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

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 2.1-1A-Unit 1. If the actual pressure/temperature point is below
2.1-1B-Unit 2
2.1-1C-Unit 3
and to the right of the line, the safety limit is exceeded.

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 2.1-2A-Unit 1. If the actual reactor-thermal-power/power
2.1-2B-Unit 2
2.1-2C-Unit 3
imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases - Unit 1

The safety limits presented for Oconee Unit 1 have been generated using BAW-2 critical heat flux (CHF) correlation(1). The reactor coolant system flow rate utilized is 106.5 percent of the design flow (131.32 x 10^6 lbs/hr) based on four-pump operation.(2)

To maintain the integrity of the fuel cladding and to prevent fission product released, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime 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 regime 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 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 2.1-1A 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 106.5 percent of 131.3 x 10^6 lbs/hr.). 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 2.1-2A 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 20.05 kw/ft for Unit 1.

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 2.1-2A correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1A is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.

The maximum thermal power for three-pump operation is 87.18 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.08 = 80.68 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

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

References

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

(2) Oconee 1, Cycle 4 - Reload Report - BAW-1447, March 1977.

Bases - Unit 2

The safety limits presented for Oconee Unit 2 have been generated using BAW-2 critical heat flux correlation[1] and the Reactor Coolant System flow rate of 106.5 percent of the design flow (design flow is 352,000 gpm for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate[2].

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 regime 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 regime 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 can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 coorelation 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 2.1-1B 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 374,880 gpm). This curve is based on the following nuclear power peaking factors with potential fuel densification and fuel rod bowing effects:

$$F_q^N = 2.565;\ F_{\Delta H}^N = 1.71^{(3)} F_z^N = 1.50$$

The design peaking combination results in a more conservative DNBR than any other power shape that exists during normal operation.

The curves of Figure 2.1-2B are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel 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 20.15 kw/ft for Unit 2.

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 2.1-2B correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1B is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3B.

The maximum thermal power for three-pump operation is 87.18 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.08 = 80.68 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

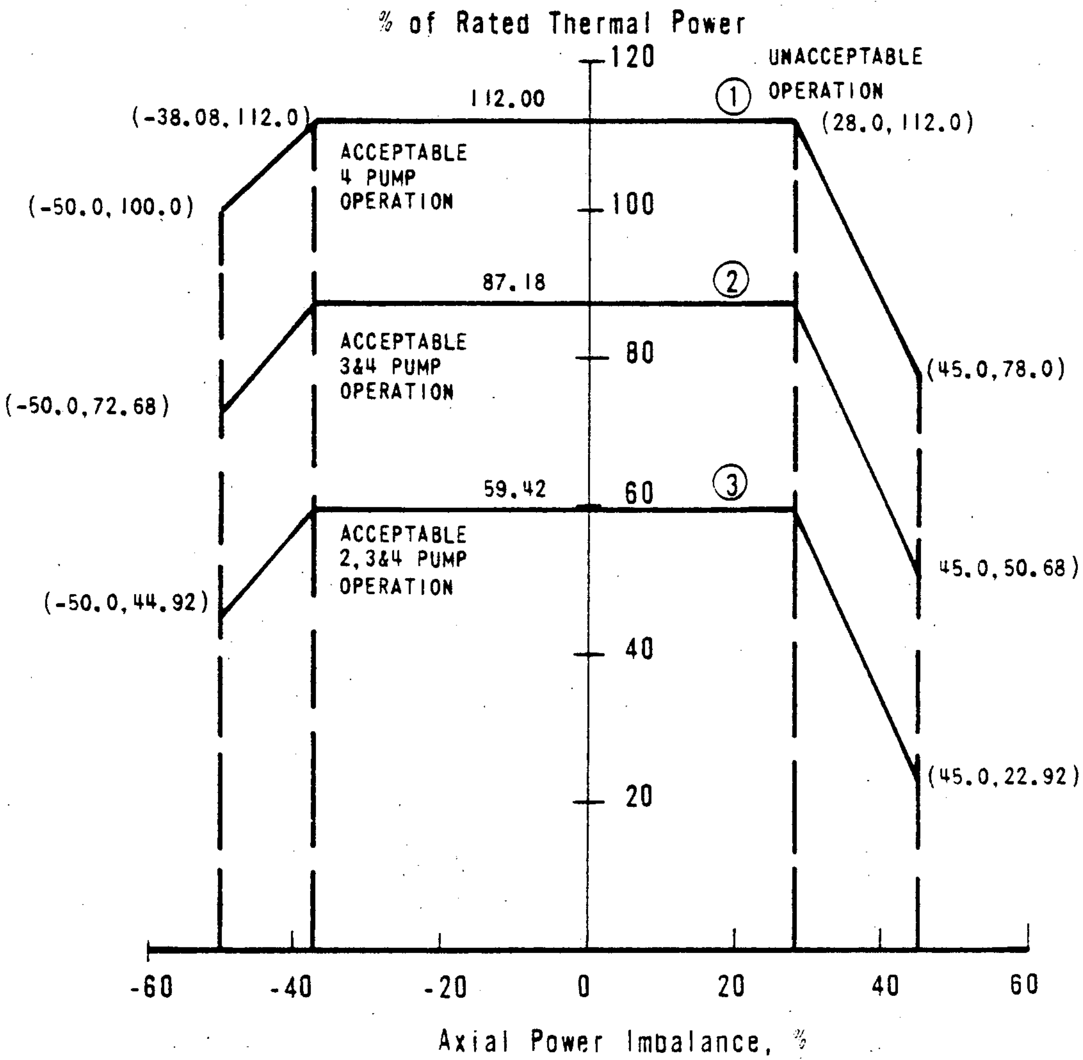
For each curve of Figure 2.1-3B, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The 1.30 DNBR curve for four-pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four-pump curve will be above and to the left of the other curves.

