2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is within the restricted region the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2. If the actual-reactor-thermal-power/reactor-power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

The safety limits presented have been generated using BAW-2 critical heat flux (CHF) correlation [1] and the actual measured flow rate[2]. This development is discussed in the Rancho Seco Unit 1, Cycle 2 Reload Report, reference [2]. The flow rate utilized is 104.9 percent of the design flow (369600 gpm) based on four-pump operation. [2,3]

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling region of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling region is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure

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can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 104.9 percent of 369,600 gpm). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and rod bowing.

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.4 KW/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which a minimum DNBR of 1.30 is predicted
at the maximum possible thermal power for the assumed design flow, or the local 70
quality at the point of minimum DNBR is equal to 15 percent, whichever condition is more restrictive.

For Figure 2.1-3, a pressure-temperature point above and to the left of the 70
curve would result in a DNBR greater than 1.30. The 1.30 DNBR curve for four-

pump operation is more restrictive than any other reactor coolant pump situation because any pressure/termperature point above and to the left of the four-pump curve will be above and to the left of the other curves.

The maximum thermal power for three-pump operation depicted in Figure 2.1-2 is 86.85 percent due to a power level trip produced by the flux-flow ratio 1.08 times 74.4 percent design flow = 80.35 percent power plus the maximum calibration and instrumentation error. The maximum thermal power for other coolant pump conditions is produced in a similar manner. The actual maximum power levels are calculated by the RPS and will be directly proportional to the actual flow during partial pump operation. 70

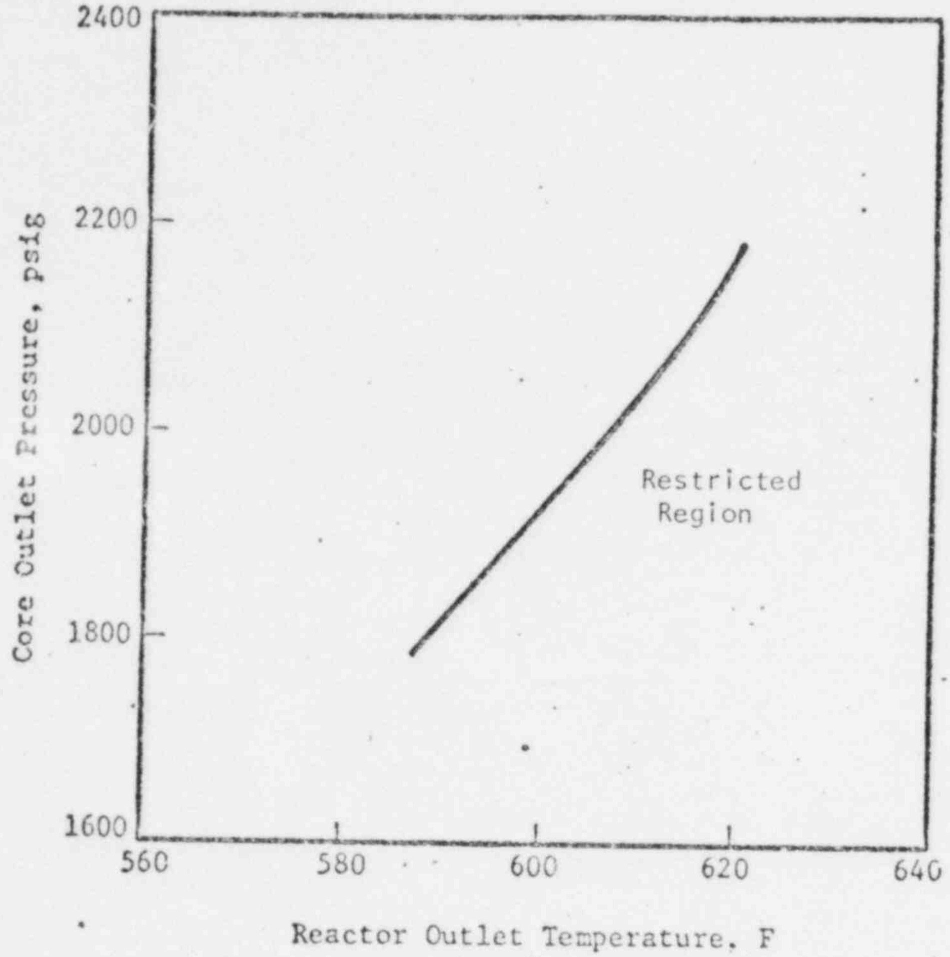
References

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March, 1970.

