March 19, 1987

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

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DSR
016

Mr. Hal B. Tucker
Vice President - Nuclear Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 155 , 155 and 152 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated December 12, 1986, as revised on January 29, 1987, and supplemented on February 11, 1987.

These amendments revise the TSs to support the operation of Oconee Unit 3 at full rated power during the upcoming Cycle 10. These amendments also revise the TSs on the Oconee Unit 1 operational power imbalance curve, quadrant power tilt and xenon reactivity. For Oconee Unit 3 only, these amendments raise the minimum boron concentration in the borated water storage tank from 1835 parts per million (ppm) to 2010 ppm to ensure that the core shutdown margin is at least 1% Δ k/k with all control rods out and the core at 70°F. Other administrative changes requested in your February 11, 1987, application are being handled separately.

In your February 11, 1987 letter, you requested that these amendments be treated as an emergency because insufficient time exists for the Commission's usual 30-day notice without extending the current outage. Because of the early shutdown of Oconee Unit 3, you determined that emergency circumstances exist for approval of these proposed revisions to support startup of Oconee Unit 3, Cycle 10.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing of the enclosed amendments will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/S/

Helen N. Pastis, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 155 to DPR-38
2. Amendment No. 155 to DPR-47
3. Amendment No. 152 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

PBD-6	PBD-6	PBD-6	~~PBD-6~~	
RIngram	HPastis:	GEdison	JStolz	GGWoodhead
3/12/87	3/12/87	3/12/87	3/12/87	3/13/87

Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

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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. 155
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 December 12, 1986, as revised on January 29, 1987 and supplemented on February 11, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 155, 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

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 19, 1987

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. 155
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 December 12, 1986, as revised on January 29, 1987, and supplemented on February 11, 1987, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 155, 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

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 19, 1987

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. 152
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 December 12, 1986 as revised on January 29, 1987, and supplemented on February 29, 1987, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 152, 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

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 19, 1987

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 155 TO DPR-38

AMENDMENT NO. 155 TO DPR-47

AMENDMENT NO. 152 TO DPR-55

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

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment numbers and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3.2-2	3.2-2
3.3-3	3.3-3
3.3-6	3.3-6
3.5-8	3.5-8
3.5-10	3.5-10
3.5-12	3.5-12
3.5-24	3.5-24
3.5-26	3.5-26
3.8-3	3.8-3

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 1020 ft^3 of 11,000 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1835* 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

* 2010 ppm boron for Unit 3, Cycle 10 only.

b. The BWST shall contain a minimum level of 46 feet of water having a minimum concentration of 1835**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 3.3.5.b(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 Specifications 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 RBC 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.

***For the "3A" RBC train, a one-time extension of inoperability is granted in order to allow for repair, provided both RBS trains are operable and that the "3A" RBC train is returned to service no later than 11:59 p.m., April 20, 1985.**

**** 2010 ppm boron for Unit 3, Cycle 10 only.**

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 1835* 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

* 2010 ppm boron for Unit 3, Cycle 10 only.

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 100% full power by 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.

3.5.2.6 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 3.5.2-10 (Unit 1). If the imbalance
3.5.2-11 (Unit 2)
3.5.2-12 (Unit 3)

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.

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

3.5.2.8 The operational limit curves of Technical Specifications 3.5.2.5.c and 3.5.2.6 are valid for a nominal design cycle length, as defined in the Safety Evaluation Report for the appropriate unit and cycle. Operation beyond the nominal design cycle length is permitted provided that an evaluation is performed to verify that the operational limit curves are valid for extended operation. If the operational limit curves are <u>not</u> valid for the extended period of the operation, appropriate limits will be established and the Technical Specification curves will be modified as required.

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.

