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June 22, 1979

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

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. 73, 73 and 70 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 and are in response to your request dated March 30, 1979, as supplemented May 17, 1979.

These amendments revise the Technical Specifications to support the operation of Oconee Unit No. 3 at full rated power during Cycle 5. The amendments also revise the Technical Specifications for Units Nos. 1, 2 and 3 in regard to power level cut-off.

The Three Mile Island, Unit No. 2 (TMI-2) accident of March 28, 1979, resulted in core damage, initiated by a loss of feedwater and apparently exacerbated by operational errors. Through IE Bulletins 79-05, 05A and 05B dated, respectively, April 1, 5 and 21, 1979, issued to Duke Power Company, the NRC Office of Inspection and Enforcement identified corrective actions to be taken at the Oconee Nuclear Station.

You responded to this group of Bulletins by letters dated April 10, 13, 21 and 25, May 4, 5, 16 and 21 (two), 1979. Our preliminary evaluation of your responses and actions taken by Duke Power Company demonstrate understanding of the TMI-2 event and its relationship to the Oconee Nuclear Station. Your actions provide added protection to the health and safety of the public during Station operation. A separate Safety Evaluation will be issued to document our review of your Bulletin responses. We will also request additional information and identify required future actions relevant to your Bulletin responses.

Constr CP

7907230495

Oconee Unit No. 3, during Cycle 4, was operating under a July 6, 1978 Exemption to 10 CFR 50.46, the Emergency Core Cooling System (ECCS) rule. The enclosed Safety Evaluation, and our letter of December 13, 1978 provide the bases for terminating the Exemption, as your ECCS modifications and operating procedures have met the provisions of the Exemption. As a consequence of the TMI-2 accident, we issued an Order dated May 7, 1979, which required a reanalysis of very small breaks. We concluded in our May 18, 1979 Safety Evaluation, which permitted restart of Unit No. 3 after the Cycle 5 reload, that the ECCS was acceptable in that it could mitigate the effects of such very small breaks.

Based on our review of the installation of two electric motor driven emergency feedwater pumps at Oconee Unit No. 3, in addition to the existing steam turbine driven emergency feedwater pump and common header design, we conclude that the installation and test of the electrically driven pumps will not degrade the emergency feedwater system during operation of Unit No. 3. We will as part of our review to lift the long term conditions of the NRC May 7, 1979 Order, prepare a Safety Evaluation with a discussion of the mechanical, structural, electrical and hydraulic aspects of the motor driven pumps.

Within 30 days of receipt of this letter kindly provide us your schedule for submittal of the model used in the analysis for potential small breaks referenced in Enclosure D of your letter of May 7, 1979.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

[illegible]

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

Enclosures:
1. Amendment No. 73 to DPR-38
2. Amendment No. 73 to DPR-47
3. Amendment No. 70 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

B&O

COVER LETTER - MOTOR DRIVEN EFW PUMPS

MBJ

OFFICE	ORB#4:DOR	ORB#4:DOR	C-ORB#4:DOR	AD-E&P:DOR	OELD	
SURNAME	RIngram*	MFairtile:rf	RReid*	BGrimes*	Olmstead*	
DATE	6/ /79	6/15/79	6/ /79	6/ /79	6/ /79	

Duke Power Company

cc w/enclosure(s):
Mr. William L. Porter
Duke Power Company
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

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700 Shoreham Building
806 15th Street, N.W.
Washington, D. C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

U. S. Nuclear Regulatory Commission
Region II
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ATTN: Mr. Francis Jape
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Seneca, South Carolina 29678

Mr. Robert B. Borsum
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Nuclear Power Generation Division
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Bethesda, Maryland 20014

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

cc w/enclosure(s) and incoming
dtd.: 3/30/79 & 5/17/79

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. 73
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 March 30, 1979, as supplemented May 17, 1979, 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 <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73 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

Robert W. Reid

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 22, 1979

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. 73
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 March 30, 1979, as supplemented May 17, 1979, 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 <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73 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

Robert W. Reid

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 22, 1979

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. 70
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 March 30, 1979, as supplemented May 17, 1979, 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 <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 70 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

Robert W. Reid

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 22, 1979

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 73 TO DPR-38

AMENDMENT NO. 73 TO DPR-47

AMENDMENT NO. 70 TO DPR-55

Revise Appendix A as follows:

Remove Pages	Insert Pages
vi thru ix	vi thru x
2.1-3d	2.1-3d
2.1-9	2.1-9
2.3-3	2.3-3
2.3-10	2.3-10
2.3-13	2.3-13
3.5-11a thru 3.5-17	3.5-12 thru 3.5-17b
3.5-20 thru 3.5-20b	3.5-20 thru 3.5-20b
3.5-23 thru 3.5-23b	3.5-23 thru 3.5-23b
3.5-23c thru 3.5-31	3.5-24 thru 3.5-33

Changes on the revised pages are identified by marginal lines.

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES (CONT'D)

LIST OF FIGURES (CONT'D)

INTRODUCTION

These Technical Specifications apply to the Oconee Nuclear Station, Units 1, 2, and 3 and are in accordance with the requirements of 10CFR50, Section 50.36. The bases, which provide technical support or reference the pertinent FSAR section for technical support of the individual specifications, are included for informational purposes and to clarify the intent of the specification. These bases are not part of the Technical Specifications, and they do not constitute limitations or requirements for the licensee. The Technical Specifications while applying to Units 1, 2, and 3 are written on a single unit basis; exceptions to this are identified.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 3.

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

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

The maximum thermal power for three-pump operation is 87.2 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.08 = 80.7 percent power plus the maximum calibration and instrument error (Reference 4). The maximum thermal power for other coolant pump conditions are produced in a similar manner.

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

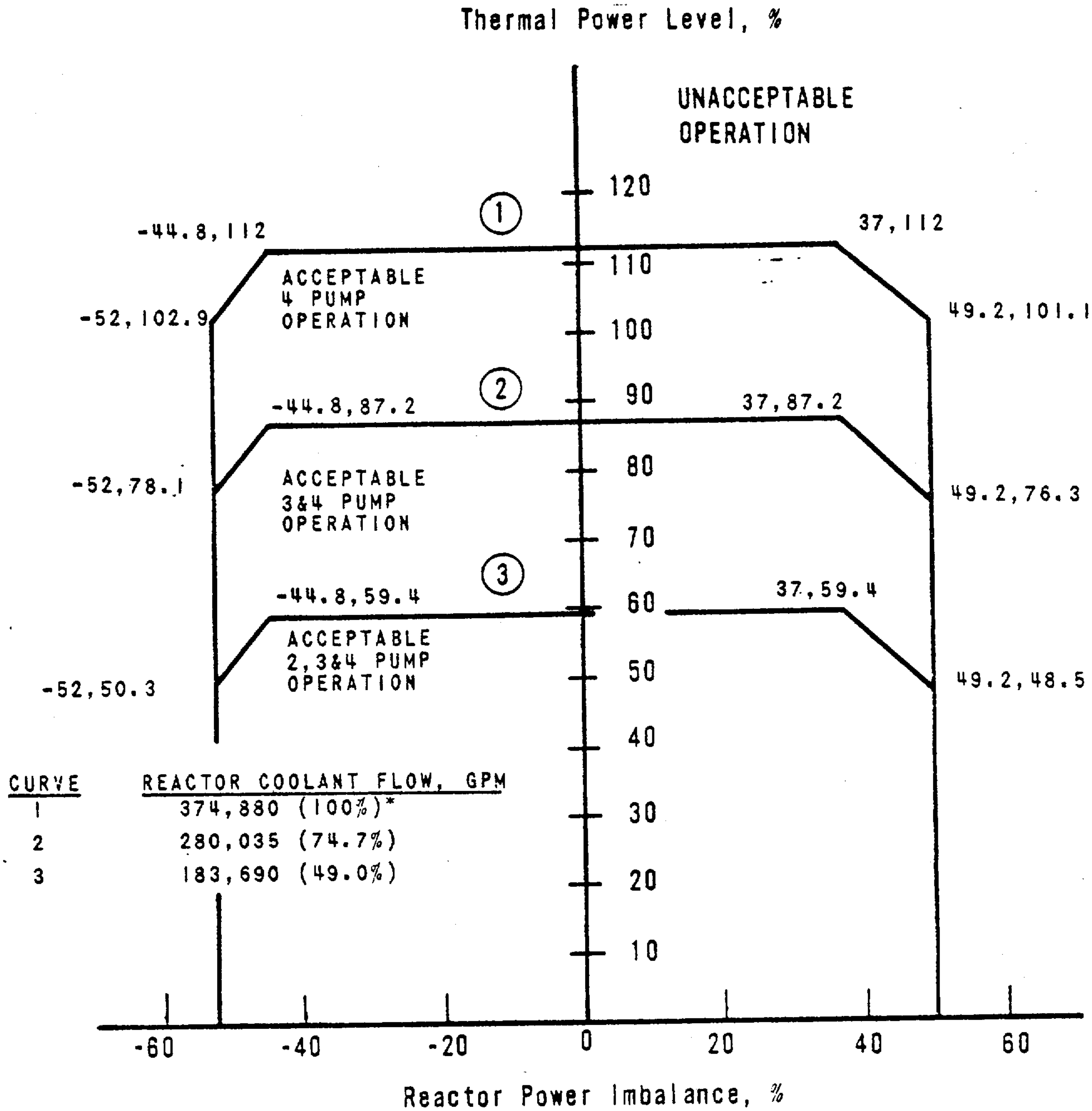
References

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

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

(3) Admendment 1 - Oconee 3, Cycle 4 - Reload Report - BAW-1486, June 12, 1978.

(4) Oconee 3, Cycle 5 - Reload Report - BAW-1522, March, 1979.



*106.5% OF FIRST CORE DESIGN FLOW

CORE PROTECTION
SAFETY LIMITS
UNIT 3
OCONEE NUCLEAR STATION
Figure 2.1-2C

Amendments Nos. 73, 73, &70

level trip and associated reactor power/reactor power-imbalance boundaries by 1.055% - Unit 1 for 1% flow reduction.
1.055% - 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 set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}$-4706) trip
(1800) psig ($11.14\ T_{out}$-4706)
(1800) psig ($11.14\ T_{out}$-4706)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

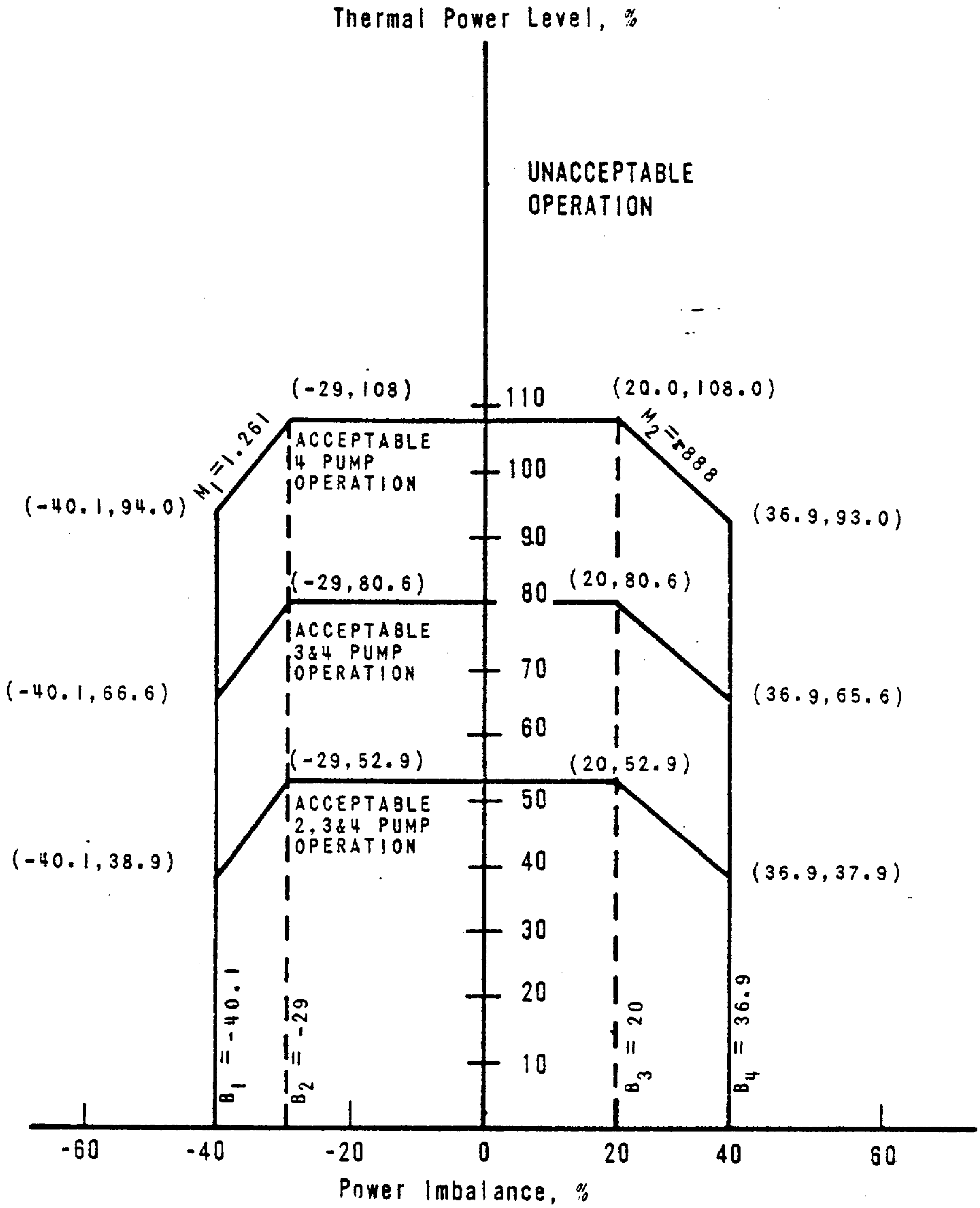
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619°F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip 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 3
OCONEE NUCLEAR STATION
Figure 2.3-2C

Table 2.3-1C
Unit 3

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 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.

Amendments Nos. 73, [illegible], & 70

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

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

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding 7.50% for Unit 1. The limits shown in Specification 3.5.2.4
7.50% for Unit 2
7.50% for Unit 3
are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.7, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

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

Technical Specification 3.5.2.6 provides the ability to prevent excessive power peaking by transient xenon at rated power.

