ADVANCED NUCLEAR FUELS CORPORATION

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WNP-2 SINGLE LOOP OPERATION ANALYSIS

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TABLE OF CONTENTS

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LIST OF TABLES

LIST OF FIGURES

ACKNOWLEDGMENT

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1.0 INTRODUCTION

Advanced Nuclear Fuels Corporation (ANF) has performed postulated transient and accident analyses for the Supply System Nuclear Project Number 2 (WNP-2) reactor with a recirculation pump (or loop) out of service. The purpose of the analyses is to demonstrate that two loop operational limits provide protection for the maximum power single loop operation (SLO) condition.

The analyses considered the following events:

- Load Rejection Without Bypass (LRWB)
- Feedwater Controller Failure (FWCF)
- Pump Trip
- Recirculation Flow Runup
- Pump Seizure Accident

The ECCS analysis for the SLO condition is reported in Reference 1. The conclusions of these analyses are applicable to future fuel cycles containing ANF/NSSS vendor fuels of the current 8x8 design.

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2.0 SUMMARY

The most limiting transient events and the pump seizure accident have been analyzed for the maximum expected power state during single loop operation (SLO) of WNP-2. The analyses were performed using current ANF transient analysis methodology for a core configuration representative of Cycle 3.

The results of the SLO transient analyses are summarized in Table 2.1. The two loop MCPR operating limits (rated conditions) bound the requirements for single loop operation. Therefore, the single loop transient analyses need not be performed on a cycle by cycle basis and the two loop MCPR operating limits applicable for a cycle are appropriate for single loop conditions.

Operation in the single loop mode results in higher uncertainties for core flow, radial power and axial power. Considering these SLO uncertainties, the MCPR safety limit was determined to increase by 0.01 to 1.07. However, the two loop MCPR limits at SLO flow conditions bound the required MCPR limits for SLO conditions including the higher safety limit MCPR.

A postulated pump seizure accident was evaluated for SLO conditions. The event is less severe than the design basis loss of coolant accident (LOCA). The radiological consequences of this accident are well within the 10 CFR 100 limits.

TABLE 2.1 SUMMARY OF SLO TRANSIENT ANALYSES

Event	Required SLO MCPR Limit GE	Required SLO MCPR Limit ANF	2 Loop MCPR Limits GE	2 Loop MCPR Limits ANF
LRWB	1.24	1.21	1.32	1.30
FWCF	1.09	1.09	1.32	1.30
Pump Trip	1.09	1.07	1.32	1.30

3.0 TRANSIENT ANALYSES

3.1 Analysis Bases

The WNP-2 single loop transient analyses were performed using ANF methodology (Reference 2) consistent with that applied to the normal reload analyses. Reference 3 requested that the analyses support plant operation to 75% of rated thermal power. Normally the transient analyses are performed at 104.2% of rated core thermal power which corresponds to the 105% steam flow condition. For consistency, these analyses were performed at 104.2% of the 75% power state or 2596.9 Mwt. The core flow for analysis purposes was assumed to be 54.0 Mlbm/hr. The steam flow at this power level was taken to be 10.79 Mlbm/hr. A conservative dome pressure of 1020 psia was used in the analyses.

The system conditions at which the SLO transients were evaluated are summarized in Table 3.1. The steam flow/feedwater enthalpy characteristic from Reference 4 was used to initialize the plant transient simulation code COTRANSA (Reference 5). A jet pump M-ratio of 3.2 was used for initialization of the COTRANSA model at 54.0 Mlbm/hr core flow. In addition, the following assumptions are made for all analyses performed:

1. Normal scram speed (NSS).
2. Technical specification scram delay.
3. Integral power multiplier of 110%[5].

3.2 Load Rejection Without Bypass

At cycle exposures equal to or greater than EOC -2000 MWd/MTU, and rated or increased (106% of rated) core flows, the pressurization transient events such as the load rejection without bypass (LRWB) and the feedwater controller failure (FWCF) usually establish the MCPR operating limits. However, at the reduced power and flow conditions for SLO relative to rated conditions, there

is a reduction in the steam flow to the turbine. With the lower steam flow, the pressurization of the reactor vessel is reduced in comparison to rated conditions when the turbine control valve is closed following the generator load rejection signal. The resulting power excursion and associated thermal margin reduction are less than that for the full power case.

Figures 3.1 and 3.2 present the time variance of critical reactor and plant parameters from the analysis of the load rejection without bypass transient at the SLO reduced power and flow condition. The analysis assumes normal scram speed and recirculation pump trip (RPT). The delta CPR and other peak conditions during the event are shown in Table 3.2. The appropriate delta CPR results from the WNP-2, Cycle 3 plant transient analysis (Reference 6) are also provided for comparison.

3.3 Feedwater Controller Failure

The other limiting pressurization event is the feedwater controller failure (FWCF) to maximum demand. This event results in the maximum amount of subcooled feedwater being introduced into the vessel causing a core power increase followed by a high water level isolation signal and a turbine trip. Because of the reduced core and recirculation flows at the SLO condition relative to rated conditions, the increased subcooling due to the high feedwater flow takes longer to reach the core and the high water level trip occurs earlier, limiting the power rise prior to the turbine trip.

As with the LRWB event, the feedwater controller failure at the SLO power and flow is less severe than that for the full power and flow condition. The time variation of the pertinent reactor and plant parameters are shown in Figures 3.3 and 3.4. The results are tabulated in Table 3.2.

