steam Isolation Valves

This calculation also assumed that six relief valves were out of service and that all four main steam isolation valves were isolated at the containment boundary within 3 seconds. At about 3.3 seconds, the reactor scram is

initiated by reaching the high flux trip setpoints. Pressures reach the recirculation pump trip setpoint (1170 psig) before the pressurization has been reversed. Loss of coolant flow leads to enhanced steam production as less subcooled water is available to absorb core thermal power. The calculated maximum pressure in the steam lines was 1289 psig, occurring near the vessel at about 5 seconds. The maximum vessel pressure was 1315 psig, occurring in the lower plenum at about 5 seconds. These results are presented in Tables 2.1 and 3.3 for the design basis point.

Since there has been almost no change in the maximum system pressure calculated for the containment isolation event for four cycles, it is reasonable to expect that the reduced reload batch size for Cyle 5 would have no impact on the Cycle 5 result given in Tables 2.1 and 3.3.

5.0 RECIRCULATION FLOW RUN-UP

The MCPR full flow operating limit is established through evaluation of anticipated transients at the design basis state. Due to the potential for large reactor power increases should an uncontrolled recirculation flow increase occur from a less than rated core flow state, the need exists for an augmentation of the operating limit MCPR (full flow) for operation at lower flow conditions.

Advanced Nuclear Fuels Corporation determined the required reduced flow MCPR operating limit by evaluating a bounding slow flow increase event. The calculations assume the event was initiated from the 104% rod line at minimum flow and terminates at 120% power at 103% flow (flow control valve wide open). This power flow relationship bounds that calculated for a constant xenon assumption. It was conservatively assumed that the event was quasi-steady and a flow biased scram does not occur.

The power distribution was chosen such that the MCPR equals the safety limit at the final power/flow run-up point. The reduced flow MCPRs were then calculated by XCOBRA[6] at discrete flow points.

The recirculation flow run-up analysis performed for WNP-2 Cycle 2 was reviewed, and the assumptions and conditions used for Cycle 2 are applicable to Cycle 5 except for the six degree reduction in feedwater temperature at full power conditions. Thus, the reduced flow MCPR operating limit for WNP-2 Cycle 5 is changed slightly from earlier cycles. For final feedwater temperature reduction (FFTR) conditions, the previously reported[1] reduced flow MCPR operating limit remains applicable. The reduced flow MCPR operating limit for Cycle 5 is presented in Figure 5.1 and tabulated in Table 5.1. The MCPR operating limit for WNP-2 shall be the maximum of this reduced flow MCPR operating limit and the full flow MCPR operating limit as summarized in Reference 2.

TABLE 5.1   REDUCED FLOW MCPR OPERATING LIMIT FOR WNP-2

Core Flow (% Rated)	Reduced Flow MCPR Operating Limit
100	1.07
90	1.13
80	1.19
70	1.26
60	1.34
50	1.44
40	1.59

NOTE: The MCPR operating limit shall be the greater of the rated condition MCPR operating limit or the value for reduced flow from this curve

TOTAL CORE RECIRCULATING FLOW (%RATED)