Operating restrictions resulting from transient xenon power peaking, including xenon-free startup, are inherently included in the limits of Sections 3.5.2.5 (Control Rod Positions) and 3.5.2.7 (Reactor power imbalance) for transient peaking behavior bounded by the following factors. For feed and bleed (unrodded) operation, a 5% peaking increase is applied to calculated peaks at equilibrium conditions for powers at and above 90% FP. A 13% increase is applied below 90% FP. For rodded operation an 8% peaking increase is applied at and above 90% FP and an 18% increase is applied below 90% FP. If these values, checked every cycle, conservatively bound the peaking effects of all transient xenon, then the need for any hold at a power level cutoff below 100% FP is precluded. If not, either the power level at which the requirements of 3.5.2.6 must be satisfied or the above listed factors will be suitably adjusted to preserve the ECCS power peaking criteria. (Reference 6)

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

(3) FSAR, SUPPLEMENT 9

(4) B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)
BAW-1396 (UNIT 2)
BAW-1400 (UNIT 3)

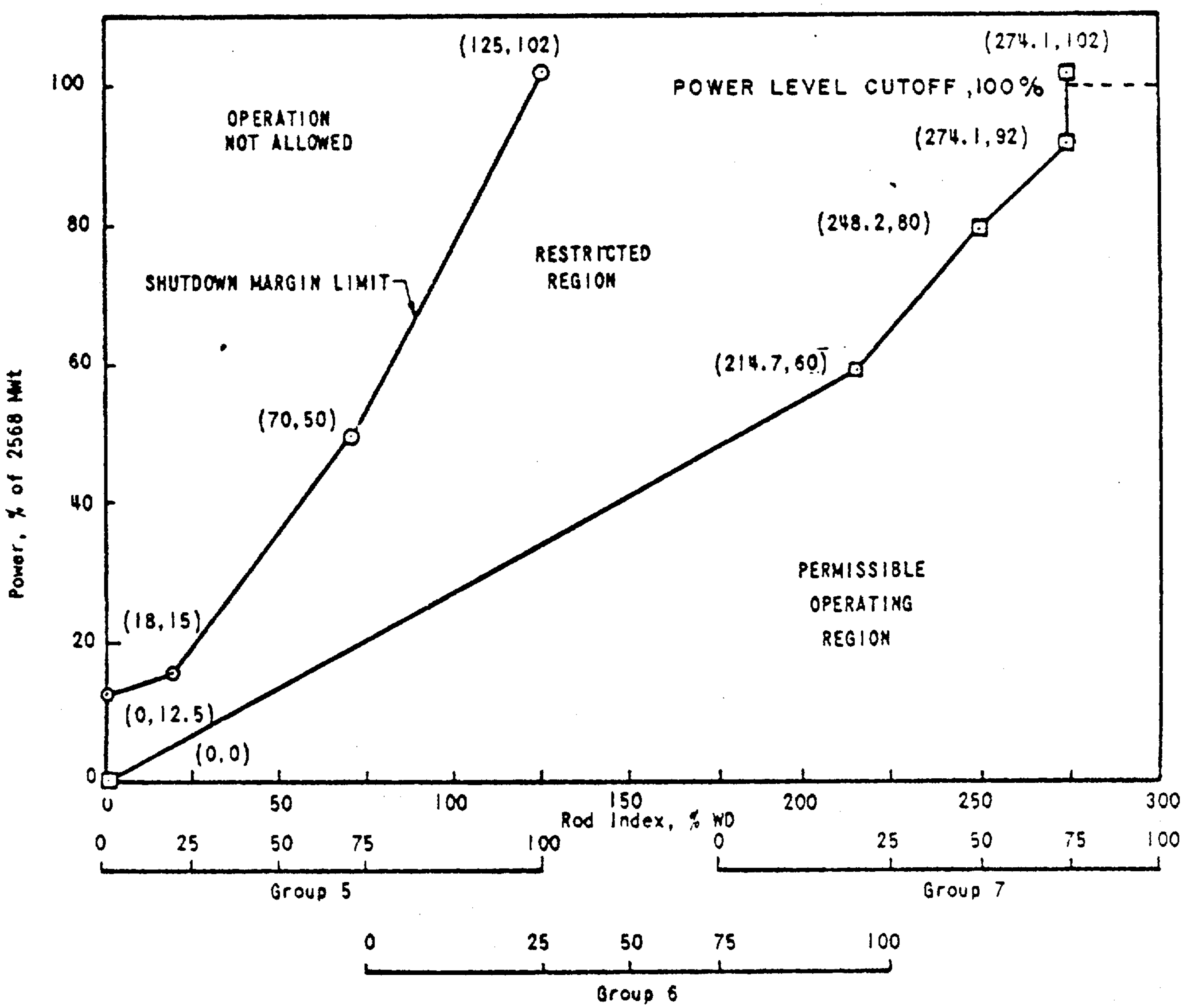
(5) Oconee 1, Cycle 4 - Reload Report - BAW 1447, March, 1977, Section 7.11

(6) Oconee 3, Cycle 5 - Reload Report - BAW-1522, March, 1979

TABLE 3.5-1

Quadrant Power Tilt Limits

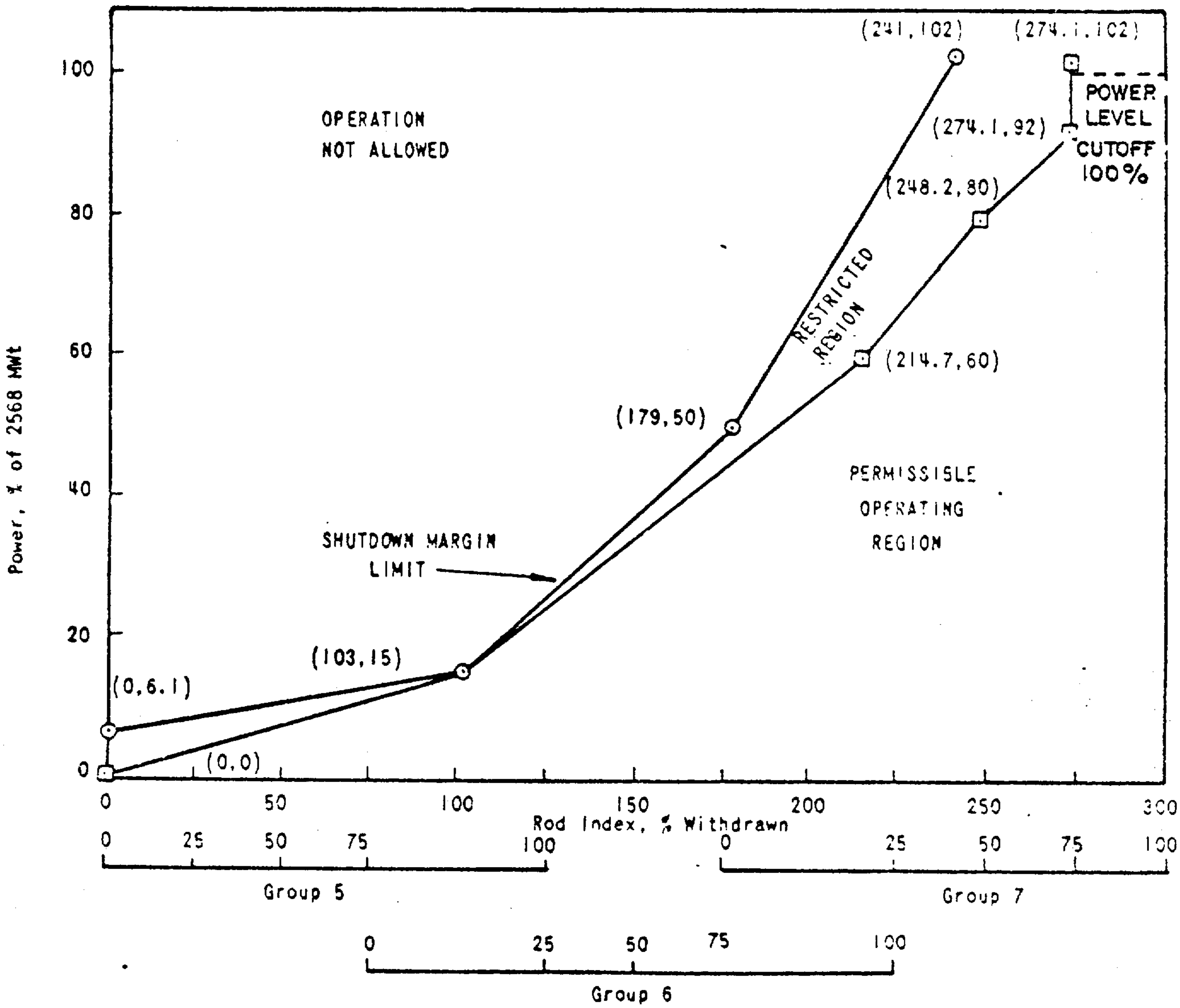
	Steady State Limit	Transient Limit	Maximum Limit
Unit 1	5.00	9.44	20.0
Unit 2	5.00	9.44	20.0
Unit 3	5.00	9.44	20.0



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 TO 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

