XN-NF-85-62

DRESDEN UNIT 3 CYCLE 10
PLANT TRANSIENT ANALYSIS

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1.0 INTRODUCTION

This report describes the plant transient analyses performed by Exxon Nuclear Co., Inc., in support of the Cycle 10 (XN-3) reload for Dresden Unit 3. This cycle is scheduled to commence operation in Summer 1986.

Cycle 10 is the third cycle during which ENC fuel will be irradiated in Dresden Unit 3. In addition to two reloads of ENC 8x8 fuel, the Cycle 10 core will contain a significant number of retrofit 8x8 assemblies fabricated by General Electric and 176 assemblies of a 9x9 lattice configuration fabricated by Exxon Nuclear. Operating limit critical power ratio values for all of these fuel types during Cycle 10 operation are established in this report. Because the ENC XN-2 8X8 fuel is mechanically identical to the ENC XN-1 8x8 fuel, the transient response of the two fuel types is very similar during anticipated operation; thus the two fuel types are considered to be the same during the transient analyses reported in this document. Similarly, since only one GE fuel type (retrofit 8x8) is to exist in this Cycle 10 core, the transient response for the General Electric fuel was explicitly modeled as retrofit 8x8 fuel.

The analyses reported in this document were performed using the same plant transient analysis methodology (Ref. 2) as was used to establish thermal margin requirements for Cycles 8 and 9 operation of Dresden Unit 3 (Ref. 1) except for use of the following:

- o the NRC approved constant flow MCPR methodology
- o the code uncertainty methodology of Ref. 8

o the COTRANSA/PTSBWR updates described in Appendix A

The analysis supports operation in the expanded power / flow operating map shown in Figure 1.1. Section 5.0 describes the results of the off-rated analysis performed to demonstrate that the MCPR operating limits together with the reduced flow MCPR allow operation throughout this map.

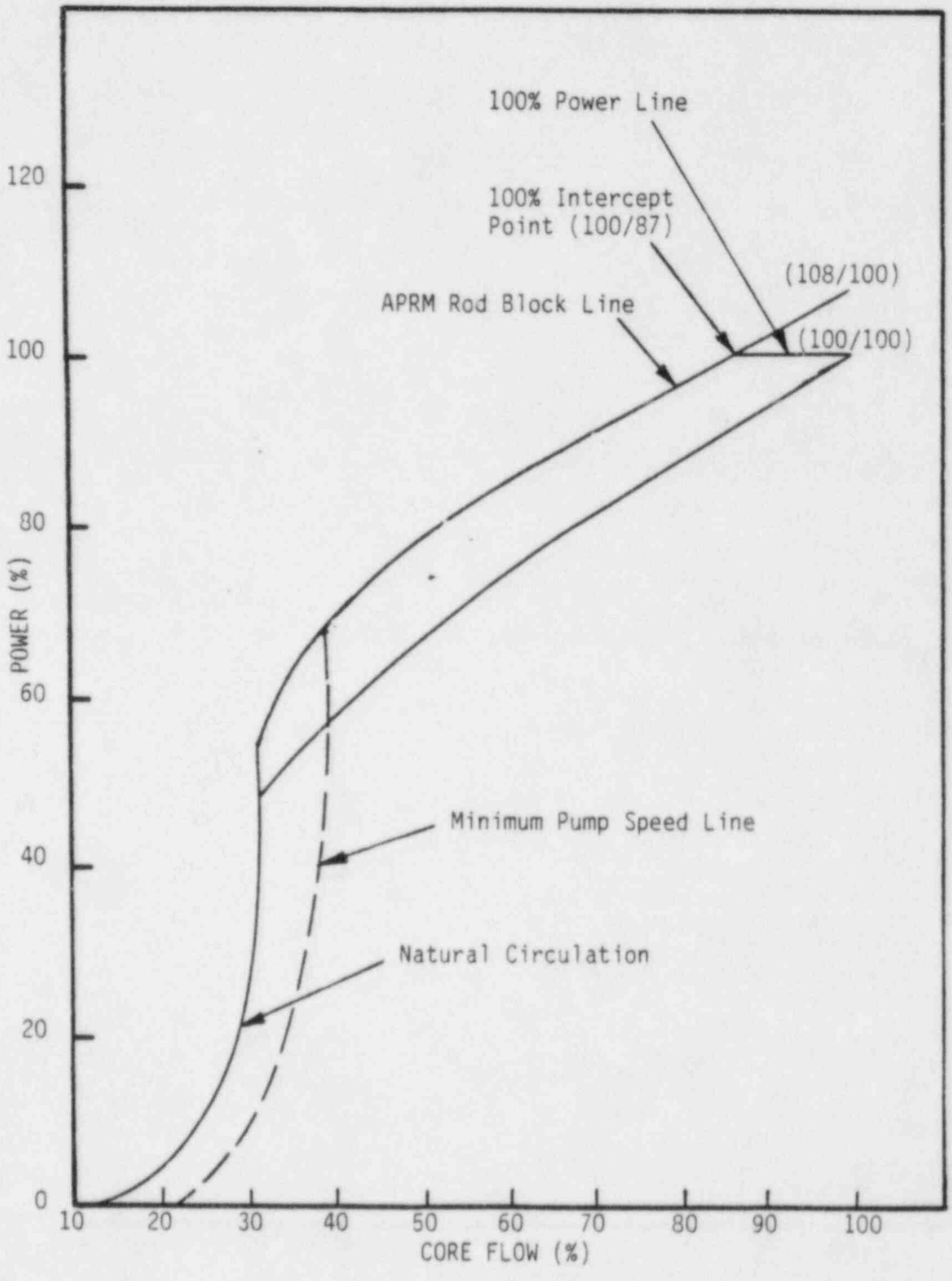

Figure 1.1 Dresden 2/3 Proposed Operating Power/Flow Map

2.0 SUMMARY

The determination of thermal margin requirements for Dresden Unit 3 Cycle 10 was based on the consideration of various operational transients. The limiting transients in each general category of event are identified in Reference 2. The most limiting transient events for determination of thermal margins in BWR/3 applications were determined to be the generator load rejection or turbine trip event without bypass to the condenser, the loss of feedwater heating event, and the feedwater controller failure (maximum demand) event. For the case of Dresden Unit 3, the most limiting of these events was found to be the generator load rejection without bypass. Table 2.1 presents the change in critical power ratio (ΔCPR) at bounding conditions for the three most limiting transients.

The safety limit for Cycle 10 conditions was calculated to be 1.05. This value is applicable to all fuel types.

The minimum thermal margin MCPR operating limit for each fuel type applicable to Cycle 10 operation of Dresden Unit 3 is contained in Table 2.2. These are obtained by adding the CPRs of the limiting transient in Table 2.1 to the 1.05 safety limit. The MCPR operating limit for 9X9 fuel is 0.04 greater than for ENC 8X8 fuel. However, for the same bundle power the operating critical power ratio (CPR) for 9X9 fuel is also about 0.04 higher than for ENC 8X8 fuel because of the larger heat transfer area in 9X9 fuel. Thus the 9X9 and 8X8 fuels have equivalent thermal margin to their respective MCPR limits or conversely, have the same bundle power at their respective MCPR operating limits.

Maximum system pressure for ASME overpressure evaluation has been calculated for the postulated closure of all main steam isolation valves (MSIVs) without activation of the MSIV position scram and without pressure relief credit for the four electromatic relief valves. The results of this analysis as shown in Table 2.2 indicates that the requirements of the ASME Code regarding overpressure protection will continue to be met for the Dresden Unit 3 Cycle 10 core, the calculated pressures are below the 1375 psig limit.

For informational purposes, Table 2.3 summarizes the maximum values obtained for other key parameters in the thermal margin and ASME overpressurization transients.

Table 2.1

Delta CPR's

Dresden Unit 3 Cycle 10

	Delta CPR		
Transient	GE 8x8	ENC 8x8	ENC 9x9
Generator Load Rejection Without Bypass(1)	0.23	0.24	0.28
Feedwater Flow Controller Failure (Maximum Demand)(2)	0.17	0.17	0.20
Loss of Feedwater Heating(2)	0.20	0.20	0.20

(1) ΔCPR on statistical basis

(2) ΔCPR on bounding basis

Table 2.2

Thermal Margin Summary

Dresden Unit 3 Cycle 10

	MCPR Operating Limit (1)		
Limiting Transient	GE 8x8	ENC 8x8	ENC 9x9
Generator Load Rejection Without Bypass	1.28	1.29	1.33

	Maximum Pressurization (psia)		
Transient	Steam Dome	Lower Plenum	Steam Lines
MSIV Closure Without Position Scram (ASME)	1316	1341	1315

1. Based on a 1.05 safety limit.

Table 2.3

Results of Plant Transient Analyses

Dresden Unit 3 Cycle 10

Event	Maximum Neutron Flux (% of Rated)	Maximum Core Average Heat Flux (% of Rated)	Maximum Vessel Pressure (psia)
Generator Load Rejection Without Bypass (1)	275	109.5	1273
Feedwater Flow Controller Failure (Maximum Demand)	185	112.1	1191
Loss of Feedwater Heating	120	120.0	1064
MSIV Closure (ASME Analysis)	494	131.7	1341

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(1) Nominal case; all other events bounding case

3.0 TRANSIENT ANALYSES FOR THERMAL MARGIN

This text section describes the analyses which were performed to determine the minimum MCPR operating limits for Dresden Unit 3, Cycle 10.

3.1 DESIGN BASIS

The plant transient analyses for Dresden Unit 3 determined that the most limiting condition for thermal margin was reactor operation at full power and flow. Reactor plant conditions for these analyses are shown in Table 3.1. The most limiting point in the cycle is the end of full power capability, at which time the control rods are fully withdrawn from the core. The thermal margins established for the end of full power capability are conservative for cases where control rods are partially inserted or reactor power is less than rated. Observance of the MCPR operating limits shown in Table 2.2 will provide adequate protection against the occurrence of boiling transition during all anticipated transients for Cycle 10 operation of Dresden Unit 3.

3.2 CALCULATIONAL MODEL

The plant transient methodology described in reference 2 and Appendix A was used for the analysis reported in this document. The COTRANSA one-dimensional core model is used to evaluate the generator load rejection and feedwater controller failure transients to model the axial power shifts associated with the system overpressurization.

The analytical models used to determine thermal margin requirements include PTSBWR3/COTRANSA (Ref 2), RODEX2 (Ref. 3), and XCOBRA (Ref. 4). The interaction of these models to define thermal limits is described in Appendix A. Fuel pellet to cladding gap conductance values used in the analyses were based on RODEX2 calcultions for the Dresden Unit 3 Cycle 10 core configuration.

In accordance with ENC methodology, possible limiting transients are evaluated using a consistant set of bounding input. From these bounding results, the limiting transient is identified as the generator load rejection without bypass. Since this is a rapid pressurization event, ENC's methodology for including code uncertainties in determining operating limits for rapid pressurization transients in BWR's (Ref. 8) is used. This methodology includes code uncertainities and uncertainties in important input variables. A conservative deterministic integral power multiplier of 110 % is used to account for code uncertainties when the statistical methodology is being applied.

Table 3.2 summarizes the values used for these important parameters in the analysis. Table 3.3 provides the feedwater flow, recirculating coolant flow, and pressure regulation system settings used in the analysis.

3.3 ANTICIPATED TRANSIENTS

Eight major categories of transients were considered generically in Reference 2. For Cycle 10 operation of Dresden Unit 3, specific events have been evaluated for thermal margin. These events are the generator load rejection transient without bypass to the condenser, and the feedwater flow controller failure to maximum demand and loss of feedwater heating. In the analysis, it was assumed that a relief valve was out-of-service. For BWR/3 plants, other categories of transients are either inherently self-limiting or bounded by one of these.

3.3.1 Load Rejection Without Bypass

This event is the most limiting of the rapid pressurization transients for Dresden Unit 3. This conclusion was verified through comparison with the results of the analysis of the turbine trip transient without condenser bypass for the Dresden plants.

In the load rejection and turbine trip transients, steam flow is interrupted by an abrupt closure of either the turbine control valve in the case of the load rejection or the turbine stop valve in the case of the turbine trip.

The resulting pressure increase causes a decrease in the void level in the core, which in turn creates a power excursion. This excursion is mitigated in part by Doppler broadening and pressure relief, but the primary mechanisms for termination of the event are control rod insertion and revoiding.

The important parameters for this transient include the power transient (integral power) determined by the void reactivity, which affects the initial power excursion rate and part of the intrinsic shutdown mechanism, and the control rod worth, which determines the value of the scram reactivity. Other important inputs include the control rod movement parameters (scram delay and insertion speed), which determine the event characteristics following the initial mitigation of the power excursion. The bounding case resulted in delta CPR's of 0.33, 0.34 and 0.40 for the GE 8X8, ENC8X8, and the ENC 9X9 fuels respectively as shown in Table 5.1.

