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

February 10, 1981

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

RECEIVED FEB 17 1981 U.S. NUCLEAR REGULATORY COMMISSION

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. 93, 93, and 90 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated August 25, 1980, as supplemented December 22, 1980 and January 22, 1981.

These amendments revise the TSs to support the operation of Oconee Unit No. 3 at full rated power during Cycle 6. The amendments also add a new TS 3.1.11, Shutdown Margin, and a new Section 3.5.2.9 to TS 3.5.2, Control Rod Group and Power Distribution Limits, for Oconee Units Nos. 1, 2 and 3.

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

Sincerely,

Robert W. Reid

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

Enclosures:
1. Amendment No. 93 to DPR-38
2. Amendment No. 93 to DPR-47
3. Amendment No. 90 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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Charlotte, North Carolina 28242

Oconee County Library
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Honorable James M. Phinney
County Supervisor of Oconee County
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8/25/80, 12/22/80 & 1/22/81

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. 93
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 August 25, 1980, as supplemented December 22, 1980, and January 22, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 93 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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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 Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 10, 1981

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. 93
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 August 25, 1980, as supplemented December 22, 1980, and January 22, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 93 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 Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 10, 1981

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. 90
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 August 25, 1980, as supplemented December 22, 1980, and January 22, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 90 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 Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 10, 1981

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 93 TO DPR-38

AMENDMENT NO. 93 TO DPR-47

AMENDMENT NO. 90 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
ii	ii
2.1-3d	2.1-3d
2.1-9	2.1-9
2.1-12	2.1-12
2.3-10	2.3-10
3.1-23	3.1-23
3.2-2	3.2-2
3.5-10	3.5-10
3.5-13	3.5-13
3.5-17	3.5-17
3.5-17a	3.5-17a
3.5-17b	3.5-17b
3.5-20	3.5-20
3.5-20a	3.5-20a
3.5-20b	3.5-20b
3.5-23	3.5-23
3.5-23a	3.5-23a
3.5-23b	3.5-23b
3.5-26	3.5-26
3.5-26a	3.5-26a
3.5-26b	3.5-26b

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 magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup independent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis (4).

All plant operating limits are presently based on an original method of calculating rod bowing penalties that are more conservative than those that would be obtained with new approved procedures (4). For Cycle 6 operation, this subrogation results in a 10% DNBR margin, which is partially used to offset the reduction in DNBR due to fuel rod bowing.

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. 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.

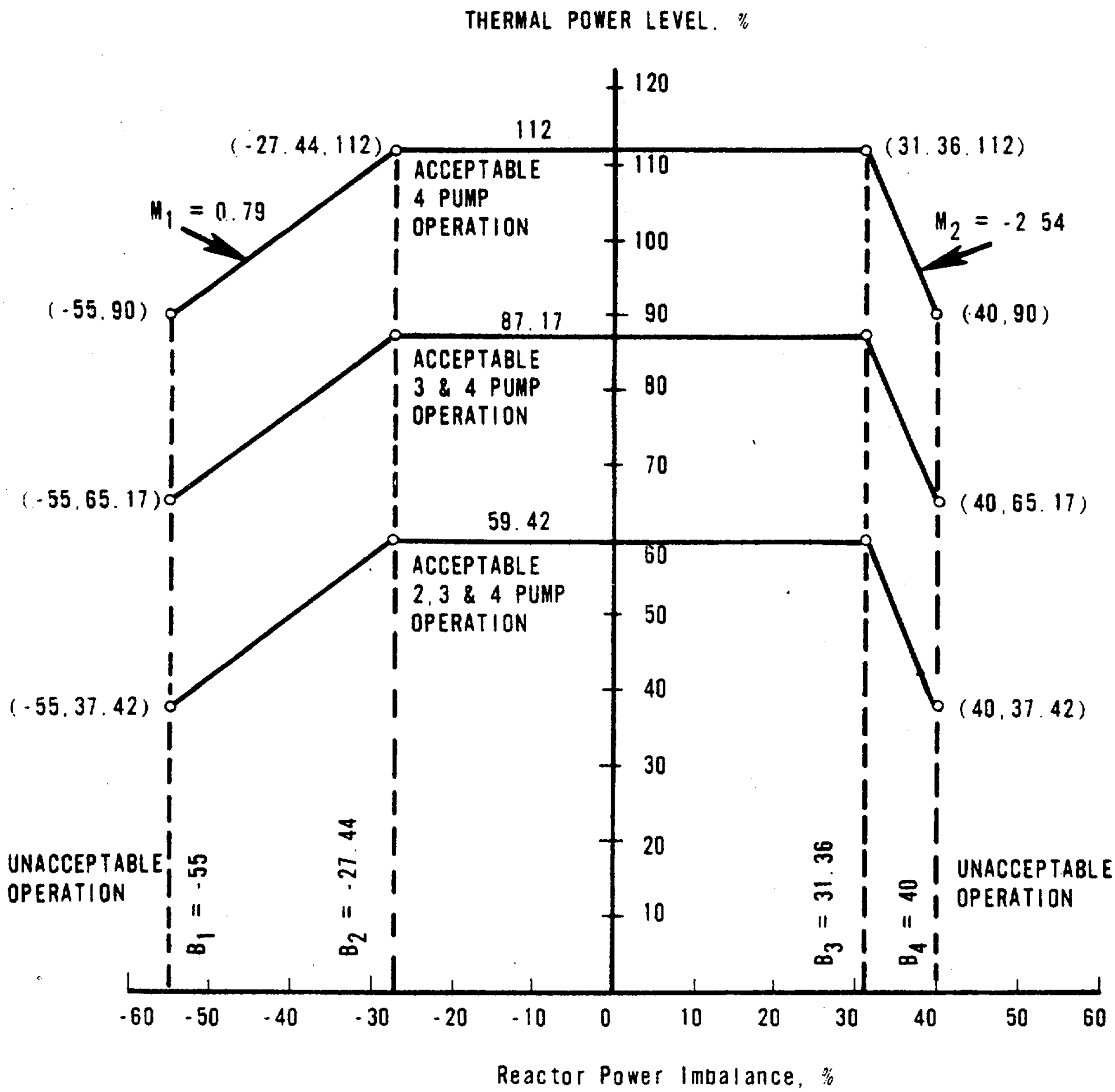
References

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

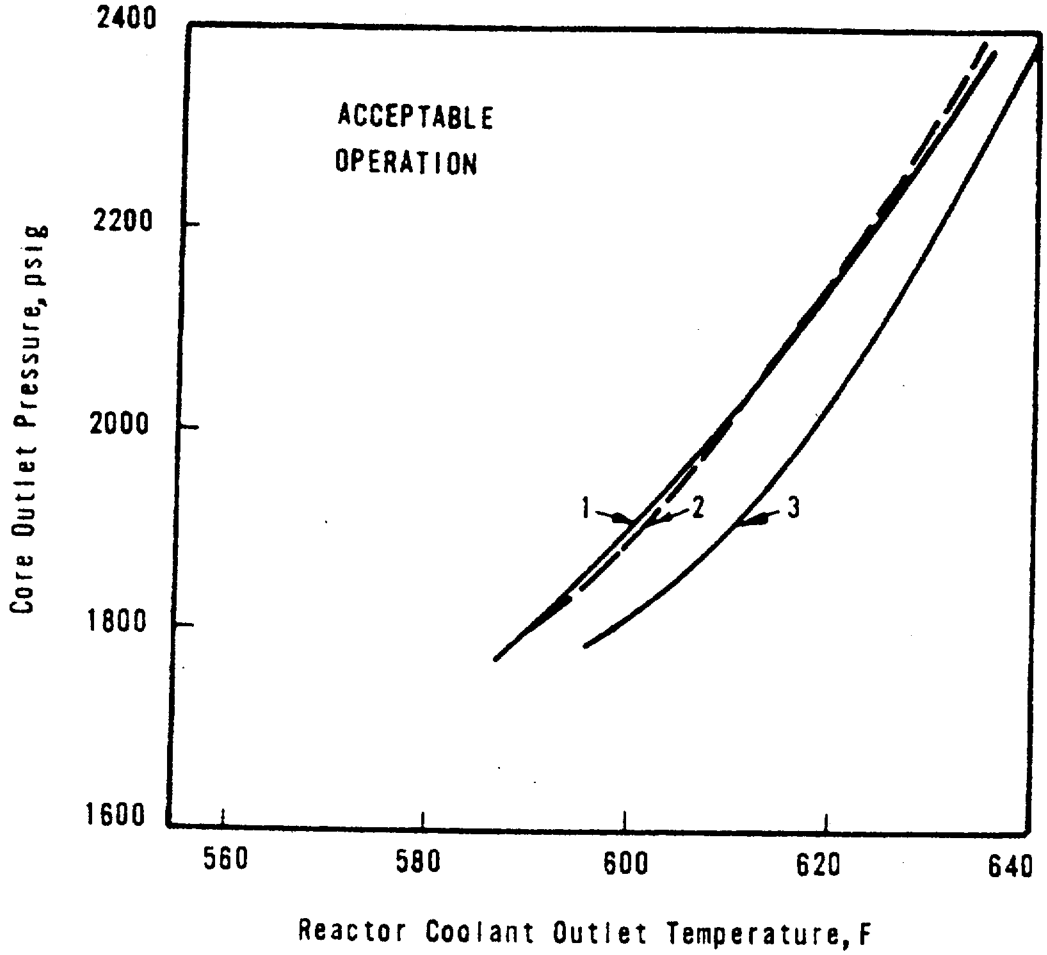
(2) Oconee 3, Cycle 3 - Reload Report - BAW-1453, August, 1977.

