The safety limits presented for Oconee Unit 3 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 (131.32×10^6 lbs/hr 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 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 2.1-1C 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 139.86×10^6 lbs/hr.). This curve is based on the following nuclear power peaking factors with potential fuel densification and fuel rod bowing effects: $F_q^N = 2.67$; $F_{\Delta H}^N = 1.78$; $F_z^N = 1.50$. The design peaking combination results in a more conservative DNBR than any other power shape that exists during normal operation.

The curves of Figure 2.1-2C 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 a nuclear peaking factor of $F_q^N = 2.67$ or 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 3.

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

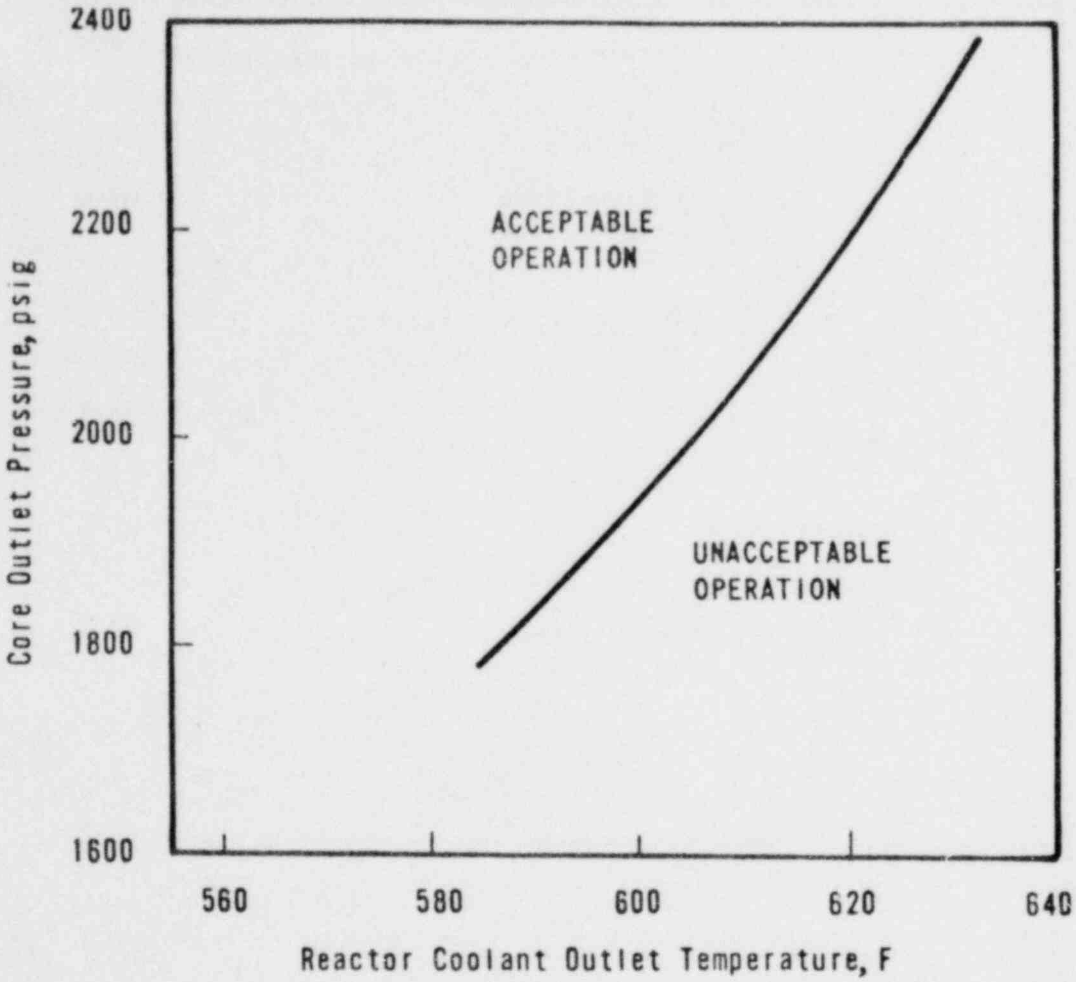
The maximum thermal power for three-pump operation is 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055= 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

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

References

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

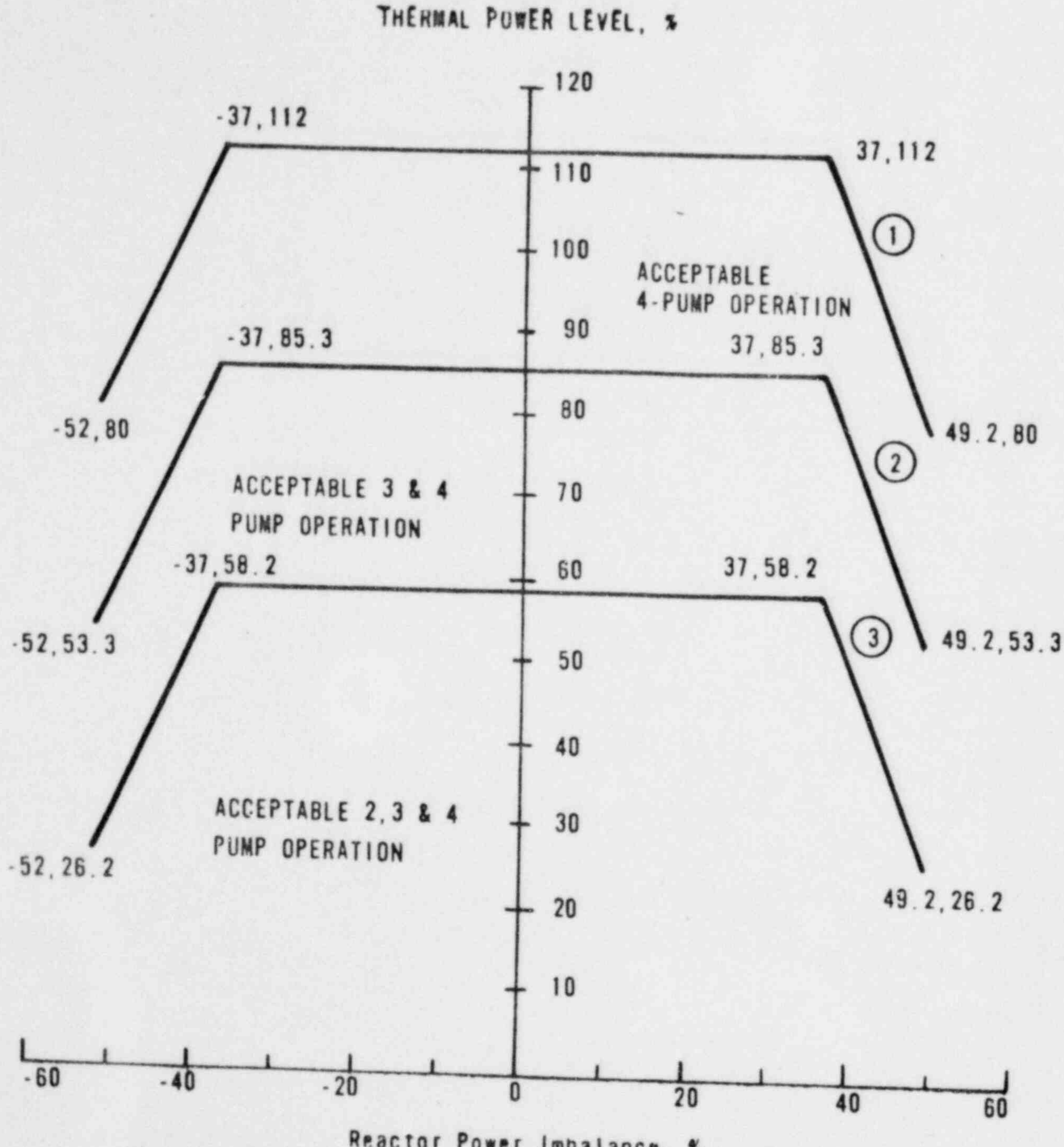
(2) Oconee 3, Cycle 3 - Reload Report - BAW-1453, August, 1977.



CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.1-1C



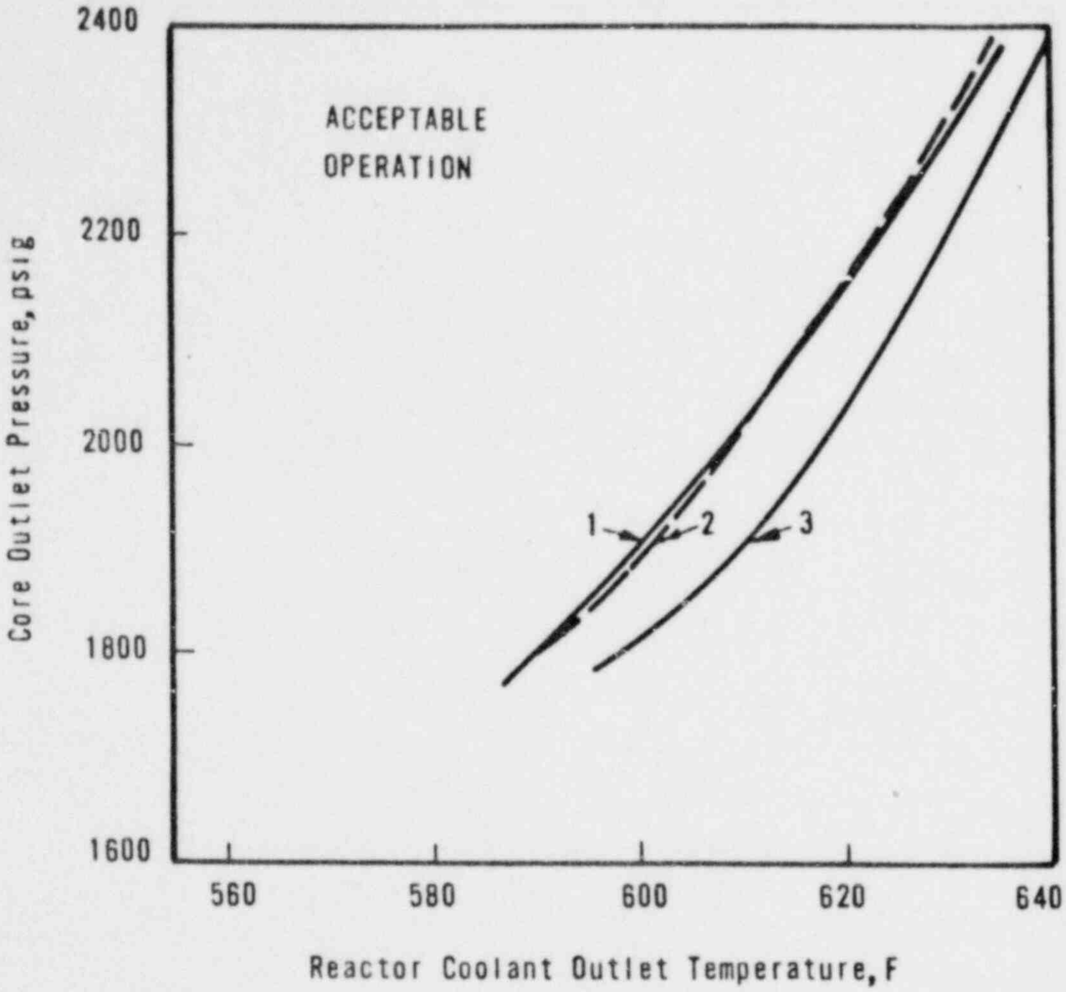
Curve	Reactor Coolant Flow, gpm
1	374,880 (100%)*
2	280,035 (74.7%)
3	183,690 (49.0%)

*106.5% of first-core design flow.

CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.1-2C



Curve	Coolant Flow, gpm	Power, %	Pumps Operating	Type of Limit
1	374,880 (100%)*	112	4	DNBR
2	280,035 (74.7%)	86.7	3	DNBR
3	183,690 (49.0%)	59.0	2	Quality

*106.5% of first-core design flow.

CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.1-3C

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. (Power/RC pump trip setpoint is reset to 55% for 2 pump operation.)

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 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.

level trip and associated reactor power/reactor power-imbalance boundaries by 1.055% for a 1% flow reduction.

Pump Monitors

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 circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
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 (1800) psig and variable low pressure ($11.14\ T_{out}-4706$) trip
(1800) psig ($11.14\ T_{out}-4706$)
(1800) psig ($11.14\ T_{out}-4706$)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
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 ($11.14\ T_{out}-4746$)
($11.14\ T_{out}-4746$)
($11.14\ T_{out}-4746$)

Coolant Outlet Temperature

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

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 loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

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 2.3-1A. Two conditions are imposed when
2.3-1B
2.3-1C

the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0% of rated power during reactor shutdown.
2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

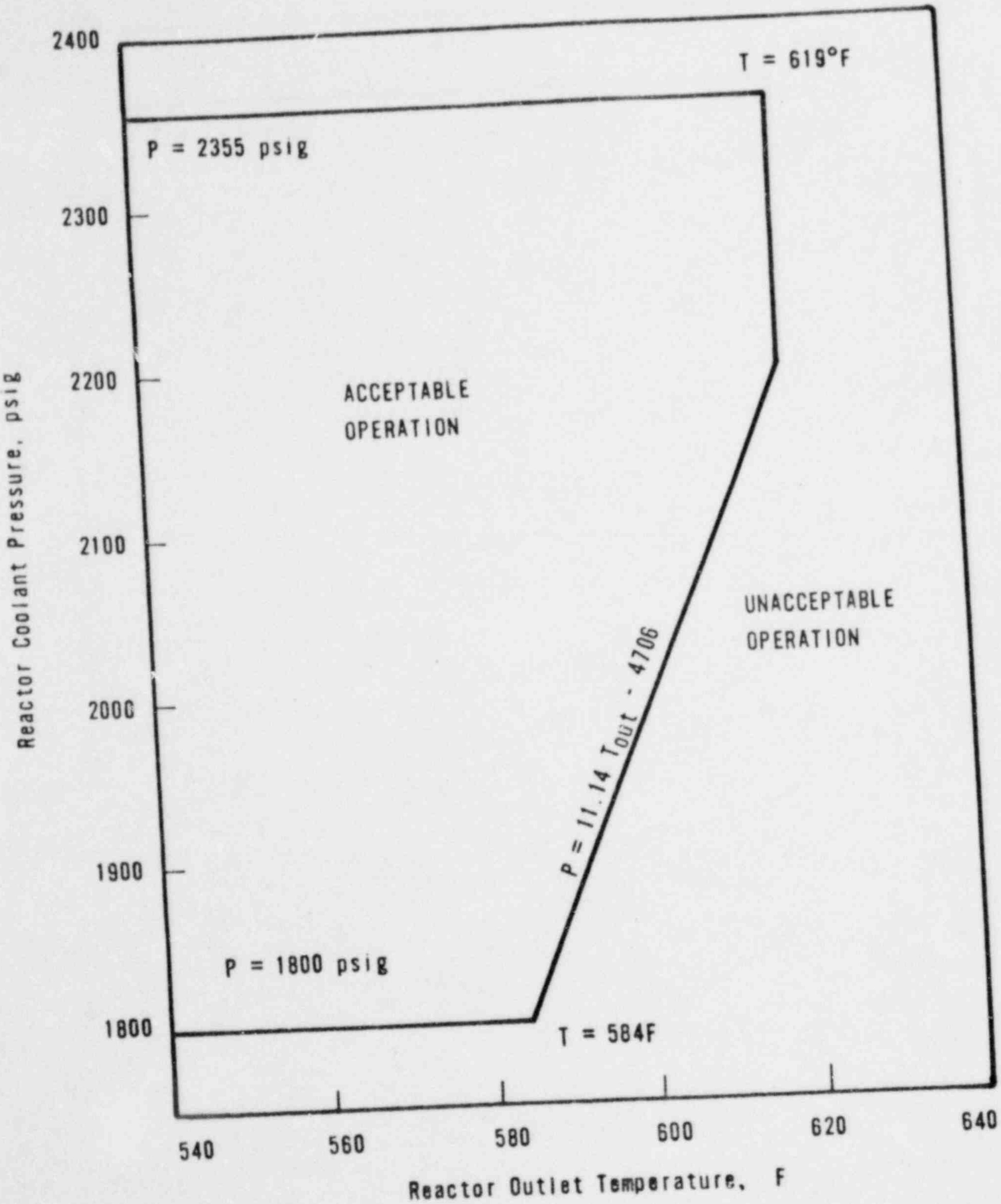
The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The over power trip set point of $\leq$ 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% of rated power if none of the reactor coolant pumps were operating.

Two Pump Operation

Operation with one pump in each loop will be allowed only following reactor shutdown. After shutdown has occurred, reset the pump contact monitor power level trip setpoint to 55.0%.

