D/F

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

June 12, 1980

Dockets Nos. 50-269, 270
and 287

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 83, 83, and 80 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 operating licenses and the Station's common Technical Specifications in response to your request dated September 22, 1976, as supplemented September 11, 1979, and your request dated November 16, 1979, as superseded March 12, 1980 and supplemented April 30, 1980.

These amendments revise the Technical Specifications to support the operation of Oconee Unit No. 2 at full rated power during Cycle 5. The amendments also incorporate monitoring conditions for secondary water chemistry in the body of the licenses for Units Nos. 1, 2 and 3 in response to your request dated September 11, 1979, and staff discussions.

Oconee Unit No. 2, during Cycle 4, was operating under a December 18, 1978 Exemption to 10 CFR 50.46, the Emergency Core Cooling System (ECCS) rule. The enclosed Safety Evaluation provides the bases for terminating the Exemption, as your ECCS modifications and operating procedures have met the provisions of the Exemption as confirmed by your letter of May 29, 1980.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 83 to DPR-38
2. Amendment No. 83 to DPR-47
3. Amendment No. 80 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

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Duke Power Company

cc w/enclosure(s):

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Duke Power Company
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Oconee Public Library
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County Supervisor of Oconee County
Walhalla, South Carolina 29621

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Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
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cc w/enclosure(s) and incoming dtd.:
9/11/79, 3/12 & 4/30/80

Office of Intergovernmental Relations
116 West Jones Street
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. 83
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 September 22, 1976, as supplemented September 11, 1979, and the application dated November 16, 1979, as superseded March 12, 1980, and supplemented April 30, 1980, 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 applications, 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, Facility Operating License No. DPR-38 is hereby amended by revising paragraph 3.B and adding paragraph 3.G. as follows and by changing the Technical Specifications as indicated in the attachment to this license amendment:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 83 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   3.G The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions of off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

3. Except for paragraph 3.G this license amendment is effective as of the date of its issuance. Paragraph 3.G is effective within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1980

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. 83
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 September 22, 1976, as supplemented September 11, 1979, and the application dated November 16, 1979, as superseded March 12, 1980, and supplemented April 30, 1980, 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 applications, 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, Facility Operating License No. DPR-47 is hereby amended by revising paragraph 3.B and adding paragraph 3.G. as follows and by changing the Technical Specifications as indicated in the attachment to this license amendment:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 83, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   3.G The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions of off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

3. Except for paragraph 3.G this license amendment is effective as of the date of its issuance. Paragraph 3.G is effective within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1980

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. 80
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 September 22, 1976, as supplemented September 11, 1979, and the application dated November 16, 1979, as superseded March 12, 1980, and supplemented April 30, 1980, 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 applications, 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, Facility Operating License No. DPR-55 is hereby amended by revising paragraph 3.B and adding paragraph 3.G. as follows and by changing the Technical Specifications as indicated in the attachment to this license amendment:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 80 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   3.G The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions of off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

3. Except for paragraph 3.G this license amendment is effective as of the date of its issuance. Paragraph 3.G is effective within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1980

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 83 TO DPR-38

AMENDMENT NO. 83 TO DPR-47

AMENDMENT NO. 80 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
viii & ix	viii & ix
2.1-2 & 2.1-3	2.1-2 & 2.1-3
2.1-3b	2.1-3b
2.1-8	2.1-8
2.3-2 & 2.3-3	2.3-2 & 2.3-3
2.3-9	2.3-9
2.3-12	2.3-12
3.5-9 & 3.5-10	3.5-9 & 3.5-10
3.5-16 & 3.5-16a	3.5-16 & 3.5-16a
3.5-16b	----------------
3.5-19 & 3.5-19a	3.5-19 & 3.5-19a
3.5-19b	----------------
3.5-22	3.5-22
3.5-22a & 3.5-22b	----------------
3.5-25 & 3.5-25a	3.5-25 & 3.5-25a
3.5-25b	----------------

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Figure		Page
3.1.2-3A	Reactor Coolant System Inservice Leak and Hydrostatic Test Heatup and Cooldown Limitation - Unit 1	3.1-7c
3.1.2-3B	Reactor Coolant System Inservice Leak and Hydrostatic Test Heatup and Cooldown Limitation - Unit 2	3.1-7d
3.1.2-3C	Reactor Coolant System Inservice Leak and Hydrostatic Test Heatup and Cooldown Limitation - Unit 3	3.1-7e
3.1.10-1	Limiting Pressure vs. Temperature Curve for 100 STD cc/Liter H_2O	3.1-22
3.5.2-1A1	Rod Position Limits for Four Pump Operation - Unit 1	3.5-15
3.5.2-1A2	Rod Position Limits for Four Pump Operation - Unit 1	3.5-15a
3.5.2-1B1	Rod Position Limits for Four Pump Operation - Unit 2	3.5-16
3.5.2-1B2	Rod Position Limits for Four Pump Operation - Unit 2	3.5-16a
3.5.2-1B3	Deleted	
3.5.2-1C1	Rod Position Limits for Four Pump Operation - Unit 3	3.5-17
3.5.2-1C2	Rod Position Limits for Four Pump Operation - Unit 3	3.5-17a
3.5.2-1C3	Rod Position Limits for Four Pump Operation - Unit 3	3.5-17b
3.5.2-2A1	Rod Position Limits for 2 and 3 Pump Operation - Unit 1	3.5-18
3.5.2-2A2	Rod Position Limits for 2 and 3 Pump Operation - Unit 1	3.5-18a
3.5.2-2B1	Rod Position Limits for 2 and 3 Pump Operation - Unit 2	3.5-19
3.5.2-2B2	Rod Position Limits for 2 and 3 Pump Operation - Unit 2	3.5-19a
3.5.2-2B3	Deleted	
3.5.2-2C1	Rod Position Limits for 2 and 3 Pump Operation - Unit 3	3.5-20
3.5.2-2C2	Rod Position Limits for 2 and 3 Pump Operation - Unit 3	3.5-20a
3.5.2-2C3	Rod Position Limits for 2 and 3 Pump Operation - Unit 3	3.5-20b
3.5.2-3A1	Operational Power Imbalance Envelope - Unit 1	3.5-21
3.5.2-3A2	Operational Power Imbalance Envelope - Unit 1	3.5-21a
3.5.2-3B1	Operational Power Imbalance Envelope - Unit 2	3.5-22
3.5.2-3B2	Deleted	

LIST OF FIGURES (CONT"D)

Figure		Page
3.5.2-3B3	Deleted	
3.5.2-3C1	Operational Power Imbalance Envelope - Unit 3	3.5-23
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3.5.2-3C3	Operational Power Imbalance Envelope - Unit 3	3.5-23b
3.5.2-4A1	APSR Position Limits - Unit 1	3.5-24
3.5.2-4A2	APSR Position Limits - Unit 1	3.5.24a
3.5.2-4B1	APSR Position Limits - Unit 2	3.5-25
3.5.2-4B2	APSR Position Limits - Unit 2	3.5-25a
3.5.2-4B3	Deleted	
3.5.2-4C1	APSR Position Limits - Unit 3	3.5-26
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3.5.2-5	LOCA - Limited Maximum Allowable Linear Heat	3.5-27
3.5.4-1	Incore Instrumentation Specification Axial Imbalance Indication	3.5-31
3.5.4-2	Incore Instrumentation Specification Radial Flux Tilt Indication	3.5-32
3.5-4-3	Incore Instrumentation Specification	3.5-33
4.5-1-1	High Pressure Injection Pump Characteristics	4.5-4
4.5-1-2	Low Pressure Injection Pump Characteristics	4.5-5
4.5.2-1	Acceptance Curve for Reactor Building Spray Pumps	4.5-9
6.1-1	Station Organization Chart	6.1-7
6.1-2	Management Organization Chart	6.1-8

can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 106.5 percent of 131.3 x 10^6 lbs/hr.). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2A are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and rod bowing:

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.05 kw/ft for Unit 1.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, and 3 of Figure 2.1-2A correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

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

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

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

References

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

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

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 2.

