PERMANENT PLANT CHANGES TO ACCOMMODATE EQUILIBRIUM CORE

SCRAM REACTIVITY INSERTION CHARACTERISTICS

MONTICELLO NUCLEAR GENERATING PLANT

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General Electric Company

Boiling Water Reactor Projects Department

RETURN TO REGULATORY CENTRAL FILES
ROOM 016

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INTRODUCTION

Analytical models have been effectively used for many years to describe the various operations, transients and hypothetical accidents associated with GE BWR's. As experience has accumulated, actual plant performance has been compared to model predictions and, where appropriate, changes were made to ensure model validity. Additionally, the evaluation of new concepts, methods and operational factors is a continuing program in GE directed toward increased confidence in the analytical methods. The development of increasingly more sophisticated analytical methods has provided a higher level of understanding of BWR dynamics which in turn has permitted the adjustment of operating techniques with the ultimate goals of greater plant safety and improved plant performance.

One such development has led to a better definition of the scram reactivity insertion function in exposed cores and the effects of changes in core dynamic characteristics resulting from the exposure dependent change in the scram reactivity function. Recent analyses indicate that this change could lead to consequences more severe than those previously analyzed single-failure-caused abnormal operational transients. Improvements in analytical techniques as well as changes to plant equipment, procedures and Technical Specifications have been applied to ensure the continuation of plant operation within previously established limits. When necessary, restrictions have been introduced to satisfy these requirements for specific intervals of operation. Meanwhile, further evaluations were progressing to determine what permanent action should be effected for long term elimination of the concerns resulting from the scram reactivity insertion function change.

Evaluations thus far have been sufficient to define the plant requirements for the initial fuel cycles. However, to eliminate the potential for exceeding established boundaries during future fuel cycles, an outer bound (equilibrium exposed core) scram reactivity function has been determined that represents the maximum degree to which the function could deteriorate. Based on this Design Basis Scram Reactivity Curve (which has been defined generically as the "D" curve) a combination of plant modifications has been defined which maintains the consequences of single-failure-caused abnormal operational transients within established limits for the expected lifetime of the plant.

PURPOSE AND SCOPE

This report is intended to provide the necessary justifications for adoption and implementation of the proposed plant modifications. While this report is considered a specific submittal on these modifications, contents relate directly to the Monticello Second Reload Submittal and should be treated as a supplemental supporting document.

Because the history of scram reactivity insertion function changes has been reported in great detail in previous submittals, a background discussion is not required in this report. The earlier reports are, however, included by reference.

Over the course of several fuel cycles the core average exposure will increase until an equilibrium value is attained. Subsequent replacement of high exposure fuel with new fuel will lower the average exposure, where upon the progressive buildup of exposure on the refueled core will again approach the equilibrium value. This equilibrium exposure thus represents the maximum core average exposure that will be attained for each fuel cycle once an equilibrium cycle occurs. Analyses of the core parameters at this exposure were made to ensure an outer bound was established. The Design Basis Scram Reactivity Insertion Curve ("D" Curve) is more conservative than the most extreme case evaluated. The "D" curve is therefore considered to be a boundary in defining the worst case curve expected in the life of the plant since the determination has been made (with 95% confidence) that it will not be exceeded over the expected lifetime of the plant. The "D" curve is used in the analyses of this report.

B. SAFETY RELIEF/VALVES

The primary concerns resulting from the changing of the scram reactivity curve have been in satisfying required pressure margins and minimizing fuel thermal duty as a result of the occurrence of abnormal operational transients, particularly events associated with turbine and generator trips with concurrent failure of the bypass valves. While other transient events are affected, the consequences are such that recommended margins are maintained.

GE recommends the maintenance of a minimum margin of 25 psi between the peak pressure resulting from the worst case pressurization event (turbine trip without bypass) and the setpoint of the lowest set spring safety valve throughout the lifetime of the plant. Previous analyses have revealed possible difficulty in maintaining this margin. Resetting the safety valve setpoints and other changes have been effected in the past to alleviate this problem.

For Monticello, the replacement of the four spring safety valves whose discharge is directly to the drywell with four combination safety/relief valves piped to the torus suppression pool eliminates two potential concerns, the meeting of the minimum margin and ASME overpressure protection requirements.

The recommended 25 psi margin is intended to preclude lifting of the spring safety valves during transients, the result of which would be the admission of reactor steam to the drywell air space. This does not present a plant safety problem but could introduce operating inconveniences. Replacement of the spring safety valves with combination safety/relief valves having piped discharges eliminates this concern.

The required reactor vessel overpressure protection as defined by the ASME Boiler and Pressure Vessel Code is satisfied through self actuation of the combination safety/relief valves. Analyses have shown that ASME overpressure protection requirements are fully met with the proposed change.

The addition of four combination safety/relief valves to the four already installed* increases the steam release capability from the reactor at lower pressures during isolation-type pressurization transients, with a resultant reduction in peak pressure.

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*Analyses deomonstrate that established limits are met with 6, 7, or 8 safety/relief valves. During the outage for the second reload, all safety valves will be removed and two safety/relief valves added bringing the plant to an acceptable configuration. The other two will temporarily be blanked off. At a subsequent outage, the other two safety/relief valves will be added so that the plant will in effect have two installed spares.

D. PROMPT RELIEF TRIP (PRT) SYSTEM

The most significant modification to be effected for the coming fuel cycle (Cycle 3) is the Prompt Relief Trip System. This system is designed to anticipate the pressure transient resulting from a turbine or generator trip with the assumed failure of the bypass valves.

Since the scram reactivity curve has been assumed for analytical purposes to reach that described by the design basis curve ("D" curve), the pressure transient caused by a turbine trip without bypass could result in pressure peaks and fuel thermal duties in excess of recommended analytical limits. During these events, the delayed negative reactivity insertion would allow a rapid buildup of energy in the fuel during the time the relief valves were "waiting" for the attainment of setpoint pressure and subsequent opening. By lifting the relief valves at the first indication of the start of this type transient, (turbine trip), blowdown is effected much sooner and the peaks of interest are correspondingly lower. The transient is most severe if the bypass system does not function. However, since the time required to determine the operability status of the bypass system is relatively too long, i.e., the event proceeds too rapidly to allow such a delay, PRT is triggered by a turbine trip whether the bypass valves open or not.

Because this is a completely new system, a full description and analysis is provided in Appendix A. A summary description is provided below:

1. Definition: Prompt Relief Trip (PRT) is the plant action whereby a predetermined number of safety/relief valves are actuated promptly following a turbine or generator trip.

2. Purpose: The objective of the PRT system is to compensate for equilibrium core scram reactivity insertion functions by minimizing the peak pressure and fuel thermal effects resulting from pressurization type abnormal operational transients.

3. Functional Description and Design Requirements: The PRT will conform to IEEE 279, and shall operate relief valves via two independent, power level biased channels, each of which is capable of operating all safety/relief valves connected to the system.

4. PRT Enable: Three conditions must be established to ENABLE the PRT system to operate; it must be manually reset, reactor power level must be in the proper range and a closure of the main turbine stop valves or fast closure of the turbine control valves must occur.

   When in the ENABLE mode the PRT system provides, within 0.100 seconds, a trip demand signal to the safety/relief valve solenoids that is sustained until a DISABLE signal is provided. An interval timer is also started concurrent with the trip demand signal.

b. Weight Analysis.

c. Safety/Relief Valve Blowdown. All eight lines are analyzed. Each line is individually modeled and analyzed. The line is considered to be a lumped mass-spring assemblage and anchored at the connections with main steam and suppression chamber. The calculation of the forcing functions are based on F J Moody's ASME paper 73-FE-23 "Time Dependent Pipe Forces Caused by Blowdown and Flow Stoppage".

d. Seismic. A response spectrum dynamic analysis is used. Each main steam line with its respective safety/relief discharge lines is modeled as one lumped mass-spring assemblage.

For conditions a., b., and d., the analysis model includes the main steam line from the reactor vessel nozzle to the flued head connection and the safety/relief valve discharge line from the main steam line to the suppression chamber.

Condition c. will only be applied to the eight safety/relief valve lines from the main steam line to the suppression chamber.

The desing loading combinations and stress limits are:

a. Design, Normal and Upset Conditions

1) $PD + DW \leq 1.0\ S_h$

2) $PD + DW + \frac{\sqrt{MS^2 + MB^2\ max}}{Z} \leq 1.2\ S_h$

3) $SE \leq Sa$

where:

Loadings are

PD = Internal design pressure

DW = Piping weight

SE = Thermal expansion

MS = Moment due to seismic response in X, Y, and Z direction

MB = Maximum moment due to blowdown forces in X, Y, and Z directions at the same time.

NOTE: Equation 2 conditions must be met in the X, Y and Z direction.

Z = Section modulus of pipe

The primary PRT disable signal is derived from an interval timer. The PRT safety/relief valves close either from the self-actuation closing set point or from the termination of the timer interval, whichever occurs later. The timer interval may be set anywhere within the range of 0-10 seconds, however, to minimize interference with the normal designed pressure regulatory functions of the valve the recommended setting is 5 seconds.

The secondary PRT disable signal is derived from dome pressure. This pressure set point must be set below the lowest operating dome pressure which is within the range of PRT operation. The lowest pressure normally corresponds to the lowest PRT power level, which is 67% when allowing for 3% tolerance at the power level boundaries. Optimizing the margins involved with this parameter, the pressure setpoint for the PRT pressure disable will be set low enough to ensure that the PRT system is available for all applicable power levels but high enough to avoid interference with other limits.

The analyses were based on the following operating conditions except as otherwise noted.

Core Model	8x8 Physics/7x7 Geometry Reload 2 Cycle 3
Thermal Power	1670 MWT (T-G Design)
T-G Design Steam Flow	6.77×10^6 lb/hr
Bypass Capacity	15% - Design Flow
Safety/Relief Valve Capacity (6 valves @ 1080 psig + 1%)	71.1% - Design Flow
Safety/Relief Valve Set Point	1080 psig + 1%
Safety/Relief Valve Time Delay	0.4 seconds
Safety/Relief Valve Stroke Time	0.1 seconds
PRT - Timer Interval	5 seconds
Scram Reactivity Curve	Figure 1 (Design Basis "D")
Scram Rod Drive	Figure 2 (3.5 seconds @ 90% stroke)
Feedwater Capacity	105% - Design Flow
Feedwater Temperature	376°F
Void Coefficient	0.37¢/% rated voids
Doppler Coefficient	0.2¢/°F

The transient is less severe than the similar case resulting for a turbine trip since stop valve closure is faster (0.1 sec) than control valve closure (0.2 sec) while the protection system response is the same for each case. Below about 15% of rated power, the bypass system will transfer steam around the turbine and avoid reactor scram. Between about 15 and 30% power, high pressure scram will result unless operator action can reduce power to within the bypass capacity.

Generator Trip with Bypass Failure with PRT

The worst single failure generator trip case occurs if the bypass system fails to open. Since adequate fuel and overpressure margin are shown for the turbine trip without bypass cases, the generator trip without bypass cases also have adequate margins because of the slower turbine steam shutoff. The former cases are covered below.

