UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT NO.1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated April 29, 1982, as supplemented May 10, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 67, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 30, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 67

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

9

9b

11

12

13

14

14a

14b

15

Using a local quality limit of 22 percent at the point of minimum DNBR as a basis for curve 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup-dependent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis. All plant operating limits are presently based on an original method (3) of calculating rod bowing penalties that are more conservative than those that would be obtained with new approved procedures (4). For the current cycle of operation, this subrogation results in a DNBR margin in excess of 3.8%, which is partially used to offset the reduction in DNBR due to fuel rod bowing.

The maximum thermal power for three-pump operation is 88.92 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.054 = 78.73 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. Curves 1 & 2 of Figure 2.1-3 are the most restrictive because any pressure/temperature point above and to the left of this curve will be above and to the left of the other curve.

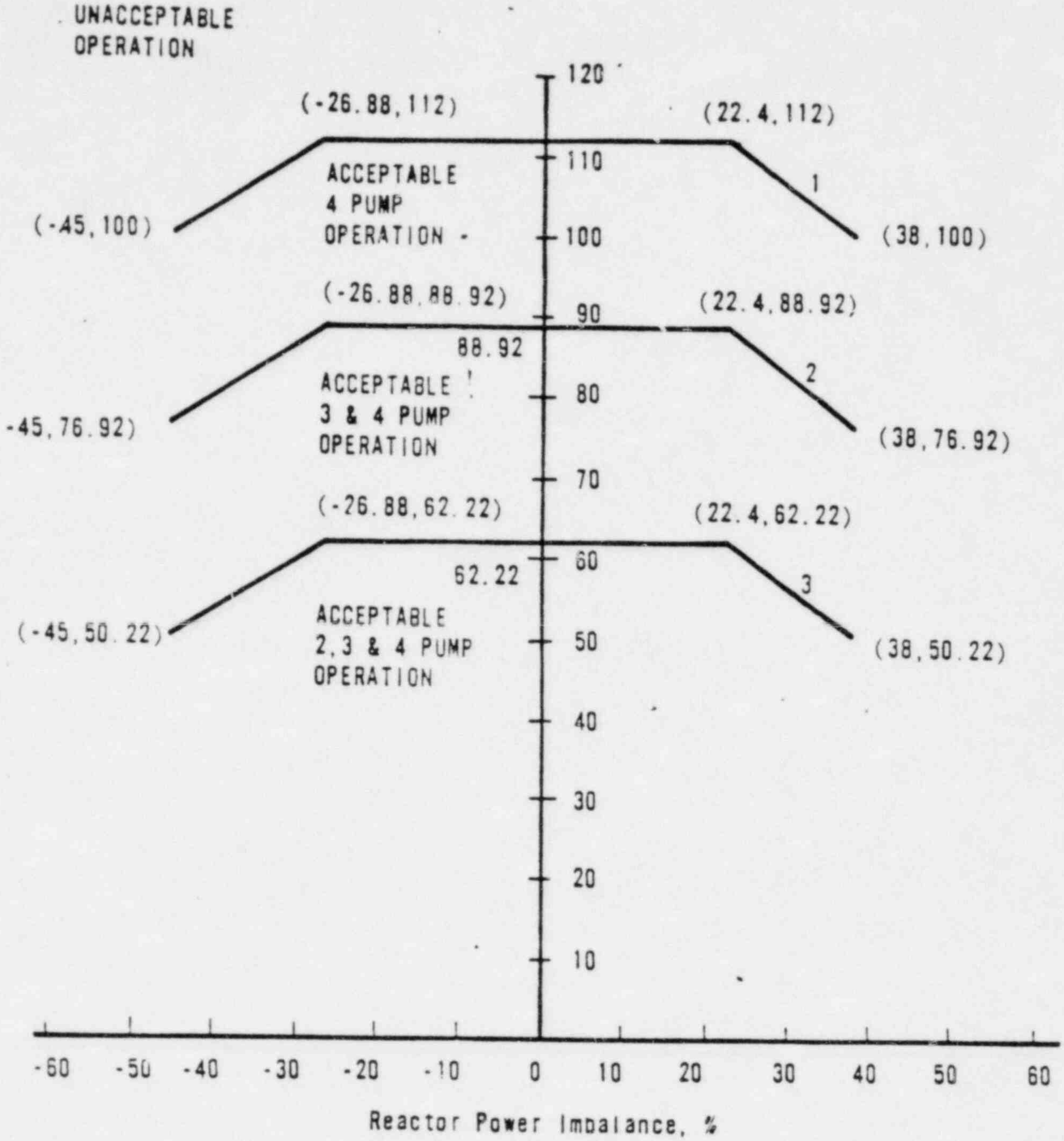
REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) FSAR, Section 3.2.3.1.1.c