REFERENCES

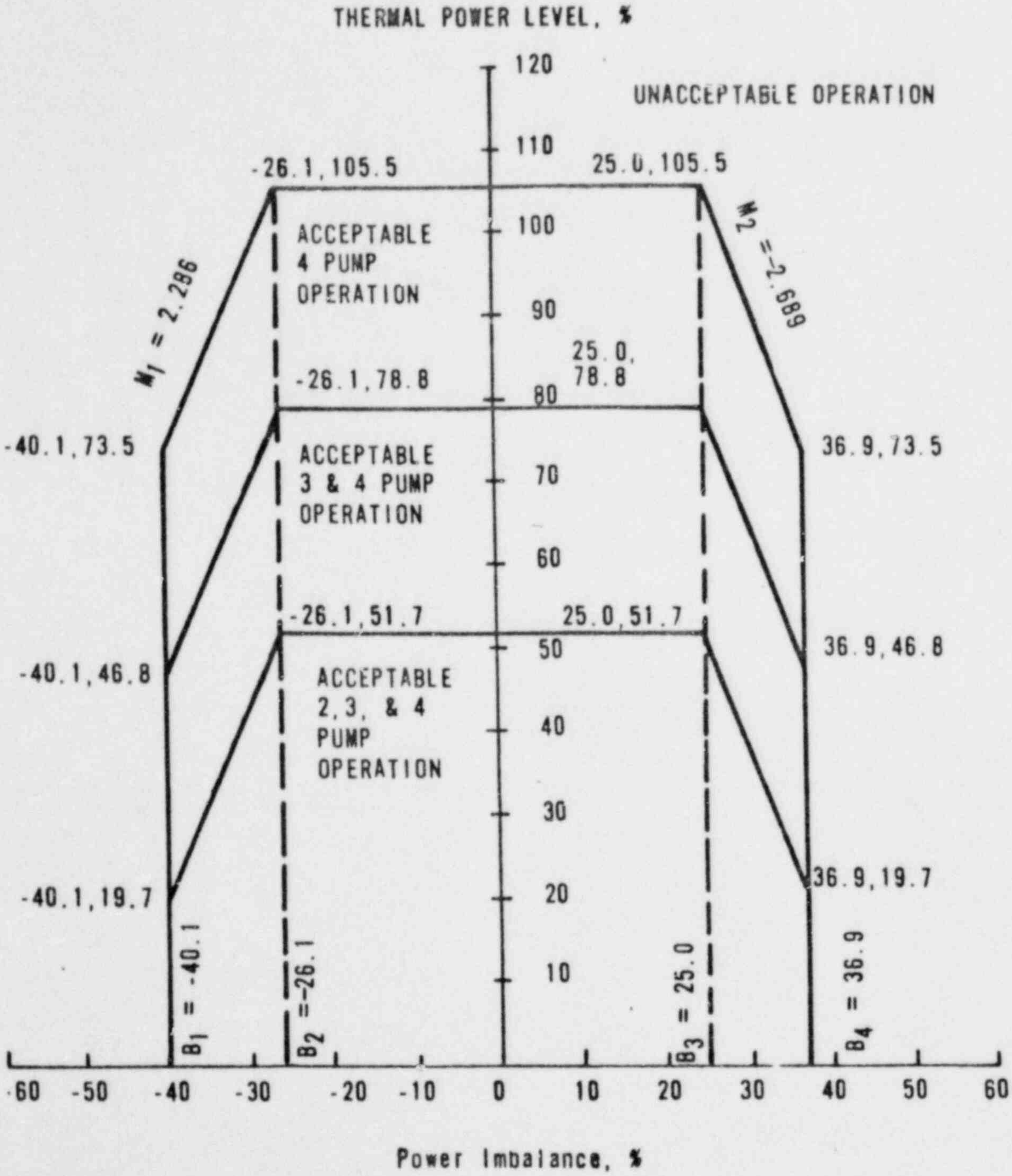
(1) FSAR, Section 14.1.2.2
(2) FSAR, Section 14.1.2.7
(3) FSAR, Section 14.1.2.8
(4) FSAR, Section 14.1.2.3
(5) FSAR, Section 14.1.2.6



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.3-1C



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.3-2C

Table 2.3-1C

Unit 3

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 -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)	1.055 times flow minus reduction due to imbalance	1.055 times flow minus reduction due to imbalance	1.055 times flow minus reduction due to imbalance	Bypassed
3. Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	N/A	55%	Bypassed
4. High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	$1720^{(4)}$
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

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

pump operati Also, excepting physics test or exercising control rods, the axial power shaping control od insertion/withdrawal limits are specified on figures 3.5.2-4A1, 3.5.2-4A2 and 3.5.2-4A3 (Unit 1), 3.5.2-4B1, 3.5.2-4B2, and 3.5.2-4B3 (Unit 2), and 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.

d. Except for physics tests, power shall not be increased above the power level cutoff as shown on Figures 3.5.2-1A1, 3.5.2-1A2 (Unit 1), 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 (Unit 2), and 3.5.2-1C!, 3.5.2-1C2, 3.5.2-1C3 (Unit 3), unless the following requirements are met.

(1) The xenon reactivity shall be within 10 percent of the value for operation at steady-state rated power.

(2) The xenon reactivity worth has passed its final maximum or minimum peak during its approach to its equilibrium value for operation at the power level cutoff.

3.5.2.6 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3A3, 3.5.2-3B1 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.

Bases

The power-imbalance envelope defined in Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3A3, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2 and 3.5.2-3C3 is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification power spike factors (Units 1 and 2 only)
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping bank)

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% Δk/k at rated power. These values have been shown to be safe by the safety analysis (2, 3, 4, 5) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% Δk/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% Δk/k at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% Δk/k ejected rod worth at rated power.

**Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Groups 6 and 7 to be partially inserted.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding 5.10% for Unit 1. The limits shown in Specification 3.5.2.4
5.10% for Unit 2
5.10% for Unit 3
are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.6, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions are included in Technical Specification 3.5.2.5d to prevent excessive power peaking by transient xenon. The xenon reactivity must be beyond its final maximum or minimum peak and approaching its equilibrium value at the power level cutoff.

REFERENCES

[1]FSAR, Section 3.2.2.1.2

[2]FSAR, Section 14.2.2.2

[3]FSAR, SUPPLEMENT 9

[4]B&W FUEL DENSIFICATION REPORT

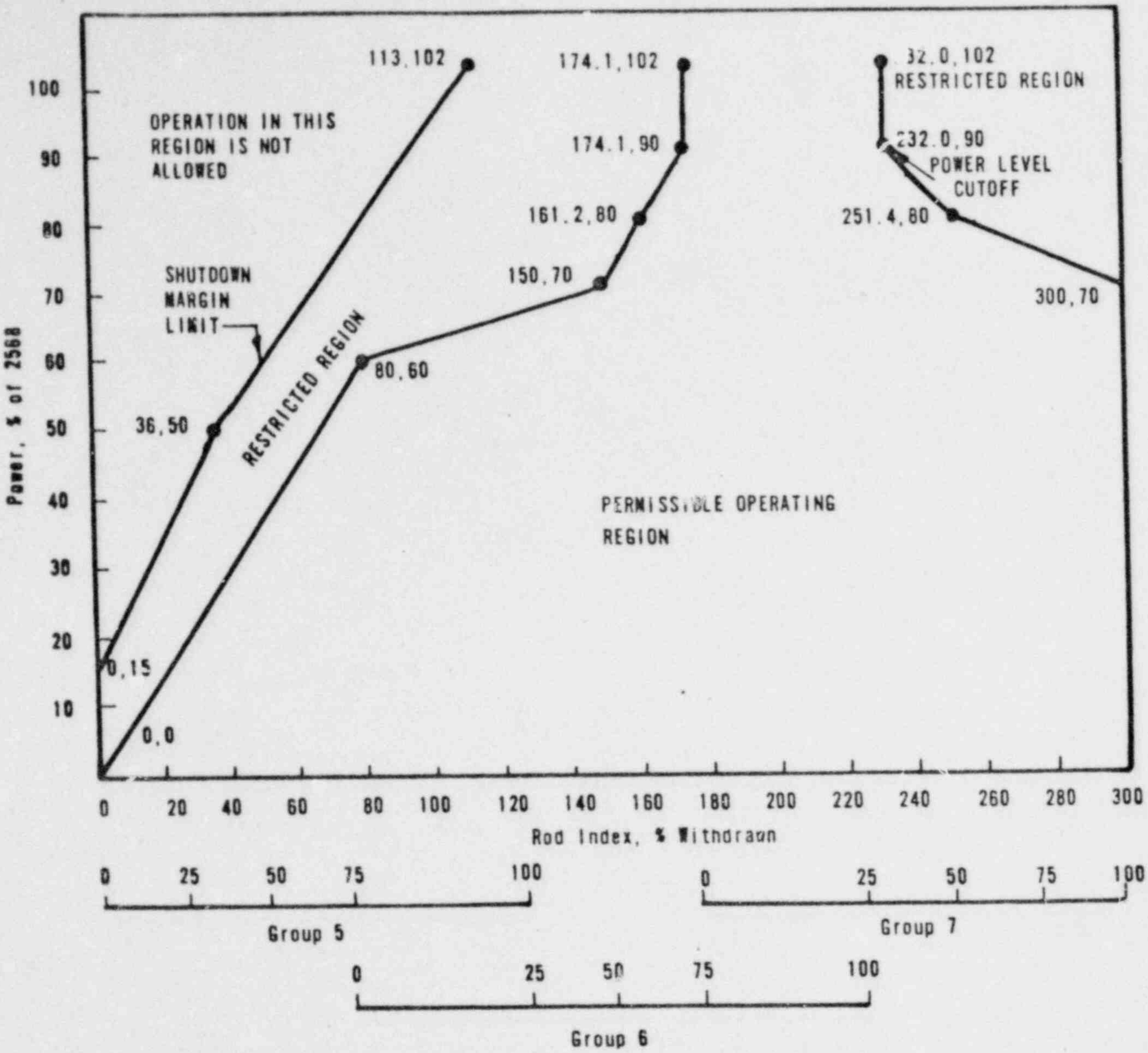
BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

