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Dockets Nos. 50-269, 50-270
and 50-287

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US NUCLEAR REGULATORY COMMISSION
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Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 111 , 111, and 108 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated November 13, 1981, as supplemented by letter dated March 24, 1982.

These amendments revise the TSs to support full power operation of Oconee Unit 2 during fuel Cycle 6.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Philip C. Wagner, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 111 to DPR-38
2. Amendment No. 111 to DPR-47
3. Amendment No. 108 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

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OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	~~AD:OR:DL~~	OELD Amend. & FR Notice only	
SURNAME	RIngram	PWagner:cf	JStolz	TNovak	R. Rawson	
DATE	3/31/82	4/1/82	4/1/82	4/5/82	4/5/82	

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Duke Power Company

cc w/enclosure(s):

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11/13/81

Office of Intergovernmental Relations
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Raleigh, North Carolina 27603

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. 111
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 November 13, 1981, 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 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 111 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 8, 1982

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. 111
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 November 13, 1981, 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 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.111 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 8, 1982

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. 108
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 November 13, 1981, 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 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 108 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: April 8, 1982

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 111 TO DPR-38

AMENDMENT NO. 111 TO DPR-47

AMENDMENT NO. 108 TO DPR-55

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

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

REMOVE PAGES	INSERT PAGES
2.1-2	2.1-2
2.1-3	2.1-3
2.1-3b	2.1-3b
2.1-8	2.1-8
2.1-11	2.1-11
2.3-9	2.3-9
2.3-11	2.3-11
3.2-2	3.2-2
3.5-9	3.5-9
3.5-10	3.5-10
3.5-16	3.5-16
3.5-16a	3.5-16a
——	3.5-16b
3.5-19	3.5-19
3.5-19a	3.5-19a
——	3.5-19b
——	3.5-19c
——	3.5-19d
——	3.5-19e

REMOVE PAGES	INSERT PAGES
3.5-22	3.5-22
——	3.5-22a
——	3.5-22b
3.5-25	3.5-25
3.5-25a	3.5-25a
——	3.5-25b

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 131.3×10^6 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 magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup independent 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 based on a minimum DNBR criteria of 1.30 plus the amount necessary to offset the reduction in DNBR due to fuel rod bow. (3)

The maximum thermal power for three-pump operation is 89.899 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.929 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions is produced in a similar manner.

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

References

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

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

(3) Oconee 1, Cycle 7 - Reload Report - BAW-1660, March, 1981

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 magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup independent 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 based on a minimum DNBR critieria of 1.30 plus the amount necessary to offset the reduction in DNBR due to fuel rod bow. (3)

The maximum thermal power for three-pump operation is 90.606 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 is produced in a similar manner.

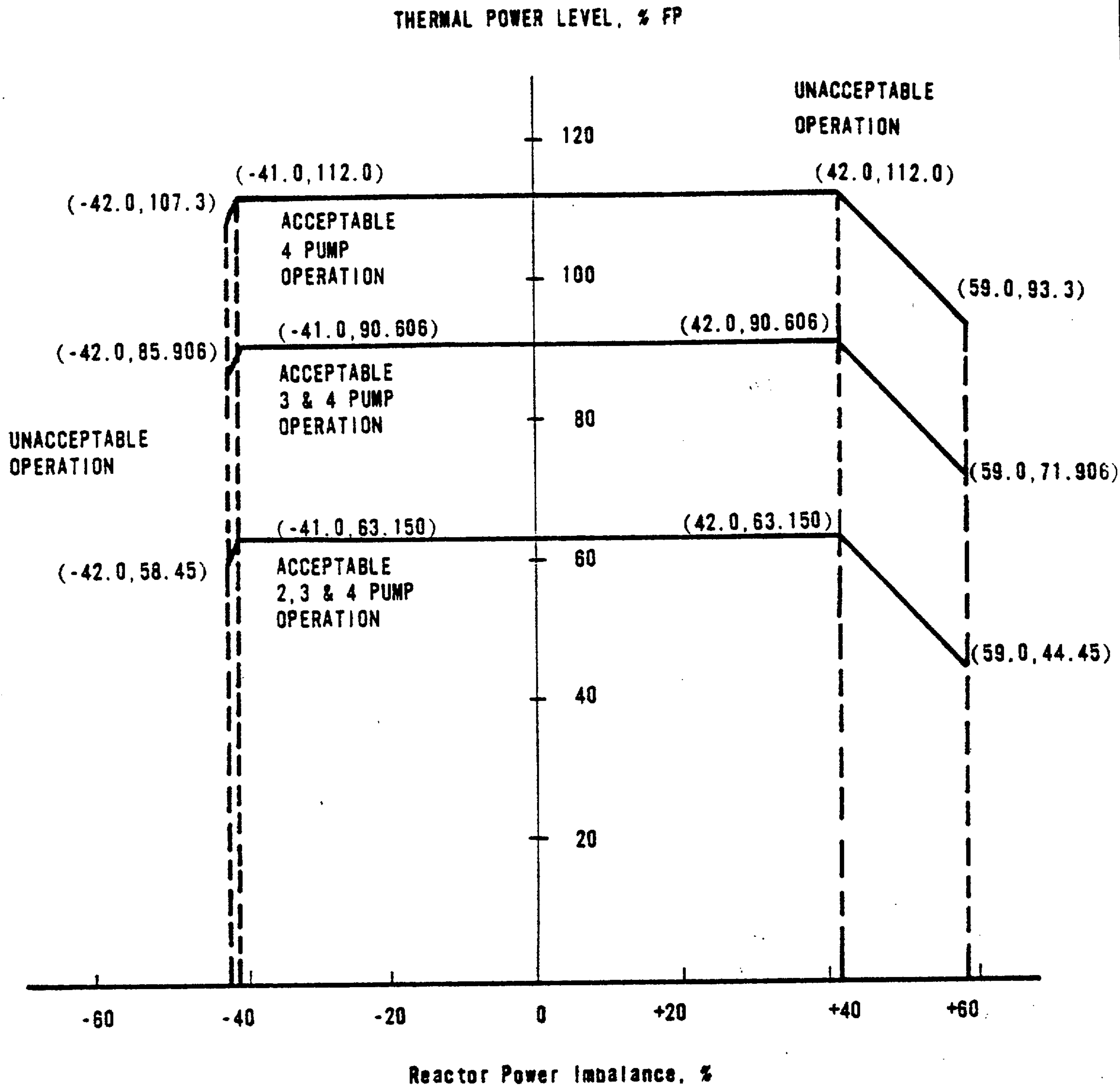
For each curve of Figure 2.1-3B, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The curve of Figure 2.1-1B is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3B.

References

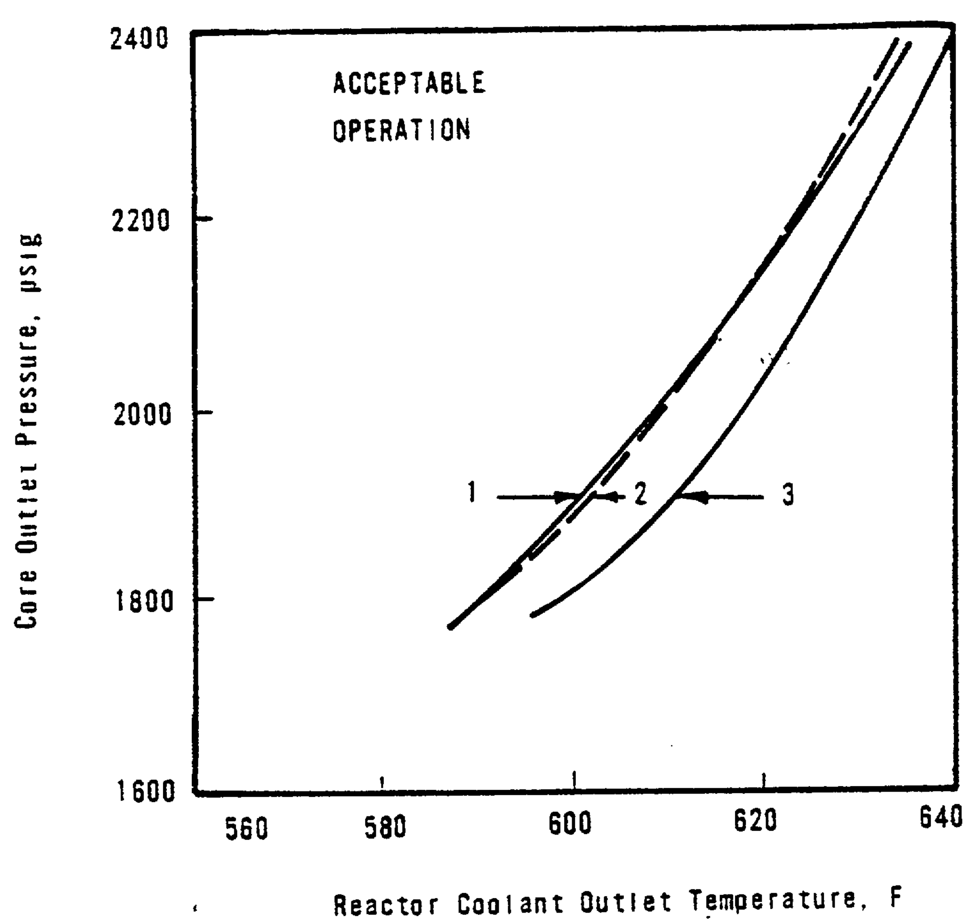
(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March 1970.

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

(3) Oconee 2, Cycle 6 - Reload Report - BAW-1691, August 1981.