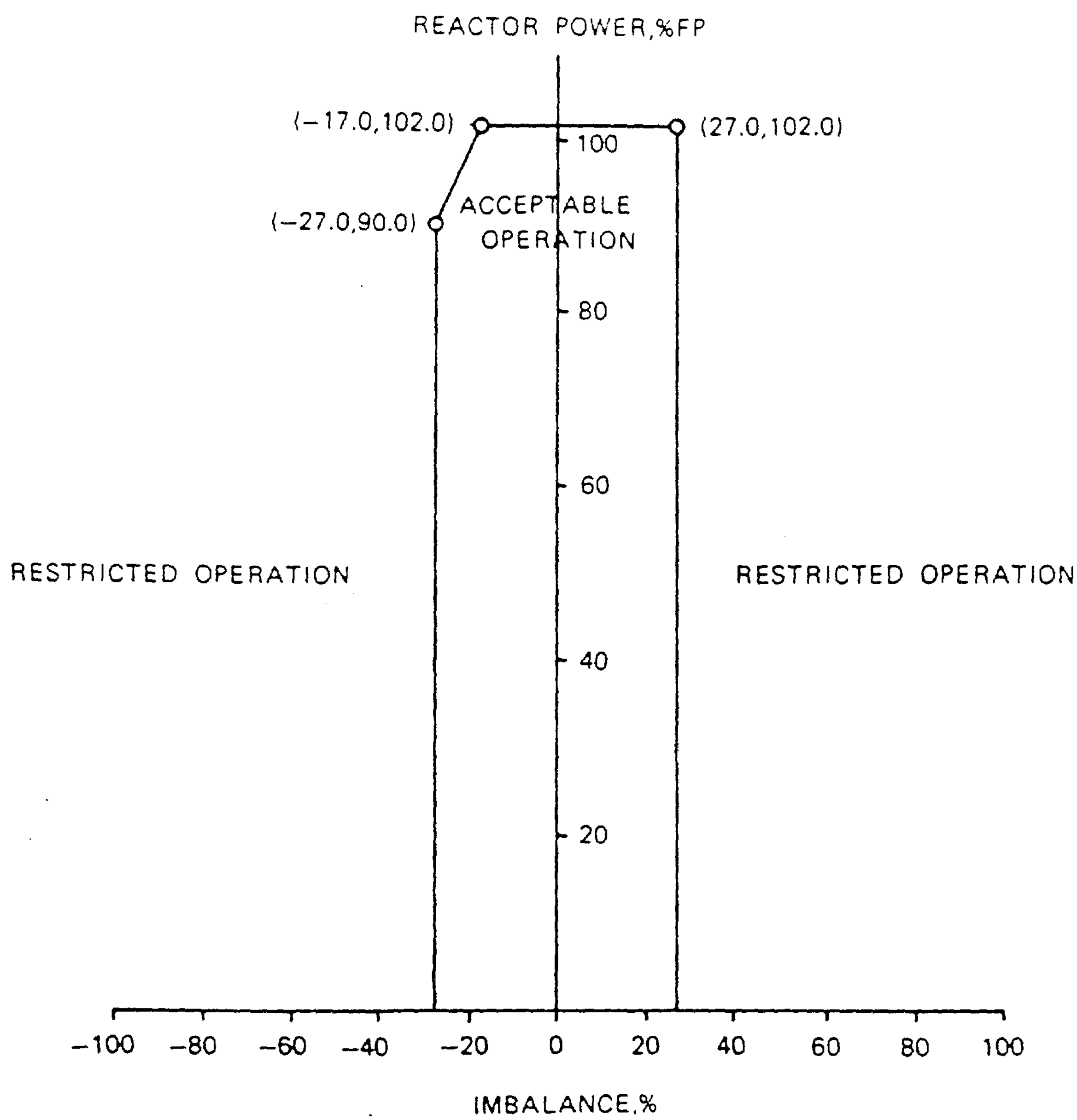
Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Group 7 to be partially inserted.

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 3.5.2.6, 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.