(2) Rancho Seco Unit 1, Cycle 2 Reload Report BAW.

(3) Rancho Seco Unit 1, Cycle 3 Reload Report BAW-1499, September, 1978.

Figure 2.1-1. Core Protection Safety Limit, Pressure Vs Temperature.



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Proposed Amendment No. 70

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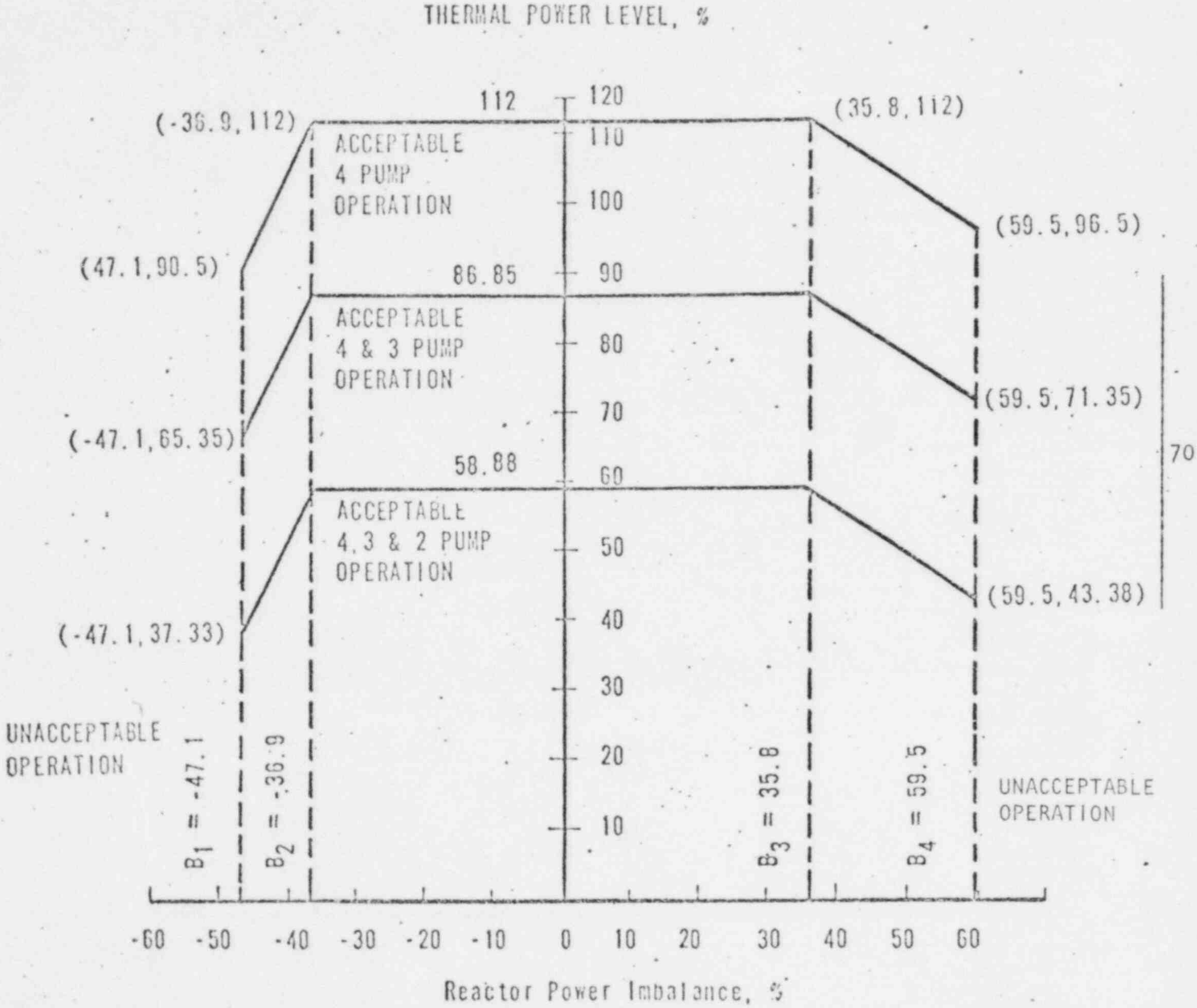
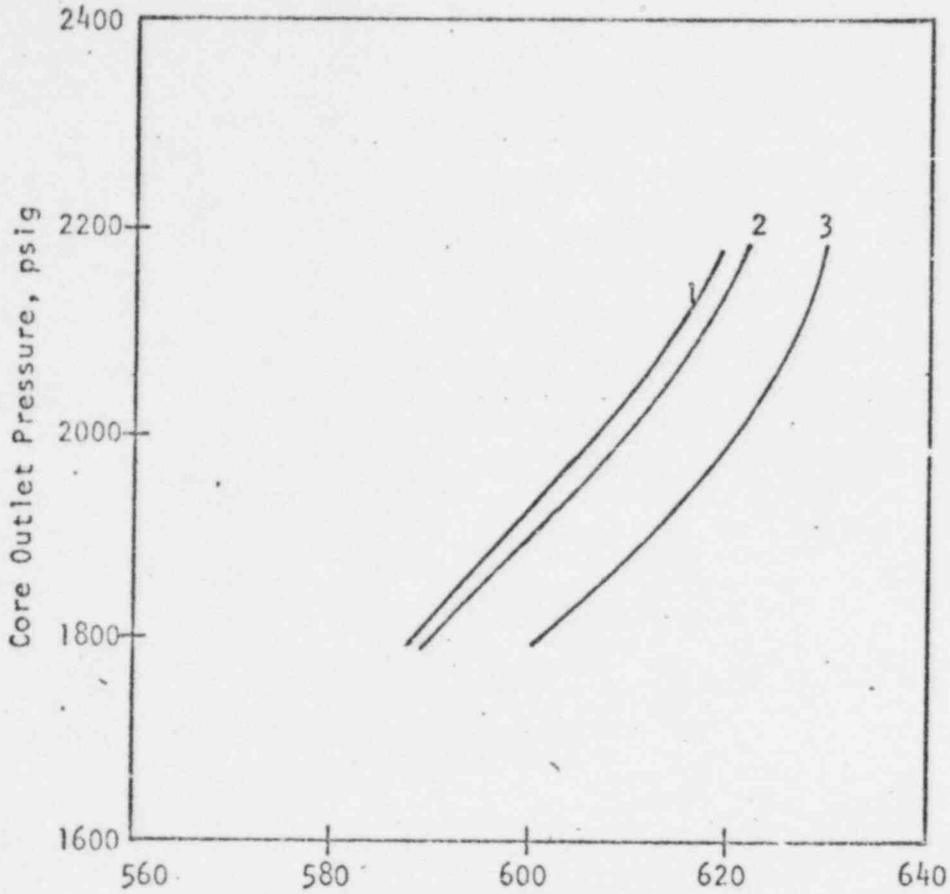
Figure 2.1-2 CORE PROTECTION SAFETY LIMITS, REACTOR POWER IMBALANCE (CYCLE 4)