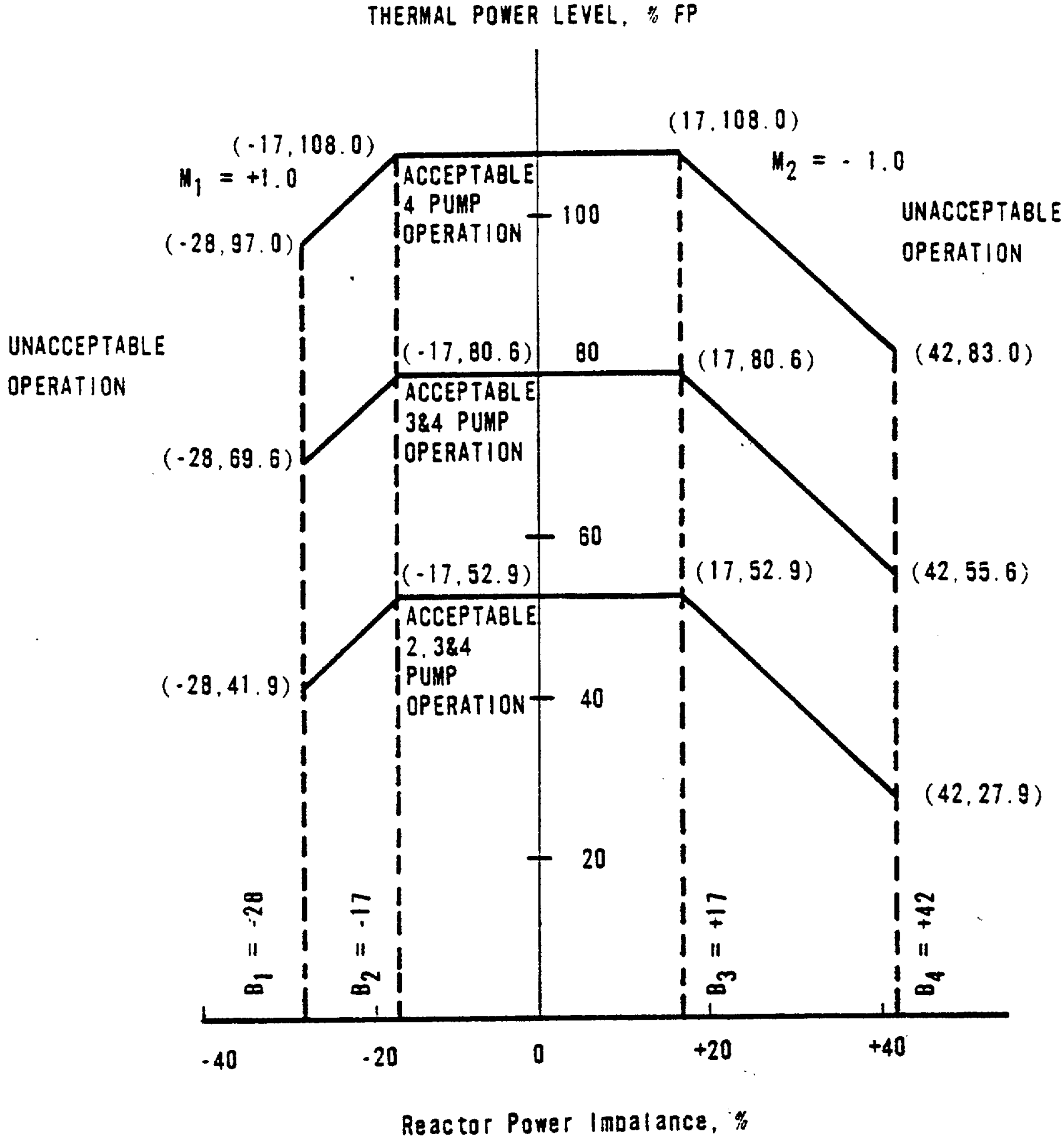


CORE PROTECTION SAFETY LIMITS
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 2.1-2B



CURVE	COOLANT FLOW, GPM	POWER, %	PUMPS OPERATING	TYPE OF LIMIT
1	374,880 (100%)	112	4	DNBR
2	280,035 (74.7%)	90.606	3	DNBR
3	183,690 (49.0%)	63.150	1 PER LOOP	QUALITY

CORE PROTECTION SAFETY LIMITS
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 2.1-3B



PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 2.3-2B

Table 2.3-1A

Unit 1

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)	104.9	104.9	104.9	$5.0^{(3)}$
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	1.07 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.	618	618	618	618
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.

Bases

The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the concentrated boric acid storage tank. An alternate method of boration will be the use of the high pressure injection pumps taking suction directly from the borated water storage tank.(2)

The quantity of boric acid in storage in the concentrated boric acid storage tank or the borated water storage tank is sufficient to borate the reactor coolant system to a 1% $\Delta k/k$ subcritical margin at cold conditions (70°F) with the maximum worth stuck rod and no credit for xenon at the worst time in core life. The current cycles for each unit, Oconee 1, Cycle 7, Oconee 2, Cycle 6, and Oconee 3, Cycle 6 were analyzed with the most limiting case selected as the basis for all three units. Since only the present cycles were analyzed, the specifications will be re-evaluated with each reload. A minimum of 1020 ft^3 of 8,700 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1835 ppm boric acid in the borated water storage tank (3) will satisfy the requirements. The volume requirements include a 10% margin and, in addition, allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. Using only one 10 gpm boric acid pump taking suction from the concentrated boric acid storage tank would require approximately 12.7 hours to inject the required boron. An alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. The required boric acid can be injected in less than six hours using only one of the makeup pumps.

The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above the crystallization temperature for the concentration present. The boric acid concentration of 8,700 ppm in the concentrated boric acid storage tank corresponds to a crystallization temperature of 77°F and therefore a temperature requirement of 87°F. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

(1) FSAR, Section 9.1; 9.2
(2) FSAR, Figure 6.2
(3) Technical Specification 3.3

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, 3.5.2-1A2, and 3.5.2-1A3 (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, on figures 3.5.2-2A1, 3.5.2-2A2, and 3.5.2-2A3 (Unit 1); 3.5.2-2B1, 3.5.2-2B2, and 3.5.2-2B3 (Unit 2) for three pump operation, on figures 3.5.2-2A4, 3.5.2-2A5, and 3.5.2-2A6 (Unit 1); 3.5.2-2B4, 3.5.2-2B5, and 3.5.2-2B6 (Unit 2) for two pump operation, and on figures 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, 3.5.2-4A2, and 3.5.2-4A3 (Unit 1); 3.5.2-4B1, and 3.5.2-4B2, and 3.5.2-4B3, (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.

3.5.2.6 Xenon Reactivity

Except for physics tests, reactor power shall not be increased above the power-level-cutoff shown in Figures 3.5.2-1A1, 3.5.2-1A2, and 3.5.2-1A3 for Unit 1; Figures 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3, for Unit 2; and Figures 3.5.2-1C1, 3.5.2-1C2, and 3.5.2-1C3 for Unit 3 unless one of the following conditions is satisfied:

1. Xenon reactivity did not deviate more than 10 percent from the equilibrium value for operation at steady state power.

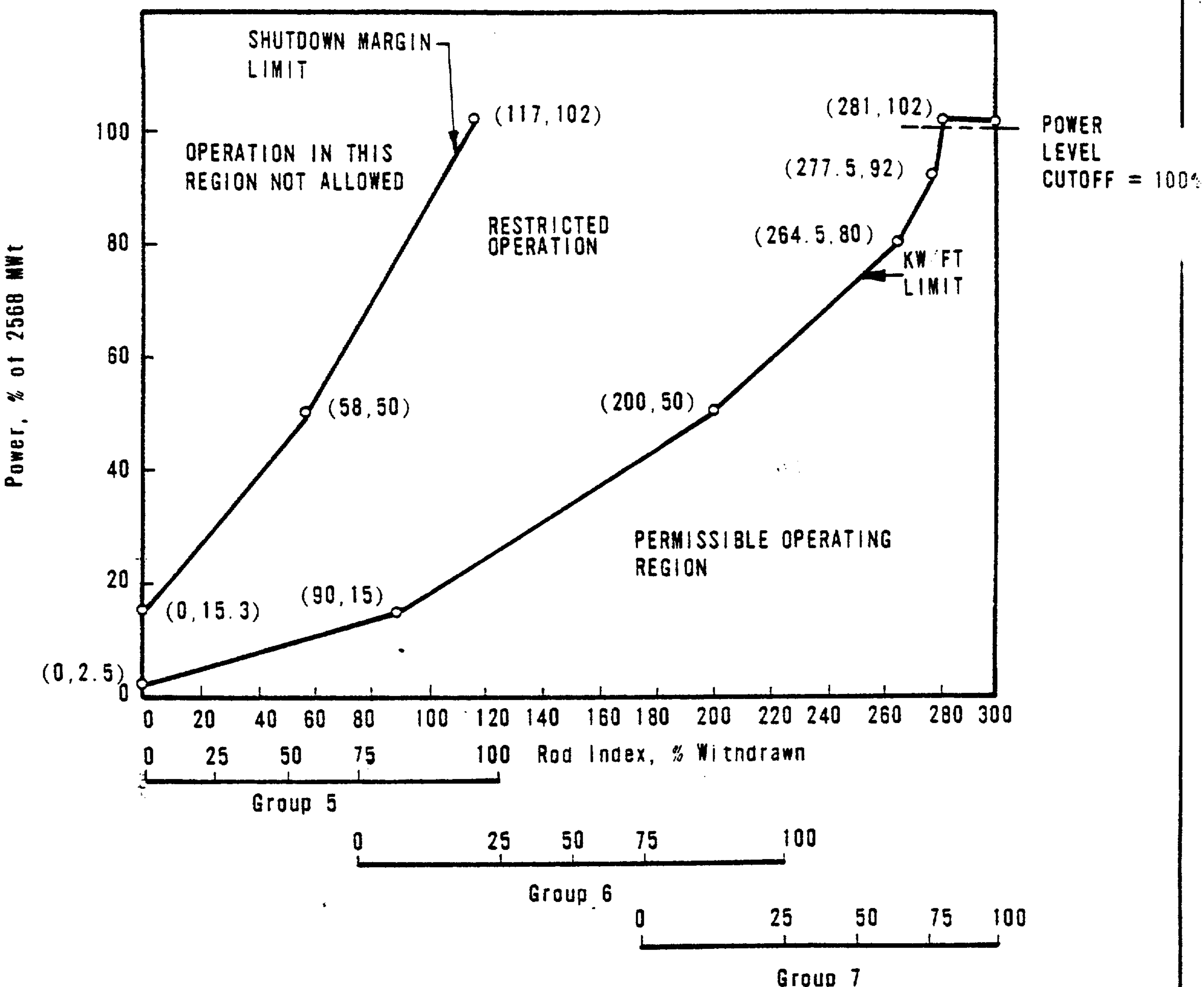
2. Xenon reactivity deviated more than 10 percent but is now within 10 percent of the equilibrium value for operation at steady state rated power and has passed its final maximum or minimum peak during its approach to its equilibrium value for operation at the power level cutoff.

3. Except for xenon free startup (when 2. applies), the reactor has operated within a range of 87 to 92 percent of rated thermal power for a period exceeding 2 hours.