3.3.2 Feedwater Flow Controller Failure

Failure of the feedwater control system could lead to a maximum increase of feedwater flow into the reactor vessel. The excessive feedwater flow increases the subcooling in the recirculating water returning to the reactor core. This reduction in average moderator temperature will result in the core power's rising to attain a new equilibrium if no other action is taken. Eventually, the level of water in the downcomer region will rise until the high water level trip is reached. The turbine then trips to prevent the transmission of liquid water to the turbine, and the turbine stop valves close. The resulting scram arrests the power increase, and the pressure pulse resulting from the stop valve closure is suppressed by the opening of the bypass valves. The analysis assumed that all of the conservative conditions of Table 3.2 were concurrent, and the calculated delta-CPR is a bounding result. The calculated values as shown in Table 2.1 of 0.17 for the 8x8 fuel types and 0.20 for the 9x9 fuel are adequate for protection of the fuel against boiling transition.

Figures 3.4, 3.5, and 3.6 illustrate the behavior of major system variables during the FWCF transient.

3.3.3 Loss of Feedwater Heating

The loss of feedwater heating leads to a gradual increase in the subcooling of the water in the lower plenum. Core power slowly rises to the overpower trip setpoint. The gradual power change allows the fuel thermal response to maintain pace with the increase in neutron flux. In this analysis, it is assumed that the plant is operating in manual control and the feedwater temperature dropped 145 degrees F over a two-minute period. Void reactivity is assumed to be 25% lower than the nominal calculated value, which resulted in a maximum value of the heat flux. Scram performance is assumed to be at Technical Specification limits, and control rod worth is assumed to be 20% less than the nominal calculated value.

Previous loss of feedwater heating analysis on Dresden have shown that the delta CPR for the transient to be less limiting than the above transients. Furthermore, Reference 9 shows that the delta CPR for the transient to be the same for 8x8 and 9x9 fuel types because it is a slow transient. This transient is not limiting for any fuel type in Cycle 10 of Dresden Unit 3. The result of the loss of feedwater heating transient analysis is a delta CPR of 0.20 for all fuel types as shown in Table 2.1.

3.3.4 Statistical Uncertainty Analysis

The bounding transient analysis showed the load rejection without bypass (LRWB) to be the limiting transient for Dresden Unit 3 Cycle 10, as in previous cycles. When rapid pressurization transients are limiting, ENC methodology for including code uncertainities in the determination of MCPR operating limits is applied (Ref. 8). This methodology uses a conservative deterministic multiplier of 110% on the calculated power transient and treats the uncertainities in the important input variables (scram speed and scram delay) statistically. The delta CPR used to establish the MCPR operating limits result from the use of the deterministic 110% integral power combined with 95% probability that the statistical variable delta CPR is not exceeded.

The uncertainty in the control rod drive performance parameters, scram delay and insertion speed, was determined from measured plant data. Incorporating the most recent plant data, the uncertainty in the scram delay time for Cycle 10 was determined to correspond to a mean value of 241 msec and a standard deviation of 28 msec. In the Cycle 9 analysis, a mean value of 240 msec and a standard deviation of 30 msec were used.

Similarly, the uncertainty in the scram insertion speed for Cycle 10 was determined to correspond to a mean value of 139.57 cm/sec and a standard deviation of 3.33 cm/sec. In the Cycle 9 analysis, a mean of 136.21 cm/sec and a standard deviation of 2.74 cm/sec were used.

The uncertainties for the Cycle 9 analyses are compared with the uncertainties for the Cycle 10 analyses in Table 3.6.

Figures 3.1, 3.2, and 3.3 illustrate the behavior of major system variables during the LRWB transient using nominal input for uncertainty variables.

3.4 MCPR SAFETY LIMIT

The MCPR safety limit for Cycle 10 operation of Dresden Unit 3 was determined using the methodology described in Reference 6. No changes were made to the methodology except for code revisions required to apply the methodology to a core of mixed 8x8 and 9x9 assemblies. This methodology was used to determine the MCPR safety limit for Cycle 9 operation of Dresden Unit 2 and for Cycle 8 and 9 operation of Dresden Unit 3. As with the transient delta CPR's, the Dresden 3 Cycle 10 analysis assumed constant flow (no flow iteration) in computation of CPR as approved in Ref. 2.

The main input parameters and uncertainties used in the safety limit analysis are listed in Table 3.5.

The design basis radial power distribution used in the analysis is shown in Figure 3.7. This power distribution was found to have the greatest number of rods near limits for Cycle 10 and is conservative in this regard relative to other expected power distributions during the cycle. The radial peaking for the four maximum power assemblies was increased above the calculated values to give a conservative operating MCPR of 1.40 and a significant portion of the core near operating limits. Four fuel types were represented in the Dresden 3 Cycle 10 safety limit analysis, i.e., ENC XN-3 9X9, ENC XN-2 8X8, ENC XN-1 8X8, and GE 8X8R.

Bounding local power distributions for each fuel type over their expected Dresden 3 Cycle 10 exposure were used. The power distributions are shown in Figures 3.8 through 3.11.

The MCPR safety limit was calculated to be 1.05. Protection of this limit will assure that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition during normal operation and anticipated operational occurrences. This limit applies to the four fuel types above.

Table 3.1

Design Reactor and Plant Conditions

Dresden Unit 3

Reactor Thermal Power	2527 MWt
Total Recirculating Flow	98.0 Mlb/hr
Core Channel Flow	87.7 Mlb/hr
Core Bypass Flow	10.3 Mlb/hr
Core Inlet Enthalpy	522.9 BTU/lbm
Vessel Pressures	
Steam Dome	1020 psia
Upper Plenum	1026 psia
Core	1035 psia
Lower Plenum	1049 psia
Turbine Pressure	964.7 psia
Feedwater/Steam Flow	9.8 Mlb/hr
Feedwater Enthalpy	320.6 BTU/lbm
Recirculating Pump Flow (per pump)	17.1 Mlb/hr

Table 3.2

Significant Parameter Values Used in Analysis (1)

Dresden Unit 3

High Neutron Flux Trip	3032.4 MW
Control Rod Insertion Time	3.5 sec/90% inserted
Control Rod Worth	20% below nominal
Void Reactivity Feedback	10% above nominal (2)
Time to Deenergized Pilot Scram Solenoid Valves	283 msec (maximum)
Time To Sense Fast Turbine Control Valve Closure	80 msec
Time From High Neutron Flux Trip To Control Rod Motion	290 msec
Turbine Stop Valve Stroke 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)	525.2 BTU/hr-ft2-F (at 8.475 kW/ft)
Safety/Relief Valve Performance Settings	Technical Specifications
Pilot Safety/Relief Valve Capacity	166.1 lbm/sec (1080 psig)
Power Relief Valves Capacity	620.0 lbm/sec (1120 psig)
Safety Valves Capacity	1432.0 lbm/sec (1240 psig)
Pilot Operated Valve Delay/Stroke	400/100 msec
Power Operated Valves Delay/Stroke	967/200 msec

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(1) LRWB transient was evaluated statistically (See 3.3.4)

(2) 25% for calculations with point kinetics model

Table 3.2 (continued)

Significant Parameter Values Used in Analysis

Dresden Unit 3

MSIV Stroke Time	3.0 sec
MSIV Position Trip Setpoint	90% open
Condenser Bypass Valve Performance	
Total Capacity	1085.2 lbm/sec
Delay to Opening (from demand)	100 msec
Opening Time (Entire bank, max demand)	1.0 sec
Fraction of Energy Generated in Fuel	0.965
Vessel Water Level (above Separator Skirt)	
Normal	30 in
Range of Operation	20-40 in
High Level Trip	48 in
Maximum Feedwater Runout Flow	
Three pumps	4966 lbm/sec
Two pumps	3310.67 lbm/sec
Doppler Reactivity Coefficient (1)	-0.00230 $/F-void fraction
Void Reactivity Coefficient (1)	-15.14 $/void fraction
Effective Delayed Neutron Fraction	0.0052
Prompt Neutron Lifetime	0.0461 msec
Recirculating Pump Trip Setpoint	1240 psig Vessel Pressure

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(1) Nominal value

Table 3.3

Control Characteristics

Sensor Time Constants	
Pressure	100 msec
Others	250 msec
Feedwater Control Mode	One-Element
Feedwater Master Controller	
Proportional Band	100%
Reset	5 rep/min
Feedwater 100% Mismatch	
Water Level Error	60 in
Steam Flow (not used)	12 in eq.
Flow Control Mode	Master Manual
Master Flow Control Settings	
Proportional Band	200%
Reset	8 rep/min
Speed Controller Settings	
Proportional Band	350%
Reset	20 rep/min
Pressure Setpoint Adjuster	
Overall Gain	5 psi/% demand
Time Constant	15 sec
Pressure Regulator Settings	
Lead	1.0 sec
Lag	6.0 sec
Gain	3.33 %/psid

Table 3.4

Data Used in Statistical Transient Analysis

	Cycle 9		Cycle 10	
Variable	Mean	Std Dev	Mean	Std Dev
Scram Insertion Speed (cm/sec)	136.21	2.74	139.57	3.33
Scram Delay Time (msec)	240	30	241	28

Integral power was assumed 110 per cent of calculated value for all cases.

Table 3.5

Input for MCPR Safety Limit Analysis

Input Uncertainties

Parameter	Standard Deviation
XN-3 Correlation	0.0411
Assembly Radial Peaking Factor	0.0528
Fuel Rod Local Peaking Factor	0.0246
Fuel Assembly Flow Rate	0.0280

Nominal Input Values

Parameter	Mixed Core (8x8 and 9x9)
Core Pressure (psia)	1015
Core Power (MW)	3222
Core Inlet Enthalpy (BTU/lbm)	521.9
Total Core Flow (Mlbm/hr)	98.0
Feedwater Temperature (F)	345
Feedwater Flowrate (Mlbm/hr)	12.56