Turbine Stop Valve Closure - Turbine Trip

Nominal Turbine Trip

The nominal turbine trip with PRT exhibits a sharp transient with respect to pressure and power. A simultaneous logic trip initiates the recirculation pump trip. Since the Monticello valve configuration has eliminated the spring safety valves, the pressure margin of interest is between the peak vessel pressure and the vessel code limit. This margin is substantial for this transient. The data of interest for this transient at 100% power, which is the most severe condition, are tabulated below;

Peak Vessel Pressure	1123 psig
Pressure Margin (to vessel code limit)	252 psi
Peak Neutron Flux	249%
Peak Average Surface Heat Flux	109%
MCHFR	1.4

It is apparent that no limits are endangered. Figure 3 illustrates the response of this transient.

peak average surface heat flux is increased by 1% but no significant shange occurs in MCHFR. Figure 4a illustrates the response to this transient.

Isolation Valve Closure

Automatic circuitry or operator action can initiate closure of the main steam isolation valves. Position switches on the valves provide reactor scram if the valves are less than 90% open and reactor pressure is above 600 psig. However, reactor protection system logic does permit the test closure of one valve without initiating scram from the position switches. Inadvertent closure of one or all of the isolation valves from reactor scrammed conditions will be less severe than the maximum power cases (maximum stored and decay heat) which follows.

Closure of All Isolation Valves

Figure 6 shows the changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at 1670 MWt power. Reactor scram is initiated by the isolation valve position switches before the valves have traveled more than 10% from the open position. A 3-sec nonlinear valve closure was simulated, which is the fastest allowable closure time. Scram is initiated before any significant flow interruption occurs; therefore, no neutron flux, fuel center temperature, or fuel surface heat flux peaks occur. A slight reduction in CHFR occurs. The nuclear system safety/relief valves open when pressure reaches the setpoint (1080 psig + 1%) at about 2.22 sec after the start of the isolation. They close sequentially as the stored heat is dissipated and will continue to intermittently discharge the decay heat. The peak pressure in the main steam line is 1134 psig. Peak pressure at the bottom of the vessel is 1160 psig and in the dome it is 1137 psig, clearly below the pressure limits of the nuclear system process barrier.

Closure of One Main Steam Isolation Valve

Closure of one main steam isolation valve is desirable for testing purposes. Therefore, the protection system logic from the valve position switches permits full shutoff of any one of these valves without initiating scram. Normal procedures for such a test require an initial power reduction to about 80-90% of design conditions to avoid high flux or pressure scram. Figure 6 graphically shows the changes of important nuclear system variables during the simulated 3-sec closure of an isolation valve in one of the four main steam lines from a maximum power of 1670 MWt. The steam flow disturbance increases the vessel pressure and reactor power causing a high neutron flux scram. The peak surface heat flux is 104% of rated and peak center fuel temperature is increased only 76°F; however, MCHFR is above 1.8, showing that no fuel damage occurs. Peak pressures remain below the setting of the relief valves.

initial pressure regulator failure;

inadvertent opening of a relief/safety valve;

loss of feedwater flow; and

loss of auxiliary power.

Failure (Open) of the Operating Initial Pressure Regulator

The initial pressure regulator (IPR) can be assumed to fail in either of two ways: zero or maximum output. If the controlling regulator fails in a closed direction, the backup IPR takes over contro of the turbine admission valves, thereby preventing a serious transient. If either the controlling or the backup regulator fails in an open direction, the turbine admission valves can be fully opened plus at least part of the bypass capacity. The maximum control plus bypass valve demand is limited by the control system to 110% or less. This potential reactor depressurization would impose additional stresses to the nuclear system, but are avoided by means of reactor isolation and subsequent scram.

Figure 8 shows the transient which occurs when the operating IPR fails to open at 1670 MWt and the reactor is isolated when the turbine throttle pressure drops to 850 psig in order to shut off this uncontrolled release of steam and scram the reactor. Neutron flux is decreased sifnificantly as the pressure drop causes a rapid formation of voids in the core which reduces power quickly. When the turbine throttle pressure decreases to 850 psig (at about 10.7 sec), closure of the main steam isolation valves is initiated. Scram trip occurs when the isolation valve stroke has reached the 10% closed position. Steam line and vessel pressures drop over 100 psi before the isolation becomes effective near 13.7 sec and the reactor is shut down with pressure rising slowly. Eventually, the relief valves open partially to dissipate the stored heat and then close sequentially as they follow the decay heat characteristics. No reduction in fuel thermal margins occurs. The isolation limits the duration of the duration of the depressurization so that no significant thermal stresses are imposed on the nuclear system process barrier.

Inadvertent Opening of a Relief or Safety Valve

A mild depressurization transient is introduced by the inadvertent opening of a valve on the main steam line. Figure 9 shows the transient simulated for the inadvertent opening of an 11.85% capacity relief valve from maximum power. The turbine pressure regulator senses the pressure decrease and drops turbine flow to maintain pressure control. The reactor settles out at nearly the initial power; however, the automatic flow control increases core flow to its maximum capability, and since it cannot satisfy the full load demand, decreases the turbine pressure regulator setpoint adjustor to its lower limit. No significant reduction in CHFR occurs, and the turbine pressure regulator prevents the occurence of any uncontrolled depressurization.

(This event also describes the single-failure-of-the-PRT specification where a single valve opens inadvertently.)

Loss of Auxiliary Power

Loss of Auxiliary Power is defined as an event which de-energizes all buses that supply power to the unit auxiliary equipment such as recirculation pumps, condensate pumps, and circulating water pumps. Two methods of experiencing this event are postulated:

trip (s) or fault (s) occurring in the auxiliary power distribution system itself without transfer to outside sources, and

complete loss of all external connections to the grid.

Estimates of the responses of the various reactor systems provided the following simulation sequence for the first type of transient:

all pumps are tripped at time = 0. Normal coastdown times were used for the recirculation and feedwater pumps;

the protection system M/G sets were assumed to coastdown in 5 sec to the point where the scram and main steam isolation occurred; and

loss of the main condenser circulating water pumps was estimated to cause condenser vacuum to drop to the turbine trip setting by 6 sec. PRT will trip the safety/relief valves coincident with the turbine trip.

Figure 11 graphically shows the simulated transient. The initial portion of the transient is very similar to the simple loss of all feedwater. Initiation of scram, isolation valve closure, and turbine trip with PRT trip of the safety/relief valves all occur between 5 and 6 sec and the transient changes to that of an isolation. The relief and bypass valves open for a short time and then reclose as the remainder of the stored heat is dissipated. Peak pressures reach less than 100 psi above normal operating pressures; therefore, no vessel pressure limits are approached.

Note how the safety/releif valves are reopened and reclosed as the generated heat drops down into the decay heat characteristic. This pressure and relief cycle will be continued with slower frequency and shorter relief discharges as the decay heat drops off up to the time the shutdown heat exchangers can dissipate the heat.

An alternate transient results if complete connection with the external grid is lost at time = 0. The same sequence as above would be followed except that the reactor would also experience a generator load rejection with its associated trip scram and PRT trip of the safety/relief valves at the beginning of the transient. Figure 12 shows this simulated loss of auxiliary power event. Note that the peak in neutron flux is limited to 258% by scram, PRT and the pump trips. The peak fuel surface heat flux is limited to 109% of the initial value. Peak pressures are virtually identical to the previous case; however, they occur sooner during the transient.

controller less severe. Response of the plant to this transient is shown in Figure 14 which results when one speed converter swings toward full pump speed at its maximum rate of 25%/second. The initial reactor condition corresponds to 65% power and 50% flow, the lower limit of the maximum expected automatic control region extends from rated conditions to 75% power and 70% flow. the transient analysis is based upon the widest possible flow control range to fully bracket the band of expected operating conditions.

Prior to the failure, both scoop tube actuators are at about the 20% position. The good speed converter system remains near this position, while the failed actuator reaches full stroke by 4 seconds. Diffuser flow in the failed recirculation loop quickly increases. It reaches 150% by 10 seconds and levels out near 140%. Diffuser flow in the opposite jet pumps is decreased by the greater core ΔP. Core inlet flow increases rapidly to about 72% causing neutron flux to increase and scram the reactor at 2.9 seconds. The neutron flux reaches 216%, however, peak fuel surface heat flux is only 83.5%, maintaining adequate thermal margins.

It should be noted that this assumed transient will progress differently (double failure) due to a trip of the overcurrent relay of the MG set drive motor in question which freezes the scoop position terminating the transient with a much smaller neutron flux spike not causing the reactor to scram.

Pump Trips

An abrupt reduction in core flow causes an increase in the core void fraction and thereby decreases reactor power. The fuel surface heat flux decreases more slowly than the coolant flow because of the lag due to the fuel time constant, so thermal margins momentarily decrease. Therefore of primary concern are the fuel thermal margins which are experienced throughout these transients.

They finally will lead to steady-state power/flow characteristics with thermal margins greater than the initial high power condition. The rotating inertias of the recirculation flow control system are chosen to provide acceptable flow reductions for all pump trip possibilities. The worst cases, where the fuel thermal margins are of greatest concern, occur for maximum initial power, 1670 MWt. This section covers the transients where one or both recirculation pumping systems are deengerized.

Trip of One MG Set Drive Motor Breaker

Normally the trip of one recirculation loop is accomplished through the MG set drive motor breaker. However, a worse transient occurs if the generator field exciter breaker is opened, separating the pump and its drive motor from the rotating inertia of the MG set.

Figure 15 shows this transient while operating at 1670 MWt. If one of the drive pumps were to fail (equalizer line is closed), half of the jet pumps will coast down and cease to function. Core flow drops to 60% and neutron flux dips to about 46% before settling at 67%. With the initial MCHFR near 1.9, it remains well above 1.5 through the transient, therefore showing no threat to fuel thermal margins.

The idle drive pump suction and discharge-bypass valves have just been opened, but the equalizer line and the drive pump main discharge valve are closed.

The speed converter scoop tube in the dead recirculation loop is at a position corresponding to approximately 50% generator speed demand for loaded condition.

The cold loop startup is simulated by the following sequence:

The MG drive motor breaker is closed at t=0.

The MG set drive motor approaches synchronous speed quickly, while the unloaded generator (field breaker is still open) reaches "significant" speed near 5 sec.

The generator field breaker is automatically closed at 5 seconds, loading the generator with the starting torque of the pump drive motor and starting the recirculation pump.

After the pump has started the speed demand is sequentially reprogrammed back to 20% of rated pump speed.

The pump main discharge valve is started open as soon as its interlock clears when the MG set drive motor breaker is closed. (Normal procedure would delay valve opening to separate the two portions of the flow transient and make sure the drive loop contains properly mixed vessel-temperature water.) A nonlinear 30-sec valve opening characteristic was used.

Shortly after the pump begins to move, a surge in core flow causes the neutron flux to peak at 83%. The fuel surface heat flux responds slowly, peaking at 64.8%. No thermal limits are approached, therefore no serious threat to the fuel cladding is experienced. Throughout the transient, diffuser flow in the started up loop remains reversed because of the strong back flow effect of the other live loop. For this reason the cold water does not significantly affect the transient since it mixes in the bulkwater region, travels through the live loop and into the lower plenum before reaching the core inlet.