References

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

(2) Oconee 2, Cycle 4- Reload Report - BAW-1491, August 1978.

(3) Oconee 2, Cycle 5 - Reload Report - BAW-1565.



CURVE	REACTOR COOLANT FLOW (GPM)
1	374,880
2	280,035
3	183,690

CORE PROTECTION
SAFETY LIMITS
UNIT 2
OCONEE NUCLEAR STATION
Figure 2.1-2B

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 protective action to prevent any combination of process variables from exceeding a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3.1A-Unit 1 and
2.3-1B-Unit 2
2.3-1C-Unit 3

Figure 2.3-2A-Unit 1
2.3-2B-Unit 2
2.3-2C-Unit 3

The pump monitors shall produce a reactor trip for the following conditions:

a. Loss of two pumps and reactor power level is greater than 55% of rated power.

b. Loss of two pumps in one reactor coolant loop and reactor power level is greater than 0.0% of rated power.

c. Loss of one or two pumps during two-pump operation.

Bases

The reactor protective 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 protective system instrumentation are listed in Table 2.3-1A-Unit 1. The safety analysis has been based upon these protective
2.3-1B-Unit 2
2.3-1C-Unit 3
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 105.5% 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 more conservative than the value used in the safety analysis. (4)

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 2.3-1A are as follows:

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

2. Trip would occur when three reactor coolant pumps are operating if power is 80.68% and reactor flow rate is 74.7% or flow rate is 69.44% and power level is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.92 and reactor flow rate is 49.0% or flow rate is 45.37% and the power level is 49%.

   The flux-to-flow ratios account for the maximum calibration and instrument 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.

For safety calculations the maximum calibration and instrumentation errors for the power level trip 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 such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2
2.3-2C - Unit 3

Table 2.3-1B
Unit 2

Reactor Protective System Trip Setting Limits

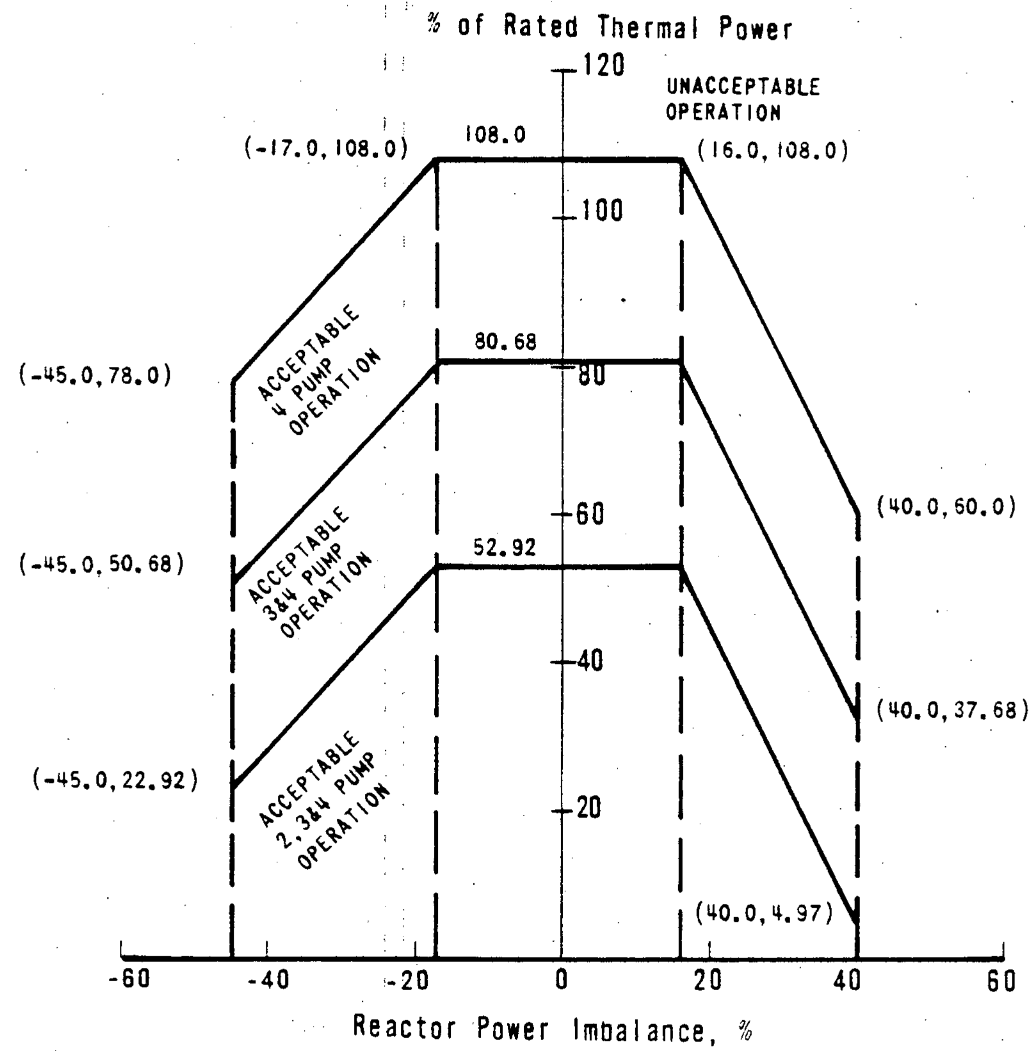
	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	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 are bypassed.