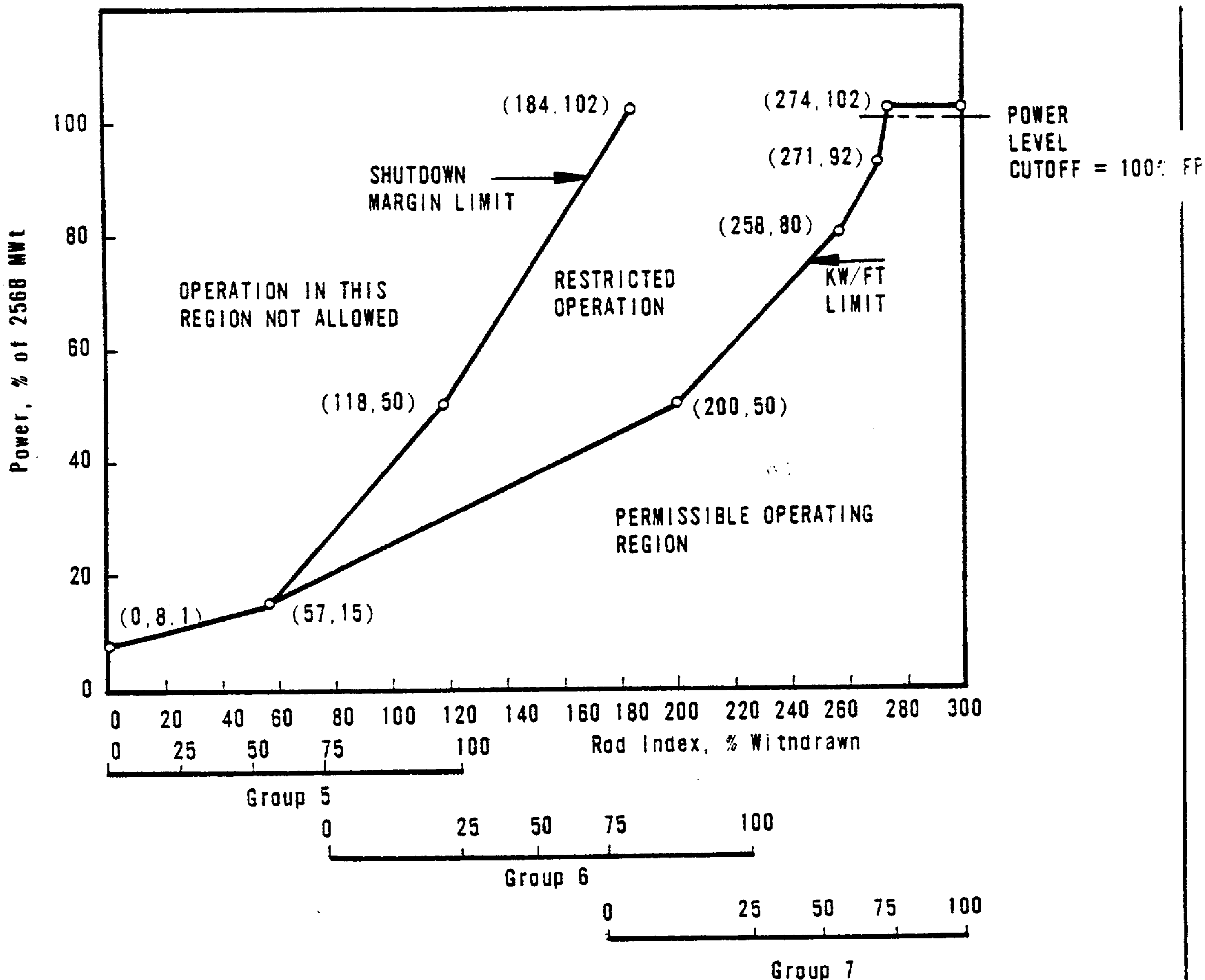
3.5.2.7 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3A3, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 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.

3.5.2.8 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.

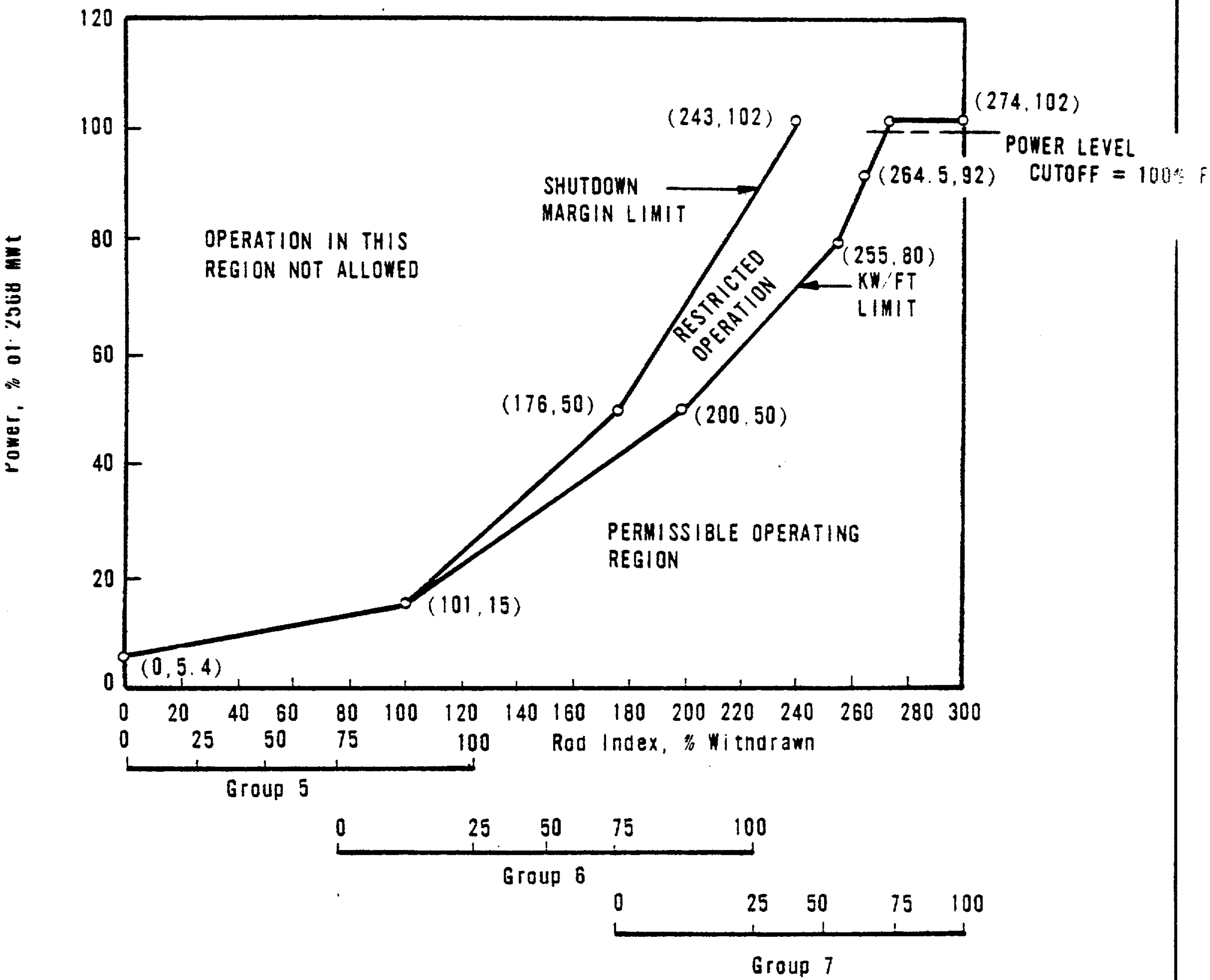
3.5.2.9 The operational limit curves of Technical Specifications 3.5.2.5.c and 3.5.2.7 are valid for a nominal design cycle length, as defined in the Safety Evaluation Report for the appropriate unit and cycle. Operational beyond the nominal design cycle length is permitted provided that an evaluation is performed to verify that the operational limit curves are valid for extended operation. If the operational limit curves are not valid for the extended period of the operation, appropriate limits will be established and the Technical Specification curves will be modified as required.



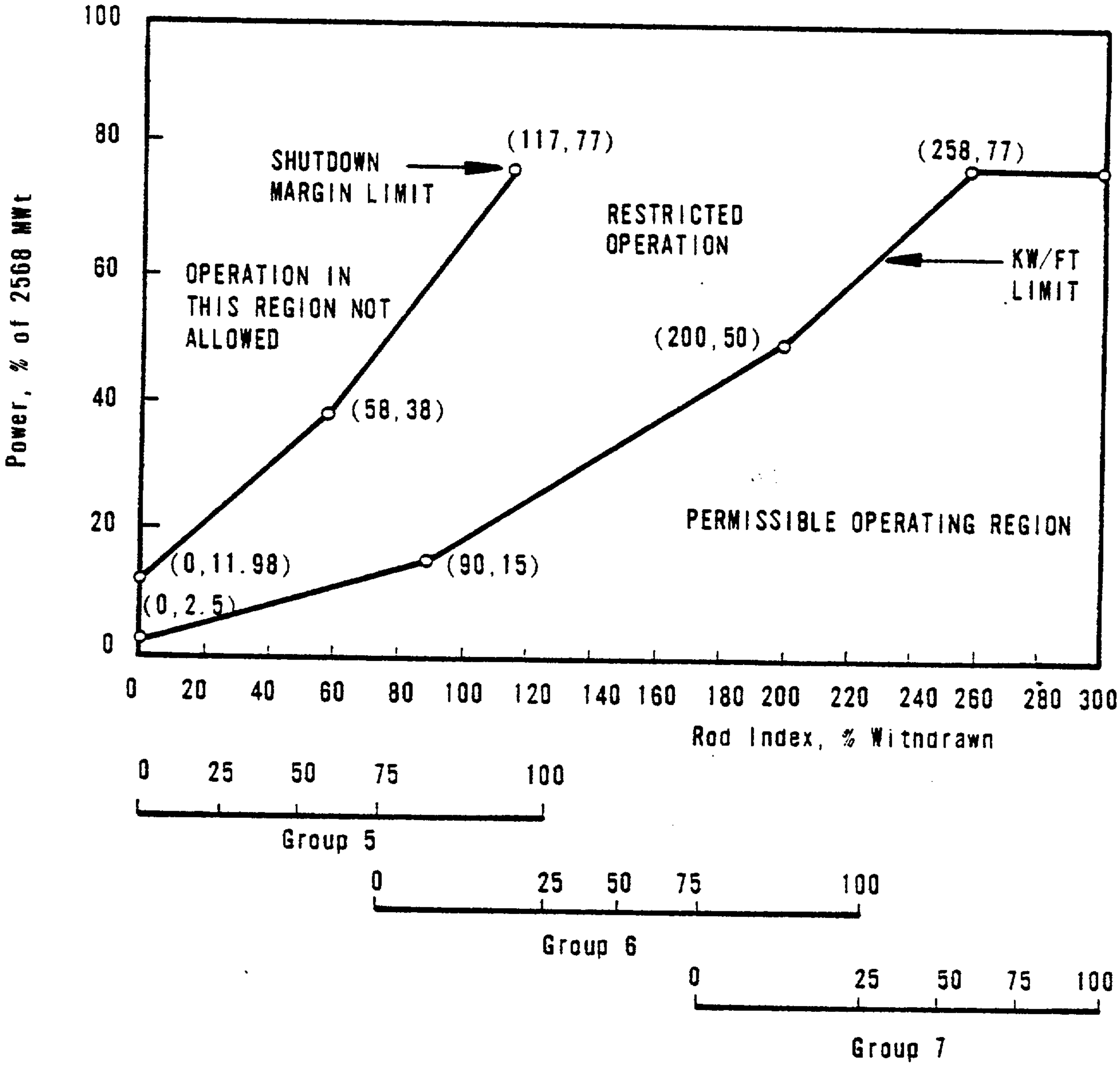
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-1B1