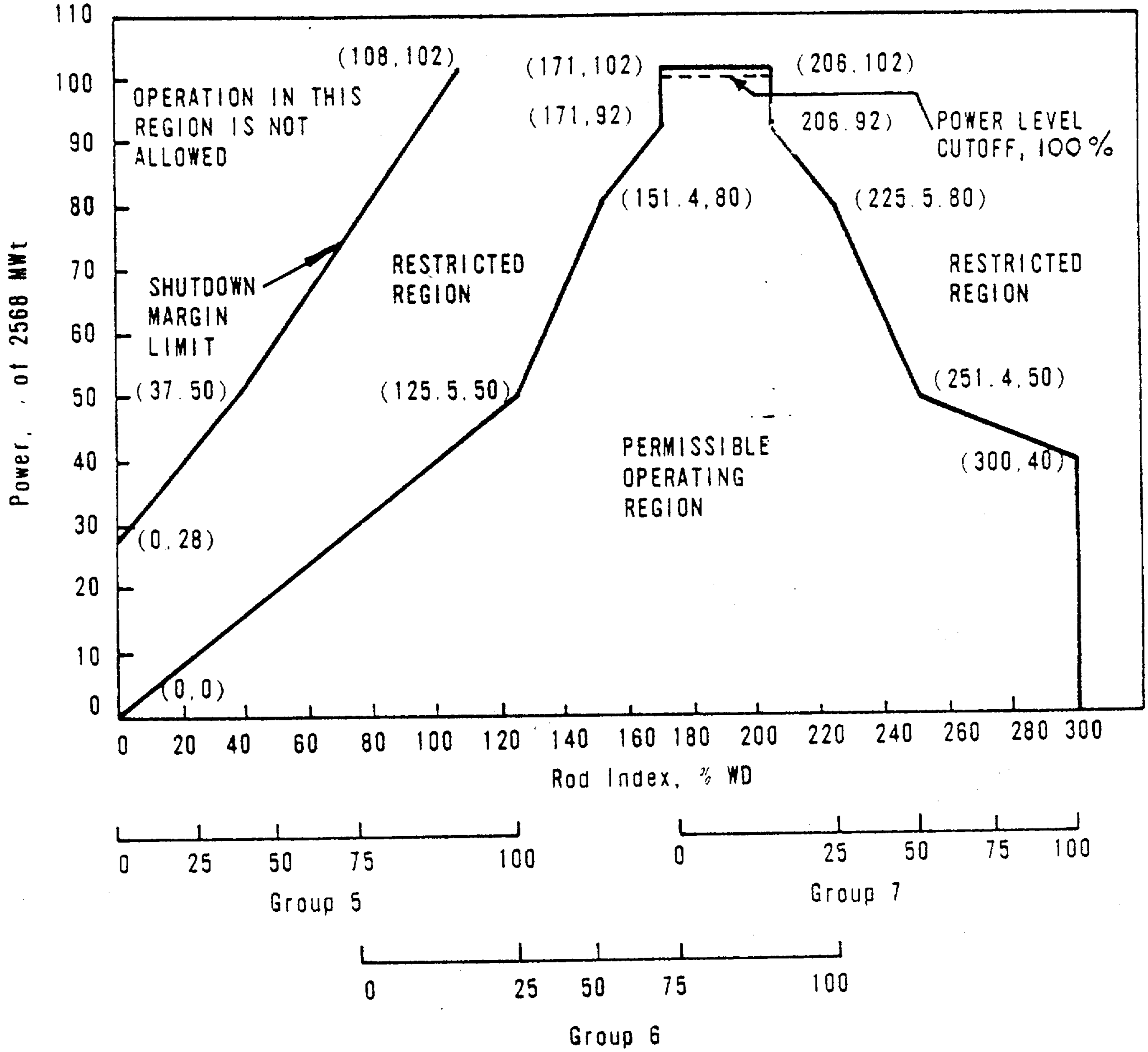
Figure 3.5.2-1A1



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 100 $\pm$ 10 EFPD
UNIT 1

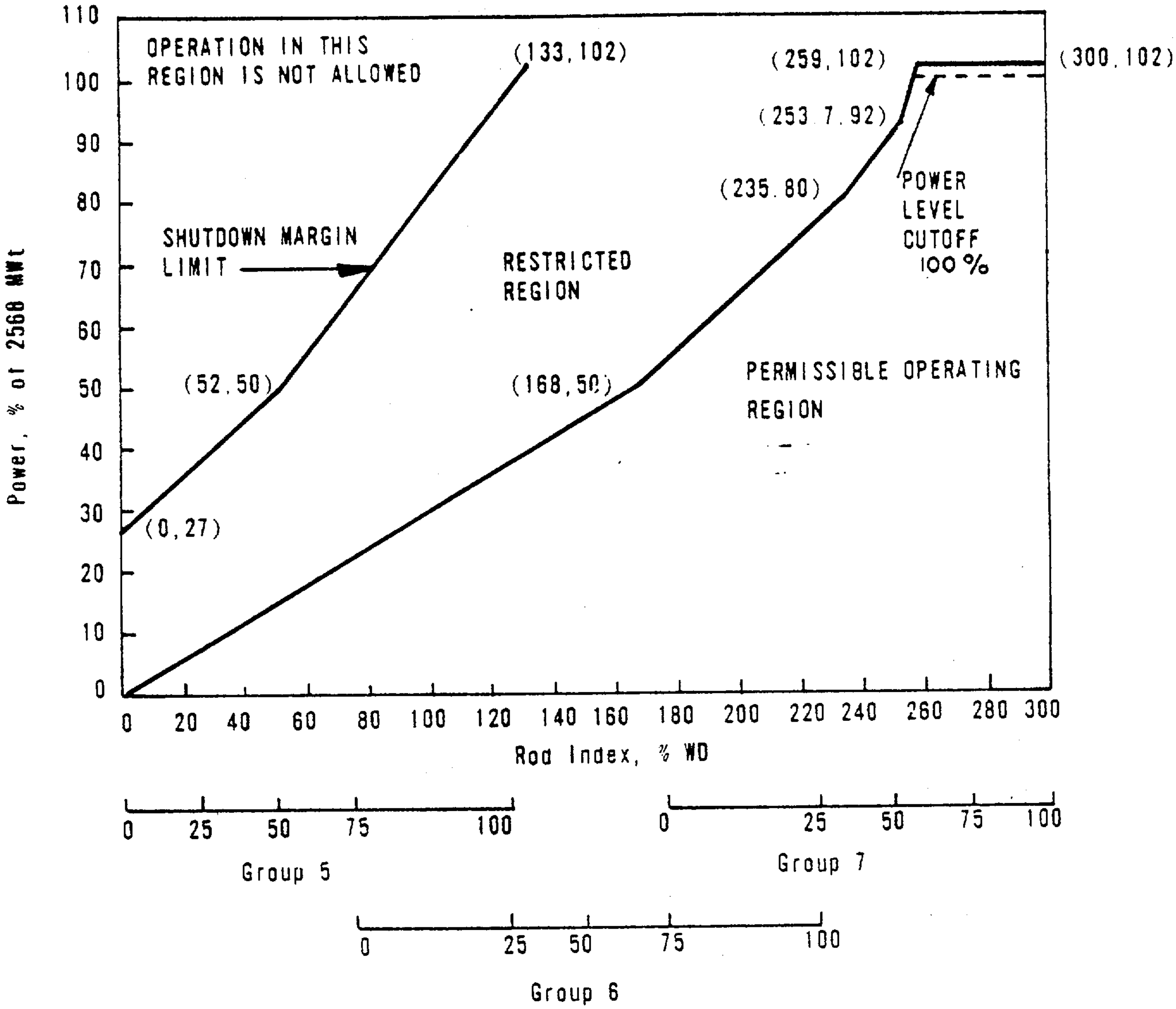
OCONEE NUCLEAR STATION

Figure 3.5.2-1A2



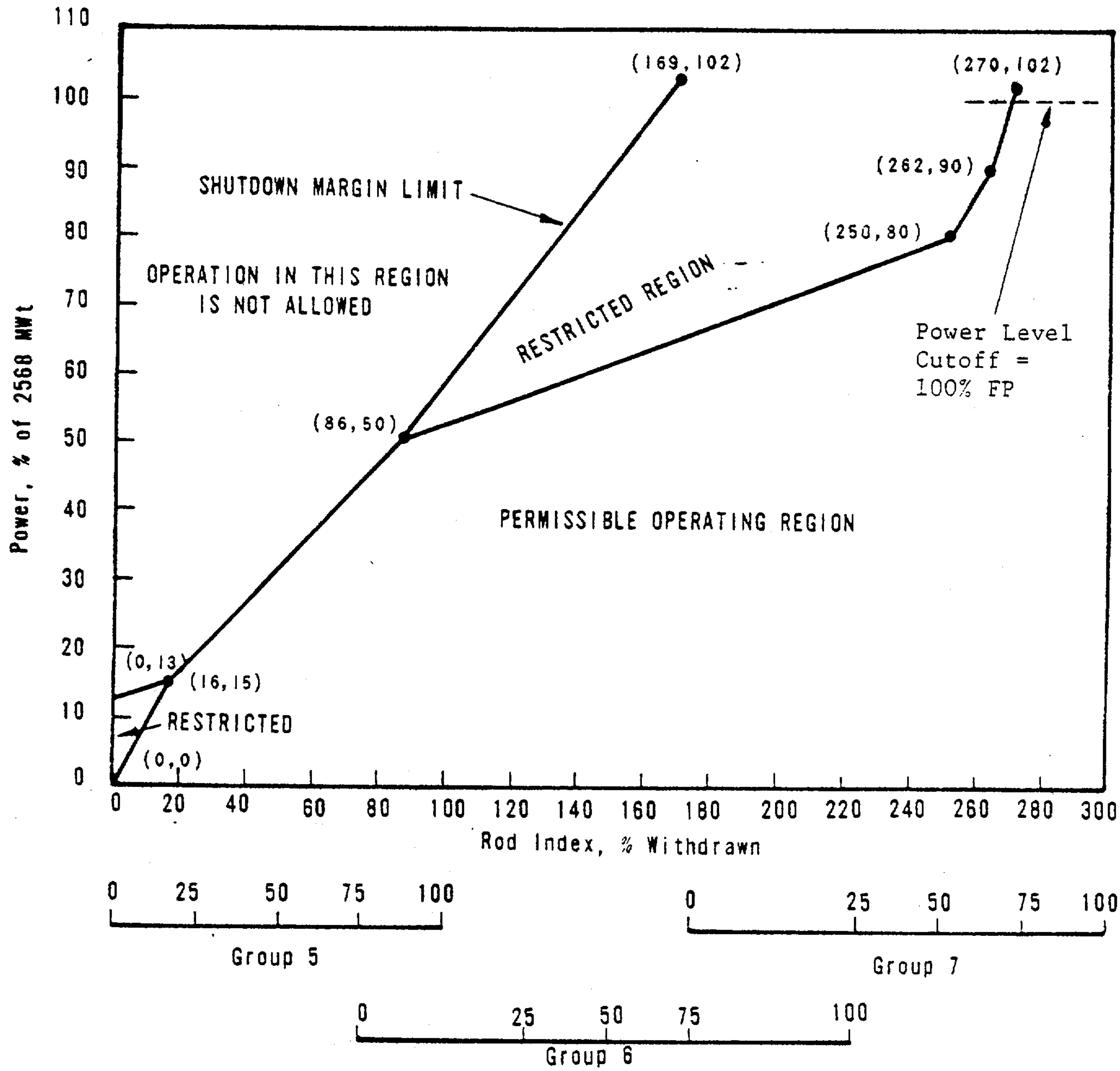
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 250 ± 10 EFPD
OCONEE 2

OCONEE NUCLEAR STATION
Figure 3.5.2-1B1

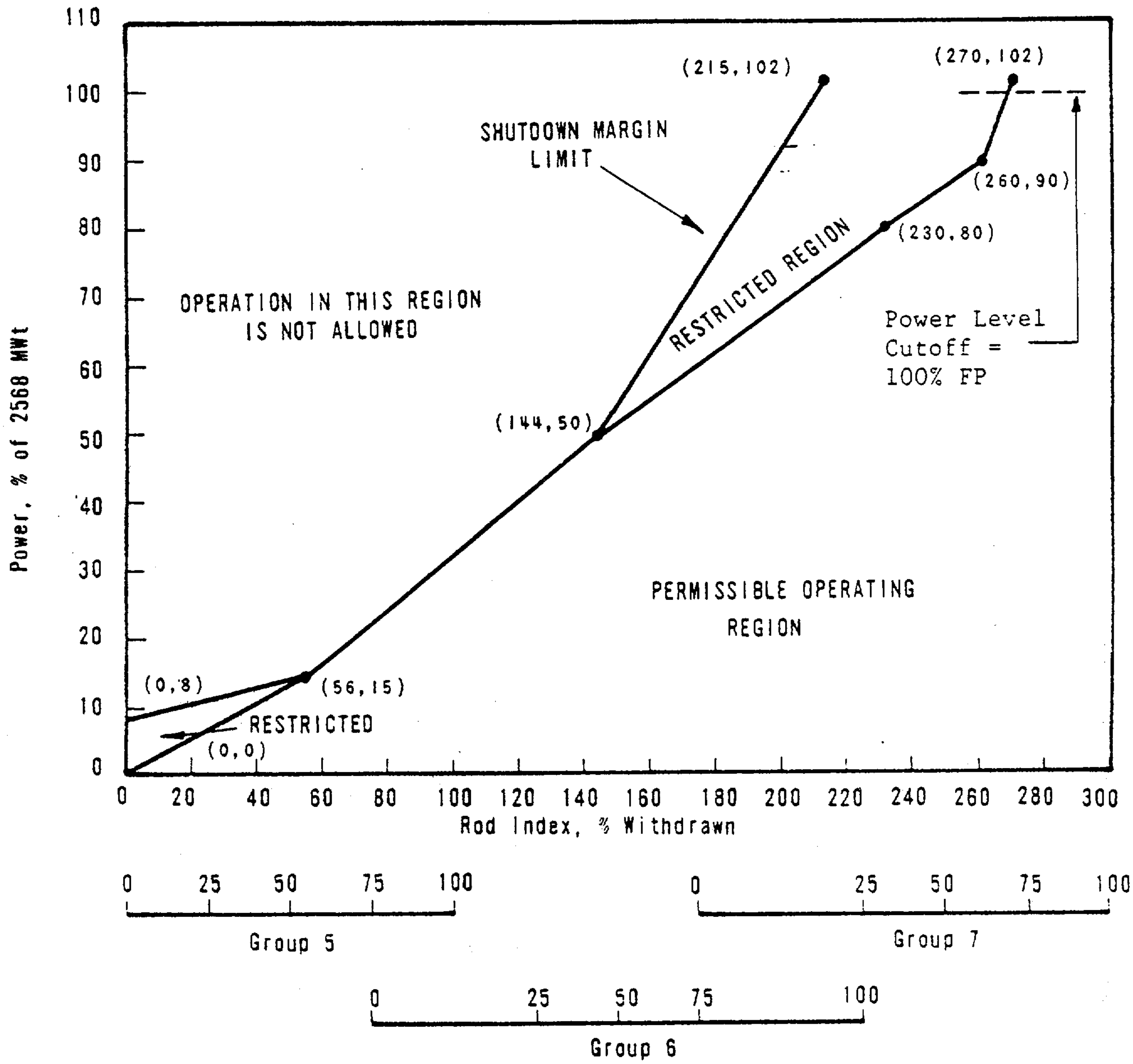


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 250 ± 10 EFPD
OCONEE 2
OCONEE NUCLEAR STATION
Figure 3.5.2-1B2

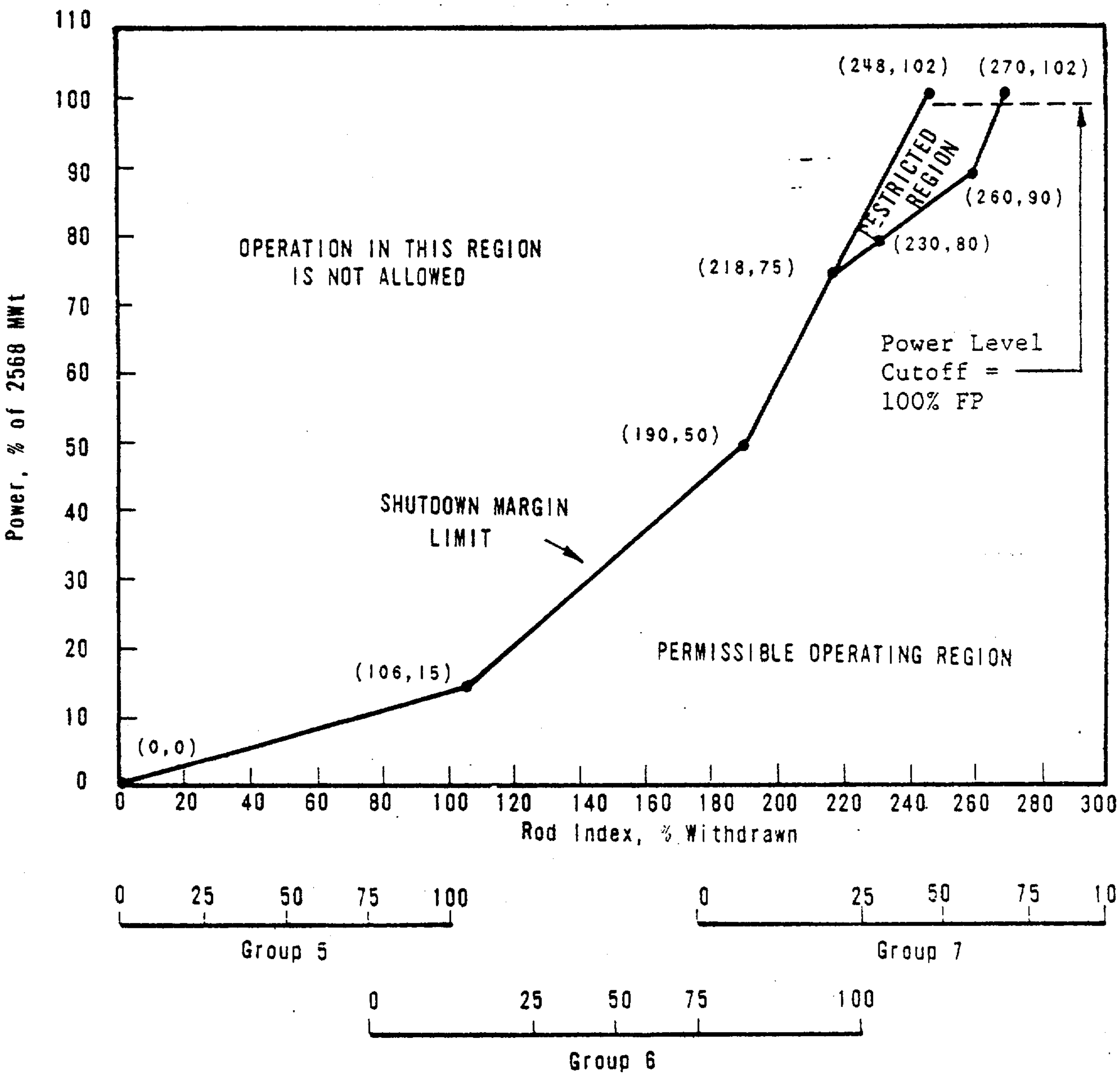
Figure 3.5.2-1B3
Deleted During Oconee Unit 2, Cycle 4 Operation



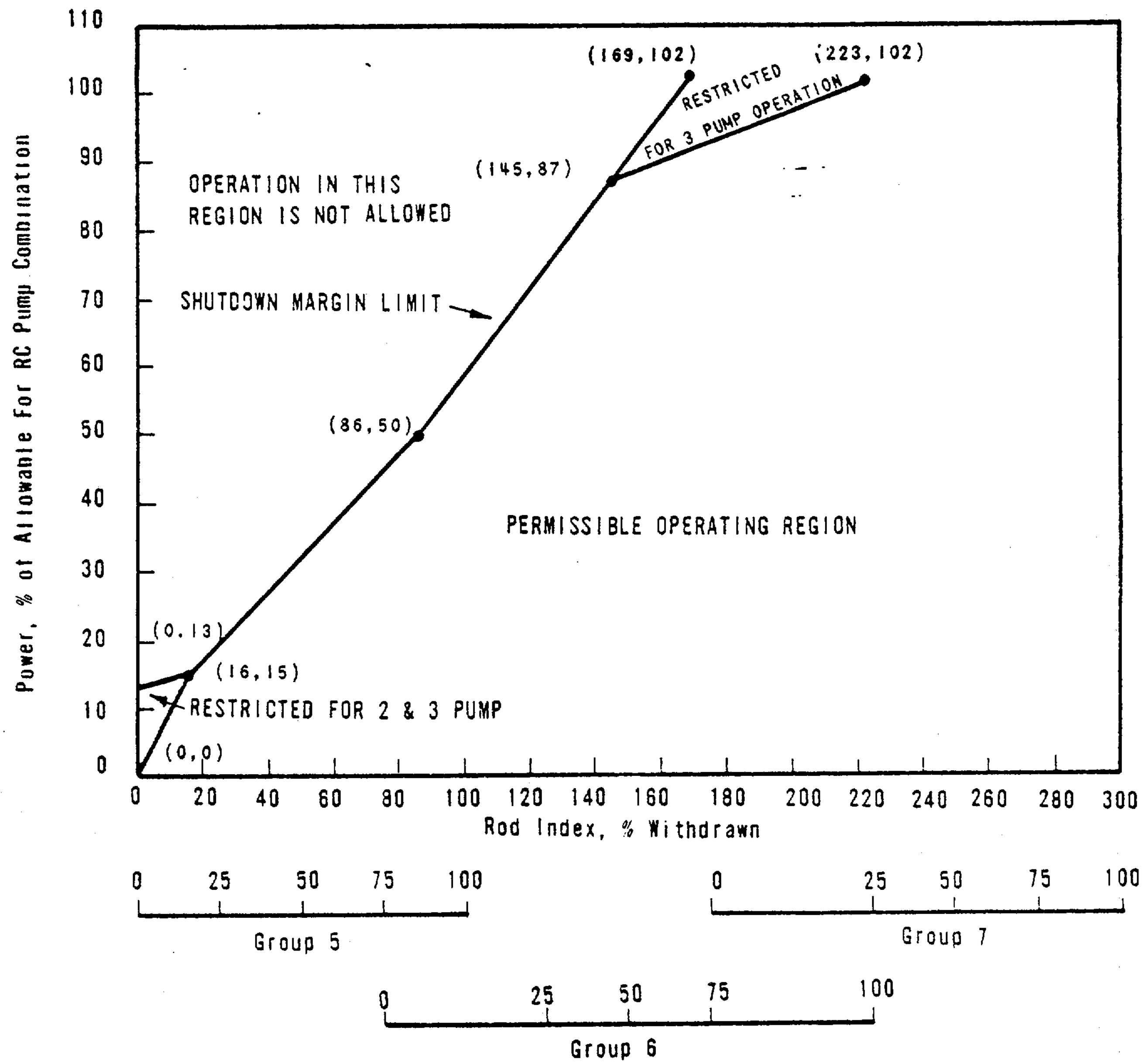
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 100 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-1C1



ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 100 ± 10 TO 200 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-1C2