MCPR OPERATING LIMIT
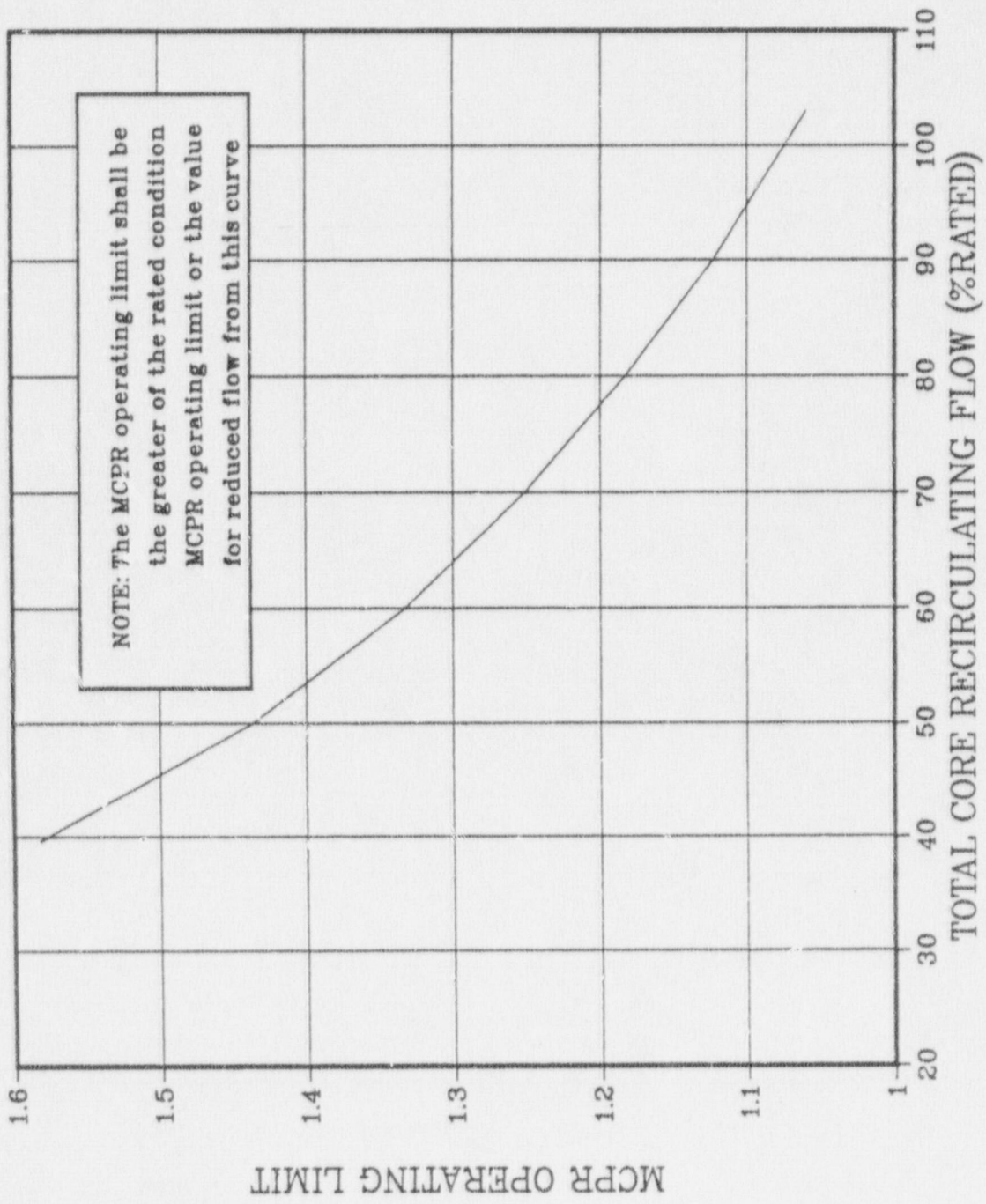
FIGURE 5.1 REDUCED FLOW MCPR OPERATING LIMIT

6.0 REFERENCES

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APPENDIX A
MCPR SAFETY LIMIT

A.1 INTRODUCTION

Bundle power limits in a boiling water reactor (BWR) are determined through evaluation of critical heat flux phenomena. The basic criterion used in establishing critical power ratio (CPR) limits is that at least 99.9% of the fuel rods in the core will be expected to avoid boiling transition (critical heat flux) during normal operation and anticipated operational occurrences. Operating margins are defined by establishing a minimum margin to the onset of boiling transition condition for steady state operation and calculating a transient effects allowance, thereby assuring that the steady state limit is protected during anticipated off-normal conditions. This appendix addresses the calculation of the minimum margin to the steady state boiling transition condition, which is implemented as the MCPR safety limit in the plant technical specifications. The transient effects allowance, or the limiting transient change in CPR (i.e., delta CPR), is treated in the body of this report.

The MCPR safety limit is established through statistical consideration of measurement and calculational uncertainties associated with the thermal hydraulic state of the reactor using design basis radial, axial, and local power distributions. Some of the calculational uncertainties, including those introduced by the critical power correlation, power peaking, and core coolant distribution, are fuel related. When ANF fuel is introduced into a core where it will reside with another supplier's fuel types, the appropriate value of the MCPR safety limit is calculated based on fuel-dependent parameters associated with the mixed core. Similarly, when an ANF-fabricated reload batch is used to replace a group of dissimilar fuel assemblies, the core average fuel dependent parameters change because of the difference in the relative number of each type of bundle in the core, and the MCPR safety limit is again reevaluated.