Power peaking is not a directly observable quantity, and, therefore, limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, and 3 of Figure 2.1-2B correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

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

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

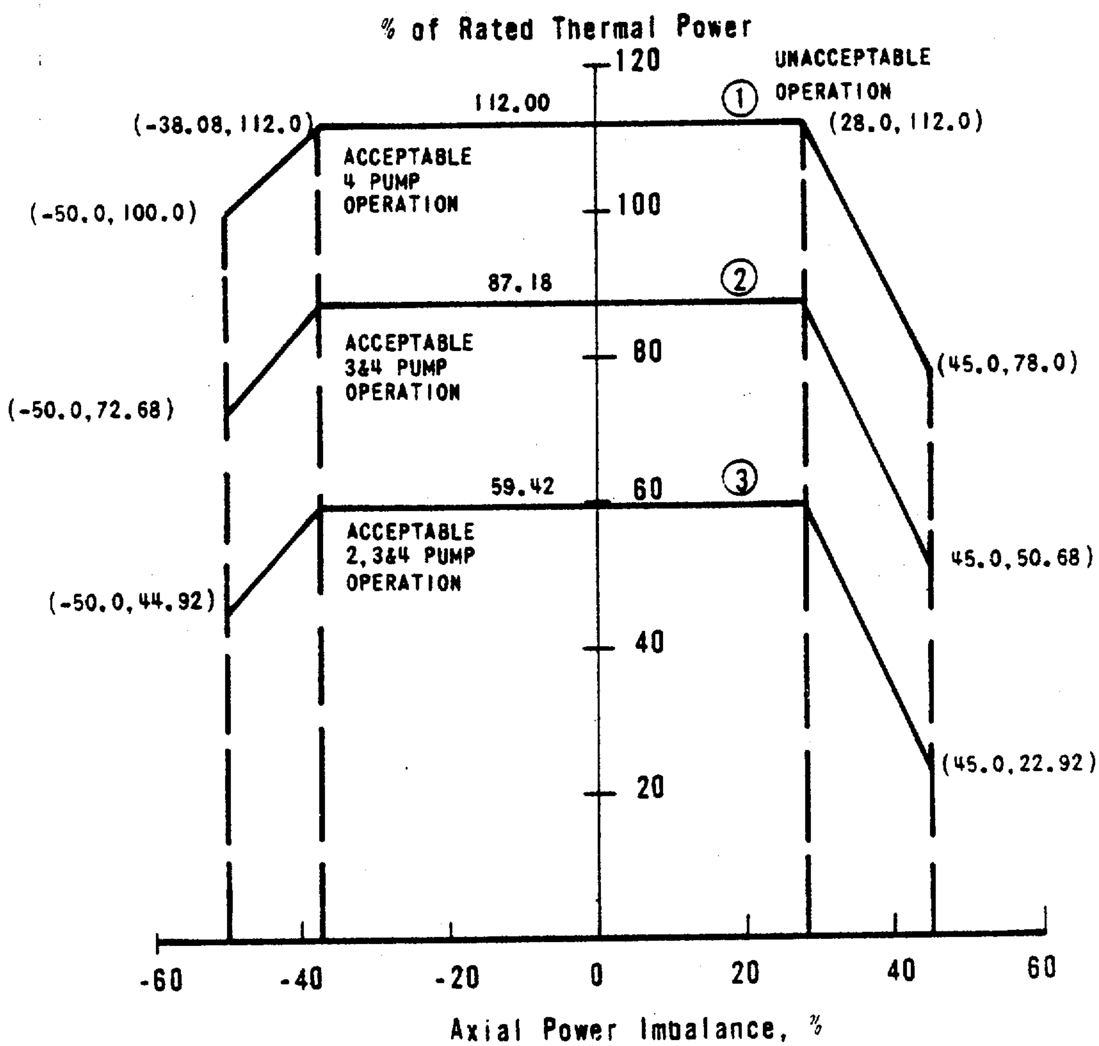
For each curve of Figure 2.1-3B, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The 1.30 DNBR curve for four-pump operation 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) Corelation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March 1970.

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

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



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

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

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis. (4)

Overpower Trip Based on Flow and Imbalance

The power level trip setpoint produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1A are as follows:

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

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

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

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

For safety calculations the maximum calibration and instrumentation errors for the power level trip were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2
2.3-2C - Unit 3

level trip and associated reactor power/reactor power-imbalance boundaries by 1.08% - Unit 1 for 1% flow reduction.
1.08% - Unit 2
1.08% - Unit 3

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 setpoint is reached before the nuclear overpower trip setpoint. 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)

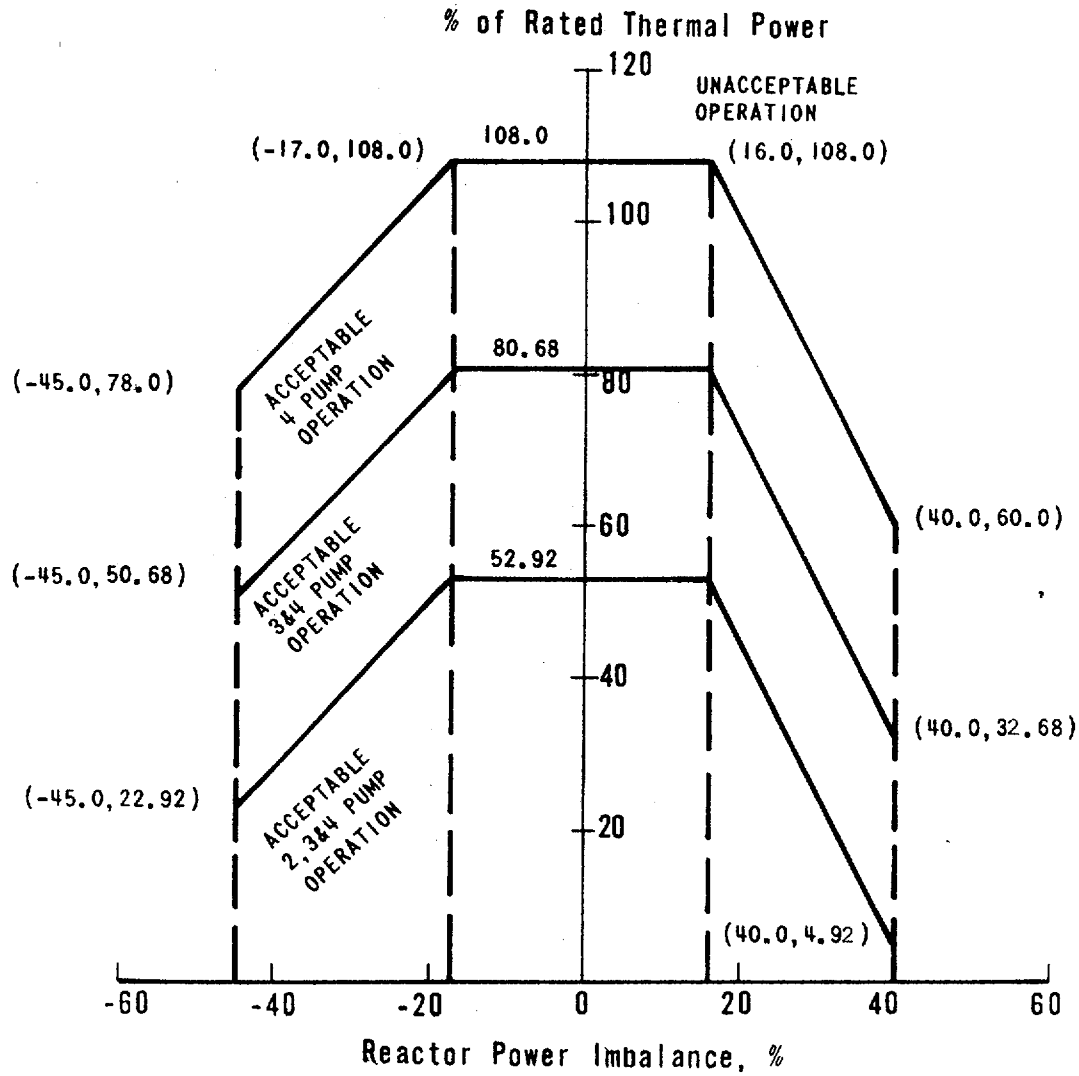
Due to the calibration and instrumentation errors, the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)

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 setpoint 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.