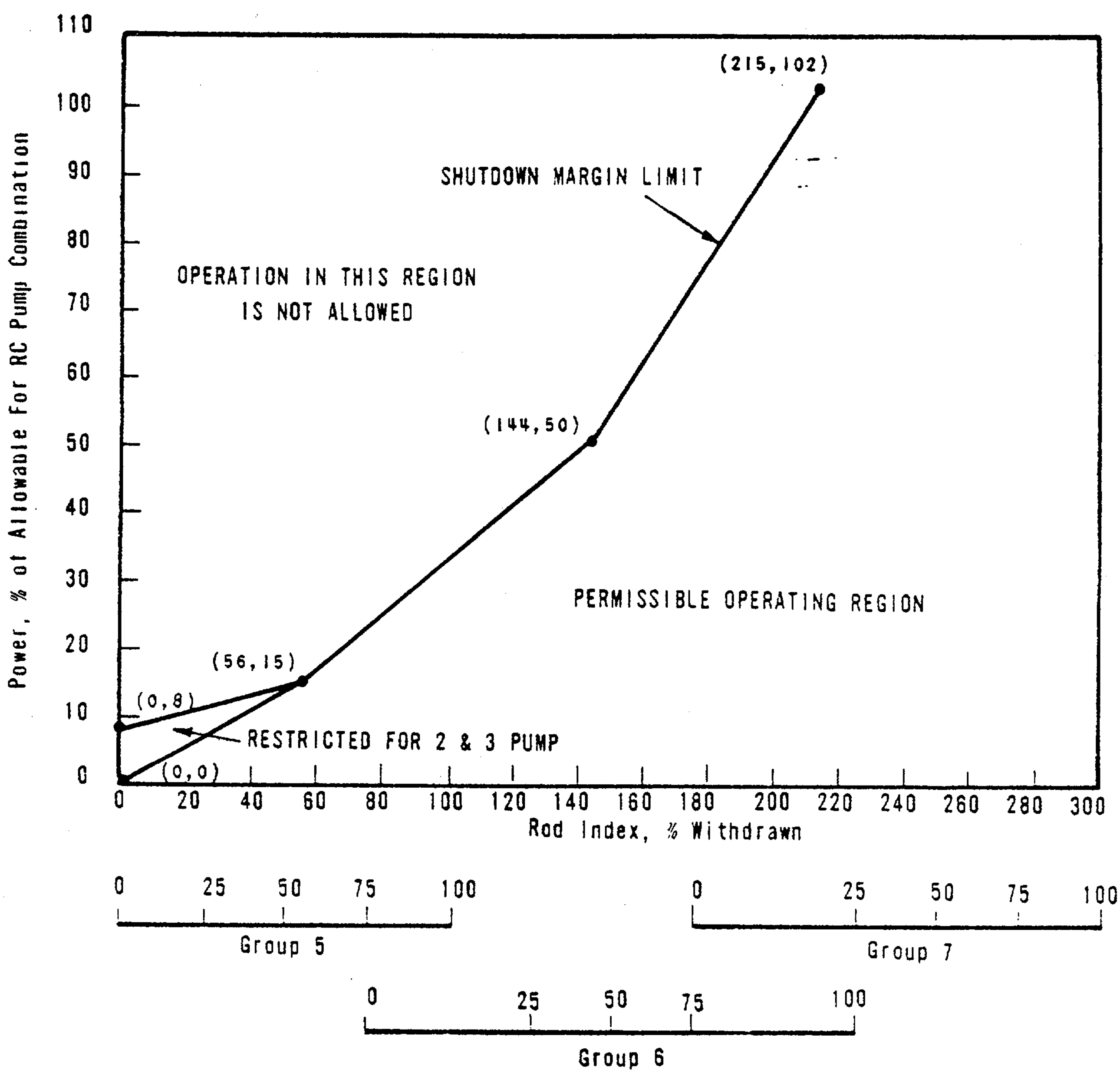


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-1C3



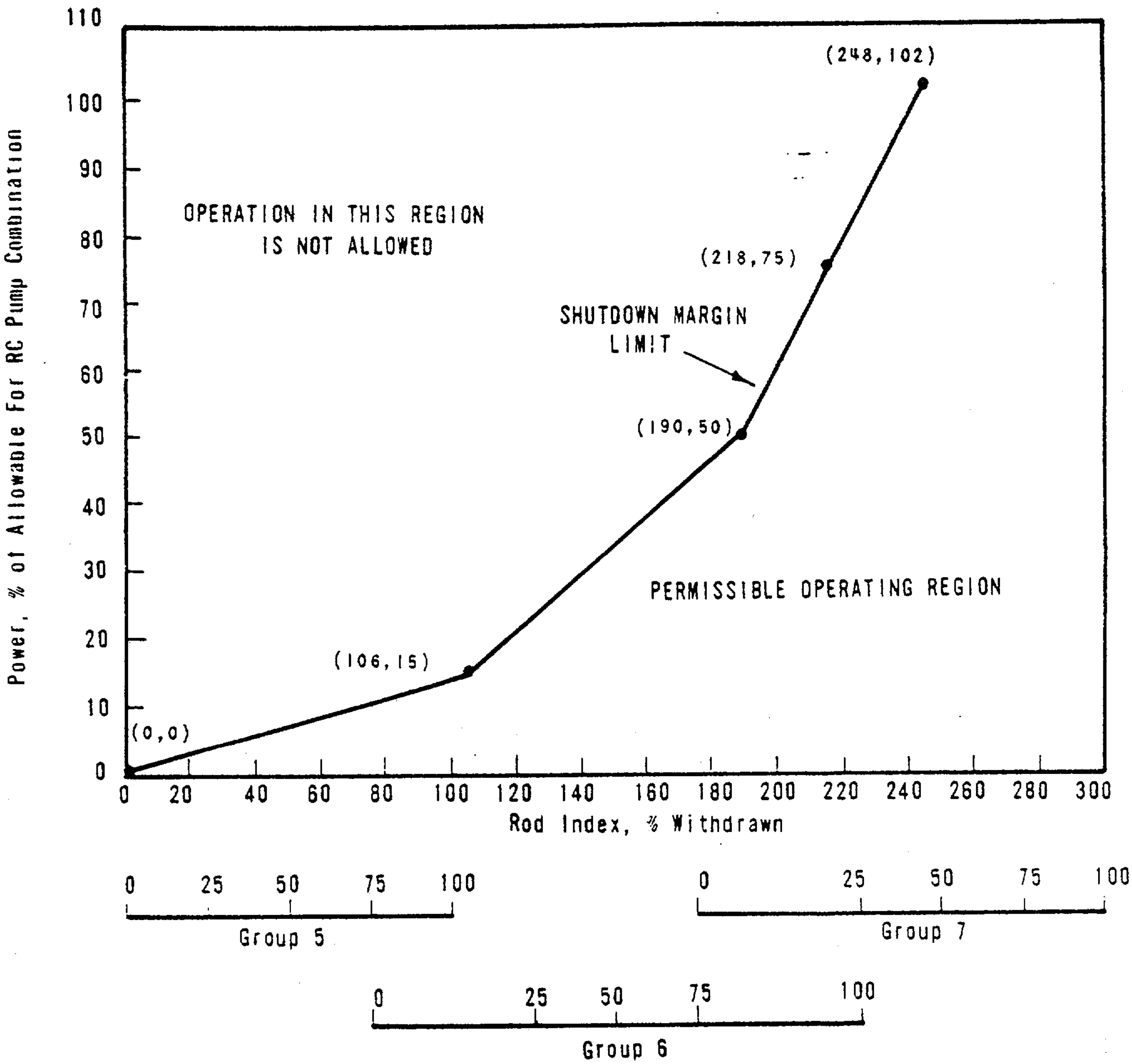
ROD POSITION LIMITS
FOR TWO- & THREE- PUMP OPERATION
FROM 0 TO 100 $\pm$ 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-2C1

Amendments Nos. 73, 73, & 70



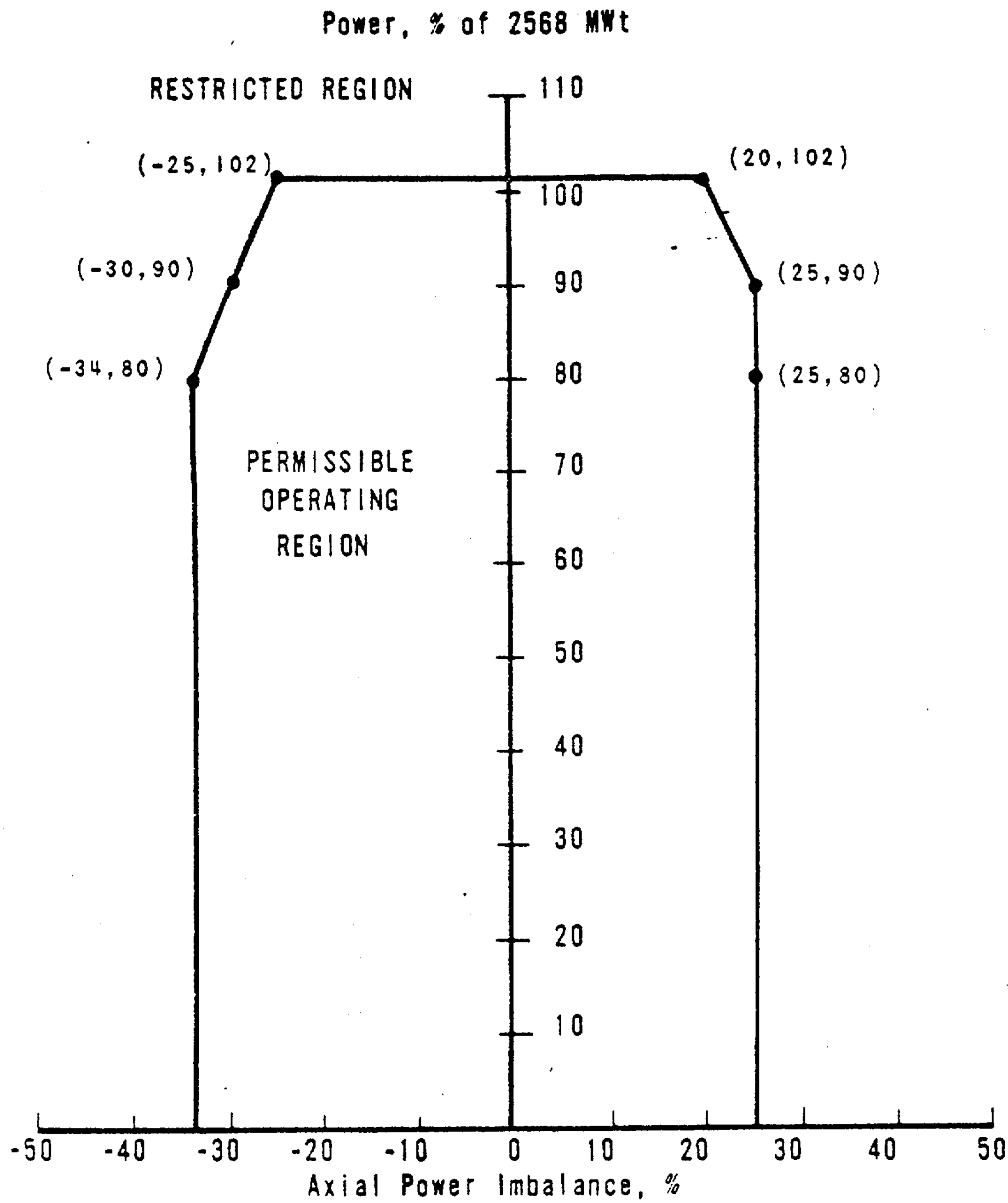
ROD POSITION LIMITS
FOR TWO- & THREE- PUMP OPERATION
FROM 100 ± 10 TO 200 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-2C2

Amendments Nos. 73, 73, & 70

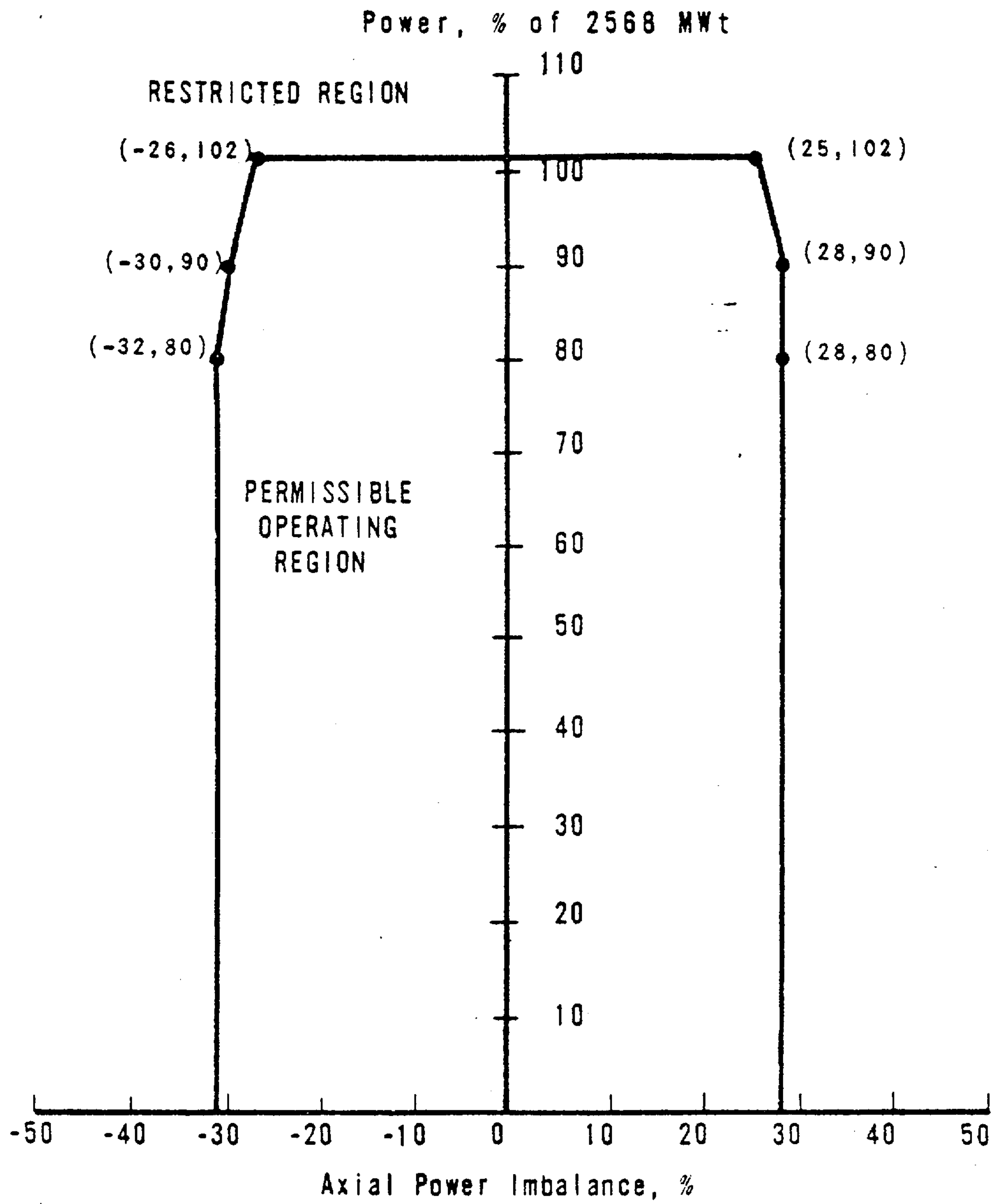


ROD POSITION LIMITS
FOR TWO- & THREE- PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-2C3