The design basis power distribution is made up of components corresponding to representative radial, axial, and local peaking factors. Where such data are appropriately available from the previous cycle, these factors are determined through examination of operating data for the previous cycle and predictions of operating conditions during the cycle being evaluated for the MCPR safety limit. If operating data are not available, either because the reactor has not been operated or because appropriate data cannot be supplied to ANF, the safety limit power distribution is determined strictly from the predicted operating conditions during the cycle being evaluated. Operating data for WNP-2 during Cycle 4 and the predicted operating conditions for Cycle 5 were evaluated to identify the design basis power distributions used in the Cycle 5 MCPR safety limit analysis.

A.2 ASSUMPTIONS

A.2.1 Design Basis Power Distribution

The local and radial power distributions which were determined to be conservative for use in the safety limit analysis are shown in Figures A-1 through A-5.

A.2.2 Hydraulic Demand Curve

Hydraulic demand curves based on calculations with XCOBRA were used in the safety limit analysis. The XCOBRA calculation is described in ANF topical reports XN-NF-79-59(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," and XN-NF-512(A), "The XN-3 Critical Power Correlation."

A.2.3 System Uncertainties

System measurement uncertainties are not fuel dependent. The values reported by the NSSS supplier for these parameters remain valid for the insertion of ANF fuel. The values used in the safety limit analysis are tabulated in the topical report XN-NF-524(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors."

A.2.4 Fuel Related Uncertainties

Fuel related uncertainties include power measurement uncertainty and core flow distribution uncertainty. The values used in the safety limit analysis are tabulated in the topical report XN-NF-524(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors." Power measurement uncertainties are established in the topical report XN-NF-80-19(A), Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors; Neutronics Methods for Design and Analysis."

A.3 SAFETY LIMIT CALCULATION

A statistical analysis for the number of fuel rods in boiling transition was performed using the methodology described in ANF topical report XN-NF-524(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors." With 500 Monte Carlo trials it was determined that for a minimum CPR value of 1.06 at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition with a confidence level of 95%.

```
**********************************************************
*
*  ---------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : .936  : .977  : 1.023 : 1.015 : 1.011 : 1.041 : 1.076 : 1.052 :
*  :       :       :       :       :       :       :       :       :
*  ---------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : .977  : 1.011 : .907  : 1.042 : 1.035 : .932  : .962  : 1.075 :
*  :       :       :       :       :       :       :       :       :
*  ---------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : 1.023 : .907  : 1.017 : .988  : .974  : .996  : .931  : 1.040 :
*  :       :       :       :       :       :       :       :       :
*  ---------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : 1.015 : 1.042 : .988  : .000  : .850  : .972  : 1.033 : 1.009 :
*  :       :       :       :       :       :       :       :       :
*  ---------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : 1.011 : 1.035 : .974  : .850  : .000  : .985  : 1.038 : 1.011 :
*  :       :       :       :       :       :       :       :       :
*  ---------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : 1.041 : .932  : .996  : .972  : .985  : 1.012 : .901  : 1.043 :
*  :       :       :       :       :       :       :       :       :
   ---------------------------------------------------------------
   :       :       :       :       :       :       :       :       :
   : 1.076 : .962  : .931  : 1.033 : 1.038 : .901  : .976  : 1.078 :
   :       :       :       :       :       :       :       :       :
   ---------------------------------------------------------------
   :       :       :       :       :       :       :       :       :
   : 1.052 : 1.075 : 1.040 : 1.009 : 1.011 : 1.043 : 1.078 : 1.054 :
   :       :       :       :       :       :       :       :       :
   ---------------------------------------------------------------
```