3.4 Recirculation Pump Trip

The recirculation pump trip transient is modelled to create the most rapid decrease in pump speed. Figures 3.5 and 3.6 illustrate the time variation of the plant parameters for this event. The peak conditions are tabulated in Table 3.2. As expected, this event is less severe than all the other transients and is bounded by the Cycle 3 reload analysis.

3.5 Recirculation Flow Runup

In the single loop configuration, the additional constraint of the reduced flow MCPR operating limits is no longer required. The 2 pump flow runup which would encroach upon the MCPR safety limit is not possible as the pump in the idle loop is not running. An inadvertent start of the idle pump cannot affect flow appreciably as the pump is interlocked to prevent starting unless it's associated flow control valve is at the minimum position.

Operation in single loop is precluded above the 80% rod line when the total core flow is less than 39% of rated core flow. This limits the most severe pump runup to a flow increase from 39% to approximately 50% core flow. This flow increase is not of sufficient magnitude to violate the MCPR safety limit if the transient initiates from the two loop MCPR operating limits. This can be seen from the WNP-2, Cycle 3 plant transient analysis (Reference 6, Figure 5.1).

TABLE 3.1 ANALYSIS CONDITIONS FOR SLO OPERATION

Reactor Thermal Power (1.042 x 75%)	2596.9 Mwt
Core Flow	54.0 Mlbm/hr
Core In-Channel Flow	48.12 Mlbm/hr
Core Bypass Flow	5.88 Mlbm/hr
Idle Jet Pump Back Flow	11.9 Mlbm/hr
Core Inlet Enthalpy	510.8 Btu/lbm
Vessel Pressures	
Steam Dome	1020.0 psia
Upper Plenum	1025.0 psia
Core Pressure	1029.7 psia
Lower Plenum	1035.0 psia
Jet Pump M-Ratio	3.2
Recirculation Pump Flow	15.7 Mlbm/hr
Turbine Pressure	960.5 psia
Feedwater/Steam Flow	10.79 Mlbm/hr

TABLE 3.2 RESULTS OF SLO PLANT TRANSIENT ANALYSES

Event	Maximum Neutron Flux (% Rated)	Maximum Core Average Heat Flux (% Rated)	Maximum System Pressure (psig)	Delta CPR GE SLO	ANF SLO	GE CY3	ANF CY3
LRWB	143	82.7	1154	0.17	0.14	0.25	0.23
FWCF	80.6	79.8	1121	0.02	0.02	0.26*	0.24*
Pump Trip	78.2	78.5	1020	0.02	0.00	NA	

* Analyzed at 47% power, 106% flow.

