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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.45
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated May 6, 1977, as supplemented June 21 and July 11, 1977, and application dated March 1, 1977, as supplemented May 5, 1977, comply 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 3.B of Facility License No. DPR-38 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 29, 1977

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated May 6, 1977, as supplemented June 21 and July 11, 1977, and application dated March 1, 1977, as supplemented May 5, 1977, comply 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 3.B of Facility License No. DPR-47 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 29, 1977

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated May 6, 1977, as supplemented June 21 and July 11, 1977, and application dated March 1, 1977, as supplemented May 5, 1977, comply 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 3.B of Facility License No. DPR-55 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 42, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 29, 1977

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 45 TO DPR-38

AMENDMENT NO. 45 TO DPR-47

AMENDMENT NO. 42 TO DPR-55

DOCKETS NOS. 50-269, 50-270 AND 50-287

Revise Appendix A as follows:

Remove the following pages and insert revised identically numbered pages.

2.1-3a
2.1-3b
2.1-5
2.1-8
2.1-11
2.3-3
2.3-4
2.3-6
2.3-9
2.3-12
3.5-8
3.5-9
3.5-10
3.5-11
3.5-14
3.5-14a
3.5-15
3.5-19
3.5-19a
3.5-19b
3.5-22
3.5-22a
3.5-22b
3.5-23f
3.5-23g
3.5-23h
4.1-9
4.2-3
4.20-1

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 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 2.1-1B 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 densitication 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 2.1-2B 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 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 19.8 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.

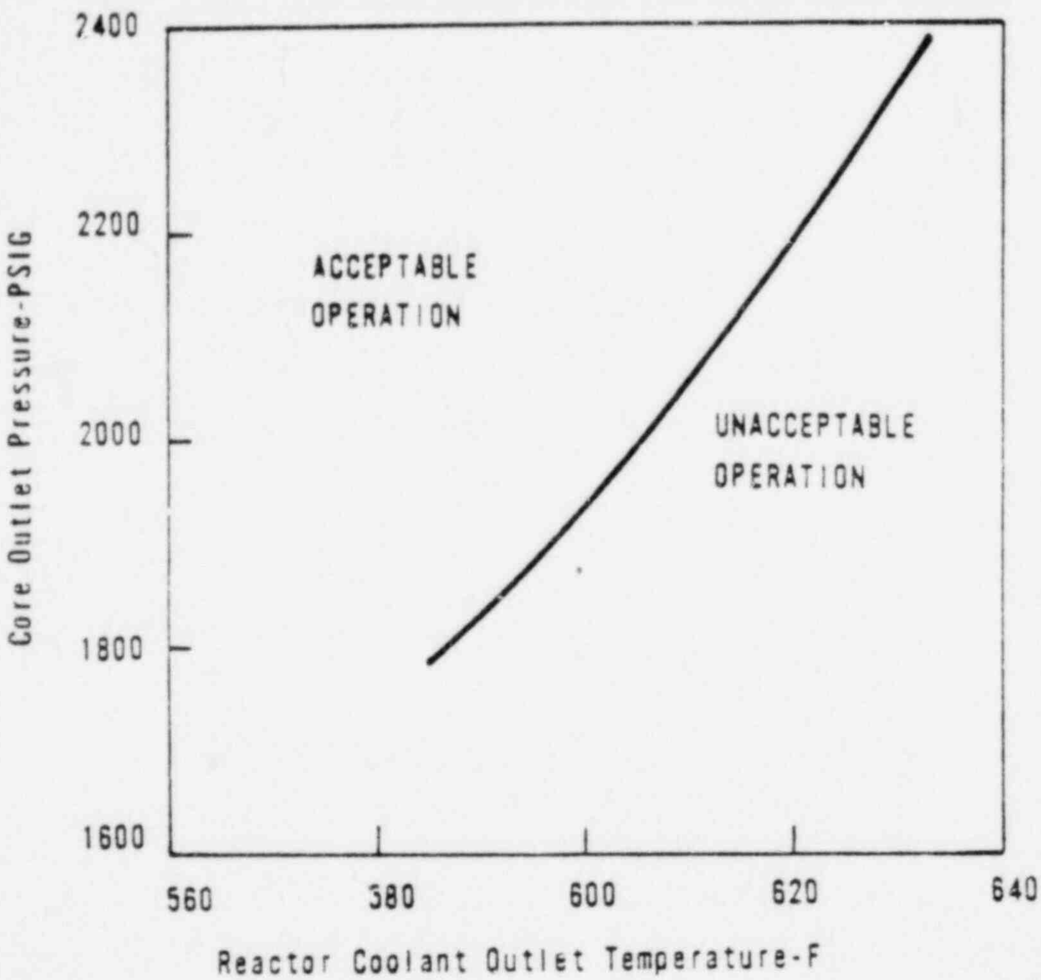
The maximum thermal power for three-pump operation is 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055= 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar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

For each curve of Figure 2.1-3B, 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.

<u>References</u>

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

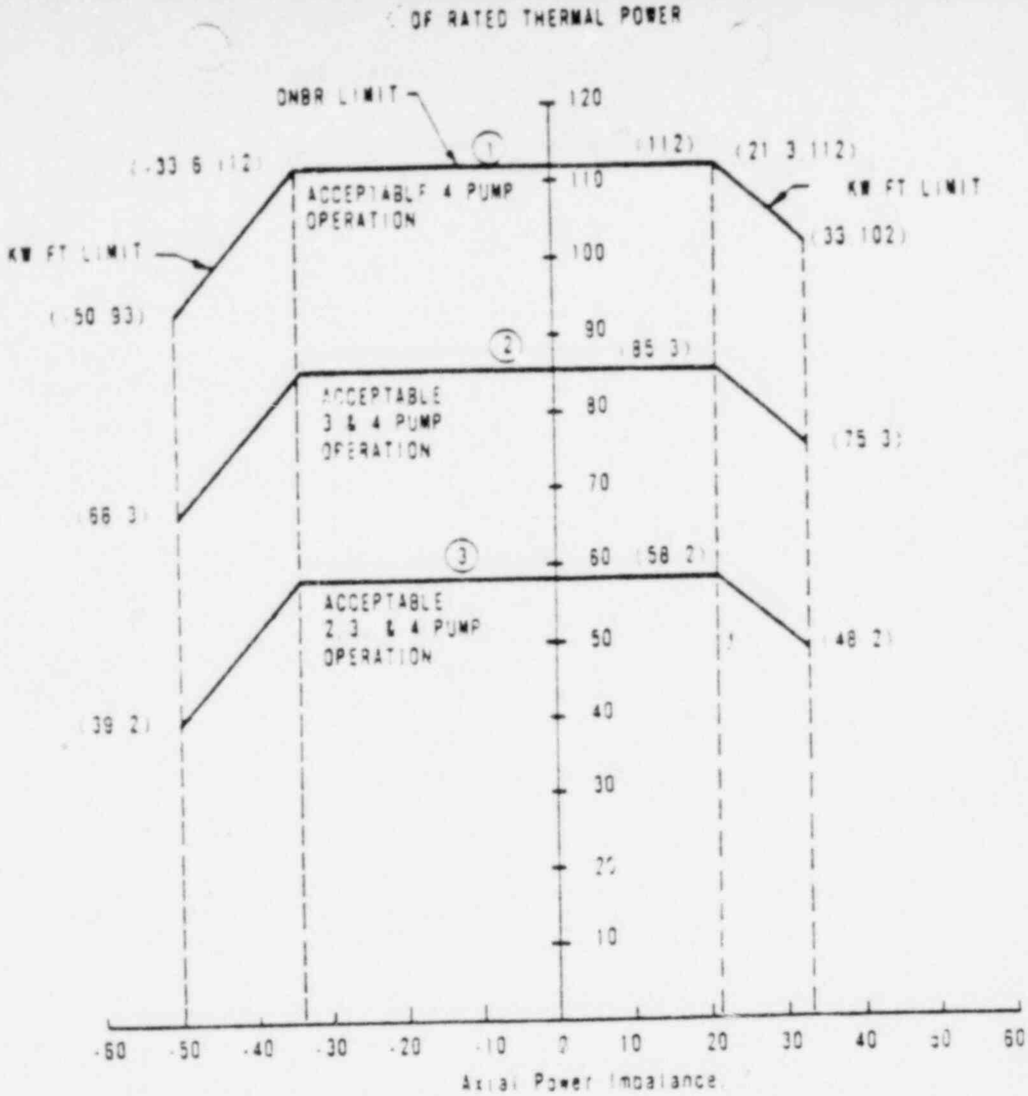
(2) Oconee 2, Cycle - Reload Report - BAW-1452, April, 1977.