Amendments Nos. 73, 73, & 70

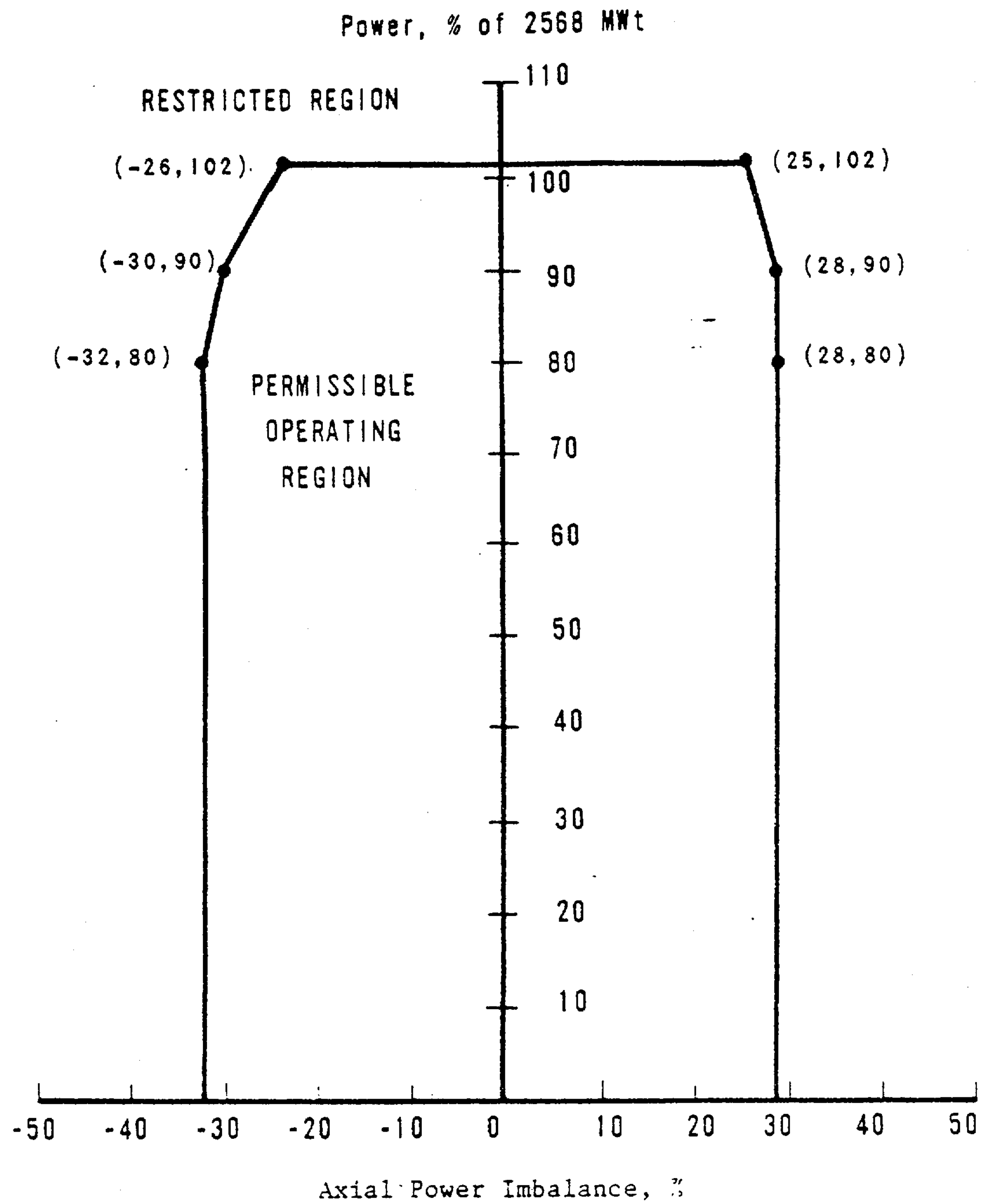


OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 0 TO 100 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3C1



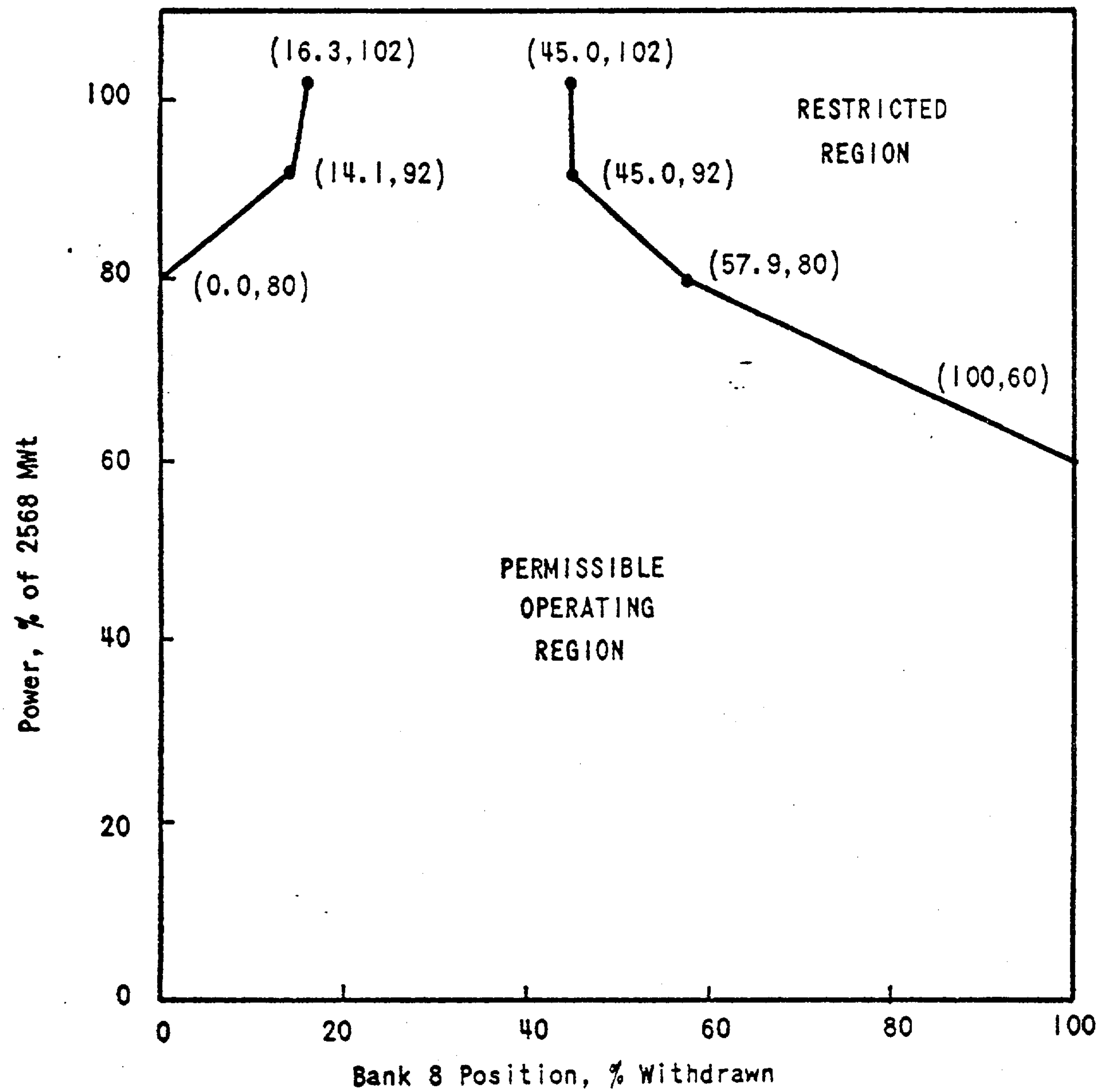
OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 100 $\pm$ 10 TO
200 $\pm$ 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3C2

Amendments Nos. 73, 73, & 70



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 200 $\pm$ 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3C3

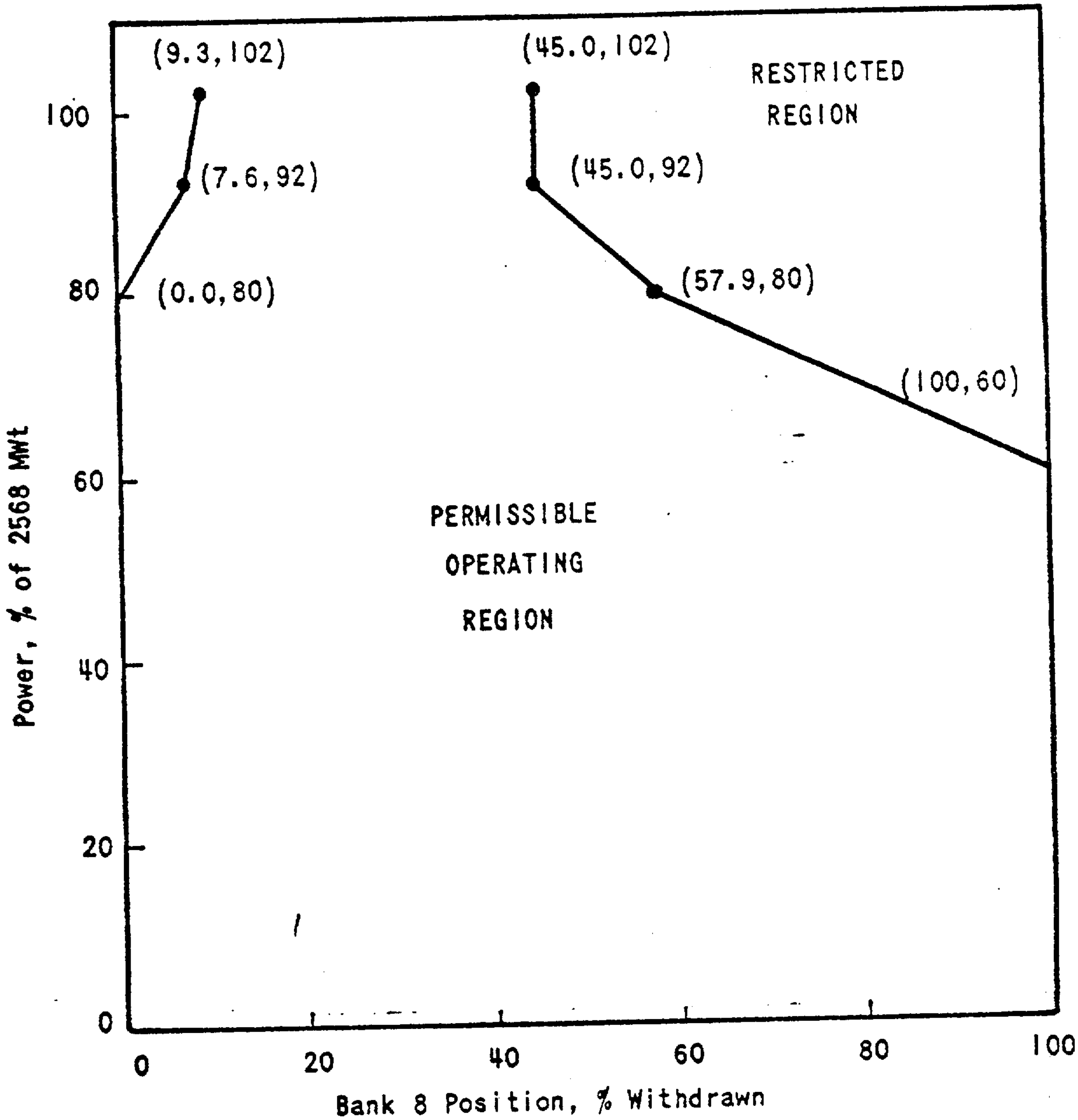
Amendments Nos. 73, 73, & 70



APSR POSITION LIMITS
FOR OPERATION FROM 0 to 100 $\pm$ 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-4A1