BAW-1400 (UNIT 3)

[5]OCONEE UNIT 1, CYCLE 4 RELOAD REPORT,

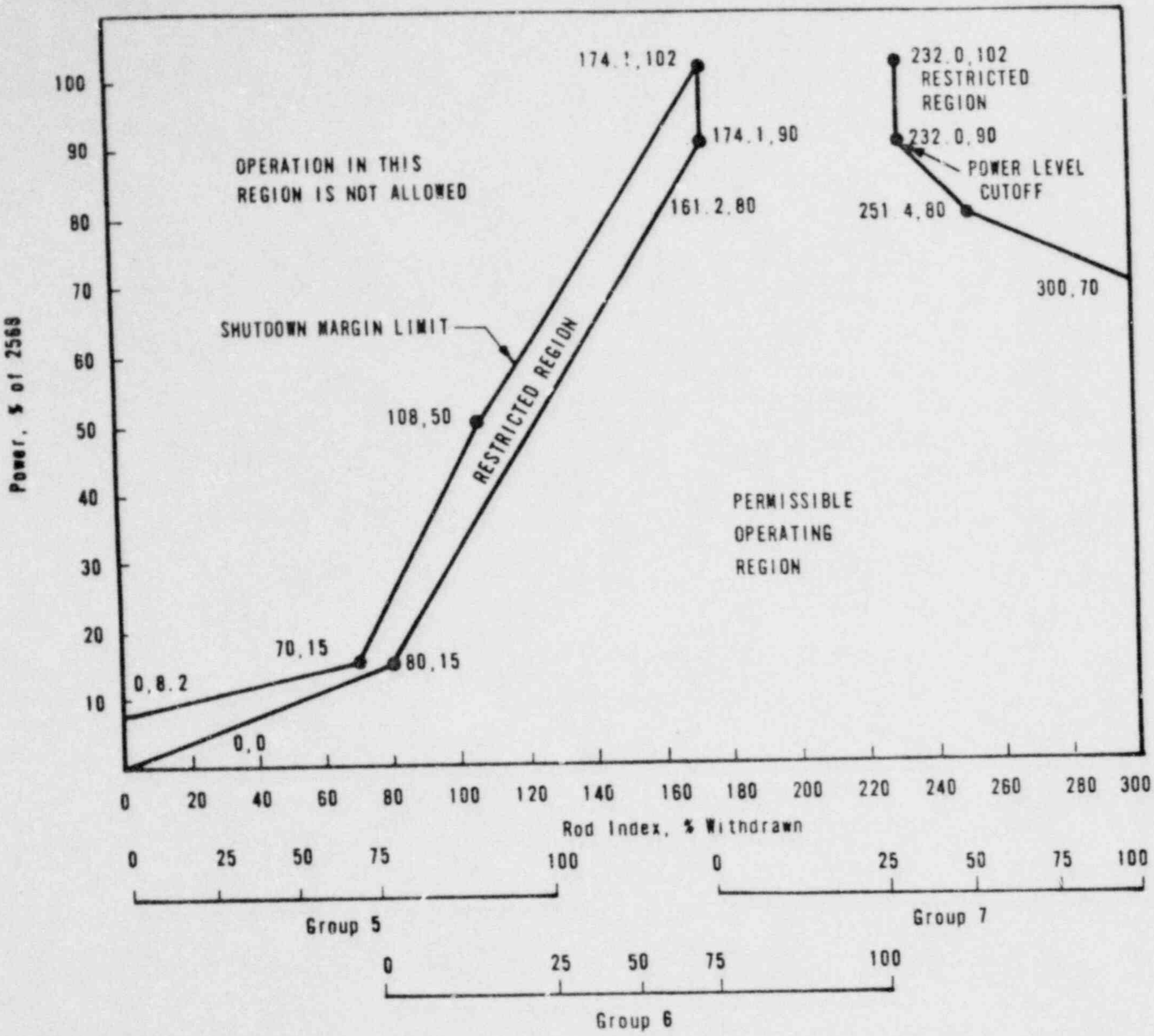
BAW-1447, March, 1977 Section 7.11



ROD POSITION LIMITS FOR
FOUR PUMP OPERATION FROM
0 TO 100 ($\pm$ 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

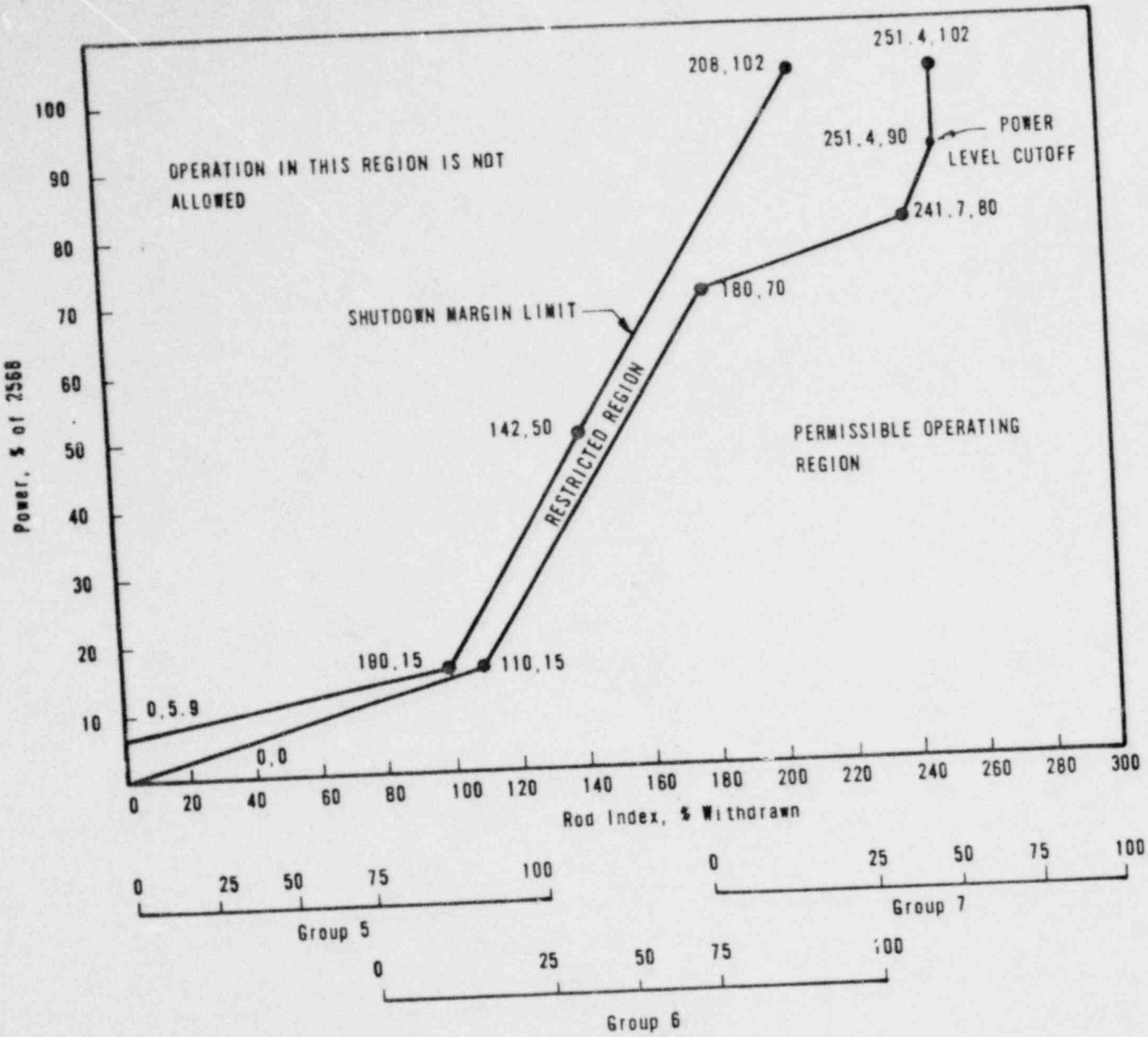
Figure 3.5.2-1C1



ROD POSITION LIMITS FOR FOUR PUMP OPERATION FROM 100 ($\pm$ 10) EFPD TO 235 ($\pm$ 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