CORE PROTECTION SAFETY LIMITS
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.1-1B



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

UNIT 2
CORE PROTECTION SAFETY LIMITS

OCONEE NUCLEAR STATION

Figure 2.1-2B

Amendments 45, 45 & 42

level trip and associated reactor power/reactor power-imbalance boundaries by 1.07% -Unit 1 for a 1% flow reduction.

1.055%-Unit 2

1.07%-Unit 3

For Unit 2, the power-to-flow reduction ratio is 0.949, and for Units 1 and 3 the power-to-flow reduction factor is 0.961 during single loop operation.

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 ($10.79\ T_{out}-4539$)

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 ($11.14\ T_{out}-4746$)

($11.14\ T_{out}-4746$)

($10.79\ T_{out}-4579$)

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.

POOR ORIGINAL

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 ≤ 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 ≤ 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

A. Two Loop 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%.

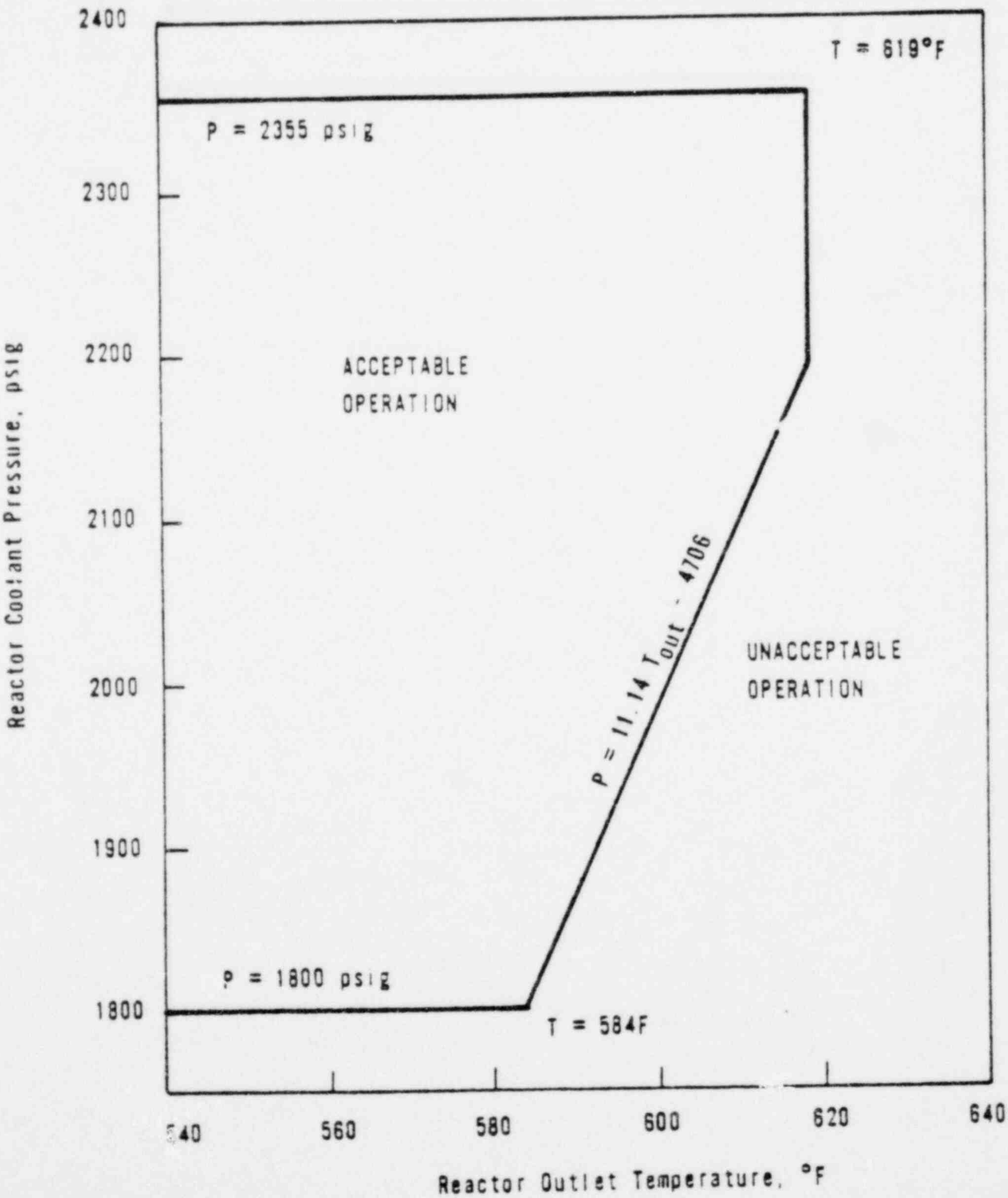
B. Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped. After the pump contact monitor trip has occurred, the following actions will permit single loop operation:

1. Reset the pump contact monitor power level trip setpoint to 55.0%.
2. Trip one of the two protective channels receiving outlet temperature information from sensors in the Idle Loop.
3. Reset flux-flow setpoint to 0.961 (Unit 1)
   0.949 (Unit 2)
   0.961 (Unit 3)

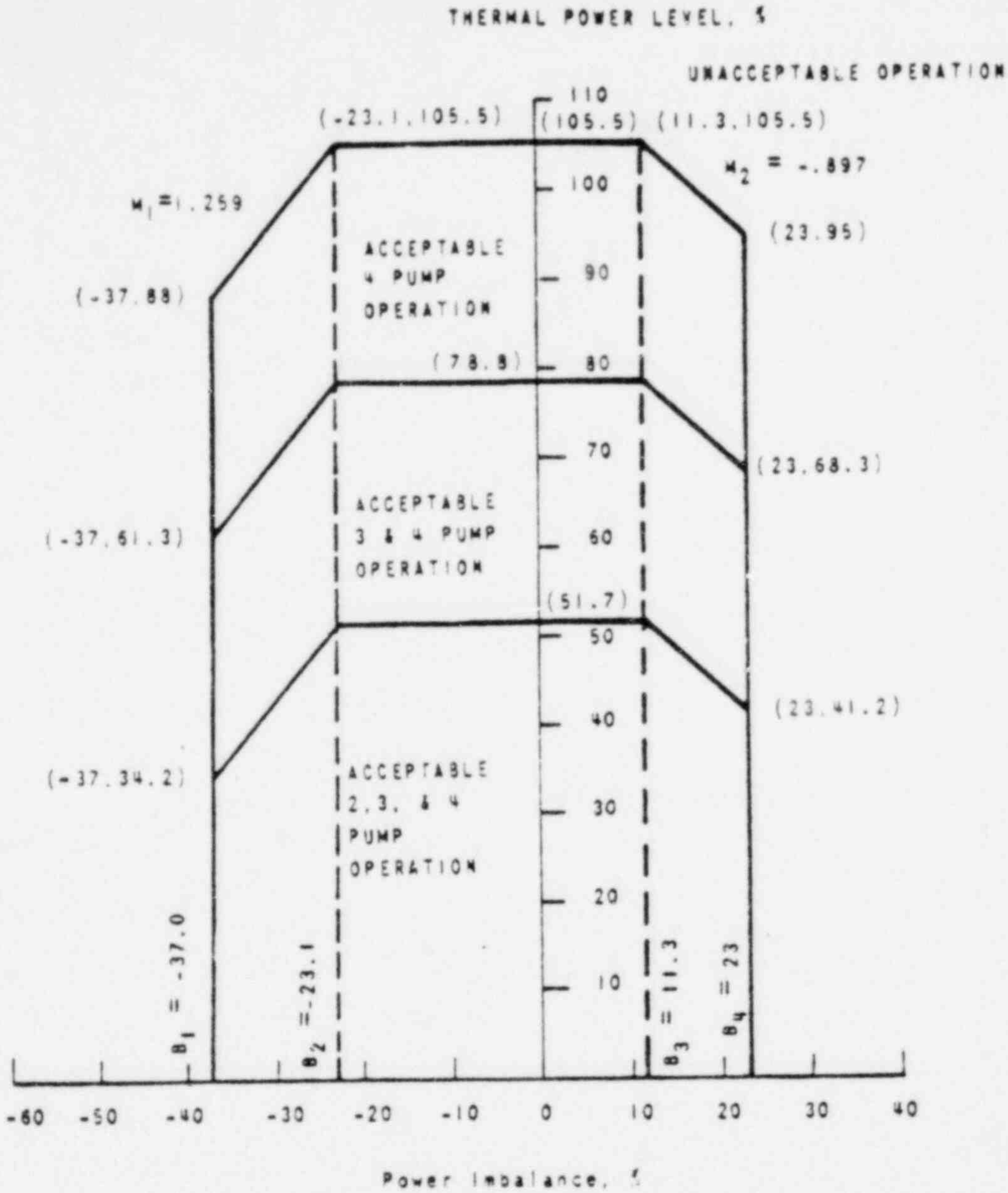
REFERENCES

(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

OCONEE NUCLEAR STATION



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 2

OCONEE NUCLEAR STATION
Figure 2.3-2B

(3) Except as provided in specification 3.5.2.4.b, the reactor shall be brought to the hot shutdown condition within four hours if the quadrant power tilt is not reduced to less than
3.41% Unit 1 within 24 hours.
3.41% Unit 2
3.41% Unit 3

b. If the quadrant tilt exceeds +3.41% Unit 1 and there is simultaneous
3.41% Unit 2
3.41% Unit 3
indication of a misaligned control rod per Specification 3.5.2.2, reactor operation may continue provided power is reduced to 60% of the thermal power allowable for the reactor coolant pump combination.