CONCLUSION

The significant changes associated with the reload 2, cycle 3 core for Monticello which have required a reanalysis of the FSAR abnormal operational transients are as follows:

1) 8x8 Fuel
2) Scram Reactivity (D" Curve)
3) Scram Rod Time (67B)
4) Six combination safety/relief valves
5) PRT

Depressurization operation

The required number of valves are opened by indirectly operated devices as part of the protection system for small line breaks

The first four valves were manufactured by the Target Rock Corporation to ASME Section III with Summer 1966 Addenda. The quality of additional valves will be the equivalent of the first four valves.

Relief valve set points and capacity

Nominal Set Point (psig)	ASME rated capacity @ 103% of set pressure (lb/hr)
1080	792,000

The pressure set points used in analysis for the safety/relief valves are shown in the Table below:

	Capacity (Based on valve setpoint)	Setpoint for 1670 Mwt Power at 980 psig Turbine Inlet Pressure
Safety/Relief Vlaves (6 valves)	71% of Nuclear boiler rated steam flow	1080 psig + 1%

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from the consequences of pressure in excess of the vessel design pressure.

A peak allowable pressure of 110% of the vessel design pressure is allowed (1375 psig for a 1250 psig vessel.)

The lowest qualified safety valves setpoint must be at or below vessel design pressure.

The highest safety valve setpoint must not be greater than 150% of vessel design pressure (1313 psig for a 1250 psig vessel).

The safety valve capacity is determined in order to accommodate the pressure rise accompanying the following hypothetical event.

The plant is operating at turbine design power conditions 1670 MWt with 980 psig turbine inlet pressure.

The bypass valves do not open.

Direct reactor scram is neglected.

The PRT does not affect the capacity of the suppression system in that the suppression system was analyzed and designed to accommodate the total energy present in the primary system; the energy to be dissipated is not changed.

Hydraulic analyses have been performed on the torus previously. In these analyses, torus sections or segments are treated as individual hydraulic cells where relief valve discharges (through "rams head") into adjacent cells are evaluated. Because of the additional relief valve discharge piping required for the modification described in this report will be installed in presently "unused" torus segments, and the basic geometric arrangement and spacing will be unchanged, the previous analyses will remain applicable. A larger number of discharge interfaces will exist, but the basic analytical relationships will not change so that an unreviewed condition will not result from the modifications.

Structural effects of the RV discharges on the torus will not be different from previously analyzed events; a more symmetrical discharge pattern in the torus is expected to reduce overall stress due to the balancing of the forces involved.

F. TECHNICAL SPECIFICATIONS

Proposed Technical Specification changes are now being written and reviewed; they will be submitted at a later date.

The proposed changes to the Technical Specifications will include all changes required for Cycle 3 operation. These changes include consideration of the proposed modifications described in this report (PRT, RV/SV changes, control rod scram times) as well as the insertion of 8 x 8 reload fuel, and all transient analyses discussions.

G. REFERENCES

1. Monticello Nuclear Generating Plant, First Reload License Submittal, February 1973.

2. "Results of Transient Reanalysis for Monticello Nuclear Generating Plant with End-of-Cycle Core Dynamic Characteristics," February 1973.

3. "Monticello -Safety Valve Setpoint Increase Analysis," Change Request dated September 3, 1973.

4. "Monticello - Cycle 2 Scram Reactivity Considerations, Analyses and Modifications," October 1973.

5. "Monticello - Second Reload Submittal," November 1973.