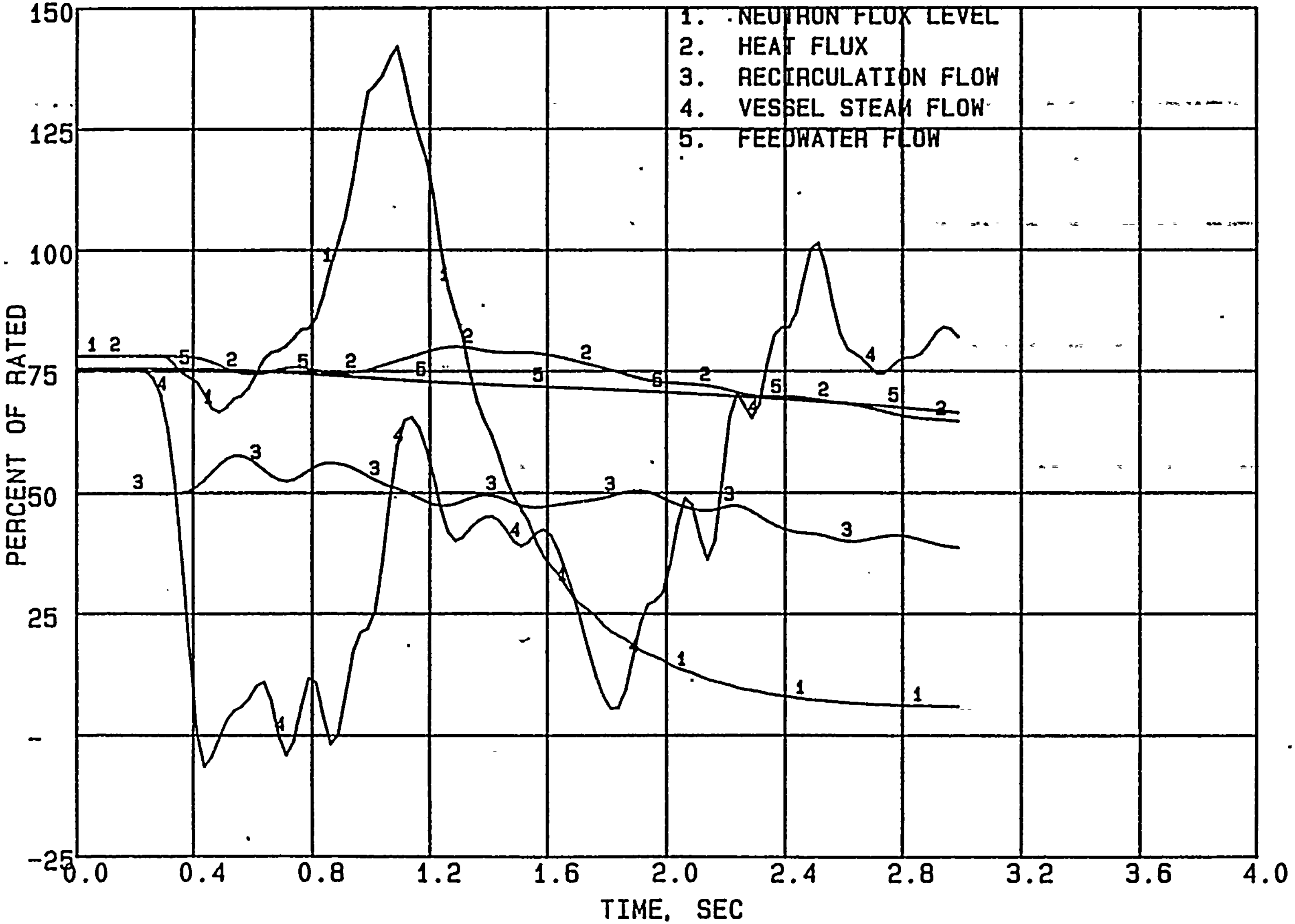

FIGURE 3.1 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT OPERABLE,
NORMAL SCRAM SPEED