PROTECTION SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 2
OCONEE NUCLEAR STATION
Figure 2.3.2B

f. If the maximum positive quadrant power tilt exceeds the Maximum Limit of Table 3.5-1, the reactor shall be shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.

g. Quadrant power tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% full power.

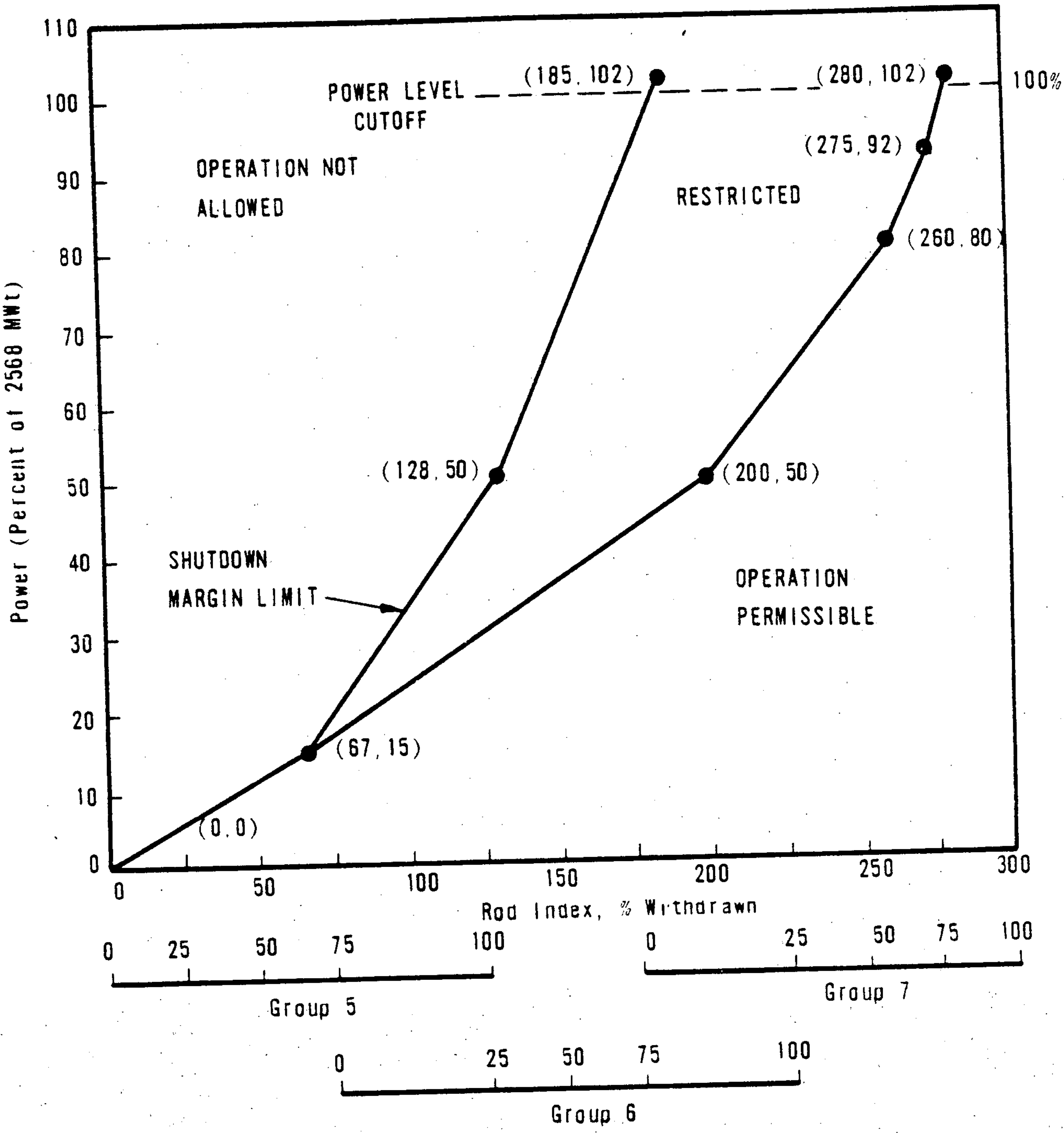
3.5.2.5 Control Rod Positions

a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

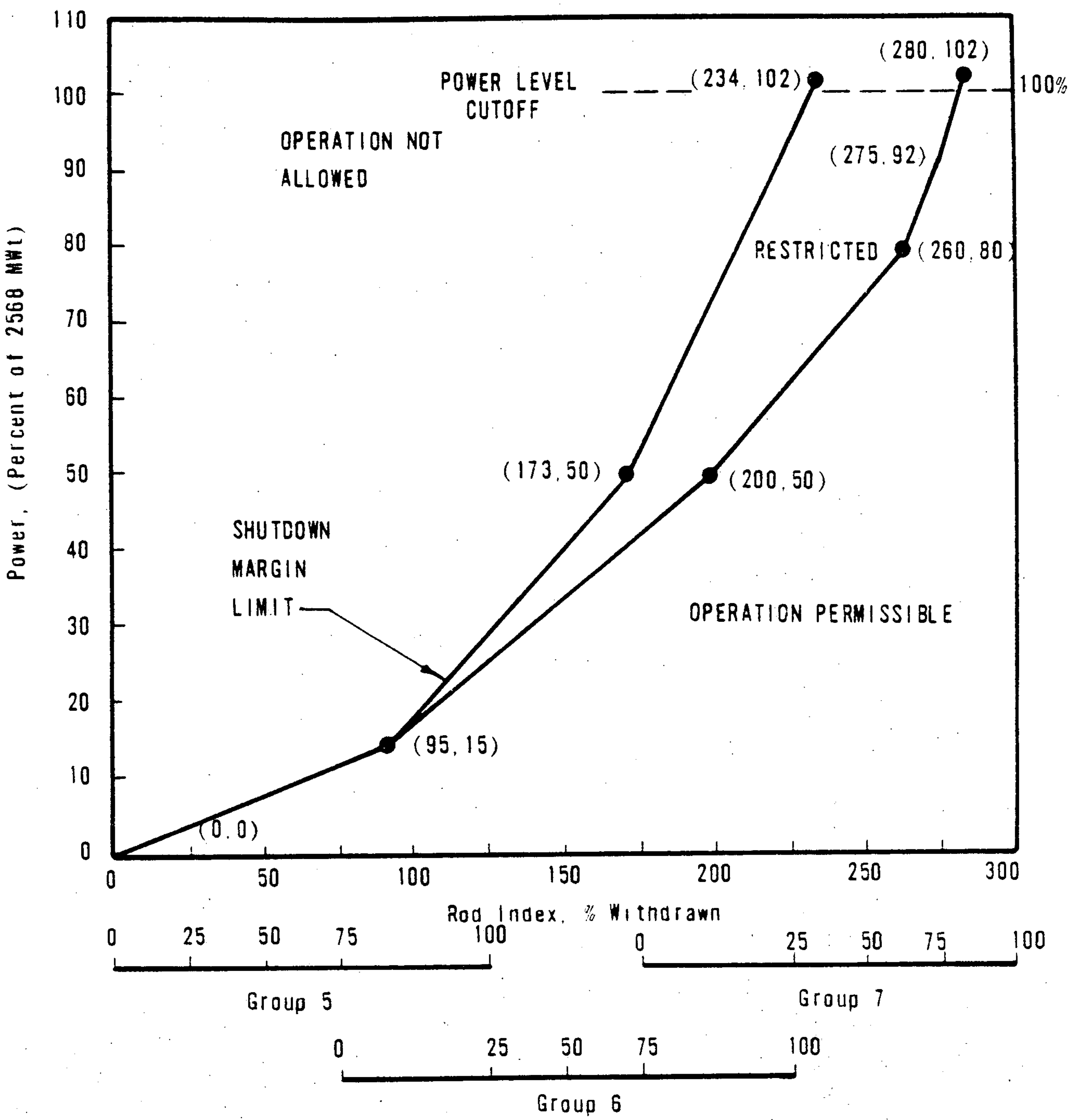
b. Except for physics tests, operating rod group overlap shall be 25% ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours or the reactor shall be placed in a hot shutdown condition within an additional 12 hours.

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on figures 3.5.2-1A1 and 3.5.2-1A2 (Unit 1); 3.5.2-1B1 and 3.5.2-1B2 (Unit 2); 3.5.2-1C1, 3.5.2-1C2 and 3.5.2-1C3 (Unit 3) for four pump operation, and on figures 3.5.2-2A1 and 3.5.2-2A2 (Unit 1), 3.5.2-2B1 and 3.5.2-2B2 (Unit 2); 3.5.2-2C1, 3.5.2-2C2 and 3.5.2-2C3 (Unit 3) for two or three pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-4A1, and 3.5.2-4A2 (Unit 1); 3.5.2-4B1 and 3.5.2-4B2 (Unit 2); 3.5.2-4C1, 3.5.2-4C2, and 3.5.2-4C3 (Unit 3).