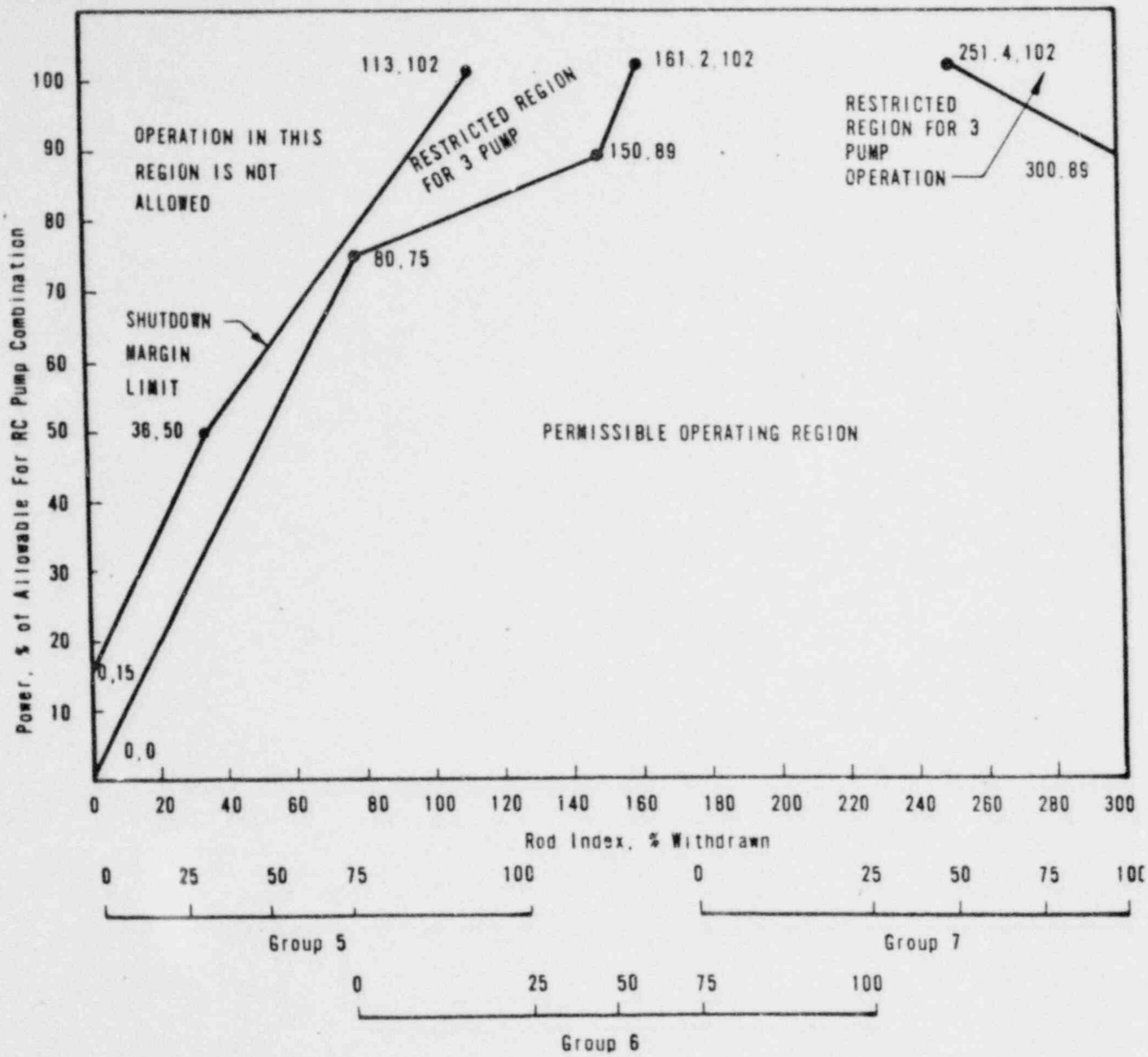
Figure 3.5.2-1C2



ROD POSITION LIMITS FOR FOUR
PUMP OPERATION AFTER 235
(+ 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

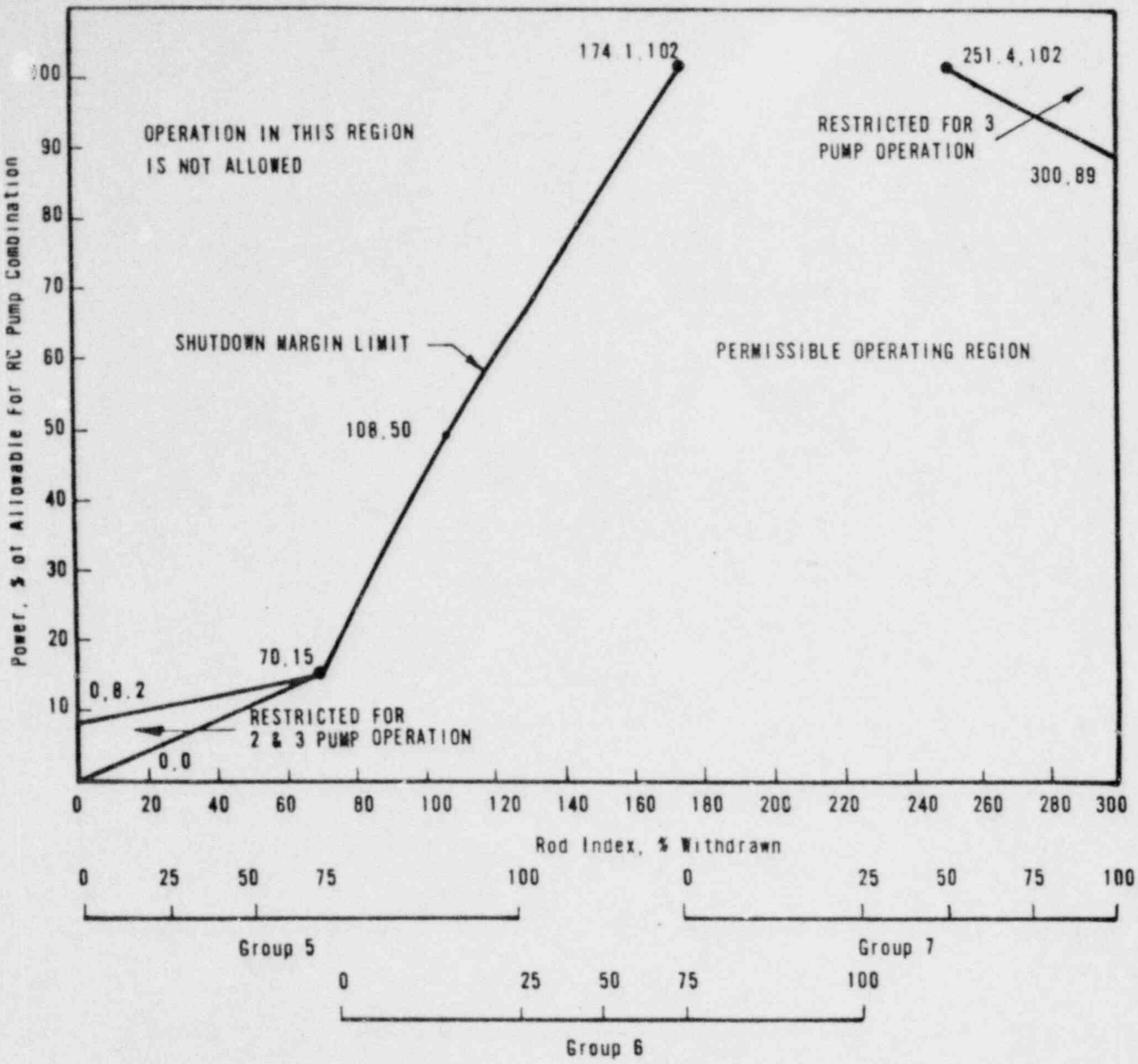
Figure 3.5.2-1C3



ROD POSITION LIMITS FOR TWO- AND THREE-PUMP OPERATION FROM 0 TO 100 ($\pm$ 10) EFPD UNIT 3

OCONEE NUCLEAR STATION

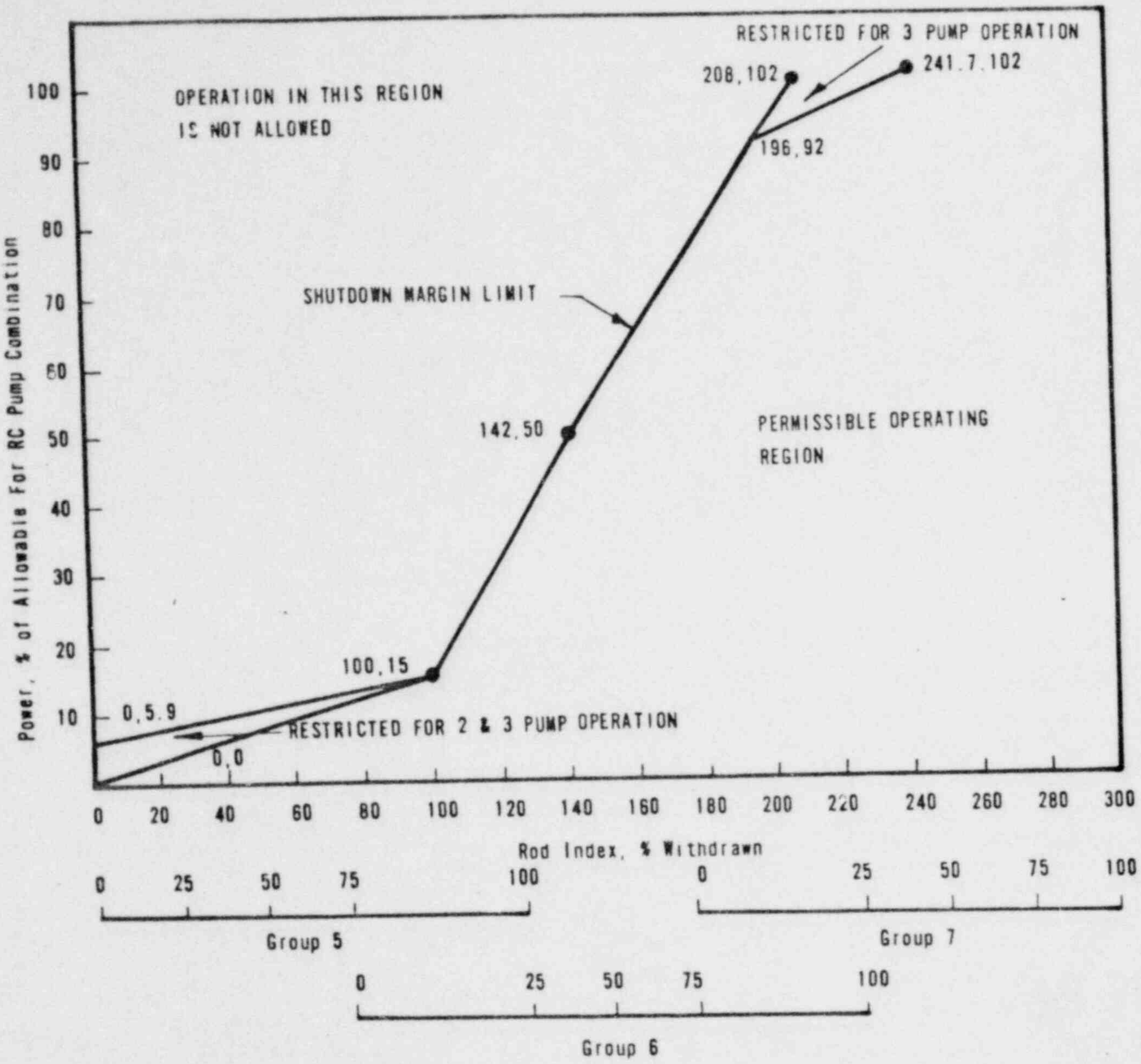
Figure 3.5.2-2C1