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

Table 2.3-1B
Unit 2

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0[3]
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	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

(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.

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

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

3.5.2.5 Control Rod Positions

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

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

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

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

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, and 3.5.2-1A2 for Unit 1; Figures 3.5.2-1B1, and 3.5.2-1B2, 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:

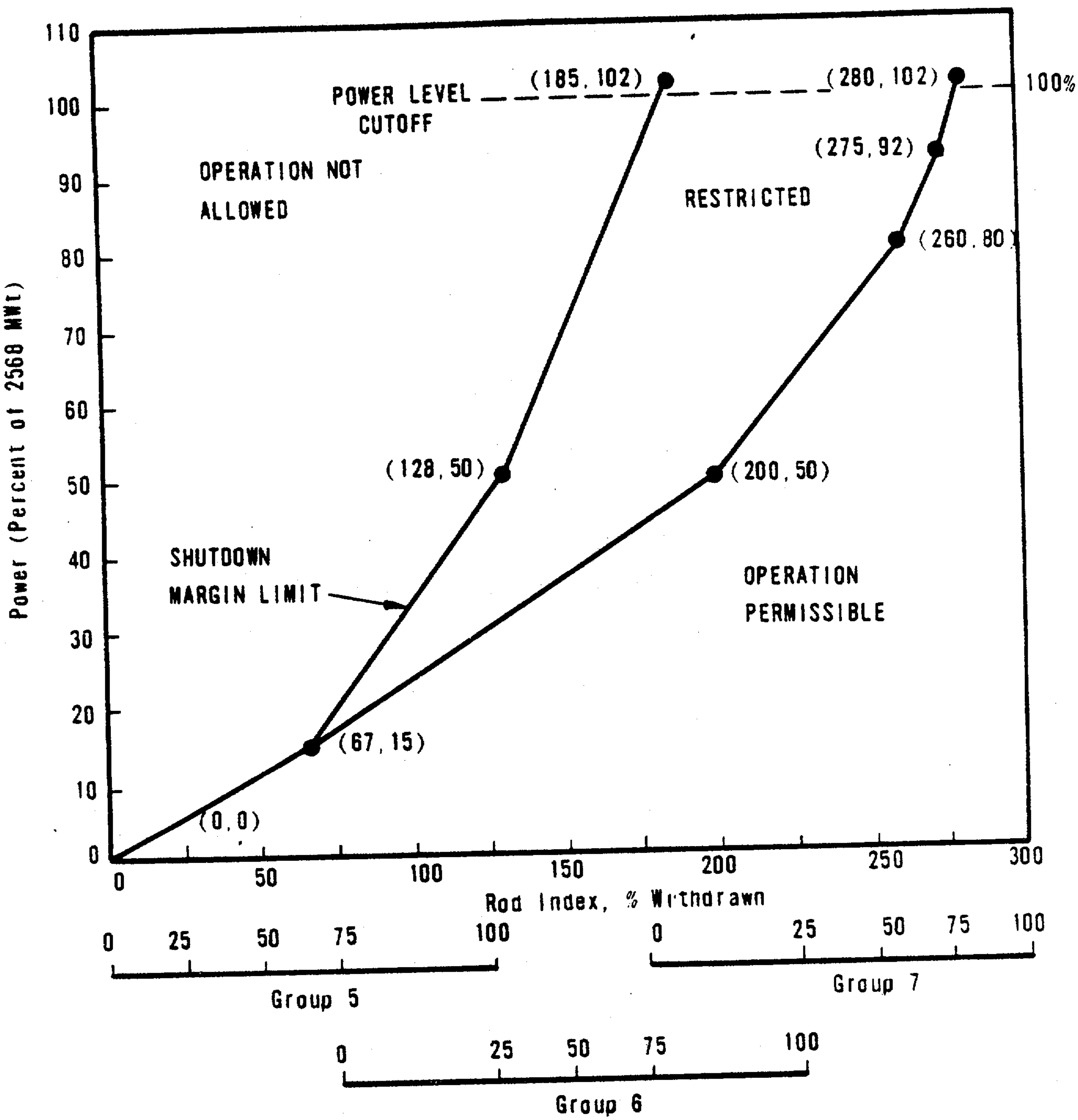
1. Xenon reactivity did not deviate more than 10 percent from the equilibrium value for operation at steady state power.

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 is approach to its equilibrium value for operation at the power level cutoff.

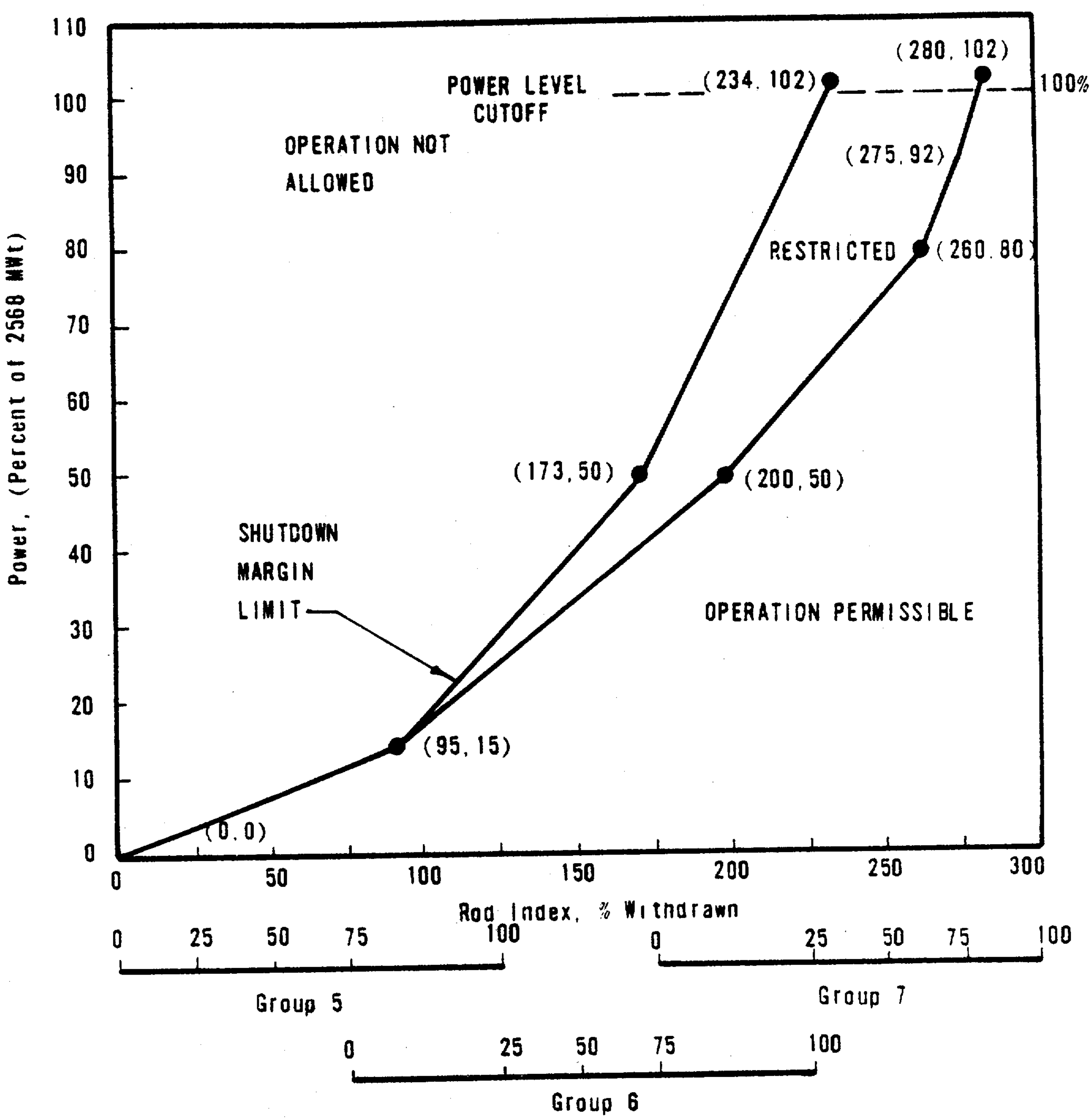
3. Except for xenon free startup (when 2. applies), the reactor has operated within a range of 87 to 92 percent of rated thermal power for a period exceeding 2 hours.