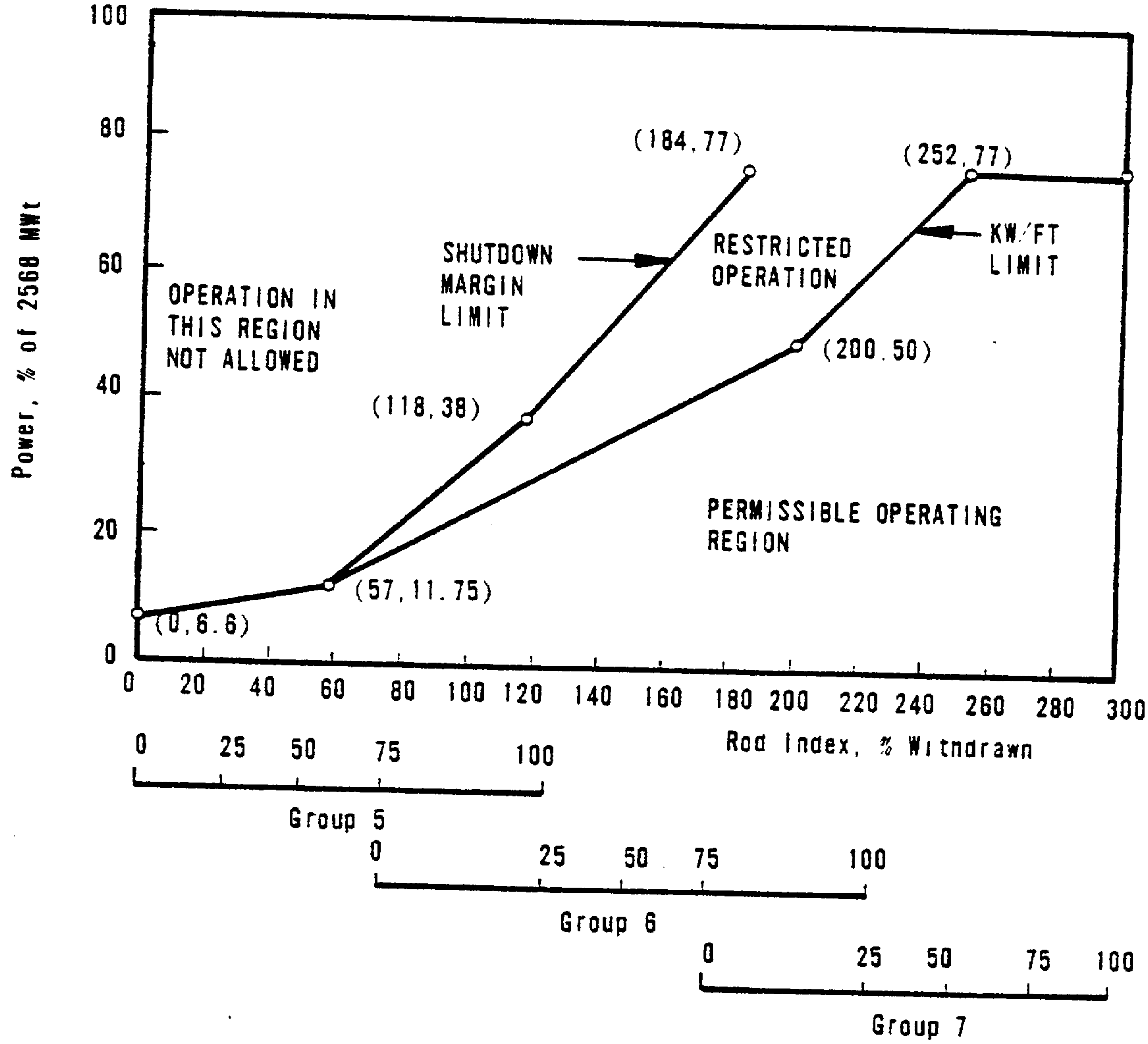
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 50 (+10, -0) to 225 ±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-1B2



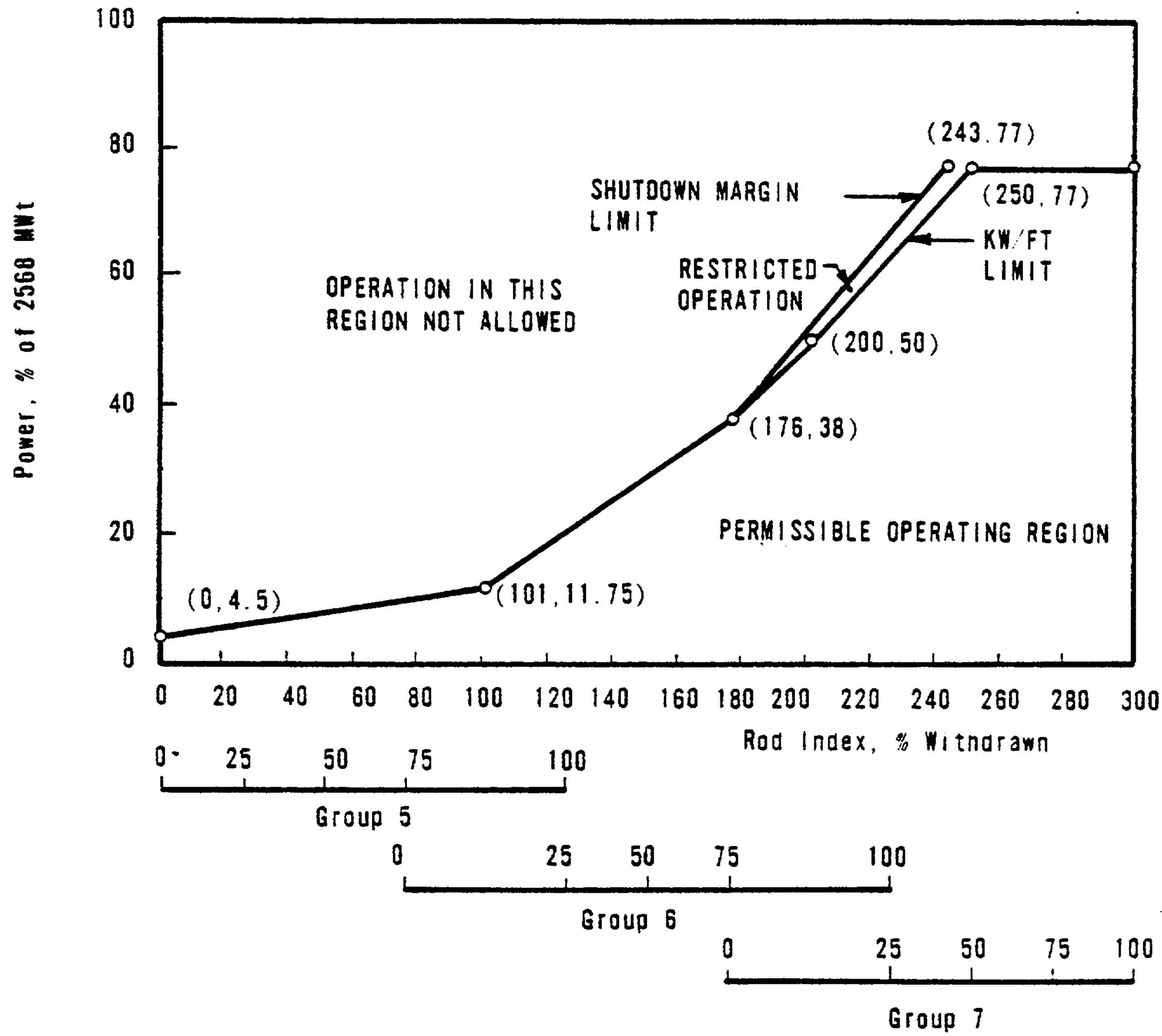
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 225 $\pm$10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-1B3