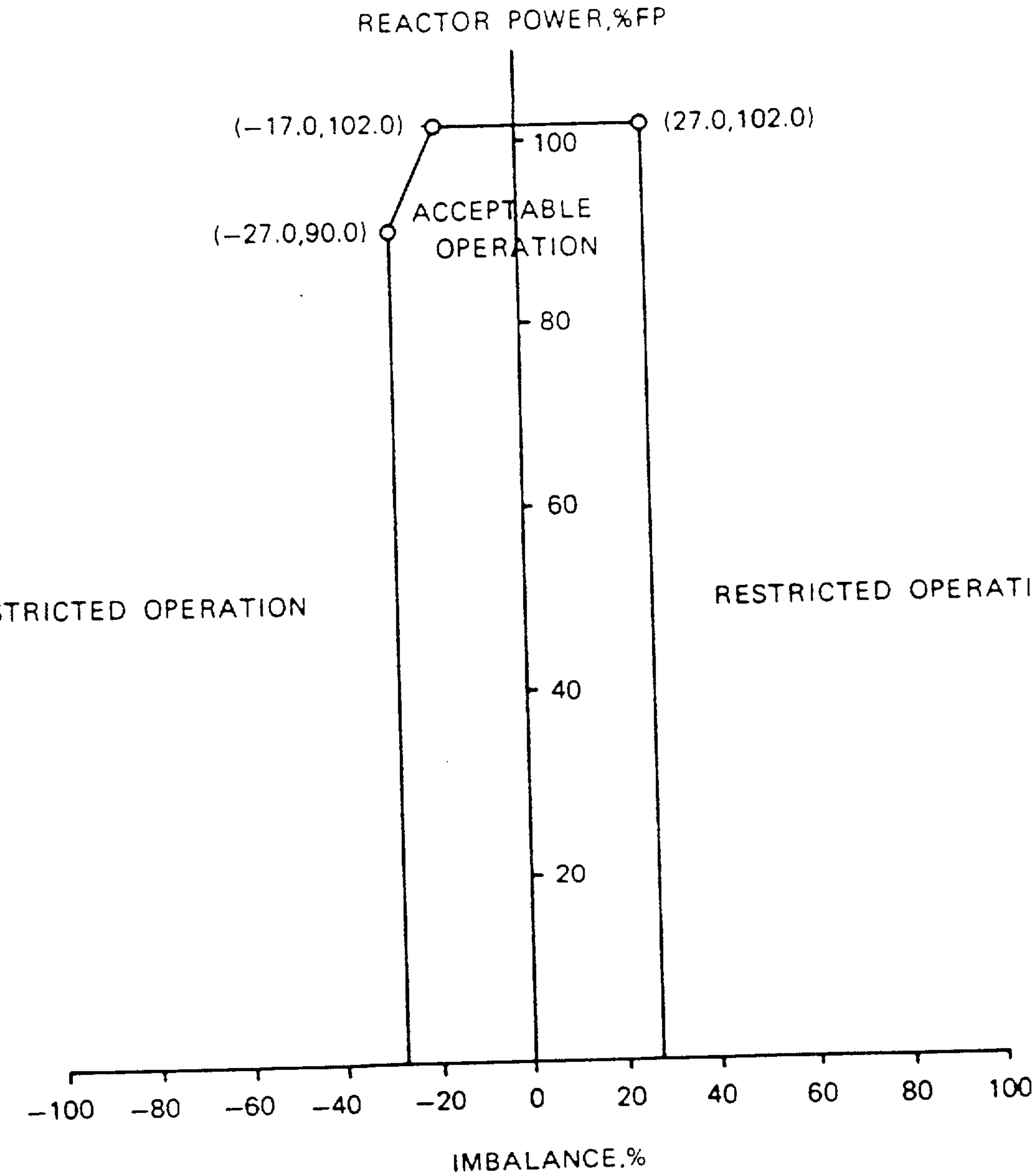
Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions resulting from transient xenon power peaking are implicitly included in the limits of Section 3.5.2.5 (control rod positions) and 3.5.2.6 (reactor power imbalance). Since these limits are set during the cycle-specific maneuvering analysis to prevent excessive power peaking by transient xenon at all power levels, there is no need for any hold at a power level cutoff below 100% FP.