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, 3.5.2-3B1, 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.

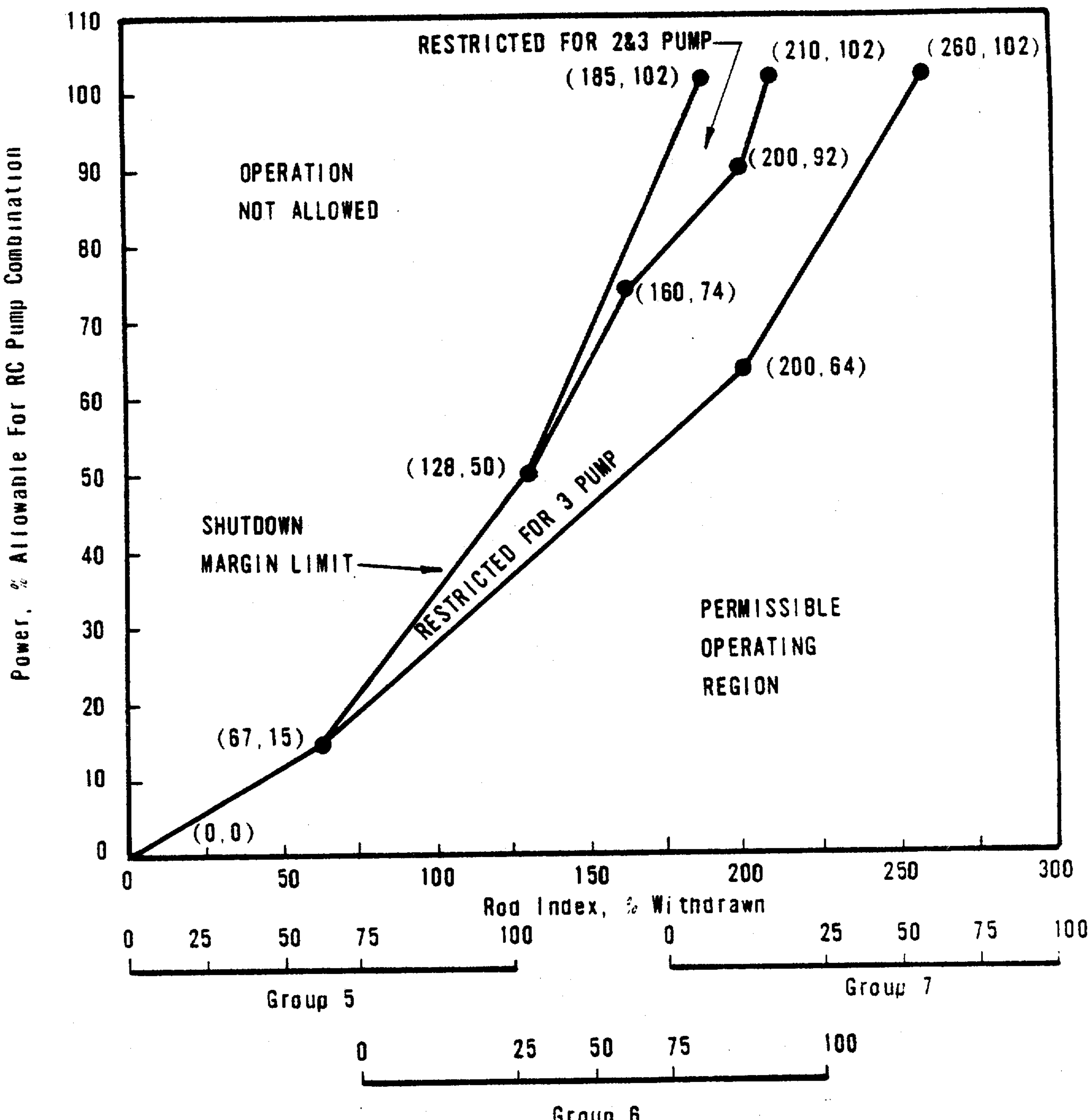
3.5.2.8 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.



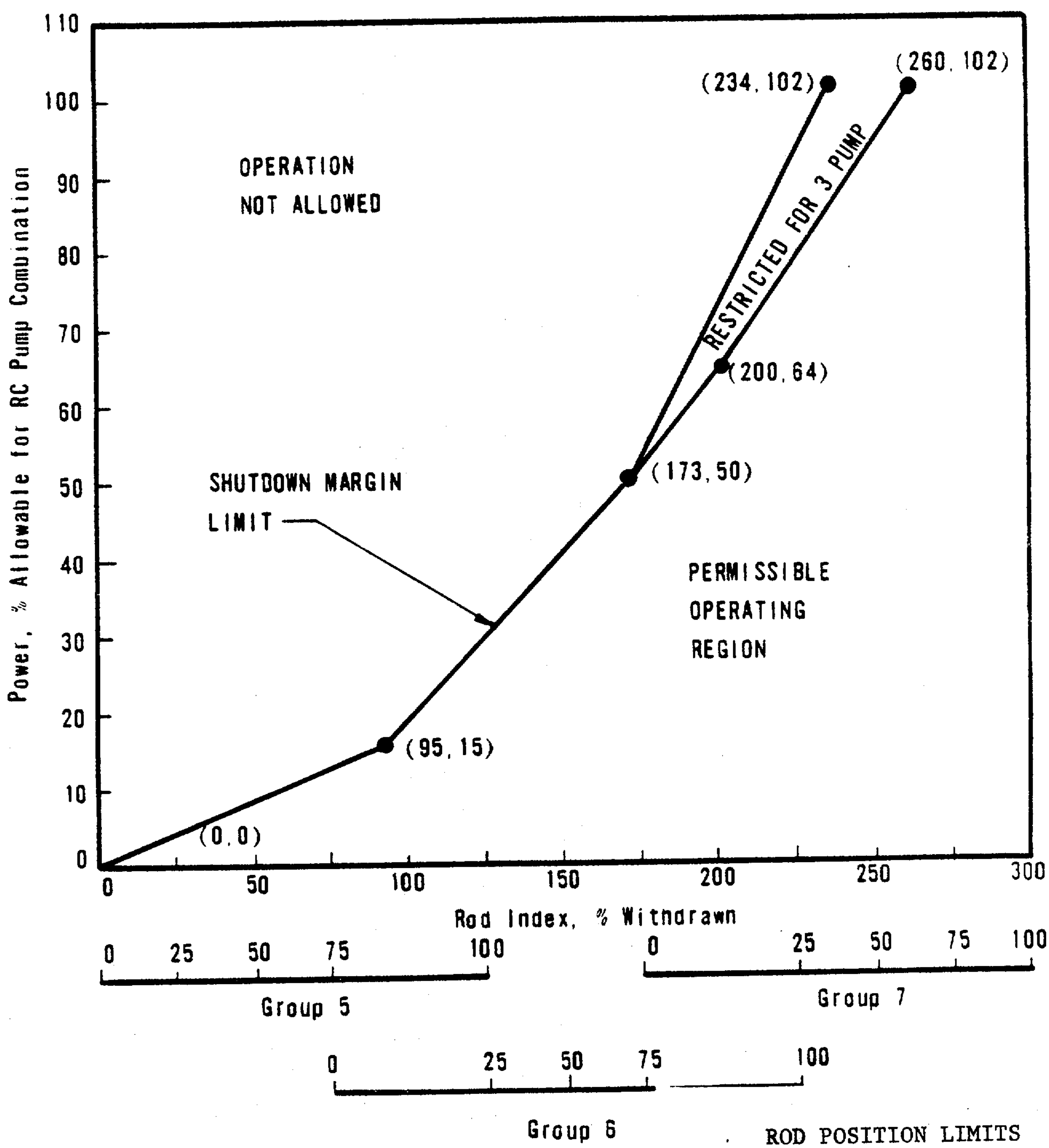
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 150 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-1B1