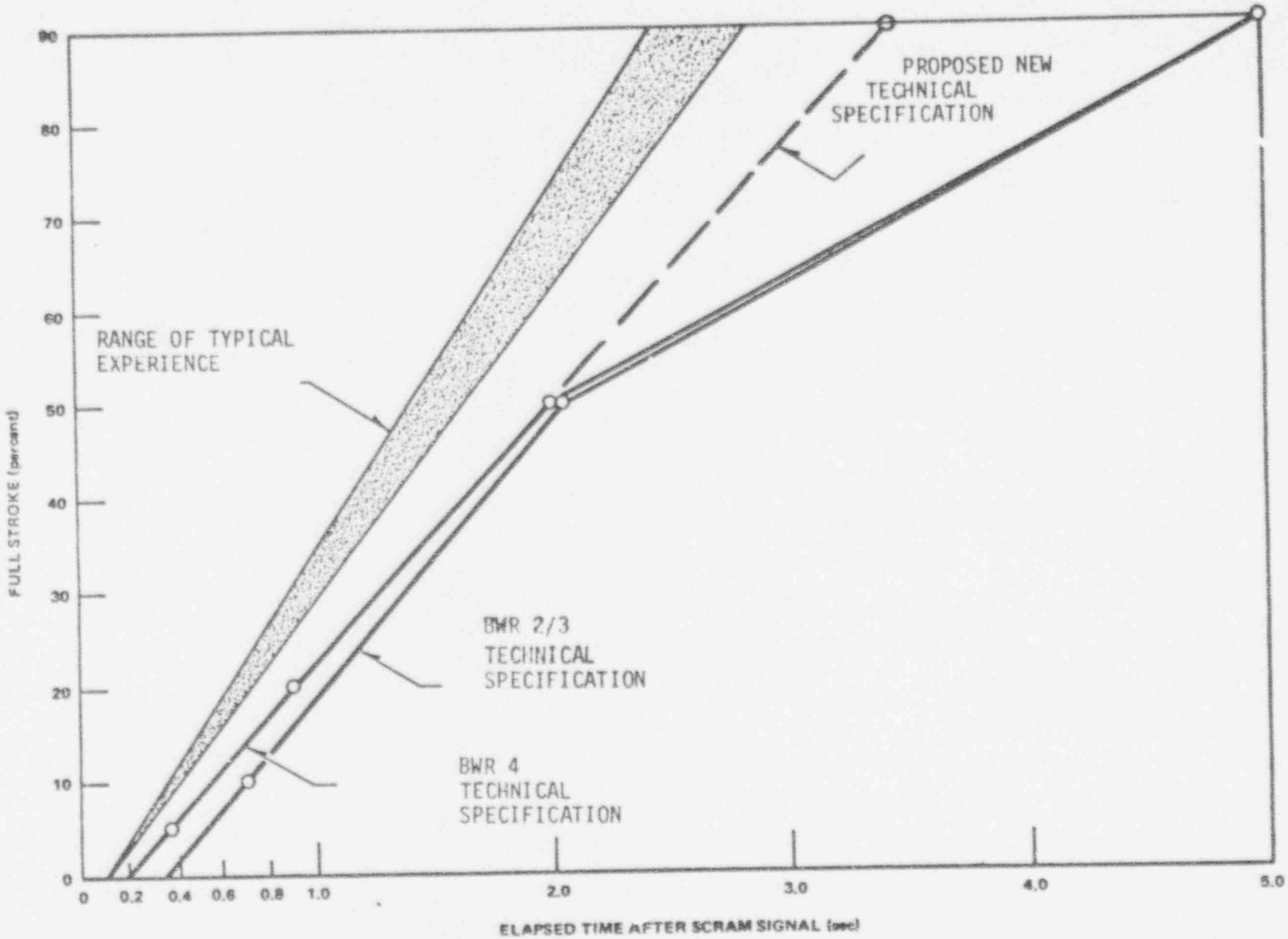


FIGURE 2

CONTROL ROD DRIVE SCRAM TIMES

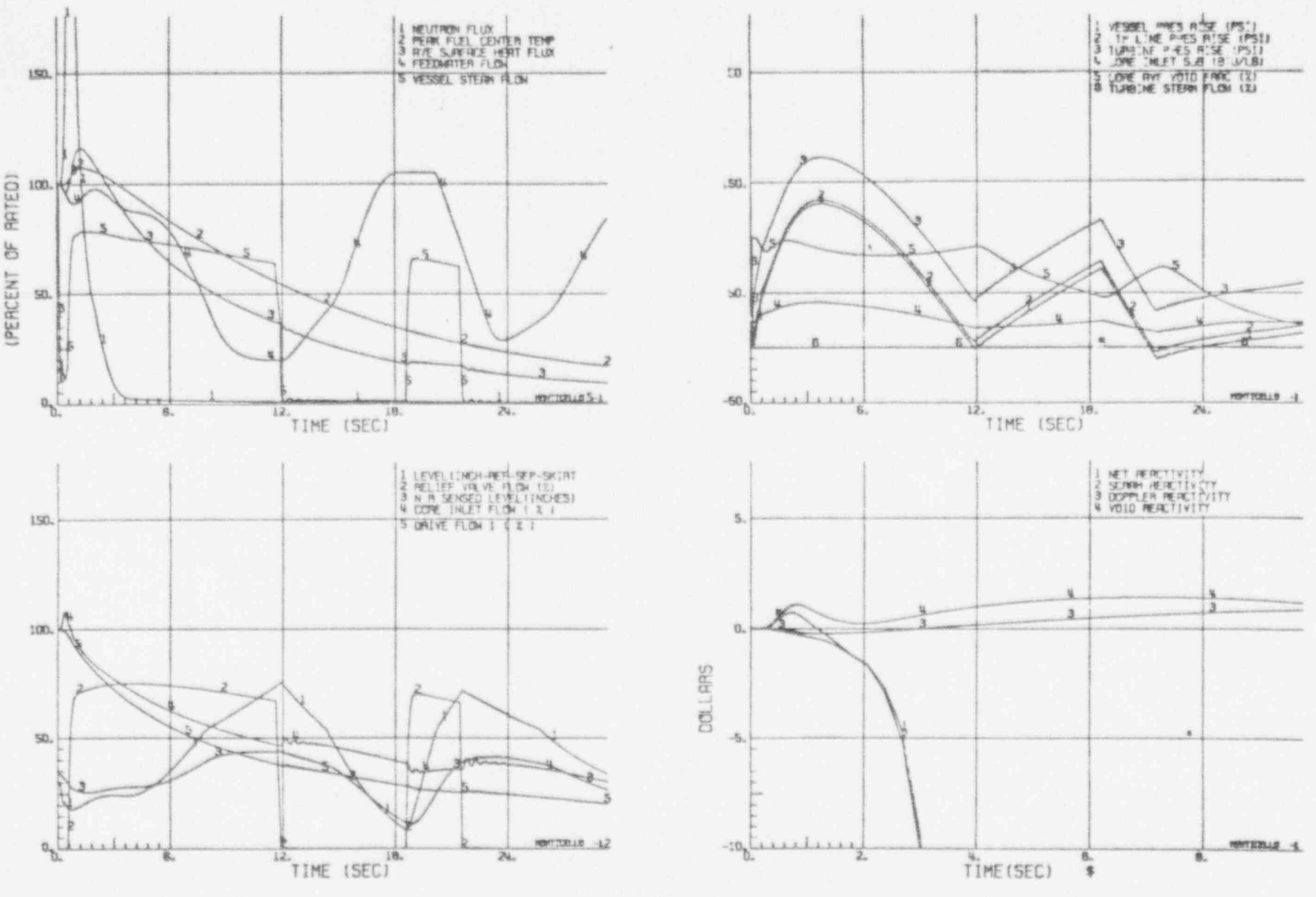


FIGURE 4. TURBINE TRIP WITHOUT BYPASS WITH PRT TWO PUMP TRIP

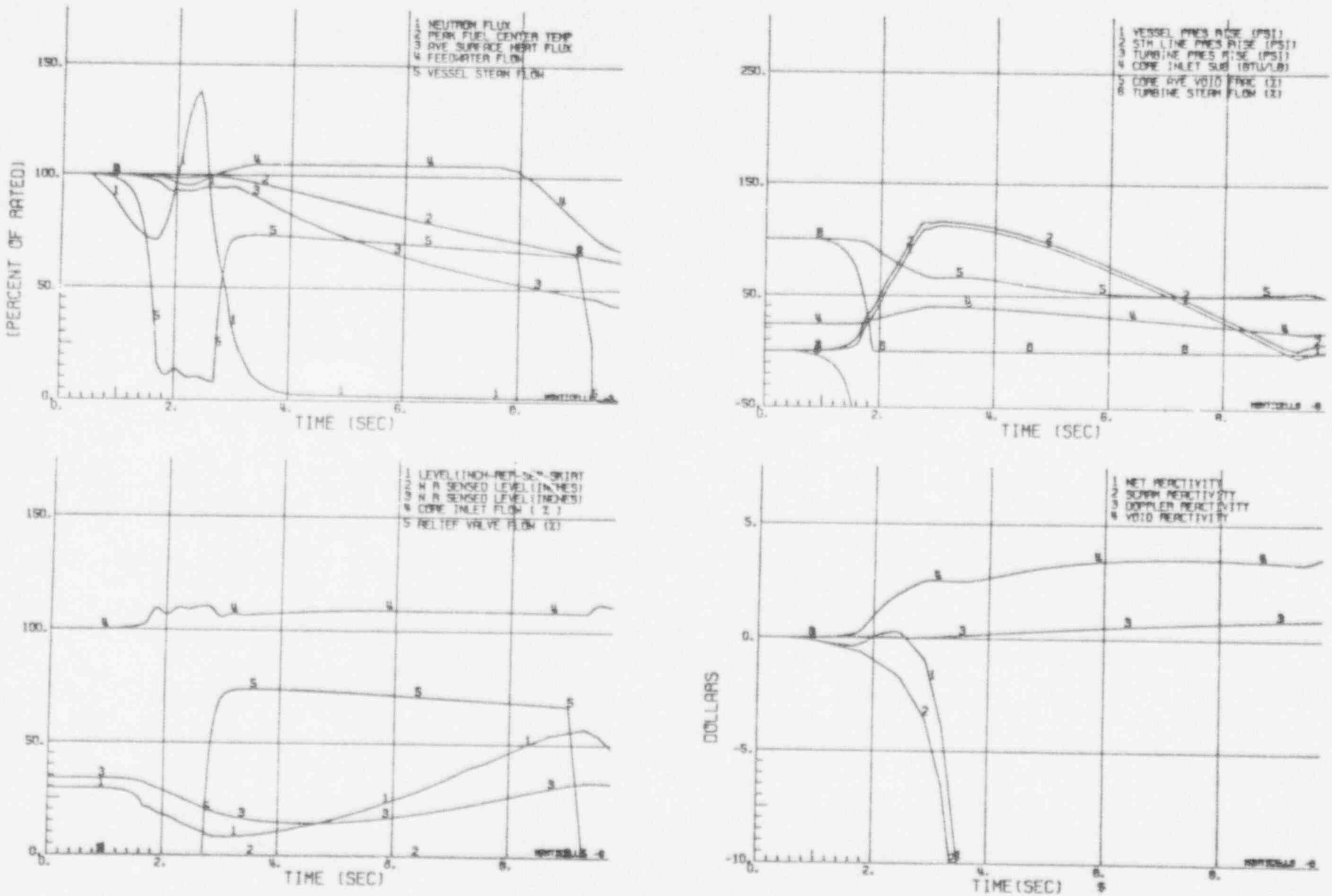


Figure 5. Main Steam Isolation Valve Closure, Trip Scram

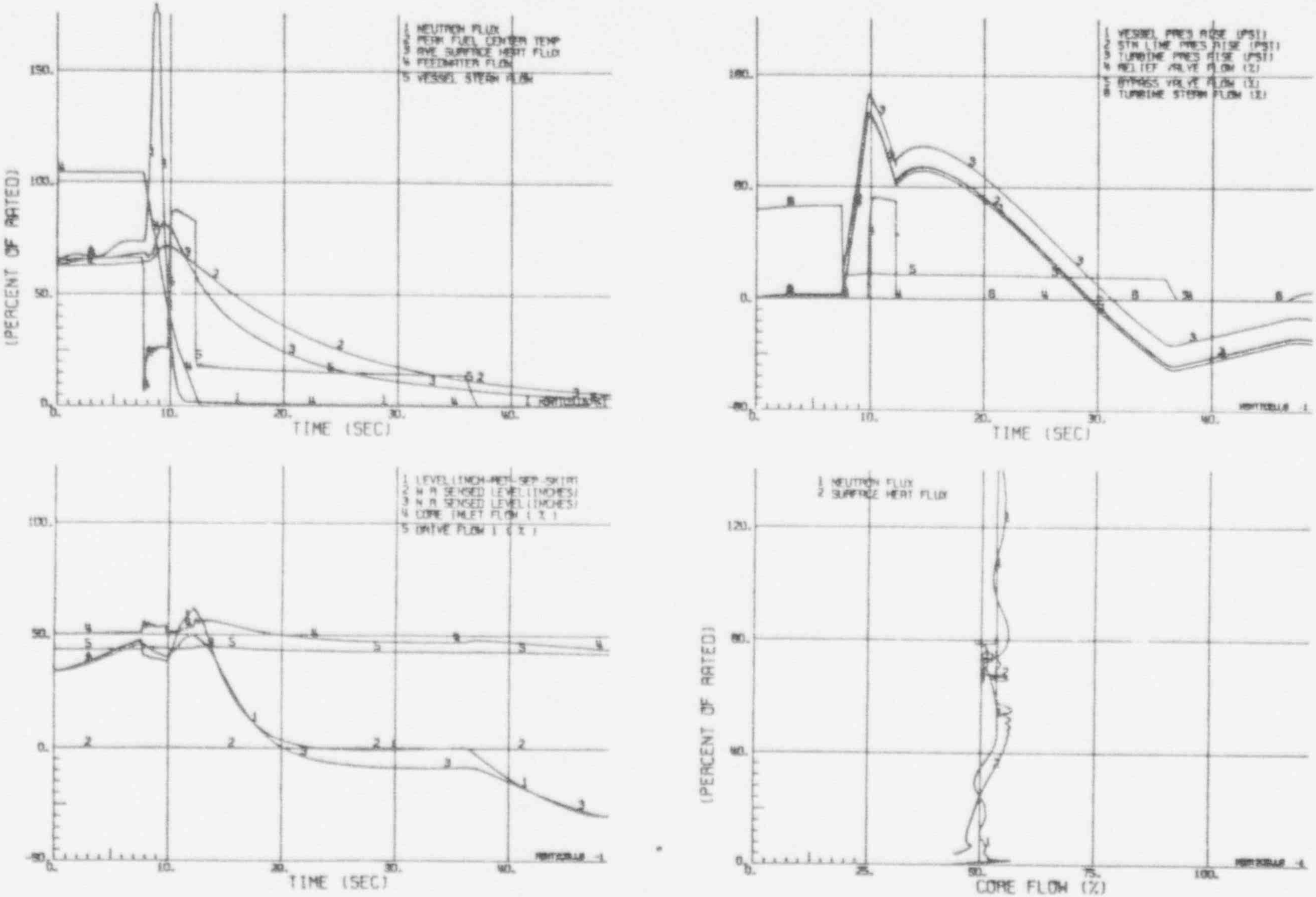


Figure 7. FW Controller Failure - Maximum Demand

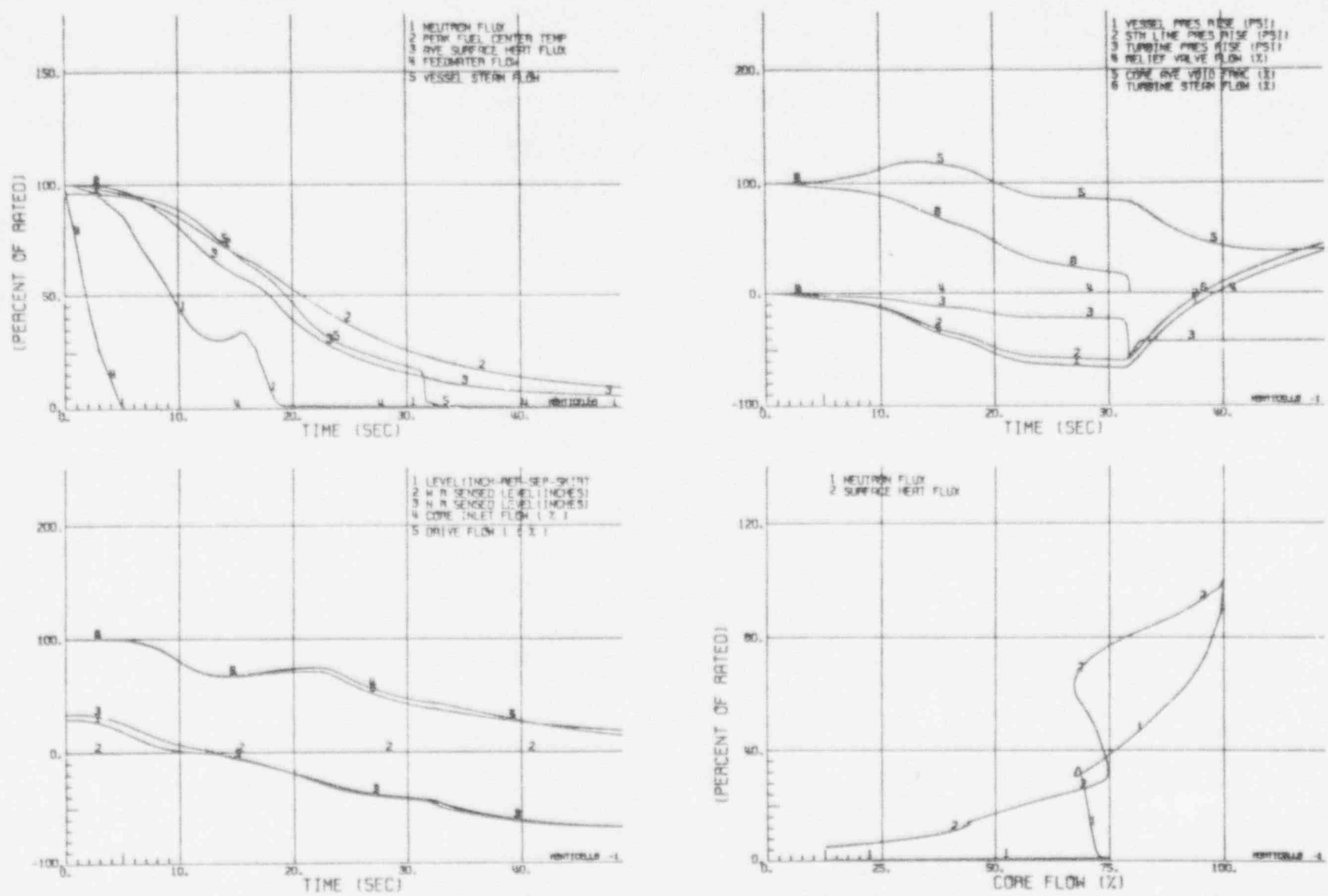


Figure 10. Loss of Feedwater

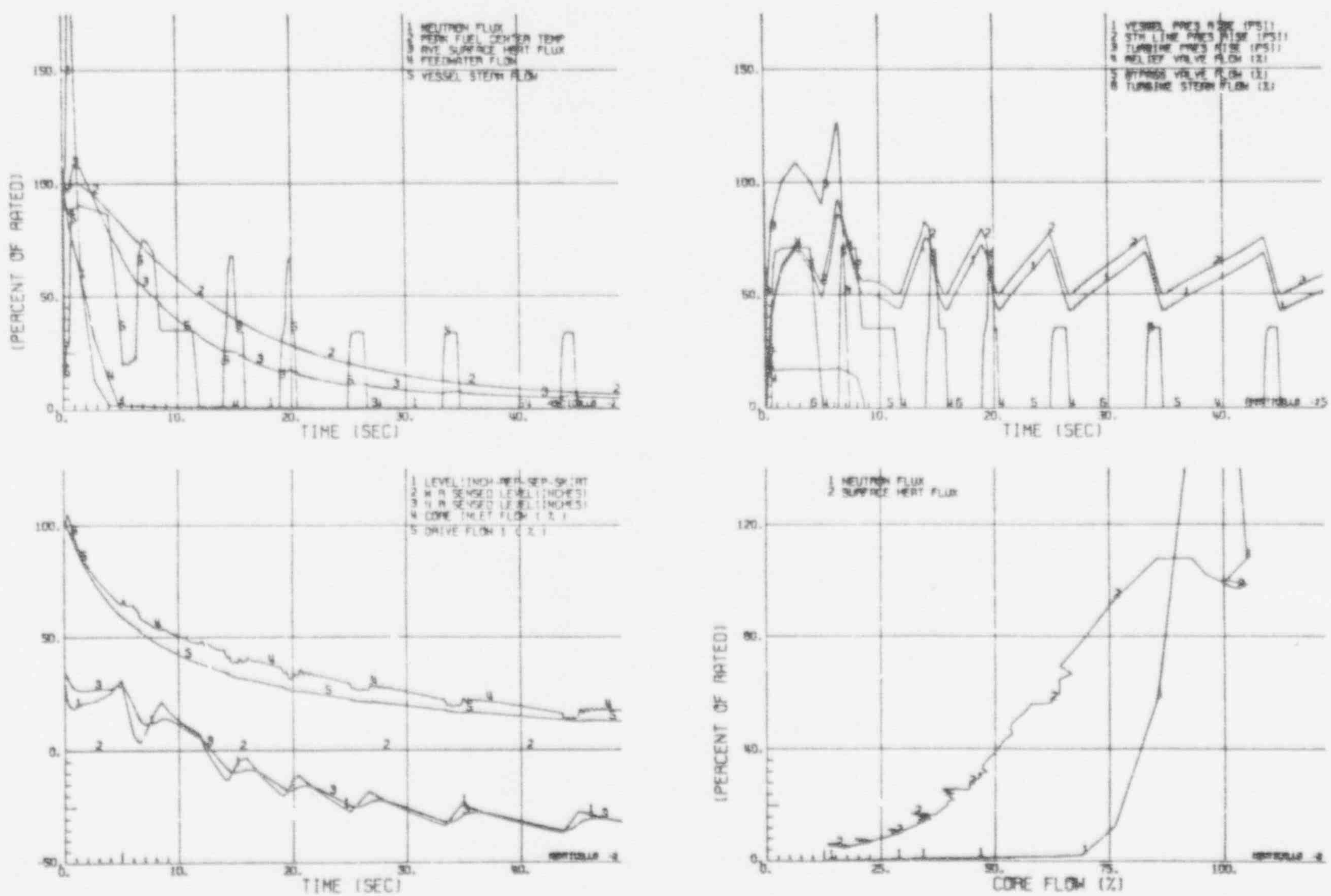


Figure 12 Loss of Auxiliary Power–Grid

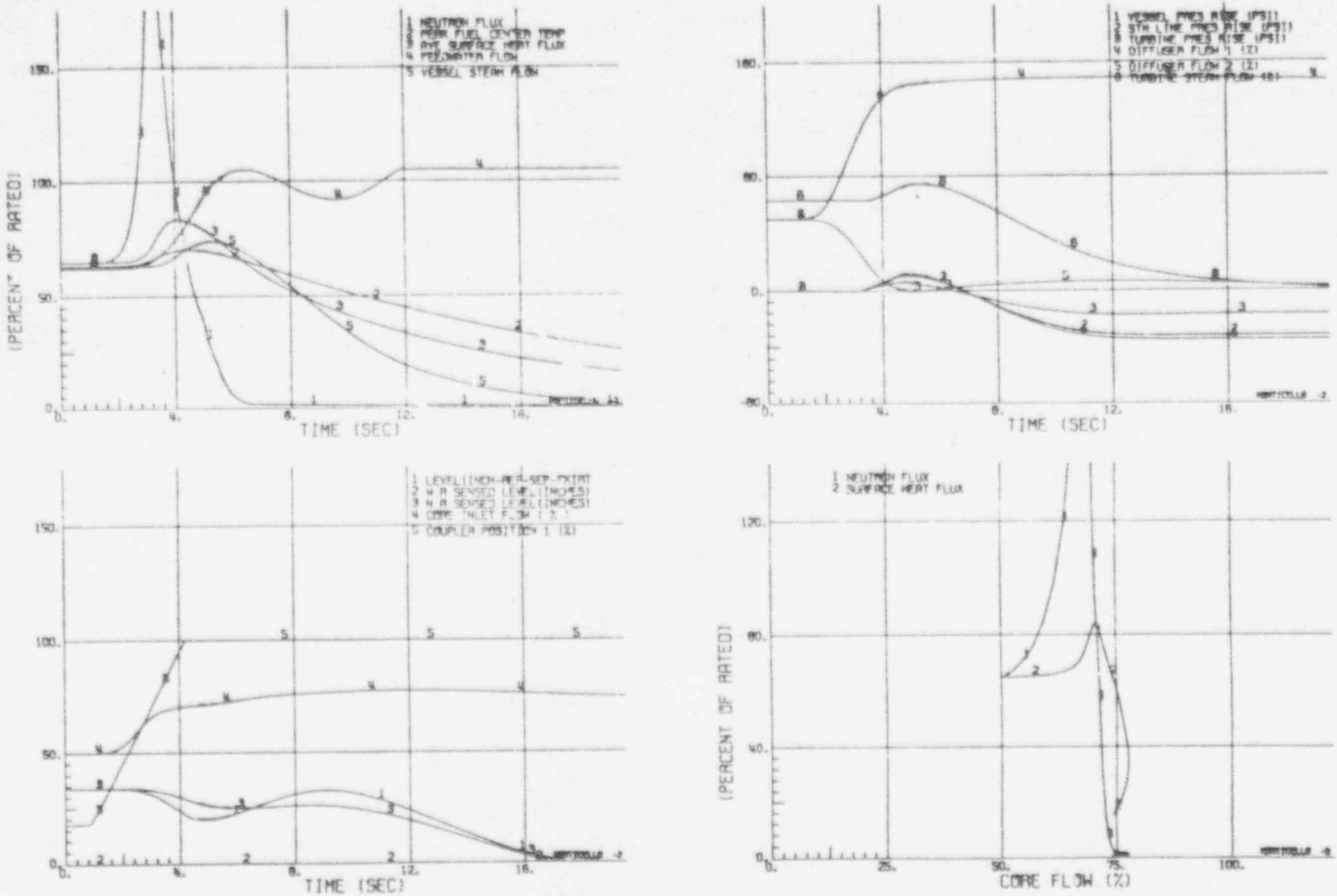


Figure 14. Recirc Speed Control Failure–Full Demand

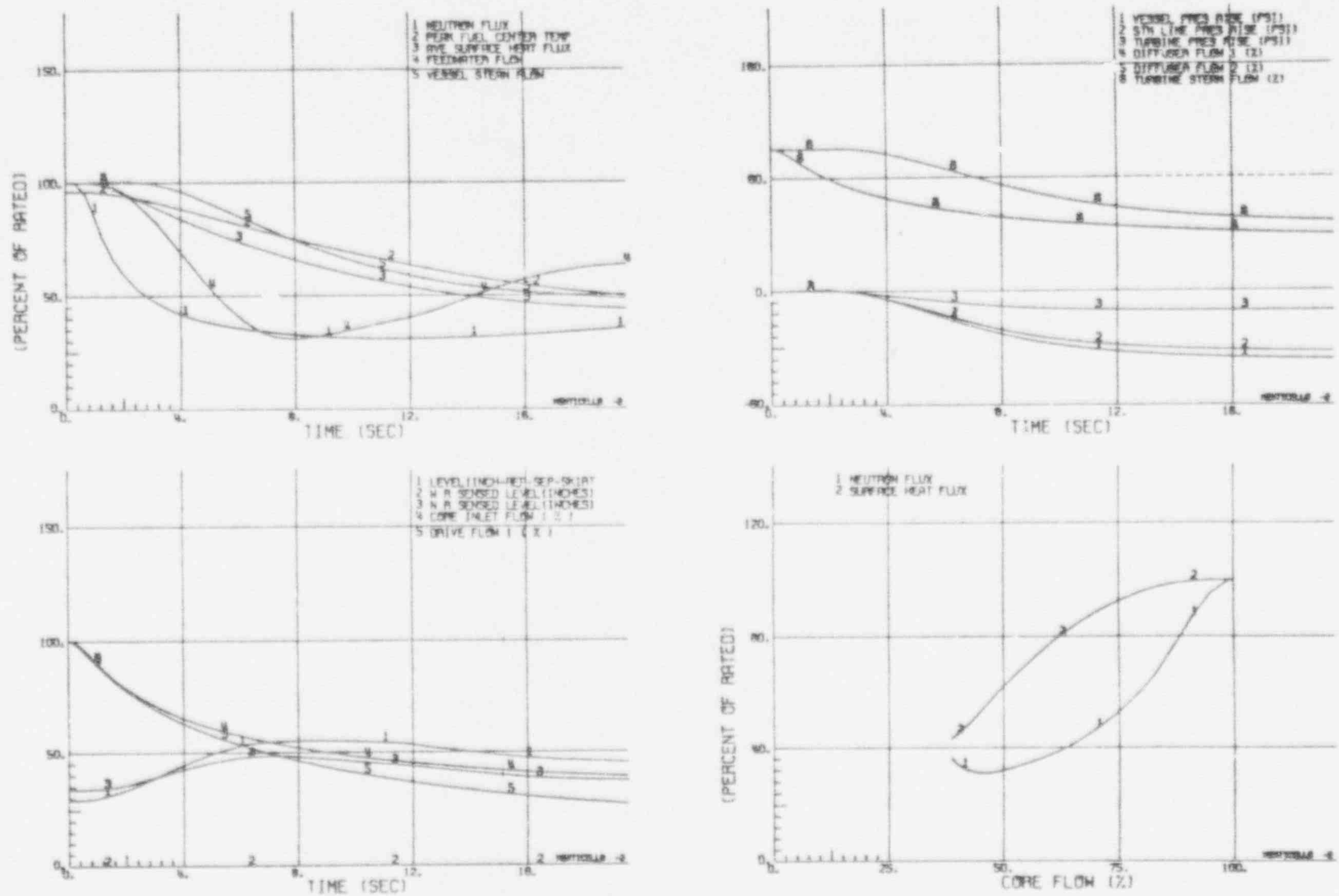


Figure 16. Trip of Two MG Breakers

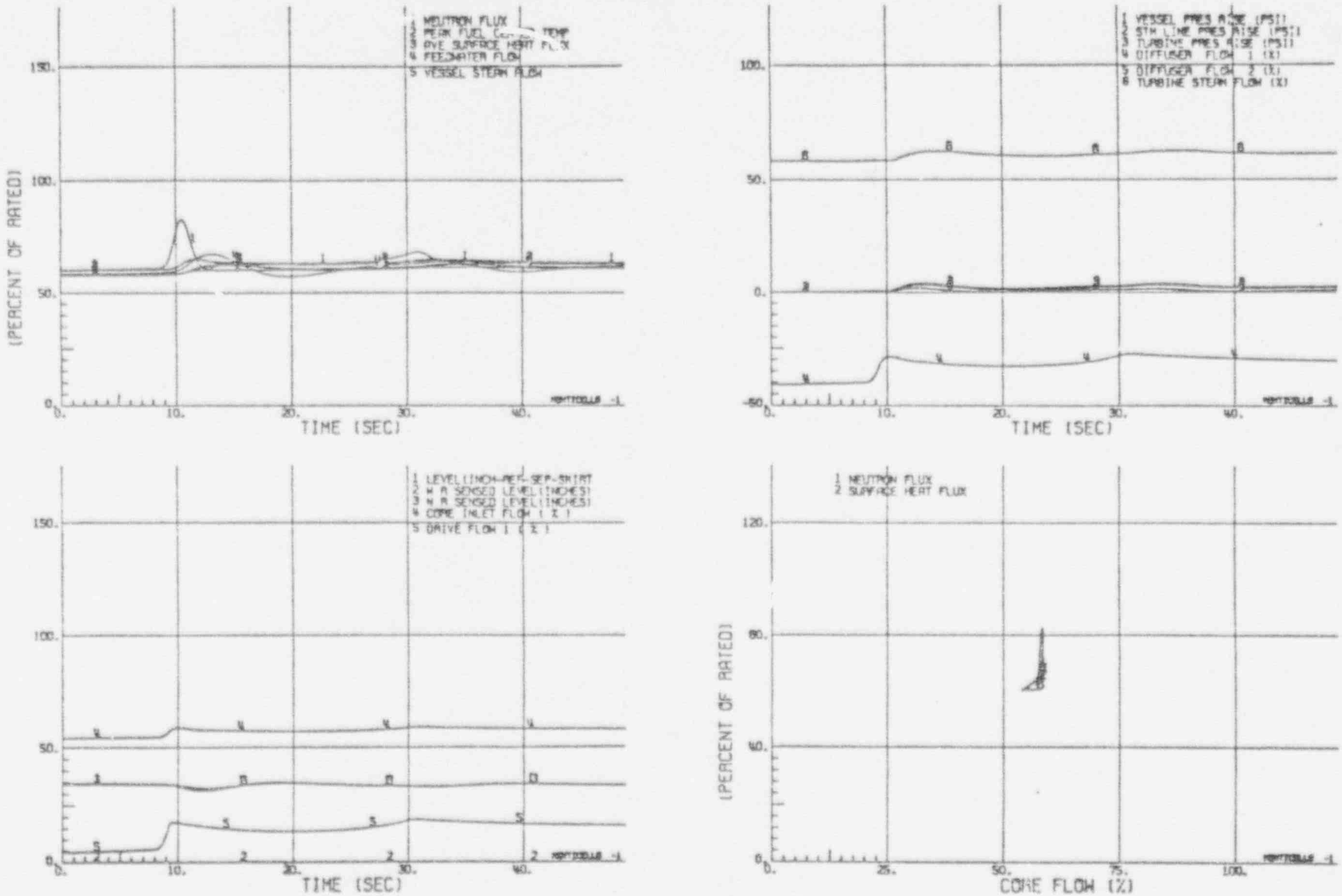


Figure 18. Cold Loop Startup

APPENDIX A

PROMPT RELIEF TRIP SYSTEM FUNCTIONAL DESCRIPTION

1.0 DEFINITION

Prompt Relief Trip (PRT) is defined as the mode of operation whereby a scheduled number of relief valves are tripped open promptly following initiation of a turbine or generator trip.

2.0 PURPOSE

The objective of PRT is to increase the effectiveness of the relief valves to compensate for the consequences of changes in the reload scram reactivity curves and thereby;

a. Avoid lifting the safety valves, if applicable, and

b. Maintain the fuel duty cycle within acceptable limits,

when subjected to the most severe single failure transients with direct scram trip. The most severe single failure transients with respect to pressure and heat flux are turbine trip without bypass and generator trip with the assumed failure of the bypass system considering both with direct scram trip (trip scram).

3.0 FUNCTIONAL DESCRIPTION AND DESIGN REQUIREMENTS

3.1 PRT Configuration

a) The PRT system configuration shall conform to the requirements of IEEE-279

b) The PRT system configuration shall define two independently adjustable power ranges.

c) An equal number of PRT coupled relief valves shall be assigned to and governed by the specifications of the power ranges.

d) An odd relief valve may be assigned to either power range according to the requirements of the plant.

3.2 PRT Enable

3.2.1 PRT Permissive Signals

The PRT enable mode of operation shall be contingent on the following permissive signals:

a) Manual reset

b) Initiation of a closure of the main turbine stop valves or the initiation of a fast closure of the turbine control valves.

c) Power level is within the adjustable PRT power range.

3.2.2 PRT Enable Mode

