December 15, 1978

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Docket 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. 66, 66 and 63 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 September 18, 1978, as supplemented September 25, and November 1, 1978.

These amendments revise the Technical Specifications to support the operation of Oconee Unit No. 2 at full rated power during Cycle 4 after core reload and removal of the orifice rod assemblies from the core. The amendments also revise the Technical Specifications for Units 1, 2 and 3 in regard to control rod operability.

In accordance with your letter dated September 18, 1978, the Commission has also issued the enclosed Exemption for Oconee Unit No. 2 from the requirements of 10 CFR 50.46(a)(1) that Emergency Core Cooling System (ECCS) performance be calculated in accordance with an acceptable calculation model which conforms to the provisions in Appendix K to 10 CFR 50.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed. A copy of the Exemption is also being filed with the Office of the Federal Register for publication.

Sincerely,

Original Signed by

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

7812280238

CP-1

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

cc w/enclosures:
See next page

OFFICE➤						
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Enclosures:
1. Amendment No. 66 to DPR-38
2. Amendment No. 66 to DPR-47
3. Amendment No. 63 to DPR-55
4. Exemption
5. Safety Evaluation
6. Notice

cc w/enclosures:
See next page

Duke Power Company

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

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DeBevoise & Liberman
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

Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U.S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S.W.
Washington, D.C. 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
Office of Inspection and Enforcement
ATTN: Mr. Francis Jape
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

cc w/enclosure(s) and incoming
dtd.: 9/18 & 25 and 11/1/78

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. 66
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated September 18, 1978, as supplemented September 25 and November 1, 1978, 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.

7812280247

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. 66 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

Gerald B. Zwetzig
for Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 15, 1978

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. 66
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated September 18, 1978, as supplemented September 25 and November 1, 1978, 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. 66 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

for Gerald B. Zwetzig
Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 15, 1978

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. 63
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated September 18, 1978, as supplemented September 25 and November 1, 1978, 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. 63 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

Gerald B. Zwetzig
for Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 15, 1978

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 66 TO DPR-38

AMENDMENT NO. 66 TO DPR-47

AMENDMENT NO. 63 TO DPR-55

Revise Appendix A as follows:

Remove the following pages and insert the revised identically numbered pages.

2.1-3a & 2.1-3b
2.1-8 (Figure 2.1-2B)
2.3-9 (Figure 2.3-2B)
3.2-1 & 3.2-2
3.5-5 - 3.5-11
3.5-11a & 3.5-11b
3.5-11c* (Table 3.5-1)
3.5-14 (Figure 3.5.2-1B1)
3.5-14a (Figure 3.5.2-1B2)
3.5-15
3.5-19 (Figure 3.5.2-2B1)
3.5-19a (Figure 3.5.2-2B2)
3.5-19b
3.5-22 (Figure 3.5.2-3B1)
3.5-22a (Figure 3.5.2-3B2)
3.5-22b
3.5-23f (Figure 3.5.2-4B1)
3.5-23g (Figure 3.5.2-4B2)
3.5-23h
4.1-9

Changes on the revised pages are identified by marginal lines. Page 3.5-5 is unchanged and is included for convenience only.

*New Page

Bases - Unit 2

The safety limits presented for Oconee Unit 2 have been generated using BAW-2 critical heat flux correlation[1] and the Reactor Coolant System flow rate of 106.5 percent of the design flow (design flow is 352,000 gpm for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate[2].

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1B represents the conditions at which a minimum DNBR of 1.30 is predictecd for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 374,880 gpm). This curve is based on the following nuclear power peaking factors with potential fuel densification and fuel rod bowing effects:

$$F_q^N = 2.565; \; F_{\Delta H}^N = 1.71^{(3)} \; F_z^N = 1.50$$

The design peaking combination results in a more conservative DNBR than any other power shape that exists during normal operation.

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

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

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 19.8 kw/ft for Unit 2.

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

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

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

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

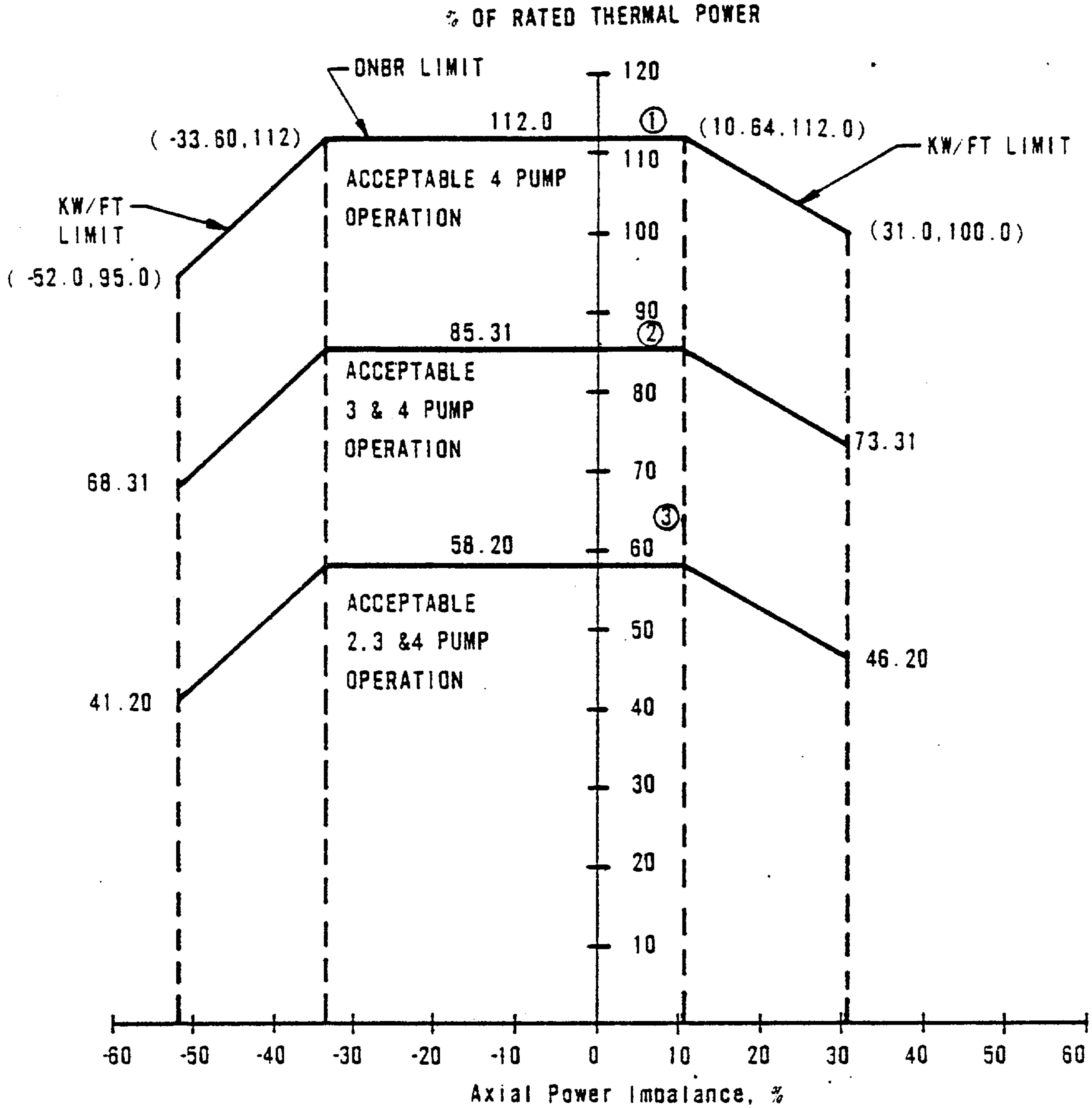
For each curve of Figure 2.1-3B, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The 1.30 DNBR curve for four-pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four-pump curve will be above and to the left of the other curves.

References

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

(2) Oconee 2, Cycle 3 - Reload Report - BAW-1452, April, 1977.

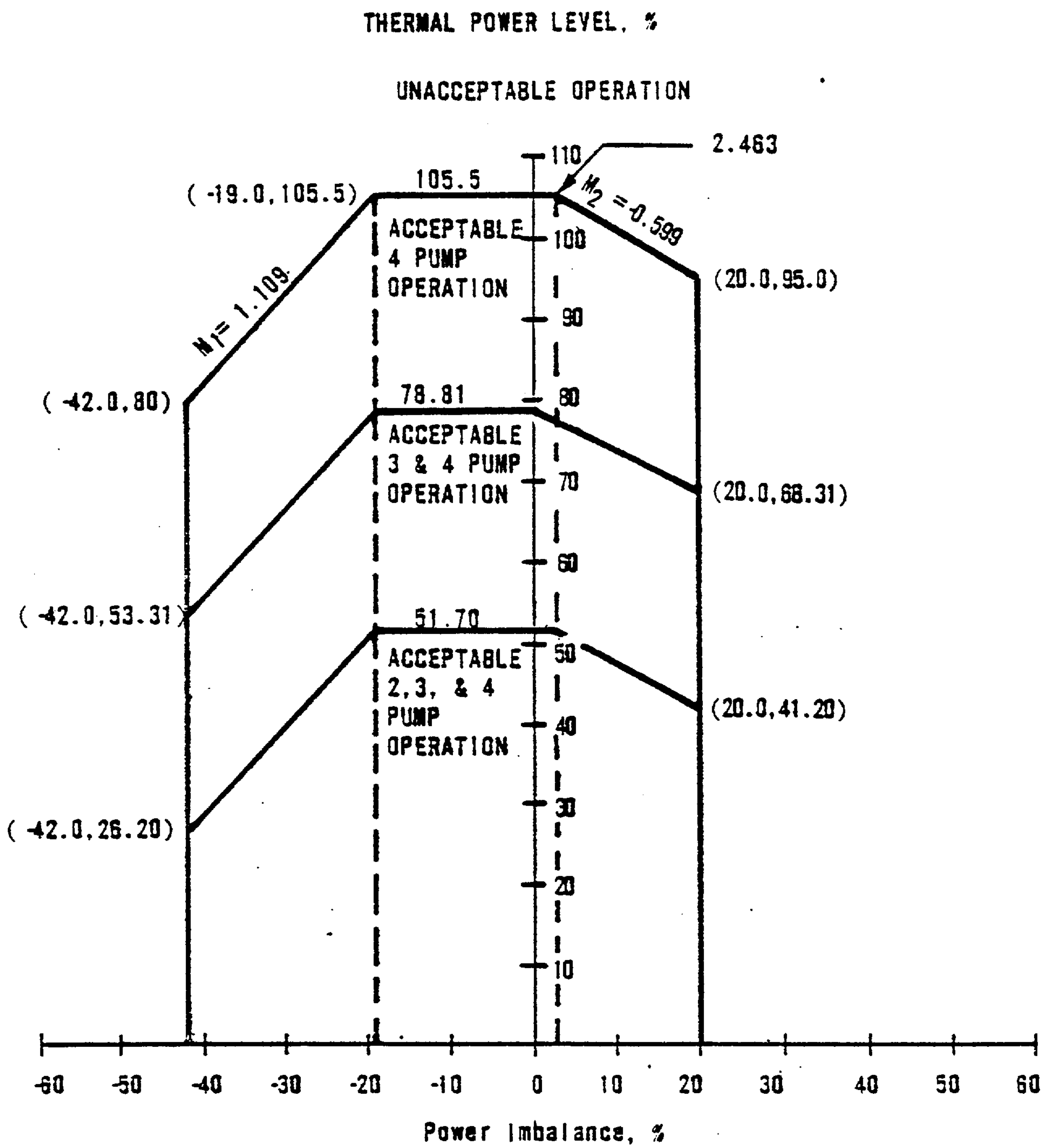
(3) Oconee 2, Cycle 4 - Reload Report - BAW-1491, August, 1978.



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

CORE PROTECTION
SAFETY LIMITS
UNIT 2

OCONEE NUCLEAR STATION
Figure 2.1-2B



PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.3-2B

3.2 HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the high pressure injection and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification

The reactor shall not be critical unless the following conditions are met:

3.2.1 Two high pressure injection pumps per unit are operable except as specified in 3.3.

3.2.2 One source per unit of concentrated soluble boric acid in addition to the borated water storage tank is available and operable.

This source will be the concentrated boric acid storage tank containing at least the equivalent of 995 ft^3 of 8700 ppm boron as boric acid solution with a temperature at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall be operable and shall have the same temperature requirement as the concentrated boric acid storage tank. At least one channel of heat tracing capable of meeting the above temperature requirement shall be in operation. One associated boric acid pump shall be operable.

If the concentrated boric acid storage tank with its associated flowpath is unavailable, but the borated water storage tank is available and operable, the concentrated boric acid storage tank shall be restored to operability within 72 hours or the reactor shall be placed in a hot shutdown condition and be borated to a shutdown margin equivalent to 1% $\Delta k/k$ at 200°F within the next twelve hours; if the concentrated boric acid storage tank has not been restored to operability within the next 7 days the reactor shall be placed in a cold shutdown condition within an additional 30 hours.

If the concentrated boric acid storage tank is available but the borated water storage tank is neither available nor operable, the borated water storage tank shall be restored to operability within one hour or the reactor shall be placed in a hot shutdown condition within 6 hours and in a cold shutdown condition within an additional 30 hours.

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 5, Oconee 2 Cycle 4, and Oconee 3 Cycle 4 were analyzed with the most limiting case selected as the basis for all three units. Since only the present cycles were analyzed, the specifications will be re-evaluated with each reload. A minimum of 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

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (Cont'd)

	Functional Unit	(A) Minimum Operable Analog Channels	(B) Minimum Degree Of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
	b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
15.	Turbine Stop Valves Closure	2	1	Bring to hot shutdown within 12 hours (f)

(a) For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

(d) Single loop operation at power (after testing and approval by the AEC/DOL) is not permitted unless the operating channels are the two receiving Reactor Coolant Temperature from operating loop.

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in the cold shutdown condition within 24 hours.

(f) One operable channel with zero minimum degree of redundancy is allowed for 24 hours before going to the hot shutdown condition.

3.5.2 Control Rod Group and Power Distribution Limits

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 Shutdown Margin

a. The available shutdown margin shall be greater than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

b. If the shutdown margin is less than 1% $\Delta k/k$, then within 1 hour initiate and continue boration until the required shutdown margin is restored. The requirements of specification 3.5.2.5.c shall be met.

3.5.2.2 Movable Control Assemblies

a. All control (safety and regulating) rods shall be operable and positioned within nine (9) inches of their group average height.

b. A control rod shall be declared inoperable if any of the following conditions exist for that rod:

1. The control rod cannot be moved due to excessive friction or mechanical interference, or cannot perform its intended trip function.

2. The control rod cannot be located by either absolute or relative position indication or by in or out limit lights.

3. The control rod is misaligned with its group average by more than nine (9) inches.

4. The control rod does not meet the exercise requirements of Specification 4.1.

5. The control rod does not meet the rod trip insertion times of Specification 4.7.1.

6. The control rod does not meet the rod program verification of Specification 4.7.2.

c. If a control rod is declared inoperable by being immovable due to excessive friction or mechanical interference or known to be untrippable then:

   1. Within 1 hour verify that the shutdown margin requirement of Specification 3.5.2.1 is satisfied, and

   2. Within 12 hours place the reactor in the hot standby condition.

d. If a control rod is declared inoperable due to causes other than addressed in 3.5.2.2.c above then:

   1. Within 1 hour either restore the rod to operable status, or

   2. Continue power operation with the control rod declared inoperable, and

      a. Within 1 hour verify the shutdown margin requirement of Specification 3.5.2.1 with an additional allowance for the withdrawn worth of the inoperable rod, and

      b. Either reactor thermal power shall be reduced to less than 60% of the allowable power for the reactor coolant pump combination within 1 hour and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow/imbalance, shall be reduced within the next 4 hours to 65.5% of thermal power value allowable for the reactor coolant pump combination, or

      c. Position the remaining rods in the affected group such that the inoperable rod is maintained within allowable group average limits of Specification 3.5.2.2.a and the withdrawal limits of Specification 3.5.2.5.c.

e. If more than one control rod is inoperable or misaligned, the reactor shall be shut down to the hot standby condition within 12 hours.

3.5.2.3 The worths of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the control rod position limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Power Tilt

a. Except for physics tests, the maximum positive quadrant power tilt shall not exceed the Steady State Limit of Table 3.5-1 during power operation above 15% full power.

b. If the maximum positive quadrant power tilt exceeds the Steady State Limit but is less than or equal to the Transient Limit of Table 3.5-1, then:

1. Either the quadrant power tilt shall be reduced within 2 hours to within its Steady State Limit, or

2. The reactor thermal power shall be reduced below the power level cutoff (as specified in Specification 3.5.2.5) and further reduced 2% thermal power for each 1% of quadrant power tilt in excess of the Steady State Limit, and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within 4 hours by 2% thermal power for each 1% tilt in excess of the Steady State Limit. If less than four reactor coolant pumps are in operation, the allowable thermal power for the reactor coolant pump combination shall be reduced by 2% for each 1% excess tilt.

c. Quadrant power tilt shall be reduced within 24 hours to within its Steady State Limit, or

1. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

d. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit of Table 3.5-1 and if there is a simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 30 minutes at least 2% for each 1% of the quadrant power tilt in excess of the Steady State Limit.

2. Either quadrant power tilt shall be reduced within 2 hours to within its Transient Limit, or

3. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

e. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit of Table 3.5-1, due to causes other than simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 2 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

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

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

3.5.2.5 Control Rod Positions

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

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

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on figures 3.5.2-1A1 and 3.5.2-1A2 (Unit 1); 3.5.2-1B1, 3.5.2-1B2 and 3.5.2-1B3 (Unit 2); 3.5.2-1C1, 3.5.2-1C2 and 3.5.2-1C3 (Unit 3) for four pump operation, and on figures 3.5.2-2A1 and 3.5.2-2A2 (Unit 1); 3.5.2-2B1, 3.5.2-2B2 and 3.5.2-2B3 (Unit 2); 3.5.2-2C1, 3.5.2-2C2 and 3.5.2-2C3 (Unit 3) for two or three pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-4A1, and 3.5.2-4A2 (Unit 1); 3.5.2-4B1, 3.5.2-4B2, and 3.5.2-4B3 (Unit 2); 3.5.2-4C1, 3.5.2-4C2, and 3.5.2-4C3 (Unit 3).

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

3.5.2.6 Xenon Reactivity

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

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

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

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

3.5.2.7 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

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

Bases

Operation at power with an inoperable control rod is permitted within the limits provided. These limits assure that an acceptable power distribution is maintained and that the potential effects of rod misalignment on associated accident analyses are minimized. For a rod declared inoperable due to misalignment, the rod with the greatest misalignment shall be evaluated first. Additionally, the position of the rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments. When a control rod is declared inoperable, boration may be initiated to achieve the existence of 1% $\Delta k/k$ hot shutdown margin.

The power-imbalance envelope defined in Figures 3.5.2-3A1 and -3A2, 3.5.2-3B1, -3B2 and -3B3, 3.5.2-3C1, -3C2 and -3C3 is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification power spike factors (Units 1 and 2 only)
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping bank)

**Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

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.

Operating restrictions are included in Technical Specification 3.5.2.6 to prevent excessive power peaking by transient xenon. For Unit 1, a 5% peaking increase is applied to calculated peaks at equilibrium conditions for powers above the power level cutoff. For Units 2 and 3, an 8% peaking increase is applied. These values conservatively bound the peaking effects of transient xenon once the applicable requirement of 3.5.2.6 has been satisfied.

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)

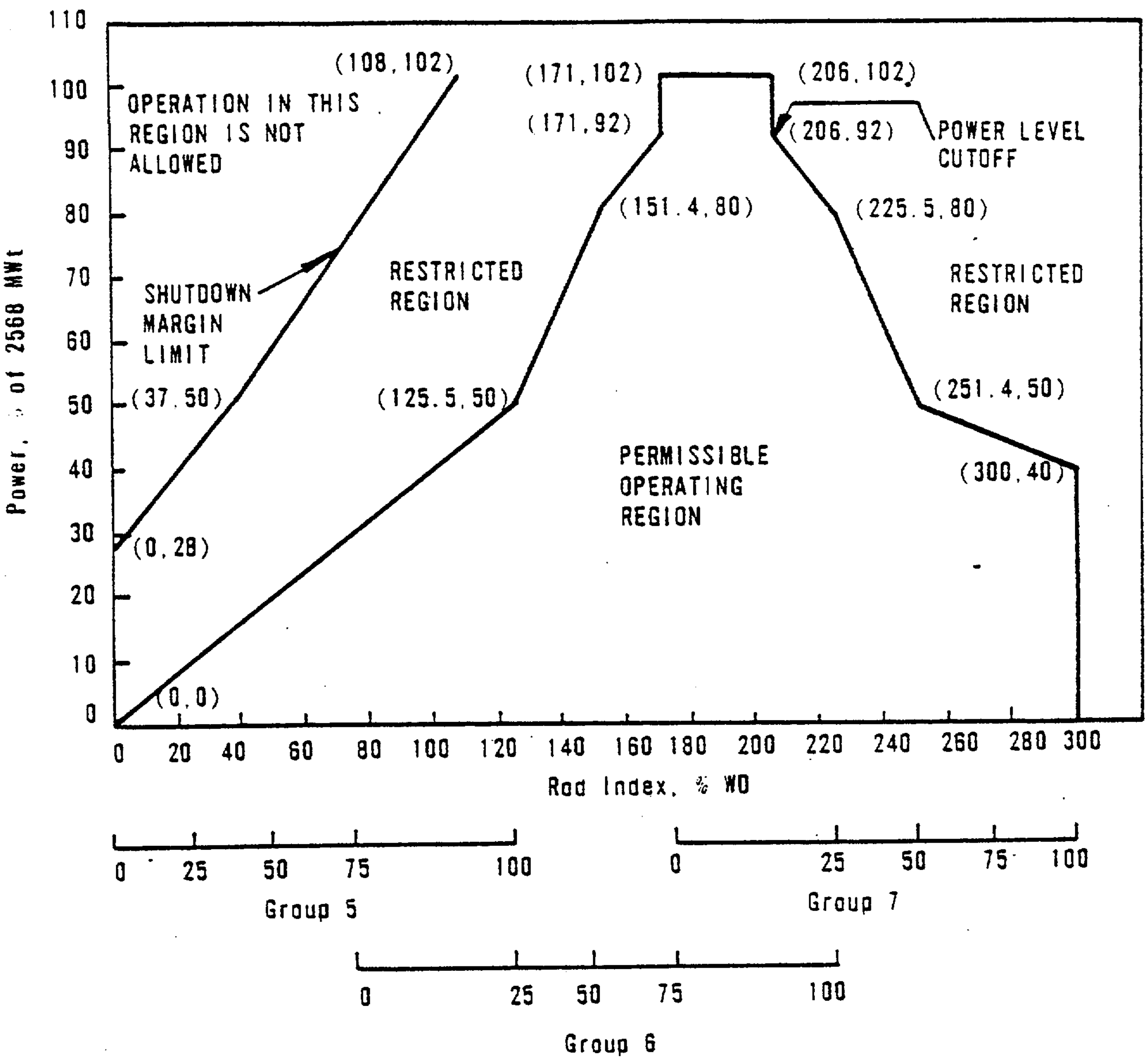
BAW-1400 (UNIT 3)

[5]Oconee 1, Cycle 4 - Reload Report - BAW 1447, March 1977, Section 7.11

TABLE 3.5-1

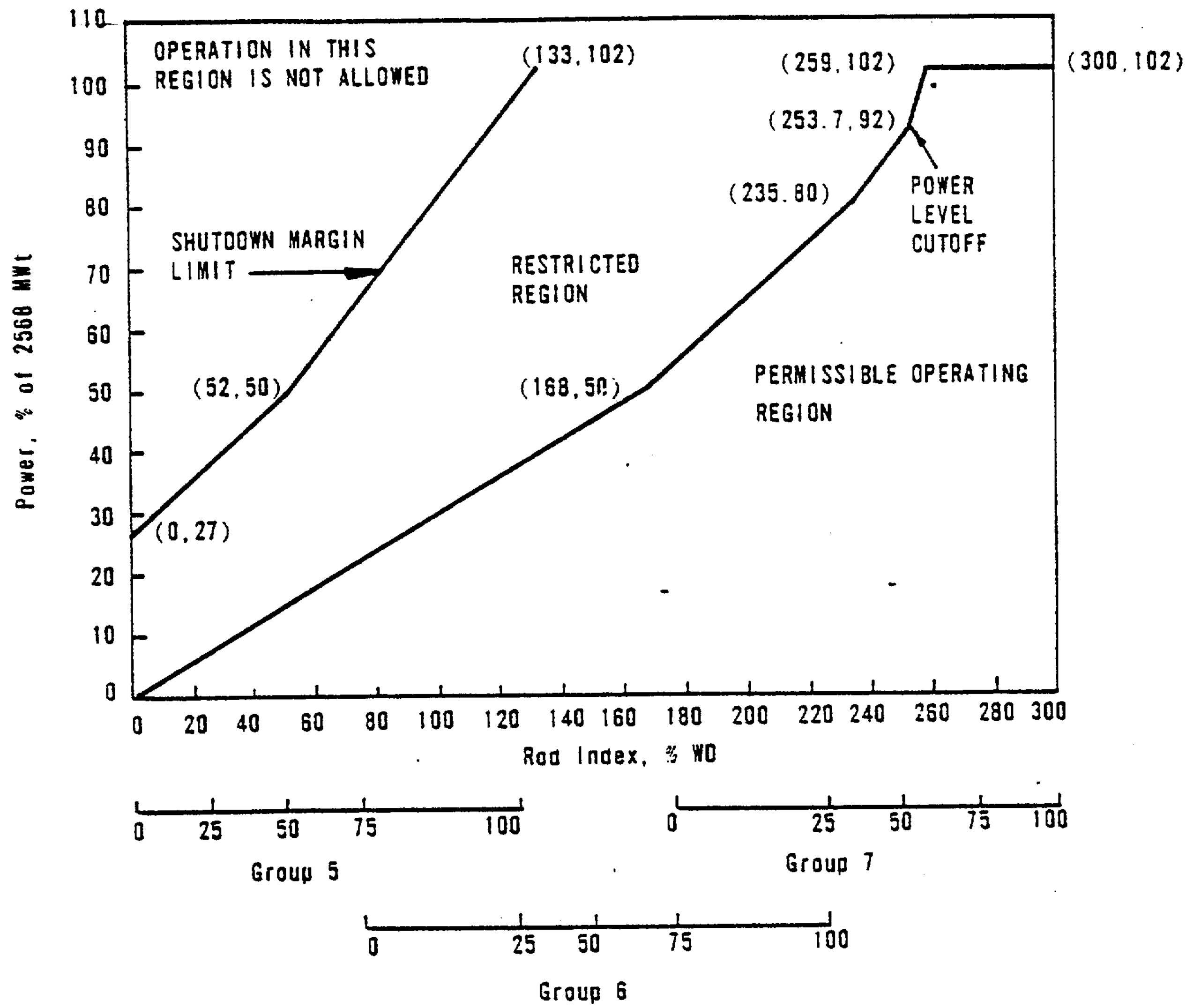
Quadrant Power Tilt Limits

	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



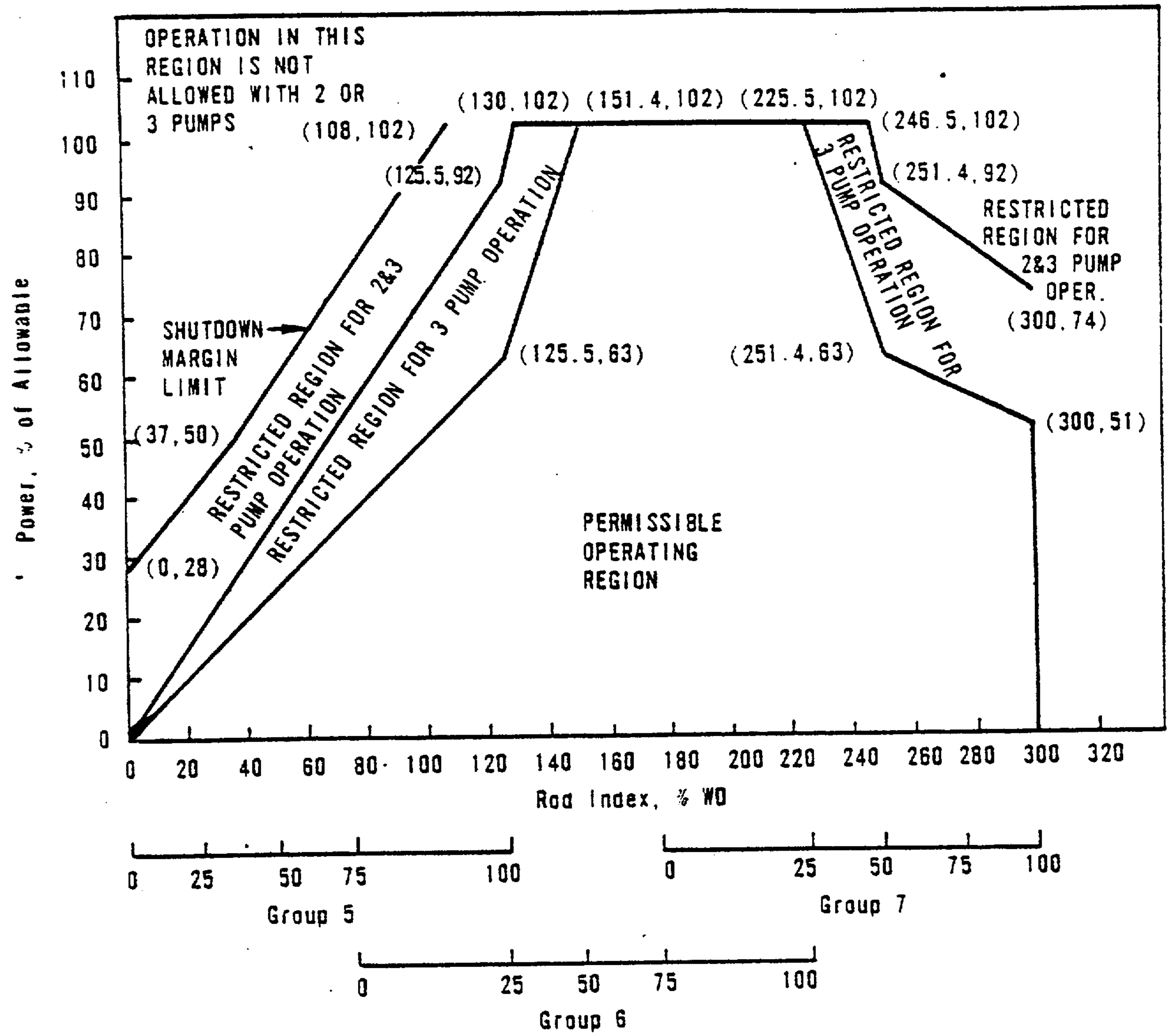
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 250 ± EFPD
OCONEE 2

OCONEE NUCLEAR STATION
Figure 3.5.2-1B1



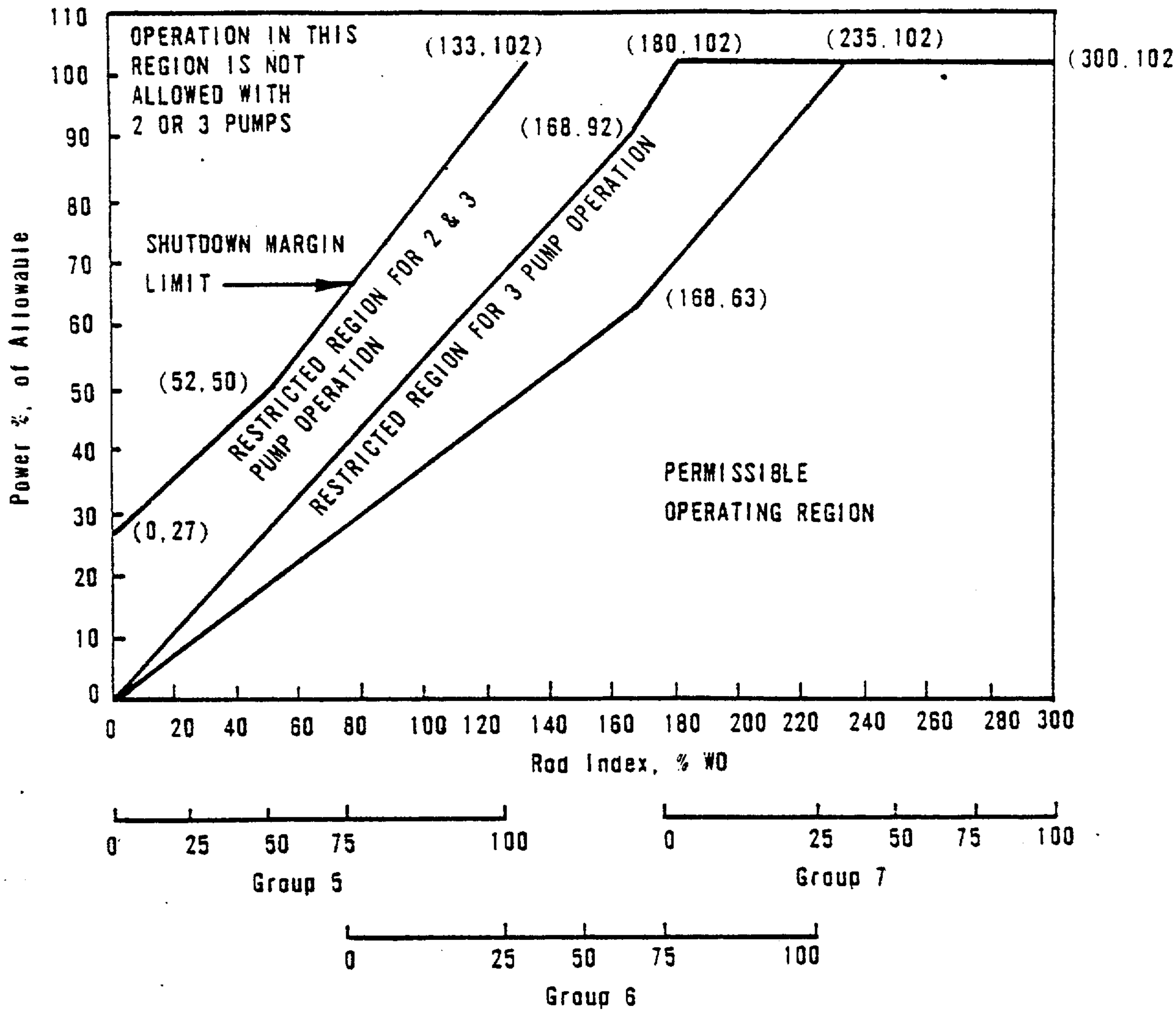
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 250 ± EFPD
OCONEE 2
OCONEE NUCLEAR STATION
Figure 3.5.2-1B2

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



ROD POSITION LIMITS
FOR TWO AND THREE PUMP OPERATION
FROM 0 TO 250 ± 10 EFPD
OCONEE 2

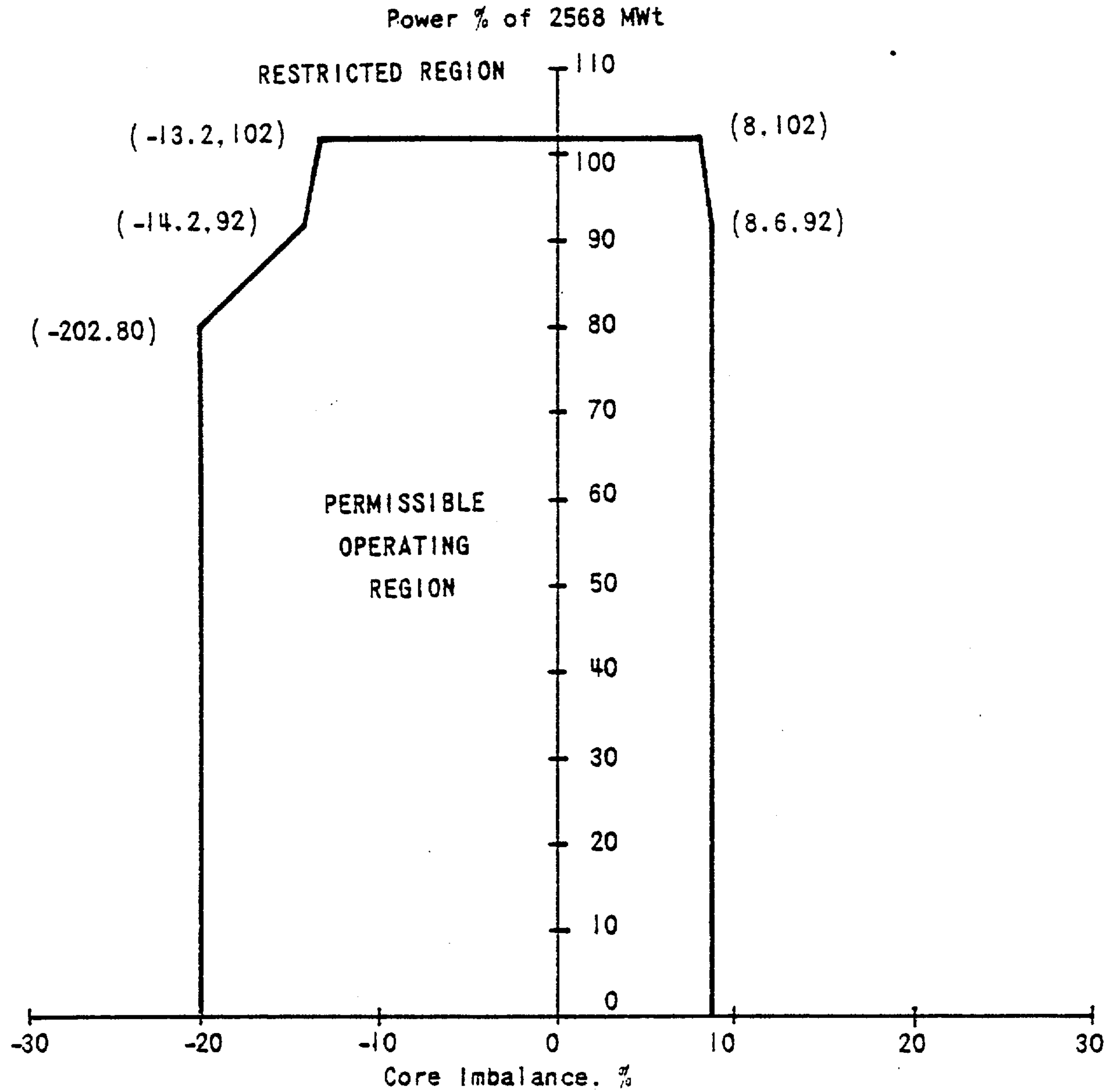
OCONEE NUCLEAR STATION
Figure 3.5.2-2B1



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

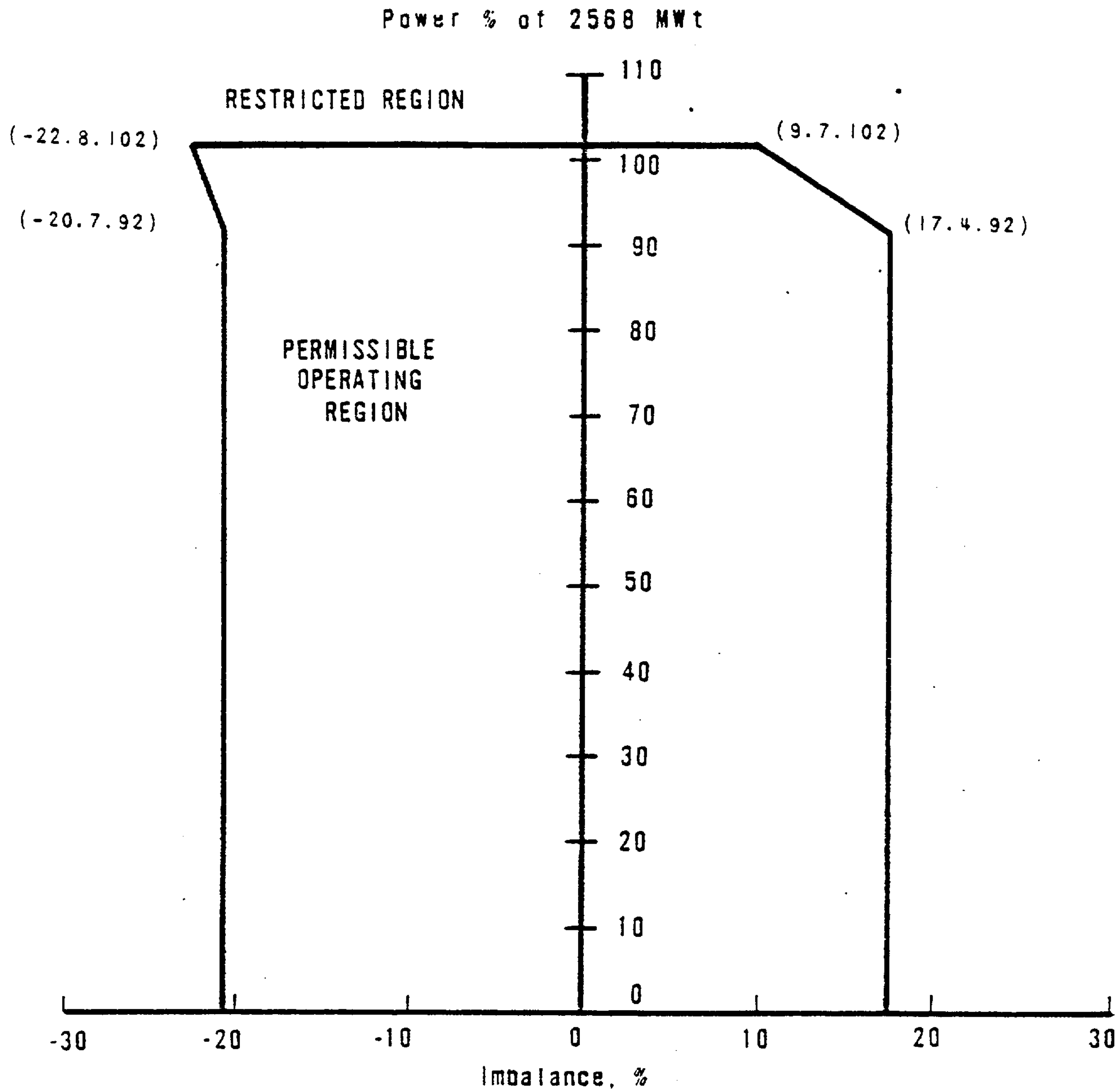
OCONEE NUCLEAR STATION
Figure 3.5.2-2B2

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



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 0 TO 250 ± 10 EFPD
OCONEE 2

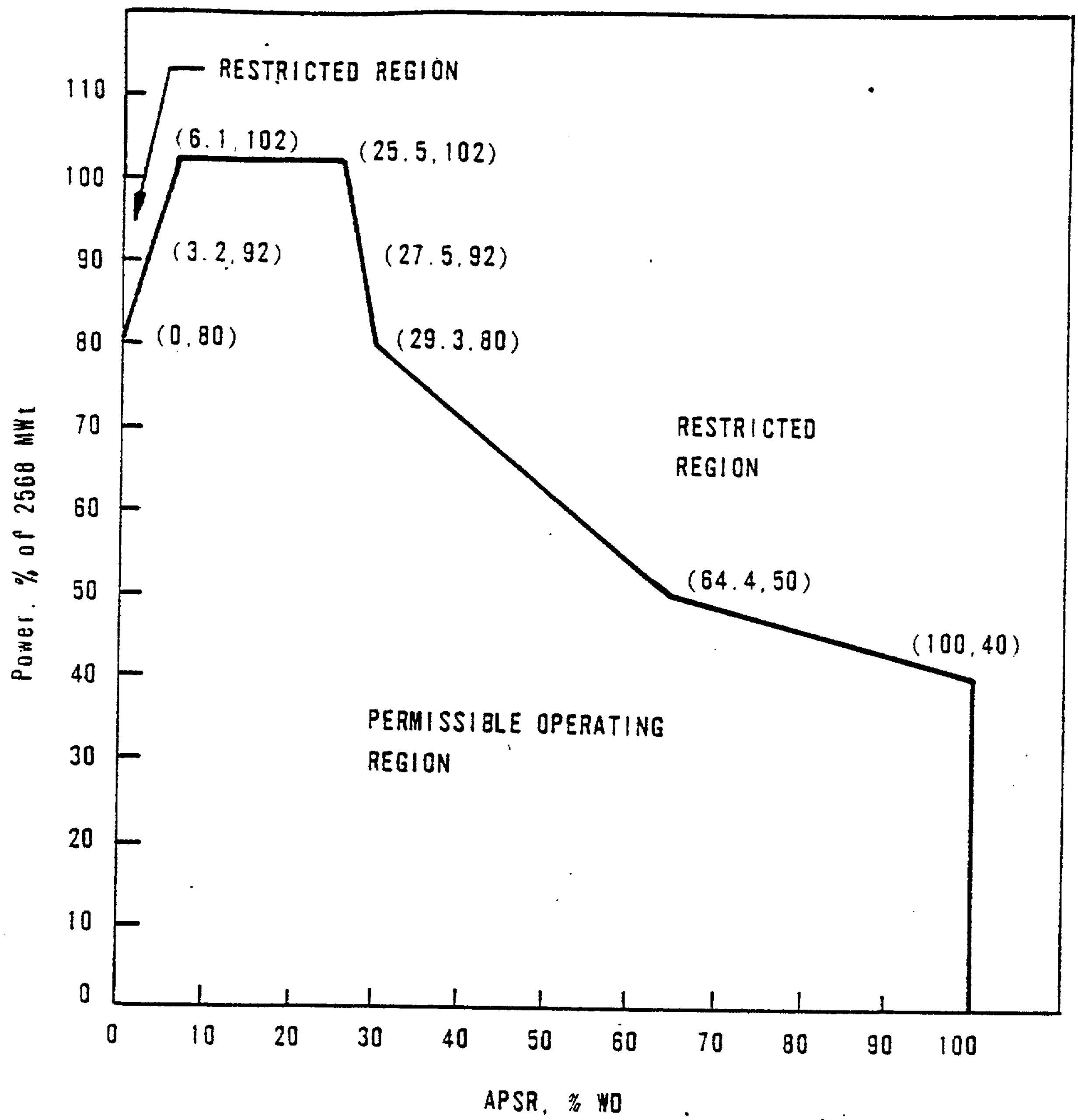
OCONEE NUCLEAR STATION
Figure 3.5.2-3B1



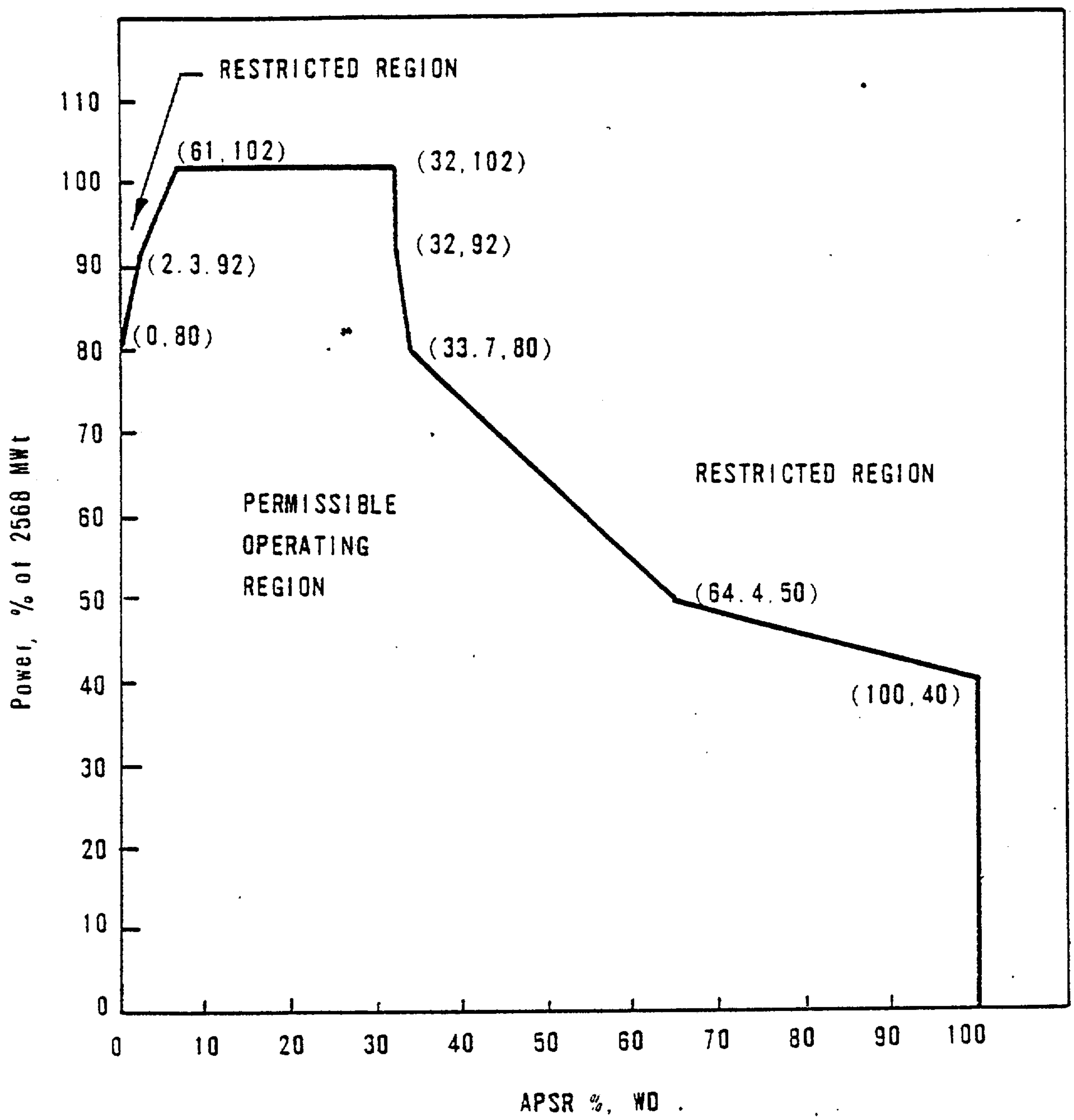
OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 250 ± 10 EFPD
OCONEE 2

OCONEE NUCLEAR STATION
Figure 3.5.2-3B2

Figure 3.5.2-3B3
Deleted During Oconee Unit 2, Cycle 4 Operation



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



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

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

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rod Movement (1)	Movement of Each Rod	Monthly
2.	Pressurizer Safety Valves	Setpoint	50% Annually
3.	Main Steam Safety Valves	Setpoint	25% Annually
4.	Refueling System Interlocks	Functional	Prior to Refueling
5.	Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly
6.	Reactor Coolant System Leakage (2)	Evaluate	Daily
7.	Condenser Cooling Water System Gravity Flow Test	Functional	Annually
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9.	Spent Fuel Cooling System	Functional	Prior to Refueling
10.	High Pressure and Low Pressure Injection System (3)	Vent Pump Casings	Monthly and Prior to Testing

(1) Applicable only when the reactor is critical

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
Duke Power Company	)	DOCKET NO. 50-270
Oconee Nuclear Station Unit No. 2	)	

EXEMPTION

I.

Duke Power Company (the licensee) is the holder of Facility Operating License No. DPR-47 which authorizes the operation of the nuclear power reactor known as Oconee Nuclear Station, Unit No. 2 (the facility), at steady reactor power levels not in excess of 2568 megawatts thermal (rated power). The facility consists of a Babcock & Wilcox (B&W) designed pressurized water reactor (PWR) located at the licensee's site in Oconee County, South Carolina.

II.

In accordance with the requirements of the Commission's Emergency Core Cooling System (ECCS) Acceptance Criteria, 10 CFR 50.46, the licensee submitted on July 9, 1975 an ECCS evaluation for the facility. The ECCS performance submitted by the licensee was based upon an ECCS Evaluation Model developed by B&W, the designer of the Nuclear Steam Supply System for this facility. The B&W ECCS Evaluation Model had been previously found to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50.46, and Appendix K. The evaluation indicated that with the limits set forth in

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the facility's Technical Specifications, the ECCS cooling performance for the facility would conform with the criteria contained in 10 CFR 50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

On April 12, 1978, B&W informed the NRC that it had determined that in the event of a small break Loss of Coolant Accident (LOCA) on the discharge side of a reactor coolant pump, high pressure injection (HPI) flow to the core could be reduced somewhat. Subsequent calculations indicated that in such a case the calculated peak clad temperature might exceed 2200°F.

Previous small break analyses for B&W 177 fuel assembly (FA) lowered loop plants had identified the limiting small break to be in the suction line of the reactor coolant pump. Recent analyses have shown that the discharge line break is more limiting than the suction line break.

The Oconee Nuclear Station Unit No. 2 has an ECCS configuration which consists of two HPI trains which are supplied by three HPI pumps. Each train injects into two of the four reactor coolant system (RCS) cold legs on the discharge side of the RCS pump. The two parallel HPI trains are connected but are kept isolated by manual valves (known as the cross-over valves) that are normally closed.

Duke Power has proposed by letter dated April 21, 1978, to maintain all three pumps in an operable status. The Oconee emergency power system is designed with sufficient capacity for this mode of operation. Upon receiving a safety injection signal the HPI pumps are started and valves in the injection lines are opened. Assuming loss of offsite power and the worst single failure (the HPI pump C or the HPI valve HP26), two HPI pumps would still be available and only one of the two injection valves would fail to open.

If a small break is postulated to occur in the RCS piping between the RCS pump discharge and the reactor vessel, the high pressure injection flow injected into this line (about 50% of the output of two high pressure pumps) could flow out the break. Therefore, for the worst combination of break location and single failure, 50% of the flow rate of two high pressure ECCS pumps would contribute to maintaining the coolant inventory in the reactor vessel. This situation had not been previously analyzed and B&W had indicated that the limits specified in 10 CFR 50.46 may be exceeded.

B&W has stated that they have analyzed a spectrum of small breaks in the pump discharge line and have determined that to meet the limits of 10 CFR 50.46(b), operator action is required to open the two manual operated crossover valves and to manually align the motor driven isolation valve which had failed to open. This would allow the flow from the two HPI pumps to feed all four reactor coolant legs. B&W has assumed that 30% of the flow would be lost through the break and 70% would enter the core. The licensee has committed to provide for the necessary operator actions within the required time frame. That is, in the event of a small break and a limiting single failure, manual action will be taken to begin opening these valves within five minutes and have them fully opened and an adequate flow split obtained within the following 10 minutes. The analyses performed by B&W assumed that the flow split was established at 650 seconds by operator action. We conclude that the analyses are a reasonable approximation of the operator action that actually will be taken, provided specific procedures are prepared and followed to assure such action.

B&W has prepared a summary entitled "Analysis of Small Breaks in the Reactor Coolant Pump Discharge Piping for the B&W Lowered Loop 177 FA Plants," April 24, 1978 (the B&W Summary), which describes the methods used and the results obtained in the above analysis. The analysis models operator action by assuming a step increase in flow to the reactor vessel (with balanced flow in the three intact loops) ten minutes after the LOCA reactor protection system trip signal occurs.

On April 26, 1978, the Commission issued an Order for Modification of License which amended the license for Oconee Unit 2 requiring (1) submission of a reevaluation of the emergency core cooling system calculated in accordance with the B&W Evaluation Model for operation with operating procedures described in the licensee's letter of April 21, 1978 and (2) operation in accordance with the procedures described in the licensee's letter of April 21, 1978.

By letter dated May 16, 1978, the licensee submitted a copy of the B&W Summary for our review. In their submittal the licensee stated that the analysis indicates that the ECCS cooling performance calculated in accordance with the B&W Evaluation Model for operation of Oconee units at the rated core thermal power of 2568 Mwt with operating procedures described in their letter of April 21, 1978, is wholly in conformance with the provisions of 10 CFR 50.46. We have reviewed the B&W Summary and find that the methods of analysis meet the requirements of 10 CFR Part 50.46.

By letter dated April 20, 1978 and as supplemented on April 27, 1978, the licensee submitted proposed Technical Specifications to implement the operating procedures and maintenance of all three HPI pumps in an operable status as described in the licensee's April 21, 1978 letter. We have issued these Technical Specifications in a license amendment dated October 23, 1978.

In the licensee's submittal of June 8, 1978, it was stated that to meet the limits of 10 CFR 50.46, operator action at the valve locations is required to open High Pressure Injection (HPI) Pump B-C discharge header cross over valves (HP-116 and HP-117) and the HPI injection line A engineering safeguards valve (HP-26) within 10 minutes.

Reliance on local operation of valves this soon after the onset of a loss-of-coolant accident is not desirable on a permanent basis. The licensee has requested an exemption from the requirements of 10 CFR

50.46 by letter dated September 18, 1978, for operation at Oconee 2 during Cycle 4 until such time as a permanent solution to this problem can be implemented.

The original concern derived from an unexpected but nevertheless inadequate assessment of a spectrum of breaks. This deviation from 10 CFR 50.46 has been ameliorated on a temporary basis by the actions discussed herein. However, combined reliance on prompt operator action to perform the required steps to assure plant safety over a period of years into the future is undesirable and will be replaced as promptly as possible by returning the system to simple control room actuation. To this extent, the original defect still remains until the modifications are made to eliminate the reliance on prompt operator actions.

We have reviewed the effects of changes made to the facility during the current refueling outage and have concluded that operation of Oconee Unit 2 at power levels of up to 2568 Mwt and in accordance with the Technical Specifications will assure that the ECCS system will conform to the performance criteria of 10 CFR 50.46. Accordingly, until modifications are completed to achieve full compliance with 10 CFR 50.46, operation of the facility at power levels up to 2568 Mwt with appropriate operating procedures will not endanger life or property or the common defense and security.

By letter dated July 14, 1978, the licensee submitted a proposed modification to the HPI system to eliminate the need for operator action outside the control room. Based on our review and Safety Evaluation, dated December 13, 1978, of the licensee's July 14 submittal we concluded that upon installation of the modification and upon completion of testing to verify the required flow split, the emergency core cooling system will fully conform to the requirements of 10 CFR 50.46.
The Evaluation provides a description of the modification.
While Oconee Unit No. 2 does not comply with our requirements for ECCS, appropriate actions, as previously described, have been taken to mitigate the consequences of any accidents at this plant. The Technical Specifications will provide protection against the subject small break LOCA and will bring plant operation wholly in conformance with 10 CFR 50.46. These Technical Specifications will be in force only for the brief interval of time until the proposed modifications of the ECCS are completed. The public interest is served in that by issuing this exemption for Unit No. 2 a significant power reduction with no concommitant increase in safety is avoided. Such a power reduction could affect system reliability, cause unemployment and increase consumer power costs in the area.

III.

Copies of the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, Washington, D.C. 20555, and are being placed in the Commission's local public document room at the Oconee County Library, 201 South Spring, Walhalla, South Carolina.

(1) The application for exemption dated September 18, 1978, and

(2) This Exemption in the matter of Duke Power Company, Oconee Nuclear Station, Unit No. 2.

IV.

WHEREFORE, in accordance with the Commission's regulations as set forth in 10 CFR 50.12, the licensee is hereby granted an exemption from the provisions of 10 CFR Part 50, Paragraph 50.46(a). With respect to Oconee Unit 2 this exemption supersedes the conditions of the Commission's Order for Modification of License dated April 26, 1978, and is conditioned as follows:

(1) The licensee has submitted the plans and schedules to modify the facility to eliminate reliance on prompt operator action described herein. Additional guidance in these areas has been provided by the NRC letter of September 26, 1978 to Duke Power Company. The staff approved the modification by letter dated December 13, 1978.

(2) The licensee shall complete such modifications prior to startup after the next scheduled refueling outage or during any scheduled outage of sufficient duration and occuring after six months from December 13, 1978 whichever occurs first.

(3) This exemption shall be terminated upon completion of the modifications in accordance with the conditions above.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr.

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 15th day of December 1978.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. DPR-38,

AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. DPR-47,

AND AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

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

1.0 Introduction

By letters dated September 18, 1978 and September 25, 1978 (References 1 and 2 respectively) Duke Power Company (DPC) has proposed changes to the Oconee Nuclear Station (ONS) Technical Specifications. Table 1 summarizes the proposed changes and indicates the applicability of each to changes to the three Oconee Units, ONS-1, ONS-2, or ONS-3.

Most of the proposed Technical Specification modifications are associated with the refueling of ONS-2 for Cycle 4 operation. The information submitted by DPC in connection with this refueling is presented in References 3 and 4 which describe the fuel system design, nuclear design, thermal-hydraulic design, accident analyses, and startup test program.

The refueling of ONS-2 for Cycle 4 will result in a core loading consisting of 56 fresh Mark B-4 assemblies, 108 previously burned Mark B-4 assemblies, nine previously burned Mark B-2, and four demonstration Mark C or Mark CR assemblies. In addition, the remaining (70) orifice rod assemblies will be removed from the core during the refueling outage. This will leave 106 vacant fuel positions which originally contained such orifice rod assemblies. The changes in the core loading and the removal of the orifice rod assemblies 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. For ONS-2, 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 4.

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Table 1. Proposed Technical Specification Changes For Oconee Nuclear Station

For Unit 2 Only

1. Modification to Core Protection Safety Limits (Figure 2.1-2B)*

2. Modification to Protective System Maximum Allowable Setpoints (Figure 2.3-2B)

3. Modifications to Rod Position Limits (Figures 3.5.2-1B1, 1B2, 2B1, and 2B2)

4. Modifications to Operational Power Imbalance Envelope (Figures 3.5.2-3B1 and 3B2)

5. Modifications to APSR Position Limits (Figures 3.5.2-4B1 and 4B2)

6. Reduction in $F_{\Delta H}$ from 1.78 to 1.71

7. Increase in the allowed steady state quadrant tilt to 5% and in the linear heat rate peaking increase associated with positive tilt to 7.50%.

For Units 2 and 3

8. Extension to Units 2 and 3 an allowance for operating above the power level cutoff associated with the rod position limits, provided the reactor has operated within 5% of the cutoff for more than two hours.

9. Adoption of an 8% peaking increase in linear heat rate associated with transient xenon.

Table 1. Proposed Technical Specification Changes For Oconee Nuclear Station (Contd)

For Units 1, 2, and 3

10. Increase in the volume of boric acid solution in the Boric Acid Storage Tank from 980 Ft^3 to 995 Ft^3.

11. Modifications to Control Rod Operability and Surveillance Requirements

*All figures are in Reference 3.

2.0 Evaluation of Modifications to ONS-2 Core Design

2.1 Fuel System Design

We have evaluated the implications of introducing the 56 fresh Mark B-4 fuel assemblies and the nine once-burned Mark B-2 fuel assemblies into the ONS-2 core and the subsequent operation at rated power for the intended 292 effective full power days.

Tables 4-1 and 4-2 of Reference 3 summarize the design characteristics of the Mark B-4, Mark B-2, Mark C and Mark CR fuel types. The fresh Mark B-4 assemblies are identical to the previously burned Mark B-4 fuel with regard to assembly mechanical design, fuel rod design and thermal design. The fuel designs of Mark B-4, Mark C and Mark CR fuel types have been evaluated for ONS-2 in association with earlier refuelings and found acceptable (References 5 and 6). The Mark B-2, which fuel has been analyzed in the ONS-2 Densification Report (Reference 10), was part of several earlier ONS core loadings.

2.1.1 Cladding Creep Collapse

Fuel rod cladding creep collapse analyses have been performed for the most limiting (i.e., most highly exposed) Mark B and Mark C assemblies to be included for Cycle 4. The analyses were performed according to the conservative methods and assumptions described in References 7 and 8 and approved by the NRC staff in Reference 9. These analyses show that the time to rod cladding collapse will be in excess of 30,000 effective full power hours. Because no Mark B or Mark C assembly will reach a total exposure as high as 30,000 EFPH during Cycle 4 (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

With regard to cladding stress and strain, the Mark B-2 fuel is most limiting for Cycle 4 because of its low prepressurization and density. For this fuel, 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 during Cycle 4 (Reference 10). In Reference 7, the anticipated cladding strain for Mark B-2 fuel 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 4. We previously accepted these conclusions regarding cladding stress and strain for ONS-2 Cycle 3 (Reference 6) and we conclude that they are valid for Cycle 4 also.

2.1.3 Fuel Thermal Design

The thermal linear heat rate (LHR) limits have been established for the Cycle 4 fuel using the TAFY code (Reference 11) and assumed fuel densification to 96.5% of theoretical density. These limits are stated in Table 4-2 of Reference 3. The thermal LHR limits which ensure that fuel center melting does not occur are less restrictive than the LOCA LHR limits. Because the LOCA LHR limits will be met by operating within the limiting conditions for operations contained in the ONS-2 Technical Specifications, the thermal LHR limits will also be met.

We conclude that the indicated thermal LHR limits are acceptable for preventing center melting of the Cycle 4 fuel and that the limits will not be exceeded.

2.2 Nuclear Design

Figure 3-1 of Reference 3 indicates the core loading arrangement for ONS-2 Cycle 4; the initial enrichments and burnup distributions are given in Figure 3-2. Most of the fresh Mark B-4 assemblies will be loaded into locations on the edge of the core and will be below fuel thermal limits. Similarly, the Mark C and Mark CR demonstration assemblies will be in non-limiting locations.

Reactivity control and power distribution control will be maintained by control rods, axial power shaping rods and boron shim. The rod locations are given in Figure 3-3 of Reference 3.

The projected Cycle 4 length is 292 effective full power days with a cycle burnup of 9138 MWd/MTU.

Cycle 4 nuclear parameters including critical boron concentrations, control rod worths, Doppler coefficients, moderator coefficients, xenon worth and effective delayed neutron fractions have been calculated using the approved PDQ07 code (Reference 12). These are presented in Table 5-1 of Reference 3 and compared to the Cycle 3 values.

Shutdown margins have been calculated for beginning of cycle (BOC) and end of cycle (EOC) (Table 5-2 of Reference 3). The calculated minimum shutdown margin during Cycle 4 is 1.45% $\Delta k/k$ which is larger than the required value of 1% $\Delta k/k$ by an adequate margin.

We conclude that the Cycle 4 nuclear design does not differ in a significant way from earlier cycles, that the nuclear parameters of Cycle 4 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-2 Cycle 4 is, therefore, acceptable.

2.3 Thermal Hydraulic Design

The thermal-hydraulic design conditions for ONS-2 Cycle 4 are included in Table 6-1 of Reference 3. Only the reference design radial-local power peaking factor and anticipated minimum departure from nucleate boiling ratio (DNBR) differ from the Cycle 3 values. The first of these differences is discussed below. The second is acceptable in that the minimum departure from nucleate boiling ratio, with densification penalty, increases from 1.91 in Cycle 3 to 1.98 for Cycle 4; 1.30 is the safety limit, thus the current Cycle 4 in this regard represents a slight increase in margin to the safety limit.

The effect of the demonstration Mark C and Mark CR assemblies on the ONS-2 thermal hydraulic design have been evaluated for earlier cycles (References 5 and 6). The continued use of the demonstration assemblies does not involve any physical effect not previously considered and is acceptable.

2.3.1 Removal of Orifice Rod Assemblies

The most significant difference between the thermal hydraulic design for Cycle 4 and that for Cycle 3 is the removal of the 70 orifice rod assemblies (ORAs). This will leave a total of 106 vacant fuel assemblies and will result in an increase in bypass flow from 8.34% for Cycle 3 to 10.4% for Cycle 4. The increased bypass flow also involves a decreased flow to fuel assemblies, and DPC has re-evaluated the effect of this modification on the reactor core DNBR safety limit. The re-evaluation indicated that a decrease in the reference design radial-local peaking factor ($F_{\Delta H}$) from 1.78 to 1.71 compensates for the larger bypass flow so that no change in the DNBR safety limit will be necessary. The DNBR safety limit was derived using the BAW-2 critical heat flux correlation (Reference 14). Based on the sensitivity of the heat flux correlations, such as BAW-2, to small changes in flow, we have concluded that the proposed reduction in $F_{\Delta H}$ to 1.71 is adequate to offset the increased bypass flow.

2.3.2 Effect of Rod Bow on Thermal Design

The effect of fuel rod bow has been reviewed generically in Reference 13. Based on the rod bow model approved by the NRC staff, DPC has applied a rod bow DNBR penalty of 11.2% to all analyses that define plant operating limits and to design transients (Reference 3).

The 11.2% penalty which has been applied includes a 1% contribution associated with pitch reduction due to fabrication tolerances and initial rod bow, and a 10.2% contribution from burnup dependent bowing. The 11.2% penalty is valid for a maximum burnup of 33,000 MWd/MTU and, therefore, bounds the burnup expected for Cycle 4.

Based on the use of an approved model and a bounding assumed burnup, we conclude that DPC has adequately taken fuel rod bowing into account for the thermal design of ONS-2 Cycle 4.

3.0 Evaluation of Accidents and Transients

The refueling of ONS-2 for Cycle 4 will not involve a change in the DNBR safety limit (Section 2.3 of this Report). Plant operating limits, as proposed in Reference 2, have been established to compensate for the effect of fuel rod bowing on DNBR.

The two pump coastdown, which is the limiting event with regard to reduction in DNBR, has been analyzed from an initial power level of 102% with a flux/flow trip set-point of 1.055. The combined reduction in DNBR due to the transient and fuel rod bowing would not result in a DNBR below the safety limit value of 1.30 Other transients are discussed below.

As discussed in Section 2.2 of this Report, the nuclear parameters, which comprise a portion of the input to the accident and transient analyses, have been evaluated using acceptable methods. Of the transients and accidents considered in the ONS-2 FSAR (Reference 16), the loss of electric power, steam generator tube failure, fuel handling accident, waste gas tank rupture, maximum hypothetical accident, and LOCA do not depend on nuclear parameters. Rod withdrawal accidents, the cold water accident, stuck or dropped rod accidents, steamline failure, and the rod ejection accident do depend on nuclear parameters. We conclude, based on Tables 6-1 and 7-1 of Reference 3, that the Cycle 4 nuclear parameters are bounded by values assumed for accident analyses in the FSAR (Reference 16) and the ONS-2 Densification Report (Reference 10).

The applicable LOCA analyses for ONS-2 have been presented in Reference 17 which has been accepted by the NRC staff for generic application to B&W plants of the ONS-2 class (177-FA Lowered Loop Plants). The fuel densification report (Reference 10) describes the effect of densification on LOCA analyses and the use of the TAFY code (Reference 11) to calculate fuel rod internal pressure and pellet volumetric average temperature. The latter parameters, which are part of the LOCA input, are also affected by enhanced fission gas release, but the original TAFY calculations did not include the effect. Calculations using the B&W code, TACO, (Reference 21) have shown that the internal pressures and average temperatures calculated using TAFY adequately bound the effects of enhanced fission gas release for up to 42,000 MWd/MTU fuel rod burnup, a higher burnup than will be attained during Cycle 4 operation.

Technical Specification proposals associated with LOCA LHR limits were presented in Reference 1. These proposed limits include a statistical combination of nuclear uncertainty factor, engineering hot channel factor and rod bow peaking penalty amounting to a 9% net peaking penalty (Reference 15). B&W has demonstrated that power spikes caused by densification need not be considered in LOCA or DNB analyses. These tests and analyses show that for LOCA the radiant heat transfer to the cool cladding surrounding the gap, where the peaking occurs, more than offsets the heat generated by the power spike. For DNB, which is a function of critical heat flux, B&W has shown that heat flux power spikes have a negligibly small effect on critical heat flux thus the effect on DNB is negligible. The staff has accepted these demonstrations and analyses in Reference 22.

We conclude that the refueling of ONS-2 for Cycle 4 will not result in kinetics parameters outside the bounds assumed for the FSAR analysis, and that no change in the DNBR safety limit is required. Furthermore, the effects of fuel row bowing, fuel densification, and enhanced fission gas release on safety limits and on all transients and accidents, including LOCA, have adequately been taken into account.

Fuel misloadings for Cycle 4 which could result in departure from nucleate boiling (DNB) will be detected during the physics startup testing to be performed at the BOC. These tests have been described in References 3 and 4 and evaluated in Section 4.0 for this report.

Based on these conclusions, and the fact that the dose calculations of the FSAR assumed maximum peakings and burnups which bound all reloads, we further conclude that the consequences of transients or accidents during Cycle 4 will be no greater than previously evaluated. There will be no increase in the probability of occurrence of any accident or transient, and no new type of accident or transient will be introduced as a result of the refueling. We, therefore, accept the transient and accident analyses presented for ONS-2 Cycle 4.

4.0 Startup Tests

Startup tests have been proposed by DPC to provide assurance that ONS-2 has been loaded as intended. The tests are described in References 3 and 4 and are consistent with the startup tests performed in association with recent B&W reloads. We have reviewed the tests and consider them acceptable.

5.0 Evaluation of Technical Specification Changes

Proposed modifications to the ONS-1, ONS-2 and ONS-3 Technical Specifications are listed in Table 1.

The changes indicated in Items 1 through 5 of Table 1 are based on FLAME code calculations (Reference 18) applied according to the descriptions in References 19 and 20. For these calculations, the statistical combination of nuclear uncertainty, engineering uncertainty and rod bow peaking, as approved in Reference 15, was applied to the linear heat rate peaking. Change No. 6 is discussed and justified in Section 2.3.1 of this Safety Evaluation.

The relation between linear heat rate peaking increase and quadrant tilt implied in Item 7 of Table 1 is based on information in References 19 and 23. Reference 23 was provided in connection with the review of the Unit 1 quadrant tilt technical specification. We believe that the information in References 19 and 21, which shows that the quadrant tilt linear heat rate peaking increase is related to the quadrant tilt by a multiplication factor of 1.495, includes a sufficiently broad data base to apply to ONS-2. The licensee has proposed to increase the current quadrant tilt Technical Specification limit to 5% from 3.4%. The quadrant tilt Technical Specification in conjunction with the control rod insertion limit and power imbalance limit Technical Specifications ensure that plant limiting conditions for operation are not exceeded. These conditions ensure that limiting values of linear heat generation rate and peak enthalpy rise assumed in the safety analysis are not exceeded. These limiting values are not altered by the proposed Technical Specification change. The margin to safety and operating limits have not been altered. Hence, Change No. 7 is acceptable. The increased tilt limit permits greater operating flexibility with no decrease in safety margin.

Based on our acceptance of the 1.075 peaking increase for quadrant tilt and the maximum allowed quadrant tilt of 5% just discussed, the acceptance of the 8% transient xenon peaking increase discussed below, the previous acceptance of other peaking factors, and the use of the approved FLAME code to derive the limits associated with Items 1 through 5, we conclude that these proposed Technical Specification changes are acceptable.

Items 8 and 9 in Table 1 are related to analyses of the design basis maximum xenon transient described in Reference 19 and performed using the FLAME code (Reference 18). Reactor power levels, except for physics tests, are not permitted by Technical Specification 3.5.2.6 to be increased above the power level cutoff curves of the Rod Position Limits of Figures 3.5.2-1B and 1B2 of the Technical Specifications, unless xenon reactivity transients and the associated change in power distribution during power operation is limited by restricting the nonequilibrium xenon. The Reference 19 calculations show that if the provisions of Technical Specification 3.5.2.6 (including modification 8 of Table 1) are met, the transient xenon peaking increase need be no greater than 8% to assure that linear heat rate limits are not exceeded. The transient xenon peaking factor of 1.08 was used in deriving the limits associated with Items 1 through 5.

Based on the use of the accepted design basis maximum xenon transient and the application of an accepted calculational method, we conclude that the modifications proposed in Items 8 and 9 are acceptable.

Modification 10 of Table 1 applies to ONS-1, 2, and 3. The increase in the volume of boric acid in the boric acid storage tank has been proposed to assure that an adequate cold shutdown capability will be maintained. The PDQ07 code (Reference 12) was used to evaluate the negative reactivity effects of the boric acid for this purpose. PDQ07 has been accepted by the NRC staff for calculations of this type and we consider it acceptable for the current application. We, therefore, conclude that modification 10 should be adopted.

Proposed modification 11 of Table 1 applies to ONS-1, 2 and 3. The control rod drive operability history, with one exception, has been favorable at the Oconee Station. The drive system has not experienced any binding or frictional problems nor has it failed to perform its intended trip (scram) function. An electrical component of the drive system, the stator coil, has failed in the past due to an electrical short in the coil. Stator failures have not prevented the affected rod from performing its required safety function, namely the trip function. A shorted stator makes it difficult to move a rod and occasionally an attempt to move such a rod causes it to drop into the core. Control rod drop events have been analyzed in the FSAR (Reference 16). They do not result in fuel damage. The stator is coupled to the rod only by a magnetic field. The licensee proposes to extend the periodic rod exercise interval from two weeks to one month, thus avoiding a situation where the rod must be exercised possibly causing the rod to drop into the core and at a time of possible high power demand from the electrical distribution system. In a previous NRC staff evaluation of this problem regarding Rod 6 of Group 4 in Oconee Unit No. 2, issued with the July 6, 1978 License Amendment,we stated, "...., we agree with the licensee's conclusion that the circuit fault (i.e., stator short) discovered in Rod 6 would not prevent the rod from performing its assigned safety function." In our letter transmitting the License Amendment, we noted that the request for that amendment could have been avoided if the licensee had previously adopted the Standard Technical Specifications of Babcock & Wilcox designed reactors. The requested change puts the test interval for control rod movement in parallel with the Standard Technical Specifications.

As the previous history of rod motion has been favorable, as discussed above, we find the change in surveillance of rod motion from two weeks to one month to be acceptable. The remainder of the

Technical Specification changes in this section are of an editorial nature and since they clarify the meaning of Section 3.5.2, we find the changed wording acceptable. The definition of shutdown margin, and the accompanying limiting condition of operation, are unaffected by the changes.

6.0 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.

7.0 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: December 15, 1978

References

1. Letter from William O. Parker, Jr., Duke Power Company (DPC), to Harold R. Denton, NRC, September 18, 1978.

2. Letter from W. O. Parker, Jr., DPC, to H. R. Denton, NRC, September 25, 1978.

3. Oconee Unit 2 Cycle 4 Reload Report, BAW 1491, August 1978.

4. Letter from W. O. Parker, Jr., DPC, to H. R. Denton, NRC, November 1, 1978.

5. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendments Nos. 27, 27 and 23 to Facilty License Nos. DPR-38, DPR-47, and DPR-55, Duke Power Company, Oconee Nuclear Station 2, June 30, 1976.

6. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendments Nos. 45, 45 and 42 to Facility License Nos. DPR-38, DPR-47, and DPR-55, Duke Power Company, Oconee Nuclear Station 2.

7. Oconee 2 Cycle 3 Reload Report, BAW-1452, Babcock & Wilcox, Lynchburg, Va., April 1977.

8. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084, Rev. 1, Babcock & Wilcox, Lynchburg, Va., November 1976.

9. Letter from A. Schwencer (NRC) to J. F. Mallary (B&W) dated January 29, 1975.

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

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

12. H. A. Hassan, et al., Babcock & Wilcox's Version of PDQ07 - User's Manual, BAW-10117, Babcock & Wilcox, Lynchburg, Va., June 1976.

13. Memo to D. B. Vassallo (NRC) from D. F. Ross (NRC), Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors, February 16, 1977.

14. Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May 1976.

15. Letter, S. A. Varga (NRC) to J. H. Taylor (B&W), "Comments on B&W's Submittal on Combination of Peaking Factors," May 13, 1977.

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

17. 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.

18. C. W. Mays, FLAME 3 - A Three-Dimensional Nodal Code for Calculating Core Reactivity and Power Distributions, BAW-10124, Babcock & Wilcox, Lynchburg, Va., May 1976.

19. Operational Parameters for B&W Rodded Plants, BAW-10078, September 1973.

20. Normal Operating Controls, BAW-10122, July 1978.

21. TACO - Fuel Pin Performance Analysis, BAW-10087.

22. Letter, S. A. Varga (NRC) to J. H. Taylor (B&W), "Update of BAW-10055 - Fuel Densification Report," December 5, 1977.

23. Letter from W. O. Parker, Jr., DPC, to E. G. Case, (NRC), March 16, 1978.

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. 66, 66 and 63 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. 2 at full rated power during Cycle 4 after core reload and removal of the orifice rod assemblies from the core. These amendments also revise the Technical Specifications for Units 1, 2 and 3 in regard to control rod operability.

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.

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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 September 18, 1978, as supplemented September 25, and November 1, 1978, (2) Amendments Nos. 66, 66 and 63 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 15th day of December 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Gerald B. Zwetzig

Gerald B. Zwetzig, Acting Chief
Operating Reactors Branch #4
Division of Operating Reactors