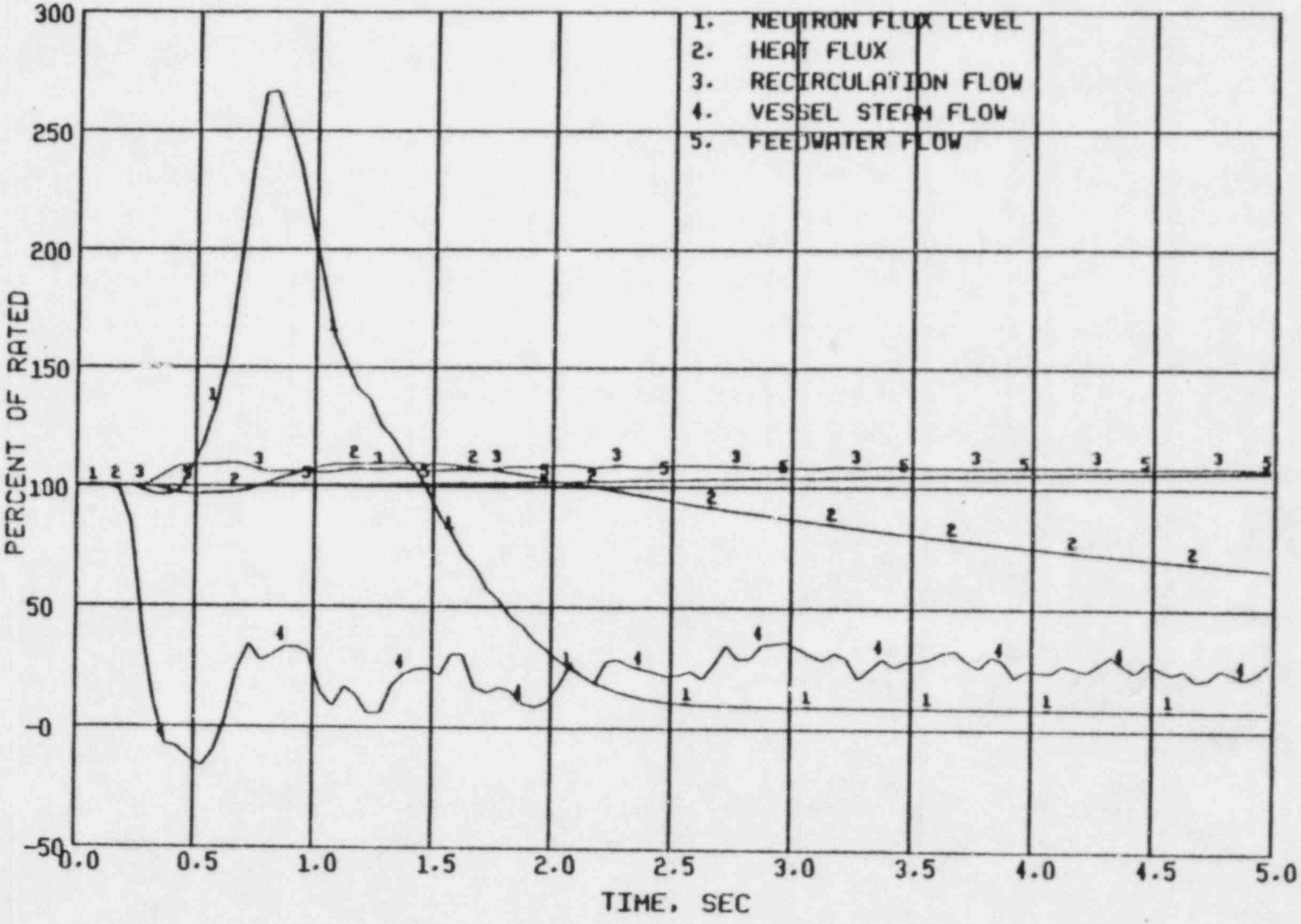

Figure 3.1 Load Rejection Without Bypass

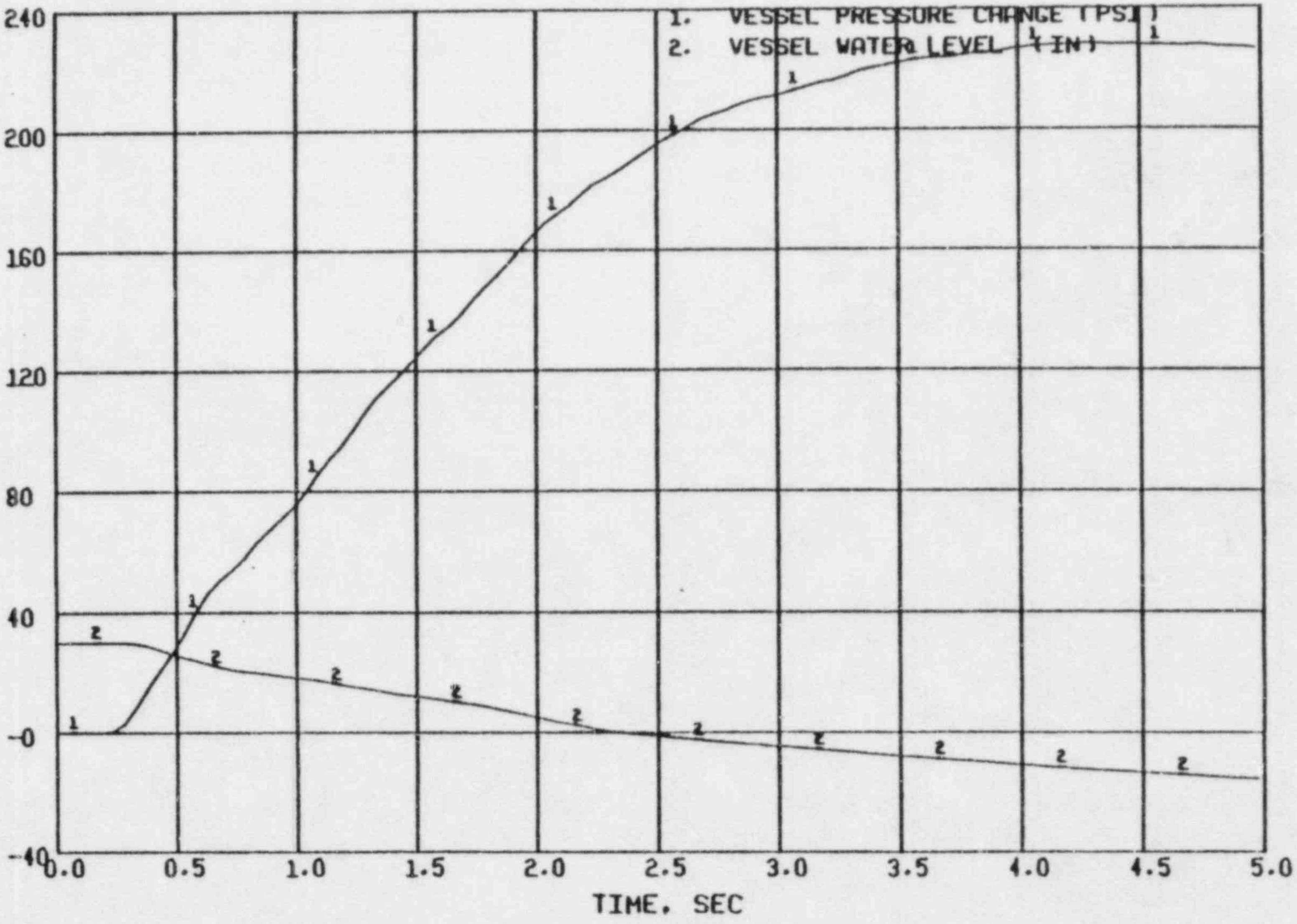

Figure 3.2 Load Rejection Without Bypass

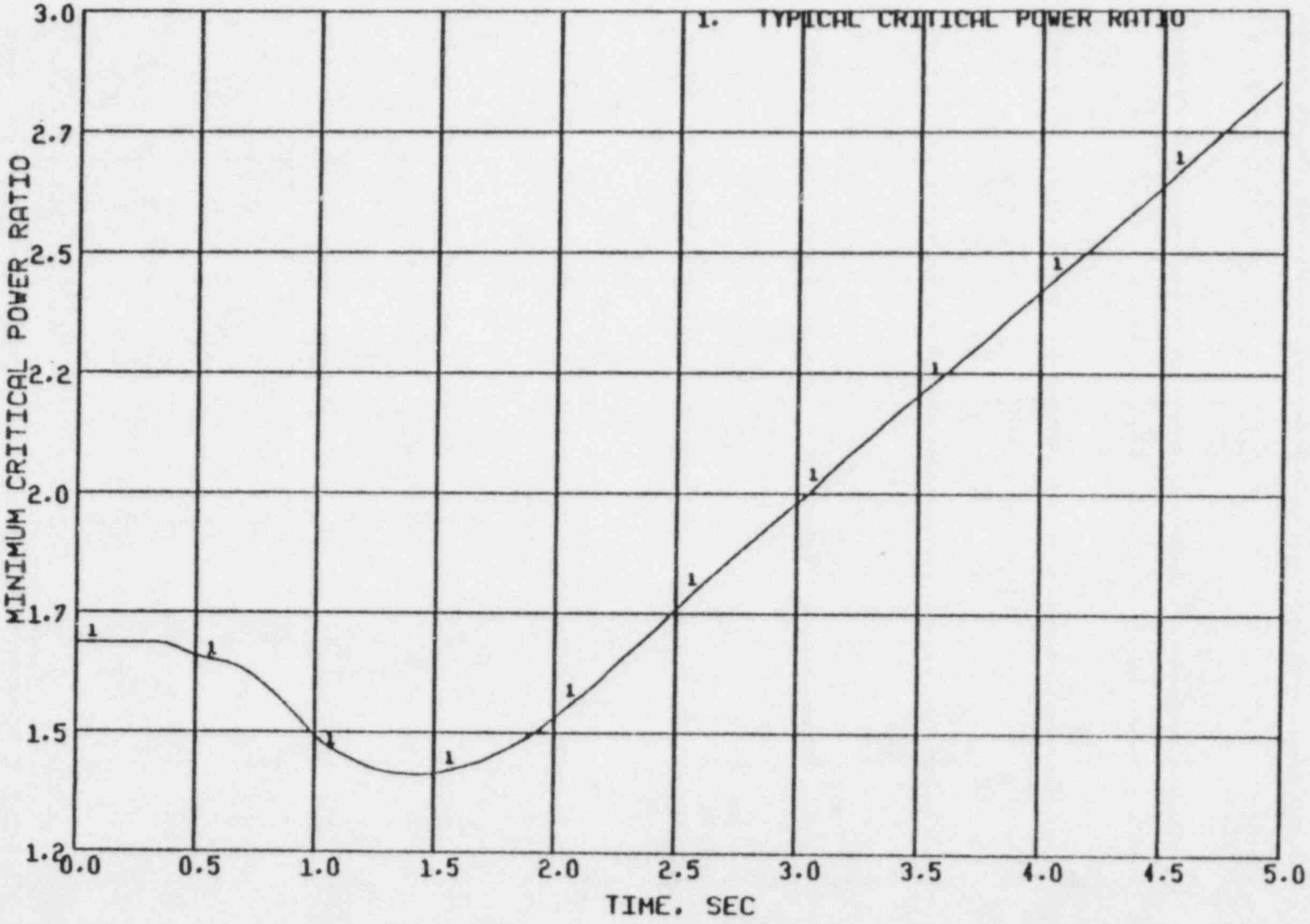

Figure 3.3 Load Rejection Without Bypass

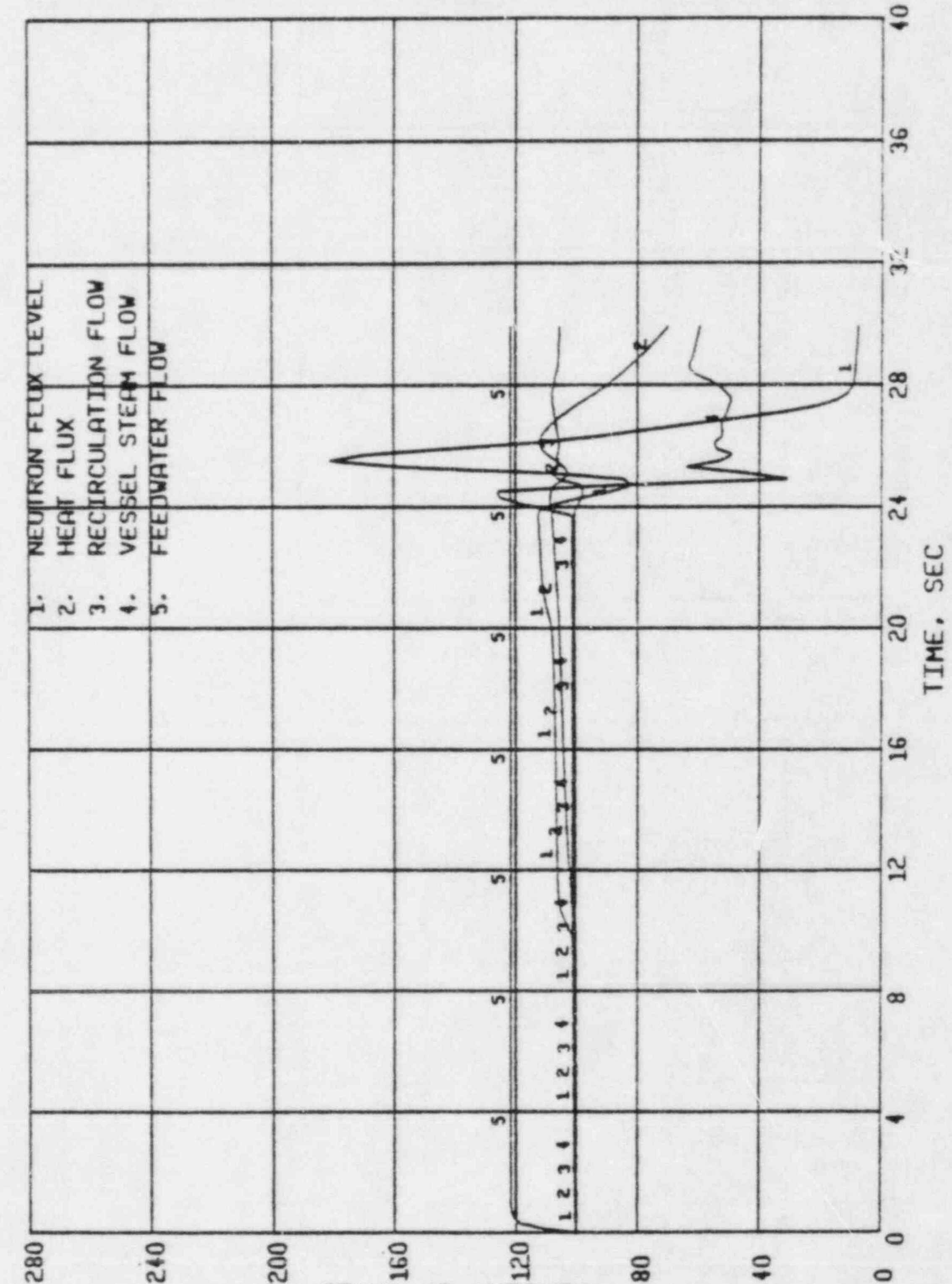

Figure 3.4 Feedwater Controller Failure

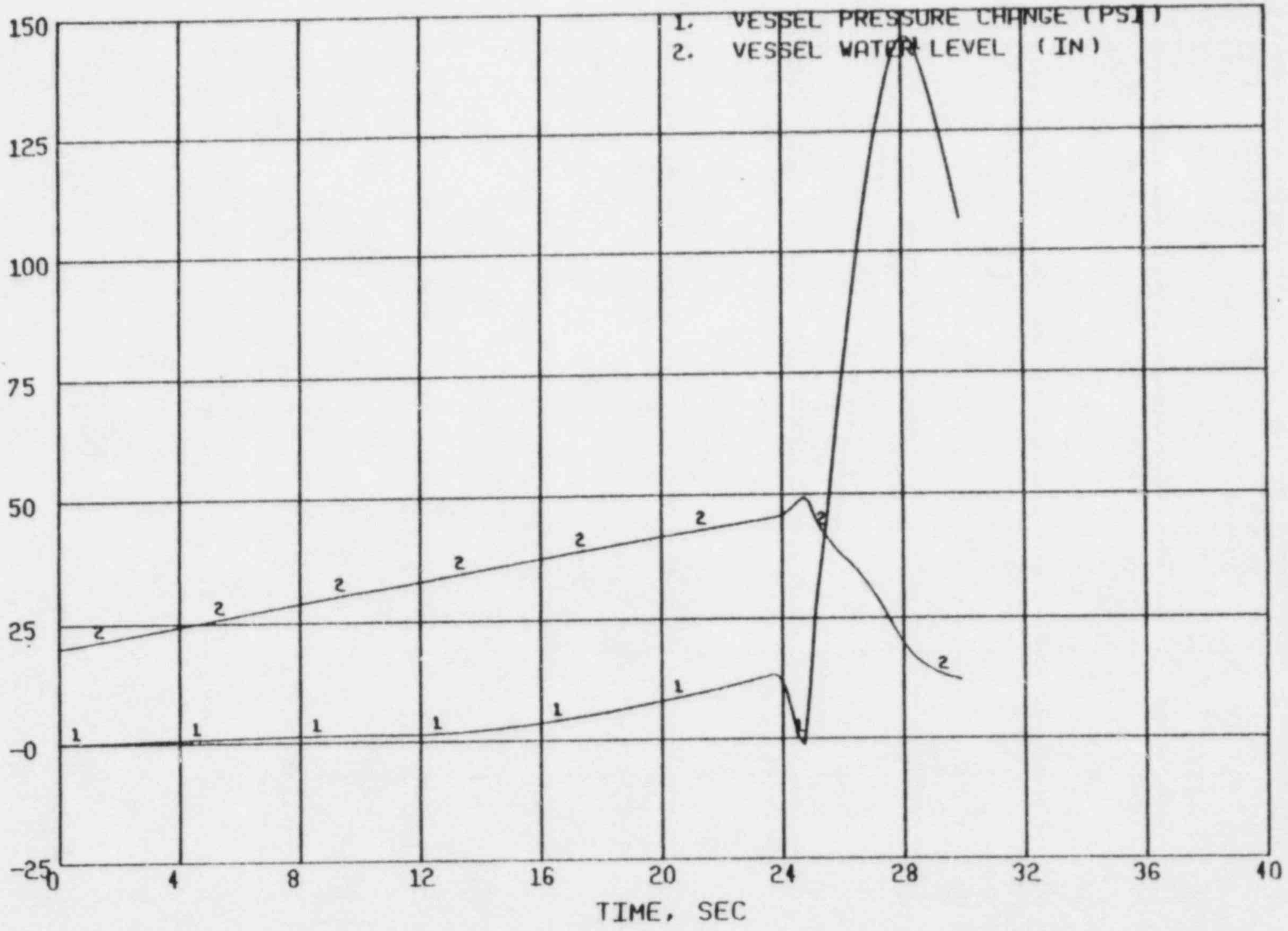

Figure 3.5 Feedwater Controller Failure

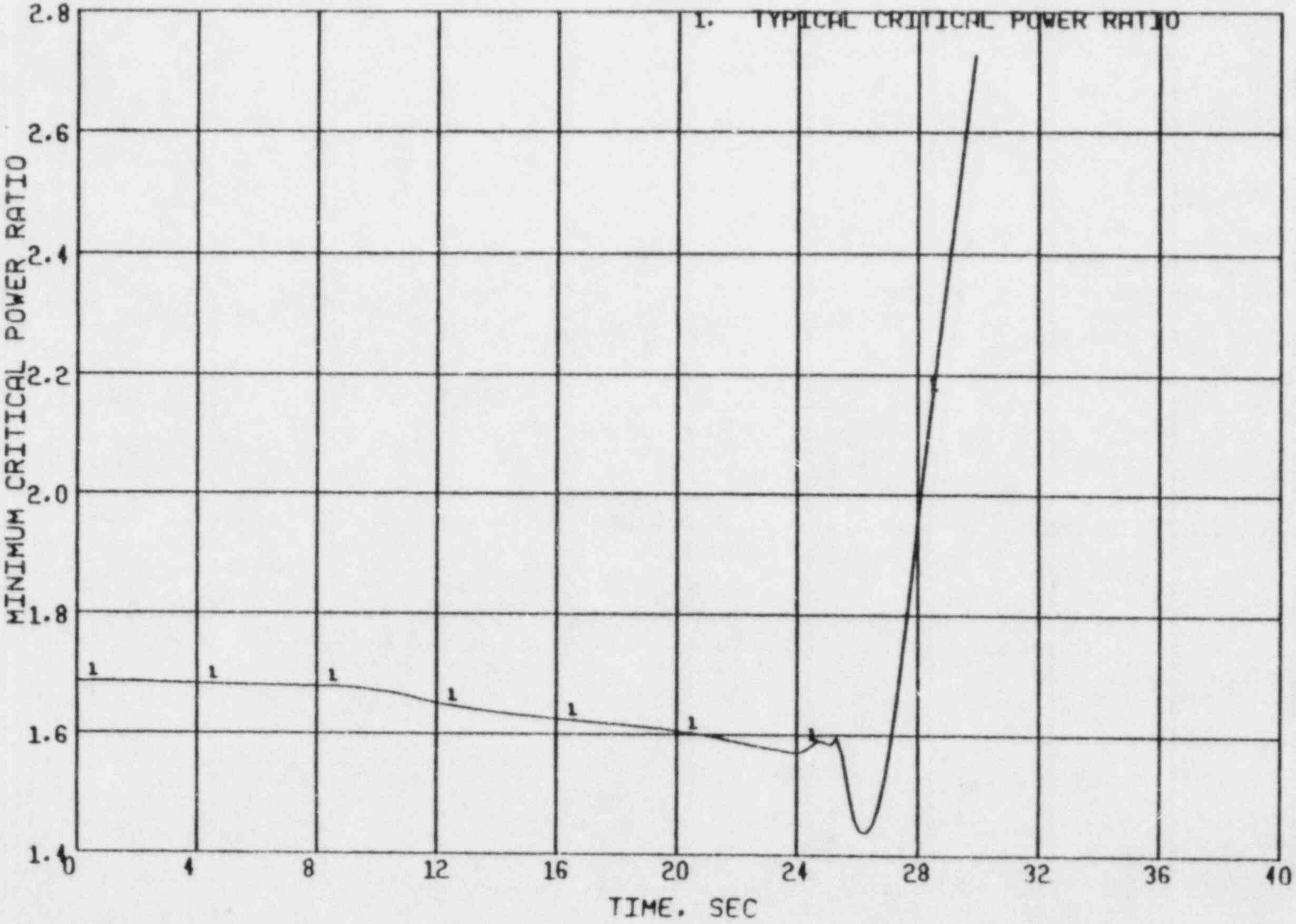

Figure 3.6 Feedwater Controller Failure

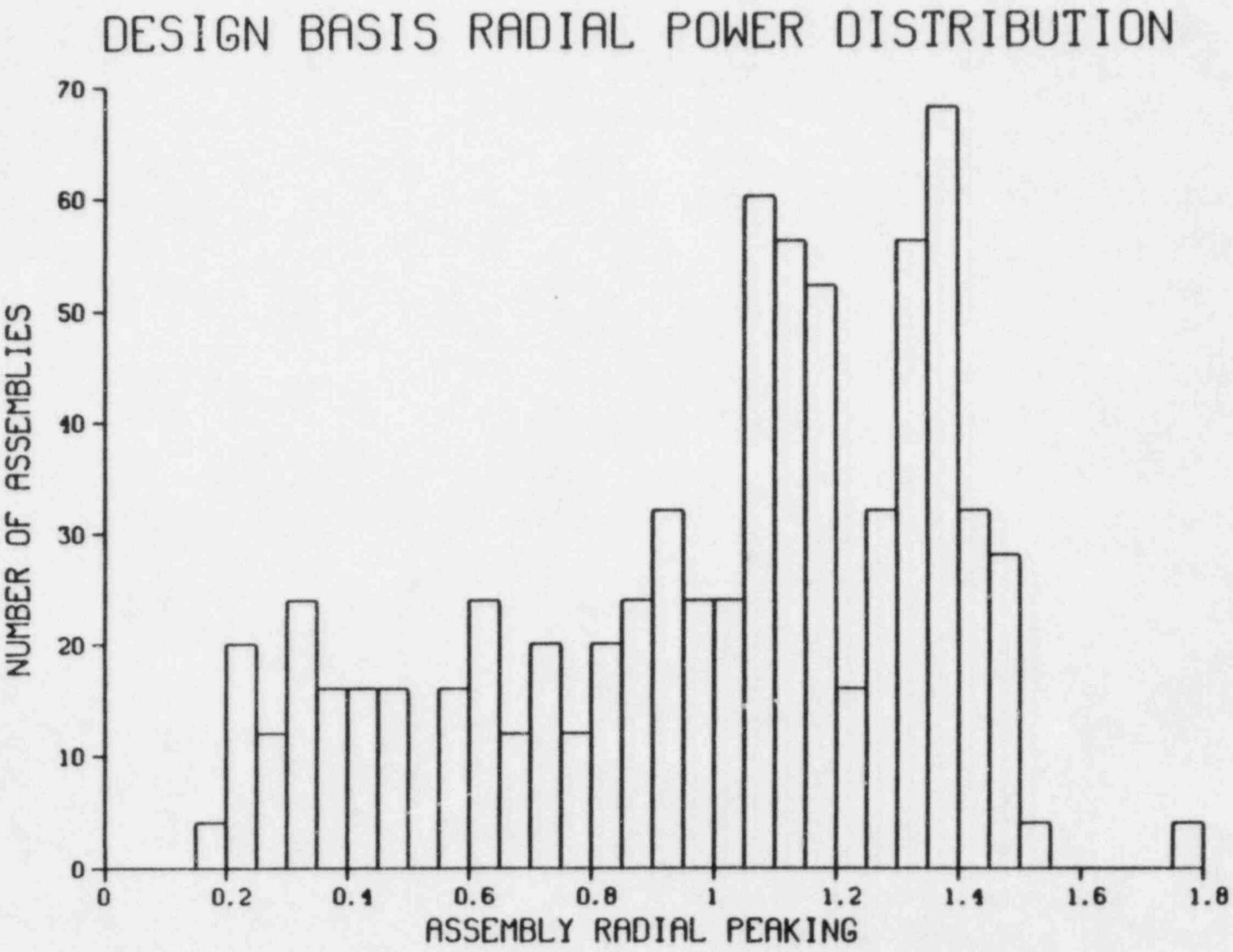

Figure 3.7 Design Basis Radial Power Distribution