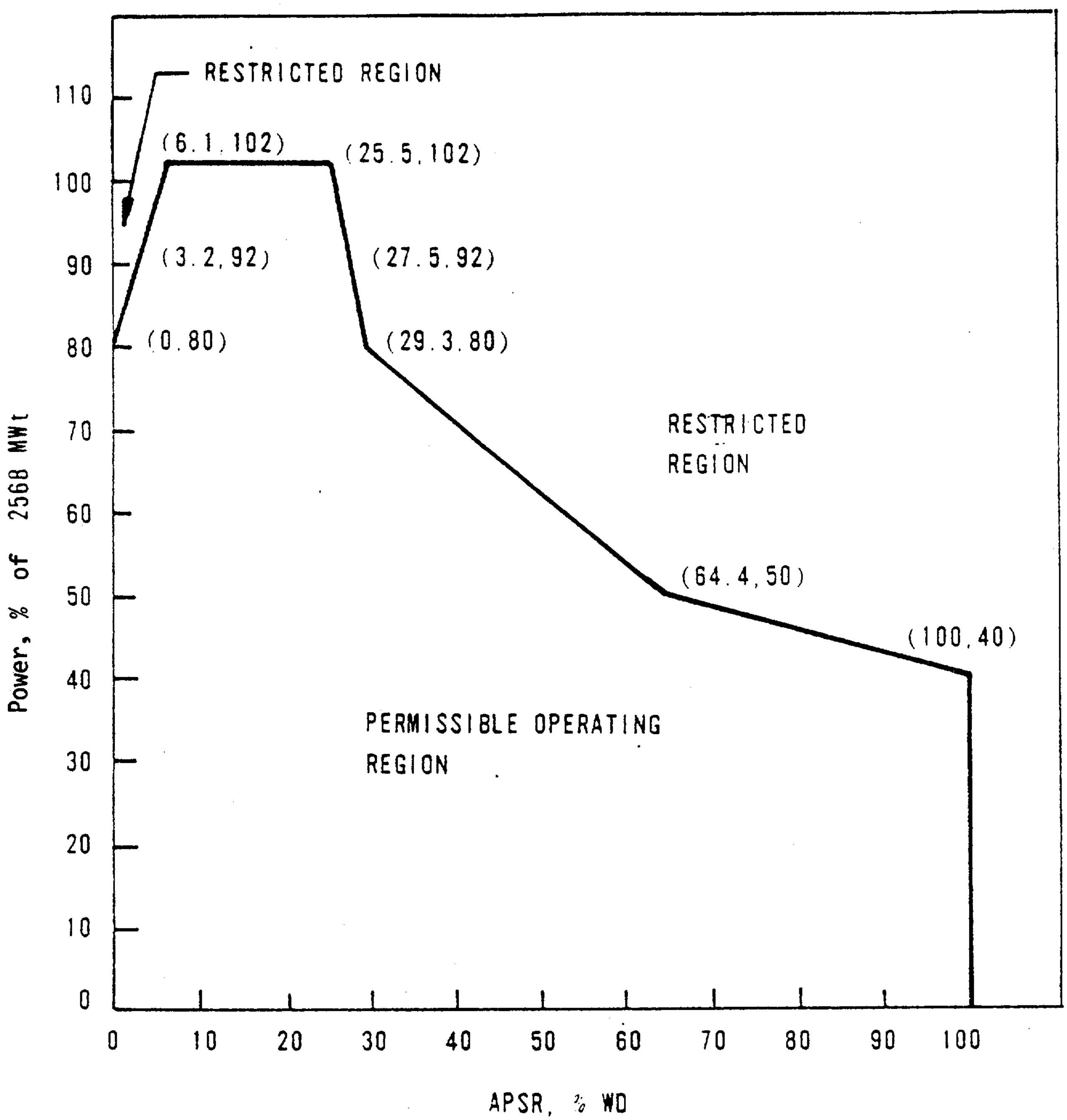


APSR POSITION LIMITS
FOR OPERATION AFTER 100 ± 10 EFPD
Unit 1

OCONEE NUCLEAR STATION

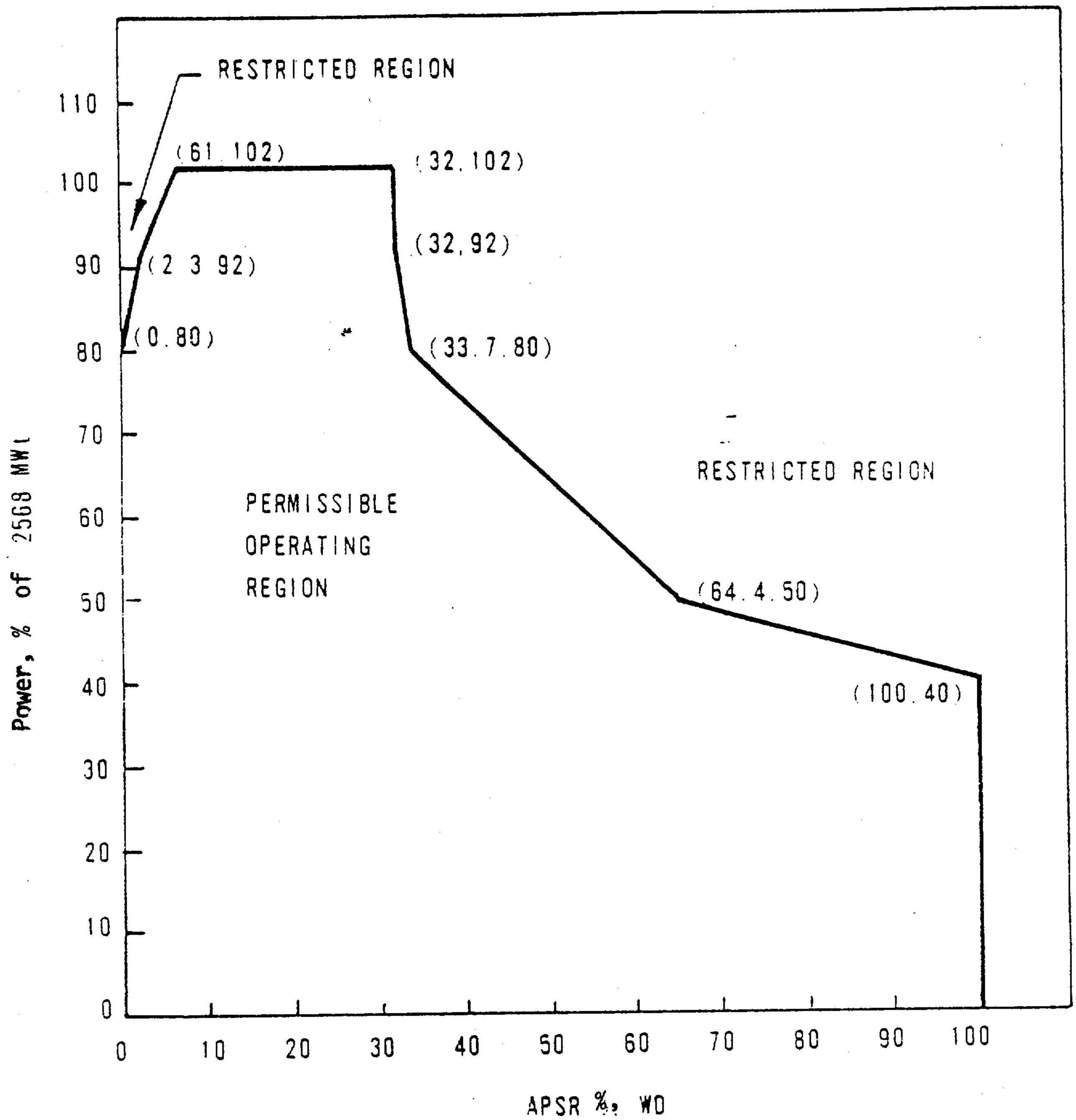
Figure 3.5.2-4A2

Amendments Nos. 73, 73, & 70



APSR POSITION LIMITS
FOR OPERATION
FROM 0 TO 250 ± 10 EFPD
OCONEE 2
OCONEE NUCLEAR STATION
Figure 3.5.2-4B1

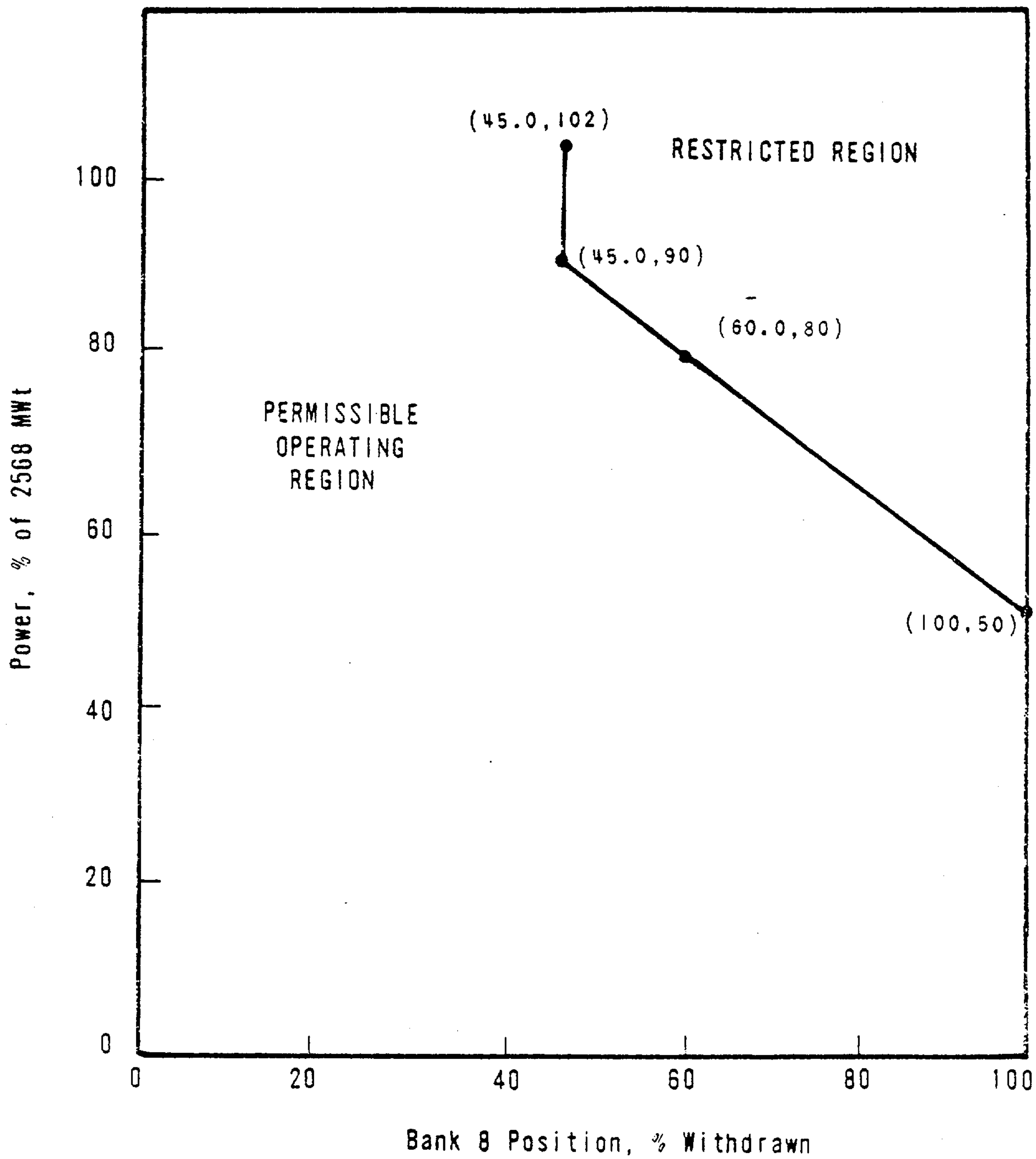
Amendments Nos. 73 , 73, & 70



APSR POSITION LIMITS FOR OPERATION
AFTER 250 ± 10 EFPD
OCONEE 2
OCONEE NUCLEAR STATION
Figure 3.5.2-4B2

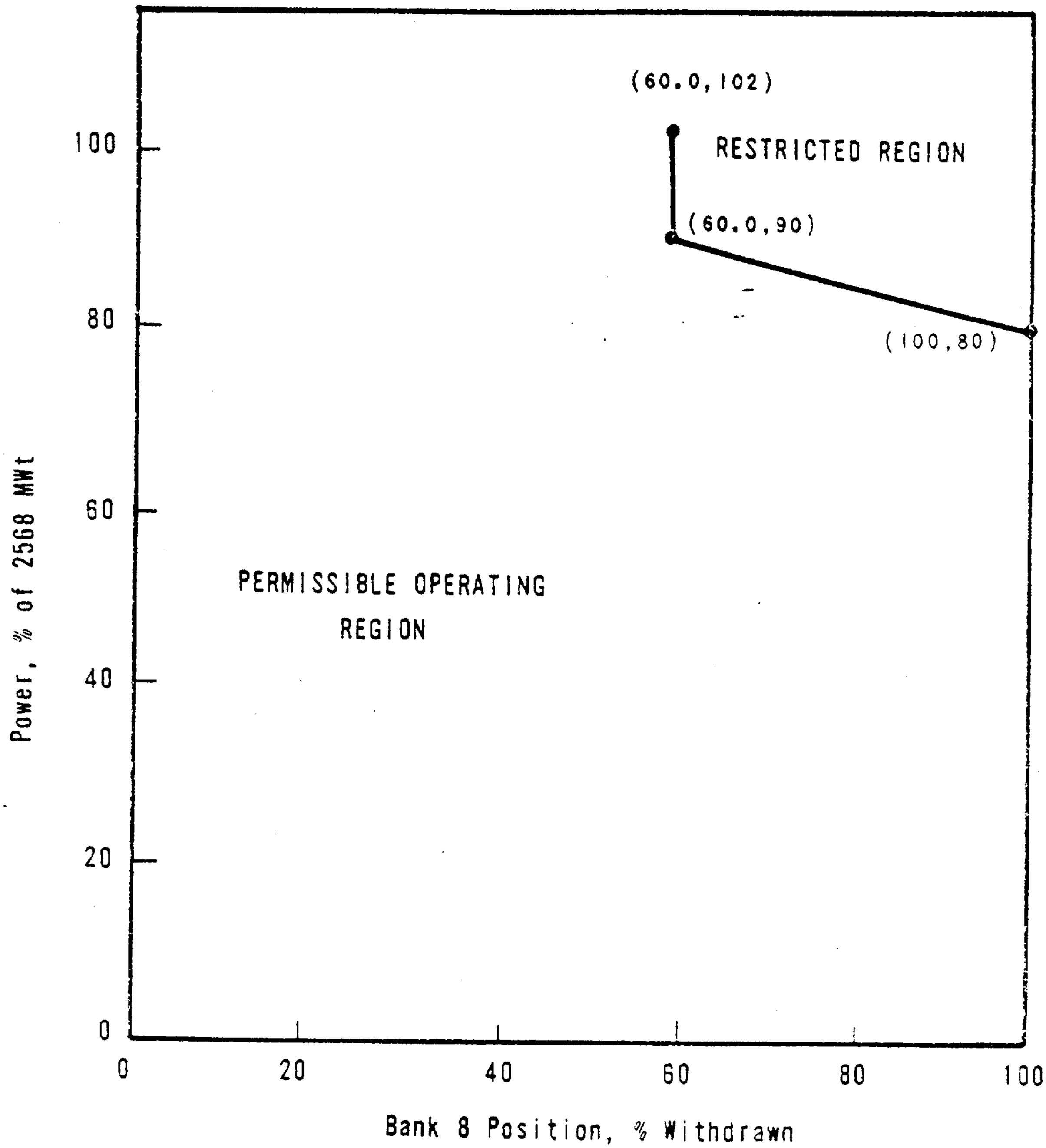
Amendments Nos. 73, 73, & 70

Figure 3.5.2-4B3
Deleted during Oconee Unit 2, Cycle 4 Operation



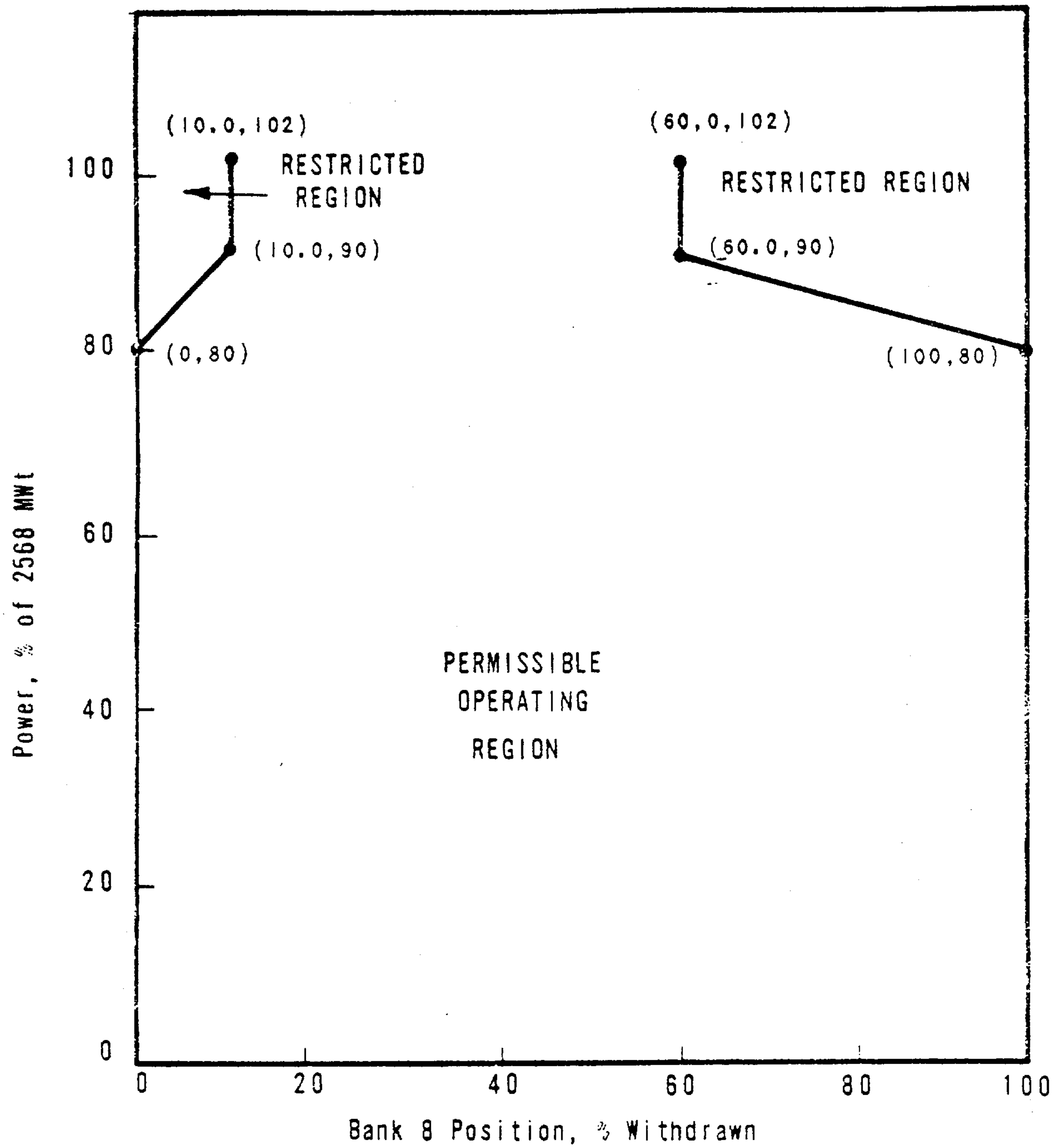
APSR POSITION LIMITS FOR OPERATION
FROM 0 TO 100 $\pm$ 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-4C1