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

(4) Oconee 3, Cycle 6 - Reload Report - BAW- 1634, August, 1980



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

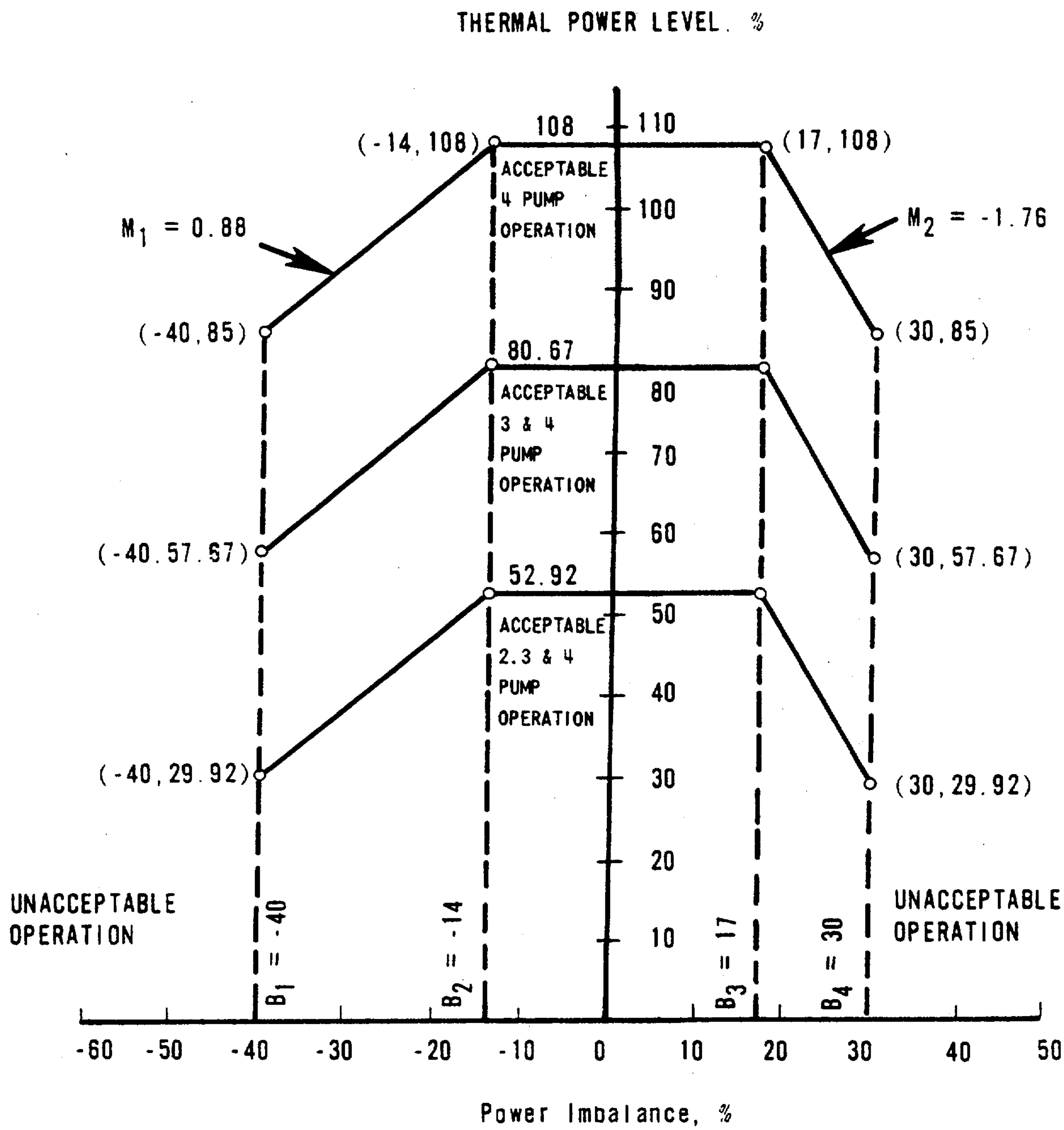


Curve	Coolant Flow, gpm	Power, %	Pumps Operating	Type of Limit
1	374,880 (100%)*	112	4	DNBR
2	280,035 (74.7%)	87.2	3	DNBR
3	183,690 (49.0%)	59.4	2	Quality

*106.5% of first-core design flow.

CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION
Figure 2.1-3C



PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 3 .

OCONEE NUCLEAR STATION

Figure 2.3-2C

3.1.11 Shutdown Margin

Specification

The available shutdown margin during all system conditions except refueling shall be greater than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

Bases

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

During power operation and startup the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits specified in Specification 3.5.2.

During refueling conditions equivalent protection is provided in the requirements of Specification 3.8.4.

Bases

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

The quantity of boric acid in storage in the concentrated boric acid storage tank or the borated water storage tank is sufficient to borate the reactor coolant system to a 1% $\Delta k/k$ subcritical margin at cold conditions (70°F) with the maximum worth stuck rod and no credit for xenon at the worst time in core life. The current cycles for each unit, Oconee 1 Cycle 6, Oconee 2 Cycle 5, and Oconee 3 Cycle 6 were analyzed with the most limiting case selected as the basis for all three units. Since only the present cycles were analyzed, the specifications will be reevaluated with each reload. A minimum of 995 ft.^{3} of 8,700 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1800 ppm boric acid in the borated water storage tank (3) will satisfy the requirements. The volume requirements include a 10% margin and, in addition, allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. Using only one 10 gpm boric acid pump taking suction from the concentrated boric acid storage tank would require approximately 12.25 hours to inject the required boron. An alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. The required boric acid can be injected in less than six hours using only one of the makeup pumps.