ROD POSITION LIMITS FOR TWO- AND THREE-PUMP OPERATION FROM 100 (± 10) TO 235 (± 10) EFPD UNIT 3

OCONEE NUCLEAR STATION

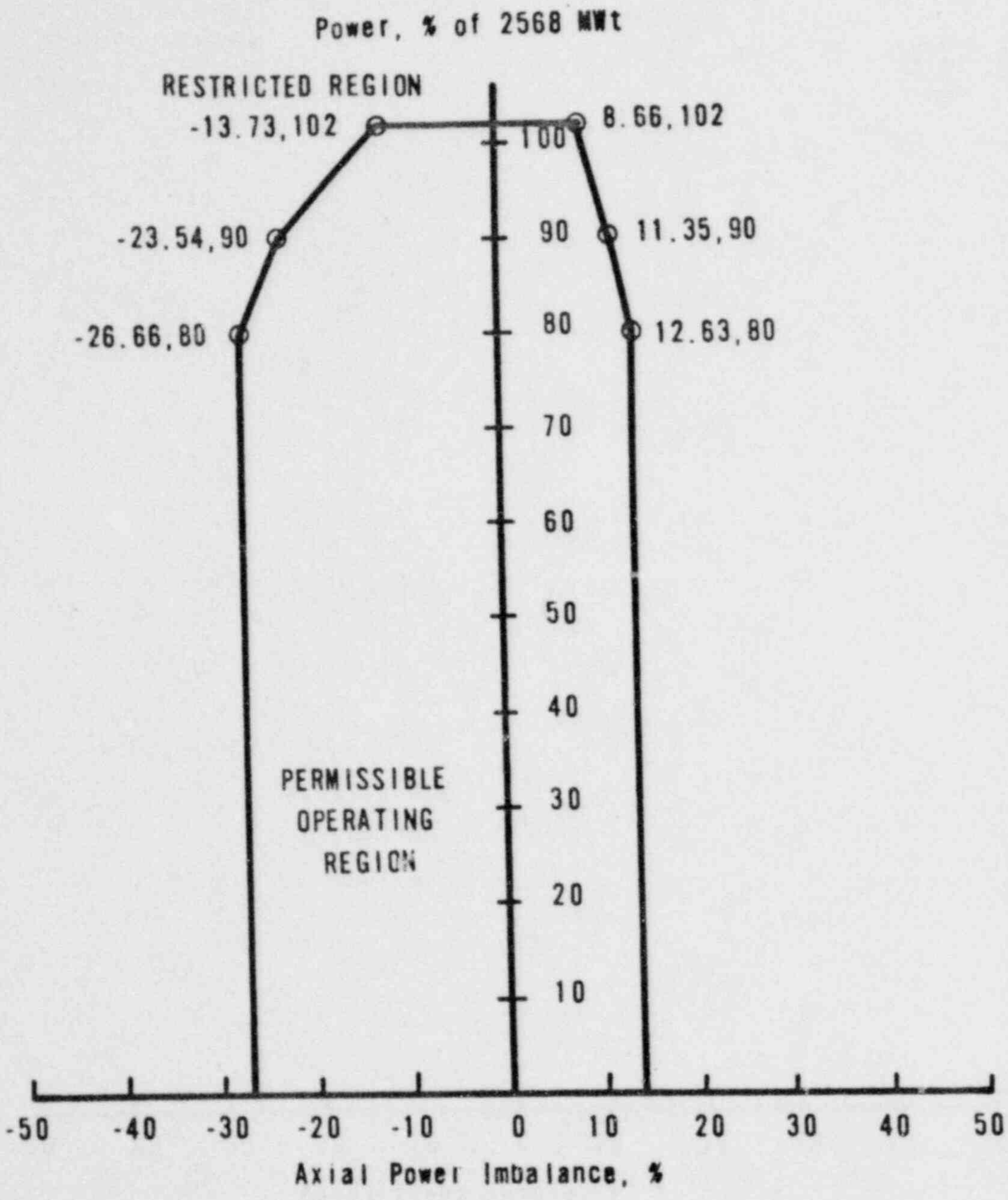
Figure 3.5.2-2C2



ROD POSITION LIMITS FOR TWO-
AND THREE-PUMP OPERATION
AFTER 235 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

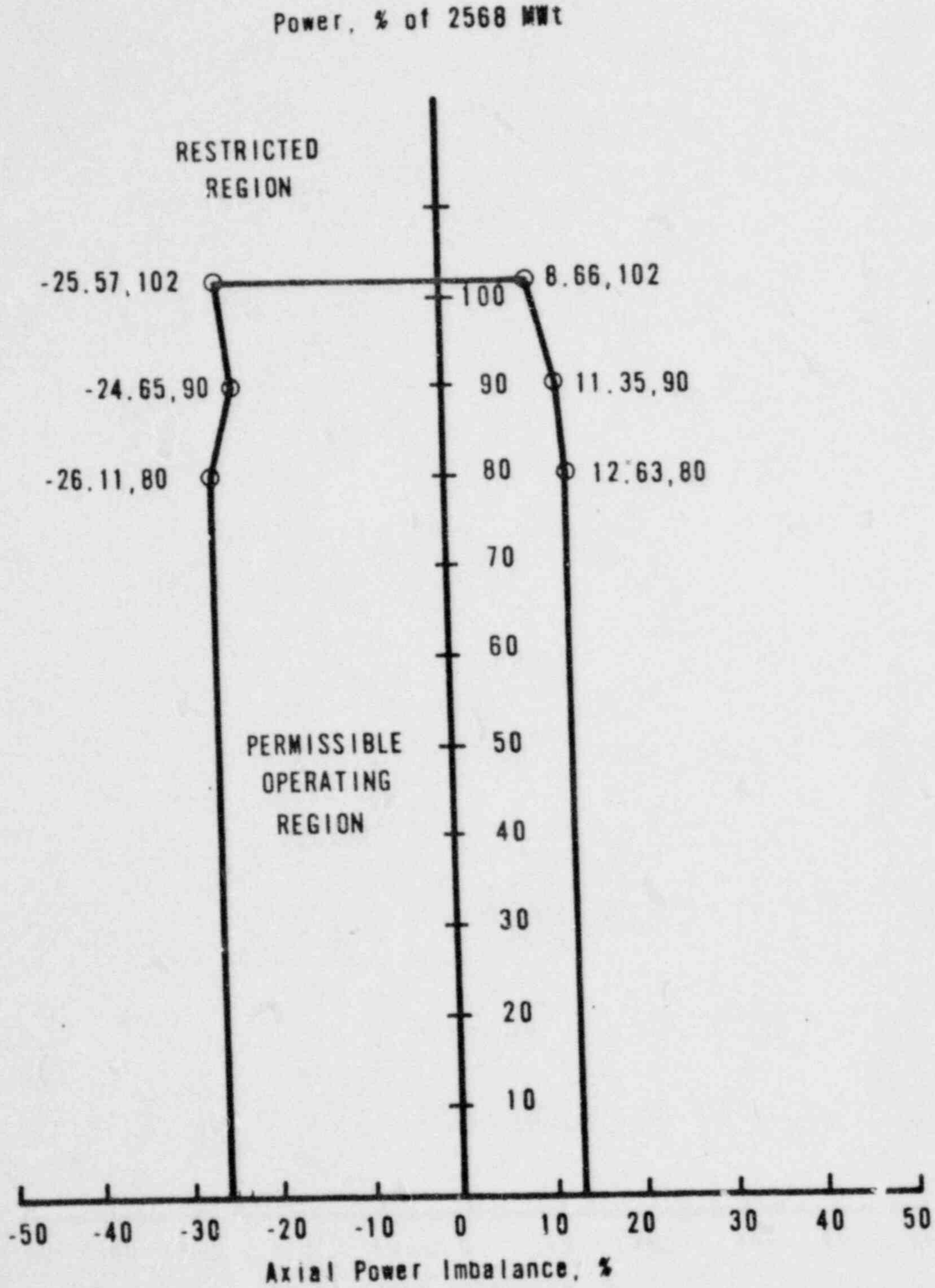
Figure 3.5.2-2C3



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