c. Except for physics test, if quadrant tilt exceeds 9.44% Unit 1,
9.44% Unit 2
9.44% Unit 3
a controlled shutdown shall be initiated immediately, and the reactor shall be brought to the hot shutdown condition within four hours.

d. Whenever the reactor is brought to hot shutdown pursuant to 3.5.2.4.a(3) or 3.5.2.4.c above, subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the power range high flux setpoint allowable for the reactor coolant pump combination are restricted by a reduction of 2 percent of full power for each 1 percent tilt for the maximum tilt observed prior to shutdown.

e. Quadrant power tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15 percent of rated 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, 3.5.2-1B2 and 3.5.2-1B3 (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 3.5.2-2B2 and 3.5.2-2B3 (Unit 2); 3.5.2-2C1, 3.5.2-2C2 and 3.5.2-2C3 (Unit 3) for two or three pump opera..on. Also, excepting physics t .ts or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-4B1, 3.5.2-4B2, and 3.5.2-4B3 (Unit 2). 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 3.5.2-1C1, 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, 3.5.3-3B1, 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, 3.5.2-3B1, 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 effects
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing effects

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 0.5% $\Delta k/k$ (Unit 1) or 0.65% $\Delta k/k$ (Units 2 and 3) 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% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.5% $\Delta k/k$ (Unit 1) or 0.65% $\Delta k/k$ (Units 2 and 3) 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 with consideration of potential effects of rod bowing and fuel densification 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

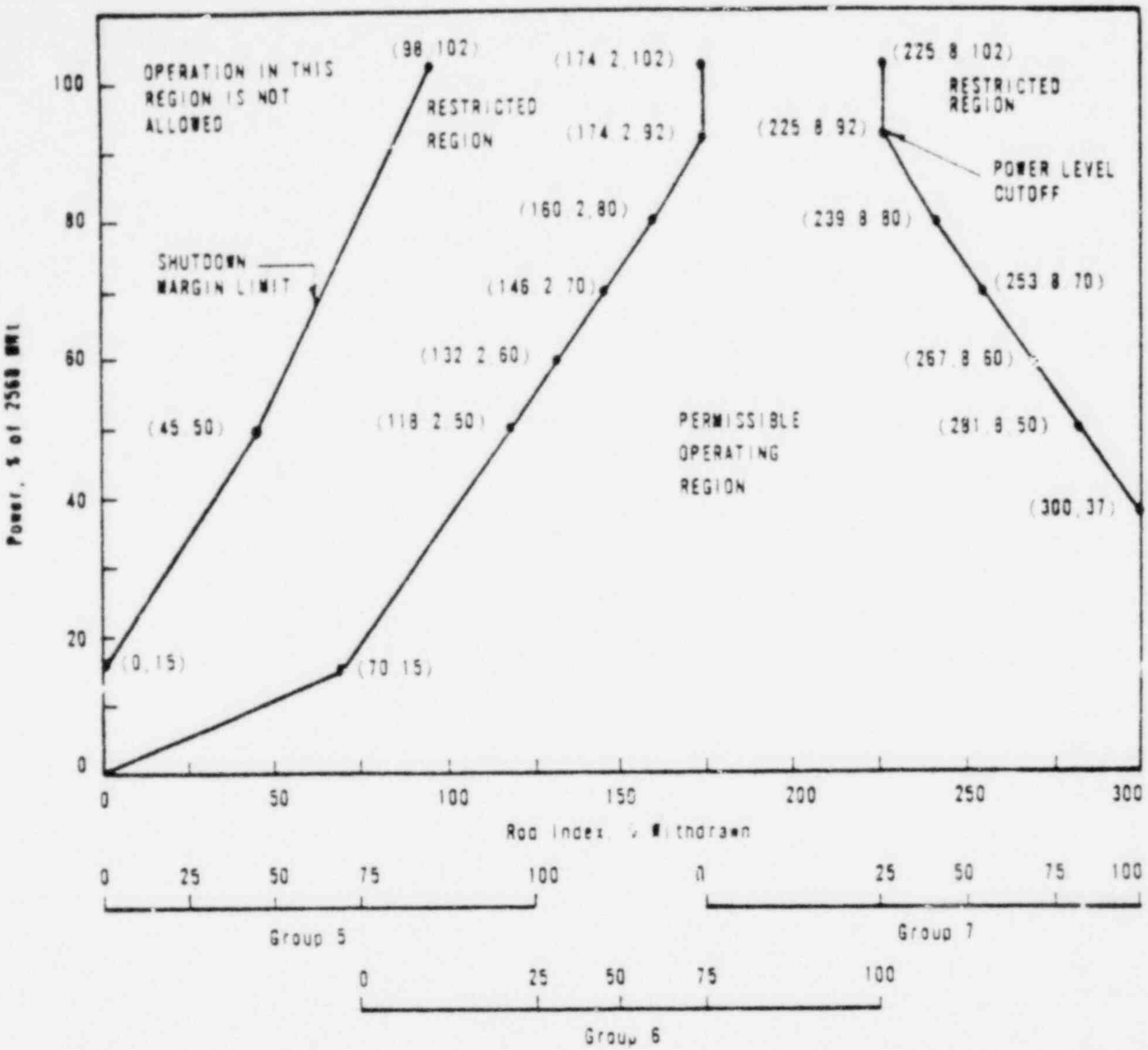
[3] FSAR, SUPPLEMENT 9

[4] B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

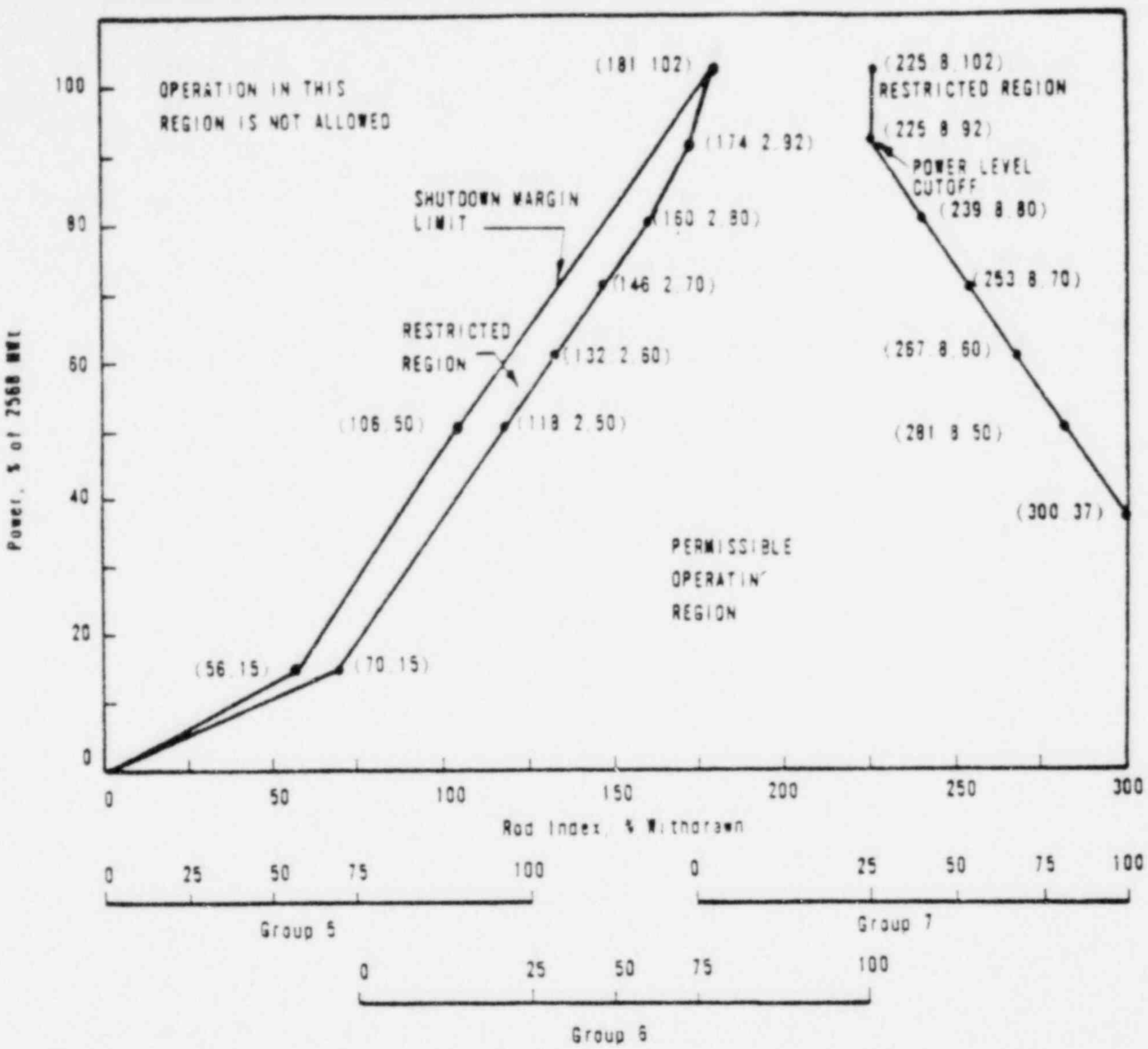
BAW-1400 (UNIT 3)