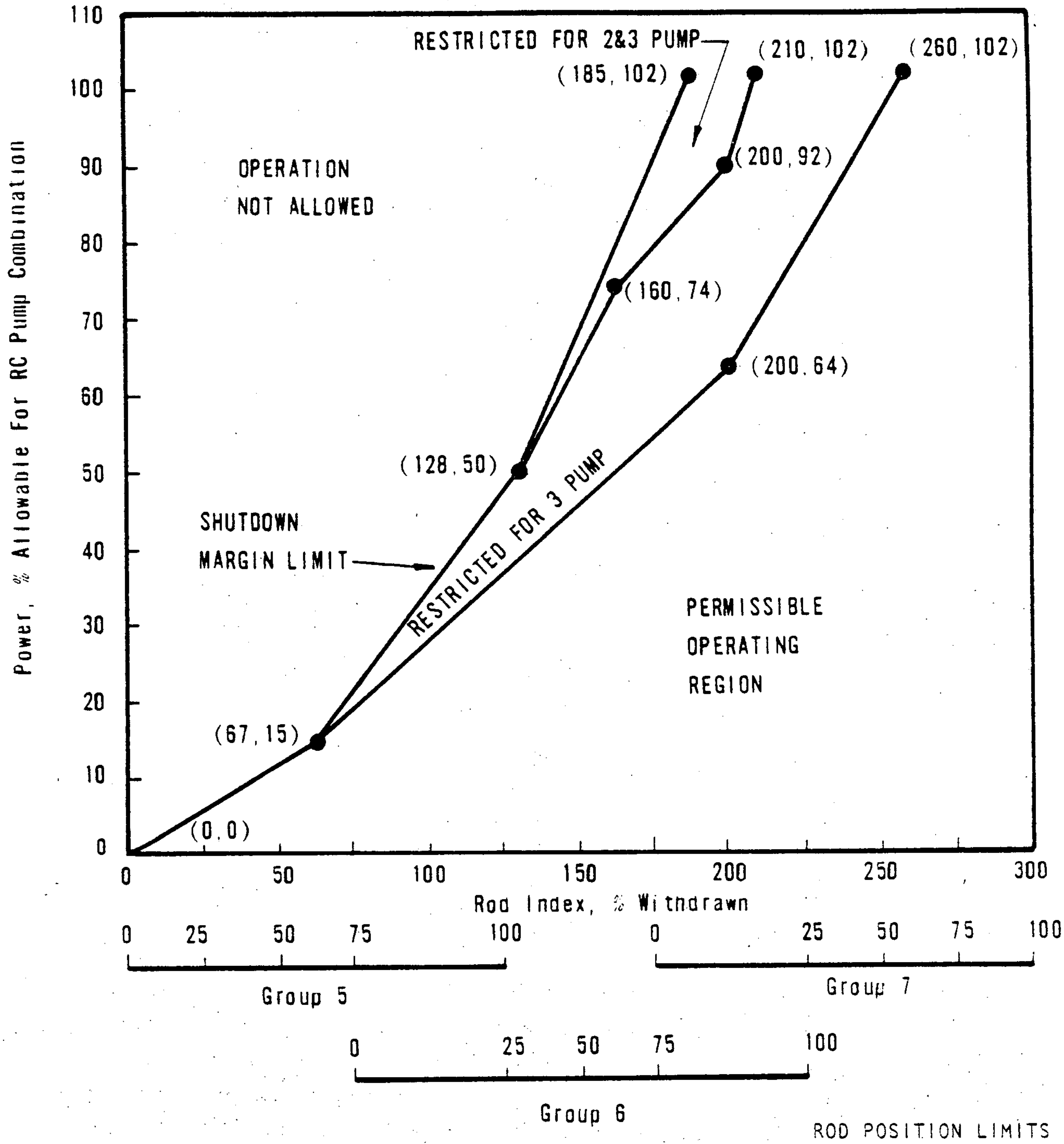
If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times.