0 TO 100 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

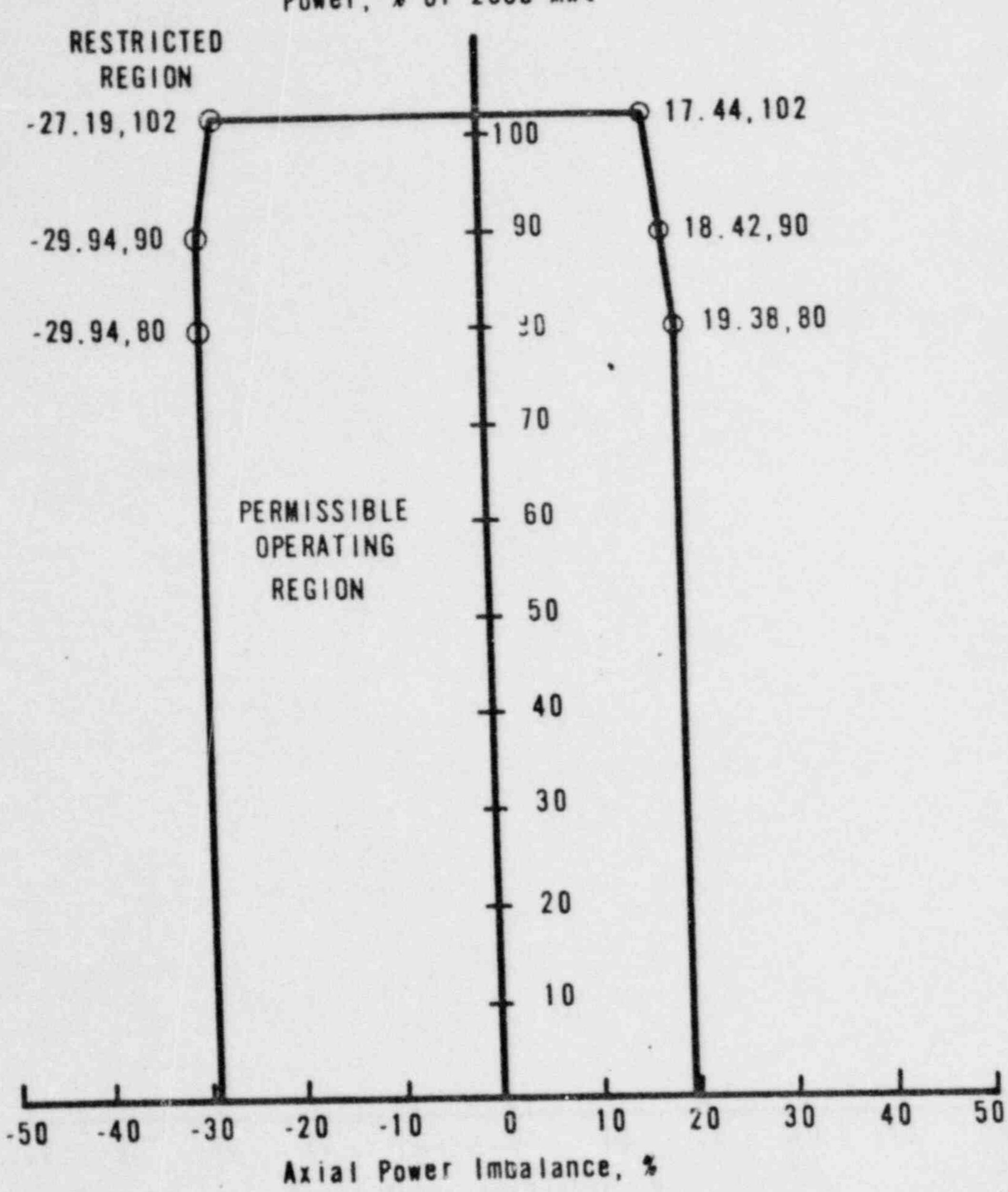
Figure 3.5.2-3C1



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
100 (+) 10 EFPD TO 235 (± 10)
EFPD UNIT 3

OCONEE NUCLEAR STATION

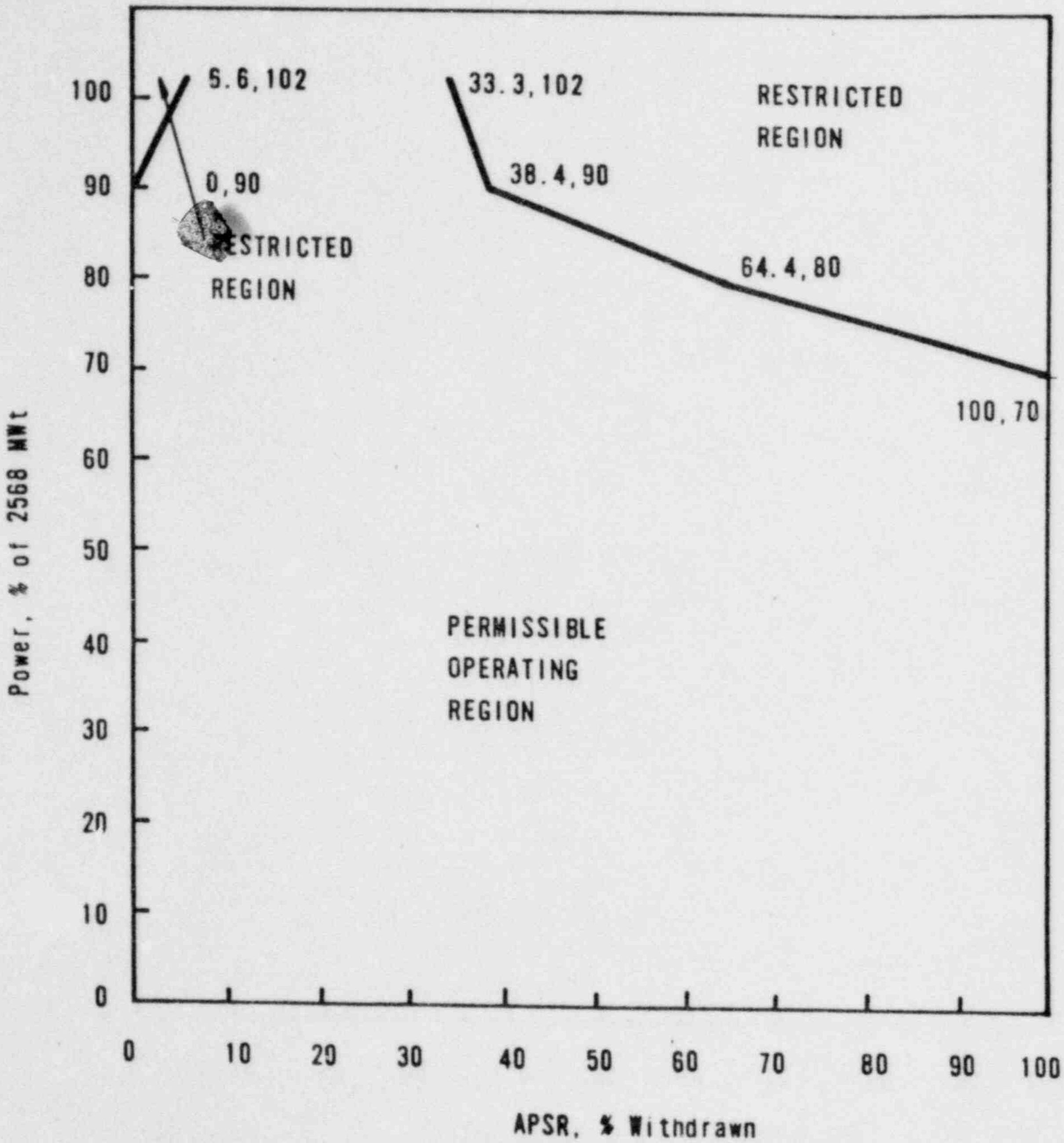
Figure 3.5.2-3C2



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION
AFTER 235 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-3C3

