ADVANCED NUCLEAR FUELS CORPORATION

ANF-88-01
Issue Date: 1/15/88

WNP-2 CYCLE 4 PLANT TRANSIENT ANALYSIS

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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 4 operation. For this analysis the Cycle 4 core was assumed to contain 428 ANF 8x8 and 336 GE P8x8R fuel assemblies.

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

2.0 SUMMARY

The Minimum Critical Power Ratios (MCPR) for potentially limiting plant system transient events at increased core flow* are shown in Table 2.1 for powers that bound allowable values (47 to 104% power) at increased core flow. The system transient MCPR values of Table 2.1 for the load rejection without bypass (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 MCPR values for the cases of Table 2.1 are 1.31 for GE and 1.30 for ANF fuel.

Also, an analysis was performed for the LRNB event at a cycle exposure of EOC -2000 MWd/MTU when a large number of control blades are still inserted in the core. The analysis showed that this system transient was insignificant relative to the control rod withdrawal event (CRWE)[2]. Thus, plant operating limits can be based on CRWE event for cycle exposures up to EOC -2000 MWd/MTU. 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) out of service and assuming technical specification scram speed (TSSS). The delta CPR results for these events are presented in Section 3.

The maximum system pressure was calculated for the containment isolation event, which is a rapid closure of all main steam isolation valves, using the scenario as specified by the ASME Pressure Vessel Code. This analysis shows that for WNP-2 Cycle 4 operation the safety valves have sufficient capacity and performance to prevent the pressure from reaching the established transient pressure safety limit of 110% of design pressure. The maximum

*The Cycle 2 transient events analyzed at the design basis power condition (104%) with increased core flow were found to bound the same transients analyzed at the design basis power (104%) and flow condition (100%) for WNP-2 Cycle 2. These results are shown in Reference 5.

system pressures predicted during the event are below the ASME limit of 1375 psig (110% of design pressure) 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 4 was confirmed for all fuel types using the methodology of Reference 6.

TABLE 2.1 THERMAL MARGIN SUMMARY FOR CYCLE 4

Transient	% Power/% Flow	Delta CPR/MCPR*	
		GE Fuel	ANF Fuel
Load Rejection** Without Bypass	104/106	0.25/1.31	0.24/1.30
Feedwater Controller** Failure	47/106	0.12/1.18	0.11/1.17
Loss of Feedwater*** Heating	Not Applicable	0.09/1.15	0.09/1.15

MAXIMUM VESSEL PRESSURE (PSIG)

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

*MCPR value using the 1.06 safety limit justified herein.

**These transients were evaluated with normal scram speed with RPT operable.

***WNP-2 plant specific bounding value, Reference 10.

3.0 TRANSIENT ANALYSIS FOR THERMAL MARGIN

3.1 Design Basis

System transient analyses to determine the most limiting type of thermal margin transient were performed at the increased core flow condition of 106%. As shown in Reference 5, system transients from the increased core flow condition bound thermal margin analyses 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 determine thermal margin 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 considers eight categories of potential system transient occurrences for Jet Pump BWRs in XN-NF-79-71[3]. The three most limiting transients are described here in detail to show the thermal margin for Cycle 4 of WNP-2. 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 analysis of the load rejection transient from the design basis power and increased core flow point for 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) and 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 LRWB transients with the RPT out of service with both normal scram speed (NSS) and technical specification scram speed (TSSS).

Because a significant number of control rods are inserted into the core at exposures less than end-of-cycle (EOC) minus 2000 MWd/MTU, the system transients are expected to be insignificant for cycle exposures less than this value. To confirm this, the LRWB was analyzed at the same 104% power/106% flow condition point for the end-of-cycle (EOC) minus 2000 MWd/MTU exposure condition for the bounding case of the RPT inoperable with TSSS. The delta CPR values for the GE and ANF fuels for this EOC minus 2000 MWd/MTU case are both 0.05. These delta CPR values are also shown in Table 3.3 and are significantly less than the delta CPR values for the control rod withdrawal error (CRWE) event reported in Reference 2. This shows that the delta CPR for the CRWE bounds plant operation up to EOC minus 2000 MWd/MTU. For Cycle 4 exposures greater than EOC minus 2000 MWd/MTU, the other 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 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 3.2.1. Sensitivity results have shown that the calculated delta CPR is insensitive to the rate of feedwater flow increase, that EOC conditions are bounding because rods are inserted for lower cycle exposure, and that 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 transient event was analyzed from the lowest allowed power (47%) a increased core flow. Figures 3.11 through 3.16 present key variables. The delta CPR values for the co-resident fuel types for these three 47% power/106% flow transients are shown in Table 3.3.

It has been determined that for a FWCF event that the control system signal to open the bypass valves passes directly to the bypass valves rather than be delayed by the pressure regulator. Thus, the bypass valves are opened earlier in the Cycle 4 analyses than in the earlier cycle analyses and results in a more realistic representation of the FWCF event. The effect on the FWCF analysis is that the compression wave produced by the turbine control valve closure is mitigated by the earlier opening of the bypass valves, and the core power excursion due to void collapse is diminished which reduces the magnitude of all calculated FWCF delta CPR's. Table 3.3 shows that all of the delta CPR/MCPR values are less than the delta CPR/MCPR value for the 104/106 LRNB event with RPT operable and normal scram speed.

3.2.3 Loss Of Feedwater Heating

Loss of Feedwater Heating (LOFH) events were evaluated 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 operating 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 noted in Section 2.0 of this report, the WNP-2 MCPR safety limit for Cycle 4 continues to be 1.06; hence the LOFH transient requires a MCPR operating 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 4 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. The FFTR analysis was performed for a 65°F temperature reduction. This FFTR analysis is 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 FFTR transients of plus 0.02 and minus 0.01 are applicable to these respective anticipated operational occurrence (AOO) events. That is, these LRNB and FFTR limit changes are applicable when Cycle 4 reactor operation is being extended with thermal coastdown at FFTR conditions.

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	101.8 Mlb/hr
Core Bypass Flow	13.2 Mlb/hr
Core Inlet Enthalpy	529.2 BTU/lbm
Vessel Pressures	
Steam Dome	1035. psia
Upper Plenum	1049. psia
Core	1055. psia
Lower Plenum	1072. psia
Turbine Pressure	974. psia
Feedwater/Steam Flow	15.0 Mlb/hr
Feedwater Enthalpy	403.5 BTU/lbm
Recirculating Pump Flow (per pump)	17.3 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
Scram Insertion Times (normal)**	0.404 sec to Notch 45
	0.660 sec to Notch 39
	1.504 sec to Notch 25
	2.624 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)	580. BTU/hr-ft2-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

*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 sec
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*	373	119	1170	0.25	0.24
LRNB RPT Inoperable, NSS	505	125	1181	0.32	0.29
LRNB RPT Operable, TSSS**	442	125	1175	0.32	0.30
LRNB RPT Inoperable, TSSS	574	131	1189	0.38	0.35
LRNB EOC -2000 MWD/MTU RPT Inoperable, TSSS	284	110	1168	0.05	0.05
FWCF (47% Power/106% Flow), NSS RPT Operable	103	50	1010	0.12	0.11
FWCF (47% Power/106% Flow), NSS RPT Inoperable	129	52	1020	0.15	0.14
FWCF (47% Power/106% Flow), TSSS RPT Operable	110	51	1013	0.14	0.12
MSIV Closure With Flux Scram	669	133	1315	N/A	

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

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*Normal Scram Speed (NSS)

**Technical Specification Scram Speed (TSSS)

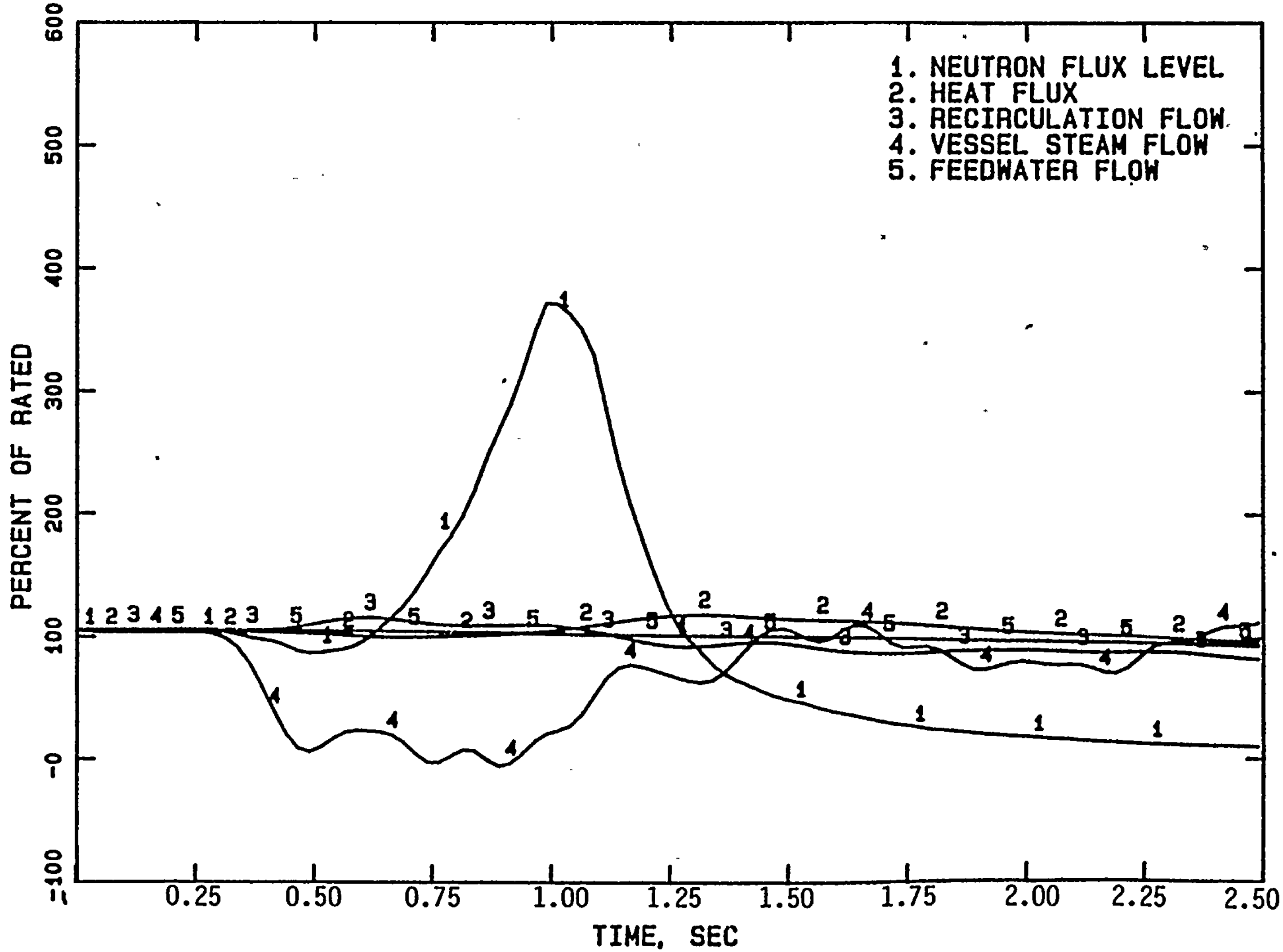

Figure 3.1 Load Rejection Wi[illegible]t Bypass Results, RPT Operable, Normal Scram Spe[illegible]