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

REFERENCES

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

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

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

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

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

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

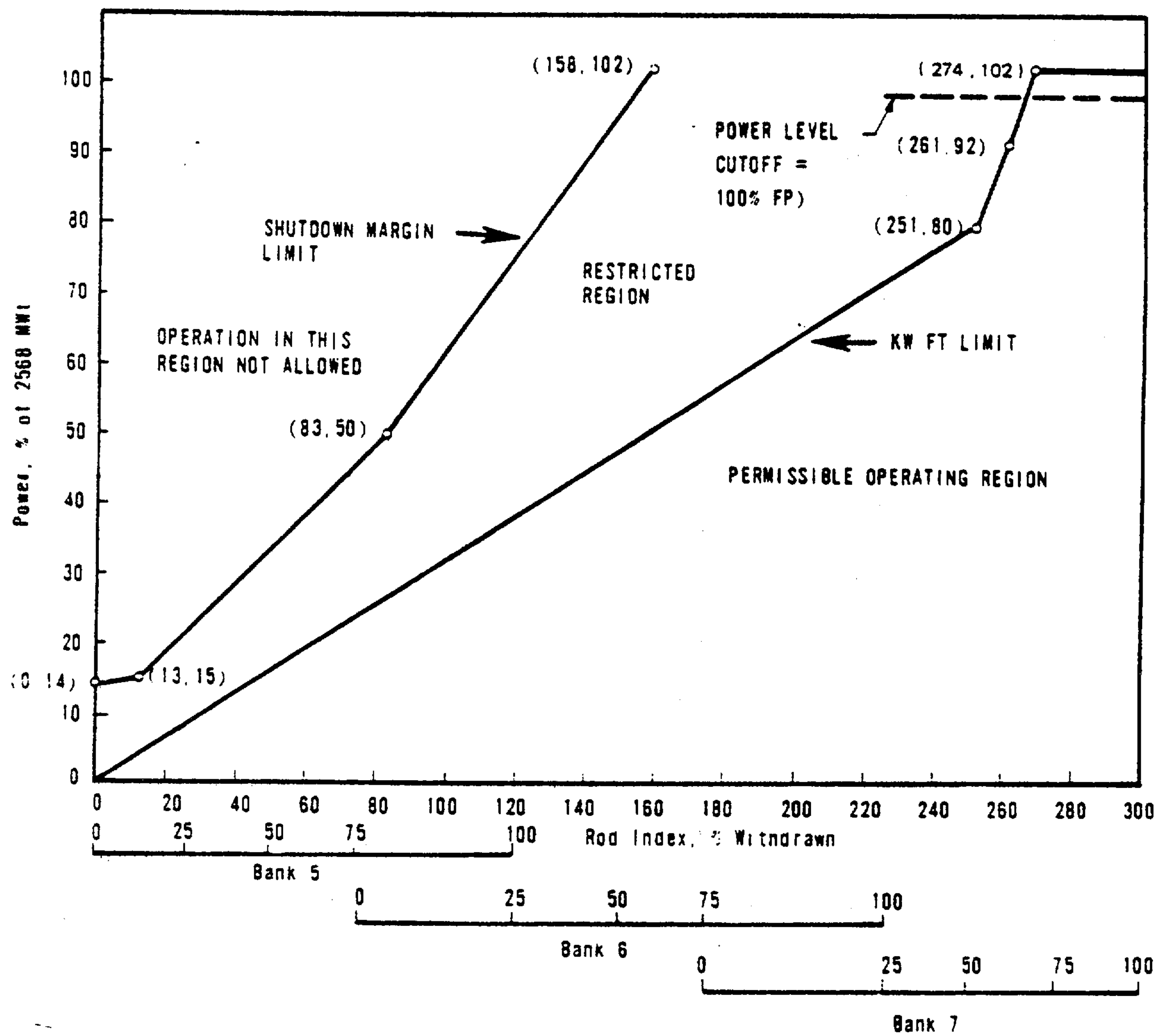
(3) FSAR, SUPPLEMENT 9

(4) B&W FUEL DENSIFICATION REPORT

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

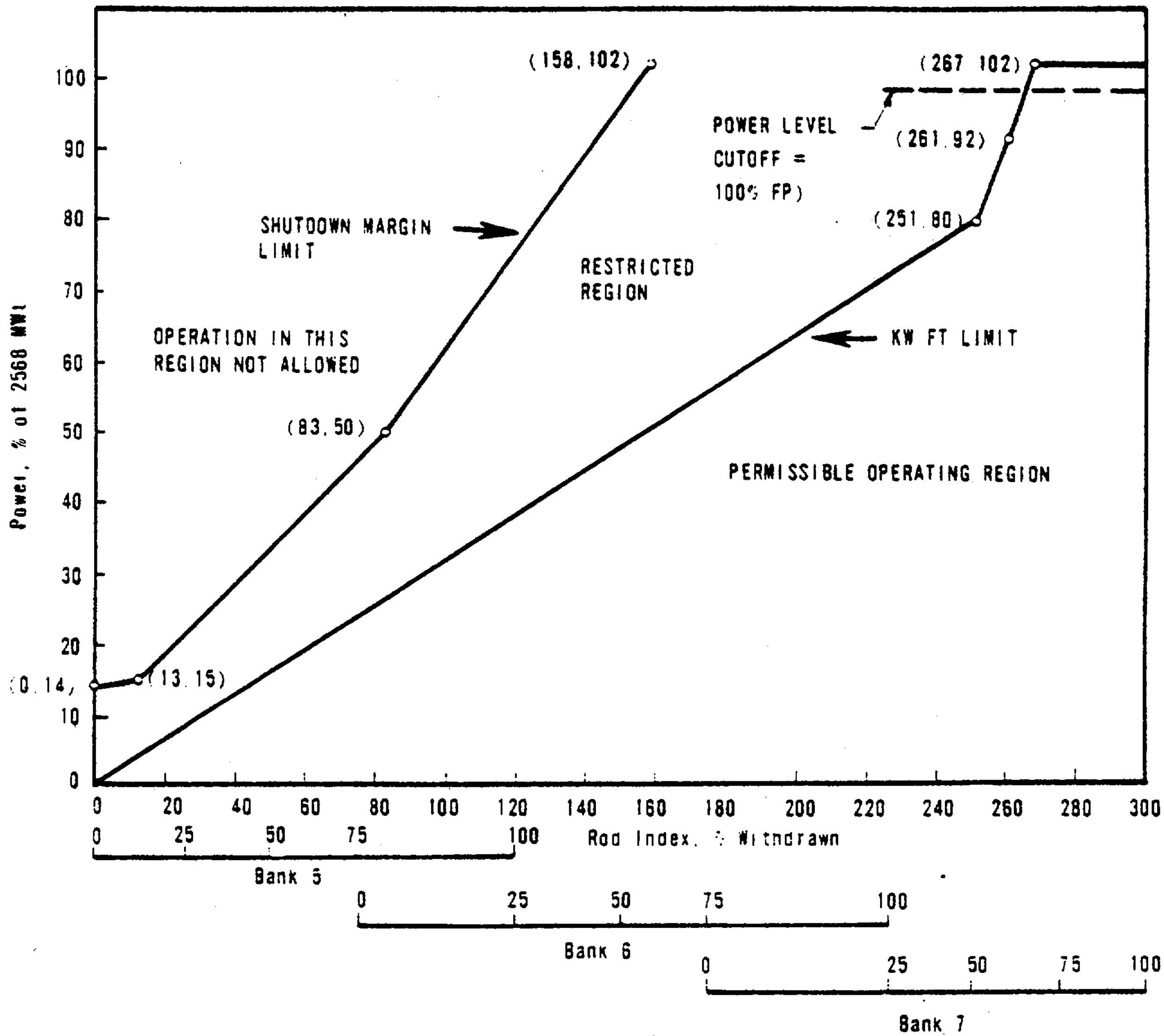
(5) Oconee 1, Cycle 4 - Reload Report - BAW 1447, March, 1977, Section 7.11

(6) Oconee 3, Cycle 6 - Reload Report - BAW-1634, August, 1980.