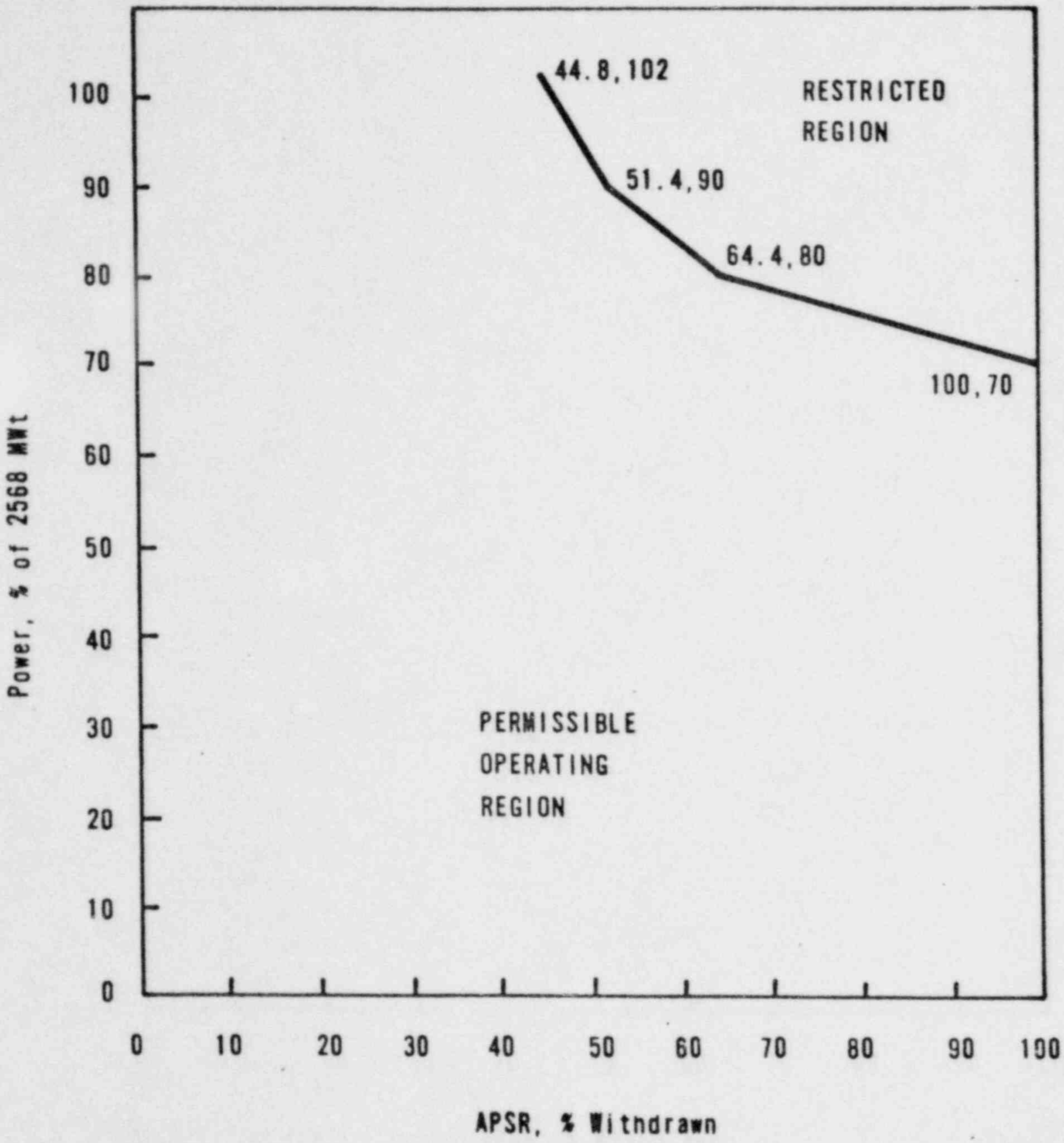


APSR POSITION LIMITS FOR
OPERATION FROM 0 TO 100 ± 10
EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-4C1

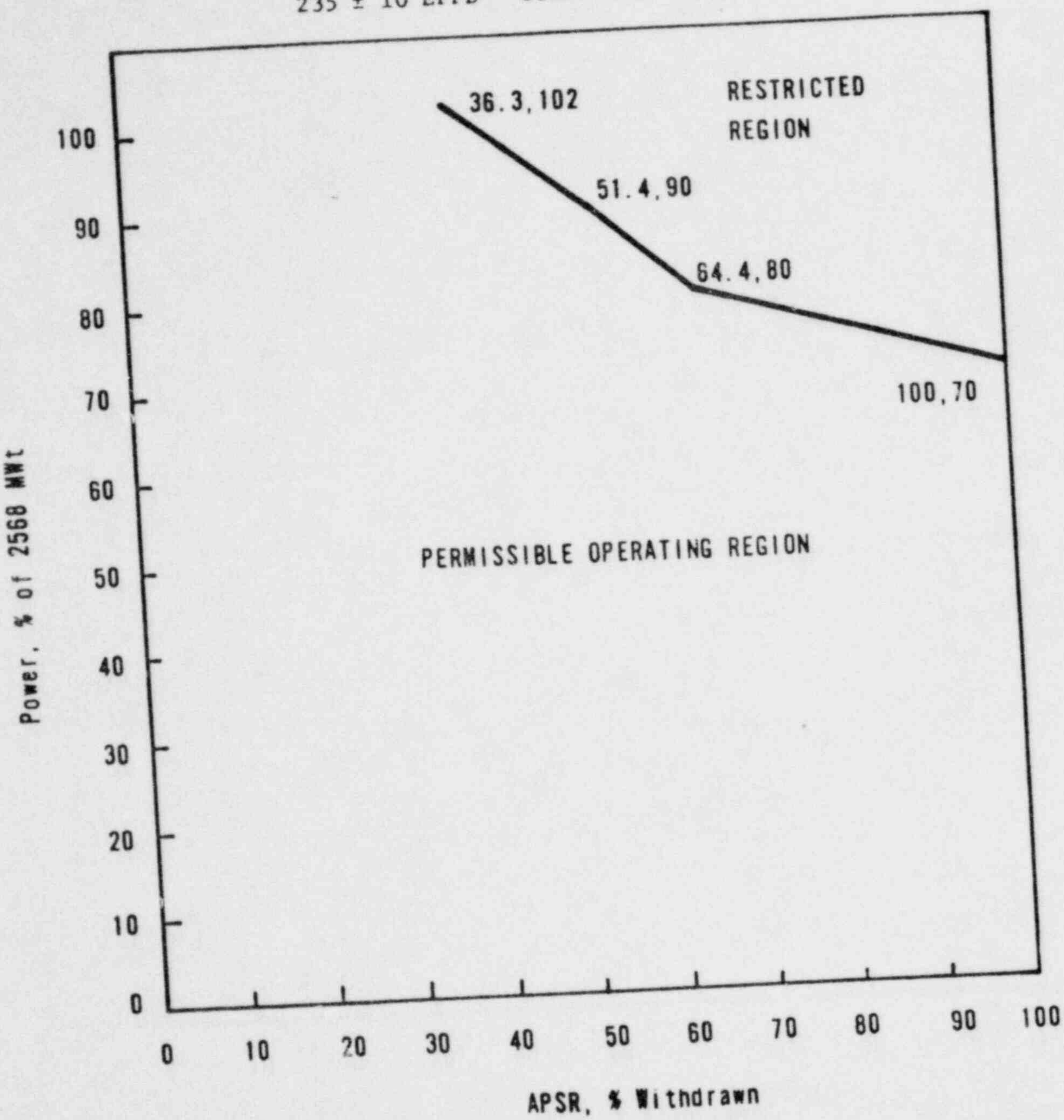
Figure 8-16. APSR Position Limits for Operation From 100 ± 10 to 235 ± 10 EFPD — Oconee 3, Cycle 3



APSR POSITION LIMITS FOR OPERATION FROM 100 ± 10 TO 235 ± 10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 8-17. APSR Position Limits for Operation After 235 ± 10 EFPD — Oconee 3, Cycle 3



APSR POSITION LIMITS FOR OPERATION AFTER 235 $\pm$ 10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-4C3