LL 0.88	L 0.93	ML 0.98	M 1.07	H 1.12	H 1.12	M 1.06	ML 0.97	L 0.92
L 0.97	ML 0.99	M 1.04	ML* 0.85	H 1.03	H 1.03	ML* 0.85	H 1.12	ML 0.97
L 0.94	M 1.09	M 0.99	H 1.01	H 0.99	H 1.00	H 1.02	ML* 0.85	M 1.06
ML 1.04	ML* 0.92	H 1.05	H 0.99	H 1.00	W 0.00	H 1.00	H 1.03	H 1.12
ML 1.04	M 1.07	H 1.05	H 1.01	W 0.00	H 1.00	H 0.99	H 1.03	H 1.12
L 0.95	M 1.10	M 1.00	H 1.03	H 1.01	H 0.99	H 1.01	ML* 0.85	M 1.07
L 0.98	ML* 1.01	ML 0.94	M 1.00	H 1.05	H 1.05	M 0.99	M 1.04	ML 0.98
LL 0.89	L 0.95	ML* 1.01	M 1.10	M 1.07	ML* 0.92	M 1.09	ML 0.99	L 0.93
LL 0.94	LL 0.89	L 0.98	L 0.95	ML 1.04	ML 1.04	L 0.94	L 0.97	LL 0.88