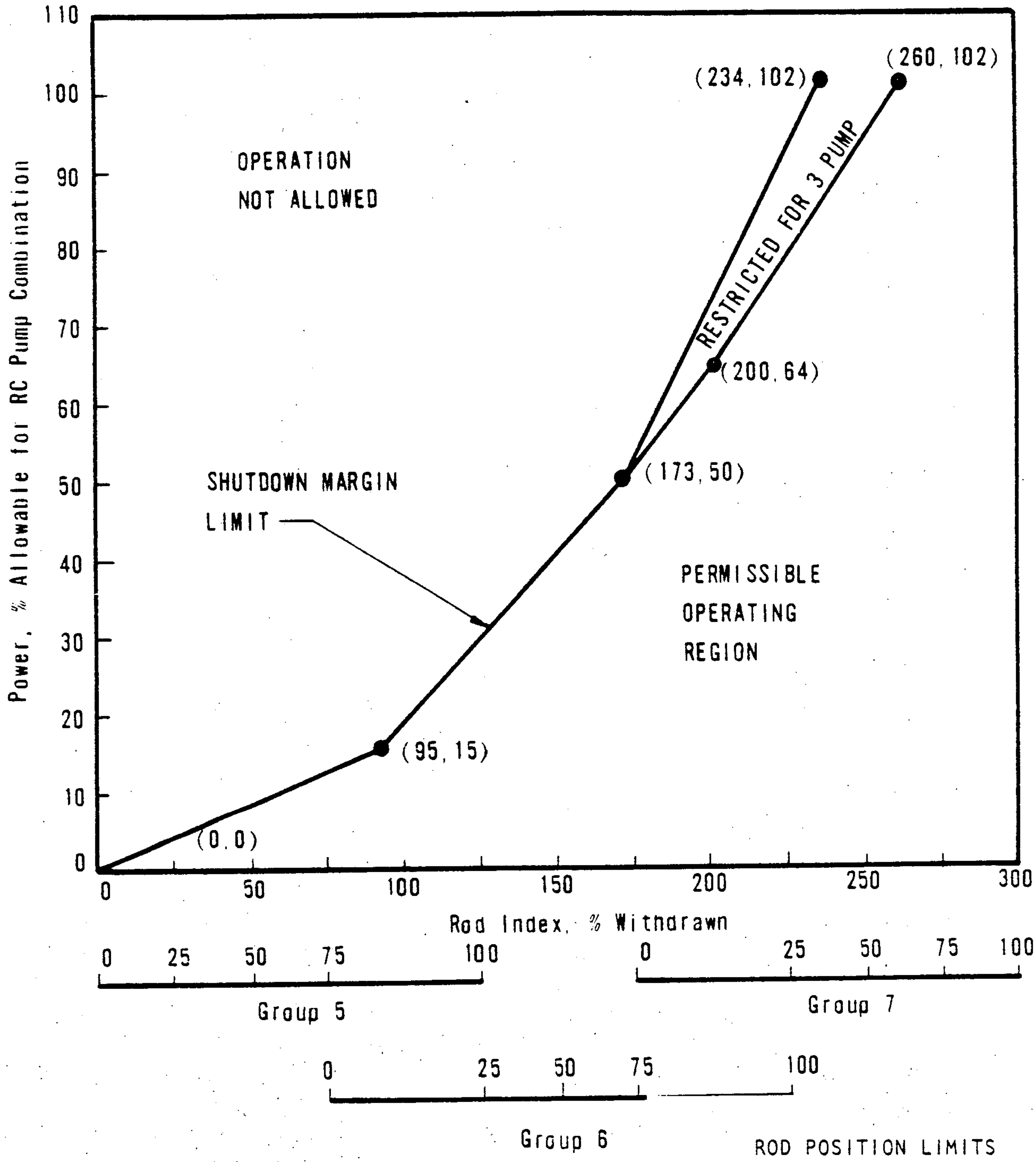
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 150 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-1B1



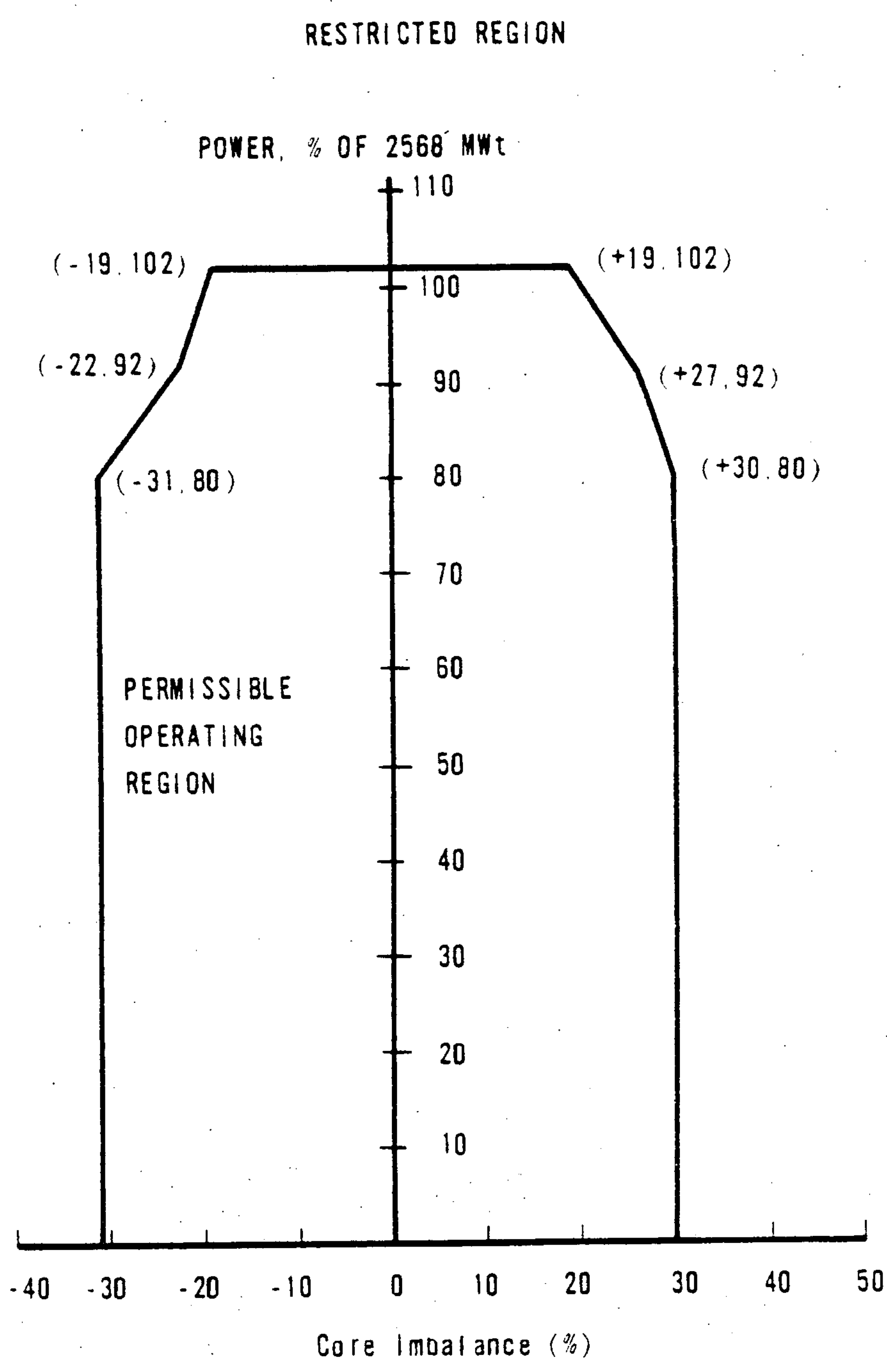
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 150 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-1B2