OPERATIONAL POWER
IMBALANCE ENVELOP
FROM 0 EFPD TO EOC UNIT 1
OCONEE NUCLEAR STATION

FIGURE 3.5.2-10



OPERATIONAL POWER
IMBALANCE ENVELOP
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 1835* 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

* 2010 ppm boron for Unit 3, Cycle 10 only.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

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

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

INTRODUCTION

By letter dated December 12, 1986 (Ref. 1), as revised on January 29, 1987 (Ref. 2) and supplemented on February 11, 1987 (Ref. 6), Duke Power Company (the licensee) proposed changes to the Technical Specifications (TSs) of Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments would consist of changes to the Station's common TSs. Oconee Unit 3 is currently completing a refueling outage.

These amendments would authorize changes to the Oconee Nuclear Station TSs which are required to support the operation of Oconee Unit 3 at full rated power during the upcoming Cycle 10. The amendments would change Figure 3.5.2-12, the Unit 3 Operational Power Imbalance Envelope curve. The Figure would be updated to reflect current cycle operating characteristics.

These amendments would also provide a more conservative curve for Oconee Unit 1 Operational Power Imbalance Envelope (Figure 3.5.2-10) to allow 10 CFR Part 50.59 reviews of future core reloads; update TS 3.5.2.4.b.2 (quadrant power tilt) to reflect the fact that power level cutoffs (other than 100%) are no longer applicable to Oconee; delete TS 3.5.2.6 (xenon reactivity) because operating restrictions resulting from transient xenon power peaking are implicitly included in the limits of TS 3.5.2.5 (control rod positions) and proposed TS 3.5.2.6 (reactor power imbalance) and note this in the bases of TS 3.5; and change TSs 3.5.2.7, 3.5.2.8 and 3.5.2.9 to reflect the deletion of TS 3.5.2.6 (xenon reactivity).

For Oconee Unit 3 only, these amendments would raise the minimum boron concentration in the borated water storage tank (BWST) from 1835 parts per million (ppm) to 2010 ppm to ensure that the core is at one percent delta k over k, 1% Δ k/k or shutdown margin, at 70°F without any control rods in the core. Other administrative type changes requested in the February 11, 1987 application are being handled separately.

To support the license amendment application, the licensee submitted "Oconee Unit 3, Cycle 10 Reload Report" as an attachment to the December 12, 1986 application. A summary of the Cycle 10 operating parameters is included in the report, along with safety analyses. On January 29, 1987, the licensee revised the reload report because Oconee Unit 3 was shutdown on December 17, 1986--earlier than scheduled because of possible wear indications in the 3B2 reactor coolant pump. The Oconee Unit 3 Cycle 10 core was then redesigned based on the shortened Cycle 9 length of 349 effective full power days. Results of this redesign indicated that to ensure the core will be shutdown in conformance with applicable criteria, the beginning of cycle, all rods out, 70°F 1% Δ k/k shutdown boron concentration should be increased from the present 1835 to 2010 ppm. In a letter dated February 11, 1987, the licensee proposed revisions to the TSs to raise the minimum boron concentration in the BWST.

The Cycle 10 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. The fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches have an average nominal fuel loading of 463.6 kilograms (kg) uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 4-1 (Ref. 3). The Cycle 10 core loading diagram, enrichments, control rods and burnable poison rod assembly (BPRA) locations and enrichments are also given in Reference 3.

Cycle 10 will operate in a rods-out, boron feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 60 BPRAs. In addition to the full-length control rods, eight Inconel gray axial power shaping rods (APSRs) are provided for additional control of axial power distribution. Since gray APSRs are being utilized, there are eight control rods in group seven and twelve in group five to reduce the negative offset response to the group seven rod movement.

The present reload involves no significant changes in core fuel design or methodology. Proposed revisions to the TSs required for Cycle 10 operation were made in accordance with methods and procedures found acceptable in connection with previous reloads (Ref. 4) and are the result of minor cycle-to-cycle fuel changes.