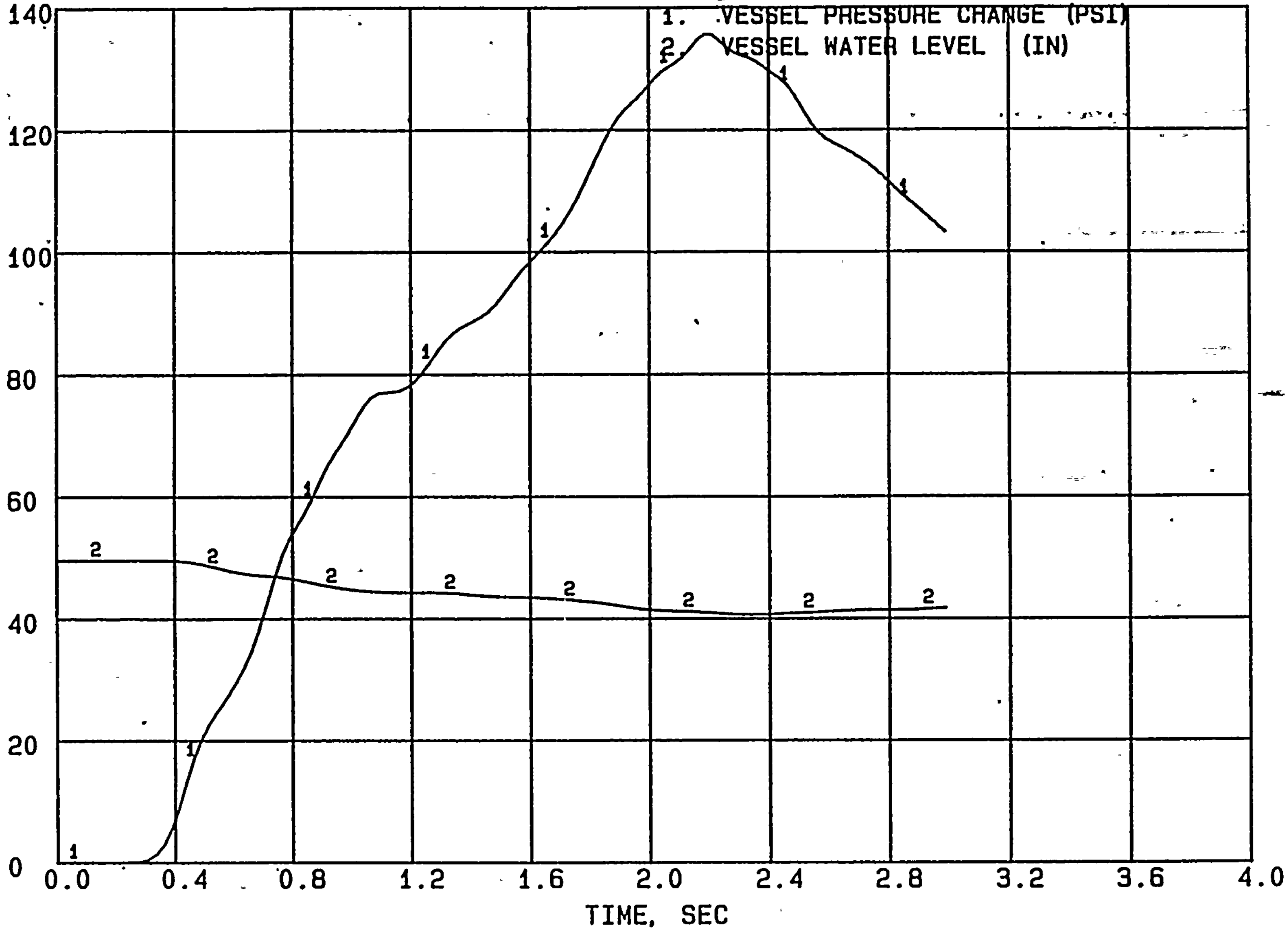

FIGURE 3.2 LOAD REJECTION WITHOUT BYPASS RESULTS, RPT OPERABLE, NORMAL SCRAM SPEED

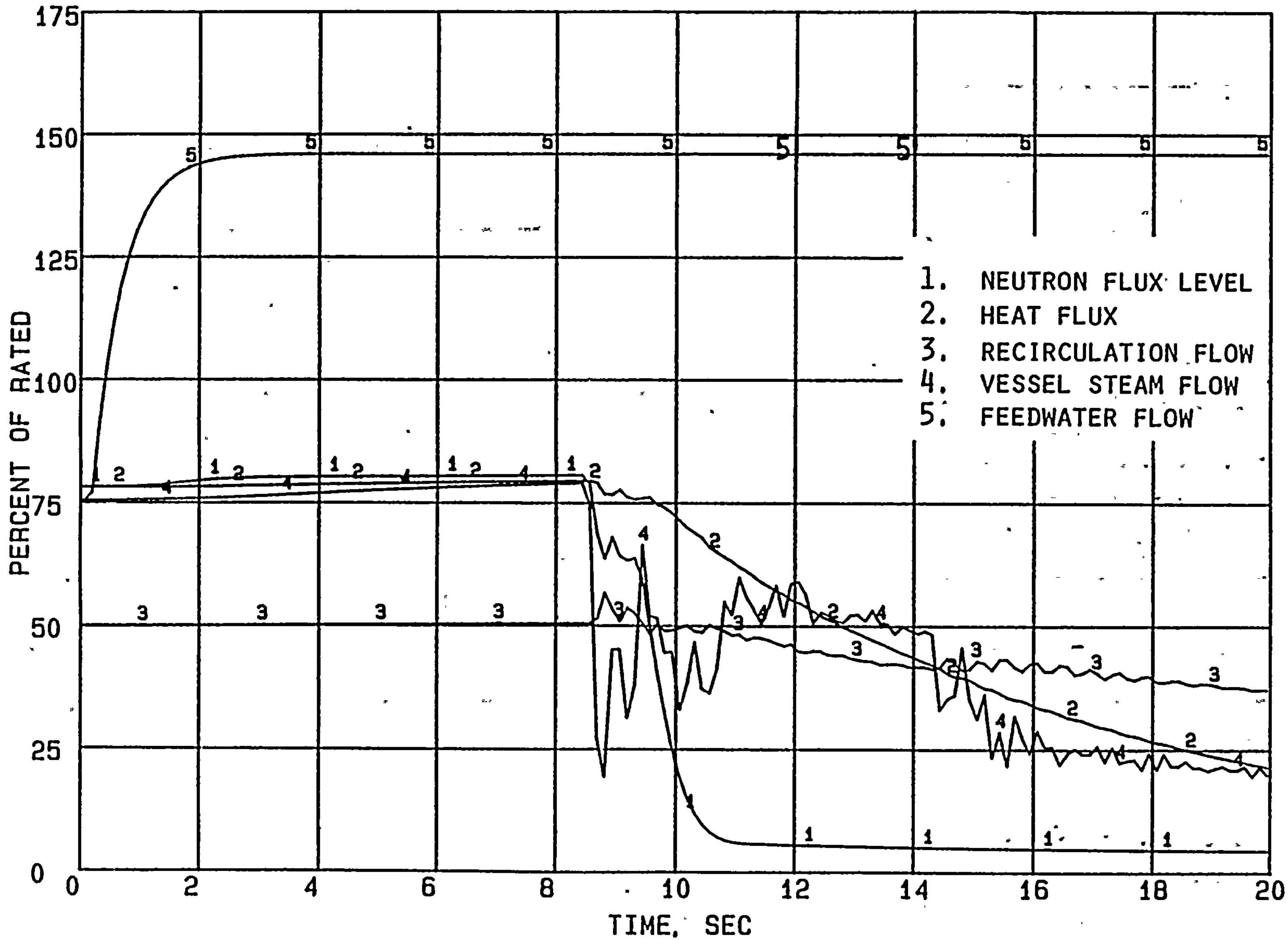

FIGURE 3.3 FEEDWATER CONTROLLER FAILURE RESULTS, RPT OPERABLE, NORMAL SCRAM SPEED

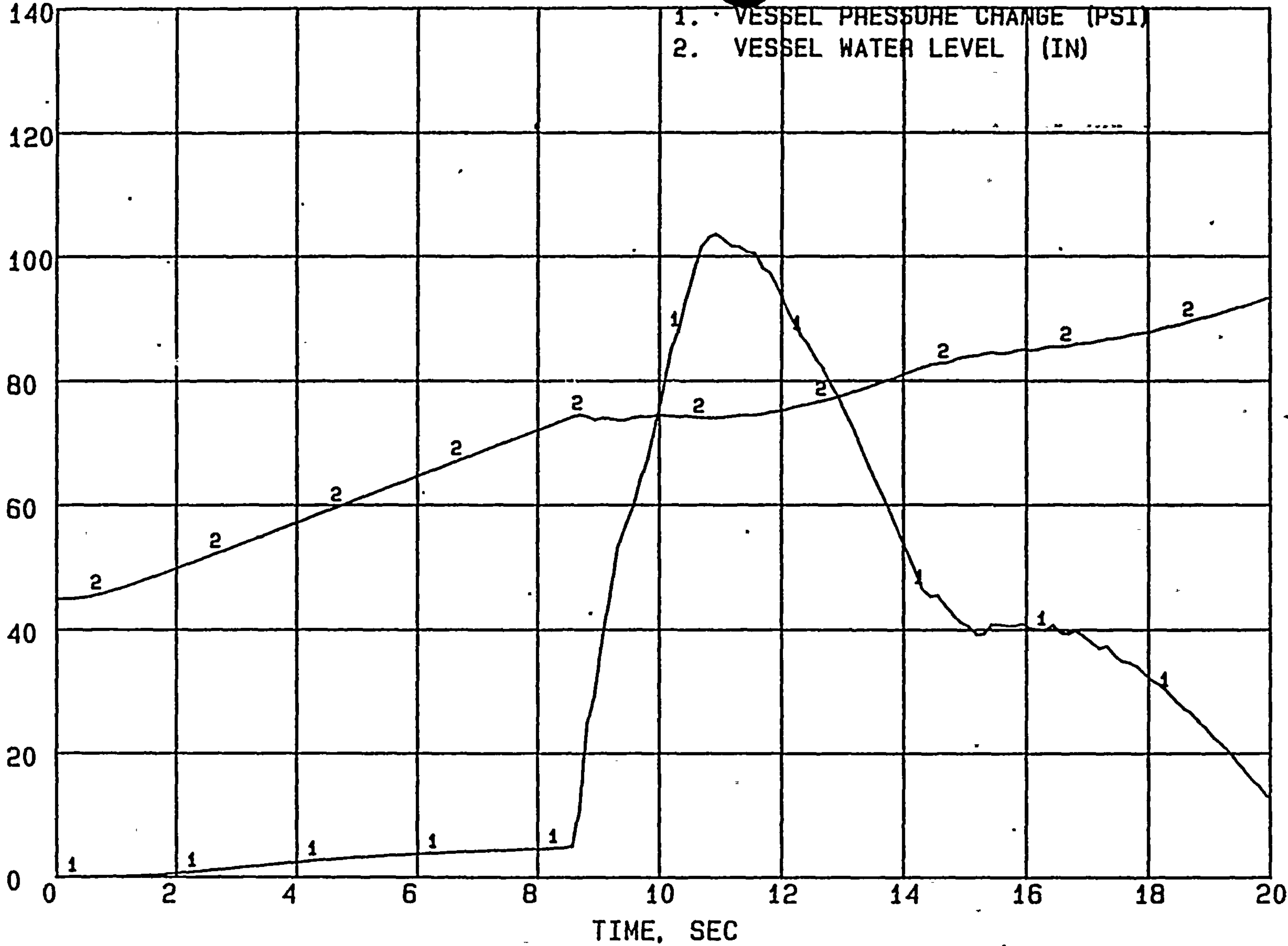

FIGURE 3.4 FEEDWATER CONTROLLER FAILURE RESULTS, RPT OPERABLE, NORMAL SCRAM SPEED

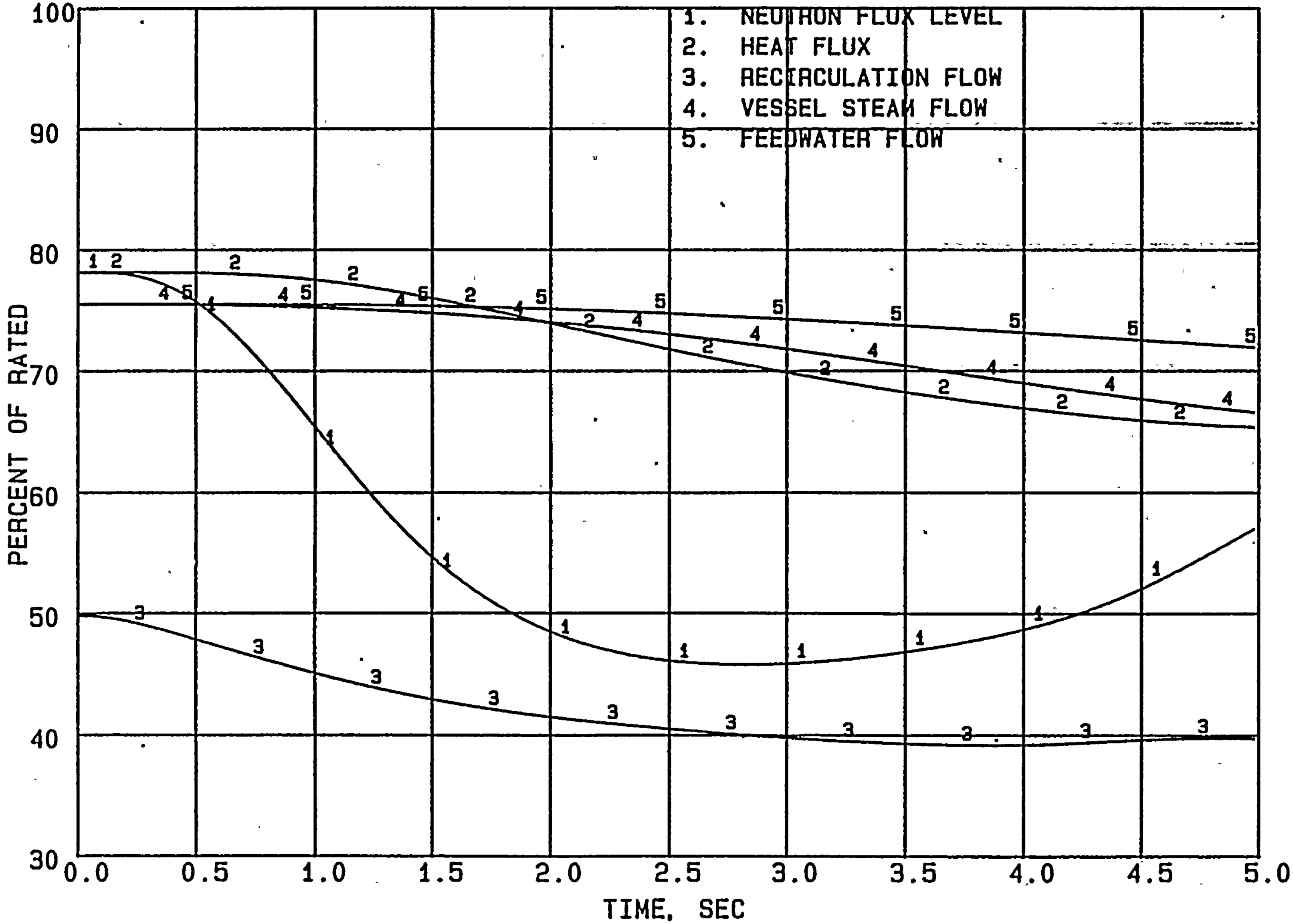