Amendments Nos. 73, 73, & 70

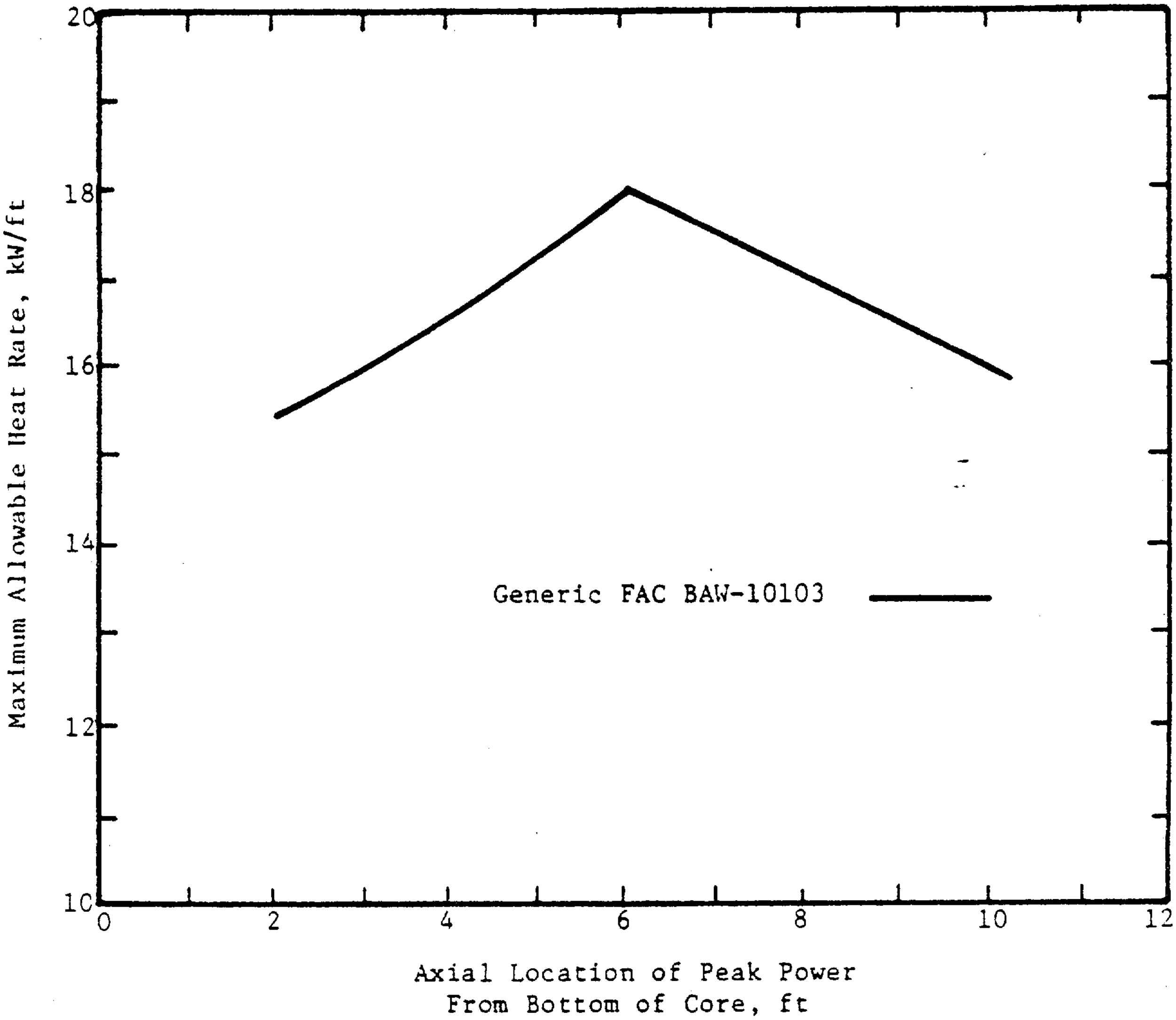


APSR POSITION LIMITS FOR OPERATION
FROM 100 ± 10 TO 200 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-4C2

Amendments Nos. 73, 73, & 70



APSR POSITION LIMITS FOR OPERATION
AFTER 200 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-4C3



LOCA-LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT

OCONEE NUCLEAR STATION

Figure 3.5.2-5

3.5.3 Engineered Safety Features Protective System Actuation Setpoints

Applicability

This specification applies to the engineered safety features protective system actuation setpoints.

Objective

To provide for automatic initiation of the engineered safety features protective system in the event of a breach of RCS integrity.

Specification

The engineered safety features protective actuation setpoints and permissible bypasses shall be as follows:

Functional Unit	Action	Setpoint
High Reactor Building Pressure	Reactor Building Spray	≦30 psig
	High-Pressure Injection	≦4 psig
	Low-Pressure Injection	≦4 psig
	Start Reactor Building Cooling & Reactor Building Isolation	≦4 psig
	Penetration Room Ventilation	≦4 psig
Low Reactor Coolant System Pressure	High Pressure Injection(1)	≧1500 psig
	Low Pressure Injection(2)	≧500 psig

(1) May be bypassed below 1750 psig and is automatically reinstated above 1750 psig.

(2) May be bypassed below 900 psig and is automatically reinstated above 900 psig.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached immediately in the event of a DBA, cover the entire spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1500 psig low reactor coolant pressure setpoint for high pressure injection initiation and 500 psig for low pressure injection is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation.[1]

REFERENCES

(1) FSAR, Section 14.2.2.3.

3.5.4 Incore Instrumentation

Applicability

Applies to the operability of the incore instrumentation system

Objective

To specify the functional and operational requirements of the incore instrumentation system.

Specification

3.5.4.1 At or above 80 percent of the power allowable for the existing reactor coolant pump operating combination, incore detectors shall be operable as necessary to meet the following:

a. For axial imbalance measurements:

At least three detectors in each of at least three strings shall lie in the same axial plane, with one plane in each axial core half. The axial planes in each core half shall be symmetrical about the core mid-plane. The detector strings shall not have radial symmetry.

b. For quadrant power tilt measurements:

At least two sets of at least four detectors shall lie in each axial core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane. Detectors in the same plane shall have quarter core radial symmetry.

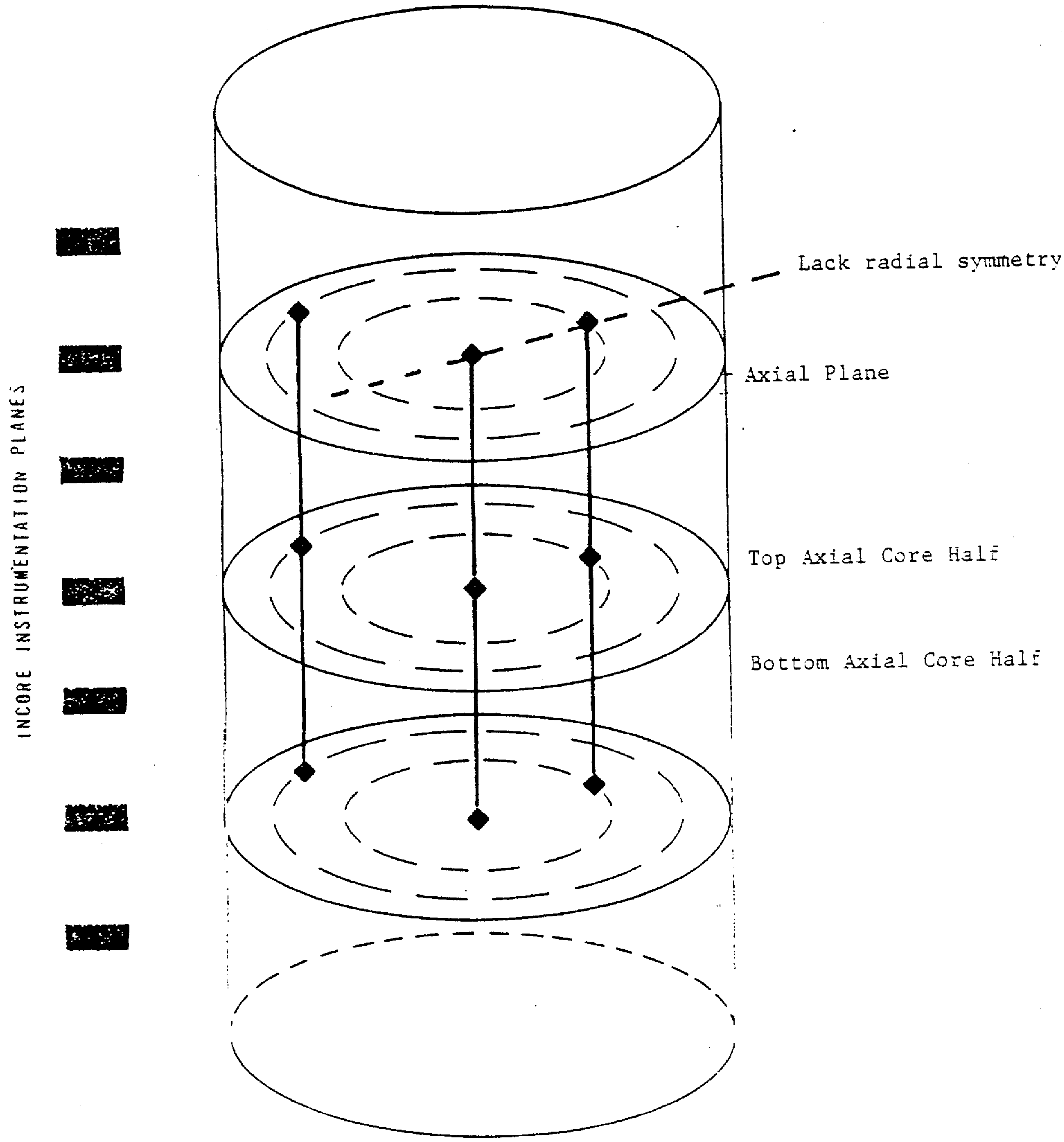
3.5.4.2 If requirements of 3.5.4.1 are not met, power shall be reduced below 80 percent of the power allowable for the existing reactor coolant pump combination within eight hours and incore detector measurements shall not be used to determine axial imbalance or quadrant power tilt.

Bases

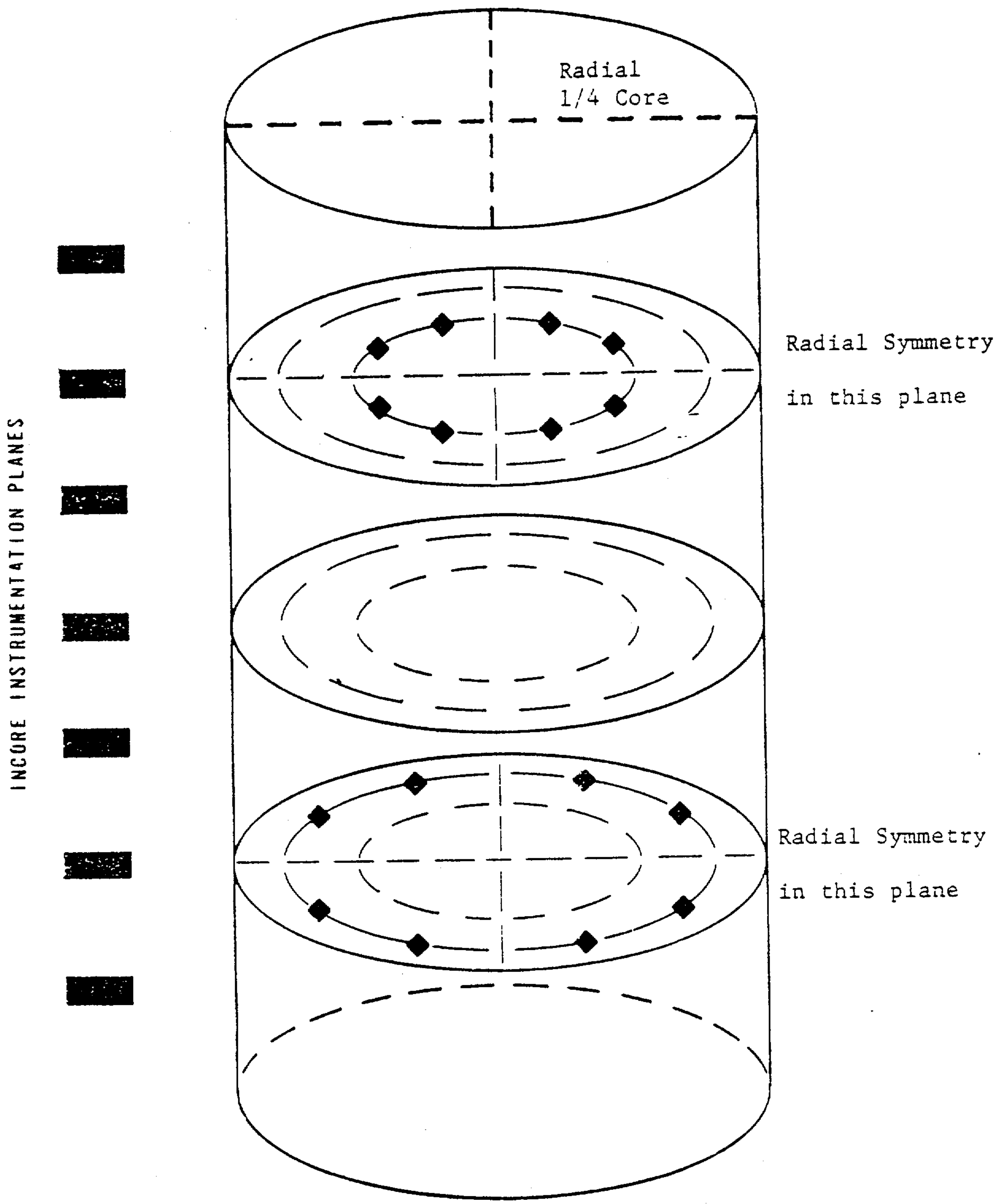
The operability of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Figures 3.5.4-1, 3.5.4-2, and 3.5.4-3 for satisfactory incore detector arrangements.

The safety of reactor operation at or below 80 percent of the power allowable for the reactor coolant pump combination(1) without the axial imbalance trip system has been determined by extensive 3-D calculations, and was verified during the physics startup testing program.