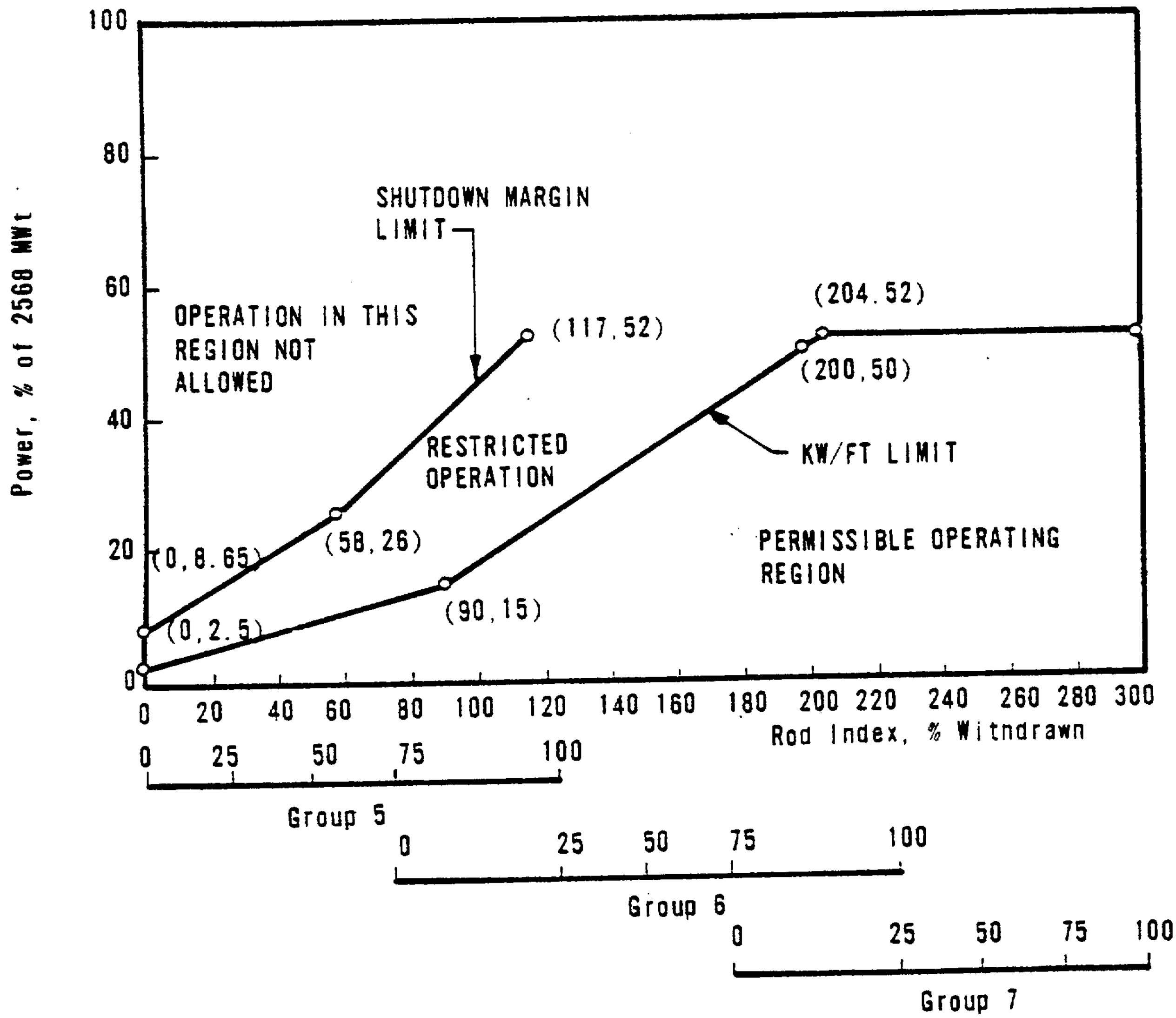
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2B1



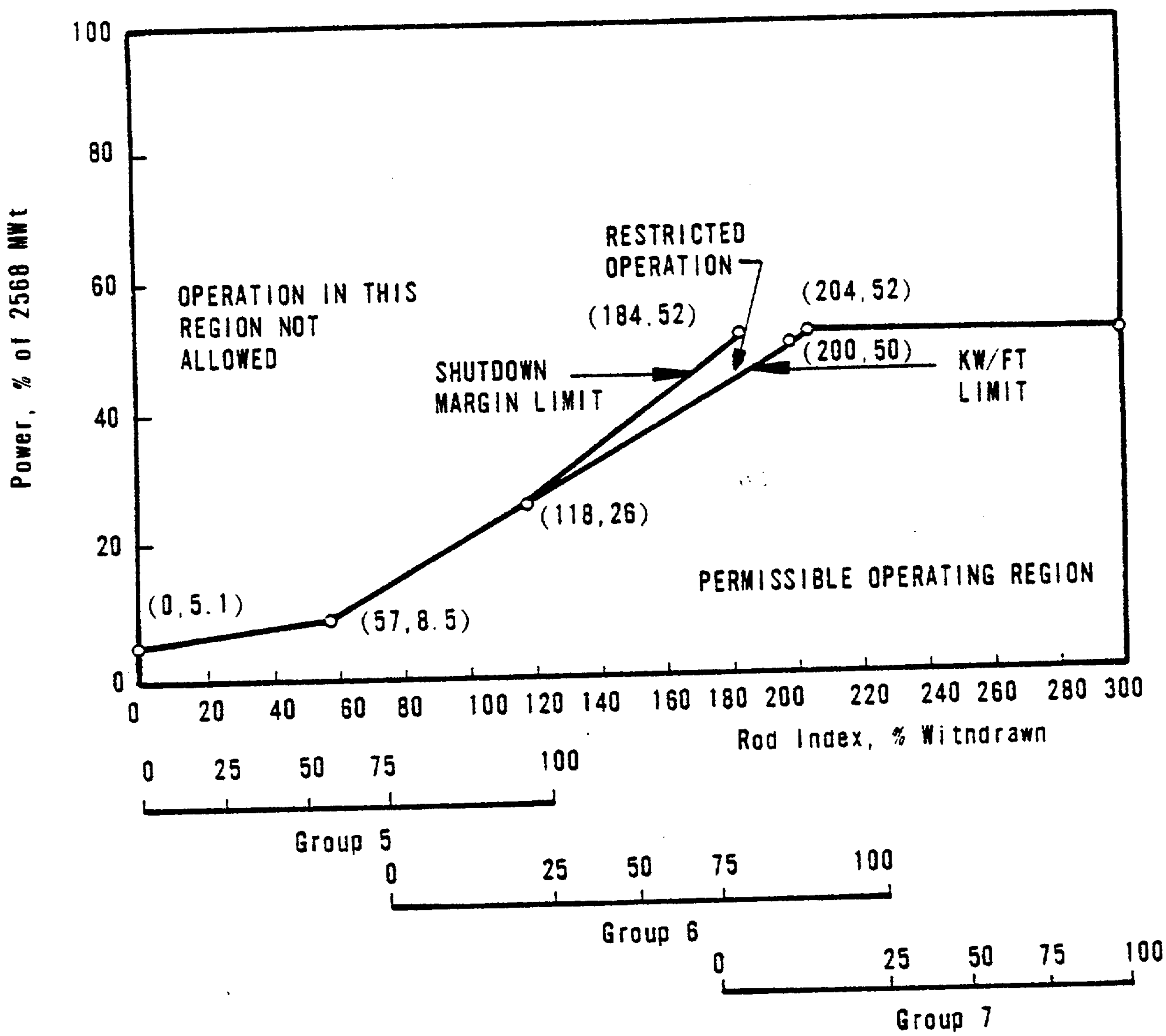
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 50 (+10, -0) to 225 ±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2B2



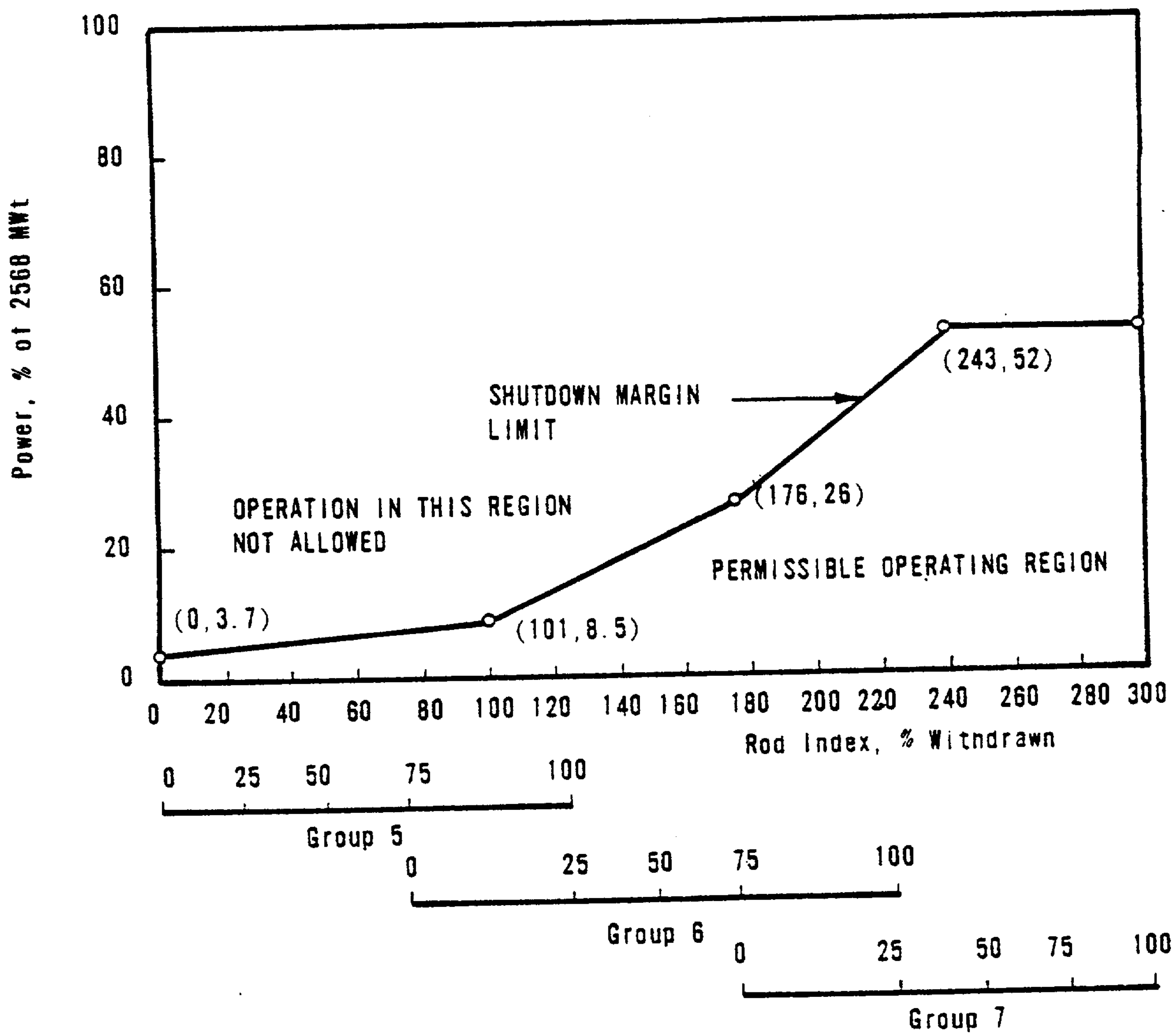
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
AFTER 225 ±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2B3