ROD POSITION LIMITS FOR
4 PUMP OPERATION FROM
0-100 ± 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

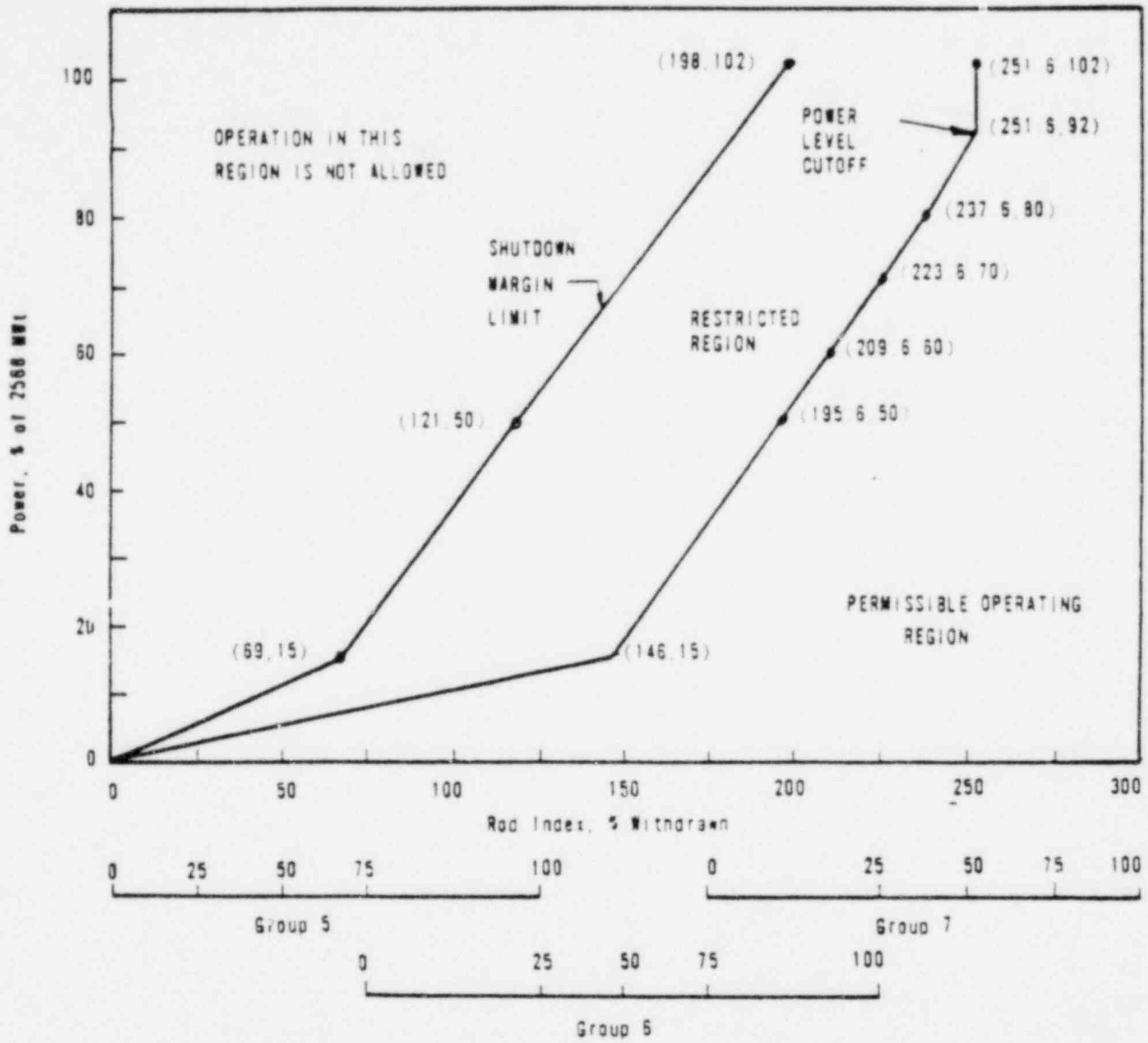
Figure 3.5.2-1B1



ROD POSITION LIMITS FOR
4 PUMP OPERATION FROM
100 ± 10 TO 250 ± 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

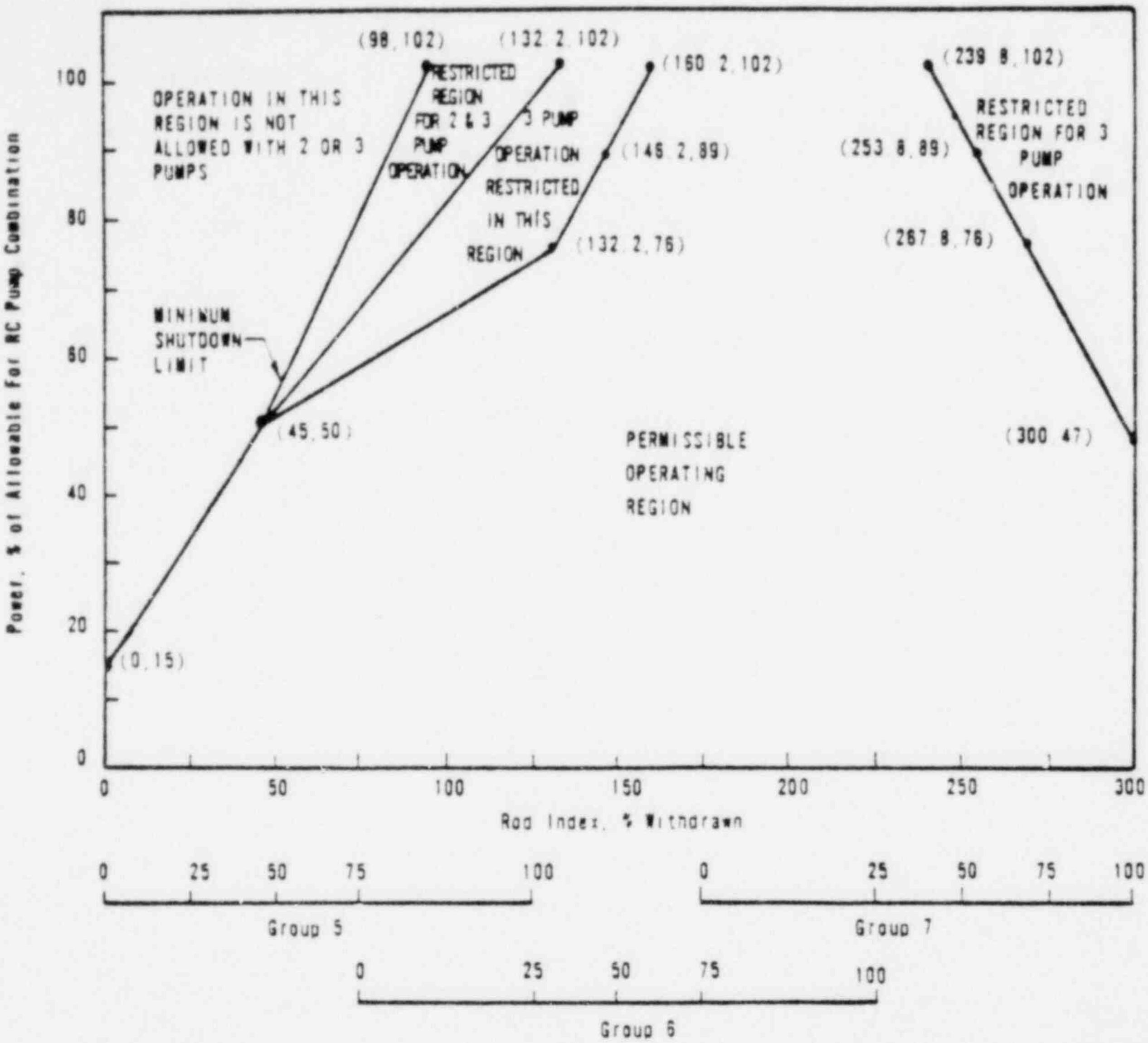
Figure 3.5.2-1B2



ROD POSITION LIMITS
FOR 4 PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 2