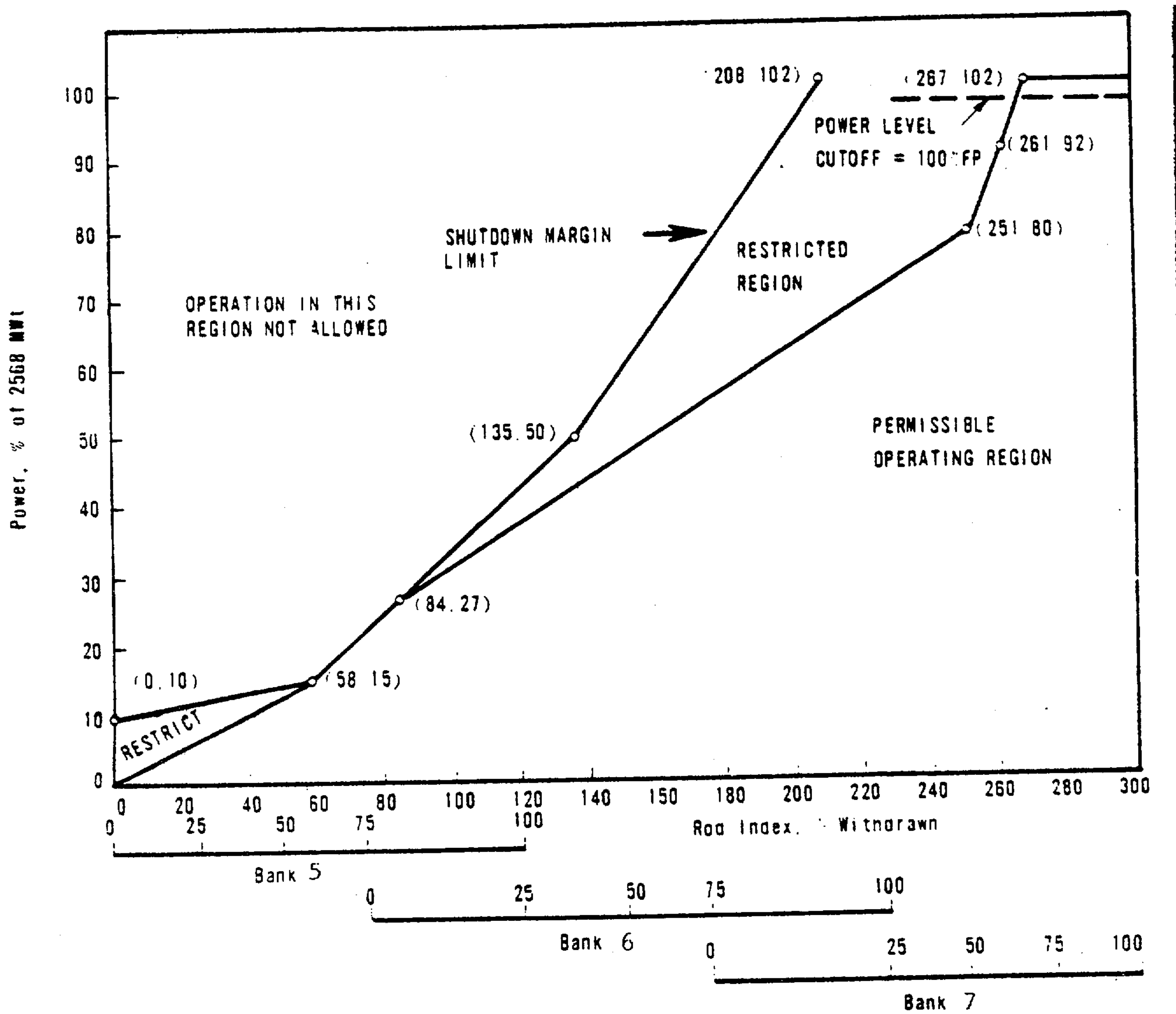


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 50 (+10, -0) EFPD
UNIT 3

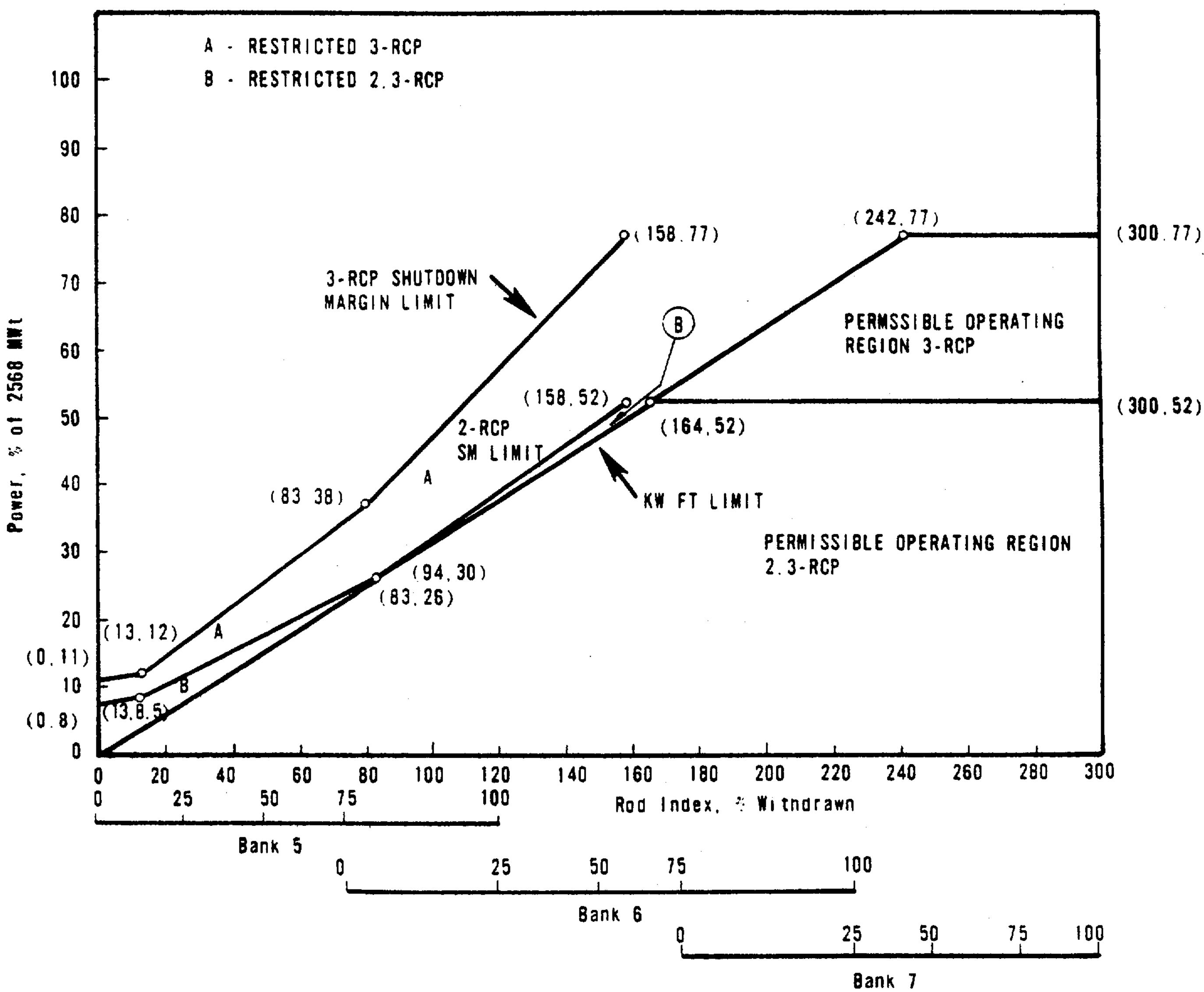
OCONEE NUCLEAR STATION
Figure 3.5.2-1C1



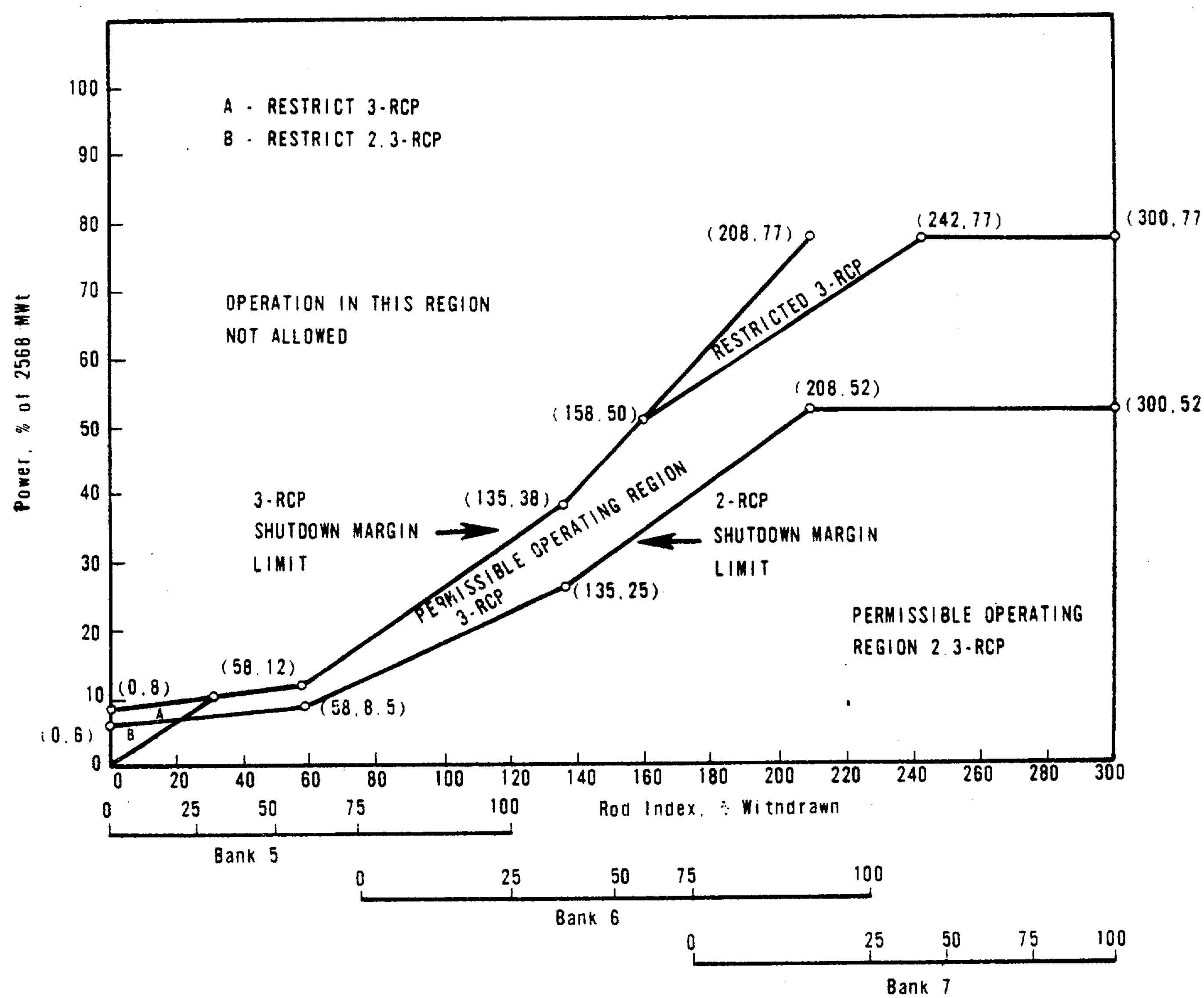
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 50 (+ 10, -0) TO 200 ($\pm$ 10) EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-1C2