WIDE

WIDE

Figure 3.8 Design Basis Local Power Distribution for ENC XN-3 9x9 Fuel

L 0.99	ML 0.99	ML 0.94	M 1.06	M 1.05	M 1.07	ML 0.94	ML 0.98
ML 1.01	ML* 0.93	M 1.04	H 1.02	H 1.01	ML* 0.83	M 1.04	ML 0.94
ML 0.99	M 1.07	H 1.02	H 0.98	H 0.99	H 1.00	ML* 0.83	M 1.07
ML 0.98	M 1.06	H 1.01	H 0.98	W 0.00	H 0.99	H 1.01	M 1.05
ML 0.99	M 1.07	H 1.03	H 0.98	H 0.98	H 0.98	H 1.02	M 1.06
ML 1.01	ML 0.93	M 1.05	H 1.03	H 1.01	H 1.02	M 1.04	ML 0.94
L 0.99	ML* 0.99	ML 0.93	M 1.07	M 1.06	M 1.07	ML* 0.93	ML 0.99
LL 0.95	L 0.99	ML 1.01	ML 0.99	ML 0.98	ML 0.99	ML 1.01	L 0.99

WIDE

WIDE

Figure 3.9 Design Basis Local Power Distribution for ENC XN-2 8x8 Fuel

L 0.99	ML 0.97	ML 0.95	M 1.05	M 1.04	M 1.05	ML 0.95	ML 0.98
ML 0.99	ML* 0.94	M 1.03	H 1.03	H 1.03	ML* 0.88	M 1.04	ML 0.95
ML 0.97	M 1.05	H 1.04	H 1.01	H 1.01	H 1.02	ML* 0.88	M 1.05
ML 0.97	M 1.04	H 1.03	H 1.01	W 0.00	H 1.01	H 1.03	M 1.04
ML 0.97	M 1.04	H 1.03	H 1.01	H 1.01	H 1.01	H 1.03	M 1.05
ML 0.98	ML 0.94	M 1.03	H 1.03	H 1.03	H 1.04	M 1.03	ML 0.95
L 0.98	ML* 0.95	ML 0.94	M 1.04	M 1.04	M 1.05	ML* 0.94	ML 0.97
LL 0.99	L 0.98	ML 0.98	ML 0.97	ML 0.97	ML 0.97	ML 0.99	L 0.99

WIDE

WIDE

Figure 3.10 Design Basis Local Power Distribution for ENC XN-1 8x8 Fuel

L 0.97	ML 0.95	MH 1.02	MH 1.01	MH 1.00	MH 1.01	M 0.96	ML 0.95
ML 0.96	MH 1.02	MH* 0.99	H 1.06	MH* 0.96	H 1.05	MH 0.99	M 0.96
ML 0.95	MH 1.01	H 1.07	H 1.05	H 1.03	H 1.03	H 1.05	MH 1.01
ML 0.95	MH 1.01	H 1.07	W 0.00	H 1.05	H 1.03	MH* 0.96	MH 1.00
ML 0.95	MH* 1.01	MH 0.99	MH 0.99	W 0.00	H 1.05	H 1.06	MH 1.01
ML 0.96	MH 1.02	MH 1.00	MH 0.99	H 1.07	H 1.07	MH* 0.99	MH 1.02
L 0.96	ML 0.94	MH 1.02	MH* 1.01	MH 1.01	MH 1.01	MH 1.02	ML 0.95
LL 0.99	L 0.96	ML 0.96	ML 0.95	ML 0.95	ML 0.95	ML 0.96	L 0.97

WIDE

WIDE

Figure 3.11 Design Basis Local Power Distribution for G.E. 8x8R Fuel

4.0 MAXIMUM OVERPRESSURE ANALYSIS

This section describes the analysis of the maximum overpressurization accident performed for compliance with the ASME code.

4.1 DESIGN BASIS

The reactor conditions used in the evaluation of the maximum pressurization transient are summarized in Table 3.1. These conditions are the same as those used in the transient analyses for thermal margin. In addition to these conservative assumptions, further conservatism was added by disallowing the operation of the four power-actuated relief valves as required by the ASME code. In further compliance, failure of the most critical active component was assumed. In this instance, the most critical active component is the reactor trip associated with the position of the Main Steam Isolation Valves (MSIVs).

4.2 PRESSURIZATION TRANSIENTS

Based on earlier analyses (Ref. 7), it has been determined that the maximum pressurization transient for the Dresden plants is the inadvertent closure of all MSIVs with failure of the MSIV position scram. The position scram, which commands reactor shutdown almost immediately upon MSIV movement, mitigates the effects of this event to the point that it does not contribute to the determination of thermal margins. Delaying the scram until the high pressure trip setpoint is reached results in a substantially more severe transient.

Although the closure rate of the MSIVs is substantially slower than that of the turbine stop or control valves, the compressibility of the fluid in the steam lines provides significant damping of the compression wave associated with the turbine trip events to the point that the slower MSIV closure without direct scram results in nearly as severe a compression wave. Once the containment is isolated, the subsequent core power production must be contained within a smaller system volume than that associated with the turbine trip events. Comparative analyses have demonstrated that the containment isolation event under these conservative assumptions results in a higher overpressure than either the turbine trip or the generator load rejection without bypass.

4.3 CLOSURE OF ALL MAIN STEAM ISOLATION VALVES

This calculation assumed that all four steam lines were isolated at the containment boundary within three seconds. The valve characteristics and steam compressibility combine to delay the arrival of the compression wave at the core until approximately three seconds from the initiation of the MSIV stroke. Effective shutdown is delayed until approximately 5 seconds following initiation of the MSIV stroke because control rod performance is assumed to be at the Technical Specification limits.

The maximum vessel pressure of 1341 psig was observed at approximately 6.5 seconds. The maximum steam line pressure of 1315 psig was observed at approximately 6.5 seconds. The maximum value of the sensed pressure in the steam dome was 1316 psig. The relative values of maximum pressure during the containment isolation transient indicate that the vessel and steam lines will be protected against overpressure limits defined in the ASME Code if a pressure safety limit of 1345 psig in the steam dome is protected.

Figures 4.1 and 4.2 illustrate the performance of major system variables during the overpressurization accident. This calculation was performed with COTRANSA.