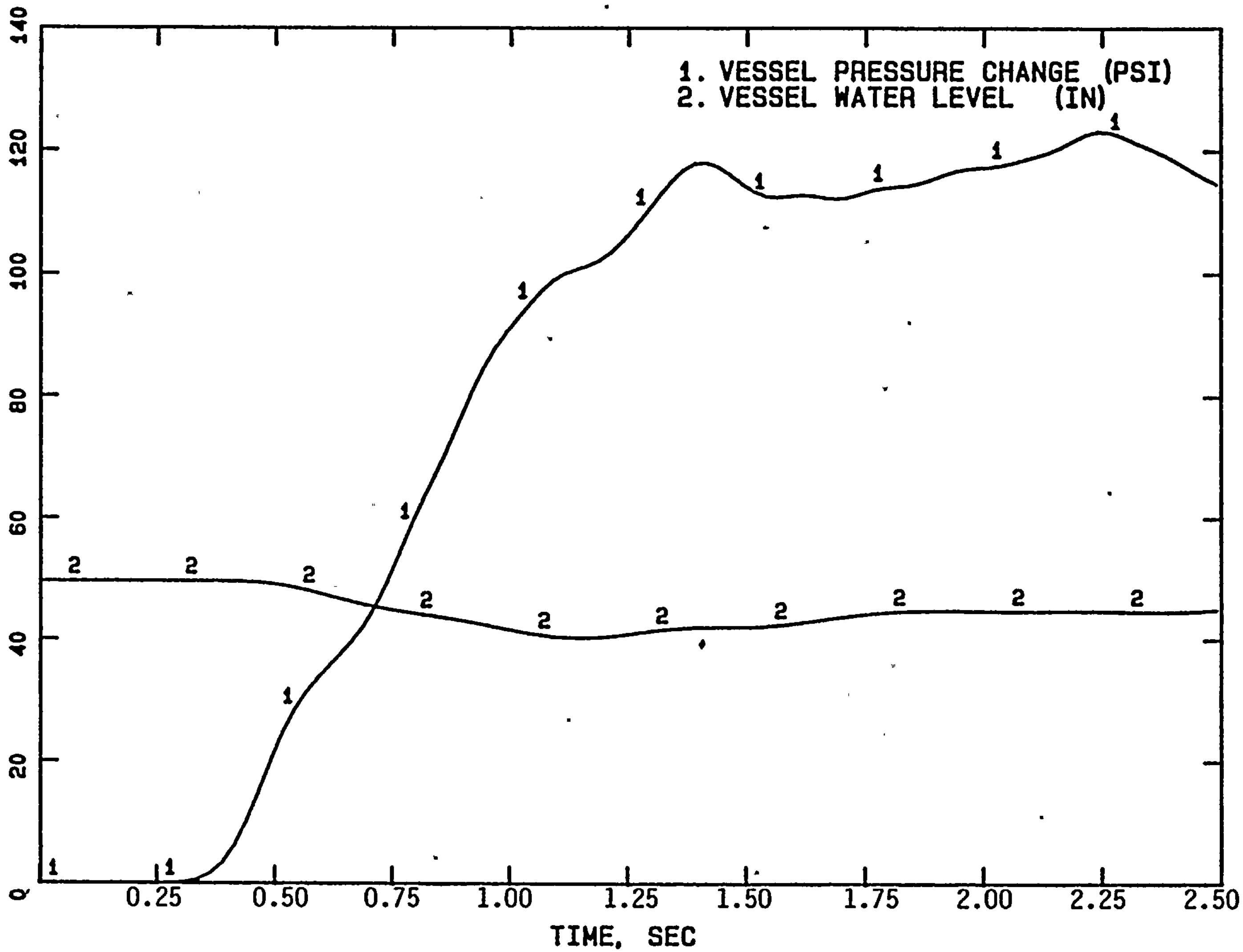

Figure 3.2 Load Rejection Without Bypass Results, RPT Operable, Normal Scram Speed

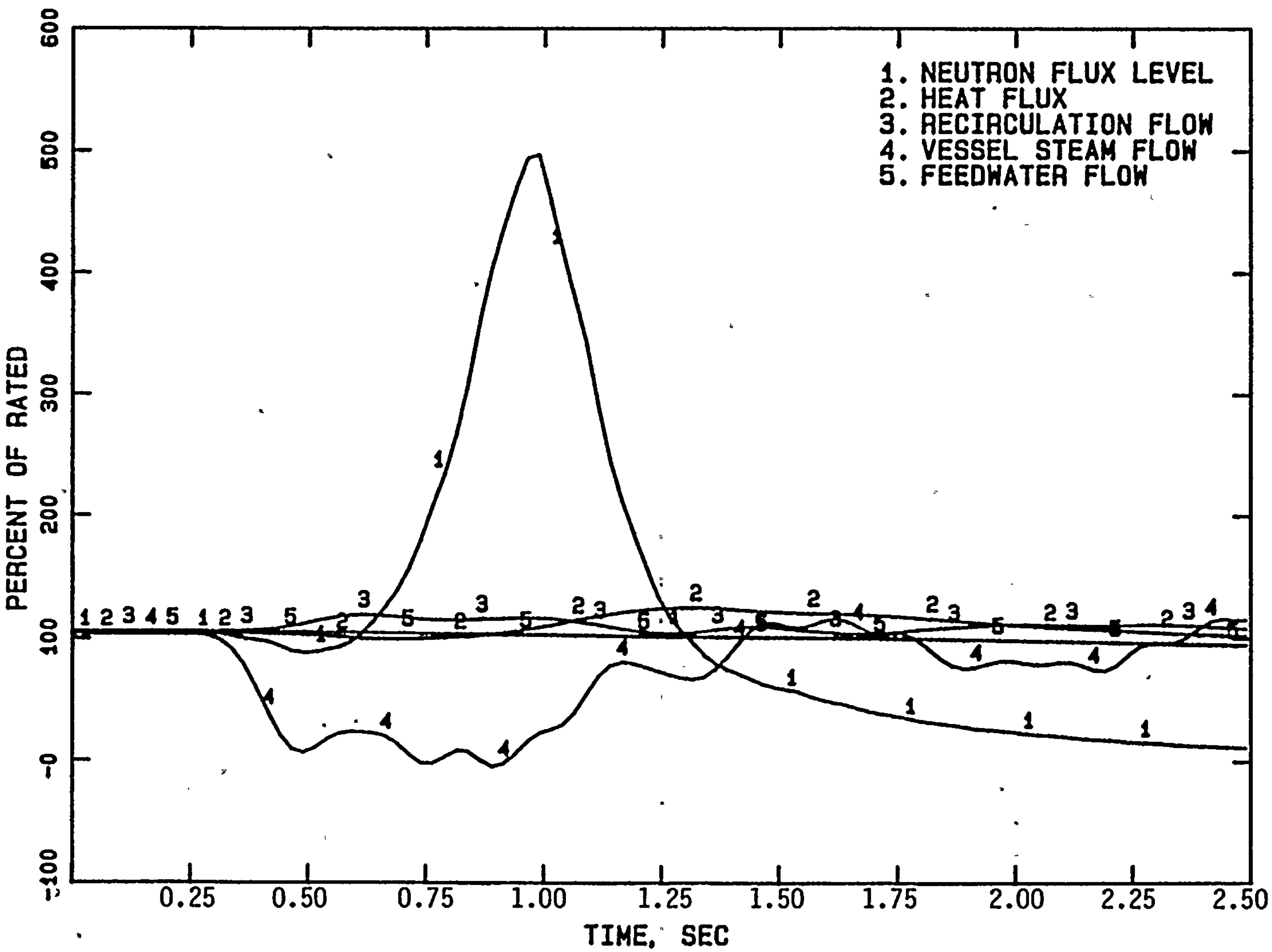

Figure 3.3 Load Rejection Without Bypass Results, RPT Inoperable, Normal Scram Speed