FIGURE 3.5 RECIRCULATION PUMP TRIP RESULTS

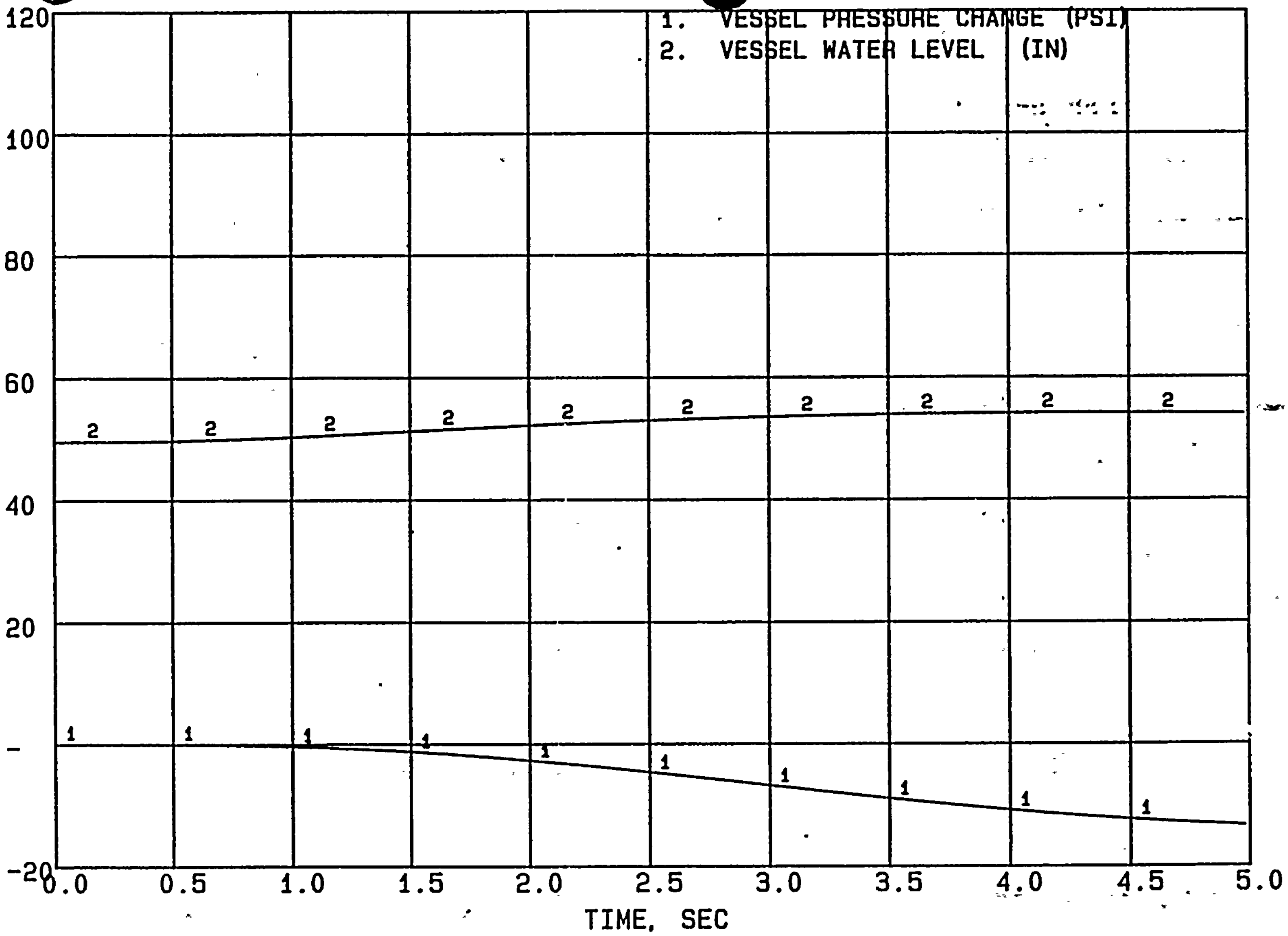

FIGURE 3.6 RECIRCULATION PUMP TRIP RESULTS

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4.0 PUMP SEIZURE ACCIDENT

The seizure of a recirculation pump is considered as a design basis accident event. It is a very mild accident relative to other design basis accidents such as the loss of coolant accident (LOCA). The pump seizure event is a postulated accident in which the recirculation pump impeller speed is rapidly reduced to zero (in 0.1 seconds). This causes a rapid decrease in core flow and a decrease in the heat removal rate from the fuel rods. Although the vessel water level increases, a high level trip did not occur in the analysis. However, even without a scram the power decreases consistent with the core flow decrease until natural circulation conditions occur.