CURVE	REACTOR COOLANT DESIGN FLOW, GPM
1	387,600
2	288,374
3	187,986

FIGURE 2.1-3. Core Protective Safety Bases



Curve	Reactor coolant flow, gpm	Power,%	Pumps operating (type of limit)
1	387600	112	Four (DNBR Limit)
2	288374	86.85	Three (DNBR Limit)
3	187986	58.88	One in each loop (quality limit)

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Proposed Change No. 70

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high Reactor Building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in table 2.3-1 and figure 2.3-2.

Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5 percent of rated power. Adding to this the possible variation in trip set points due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112 percent, which was used in the safety analysis.(4)

A. Overpower trip based on flow and imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. The analysis in section 14 demonstrates the adequacy of the specified power-to-flow ratio.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power to flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of table 2.3-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108 percent and reactor flow rate is 100 percent, or flow rate is 92.59 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating
if power is 80.35 percent and reactor flow rate is 74.4 percent 70
or flow rate is 69.44 percent and power level is 75 percent.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.38 percent and reactor flow rate is 48.5 percent or flow rate is 45.37 percent and the power level is 49 percent.

For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of figure 2.3-2 are produced. The power-to-flow ratio reduces the power level
trip and associated reactor-power reactor-power-imbalance boundaries by 1.08 70
percent for a 1 percent flow reduction.

B. Pump monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to (a) the loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss one one or two reactor coolant pumps during two-pump operation. The pump monitors also restrict the power level to 55 percent for one reactor coolant pump operation in each loop.

C. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip set point. The trip setting limit shown in figure 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1900 psig) and variable low pressure ($12.96\ T_{out} - 5834$) trip set point shown in figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($12.96\ T_{out} - 5884$).

D. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620 F.

E. Reactor Building pressure

The high Reactor Building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the Reactor Building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in table 2.3.1. Two conditio are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1820 psig is automatically imposed.

The purpose of the 1820 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of $\leq$5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

REFERENCES

(1) FSAR, paragraph 14.1.2.2.