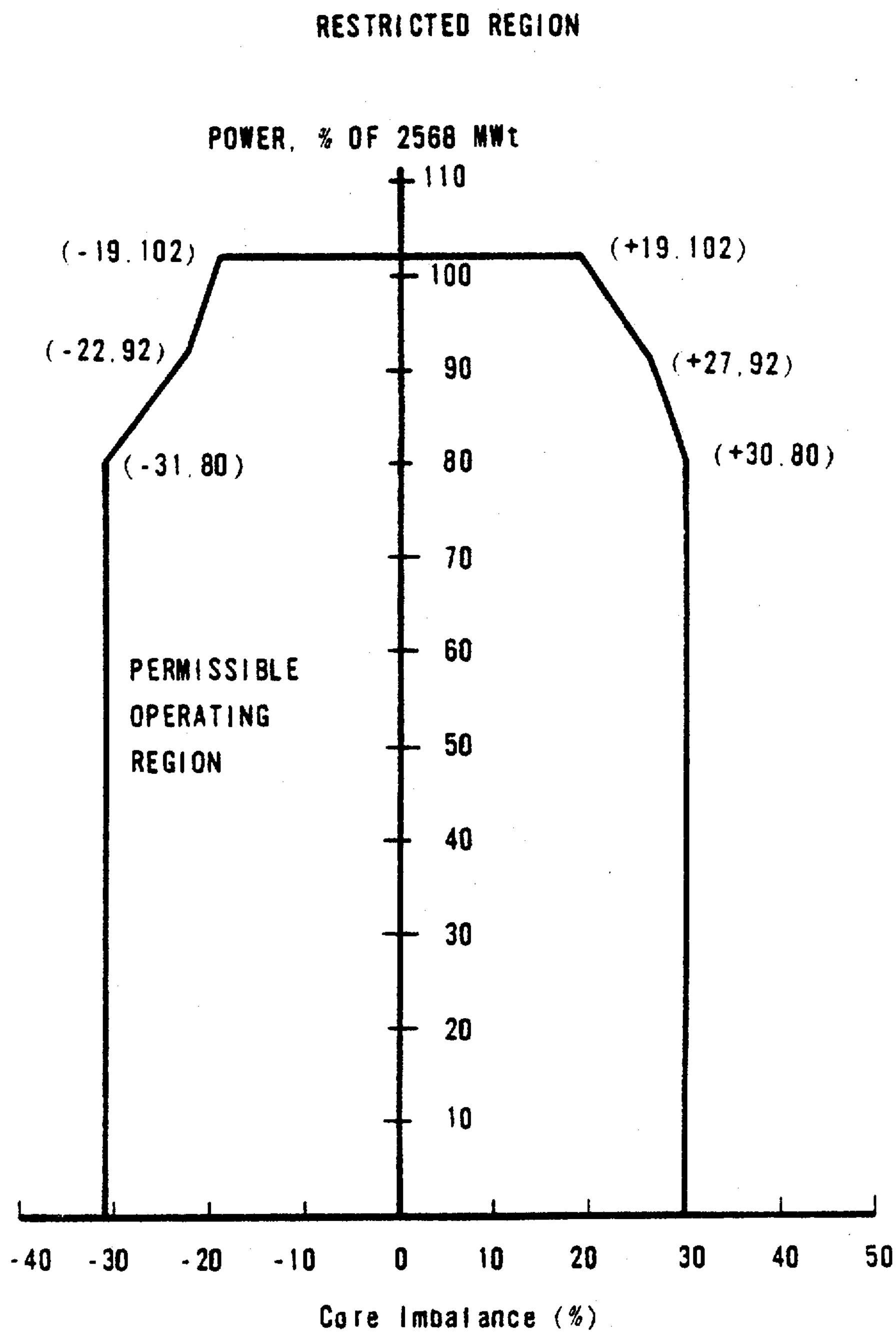
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 150 ± 10 TO 360 ± 10
EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-1B2



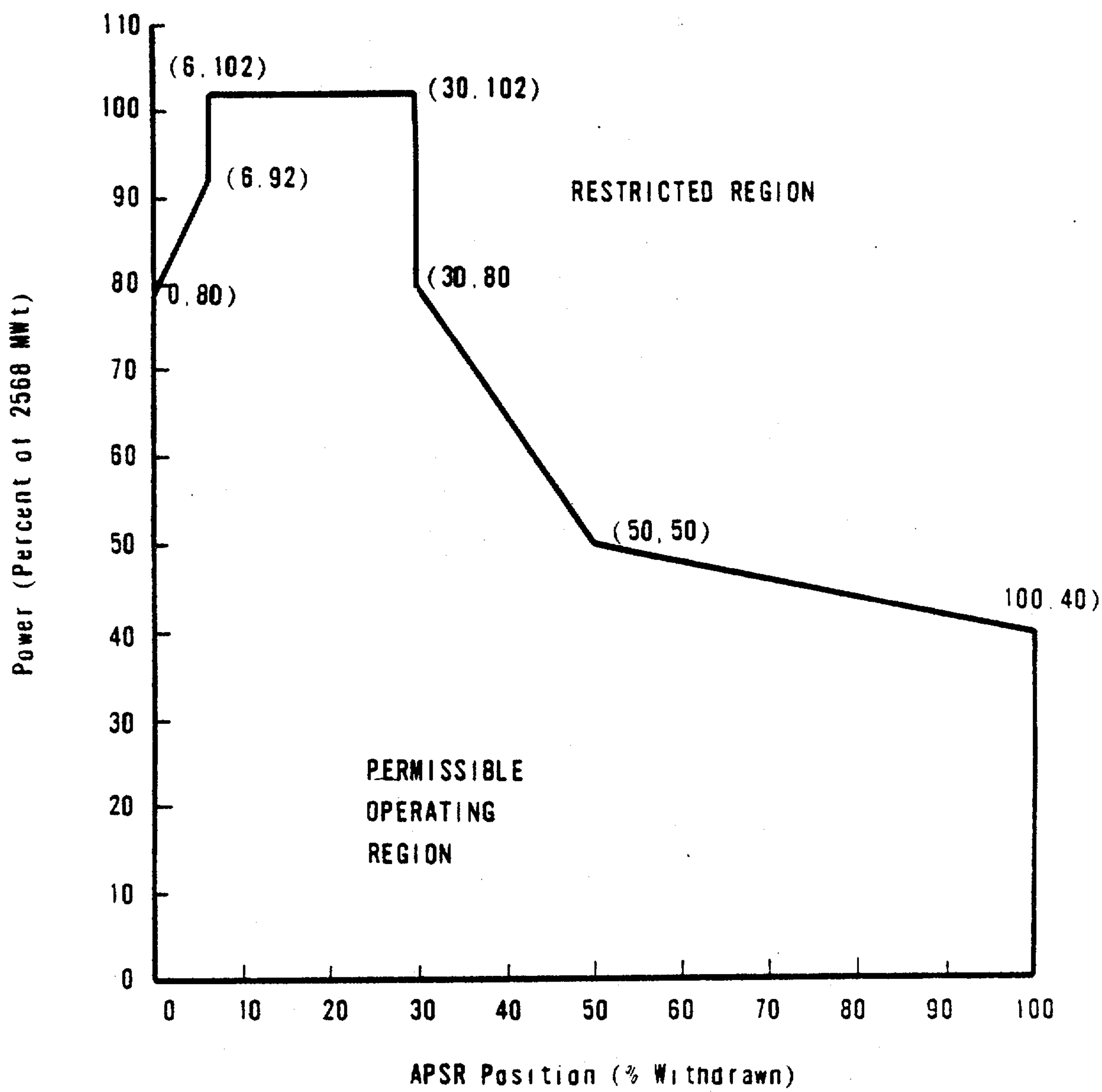
ROD POSITION LIMITS
FOR TWO AND THREE PUMP
OPERATION FROM
0 TO 150 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-2B1