a) The PRT system shall be enabled and provide a trip demand signal to the corresponding relief valve solenoids if the proper channel PRT permissive signals are present.

b) The PRT trip demand signal shall be sustained until it is terminated by a disable signal.

c) The time delay from initiation of a closure of the turbine stop valves of a fast closure of the turbine control valves to the generation of a PRT trip demand signal shall not exceed 100 milliseconds.

3.2.3 PRT Permissive Power Range

a) The PRT permissive power ranges shall be defined as follows:

$$\Delta P_{R1} \geq P_{L1}$$

$$\Delta P_{R2} \geq P_{L2}$$

where ΔP_R = Power Range

P_L = Power Limit

b) The power limit shall be adjustable from 70% to the maximum operating power level.

c) The accuracy of the adjustable power setting shall be within $\pm$ 3% of full power.

3.3 PRT Disable

3.3.1 PRT Inhibit

The PRT system shall be inhibited if any of the following conditions exist:

a) Manual inhibit for test of that portion of the reactor protection system which requires closure of the turbine stop valves to perform.

b) The power level is below the permissive power range.

3.3.2 PRT Disable Mode

a) A PRT trip demand shall be terminated in response to a disable signal

b) A PRT disable signal shall not disable any other mode of operation of the relief valves (Pressure actuation, manual or automatic depressurization (ADS).

3.3.3 PRT Disable Signals

a) The source of the primary disable signal shall be an interval timer.

1) The interval timer shall be initiated coincident with the PRT trip demand signal

2) The interval timer shall be terminated after an elapsed time interval determined by an adjustable setting.

3) The range of the interval timer shall be at least 10 seconds.

4) The timer setting shall be adjustable over the full range of the timer range.

5) The timer setting shall have an accuracy of at least ± 0.5 second.

b) The source of the secondary disabling signal shall be dome pressure or a steamline pressure at a location upstream of the relief valves.

1) The adjustable range of the pressure signal shall extend from 800 to 1100 psig.

2) The accuracy of the setpoint shall be within ± 1% of pressure setpoint.

3.4 PRT Failure Mode

a) The logic design of the PRT system shall provide that a single failure shall cause the inadvertent opening of no more than one relief valve.

b) The logic design of the PRT system shall provide that a single failure shall not prevent the operation of PRT.

4.0 DETAILED DESCRIPTION OF THE PRT