EVALUATION

Evaluation of Fuel System Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 3 Cycle 10 are listed in Table 4-1 (Ref. 3). All fuel assemblies are mechanically interchangeable. Two regenerative neutron sources will be used in the Mark BZ fuel assemblies. The Cycle 10 core contains only fuel designs which have been previously loaded in the Oconee Unit 3 reactor and have been previously approved by the NRC staff. The fuel rod design, cladding collapse, cladding stress and strain, and the thermal design fuel analyses for Cycle 10 fuel designs, including the gray APSRs, are either bounded by conditions previously analyzed for Oconee 3 or were analyzed specifically for Cycle 10 using methods and limits previously reviewed and approved by the NRC staff. Therefore, we conclude that the overall fuel system design for Oconee 3 Cycle 10 is acceptable.

Nuclear Design

Table 5-1 (Ref. 3) compares the core physics parameters of Cycle 10 with those of the reference Cycle 9. The values for Cycle 9 and Cycle 10 were generated by Duke Power Company using the reload design methods described in Reference 5 which have been reviewed and approved by the NRC staff.

We have determined that approved methods have been used, and the nuclear design parameters meet the acceptance criteria of Standard Review Plan, Section 4.3, Part II, and, therefore, conclude that the nuclear design of Oconee 3 Cycle 10 is acceptable.

Evaluation of Thermal-Hydraulic Design

The generic Mark B and Mark BZ thermal-hydraulic design analyses supporting Cycle 10 operation were performed by Duke Power Company using the methods described in Reference 5. The Cycle 9 and Cycle 10 thermal-hydraulic design conditions are summarized in Table 6-1 (Ref. 3).

The Cycle 10 core will include 60 fresh Mark BZ Batch 12 fuel assemblies, all of which will contain BPRAs. This results in a core bypass flow of 7.9% of the total system flow, which is less than the bypass flow assumed in the generic thermal-hydraulic analyses.

The Mark BZ fuel assembly has a slightly higher pressure drop than the Mark B assembly as a result of the increased flow resistance of the Zircaloy spacer grids. The presence of Mark BZ and Mark B assemblies in a core results in less coolant flow in the Mark BZ fuel than would occur in an all Mark BZ core. The generic Mark BZ analyses conservatively account for this transition core effect.

In a Mark BZ transition core, the limiting Mark B hot channel will receive more coolant and yield better departure from nucleate boiling (DNB) performance than would be predicted for a full Mark B core. Thus, the generic Mark B analyses, based on the B&W-2 critical heat flux (CHF) correlation, are bounding and are applicable to the Cycle 10 transition core.

We have determined that approved methods have been used, and the thermal-hydraulic design parameters meet the departure from nucleate boiling ratio (DNBR) safety limit using approved CHF correlations and, therefore, conclude that the thermal-hydraulic design of Oconee 3 Cycle 10 is acceptable.

Safety Analyses

The important kinetics parameters for Cycle 10 have been compared to the values used in the Final Safety Analysis Report (FSAR) and/or the densification report. The licensee has shown that the Cycle 10 values are bounded by those previously used. The licensee has also determined that the initial conditions of the transients in Cycle 10 are bounded by either the FSAR, the fuel densification report, previous reload analyses, or analyses using approved methods.

Babcock & 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 using the final acceptance criteria Emergency Core Cooling System evaluation model. The combination of average fuel temperature as a function of linear heat rate (LHR) and the lifetime pin pressure data used is conservative relative to those calculated for this cycle. These results are based upon a bounding analytical assessment of NUREG-0630 on LOCA and operating LHR limits performed by B&W. The B&W analyses have been approved by the NRC staff and the LHR limits are satisfactorily incorporated into the TSs for Cycle 10 through the operating limits on rod index and axial power imbalance and, therefore, are acceptable.