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



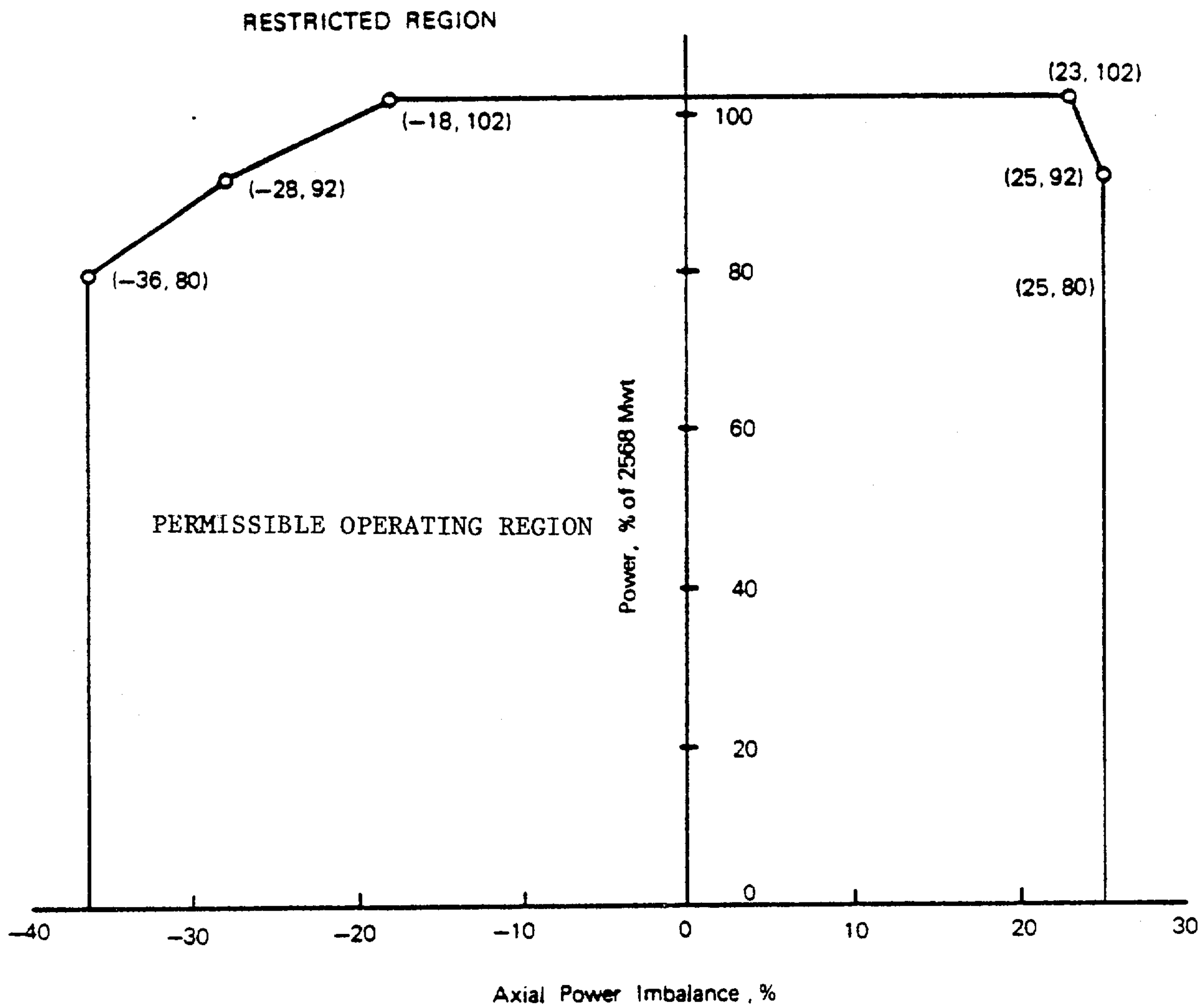
ROD POSITION LIMITS
FOR TWO- & THREE- PUMP OPERATION
FROM 0 to 200 ± 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-2C1



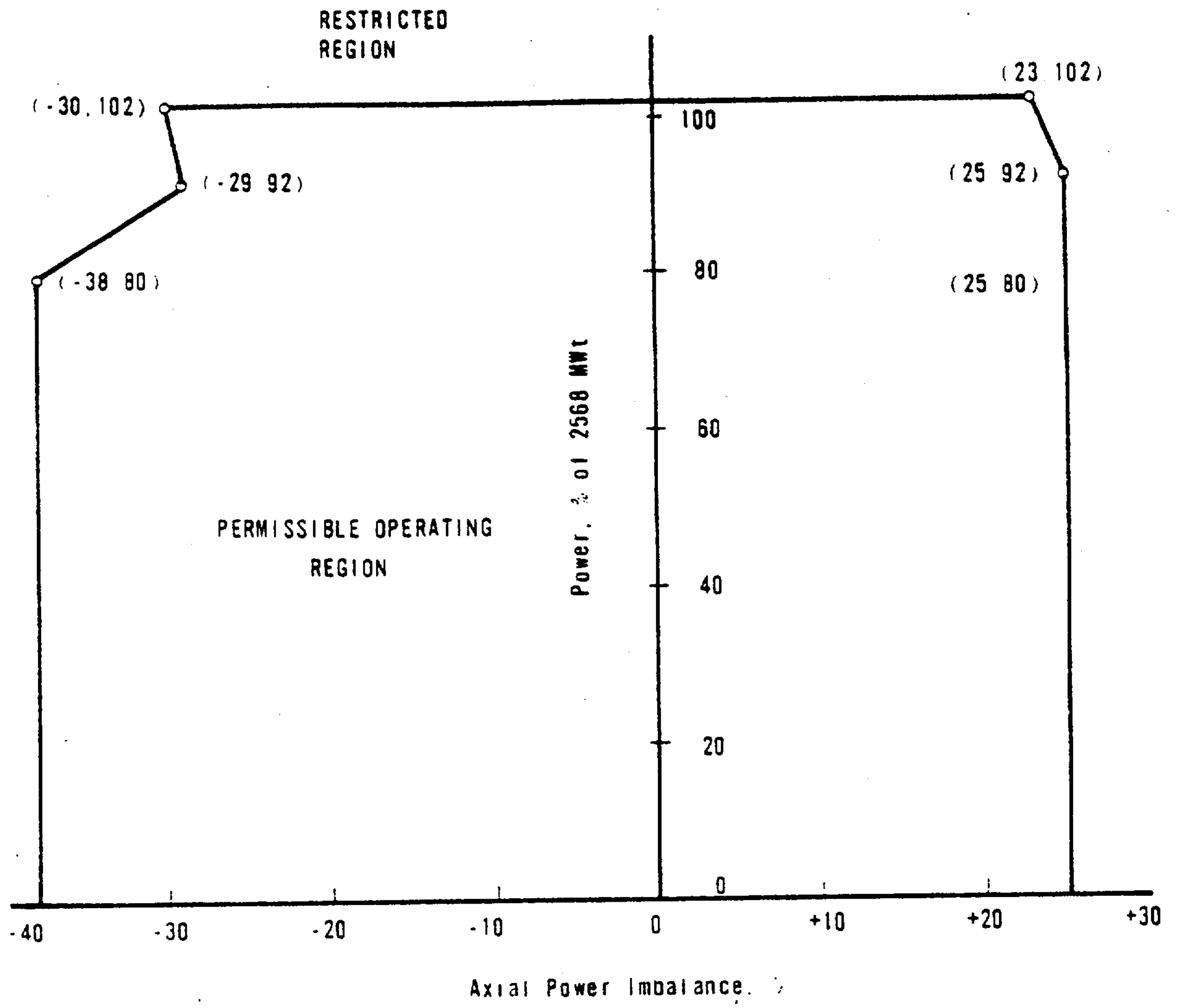
ROD POSITION LIMITS
FOR TWO- & THREE- PUMP OPERATION
AFTER 200 $\pm$ 10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-2C2

Figure 3.5.2-2C3

Deleted During Oconee Unit 3, Cycle 6 Operation



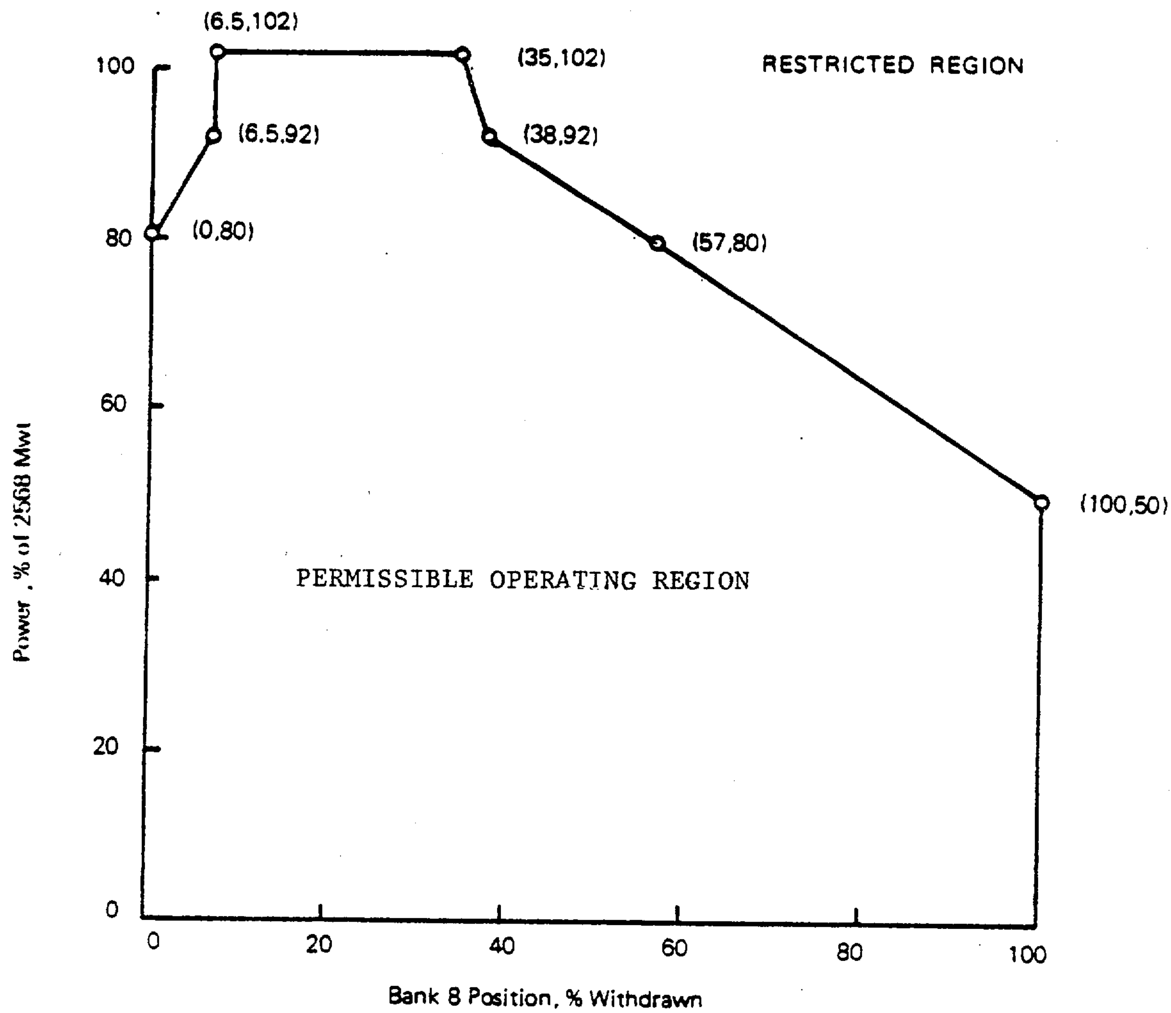
OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 50 (+ 10, -0) EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3C1