A pump seizure accident event was analyzed for WNP-2 to confirm the insignificance of this event relative to the design basis LOCA. The plant response to the pump seizure accident is shown in Figure 4.1. The core remains covered and natural circulation conditions are approached within three seconds. Any fuel rods which experience boiling transition would be expected to be in the film boiling mode for a short period. In addition, the film boiling would be limited to small, localized areas in the affected fuel assemblies. Because of this short duration, fuel failures due to overheating or clad strain would not be expected as a result of this accident. Thus, the consequences of this event are bounded by the LOCA where fuel failures are assumed to be extensive.

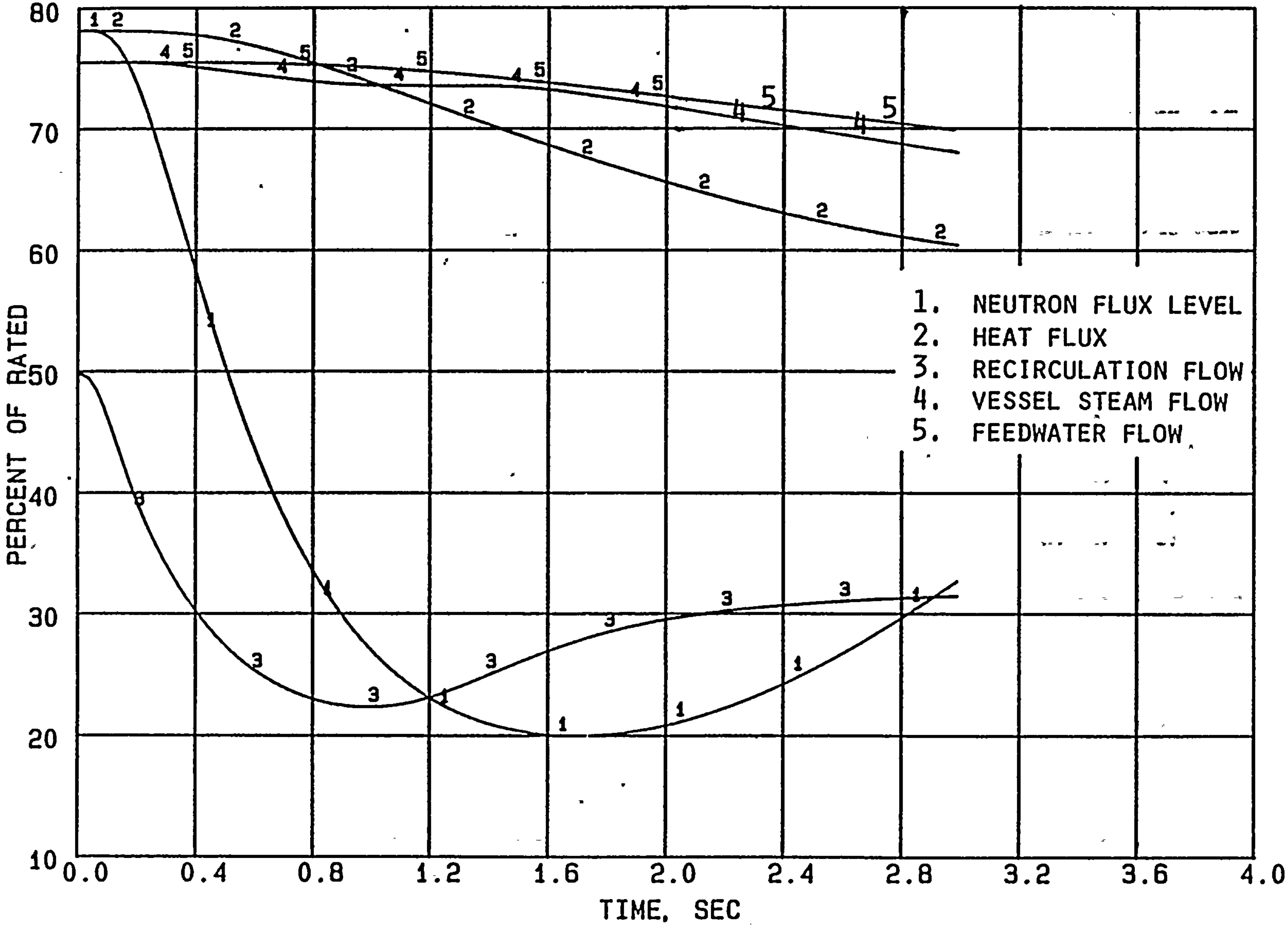

FIGURE 4.1 PUMP SEIZURE ACCIDENT

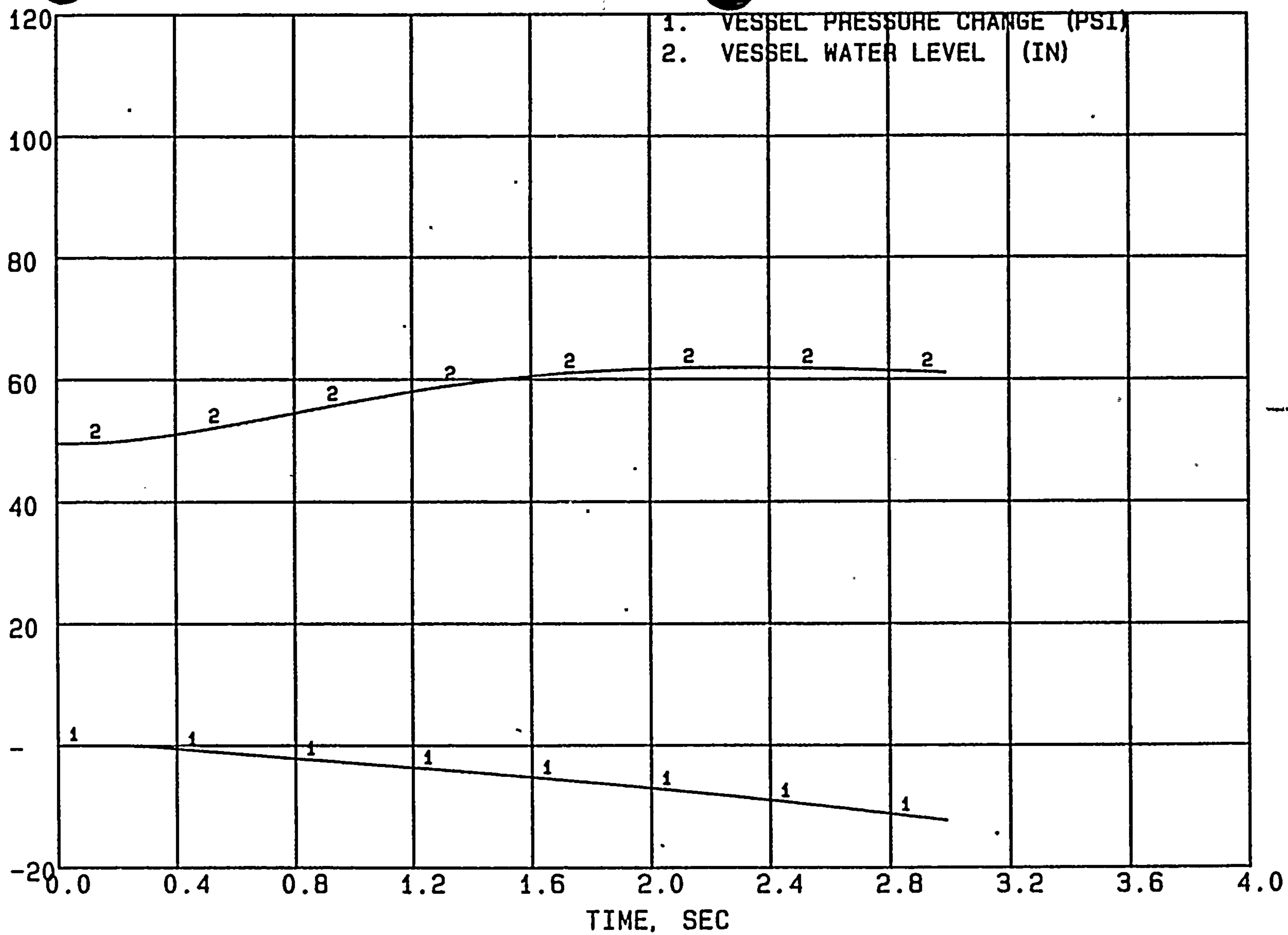

FIGURE 4.2 PUMP SEIZURE ACCIDENT

5.0 SAFETY LIMIT

The MCPR fuel cladding integrity safety limit for single loop operation was calculated using the methodology described in Reference 7. In this methodology, a Monte Carlo procedure is used to evaluate the impact on the safety limit of plant measurement and power prediction uncertainties. At the reduced flow state with single loop operation, the plant measurement and prediction uncertainties of core flow, radial power distribution, and axial power distribution increase. The uncertainties used in the SLO safety limit evaluation are shown in Table 5.1. The XN-3 correlation, Reference 8, is then used to predict the critical heat flux phenomena. Non-parametric tolerance limits, Reference 9, are used to determine the expected number of rods in boiling transition.

During sustained SLO operation at a MCPR of 1.07 with the design basis power distribution described below, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition at a confidence level of 95%. This supports a safety limit of 1.07, an increase of 0.01 over that for the normal operating state for all fuel types.