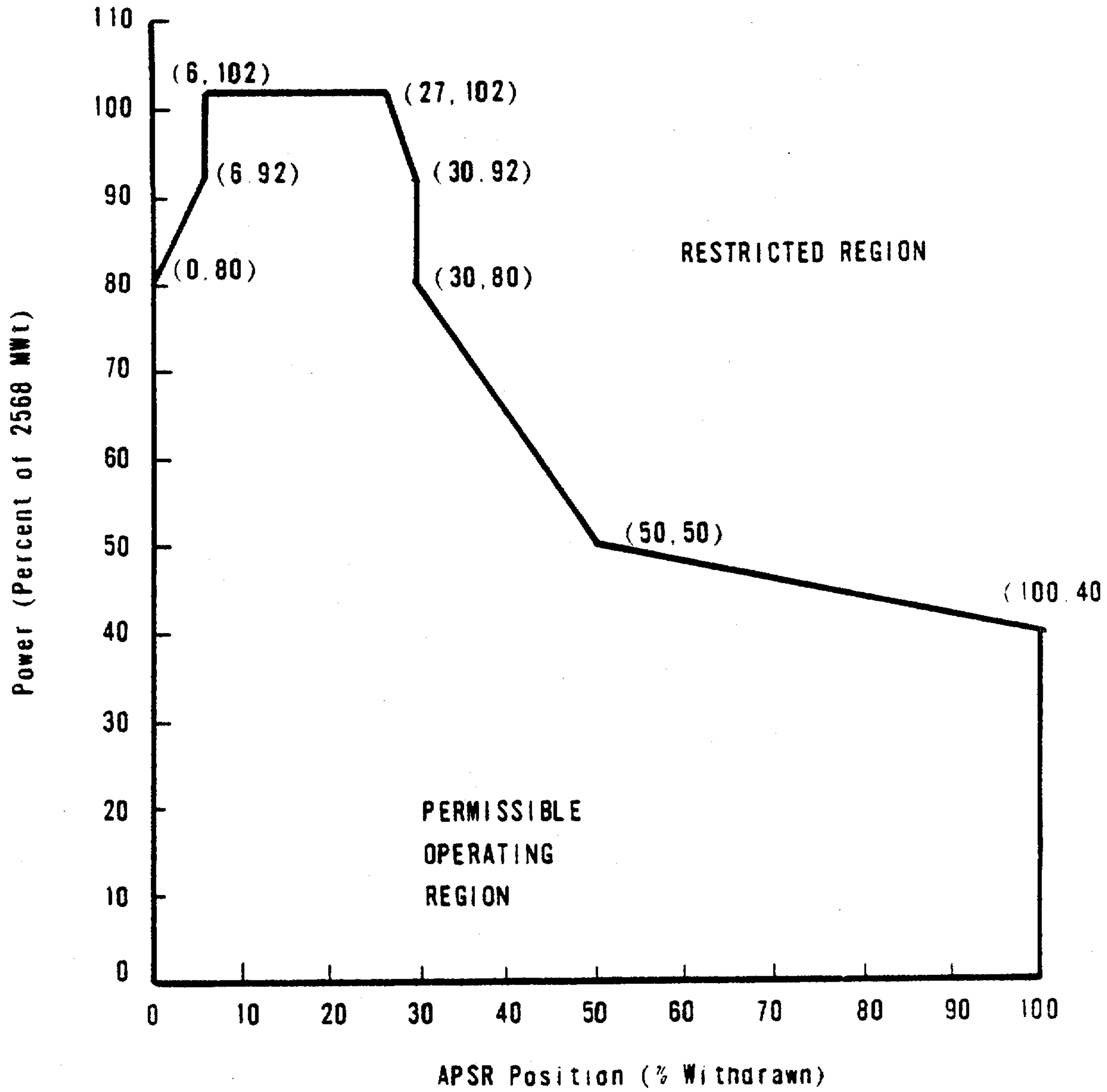
ROD POSITION LIMITS
FOR TWO AND THREE PUMP
OPERATION FROM
150 ± 10 TO 360 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-2B2



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



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



APSR POSITION LIMITS FOR OPERATION
FROM 150 ± 10 EFPD TO
360 ± 10 EFPD
Unit 2
OCONEE NUCLEAR STATION
Figure 3.5.2-4B2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 80 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 letters dated November 16, 1979, and March 12, 1980 (1, 2), Duke Power Company (DPC or the licensee) requested amendment to Appendix A of License No. DPR-47 for the Oconee Nuclear Station, Unit No. 2. The DPC submittal of March 12, 1980, was provided to substitute for the November 16, 1979 application in its entirety due to the extended Cycle 4 operation (350 Effective Full Power Days (EFPDs) vs. 297 EFPDs) provided by our License Amendment of January 4, 1980. This increased cycle length necessitated a complete core redesign. Therefore, the proposed amendment to the Technical Specifications (TSs) allowing full power operation of Unit 2 based on an extended Cycle 4, and the Babcock and Wilcox (B&W) Topical Report BAW-1565, Rev. 1 (2), were presented in support of Cycle 5 operations. The topical report describes the fuel system design, accident analyses, and the startup test program. The design length of the proposed Cycle 5 operation is 360 EFPDs. At the end of the current cycle, Cycle 4, a total of 68 burned fuel assemblies will be discharged and 68 fresh fuel assemblies will be loaded in the core in a checkerboard pattern. The Cycle 5 operational mode will be changed from rodded to feed-and-bleed. This change is not regarded as a major change in operating mode since Oconee 2 was operated essentially in a rods-out configuration during the latter part of the previous cycle. A similar change from rodded to feed-and-bleed mode was approved for Oconee Unit 3, Cycle 5[3]. Reactivity is controlled by 61 full-length Ag-In-Cd control rods, soluble boron shim, and 56 burnable poison rod assemblies (BPRAs). The latter is required to offset the increased excess reactivity built into the longer cycle length. In addition to the full-length control rods, eight axial power shaping rods are provided for axial power distribution control. The following sections present the evaluations of any changes to the fuel system design, accident and transient analysis, startup physics testing, nuclear and thermal-hydraulic design, and the proposed TS changes required for Cycle 5 operation.

By letter dated September 11, 1979(20) the licensee also requested an amendment to all three Oconee Units to incorporate a secondary water chemistry monitoring program in the body of the license.

2.0 Evaluation of Core Design Modifications

2.1 Fuel System Design

2.1.1 General

To achieve the longer cycle (Cycle 5) length and to utilize an in-out-in fuel management scheme, the operational mode is being changed from rodded to feed-and-bleed and an average core fuel enrichment increase is applied. To ensure that achieved core power distributions conform with values assumed in the safety and setpoint analyses, monthly incore power maps are to be compared with predicted distributions and deviations are to be reported to the NRC. In addition to the longer cycle length and fuel management changes, two fuel assemblies in Cycle 5 are demonstration 17 X 17 Mark-CR assemblies. The Mark-CR demonstration assemblies of batch 5 are mechanically identical in function to the Mark-C assemblies of batch 4 described in Reference 4; these assemblies have been twice burned in previous cycles. One Mark-BZ demonstration fuel assembly is included in batch 7. The Mark-BZ is a 15 X 15 fuel assembly similar to the Mark-B assembly described in Reference 5 except that six intermediate spacer grids are of Zircaloy material, and an Inconel 718 spring replaced the Inconel X750 holddown spring. The Mark-BZ assembly is described in Reference 6, which also states that reactor safety and performance are not adversely affected by the presence of the one Mark-BZ demonstration assembly.

2.1.2 Rod Design

The fuel pellet end configuration has changed from a spherical dish for batches 1 through 6 to a truncated cone dish for batch 7. This minor design change facilitates manufacturing while maintaining the same end void volume. We conclude that fuel performance will not be adversely affected by this change and it is thus acceptable.

2.1.3 Cladding Creep Collapse

Due to its longer accumulated incore exposure, the fuel of batch 5 is more limiting than the fuel in other batches. The batch 5 assembly power histories were analyzed and the most limiting Mark-B and Mark-C assemblies were used to perform the creep collapse analysis using the CROV computer code and procedures described in Reference 7. The collapse time for the most limiting assemblies were both conservatively determined to be more than 30,000 effective full power hours (EFPHs), which is greater than the maximum projected residence time for Cycle 5 operation. We conclude that cladding collapse has been adequately considered.