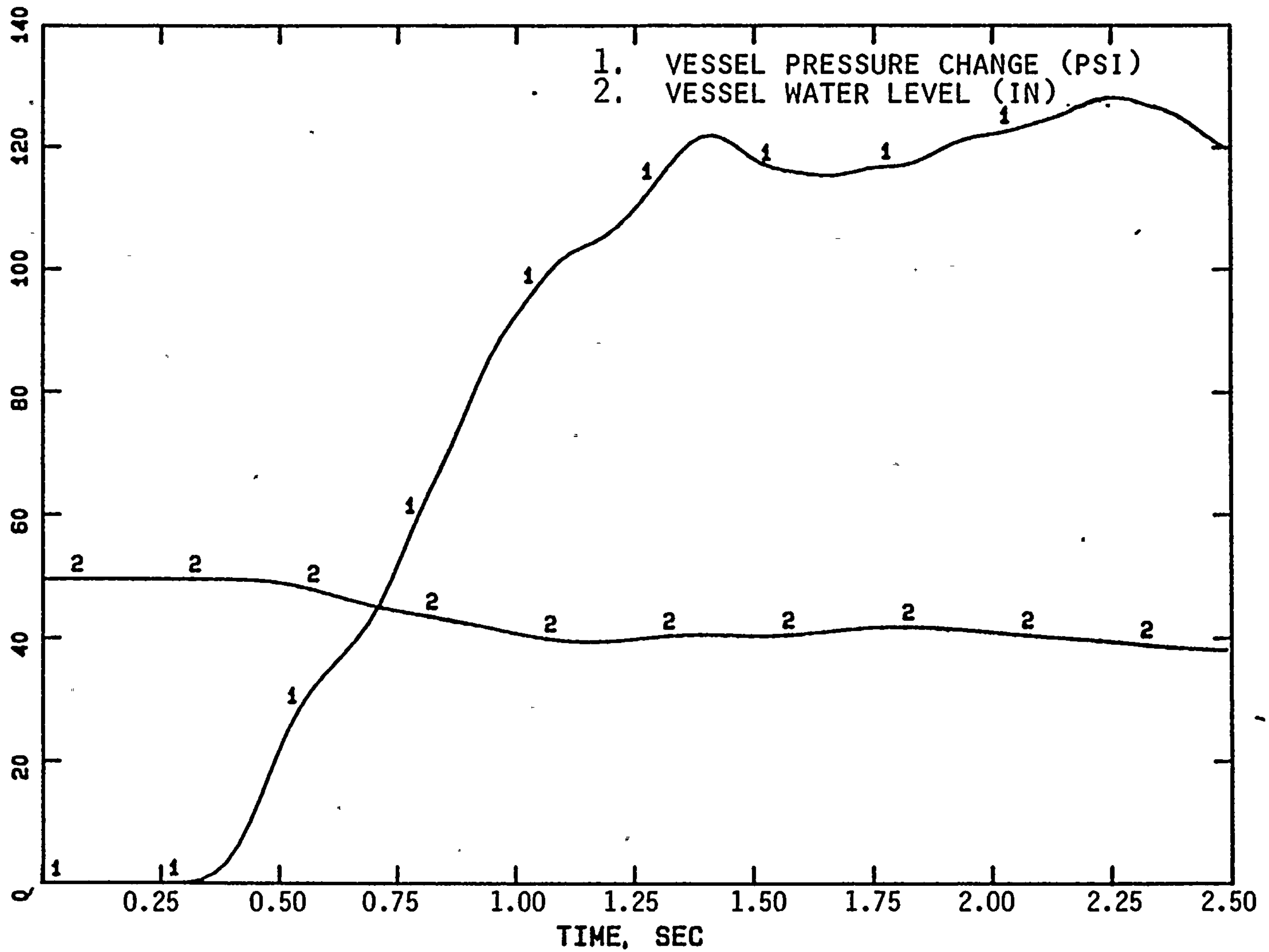

Figure 3.4 Load Rejection Without Bypass Results, RPT Inoperable, Normal Scram Speed

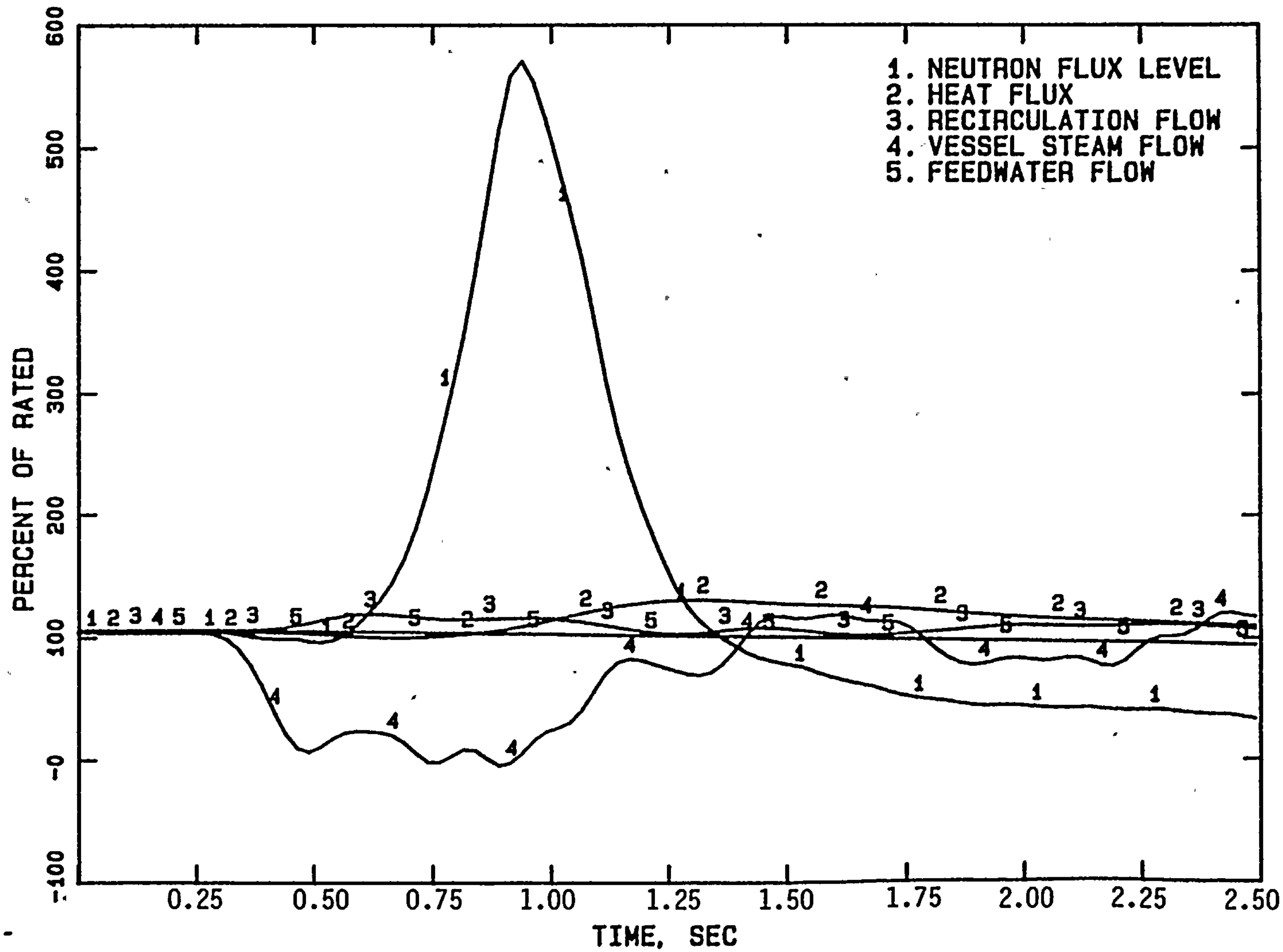

Figure 3.5 Load Rejection Without Bypass Results, RPT Operable, Tech. Spec. Scram Speed

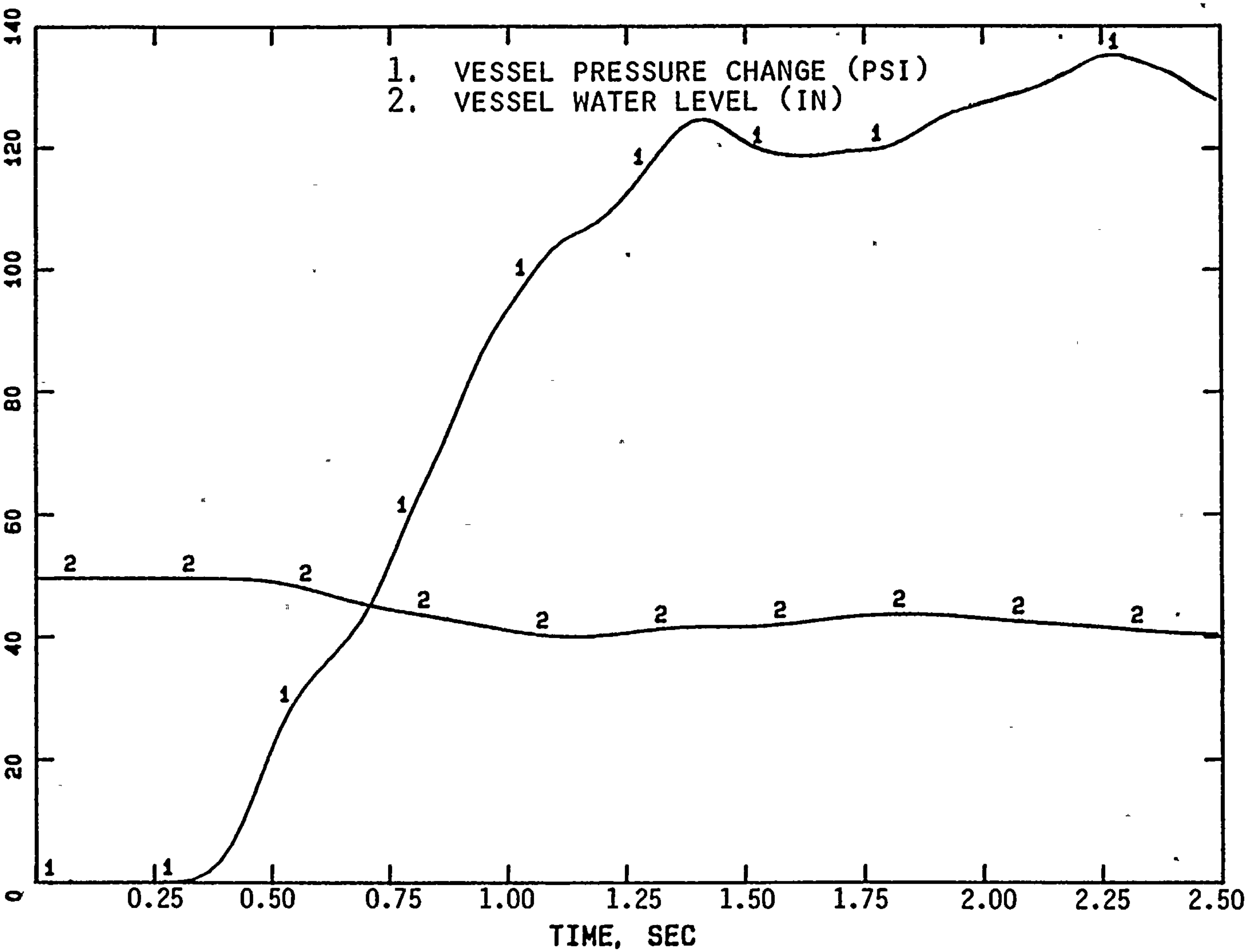

Figure 3.6 Load Rejection Without Bypass Results, RPT Operable, Tech. Spec. Scram Speed