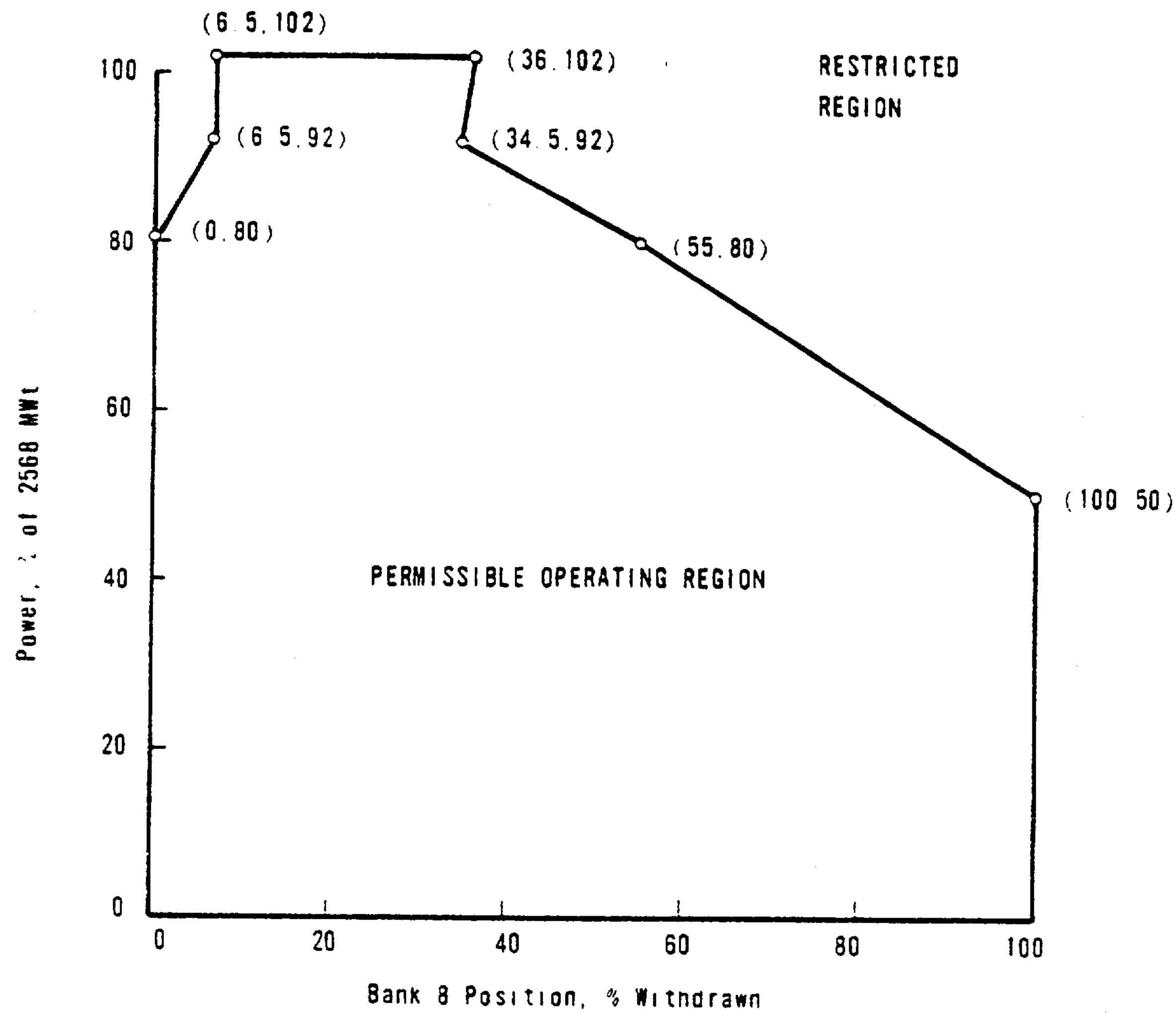
OPERATIONAL POWER
IMBALANCE ENVELOPE
AFTER 50 (+ 10, -0) EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3C2

Figure 3.5.2-3C3

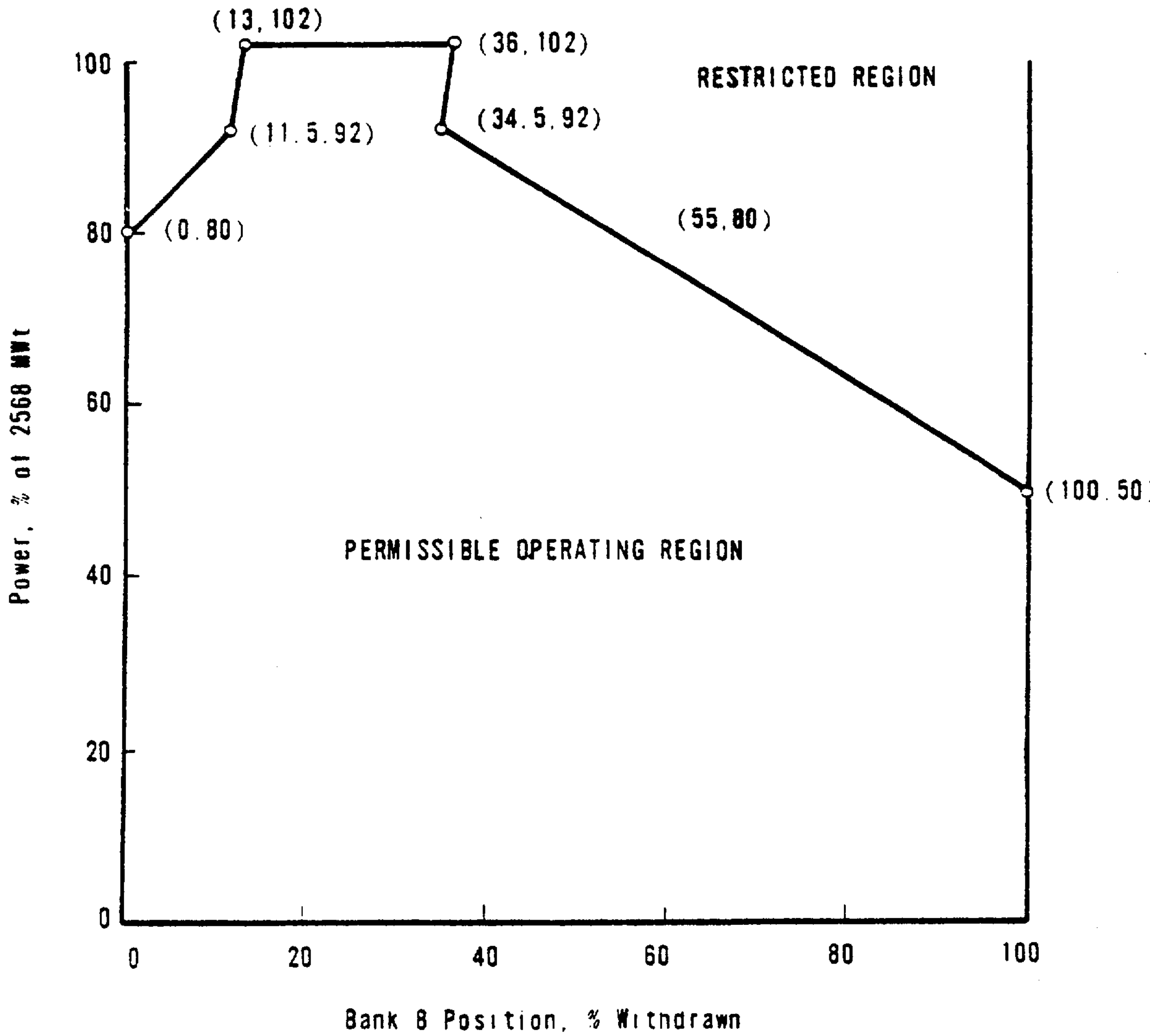
Deleted During Oconee Unit 3, Cycle 6 Operation



APSR POSITION LIMITS
FOR OPERATION
FROM 0 TO 50 (+ 10, -0) EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-4C1



APSR POSITION LIMITS
FOR OPERATION
FROM 50 (+10, -0) TO 200 (± 10) EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-4C2



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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

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

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

1.0 Introduction

By letter dated August 25, 1980(1), as supplemented December 22, 1980(2) and January 22, 1981(8), Duke Power Company (DPC or the licensee) requested amendments to the Appendix A Technical Specifications (TSs) of the Oconee Nuclear Station, Units 1, 2 and 3, Licenses Nos. DPR-38, DPR-47, and DPR-55. One request was to support the operation of Oconee Unit No. 3 at full rated power during Cycle 6. There were two other requested changes: one to add a new TS 3.1.11, Shutdown Margin, and the second to add a new Section 3.5.2.9 to TS 3.5.2, Control Rod Group and Power Distribution Limits; both apply to Oconee Units 1, 2 and 3.