2.1.4 Cladding Stress and Strain

For design evaluation, the primary stress is less than two-thirds of the minimum specified unirradiated yield strength, and all stresses (primary and secondary) are less than the minimum specified unirradiated yield strength. The licensee states that the stress analysis has a margin in excess of 30%.

The fuel design criteria specify a limit of 1.0% on cladding circumferential plastic strain. The pellet design is established for plastic cladding strain of less than 1.0% at maximum design local pellet burnup (55,000 MWD/MTU) and heat generation rate (20.15 KW/ft). Oconee 2 fuel will not operate up to these maximum allowable values. We conclude that the cladding stresses and strains to be experienced by the Cycle 5 fuel are acceptable.

2.1.5 Thermal Design

All fuel assemblies in this core are thermally similar. The fresh batch 7 fuel inserted for Cycle 5 operation introduces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. Linear Heat Rate (LHR) capabilities are based on centerline fuel melt and were established using TAFY 3 code (8) with consideration for fuel densification. We conclude that the fuel thermal design is acceptable.

2.2 Nuclear Design

The design core physics parameters for Cycle 5 are generated using the B&W version of PDQ07 (9, 10, 11) and are compared to the Cycle 4 parameters (2, Table 5-1). The boron concentrations for Cycle 5 are higher due to the additional reactivity necessary for the longer cycle which is not completely offset by the BPRAs. The control rod worth differs between cycles due to changes in burnup and radial flux distributions. Cycle 5 shutdown margin is calculated to be 3.27% $\Delta K/K$ and 2.46% $\Delta K/K$ for beginning of cycle (BOC) and end of cycle (EOC) conditions, respectively, with the maximum worth rod stuck out of the core. The required shutdown margin is 1.0% $\Delta K/K$; thus, the BOC and EOC shutdown margins are acceptable.

2.3 Thermal-Hydraulic Design

The incoming batch 7 fuel is hydraulically and geometrically similar to the fuel remaining from previous cycles. The thermal-hydraulic models and methodologies used to support Cycle 5 operation are described in References 5, 12, and 13. The main differences between Cycle 5 and the reference Cycle 4 are discussed below.

2.3.1 Core Bypass Flow

The maximum core bypass flow due to the removal of all orifice rod assemblies (ORAs) in Cycle 4 was 10.4%. For Cycle 5 operation, 56 BPRAs will be inserted, leaving 50 vacant assemblies, resulting in a decrease in calculated maximum core bypass flow to 8.1% (i.e., net increase in core flow). A flux/flow trip setpoint of 1.08 was established to compensate for the increase in core flow.

2.3.2 BPRA Retainers

The retainers added to provide positive holddown of BPRAs introduce a small Departure from Nucleate Boiling Ratio (DNBR) penalty as discussed in Reference 14. However, the increase in core flow due to the BPRA insertion (Section 2.3.1 above) more than compensates for the decrease in DNBR due to the BPRA retainers.

2.3.3 Mark-CR and Mark-BZ Demonstration Assemblies

The two Mark-CR demonstration assemblies will be limited to a design peak of 1.50, and the Mark-BZ low-absorption demonstration assembly will be limited to a 1.40 design peak. This will assure both peaking and DNBR margin for Mark-CR and Mark-BZ assemblies and certify that they are not limiting for reactor protection. The 1.71 design radial-local peak remains valid for all other assemblies.

2.3.4 Rod Bow DNBR Penalty

The rod bow penalty applicable to Cycle 5, according to the licensee, was calculated using the interim rod bow penalty evaluation procedure approved in Reference 15. The limiting (maximum radial x local peak) fuel assembly for Cycle 5 is a batch 7 assembly at a projected burnup of 15,219 MWD/MTU. The calculated rod bow penalty using this procedure is 0.5%. Utilizing the 1% Departure from Nucleate Boiling (DNB) credit for the flow area reduction factor, the actual penalty is zero; therefore, no penalty is applied to the DNB calculations and they are thus acceptable.

3.0 Evaluation of Accidents and Transients

The licensee has examined each Final Safety Analysis Report (FSAR) accident analysis with respect to changes in Cycle 5 parameters to determine their effect on the plant thermal performance during the analyzed accidents and transients. The key parameters having the greatest effect on the outcome of a transient or accident are the core thermal parameters, thermal-hydraulic parameters, and physics and kinetics parameters. Fuel thermal analysis values are listed in Table 4-2 of Reference 2 for all fuel batches in Cycle 5. Table 6-1 of the same reference compares the thermal-hydraulic parameters for Cycles 4 and 5. These parameters are the same for both cycles with the exception of the higher value of design Maximum Departure from Nucleate Boiling Ratio (MDNBR) for Cycle 5 (2.05 as compared to 1.98 for Cycle 4). A comparison of the key kinetic parameters from the FSAR and Cycle 5 is provided in Table 7-1 of Reference 2. These comparisons indicate no significant changes (Table 4-1 of Ref. 2 compared to Table 4-1 of Reference 12) or changes in the conservative direction (Tables 6-1, 7-1 of Reference 2). The effects of fuel densification on the FSAR accident analyses have been evaluated in Reference 13.

A generic Loss of Coolant Accident (LOCA) analysis for the B&W 177-fuel assembly, lowered loop Nuclear Steam Supply System (NSSS) has been performed using the final acceptance criteria Emergency Core Cooling System (ECCS) evaluation model (Reference 17). That analysis used the limiting values of key parameters for all plants in the 177-FA lowered loop category, and therefore is bounding for the Oconee 2 Cycle 5 operation.

We conclude from the examination of Cycle 5 core thermal and kinetic properties, with respect to acceptable previous cycle values and with respect to the FSAR values, that this core reload will not adversely affect the Oconee 2 plant's ability to operate safely during Cycle 5.

4.0 Emergency Core Cooling System

An Exemption was granted on December 18, 1978, to 10 CFR 50.46(a), "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." The Exemption provided for its own termination upon completion of the modifications required by the Exemption. By letter dated December 13, 1978 (18), we found the design of the modifications to be acceptable. DPC has installed the modifications at Oconee 2 (19) and prepared acceptable operating procedures; thus, we conclude that the as-modified ECCS required by the Exemption of December 18, 1978, is acceptable.

5.0 Startup Test Program

Startup tests have been proposed by DPC to provide assurance that Oconee 2 has been loaded as intended. This test program is similar to that used at Oconee Nuclear Station and other B&W reactors. We have reviewed the test program and find it acceptable.

6.0 Technical Specification Changes

Proposed modifications to the Oconee 2 TSs needed to support Cycle 5 operation are described below (2):

(1) The effect of transient xenon on power peaking is conservatively accounted for by the xenon penalty factor of 5%.

(2) Primarily due to the decrease in core bypass flow, the power-to-flow ratio has been increased to 1.08. Reactor trip setpoints of power based on flow have been correspondingly changed.

(3) Oconee 2 will be changed from a rodded to a feed-and-bleed mode of operation for Cycle 5. A similar change was approved for Oconee Unit 3 for Cycle 5 (3).

(4) The following limits have been changed:

  a. Axial power and reactor power imbalance safety limits and trip setpoints for 2, 3, and 4 reactor coolant pump operation.

  b. Rod position limits for 2, 3, and 4 reactor coolant pump operation for less than 150 EFPDs and for more than 150 EFPDs during the cycle life.

c. Axial power shaping rod position limits for less than 150 EFPDs and for more than 150 EFPDs.

d. Operational power imbalance limits for 0 to 360 EFPDs.

We have evaluated the reload report for the Oconee 2 Cycle 5 operation and the proposed TS changes that reflect the changed parameters for the new cycle and find the revised TSs acceptable.