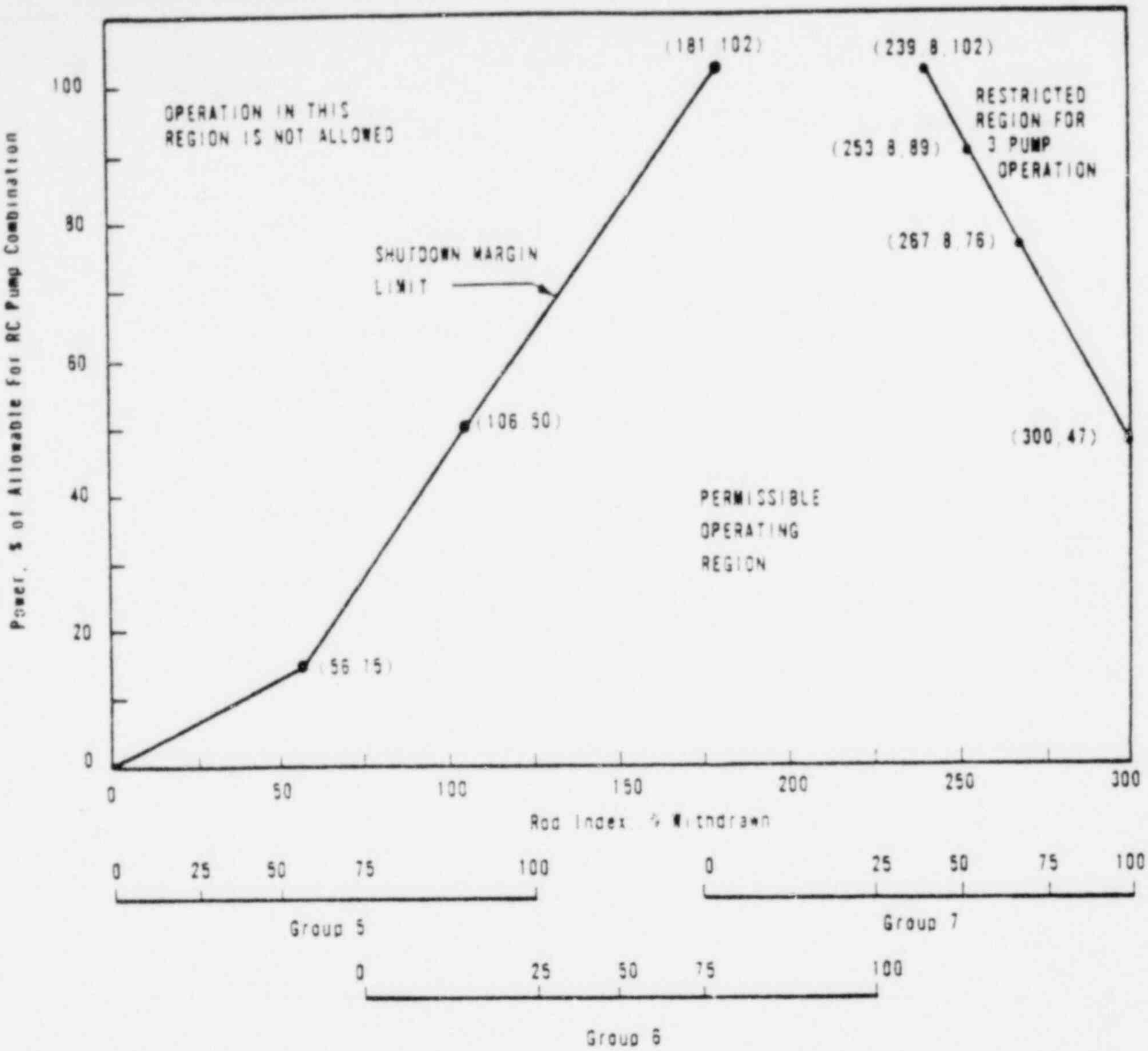
OCONEE NUCLEAR STATION

Figure 3.5.2-1B3



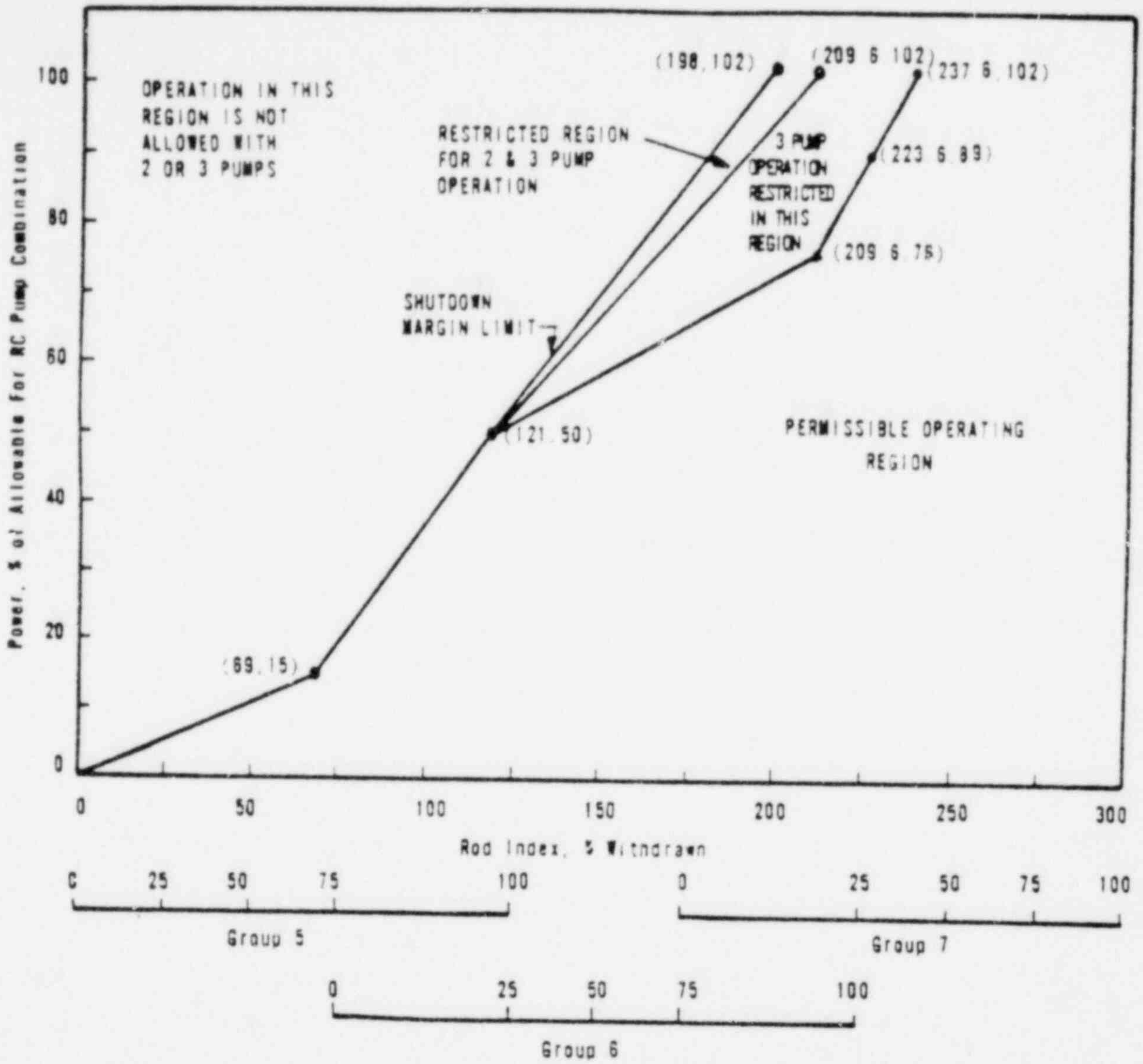
ROD POSITION LIMITS
FOR TWO AND THREE PUMP
OPERATION FROM 0 TO 100 EFPD
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-2B1