(1) FSAR, Section 4.1.1.3

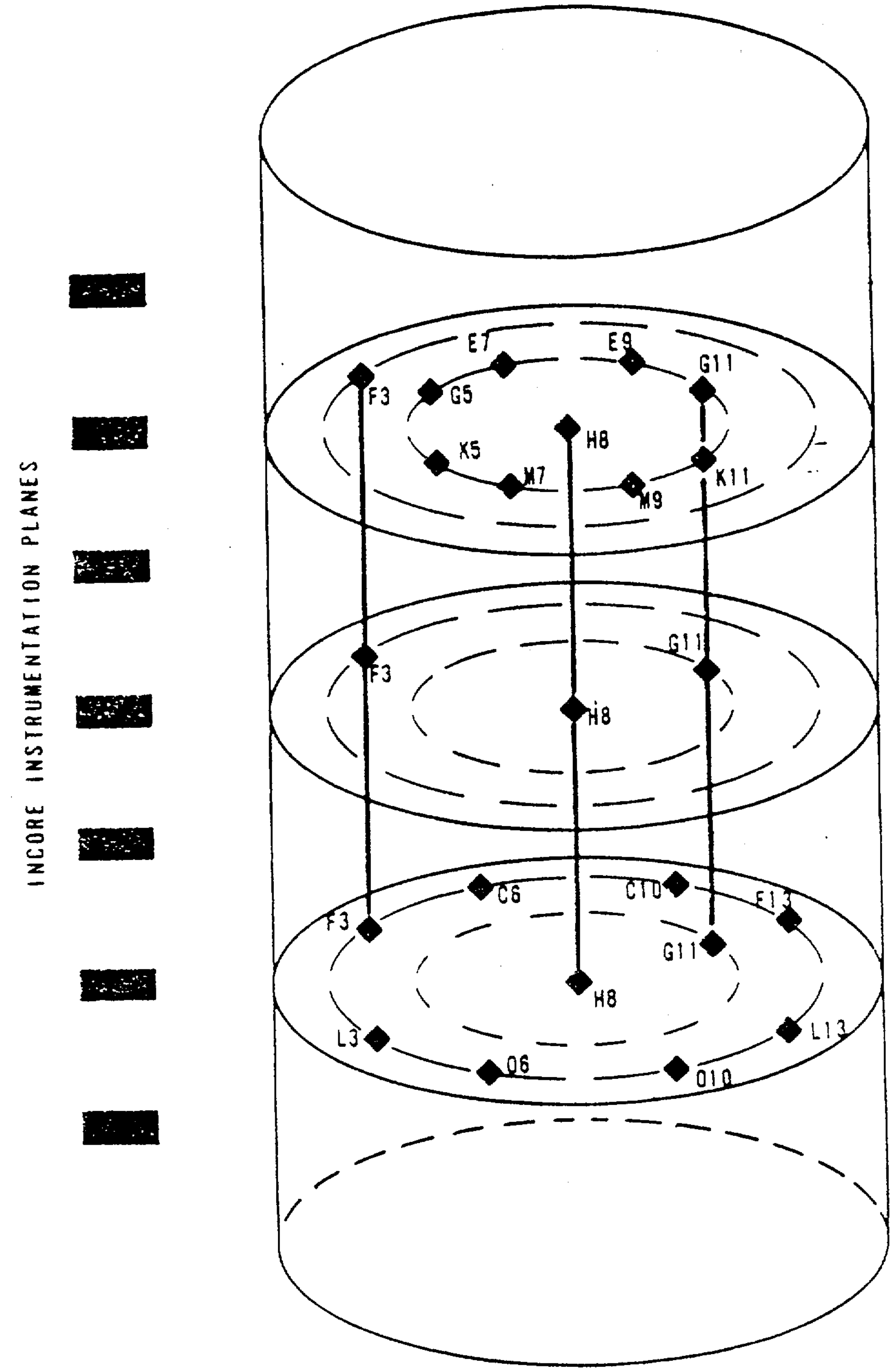


INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION



INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX TILT INDICATION

OCONEE NUCLEAR STATION
Figure 3.5.4-2



INCORE INSTRUMENTATION SPECIFICATION

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. DPR-38, AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. DPR-47, AND AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

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

1.0 Introduction

By letter dated March 30, 1979 (Reference 1) and supplement dated May 17, 1979(2), Duke Power Company (DPC) has proposed changes to the Oconee Nuclear Station Technical Specifications. The proposed changes include modifications for Oconee Nuclear Station Unit 3 (ONS-3) operation after reload for Cycle 5 and page number modifications for administrative purposes.

Most of the proposed Technical Specification modifications and review effort have been associated with the refueling of ONS-3 for Cycle 5 operation. The information submitted by DPC in connection with this refueling is presented in Reference 1 which describes the fuel system design, nuclear design, thermal-hydraulic design, accident analyses, and startup test program. The referenced supplement provides confirmation that the presented analysis and specification of Reference 1 are applicable for actual previous cycle exposure.

The refueling of ONS-3 for Cycle 5 will result in a core loading of 56 fresh Mark B4 assemblies, 108 previously burned Mark B4 assemblies, and nine previously burned Mark B2 fuel assemblies. For ONS-3, this evaluation has taken into consideration the proposed refueling of the core as described in Reference 3 and subsequent operation for the targeted 292 effective full power days (EFPDs) during Cycle 5.

The changes in the core loading and mode of operation are the only physical modifications associated with the refueling. The evaluation of DPC's proposed modifications to the Technical Specifications of ONS-1, 2 and 3 is presented in the following sections.

2.0 Evaluation of Modifications to ONS-3 Core Design

2.1 Fuel System Design

We have evaluated the proposed fuel loading and operation. Tables 4-1 and 4-2 of Reference 3 summarize the design characteristics of the reload fuel types. The fresh Mark B4 assemblies are identical to the previously burned Mark B4 fuel with regard to assembly mechanical design, fuel rod design and thermal design. The fuel designs of Mark B4 fuel types have been evaluated for ONS-3 in association with earlier refuelings and found acceptable (References 4, 5 and 6).

2.1.1 Cladding Creep Collapse

Fuel rod cladding creep collapse analyses have been performed for the most limiting (i.e., most highly exposed) fuel assemblies to be included for Cycle 5. The analyses were performed according to the conservative methods and assumptions described in Reference 7 which has been accepted by the NRC staff. (This reference is a proprietary version, but nonproprietary versions are available also.) These analyses show that the time to rod cladding collapse will be in excess of 30,000 effective full power hours (EFPHs). Because no assembly will reach a total exposure as high as 30,000 EFPH during Cycle 5 (Table 4-1 of Reference 3), we conclude that cladding creep collapse will not occur during the cycle.

2.1.2 Cladding Stress and Strain

For this cycle, the cladding stress due to differential pressure, temperature gradient or axial loads and restraints will not exceed the yield stress or ultimate strength of the material. The anticipated cladding strain was shown to be less than the 1% plastic cladding strain limit for up to 55,000 MWd/MTU, well below the exposure to be accumulated by the end of cycle. We previously accepted these criteria for cladding stress (Reference 8) and strain (Reference 6) and we conclude that they are also valid for this cycle.

2.1.3 Fuel Thermal Design

The thermal linear heat rate (LHR) limits have been established with the TAFY Code (Reference 9) and assumed fuel densification to 96.5% of theoretical density. These limits are stated in Table 4-2 of Reference 2. The thermal LHR limits which ensure that fuel center melting does not occur are less restrictive than the Loss of Coolant Accident (LOCA) LHR limits. Because the LOCA LHR limits will be met by operating within the limiting conditions for operations, the thermal LHR limits will also be met. We conclude that the indicated thermal LHR limits are acceptable for preventing center melting and that the limits will not be exceeded.

2.2 Nuclear Design

Reference 3 indicates the proposed core loading arrangement, the initial enrichments and burnup distributions. Most of the fresh Mark B4 assemblies will be loaded into locations on the edge of the core and will be below fuel thermal limits.

Reactivity control will be supplied by soluable boron in the reactor coolant which will be supplemented by 61 full length control rods. Also, APSRs will provide axial power distribution control profile.

Nuclear parameters, e.g., critical boron concentrations, control rod worths, Doppler coefficients, moderator coefficients, xenon worth and effective delayed neutron fractions have been calculated using the same techniques as accepted for the previous cycle in Reference 6. These parameters are presented and compared in Reference 3 to the previous cycle values.

Shutdown margins have been calculated for beginning of cycle (BOC) and end of cycle (EOC). The calculated minimum shutdown margin is larger than the required value.

We conclude that the nuclear design does not differ in a significant way from earlier cycles, that the nuclear parameters have been calculated by acceptable methods and are within the range of values expected for a cycle approaching an equilibrium cycle, and that the nuclear design has resulted in an adequate shutdown margin. The nuclear design for ONS-3 Cycle 5 is, therefore, acceptable.

2.3 Thermal Hydraulic Design

The new fuel is thermal-hydraulically identical to that currently in use. The thermal-hydraulic design evaluation used methods and models previously accepted in References 5 and 6. The results of this evaluation are included in Table 6-1 of Reference 3.

The flux/flow trip setpoint has been revised for this cycle to 1.08 from 1.05 of the previous cycle. This modification is based on the transient analysis of the loss of two reactor coolant pumps. The methods used are identical to those for previous cycle evaluations as found acceptable in References 4 through 6. The only change to this thermal hydraulic analysis is the use of reactor coolant pump flow rate coastdown values which are specific to this plant. Previously generic values were used. This revision and revisions to the $F_{\Delta H}$ Technical Specification (Reference 5) provide sufficient margin to adjust the flux flow trip setpoint and still maintain reactor safety limits, e.g., Departure from Nucleate Boiling Ratio (DNBR) equal to or less than a ratio of 1.3.

The licensee accounted for the reduction in DNBR due to fuel rod bowing based on Reference 10, which is a Babcock & Wilcox (B&W) interim method. It is the present staff position that a future modification to the methods of Reference 10 are required. The licensee has included a 10.2% DNBR margin in the setpoint calculations for Oconee Unit No. 3. We consider this sufficient to account for the increased reduction in DNBR due to future modifications to the methods described in Reference 10. Therefore, we consider the licensee's fuel rod bowing calculations to be acceptable.

3.0 Evaluation of Accidents and Transients

Each Final Safety Analysis Report (FSAR) accident analysis (Reference 11) has been examined with respect to cycle-dependent parameter changes to determine the effect of the reload and to ensure that reactor performance during hypothetical transients is not degraded.

Fuel thermal analysis parameters are given in Reference 3. That reference compares the Cycle 4 and 5 thermal-hydraulic maximum design conditions. Reference 2 compares the key kinetics parameters from the FSAR and Cycle 5.

From the examination of Cycle 5 core thermal properties and kinetics properties with respect to acceptable previous cycle values, it is concluded that this core reload will not adversely affect the safe operation of ONS-3 during Cycle 5. The only parameter which is potentially less conservative for Cycle 5 than for the FSAR value is the Doppler coefficient. We have estimated the effect of this nonconservatism for the FSAR transients and have concluded that there is sufficient conservatism in the FSAR analyses to compensate for this potential nonconservatism.

4.0 Emergency Core Cooling System (ECCS)

On July 9, 1975, DPC submitted an acceptable ECCS evaluation (Reference 12) for ONS-3. On April 12, 1978, we were informed of a potentially more severe limiting small break location than previously analyzed. By a Commission Order for Modification of License dated April 26, 1978, certain modifications to the ECCS and a procedure for prompt operator action were required for ONS-3 to permit future operation. An Exemption was granted on July 6, 1978 to 10 CFR 50.46(a), "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," which superseded the Order. The Exemption provided for its own termination upon completion of the modifications required by the Exemption and for prior NRC staff approval of the design. By letter dated December 13, 1978, we found the design of the modifications to be acceptable. DPC has installed the modifications and prepared acceptable operating procedures; thus, we conclude that the as-modified ECCS required by the Exemption of July 6, 1978 is acceptable.

Due to the accident at Three Mile Island, Unit No. 2 (TMI-2) on March 28, 1979, where a pilot-operated relief valve on the primary system failed to close after opening, thus inducing the equivalent of a very small break LOCA, in conjunction with a complete loss of feedwater, the NRC is reexamining the ECCS of all pressurized water reactors in terms of very small breaks.

Our Order of May 7, 1979, issued in the aftermath of TMI-2, required DPC to: "Complete analyses for potential small breaks and implement operating instructions to define operator action." B&W, by a report dated May 1979 and a supplementary letter dated May 12, 1979, responded on the Oconee dockets to the requirements of our Order. Our May 18, 1979 Safety Evaluation of these submittals stated, "A principal finding of our generic review is a reconfirmation that LOCA analyses of breaks at the lower end of the small break spectrum (smaller than 0.04 sq. ft.) demonstrate that a combination of heat removal by the steam generators, high pressure injection system and operator action ensure adequate core cooling." We concluded that DPC complied with the analysis portion of the quoted paragraph of the Order. We further stated that to support longer term operation of the Oconee Station, requirements will be developed for additional and more detailed analyses of loss of feedwater and small break LOCA events. We concluded in our letter of May 18, 1979, that DPC could restart ONS-3 in that it met the requirements of our May 7, 1979 Order. Based on the above, we conclude that the emergency core cooling system for Oconee Unit No. 3 is acceptable.

5.0 Startup Tests

Startup tests have been proposed by DPC to provide assurance that ONS-3 has been loaded as intended. This test program is very similar to that used for ONS-3 Cycle 4. We have reviewed the test program and consider it acceptable.

6.0 Evaluation of Technical Specification Changes

The Technical Specifications have been revised for Cycle 5 operation in accordance with the methods of Technical Specification bases to account for minor changes in power peaking and control rod worths inherent with non-equilibrium cycles. In addition, the power level cutoff restriction applied in previous cycles to the control rod position limits has been revised, not only for ONS-3 but also for ONS-1 and 2 operation. The change has been accomplished by designating the power level cut-off at 100% full power. Any operating restrictions resulting from transient xenon-induced power peaks, including xenon-free startup, are inherently included in the control rod position and axial imbalance limits. The remaining changes have been discussed in the previous text. We have reviewed these changes and found them acceptable.

7.0 Environmental Consideration

We have determined that these 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 these 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.

8.0 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: June 22, 1979

References

1. Letter from William O. Parker, Jr., DPC, to Harold R. Denton, NRC, March 30, 1979.

2. Letter from William O. Parker, Jr., DPC, to Harold R. Denton, NRC, May 17, 1979.

3. Oconee Unit 3 Cycle 5 Reload Report, BAW 1522, March 1979.

4. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendments Nos. 34, 34 and 31 to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55, Duke Power Company, Oconee Nuclear Station 3, October 22, 1976.

5. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendments Nos. 52, 52 and 49 to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55, Duke Power Company, Oconee Nuclear Station 3, November 21, 1977.

6. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendments Nos. 63, 63 and 60 to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55, Duke Power Company, Oconee Nuclear Station 3, July 6, 1978.

7. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084P-A, Rev. 2, Babcock & Wilcox, Lynchburg, Va., January 1979.

8. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendments Nos. 45, 45 and 42 to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55, Duke Power Company, Oconee Nuclear Station 3, July 29, 1977.

9. C. D. Morgan and H. S. Kao, TAFY - Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, Lynchburg, Va., May 1972.

10. Letter from J. H. Taylor, B&W, to D. B. Vassallo, NRC, "Determination of the Fuel Rod Bow DNB Penalty," December 13, 1978.

11. Oconee Nuclear Station, Units 1, 2, and 3 Final Safety Analysis Report, Docket Nos. 50-269, 50-270, and 50-287.

12. R. C. Jones, J. R. Biller, and B. M. Dunn, ECCS Analysis of B&W's 177-FA Lowered Loop NSS, BAW-10103A, Rev. 3, Babcock & Wilcox, Lynchburg, Va.

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. 73, 73 and 70 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised Technical Specifications 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.

The amendments revise the Station's common Technical Specifications to support the operation of Oconee Unit No. 3 at full rated power during Cycle 5 after core reload. These amendments also revise the Technical Specifications for Units Nos. 1, 2 and 3 in regard to power level cut-off. Also included in this action is the termination of the July 6, 1978 Exemption (43 F.R. 31074, July 19, 1978) for Oconee Unit No. 3 from the requirements of 10 CFR 50.46(a)(1).

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 these amendments.

For further details with respect to this action, see (1) the application for amendments dated March 30, 1979, as supplemented May 17, 1979, (2) Amendments Nos. 73, 73 and 70 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 Operating Reactors.

Dated at Bethesda, Maryland, this 22nd day of June 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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