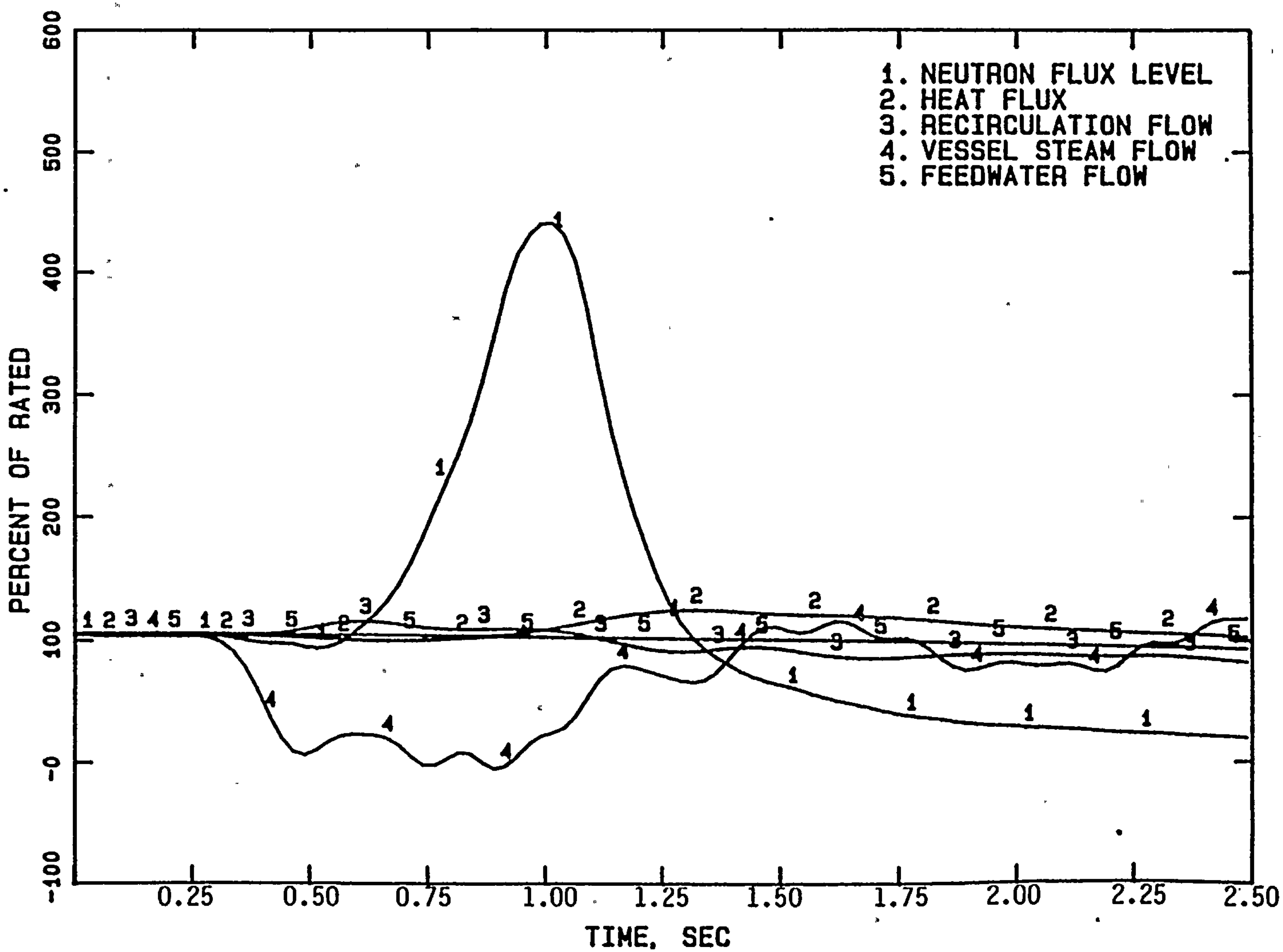

Figure 3.7 Load Rejection Withou[illegible]pass Results, RPT Inoperable,
Tech. Spec. Scram Spe[illegible]

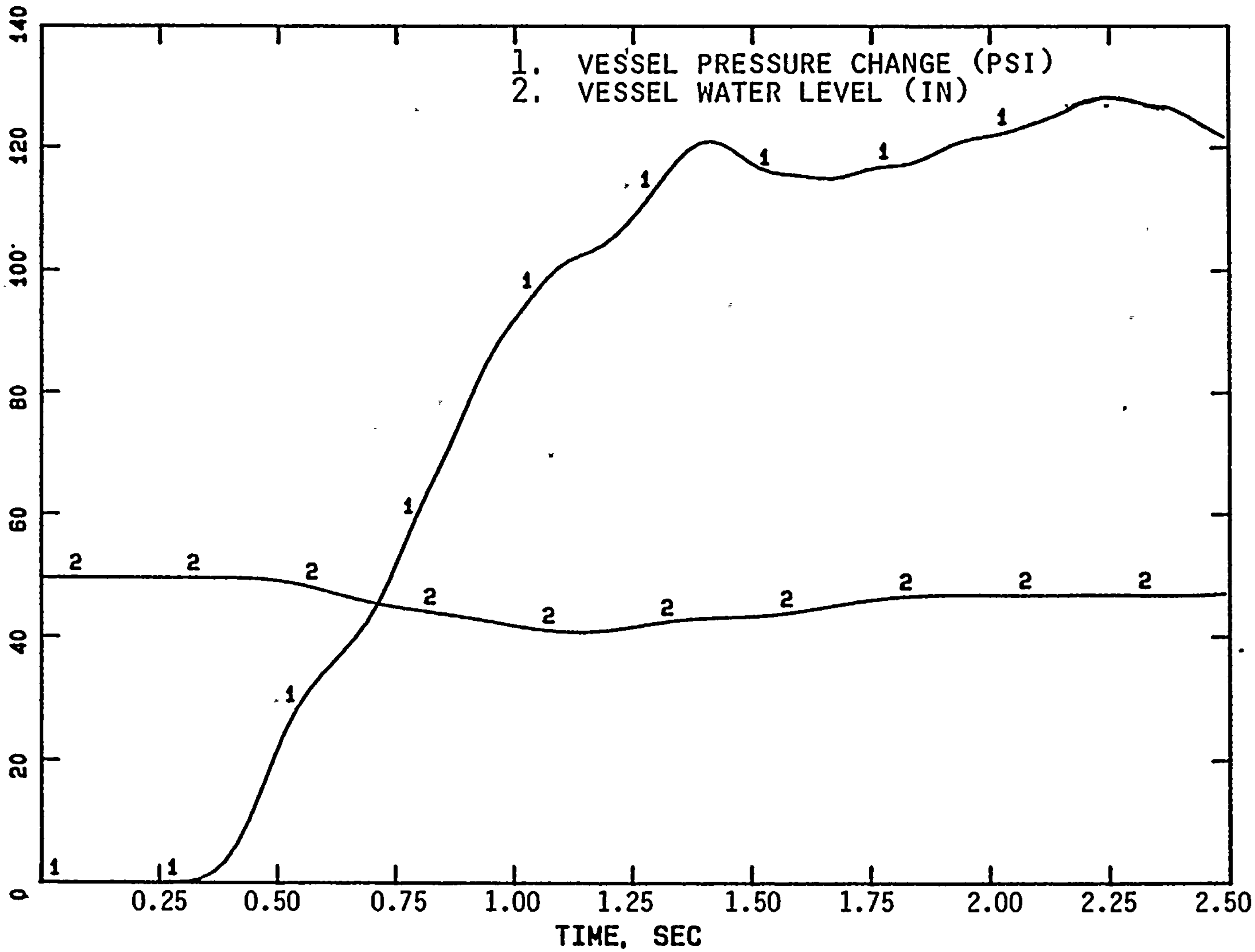

Figure 3.8 Load Rejection Without Bypass Results, RPT Inoperable, Tech. Spec. Scram Speed

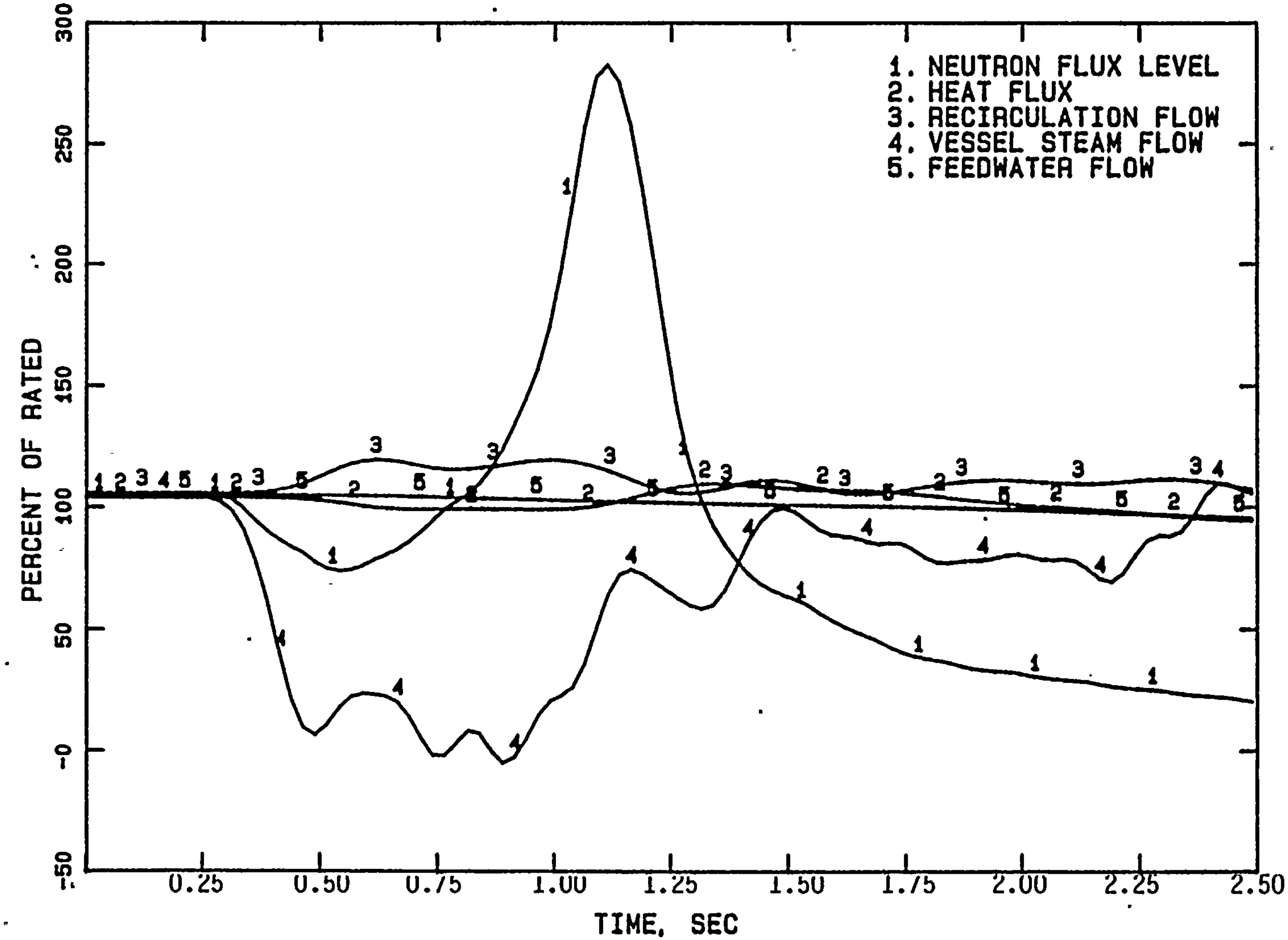

Figure 3.9 Load Rejection Without Bypass Results, End-Of-Cycle Minus 2000 MWD/MTU Exposure, RPT Inoperable, Tech. Spec. Scram Speed