ROD POSITION LIMITS FOR TWO
AND THREE PUMP OPERATION
FROM 100 ± 10 TO 250 ± 10 EFPD
UNIT 2

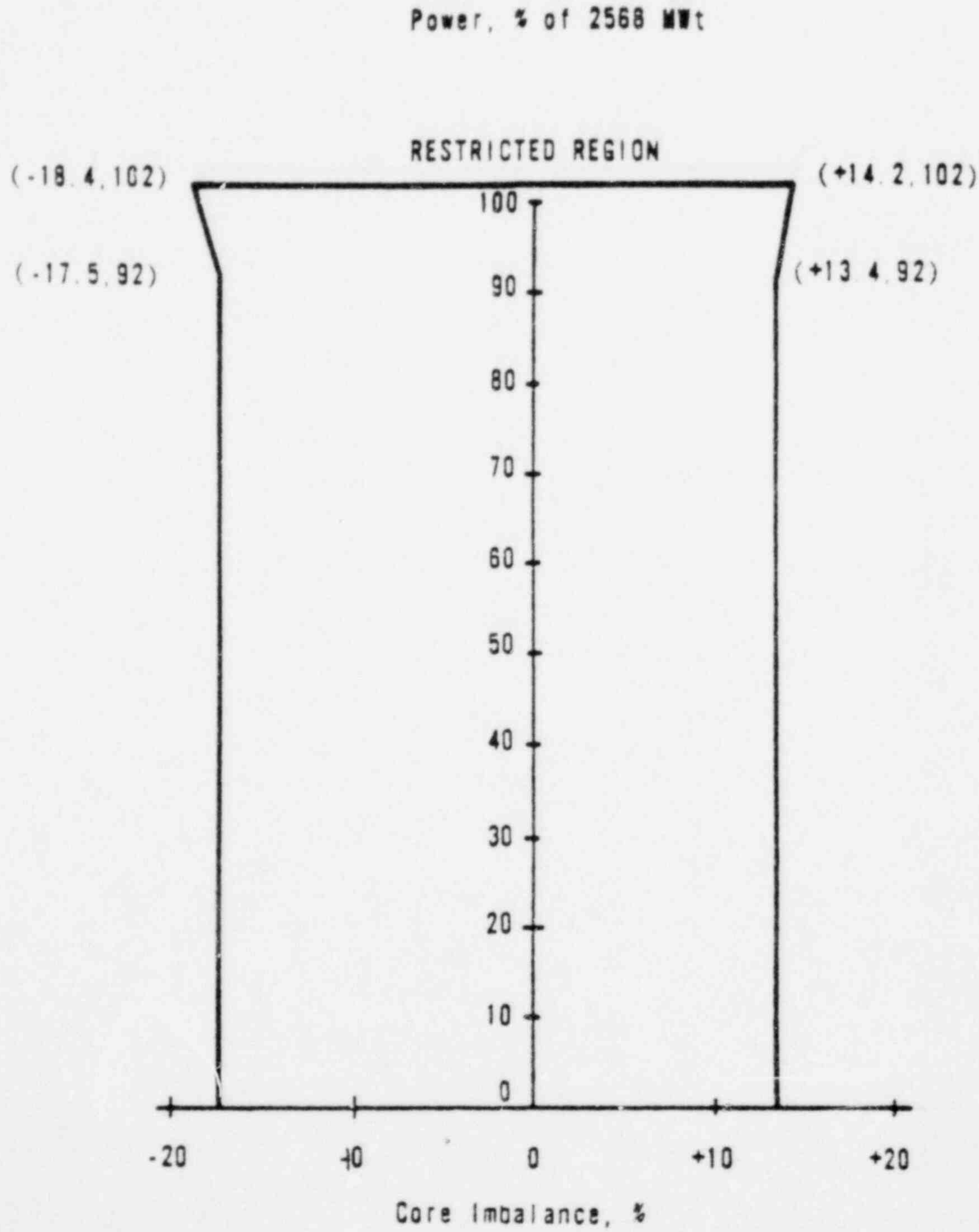
OCONEE NUCLEAR STATION
Figure 3.5.2-2B2



ROD POSITION LIMITS FOR TWO
AND THREE PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 2