(3) D. F. Ross and D. G. Eisenhut (NRC) memorandum to D. B. Vassallo and K. R. Goller (NRC) on "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors" dated December 8, 1976.

(4) L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) on "Evaluation of Interim Procedure for Calculating DNBR Reduction Due to Rod Bow" dated October 18, 1979.



CURVE	REACTOR COOLANT FLOW (GPM)
1	374,880
2	280,035
3	184,441

CORE PROTECTION SAFETY LIMITS

Figure 2.1-2

Amendment No. ~~5~~, ~~27~~, ~~37~~, ~~43~~, ~~52~~, 67

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 preselected 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 on these protection system instrumentation trip setpoints 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 104.9 percent of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis.

A. Overpower Trip Based on Flow and Imbalance

The power level trip setpoint 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. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip setpoint 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 setpoint 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 105.4 percent and reactor flow rate 100 percent or flow rate is 94.88 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 78.73 percent and reactor flow rate is 74.7 percent or flow rate is 71.16 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 51.85 percent and reactor flow rate is 49.2 percent or flow rate is 46.49 percent and the power level is 49.0 percent.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. 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 top half of core minus power in 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 associated with reactor power-to-reactor power imbalance boundaries by 1.054 percent for a 1 percent flow reduction.

B. Pump Monitors

In conjunction with the power imbalance/flow trip, the pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high-pressure trip setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high RCS pressure (2300 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(2)

The low-pressure (1800 psig) and variable low-pressure ($11.75\ T_{out} - 5103$) trip setpoints 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 ($11.75\ T_{out} - 5143$).

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) 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 setpoint of 620F.

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 that can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