(2) FSAR, paragraph 14.1.2.7

(3) FSAR, paragraph 14.1.2.8

(4) FSAR, paragraph 14.1.2.3

(5) FSAR, paragraph 14.1.2.6

TABLE 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear power, % of rated, max.	105.5	105.5	105.5	5.0(3)
2. Nuclear power based on flow(2) and Imbalance, % of rated, max.	1.08 times flow minus reduction due to Imbalance(s)	1.08 times flow minus reduction due to Imbalance(s)	1.08 times flow minus reduction due to Imbalance(s)	Bypassed
3. Nuclear power based on pump monitors, % of rated, max.	NA	NA	55	Bypassed
4. High reactor coolant system pressure, psig. max.	2355	2355	2355	1820(4)
5. Low reactor coolant system pressure, psig. min.	1900	1900	1900	Bypassed
6. Variable low reactor coolant system pressure, psig, min.	12.96 T_{out} - 5834	12.96 T_{out} - 5834	12.96 T_{out} - 5834	Bypassed
7. Reactor coolant temp. F., max.	619	619	619	619
8. High Reactor Building pressure, psig, max.	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (F).

(2) Reactor coolant system flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS (as specified) are bypassed.

(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.

Figure 2.3-1. Protective System Maximum Allowable Setpoints, Pressure Vs Temperature