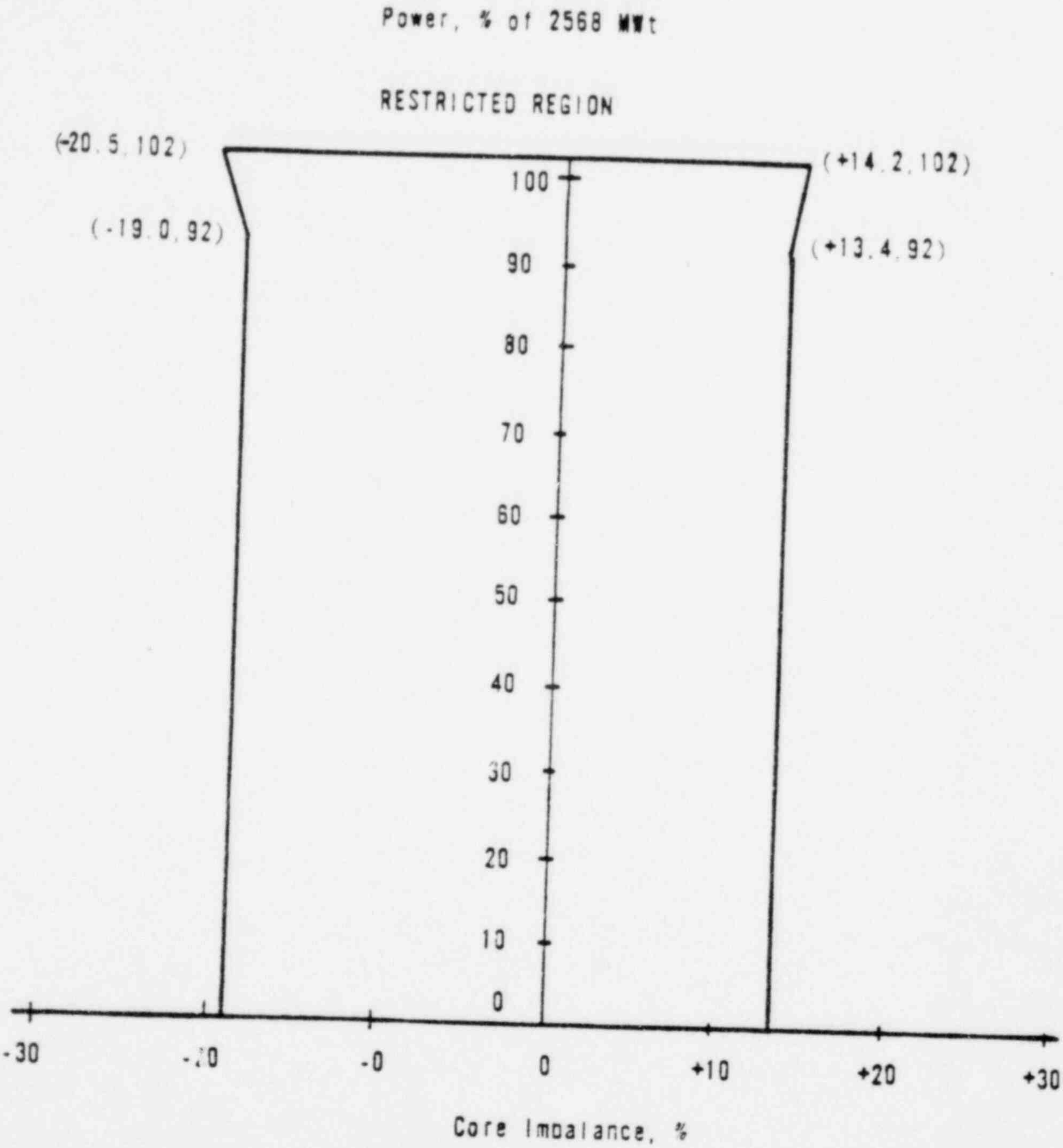
OCONEE NUCLEAR STATION

Figure 3.5.2-2B3



OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 TO 100 ± 10 EFPD, UNIT 2

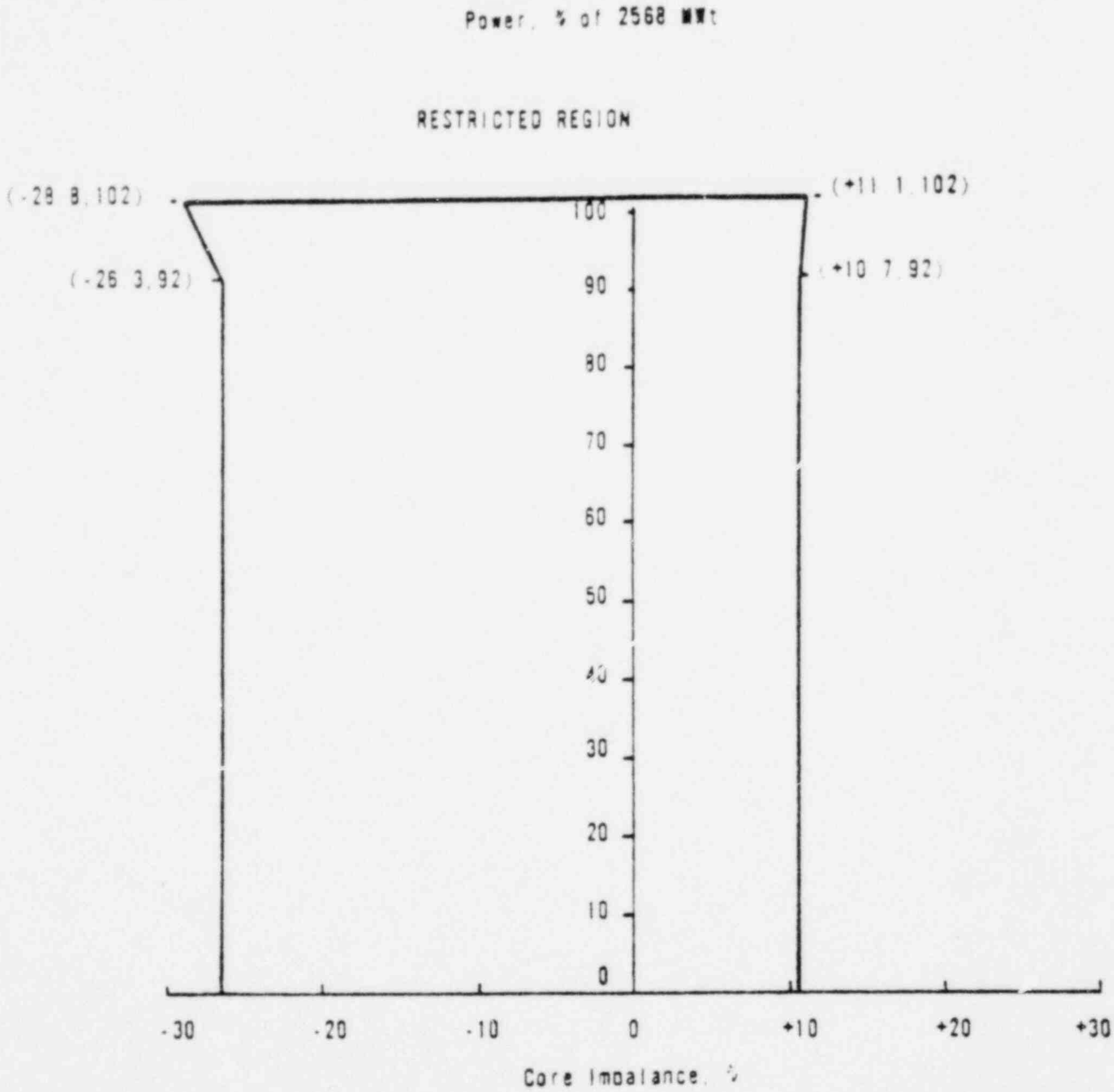
OCONEE NUCLEAR STATION



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
100 ± 10 TO 250 ± 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

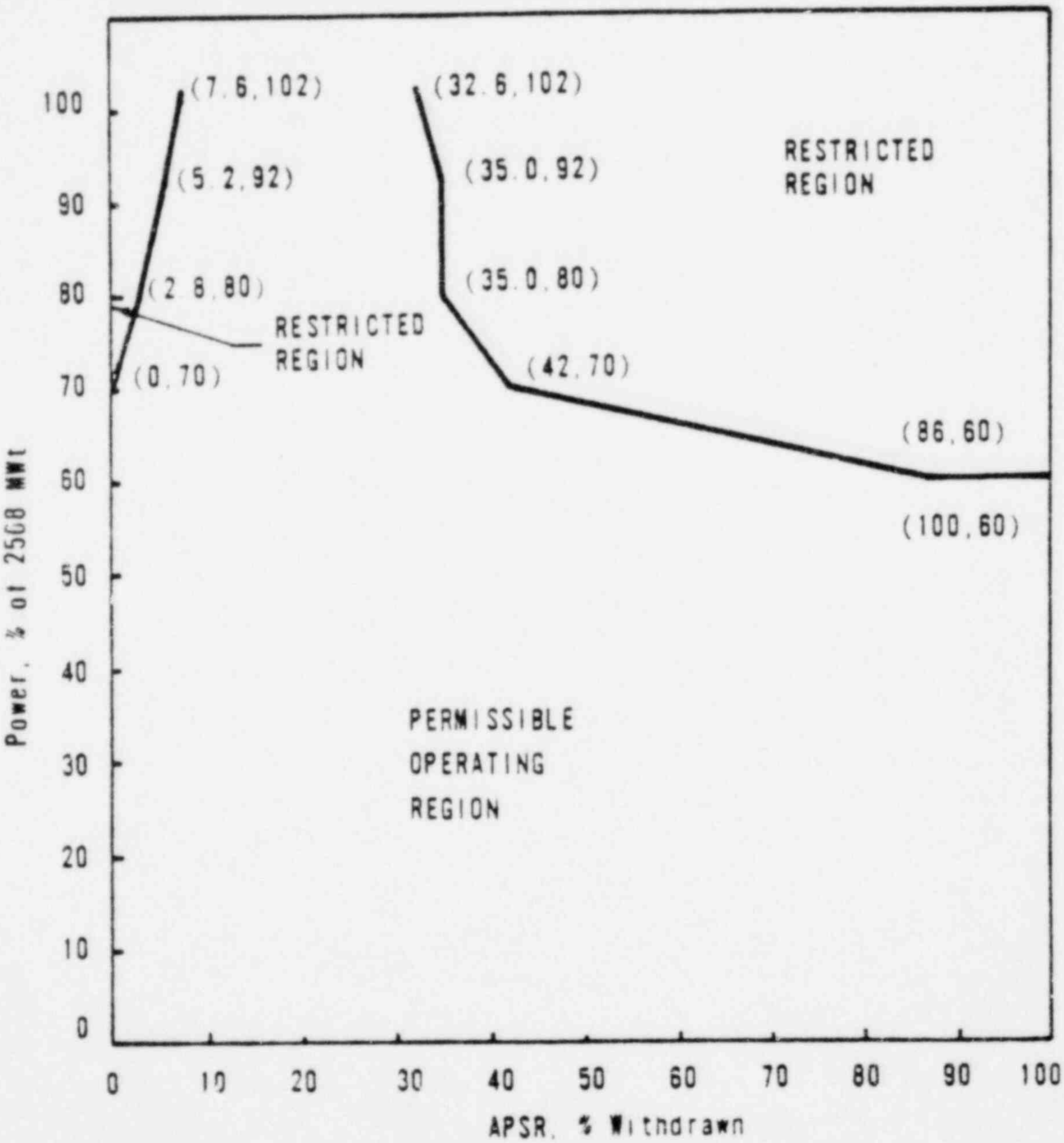
Figure 3.5.2-3B2



OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION AFTER 250 ± 10 EFPD, UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-3B3