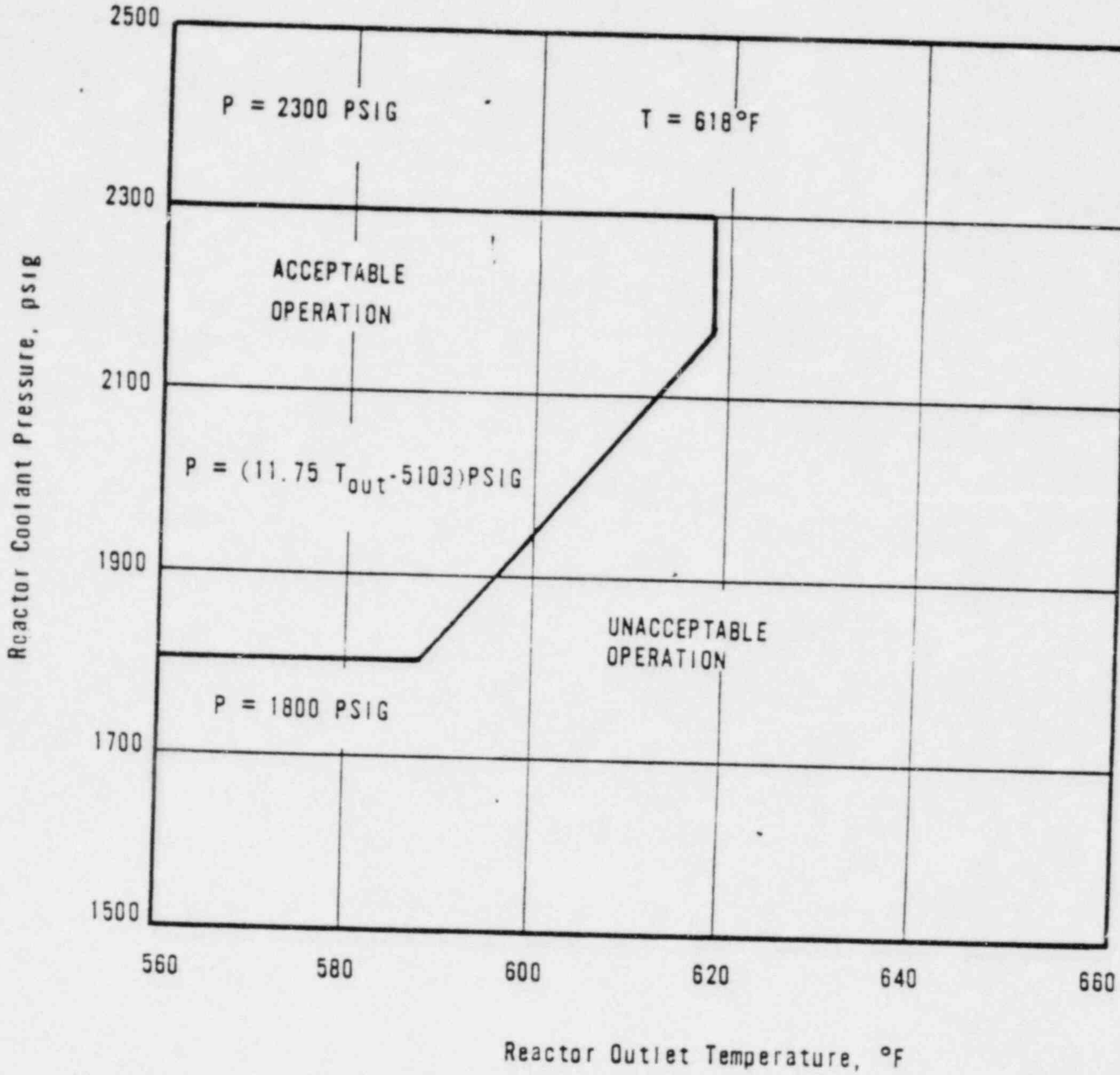
1. A nuclear overpower trip setpoint of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

The purpose of the 1720 psig high-pressure trip setpoint is to prevent normal operation with part of the reactor protection system bypassed. This high-pressure trip setpoint is lower than the normal low-pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The overpower trip setpoint of ≦5.0 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