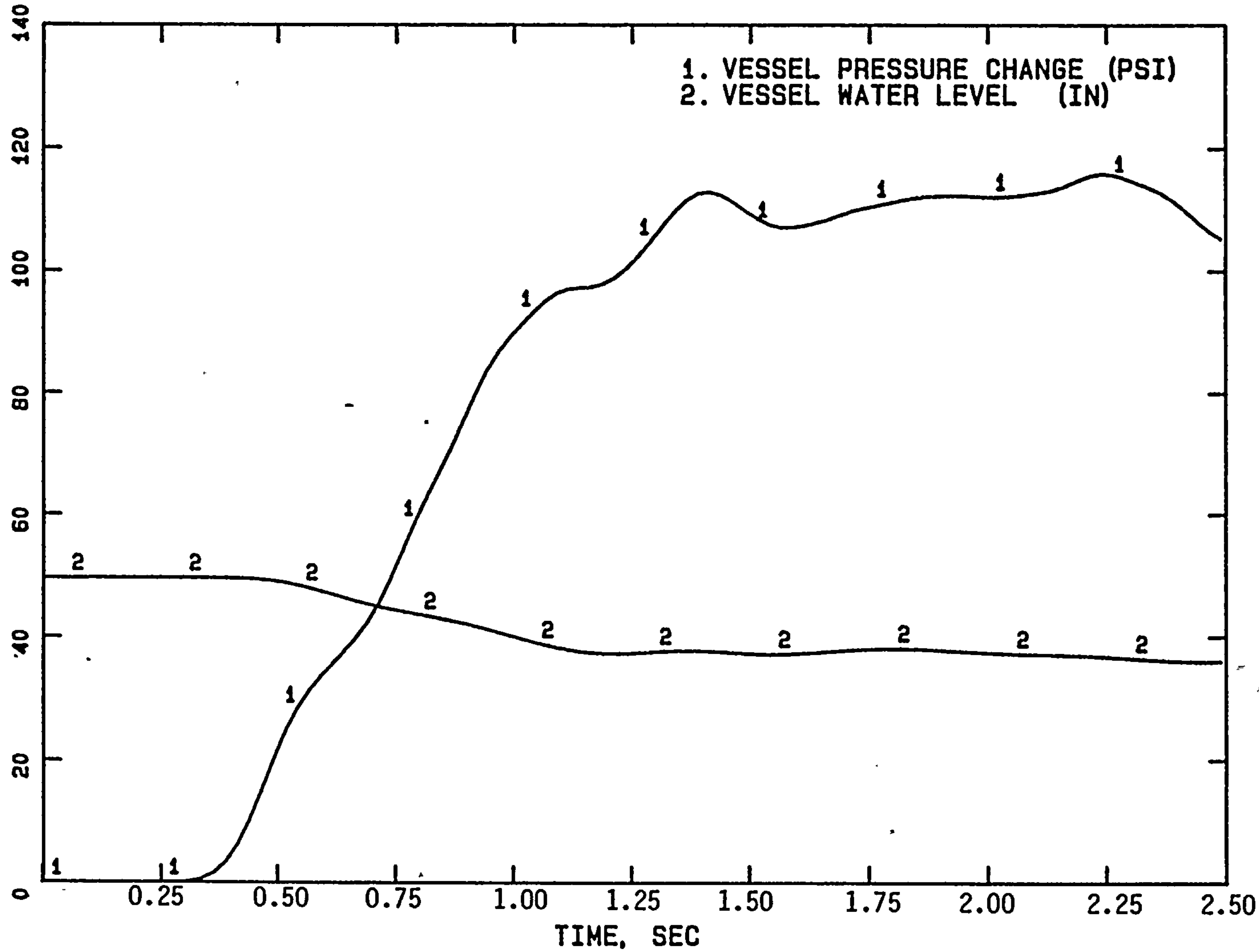

Figure 3.10 Load Rejection Without Bypass Results, End-Of-Cycle Minus 2000 MWD/MTU Exposure, RPT Inoperable, Tech. Spec. Scram Speed

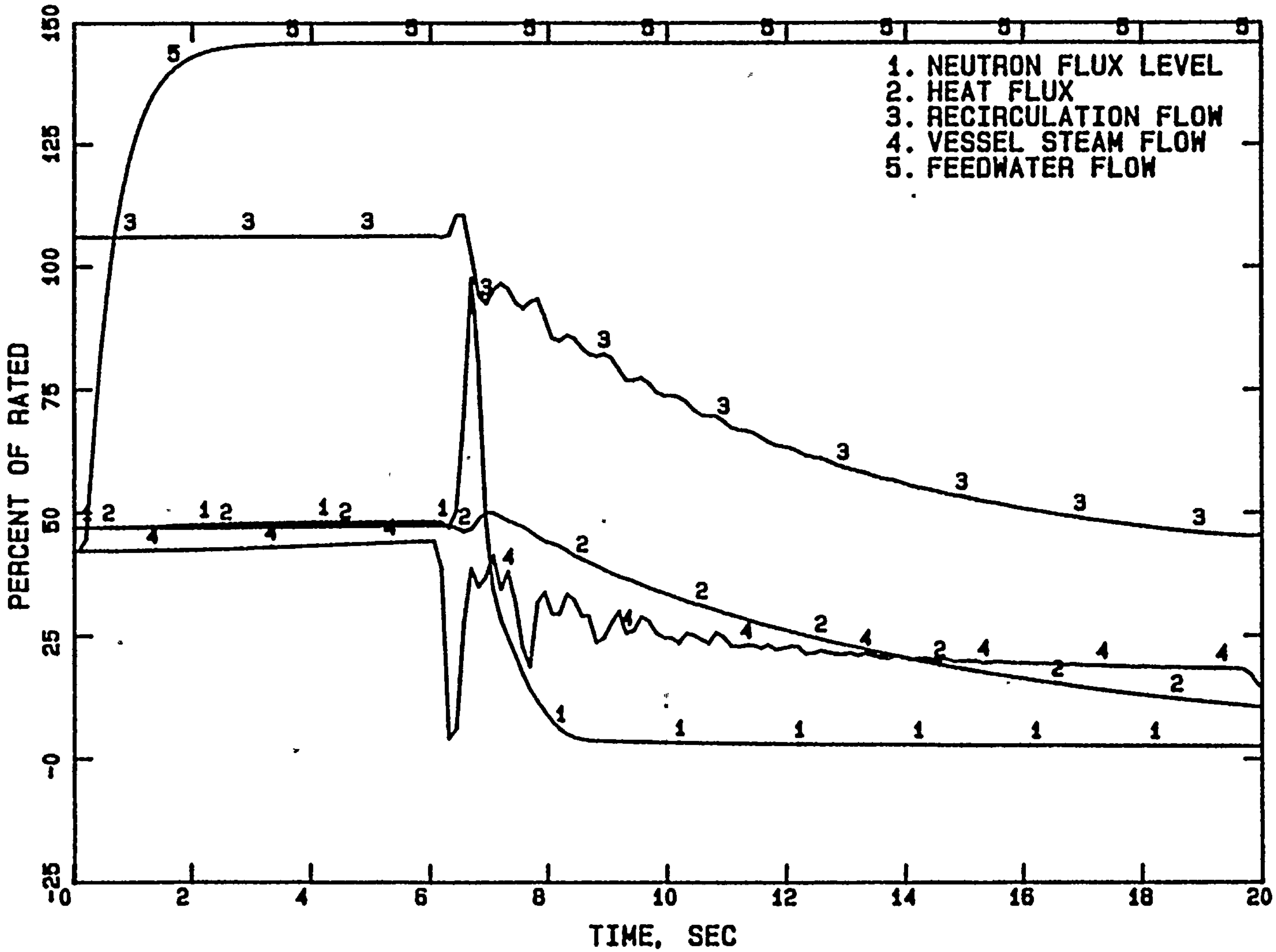

Figure 3.11 Feedwater Controller Failure Results For 47% Power And 106% Flow With Norm[illegible]cram Speed