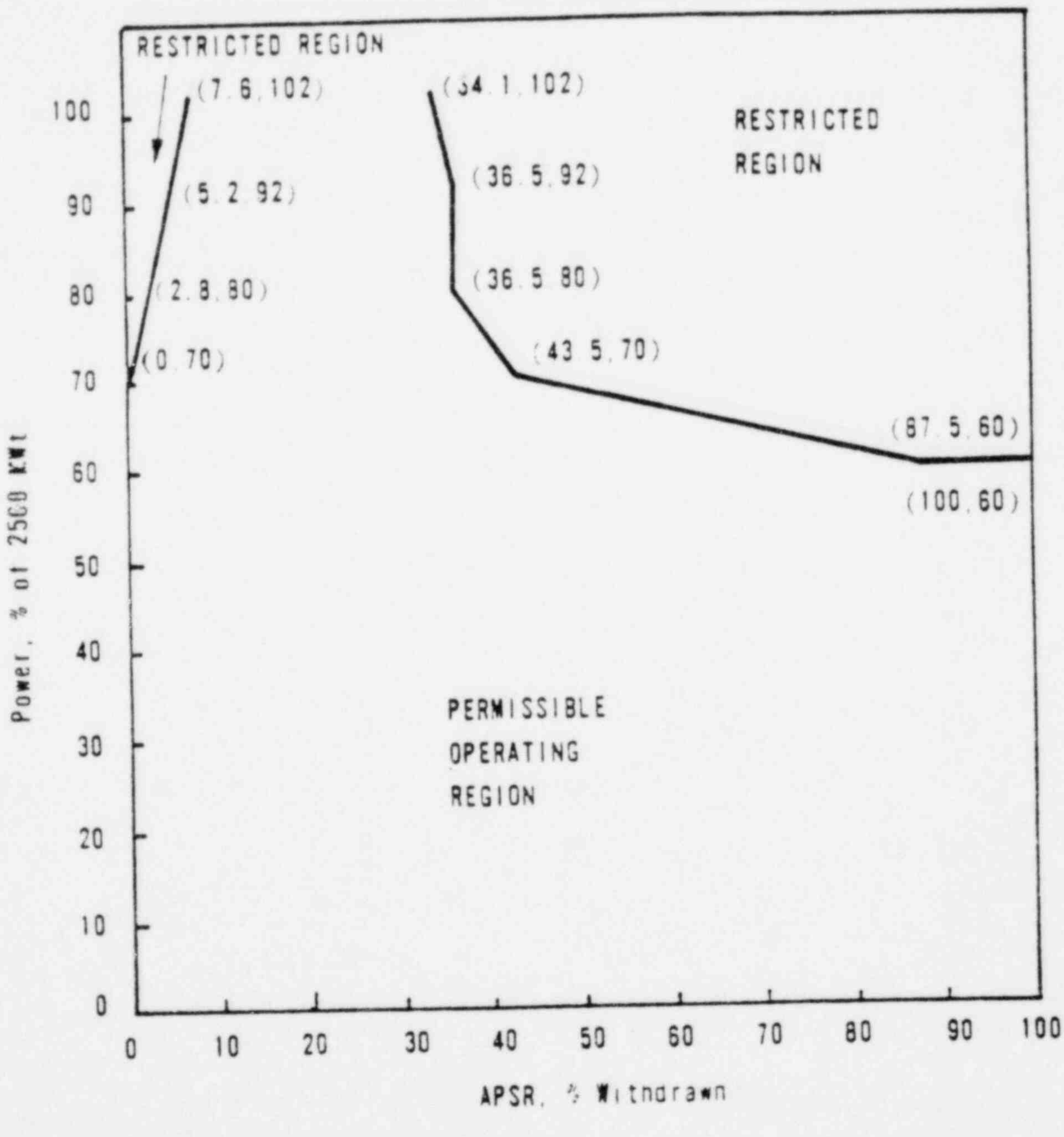


APSR POSITION LIMITS FOR OPERATION FROM 0 TO 100 ± 10 EFPD, UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-4B1

Amendments 45, 45 & 42

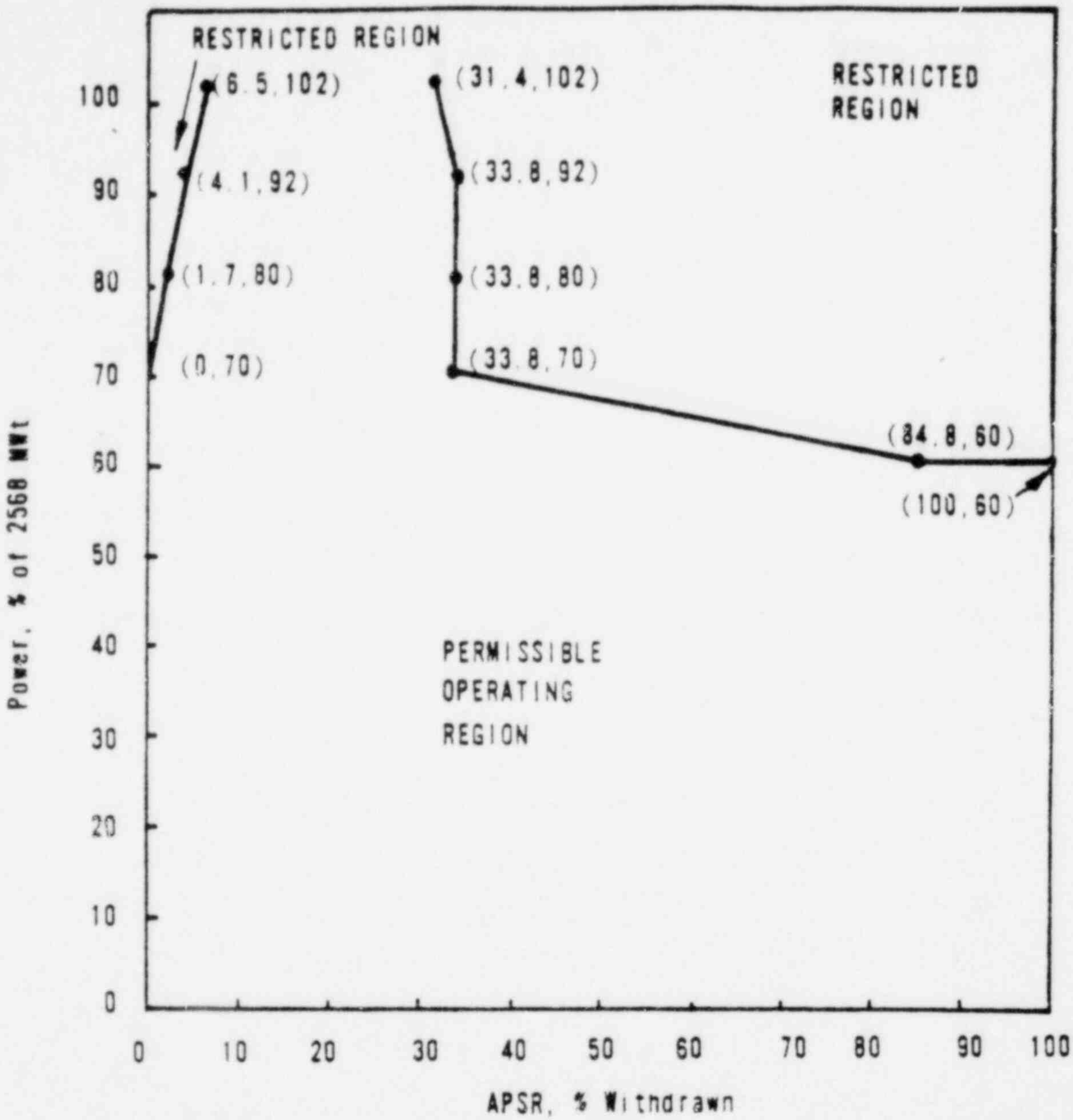


APSR POSITION LIMITS FOR OPERATION FROM 100 ± 10 TO 250 ± 10 EFPD, UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-4B2

Amendments 45, 45 & 42



APSR POSITION LIMITS FOR
OPERATION AFTER 250 ± 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-4B3

4.2.10 The licensee shall submit a report or application for license amendment to the NRC within 90 days after the occurrence of the following: After March 13, 1978, any time that Crystal River Unit No. 3 fails to maintain a cumulative reactor utilization factor of greater than 45%.

The report shall provide justification for continued operation of Oconee Nuclear Station Units 1, 2 and 3 with the reactor vessel surveillance program conducted at Crystal River Unit No. 3 or the application for license amendment shall propose an alternative program for conduct of the reactor vessel surveillance program.

4.2.11 During the first two refueling periods, two reactor coolant system piping elbows shall be ultrasonically inspected along their longitudinal welds (4 inches beyond each side) for clad bonding and for cracks in both the clad and base metal. The elbows to be inspected are identified in B&W Report 1364 dated December 1970.

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, 1970, including 1970 winter addenda, edition. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The number of reactor vessel specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

For the purpose of Technical Specification 4.2.10. Cumulative reactor utilization factor is defined as: [(Cumulative thermal megawatt hours since attainment of commerical operation at 100% power) x 100] ÷ [(licensed thermal power) x (cumulative hours since attainment of commercial operation at 100% power)]. The definition of Regulatory Guide 1.16, Revision 4 (August 1975) applies for the term "commerical operation".

Early inspection of Reactor Coolant System piping elbows is considered desirable in order to reconfirm the integrity of the carbon steel base metal when explosively clad with sensitized stainless steel. If no degradation is observed during the two annual inspections, surveillance requirements will revert to Section XI of the ASME Boiler and Pressure Vessel Code.

4.20 REACTOR VESSEL INTERNALS VENT VALVES

Applicability

Applies to reactor vessel internals vent valves used to prevent vapor lock in the reactor vessel following a postulated reactor coolant inlet pipe rupture.

Objective

To verify that the reactor vessel internals vent valves operate as required.

Specification

At least once each refueling cycle, each reactor vessel internals vent valve shall be demonstrated operable by:

a. Conducting a remote visual inspection of visually accessible surfaces of the value body and disc sealing faces and evaluating any observed surface irregularities.

b. verifying that the value is not stuck in an open position, and

c. Verifying that the valve can be fully opened with force equivalent to or less than 1.00 psid.

Bases

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves (1) assures operability, (2) assures that the valves are not open during normal operation, and (3) demonstrates that the valves are fully open at the forces equivalent to the differential pressures justifiable by the ECCS analysis.