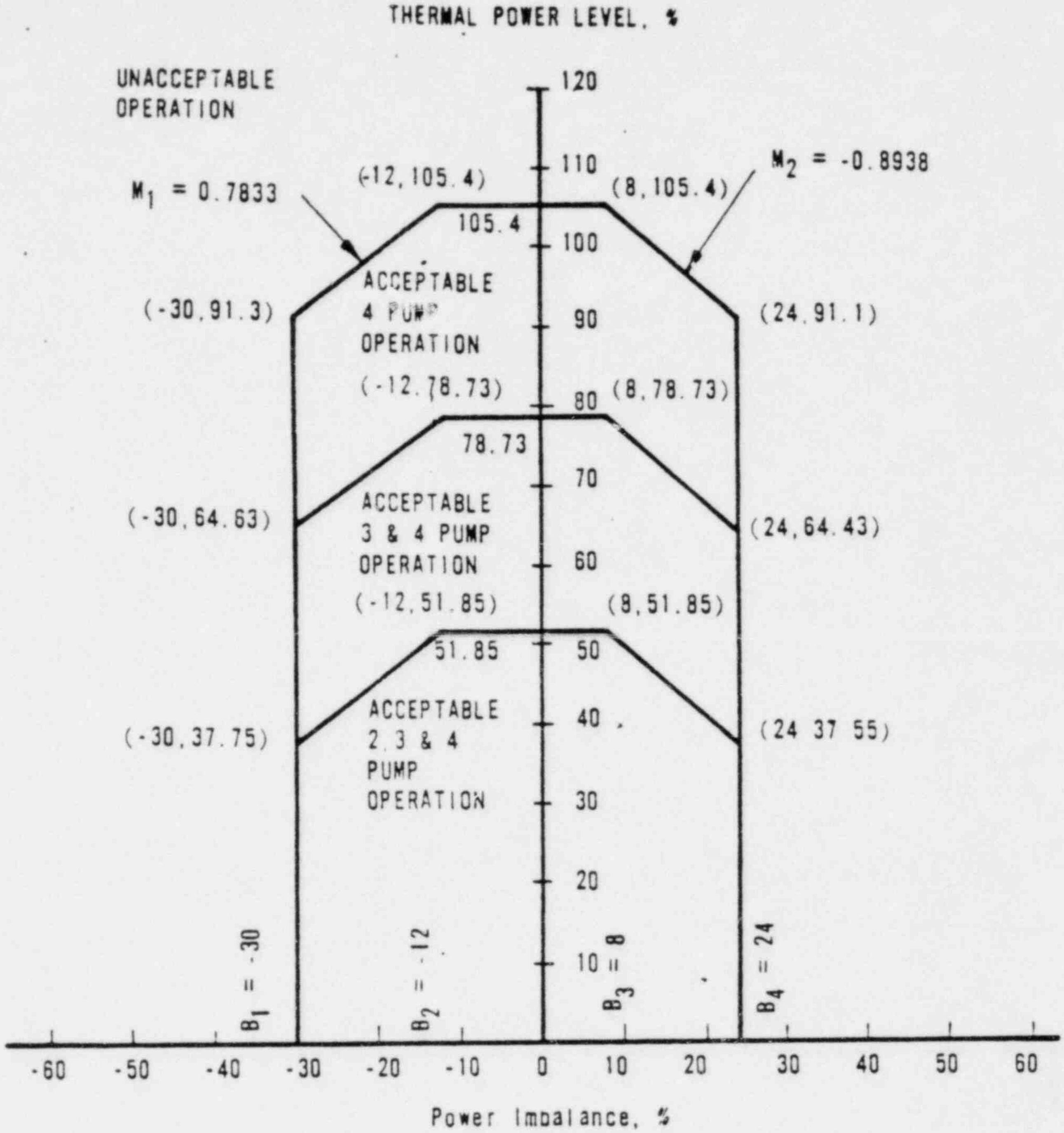
(1) FSAR, Section 14.1.2.3
(2) FSAR, Section 14.1.2.2
(3) FSAR, Section 14.1.2.7
(4) FSAR, Section 14.1.2.8
(5) FSAR, Section 14.1.2.6



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINT

Figure 2 3-1

Amendment No. 27, 49, 57



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS

Figure 2.3-2

Amendment No. ~~3~~, ~~27~~, ~~37~~, ~~43~~, ~~52~~, 67

Table 2.3-1 Reactor Protection System Trip Setting Limits

	Four RC Pumps Operating (Nominal Operating Power 100%)	Three RC Pumps Operating (Nominal Operating Power 75%)	One RC Pump Operating in each loop (Nominal Operating Power 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear power based on flow[b] and imbalance, % of rated, max	1.054 times flow minus reduction due to imbalance(s).	1.054 times flow minus reduction due to imbalance(s).	1.054 times flow minus reduction due to imbalance(s).	Bypassed
Nuclear power based on pump monitors, % of rated; max[c]	NA	NA	55	Bypassed
High RC system pressure, psig, max.	2300	2300	2300	1720[a]
Low RC system pressure, psig, min.	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min.	$11.75\ T_{out}-5103$[d]	$11.75\ T_{out}-5103$[d]	$11.75\ T_{out}-5103$[d]	Bypassed
RC temp, F, max	618	618	618	618
High reactor bldg. pressure, psig, max.	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

[a] Automatically set when other segments of the RPS (as specified) are bypassed.

[b] Reactor coolant system flow.

[c] The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

[d] T_{out} is given in degrees Fahrenheit (F).