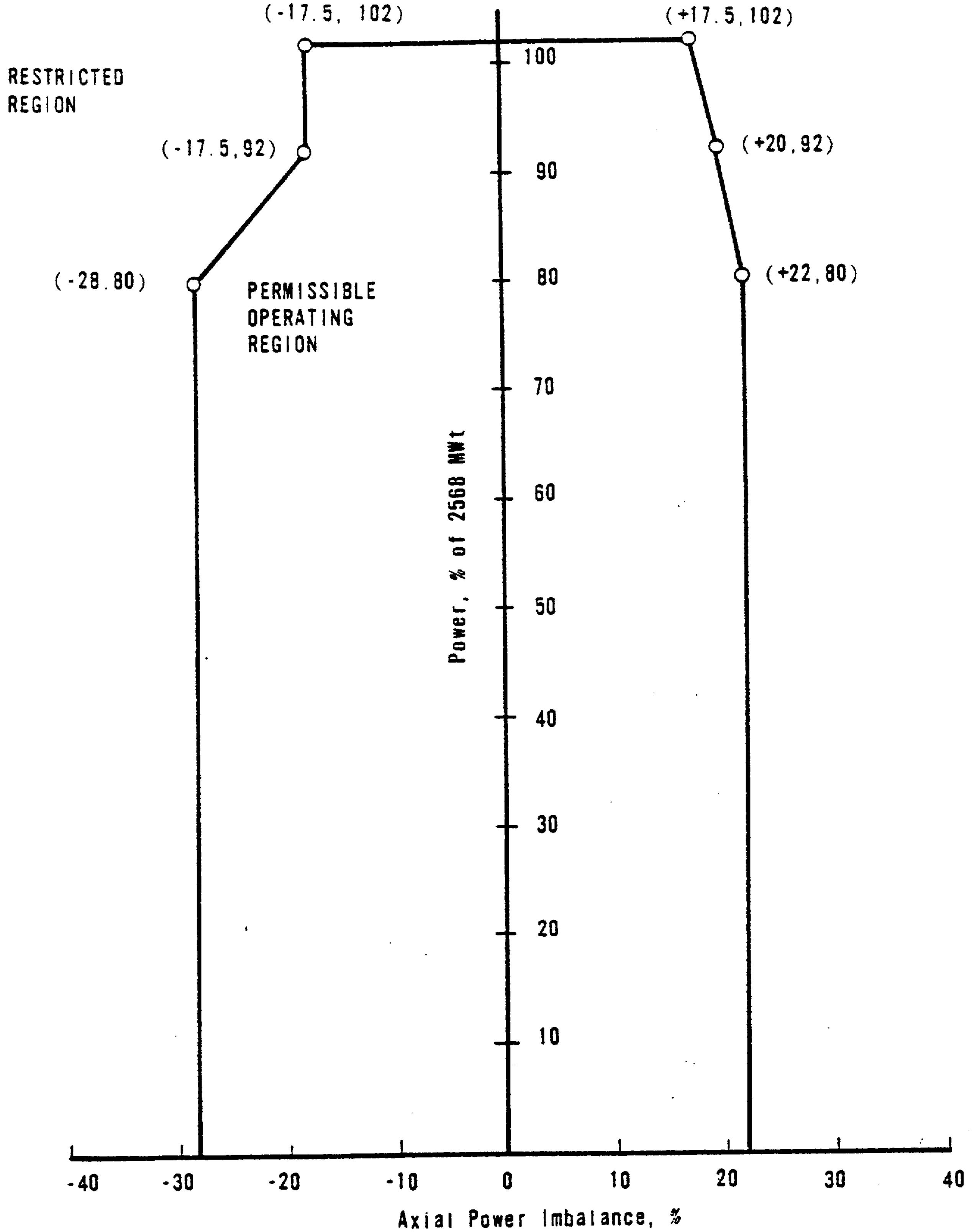
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2B4



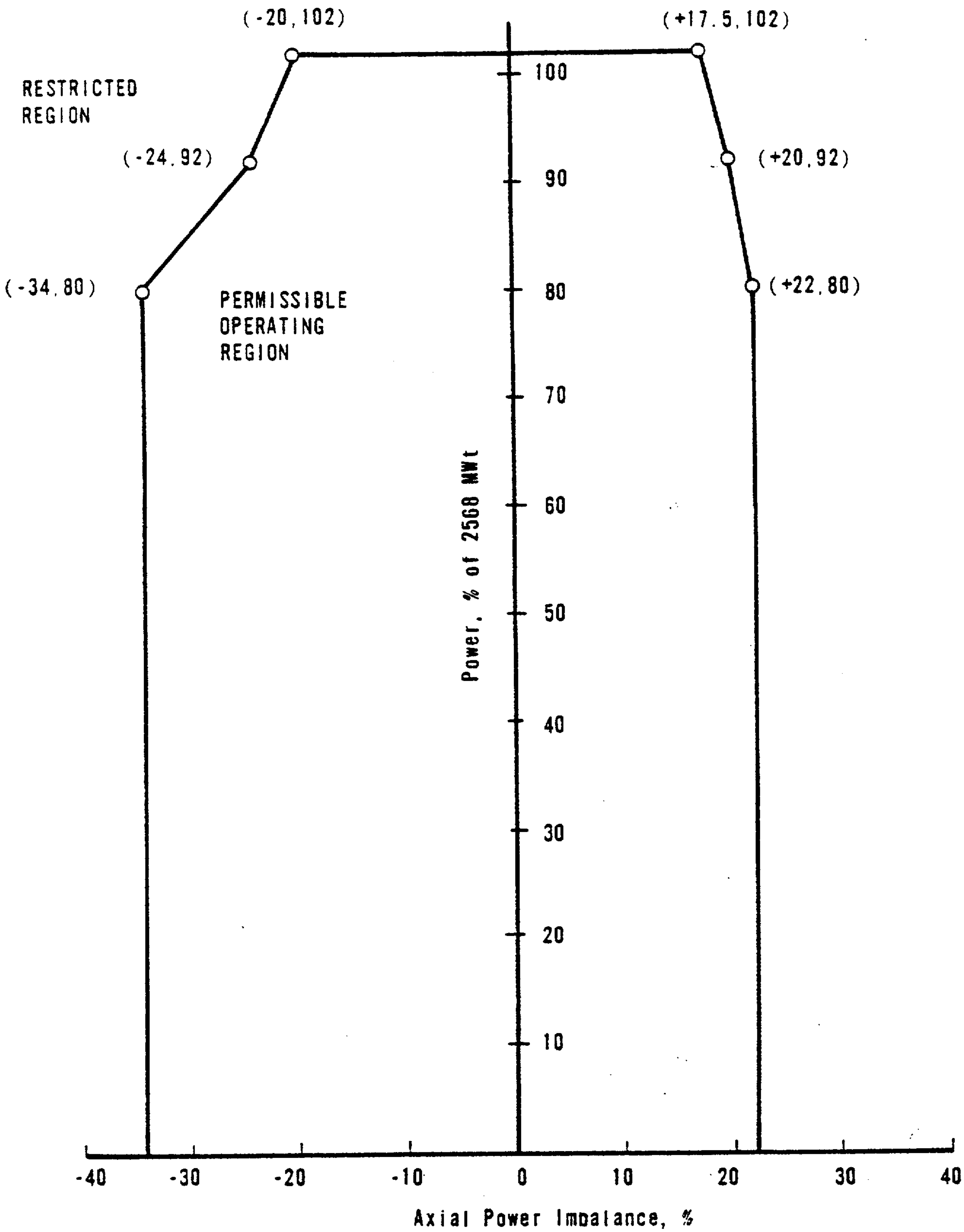
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 50 (+10, -0) to 225 ±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2B5



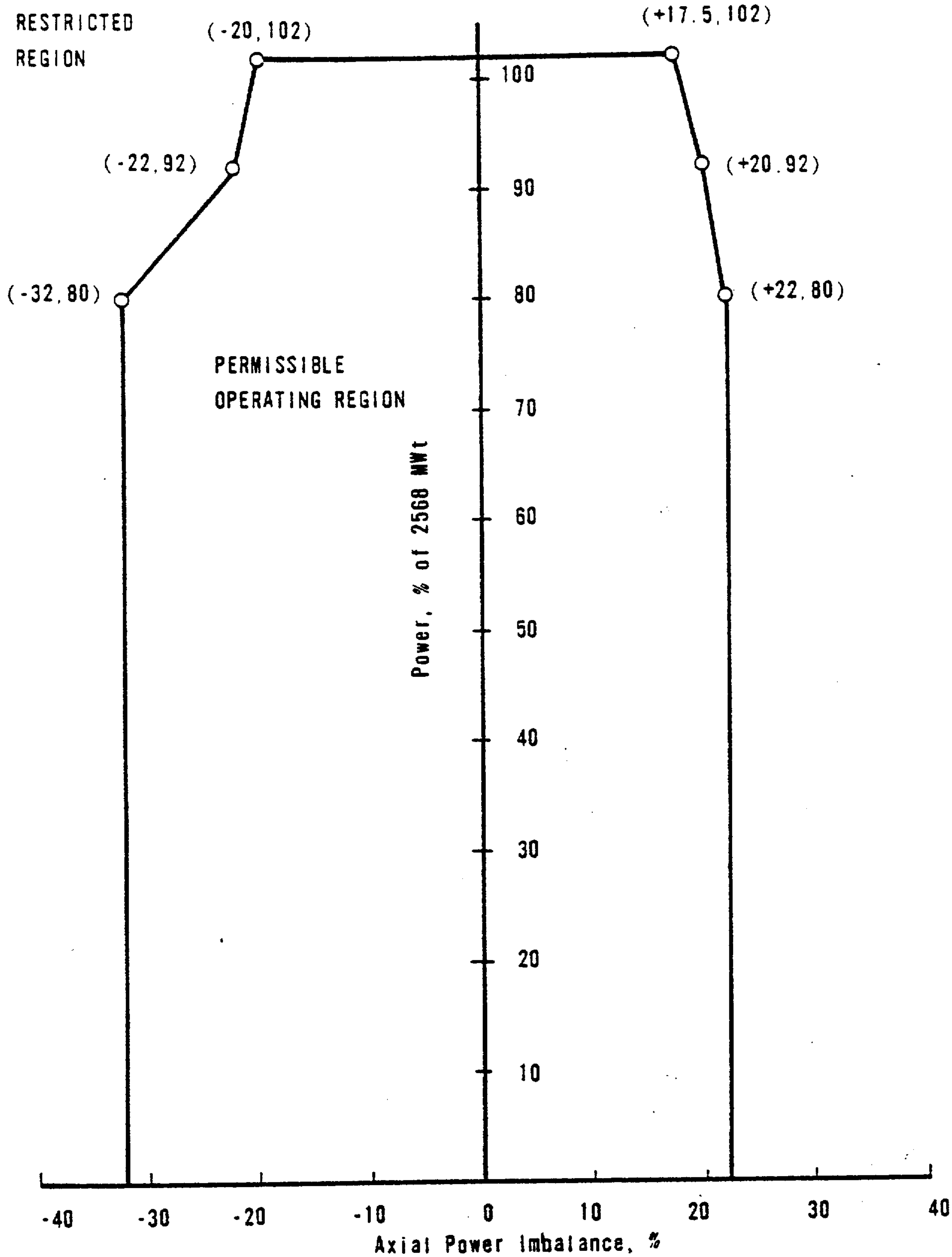
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 225 ±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2B6