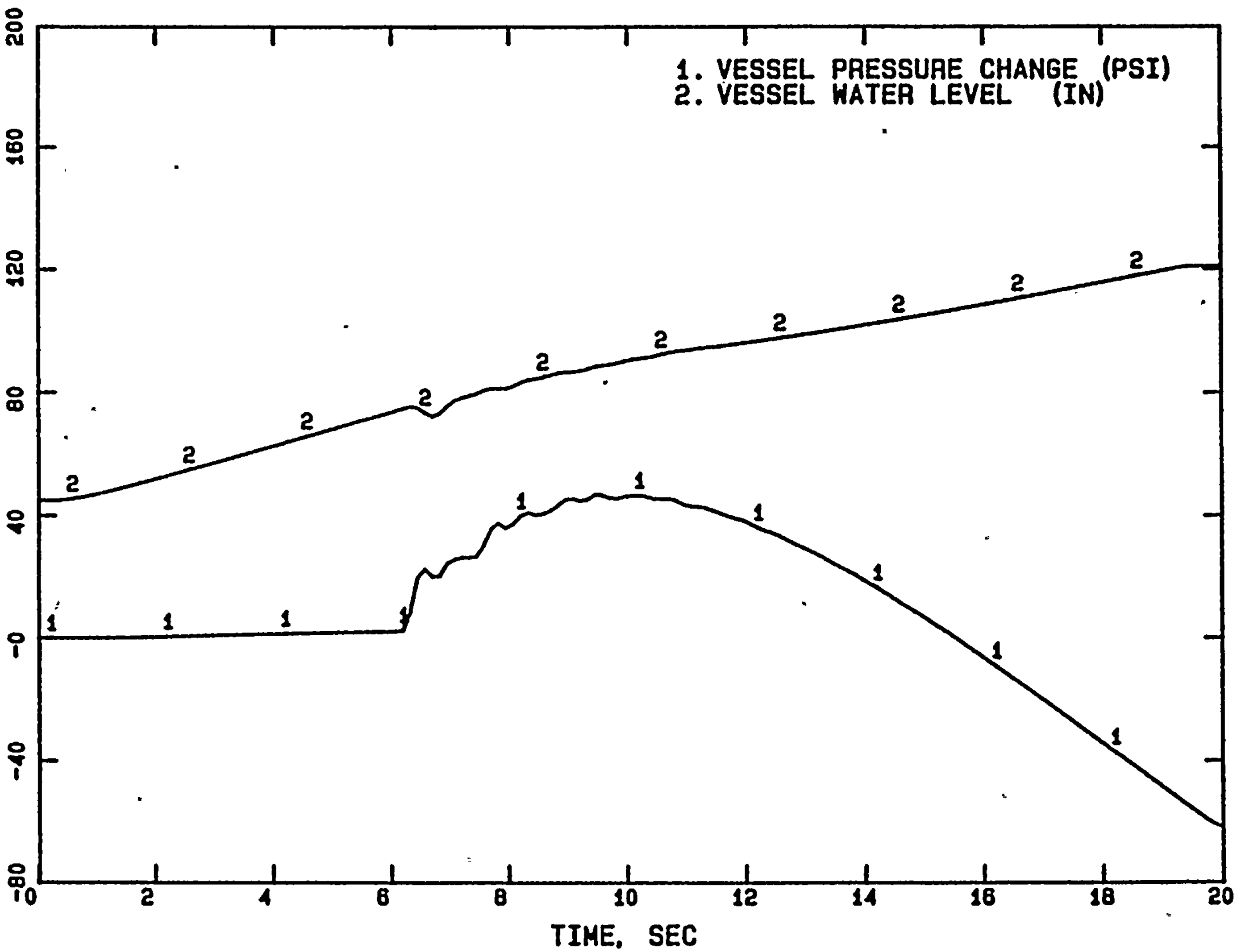

Figure 3.12 Feedwater Controller Failure Results For 47% Power And 106% Flow With Normal Scram Speed

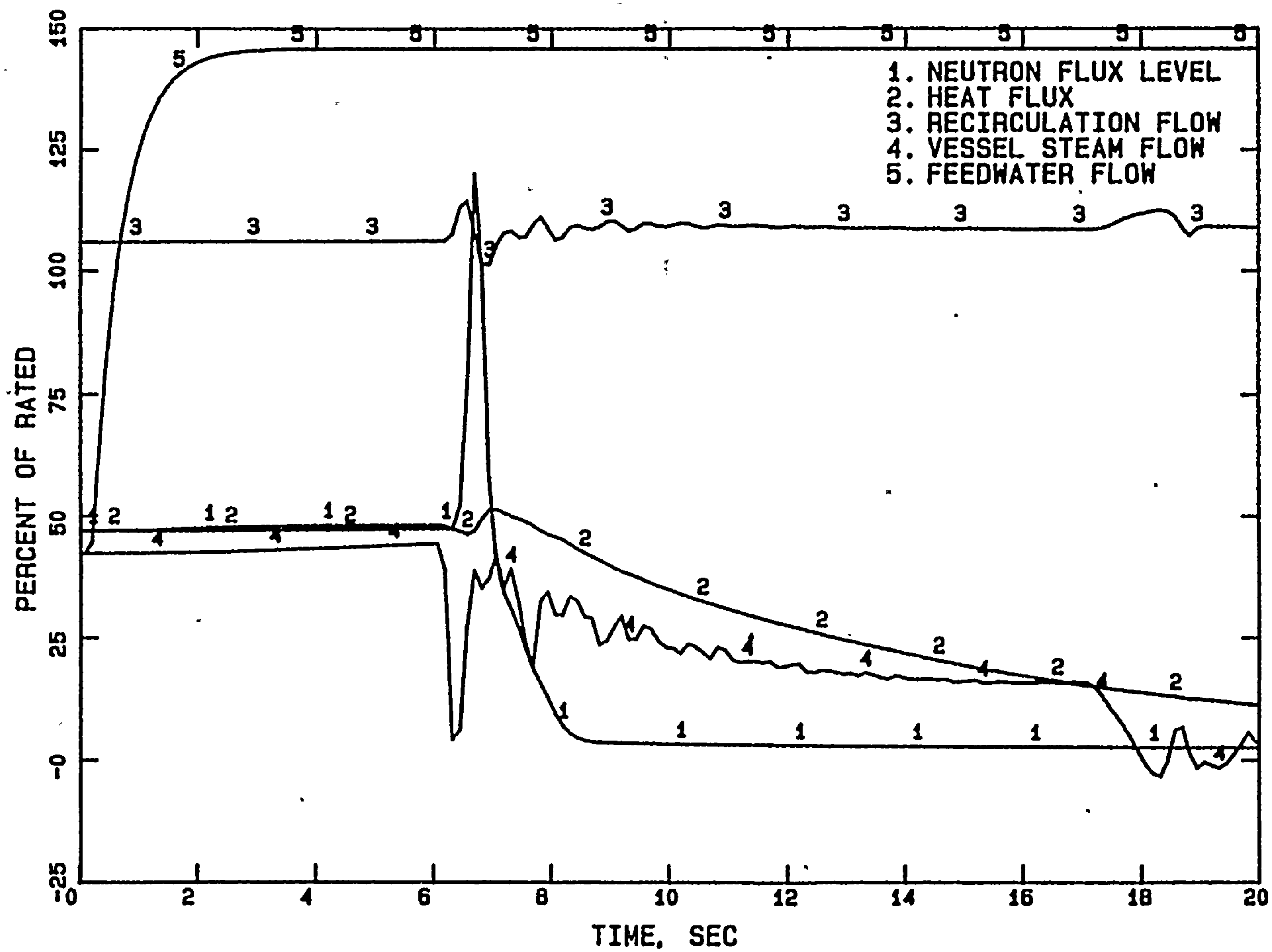

Figure 3.13 Feedwater Controller Failure Results, RPT Inoperable, Normal Scram Speed

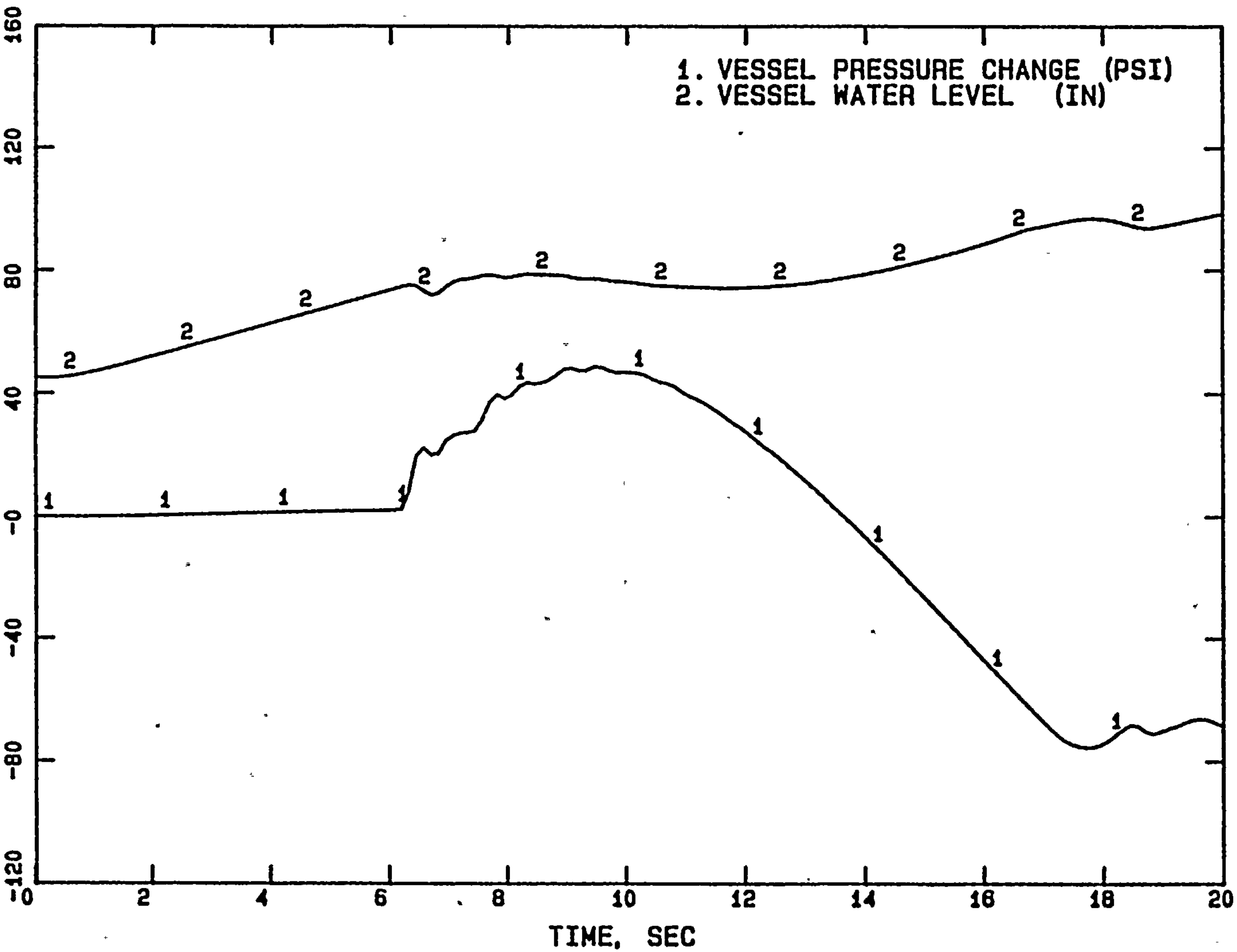

Figure 3.14 Feedwater Controller Failure Results, RPT Inoperable, Normal Scram Speed