The PRT consists of two identical channels. An overall block diagram is shown on Figure A-1. The sensors are generally from the Reactor Protection System with additional sensors provided for power level and reactor pressure as shown on Figure A-2. The manual controls consist of the reset and test bypass functions. Indication and annunciation are provided for the operator to indicate status. The connection to the valves is in some cases through the Automatic Despressurization System (ADS). The logic per channel is 2/2, thus the confirmation within one channel meets the criteria that a single failure will not cause actuation of the PRT. The two channels meet the criteria that a single failure will not prevent a valid requirement for PRT operation. Either channel will actuate all the relief valves that are connected to the system through separate air solenoid valves.

The block diagram of Figure A-3 gives the detail of a channel. The signal sources are shown as previously defined. The deactivating logic from a timer is backed up by reactor pressure thus providing redundancy in preventing vessel blowdowns.

Figure A-4 shows the panel arrangement indicating the barrier separating the two channels. One channel is supplied by one of the 125 V DC supplies while the other is supplied from the other 125 V DC, thus maintaining power separation and meeting single failure criteria.

The first stage pressure comes to the individual legs of the channels from independent sensors. The deactivating pressure sensors are independent and the timers are likewise independent.

Each PRT channel is connected to a separate air solenoid valve provided to actuate the Target Rock valve. The cabling from one channel is separated from the other channel to maintain separation from the air solenoid valve to the PRT cabinet. Three of the four existing Target Rock valves in Monticello are associated with the ADS system and their cabling has been separated from the HPCI system. Also these valves have test switches mounted on the control benchboards. To utilize these cables and maintain the separation for ADS and PRT the following is done:

1. One group of cables associated with one PRT channel and test switches for the additional valves will be grouped with the present air solenoid valve on present Target Rock valves and be located with the ADS. One air solenoid valve on the new Target Rock valves is connected to this group.

2. The second PRT channel with cables to the second air solenoid valve on all Target Rock valves (this is to be mounted on existing valves) is run separate from the ADS.