FIGURE A.1   WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF-4 FUEL)

```
****************************************************
*
*  ------------------------------------------------------------------------
*  :        :        :        :        :        :        :        :        :
*  :  .944  :  .962  : 1.011  : 1.044  : 1.043  : 1.010  :  .960  :  .943  :
*  :        :        :        :        :        :        :        :        :
*  ------------------------------------------------------------------------
*  :        :        :        :        :        :        :        :        :
*  :  .962  :  .980  : 1.064  :  .894  : 1.033  : 1.059  : 1.034  :  .961  :
*  :        :        :        :        :        :        :        :        :
*  ------------------------------------------------------------------------
*  :        :        :        :        :        :        :        :        :
*  : 1.011  : 1.064  : 1.010  :  .994  :  .982  : 1.002  :  .915  : 1.010  :
*  :        :        :        :        :        :        :        :        :
*  ------------------------------------------------------------------------
*  :        :        :        :        :        :        :        :        :
*  : 1.044  :  .894  :  .994  :  .000  :  .907  :  .980  : 1.032  : 1.042  :
*  :        :        :        :        :        :        :        :        :
*  ------------------------------------------------------------------------
*  :        :        :        :        :        :        :        :        :
*  : 1.043  : 1.033  :  .982  :  .907  :  .000  :  .988  :  .952  : 1.041  :
*  :        :        :        :        :        :        :        :        :
*  ------------------------------------------------------------------------
*  :        :        :        :        :        :        :        :        :
*  : 1.010  : 1.059  : 1.002  :  .980  :  .988  : 1.004  : 1.060  : 1.065  :
*  :        :        :        :        :        :        :        :        :
   ------------------------------------------------------------------------
   :        :        :        :        :        :        :        :        :
   :  .960  : 1.034  :  .915  : 1.032  :  .952  : 1.060  :  .966  : 1.053  :
   :        :        :        :        :        :        :        :        :
   ------------------------------------------------------------------------
   :        :        :        :        :        :        :        :        :
   :  .943  :  .961  : 1.010  : 1.042  : 1.041  : 1.065  : 1.053  : 1.019  :
   :        :        :        :        :        :        :        :        :
   ------------------------------------------------------------------------
```

FIGURE A.2 WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF XN-3 FUEL)

```
************************************************************
*
*    ---------------------------------------------------------------
*    :        :        :        :        :        :        :        :
*    : .950 :   .963 : 1.000 : 1.027 : 1.026 :   .999 :   .963 :   .950 :
*    :        :        :        :        :        :        :        :
*    ---------------------------------------------------------------
*    :        :        :        :        :        :        :        :
*    : .963 :   .981 : 1.052 :   .920 : 1.033 : 1.049 : 1.020 :   .963 :
*    :        :        :        :        :        :        :        :
*    ---------------------------------------------------------------
*    :        :        :        :        :        :        :        :
*    : 1.000 : 1.052 : 1.017 : 1.005 :   .997 : 1.011 :   .936 : 1.000 :
*    :        :        :        :        :        :        :        :
*    ---------------------------------------------------------------
*    :        :        :        :        :        :        :        :
*    : 1.027 :   .920 : 1.005 :   .000 :   .935 :   .996 : 1.033 : 1.027 :
*    :        :        :        :        :        :        :        :
*    ---------------------------------------------------------------
*    :        :        :        :        :        :        :        :
*    : 1.026 : 1.033 :   .997 :   .935 :   .000 : 1.002 :   .971 : 1.027 :
*    :        :        :        :        :        :        :        :
*    ---------------------------------------------------------------
*    :        :        :        :        :        :        :        :
*    : .999 : 1.049 : 1.011 :   .996 : 1.002 : 1.016 : 1.054 : 1.042 :
*    :        :        :        :        :        :        :        :
*    ---------------------------------------------------------------
     :        :        :        :        :        :        :        :
     : .963 : 1.020 :   .936 : 1.033 :   .971 : 1.054 :   .973 : 1.029 :
     :        :        :        :        :        :        :        :
     ---------------------------------------------------------------
     :        :        :        :        :        :        :        :
     : .950 :   .963 : 1.000 : 1.027 : 1.027 : 1.042 : 1.029 : 1.003 :
     :        :        :        :        :        :        :        :
     ---------------------------------------------------------------
```