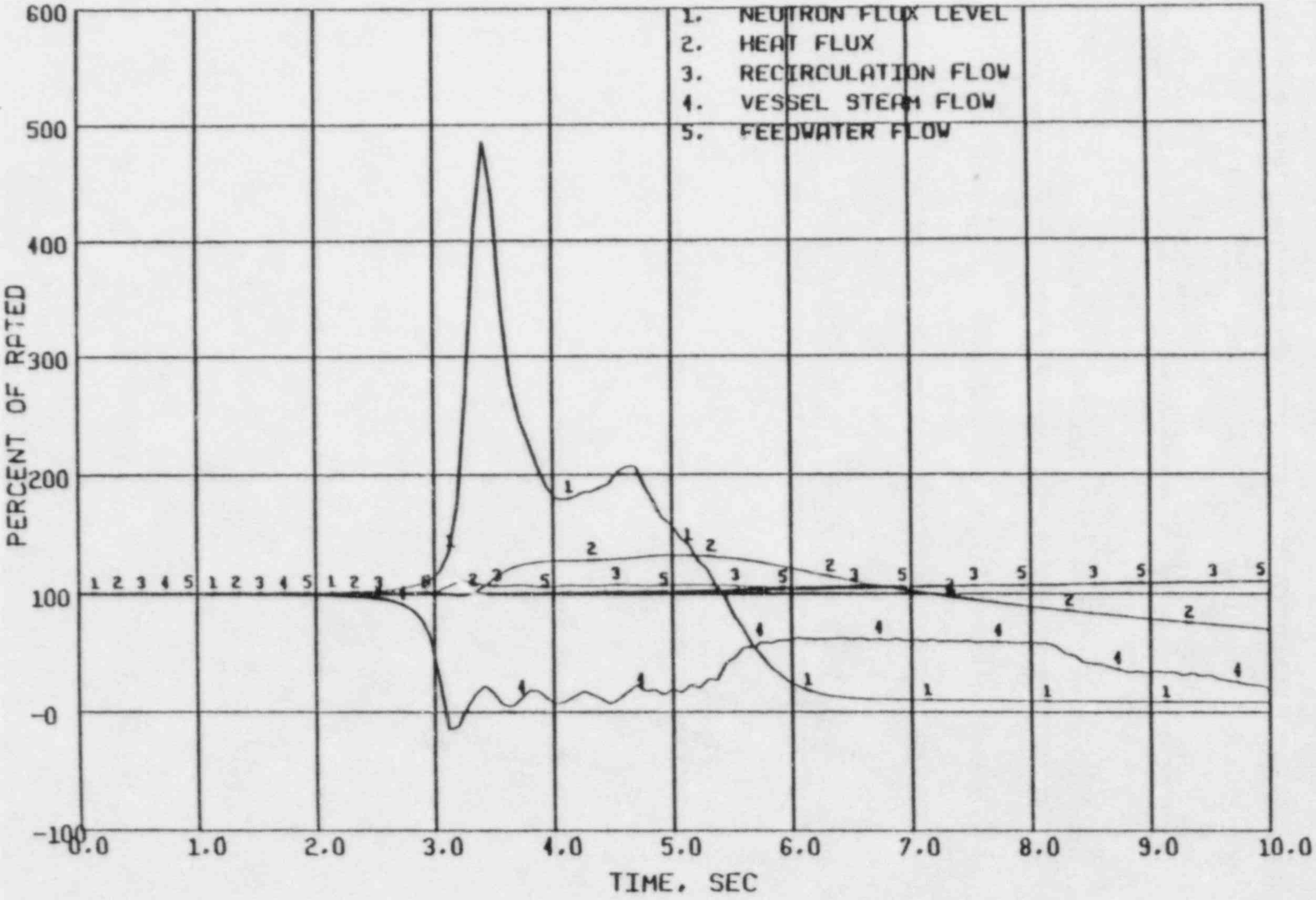

Figure 4.1 MSIV Closure

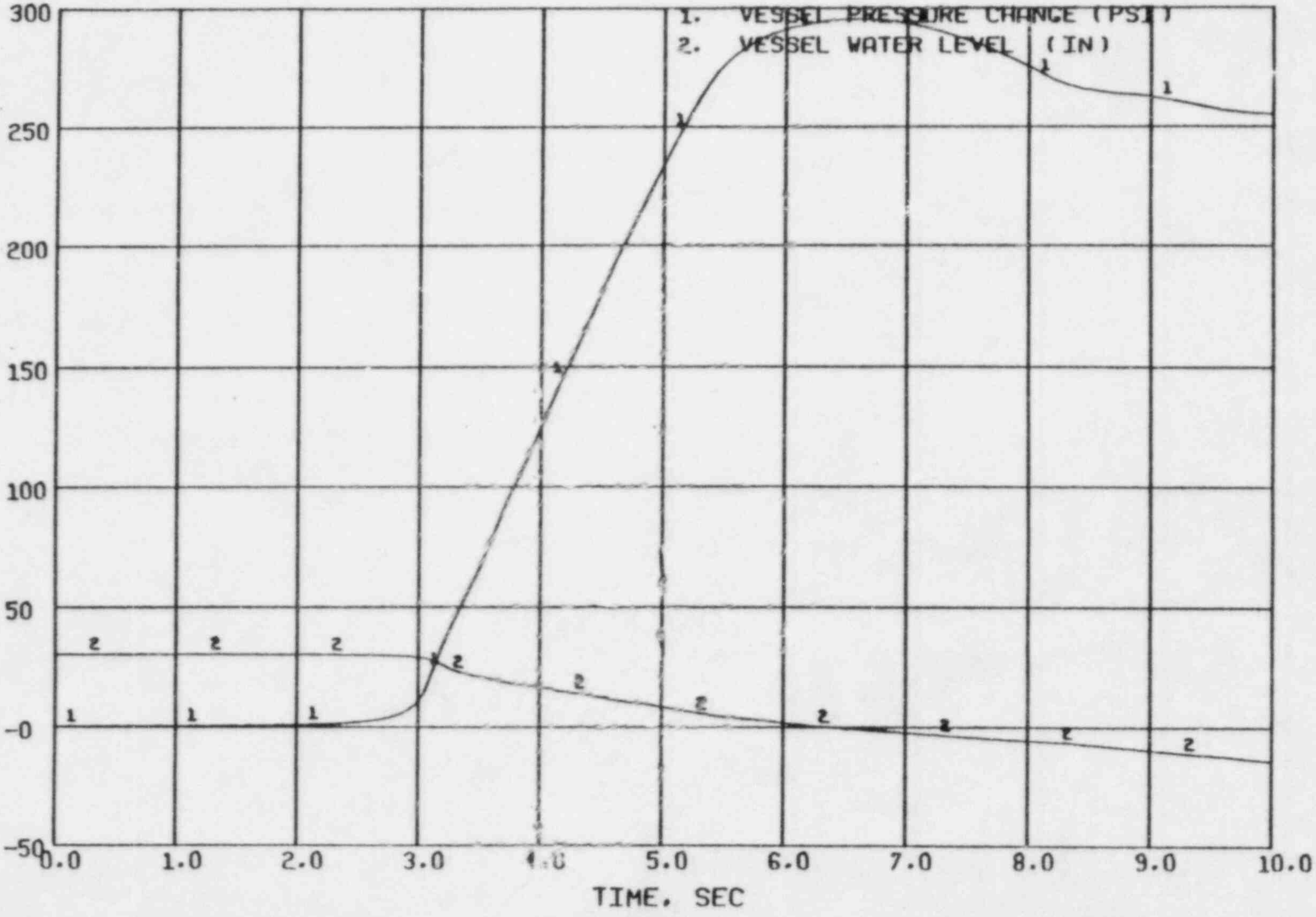

Figure 4.2 MSIV Closure

5.0 ANALYSIS OF OFF-RATED CONDITIONS

Transient analysis of a BWR requires consideration of transients at 'off-rated' and reduced flow conditions. The MCPR full flow operating limit is established through evaluation of anticipated transients which are expected to be most limiting at rated conditions. To assure that no thermal limits would be violated the generator load rejection without bypass and the feedwater controller faliure were also evaluated at three off-rated points on the expaned power to flow map of Fig. 1.1. The analysis indicates that all three points are bounded by the generator load rejection without bypass at the 100% power 100% flow position on the operating map. The results of the analysis are summarized in Table 5.1.

Analysis for pump runup events for operation at less than rated recirculation pump capicity indicates the need for an augmentation of the full flow MCPR operating limit for lower flow conditions. This is due to the potential for large reactor power increases should an uncontrolled pump flow increase occur.

The present analysis establishes the necessary reduced flow MCPR operating limit to protect the reactor fuel against boiling transtion during anticipated pump run-up events from off rated core flow conditions for both automatic flow control and manual flow control. These limits are shown in Figure 5.1 and 5.2, respectively. The cycle specific MCPR limit for Dresden Unit 3 shall be the maximum of the reduced flow MCPR operating limit depicted in these figures for the appropriate control mode and full flow cycle specific MCPR operating limit.

5.1 AUTOMATIC FLOW CONTROL

If the reactor is operated in the automatic flow control mode (AFC), variations in core power should not result in critical power ratios less than the established MCPR operating limit for rated conditions. If the rated condition MCPR limit is observed in a reduced flow condition, a subsequent increase in power to full power along the AFC control line may result in inadvertent degradation of fuel critical power ratios to below this reference(full power) MCPR operating limit. The probability of boiling transition conditions occurring during a subsequent anticipated event may increase beyond acceptable levels if this were the case.

Exxon Nuclear Company has determined the required reduced flow MCPR operating limit for off-rated conditions to prevent the MCPR from degrading below the Cycle 10 MCPR (full flow) operating limits during AFC operation. This was determined by evaluating the MCPR for a given reactor power distribution at varying total reactor power and flow conditions. The variations in total core power and flow were assumed to follow the expected relationship (Table 5.2) for automatic flow control operation (100% rod line). The power distribution chosen was such that MCPR equaled the referenced MCPR operating limit at rated conditions of power and flow. The expected variation of core pressure and inlet coolant subcooling with reactor power level was also considered.

The reduced flow MCPR's were then calculated by XCOBRA (Ref. 6) along the 100% rod line. The reduced flow MCPR limit for AFC is presented in Figure 5.1 and Table 5.4.

5.2 MANUAL FLOW CONTROL

This section dicusses pump excursions when the plant is not in automatic flow control operation mode.i.e., manual flow control. Based on the results obtained from the Dresden Unit 3 analysis which showed two pump excursions were the limiting pump run-up event, only two pump excursions are evaluated for Dresden Unit 3 Cycle 10. These results indicate that MCPR would decrease below the safety limit if the full flow reference MCPR was observed at initial conditions. Thus, an augmented MCPR is needed for partial flow operation to protect the two pump excursion event.

The evaluation of the two recirculation pump flow excursion for Dresden Unit 3 showed that establishment of MCPR limits for this event which prevents boiling transition will also bound single pump failures. The analysis of the two pump flow excursion indicates that the limiting event scenario is a gradual quasi-steady run-up due to the inlet enthalpy lag associated with a more rapid run-up.

The analysis conservatively assumed the reactor reaches 120% rated power at 110% rated flow. This power to flow relationship bounds that calculated by XTGBWR for the constant Xenon assumption. The reduced flow MCPR calculations were performed assuming the event was initiated from the APRM Rod Block Line as well as the 100% flow control line. The results show that pump run-up events initiated from the 100% flow control line are bounding.

The results of the two pump run-up analyses for manual flow control are presented in Figure 5.2 and Table 5.4. The cycle specific MCPR limit for Dresden Unit 3 shall be the maximum of the reduced flow MCPR operating limit and the full flow MCPR operating limit.

Table 5.1

Bounding Delta CPR's at Off Rated Conditions

DRESDEN UNIT 3 CYCLE 10

	Bounding Delta CPR		
Transient	GE 8x8	ENC 8x8	ENC 9x9
100% POWER 100% FLOW			
Generator Load Rejection Without Bypass	0.33	0.34	0.40
Feedwater Flow Controller Failure (Maximum Demand)	0.17	0.17	0.20
100% POWER 87% FLOW			
Generator Load Rejection Without Bypass	0.30	0.31	0.37
Feedwater Flow Controller Failure (Maximum Demand)	0.13	0.13	0.15
85% POWER 61% FLOW			
Generator Load Rejection With Bypass	0.31	0.31	0.38
Feedwater Flow Controller Failure (Maximum Demand)	0.18	0.19	0.24
67% POWER 39% FLOW			
Generator Load Rejection Without Bypass	0.12	0.12	0.14
Feedwater Flow Controller Failure (Maximum Demand)	0.08	0.08	0.09

Table 5.2 Automatic Flow Control

Recirculating Flow (% Rated)	Power (% Rated)
100	100
90	94
80	88
70	81
60	74
50	67
40	58

Table 5.3 Reduced Flow MCPR Limits for Automatic Flow Control

Recirculating Flow % Rated)	MCPR Limit	
	8x8	9x9
100	1.29	1.33
90	1.32	1.35
80	1.35	1.38
70	1.40	1.42
60	1.45	1.48
50	1.52	1.54
40	1.65	1.67

Table 5.4 Reduced Flow MCPR Limits for Manual Flow Control

Recirculating Flow % Rated)	MCPR Limit	
	8x8	9x9
100	1.10	1.09
90	1.15	1.14
80	1.21	1.19
70	1.27	1.25
60	1.35	1.32
50	1.44	1.42
40	1.58	1.53