3. Separate electrical penetrations for the two groups of cables are required to meet IEEE-279.

PROMPT RELIEF TRIP SYSTEM
OVERALL BLOCK DIAGRAM

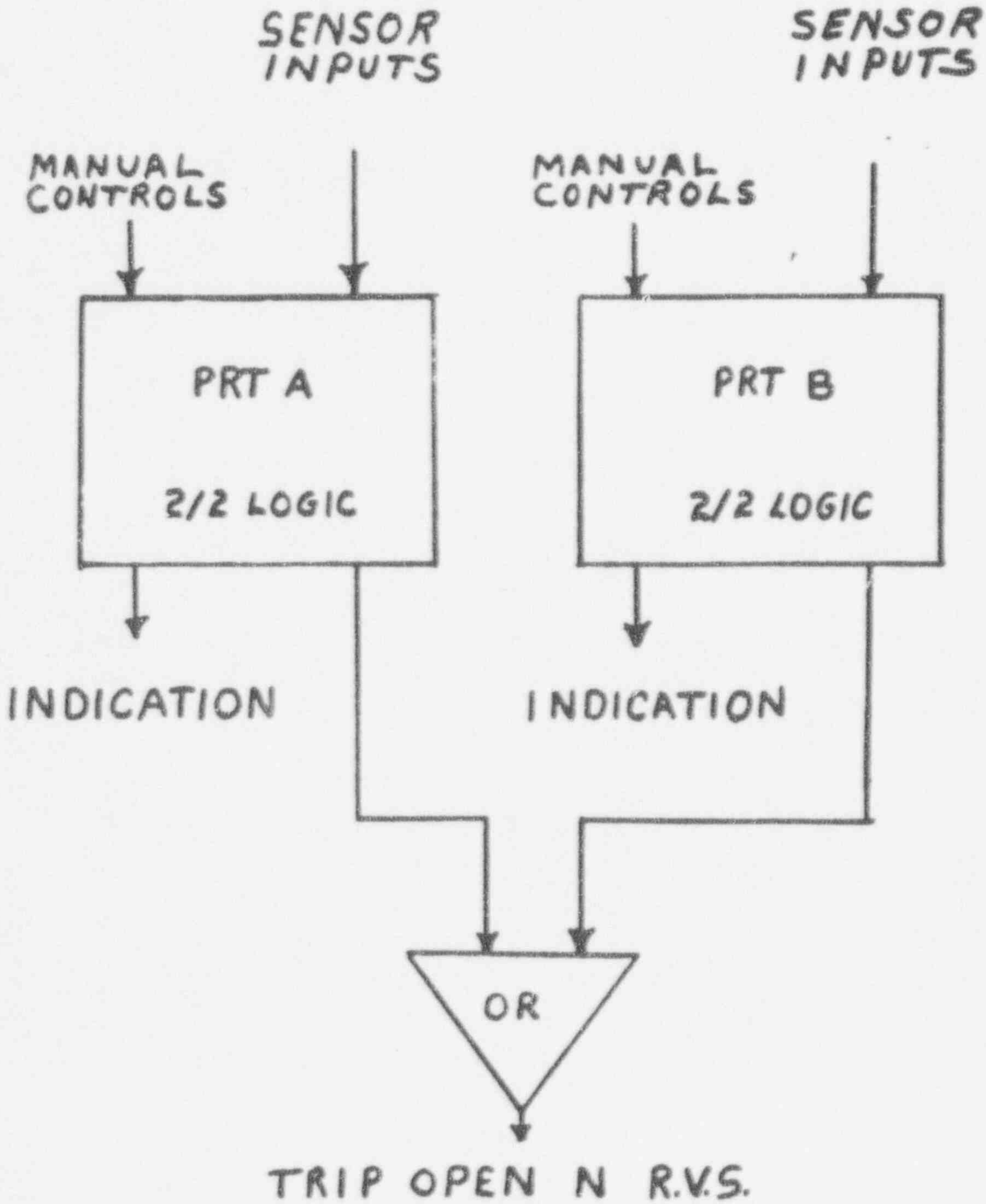
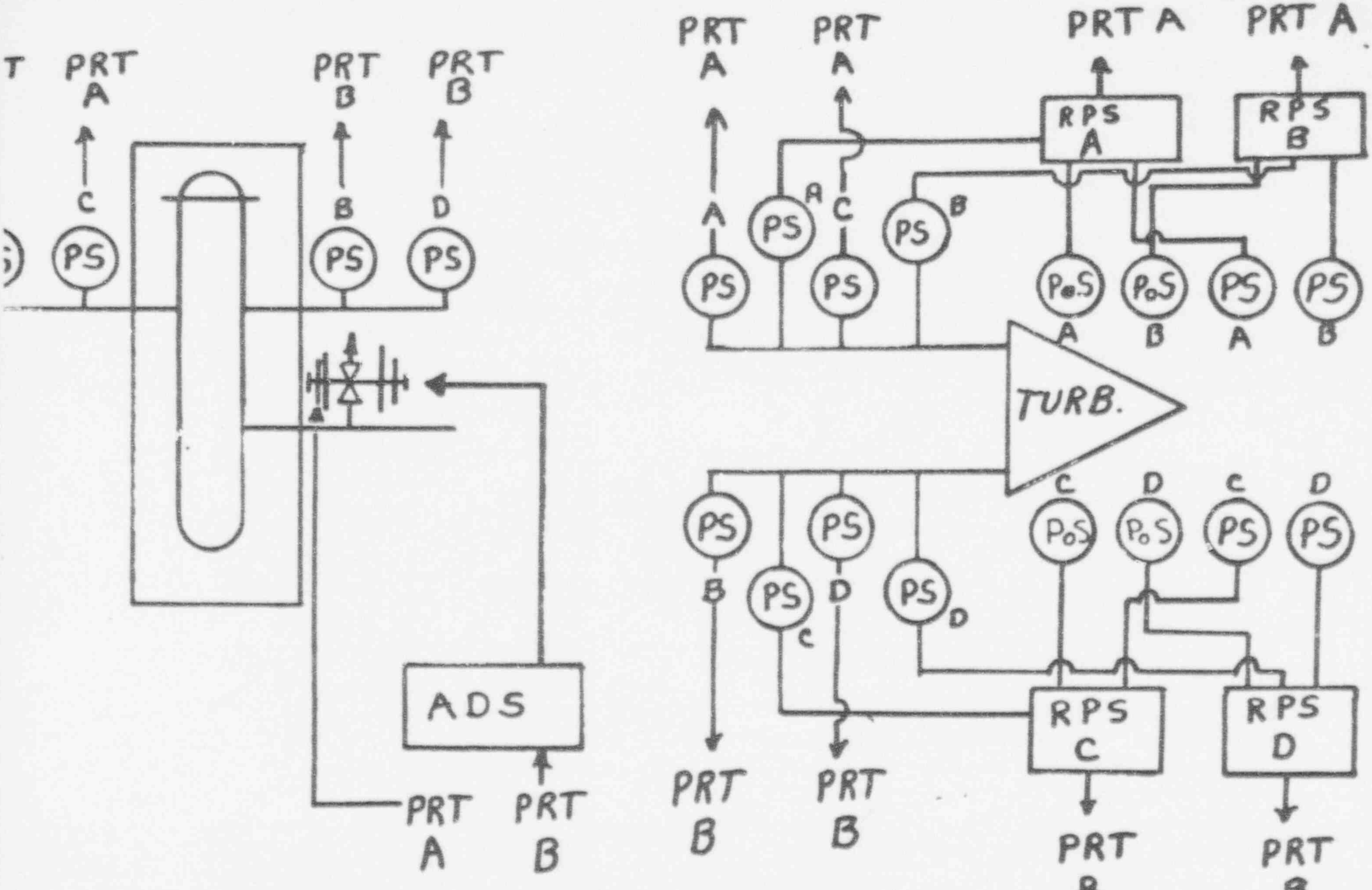
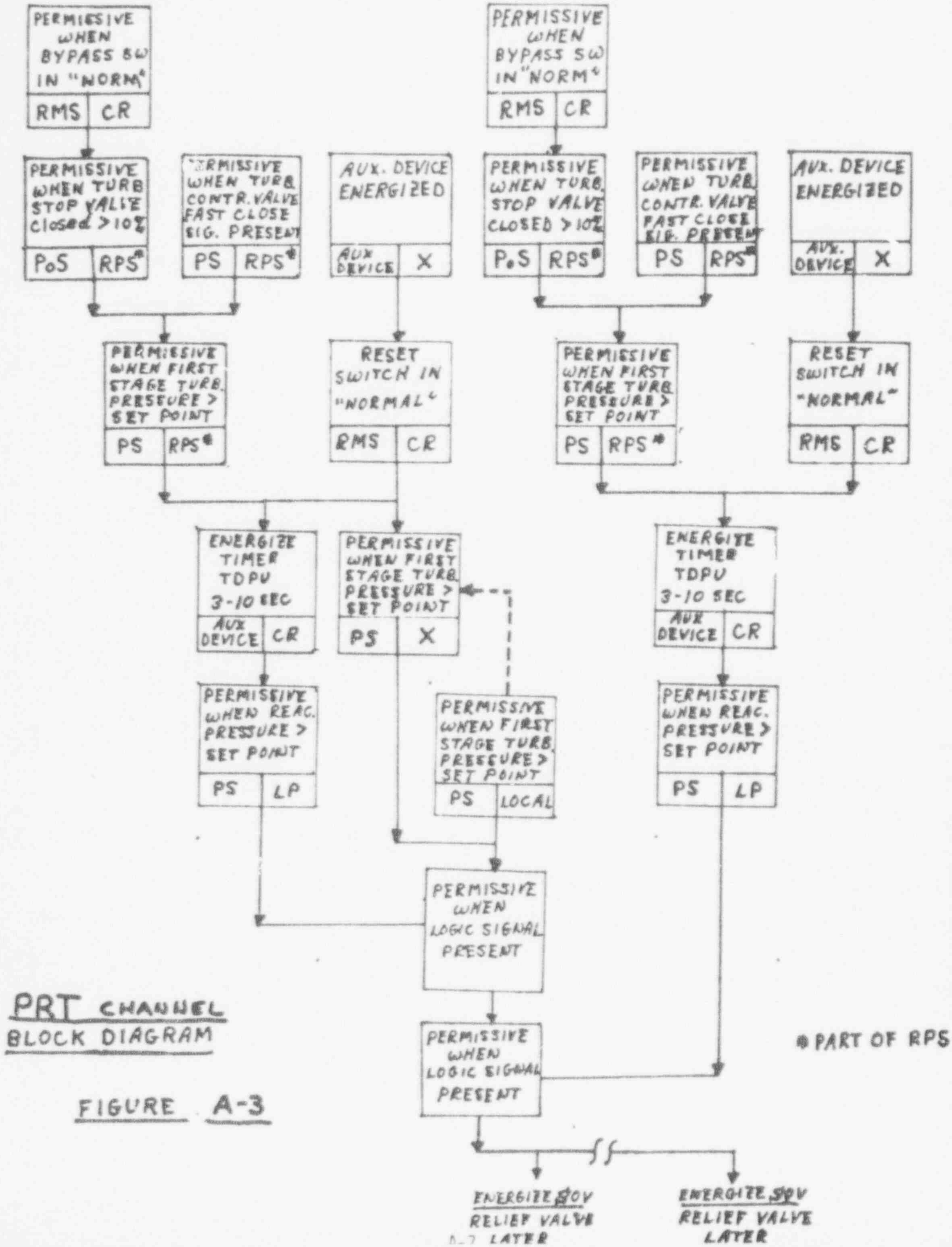