2.0 Evaluation

2.1 Fuel Assembly Mechanical Design

The sixty-eight Babcock and Wilcox (B&W) Mark B-4 15x15 fuel assemblies loaded as Batch 8 at the end of Cycle 5 (EOC 5) are mechanically interchangeable with Batches 5B, 6 and 7 fuel assemblies previously loaded at Oconee 3. Fuel assemblies of the Mark B-4 design have been used in four previous refuelings of Oconee 3. The design was most recently approved(3) for the previous cycle of operation (4) and is used in other B&W nuclear steam supply systems. Two assemblies will contain regenerative neutron sources, and retainers will be used to contain the sources. Justification for the design and use of the neutron source retainer is described in the "Burnable Poison Rod Assembly Retainer Design Report"(5). A discussion of the burnable poison rods themselves is presented in Section 2.1.1 of this evaluation.

2.1.1 Reactivity Control System

In addition to the permanent reactivity control system (soluble boron and control rods), 60 burnable poison rod assemblies (BPRAs) are being added to control reactivity changes due to fuel burnup and fission product buildup. The BPRAs are normally removed from the reactor at the end of first cycle and reinserted only for extended cycle operation, such as that proposed for Cycle 6. In April 1978, two BPRAs were accidentally ejected from the core of another B&W-designed reactor at Crystal River(6). The

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ejected BPRAs were carried out of the reactor vessel by the coolant flow to the steam generator, where damage to the steam generator tube ends resulted. B&W determined that the ejection of the BPRAs from the core resulted from fretting wear in the holddown latching mechanism. In order to avoid similar problems at other plants, B&W redesigned and replaced the BPRA holddown mechanism on all operating B&W cores. The NRC staff has generically approved (7) the new design. We therefore conclude that changes to the core reactivity control system have been adequately considered for Cycle 6 operation.

2.1.2 Fuel Rod Design

Although all batches in Oconee 3 Cycle 6 utilize the same Mark B-4 fuel, the Batch 8 assemblies incorporate a slightly higher initial fuel density. The change, from 94 to 95 percent of theoretical density, is a consequence of using a modified fuel fabrication process. The stability (densification resistance) of both fuel types is similar. As a consequence, the densified fuel stack height is virtually unchanged for the Batch 8 assemblies. Densification in Oconee 3 Cycle 6 fuel is discussed further in Section 2.3 of this report.

2.2.1 Cladding Collapse

The licensee has stated(8) that the cladding collapse analysis in the Cycle 6 Reload Report(1) is bounded by conditions previously analyzed in the Oconee Unit 3 Final Safety Analysis Report (FSAR) or analyzed specifically for Cycle 6 conditions using methods and limits previously reviewed and approved by the NRC. We conclude that additional NRC staff review of the cladding collapse analysis is unnecessary for Cycle 6 operation due to the similarity of Cycle 6 fuel to previous fuel.

2.2.2 Cladding Stress

The licensee has stated(8) that the cladding stress analysis described in the Cycle 6 Reload Report(1) is bounded by conditions previously analyzed in the Oconee 3 FSAR or analyzed specifically for Cycle 6 conditions using methods and limits previously reviewed and approved by the NRC. We conclude that additional NRC staff review of the cladding stress analysis is unnecessary for Cycle 6 operation.

2.2.3 Cladding Strain

The licensee has stated(8) that the cladding strain analysis described in the Cycle 6 Reload Report(1) is bounded by conditions previously analyzed in the Oconee 3 FSAR or analyzed specifically for Cycle 6 conditions using methods and limits previously reviewed and approved by the NRC. We conclude that additional NRC staff review of the cladding strain analysis is unnecessary for Cycle 6 operation.

2.2.4 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (SRP)(9) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Among those which may affect the operation of the fuel rod is the internal pressure limit. The acceptance criterion (SRP 4.2, Section II.A.1(f)) is that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified.

DPC has stated(1) that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 6. This analysis is based on the use of the B&W TAFY code(10) rather than a newer B&W code called TACO(11). Although both of these codes have been approved for use in safety analyses, we believe(12) that only the newer TACO code is capable of correctly calculating fission gas release (and therefore rod pressure) at very high burnups. B&W has responded(13) to this concern with an analytical comparison between both codes. In this response, they have stated that the internal fuel rod pressure predicted by TACO is lower than that predicted by TAFY for fuel rod exposures of up to 42,000 MWd/MTu. Although we have not examined the comparison, we note that the analyses exceed the expected exposure (37,000 MWd/MTu) in Oconee 3 at EOC 6. Therefore, we conclude that the rod internal pressure limits have been adequately considered.

2.3 Fuel Thermal Design

The average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the Loss of Coolant Accident (LOCA) analysis (Section 7.2 of the Reload submittal) are also calculated with the TAFY code(10). B&W has stated(1) that the fuel temperature and pin pressure data used in the generic LOCA analysis(14) are conservative compared with those calculated for Cycle 6 at Oconee 3.

As previously mentioned in Section 2.2.4 of this evaluation, B&W currently has two fuel performance codes, TAFY(10) and TACO(11), which could be used to calculate the LOCA initial conditions. The older code, TAFY, has been used for the Cycle 6 LOCA analysis. Recent information(15) indicates that the TAFY code predictions do not produce higher peak cladding temperatures than TACO for all Cycle 6 conditions as suggested in Ref. 13. The issue involves calculated fuel rod internal gas pressures that are too low at beginning of life. The rod internal pressures are used to determine swelling and rupture behavior during LOCA. B&W has proposed(16) a method of resolving this issue which we accepted(17). The method involves the use of reduced LOCA kW/ft limits at low core elevations during the first 50 effective full power days (EFPD) of operation. The licensee has incorporated(2) these changes into the Oconee Nuclear Station TSs to support the operation of Oconee 3 at full rated power during Cycle 6. We have reviewed (18) these changes and find them acceptable. We conclude that the initial thermal conditions for LOCA analysis have been appropriately considered for Cycle 6 operation.

2.4 Material Compatibility

The chemical and material compatibility of possible fuel, cladding and coolant interactions is unchanged from the previous cycle of operation. The impact of this issue on the operational safety of Oconee 3 need not be reconsidered for Cycle 6 operation.

2.5 Operating Experience

B&W has accumulated operating experience with the Mark B 15x15 fuel assembly at all of the eight operating B&W 177-fuel assembly plants. A summary of this operating experience as of April 30, 1980, is given on page 4-3 of Ref. 1.

2.6 Fuel Rod Bowing

The licensee has stated that a fuel rod bowing penalty has been calculated according to the procedure that was approved in Ref. 19. The burnup used in that calculation was the maximum fuel assembly burnup of the batch that contains the limiting fuel assembly. For Cycle 6, this burnup is 23,411 MWd/MTu in a Batch 7 assembly. The resultant rod bowing penalty was found to be a 2.1% reduction in Departure from Nucleate Boiling Ratio (DNBR).

To offset the 2.1% penalty, the licensee has drawn upon both generic and plant-specific margins. The generic margin employed was a thermal credit equivalent to 1% DNBR. This credit is a result of the standard flow-area-reduction factor included in all B&W hot-channel thermal-hydraulic analyses. The plant-specific margin employed was a 10% DNBR credit available because plant operating limits were set at conservative values that correspond to the original method [20] of calculating rod bowing penalties rather than the new procedure.

During our review of this reload application, we audited the Cycle 6 DNBR penalty due to fuel rod bowing. We were unable to reproduce the penalty, and therefore requested additional information from the licensee. Based on information supplied in the response[8] from DPC, we were able to duplicate the DNBR penalty as previously specified in Section 6 of the reload report[1]. We conclude that the DNBR reduction due to fuel rod bowing has been conservatively calculated for Cycle 6 operation. In order to provide for a proper accounting of margins used to offset the DNBR penalty, we required--as on other operating reactors--that the bases for the TSs for Oconee Unit 3 be amended to identify each generic or plant-specific margin that was used. The licensee provided such an amendment that identified the generic margin, and we provided the plant-specific margin in the TS bases.

2.7 Nuclear Design

We have reviewed the effect on the rod insertion limit and axial imbalance limiting conditions of operation caused by the reduction in allowable heat generation rate at the bottom of the core due to the TAFY-TACO conversion. In order to meet the reduced limit on the power in the lower half of the core during the first 50 EFPD of the cycle, the allowable negative imbalance has been reduced, the amount of control rod insertion allowed at full power has been decreased, and the amount of permitted withdrawal of the axial power shaping rods has been reduced. All of these actions are in a direction to reduce the power at the bottom of the core. The techniques used to obtain the revised limiting conditions of operation are the same as have been previously used to obtain limiting operating conditions. On the basis of our review, which is discussed above, we conclude that the revised TSs are acceptable.

A further TS (3.5.2.9) specifies that the curves shown in the various Specifications shall be valid only to the end of the nominal cycle length (in spite of the open ended nature, e.g., Figure 3.5.2-1C3 which is designated for use after 200 $\pm$ 10 EFPD).

However, use of these curves would be permitted after the end of the nominal cycle if analyses are performed which confirm their suitability. Such use would not, therefore, involve a TS change. If analyses failed to confirm the suitability of the curves, a TS change would have to be obtained to continue operation beyond the nominal cycle length. We find this approach to be acceptable.

2.8 Shutdown Margin

The licensee proposed adding a new TS 3.1.11, Shutdown Margin. The current TSs included a shutdown margin only for the operating and refueling conditions. TS 3.1.11 provides for a shutdown margin greater than 1% $\Delta k/k$ with the highest worth control rod withdrawn for all modes of operation and ensures the reactor can remain subcritical during various shutdown conditions. We conclude that TS 3.1.11 provides conservative shutdown margins for all modes of reactor operation and is thus acceptable.

2.9 Thermal and Hydraulic Design

The thermal and hydraulic design of the reload core was reviewed to confirm that it uses acceptable analytical methods, is equivalent to or is a justified extrapolation from previously approved core designs, and provides an acceptable margin of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational transients. Oconee 3 Cycle 6 consists of 68 new Mark B-4 Batch 8 fuel assemblies. There are 60 BPRAs inserted for Cycle 6 operation. Retainers are used on these assemblies as described in Ref. 5. Two assemblies contain regenerative neutron sources. The number of open assemblies is 46. The Cycle 5 and 6 maximum design conditions are provided in Table 6-1 of Ref. 1. The burnup used to calculate the rod bow penalty is the highest Batch 7 burnup of 23,411 MWd/MTu.

2.9.1 Evaluation of Thermal-Hydraulic Design

The incoming Batch 8 fuel is hydraulically and geometrically similar to the fuel remaining from the previous cycles. The thermal-hydraulic models and methodologies used to support Cycle 6 operation are described in Ref. 21, 22 and 23. The main differences between Cycle 6 and the Reference Cycle 5 are discussed below.

Core Bypass Flow

The maximum core bypass flow in Cycle 5 was 10.4%. For Cycle 6 operation, 60 BPRAs will be inserted, leaving 46 open assemblies, resulting in a decrease in calculated maximum core bypass flow to 8.1% (i.e., net increase in core flow).

BPRA Retainers

The retainers added to provide positive hold-down of BPRAs introduce a small DNBR penalty discussed in Ref. 5. However, the increase in core flow due to the BPRA insertion more than compensates for the decrease in DNBR due to the BPRA retainers.

Rod Bow DNBR Penalty

The rod bow DNBR penalty applicable to Cycle 6, according to the licensee, was calculated using the interim rod bow penalty evaluation procedure approved in Ref. 19. The burnup used to calculate the penalty was the highest Batch 7 burnup, 23,411 MWd/MTu. The calculated rod bow penalty using this procedure is 2.1%. Utilizing the 1% DNB credit for the flow area reduction hot channel factor, the actual penalty is 1.1%. However, according to the licensee, all plant operating limits based on DNBR criteria include a minimum of 10% DNBR margin available due to the plant operating limits being set at conservative values that correspond to the original method (20) of calculating rod bow penalties rather than the new procedure given in Ref. 19. The licensee wants to do this for their convenience of establishing the set points once for all the future reloads. Therefore, we find the licensee's minimum DNBR limit value of 1.43 to be conservative and acceptable.

3.0 Evaluation of Transients and Accidents

The licensee has examined each FSAR (21) accident analysis with respect to the changes in Cycle 6 parameters to determine their effect on the plant performance during the analyzed transients. The parameters having an effect on the outcome of a transient are the core thermal parameters, thermal-hydraulic parameters, and the physics and kinetics parameters. The kinetics parameters, including reactivity feedback coefficients and control rod worths, have the greatest effect on the outcome of a transient. The licensee, in Table 7-1 of Ref. 1, compared the Cycle 6 input parameters to the FSAR values. Our review of these input parameters indicate that Cycle 6 is bounded by the FSAR values. Fuel thermal analysis values are listed in Table 4-2 of Ref. 1 for all fuel batches in Cycle 6. Table 6-1 of Ref. 1 compares the thermal-hydraulic parameters for Cycles 5 and 6. These parameters are the same for both cycles with the exception of the higher value of design maximum DNBR for Cycle 6 (2.05 as compared to 1.98 for Cycle 5). According to the FSAR (Ref. 21), loss of flow (2 pump coast down) is the worst transient and the minimum DNBR is 1.4326, which is within the licensee's acceptable limit of 1.43.

We conclude from our review of the Cycle 6 core accident-related parameters, with respect to acceptable previous cycle values and with respect to the FSAR values, that this core reload design will enable safe operation of Oconee 3 during Cycle 6.

4.0 TS Changes

The proposed modifications to the Core Protection Safety Limits of Specification 2.1 (Figure 2.1-3C, Page 2.1-12 of Ref. 1) have been reviewed for the Oconee 3 Cycle 6 operation, and we find the revised TSs acceptable. The TS changes related to cycle length and shutdown margin have been reviewed in Sections 2.7 and 2.8 of this Safety Evaluation.

5.0 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.

6.0 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: February 10, 1981

REFERENCES

1. W. O. Parker, Jr. (Duke) letter dated August 25, 1980, to R. W. Reid (NRC) transmitting Oconee Unit 3, Cycle 6 Reload Report (BAW-1634) dated August 1980.

2. W. O. Parker, Jr. (Duke) letter to H. R. Denton (NRC) dated December 22, 1980.

3. R. W. Reid (NRC) letter to W. O. Parker, Jr. (Duke) dated June 22, 1979.

4. Oconee Unit 3 Cycle 5 Reload Report, Babcock & Wilcox Company Report BAW-1522, March 1979.

5. BPRA Retainer Design Report, Babcock & Wilcox Company Report BAW-1496, May 1978.

6. W. P. Stewart (Florida Power Corporation) letter to C. Nelson (NRC) on "Crystal River Unit Three Status Report - May 1, 1978," dated May 4, 1978.

7. T. M. Novak (NRC) memorandum to E. L. Jordon (NRC) dated December 22, 1980.

8. W. O. Parker, Jr. (Duke) letter to H. R. Denton (NRC) dated January 22, 1981.

9. Standard Review Plan, Section 4.2 (Rev. 1), "Fuel System Design," U. S. Nuclear Regulatory Commission Report NUREG-75/087.

10. C. D. Morgan and H. S. Kao, "TAFY-Fuel Pin Temperature and Gas Pressure Analysis," Babcock and Wilcox Company Report BAW-10044, May 1972.

11. "TACO-Fuel Pin Performance Analysis," Babcock and Wilcox Company Report BAW-10087P-A, Rev.2, August 1977.

12. D. F. Ross, Jr. (NRC) letter to J. H. Taylor (B&W) dated January 18, 1978.

13. J. H. Taylor (B&W) letter to P. S. Check (NRC), dated July 18, 1978.

14. W. L. Bloomfield, et al., "ECCS Analysis of B&W's 177-FA Raised-Loop NSS," Babcock and Wilcox Company Report BAW-10105, June 1975.

15. R. O. Meyer (NRC) memorandum to L. S. Rubenstein (NRC) on "TAFY/TACO Fuel Performance Models in B&W Safety Analyses," dated June 10, 1980.

16. J. H. Taylor (B&W) letter to L. S. Rubenstein (NRC) dated September 5, 1980.

17. L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) dated October 28, 1980.

18. W. V. Johnston (NRC) memorandum for R. W. Reid (NRC) on "Oconee Unit 3 Reload for Cycle 6" dated January 13, 1981.

19. L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) on "Evaluation of Interim Procedure for Calculating DNBR Reduction Due to Rod Bow," dated October 18, 1979.

20. D. F. Ross and D. G. Eisenhut (NRC) memorandum to D. B. Vassallo and K. R. Goller (NRC) on "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," dated December 8, 1976.

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

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

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

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. 93 , 93 , and 90 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the Technical Specifications (TSs) for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments revise the Station's common TSs to support the operation of Oconee Unit No. 3 at full rated power during Cycle 6. The amendments also add a new TS 3.1.11, Shutdown Margin, and a new Section 3.5.2.9 to TS 3.5.2,Control Rod Group and Power Distribution Limits, for Oconee Units Nos. 1, 2 and 3.

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

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

For further details with respect to this action, see (1) the application for amendments dated August 25, 1980, as supplemented December 22, 1980, and January 22, 1981, (2) Amendments Nos. 93 , 93 , and 90 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oconee County Library, 501 West Southbroad Street, Walhalla, South Carolina. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 10th day of February 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

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