FIGURE A.3   WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF XN-2 FUEL)

```
****************************************************
*
*  -------------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : .967  : .969  : .997  : 1.019 : 1.019 : .996  : .968  : .966  :
*  :       :       :       :       :       :       :       :       :
*  -------------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : .969  : .981  : 1.044 : .932  : 1.030 : 1.042 : 1.013 : .968  :
*  :       :       :       :       :       :       :       :       :
*  -------------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : .997  : 1.044 : 1.017 : 1.008 : 1.001 : 1.012 : .944  : .997  :
*  :       :       :       :       :       :       :       :       :
*  -------------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : 1.019 : .932  : 1.008 : .000  : .947  : 1.000 : 1.030 : 1.019 :
*  :       :       :       :       :       :       :       :       :
*  -------------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : 1.019 : 1.030 : 1.001 : .947  : .000  : 1.006 : .976  : 1.020 :
*  :       :       :       :       :       :       :       :       :
*  -------------------------------------------------------------------
*  :       :       :       :       :       :       :       :       :
*  : .996  : 1.042 : 1.012 : 1.000 : 1.006 : 1.017 : 1.047 : 1.032 :
*  :       :       :       :       :       :       :       :       :
   -------------------------------------------------------------------
   :       :       :       :       :       :       :       :       :
   : .968  : 1.013 : .944  : 1.030 : .976  : 1.047 : .975  : 1.020 :
   :       :       :       :       :       :       :       :       :
   -------------------------------------------------------------------
   :       :       :       :       :       :       :       :       :
   : .966  : .968  : .997  : 1.019 : 1.020 : 1.032 : 1.020 : 1.003 :
   :       :       :       :       :       :       :       :       :
   -------------------------------------------------------------------
```

FIGURE A.4 WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF XN-1 FUEL)

```
*********************************************************
*
* -------------------------------------------------------------------
* :      :      :      :      :      :      :      :      :
* : 1.03 : 1.00 :  .99 :  .99 :  .99 :  .99 : 1.00 : 1.03 :
* :      :      :      :      :      :      :      :      :
* -------------------------------------------------------------------
* :      :      :      :      :      :      :      :      :
* : 1.00 :  .97 :  .99 : 1.02 : 1.03 : 1.03 :  .99 : 1.00 :
* :      :      :      :      :      :      :      :      :
* -------------------------------------------------------------------
* :      :      :      :      :      :      :      :      :
* :  .99 :  .99 : 1.02 : 1.01 : 1.02 :  .91 : 1.03 :  .99 :
* :      :      :      :      :      :      :      :      :
* -------------------------------------------------------------------
* :      :      :      :      :      :      :      :      :
* :  .99 : 1.02 : 1.01 :  .91 :  .00 : 1.02 : 1.02 :  .99 :
* :      :      :      :      :      :      :      :      :
* -------------------------------------------------------------------
* :      :      :      :      :      :      :      :      :
* :  .99 : 1.03 : 1.02 :  .00 : 1.02 : 1.01 :  .99 :  .99 :
* :      :      :      :      :      :      :      :      :
* -------------------------------------------------------------------
* :      :      :      :      :      :      :      :      :
* :  .99 : 1.03 :  .91 : 1.02 : 1.01 :  .98 :  .99 :  .99 :
* :      :      :      :      :      :      :      :      :
* -------------------------------------------------------------------
  :      :      :      :      :      :      :      :      :
  : 1.00 :  .99 : 1.03 : 1.02 :  .99 :  .99 :  .97 : 1.00 :
  :      :      :      :      :      :      :      :      :
  -------------------------------------------------------------------
  :      :      :      :      :      :      :      :      :
  : 1.03 : 1.00 :  .99 :  .99 :  .99 :  .99 : 1.00 : 1.03 :
  :      :      :      :      :      :      :      :      :
  -------------------------------------------------------------------
```