FIGURE A-1

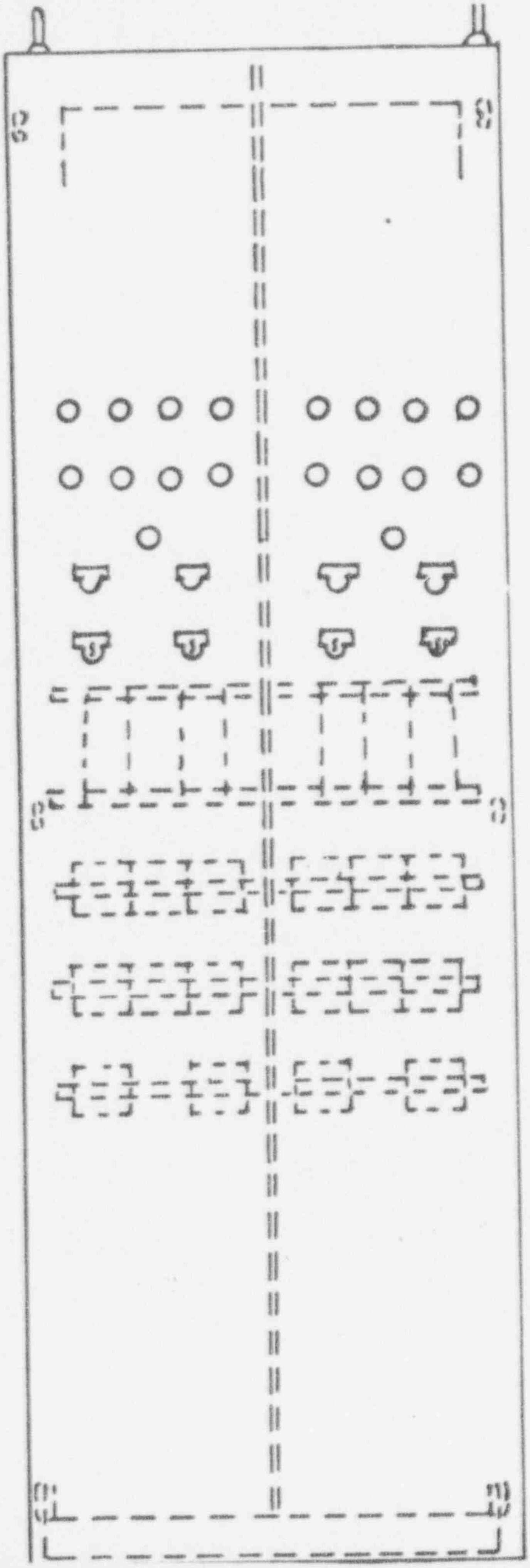


PRT SENSORS
FIGURE A-2



PRT CHANNEL BLOCK DIAGRAM

FIGURE A-3



PROMPT RELIEF
TRIP CABINET
FIGURE A-4

APPENDIX B

DESCRIPTION OF ANALYTICAL CONSERVATISM FACTORS

This submittal provides the analytical justification for licensing the proposed plant modifications in that the effects of abnormal operational transients occurring after the modifications lie within a boundry limit based on MCHFR 1.0 and peak transient pressure less than the ASME code limit, given adequate Operational Conservatism Factors (OCF) to allow for analytical uncertainties

Because the proposed modifications are long term (greater than one fuel cycle), Design Conservatism Factors (DCF) have been applied to add margin for possible future variations that might otherwise require additional major modifications

The DCF reflect allowances for uncertainties, and are generally applied to analyses of a long term nature prior to putting a change into effect. The analytical results of DCF are a pessimistic estimate of the actual consequences that could occur. The limiting boundary for operation is established through the application of Operational Conservatism Factors in the analyses that yield analytical transient effects more conservative than has been observed during BWR startup tests.

The key input parameters of the abnormal operational transient analyses are the scram negative reactivity insertion function, void coefficient and Doppler coefficient. For long term design analyses, these factors are multiplied (or "discounted") by 0.80, 1.25 and 0.90 respectively. By doing so, a considerable margin below the bounding limit is ensured.

For subsequent analyses, such as those performed for a reload license, an evaluation is made to determine whether the new input parameters compare favorably with the previously used inputs.

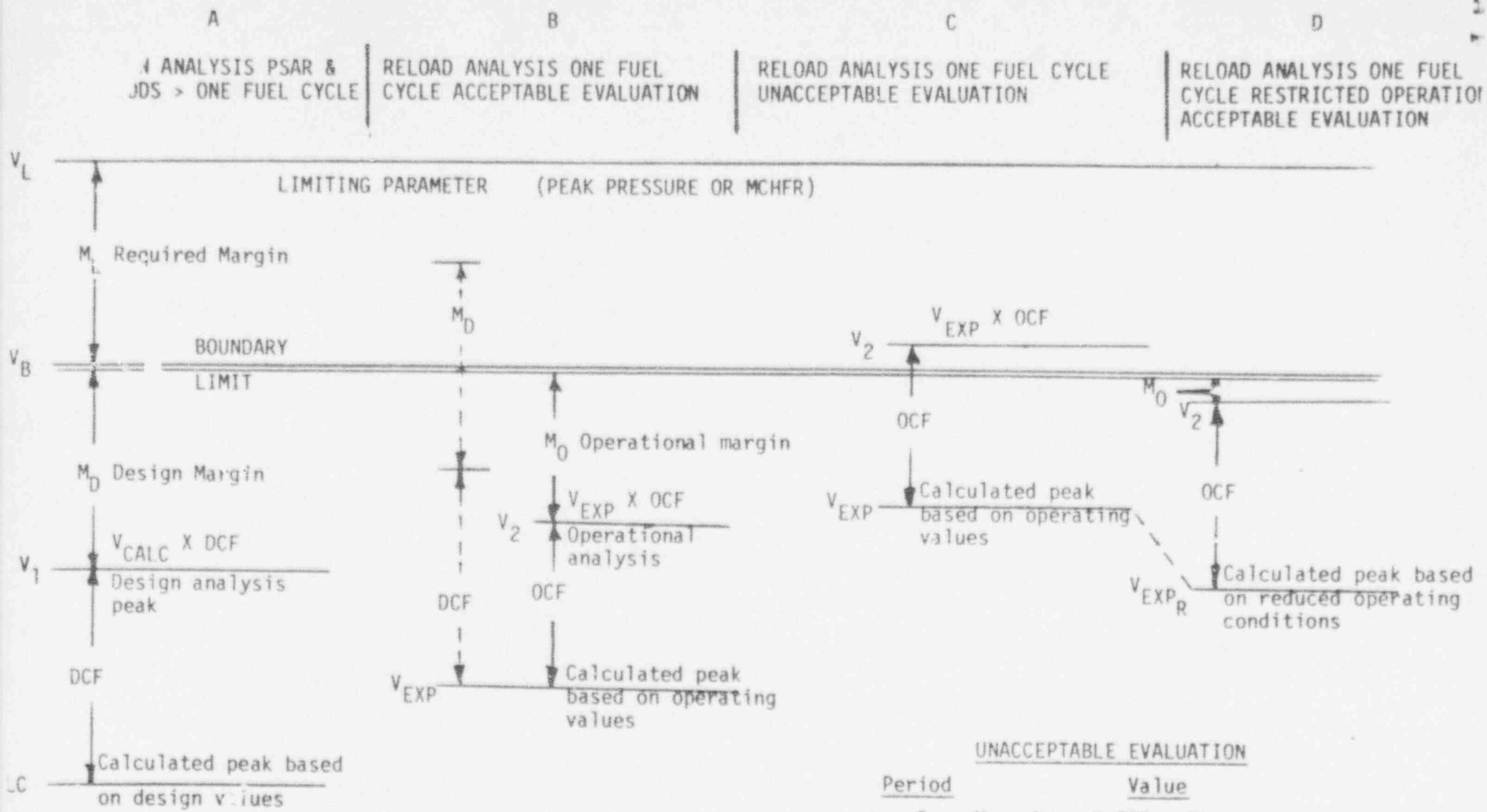
Because the knowledge of actual plant performance is well established for reloads and the parametric variations can be more accurately calculated, the need for reanalysis of transient events will be totally eliminated in most cases. Where they are not totally eliminated, analyses may be performed using Operational Conservatism Factors where the multipliers on the scram reactivity insertion function, void coefficient and Doppler coefficient are 0.95, 1.15 and 0.95 respectively. Key analyses, (e.g., RV sizing) can be made with OCF and, upon determination that the required margins are satisfied, the conclusion drawn that lesser events will not be limiting.

Figure B-1 shows the progression discussed above. In this diagram, V_L represents the limiting parameter (peak transient pressure of MCHFR); V_B represents the required margin (M_L) that establishes the boundry for the events considered. Early in licensing of the plant, (and at times later, when long term changes are contemplated), design values are used and an anticipated peak value for transient events can be calculated (V_{calc}); however, because uncertainties exist, a large conservative element (DCF) is applied to the input parameters and an additional design margin (M_D) recommended resulting in an analyzed peak transient value (V_1) that is far below the boundary value (V_B). This ensures the creation of a considerable allowance for future changes to equipment or analytical methods, as well as coverage for uncertainties.

Later analyses, e.g., reload licenses, follow a period of plant testing and operation during which many of the uncertanties that warranted the design margin (M_D) and DFC are eliminated entirely or their values can be more accurately determined. This permits a more realistic, operationally oriented analysis where-in the boundary (V_B) is the limiting value and Operational Conservatism Factors (OCF) provide adequate conservatism for uncertainties. In these situations, an expected value (V_{exp}) can be calculated that, when modified by the OCF, results in a peak transient value (V_2) that must be less than the boundary value (V_B). The resultant operational margin (M_O), although not required, provides additional conservatism.

Should V_2 exceed V_B, additional measures, such as operating restrictions or plant modifications would be required. Following such action, a new, lower, expected value (V_{exp_R}) is derived and modified by OCF to ensure the peak transient value (V_2) is below the boundary (V_B) with sufficient allowance for operational uncertainties.

Operation of the plant over an extended period and the occurrence of events similar to the analyzed transients will provide the confirming data to permit future analyes based on the actual values of the input parameters. Present experience shows that for many analyzed transients, sufficient conservatism exists in the transient analytical model to allow for uncertainties without the application of OCF. However, until a larger bank of data and experience is gained, the more conservative OCF approach will be used.



UNACCEPTABLE EVALUATION

Period	Value
C	$V_2 = V_{EXP} \times OCF > V_B$

RESTRICTED OPERATION

Period	Value
D	$V_2 = V_{EXP_R} \times OCF \leq (V_L - M_L) = V_B$

MEASURE OF ACCEPTABILITY

Period	Value
A	$V_1 = V_{CALC} \times DCF \leq V_L - (M_L + M_D)$
B	$V_2 = V_{EXP} \times OCF \leq (V_L - M_L) = V_B$

FIGURE B-1