The design basis power distribution used in this analysis for single loop operation was based on the predicted power distributions which were determined to be the most severe or conservative with respect to the number of rods subject to boiling transition considerations.

5.1 SAFETY LIMIT EVALUATION UNCERTAINTIES

Parameter	Standard Deviation*
Feedwater Flow Rate	.0176
Feedwater Temperature	.0076
Core Pressure	.0050
Total Core Flow Rate	.0600
Core Inlet Enthalpy	.0024
XN-3 Critical Power Correlation	.0411
Assembly Flow Rate	.0280
Power Distribution	
Radial Peaking Factor	.0551
Local Peaking Factor	.0246

*Fraction of Nominal Value.

6.0 STABILITY ANALYSIS

Single loop stability analyses have been performed with COTRAN for Cycle 3. The calculations were performed using the two loop APRM rod block equation as specified in Reference 3. The decay ratio calculations were performed with the COTRAN code at the limiting Cycle 3 power and flow conditions. The most limiting Cycle 3 single loop decay ratios are identical to the Cycle 3 decay ratios for two loop operation. The limiting single loop decay ratios and the corresponding power/flow conditions are as follows:

Power % Rated	Flow % Rated	Decay Ratio
65.0	45.0	0.49*
48.0	27.6	0.84**

Since the Cycle 3 single loop decay ratios are no larger than the Cycle 3 two loop decay ratios, the cycle dependent two loop stability analysis performed for future cycles will bound single loop operation.

*At right hand boundary of Detect and Suppress region.

**Within Detect and Suppress region.

[illegible]

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WNP-2 SINGLE LOOP OPERATION ANALYSIS

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