OPERATIONAL POWER IMBALANCE LIMITS
FOR OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-3B1

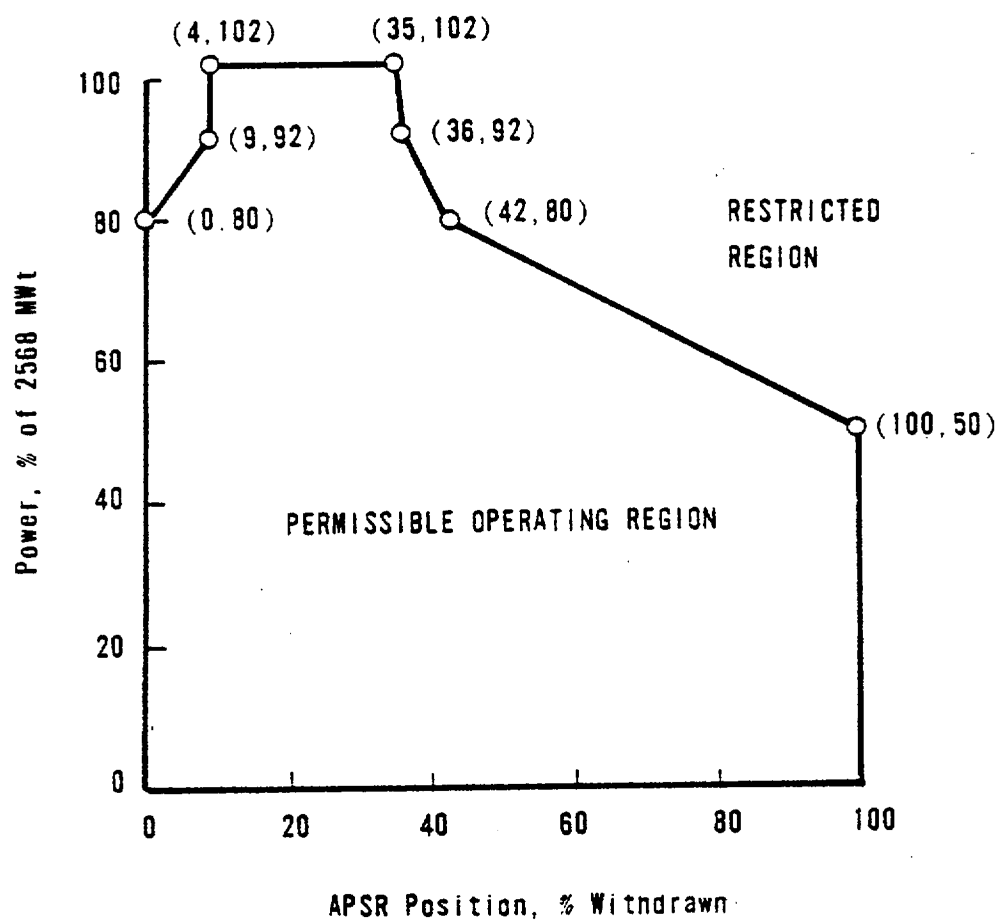


OPERATIONAL POWER IMBALANCE LIMITS
FOR OPERATION
FROM 50 (+10, -0) to 225 ±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-3B2

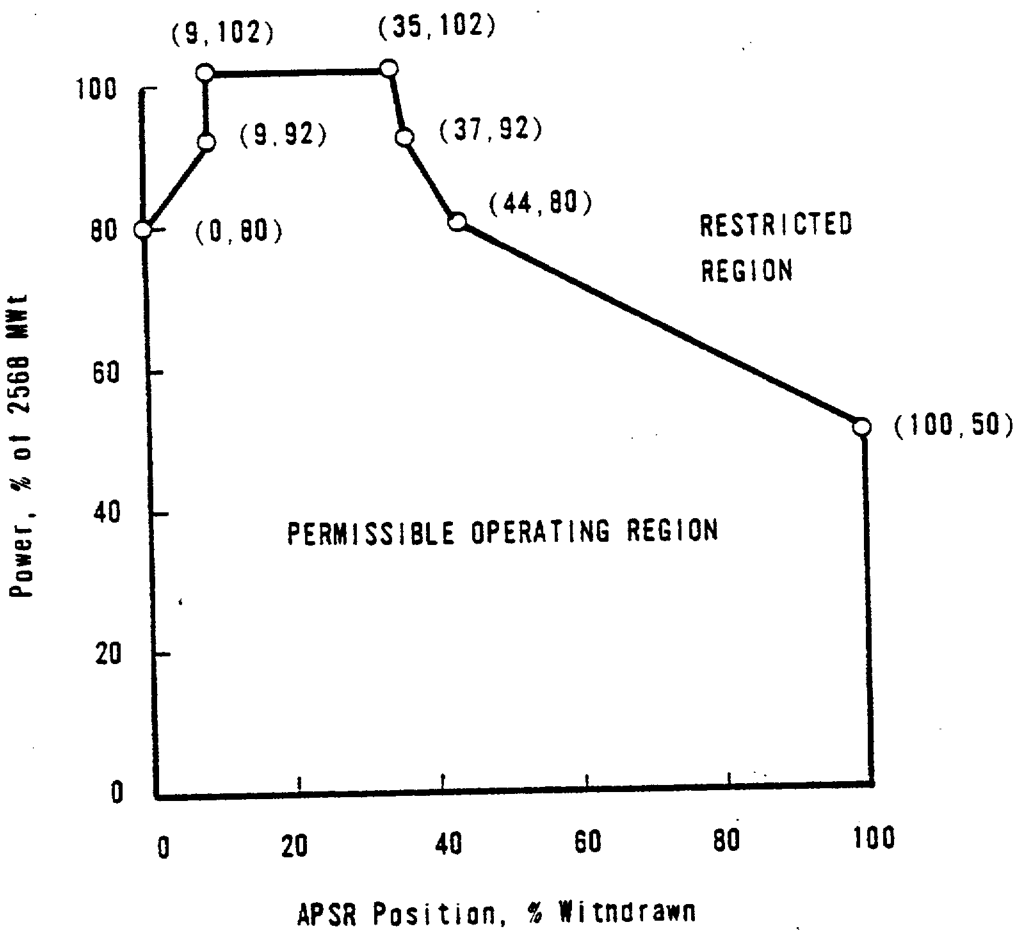


OPERATIONAL POWER IMBALANCE LIMITS
FOR OPERATION
AFTER 225 $\pm$10 EFPD
UNIT 2

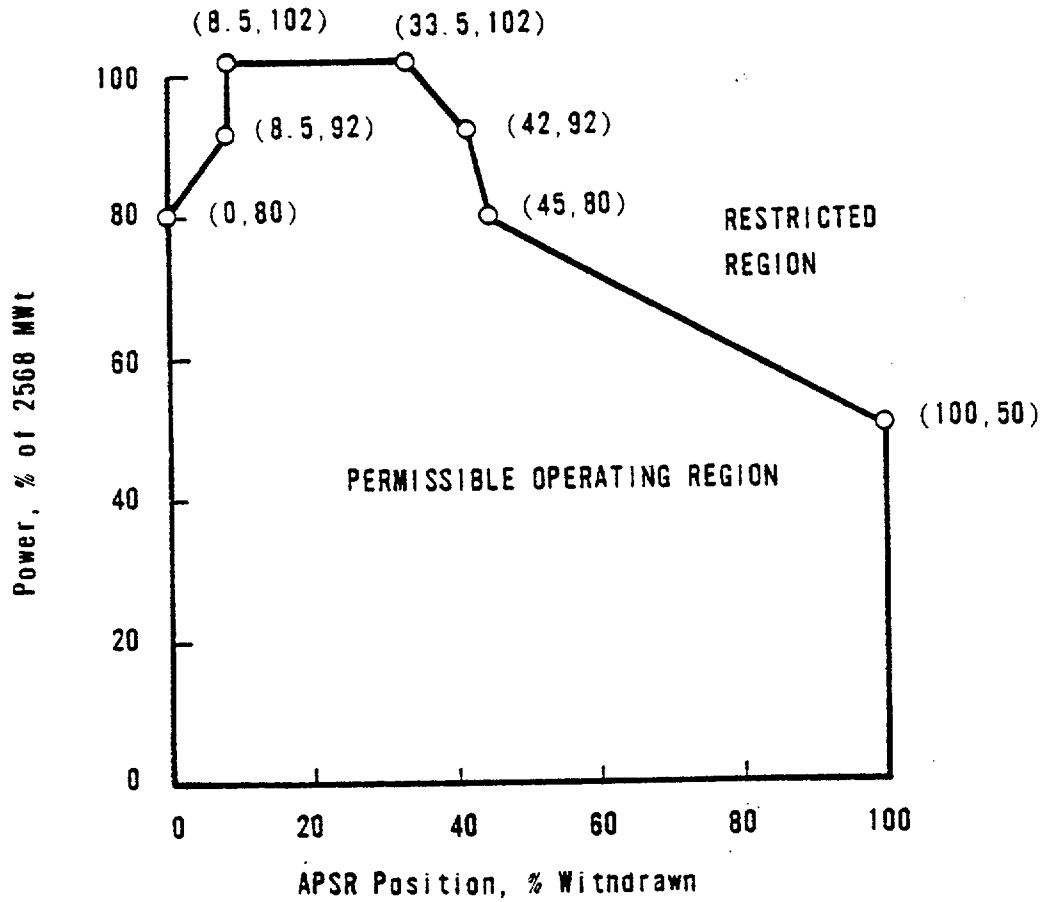
OCONEE NUCLEAR STATION
FIGURE 3.5.2-3B3



APSR POSITION LIMITS
FOR OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-4B1



APSR POSITION LIMITS
FOR OPERATION
FROM 50 (+10, -0) to 225±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-4B2



APSR POSITION LIMITS
FOR OPERATION
AFTER 225 $\pm$ 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

FIGURE 3.5.2-4B3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

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

1.0 Introduction

By letter dated November 13, 1981, Duke Power Company (Duke or the licensee) submitted an application (the reload application) to amend the common Oconee Nuclear Station (ONS) Technical Specifications (TSs) to support full power operation of Unit 2 during fuel Cycle 6. The reload application also provided, as an enclosure, Babcock and Wilcox Report BAW-1691, "Oconee Unit 2, Cycle 6 Reload Report," August 1981, in support of the proposals. This report includes a summary of the operating parameters and contains the safety analyses supporting Unit 2 operation during Cycle 6.