Technical Specification Modifications

Oconee Unit 3 Cycle 10 TSs have been modified to account for normal cycle-to-cycle fuel changes in power peaking and control rod worths. We have reviewed the proposed specification revisions for Cycle 10. These changes concern the Operational Power Imbalance Envelope (Figure 3.5.2-12). In addition, the licensee has provided a more conservative curve for the Unit 1 Operational Power Imbalance Envelope (Figure 3.5.2-10) in order to reduce future TS changes and to allow more of their future reload cores to be reviewed under 10 CFR 50.59. On the basis that approved methodology was used to obtain these limits which assure that general design criteria 10 and 12 are satisfied, we find these TS modifications acceptable.

The licensee also proposed administrative changes to TSs 3.5.2.4.b.2, 3.5.2.6, 3.5.2.7, 3.5.2.8 and 3.5.2.9 which are common to all three Oconee units. These changes reflect the fact that power level cutoff is no longer applicable to Oconee and operating restrictions resulting from transient xenon power peaking are implicitly included in the control rod position and reactor power imbalance limits. Therefore, we find these changes acceptable.

Increase in Boron Concentration in the Borated Water Storage Tank

As a result of a shortened Cycle 9 of Oconee Unit 3 the design of the Cycle 10 core will require an increase in the BWST boron concentration to ensure the core will be shutdown in conformance with TS 3.8.4 and TS 3.3 criteria. By letter dated February 11, 1987 (Ref. 3), as supplemented on February 27, 1987 (Ref. 6), Duke Power Company presented the results of its analysis which indicates that the beginning of cycle, all rods out, 70°F, 1 percent delta k over k shutdown boron concentration should be increased from the present 1835 ppm to 1873 ppm in order to meet the 1 percent subcritical acceptance criteria. Duke has requested TS changes which will conservatively increase the minimum concentration in the Oconee Unit 3 BWST to 2010 ppm for Cycle 10.

We have determined that approved methods have been used to insure that the 1 percent subcritical acceptance criteria are conservatively met, and that the plant will remain bounded by the FSAR safety analyses. Therefore, we conclude that the increase in the BWST boron concentration to 2010 ppm for Oconee Unit 3 Cycle 10 is acceptable.

EMERGENCY CIRCUMSTANCES

In its February 11, 1987 letter, the licensee requested that these amendments be treated as an emergency because insufficient time exists for the Commission's usual 30-day notice without extending the current outage. Because of the early shutdown of Oconee Unit 3, the licensee determined that emergency circumstances exist for approval of these proposed revisions to support startup of Oconee Unit 3, Cycle 10.

The licensee revised the reload report because Oconee Unit 3 was shutdown on December 17, 1986 - earlier than scheduled because of possible wear indications on a reactor coolant pump. The Oconee Unit 3 Cycle 10 core was then redesigned based on the shortened cycle. Results of this redesign indicated that to ensure the core shutdown margin, the boron concentration in the BWST would need to be increased from the present 1835 to 2010 ppm. In its February 11, 1987 letter, the licensee proposed revisions to the TSs to raise the minimum boron concentration in the BWST.

The Commission has determined that emergency circumstances exist in that swift action is necessary to avoid a delay in startup not related to safety and finds that, for the reason stated above, emergency circumstances exist.

In connection with a request indicating an emergency, the Commission expects its licensees to apply for license amendments in a timely fashion. However, with this consideration in mind, it has been determined that a circumstance has arisen where the licensee and the Commission must act quickly, and the licensee has made a good effort to make a timely application.

FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

We have confirmed the basis of the no significant hazards consideration findings described in the notice published in the FEDERAL REGISTER on March 6, 1987 (52 FR 7050). The amendments change the TSs to reflect new operating limits based on the fresh fuel to be inserted into the core. These parameters are based on the new physics of the core and fall within the acceptance criteria. There are no significant changes in the fuel being used, or the fuel assembly design. We have previously reviewed postulated fuel-related transients and accidents. As part of these analyses, bounding parameters were used, for example, power peaking limits and reactor system pressure. Accident analyses previously submitted by the licensee and approved by the NRC staff for Oconee 3 utilized input values of physics parameters which are designed to be bounding for various operating cycles and operating conditions. The power imbalance limit curve for Cycle 10 was derived by the licensee so that the previous analyses for the postulated accidents would remain valid for Cycle 10. Therefore, it was unnecessary to analyze any accident for Cycle 10 of Oconee 3. Since the postulated accidents previously analyzed remain applicable to the new core (i.e., continue to be bounding), the probability or consequences of an accident previously evaluated have not increased. Because of the fundamental identity of the new fuel in terms of its nuclear and fuel assembly design, the possibility of a new or different kind of accident from any accident previously evaluated has not been created. Finally, the power imbalance curve ensures that the licensed margin of safety has not been reduced.