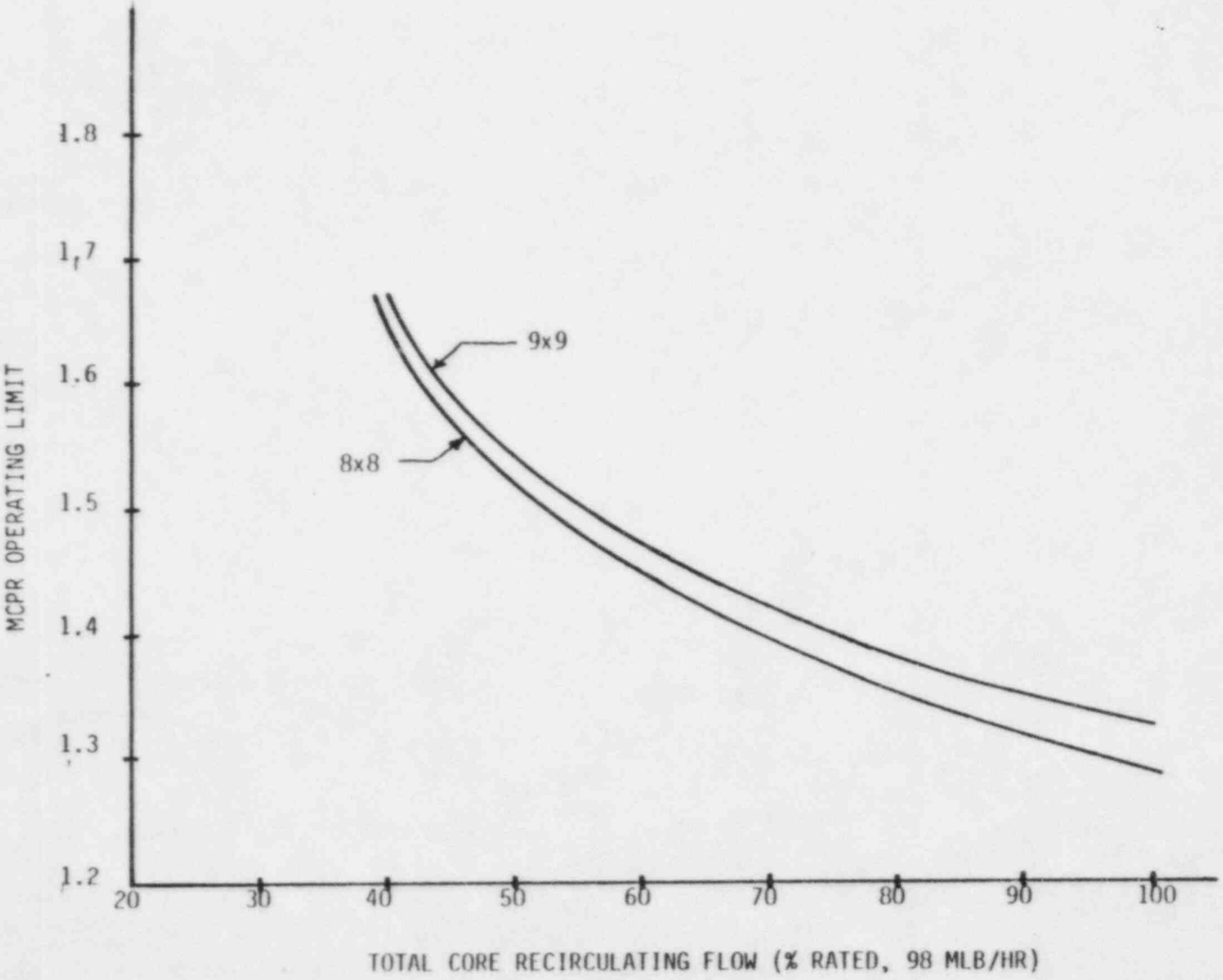

Figure 5.1 Reduced Flow MCPR for Auto Flow Control

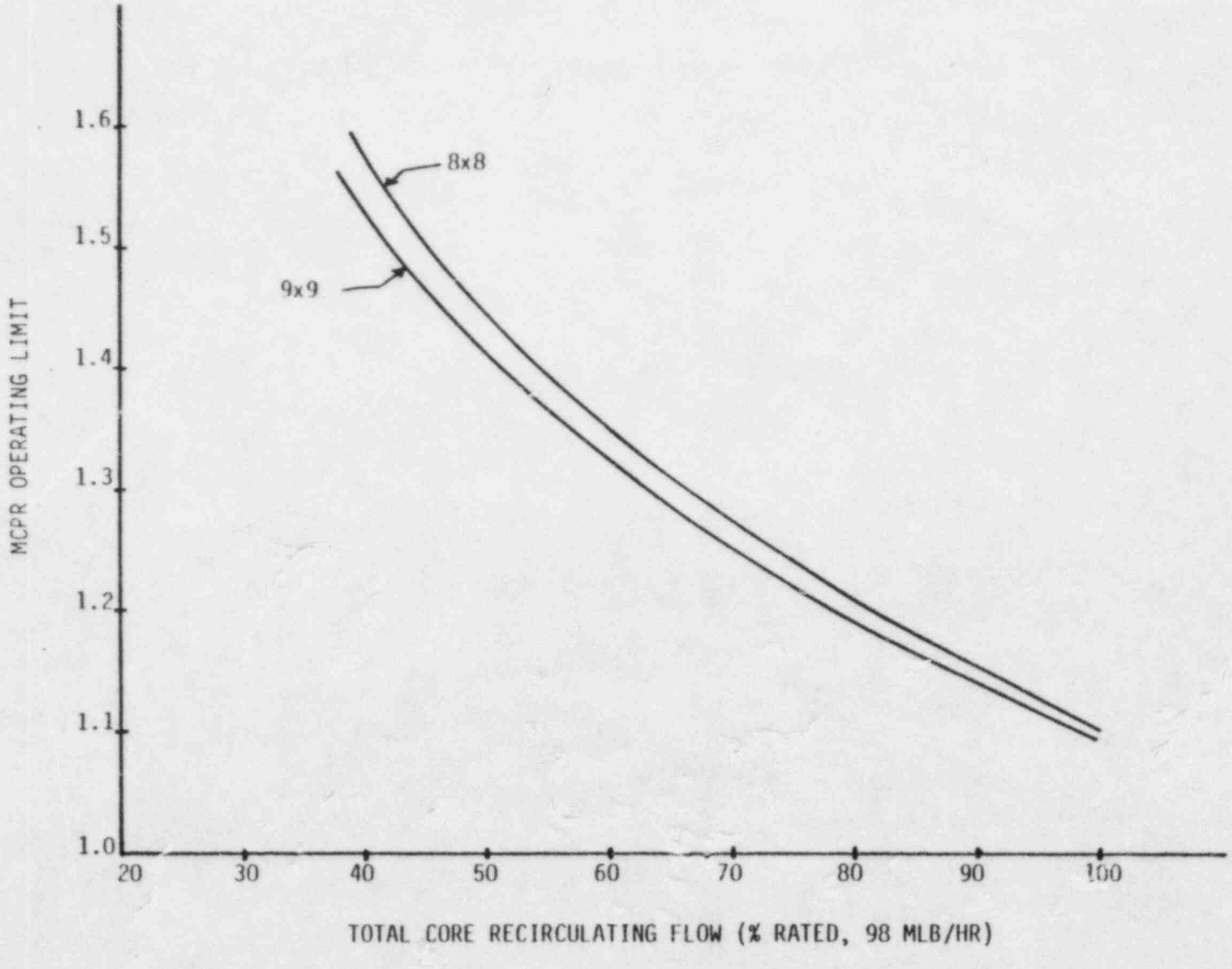

Figure 5.2 Reduced Flow MCPR for Manual Flow Control

6.0 REFERENCES

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

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

4. T.L. Krysinski and J.C. Chandler, "Exxon Nuclear Methodology for Boiling Water Reactors; THERMEX Thermal Limits Methodology; Summary Description," XN-NF-80-19(P), Volume 3, Revision 1, Exxon Nuclear Co., Inc., Richland, WA 99352 (April 1981).

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8. S. E. Jensen, "Revised Methodology for Including Uncertainties in Determining Operating Limits for Rapid Pressurization Transients in BWRs," XN-NF-79-71(P),Revision 2, Supplement 3, Exxon Nuclear Co., INc., Richland, Wa. 99352 (March 1985).

APPENDIX A

MODIFICATIONS TO COTRANSA/PTSBWR3

1.0 GENERIC CODE UPDATES

COTRANSA originated with the coupling of a plant transient simulator code, PTSBWR3, and a one dimensional, coupled neutronic hydraulic code, COTRAN. Subsequent to the licensing of Dresden-3 Cycle 9 the following modifications have been introduced into ENC's BWR plant transient model:

- o latest version of COTRAN replaced the original COTRAN
- o control system module introduced to coding
- o codes COTRAN, COTRANSA, PTSBWR3, and CONTROL all reside in the same program library

The latest version of COTRAN (JUL83) replaced the original COTRAN because the numerical convergence features had been upgraded to increase code execution efficiency. In addition the code core outflow is now deterministically calculated instead of assumed equal to the inlet flow. A Control System Module has replaced the original control system model so that all operations are handled through the input stream and may be tailored to the user's specific needs. This allows the user to simulate a reactor control system different than a pre-defined model. Having COTRAN, COTRANSA, PTSBWR3, and CONTROL in the same program library permits stand alone or grouped execution of each of the codes. It also allows individual or grouped modifications as required for application purposes.

2.0 PLANT SPECIFIC CHANGES TO COTRANSA/PTSBWR3

2.1 SUMMARY

The ENC plant transient analysis codes incorporata plant specific features through input and coding changes. This section discusses the required Dresden Unit 3 plant specific changes and control system input to the COTRANSA/PTSBWR3 plant simulator codes. These modifications were incorporated for exclusive use to Dresden Units 2 and 3. Unless otherwise stated, modifications that were made to the PTSBWR3 portion of the code were paralled in the COTRANSA portion.

2.2 FEEDWATER CONTROLLER

Because of the modifications made to COTRANSA the feedwater control system is incorporated into the code through the input stream. Figure A-1 gives a schematic of the control system used.

The feedwater control system maintains a pre-established level in the reactor vessel during normal plant operation by varying the speed of the steam turbine driven feed pumps. Steam flow and feed flow are compared and an error signal is sent to the mismatch gain amplifier. The sensed reactor water level is compared to the level setpoint, this error signal is summed with the mismatch gain amplfier signal to provide the input signal to the flow controller. The flow controller provides the input to the function generator after going through an output limiter and a lead/lag compensator. The function generator signal is then sent to the turbine feed pumps.

2.3 RECIRCULATION FLOW CONTROL SYSTEM

To determine if the manual or automatic mode of recirculation is more limiting, the loss of feedwater heating transient is analyzed in both methods of control. The recirculation flow control system is modelled as described in Reference 2. A diagram of the system is shown in figure A-2.

2.4 PRESSURE REGULATOR CONTROL SYSTEM

As is discussed in Appendix A, the pressure regulator control system is entered into COTRANSA/PTSBWR3 as input data. The model as it is input is shown in Figure A-3. Functionally, the pressure regulator adjusts turbine and bypass flow to maintain turbine pressure at a desired setpoint. Essentially, the system produces an error signal by comparing a sensed pressure with a pressure setpoint. This error signal is conditioned by the lead/lag characteristics of the control valve and produces a steam flow based on the pressure setpoint.

3.0 COTRANSA HOT CHANNEL MODEL

3.1 SUMMARY

The original COTRANSA hot channel model was used to give a figure of merit delta CPR used to determine the limiting transient. The limiting delta CPR was then determined by the user using a cumbersome XCOBRA-RODEX2-HUXY manual iteration. This involved a time consuming calculation where the user manipulated a considerable amount of data between codes. Furthermore, it also resulted in a transient condition being analyzed using steady state approximations. The COTRANSA hot channel model has been modified to automate the delta CPR calculation and to give a transient delta CPR. Each fuel type is modeled, and a delta CPR specific to that fuel type is determined. XCOBRA and RODEX2 are used to determine the input for each hot channnel. COTRANSA then calculates the delta CPR for each time step. The largest delta CPR is then reported.

3.2 MODIFICATIONS TO THE HOT CHANNEL

3.2.1 Flow Responce Surface

The modifications to COTRANSA include a time dependent calculation of the flow rate to the hot assembly of each fuel type. The initial and transient flow to the hot channel is determined using XCOBRA, ENC's approved subchannel code for BWR's. A steady state response surface for the hot assemblies' flow rates are determined for four key variables:

- o Relative assembly thermal power
- o Core average thermal power
- o Core average active flow
- o Core pressure

A quadratic equation is then determined for each hot assembly flow.

3.2.2 Fuel Temperature Model

The fuel temperature model for the hot rod is as described in the approved HUXY (XN-NF-79-71, Rev 2) with the clad gap conductance based on RODEX2 calculations. Each fuel type is run to the end of cycle, at the end of cycle the power is increased and the relationship of gap conductance to average fuel temperature is then determined. This gap conductance is then used in the hot channel model.

3.2.3 Critical Power Ratio Calculations

The MCPR calculation model used in the hot channel model is the approved XN-3 correlation as described in XN-NF-512, Rev 1. The hot channel model calculations do not interact with the core average solutions since the impact of the hot assembly is so small. Therefore, the boundary conditions which drive the hot channel model are stored and used iteratively. These boundary conditions are:

- Power
- Core inlet enthalpy
- Pressure
- Inlet flow rate
- Outlet flow rate
- Bulk fluid temperature
- Clad to fluid heat transfer coefficient
- Heat flux
- Axial power distribution
- Enthalpy rise

The purpose of the calculation is to determine the maximum allowable assembly power which will not exceed critical heat flux conditions during the transient.

The initial power used in the calculation is only an estimate. After the completion of the transient simulation the lowest calculated CPR is compared to 1.0 and the power of the fuel rod is modified. This new power is assumed as an initial condition. The flow to the limiting assembly is determined from the response surface and the enthalpy rise is adjusted to be consistent with the new conditions. The hot channel model calculations are repeated and the lowest CPR is again compared to 1.0. The process is then repeated until the lowest CPR is 1.0. The initial CPR minus the lowest CPR is the delta CPR for the transient consistent with ENC's reported methodology.

4.0 VERIFICATION OF THE HOT CHANNEL

Two different checks were made to insure the adequacy of the COTRANSA hot channel model's delta CPR calculation. The standard XCOBRA - RODEX2 - HUXY iteration was performed, and steady state conditions were input into the hot channel. The XCOBRA - RODEX2 - HUXY iteration resulted in delta CPR's that were more conservative than those for the hot channel model. This is what was expected because of the steady state nature of the method. When quasi-steady state conditons were forced into the hot channel, the results of the comparision were favorable and as expected.