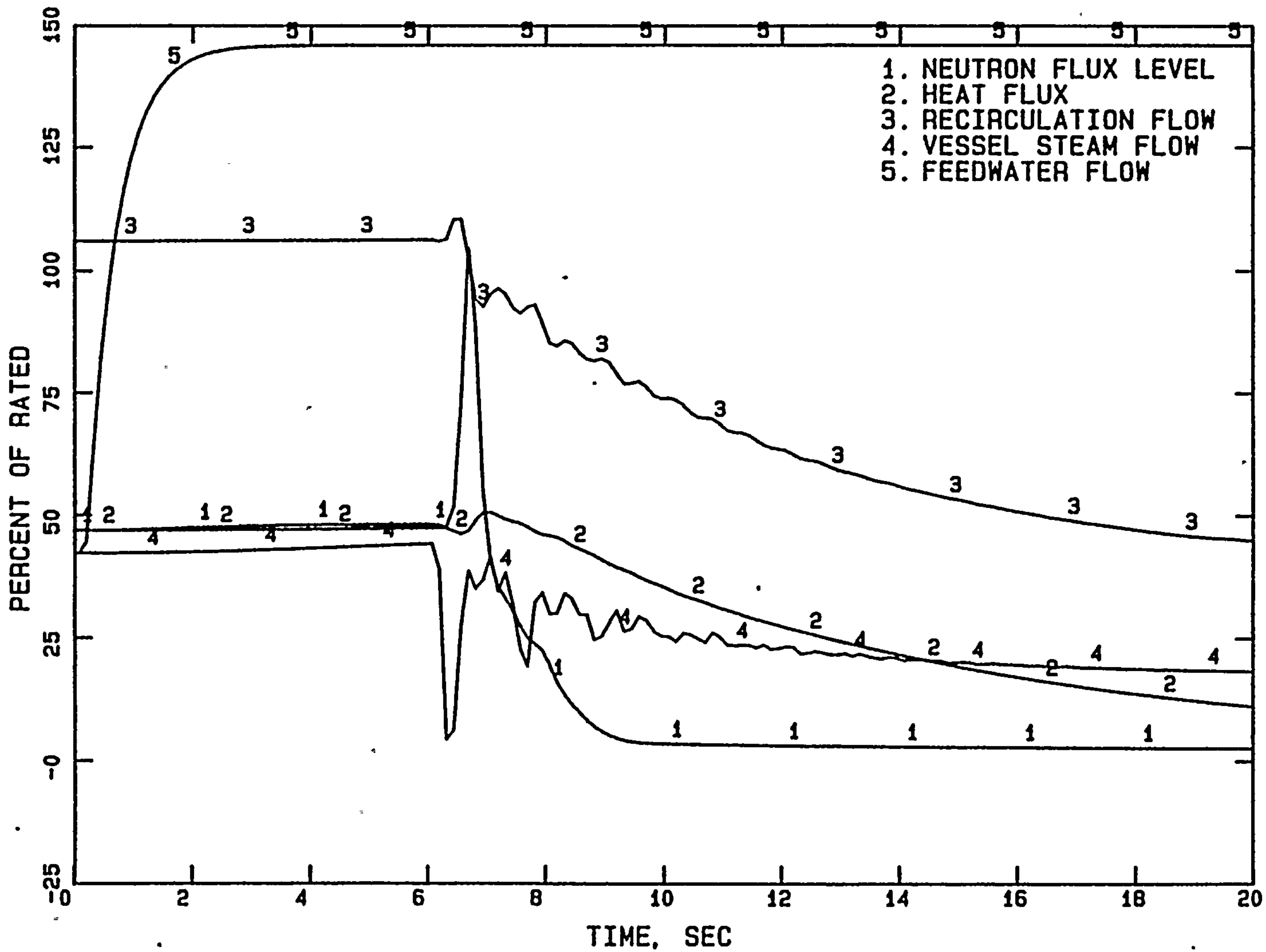

Figure 3.15 Feedwater Controller Failure Results For 47% Power And 106% Flow With Tech. Spec. Scram Speed

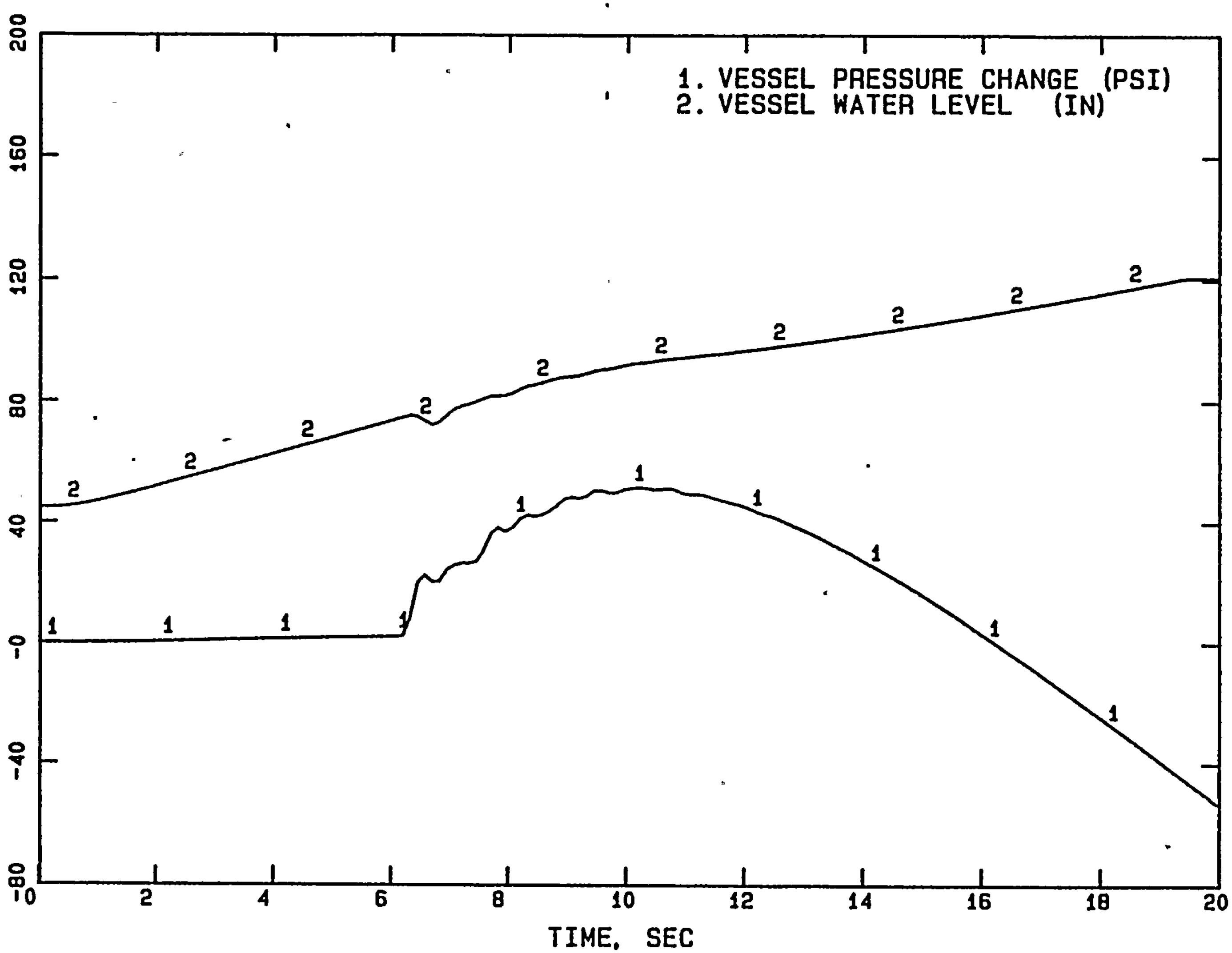

Figure 3.16 Feedwater Controller Failure Results For 47% Power And 106% Flow With Tech. Spec. Scram Speed

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 maximum pressure calculated 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 Table 2.1 and 3.3 for the design basis point.

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 4. Thus, the reduced flow MCPR operating limit for WNP-2 Cycle 2 is applicable to Cycle 4. This reduced flow MCPR operating limit 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	1.12
80	1.17
70	1.23
60	1.32
50	1.42
40	1.55

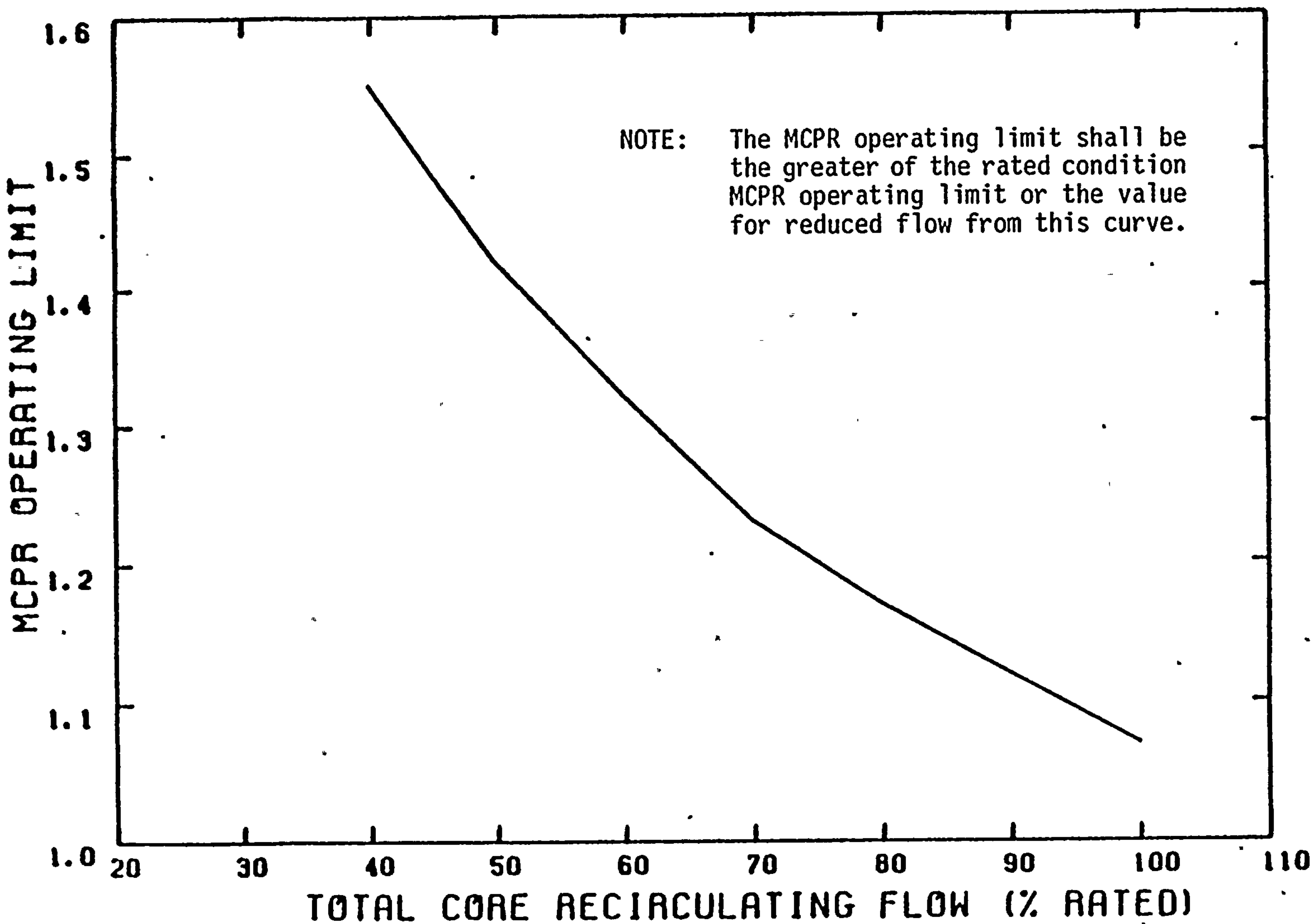

Figure 5.1 Reduced Flow MCPR Operating Limit

6.0 REFERENCES

1. J. E. Krajicek, "WNP-2 Plant Transient Analysis With Final Feedwater Temperature Reduction," XN-NF-87-92, Advanced Nuclear Fuels Corporation, Richland, WA 99352, June 1987.