By letter dated March 24, 1982, Duke provided information on the degraded condition discovered on a Mark BZ demonstration fuel assembly which was to be reinserted in the core for Cycle 6 operation. As a result of the movement detected in zircaloy spacer grids on this assembly, a decision was made by Duke not to reinsert it for Cycle 6 but instead to insert an assembly with similar reactivity from the spent fuel pool with Inconel spacer grids. Analyses were performed which showed that the conclusions of the reload application remain valid.

2.0 Discussion and Evaluation

2.1 Evaluation of Fuel System Design

The reload application described the core loading to be used in Cycle 6. Seventy-two fresh assemblies having an initial enrichment of 3.17 weight percent U-235 will be loaded. Cycle 6 is to have an extended length of approximately 400 effective full power days. For this reason burnable poison assemblies are used to limit the required beginning of cycle soluble boron concentration.

The seventy-two Babcock and Wilcox (B&W) Mark B4 15x15 fuel assemblies loaded as Batch 8 at the end of Cycle 5 (EOC 5) are mechanically interchangeable with Batches 6B and 7 fuel assemblies previously loaded at Oconee 2. The Cycle 6 core will also contain four previously irradiated burnable poison rod assemblies (BPRAs).

Although all batches in the Oconee 2 Cycle 6 core will utilize the same Mark B fuel design, the Batch 8 assemblies incorporate a slightly different active fuel length. The change, based on undensified fuel length, is a consequence of a minor modification in the fuel fabrication process. The stabilities (densification resistance) of all fuel types are almost identical. As a consequence, the densified fuel stack height is nearly the same for all Cycle 6 assemblies.

In addition to the permanent reactivity control system (soluble boron and control rods), 52 previously-irradiated BPRAs will be discharged and 64 fresh BPRAs will be added to control reactivity changes due to fuel burnup and fission product buildup. The irradiated BPRAs are normally removed from the reactor at the end of each cycle and fresh BPRAs are inserted for the subsequent cycle of operation, particularly where extended cycle operation is anticipated. Four previously irradiated BPRAs will remain in the Cycle 6 core for a second cycle to gather burnup data on these assemblies. The licensee has considered the impact of these four assemblies on the operation of Oconee 2 and has determined that they will not adversely affect Cycle 6 operation.

The reload application states that the cladding collapse, stress and strain analyses are bounded by conditions previously analyzed and approved by the NRC. We agree with these conclusions.

The reload application also states that fuel rod internal pressure will not exceed normal system pressure during normal operation for Cycle 6. The analysis is based on the use of the B&W TAFY code rather than a newer B&W code called TACO-1. Although both of these codes have been approved for use in safety analysis, we believe that the newer TACO code is capable of more correctly calculating fission gas release (and therefore rod pressure) at very high burnups. B&W has stated that the internal fuel rod pressure predicted by TACO is lower than that predicted by TAFY for fuel rod exposures of up to 42,000 MWd/MtU. Although we have not examined the comparison, we note that the maximum expected exposure (37,046 MWd/MtU), in Oconee 2 at EOC 6, for all assemblies is lower than this predicted value. We, therefore, conclude that the rod internal pressure limits have been adequately considered for Cycle 6 operation.

The average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the Loss of Coolant Accident (LOCA) analysis (Section 7.2 of the reload application) are also calculated with the TAFY code. Duke has stated that the fuel temperature and pin pressure data used in the generic LOCA analysis are conservative compared with those calculated for Cycle 6 at Oconee 2.

The chemical and material compatibility of possible fuel, cladding and coolant interactions is unchanged from the previous cycle of operation. The impact of material compatibility on the operational safety of Oconee 2 need not be reconsidered for Cycle 6 operation.

The licensee has calculated a fuel rod bowing penalty with a method similar to that previously approved. The rod bowing magnitude correlation used in that method is approved, and we conclude that it adequately accounts for gap closure as a function of burnup in the Mark B fuel design.

We have reviewed those sections of the reload application for Oconee 2, Cycle 6, dealing with the fuel system design. We find those portions of the application acceptable.

2.2 Evaluation of Nuclear Design

The nuclear characteristics of the core have been computed by methods previously used and approved for B&W reactors. Comparisons are made between the physics parameters for Cycles 5 and 6. The differences that exist between the parameters are due to the increased cycle length which tends to increase values of critical boron concentrations. Changes in the radial flux and burnup distributions between cycles also accounts for the differences in control rod worths, including ejected and stuck rod worths. All safety criteria are still met. Shutdown margin values at beginning and end of cycle are 3.74 and 2.40 percent $\Delta k/k$ respectively compared to the required 1.0 percent. Beginning of cycle radial power distributions show acceptable margins to limits. Based on our review, we conclude that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of Cycle 6 is acceptable.

The key kinetics parameters for Cycle 6 have been compared to the values used in the Final Safety Analysis Report (FSAR) and densification report. It is shown that in all cases Cycle 6 values are bounded by those previously used. We conclude that the FSAR transient and accident analyses are valid.

We have reviewed the proposed TSs for Cycle 6. The limiting safety systems settings and the limiting conditions for operation have been established by previously used and approved methods. The rod withdrawal limits for the various pump combinations and times in life are presented. On the basis that previously approved methods were used to obtain the limits, we find them acceptable.

The effects of the recently discovered under-estimate of the errors in certain modules of the reactor protection system have been included. The nuclear overpower trip setpoint was reduced from 105.5 to 104.9 percent full power, and the high reactor coolant temperature trip was reduced from 619 to 618 degrees Fahrenheit. On the basis that these setpoints were established by previously accepted methods, we conclude that the revised limits are acceptable.

2.3 Evaluation of Thermal-Hydraulic Design

The incoming Batch 8 fuel is hydraulically and geometrically similar to the fuel remaining in the core from the previous cycles. For Cycle 6 reload, 68 BPRAs will be inserted, 12 more than the previous cycle. The fewer number of unplugged guide tubes in this reload results in a decrease of maximum bypass flow to 7.6% from 8.1% for Cycle 5. The decreased bypass flow and consequent increase in core flow indicates that with other core parameters unchanged, the safety margin for Cycle 6 is at least comparable to that of Cycle 5. The reinserted BPRAs have been designed to ensure that the impact on thermal-hydraulic analysis is insignificant.

The rod bow Departure from Nucleate Boiling Ratio (DNBR) compensation applicable to Cycle 6 was calculated using the interim rod bow compensation evaluation procedure similar to that previously approved. The burnup used to calculate the rod bow compensation was the highest assembly burnup in Batch 8, 19,100 MWd/MtU, which contains the limiting (maximum radial peak factor) fuel assembly. The resultant hot rod bow compensation factor after inclusion of the one percent flow area reduction factor credit is 0.4 percent reduction in DNBR. To demonstrate that the rod bow compensation for Batch 8 fuel is the most limiting, the licensee performed a series of thermal-hydraulic analyses. The analytical results for the limiting assemblies in fuel Batches 6B and 7, based on steady state power distributions, demonstrate that the increase in DNBR associated with the lower peaking of these assemblies relative to the limiting Batch 8 assembly offsets the increased rod bow DNBR compensation that would be calculated on the basis of maximum assembly burnup values for these batches. We, therefore, conclude that the available margin for Cycle 6 more than offsets the 0.4 percent DNBR rod bow compensation and that the thermal-hydraulic design is, therefore, acceptable.

To support the operation of Oconee 2 at full power during Cycle 6, the licensee proposed modifications to core protection safety limits of TS 2.1 (Figures 2.1-2B, 2.1-3B and 2.3-2B). The changes reflect a revised flux/flow setpoint of 1.08 which remains within the safety limit DNBR criterion of 1.30 and is, therefore, acceptable.

The pertinent thermal-hydraulic parameters are summarized in Table 6.1 of the reload application and are identical for Cycles 5 and 6. The fuel in Cycle 6 is geometrically similar to Cycle 5 fuel which has been previously approved for Oconee 2, and the thermal-hydraulic models and methodology used for Cycle 6 have been previously approved. We conclude that this core reload will not adversely affect the capability to operate Oconee 2 safely during Cycle 6.

2.4 Additional Changes

As mentioned in Section 2.2 of this Safety Evalution, recently discovered under-estimate of the errors in certain modules of the reactor protection system require changes to some setpoints in the TSs. By license amendments dated November 2,1981, the NRC approved similar changes for the Oconee 1 TSs. One item, Maximum Nuclear Power, was not revised on page 2.3-11 although it was changed in all other areas. This inconsistency was recently discovered, and we have included a revised page indicating the correct value of 1.07 vs. 1.08. Since this change only corrects a previous omission, we consider it to be an acceptable, administrative change.

An additional change related to Oconee 1 involves the clarification of the Bases on page 2.1-2 related to the DNBR margin. The bases indicate that a 10% margin results when all penalties are taken into account for Cycle 7 operation. However, as a result of the under-estimate of module errors discussed above, this is no longer the case. Therefore, we have removed this statement to eliminate any confusion. Since this change is only a clarification of a Bases, we consider it to be an acceptable, administrative change.

2.5 Summary

We have reviewed the physics, fuels, thermal-hydraulic and transient and accident information presented in the Oconee 2 Cycle 6 reload application and find the proposed reload and the associated modified TSs to be acceptable. We have also reviewed the additional changes to the TSs and find them to be acceptable.

3.0 Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 8, 1982

The following NRC staff personnel have contributed to this Safety Evaluation: P. C. Wagner, L. Kopp, S. Sun, J. Vogelwede.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 111 , 111 and 108 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the Technical Specifications (TSs) for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments revise the TSs to support full power operation of Oconee 2 during fuel Cycle 6.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated November 13, 1981, as supplemented March 24, 1982, (2) Amendments Nos. 111 , 111 , and 108 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Oconee County Library, 501 West Southbroad Street, Walhalla, South Carolina. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 8th day of April 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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