FIGURE A.5 WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS
(GE FUEL)
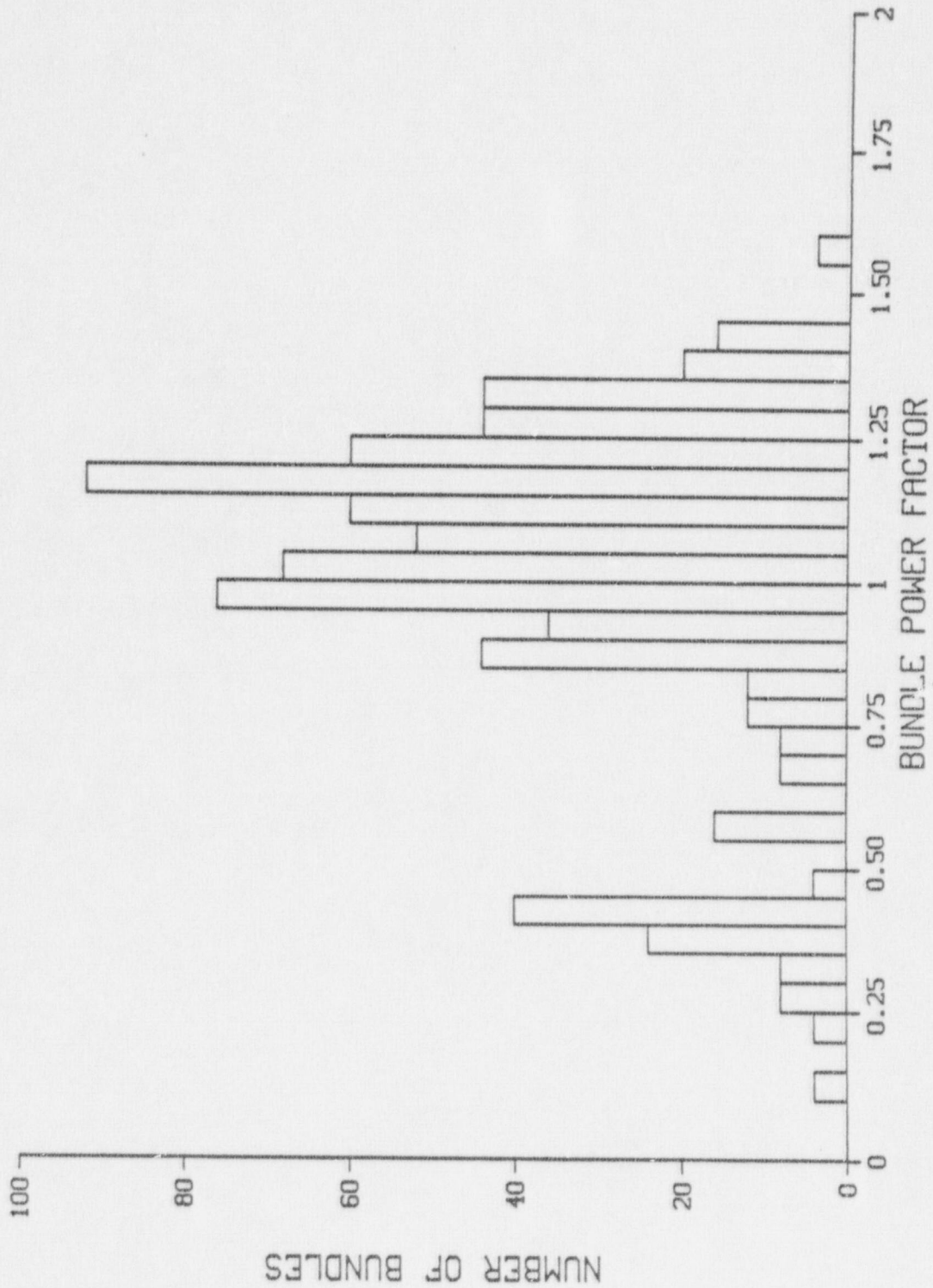
FIGURE A.6 RADIAL POWER HISTOGRAM FOR 1/4 CORE SAFETY LIMIT MODEL

WNP-2 CYCLE 5 PLANT TRANSIENT ANALYSIS

Distribution:

O. C. Brown
M. E. Byram
R. E. Collingham
S. J. Haynes
S. E. Jensen
J. E. Krajicek
S. L. Leonard
J. L. Maryott
L. A. Nielsen
G. L. Ritter
R. B. Stout/D. C. Kilian
H. E. Williamson

Y. U. Fresk/WPPSS (51)
Document Control (5)