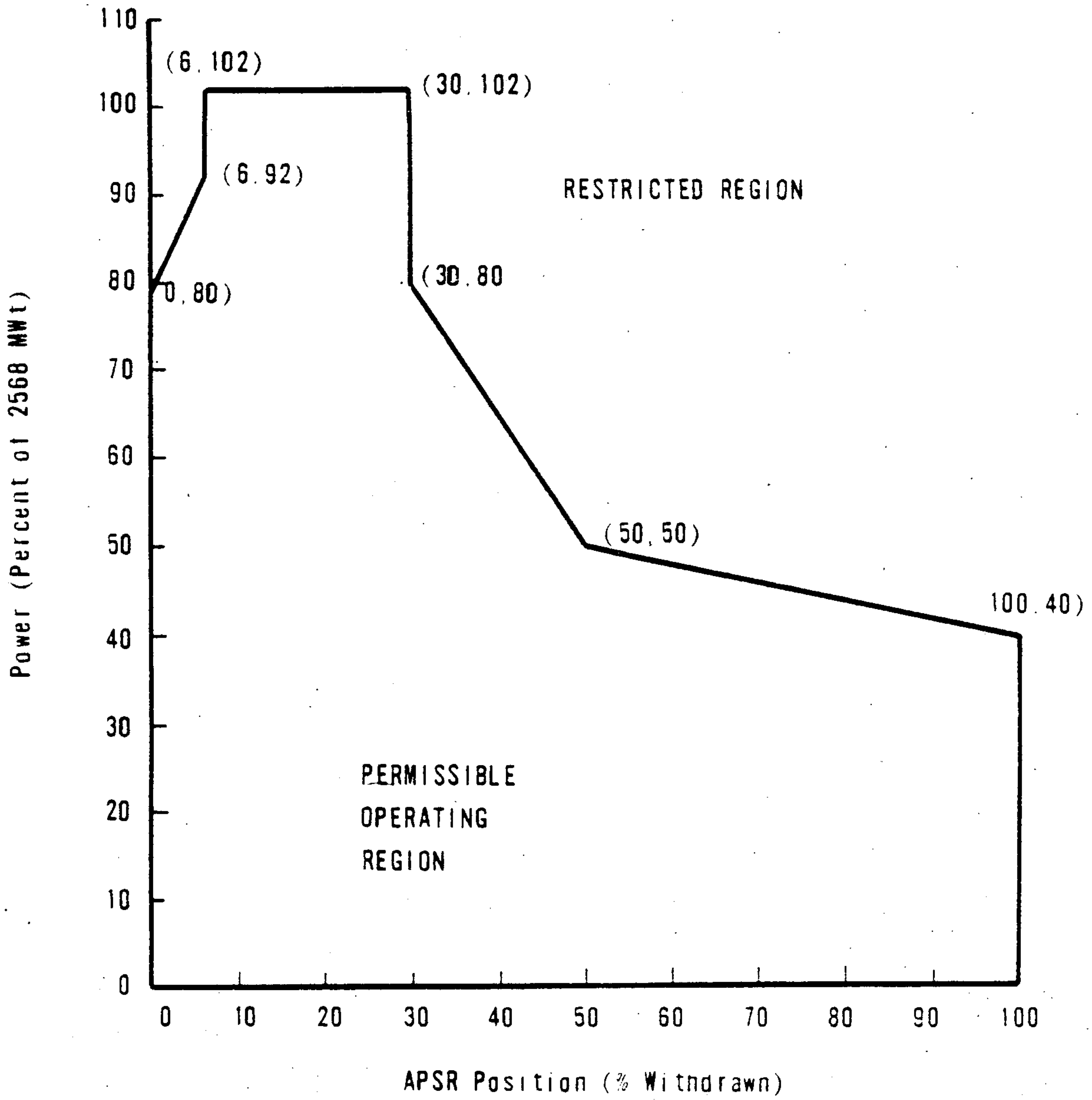
ROD POSITION LIMITS
FOR TWO AND THREE PUMP
OPERATION FROM
0 TO 150 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-2B1