2. J. E. Krajicek, "Supply System Nuclear Project Number 2 (WNP-2) Cycle 4 Reload Analysis," ANF-88-02, Advanced Nuclear Fuels Corporation, Richland, WA 99352, January 1988.

3. R. H. Kelley, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," XN-NF-79-71(P), Revision 2 (as supplemented), Exxon Nuclear Company, Inc., Richland, WA 99352, November 1981.

4. M. J. Ades and B. C. Fryer, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," XN-NF-84-105(A), Volume 1, Volume 1 Supplement 1, Volume 1 Supplement 2 and XN-NF-84-105(P), Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, Richland, WA 99352, February 1987 and July 1987.

5. J. B. Edgar, Letter to WPPSS, Supplemental Licensing Analysis Results, ENWP-86-0067, Exxon Nuclear Company, Inc., Richland, WA 99352, April 15, 1986.

6. T. W. Patten, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," XN-NF-524(A), Revision 1, Exxon Nuclear Company, Inc., Richland, WA 99352, November 1983.

7. T. L. Krysinski and J. C. Chandler, "Exxon Nuclear Methodology for Boiling Water Reactors; THERMEX Thermal Limits Methodology; Summary Description," XN-NF-80-19(A), Volume 3, Revision 2, Exxon Nuclear Company, Inc., Richland, WA 99352, January 1987.

8. K. R. Merckx, "RODEX2 Fuel Rod Mechanical Response Evaluation Model," XN-NF-81-58(A), Revision 2, Exxon Nuclear Company, Inc., Richland, WA 99352, March 1984.

9. S. E. Jensen, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors: Revised Methodology for Including Code Uncertainties in Determining Operating Limits for Rapid Pressurization Transients in BWRs," XN-NF-79-71(A), Revision 2, Supplements 1, 2, and 3, Exxon Nuclear Company, Inc., Richland, WA 99352, March 1986.

10. "Exxon Nuclear Methodology for Boiling water Reactors Neutronics Methods for Design Analysis," XN-NF-80-19(A), Volume 1, Supplements 1 and 2, Exxon Nuclear Company, Inc., Richland, WA 99352, March 1983.

11. J. E. Krajicek, "WNP-2 Cycle 2 Plant Transient Analysis," XN-NF-85-143, Revision 1, Exxon Nuclear Company, Inc., Richland, WA 99352, February 1986.

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 3 and the predicted operating conditions for Cycle 4 were evaluated to identify the design basis power distributions used in the Cycle 4 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-4.

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 Rectors; 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%.

LL 0.93	L 0.95	ML 1.02	M 1.06	M 1.06	ML 1.02	L 0.95	LL 0.92
L 0.95	ML 0.97	H 1.08	ML 0.87	H 1.04	H 1.07	M 1.04	L 0.95
ML 1.02	H 1.08	H 1.01	H 1.00	H 0.98	H 1.00	ML 0.90	ML 1.02
M 1.06	ML 0.87	H 1.00	W 0.00	M 0.90	H 0.97	H 1.03	M 1.06
M 1.06	H 1.04	H 0.98	M 0.90	W 0.00	H 0.99	M 0.93	M 1.05
ML 1.02	H 1.07	H 1.00	H 0.97	H 0.99	H 1.00	H 1.06	M 1.08
L 0.95	M 1.04	ML 0.90	H 1.03	M 0.93	H 1.06	ML 0.96	ML 1.07
LL 0.92	L 0.95	ML 1.02	M 1.06	M 1.05	M 1.08	ML 1.07	L 1.03

Figure A.1 WNP-2 Cycle 4 Safety Limit Local Peaking Factors (ANF XN-3 Fuel)

LL 0.95	L 0.96	ML 1.00	M 1.03	M 1.03	ML 1.00	L 0.96	LL 0.95
L 0.96	ML 0.98	H 1.05	ML 0.92	H 1.03	H 1.05	M 1.02	L 0.96
ML 1.00	H 1.05	H 1.02	H 1.01	H 1.00	H 1.01	ML 0.94	ML 1.00
M 1.03	ML 0.92	H 1.01	W 0.00	M 0.93	H 1.00	H 1.03	M 1.03
M 1.03	H 1.03	H 1.00	M 0.93	W 0.00	H 1.00	M 0.97	M 1.03
ML 1.00	H 1.05	H 1.01	H 1.00	H 1.00	H 1.02	H 1.05	M 1.04
L 0.96	M 1.02	ML 0.94	H 1.03	M 0.97	H 1.05	ML 0.97	ML 1.03
LL 0.95	L 0.96	ML 1.00	M 1.03	M 1.03	M 1.04	ML 1.03	L 1.00

Figure A.2 WNP-2 Cycle 4 Safety Limit Local Peaking Factors (ANF XN-1, -2 Fuel)

LL 1.03	L 1.00	ML 0.99	M 0.99	M 0.99	ML 0.99	L 1.00	LL 1.03
L 1.00	M 0.99	H 1.03	H 1.02	MH 0.99	MH 0.99	ML 0.97	L 1.00
ML 0.99	H 1.03	L 0.91	H 1.02	H 1.01	MH 0.98	MH 0.99	ML 0.99
M 0.99	H 1.03	H 1.02	W 0.00	H 1.02	H 1.01	MH 0.99	M 0.99
M 0.99	H 1.02	H 1.01	L 0.91	W 0.00	H 1.02	H 1.02	M 0.99
ML 0.99	MH 0.99	H 1.02	H 1.01	H 1.02	L 0.91	H 1.03	ML 0.99
L 1.00	ML 0.97	MH 0.99	H 1.02	H 1.03	H 1.03	M 0.99	L 1.00
LL 1.03	L 1.00	ML 0.99	M 0.99	M 0.99	ML 0.99	L 1.00	LL 1.03

Figure A.3 WNP-2 Cycle 4 Safety Limit Local Peaking Factors (GE Fuel)

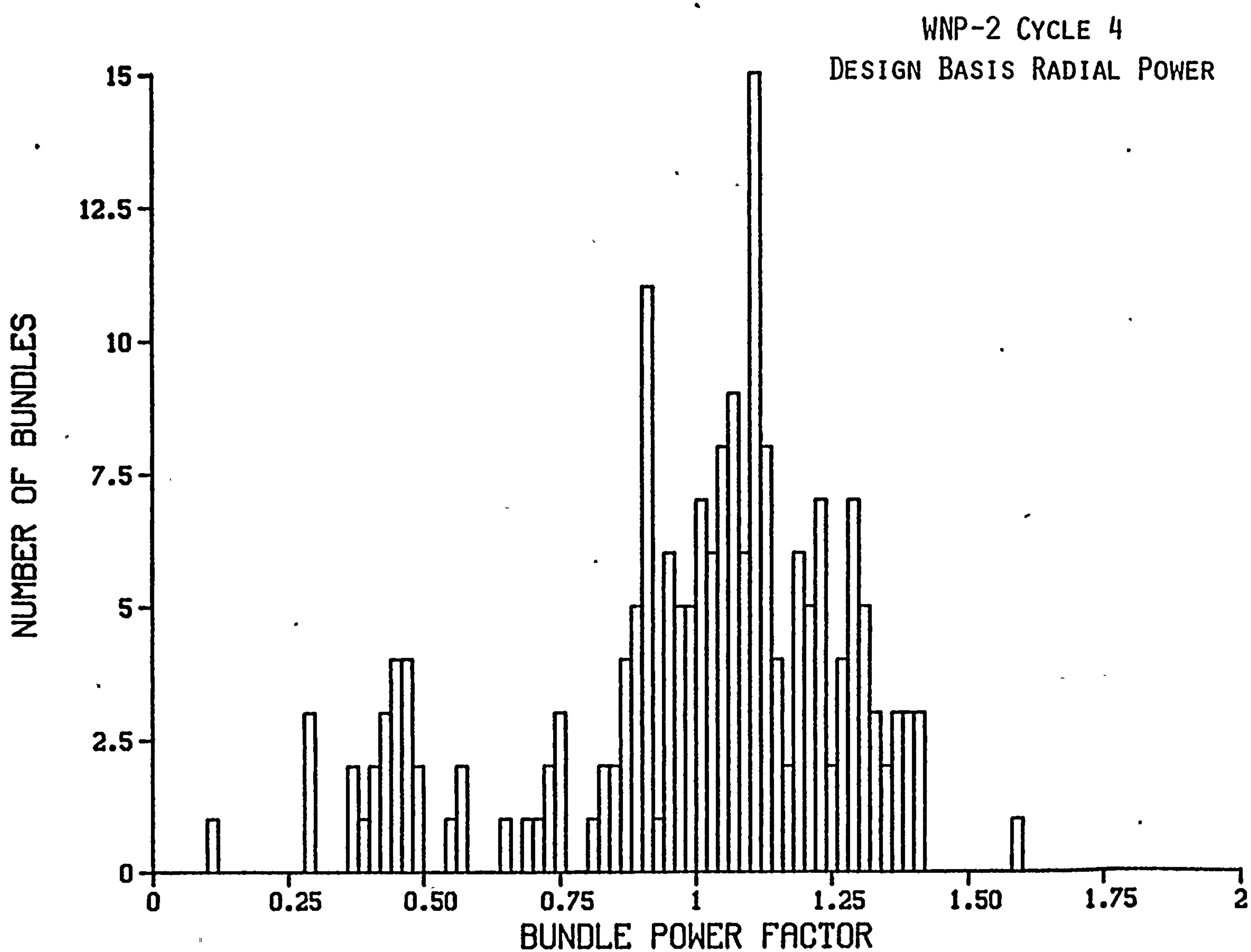

Figure A.4 Radial Power Histogram For 1/4 Core Safety Limit Model

ANF-88-01
Issue Date: 1/15/88

WNP-2 CYCLE 4 PLANT TRANSIENT ANALYSIS

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