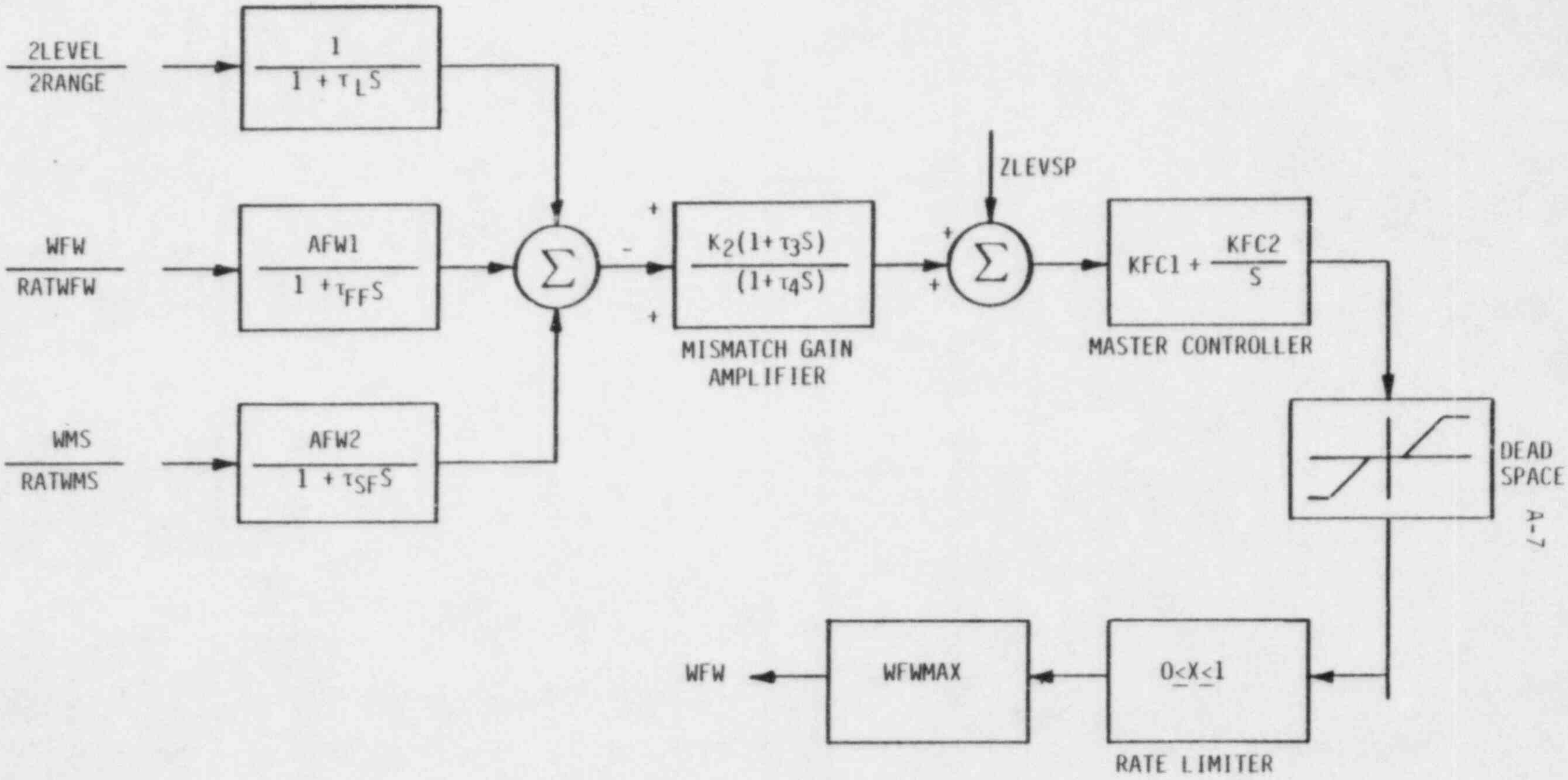

Figure A-1 Feedwater Controller

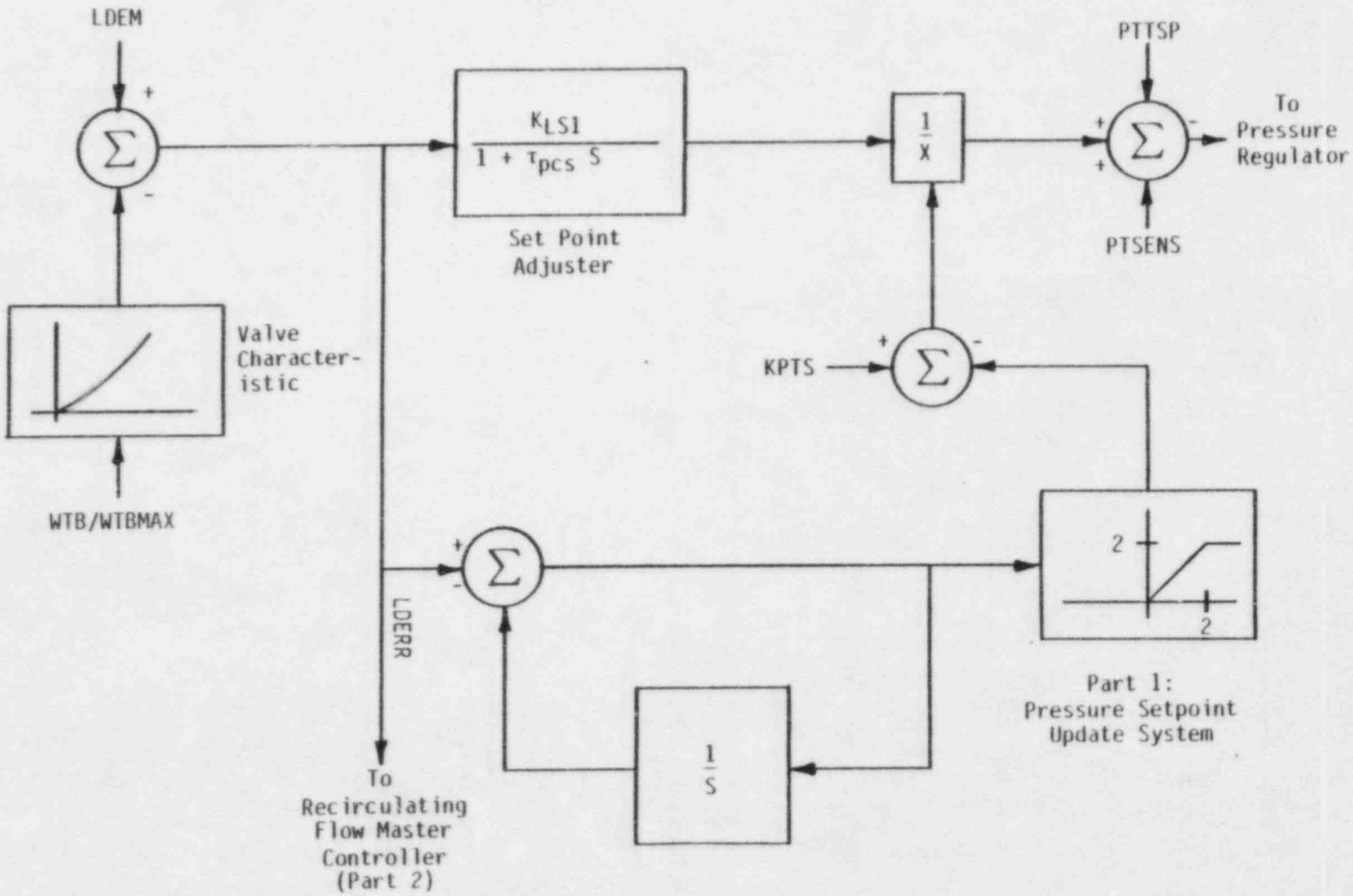

Figure A-2 Recirculation Flow Controller

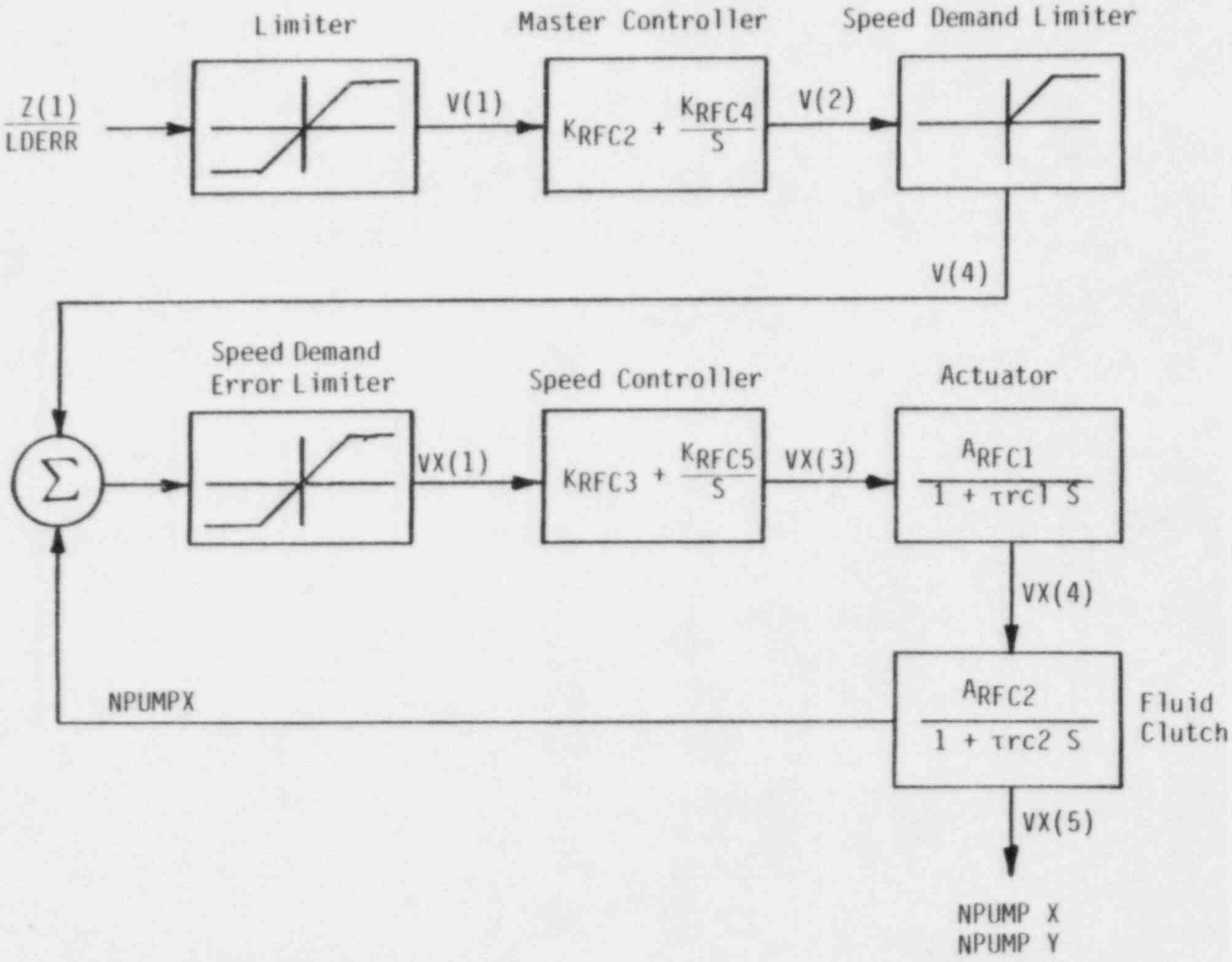

Figure A-2 (Cont.) Recirculation Flow Controller

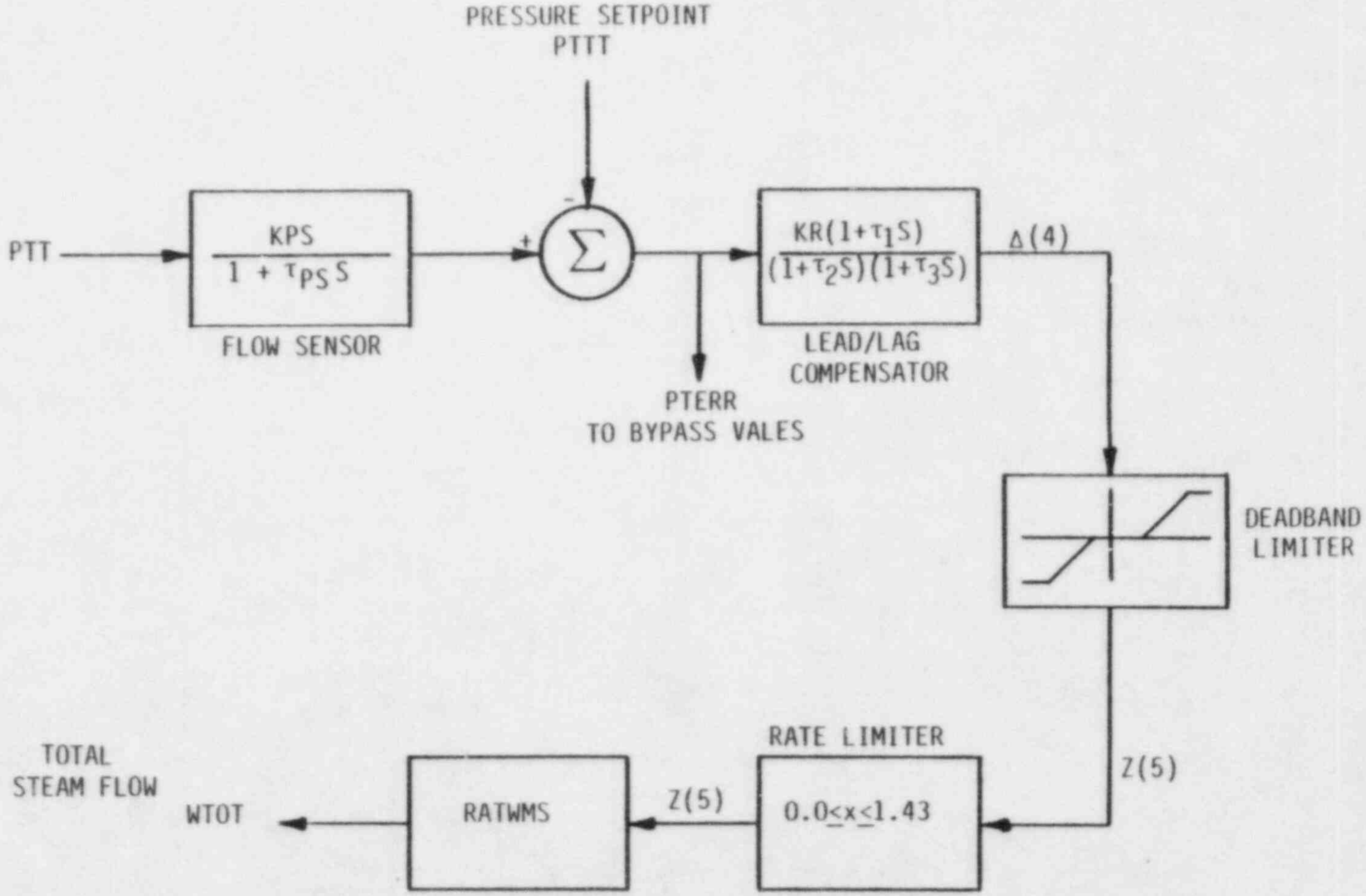

Figure A-3 Pressure Regulator

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DRESDEN UNIT 3 CYCLE 10 PLANT TRANSIENT ANALYSIS

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