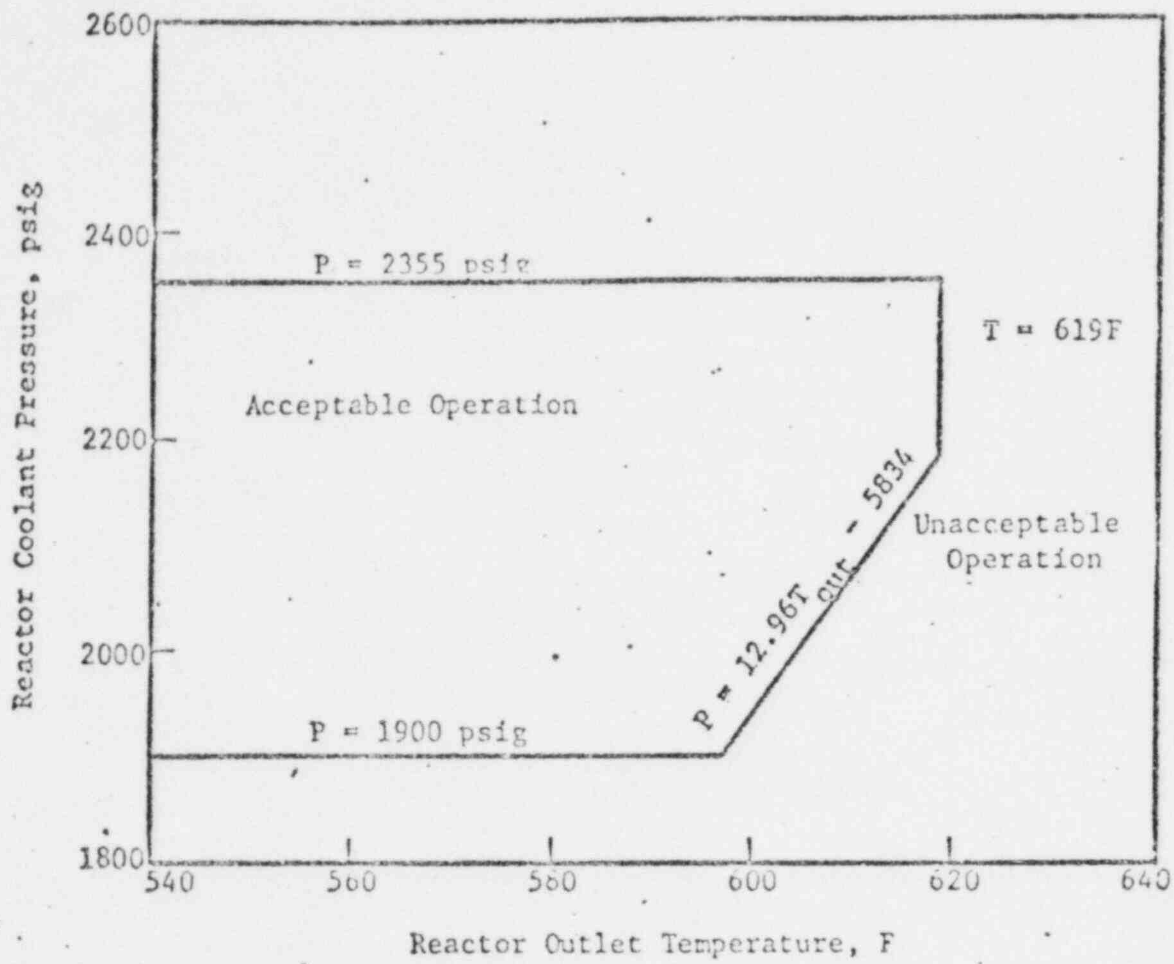
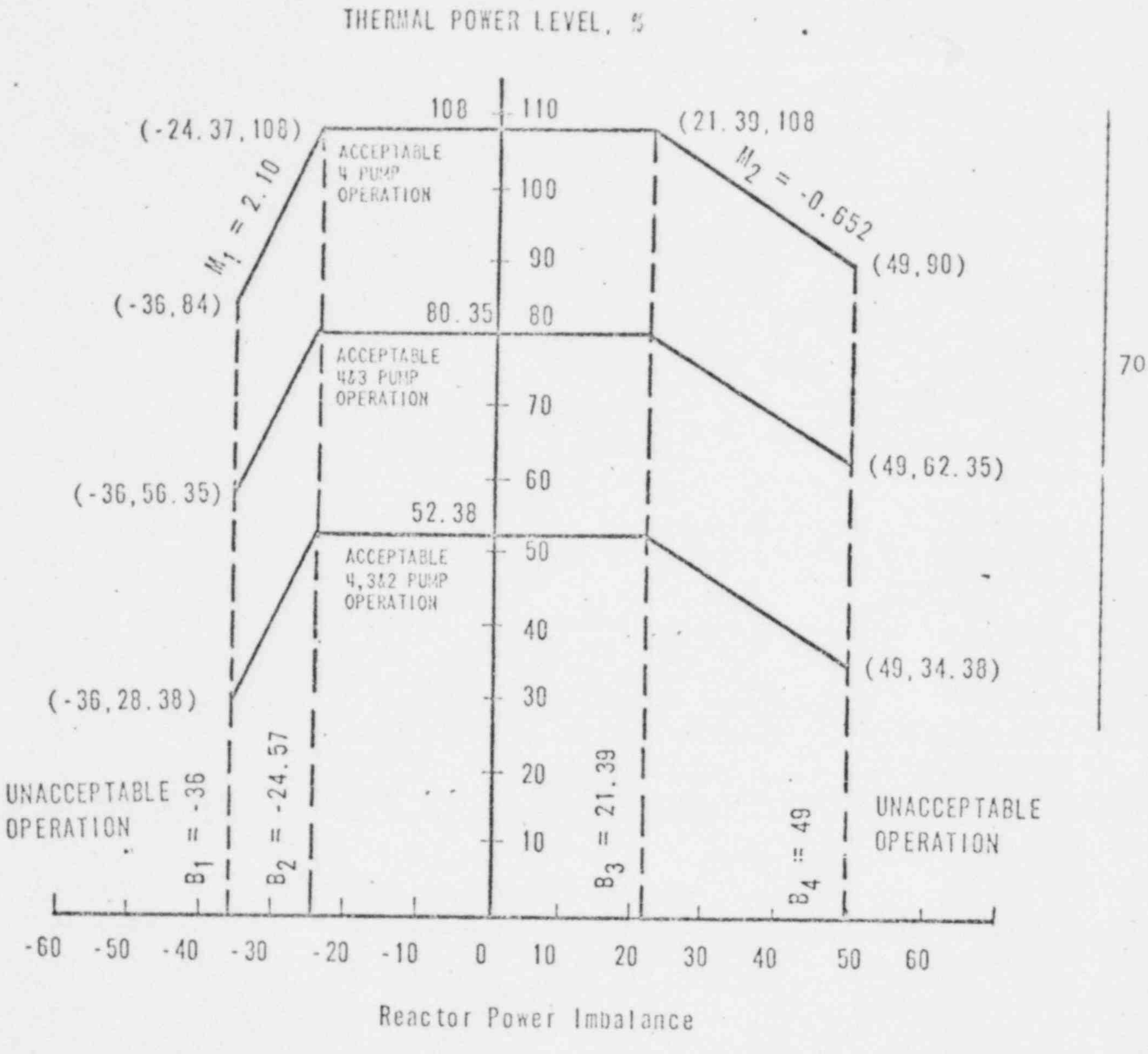


Figure 2.3-2 PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS, REACTOR POWER IMBALANCE (CYCLE 4)



CURVE	REACTOR COOLANT DESIGN FLOW, GPM
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