To ensure that the core shutdown margin is 1 percent delta k over k, at 70°F without any control rods in the core, the minimum boron concentration in the BWST will have to be raised from 1835 ppm to 2010 ppm. We have confirmed that approved methods have been used to ensure that the 1 percent subcritical acceptance criteria are conservatively met, and that the plant will remain bounded by the FSAR safety analyses. Therefore, the probability of any Design Basis Accident (DBA) is not affected by this change, nor are the consequences of a DBA affected by this change. The key physics parameter affected by the Oconee Unit 3 Cycle 10 redesign is the BOC boron concentration. The limiting FSAR transient with respect to changes in the boron concentration is the moderator dilution transient at power. Only the non-LOCA boron dilution transient was found to have a more potentially severe result due to increased boron concentration. This event is bounded by the values assumed in the FSAR. Therefore, the moderator dilution transient presented in the FSAR remains conservative for Oconee Unit 3, Cycle 10. Analysis of the increase in the Oconee Unit 3, Cycle 10, minimum BWST boron concentration has indicated that the 2010 ppm concentration is well within all acceptance criteria. For refueling and LOCA conditions, the proposed concentration is sufficient to maintain the core 1 percent subcritical at 70°F with all control rods removed; this change affects only previously evaluated accidents, discussed above, and does not create the possibility of a new or different kind of accident from any accident previously evaluated. The predicted boron concentration required to maintain the core 1 percent subcritical at 70°F with all rods out of the core during refueling or a LOCA has been compared to the current TS value for the BWST. The predicted BOC, all rods out, 70°F, 1 percent subcritical boron concentration of 1873 ppm has necessitated a change in the required boron concentration for the BWST from 1835 ppm. To provide additional shutdown margin during refueling or a LOCA, a more conservative BWST boron concentration of 2010 ppm will be used. For the non-LOCA events, the moderator dilution transient has been shown to be bounded by the FSAR analysis and involves no significant reduction in a margin of safety.

Therefore, we conclude that:

(1) Operation of the facilities in accordance with the amendments would not significantly increase the probability or consequences of an accident previously evaluated.

(2) Operation of the facilities in accordance with the amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facilities in accordance with the amendments would not involve a significant reduction in a margin of safety.

Accordingly, we conclude that the amendments to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 to support operation of Oconee Unit 3 at full rated power during the upcoming Cycle 10, involve no significant hazards considerations.

STATE CONSULTATION

In accordance with the Commission's regulations, consultation was held with the State of South Carolina by telephone. The State expressed no concern either from the standpoint of safety or of our no significant hazards consideration determination.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

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.

Dated: March 19, 1987

Principal Contributor:
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REFERENCES

1. Letter, H. B. Tucker (Duke) to H. R. Denton (NRC), "Oconee Nuclear Station Unit 3," December 12, 1986.

2. Letter, H. B. Tucker (Duke) to H. R. Denton (NRC), "Oconee Nuclear Station," January 29, 1987.

3. Report, "Oconee Unit 3, Cycle 10 Reload Report," DPC-RD-2008, Duke Power Company, January 1987.

4. Letter, H. Nicolaras (NRC) to H. B. Tucker (Duke), September 19, 1985.

5. Report, "Oconee Nuclear Station Reload Design Methodology II," DPC-NE-1002, Duke Power Company, Charlotte, North Carolina, March 1985.

6. Letter, H. B. Tucker (Duke) to H. R. Denton (NRC) "Oconee Nuclear Station", February 27, 1987.