7.0 Boron Dilution Accidents

These accidents were not addressed in the DPC March 12, 1980(2) license amendment submittal as the licensee previously submitted his analyses of these accidents in the FSAR. The Oconee FSAR did not specifically include the analysis for moderator dilution after the reactor vessel was drained down to the bottom of the lowest nozzle to enable maintenance work, such as steam generator repairs or reactor coolant pump repairs, and subsequent refilling of the primary system. The system during such work is in the cold shutdown mode and the accident of concern is refilling the system with unborated water which would result in a return to criticality, possibly, with all control rod assemblies fully inserted. While such an accident would not result in fuel failures nor release any radiation outside the primary system and surely none to the environment it is still an undesirable event. Our review of the licensee's Technical Specifications 3.5.1 and 3.5.2 indicate that the source range nuclear instrumentation channel is maintained in an operable condition during the cold shutdown mode and that a shutdown margin of 2.5% $\Delta k/k$ is maintained in this mode. Our position for plants currently receiving an operating license is for a shutdown margin of 1% $\Delta k/k$. Based on the above, we conclude that the Oconee Station is operated to mitigate the effects of a boron dilution accident. DPC is engaged in a program to further diminish the probability of a boron dilution accident by strengthening the Technical Specifications and Station Operating Procedures. This program will be implemented prior to any maintenance work requiring drainage of the primary system to the nozzles for any of the three Oconee reactors.

8.0 Control Rod Guide Tube Wear

By letter dated November 23, 1979 we requested DPC to inspect control rod guide tubes for wear. B&W performed the inspections for DPC on spent fuel assemblies in the Oconee spent fuel pools. The results of these tests, performed by eddy-current techniques, indicated negligible wear. Similar inspections were performed on spent fuel at the Rancho Seco plant with results that confirmed the Oconee results. We conclude that operation of Oconee 2 in Cycle 5 will not result in guide tube wear beyond design limits and is thus acceptable. Continued testing, particularly for fuel in the final cycle of extended cycles may still be needed before we can complete our evaluation of this problem.

9.0 Secondary Water Chemistry Monitoring

By letter dated September 11, 1979 DPC requested amendment to the Facility Operating Licenses to incorporate the monitoring program for secondary water chemistry in the body of the license.

In 1976, we sent letters to the licensees who operate Pressurized Water Reactors (PWRs) regarding the control of secondary water chemistry to inhibit corrosion of steam generator tubes. The letters requested the licensees to propose Technical Specification changes to incorporate limiting conditions for operation and surveillance requirements for secondary water chemistry parameters. This request was sent to DPC by letter dated August 18, 1976.

Many licensees objected to the Model Technical Specifications principally on the basis that they could unnecessarily restrict plant operation. The majority of these licensees submitted alternative approaches that were directed more toward monitoring and record keeping rather than specific limits on chemistry parameters. At the time of our request, we recognized that a major disadvantage of the Technical Specifications was a potential decrease in operational flexibility, but our request was motivated by an overriding concern for steam generator tube integrity. Our objective was to provide added assurance that licensees would properly monitor and control secondary water chemistry to limit corrosion of steam generator tubes.

However, based on the experience and knowledge gained since 1976, we concluded in mid-1979 that Technical Specification limits would not be the most effective way of accomplishing this objective. Due to the complexity of the corrosion phenomena involved, and the state-of-the-art as it exists today, we believe that a more effective approach would be to institute a license condition that requires the implementation of a secondary water chemistry monitoring and control program containing appropriate procedures and administrative controls. The required program and procedures would be developed by the licensees, with any needed input from their reactor vendors or other consultants, and thus could more readily account for site and plant specific factors that affect chemistry conditions in the steam generators. In our view, such a license condition would provide assurance that licensees would devote proper attention to controlling secondary water chemistry while also providing the needed flexibility to allow them to more effectively deal with any off-normal conditions that might arise.

Consequently, by letter dated July 23, 1979, we requested the licensee to propose such a license condition for the Oconee Station. The licensee responded on September 11, 1979 to our request and agreed to implement the program within 60 days from the issuing date of the proposed amendment. The proposed amendment complies with the guidance we provided to the licensee in our July 23, 1979 request. The NRC staff has made minor changes to the wording of the proposed license condition for the purpose of clarification. These changes were discussed with and concurred in by the licensee.

Based on our review, we have concluded that the addition of this license condition in conjunction with existing Technical Specifications on steam generator tube leakage and inservice inspection, will provide the most practical and comprehensive means of assuring that steam generator tube integrity is maintained; and thus, the proposed amendment is acceptable.

10.0 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.

11.0 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 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: June 12, 1980

REFERENCES

1. Letter, W. O. Parker (Duke Power Company) to H. R. Denton (NRC), dated November 16, 1979.

2. Letter, W. O. Parker (Duke Power Company) to H. R. Denton (NRC), dated March 12, 1980.

3. Letter, R. W. Reid, (NRC) to W. O. Parker (Duke Power Company), dated June 23, 1979, transmitting License Amendment for Oconee Unit 3, Cycle 5 reload.

4. Irradiation of Two 17 X 17 Demonstration Assemblies in Oconee 2, Cycle 2-Reload Report, BAW-1424, Babcock & Wilcox, Lynchburg, Virginia, January 1976.

5. Oconee Nuclear Station, Units 1, 2, and 3 - Final Safety Analysis Reports, Dockets Nos. 50-269, 50-270, and 50-287, Duke Power Company.

6. Mark-BZ Demonstration Assembly - Licensing Report, BAW-1533, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, March 1980.

7. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084, Rev. 2, Babcock & Wilcox, Lynchburg, Virginia, October 1978.

8. C. D. Morgan and H. S. Kao, TAFY-Fuel Pin Temperature and Gas Pressure Analysis, BAW-10084, Rev. 2, Babcock & Wilcox, Lynchburg, Virginia, October 1978.

9. B&W Version of PDQ 07 Code, BAW-10117A, Babcock & Wilcox, Lynchburg, Virginia, Janaury 1977.

10. Core Calculational Techniques and Procedures, BAW-10118, Babcock & Wilcox, Lynchburg, Virginia, October 1977.

11. Assembly Calculations and Fitted Nuclear Data, BAW-10116A, Babcock & Wilcox, Lynchburg, Virginia, May 1977.

12. Oconee Unit 2, Cycle 4 Reload Report, BAW-1491, Babcock & Wilcox, Lynchburg, Virginia, August 1978.

13. Oconee 2, Fuel Densification Report, BAW-1395, Babcock & Wilcox, Lynchburg, Virginia, June 1973.

14. BPRA Retainer Design Report, BAW-1496, Babcock & Wilcox, Lynchburg, Virginia, May 1978.

15. L. S. Rubenstein (NRC) to J. H. Taylor (B&W), Letter, "Evaluation of Interim Procedure for Calculating DNBR Reductions Due to Rod Bow," October 18, 1979.

16. Letter, W. O. Parker (Duke Power Company) to H. R. Denton (NRC), Revised Pages to BAW-1565, Rev. 1, "Oconee Unit 2, Cycle 5 Reload Report, " dated April 30, 1980.

17. ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103A, Rev. 2, Babcock & Wilcox, Lynchburg, Virginia, Spetember 1977.

18. Letter, R. W. Reid (NRC) to W. O. Parker (DPC) dated December 13, 1978.

19. Letter, W. O. Parker (DPC) to H. R. Denton (NRC) dated May 29, 1980.

20. Letter, W. O. Parker (DPC) to H. R. Denton (NRC) dated September 11, 1979.

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. 83, 83, and 80 to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55, respectively, issued to Duke Power Company (the licensee), which revised the licenses and the Station's common Technical Specifications for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. Except for the incorporation of the secondary water chemistry monitoring conditions, the amendments are effective as of the date of issuance. The secondary water chemistry monitoring conditions become effective within 60 days from the date of issuance.

These amendments revise the Technical Specifications to support the operation of Oconee Unit No. 2 at full rated power during Cycle 5. The amendments also revise the licenses for Units Nos. 1, 2, and 3 to incorporate monitoring conditions for secondary water chemistry.

The applications for the amendments comply 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 issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated September 22, 1976, as supplemented September 11, 1979, and the application for amendments dated November 16, 1979, as superseded March 12, 1980, and supplemented April 30, 1980, (2) Amendments Nos. 83, 83 , and 80 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, 201 South Spring 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 12th day of June 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing