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September 19, 1988

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Posted
Amdt 170
to DPR-47

Docket Nos.: 50-269, 50-270
and 50-287

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:
SUBJECT: ISSUANCE OF AMENDMENT NOS. 170, 170, and 167 TO FACILITY OPERATING LICENSES DPR-38, DPR-47, and DPR-55 - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TACS 68351/68352/68353)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 170, , 170, and 167 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments consist of changes to the Station's Technical Specifications (TS) in response to your request dated May 16, 1988.

The amendments revise the TS to support operation of Unit 3, Cycle 11, at full rated power.

A copy of our Safety Evaluation is also enclosed in addition to a copy of the notice of issuance of the amendments which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

ORIGINAL SIGNED BY:

Helen N. Pastis, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II

Enclosures:
1. Amendment No. 170to DPR-38
2. Amendment No. 170to DPR-47
3. Amendment No. 167to DPR-55
4. Safety Evaluation
5. Federal Register Notice

cc w/enclosures:
See next page

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9/12/88	9/12/88	9/15/88

Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

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Honorable James M. Phinney
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated May 16, 1988, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachments 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. 170, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: September 19, 1988

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	HPastis:sw		DMatthews
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated May 16, 1988, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachments 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. 170, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: September 19, 1988

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	HPastis:sw		DMatthews
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated May 16, 1988, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachments 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. 167 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: September 19, 1988

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	HPastis:sw		DMatthews
9/12/88	9/14/88	9/14/88	9/15/88

ATTACHMENT TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Amended Page

2.1-2
3.2-1
3.2-2
3.3-3
3.3-6
3.5-24
3.5-25
3.5-26
3.8-3

The curve presented in Figure 2.1-1[3] represents the conditions at which the minimum allowable DNBR is predicted to occur for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based upon the design nuclear peaking factors which include the potential effects of fuel densification[4]:

$$F^N_{\Delta H} = 1.71$$

$$F^N_Z = 1.50$$

Since power peaking is not a directly measurable quantity, DNBR limited power peaks and fuel melt limited power peaks are separately correlated to measurable reactor power and power imbalance. The reactor power imbalance limits, Figure 2.1-2[5], define the values of reactor power as a function of axial imbalance that correspond to the more restrictive of two thermal limits - MDNBR equal to the DNBR limit or the linear heat rate equal to the centerline fuel melt limit.

The core protection safety limits are based on an RCS flow less than or equal to 385,440 gpm (4 pump operation). Three and two pump operation are analyzed assuming 74.7 percent and 49.0 percent of four pump flow, respectively. The maximum thermal power for three pump operation is 88.07 percent (Figure 2.1-2) due to a power level trip produced by the flux/flow ratio (74.7 percent flow x 1.07 = 79.92 percent power = 88.07 percent power adding the maximum calibration and instrument error). The maximum thermal power for 2 pump operation, 60.63 percent, is produced in a similar manner.

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 1100 ft^3 of 11,000 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 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 1100 ft^3 of 11,000 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1950 ppm boric acid in the borated water storage tank (3) will satisfy the requirements. The volume requirements include a 10% margin and, in addition, allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. Using only one 10 gpm boric acid pump taking suction from the concentrated boric acid storage tank would require approximately 12.7 hours to inject the required boron. An alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. The required boric acid can be injected in less than six hours using only one of the makeup pumps.

The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above the crystallization temperature for the concentration present. The boric acid concentration of 11,000 ppm in the concentrated boric acid storage tank corresponds to a crystallization temperature of 88°F and therefore a temperature requirement of 98°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, Sections 9.3.1, and 9.3.2
(2) FSAR, Figure 6.0.2
(3) Technical Specification 3.3

b. The BWST shall contain a minimum level of 46 feet of water having a minimum concentration of 1950 ppm boron at a minimum temperature of 50°F. The manual valve, LP-28, on the discharge line shall be locked open. If these requirements are not met, the BWST shall be considered unavailable and action initiated in accordance with Specification 3.2.

3.3.5 Reactor Building Cooling (RBC) System

a. Prior to initiating maintenance on any component of the RBC system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical:

(1) Two independent RBC trains, each comprised of an RBC fan, associated cooling unit, and associated ESF valves shall be operable.

(2) Tests or maintenance shall be allowed on any component of the RBC system provided one train of the RBC and one train of the RBS are operable. If the RBC system is not restored to meet the requirements of Specification 3.3.5b(1) above within 24 hours, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

c. When the reactor is critical:

(1) In addition to the requirements of Specification 3.3.5.b(1) above, the remaining RBC fan, associated cooling unit, and associated ESF valves shall be operable.

(2) Tests or maintenance shall be allowed on one RBC train under either of the following conditions:

(a) One RBC train may be out of service for 24 hours.

(b) One RBC train may be out of service for 7 days provided both RBS trains are operable.

(c) If the inoperable RBC train is not restored to meet the requirements of Specification 3.3.5.c(1) within the time permitted by Specification 3.3.5.c(2) (a) or (b), the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.5.c(1) are not met within an additional 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70°F without any control rods in the core. The minimum value specified in the tanks is 1950 ppm boron.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the Reactor Building design pressure will not be exceeded with one spray and two coolers operable. (4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor Building spray systems are operable for seven days or one Reactor Building spray system provided all three Reactor Building cooling units are operable.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Unit 1, 2, and 3. One low pressure service water pump per unit is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

Prior to initiating maintenance on any of the components, the redundant component(s) shall be tested to assure operability. Operability shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. The 24 hour period prior to removal is adequate to permit efficient scheduling of manpower and equipment testing while ensuring that the testing is performed directly prior to removal. The basis of acceptability is the low likelihood of failure within a clearly defined 48 hours following redundant component testing.

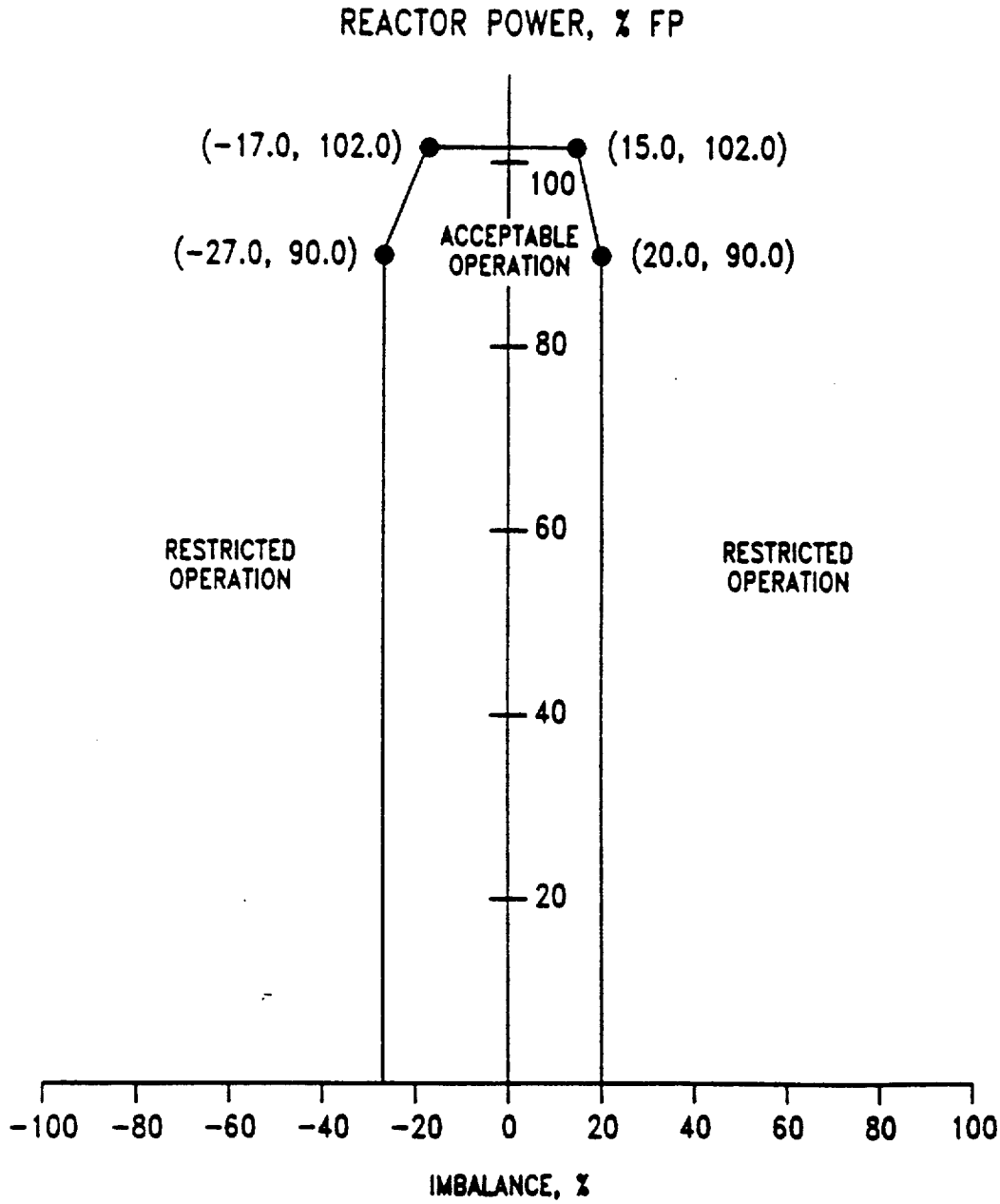
REFERENCES

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

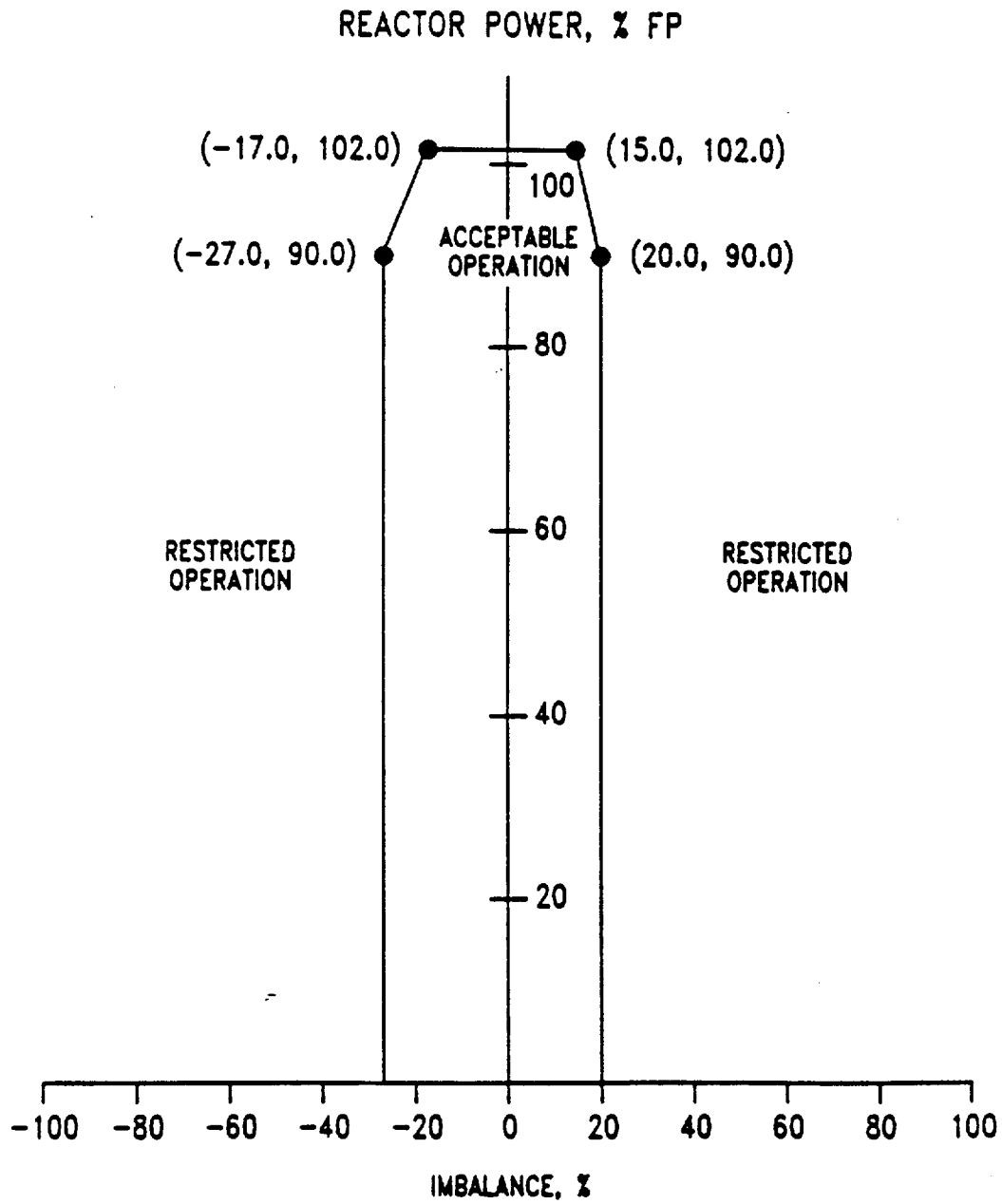
(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) FSAR, Section 9.3.3.2

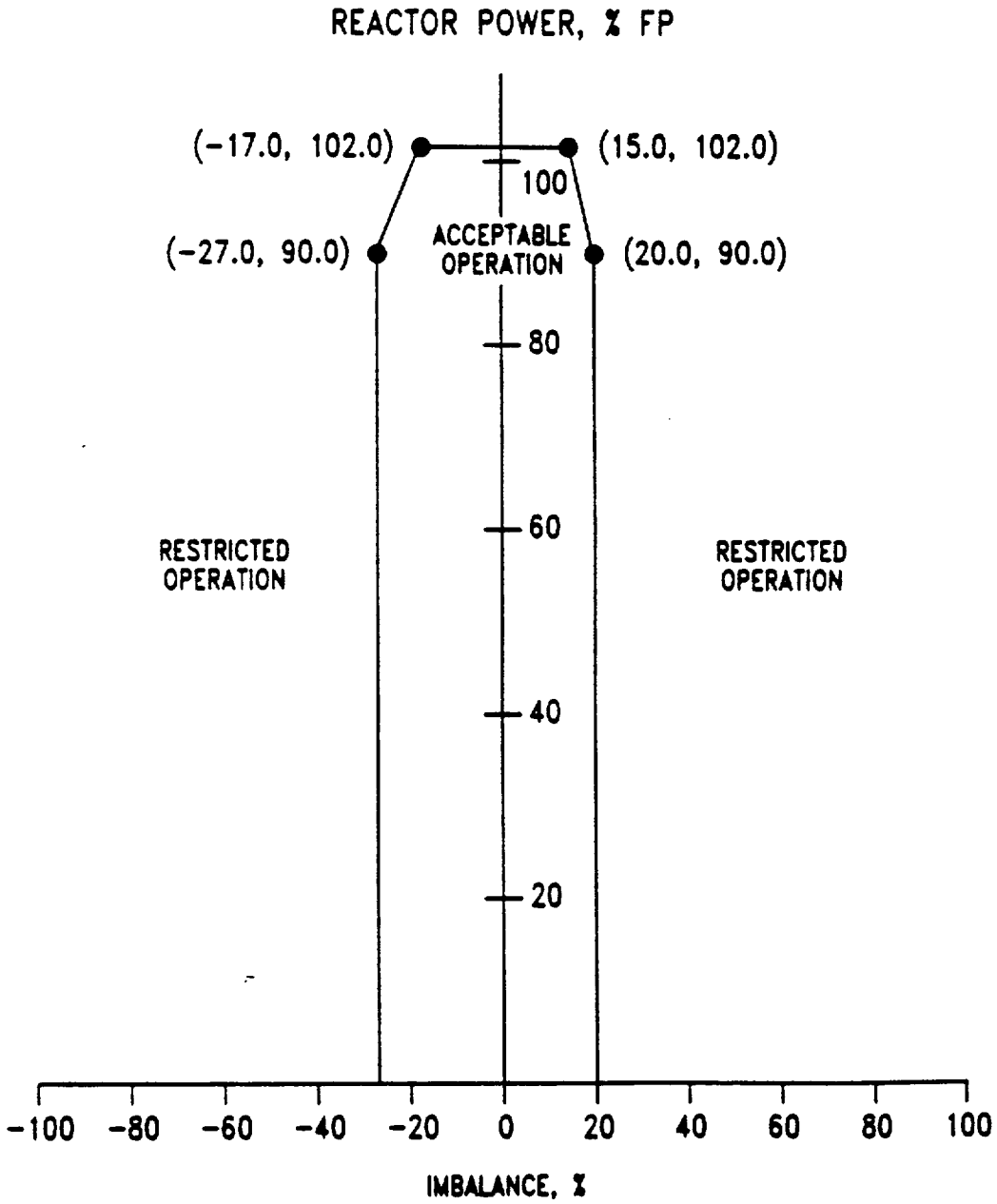
(4) FSAR, Section 15.14.5



OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 EFPD TO EOC UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-10



OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 EFPD TO EOC UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-11



OPERATIONAL POWER IMBALANCE ENVELOPE FROM 0 EFPD TO EOC UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-12

These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.1.4 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation.

Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The low pressure injection pump is used to maintain a uniform boron concentration. (1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. (2) The boron concentration will be maintained above 1950 ppm. Although this concentration is sufficient to maintain the core $K_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The K_{eff} with all rods in the core and with refueling boron concentration is approximately 0.90. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing of the Reactor Building purge isolation is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.11 is required, as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.(3)

The off-site doses for the fuel handling accident are within the guidelines of 10 CFR 100; however, to further reduce the doses resulting from this accident, it is required that the spent fuel pool ventilation system be operable whenever the possibility of a fuel handling accident could exist.

Specification 3.8.13 is required as the safety analysis for a postulated cask handling accident was based on the assumptions that spent fuel stored as indicated has decayed for the amount of time specified for each spent fuel pool.

Specification 3.8.14 is required to prohibit transport of loads greater than a fuel assembly with a control rod and the associated fuel handling tool(s).

REFERENCES

(1) FSAR, Section 9.1.4
(2) FSAR, Section 15.11.1
(3) FSAR, Section 15.11.2.1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE DPR-47

AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

By letter dated May 16, 1988 (Ref. 1), Duke Power Company, the licensee, submitted an application to reload Unit No. 3 of the Oconee Nuclear Station and operate it for Cycle 11. To support the application the licensee submitted report DPC-RD-2011 (Ref. 2) entitled "Oconee Unit 3 Cycle 11 Reload Report" and proposed changes to the Unit 3 Technical Specifications.

The Cycle 11 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 11 is to have an operating length of approximately 410 effective full power days (EFPD). As has been the case for Cycle 10, Cycle 11 will be operated in a rods out, feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length silver-indium-cadmium (Ag-In-Cd) control rods and 60 burnable poison rod assemblies (BPRAs). In addition, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The licensed core full power level remains at 2568 MWt.

2.0 EVALUATION

2.1 Fuel Design

Cycle 11 will contain 16 Mark B4 fuel assemblies in Batches 8C and 9C, 25 Mark B5Z assemblies in Batch 11B, 60 Mark B5Z assemblies in Batch 12, and 60 Mark B7 assemblies in Batch 13. All of these fuel assemblies are mechanically interchangeable. The Mark B7 fuel incorporates slightly longer fuel rods and a shorter lower end fitting. The longer fuel rods have increased plenum volume allowing for higher fuel burnup. The shoulder gap has also been increased to provide additional margin for fuel rod growth.

Because of the previous incore exposure of Batch 9C fuel, it is the most limiting in terms of cladding creep collapse. The licensee has stated that the cladding collapse time for the most limiting Cycle 11 assembly was conservatively determined to be greater than the maximum projected residence time for any Cycle 11

assembly. The methods and procedures used for the analyses (Ref. 3) have been previously reviewed and approved by the staff. The staff concludes that cladding collapse has been appropriately considered and will not occur for Cycle 11 operation.

The cladding stress and strain analyses for the Cycle 11 fuel designs were calculated using methods and limits previously reviewed and approved by the NRC. The staff concludes that the analysis of cladding stress and strain has been appropriately considered for Cycle 11 operation and is acceptable.

The thermal behavior of all fuel in the Cycle 11 core is virtually identical. The thermal analysis was performed with the approved TACO2 code (Ref. 4) and the Cycle 11 core protection limits were based on the calculated linear heat rate (LHR) to centerline fuel melt limits. These limiting values are satisfactorily incorporated into the Technical Specifications for Cycle 11 through the operating limits on rod index and axial power imbalance.

Standard Review Plan 4.2, Section II.A.1(f), contains the requirement that the fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Based on TACO2 analyses, the licensee has stated that the internal pressure in the highest burnup rod of each fuel type will not reach the nominal reactor coolant system (RCS) pressure of 2200 psia. The staff finds this acceptable and concludes that the fuel rod internal pressure limits have been adequately considered for Cycle 11 operation.

Based on its review, the staff concludes that approved methods have been used, that the fuel design parameters meet applicable criteria and that the fuel design for Oconee Unit 3 Cycle 11 is acceptable.

2.2 Nuclear Design

The nuclear design parameters characterizing the Oconee Unit 3 Cycle 11 core have been computed by methods previously used and approved for B&W reactors (Ref. 5). Comparisons have been made between the parameters for Cycle 10 and Cycle 11. Differences in the fuel shuffle patterns and the 32 Mark B assemblies inserted in Cycle 11 from the spent fuel pool account for the differences in ejected and stuck control rod worths, critical boron concentrations, and moderator temperature coefficients.

Shutdown margin calculations for Cycle 11 include the effects of poison material depletion, a 10% calculational uncertainty, allowance for rod bite, and neutron flux redistribution as well as a maximum worth stuck rod. Beginning of cycle (BOC) and end of cycle (EOC) shutdown margins show adequate reactivity worth exists above the total required worth during the cycle. Shutdown margins at BOC and EOC are 3.15% delta k/k and 2.99% delta k/k, respectively, compared to the minimum required value of 1.0% delta k/k.

Based on its review, the staff concludes that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of Oconee Unit 3 Cycle 11 is acceptable.

2.3 Thermal-Hydraulic Design

Although a full Mark BZ core and a full Mark B core provide practically the same departure from nucleate boiling (DNB) margin for both steady-state and transient conditions (Ref. 6), incompatibility in the hydraulic characteristics has an effect on thermal margin during transitional mixed core cycles when both Mark BZ and Mark B fuel assemblies co-exist in the core. Since the Mark BZ assemblies have a higher hydraulic resistance due to the BPRA retainers and the Zircaloy intermediate spacer grids, some of the coolant flow is diverted from Mark BZ fuel to the lower power Mark B fuel. The fact that the Mark BZ assemblies have less flow in a mixed core results in lower maximum allowable power peaking and a lower enthalpy rise factor required in order to maintain the same DNBR limit compared to a whole core of Mark BZ fuel. This transition core effect has been suitably accounted for in the generic Mark BZ thermal-hydraulic design analyses performed by Duke Power Company to support Cycle 11 operation.

The thermal design flow for Unit 3 Cycle 11 is no greater than 385,440 gpm (109.5% of design flow). This has been increased from 106.5% in order to offset BOC power distribution predictions. The thermal-hydraulic analyses for Units 1 and 2 will continue to be based on 106.5% of design flow. The licensee has calculated that the minimum departure from nucleate boiling ratio (DNBR) will remain greater than the applicable BWC critical heat flux correlation limit of 1.18 or the BAW-2 correlation limit of 1.30 with this thermal design flow. Since the analysis was performed with NRC approved methodology and RCS flow is monitored to assure that actual flow is greater than that assumed in the analysis, the staff finds the increased thermal design flow and the resulting thermal limits acceptable.

2.4 Accident and Transient Analysis

The important physics, thermal-hydraulic, and kinetics parameters for Cycle 11 have been compared to the values used in the previous cycle, the FSAR (Ref. 7) and/or for the fuel densification report (Ref. 8). The licensee has shown that the Cycle 11 values are bounded by those previously used and, therefore, the transient and accident evaluation of Cycle 11 are considered to be bounded by previously accepted analyses.

Babcock and Wilcox (B&W) has performed a generic loss of coolant accident (LOCA) analysis for the B&W 177-FA lowered-loop nuclear steam supply system (NSSS) using the final acceptance criteria emergency core cooling system (ECCS) evaluation model (Ref. 9). The combination of average fuel temperature as a function of LHR and the lifetime pin pressure data used is conservative relative to these calculated for this cycle. Two sets of bounding values for allowable LOCA peak LHRs for Cycle 11 are given as a function of core height.

These limits apply during the periods 0 to 25 EFPD and for the balance of the cycle. These results are based upon a bounding analytical assessment of NUREG-0630 on LOCA and operating LHR limits performed by B&W (Ref. 10). The B&W analyses have been approved by the NRC staff and the LHR limits are satisfactorily incorporated into the Technical Specifications for Cycle 11 through the operating limits on control rod withdrawal index and axial power imbalance.

2.5 Technical Specification Changes

Oconee Unit 3 Cycle 11 Technical Specifications have been modified by these amendments to reflect changes in power peaking and control rod worths, and BOC boron concentration requirements. The specific changes are evaluated below.

(1) Page 2.1-2. The Bases of Specification 2.1, "Safety Limits, Reactor Core," have been updated to state that the core protection safety limits are based on a RCS flow less than or equal to 385,440 gpm (109.5% of design flow). As discussed in Section 4.0 of this safety evaluation, the thermal design flow for Unit 3 Cycle 11 was increased from 106.5% of design flow to 109.5% of design flow in order to offset BOC power distribution predictions. The resulting minimum DNBRs were found to be greater than the applicable limits and, therefore, this change is acceptable.

(2) Page 3.2-1. Specification 3.2, "High Pressure Injection and Chemical Addition Systems," has been updated to increase the minimum required concentrated boric acid storage tank (CBAST) volume from 1020 ft^3 to 1100 ft^3. This increase will ensure that the CBAST can borate the RCS to 1% delta k/k subcritical at cold conditions with the maximum worth stuck rod and no credit for xenon at the most limiting time in core life. The Oconee 3 Cycle 11 moderator dilution accident would still remain bounded by the FSAR analyses. The change is, therefore, acceptable.

(3) Page 3.2-2. The Bases of Specification 3.2 have been updated to provide consistency with the revised CBAST volume requirements of Specification 3.2.2 and the minimum required boron concentration in the borated water storage tank (BWST) given in Specification 3.3.4.b. These changes are acceptable.

(4) Page 3.3-3. Specification 3.3.4.b, "Borated Water Storage Tank," has been updated to increase the minimum required BWST boron concentration to 1950 ppm. As mentioned in item 2, this increase will ensure that adequate shutdown margin exists during refueling and is acceptable. The deletion of the footnote regarding Unit 3 Cycle 10 BWST boron concentration is acceptable since the 2010 ppm limit is not applicable to Cycle 11. The changes to the associated Bases are also acceptable.

8. "Oconee 3 Fuel Densification Report," BAW-1399, November 1973.

9. "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," BAW-10103, Rev. 3, July 1977.

10. "Bounding Analytical Assessment of NUREG-0630 on LOCA kw/ft Limits with use of FLECSET," BAW-1915P, May 1986.

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Dated: September 19, 1988

(5) Pages 3.5-24, -25, -26. Figures 3.5.2-20, -11, -12, "Operational Power Imbalance Envelope," have been updated to provide an envelope for all 3 Oconee units. These updated limits are acceptable since they are derived from analyses performed using approved methods and have been appropriately considered in the safety analyses.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, the Commission has determined that the issuance of these amendments will have no significant impact on the environment. (53 FR 36140).

4.0 CONCLUSION

The Commission issued a Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for Hearing which was published in the Federal Register (53 FR 28735) on July 29, 1988, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

5.0 REFERENCES

1. Letter from W. H. Owen (Duke) to USNRC, "Oconee Nuclear Station Unit 3 Cycle 11 Reload Technical Specifications," May 16, 1988.

2. "Oconee Unit 3 Cycle 11 Reload Report," DPC-RD-2011, May 1988.

3. "Program to Determine In-Reactor Performance of B&W Fuels, Cladding Creep Collapse," BAW-10084A, Revision 2, October 1978.

4. "TACO2: Fuel Performance Analysis," BAW-10141P-A, Revision 1, June 1983.

5. "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," BAW-10152-A, June 1985.

6.. "Rancho Seco Nuclear Generating Station, Evaluation of Mark BZ Fuel Assembly Design," USNRC, November 16, 1984.

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

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY

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

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 170 , 170 , and 167 to Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 issued to Duke Power Company, (the licensee) which revised the Technical Specifications for operation of the Oconee Nuclear Station, Units 1, 2, and 3 (the facility) located in Oconee County, South Carolina. The amendments were effective as of the date of issuance.

The amendments revise the Technical Specifications to support operation of Unit 3, Cycle 11, at full rated power.

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.

Notice of Consideration of Issuance of Amendments and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on July 29, 1988 (53 FR 28735). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of these amendments will not have a significant effect on the quality of the human environment. (53 FR 36140)

For further details with respect to the action see (1) the application for amendments dated May 16, 1988, (2) Amendment Nos. 170, 170, and 167 to License Nos. DPR-38, DPR-47, and DPR-55 and (3), the Commission's related Safety Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., and at the Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691. 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 Reactor Projects I/II.

Dated at Rockville, Maryland this 19th day of September 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

Helen N. Pastis, Project Manager
Project Directorate II-3
Division of Reactor Projects -I/II

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