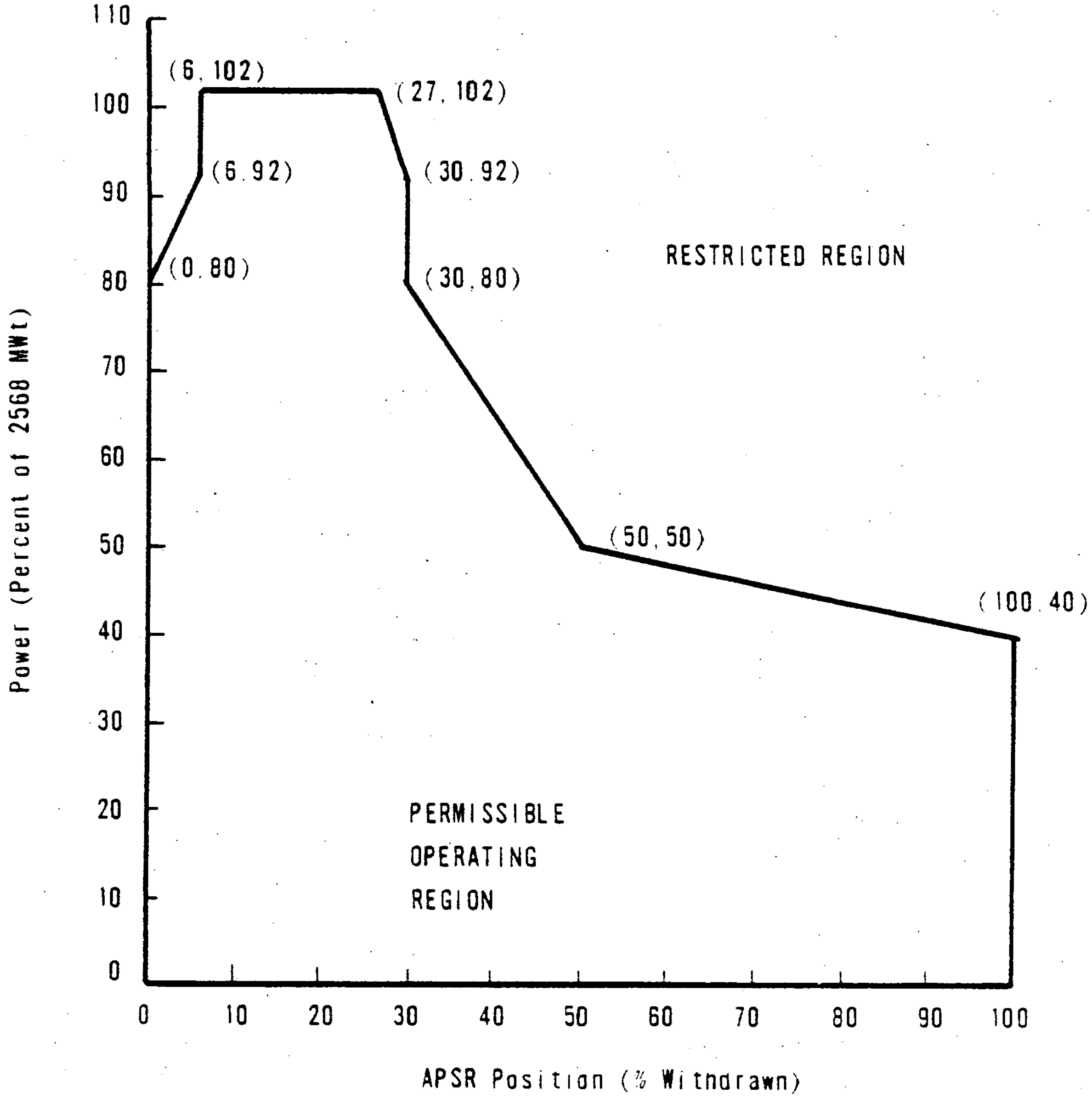
ROD POSITION LIMITS
FOR TWO AND THREE PUMP
OPERATION FROM
150 ± EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-2B2



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 0 TO 360 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-3B1



APSR POSITION LIMITS
FOR OPERATION
FROM 0 TO 150 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-4B1



APSR POSITION LIMITS FOR
OPERATION
AFTER 150 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-4B2